Stephen A. Byrne
Senior Vice President, Nuclear Operations
803.345.4622

SCE&G
A SCANA COMPANY

June 12, 2003
RC-03-0112

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Ladies and Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION (RAI) FOR THE REVIEW OF THE LICENSE RENEWAL APPLICATION FOR VIRGIL C SUMMER NUCLEAR STATION

In Attachments I through XII you will find responses to the Requests for Additional Information (RAIs) regarding the License Renewal Application for the V. C. Summer Nuclear Station (VCSNS).

Please contact Al Paglia at (803) 345-4191 if you have additional questions or comments.

I certify under penalty of perjury that the foregoing is true and correct.

6/12/03
Executed on

Stephen A. Byrne
Senior Vice President, Nuclear Operations

AMP/SAB/mbb
Attachments

c:	N. O. Lorick	(w/o attachment)	R. C. Auluck	(CD)
	N. S. Carns	(w/o attachment)	T. B. Doerr	(w/o attachment)
	T. G. Eppink	(w/o attachment)	T. P. O'Kelley	(CD)
	R. J. White	(CD)	P. Ledbetter	(w/o attachment)
	R. B. Clary	(Hard Copy)	K. M. Sutton	(CD)
	L. A. Reyes	(CD)	NSRC	(CD)
	K. R. Cotton	(w/o attachment)	File (821.00)	(CD)
	NRC Resident Inspector	(CD)	DMS (RC-03-0112)	(CD)

A094

Attachments:

I. Response to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G), Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 2.1, 2.2, 2.3, 3.3, and Appendix B, March 28, 2003, Accession No. ML030900653

II. Response to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 2.4, 2.5, 3.6, and Appendix B, March 28, 2003, Accession No. ML030900596

III. Response to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 3.1 and Appendix B, March 28, 2003, Accession No. ML030900279

IV. Respnse to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 3.2, 3.3, 3.4, 3.5, 4.0 and Appendix B, March 28, 2003, Accession No. ML030900096

V. Response to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Section 2.3, April 9, 2003, Accession No. ML030990546

VI. Response to Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the V.C.Summer Nuclear Station (VCSNS) License Renewal Application - Section 2.2, June 2, 2003, Accession No. ML031540250

VII. ATTACHMENT TO RESPONSE; RAI 3.6-1

VIII. ATTACHMENT TO RESPONSE; RAI 3.6-2

IX. ATTACHMENT TO RESPONSE; RAI 3.6-3

X. 2002 Yearly Review of Cycle Count from WESTEMS

XI. Containment Building Post-Tensioning Plots

XII. Structural Technical Report, TR00170-003, Revision 0

References:

1. Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G), Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 2.1, 2.2, 2.3, 3.3, and Appendix B, March 28, 2003, Accession No. ML030900653

2. Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 2.4, 2.5, 3.6, and Appendix B, March 28, 2003, Accession No. ML030900596

3. Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 3.1 and Appendix B, March 28, 2003, Accession No. ML030900279

4. Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Sections 3.2, 3.3, 3.4, 3.5, 4.0 and Appendix B, March 28, 2003, Accession No. ML030900096

5. Rajender Auluck (NRC) letter to Stephen A. Byrne (SCE&G); Request for Additional Information for the Review of the VCSNS License Renewal Application - Section 2.3, April 9, 2003, Accession No. ML030990546

6. Stephen A. Byrne (SCE&G) letter (RC-02-0159) to Document Control Desk dated September 12, 2002; Criteria 2 Supplement to the Application for Renewed Operating License, Accession No. 022630347

7. Christopher I. Grimes letter dated March 15, 2002 to Mr. Alan Nelson and Mr. David Lochbaum; "License Renewal Issue: Guidance on the Identification and Treatment of Structures, Systems, and Components Which Meet 10 CFR 54.4(a)(2)", Accession No. ML020770026

8. Karen R. Cotton (NRC) letter to Stephen A. Byrne (SCE&G) dated February 20, 2001; Safety Evaluation Report - WCAP-15615, "Integrity Evaluation for Future Operation: Virgil C. Summer Nuclear Plant Reactor Vessel Nozzle to Pipe Weld Regions" (TAC NO. MB0251) Accession No. ML010510338

9. Karen R. Cotton (NRC) letter to Stephen A. Byrne (SCE&G) dated October 1, 2002; Safety Evaluation of Flaws Detected in V. C. Summer Nozzle-to-Pipe Welds in the Hot Legs of Loops B and C (TAC NO. MB4870) Accession No. ML022740071

10. G. J. Taylor (SCE&G) letter (RC-98-0185) to Document Control Desk, dated October 9, 1998; "Reactor Vessel Radiation Surveillance Program"; WCAP-15101, Rev. 0 and WCAP-15103, Rev. 0,

11. Stephen A. Byrne (SCE&G) letter (RC-03-0016) to the Document Control Desk dated January 24, 2003; "Response for Additional Information Regarding 60 Day Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity", Accession No. ML030300390

12. Stephen A. Byrne (SCE&G) letter (RC-02-0115) to the Document Control Desk dated July 3, 2002; "Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity", Accession No. ML021900011

13. Stephen A. Byrne (SCE&G) letter (RC-02-0160) to the Document Control Desk dated September 12, 2002; "Response to NRC Bulletin 2002-02 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs", Accession No. ML022590065

14. O. S. Bradham (SCE&G) letter to the Document Control Desk dated November 4, 1988, "Response to NRC Bulletin 88-09"

15. Stephen A. Byrne (SCE&G) letter (RC-02-0094) to the Document Control Desk dated May 13, 2002; "Submission of Information Requested by NRC for Integrity Evaluation for Future Operation Virgil C. Summer Nuclear Station (VCSNS) Reactor Vessel Nozzle to Pipe Weld Regions", Accession No. ML021360139

16. O. S. Bradham (SCE&G) letter to the Document Control Desk dated February 2, 1989, "Response to Generic Letter 88-14"

Attachment I
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Sections 2.1, 2.2, 2.3, 3.3, and Appendix B
Accession No. ML030900653

2.1 SCOPING AND SCREENING METHODOLOGY

RAI 2.1-1: LRA Section 2.1.1.2, "Safety-Related Criteria Pursuant to 10 CFR 54.4(a)(1)," appropriately states in the table that plant systems, structures, and components within the scope of this part are--

a. Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions--

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents. which could result in potential offsite exposures comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or § 100.11 of this chapter, as applicable [emphasis added].

During the audit, however, the staff noted that Procedures TR00160-001, "Mechanical Systems Scoping for License Renewal," dated July 3, 2002; TR00170-001, "Structures Scoping for License Renewal," dated July 3, 2002, and TR00150-001, "Electrical Systems Scoping for License Renewal," dated July 3, 2002, currently cite superseded regulatory text in establishing the scoping criteria to be used in identifying structures, systems, and components in accordance with §54.4(a)(1) requirements. Specifically, these specifications cite the following criteria in reference to §54.4(a)(1) scoping requirements:

10 CFR Part 54.4(a) -

(a) Plant systems, structures, and components within the scope of this part are--

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design bases events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions--

(i) The integrity of the reactor coolant pressure boundary,

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines [emphasis added].

Therefore, the applicant needs to provide a written evaluation that addresses the impact, if any, of not having explicitly considered in its scoping methodology those structures, systems, or components that are relied upon to ensure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable [emphasis added], consistent with the CLB.

VCSNS Response RAI 2.1-1

The text is different in the Technical Report that support the License renewal effort than the current version of the rule. The text in the technical reports *(comparable to the 10 CFR Part 100 guidelines)* was taken from the text of the original rule (May 8, 1995) as published in NEI 95-10. The words were amended December 11, 1996 to read *"comparable to those referred to in §50.34(a)(1) or §100.11 of this chapter, as applicable."* Another change was made December 23, 1999 to make the Rule read *"comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable."* These changes have no impact on how scoping was done for VCSNS. 10 CFR Part 100.11 is the specific section of Part 100 that VCSNS uses to define the site boundary and allowable doses to the public.

<u>RAI 2.1-2:</u> Long-Term Implementation

During the scoping and screening methodology audit, the team determined that the procedures reviewed, in combination with the review of a sample of scoping and screening products, provided adequate evidence that the scoping and screening process was conducted in accordance with the requirements of 10 CFR 54.4, "Scope," and 10 CFR 54.21, "Contents of Application - Technical Information." Additionally, the staff discussed the applicant's position concerning the potential long-term program implementation of the License Renewal Application (LRA) methodology and guidance into the operational phase of the plant during the extended period of operation. As a result, the team concluded that the applicant needs to formally document the process it intends to implement to capture the LRA methodology and guidance upon which the applicant will rely during the period of extended operation to satisfy the requirements of 10 CFR 54.35, "Requirements During the Term of Renewed License." The discussion should include, as appropriate, a description of the current configuration and design control processes including references to implementation guidance for those processes which are currently being reviewed for potential impact, and identification of any new process or procedures planned to address the integration of the LRA methodology and guidance into the operational phase of the plant.

VCSNS Response RAI 2.1-2

VCSNS is developing a process which will be implemented to capture the LRA methodology and guidance for use during the period of extended operation to satisfy the requirements of 10 CFR 54.35:

1) **Existing plant programs and procedures (associated with aging management) will be revised and/or enhanced to identify those commitments (governed by the license / CLB) which cannot be altered without prior review against the LRA criteria.**

2) **New "one-time inspection" aging management programs will be developed in accordance with the LRA, incorporating the commitment process identified above.**

3) **Plant procedures which impact "control of facility changes", including modifications and documentation, will be reviewed to determine an acceptable screening review process against the 10 CFR 54 requirements to ensure consistency with the LRA methodology and guidance.**

In order to support these processes, a License Renewal DBD will be developed as a guidance document which can be used for all future plant procedure, documentation and modification changes to ensure consistency with 10 CFR 54. All Technical Reports, which have been developed to substantiate the LRA submittal, are filed as permanent records and will be available for future reference and/or update.

<u>RAI 2.1-3:</u> Quality Assurance Program Attributes in Appendix A, "FSAR Chapter 18" and Appendix B, "Aging Management Program and Activities"

During the audit, the staff reviewed the applicant's programs described in Appendix A, "FSAR Chapter 18," and Appendix B, "Aging Management Program and Activities," to assure that the aging management activities were consistent with the staff's guidance described in Section A.2, "Quality Assurance for Aging Management Programs" and Branch Technical Position IQMB-1, regarding quality assurance of the LR-SRP.

Based on the staff's evaluation, the descriptions and applicability of the aging management programs and their associated attributes to all safety-related and non safety-related structures and components provided in Appendix A and Appendix B of the LRA are consistent with the staff's position regarding quality assurance for aging management. However, the applicant has not sufficiently described the applicability of the quality assurance program and its associated attributes (corrective action, confirmation process, and document control) in Appendix A of the LRA. The staff requests that the applicant revise Appendix A to include a description of the quality assurance program attributes, including references to pertinent implementing guidance. This description should be consistent with the level of detail provided in Appendix B of the LRA.

VCSNS Response RAI 2.1-3

The applicability of the VCSNS QA Program applies equally to existing programs as to new programs being developed for license renewal. Generic statements regarding the applicability of the VCSNS QA Program will be made to the FSAR Section 18.1 for all of the programs credited to manage aging effects for in-scope SSCs as follows:

> **The implementing documents are subject to administrative controls, including a formal review and approval process, and are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in the FSAR. The confirmation process is part of the Aging Management Program implementing procedures and the Corrective Action Program, which are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in the FSAR. The aging management activities required by this program would also reveal any unsatisfactory condition due to ineffective corrective action.**
>
> **The implementing documents are subject to administrative controls, including a formal review and approval process, are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in the FSAR.**

2.2 PLANT LEVEL SCOPING RESULTS

<u>RAI 2.2.2-1:</u> Tables 2.2-1, "Mechanical Scoping Results," and 2.2-2, "Structural Scoping Results," of the license Renewal Application (LRA) list the mechanical systems and plant structures that are not within the scope of license renewal. Since no descriptions and intended functions of these mechanical systems and structures are provided in the LRA and tables, the staff is unable to determine the scoping results whether the following mechanical systems and plant structures should be included in the scope of license renewal:

Mechanical Systems: emergency offsite facility (EO), emergency equipment (EQ), liquid effluent from nuclear plant to pent stock (LW), radwaste solidification, and solid handling (WD).

Structures: auxiliary fire pump house, containment access building, plant lighting masts, radiological maintenance building.

Describe these passive and long-lived mechanical components and structures and their intended functions. Justify why they should not be in scope.

VCSNS Response RAI 2.2.2-1

The plant systems, as listed in plant documents, were reviewed against the scoping criteria in 10 CFR Part 54.4(a)(1), (2) and (3). System designations that contain no permanent plant equipment were included in the list of evaluated systems. Some system designations are not in-scope as they are "inactive" designations, have no permanent plant equipment, and are not used for any reason. Some system designations are not in-scope as they have no permanent plant equipment but are used to generate repetitive task such as inspections or maintenance.

The evaluation for mechanical systems does not include the structural commodities, such as system supports, that are evaluated and presented separately. Supports were assumed to be a sufficient mitigative feature for the prevention of adverse seismic II/I system interactions. This original position categorized mechanical systems with only seismic II/I concerns as not in scope. A re-review of the license renewal scope was conducted and the results were submitted to the staff in a Supplement (Reference 6). The re-review for Criteria 2 considered seismic II/I piping within the scope of license renewal and added several mechanical systems to the scope of license renewal.

The following information is provide for the listed mechanical systems:

- **Total loss of the Emergency Offsite Facility [EO] system function will not result in the loss of any safety related functions. The components in this system are for emergency plan activities and have no direct license renewal function.**

- **Emergency Equipment [EQ] System listing, has no permanent plant equipment. This designation is used to generate a repetitive task.**

- **In the scoping for the application, Liquid Effluents from Nuclear Plant to PenStock [LW] was excluded from Mechanical LR and included in Civil/Structural screening commodities. The evaluation for Criteria 2 Supplement included LW in scope because of potential special interactions with safety related components.**

- **In the scoping for the application, RW Solidification & Solids Handling [WD] was excluded from Mechanical LR and included in Civil/Structural screening commodities. The evaluation for Criteria 2 Supplement included LW in scope because of potential special interactions with safety related components.**

The plant structures were reviewed against the scoping criteria in 10 CFR Part 54.4(a)(1), (2) and (3). The following information is provide for the listed structures:

- <u>**Auxiliary Fire Pump House**</u> **- This building is a non-nuclear safety (NNS) structure which houses auxiliary back up fire pumps, which were used during construction. There are no mechanical or electrical components in this structure, which are within license renewal scope. This structure has no effect on safety-related**

structures or components and is not used to support any of the regulated events, thus performing no intended functions for license renewal.

- **<u>Containment Access Building (CAB)</u> - The CAB is a NNS structure, which is located exterior to and away from the Containment (Reactor Building) structure. This structure was constructed to facilitate containment access during the steam generator replacement project and no longer serves a direct plant operational or access function. It remains for storage as part of the radiological maintenance area. This structure has no effect on safety-related structures or components and is not used to support any of the regulated events, thus performing no intended functions for license renewal.**

- **<u>Plant Lighting Masts</u> - There are seven (7) NNS high mast light pole structures located around the plant site which primarily function as security lighting, with their individual failure having no effect on safety related structures or components. They are not used to support accident conditions or any of the regulated events and thus perform no intended functions for license renewal. In addition to these high mast light poles, exterior lighting also consists of standard height light poles and wall mounted lights along the perimeter of each structure within the protected area of the plant. All of the exterior lighting is supplied by 480 volt, single-phase power from the nearest available 480 volt load center. Since none of the exterior lights are credited for accident or regulated event mitigation, they are not considered within the scope of license renewal.**

- **<u>Radiological Maintenance Building (RMB)</u> - The RMB is a NNS structure, which is located adjacent to the Hot Machine Shop. This structure serves only as a maintenance facility for contaminated components and tools. This structure has no effect on safety-related structures or components and is not used to support any of the regulated events, thus performing no intended functions for license renewal.**

Section 2.3.3.1 Air Handling And Local Ventilation And Cooling Systems

<u>RAI 2.3.3.1-1:</u> Ventilation damper housings are highlighted on the ventilation flow diagrams identified in the LRA as within the 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 LRA that relates ventilation system component types subject to an aging management review (AMR) and their intended functions. Examples of the ventilation damper housings highlighted on the 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 flow 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 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. If the components are not in scope or not subject to an AMR, provide justification for their exclusion.

VCSNS Response RAI 2.3.3.1-1

Damper housings are considered subject to an AMR if they are in an in-scope (pressure boundary) portion of a system. These damper housings are highlighted on the mechanical scoping drawings and are included with ductwork in the application. Table 3.3-2 Item 2 for stainless steel and Item 3 for carbon and galvanized steel provides a more complete description.

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

- Ductwork turning vanes
- Ventilation system elastomer seals
- 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.

VCSNS Response RAI 2.3.3.1-2

Ductwork turning vanes are not specifically called out as a commodity. Turning vanes are a subcomponent of the ductwork and are made of the same material as the ductwork (galvanized steel or stainless steel) and is bounded by the AMP for the ductwork.

Flexible seals between duct and housings are in scope and included in Table 2.3-18. Door seals are considered a consumable subcomponent and are not credited for the license renewal intended of pressure boundary and are not subject to Aging Management Review.

Ductwork flexible connections, they are in scope and included on Table 2.3-18. VCSNS does not mount in scope components with "vibration isolator flexible connections."

Ductwork test connections and ductwork access doors are considered part of the ductwork and are included in that component grouping. Ductwork test connections are typically holes that are normally filled with a "push penny." The push penny is not required for the system to meet its license renewal intended function.

RAI 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 AMR. 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.

VCSNS Response RAI 2.3.3.1-3

VCSNS does not specifically use the terminology "structural sealants" as identified in this RAI and NUREG-1800. However, in the context of NUREG-1800, structural sealants at VCSNS would include: fire seals and coatings, pressure seals, expansion joints, etc., all of which are addressed within the Application. VCSNS recognizes that locations exist where these materials or component types are important in maintaining the integrity of the component to which they are connected. For these situations, the license renewal or component intended function supported by the sealant is to maintain the building pressure boundary envelope. The pressure boundary function is addressed by surveillance testing to demonstrate compliance with Technical Specifications.

Additionally, these materials (expansion joints, caulking, seals, etc.) are considered to be consumables. Various inspection programs as described in the Application (B.1.5, B.1.11, B.1.12, B.1.16, B.1.18, B.1.20) are used to determine their replacement. The life of these materials is based on identification of wear or damage during these inspections. Programmatic actions are not to manage their life, but rather to replace them when their condition indicates that they are no longer acceptable for service. Therefore, these materials are replaced based on condition monitoring and are not subject to aging management review.

RAI 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 results table. If filter housings are not subject to an AMR, provide justification for their exclusion.

VCSNS Response RAI 2.3.3.1-4

Filter housings are considered subject to an AMR if they are in an in-scope (pressure boundary) portion of a system. The filter housing (with a license renewal intended function) are highlighted on the mechanical scoping drawings and are included in Table 2.3-18 (Ductwork, Fan and Plenum Housings.) More discussion on this subject is found in Section 2.1.2.1.4 of the Application.

<u>RAI 2.3.3.1-5:</u> 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 safe shutdown controls are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4(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.

VCSNS Response RAI 2.3.3.1-5

The "safe shutdown controls and panels" at VCSNS are the Control Room Evacuation Panels (CREP). The CREP panels are located in the speed switch room area of the plant and the cooling for this area is shown on drawing D-912-157. The speed switch room area cooling is within the scope of license renewal and subject to aging management review.

Alternate safe shutdown (requiring control room evacuation) is achieved using the train "B" equipment and controls by a variety of means including; controls at the CREP (VCSNS Name for Alternate or Remote Safe Shutdown Panels), controls at switchgear and motor control centers, and controls mounted on the local panels for the "B" diesel generator and "B" water chiller".

<u>RAI 2.3.3.1-6:</u> The air handling and local ventilation and cooling systems 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.

VCSNS Response RAI 2.3.3.1-6

Instruments and instrument tubing considered in-scope are highlighted on the applicable drawings. The instrument tubing is listed in Application Table 2.3-18. The instruments are active components and are not listed as subject to aging management review. More discussion on instruments can be found in Application Section 2.1.2.1.4.

RAI 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 the 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 justification for their exclusion.

VCSNS Response RAI 2.3.3.1-7

The VCSNS position on consumables is consistent with NUREG-1800 Table 2.1-3.

a) **Packing, gaskets, component seals, and O-rings are subcomponents and are excluded for several reasons. ASME code pressure does not rely on gasket, packing and O-rings as pressure boundary components and they are therefore excluded from aging management review. Seal material on components, such as doors, is either not credited for system intended function or it is replaced on condition based on testing results and is not subject to aging management review.**

Seals and O-rings for structural components are not treated individually as consumables, but rather as parts of their host components (doors, airlocks, hatches, etc.) which are managed under Aging Management Programs and plant procedures.

b) **Structural sealants: VCSNS does not specifically use the terminology "structural sealants" as identified in this RAI and NUREG-1800. However, in the context of NUREG-1800, structural sealants at VCSNS would include: fire seals and coatings, pressure seals, expansion joints, etc., all of which are addressed within the Application. VCSNS recognizes that locations exist where these materials or component types are important in maintaining the integrity of the component to which they are connected. For these situations, the license renewal or component intended function supported by the sealant is to maintain the building pressure boundary envelope. The pressure boundary function is addressed by surveillance testing to demonstrate compliance with Technical Specifications.**

Additionally, these materials (expansion joints, caulking, seals, etc.) are considered to be consumables. Various inspection programs as described in the Application (B.1.5, B.1.11, B.1.12, B.1.16, B.1.18, B.1.20) are used to determine their replacement. The life of these materials is based on identification of wear or damage during these inspections. Programmatic actions are not to manage their life, but rather to replace them when their condition indicates that they are no longer acceptable for service. Therefore, these materials are replaced based on condition monitoring and are not subject to aging management review.

c) **Oil, grease, and component filters are short lived with periodic replacement and are excluded from aging management review.**

d) **System filters, fire extinguishers, fire hoses, and air packs, are discuss in Section 2.1.2.1.4 of the Application.**

RAI 2.3.3.1-8: Fuel handling building charcoal exhaust system and air supply distribution ductwork are not highlighted on the ventilation flow diagram identified in the license renewal application as within the scope of license renewal (D-912-131, zones A-1 thru A-5). State whether this exhaust ductwork is within the scope of license renewal and subject to an AMR. If so, provide the relevant information about the exhaust ductwork in order to provide the staff with ability to coordinate between the component/commodity tables and the referenced flow diagram drawing and complete the aging management review Table 2.3-18 of the LRA. If the exhaust ductwork is not in scope or subject to an AMR, provide justification for their exclusion.

VCSNS Response RAI 2.3.3.1-8

The Fuel Handling Building exhaust system ductwork down stream of the fans does not need to remain intact for the system to perform its intended function. The system function is to maintain ability to provide negative pressure as well as removal of airborne particulate and radioiodines within the Fuel Handling Building during fuel handling activities and blackout conditions within acceptable limits. The pressure boundary of ductwork downstream of the fans is located in the Auxiliary Building and not needed to meet this function. Some of this ductwork is included in scope for special interaction concerns and was included in the Supplement to the Application (Reference 6).

Section 2.3.3.6 Component Cooling Water

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

venturi	Drawing D-302-612, locations D4, D5, D6, and D7
radiation monitor housing	Drawing D-806-005, locations A5 and A8

However, the staff is unable to locate these component types in LRA Table 2.3-22. 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)(1) and 10 CFR 54.21(a)(1).

VCSNS Response RAI 2.3.3.6-1

1) **The flow elements shown on drawing D-302-612 are venturis. The venturis are listed as "Orifices" and are included in Table 2.3-22 of the Application.**

2) **Radiation monitors are contained in Section 2.3.3.17 of the Application and the instrument pressure boundary component is a listed component type. Drawing D-806-005 is listed in Section 2.3.3.6 of the Application for a small spool of piping that is included in the CC system. Drawing D-806-005 should have also been listed in section 2.3.3.17 for the highlighted valves, tubing and instruments included in the RM system. Section 2.3.3.17 of the Application does contain the pressure boundary portions of the radiation monitors.**

RAI 2.3.3.6-2: The license renewal boundary drawings referenced in LRA Section 2.3.3.6 show numerous lines connecting temperature elements or temperature indicators to piping segments that are within the scope of license renewal. However, although they often include dimensional markings indicating they represent piping stubs, these connecting lines are not identified as being within the license renewal scope. 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).

VCSNS Response RAI 2.3.3.6-2

Thermowells are used in temperature monitoring and are included in the scope of license renewal. They are listed in Table 2.3-22 of the Application. The lines shown on the drawings are not piping; they indicate the relative location of the thermowell or sensing location. The RTDs are depicted as TE on these drawings. Temperature indicators (depicted as TI on the drawings) may be local (in a thermowell) or control board mounted. (Thermowells without a device of some kind in them are shown as TW. Thermo wells with a device in them will not have TW.) The drawings were used for mechanical scoping and electrical signals were not highlighted.

Section 2.3.3.7 Diesel Generators Services Systems

RAI 2.3.3.7-1: With regard to diesel generator (DG) fuel oil storage and transfer system, the following components are neither identified in LRA drawing D-302-351 as being within the scope of license renewal nor included in LRA Table 2.3-23 as being subject to an AMR:

- the vent line with flame arrestor for each fuel oil storage tank and each day tank,
- the manway for each fuel oil storage tank,
- the fuel oil fill lines.

The staff believes that these components are the long-lived components with a passive function as described in 10 CFR 54.4 and, therefore, should be subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

VCSNS Response RAI 2.3.3.7-1

These components do not have a license renewal intended function. They are not required to contain the diesel fuel oil in the system as they are located above the minimum fuel oil level in the storage tanks. Their failure would not prohibit the diesel fuel from being supplied with the required amount of fuel nor would it allow a loss of fuel oil inventory.

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 LRA drawings (D-302-353 and IMS-32-005, sht. 7) as being within the scope of license renewal nor included in LRA Table 2.3-23 for an AMR. The staff believes that these components are the long-lived components with a passive function as described in 10 CFR 54.4, and therefore, should be subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

VCSNS Response RAI 2.3.3.7-2

These components do not have a license renewal intended function. Crankcase vacuum is not required for the diesel to operate. Crankcase vacuum is required for pollution control. The failure of the non-highlighted components would not prohibit the diesel from operating and meeting its license renewal intended function of supplying electric power.

RAI 2.3.3.7-3: The components (e.g., expansion tanks, sight glasses, flex connectors, valves, piping, etc.) of DG jacket water system are neither identified in LRA drawings (D-302-353 and IMS-32-005, sht. 4) as being within the scope of license renewal nor included in LRA Table 2.3-23 for an AMR. The staff believes that these components are the long-lived components with a passive function as described in 10 CFR 54.4, and therefore, should be subject to an AMR. Please clarify whether these components are subject to an AMR or justify their exclusion.

VCSNS Response RAI 2.3.3.7-3

These components do not have a license renewal intended function. The 12" standpipe meets the volume requirements for containing the jacket cooling water. The jacket cooling system is maintained in a hot condition. The NNS expansion tank and components are provided for extra surge capacity and static head to ensure proper filling after maintenance. The tank is restrained such that it cannot fall and impact the D/G or any of its required auxiliaries. The failure of these components would not prohibit the diesel from being supplied with the required amount of coolant to meet its license renewal intended function.

Section 2.3.3.8 Fire Service System

RAI 2.3.3.8-1: The license renewal boundary drawings referenced in Section 2.3.3.8 did not identify the following fire protection (FP) systems and components as being within the scope of license renewal and subject to an AMR. The staff believes that the FP systems and components described below are passive and long-lived and perform a function that demonstrates compliance with 10 CFR 50.48 for fire protection. Provide basis for excluding the following FP systems and components from the scope of license renewal and subject to an AMR:

(1) LRA Drawing D-302-231, sht. 1 - Fire Service - pumps

The FP piping leading to the Alt. F.S. pump house, turbine building, a portion of the CW pump house, and the FP components (including Jockey Pumps, valves, piping, fittings, and diesel fuel tanks) are not highlighted in the system flow diagram (D-302-231, sht. 1) as components within the scope of license renewal and subject to an AMR. The staff believes that these FP components perform a pressure boundary intended function consistent with rest of the FP system in scope. Clarify whether the FP piping and components should be in scope or justify their exclusion.

(2) LRA Drawing D-302-231, sht. 2 - Fire Service - hydrants and loops

The fire hydrants (at locations H12, K8, K9, K10, K11, and K12) are not highlighted in the system flow diagram (D-302-231, sht 2) as components within the scope of license renewal and subject to an AMR. The staff believes that these components have the FP intended functions required to be compliance with 10 CFR 50.48 as stated in 10 CFR 54.4. The fire hydrants also serve as the pressure boundary for the FP water supply system. LRA Section 2.1.1.4.1, "Fire Protection," states that the plant's fire protection program meets the guidance of Appendix A to BTP 9.5-1. For the fire hydrants, Appendix A to BTP 9.5-1, Section E.2.g, states that outside manual hose installation should be sufficient to reach any location with an effective hose stream. To accomplish this, hydrants should be installed approximately every 250 feet on the yard main system. In addition, hydrants are the integral components for performing system flow tests. Lack of maintenance on fire hydrants over time can result in partially close or shutting valves and clogging hydrants with debris, all of these will affect system flow. Furthermore, fire hydrants are subject to an AMR in accordance with 10 CFR 54.21, because they are the passive and long-lived components. Clarify whether the fire hydrants should be in scope or justify their exclusion.

(3) LRA Drawing D-302-231, sht. 3 - Fire Service - reactor building, intermediate building, diesel generator building, fuel handling building, and control building.

The FP piping, fittings, and valves in the reactor building (at locations E5, E7, and E8), fire hose connections in the fuel handling building (at location B4), fire hose connection in the auxiliary building (at location B13), fire hose connection in intermediate building (at location H4), and fire hose connections in reactor building (at location E9) are not highlighted in the system flow diagram (D-302-231, sht. 3) as components within the scope of license renewal and subject to an AMR. However, the FP components perform a pressure boundary intended function with rest of the FP water supply system. Clarify whether the FP piping, fittings, and valves and fire hose connections in the reactor building, intermediate building, fuel handling building, and auxiliary building at these locations should be in scope or justify their exclusion.

(4) LRA Drawing D-302-231, sht. 4 - Fire Service - turbine building and water treatment building

The FP piping, fittings, valves, and fire hose connections in the turbine building (at locations D6, E6, E7, E8, E9, E10, F7, F8, F9, and F10) are not highlighted in the system flow diagram drawing (D-302-231, sht. 4) as components within the scope of license renewal and subject to an AMR. However, these FP components perform a pressure boundary intended function with rest of the FP water supply system in the turbine building that is in scope. Clarify whether the FP piping, fittings, and valves in the turbine building at these locations should be in scope or justify their exclusion.

(5) LRA Drawing D-302-231, sht. 5 - Valve Manifolds

The FP piping, fittings, and valves in turbine building south, El 412' (at locations J6 to J9) are not highlighted in system flow diagram (D-302-231, sht. 5) as components within the scope of license renewal and subject to an AMR. However, these FP components at level EL. 412' of turbine building south area perform a pressure boundary intended function with rest of the FP water supply system that is in scope. Clarify whether the FP piping, fittings, and valves in turbine building south area at El 412' should be in scope or justify their exclusion.

(6) LRA Drawing D-302-232 - Fire Service - halon and low pressure CO_2

As shown in LRA drawing D-302-232, the computer BOP relay and technical support center (TSC) equipment rooms are protected by a total flooding carbon dioxide (CO_2) fire extinguishment system utilizing 5-ton low pressure CO_2 storage tanks as the supply source. However, the CO_2 system electric control panels and the IF&S system are not highlighted in the system flow diagram as components within the scope of license renewal and subject to an AMR. These components perform a pressure boundary intended function for the total flooding CO_2 fire extinguishment system in the computer BOP relay and TSC equipment rooms. Clarify whether these components should be in scope or justify their exclusion.

(7) LRA Drawing IMS-55-059, sht. 1 - Deluge Water Spray Systems - deluge valve station in turbine building

The FP components for the valve station system in the turbine building are not highlighted in the system flow diagram (IMS-55-059, sht. 1) as components within the scope of license renewal and subject to an AMR. These FP components perform a pressure boundary intended function with rest of the FP system that is in scope and subject to an AMR for license renewal. Clarify whether these components should be in scope or justify their exclusion.

(8) LRA Drawing IMS-55-085, sht. 26 - diesel fire pump room in diesel generator building

The fire suppression system (including sprinklers with heat collectors) are installed in the diesel fire pump room of the diesel generator building. The fire suppression system is not highlighted in the system flow diagram (IMS-55-085, sht. 26) as components within the scope of license renewal and subject to an AMR. However, the fire suppression system and its components perform a pressure boundary intended function with rest of the FP system that is in scope and subject to an AMR for license renewal. Clarify whether the fire suppression system and its components should be in scope or justify their exclusion.

(9) LRA Drawing IMS-55-085-27-2 - Charcoal Filter Plenum Systems in auxiliary building, control building, and reactor building

The manual deluge sprinkler system for the charcoal filter plenums (XAA-40A-AH and XAA-40B-AH) in the auxiliary building are not highlighted in the system flow diagram (IMS-55-085-27-2) as components within the scope of license renewal and subject to an AMR. This manual deluge system provide a pressure boundary intended function for the charcoal filter plenums in the auxiliary building. Clarify whether the deluge sprinkler system and its components should be in scope or justify their exclusion.

(10) LRA Section 2.3.3.8 and the flow diagram drawings referenced in the LRA do not identify the pre-action sprinkler system installed in the diesel generators room as systems within

the scope of license renewal and subject to an AMR. Section 5.0 (F) (9), "Diesel Generator Area" (Item "d" on page 5.0-40) of the VCSNS FPER (Amendment 02-01), states that the emergency diesel generators are protected by a pre-action sprinkler system. Since this system performs a pressure boundary intended function for the FP water supply system in the emergency diesel generator room, the pre-action sprinkler system and its components should be within the scope of license renewal. Provide a technical justification for their exclusion.

VCSNS Response RAI 2.3.3.8-1

Appendix A to BTP 9.5-1 included ten specific quality assurance criteria (Section C of Appendix A). VCSNS provided a detailed comparison of the provisions at VCSNS with these criteria in the FPER, Section 5 Point-by-Point Comparison with Appendix A. The FPER addresses Section C (Quality Assurance) of Appendix A by referring to responses to the NRC questions 421.77 and 421.78 in initial licensing of VCSNS. The VCSNS response to 421.77 states in part "although the fire protection equipment is not considered safety related, the QA Program for fire protection is part of the overall program, and installation, testing, and subsequent operations for areas containing safety related equipment are processed by procedures similar to those utilized for safety related work." These components serve to ensure the capability to shutdown the reactor and maintain it in a safe shutdown condition and to minimize radioactive releases to the environment in the event of a fire. These Quality Related (QR) components are identified by "QR" code flags on flow diagrams or are listed in the VC Summer CHAMPS Program by tag number and identified by a "QR" in the safety class designation column.

The VCSNS Technical Specifications originally contained Fire Protection Program commitments and reporting requirements. Amendment 79 to the VCSNS Technical Specifications removed fire protection elements from the Technical Specifications. Consistent with the recommendations of Generic Letter 86-10, Amendment 79 resulted in the transfer of Fire Protection Program commitments and reporting requirements from the Technical Specifications to plant procedures. This change resulted in the VCSNS Fire Protection Program being completely described and controlled through the combination of the FSAR/FPER and plant procedures rather than through the combination of FSAR/FPER and Technical Specifications. This transfer maintained the program requirements in appropriate plant procedures and provided flexibility by allowing modifications to the program under the provisions of 10CFR50.59.

The portions of the FS system outside of the QR flags are isolatable by manual valves. If required to be isolated, these portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements, only insurance carrier requirements, for their maintenance. Should degradation occur in these portions to the extent that a leak occurs such that pressure is reduced beyond the capacity of the components used for FS pressure maintenance, the FS pumps will start, thereby setting off an alarm in the Control Room. The capacity of each FS pump is 2500 gpm at 125 psig. The electric FS pump starts at 95 psig decreasing. Should FS system pressure continue to decrease, the diesel FS pump starts in parallel to the electric FS pump at 85 psig. Upon receipt of the alarm in the control room, Auxiliary Operators

would be sent to find and isolate the leak. Operability of the FS system required for Appendix A, BTP 9.5-1 is thereby maintained.

(1) The Alternate Fire Service Pumps and all associated valves, piping, fittings and the diesel fuel tanks for the Alternate Diesel FS Pumps are not credited for fire protection and therefore should not be in scope for license renewal. These pumps were installed for fire service needs during the construction of the station but are no longer used. They are isolated from the FS system by two closed manual isolation valves. They are, however, maintained by the Preventive Maintenance Program so that they could be available in an emergency.

The piping leading to the Turbine Building shown on drawing D-302-231, Sheet 1 should not be in scope for license renewal. These 1-1/2" diameter lines provide FS system pressure maintenance directly from the Filtered Water (FI) system or from the FS Jockey pump (also supplied from the FI system). Both of these FS pressure maintenance supply lines contain backflow preventers, as mandated by SC DHEC to prevent raw water from entering into a potable water supply. Loss of pressure boundary upstream of these pressure boundaries, including the FS Jockey pump casing, would only affect the FI system and would not impact the FS system. A leak in the portion of the 1-1/2" piping that lies between these preventers and the highlighted QR boundary valve can be readily isolated and would not challenge the capacity of the FS pumps to provide adequate pressure and flow to the QR portions of the FS system.

The portions of FS piping in the Circulating Water (CW) Pumphouse that are not highlighted on drawing D-302-231, Sheet 1 should not be in scope for license renewal. The test header piping for the FS pumps is isolated from the FS system by closed manual isolation valves. The 1-1/2" pump casing drains and vacuum breakers would not challenge the capacity of the FS pumps to provide adequate pressure and flow to the QR portions of the FS system.

As a result of discussions with the Staff on this issue, VCSNS will expand the scope for license renewal to include the Fire Service Jockey pump and associated piping and components. Associated piping and components are bounded by XVT-6955-FS and XVM-6956-FS. Components added by this expansion of scope are subject to screening. If screened in, the Fire Protection Program will manage aging of these components.

(2) The fire hydrants in question are not within scope and should not be highlighted on drawing D-302-231, Sheet 2. The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The portions of the FS system outside of the QR flags are isolatable by manual valves. These portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements for their maintenance, only insurance carrier requirements. The fire hydrants in question are associated with Fire Hose Houses 8, 9, 10, 16, 17, 18, 19, and 20. All of these hose houses

are located outside of the protected area. The fire hydrants in question are "non-FPER" per present fire protection plant procedures. The requirements contained in the present plant procedures concerning fire hydrants have not changed since these requirements were transferred from the VCSNS Technical Specifications.

(3) The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The portions of the FS system outside of the QR flags are isolatable by manual valves. These portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements for their maintenance, only insurance carrier requirements. The portion of piping in question in the reactor building (locations E5, E7, and E8 on D-302-231 Sheet 3) is normally isolated per Criterion 56. The highlighted portion of this piping is in scope for containment isolation only. The unhighlighted fire hose stations at locations B4, B13, H4, and E9 of D-302-231 Sheet 3 provide fire protection to areas not containing safety related equipment.

The hose stations in question are "non-FPER" per present fire protection plant procedures because they provide protection for areas not containing safety related equipment. The requirements contained in the present plant procedures concerning these hose stations have not changed since these requirements were transferred from the VCSNS Technical Specifications.

As a result of discussions with the Staff on this issue, VCSNS will expand the scope for license renewal to include fire hose connections identified by the Staff on drawing D-302-231, Sheet 3 in the Fuel Handling Building (at location B4), in the Auxiliary Building (at locations B13 and E9), and in the Intermediate Building (at location H4). Components added by this expansion of scope are subject to screening. If screened in, the Fire Protection Program will manage aging of these components.

(4) The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The FS system components not highlighted on drawing D-302-231 sheet 4 are not in scope for license renewal, because they provide suppression capability only for areas not containing safety related equipment. The Turbine Building is separated from areas containing safety related equipment by three-hour rated fire barriers. The hose stations in question are "non-FPER" per present fire protection plant procedures. The requirements contained in the present plant procedures concerning these hose stations have not changed since these requirements were transferred from the VCSNS Technical Specifications. The portions of the FS system outside of the QR flags are isolatable by manual valves. These portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements, only insurance carrier requirements, for their maintenance.

As a result of discussions with the Staff on this issue, VCSNS will expand the scope for license renewal to include fire hose stations identified by the Staff on drawing D-302-231, Sheet 4 in the turbine building (at locations D6, E6, E7, E9, D10, F7, F8, F9, and F10). Components added by this expansion of scope are subject to screening. If screened in, the Fire Protection Program will manage aging of these components.

(5) The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The FS system components not highlighted on drawing D-302-231 sheet 5 are not in scope for license renewal, because they provide suppression capability only for areas not containing safety related equipment. The deluge systems in question are "non-FPER" per present fire protection procedures. There were no requirements for these systems in the Technical Specifications when the Fire Protection Program requirements were transferred from the Technical Specifications.

As a result of discussions with the Staff on this issue, VCSNS will expand the scope for license renewal to include the valve manifolds identified by the Staff on drawing D-301-231, Sheet 5 in the turbine building (at locations J6 to J9). C Components added by this expansion of scope are subject to screening. If screened in, the Fire Protection Program will manage aging of these components.

(6) The CO_2 system electric control panels and the IF&S system are not within scope for mechanical components in license renewal and should not be highlighted on this drawing. The dashed lines on this drawing indicate electrical signals only. The only mechanical components in these panels are the ¼" CO_2 sensing lines feeding into the electric control panels that give indication that the system has been charged. These lines terminate in a sealed pneumatic actuation assembly that is an active component. There are no mechanical components on this drawing that interface with the IF&S system. Electrical and instrument panel enclosures are covered in the VCSNS Application in Table 2.4.4 of the Structures and Structural Components Scoping and Screening Results.

(7) Location D-13 on drawing D-302-231, Sheet 5, shows that the in scope Fire Service piping to the Control Building charcoal plenums is continued on drawing 1MS-55-059-1. This drawing is provided, but does not show the actual continuation of the FS system beyond the deluge valves. It only shows the deluge valves again, which is the only thing highlighted on the drawing.

The other components on 1MS-55-059-1 are not in scope for license renewal. The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The FS system components not highlighted on drawing 1MS-55-059, sheet 1 are not in scope for license renewal, because they provide suppression capability only for areas not containing safety related equipment. The deluge systems in question are "non-FPER" per present fire protection

procedures. There were no requirements for these systems in the Technical Specifications when the Fire Protection Program requirements were transferred from the Technical Specifications. The portions of the FS system outside of the QR flags are isolatable by manual valves. These portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements, only insurance carrier requirements, for their maintenance.

As a result of discussions with the Staff on this issue, VCSNS will expand the scope for license renewal to include the valve manifolds identified by the Staff on drawing 1MS-55-059, Sheet 1 in the Turbine Building. Components added by this expansion of scope are subject to screening. If screened in, the Fire Protection Program will manage aging of these components.

(8) Although the suppression system for the Diesel Fire Pump Room is shown on the same drawing with the suppression system for the Diesel Generator Building, the Diesel Fire Pump Room is not in the Diesel Generator Building. The fire suppression system for the Diesel Generator Building on this drawing (1MS-55-085, Sheet 26) should be highlighted as in scope. The components in this system are subject to an AMR and the aging of the components will be managed by the Fire Protection Program. The fire suppression system for the Diesel Fire Pump Room is in scope and is highlighted on the drawing.

There is no specification or requirement for heat collectors in the NFPA code. Their installation was originally considered to be a good practice. These devices are installed on sprinkler heads in areas where the air flow may delay the buildup of heat necessary to actuate spray. The material is sheet metal. Because there is no requirement or specification for these devices, they should not be included in scope for license renewal. However, these devices are inspected when the associated sprinkler head is inspected per the Fire Protection Program.

(9) The only filter systems that have FS piping in scope beyond the manual deluge valve are the Emergency Safeguards Feature (ESF) filter systems: the Control Room Emergency Filter Plenums and the FHB Charcoal Exhaust system. The other charcoal filter plenums are not in scope. The mechanical components of the FS system needed for compliance to Appendix A to BTP-9.5-1 are located within the QR code flags on the highlighted drawings, D-302-231, Sheets 1 - 5. The portions of the FS system outside of the QR flags are isolatable by manual valves. These portions can remain isolated indefinitely without affecting the requirements of Appendix A to BTP 9.5-1. These portions are maintained by the Fire Protection Program, however there are no Appendix A, BTP 9.5-1 requirements, only insurance carrier requirements, for their maintenance. Also, a manual deluge valve for each filter system normally provides pressure boundary isolation from the rest of the FS system.

(10) The fire suppression system for the Diesel Generator Building on drawing 1MS-55-085, Sheet 26 should be highlighted as in scope. The system is listed as an

"FPER" system by the plant procedures that control the requirements for the Fire Protection Program. The components in this system are subject to an AMR and are encompassed by the component types listed in Table 2.3-24 of the LRA.

Section 2.3.3.9 Fuel Handling System

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, LRA Table 2.3-25 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 LRA Table 2.3-25 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).

VCSNS Response RAI 2.3.3.9-1

Section 2.3.3.9 of the Application is Fuel Handling. The mechanical component in the FH system that is in scope is the transfer tube and it is shown on Drawing D-302-651. The Fuel Transfer Tube [XNF0009-FH], including pipe, blind flange and slip-on flange is installed inside a penetration sleeve. The Fuel Transfer Tube is welded to the penetration sleeve which connects the Fuel Transfer Canal in the sheltered environment of the Fuel Handling Building to the Refueling Cavity inside the Reactor Building. The penetration sleeve itself is a civil/structural commodity. This drawing (D-302-651) shows the Spent Fuel Cooling System (SF) that is described in Section 2.3.3.22 of the Application. FH only included the tube proper and the flange located in the Reactor Building. The associated gate valve (XVM-06737-SF), the test valve (XVG-06657-SF), and test valve pipe are included in the SF section of the Application (Section 2.3.3.22). Bolting is not considered a separate component at VCSNS, but is subject to inspections required by ASME code.

Section 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 continuing to drawing E-302-743. However, drawing E-302-743 is not included in the submittal nor referenced in Section 2.3.3.10 of the LRA. Explain whether the license renewal boundary of gaseous waste processing system extends to drawing E-302-743. Please supply this drawing.

VCSNS Response RAI 2.3.3.10-1

Drawing E-302-743 was not supplied with the Application. This drawing can be found in the FSAR as Figure 11.3-4 sheet 2. If it were supplied as a markup, it would be the same as the A recombiner shown on Drawing E-302-742. Drawing E-302-744 does provide the detail of the A Catalytic Hydrogen Recombiner. Drawing E-302-745 provides the detail of the B Catalytic Hydrogen Recombiner.

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 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 AMR. Justify exclusion of these components from the scope of license renewal and aging management review.

VCSNS Response RAI 2.3.3.10-2

The piping up to and including valves 7938A and 7938B are in scope. This piping and valves are pressure boundary for the CC system. Drawings E-302-742, 743, 744 and 745 incorrectly show the safety class as "QRG" instead of "safety class 3".

Section 2.3.3.12 Instrument Air Supply System

RAI 2.3.3.12-1: FSAR Section 9.3.1.3 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, with the exception of the valve air operators, the applicant did not identify the accumulators and the related components necessary for operation of the feedwater isolation valves among the components identified in the drawings referenced in LRA Sections 2.3.3.12 and 2.3.4.5 as being within the scope of license renewal. The air operators for the feedwater isolation valves were identified as being within the scope of license renewal on license renewal drawing 1MS-25-898. Clarify whether the accumulators and the 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).

VCSNS Response RAI 2.3.3.12-1

Many components are typically supplied with various combinations of external-to-actuator air accumulator tanks, check valves, pressure regulators and solenoid valves, which may not be IA System designated, but were evaluated with the IA System. These components are identified by the referenced drawings in Section 2.3.3.12 of the Application. These components are included in the various line items of the Application Table 2.3-27. The accumulator for the Feedwater Isolation Valve is incorporated into the design of the valve actuator. The "accumulator" is included as a tank. Actuators are

considered "active" components and are screened out as "not subject to Aging management review."

RAI 2.3.3.12-2: The license renewal drawings referenced in LRA Section 2.3.3.12 identified the air accumulators and their associated components for the following valves and dampers as components 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 actuator housings associated with the above dampers and valves were not included in the scope of license renewal. For the listed dampers and valves, clarify whether the portions of the associated actuator housings 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).

VCSNS Response RAI 2.3.3.12-2

Valve actuators are considered "active" components and are screened out as "not subject to Aging management review" based on 10 CFR 54.21(a)(1)(i).

The internal environment of air actuators is dried air. There are no aging effects that require management for these carbon steel actuators in dry air.

Most of the actuators are in areas where leaking boric acid is not credible. Those in areas where leaking boric acid may be found would be subject to Boric Acid Corrosion Surveillances (B.1.2). These surveillances are performed to monitor the effect of leaking acid not specific components.

The external environment of the actuators is considered moist air. These actuators are located in a sheltered (i.e. indoors, non-condensing) environment. In this type of environment pitting and crevice corrosion are not considered aging effects that require management. The actuator will remain dry and even if some general corrosion is experienced it would not be severe enough to challenge the actuators ability to perform its intended function.

RAI 2.3.3.12-3: FSAR Section 9.2.1.2 states that the fire protection system serves as a standby means of cooling the diesel generators. When the diesel generator is operating in the emergency mode, the cross-connect valve automatically opens on high lube oil temperature or high jacket water temperature. FSAR Section 9.3.1.3 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.

VCSNS Response RAI 2.3.3.12-3

These accumulators and associated components for these valves (XVG-03105A and B-SW) are shown on Drawing B-817-048. They are highlighted and in scope. Accumulators are listed as tanks in Application Table 2.3-27.

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

VCSNS Response RAI 2.3.3.14-1

Besides components for the Reactor Coolant Drain Tank, components for the Waste Evaporator are also included for this section. The components for the Waste Evaporator can be found on the Drawing 1MS-09-238 that was provided as a mechanical scoping drawing. Drawing 1MS-09-238 should have been listed in Application Section 2.3.3.14. The Waste Evaporator is shown on Drawing E-302-736, which is FSAR Figure 11.2-2, sheet 2. This drawing was not provided as a mechanical scoping drawing as no components shown are in scope.

Section 2.3.3.17 Radiation Monitoring System

RAI 2.3.3.17-1: LRA Section 2.3.3.17 states that the mechanical license renewal functions of the radiation monitoring system are to provide post accident monitoring capability for the containment activities and maintain system boundaries with the component cooling, spent fuel cooling, and chemical and volume control systems. However, in system flow diagram drawings D-302-611 (component cooling), D-302-651 (spent fuel cooling), and D-302-771 (nuclear sampling), the license renewal boundaries of the radiation monitoring system are not defined in these drawings. Also, Section 2.3.3.17 does not provide adequate information for the system components related to these drawings. Please define the license renewal boundaries for the system in these system flow diagram drawings.

VCSNS Response RAI 2.3.3.17-1

The only license renewal intended functions for the Liquid Radiation Monitors shown on Drawings D-302-611, D-302-651, and D-302-771 are as pressure boundaries for the Component Cooling, Spent Fuel Cooling, and Nuclear Sampling systems. Drawing D-806-005 is a radiation monitoring system drawing that shows all of the components of the monitors for the Component Cooling, Spent Fuel Cooling, and Nuclear Sampling

systems. For Application Section 2.3.3.17, the Radiation Monitoring System Drawing D-806-005 should have been the reference for Liquid Monitors rather than Drawings D-302-611, D-302-651, and D-302-771.

The Area Radiation Monitors shown on Drawings D-806-010 and D-806-011 provide required post accident containment monitoring capability and are environmentally qualified. These radiation monitors perform this function using an ion chamber probe inserted into the Reactor Building atmosphere; therefore, the intended function is being performed by instrumentation, not by mechanical components. Instrumentation is excluded from a mechanical aging review as outlined in 10CFR54.21(a)(1)(i) and (ii).

Section 2.3.3.18 Reactor Makeup Water Supply System

RAI 2.3.3.18-1: System flow diagram drawing D-302-791 (reactor makeup water supply system) highlights the flow restrictors (XPS-009-MU, XPS-158-MU, XPS-93A-MU, XPS-93B-MU) as components within the scope of license renewal. However, these passive and long-lived components are not included in LRA Table 2.3-32 as components subject to an AMR. These flow restrictors serve as a pressure boundary with the system piping in scope. Therefore, the flow restrictors should be subject to an AMR. Please justify their exclusion.

VCSNS Response RAI 2.3.3.18-1

These components are listed in Application Table 2.3-32 as orifices and are subject to an Aging management review.

Section 2.3.3.21 Service Water System

RAI 2.3.3.21-1: The license renewal boundary drawings referenced in LRA Section 2.3.3.21 show numerous lines connecting temperature elements or temperature indicators to piping segments that are within the scope of license renewal. However, although they often include dimensional markings indicating they represent piping stubs, these connecting lines are not identified as being in scope. Describe the typical configuration used to monitor flow stream temperature in the service water system using temperature elements or temperature indicators. 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).

VCSNS Response RAI 2.3.3.21-1

Thermowells are used in temperature monitoring and are included in scope. They are listed in Table 2.3.35 of the Application. The lines shown on the drawings are not piping, they indicate the relative location of the thermowell or sensing location. The RTDs are depicted as TE on the drawings. Temperature indicators (depicted as TI on the drawings) may be local (in a thermowell) or control board mounted. Thermowells without a device of some kind in them are shown as TW. Thermowells with a device in them will not have TW. The drawings were used for mechanical scoping and electrical signals were not highlighted.

RAI 2.3.3.21-2: License renewal boundary drawing (D-302-222) shows that the 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, LRA Section 2.3.3.21 fails to reference drawings D-302-085 and D-302-611 to include service water piping on these flow diagrams within the aging management programs identified for the service water system. LRA Tables 2.3-22 and 2.3-40, which present aging management results for the component cooling water and emergency feedwater systems respectively, do not reference aging management programs consistent with the 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.

VCSNS Response RAI 2.3.3.21-2

The EF piping from SW up to the Check valves XVC-01034A and B-EF and XVC-01022A and B-EF is included in the scope of the Service Water System Reliability and In-Service Testing Program.

The CC piping from SW up to the Check Valves XVC-09680A and B-CC is included in the scope of the Service Water System Reliability and In-Service Testing Program.

The SW piping starts at the first break down orifices XPS-0146A and XPS-147A and is included in the scope of the Service Water System Reliability and In-Service Testing Program. Upstream of the orifices XPS-0146A and XPS-147A is included in the scope of the Fire Protection Program.

Section 2.3.3.23 Thermal Regeneration System

RAI 2.2.3.23-1: LRA Section 2.3.3.23 states that the (boron) thermal regeneration system (BTRS) is used as the deborating demineralizer to reduce reactor coolant boron concentration towards the end of core life. LRA Table 2.3-37 lists heat exchangers (HX)-channel head, HX (shell), HX (tube), and HX (tube-sheet) as the component types of the thermal regeneration system subject to an AMR. System flow diagram drawing E-302-676, which contains the thermal regeneration system and portion of the chemical and volume control system, shows the letdown reheat HX, letdown chiller HX, and moderating HX in the boundary of the chemical and volume control system. The drawing does not show any heat exchangers in the boundary of the thermal regeneration system. LRA Table 2.3-8 lists HX-channel head, HX-shell, HX-tubes, and HX-tube sheet as the components of the chemical and volume control system subject to an AMR. Clarify whether the heat exchangers in LRA Table 2.3-37 for the thermal regeneration system are those shown in LRA Table 2.3-8 for the chemical and volume control system. If so, explain why the same heat exchangers should be listed in both tables.

VCSNS Response RAI 2.3.3.23-1

Drawings were highlighted during the screening process according to individual systems. Because there may be more than one system on a particular drawing, as in the case of Drawing E-302-676, the screening process resulted in multiple copies of a drawing showing highlighting for each system. These working copies are available on site for inspection. The drawings supplied to the NRC are composite drawings showing highlighting, in some instances, for multiple systems.

The Letdown Reheat, Letdown Chiller, and Moderating heat exchangers are BTRS components and are the heat exchanger components listed in Application Table 2.3-37. The only license renewal function for these components is pressure boundary for the Chemical and Volume Control (CS) system. The heat exchanger components listed in Table 2.3-8 are the CS heat exchangers: Regenerative, Excess Letdown, Seal water, and Letdown.

Section 2.3.4.1 Auxiliary Boiler Steam and Feedwater System

RAI 2.3.4.1-1: In LRA drawing D-302-051, the license renewal boundary of the auxiliary boiler steam piping terminates at valves PCV 337 (at locations G11) and PCV 316 (at locationJ8), upstream of the two evaporators. In the same drawing, the license renewal boundary terminates at mid-pipe of the 2 " line that supplies the preheater, downstream at valves TCV 325 (at location G10) and TCV 304 (at location G8). Typically, the license renewal boundary for the lines upstream of large components with a significant amount of stored energy terminates at the components that provide physical pressure boundaries per criterion 10 CFR 54.4(a)(2). Justify the differences in the license renewal boundary between the lines supplying steam to the preheaters and the lines supplying steam to the evaporators.

VCSNS Response RAI 2.3.4.1-1

Highlighting on Drawing D-302-051 partially obscures the license renewal boundaries. The license renewal boundaries terminate at the connections to the components, not at mid-pipe or at the valves. The portions of the system in scope for license renewal are found within the "QR" flags. Piping within the "QR" flags has been evaluated in the current licensing basis to maintain system pressure integrity to prevent adverse reactions with NSR equipment in certain areas of the Auxiliary Building for pipe rupture considerations.

Section 2.3.4.2 Condensate System

RAI 2.3.4.2-1: The condensate storage tank is the primary source of water for the emergency feedwater system. In LRA drawing D-302-101, it appears that the10" atmospheric vent pipe on the condensate storage tank (at location A11) provides vacuum protection for the tank. This vent pipe is not highlighted in the drawing as being in scope. The vent pipe has an intended function to protect the tank and should be included in the scope of license renewal. Justify its exclusion from being within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4(a)(1) and 10 CFR 54.21. Also explain why this 10" vent pipe is not shown on the condensate storage tank in LRA drawing D-302-085.

VCSNS Response RAI 2.3.4.2-1

The 10" vent does not provide a license renewal intended function (the vent performs its function by not being a pressure boundary). Plugging of this large vent is not a credible aging effect; however, the Inspections for Mechanical Components, by inspecting the exterior of the tank, will detect any degradation of this vent.

P&ID drawings (302 system flow diagrams) provide details with respect to piping and piping components, but not necessarily to other components. Details for other components are provided insomuch as they are pertinent to the system depicted on the drawing. For instance, Drawing D-302-101 (Condensate) shows condensate piping connections to the Condensate Storage Tank (CST), whereas Drawing D-302-085 (Emergency Feedwater) shows only Emergency Feedwater connections.

RAI 2.3.4.2-2: The condensate storage tank is a safety-related, safety class 2B component. It appears that the piping attached to the condensate storage tank should be within the scope of license renewal up to the first isolation valve to meet the requirements of 10 CFR 54.4(a)(2). However, in LRA drawing D-302-101, the piping is not shown to be within the license renewal boundary. Justify why the piping attached to the condensate storage tank is not considered to be in scope and subject to an AMR for license renewal.

VCSNS Response RAI 2.3.4.2-2

The Condensate Storage Tank (CST) is in scope for license renewal because one of its functions is to supply the Emergency Feedwater (EF) system. The CST is designed to have a reserve volume dedicated for use by the EF system. This volume is maintained by having the tank connections, except those required for the EF system, located above this volume. All connections above this volume are not in scope for license renewal. All connections below this volume are designated as EF components and are in scope for license renewal.

Section 2.3.4.3 Emergency Feedwater System

RAI 2.3.4.3-1: In LRA drawing D 302-085, the license renewal boundaries terminate at locked open valves, 1026 EF (at location G5), 1025A-EF (at location A5), and 1025B-EF (at location E5). It appears that the 2" and 3" lines that extend upstream of these valves should be within the scope of license renewal to meet the requirements of 10 CFR 54.4(a)(2). Please explain why these 2" and 3" lines, downstream of these valves, are not highlighted up to, and including, check valve 1027-EF (at location C4) as being within the scope of license renewal.

VCSNS Response RAI 2.3.4.3-1

Each pump has its own recirculation line with a breakdown orifice. Each breakdown orifice ensures that the pump can deliver required minimum flow for that pump with no

other flow path established. The breakdown orifices are also small enough to ensure flow to the Steam Generators without their isolation. After the breakdown orifice isolation valves, the line classification changes to non-safety related and ties together in a common line with a common check valve before flowing back to the Condensate Storage Tank above the dedicated inventory. Although unlikely, failure to establish recirculation flow (such as failure of the check valve to open) has been examined. This type of failure is mainly of concern when approaching Hot Shutdown (RHR conditions) when EF flow is being throttled back. In other words, failure of the recirculation line to allow flow will not affect the ability of EF to deliver 380 gpm to 2/3 SG's within one minute of initiation. This position was accepted as a part of the licensing basis of the plant by the NRC.

Loss of condensate quality water due to postulated breakage of the non-safety classed recirculation piping, downstream of the flow restriction orifices will not compromise safe shutdown based on the provision of two trains of Service Water as backup. The non-safety classed recirculation piping is low pressure moderate energy piping because the orifice reduces the pressure and it is open ended into the CST. Under the rules for postulated moderate energy piping cracks/leaks, cracks in the non-seismic connections are considered as single initiating events and do not need to be postulated concurrently with the effects of other initiating events.

The evaluation for "Criteria 2 Supplement To The Application" added portions of EW in scope because of potential special interactions with safety related components.

Section 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 housing of stop valve, S.V. # 1. This valve housing provides a pressure retaining function and is passive and long-lived. Therefore, this component appears to be within the scope of license renewal and subject to an AMR. Justify exclusion of the valve housing from the scope of license renewal and aging management review.

VCSNS Response RAI 2.3.4.6-1

The stop valves are in scope. They are shown on Drawing D-302-012 and included in Application Section 2.3.4.7 Main Steam System. (The configuration of the Drawing D-302-141 is different than other 302 drawings as it is based on a vendor drawing.)

Section 2.3.5 Criterion 2 Supplement to the License Renewal Application

RAI 2.3.5-1: The technical Report, "Criteria 2 Supplement to the Application for Renewed Operating License (RC-02-0159)," did not fully address non-fluid containing component groups (e.g., ventilation ducts, instrument air valves, valve actuators, etc.) that are spatially orientated near safety-related components. Although, LRA Section 2.3.3.1 has identified the components of air handling and local ventilation duck-work that perform intended safety functions in scope. However, certain non-fluid containing components may not have safety functions but have a spatial relationship with safety-related piping, such that there failure could adversely impact the performance of an intended safety function. Explain whether any components of these groups should be identified and treated as seismic II/I components.

VCSNS Response RAI 2.3.5-1

Piping and piping system components, ventilation ductwork, and piping and component insulation were specifically included in the supplement to the application. The review conducted for the supplement included all system piping and ductwork regardless of the internal environment, i.e. steam, treated water, raw water, gasses, air, etc. Piping and piping system components do include valves, fittings and various piping components located in the seismic portion of the piping. Ventilation ductwork includes damper housing when contained in the seismic portions of the system. Piping and component insulation was included as the portions or sections of insulation may support other sections.

The following information is from Christopher I. Grimes letter dated March 15, 2002 to Mr. Alan Nelson and Mr. David Lochbaum subject; "License Renewal Issue: Guidance On The Identification And Treatment Of Structures, Systems, And Components Which Meet 10 CFR 54.4(a)(2)" (Reference 7):

> ***"For non safety-related SSCs which are not connected to safety-related piping or components or are beyond the first seismic anchor past the safety/non-safety interface, but have a spatial relationship such that their failure could adversely impact on the performance of a safety-related SSC's intended function, the applicant has two options when performing its scoping evaluation; a mitigative option or a preventive option. With the mitigative option, the applicant should demonstrate that plant mitigative features (e.g., pipe whip restraints, jet impingement shields, spray and drip shields, seismic supports, flood barriers) are provided which protect safety-related SSCs from failures of non safety-related SSCs. This demonstration should show that the mitigating devices are adequate to protect safety-related SSCs from failures of non safety-related SSCs regardless of failure location (consideration can be given to the likelihood of failure at a particular location based on sound engineering judgment). If this level of protection can be demonstrated, then only the mitigative features need to be included within the scope of license renewal. However, if an applicant cannot demonstrate that the mitigative features are adequate to protect safety-related SSCs from the consequences of failures of non safety-related SSC's, then the applicant should utilize the preventive option, which requires that the entire non safety-related SSC be brought into the scope of license renewal. An applicant may determine that, in order to ensure adequate protection of the safety-related SSC, a combination of mitigative features and non safety-related SSCs must be brought within scope. Again, it is incumbent upon the applicant to provide adequate justification for the approach taken with respect to scoping of non safety-related SSCs in accordance with the Rule."***

VCSNS considers seismic supports for non-safety related components in scope as structural commodities. These supports are "mitigative features" for components other

than those listed above (piping and piping system components, ventilation ductwork, and piping and component insulation). These mitigative features prohibit the component from adversely effecting safety related equipment.

The environment for these seismically supported components is ambient air. These seismically supported components are located in a sheltered (i.e. indoors, non-condensing) environment. In this type of environment pitting and crevice corrosion are not considered aging effects that require management. The seismically supported components will remain dry and even if some general corrosion is experienced it would not be severe enough to challenge the actuators ability to remain intact. Therefore no aging management program would be required for the questioned seismically supported components.

The seismically supported components may be in areas where leaking boric acid is credible. In areas, seismically supported components would be subject to Boric Acid Corrosion Surveillances (B.1.2). These surveillances are performed to monitor the effects of leaking acid not specific components.

RAI 2.3.5-2: On page 5 of 56 of the technical report (RC-02-0159), the applicant stated that code break piping is within the scope of license renewal to preclude adverse affects on safety-related equipment and function. Please define the code break piping and in what situation a pipe is considered as code break piping.

VCSNS Response RAI 2.3.5-2

Code break supports are those pipe supports on Non-Nuclear Safety piping which are designed to ensure that significant stresses are not induced into Safety Related piping at safety class boundaries. Specifically, the seismic effects of the Non-Nuclear Safety piping are isolated from the Safety Related pipe. Code break supports protect essential equipment by extending the design requirements for Nuclear Safety Related piping beyond the class change until one support (at a minimum) in each of the three mutually perpendicular transverse directions is provided (or the equivalent). Code break piping is the piping in the Non-Nuclear Safety piping from the code pipe to the outer most code break support.

3.3 AUXILIARY SYSTEMS

Section 3.3.2.4.8 Fire Service System

RAI 3.3.2.4.8-1: (1)LRA Table 3.3-1 (Item 6) lists components in reactor coolant pump oil collect system of fire protection as a component group in the aging management program (AMP). However, the AMP only requires one time inspection for these components. Explain why these components should not be inspected periodically for managing aging.

(2) LRA Table 3.3-2 (Item 18) lists nozzles, piping, and fire hydrants of the fire service system as a component type subject to aging management evaluation. This table does not

identify any aging effect or mechanism to be evaluated for these components. However, the components in this component type expose to outside environment (such as fire hydrants) and are subject to corrosion that may results in loss of material due to pitting and microbiological influenced corrosion. Provide basis for not identifying any aging effect/mechanism for Item 18 aging management evaluation.

VCSNS Response RAI 3.3.2.4.8-1

The first five columns of Table 3.3-1 are NUREG-1801 listings. The sixth column is the VCSNS response.

NUREG-1801 recommends a one-time inspection for Reactor Coolant Pump Oil Collection System components that are composed of carbon steel, copper, and brass. The Reactor Coolant Pump Oil Collection System components at VCSNS are composed of stainless steel. The purpose of the oil collection system is to contain oil should there be a problem with a reactor coolant pump such that it leaks oil; therefore, it rarely contains oil. Should the system collect any oil, the temperature of the oil would be at Reactor Building ambient temperature, or above, such that moisture would not condense out of the oil to pool in the system. For these reasons, VCSNS maintains that the Reactor Coolant Pump Oil Collection system will not experience any aging effects requiring management. This fact is consistent with the operating experience reviews conducted at VCSNS.

Table 3.3-2 concerns Auxiliary System components or material/environment combinations not addressed by NUREG-1801. Item 18 of this table addresses components that are normally in a standby mode where air is the predominant internal environment. VCSNS external environments for these components are addressed in Application Table 3.3-1, Items 5 and 20.

APPENDIX B.1.5 FIRE PROTECTION PROGRAM

RAI B.1.5-1: LRA Appendix B.1.5, "Fire Protection Program," states that the fire protection program is consistent with XI.M26, "Fire Protection," XI.M27, "Fire Water System", and XI.M33, "Selective Leaching of Materials," as identified in NUREG-1801 and is enhanced in a specified table of this appendix. In order for the staff to evaluate the adequacy of the applicant's fire protection AMP and reach a conclusion that it is consistent with NUREG-1801, the staff requests the applicant to confirm the following:

(1) The additional guidance, which will be added to the diesel fire pump maintenance procedures during enhancements in accordance with Chapter XI. M26 of NUREG-1801, should ensure that the diesel-driven fire pump is under observation during the performance tests for detecting any degradation of the fuel supply line (such as flow and discharge tests, sequential starting capability tests, and controller function tests).

(2) The guidance, which will be added to the carbon dioxide (CO_2) fire suppression systems and fire damper inspection procedures in accordance with Chapter XI. M26 of the NUREG-1801, should include periodic visual inspection to examine signs of degradation. Material conditions that may affect the performance of the system (such as corrosion, mechanical

damage, or damaged damper) are observed during inspection. Inspection should be performed at least once every month to verify that the extinguishing agent supply valves are open, and the system is in an automatic mode.

(3) The specific guidance, which will be added related to fire door inspection, will ensure that hollow metal fire doors are visually inspected for holes in the skin of the door based on the plant specific frequency. Fire door clearances are also checked as part of an inspection. Functional tests of fire doors are performed daily, weekly, or monthly (which may be plant-specific) to verify the operability of the automatic hold-open, release, closing mechanism, and latches. The visual inspections should detect any degradation of the fire doors prior to loss of the intended function.

(4) The NRC staff has issued an Interim Staff Guidance (ISG)-04, "Aging Management of Fire Protection Systems for License Renewal" (in ADAMS Accession # 022260137, dated December 3, 2002), to modify the FP AMP described in NUREG-1801 Chapter XI. M27. The relevant portions of the ISG-04 are summarized below:

Staff Position for Wall Thinning of FP Piping due to Internal Corrosion

Fire protection piping is typically designed for a 50-year life in industrial applications. The limiting aging mechanism is general corrosion. Because the general corrosion of FP piping is typically very uniform, loss of intended function as a result of catastrophic failure caused by wall thinning throughout the system is possible and needs to be managed. However, internal inspections performed during each refueling cycle by disassembling portions of the FP piping, as stated in NUREG-1801, Chapter XI. M27, "Fire Water Systems," may not be most effective means to detect this aging effect. Each time the system is opened, oxygen is introduced into the system and this accelerates the potential for general corrosion. Therefore, the staff recommends that the applicant perform a baseline pipe wall thickness evaluation of the fire protection piping using a non-intrusive means of evaluating wall thickness, such as volumetric inspection, to detect this aging effect before the current license term expires. The staff also recommends that the applicant performs pipe wall thickness evaluations at plant-specific intervals during the period of extended operation. The plant-specific inspection intervals are determined by engineering evaluation performed after each inspection of the fire protection piping to detect degradation prior to the loss of intended function. As an alternative to pipe wall thickness evaluations, an applicant may use the existing Chapter XI. M27.

As part of the review of this issue and the above stated approach, a concern was raised as to the inspection specifications of the internal surface of below grade FP piping. The staff acknowledges that some applicants may be able to demonstrate that the environmental and material conditions that exist on the interior surface of below grade FP piping are similar to the conditions that exist within the interior surface of the above grade FP piping. If an applicant makes such a demonstration, the staff agrees that the results of the interior inspections of the above grade FP piping can be extrapolated to evaluate the interior condition of the below grade FP piping. If not, additional inspection activities are needed to provide reasonable assurance that the intended function of below grade FP piping will be maintained consistent with an applicant's current licensing basis for the period of extended operation.

Staff Position for Testing of Sprinkler Heads

NFPA 25 (1999 Edition) Section 2.3.3.1, "Sprinklers," states that where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing. NFPA 25 also contains guidance to perform this sampling every 10 years after the initial field service testing.

The 50-year service life of sprinkler heads does not necessarily occur at the 50th year of operation in terms of licensing. The service life is defined from the time the sprinkler system is installed and functional. In most cases, sprinkler systems are in place several years before the operating license is issued. However, sprinkler systems in some plants may have been installed after the plant was placed in operation. The staff recommends, in accordance with NFPA 25, that sprinkler head testing should be performed at year 50 of sprinkler system service life, not at year 50 of plant operation, with subsequent sprinkler head testing every 10 years thereafter.

In order to adequately managing the water-based FP systems and components (including sprinklers, nozzles, fittings, valves, hydrants, hose stations, stand-pipes, water storage tanks, and aboveground and underground piping), the staff requests the applicant to revise LRA Appendix B.1.5 in accordance with ISG-04 and revise NUREG-1801 Chapter XI. M27 to assure maintenance of the structures and components intended function during the period of extended operation. The staff also requests the applicant to discuss (1) how it plans to follow the guidance of the ISG-04, and (2) how this will be reflected in LRA Appendix B 1.5 and conforms with the staff position, as outlined above.

VCSNS Response RAI B.1.5-1

(1) In the present monthly surveillance test procedure for the Diesel Fire Pump, a visual inspection of any leaks or abnormalities is required and documented. Any degradation to the diesel fire pump fuel oil line would be detected during this pre-starting visual inspection.

(2) The present surveillance test procedures for fire dampers require visual inspections of fire dampers that specifically look for changes in appearance or abnormal degradation. These surveillance test procedures are performed every 18 months. No aging effects have been identified for the internal surfaces for carbon dioxide suppression system components. Aging of the external surfaces for the components will be managed by the Inspections for Mechanical Components. At VCS, the Carbon Dioxide Fire Suppression System valve lineup is required by the Fire Protection Program to be performed every 92 days. The interval for the Carbon Dioxide Fire Suppression System valve lineup was changed from monthly to quarterly under the provisions of 10CFR50.59.

(3) Current plant surveillance test procedures are performed on fire doors on a minimum frequency of six months. These procedures require visual inspections of the following: (a) Automatic Closing Mechanisms - to verify no oil leaks, hardware fasteners are secure, and adjusting rods are in place and secure; and (b) Door Integrity - to verify

latches are securely in place, free movement of bolts, bolt engages door strike, knobs and surface hardware are firmly attached, door closes and latches on its own power, no holes or breaks in the door skin, and no broken, damaged or cracked door glass.

As noted in Application Section B.1.5, VCSNS fire rated doors are inspected (as specified above) at a frequency of every 6 months under the current licensing basis rather than the bi-monthly frequency recommended in NUREG-1801, Section XI.M26. Based on VCSNS and industry operating experience, the 6 month inspection frequency provides reasonable assurance that degradation of a door is detected prior to loss of function.

(4) Section B.1.5 of the LRA lists the wall thickness evaluations as an enhancement to the Fire Protection Program. VCS will perform the wall-thickness evaluations of above ground fire protection piping prior to the end of the current operating term (August 6, 2022). Subsequent evaluations will occur at 10-year intervals thereafter. At VCSNS, the internal surfaces of underground piping for fire service is cement lined. No aging effects have been identified for the internal surfaces of cement lined piping in a raw water environment.

Section B.1.5 of the LRA lists the sprinkler testing/replacement as an enhancement to the Fire Protection Program. Testing/replacement will be performed in accordance with NFPA Code 25, which states that this should be done prior to year 50 of sprinkler system life, with subsequent testing performed at 10 year intervals. To ensure testing is performed prior to year 50 of sprinkler system life, VCS will perform this testing prior to the end of the current operating term (August 6, 2022).

RAI B.1.5-2: The staff is concerned that the applicant's AMP for FP systems and components may not adequately manage the aging of the protective coatings in steel structure, since neither NUREG-1801 Chapters XI. M26 nor XI. M27 address aging effects for the protective coating. On this basis, the staff requests the applicant to identify any steel structures within the scope of license renewal and subject to an AMR which depend on coatings to protect steel structures from aging-related degradation. For any such coatings, describe the aging management activities that manage the aging effects for the coatings and identify what AMP performs these activities.

VCSNS Response RAI B.1.5-2

The VCSNS Fire Protection Program (as described in Application Section B.1.5) is focused primarily on the fire protection system components, fire barriers and seals, and fire doors (consistent with GALL Sections XI.M26 and XI.M27). Steel structures (including structural steel components) within the scope of license renewal are identified by building in Application Section 2.4 and TR00170-003. Additionally, all structural steel has a protective coating which provides protection against age-related degradation. As noted in Application Table 3.5-1 (Item 16), aging of steel components is managed by the Maintenance Rule Structures Program as described in Application Section B.1.18. This program inspects structural steel for integrity via visual inspections of coatings for degradation such as peeling, flaking, blistering, rusting , scaling, etc. For

containment steel structures (liner), AMPs described in Application Sections B.1.11, B.1.15 and B.1.16 also apply.

Attachment II
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Sections 2.4, 2.5, 3.6, and Appendix B
Accession No. ML030900596

2.4 SCOPING AND SCREENING RESULTS: STRUCTURES

2.4.1 Reactor Building

RAI 2.4.1-1: LRA Section 2.4.1, "Reactor Building," 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. 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, lists reduction in foundation strength due to erosion of porous concrete subfoundation as an AMR result for the foundations in Table 2.4-2, the staff interprets that component type "foundations" to include the foundation mat and the fill concrete, which is the subfoundation. Verify whether the staff's interpretation is correct. If not correct, 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 these structures. 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.

VCSNS Response RAI 2.4.1-1

Table 3.5-1, Item 9, is addressed in the Application only for completeness in using the GALL tabular format and listings. Porous concrete is not used at VCSNS.

1) **Fill concrete is addressed in detail in Response to RAI 3.5-6, concluding that it does not perform an intended function and does not require evaluation under any aging management programs. "Foundations" as listed in Application Table 2.4-2 include only the design structural foundations which are above the fill concrete.**

2) **Table 3.5-1, Item 9, is not listed as an AMR result for the Auxiliary, Control, Fuel Handling, Intermediate, Turbine, and Service Water Discharge Structures since the GALL did not identify this specific aging effect (erosion of porous concrete) in the tabular listing for "Class 1 Structures". In alignment with the GALL, this item was only addressed under Reactor Building. [Note that only the Reactor, Auxiliary and Control Buildings have underlying fill concrete.]**

RAI 2.4.1-2: 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 are not considered a long-lived components because they are tested and replaced when warranted by their condition, and therefore do not require an AMR. According to 10 CFR 54.21 (a)(1)(ii) 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 intervals 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).

VCSNS Response RAI 2.4.1-2

Containment hatches are "components" that meet the requirement of 10 CFR 54.21 (a)(1)(i) and are subject to an AMR as described in the Application. O-rings are considered as "parts" of these components and are not individually identified as meeting the requirements of 10 CFR 54.21(a)(1)(ii). Regardless, aging management of containment hatches (including all parts) is required to meet 10 CFR 50 Appendix J; therefore, implementation is under the Appendix J Leak Rate Testing Program (Application Appendix B.1.12). Plant procedures require that hatch seal leakage be tested within seven days following any door operation to ensure that containment integrity is achieved, thus ensuring functional integrity of the seals.

RAI 2.4.1-3: 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. Since the first half of the component type Anchorage / Embedments is Anchorage, which is identical to the component type Anchorage, and the second half is identical to the component type Embedments, This is confusing. Clearly describe each component type so that staff can distinguish the three component types.

VCSNS Response RAI 2.4.1-3

At VCSNS, general definitions of these component types are as follows:

1) **Anchorage - Cast in-place anchor bolts.**

2) **Anchorage / Embedments (exposed surfaces) - Includes support bearing plates, other anchor bolts such as Hilti Bolts or embedments for attachment such as Unistrut.**

3) **Embedments - Flat plates embedded in concrete surfaces (walls, ceilings, etc.) which are anchored with Nelson Studs. Flat plates are used as attachment plates for welded supports.**

RAI 2.4.1-4: Indicate whether there are any masonry block walls in the reactor building which are subject to AMR.

VCSNS Response RAI 2.4.1-4

1) **There are no masonry block walls in the Reactor Building, nor does Application Table 2.4-2 (Reactor Building) include masonry block walls.**

2) **As noted in Application Table 3.5-1 (Item 20), masonry block walls are not used in nuclear safety related structures at VCSNS.**

3) **Note that Application Table 2.4-4 (Control Building) and Table 2.4-6 (Fuel Handling Building) are slightly confusing in that " Masonry Block, Brick Walls, or Knockdown Walls" are listed. These structures do not contain masonry block walls, rather knockdown walls constructed of reinforced concrete within an embedded structural steel sleeve. This design concept was used at VCSNS to allow for future openings within walls.**

2.4.2 Other Structures

RAI 2.4.2-1: LRA Section 2.4.2.1, "Auxiliary Building," of the LRA states: "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 mean 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 including the Hot Machine Shop from scope, but not for excluding it from an AMR. LRA Table 2.2-2, "Structural Scoping Results," lists the Hot Machine Shop as in scope, the reason being that its intended functions are those that meet the requirements of 10 CFR 54.4(a)(2), and involve a seismic II/I concern. Your statement in Section 2.4.2.1 appears to be inconsistent with the intended functions listed in Table 2.2-2. Clarify whether or not the Hot Machine Shop is in scope and requires an AMR, and provide a justification for your determination.

VCSNS Response RAI 2.4.2-1

This RAI is correct in that the Application is contradictory for including the Hot Machine Shop in scope. Application Table 2.2-2 was extracted from the VCSNS Scoping Report which identified the Hot Machine Shop as <u>initially</u> in scope due to the <u>potential</u> for seismic interaction with the Auxiliary Building. During the Screening process, it was subsequently determined that failure of the Hot Machine Shop would have an insignificant impact on the Auxiliary Building, and would not prevent satisfactory

accomplishment of any safety related functions. Therefore, since it does not actually perform an intended function, it was taken out of scope.

The statement in Application Section 2.4.2.1 is correct in that the failure of the Hot Machine Shop will not prevent the satisfactory accomplishment of any required safety related functions, and is thus not subject to an aging management review. Supporting Technical Reports will be revised to delete the Hot Machine Shop from the scope of license renewal.

RAI 2.4.2-2: LRA Section 2.4.2.1, "Auxiliary Building," 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". If you determine that these two tanks are subject to an AMR, provide the information on component type, intended functions, and AMR results for these two tanks. If not, provide a justification for their exclusion.

VCSNS Response RAI 2.4.2-2

The Refueling Water Storage Tank and the Reactor Make-Up Water Storage Tank are both in scope and included in the Application with their respective mechanical systems. The Refueling Water Storage Tank is discussed in Section 2.3.2.5 and Table 3.2-2, Items 1 and 7. The Reactor Make-Up Water Storage Tank is discussed in Section 2.3.3.18 and Table 3.3-2, Items 1 and 20.

RAI 2.4.2-3: 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 you determine that grout is subject to an AMR, provide the information on component type, intended functions, and AMR results for the grout. If not, provide a justification for its exclusion.

VCSNS Response RAI 2.4.2-3

In the Application 2.4 Tables, grout is generically included as a component type under "Equipment Pads" for each structure even though it is not specifically listed as an individual component type. In the supporting technical reports, grout is not identified as an individual commodity type, rather included under the commodity grouping of "concrete", and subject to the same AMPs.

2.5 SCOPING AND SCREENING RESULTS: ELECTRICAL

RAI 2.5-1: Appendix B of NEI 95-10 identifies uninsulated ground conductors, isolated-base bus, non-segregated-phase bus, and segregated-phase bus, as passive components. In Section 2.5 of the LRA, you indicate that these components were screened out. In addition, you indicate that they are considered out of scope for license renewal because they do not perform any intended functions. Explain why each of these passive components performs no intended function at VCSNS defined in 10 CFR 54.4. For uninsulated ground conductors, amplify in your response how each of the following criteria is addressed.

(a) GDC 3, "Fire Protection," states: "SSCs important to safety shall be designed ... to minimize ... the probability ... of fires." Explain why uninsulated ground conductors are not relied on (or credited) in safety analyses or plant evaluations in the design of SSCs important to safety to minimize the probability of fires pursuant to GDC 3.

(b) GDC 17, "Electric Power Systems," states "Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies." Explain why uninsulated ground conductors are not relied on (or credited) in safety analyses or plant evaluations in the design of SSCs important to safety pursuant to GDC 17. As a part of this explanation clarify (1) why unisulated ground conductors are not relied on to meet GDC 17 in the design of loss of power instrumentation for opening the offsite power supply breaker to the 7200 volt Class 1E bus, (2) the preferred offsite power system including the switchyard and transmission network to assure onsite electrical safety system are capable of withstanding electrical system disturbances (e.g., electrical faults, lightning surges), and (3) the EG - Generator & Main Transformer system to assure onsite electrical safety systems are capable of withstanding electrical system disturbances (e.g., electrical faults, lightning surges)].

(c) Section 2.1.1.3.3 of LRA states "Electrical systems and portions of electrical systems that are non-safety-related but whose failure could prevent the satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), and (iii) [Reference 2.1-1] are within the scope of license renewal (as outlined in 10 CFR 54.4(a)(2))." And the LRA further indicates in table 2.2-3 that the "EC - Grounding & Cathodic Protection" system is out of scope of license renewal. However, the following statements in the FSAR imply that failure of grounding systems could prevent the satisfactory accomplishment of a function identified in 10 CFR 54.4(a)(1)(i), (ii), and (iii) and should therefore be considered within the scope of license renewal (as outlined in 10 CFR 54.4(a)(2)). The FSAR on page 8.3-17 states: "A low impedance ground return path is provided to facilitate the operation of ground fault detection or protective devices in the event of ground fault or insulation failure on any electrical load or circuit." In addition, the FSAR states: "... over-current protection exists for the cables in the non-Class 1E trays so that they cannot be a hazard to the Class 1E trays whose separation distance has been violated." Clarify why grounding systems should be considered outside the scope of license renewal.

A followup question was received on May 15, 2003, from the NRC project manager as follows: "In addition to passive component commodity groups of uninsulated ground conductors, isolated-phase bus, nonsegregated-phase bus, and segregated-phase bus identified in RAI 2.5-1, Appendix B of NEI 95-10 also identifies as passive; Elements, RTDs, Sensors, Thermocouples, Transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors, watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, var. transducers, vibration transducers, voltage transducers). Expand the response to question 2.5-1 to explain why each of these passive components are not passive as stated in the NEI guidance or do not perform any intended functions pursuant to 10 CFR 54.4(a)(2)."

VCSNS Response RAI 2.5-1

VCSNS has only one application for bus duct, the isolated phase bus duct from the Main Generator to the Main Power Transformer in the Generator & Main Transformer (EG) System. This application is not in scope, as it is not credited as one of the two preferred sources for providing offsite power. See response to RAI 2.5-4 for further detail. Insulated cables are credited for providing offsite ESF power. These insulated cables on the plant system portion of the offsite power grid will be included in the Non-EQ Insulated Cable and Connection Inspection Program.

Regarding the NEI 95-10 Appendix B commodity group of "Elements, RTDs, Sensors, Thermocouples, Transducers", NEI 95-10, Revision 3, Appendix B, Item 84 states that from an electrical standpoint, the "Elements" commodity group is considered active and therefore screened out of consideration. From a pressure boundary standpoint, these elements are not considered in LR scope because they are not pressure boundary at VCSNS. RTDs at our plant, for example, are contained within thermowells and not subject to pressure boundary considerations. Any of the NEI 95-10 listed examples of elements, as well as other electrical devices that maintain a pressure boundary, are addressed within the Mechanical or Structural LR reviews.

Regarding uninsulated ground conductors (reference the EPRI License Renewal Electrical Handbook 1003057 December 2001), these are electrical conductors (e.g., copper cable, copper bar, steel bar) that are uninsulated (bare) and are used to make ground connections for electrical equipment. Uninsulated ground conductors are connected to electrical equipment housings and electrical enclosures as well as metal structural features such as the cable tray system and building structural steel. Uninsulated ground conductors are isolated or insulated from the electrical operating circuits. Uninsulated ground conductors enhance the capability of the electrical system to withstand electrical system disturbances (e.g., electrical faults, lightning surges) for equipment and personnel protection.

Uninsulated ground conductors do not include instrument grounding conductors or computer grounding conductors since these grounding conductors are insulated. Being insulated, in-scope instrument and computer grounding conductors are included in the aging management review of the general population of non-EQ insulated cables and connections.

Uninsulated ground conductors are installed in the Grounding and Cathodic Protection System (EC). This system is designated as non safety-related and does not meet the criteria of §54.4(a)(1).

Guidance for the application of the criteria of §54.4(a)(2) is provided in Section III.c.(iii) of the SOC to 10 CFR Part 54 and is provided below:

<u>SOC to 10 CFR Part 54, Section III.c.(iii) [60FR22467]</u>
"Pre-application rule implementation has indicated that the description of systems, structures, and components subject to review for license renewal could be broadly interpreted and result in an unnecessary expansion of the review. To limit this possibility for the scoping category relating to nonsafety-related systems, structures, and components, the Commission intends this nonsafety-related category (§54.4(a)(2)) to apply to systems, structures, and components whose failure would prevent the accomplishment of an intended function of a safety-related system, structure, and component. An applicant for license renewal should rely on the plant's CLB, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those nonsafety-related systems, structures, and components that are the initial focus of the license renewal review. Consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required."

The consideration of uninsulated ground conductors and multi point grounding connections within the EC system as a non-safety-related system which may impact a safety function would be based upon hypothetical failures resulting from system interdependencies, which are not a part of the CLB and have not been experienced.

Per the nonsafety-related criterion of §54.4(a)(2), all nonsafety-related electrical systems and components whose failure could prevent satisfactory accomplishment of any of the functions identified in §54.4(a)(1)(i), (ii) or (iii) are in scope.

The nonsafety-related scoping criterion of §54.4(a)(2) is not a function-based criterion but a failure-based criterion. To further understand this scoping criterion and how a nonsafety-related system or component could be within scope, the language of this criterion is expanded in Chapter 6 of the *License Renewal Electrical Handbook*, EPRI 1003057, (page 6-6) as follows:

License Renewal Electrical Handbook
"A nonsafety-related system or component is not in scope (per §54.4(a)(2)) unless its failure would:

- **cause a loss of the integrity of the reactor coolant pressure boundary,**
- **cause a loss of the capability to shut down the reactor or the capability to maintain it in a safe shutdown condition, or**
- **cause a loss of the capability to prevent or mitigate the consequences of accidents that could result in the potential offsite exposure specified in §54.4(a)(1)(iii)."**

This nonsafety-related failure is a single failure as discussed in licensing and station design documents. Single failures are considered as part of the current licensing basis for VCSNS. VCSNS is in conformance with licensing commitments concerning single failure as contained in Section 3.1, "Conformance with General Design Criteria" of the FSAR. Criterion 17 - Electrical Power Systems is excerpted below:

FSAR Section 3.1, Conformance with General Design Criteria
Criterion 17 - Electrical Power Systems

"...The onsite electrical power supplies...and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure...."

Based on conformance with single failure criteria as outlined in the VCSNS FSAR, no uninsulated ground conductor failure would prevent satisfactory accomplishment of any of the safety-related functions identified in §54.4(a)(1)(i), (ii) or (iii). Uninsulated ground conductors, therefore, do not meet the nonsafety-related scoping criterion of §54.4(a)(2).

With regard to the quote from the VCSNS FSAR page 8.3-17, and subsequent mention of overcurrent protection in cases of cable tray separation violation, the acceptability of the separation violations are well documented as discussed in the FSAR Section 8.3.1.4. Based on conformance with single failure criteria as outlined in the VCSNS FSAR and discussed above, no uninsulated ground conductor failure would prevent satisfactory accomplishment of any of the safety-related functions of those overcurrent protective devices relied upon for cable protection. In addition, these overcurrent protective devices are periodically tested IAW electrical breaker surveillance testing requirements at VCSNS. (Reference FSAR Section 8.3.1.4.1 (4).) Uninsulated ground conductors do not provide circuit protection (over-current protection) - electrical circuit protection is provided by circuit breakers and isolation devices to prevent single (common-mode) failures. Equipment grounds are more directed at personnel safety than equipment protection.

Uninsulated ground conductors are not relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (FP), environmental qualification (EQ), pressurized thermal shock (PTS), anticipated transients without scram (ATWS), or to address station blackout (SBO). Given this, uninsulated ground conductors are not relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations identified in §54.4(a)(3).

Unlike Turkey Point and St. Lucie plants, other plants undergoing license renewal activities, including VCSNS, have not found uninsulated ground conductors to be within the scope of license renewal. At Turkey Point and St. Lucie, uninsulated ground conductors are specifically identified in their Fire Protection commitments (for lightning protection in the switchyard) and are in scope only for the Fire Protection scoping criterion. VCSNS has no such commitment.

NRC reference to excerpts of the introductory sentence of GDC 3 as implied reason to treat uninsulated ground conductors as potentially impacting safety systems from a fire protection standpoint seems to be taken out of context. GDC 3 is referring to the design related to materials and location of SSCs to preclude the probability and minimize the effect of fires. Since uninsulated ground conductors have no combustible organic materials associated with them, they were not a consideration within the first sentence

of GDC 3. Fire protection analysis did not consider uninsulated grounds because they are not relied upon for fire protection/prevention.

In conclusion, uninsulated ground conductors are not within the scope of license renewal because the scoping criteria of §54.4(a)(1), §54.4(a)(2) or §54.4(a)(3) are not met.

RAI 2.5-2: Section 2.5 of the LRA does not identify fuse holders as part of any commodity group or in scope for aging management review. Clarify which commodity group fuse holders belong to.

VCSNS Response RAI 2.5-2

Due to recent industry and NRC meetings and the issuance of a fuse holder ISG, in-scope, non EQ, passive fuse blocks, not a part of a larger active equipment or active device enclosure, will be included in-scope, within the same commodity group as terminal blocks. This is the Cables and Connections commodity group of NEI 95-10. Fuse holders will be specifically included in our scoping, screening, and aging management review methodology and will be handled in a manner consistent with the recently approved ISG-5 on Fuse Holders. Also see the Response to RAI 3.6-5.

RAI 2.5-3: Table 2.2-3 of the LRA indicates that the EP - Emergency Power system is out of scope for license renewal. Explain why the EP - Emergency Power system is out of scope for license renewal.

VCSNS Response RAI 2.5-3

System EP is a system designator with no assigned set of installed permanent plant components. The system functions are covered in other systems. The CHAMPS database shows that this system has no equipment or components assigned to it, while the designation may be used to generate a repetitive task in the database (i.e., a dummy field). This system's functions are covered in other systems.

RAI 2.5-4: Table 2.2-3 of the LRA indicates that the EG - Generator & Main Transformer system is out of scope for license renewal. Explain why the EG - Generator & Main Transformer system is out of scope for license renewal. With main transformer and EG-Generator system bus duct failure, clarify how the two preferred offsite circuits are isolated.

VCSNS Response RAI 2.5-4

The EG system provides for the transmission of power from the site. The handling of plant loads, which are in the LR scope, is provided by one of the two preferred paths of offsite power, which do not include system EG [reference FSAR 8.1]. The Main Generator bus is not used by either of the two preferred sources of offsite power and is isolated by the associated substation 230 KV circuit breaker OCB-8892. The main electrical generator bus is not subject to aging management because it does not meet

any of the criteria in 10 CFR 54.4(a). The main transformer is in the same category, and system EG is not relied upon for any in-scope electrical back feed in response to an SBO event. The system is therefore not in the scope of license renewal consideration.

The boundary of the plant systems portion of the offsite power grid for the two preferred sources of offsite power is shown on a drawing, which has been furnished for your information as requested.

It should be noted that the 230KV preferred source of offsite power comes from switchyard 230KV bus 3. A mistake was made in the LRA Section 2.1.1.1.4, Table 2.2-2 [Electrical Substation; Transmission Towers and Foundations], and Section 2.5.4, which refer to 230KV bus 1. The correct 230KV preferred source of offsite power is 230KV bus 3.

4.4. Environmental Qualification (EQ)

RAI 4.4-1: Section 4.4 (3rd paragraph) of the LRA indicates that each of the EQ documentation binders contains or references either a calculation of qualified life or an evaluation to justify a qualified life. For components that justify their qualified life based on an evaluations (versus a calculation) and whose qualified life evaluation meets the 10 CFR 54.3 definition for TLAA and is thus considered a TLAA for license renewal, describe and justify the method (when the Analytical EQ reanalysis method using calculations described in NUREG-1801 is not used) for extending the qualified life from 40 to 60 years.

VCSNS Response RAI 4.4-1

The Arrhenius methodology is the approved model used for calculating thermal qualified life of 10 CFR 50.49 equipment at VCSNS. As discussed in the LRA, Section 4.4.1.2, there may be some reduction of excess conservatism in service conditions from previous evaluations when sufficient information is available; however, the Arrhenius model is used in processing thermal qualified life determinations in accordance with approved Engineering Services calculation procedures.

RAI 4.4-2: Section B.3.1.2 of the LRA states: "The EQ Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation." Clarify the intent of this concluding statement given that Section 2.5 (3rd paragraph) of the LRA indicates that components in the EQ program are not subject to an aging management review.

VCSNS Response RAI 4.4-2

LRA Section B.3.1.2 should read: "The EQ Program provides reasonable assurance that the aging effects will be managed such that the components subject to an EQ TLAA will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation."

RAI 4.4-3: Explain why (and to what extent) electrical penetration assemblies are not subject to the aging (or qualified life) requirements of 10 CFR 50.49. Assuming all electrical penetrations are included within the VCSNS Harsh EQ Program and required to meet 10 CFR 50.49, does this mean that the non-1E electrical penetrations qualified life analysis is considered a TLAA which will be re-analysed for 60 years? And does this mean non-1E electrical penetrations will also be subject to an aging management review?

VCSNS Response RAI 4.4-3

All electrical penetrations are included within the VCSNS Harsh EQ Program and meet the requirements of 10 CFR 50.49. All VCSNS electrical penetrations are subject to a TLAA. Non-Class 1E electrical penetrations were previously conservatively listed as requiring an aging management review because of their non-Class 1E status [Reference LRA 3.6.1.4]. The Aging management review is not required as these electrical penetrations have a qualified life that is administratively controlled within the EQ Program, are handled the same as class 1E penetrations, and are screened out in 54.21(a)(1)(ii).

Although the non-1E electrical penetrations are covered within the EQ Program as a TLAA, the EQ Program does not include the non-1E cables leading up to the electrical penetrations, the splices or connections inside and outside the RB. These non-EQ components will be included in the Cables and Connections Aging Management Program.

RAI 4.4-4: With regard to the attribute, Data Collection & Reduction Methods, for reanalysis of the EQ aging evaluation, Section X.E1 of NUREG-1801 states: "A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a re-analysis are to be justified on a plant-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging." Clarify that the Virgil C. Summer EQ program is consistent with this attribute or justify the extent to which the EQ program is inconsistent with this attribute.

VCSNS Response RAI 4.4-4

The VCSNS EQ Program is fully consistent with this attribute contained within Section X.E1 of NUREG-1801.

3.6 Aging Management of Electrical and Instrumentation and Controls

RAI 3.6-1: In LRA Section B 2.9, the applicant states that the Non-EQ Insulated Cables and Connections Inspection Program will be consistent with GALL program XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements, as identified in NUREG-1801 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 GALL program XI.E1 and the applicant's AMP) how the applicant's AMP is consistent with the GALL program XI.E1.

VCSNS Response RAI 3.6-1

Section B.2.9 of the LRA has been revised to (1) clarify the consistency of the VCSNS AMP to the XI.E1 Program as identified in NUREG-1801, (2) incorporate the AMP for fuse holders, and (3) clarify which instrumentation cables are included in the XI.E1 and XI.E2 AMPs. The new B.2.9 is designated Attachment VII herein.

RAI 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 range or lower in range, 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.

VCSNS Response RAI 3.6-2

VCSNS will establish a GALL type program for relevant, non-EQ, in-scope I&C cables with sensitive, low-level signals for the NI and RM systems. A description of this program is attached herein. The program will use the guidance of the GALL program as well as considering the proposed changes to the GALL program as has recently been presented to the NRC in meetings with the License Renewal Electrical Working Group (LREWG). A description of this new program is attached and is considered consistent with the NUREG-1801 Program XI.E2. For those relevant, non-EQ, in-scope I&C cables with sensitive, low-level signals for which the cable is not specifically included in the loop calibration process, an "Alternate XI.E2" Program is being reviewed by the LREWG. Implementation of an alternate program will be considered, when appropriate, for low signal level NI and RM circuit cables without loop calibrations, after the industry finalizes the approach. See the summary description of the XI.E2 GALL type program followed by the alternate XI.E2 Program, both designated as Attachment VIII herein.

RAI 3.6-3: LRA Table 3.6-1, Item 4 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.

VCSNS Response RAI 3.6-3

Water treeing phenomenon must be addressed when relevant in-scope medium voltage underground cables are exposed to moisture together with significant voltage. VCSNS recognizes the potential uncertainties involved with water treeing, even with ducts that are sloped to preclude moisture accumulation, and will create a program consistent with NUREG-1801 section XI.E3. The program description is designated as Attachment IX herein. Relevant cables are limited to that supplying 7.2kv to the Service Water Pumphouse motors via two underground ducts using Okonite EPR cable with a Hypalon jacket. All other underground 7.2kv cables are either not in the LR Scope or are energized less than 25% of the time. The underground ducts for the relevant cables are sloped to provide drainage. Cable and manhole inspections have shown indications that the relevant cables have been exposed to moisture with significant voltage. The VCSNS program described herein will result in a 10-year test interval by an appropriate industry approved testing method selected to validate the satisfactory condition of the cable insulation and to give some assurance of the remaining life of the cable, while not damaging the cable itself. The specific type of test performed will be determined prior to the initial test. The 10-year interval will commence prior to the start of the period of extended operation. A description of this new program is designated Attachment IX herein and is consistent with the NUREG-1801 Program XI.E3.

RAI 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 (in ultraviolet-sensitive materials only) of organic, radiation-induced oxidation, and 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.

Response RAI 3.6-4

All VCSNS Electrical Penetrations are included within the VCSNS Harsh EQ Program and meet the requirements of 10CFR50.49. The non-Class 1E as well as the Class 1E electrical penetrations are considered subject to a TLAA and will be re-analyzed for a 60-year life under the EQ Program. All Electrical Penetrations have a definitive long-lived qualified life assigned within the EQ Database "HARSH EQ Maintenance Manual" the same as all Harsh EQ related equipment. Non-Class 1E electrical penetrations were previously conservatively listed as requiring an Aging management review because of their non-Class 1E status [Reference LRA 3.6.1.4]. The Aging management review is not required as these electrical penetrations are to receive a TLAA for consideration of a 60-year life. There will be no aging management program for electrical penetrations as these electrical penetrations have a qualified life that is administratively controlled within the EQ Program and are screened out in 54.21(a)(1)(ii).

<u>RAI 3.6-5:</u> 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 are considered a piece parts of the larger assembly and not subject to an AMR.

As discussed in NUREG-1760 "Aging Assessment of Safety-Related Fuses Used in Low- and Medium-Voltage Applications in Nuclear Power Plants." operating experience 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 an AMR and require an 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 overtemperature, 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. Provide a commitment to Implement the fuse holder ISG.

VCSNS Response RAI 3.6-5

Based upon recent industry and NRC discussion and the issuance of a fuse holder ISG, in-scope, non-EQ, passive fuse blocks will be included within the same commodity group as terminal blocks, i.e. the Cables and Connections commodity group of NEI 95-10. Fuse holders are specifically included in scoping, screening, and aging management review methodology and are handled in a manner consistent with the recently approved ISG-5. In addition to the visual inspection of in-scope, passive fuse holders on a 10-year periodicity for indication of age related degradation, the metallic fuse clip portion of the in-scope, passive fuse holders that are found to be susceptible to age related degradation, will receive a continuity check or will undergo thermography or other appropriate test on a representative sample basis to assure the metallic fuse clip is still making a good connection. This test or inspection will serve to give additional assurance that evidence of age related degradation from fatigue, mechanical stress, vibration, chemical contamination, and corrosion will be discovered prior to a loss of intended function.

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

VCSNS Response RAI 3.6-6

For the ambient environmental conditions at V. C. Summer, no aging effects have been identified that could cause a loss of intended function for the extended period of operation. Therefore, there are no applicable aging effects for the switchyard bus.

At VCSNS, the switchyard bus is comprised of copper or 5" schedule 80 aluminum tube. Organic materials are not involved. Connections to the switchyard bus are welded. Conductor connections are generally of the compression bolted category. The switchyard bus is located in the yard, which is the ambient environment of the plant. The temperature ranges from a historic low of -4°F to a high of 108°F. The environment is periodically wet (from rainfall). The copper and aluminum materials do not experience any appreciable aging effects in this environment, except for minor oxidation, which does not impact the ability of the switchyard bus to perform its design function. In order to validate aging effects, a review of industry experience was performed. This review included NRC generic communications, LERs, and NRC NUREGs related to switchyard bus. No documents involving switchyard bus were identified. VCSNS operating experience was also reviewed to validate aging effects for switchyard bus and connections. This review included CERs (condition evaluation reports) and NCNs (non-conformance notices) for any documented instances of switchyard bus aging, in addition to interviews with responsible SCANA Substation Department and VCSNS engineering and maintenance personnel. No instance of aging related problems with in-scope switchyard bus and connections due to contaminants or oxidation was uncovered.

RAI 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 have been identified at VCSNS.

VCSNS Response RAI 3.6-7

From the EPRI License Renewal Electrical Handbook 1003057 December 2001 in summary:

Regarding HV transmission conductor strength, tests performed by Ontario Hydroelectric showed a 30% loss of composite conductor strength of an 80-year-old ACSR conductor due to corrosion. There is a set percentage of composite conductor strength established at which a transmission conductor is replaced. As illustrated below, there is ample strength margin to maintain the transmission conductor intended function through the extended period of operation.

The National Electrical Safety Code (NESC) requires that tension on installed conductors be a maximum of 60% of the ultimate conductor strength. The NESC also sets the maximum tension a conductor must be designed to withstand under various load requirements, which includes consideration of ice, wind and temperature. These requirements were reviewed concerning the specific conductors included in the AMR. The conductors with the smallest ultimate strength margin (4/0 ACSR) will be used as an illustration. VCSNS is in the medium loading zone; therefore, the Ontario Hydroelectric heavy loading zone study is conservative.

The ultimate strength and the NESC heavy load tension requirements of 4/0 ACSR are 8350 lbs. and 2761 lbs. respectively. The margin between the NESC Heavy Load and the ultimate strength is 5589 lb.; i.e., there is a 67% of ultimate strength margin. The Ontario Hydroelectric study showed a 30% loss of composite conductor strength in an 80-year-old conductor. In the case of the 4/0 ACSR transmission conductors, a 30% loss of ultimate strength would mean that there would still be a 37% ultimate strength margin between what is required by the NESC and the actual conductor strength.

The 4/0 ACSR conductors have the lowest initial design margin of any transmission conductors included in the AMR. This illustrates with reasonable assurance that transmission conductors will have ample strength through the period of extended operation. Corrosion of ACSR conductors is a very slow acting aging effect that is even slower for rural areas with generally less suspended particles and SO_2 concentrations in the air than urban areas.

At VCSNS, the transmission conductors are constructed of ACSR (aluminum conductor steel reinforced) material, either 795 kcmil or 1590 kcmil. The shield wire is 3/16" H.S. steel. There are no organic materials involved and no appreciable aging effects for the transmission conductors. The connections used for these conductors likewise have no

organic materials. There are no applicable aging effects that could cause loss of the intended function of the transmission conductors for the period of extended operation.

Regarding wind loading vibration and sway on HV connections, wind loading that can cause a transmission line and insulators to vibrate is considered in the design and installation. Loss of material (wear) and fatigue that could be caused by transmission conductor vibration or sway are found not to be applicable aging effects in that they would not cause a loss of intended function if left unmanaged for the extended period of operation.

In order to validate aging effects, a review of industry experience was performed. This review included NRC generic communications, LERs, and NRC NUREGs related to transmission conductors. No documents involving transmission conductors were identified.

VCSNS operating experience was also reviewed to validate aging effects for transmission conductors. This review included CERs (condition evaluation reports) and NCNs (non-conformance notices) for any documented instances of transmission conductor aging, in addition to interviews with responsible SCANA Substation Department and VCSNS engineering and maintenance personnel. No instance of aging related problems with transmission conductors was uncovered.

RAI 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 are greater concentrations of airborne particles such as near facilities that discharge soot or near the seacoast where salt spray is prevalent. Cracks in insulators have been known to occur 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 because 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 detailed 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.

VCSNS Response RAI 3.6-8

From the EPRI License Renewal Electrical Handbook 1003057 December 2001 in summary:

Regarding the potential for contamination of insulators, the buildup of surface contamination is gradual and in most areas such contamination is washed away by rain; the glazed insulator surface aids this contamination removal. 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 a problem in areas where

there are greater concentrations of airborne particles such as near facilities that discharge soot or near the seacoast where salt spray is prevalent. VCSNS is located in an area with moderate rainfall where airborne particle concentrations are comparatively low; consequently, the rate of contamination buildup on the insulators is not significant. At VCSNS, as in most areas of the SCE&G transmission system, contamination build-up on insulators is not a problem due to rainfall periodically "washing" the insulators. Additionally, there is no nearby heavy industry or other producers of industrial effluents, which could cause excessive contamination. There is no salt spray at VCSNS as the plant is over 100 miles from the ocean. Therefore, surface contamination is not an applicable aging effect for the insulators in the service conditions they are exposed to at VCSNS.

Regarding HV porcelain insulator cracking, porcelain is essentially a hardened, opaque glass. As with any glass, if subjected to enough force it will crack or break. The most common cause for cracking or breaking of an insulator is being struck by an object (e.g., a rock or bullet). Cracking and breaking caused by physical damage is not an aging effect and is not subject to an AMR. Cracks have been known to occur with insulators when the cement that binds the parts together expands enough to crack the porcelain. This phenomenon, known as cement growth, occurs mainly because of improper manufacturing processes or materials, which make the cement more susceptible to moisture penetration, and the specific design and application of the insulator. The string insulators susceptible to porcelain cracking caused by cement growth are isolated to bad batches (specific, known brands and manufacture dates) of string insulators used in strain application. The post insulators most susceptible to this aging effect are multi-cone (post) insulators used in cantilever applications. Research of NCNs and CERs within the VCSNS database and discussions with the Substation Department personnel revealed no instance of insulator cracking or failure related to cement growth at the VCSNS switchyard. Accordingly, cracking due to cement growth is not an applicable aging effect for the HV switchyard insulators in the service conditions they are exposed to at VCSNS.

Regarding mechanical wear, this is an aging effect for strain and suspension insulators in that they are subject to movement. Movement of the 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. Although this mechanism is possible, experience has shown that the transmission conductors do not normally swing and that when they do, due to a substantial wind, do not continue to swing for very long once the wind has subsided. Wind loading that can cause a transmission line and insulators to vibrate or sway is considered in the design and installation. The loss of material due to wear concern will not cause a loss of intended function of the insulators at VCSNS; therefore, loss of material due to wear is not an applicable aging effect for insulators.

VCSNS operating experience was reviewed to validate aging effects for switchyard insulators. This review included CERs (condition evaluation reports) and NCNs (non-conformance notices) for any documented instances of switchyard insulator aging, in

addition to interviews with responsible SCANA Substation Department and VCSNS engineering and maintenance personnel. No instance of aging related problems with in-scope switchyard insulators due to contaminants, cracking, cement growth, or mechanical wear was uncovered.

Attachment III
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Sections 3.1, and Appendix B
Accession No. ML030900279

3.1.2.2 Aging Management Evaluations in The GALL Report That Are Relied on License Renewal, For Which GALL Recommends Further Evaluation

3.1.2.2.2 Loss of Material Due to Pitting and Crevice Corrosion

RAI 3.1.2.2.2-1: In LRA Table 3.1-1, AMR Item 2, the applicant states that cracking in the steam generator shell caused by flaw growth is managed by the in-service inspection plan (LRA Appendix B.1.7). The plan is stated to be consistent with NUREG-1801, Section XI.M1, which, in turn, is based upon ASME. Loss of material caused by general, pitting, and crevice corrosion is managed by the chemistry program (LRA Appendix B.1.4). The staff notes that the applicant's in-service inspection program cites Section XI, Subsections IWB, IWC, and IWD. However, Section XI.M1 of NUREG-1801 states, "In certain cases, the ASME inservice inspection program is to be augmented to manage effects of aging for license renewal...." For the case of pitting corrosion at the surfaces of the steam generator shell assembly, NRC Information Notice (IN) 90-04 recommends the use of such augmented procedures, including the use of enhanced ultrasonic techniques and additional visual and magnetic particle examinations as required.

a. Provide an enhanced condition-monitoring program that can reliably detect loss of material due to pitting and crevice corrosion, and detect cracks at the inside surface of the VCSNS steam generator girth welds and shell plates so that loss of material and cracking are effectively managed.

b. The applicant states that its AMR results are consistent with NUREG-1801, which references to NRC IN 90-04 as the basis for enhanced detection of aging effects in the shell. Confirm whether the 100% secondary-side inspection during refueling outage 12 included an examination of the steam generator shell assembly for the presence of degradation (e.g., pitting). If such examinations were performed, identify the inspection methods used, list the exact shell components inspected, confirm whether these methods were adequate to detect pitting, and describe the inspection results.

VCSNS Response RAI 3.1.2.2.2-1

GALL Item IV.D1.1-c addresses corrosion of the upper and lower shell and the transition cone. The sub-component AMR results are consistent with the identified GALL Item with regards to material, environment, aging effects requiring management, and credited programs, except as clarified. Certain aging effects (cracking due to flaw growth) not specified for the identified GALL Item are managed by the credited programs.

NRC IN 90-04, which is referenced for the GALL Item as the basis for enhanced detection of aging effects in the shell, addressed flaw growth at girth welds due to service loading.

The IN contains only general indication that pits on the surface served as crack initiation sites, and not that pitting corrosion resulted in sufficient degradation to cause loss of component function. No subsequent industry experience has further identified pitting corrosion resulting in reportable indications for the shell. Cracking due to flaw growth is managed at VCSNS by the ISI Plan.

3.1.2.2.4 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement

RAI 3.1.2.2.4-1: The reactor vessel internals program assumes sufficient redundancy in bolt function that the plant can continue to function safely with 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 the 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. A similar question is asked in RAI 3.1.2.2.12; however, it addresses irradiation-assisted stress corrosion cracking.

VCSNS Response RAI 3.1.2.2.4-1

It is the intent of VCSNS to follow industry initiatives for these inspections. At present, the specifics and details of these activities are not available. Effective and proven volumetric and visual examination techniques will be selected for use in performing the inspection.

The Reactor Vessel Internals Inspection includes the following inspection activities, which will be conducted on a sample of the most susceptible components, as determined by engineering evaluation:

- **For those components that are accessible or can be rendered accessible by the removal of the core and/or other internals for examination, a visual inspection will be performed to detect the presence and extent of cracking due to IASCC (and enhanced by reduction of fracture toughness due to irradiation embrittlement) and loss of material due to wear.**

- **For bolts and other inaccessible components, a volumetric inspection will be performed to detect the presence and extent of changes in dimensions due to irradiation creep and void swelling, cracking due to IASCC, loss of preload due to stress relaxation, and reduction of fracture toughness due to irradiation embrittlement and void swelling.**

- **With respect to changes in dimension due to void swelling, industry activities [including WOG and EPRI] are under way to determine whether this is an aging effect requiring management for license renewal, and, if necessary, to develop**

and qualify methods for detection and management. These activities will be monitored by VCSNS, and will be performed if necessary.

The Reactor Vessel Internals Inspection will include the following acceptance criteria:

- **For all subject components, critical crack size will be determined by analysis prior to the inspection.**
- **For bolts, any detectable crack indication is unacceptable for a particular bolt. However, the intended function(s) of reactor vessel internals can be maintained with fewer than 100% of the bolts intact. That quantity, and specific bolt locations, will be determined by analysis prior to the inspection.**
- **Specific acceptance criteria for changes in dimensions 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.**
- **Inspection results may also be compared with the acceptance standards of ASME Section XI, Subsections IWB-3400 and IWB-3500.**

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

RAI 3.1.2.2.6-1: 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 all on-going industry activities related to failure mechanisms for small-bore piping including the recommendations of the EPRI sponsored Materials Reliability Program (MRP) Industry Task Group (ITG) on Thermal Fatigue will be followed and whether changes to inspection activities based on industry recommendations will be evaluated. 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.

VCSNS Response RAI 3.1.2.2.6-1

It is the intent of VCSNS to follow and implement the recommendations of industry initiatives by the EPRI sponsored Materials Reliability Program (MRP) Industry Task Group (ITG) on Thermal Fatigue on small-bore piping. The selection locations for inspections will be selected to be representative of the bounding locations.

3.1.2.2.7 Crack Growth due to Cyclic Loading

RAI 3.1.2.2.7-1: 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 cracking 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 and if growth of underclad cracking is an aging effect, indicate what program will manage the applicable aging effect.

VCSNS Response RAI 3.1.2.2.7-1

Application Table 3.1-1, Item 7 includes only NUREG-1801 items for the three cylindrical shell sections (A2.5.1 and A2.5.2). This item does not include the flange (A2.5.3), the bottom head (A2.5.4), and the nozzles (A2.3.1 and A2.3.2). The vessel nozzles and vessel flange are SA-508 material. SA-508 material was not included in Application Table 3.1-1, Item 7, based on the referenced Item Number in GALL listing in Table 1 of NUREG-1800. During the license renewal evaluation VCSNS determined that underclad cracking is not an aging effect requiring management for the VCSNS reactor vessel.

The underclad cracking in the heat affected zones (HAZ) of SA-508, Class 2 forgings and plate due to reheat cracking as a result of post weld heat treatment is an aging effect requiring evaluation for stainless steel clad alloy steel Reactor Vessel sub-components. Two types of underclad cracking have been identified. Reheat cracking has occurred as a result of postweld heat treatment of austenitic steel cladding applied using high-heat-input welding processes on ASME SA-508, Class 2 forgings. Cold cracking has occurred in ASME SA-508, Class 3 forgings after deposition of the second and third layers of cladding that had no pre-heating nor post-heating applied during the cladding procedure. The VCSNS Reactor Vessel is not constructed of ASME SA-508, Class 3 material. The inlet/outlet nozzles, closure head flange, and Reactor Vessel flange are sub-components that are SA-508, Class 2 forgings. The high-heat-input welding processes affecting reheat cracking, based upon tests of both laboratory samples and clad nozzle cutouts, include strip clad and manual inert gas (MIG) cladding processes. Testing also revealed that reheat cracking did not occur with one-wire and two-wire submerged arc cladding processes. A review of available Reactor Vessel heat treatment and welding material historical data provided confirmation that high-heat-input weld process, as described above, were not utilized on VCSNS Reactor Vessel sub-components. The VCSNS position on Regulatory Guide 1.43 is included in FSAR Appendix 3A.

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

<u>RAI 3.1.2.2.9-1:</u> With respect to Ni alloy components please provide the following information:

a. 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.

b. 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 Alloy 600 aging management program (LRA Appendix B.1.1) to include all other Alloy 600 components in addition to reactor vessel head penetrations.

VCSNS Response RAI 3.1.2.2.9-1

The Alloy 600 Aging Management Program is utilized for aging management for components in addition to the vessel head penetrations. The program at VCSNS is expanded to include other components susceptible to PWSCC (ie, Alloy 600) rather than establishing an additional program. The program includes aspects of the Chemistry Program and In-Service Inspection Plan.

The components managed by this program are:

- **PZR Nozzle-Safe End Weld Metal,**
- **Steam Generator Primary Side Tubeplate,**
- **Reactor Vessel Bottom Head Penetration Tubes,**
- **Reactor Vessel Closure Head Penetration Tubes (CR, Instrument, Vent Pipe),**
- **Reactor Vessel Inlet & Outlet Nozzle Safe Ends**

RAI 3.1.2.2.9-2: 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 the following information: (1) inspection method used in detecting cracking in these components, (2) technical basis showing the adequacy of this method to detect cracking, (3) inspection frequency and its justification, and (4) acceptance criteria.

VCSNS Response RAI 3.1.2.2.9-2

The Aging Management Programs for cracking of the Core Support Pads and Bottom Head Penetrations include the Alloy 600 Aging Management Program, Chemistry Program, as well as the In-Service (ISI) Plan. ISI inspections are done in accordance with the ASME code requirements. VCSNS is active in industry groups specifically EPRI and WOG. New developments will be reviewed and if deemed appropriate incorporated into the aging management of the Core Support Pads and Bottom Head Penetrations.

RAI 3.1.2.2.9-3: The applicant has not presented the aging management review results for the 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.

VCSNS Response RAI 3.1.2.2.9-3

The pressurizer immersion heater well assemblies (heater sheaths) are made of 316 stainless steel, not alloy 600. They are not considered susceptible to PWSCC.

3.1.2.2.10 Crack Initiation and Growth due to SCC

RAI 3.1.2.2.10-1: In AMR Item 10, Table 3.1-1 of the LRA, the applicant identified crack initiation and growth due to SCC as an aging effect for the reactor coolant system (RCS) cast austenitic stainless steel (CASS) piping components. The applicant credits the chemistry program and the inservice inspection (isi) plan to manage these aging effects. Explain whether the inspection techniques included in the ISI plan are qualified for detecting and sizing cracks in the CASS components. Also, confirm whether the ISI plan includes inspection of heat-affected zones associated with any weld repairs performed on these CASS components.

VCSNS Response RAI 3.1.2.2.10-1

The In-Service Inspection (ISI) Plan detects degradation of piping welds by applying the examination and inspection requirements specified in ASME Section XI Tables IWB-2500-1 for Class 1 components. VCSNS has committed to use the 1989 Edition of ASME Code Section XI as the Code of record during the current (second) inservice inspection interval.

There have been no weld repairs to date on Class 1 CASS components.

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

RAI 3.1.2.2.12-1: 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.

VCSNS Response RAI 3.1.2.2.12-1

It is the intent of VCSNS to follow and implement the recommendations of the industry initiatives (specifically EPRI and WOG) for these inspections. At present, the specifics and details of these activities are not available. Effective and proven volumetric and visual examination techniques will be selected for use in performing the inspection.

The Reactor Vessel Internals Inspection includes the following inspection activities, which will be conducted on a sample of the most susceptible components, as determined by engineering evaluation.

- **For those components that are accessible or can be rendered accessible by the removal of the core and/or other internals for examination, a visual inspection will be performed to detect the presence and extent of cracking due to IASCC (and enhanced by reduction of fracture toughness due to irradiation embrittlement) and loss of material due to wear.**

- **For bolts and other inaccessible components, a volumetric inspection will be performed to detect the presence and extent of changes in dimensions due to irradiation creep and void swelling, cracking due to IASCC, loss of preload due to stress relaxation, and reduction of fracture toughness due to irradiation embrittlement and void swelling.**

- **With respect to changes in dimension due to void swelling, industry activities (including WOG and EPRI) are under way to determine whether this is an aging effect requiring management for license renewal, and, if necessary, to develop and qualify methods for detection and management. These activities will be monitored by VCSNS, and will be performed if necessary.**

The Reactor Vessel Internals Inspection will include the following acceptance criteria:

- **For all subject components, critical crack size will be determined by analysis prior to the inspection.**

- **For bolts, any detectable crack indication is unacceptable for a particular bolt. However, the intended function(s) of reactor vessel internals can be maintained with fewer than 100% of the bolts intact. That quantity, and specific bolt locations, will be determined by analysis prior to the inspection.**

- **Specific acceptance criteria for changes in dimensions 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.**

- **Inspection results may also be compared with the acceptance standards of ASME Section XI, Subsections IWB-3400 and IWB-3500.**

3.1.2.2.14 Loss of Section Thickness due to Erosion at the Feedwater Impingement Plate and Support

RAI 3.1.2.2.14-1: In LRA Table 3.1-1, AMR Item 14, the applicant addresses loss of section thickness due to erosion at the steam generator feedwater impingement plate and support. In its discussion, the applicant states that these components do not have a license renewal intended function for VCSNS and aging management is therefore not required. However, the applicant provides no justification for this conclusion. The staff is not clear whether the feedwater impingement plate and support are installed in the VCSNS steam generators. If these components are installed in the VCSNS steam generators, the applicant needs to provide the technical basis for excluding the steam generator feedwater impingement plate and support from the scope of license renewal.

VCSNS Response RAI 3.1.2.2.14-1

There is not an "impingement plate" on the Delta 75 Steam Generators. There is a similar component, a baffle, provided for Emergency Feedwater (Aux feed) to prevent cold water from spraying on the hot shell during filling or transients. The nozzle outlet for emergency and main feed are both below the normal water level. Emergency feed is not used in normal operation, the nozzle is normally flooded when the system is in service so erosion from the nozzle outlet in not an aging effect requiring management.

3.1.2.2.15 Crack Initiation and Growth due to PWSCC, ODSCC, and/or IGA

RAI 3.1.2.2.15-1: In LRA Appendix B.1.10, "Steam Generator Management Program," the applicant states that the steam generator management program is consistent with GALL AMP XI.M19. GALL AMP XI.M19 states that all PWR licensees have voluntarily committed to a steam generator degradation management program described in NEI 97-06, whose guidelines were under NRC staff review at the time of NUREG-1801 publication. GALL AMP XI.M19 states that the plant technical specifications, staff-approved NEI 97-06 guidelines, and any other alternate regulatory basis for steam generator degradation management that have been previously approved by the staff for that plant, are adequate to manage the effects of aging on the steam generator tubes.

a. The GALL report (NUREG 1801) was published approximately two years ago. The staff is not clear as to the applicant's commitment with respect to NEI 97-06. Confirm whether VCSNS has committed to the steam generator management program described in NEI 97-06 for the period of extended operation.

b. Identify the alternate regulatory basis for steam generator degradation mechanisms that have been approved by the staff for VCSNS and confirm whether these alternate bases are incorporated in the steam generator management program as presented in LRA Appendix B.1.10.

VCSNS Response RAI 3.1.2.2.15-1

The Steam Generator Management Program is credited with managing cracking and loss of material for the Steam Generator nickel-based alloy tubes and tube plugs on both the

primary and secondary sides. The Steam Generator Management Program is structured to meet NEI 97-06, "Steam Generator Program Guidelines" and consists of the following elements:

- **Primary and secondary chemistry control**
- **Foreign material exclusion**
- **Assessment of degradation mechanisms**
- **Inspection of steam Generator Tubing and tube plugs**
- **Tube integrity assessment**
- **Maintenance and repairs**
- **Primary to secondary leakage monitoring**

The Steam Generator tube volumetric inspection technique detects flaw size and depth, or alternatively, remaining sound tube wall thickness. Primary-to-secondary leakage is monitored to verify tube integrity during plant operation. Steam Generator tube integrity is assessed in accordance with the performance criteria in NEI 97-06. The performance criteria include structural integrity, accident-induced leakage, and operational leakage limits. Inspection activities also monitor for leakage from tube plugs.

RAI 3.1.2.2.15-2: The staff notes that after more than ten years of experience with thermally treated Alloy 690 tubes in the U.S. steam generator service, EPRI TE-106365-R14 reports virtually no incidents of wastage and pitting corrosion for these tubes. However, a small number of tubes have been plugged due to wear.

a. Describe the design features incorporated into the Westinghouse Delta-75 steam generators in VCSNS to minimize wear.

b. Describe the features of the steam generator management program to ensure that loss of material due to wear, loose parts, and other causes is effectively prevented or mitigated.

VCSNS Response RAI 3.1.2.2.15-2

The VCSNS Replacement Steam Generators incorporated many lessons learned from years of experience with U-tube Steam Generators. This information is reflected in the information supplied to the NRC for the Steam Generator Replacement Project at VCSNS. One source of information is WCAP-13480 Rev. 1.

Most of the features of the Model F Steam Generators are incorporated directly into the Delta 75 Replacement Steam Generator. Broached 405 stainless steel tube support plates, thermally treated Alloy 690 tubes, hydraulically expanded tubes, and reduced U-bend gaps are carried over from Model F Steam Generators. Several enhancements over the Model F Steam Generators were made to the anti-vibration bars.

The Steam Generator Management Program is structured to meet NEI 97-06, "Steam Generator Program Guidelines" and consists of the following elements:

- **Primary and secondary chemistry control**
- **Foreign material exclusion**
- **Assessment of degradation mechanisms**
- **Inspection of steam generator tubing and tube plugs**
- **Tube integrity assessment**
- **Maintenance and repairs**
- **Primary to secondary leakage monitoring**

Aging Management of Plant-Specific Components

3.1.2.4.2 Reactor Coolant Piping, Valves and Pumps

RAI 3.1.2.4.2-1: With respect to bolting provide the following information:

a. 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.

b. In LRA Table 3.1-1, AMR Item 22, the applicant presents the AMR results for the reactor coolant pressure boundary bolting. The applicant states that the AMR results are consistent with NUREG-1801 in materials and environments, but not in aging effects. The applicant further states that loss of closure integrity, rather than loss of preload or cracking, is the aging effect requiring management. The applicant credits the inservice inspection plan for managing these effects. Explain the difference between loss of closure integrity and loss of preload or cracking.

VCSNS Response RAI 3.1.2.4.2-1

Class 1 bolted closures (as well as non-Class 1 closures for the Steam Generators and Reactor Coolant Pumps) are covered by specific ASME Section XI activities only associated with the Reactor Coolant system (i.e. harsher conditions) such that treatment of individual sub-components is warranted for license renewal.

For all other bolted closures (i.e. pressure-retaining) of components/component types subject to aging management review, the design of critical closure joint bolting involves enough redundancy to ensure joint integrity and no aging effects unique to bolting, over the components being joined/closed, require evaluation for license renewal as discussed further below.

Although identified as an aging effect in various industry references, loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for non-Class 1 component bolted closures (i.e. pressure boundary closures) within the scope of license renewal.

Mechanical components within the scope of license renewal, both Class 1 and non-Class 1, contain bolted closures that are necessary for the pressure boundary of the components being joined/closed. Examples of these bolted closures are valve bonnet to body, pump cover to casing, heat exchanger manway and channel head (end-bell), and piping flange sets. The bolted closure is comprised of two mating surfaces, a gasket, and a fastener set (studs or bolts, washers, and nuts). By themselves, the mating set, gasket, or fastener set has no component intended function. Together, the bolted closure forms an integral part of the pressure-retaining boundary of the component. Additionally, the bolted closure is exposed to the same environment(s) as the components in the plant areas where the closure is located (process fluid for internal mating surface and ambient conditions else). As such, the bolted closure (including fastener set) was considered to be a sub-component (piece-part) of the components/ component types within the scope of license renewal and did not require evaluation separate from the component, except as clarified.

Loss of mechanical closure integrity can result in failure of the mechanical joint, as evidenced by leakage rather than joint failure, and can be attributed to one or more of the following effects:

- Loss of bolt pre-load (embedment, cyclic load embedment, gasket creep, etc.),
- Loss of bolting material (from general and/or boric acid corrosion),
- Reduction of bolting material fracture toughness, and
- Cracking of high strength bolting material (SCC).

Loss of pre-load is considered to be the result of inadequate design or improper assembly (i.e. event driven) or time dependant changes in stresses (e.g. creep). Inadequate design or improper assembly is not a license renewal concern. Changes in preload may result in leakage from the parent component. (Leakage is not considered a failure of the pressure boundary.) Leakage will be detected during system leak test or during normal operation. (Normal operation leakage surveillances detect leakage rates below one gallon per minute.) Boric Acid Corrosion Surveillances conducted each refueling will detect very small amounts of primary coolant leakage. When leakage is detected, the source is identified and repairs will be made prior to restart after the next outage.

It is recognized that loss of bolting material could ultimately result in the loss of a component's pressure boundary integrity and thus, requires evaluation for license renewal. However, loss of material is an aging effect requiring license renewal evaluation for carbon and alloy steel components/component types subject to aging management review. As such, no evaluation separate from the subject components/component types of which bolted closures are a part is necessary and, for carbon and alloy steel components/component types, the aging management programs credited for managing external general corrosion will also inherently address their fasteners.

Reduction of fracture toughness of bolting material, due to thermal/neutron effects is a license renewal concern for the fasteners of components only due to the associated elevated system operating temperatures and proximity to the reactor vessel beltline region. Reduction of fracture toughness is not a license renewal aging effect requiring management for the fasteners of components out side the reactor vessel.

Stress corrosion cracking (SCC) of bolting materials, it is a condition in which a fastener that is statically loaded well below the material yield strength may suddenly fail. SCC bolted closure fastener failures have occurred in materials with apparently normal chemical and mechanical properties. Although there have been a few instances of cracking of bolting in the industry due to SCC, these have been attributed to high yield stress materials and contaminants, such as the use of lubricants containing MoS_2. VCSNS has not and does not use lubricants containing MoS_2. However, most bolting is normally in a dry environment and is coated with a lubricant; in general, environmental conditions that could lead to SCC of bolting are not expected to occur in non-Class 1 components. For quenched and tempered low alloy steels used for closure bolting (e.g., SA193 Grade B7), material susceptibility to SCC is minimized by having a lower yield strength. EPRI Report NP-5769 (Volume I, pg. 11-5) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified yield strength is below 150 ksi. The specification for the fabrication of nuclear piping, specifies alloy steel ASME SA 193, Class B7 bolts/studs and ASME 194 Grade 2H nuts, which have minimum yield strengths below 150 ksi (105 ksi). A minimum yield strength for bolting does not, in and of itself, preclude SCC since the actual yield strength of the bolt could be above the threshold value for SCC of low alloy steel bolting/fasteners to occur (150 ksi). However, sound maintenance bolt torquing practices can control bolting material stresses and the use of appropriate material (such as ASTM A193 Gr. B7) for bolting reduces the potential for SCC to occur. A review of industry failure databases and NRC generic communications, supports the fact that proper material selection, proper maintenance and torquing procedures, and removal of contaminants from lubricants have been effective in eliminating the potential for SCC of bolting materials. Therefore, SCC of bolting materials is not an aging effect requiring evaluation for license renewal for non-Class 1 components/component types.

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.

VCSNS Response RAI 3.1.2.4.2-2

The tables in Section 2.3 of the Application list the results of the aging management review for mechanical components. For each component type and environment combination there will be at least one entry. Entries are not made for aging effects that do not require management. If no aging effects are identified that require management,

there is one entry for that. Since Table 3.1-1, Item 20, is not an aging effect requiring management, it is not referenced in Table 2.3-2. Items in Table 3.1-1 were addressed for whether they required aging management or not. For CASS material in borated water Table 2.3-2 references Table 3.1-1, Item 10.

RAI 3.1.2.4.2-3: The austenitic stainless steel and CASS 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. Discuss the controls that are in place to ensure that all insulation used on austenitic stainless steel and CASS RCS piping is free 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."

VCSNS Response RAI 3.1.2.4.2-3

Stainless steel reflective metal insulation is the most common used insulation type on stainless steel piping and components. Various other type of insulation used on stainless steel are encapsulated in stainless steel. Unlike fiberious insulation the stainless steel does not need controls for control of halogens. More detail on Reactor Building insulation can be found in the FSAR 6.3.2.6.1.

RAI 3.1.2.4.2-4: 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 butter and 182/82 weld on RCS 'A' hot leg nozzle to pipe weld. 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 discrepancy.

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.

VCSNS Response RAI 3.1.2.4.2-4

The information and commitments made by VCSNS as a result of the crack can be found in the documentation associated with the SER transmitted by NRC letter from Karen R. Cotton to Steve Byrne dated February 20, 2001 (Reference 8).

The Information associated with the follow up actions was submitted to the NRC and an SER from Karen R. Cotton to Stephen A. Byrne was issued on October 1, 2002. (See TAC NO. MB4870, Reference 9).

VCSNS has implemented mechanical stress improvement activities for the two hot legs that were not repaired. The mechanical stress improvement activities determined the LBB report did not need to be revised. The information associated with this effort was submitted to the NRC (see TAC NO. MB4870, Reference 15).

RAI 3.1.2.4.2-5: 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.

VCSNS Response RAI 3.1.2.4.2-5

The VCSNS Chemistry Program incorporates the guidelines of EPRI TR-105714 Revision 4.

RAI 3.1.2.4.2-6: 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.

VCSNS Response RAI 3.1.2.4.2-6

Table 3.1-1, Item 6, is identified from the GALL Table 1 requirements as only Class 1 piping. Table 3.1-2, Item 6, contains other system components and non-Class 1 piping. Table 3.1-2, Item 6, includes: Thermocouple seals, RC tubing and fittings, RCP Thermal Barrier flange and non-Class 1 piping. These RC components are not Class 1 piping. The Small-Bore Class 1 Piping Inspections are not applicable to non-Class 1 piping or to non-pining components.

GALL has acknowledged chemistry programs alone are effective for many areas. For treated water and borated water systems, many component-aging effect combinations that are managed by chemistry alone are not listed in the GALL tables.

From the GALL Program Description of XI.M2 WATER CHEMISTRY:

"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."

From the GALL Section VII.E1:

"The effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated borated water and, therefore, are not included in this section."

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the item IV D1.1-c. With the approval of GALL, "loss of material due to pitting and crevice corrosion" is not an aging effect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

GALL Chapter V has accepted chemistry alone as an acceptable aging management program for "crack initiation and growth" for stainless steel components.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This position is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

<u>RAI 3.1.2.4.2-7:</u> In AMR Item 20, Table 3.1-1 of the LRA, the applicant states that the CASS components in the reactor coolant piping (elbows and accumulator nozzles) are statically cast. These components have low molybdenum content and have ferrite levels of less than 20%. Therefore, based on the screening criteria set forth in the NRC letter dated May 19, 2000, from C. Grimes (NRC) to D. Walter (NEI), these CASS components are not subject to loss of fracture toughness due to thermal aging. Explain how the ferrite content was determined.

VCSNS Response RAI 3.1.2.4.2-7

Based on the material chemistry [WCAP-13206 Revision 2] for the VCSNS Reactor Coolant loop piping, the elbows in the hot legs, cold legs, and crossover legs are not susceptible to reduction in fracture toughness due to thermal aging. The elbows have a low molybdenum content and have delta ferrite levels of less than 20%.

The material chemistry for the 45-degree accumulator nozzles was not contained in WCAP-13206. The original material certifications for the nozzle heat numbers were retrieved and indicated delta ferrite levels of less than 20%. The certifications did not specify a molybdenum content (left blank) that could be construed as trace amounts or just not documented. Based on the material specification for SA-351 CF8A, the molybdenum is a maximum of 0.50% thereby classifying the nozzles as having a low molybdenum content. Therefore, the 45-degree accumulator nozzles are not susceptible to reduction in fracture toughness due to thermal aging.

RAI 3.1.2.4.2-8: 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 reactor coolant system non-Class 1 stainless steel components exposed to a moist air environment. The staff is requesting that the applicant provide additional information so that the staff can 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.

VCSNS Response RAI 3.1.2.4.2-8

VCSNS is located well inland and is located in an area where forestry is the prime commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Samples of groundwater and rainwater show chloride and sulfate concentrations less than 10 ppm. The environment for the oil collection system components is indoors (the Reactor Building) with moist air. Moist air is not dried but it is non-condensing for the oil collection components, as well as most other components. GALL did not identify aging effects that require aging management for the stainless steel in an air environment. This position is considered consistent with the conclusions of the efforts that resulted in the publication of GALL. GALL did not identify aging effects that require management for the external surface of stainless steel components or stainless steel in a gas environment.

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. Submit a technical basis for identifying cracking as an applicable aging effect for the cladding.

VCSNS Response RAI 3.1.2.4.3-1

GALL Chapter V has accepted chemistry alone as an acceptable Aging Management Program for "crack initiation and growth" for stainless steel components. VCSNS considers cracking a valid aging effect that is managed by chemistry. This position is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

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.

VCSNS Response RAI 3.1.2.4.3-2

VCSNS CRD nozzles are seal welded to the CRDM. The pressure boundary is not a bolted connection. Bolts are used for the magnetic housings, however, they are not pressure boundary and are not in scope.

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).

VCSNS Response RAI 3.1.2.4.3-3

The Table 3.1-1, Item 28, "Component Group" description was copied from "Component" column of Table 1 of GALL. The actual components included in the grouping were selected from the "Item Number in GALL" column. The items included in Table 3.1-1, Item 28, are: Incore Flux Thimbles, Clevis Inserts, Head/Vessel Alignment Pins, Radial Keys, Upper Core Plate Alignment Pins, and Reactor Vessel Closure Head & Vessel Flanges. The Reactor Vessel Closure Studs are not included in Table 3.1-1, Item 28. Table 3.1-1, Item 28, should not be listed for the Reactor Vessel Closure Studs in Table 2.3-3.

RAI 3.1.2.4.3-4: The austenitic stainless steel and Ni-alloy based 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. Discuss the controls that are in place to ensure that all insulation used on austenitic stainless steel and Ni-alloy based reactor vessel components are free from halogens.

VCSNS Response RAI 3.1.2.4.3-4

Stainless steel reflective metal insulation is the most common used insulation type on stainless steel piping and components. Various other type of insulation used on stainless steel are encapsulated in stainless steel. Unlike fiberious insulation, the stainless steel does not need controls for control of halogens. More detail on Reactor Building insulation can be found in the FSAR 6.3.2.6.1.

RAI 3.1.2.4.3-5: LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 23, and LRA Table 3.1-2, AMR Item 11, for the AMR results for the Alloy 600 reactor vessel closure head penetration tubes. Both AMR items address cracking as an aging effect for these tubes but identify different programs for managing this effect. Item 23 recommends the Alloy 600 AMP, whereas, Item 11 recommends the chemistry program. Explain why two different programs have been identified to manage the same aging effect of cracking for the Alloy 600 reactor vessel closure head penetration tubes.

VCSNS Response RAI 3.1.2.4.3-5

The following items were evaluated for loss of material due to crevice corrosion, pitting corrosion and cracking due to stress corrosion cracking (SCC) or primary water stress corrosion cracking (PWSCC): 1) incore neutron detector (flux) thimbles bottom mounted instrumentation; 2) tube & fittings / incore neutron detector conduits; 3) manway covers insert plates - primary side; 4) PZR manway cover; and 5) RV closure head penetration tubes (CR, instrument, vent pipe). Loss of material due to crevice corrosion, pitting corrosion for all five of these items is included in Table 3.1-2, Item 7. Cracking due to PWSCC of RV closure head penetration tubes is included in Table 3.1-1, Item 23. Table 3.1-2, Item 11, does not need to include cracking due to PWSCC of RV closure head penetration tubes. Table 3.1-2, Item 11, should not be referenced in Table 2.3-3 for RV closure head penetration tubes. Table 3.1-2, Item 11, should only include stainless steel (not nickel based alloy) and cracking due to SCC (not PWSCC).

RAI 3.1.2.4.3-6: The AMR results for the PWR reactor vessel flange leak detection line (GALL Item No. IV.A2.1-f) are presented in Table 1 of GALL Volume 1 (NUREG-1801). Therefore, AMR Item 9 in LRA Table 3.1-1 should also include this AMR result. Confirm whether AMR Item 9 of LRA Table 3.1-1, includes the AMR results for PWR reactor vessel flange leak detection line. If so, identify the aging management programs for this component. If not, provide an explanation or basis for not including these AMR results.

VCSNS Response RAI 3.1.2.4.3-6

VCSNS did not include Reactor Vessel flange leak detection line in Table 3.1-1, Item 9. The Reactor Vessel flange leak detection line is included in Table 3.1-2, Items 1, 5, and 6.

The Reactor Vessel flange leak detection components are classified as Code Class 2 components and are not normally filled with borated water.

3.1.2.4.4 Reactor Vessel Internals

<u>RAI 3.1.2.4.4-1:</u> 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.

VCSNS Response RAI 3.1.2.4.4-1

VCSNS Reactor Vessel Internals Inspection Program has not yet been developed but will be developed and implemented prior to the period of extended operation. VCSNS will follow the practices that are developed from industry initiatives (specifically EPRI and WOG) and operating experience for Reactor Vessel Internals Inspection.

<u>RAI 3.1.2.4.4-2:</u> The applicant credits the Reactor Vessels Internals Inspection Program (LRA Appendix B.2.4) 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 should be applied to manage 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.

VCSNS Response RAI 3.1.2.4.4-2

VCSNS Reactor Vessel Internals Inspection Program has not yet been developed but will be developed and implemented prior to the period of extended operation. VCSNS will follow the practices that are developed from industry initiatives (specifically EPRI and WOG) and operating experience for Reactor Vessel Internals Inspection.

Loose parts monitoring and neutron noise monitoring will potentially detect problems in the vessel but only after failure. These are not valid aging management programs but potential failure detection systems.

3.1.2.4.5 In-Core Instrumentation System

RAI 3.1.2.4.5-1: With respect to in-core instrumentation bolting provide the following information:

a. In LRA Table 3.1-2, AMR Item 4 identifies stainless steel as material for in-core thermocouple seal bolting. However, in the "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.

b. 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.

c. 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.

d. 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.

VCSNS Response RAI 3.1.2.4.5-1

a. Bolting for the In-core thermocouple seal is stainless steel. It is not the intent of the discussion column to state the material type. It is intended to indicate that SCC has been identified as an aging effect for some bolting.

b. Although identified as an aging effect in various industry references, loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for component bolted closures (i.e. pressure boundary closures) within the scope of license renewal. The bolted closure was considered to be a sub-component (piece-part) of the components/component types within the scope of license renewal and did not require evaluation separate from the component.

Loss of mechanical closure integrity can result in failure of the mechanical joint, is evidenced by leakage rather than joint failure, and can be attributed to one or more of the following effects:

- Loss of bolt pre-load (embedment, cyclic load embedment, gasket creep, etc.),
- Loss of bolting material (from general and/or boric acid corrosion),
- Reduction of bolting material fracture toughness, and
- Cracking of high strength bolting material (SCC).

Loss of pre-load was considered to be the result of inadequate design or improper assembly (i.e. event driven) that is not related to aging and that would manifest itself during the current operating term and be corrected prior to the period of extended operation. This component is disassembled and reassembled each refueling. Potential loss of pre-load should have already manifested itself for this component. Thus, the mechanisms associated with loss of bolting pre-load are not a license renewal concern.

The stainless steel fasteners are immune to loss of material due to general corrosion and are normally in a dry environment limiting the concerns for pitting and crevice corrosion. Stainless steel fasteners have been shown to be immune to loss of material due to boric acid wastage.

Reduction of fracture toughness of bolting material, due to thermal/neutron effects is a license renewal concern for the fasteners of components only due to the associated elevated system operating temperatures and proximity to the reactor vessel beltline region. This is not applicable to bolting of the in-core thermocouple seal.

Stress corrosion cracking (SCC) of bolting materials, it is a condition in which a fastener that is statically loaded well below the material yield strength may suddenly fail. SCC bolted closure fastener failures have occurred in materials with apparently normal chemical and mechanical properties. Although there have been a few instances of cracking of bolting in the industry due to SCC, these have been attributed to high yield stress materials and contaminants, such as the use of lubricants containing MoS_2. VCSNS has not and does not use lubricants containing MoS_2.

c) The connection between the incore instrument thimble and the guide tube is made with the standard Westinghouse style tube fitting arrangement. It is not a bolted connection.

d) Loss of pre-load was considered to be the result of inadequate design or improper assembly (i.e. event driven) that is not related to aging and that would manifest itself during the current operating term and be corrected prior to the period of extended operation. This component is disassembled and reassembled each refueling. At each refueling the bolts are inspected. Potential loss of pre-load should have already manifested itself for this component. Thus, the

mechanisms associated with loss of bolting pre-load are not a license renew al concern.

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.

VCSNS Response RAI 3.1.2.4.6-1

Pressurizer seismic lugs are included with the Pressurizer shell and are not called out as a separate component. Immersion Heater Well Assemblies is the component name used at VCSNS for the heater sheaths and they are included in Table 3.1-1, Item 24, and Table 3.1-2, Items 1 and 7. The manway pad gasket-seating surface is the stainless steel clad mating surface of the manway nozzle and is not called out as a separate component. Pressurizer safety and relief valves are included in Table 3.1-1, Items 19 and 24, and Table 3.1-2, Items 1 and 5.

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.

VCSNS Response RAI 3.1.2.4.6-2

The surge nozzle, spray nozzle, safety nozzle, relief nozzle nozzles do have safe end and they are included in the Application Table 2.3-6 as well as the instrument nozzles.

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?

Response RAI 3.1.2.4.6-3

Application Table 2.3-6, Item 4, manway cover, is a non-structural stainless steel insert that is used to prevent the borated water from contacting the manway forgings. The manway forging, included in Application Table 2.3-6, Item 7, is a carbon steel structural element that is the ASME pressure boundary that supports the stainless cover (Item 4)

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.

VCSNS Response RAI 3.1.2.4.6-4

VCSNS does not consider these aging mechanisms requiring management for these components. The components are erosion resistant material and are not normally exposed to high velocity flow or two phase flow. The Pressurizer pressure control system is operated in a manner to reduce the spray flow to a minimal amount. Spray flow is normally several gallons a minute verses the 750 gallons a minute design flow. This low flow rate into the Pressurizer through these 4-inch and 14-inch nozzle safe ends and associated thermal sleeves is insignificant to require erosion evaluations.

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.

VCSNS Response RAI 3.1.2.4.6-5

The support bracket for the heater supports are welded to the Pressurizer clad. These materials are not sensitized, see FSAR 5.2.5.5, and no special AMP is required for the welded attachments.

RAI 3.1.2.4.6-6: 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.

VCSNS Response RAI 3.1.2.4.6-6

The Pressurizer shell design considers fatigue usage. The cladding and the welds that attach internal items to the Pressurizer cladding are not sensitized, the location that is most likely to experience cracking by thermal fatigue is the connection of the surge line nozzle to the Pressurizer shell. The Pressurizer surge line nozzles are the limiting Pressurizer location from a fatigue usage perspective, and are included in the ASME Section XI inspection plan. If cracking were to occur at the surface of the surge nozzle cladding and propagate to the base metal, volumetric examinations performed in accordance with ASME Section XI examination would detect the flaw prior to loss of the Pressurizer intended function. A specific aging management program to address cracking of the Pressurizer cladding due to fatigue is not required.

RAI 3.1.2.4.6-7: 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.

VCSNS Response RAI 3.1.2.4.6-7

Loss of pre-load is considered to be the result of inadequate design or improper assembly (i.e. event driven) or time dependant changes in stresses (e.g. creep). Inadequate design or improper assembly is not a license renewal concern. Changes in preload may result in leakage from the parent component. (Leakage is not considered a failure of the pressure boundary.) Leakage will be detected during system leak test or during normal operation. (Normal operation leakage surveillances detect leakage rates below one gallon per minute.) Boric Acid Corrosion Surveillances conducted each refueling will detect very small amounts of primary coolant leakage. When leakage is detected the source is identified and repairs will be made .

3.1.2.4.7 Steam Generators

RAI 3.1.2.4.7-1: In LRA Table 3.1-1, AMR Item 9, the applicant addresses possible crack initiation and growth due to SCC and PWSCC in various nickel-based alloy components, including nozzles for the steam generator instrument and drain lines. In the discussion section of AMR Item 9, the applicant did not include the steam generator instrument and drain lines in the component group for aging management. The applicant implies that the SCC and PWSCC aging effects are not applicable to the steam generator instrument and drain lines in VCSNS.

As detailed in NUREG-1801, Table IV.D1 Item D1.1-j, a plant-specific aging management program is to be applied to pressure boundary and structural instrument nozzles that are fabricated with Alloy 600.

a. The staff is not clear whether the instrument and drain lines in VCSNS steam generators are made of Alloy 600. If the lines are not made of Alloy 600, SCC and PWSCC in GALL Table IV.D1.1-j may not be applicable; however, the applicant needs to identify appropriate aging effect(s) and AMP(s) to manage the material that was used in the fabrication of the instrument and drain lines. Clarify why steam generator instrument and drain nozzles are not included in the aging management in AMR Item 9 in Table 3.1-1 of the LRA.

b. If Inconel 82/182 weld metal is used in the nozzle welds, then these welds are susceptible to crack initiation and growth due to PWSCC. Confirm whether these welds at VCSNS are made of Alloy 82/182, and if so, provide a program for managing cracking in these welds due to PWSCC.

VCSNS Response RAI 3.1.2.4.7-1

At VCSNS only the core support pads and bottom head penetration are include in Table 3.1-1, AMR Item 9. The primary side of the steam generators does not have Alloy 600 or Alloy 82/182 weld material exposed to borated water other than the cladding of the primary side of the tube sheet. The secondary side components do not use Alloy 600 or Alloy 82/182 weld material.

<u>RAI 3.1.2.4.7-2:</u> In LRA Table 3.1-1, AMR Item 15, the applicant addresses the potential aging effects associated with steam generator tubes, sleeves, and plugs. In GALL Item IV.D.1.2-a, -b, -c, -d, -e, -f, -g, -h, -i, -j, and -k, the staff identified the following aging effects associated with tubes, plugs and sleeves: (1) crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking, and intergranular attack; (2) loss of material due to wastage and pitting corrosion; (3) deformation due to corrosion at tube support plate intersections; (4) loss of section thickness due to fretting and wear; (5) denting due to corrosion of carbon steel tube support plate; (6) loss of section thickness due to flow-accelerated corrosion; and (7) ligament cracking due to corrosion. Clarify which aging effects in the above GALL IV.D.1.2 component group is applicable to the VCSNS steam generator tubes, sleeves and plugs. For those aging effects that are not applicable to the VCSNS steam generators, provide technical justification.

VCSNS Response RAI 3.1.2.4.7-2

The VCSNS Replacement Steam Generators incorporated many lessons learned from years of experience with U-tube Steam Generators. This information is reflected in the information supplied to the NRC for the Steam Generator Replacement Project at VCSNS. One source of information is WCAP-13480 Rev. 1.

The secondary side components do not use Alloy 600 or Alloy 82/182 weld material. Thermally treated Alloy 690 tubes are used in the Delta 75 Replacement Steam Generators. Most of the features of the Model F Steam Generators are incorporated

directly into the Delta 75 Replacement Steam Generator. Broached 405 stainless steel tube support plates, thermally treated Alloy 690 tubes, hydraulically expanded tubes, and reduced U-bend gaps are carried over from Model F Steam Generators. Several enhancements over the Model F Steam Generators were made to the anti-vibration bars.

Thermally treated Alloy 690 tubes are used to eliminate concerns of PWSCC and intergranular attack. The enhancements made to the anti-vibration bars and broached 405 stainless steel tube support plates were designed to prevent fretting and wear. Phosphate chemistry control has not been used at VCSNS thus eliminating the associated concerns for wastage and pitting. Stainless steel tube support plates remove concerns for denting. Alloy 690 tube plugs have been used on the tubes that have been plugged. Alloy 600 tube plugs will not be used.

<u>RAI 3.1.2.4.7-3:</u> In LRA Table 3.1-1, AMR Item 17, the applicant addresses possible ligament cracking due to corrosion in the steam generator carbon steel tube support plate. The GALL report addresses ligament cracking due to corrosion of carbon steel steam generator tube supports. The applicant states that the tube supports at VCSNS are made of 405 stainless steel and the alloy 690 anti-vibration bars are included in this component group. The applicant states that the chemistry program alone is sufficient to manage cracking in the tube supports and anti-vibration bars. In GALL IV D.1.1-i, the staff specifies the inservice inspection program and water chemistry program to manage cracking of stainless steel components. Also, in GALL IV D.1.2-i and j, the staff specifies the inservice inspection program and water chemistry program to manage cracking of Alloy 690 components.

a. Provide the technical basis for the conclusion that the water chemistry program by itself is sufficient to manage cracking in tube supports and anti-vibration bars in light of the recommended inservice inspection program in the GALL report.

b. Discuss whether tube support plates and anti-vibration bars were inspected during refueling outage 12. If inspected, what were the inspection results. (See RAI 3.1.2.3.7-2).

VCSNS Response RAI 3.1.2.4.7-3

The Table 3.1-1, Item 16, incorrectly list the anti-vibration bar material as 690 instead of the actual 405 stainless steel. Ligament cracking due to corrosion of carbon steel steam generator tube supports is an aging effect for CE steam generators only.

The tube support plates and anti-vibration bars were inspected during RF-12 and no abnormal indications were noted.

<u>RAI 3.1.2.4.7-4:</u> In LRA Table 3.1-1, AMR Item 21, the applicant addresses potential wall thinning due to flow-assisted corrosion for the steam generator steam and feedwater nozzles and safe ends. The applicant concludes that this aging effect is not applicable to these components at VCSNS; therefore, aging management for this effect is not required, but provides no justification for this conclusion. As detailed in NUREG-1801, Table IV. D1, Item

D1.1-d, steam and feedwater nozzles and safe ends are managed under the flow accelerated corrosion program. Provide the technical basis for concluding that wall thinning due to flow-accelerated corrosion is not an applicable aging effect and that no aging management program is needed for these nozzles and safe ends.

VCSNS Response RAI 3.1.2.4.7-4

The main steam exiting the steam generators is dry (less than 0.1% moisture). Dry steam is not a concern for flow-accelerated corrosion (FAC).

FAC is not a concern for the steam generator nozzle for emergency feedwater (auxiliary feedwater). The liquid that enters the steam generator through this nozzle is significantly subcooled; heated water is not injected through this nozzle.

A review of operating experience at VCSNS has concluded the only location in the feedwater system that experience significant flow-accelerated corrosion is the piping on the outlet of the feedwater flow control valves. At full load conditions the Alloy 690 feedring nozzles provide a backpressure for the inlet nozzle and thus limit the possibility of two-phase flow into the steam generator.

<u>RAI 3.1.2.4.7-5:</u> In LRA Table 3.1-1, AMR Item 22, the applicant states that for steam generator class 2 bolting, loss of closure integrity rather than loss of preload or cracking is the aging effect requiring management and that the additional aging mechanism of stress relaxation is being managed. In GALL IV.D1.1-f, the staff identifies the aging effect of loss of preload due to stress relaxation for secondary manway and handhole bolting, for which the staff identifies the bolting integrity program as the AMP.

a. Explain the difference between loss of closure integrity and loss of preload. Explain how the management of stress relaxation differs from the management of loss of preload for bolting.

b. Confirm whether bolt retorquing is performed at VCSNS as a part of its maintenance activities and, if so, why it is not credited for managing loss of preload in bolting.

VCSNS Response RAI 3.1.2.4.7-5

Loss of pre-load is considered to be the result of inadequate design or improper assembly (i.e. event driven) or time dependant changes in stresses (e.g. creep). Inadequate design or improper assembly is not license renewal concerns. Changes in preload may result in leakage from the parent component. (Leakage is not considered a failure of the pressure boundary.) Leakage will be detected during system leak test or during normal operation. (Normal operation leakage surveillances detect leakage rates below one gallon per minute.) Boric Acid Corrosion Surveillances conducted each refueling will detect very small amounts of primary coolant leakage. When leakage is detected the source is identified and repairs will be made prior to restart after the next outage.

Bolting is torqued when components are reassembled after they are disassembled for inspections. The steam generator secondary bolted connections are disassembled for scheduled inspection activities. Retorquing of bolts is not a routine activity by itself.

RAI 3.1.2.4.7-6: In LRA Table 3.1-1, AMR Item 27, the applicant addresses loss of material due to erosion as a potential aging effect for steam generator secondary manways and handholds [handholes] that are made of carbon steel. The applicant states that "...VCSNS has Westinghouse recirculating steam generators..." without further elaboration.

a. Clarify whether there are any secondary manways and handholes in the VCSNS recirculating steam generators and if there are, specify whether loss of material due to erosion or any potential degradation may be applicable.

b. If the VCSNS replacement steam generators have no secondary side manways and handholes, discuss how the inservice inspection of the secondary side of the steam generators is performed (e.g., how is the inspection equipment accessed to the secondary side?)

VCSNS Response RAI 3.1.2.4.7-6

The referenced item in GALL is D2.1-I. D2 items are once-through (B&W) Steam Generators. Recirculating Steam Generators do not have this item identified in GALL and no industry or plant-operating experience has identified any erosion issues with secondary manways or handholes. The VCSNS Steam Generators do have these components and they are included in Table 3.1-1 item 26 and 3.1-2 item 3.

RAI 3.1.2.4.7-7: In LRA Table 3.1-1, AMR Item 32, the applicant identifies the aging effect of crack initiation and growth due to SCC and PWSCC for the channel head divider plate in the VCSNS steam generators. In LRA Table 3.1-2, AMR Item 7, the applicant identifies the aging effect of loss of material due to crevice and pitting corrosion for the same component. The applicant states that the cracking aging effect is managed by the in-service inspection plan (LRA Appendix B.1.7) and the chemistry program (LRA Appendix B.1.4), and the loss-of-material aging effect is managed by the chemistry program.

a. Confirm whether the weld on the divider plate is an Alloy 82/182 weld. Discuss industry and VCSNS operating experience for the channel head divider plate with respect to possible SCC and PWSCC of the welds joining the divider plate to the tubesheet and lower head.

b. If the weld metal used on the divider plate is Alloy 82/182, discuss whether the Alloy 600 aging management program (LRA Appendix B.1.1) includes managing of cracking of these welds due to PWSCC and SCC.

c. Clarify whether the in-service inspection plan includes the examination of the weld on the divider plate which includes the weld between the plate and tubesheet, and the plate and the steam generator lower head.

VCSNS Response RAI 3.1.2.4.7-7
The divider plate is welded with Alloy 82/182, however the final pass was made with Alloy 52/152 so the weld does not have 82/182 exposed to borated water. VCSNS has seen no evidence of cracking on these welds since the installation of the Replacement Steam Generators in the 1994 outage. The plate welds are in the ISI Program.

RAI 3.1.2.4.7-8: In LRA Table 3.1-2, the applicant identifies a number of combinations of components and aging effects that are not included in NUREG-1801 (See Table 3.1-2, AMR Items 3, 7, 8, 9, 11, and 13). To ensure that all of the applicable aging effects are identified, the applicant needs to provide industry-operating experience for these components exposed to their respective environments.

VCSNS Response RAI 3.1.2.4.7-8
The effects of pitting and crevice corrosion can be controlled on many materials with a proper chemically controlled environment. Similarly cracking may be controlled in a proper chemically controlled environment.

GALL has acknowledged the Chemistry Program alone is effective for many areas. For treated water and borated water systems many component-aging effect combinations that are managed by chemistry alone are not listed in the GALL tables.

> **From the GALL Program Description of XI.M2 WATER CHEMISTRY:**
>
> ***"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."***
>
> **From the GALL Section VII E1:**
>
> **"The effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated borated water and, therefore, are not included in this section."**

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the Item IV D1.1-c. Based on approval of GALL, the Staff has determined, "loss of material due to pitting and crevice corrosion" is

not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

GALL Chapter V has accepted chemistry alone as an acceptable aging management program for "crack initiation and growth" for stainless steel components.

From the GALL Section VIII B1 Items B1.1-a and B1.2-a list loss of material due to pitting and crevice corrosion of carbon steel components. These items have only the Chemistry Program listed as the aging management program.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

RAI 3.1.2.4.7-9: In LRA Table 3.1-2, AMR Item 3, the applicant discusses possible loss of material due to crevice, general, pitting, and galvanic corrosion in carbon steel steam generator components (components were not identified).

a. The applicant notes that these components are not specifically addressed in Chapter IV of NUREG-1801 and states that the chemistry program (LRA appendix B.1.4) alone provides adequate management for these aging effects. Discuss why inservice inspection program or other relevant AMPs are not considered to monitor the aging effects for these components. Provide a condition monitoring program to verify the effectiveness of the chemistry program in mitigating loss of material in these components due to various corrosion mechanisms.

b. Discuss industry-operating experience with the components that are covered in this AMR Item, and confirm whether these components were included in the 100% inspection of the steam generator secondary side during refueling outage 12 as discussed in LRA Appendix B.1.10.

VCSNS Response RAI 3.1.2.4.7-9

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the Item IV D1.1-c. With the approval of GALL, the Staff has determined "loss of material due to pitting and crevice corrosion" is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

From the GALL Program Description of XI.M2 WATER CHEMISTRY:

"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of

aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."

Steam generator components do not have "low flow or stagnant areas" to cause concerns.

Additionally GALL Section VIII B1 Items B1.1-a and B1.2-a list loss of material due to pitting and crevice corrosion of carbon steel components. These items have only the Chemistry Program listed as the aging management program.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

RAI 3.1.2.4.7-10: In LRA Table 3.1-2, AMR Item 8, the applicant discusses possible loss of material due to crevice and pitting corrosion in steam generator secondary-side thermal sleeves and the steam flow limiter that are fabricated with nickel based alloy, including thermally treated Alloy 690. The applicant notes that these components are not specifically addressed in Chapter IV of NUREG-1801 and states that the chemistry program (LRA Appendix B.1.4) will manage the aging effects of these components. However, the staff notes that the applicant did not consider the inservice inspection program to manage the potential aging effect for the components. The chemistry program can only mitigate or prevent the loss of material, but cannot monitor the component integrity should degradation occur. Discuss how the loss of material due to crevice and pitting corrosion of the components can be monitored without an inservice inspection program.

Response RAI 3.1.2.4.7-10

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the Item IV D1.1-c. With the approval of GALL, the Staff has determined "loss of material due to pitting and crevice corrosion" is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

From the GALL Program Description of XI.M2 WATER CHEMISTRY:

"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which

> ***the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."***

Steam generator components do not have "low flow or stagnant areas" to cause concerns.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for aging management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

RAI 3.1.2.4.7-11: In LRA Table 3.1-2, AMR Item 9, the applicant discusses possible crack initiation and growth due to SCC and flaw growth in steam generator secondary-side thermal sleeves and the steam flow limiter fabricated with nickel based alloy, including thermally treated Alloy 690. The applicant notes that these components are not specifically addressed in Chapter IV of NUREG-1801 and states that the chemistry program (LRA Appendix B.1.4) and in-service inspection plan (LRA Appendix B.1.7) will manage the aging effects of these components. However, the staff notes that ASME Section XI, Subsections IWB or IWC do not explicitly include inspection of some of the secondary side components and the attachment welds for these components. Confirm whether the in-service inspection plan include inspection of all the components in this AMR Item and the attachment welds for these components. If not, provide a condition-monitoring program for managing cracking in these components and associated attachment welds.

VCSNS Response RAI 3.1.2.4.7-11

GALL Chapter IV does not list "crack initiation and growth due to SCC and flaw growth" in steam generator secondary components for Westinghouse PWRs other than the Item IV D1.1-c. With the approval of GALL, the Staff has determined this is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

RAI 3.1.2.4.7-12: in LRA Table 3.1-2, AMR Item 10, the applicant discusses possible loss of material due to crevice and pitting corrosion and possible crack initiation and growth due to SCC in the feedwater distribution components (pipe and fittings) that are fabricated with alloy steel (chrome molybdenum). The applicant states that the chemistry program (LRA Appendix B.1.4) and inservice inspection plan (Appendix B.1.7) provide adequate management for these aging effects. Discuss industry and VCSNS experience with these components and state whether these components were included in the 100% inspection of the steam generator secondary side during refueling outage 12.

VCSNS Response RAI 3.1.2.4.7-12

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the Item IV D1.1-c. With the approval of GALL, the Staff has determined "loss of material due to pitting and crevice corrosion" is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

> **From the GALL Program Description of XI.M2 WATER CHEMISTRY:**
>
> ***"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."***

Steam generator components do not have "low flow or stagnant areas" to cause concerns.

GALL Chapter IV does not list "crack initiation and growth due to SCC and flaw growth" in steam generator secondary components for Westinghouse PWRs other than the Item IV D1.1-c. Based on approval of GALL, the Staff has determined this is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

The feedwater distribution components were inspected during RF-12 and no abnormal indications were noted.

RAI 3.1.2.4.7-13: In Table 3.1-2, AMR Item 13, the applicant discusses loss of material due to crevice and pitting corrosion and crack initiation and growth due to SCC in the components that are fabricated with stainless steel and nickel-based alloys, including tube support plates, anti-vibration bars, and flow-distribution baffle. The applicant states that the chemistry program alone (LRA Appendix B.1.4) provides adequate management for these aging effects.

a. Discuss why an inservice inspection plan is not considered to monitor the potential aging effects of cracking and loss of material on these components.

b. Discuss industry and VCSNS experience with these components.

c. Discuss whether these components were included in the inspection of the steam generator secondary side during refueling outage 12. If not, discuss a condition-monitoring program to confirm that no loss of material or cracking is occurring in these components.

VCSNS Response RAI 3.1.2.4.7-13

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the Item IV D1.1-c. With the approval of GALL, the Staff has determined "loss of material due to pitting and crevice corrosion" is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

From the GALL Program Description of XI.M2 WATER CHEMISTRY:

"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."

Steam generator components do not have "low flow or stagnant areas" to cause concerns.

GALL Chapter IV does not list "crack initiation and growth due to SCC and flaw growth" in steam generator secondary components for Westinghouse PWRs other than the item IV D1.1-c. With the approval of GALL, the Staff has determined this is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

VCSNS has chosen to list these component-aging effect combinations where chemistry alone is credited for Aging Management in Table 3.1-2. This is consistent with the operating experience reviews conducted for the development of GALL as well as the operating experience reviews conducted at VCSNS.

The accessible portions of these components were inspected during RF-12 and no abnormal indications were noted.

Time-Limited Aging Analyses

4.2 Reactor Vessel Neutron Embrittlement

4.2.2.1 Upper Shelf Energy

RAI 4.2.2.1-1: Submit a table of the V.C. Summer 60 year EOL USE values for each of the beltline materials. Tabulate the heat numbers, material ID, copper values, Initial USE, the EOL 1/4 T fluence, and the EOL 1/4 T USE. Discuss how surveillance capsule results were evaluated in your determination of the USE values.

VCSNS Response RAI 4.2.2.1-1

Information on the specimens from capsule W was submitted to the Staff in a letter (RC-98-0185, Reference 10), G. J. Taylor, SCE&G, to NRC Document Control Desk, dated October 9, 1998, "Reactor Vessel Surveillance Program". Attached to the letter was WCAP-15101, "Analysis of Capsule W From the South Carolina Electric & Gas Company V.C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," September 1998. This WCAP contains the data from the test results of capsule W that was removed after 10.78 Effective Full Power Years (EFPY) with a lead factor of 3.40.

The End of Life (EOL) for VCSNS is 54 EFPY. The 54 EFPY value allows for a lifetime capacity factor of 90%. The 54 EFPY fluence values can be found in WCAP-15101. Section 6 of the WCAP provides the radiation analysis and neutron dosimetry information. Table 6-14 of the WCAP provides both calculated and best estimate fluence values. Each set of values is derived using E > 1.0 MeV slope method and dpa slope method. Values are listed for 0, 15, 30, and 45 degree locations and for the surface ¼ thickness and ¾ thickness. The highest value listed is 6.40×10^{19} n/cm^2 (also written as 6.40E+19 n/cm^2). This value is the calculated value at the surface of 0 degrees.

From the WCAP the measured upper shelf energy at a fluence of 4.664E+19 n/cm^2 for the limiting plate is 126 ft-lb in the longitudinal direction and 74 ft-lb in the transverse

direction, for the weld material is 87 ft-lb, and for the HAZ is 103 ft-lb. The fluence value of 4.664E+19 n/cm^2 corresponds to over 36 EFPY.

The highest percent of copper (Cu) in the weld material is 0.05%. The base metal has 0.10% Cu in the intermediate shell plate which is the limiting plate material. Using the Regulatory Guide 1.99 Figure 2 curves, with these Cu values and an EOL fluence of 6.40E19 n/cm^2 the predicted decrease in USE is estimated to be 31%. This would reduce the upper shelf energy for the limiting plate from the unirradiated values of 132 ft-lbs in the longitudinal direction and 75 ft-lbs in the transverse direction to EOL values 91 ft-lbs and 51.75 ft-lbs respectively. The weld material is reduced from 91 ft-lbs to 62 ft-lbs. The HAZ is reduced from an initial 130 ft-lbs to 90ft-lbs. These USE values are greater than the required 10 CFR 50 Appendix G requirements of 50 ft-lbs.

Test values have shown the reduction of USE to be significantly less than the Regulatory Guide 1.99 predictions.

During the next refueling outage the remaining capsules will be removed from the vessel. One of these capsules will be tested and will provide bounding data for the EOL fluence of 54 EFPY.

4.2.2.2 Pressurized Thermal Shock

RAI 4.2.2.2-1: Submit a table of the V.C. Summer 60 year EOL RT_{PTS} values for each of the beltline materials. Tabulate the heat numbers, material ID, copper and nickel values, chemistry factor, Initial RT_{NDT}, margin, EOL peak fluence, fluence factor, delta RT_{PTS}, and EOL RT_{PTS}. Discuss how surveillance capsule results were applied in your determination of the RT_{PTS}. values.

VCSNS Response RAI 4.2.2.2-1

Information on the specimens from capsule W was submitted to the Staff in a letter (RC-98-0185, Reference 10), G. J. Taylor, SCE&G, to NRC Document Control Desk, dated October 9, 1998, "Reactor Vessel Surveillance Program". Attached to the letter was WCAP-15101, "Analysis of Capsule W From the South Carolina Electric & Gas Company V.C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," September 1998 and WCAP-15103, "Evaluation of Pressurized Thermal Shock for V. C. Summer Unit 1" September 1998. These WCAPs contain the data from the test results of capsule W that was remove after 10.78 Effective Full Power Years (EFPY) with a lead factor of 3.40.

The End of Life (EOL) for VCSNS is 54 EFPY. The 54 EFPY value allows for a lifetime capacity factor of 90%. The 54 EFPY fluence values can be found in WCAP-15101. Section 6 of the WCAP provides the radiation analysis and neutron dosimetry information. Table 6-14 of the WCAP provides both calculated and best estimate fluence values. Each set of values is derived using E > 1.0 MeV slope method and dpa slope method. Values are listed for 0, 15, 30, and 45 degree locations and for the surface ¼ thickness and ¾ thickness. The highest value listed is 6.40 x 10^{19} n/cm^2 (also written as 6.40E+19 n/cm^2). This value is the calculated value at the surface of 0 degrees. When

fluence is used in the equations, as the fluence factor (f), it is in the value at 10^{19} n/cm^2 or 6.40.

From the WCAP the measured RT_{PTS} at a fluence of 4.664E+19 n/cm^2 for the limiting plate is 44°F in the longitudinal direction and 86°F in the transverse direction, for the weld material is -10°F, and for the HAZ is -33°F. The initial, unirradiated, values for these materials were -22°F, 28°F, -53°F, and -93°F respectively. he as-tested changes (ΔRT_{NDT}) for these materials were 66°F, 58°F, 43°F, and 60°F, respectively. The fluence value of 4.664E+19 n/cm^2 corresponds to over 36 EFPY.

The chemical composition values for copper and nickel (as well as several other elements) can be found in WCAP-15101 Table 4-1. The highest percent of copper (Cu) in the weld material is 0.05%. The highest percent nickel (Ni) in the weld material is 0.95%. The base metal has 0.10% Cu and 0.51 % Ni in the intermediate shell plate which is the limiting plate material. These values are used to obtain a chemistry factor (CF). CF is given in 10 CFR Part 50.61. Table 1 provides the 68°F value for weld material. Table 2 provides the 65°F base metal value (plates and forgings). WCAP-15103 Table 5 and 6 also list 68°F for weld material of various seams and the 65°F intermediate shell plate.

From WCAP-15103 Table 5 and 6, the margin value (M) for the intermediate shell plate is 34°F and for the weld material is 56°F.

The initial reference temperature for the material in the pre-service or unirradiated condition is found in WCAP-15103. From Tables 5 and 6 of the WCAP, for the intermediate shell plate the initial value is 30°F and the initial value for the weld material is -44°F.

**The projected RT_{PTS} for the shell material at the 54 EFPY EOL value is 158.1°F.
The projected RT_{PTS} for the weld material at the 54 EFPY EOL value is 110.5°F.**

During the next refueling outage the remaining capsules will be removed from the vessel. One of these capsules will be tested and will provide bounding data for the EOL fluence of 54 EFPY.

4.2.2.3 Pressure-Temperature Limits

<u>RAI 4.2.2.3-1:</u> The staff requests that the applicant identify LTOP as part of the reactor vessel neutron embrittlement time-limited aging analysis, and commit to develop LTOP values for the period of extended operation, as was done for the P-T limits.

VCSNS Response RAI 4.2.2.3-1

At VCSNS, the Low Temperature Overpressure Protection (LTOP) analysis is part of the calculation that develops the heatup and cooldown curves from analysis of the Reactor Vessel specimens. Section 4.2.3 of the Application identifies pressure-temperature limits as a time-limited aging analysis and commits VCSNS to revising the calculated

value of Adjusted Reference Temperature (ART) and associated pressure-temperature limits for heatup and cooldown when one of the two remaining surveillance capsules is removed and analyzed. The LTOP analysis will be done as part of this calculation revision.

Other TLAAs

4.7.1 Reactor Coolant Pump Flywheel

RAI 4.7.1-1: Discuss how WCAP-14535 is applicable to the VCSNS reactor coolant pump flywheel. Indicate what material was used to fabricate the flywheel. Does the VCSNS flywheel belong to a specific flywheel group as defined by WCAP-14535? Has VCSNS submitted, for staff review, its assessment of the plant-specific applicability of WCAP-14535 for V.C. Summer? If so, please provide the applicable references. What is the inspection frequency for the flywheels?

VCSNS Response RAI 4.7.1-1

WGAP-14535A has not been previously docketed by VCSNS. The generic report has been docketed by Westinghouse and it is applicable to VCSNS. The VCSNS Reactor Coolant Pump (RCP) flywheels are included in group 3 of the WCAP classification. VCSNS has conservatively treated the RCP flywheel as a TLAA due to the fatigue issues of this component, the WCAP, and previous applicants' submittals. VCSNS does not intend to eliminate the current 10-year inspection through the license renewal submittal. The intent of the submittal was to show WCAP-14535A and plant operations support a 60-year life of the RCP flywheel.

4.7.2 Leak-Before-Break

RAI 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 the industry's PWR Materials Reliability Program (MRP) and received an interim report from the MRP, "PWR Materials Reliability Program, 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 V.C. Summer LRA, the NRC staff requests that the applicant provide a licensee commitment which states that for the period of extended operation of V.C. Summer, the applicant will implement actions or perform analyses, as deemed to be necessary by the NRC, to confirm continued applicability of existing V.C. 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.

b. 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 Response RAI 4.7.2-1

The commitments made by VCSNS in response to the PWSCC crack of the RCS piping will continue into the period of extended operation.

VCSNS has implemented mechanical stress improvement activities for the two hot legs that were not repaired. The information associated with this effort was submitted to the NRC (Reference 15), and an SER dated October 1, 2002 Karen R. Cotton to Stephen A. Byrne was issued (TAC NO. MB4870, Reference 9). The mechanical stress improvement activities determined the LBB report did not need to be revised.

Aging Management Programs (System Specific)

B.1.1 Alloy 600 Aging Management Program

RAI B.1.1-1: With respect to inspections of vessel head and vessel head penetration provide the following information:

a. 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.

b. 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.

c. 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.

d. IN 2000-17, Supplement 2, "Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V. C. Summer," it was mentioned that the licensee intended to make several enhancements to their leak detection capability including: 1) performing noble gas sampling, 2) performing a reactor coolant system water inventory balance once per day, 3) addition of a main control board annunciator for a 0.75 gallon per minute leak, and 4) revising the procedures for boric acid inspections to list specific components and locations to be inspected and to provide specific guidance on evaluation methodologies. Explain why VCSNS is not crediting this enhanced leak detection capability for managing cracking due to PWSCC in Alloy 82/182 welds and Alloy 600 base metal.

VCSNS Response RAI B.1.1-1

Specific information on the boric acid leakage inspections can be found in a letter from Stephen A. Byrne to the Document Control Desk dated January 24, 2003 (RC-03-0016, Reference 11). The subject of this letter is "Response for Additional Information Regarding 60 Day Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." VT-2 inspections are utilized for boric acid inspections. When leaks are found the affected components are evaluated for impact and corrective actions are implemented as appropriate. Boric acid residue is removed.

An inspection of the vessel head was conducted during RF-13. This was a remote visual examination of the area between the reactor vessel head insulation and the reactor vessel head. Specific information on the head inspection can be found in a letter from Stephen A. Byrne to the Document Control Desk dated July 3, 2002 (RC-02-0115, Reference 12). The subject of this letter is "Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

A "bare metal visual inspection" of the reactor head using the guidance of the EPRI MRP is scheduled for RF-14. This commitment is contained in the VCSNS September 12, 2002 (Reference 13) response to NRC Bulletin 2002-02. This letter also justifies that boric acid leakage inspections are capable of detecting very small leak rates. A leak rate of 10^{-3} gpm will result in release of about 500 in^3 of boric acid in an 18-month fuel cycle, which will be detectable by visual inspections. Industry experience confirms that indications of Alloy 600 PWSCC crack formation by means of observed primary coolant leakage, and/or surveillance of boric acid residue in the vicinity of affected vessel head penetrations, provide adequate opportunity to detect Alloy 600 PWSCC before cracks reach critical

length. VCSNS experience with leaks is consistent with the assumption that boric acid leakage inspection will detect leaks more reliably than leak rate surveillances.

The inspection of the underside of the vessel head, referenced to in the response to NRC Bulletin 2001-1, was conducted using a remotely operated camera. The intent of this VT-3 inspection was to detect any significant indication of cracking or loss of material.

RAI B.1.1-2: 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.

VCSNS Response RAI B.1.1-2

Industry experience confirms that indications of Alloy 600 PWSCC crack formation by means of observed primary coolant leakage, and/or surveillance of boric acid residue in the vicinity of affected vessel head penetrations, provide adequate opportunity to detect Alloy 600 PWSCC before cracks reach critical length. VCSNS discovered a crack on the 'A' hot leg nozzle at the beginning of RF-12 (October 2000), when boric acid was found on the floor of the containment building. The VCSNS crack was the subject of NRC Information Notice 2000-17, Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V. C. Summer.

RAI B.1.1-3: 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.

VCSNS Response RAI B.1.1-3

An inspection of the vessel head was conducted during RF-13. This was a remote visual examination of the area between the reactor vessel head insulation and the reactor vessel head. Specific information on the head inspection can be found in a letter from Stephen A. Byrne to the Document Control Desk dated July 3, 2002 (Reference 12). The subject of this letter is "Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

RAI B.1.1-4: 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.

VCSNS Response RAI B.1.1-4

VCSNS is participating in the industry program for managing PWSCC type aging on vessel head penetrations.

RAI B.1.1-5: 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.

VCSNS Response RAI B.1.1-5

The Alloy 600 Aging Management Program is utilized for aging management for components in addition to the vessel head penetrations. The program at VCSNS is expanded to include other components susceptible to PWSCC (i.e. Alloy 600) rather than establishing an additional program. The program includes aspects of the Chemistry Program and In-Service Inspection Plan.

The components managed by this program are:

- **PZR Nozzle-Safe End Weld Metal,**
- **Steam Generator Primary Side Tubeplate,**
- **Reactor Vessel Bottom Head Penetration Tubes,**
- **Reactor Vessel Closure Head Penetration Tubes (CR, Instrument, Vent Pipe),**
- **Reactor Vessel Inlet & Outlet Nozzle Safe Ends**

The FSAR revisions in Appendix A do not need to be revised. These FSAR descriptions do not reference GALL AMP XI.M11.

B.1.3 Bottom-Mounted Instrumentation Inspection

RAI B.1.3-1: 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 (ECT). 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.

VCSNS Response RAI B.1.3-1

The full length of each tube is inspected. The results of the RF-4 ECT were provided to the NRC in a letter from O. S. Bradham to the Document Control Desk dated November 4,

1988 (Reference 14). The detected wear is occurring at the core plate or fuel assembly bottom nozzle area.

RAI B.1.3-2: 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.

VCSNS Response RAI B.1.3-2

Wear Rate Formula that is used to project future percent through wall wear was developed by Westinghouse. This formula is: $W_a = W_d (N_a / N_d)^n$. Where W_a = % wear at future time, W_d = % wear at time of measurement, N_a = accumulated time to date of projected wear in cycles of operation, N_d = accumulated time to date of wear measurement in cycles and n = slope exponent of wear rate.

RAI B.1.3-3:

(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. Has this extrapolation of wear data been approved by the NRC? If so, identify a reference.

VCSNS Response RAI B.1.3-3

VCSNS now has 4 sets of data on the of the thimble tubes. Data has been gathered in RF-4, 5, 6 and 13. The highest recorded measurement in RF-4 was 38% and in RF-13 was 57%. The projections for wear at RF-17 are all below 75% and the highest wear predicted for RF-18 is between 75% and 80%.

RAI B.1.3-4: 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.

VCSNS Response RAI B.1.3-4

The evaluation performed by Westinghouse makes allowances for uncertainties. The Westinghouse methodology has an acceptance criteria of 80%, VCSNS uses 75% for additional conservatism.

RAI B.1.3-5: The bottom-mounted instrumentation inspection program also requires that the thimble tubes must be capped or replaced if the projected through-wall wear exceeds 80% prior to the next scheduled ECT. However, the VCSNS response to Bulletin 88-09 suggests that the thimble tube must be replaced when wall wear exceeds 60%. Explain this discrepancy.

VCSNS Response RAI B.1.3-5

Thimbles must be repositioned or capped if projected through wall wear will exceed 75% prior to the next scheduled eddy current exam. If measured or projected thimble wear exceeds 80%, the thimble must be capped or replaced. This criteria is based upon revised Westinghouse recommendations which were incorporated into our program.

RAI B.1.3-6: 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. If not, provide a discussion of and the technical basis for the scope of inspection that has been performed and how the scope of inspection is determined for future inspections. Also, discuss inspection expansion plans, if applicable to this program.

VCSNS Response RAI B.1.3-6

VCSNS now has 4 sets of data on the of the thimble tubes. Data has been gathered in RF-4, 5, 6 and 13. The highest recorded measurement in RF-4 was 38% and in RF-13 was 57%. The projections for wear at RF-17 are all below 75% and the highest wear predicted for RF-18 is between 75% and 80%.

B.1.8 Reactor Head Closure Studs Program

RAI B.1.8-1: In LRA Table 3.1-1, AMR Item 18, the applicant states that the aging effect requiring management is loss of closure integrity. The applicant states that the AMR results are consistent with NUREG-1801 in materials and environments, but not in aging effects. The applicant further states that loss of closure integrity, rather than loss of preload or cracking, is the aging effect requiring management. Explain the difference between loss of closure integrity and loss of preload or cracking.

VCSNS Response RAI B.1.8-1

Although identified as an aging effect in various industry references, loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for component bolted closures (i.e. pressure boundary closures) within the scope of license renewal.

Mechanical components within the scope of license renewal, both Class 1 and non-Class 1, contain bolted closures that are necessary for the pressure boundary of the components being joined/closed. Examples of these bolted closures are valve bonnet to body, pump cover to casing, heat exchanger manway and channel head (end-bell), and piping flange sets. The bolted closure is comprised of two mating surfaces, a gasket, and a fastener set (studs or bolts, washers, and nuts). By themselves, the mating set, gasket, or fastener set have no component intended function. Together, the bolted closure forms an integral part of the pressure-retaining boundary of the component. Additionally, the bolted closure is exposed to the same environment(s) as the components in the plant areas where the closure is located (process fluid for internal mating surface and ambient conditions else). As such, the bolted closure (including fastener set) was considered to be a sub-component (piece-part) of the components/ component types within the scope of license renewal and did not require evaluation separate from the component, except as clarified.

Loss of mechanical closure integrity can result in failure of the mechanical joint, is evidenced by leakage rather than joint failure, and can be attributed to one or more of the following effects:

- **Loss of bolt pre-load (embedment, cyclic load embedment, gasket creep, etc.),**
- **Loss of bolting material (from general and/or boric acid corrosion),**
- **Reduction of bolting material fracture toughness, and**
- **Cracking of high strength bolting material (SCC).**

B.1.10 Steam Generator Management Program

<u>RAI B.1.10-1:</u> In LRA Appendix B.1.10, the applicant states that the steam generator management program is consistent with GALL AMP XI.M19, and no deviations are noted. The applicant also describes its operating experience with the steam generator management program and states that no significant degradation was found during routine inspections. The staff notes that operating experience with the VCSNS replacement steam generators (Westinghouse Delta-75) is quite limited, but additional experience is available with other replacement steam generators of similar design with thermally treated Alloy 690 tubes and Type 405 stainless steel tube support plates. Summarize the industry operating experience with Alloy 690 tubes and Type 405 stainless steel tube support plates.

VCSNS Response RAI B.1.10-1

The VCSNS Replacement Steam Generators incorporated many lessons learned from years of experience with U-tube Steam Generators. Many of the design features of the

Delta-75 Steam Generators are also found in the Model F Steam Generators. This information is reflected in the information supplied to the NRC for the Steam Generator Replacement Project at VCSNS. One source of information is WCAP-13480 Rev. 1.

The inspections performed on the Delta-75 Steam Generators have detected no abnormal indications.

RAI B.1.10-2: In LRA Appendix B.1.10, the applicant states that a partial eddy current inspection was performed during refueling outages 9, 10, and 11. A 100% eddy current inspection and full secondary side inspection of steam generators A, B, and C were performed during refueling outage 12.

a. Submit more information about the secondary side inspection during refueling outage 12, i.e., identify the secondary side components inspected, type of inspection performed, guidance used, and the frequency of such inspection to be performed during the extended period of operation.

b. Provide detailed information about the primary side inspection of the steam generators (i.e., tubes and plugs) during refueling outages 9, 10, 11, and 12. Discuss the components inspected, inspection scope, inspection technique used, and guidelines used.

c. The applicant states that no significant degradation was detected by the inspection performed during the refueling outages 9, 10, 11, and 12. The staff would like to have more information about these inspections so that it can assess the susceptibility of Alloy 690 tubes and plugs to different aging effects. Discuss all primary and secondary side degradations that have been detected since the operation of the VCSNS replacement steam generators.

VCSNS Response RAI B.1.10-2

a. The secondary side maintenance/inspections performed during RF-12 included sludge removal and a general visual inspection of secondary side components in an effort to detect loose parts, foreign objects, or abnormal degradation. The scope and frequency of future inspections will continue to be guided by NEI 97-06 and EPRI guidelines.

b., c. Steam Generator tube inspection results were reported to the Staff in accordance with the requirements of Technical Specification 3/4.4.5.

RAI B.1.10-3: The FSAR supplement for the steam generator management program in LRA Appendix A, Section 18.2.35, states that the program implements the requirements of VCSNS Technical Specification 4.4.5 and follows the recommendations provided by NEI and EPRI guidelines, specifically their commitment to NEI 97-06.

a. Include, in the FSAR supplement, the specific references to the NEI and EPRI guidelines, specifically, their commitment to NEI 97-06.

b. If the applicant's steam generator management program is consistent with the GALL report, the applicant needs to state in LRA Appendix A, Section 18.2.35, that its steam generator management program is consistent with GALL XI.M19.

c. In LRA Appendix A, Section 18.2.35, the applicant states that "...The purpose of the Steam Generator Management Program is to perform examinations of nickel-based alloy steam generator tubes and tube plugs to ensure that cracking and loss of material are identified and corrected prior to exceeding allowable limits..." The staff believes that the applicant's steam generator management program should inspect components other than tubes and tube plugs, if the program were to be consistent with GALL XI.M19. List all the components that will be inspected in the steam generator management program.

VCSNS Response RAI B.1.10-3

Steam Generator tube Inspections are performed in accordance with the requirements of Technical Specification 3/4.4.5. There in no need to restate these requirements in Chapter 18 of the FSAR. Commitments made under the current license will carry on into the period of extended operation.

From the GALL program scope statement "The scope of the program is specific to steam Generator tubes." The FSAR section will be revised as below, to more correctly describe the program:

18.2.35 STEAM GENERATOR MANAGEMENT PROGRAM
The purpose of the Steam Generator Management Program is to perform examinations to ensure that cracking and loss of material of nickel-based alloy steam generator tubes and tube plugs are identified and corrected prior to exceeding allowable limits. The program implements the requirements of Technical Specification 4.4.5 for tube inspections. Other components, in addition to tubes, are inspected under this program. The program follows the recommendations provided by NEI and EPRI guidelines for Steam Generator component inspections and chemistry controls.

B.1.24 Reactor Vessel Surveillance Program

RAI B.1.24-1: The VCSNS reactor vessel surveillance program consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years. The applicant plans to remove the two remaining surveillance capsules during RF-14. As a result, no surveillance capsules will be left in the vessel during the extended period of operation. Confirm whether the operating restrictions will be established at the end of RF-14 to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the reactor vessel will be monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. In addition, confirm whether an alternative dosimetry will be used at VCSNS to monitor neutron fluence during the period of extended operation.

VCSNS Response RAI B.1.24-1

A program will be established at the end of RF-14 to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the Reactor Vessel will be monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. This program may be supplemented or revised by using alternative dosimetry or other effective neutron fluence monitoring techniques during the period of extended operation.

RAI B.1.24-2: Provide information regarding the fluence calculation methodology, i.e., how 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. Provide applicable references.

VCSNS Response RAI B.1.24-2

Information on the specimens from capsule W was submitted to the Staff in a letter (RC-98-0185, Reference 10), G. J. Taylor, SCE&G, to NRC Document Control Desk, dated October 9, 1998, "Reactor Vessel Surveillance Program". Attached to the letter was WCAP-15101, "Analysis of Capsule W From the South Carolina Electric & Gas Company V.C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," September 1998. This WCAP contains the data from the test results of capsule W that was removed after 10.78 Effective Full Power Years (EFPY) with a lead factor of 3.40.

RAI B.1.24-3: In LRA Appendix B.1.24, "Reactor Vessel Surveillance Program," the applicant states that VCSNS will perform a one-time analysis to demonstrate that the material in the inlet and outlet nozzles and upper shell courses will not become controlling during the period of extended operation. Describe the one-time analysis to be performed and explain why such a demonstration is needed. If the demonstration is not successful, confirm that the reactor vessel surveillance program will be reestablished. Submit the results of the analysis, for staff's review, prior to the extended period of operation.

VCSNS Response - RAI B.1.24-3

The material test reports are found in the document package for the Reactor Vessel. The nozzles are SA-508 material. No information on Cu or Ni was found in the vessel documentation. The standard values from 10 CFR 50.61 of 0.35% Cu and 1.00% Ni were used. From Table 2, the chemistry factor is 272ºF. The material test reports contain the NDT results for the vessel nozzles. The highest temperature for NDT is 0ºF for one of the inlet nozzle. No information on margin value (M) was found in the vessel documentation. Equation 2 and the standard values for its inputs from 10 CFR 50.61 could be used. However, Westinghouse has used 34ºF for plates, therefore 34ºF was used for M. The fluence falls off promptly with the distance from the core. For the nozzle, a distance of 8 feet was used from the core midplane to the edge of the nozzle. This was used to convert flux to the fluence value (f).

VCSNS has evaluated the vessel nozzles in the above-described manner and found they do not become limiting for a sixty-year plant life. A conservative projected RT_{PTS} for the nozzle material at the 54 EFPY EOL value is 145.2°F.

B.2.1 Small-Bore Class 1 Piping Inspection

RAI B.2.1-1: The applicant has committed to perform destructive examinations of small bore piping. This is of significance to the staff in order to make a reasonable assurance determination of this AMP; therefore, the applicant should include this commitment of performing destructive examinations of small bore piping in the FSAR supplement for the AMP described in LRA Appendix B.2.7.

VCSNS Response RAI B.2.1-1

VCSNS will evaluate the small-bore class 1 piping with a methodology that is approved by the Staff. The present approved methodology is to perform destructive examinations of small-bore piping. The approved method will be incorporated into the Small Bore Class 1 Piping Inspection prior to the period of extended operation. It is not appropriate to incorporate the specific inspection into the FSAR, as it is very likely new approved techniques may be developed.

B.2.4 Reactor Vessel Internals Inspection

RAI B.2.4-1: 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.

VCSNS Response RAI B.2.4-1

The capability to achieve 0.0005-inch resolution in the field has not been demonstrated. The GALL program did not state this as a requirement but stated, "...enhancing the VT-1 examinations for non-bolted components for example, to include the ability to achieve a 0.0005 inch resolution."

RAI B.2.4-2: 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.

VCSNS Response RAI B.2.4-2

The details of the Reactor Vessel Internals Inspection Program have not been developed. VCSNS will follow industry initiatives and will have a program in place prior to the period of extended operation.

RAI B.2.4-3: 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.

VCSNS Response RAI B.2.4-3

No instances of irradiation-assisted stress corrosion cracking have been reported for internal components. If this aging effect is seen, it will be seen only in close proximity to the reactor core. Inspections that will be performed will detect cracking regardless of the cause.

RAI B.2.4-4: 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 the NRC prior to the end of the initial operating term for V.C. Summer. The report should summarize the understanding of the aging effects applicable to the reactor vessel internals and should contain a description of the V.C. Summer inspection plan, including methods for detection and sizing of cracks and acceptance criteria. This should also be discussed in the FSAR supplement.

VCSNS Response RAI B.2.4-4

The details of the Reactor Vessel Internals Inspection Program have not been developed. VCSNS will follow industry initiatives and will have a program in place prior to the period

of extended operation. As these details have not been developed they cannot be placed in the FSAR.

B.1.2 Borice Acid Corrosion Surveillances (BACS) Program

RAI B.1.2-1: The definition section of Appendix B.1.2, "Boric Acid Corrosion Surveillances" (BACS), states that this program is consistent with GALL XI.M10, except for enhancements related to dissimilar metal weld inspections. The staff requests the applicant to discuss the enhancements to this program by addressing the following:

- operating experience since publication of GALL report and since submittal of the application in addition to 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.

VCSNS Response RAI B.1.2-1

Presently, the Boric Acid Corrosion Surveillances Program concerns only GL 88-05 requirements. GALL is driving the industry to make enhancements to the surveillances beyond those required for GL 88-05, to include systems outside of containment that contain boric acid solutions. Recent industry events as described in NRC Bulletins 2002-01 and 2002-02 are also driving the industry to additional inspections beyond those currently performed for GL 88-05.

For those systems outside of containment, VCS intends to enhance the Surveillance Test Procedures already required by Technical Specifications for leakage of primary coolant sources outside containment. These leakage assessment tests are for the following systems: Boron Recycle, Liquid Waste, Nuclear Sampling, Chemical and Volume Control, Residual Heat Removal, and RB Spray. In addition to these, VCS intends to enhance the leak tests performed for the SI Accumulators and the Spent Fuel Pool Cooling. These test procedures will be enhanced to specify inspecting for boric acid crystallization on the system being tested and, when boric acid is found, on surrounding systems. The enhancements to the procedures will be noted on the procedures and maintained as license renewal commitments.

The industry events that resulted in NRC Bulletins 2002-01 and 2002-02 are a current licensing basis issue that will result in additional inspections. VCS is following these developments and is involved in the EPRI MRP that is studying various inspection plans. VCS is considered to be a "low susceptibility plant" for CRDM nozzle degradation.

VCSNS is developing an overall Boric Acid Corrosion Program that will incorporate GL 88-05 requirements, license renewal commitments, and the additional inspections that will result from the NRC Bulletins.

RAI B.1.2-2: The operating experience section of Appendix B1.2, "Boric Acid Corrosion Surveillances," discusses the hot leg axial cracking at VCSNS on October 7, 2000 and further states that the BACS 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. The staff requests the applicant to list the locations of the other dissimilar metal welds exposed to borated coolant to be included within the scope of this program.

VCSNS Response RAI B.1.2-2

As a result the hot leg axial cracking at VCSNS, a new Surveillance Test Procedure was issued. Dissimilar welds are contained in an enclosure of this procedure and are as follows:

"A" Hot Leg weld to Reactor Vessel Nozzle
"A" Cold Leg weld to Reactor Vessel Nozzle
"B" Hot Leg weld to Reactor Vessel Nozzle
"B" Cold Leg weld to Reactor Vessel Nozzle
"C" Hot Leg weld to Reactor Vessel Nozzle
"C" Cold Leg weld to Reactor Vessel Nozzle
Pressurizer Surge Line Weld to Pressurizer Nozzle
Pressurizer Nozzle weld to "A" Pressurizer Safety Valve
Pressurizer Nozzle weld to "B" Pressurizer Safety Valve
Pressurizer Nozzle weld to "C" Pressurizer Safety Valve
Pressurizer Nozzle weld to PORVs
Pressurizer Nozzle weld to Spray piping
"A" Hot Leg weld to Steam Generator Nozzle
"A" Crossover weld to Steam Generator Nozzle
"B" Hot Leg weld to Steam Generator Nozzle
"B" Crossover weld to Steam Generator Nozzle
"C" Hot Leg weld to Steam Generator Nozzle
"C" Crossover weld to Steam Generator Nozzle

RAI B.1.2-3: In Table 3.1-1, "Summary of Aging Management Programs for the Reactor Coolant System Evaluated in NUREG-1801 that are Relied on for License Renewal," AMR Item 26 credits the BACS for managing loss of material due to boric acid corrosion of the pressurizer carbon steel and low alloy steel components; i.e., shell, upper and lower heads, nozzles, integral support, manway cover and bolts. The staff requests the applicant to discuss how this program will be sufficient to manage the corrosive effects of boric acid leakage on the base metal of these insulated components during the extended period of operation; i.e, postulated leakage from the pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials.

VCSNS Response RAI B.1.2-3

BACS will evaluate all leaks of boric acid. The insulation on the mechanical joints on the Pressurizer is removed and the joints are inspected for leakage each refueling. Corrective actions for boric acid leaks are required for the source as well as the adjacent

components, supports, or structures. More information on this subject can be found in a letter from Stephen A. Byrne to the Document Control Desk January 24, 2003 (Reference 11). The subject of this letter is "Response for Additional Information Regarding 60 Day Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

RAI B.1.2-4 In Table 3.1-1, "Summary of Aging Management Programs for the Reactor Coolant System Evaluated in NUREG-1801 that are Relied on for License Renewal," AMR Item 26 credits the BACS for managing the loss of material due to boric acid corrosion for the external surfaces of carbon steel components in reactor coolant system pressure boundary. In Table 2.3-7, "Steam Generators Component Types Subject to Aging Management Review and Their Intended Functions," the applicant identified the steam generator elliptical head and channel head as being within the pressure boundary and therefore, managed by the BACS. The staff requests the applicant to discuss how the BACS will manage the external surfaces of the VCSNS steam generators in light of BL 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," and GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants".

VCSNS Response RAI B.1.2-4

BACS will evaluate all leaks of boric acid. The insulation on the primary side manways of the steam generators is removed and the joint are inspected for leakage each refueling. (Only the lower portions of the steam generator shell are considered subject to leaking boric acid.) More information on this subject can be found in a letter from Stephen A. Byrne to the Document Control Desk dated January 24, 2003 (Reference 11). The subject of this letter is "Response for Additional Information Regarding 60 Day Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

B.1.4

RAI B.1.4-1: The program description section of Appendix B.1.4, "Chemistry Program", states that this program does not commit to performing one-time inspections to verify its effectiveness as suggested by NUREG-1801, XI.M2. The following are examples of components exposed to chemically treated water that will be managed by this program alone:

- From Table 3.1-2, "Summary of Aging Management Evaluations for the Reactor Coolant System that are Different From or Not Addressed in NUREG-1801 but are Relied on for License Renewal":
  - 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)
  - AMR Item 5, crevice and pitting corrosion of SS piping and piping system components

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

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

- AMR Item 14, loss of material in the RCP thermal barrier flange and loss of material due to crevice and pitting corrosion of non-Class 1 piping and valve bodies for the pressurizer relief tank spray

- 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

- From Table 3.3-1, "Summary of Aging Management Program for the Auxiliary Systems Evaluated in NUREG-1801 that are Relied on for License Renewal":

  - 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 discuss the operating history and the results of the most recent surveillances or inspections for these and similar components in the various water environments assure that the above-listed types of corrosion (crevice, general, pitting, and galvanic) are adequately managed.

The staff also requests the applicant to discuss if there a one-time inspection for the most susceptible locations; i.e., low flow and/or stagnant areas, in the components that credit this program for aging management. If so, the staff requests the applicant to describe the one-time inspection; e.g., scope, determination of sample size, identification of inspection locations, determination of examination technique, and the evaluation of the need for follow-up examinations.

VCSNS Response RAI B.1.4-1

GALL has acknowledged the Chemistry Program alone is effective for many areas. For treated water and borated water systems many component-aging effect combinations that are managed by chemistry alone are not listed in the GALL tables.

From the GALL Program Description of XI.M2 WATER CHEMISTRY:

"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which

the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."

From the GALL Section VII E1:

"The effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated borated water and, therefore, are not included in this section."

From the GALL Section VIII B1 Items B1.1-a and B1.2-a list loss of material due to pitting and crevice corrosion of carbon steel components. These items have only the Chemistry Program listed as the aging management program.

GALL Chapter IV does not list "loss of material due to pitting and crevice corrosion" for Westinghouse PWR components other than the item IV D1.1-c. With the approval of GALL the staff has determined, "loss of material due to pitting and crevice corrosion" is not an aging affect requiring management for Westinghouse PWR components listed in Chapter IV other than IV D1.1-c.

The components listed in the RAI are generic and not plant specific and have been generically evaluated during the development of GALL. VCSNS has chosen to list the component-aging effect combination where chemistry alone is credited for Aging Management.

RAI B.1.4-2: The applicant appears to have combined aspects of several GALL programs into its chemistry program. Therefore, 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.

VCSNS Response RAI B.1.4-2

The Service Water System Reliability and In Service Testing Program, not the Chemistry Program, is credited for meeting the GALL requirements for XI.M20, "Open-Cycle Cooling Water System." This program meets the intent of GL 89-13.

As a response to Recommended Action #2 of Generic Letter 89-13 various systems were reviewed and evaluated by the Design Engineering Department at VCSNS. Their evaluation was based on a review of the historical Maintenance Work Requests from the time of CHAMPS inception through August 27, 1991. These systems were the Component Cooling Water System, the Chemical Volume and Control System, the Residual Heat Removal System, the Spent Fuel Cooling System and the Chilled Water System. None of the documents revealed that the corrosion protection of these systems had been compromised. VCSNS maintains chemical concentrations within the guidelines of EPRI TR-107396 for its closed cycle cooling systems. Prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

RAI B.1.4-3: By letter dated October 1, 2002, the applicant provided a table entitled "Virgil C. Summer Nuclear Station Database AMR Query," which states in several entries that CS components; e.g., 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). Additional information is also provided in the accompanying applicant-supplied VCSNS Mechanical Database AMR Query Notes A-CC-c and A-CC-h on pages 23 and 25, respectively. However, no aging management program has been credited to manage this aging effect. According to the ASM Handbook, Vol. 11, "Failure Analysis Prevention," and EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines," October 1997, SCC occurs in carbon steels usually in the presence of hydroxides, carbonates or nitrates. Therefore, the staff requests the applicant to discuss how this aging effect is managed; i.e., name of the program(s) which manage this aging effect and how this program can prevent, detect or mitigate the effects of SCC in these carbon steel components.

VCSNS Response RAI B.1.4-3

Database AMR note A-CC-c refers to SCC of stainless steel components, not carbon steel components. Database AMR note A-CC-h states that SCC is not an aging effect requiring aging management for the Component Cooling (CC) System because the CC System does not use a nitrate-based corrosion inhibitor.

There are systems that utilize nitrates as corrosion inhibitors. These systems are the jacket water cooling portion of the Diesel Generator (DG) System, the Chilled Water (VU) System, and, through interfaces with the VU System, the Air Handling (AH) System, and the Local Ventilation (VL) System. Industry data does not exhibit widespread incidence of SCC in low strength carbon steels; however, there was a reported case suspected to be nitrate-induced SCC of carbon steel in a treated water system. VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrates are added as a corrosion inhibiter. In these closed systems there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrates are added as a corrosion inhibiter by the Chemistry Program at levels within EPRI guidelines (EPRI TR-107396); therefore, VCSNS maintains that the Chemistry Program adequately manages SCC of carbon steel components in a treated water environment.

RAI B.1.4-4: The program description section of Appendix B.1.4, "Chemistry Program", states that this program is consistent with GALL XI.M30, "Fuel Oil Chemistry"; however, the application does not discuss the verification of the program's effectiveness at locations where contaminants may accumulate as recommended in GALL. Therefore, the staff requests the applicant to discuss the basis for not including the verification of this program to manage loss of material.

VCSNS Response RAI B.1.4-4

The details of the sampling of fuel oil, contained in plant procedures, are in accordance with standards listed in GALL and thus meet the requirements of the GALL for sampling at different levels inside the fuel oil tanks. Per Technical Specification 4.8.1.1.2.i.1, every ten years the Diesel Generator Fuel Oil Storage Tanks are drained and cleaned. Operating experience at VCSNS for the fuel oil components managed by this program reveals no history of aging degradation for the internal surfaces.

RAI B.1.4-5: In Table 3.1-2, "Summary of Aging Management Evaluations for the Reactor Coolant System that are Different From or Not Addressed in NUREG-1801 but are Relied on for License Renewal", AMR Item 7 credits the chemistry program for managing loss of material due to crevice and pitting corrosion in the 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 due to high levels of oxygen, which may be present in the reactor coolant. However, if hydrogen overpressurization is maintained in the reactor coolant at sufficiently high levels, protection is provided in the creviced geometries of pressurizer's internal surfaces. The staff requests the applicant to discuss how the chemistry program will ensure a sufficient level of hydrogen overpressurization to manage crevice corrosion in the pressurizer's internal surfaces.

VCSNS Response RAI B.1.4-5

The Reactor Coolant System (RCS) environment, including the primary side of the Steam Generators, is sampled and analyzed for chloride, fluoride, and dissolved oxygen. The frequency for these samples is in accordance with EPRI guidelines. Dissolved oxygen concentrations are not permitted to exceed procedure limits for prolonged periods. Action levels are established to direct correction of any out of specification condition if any limits are exceeded. Oxygen is controlled in makeup water as well as in the RCS. Hydrogen is controlled between 25-50 cc/kg H_2O in the RCS to ensure scavenging of oxygen.

B.1.15 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 discuss these sampling procedures used 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 additional information on the 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 this element was determined to not be applicable, the staff requests a justification for this determination.

VCSNS Response RAI B.1.15-1

As stated in Application Section 18.2.11, "sampling approaches" are used for cases of inaccessibility. There are no specific "sampling procedures" used at VCSNS, rather experience of the inspection team member(s) is used to either select specific inaccessible areas (such as behind wall attachments or appurtenances) for a closer inspection, or to determine if visible accessible areas provide any indication that additional inspections are required in adjacent inaccessible areas. This process is consistent with inspection requirements of the Appendix J General Visual Inspection (B.1.11), Containment ISI Program - IWE/IWL (B.1.16) and Maintenance Rule Structures Program (B.1.18), all of which provide effective aging management of coatings under the current licensing basis and during the extended period of operation.

Consistent with confirmatory element 4 (Detection of Aging Effects), VCSNS: (a) conducts coatings inspections at a minimum frequency of each refueling outage or during other major maintenance outages; (b) ensures inspection team personnel (Quality Control or Engineering) are qualified; (c) conducts general visual inspections during walk-throughs, inspects previously designated areas, and inspects all coating in the vicinity of sumps and screens; and (d) documents the inspection results via condition evaluation reports or non-conformances.

VCSNS Operating Experience: Containment coatings have been inspected at a minimum frequency of each refueling outage by Operations, Maintenance, and /or QC personnel. Engineering personnel have also participated in these inspections since about 1996. An extensive inspection of Containment coatings was conducted by QC and Engineering personnel in 2000 as part of the Containment ISI Program - IWE/IWL (B.1.16) and Maintenance Rule Structures Program (B.1.18). Under the provisions of these programs, inspections are made of all accessible areas using direct line of sight from permanent vantage points, which also allows for random inspections of inaccessible areas such as behind structural attachments, cable trays and duct work. These inspections identified several areas with coating deficiencies (failures) in accessible areas which were documented and/or corrected/repaired. There were no observations of coating deficiencies in any inaccessible areas.

B.1.6 Flow-Accelerated Corrosion Monitoring Program

RAI B.1.6-1: The program description section of Appendix B.1.6, "Flow-Accelerated Corrosion (FAC) Monitoring Program," states that this program is consistent with GALL XI.M17, "Flow-Accelerated Corrosion". The applicant also states that the need for inspections is determined by a calculation performed in accordance with engineering procedures and that if components exhibit high wear during a cycle these are replaced with more FAC-resistant material. The

EPRI document, NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," recommends the use of a predictive method for determining the rate at which component degradation is occurring by FAC. 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 staff requests the applicant to discuss the "calculation performed in accordance with engineering procedures" to determine inspection need. Specifically, the staff requests a discussion of the methods used at VCSNS for predicting component degradation by FAC and how these predictive methods are used to determine the need and frequency of inspections.

VCSNS Response RAI B.1.6-1

The Flow-Accelerated Corrosion Monitoring Program is consistent with the basic guidelines and recommendations provided in EPRI NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program". Inspection frequency varies for each location depending on trending of inspection results (the calculated rate of material loss), analytical model review (CHECWORKS), changes in operating or chemistry conditions, pertinent industry operating experience, plant operating experience, and engineering judgment. The most probable locations of significant wear are given the highest priority with respect to scheduling inspections during each outage. This allows for sufficient lead time should any corrective action be deemed necessary.

<u>RAI B.1.6-2:</u> The staff requests a list of components within the scope of this program which are most susceptible to FAC in addition to the initial wall thickness (nominal), current (measured) wall thickness and the future predicted wall thickness to demonstrated the effectiveness of the FAC monitoring program.

VCSNS Response RAI B.1.6-2

The Flow-Accelerated Corrosion Monitoring Program will detect loss of material due to erosion-corrosion prior to loss of component intended function. The V.C. Summer program has been the subject of numerous NRC reviews, and the overall program was found satisfactory in NRC Inspection Report 50-395/92-20. This inspection report makes the following statement: "The licensee has established an effective program to maintain high energy carbon steel piping systems within acceptable wall thickness limits."

<u>RAI B.1.6-3:</u> The FAC monitoring program includes the use of a predictive method to calculate the wall thinning of components susceptible to FAC. In order for the staff to evaluate the accuracy of these predictions, the staff requests a sample list of components for which wall thinning is predicted and measured by UT or other method.

VCSNS Response RAI B.1.6-3

The Flow-Accelerated Corrosion Monitoring Program will detect loss of material due to erosion-corrosion prior to loss of component intended function. The V.C. Summer

program has been the subject of numerous NRC reviews, and the overall program was found satisfactory in NRC Inspection Report 50-395/92-20. This inspection report makes the following statement: "The licensee has established an effective program to maintain high energy carbon steel piping systems within acceptable wall thickness limits."

B.2.1 Above Ground Tank Inspection

RAI B.2.1-1: The description section of Appendix B.2.1, "Above Ground Tank Inspection," states in element 5, "Monitoring and Trending," that no actions are taken to trend inspection results. This one-time inspection program determines if further actions are required. The staff notes that the evaluation of the techniques and the 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, the staff requests that the applicant to address any changes, which are made in monitoring and trending for those components, exposed to borated water.

VCSNS Response RAI B.2.1-1

There is no action to trend the inspection results because it is a one-time inspection. The aim of one-time inspections is to determine if further actions are required. Element #7 of the program addresses additional inspections should degradation be detected. Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this research are contained within the body of technical work at VCSNS supporting the LR application. The one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS.

The Above Ground Tank Inspection at VCSNS is not the same as the GALL Program XI.M29 "Above Ground Carbon Steel Tanks." The Above Ground Tank Inspection will inspect the internal surfaces of a sampling of both carbon steel and stainless steel tanks. The aging of the external surfaces of tanks are managed by Inspections for Mechanical Components and, if susceptible to boric acid corrosion, by the Boric Acid Corrosion Surveillances.

RAI B.2.1-2: The staff notes that GALL XI.M29 "Above Ground Carbon Steel Tanks," is not credited for aging management in the VCSNS LRA. This GALL program defines preventive measures to mitigate corrosion of the external surface of carbon steel tanks with paint or coatings in accordance with standard industry practice. The staff also notes that Appendix B.1.15,"Containment Coating Monitoring and Maintenance Program," is an existing aging management program that manages the loss of material due to coating degradation but is not credited with managing the external degradation of the tanks. The staff requests the applicant to discuss how the Above Ground Tank Inspection Program adequately manages the external surface of the above ground tanks if this program only inspects the internal surfaces of the tanks.

VCSNS Response RAI B.2.1-2

The Above Ground Tank Inspection at VCSNS is not the same as the GALL Program XI.M29 "Above Ground Carbon Steel Tanks." The Above Ground Tank Inspection will inspect the internal surfaces of a sampling of both carbon steel and stainless steel tanks. The aging of the external surfaces of tanks are managed by Inspections for Mechanical Components and, if susceptible to boric acid corrosion, by the Boric Acid Corrosion Surveillances. The Inspections for Mechanical Components manages the relevant aging effects for mechanical components constructed of carbon steel, low alloy steel, and other susceptible materials. Tank foundations and supports are inspected under the Maintenance Rule Structures Program (Application Section B.1.18).

Additionally, outside above ground steel tanks are externally inspected under the Maintenance Rule Structures Program. Inspections are conducted on the Condensate Storage Tank, Refueling Water Storage Tank, Reactor Make-Up Water Storage Tank, and Sodium Hydroxide Storage Tank. This program includes visual inspections of the exterior surface of the tank, anchor bolts and attachment anchorage plates/welds, concrete foundation support pads, piping connections, and caulking between tank / foundation.

RAI B.2.1-3: The description section of Appendix B.2.1, "Above Ground Tank Inspection," states that this program will be consistent with the GALL 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:

a) Element 4, "Detection of Aging Effects" - This program does not discuss the qualification of the personnel conducting the inspection.

b) Element 6, "Acceptance Criteria" - This program does not discuss or refer to the design minimum wall thickness nor to the criteria for verifying the absence of cracking.

VCSNS Response RAI B.2.1-3

Inspections required by this program would be performed by personnel qualified in accordance with ASME code and 10CFR50 Appendix B.

Minimum wall thickness will be determined by the design of the individual component and in accordance with ASME code. Cracking will be detected by volumetric and visual inspections conducted in accordance with ASME code.

B.2.10 Buried Piping and Tanks Inspection

RAI B.2.10-1: In Appendix B.2.10, "Buried Piping and Tanks Inspection," under element 2, "Preventive Actions," the applicant states that underground components are coated and wrapped during installation to prevent direct contact with the soil environment. The staff requests the applicant to briefly describe the coating techniques used and to discuss the verification of the adequacy of these techniques in light of the inadequate cathodic protection discussed in the operating history section.

VCSNS Response RAI B.2.10-1

VCSNS coats and wraps underground components in accordance with site procedures, which are based on accepted industry standard AWWA C-203, 1973. These procedures are available on site for inspection. Operating experience for the Diesel Generator Fuel Oil Storage Tanks revealed that negligible wall thinning had occurred thereby verifying that the techniques of coating and wrapping are effective.

RAI B.2.10-2: In Appendix B.2.10, "Buried Piping and Tanks Inspection," under element 3, "Parameters Monitored or Inspected," the applicant states 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 applicant later cited operating experience with buried piping and tanks, which used UT. The staff requests the applicant to discuss if UT will be used in addition to or in place of the visual inspection and the criteria used to determine the applicability of the technique used.

VCSNS Response RAI B.2.10-2

A visual inspection of the wrapping and coating will be performed and evaluated upon initial excavation of the component. If the wrapping or coating is damaged, or are required to be removed as part of the maintenance activity, then the underlying metal will be visually inspected for degradation. Depending on the condition of the underlying metal, subsequent inspections, and types of inspections, will be determined through the VCSNS Corrective Action Program.

RAI B.2.10-3: In Appendix B.2.10, "Buried Piping and Tanks Inspection," under element 4, "Detection of Aging Effects," the applicant stated that a specific inspection frequency for buried components is not warranted. The staff requests the applicant to discuss why periodic inspection of the most susceptible locations is not needed especially in areas with the highest likelihood of corrosion and in areas with a history of corrosion problems.

VCSNS Response RAI B.2.10-3

GALL program XI.M34 allows the inspection frequency to be whenever underground piping is excavated for maintenance depending on operating experience. VCSNS operating experience has shown no history of corrosion problems for buried piping and tanks. Confirmation of this is the negligible wall thinning of the diesel fuel oil storage tanks. Based on VCSNS operating experience, frequency of inspection based upon scheduled maintenance is justified. Depending on the condition of the underlying metal, subsequent inspections, and types of inspections, will be determined through the VCSNS Corrective Action Program.

RAI B.2.10-4: In Appendix B.2.10, "Buried Piping and Tanks Inspection," under element 6, "Acceptance Criteria," the applicant states 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." The staff requests the applicant to discuss how the coating and wrapping degradation will be reported and evaluated; e.g., by site corrective actions or other procedure.

VCSNS Response RAI B.2.10-4

Any coating and wrapping degradation would be reported and evaluated according to the VCSNS Corrective Action Program.

RAI B.2.10-5: The operating history section of Appendix B.2.10, "Buried Piping and Tanks Inspection," discusses the inspection of the 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.

VCSNS Response RAI B.2.10-5

The only buried tanks in scope for license renewal are the diesel fuel oil storage tanks. VCSNS operating experience has shown no history of corrosion problems for buried piping.

Attachment IV
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Sections 3.2, 3.3, 3.4, 3.5, 4.0, and Appendix B
Accession No. ML030900096

3.2 ENGINEERED SAFETY FEATURES SYSTEM

RAI 3.2-1: In LRA Table 3.2-1, Item 3, the applicant stated 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. It is not clear to the staff 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 that there are no aging effects requiring management.

VCSNS Response RAI 3.2-1

The 1" stainless steel RWST bottom at VCSNS sits on an outdoor raised concrete foundation (see FSAR Figures 1.2-4 and 1.2-10) that ensures the tank bottom is not subjected to flowing water. VCSNS is located well inland and does not see salt or other corrosive materials in the air. The stainless steel tank bottom is in a low temperature (less than 140ºF) environment similar to stainless steel embedded in concrete. This environment has been determined not to have aging effects requiring management.

RAI 3.2-2: In LRA, Table 3.2-1, Item 4, the applicant stated that MIC has been determined not to be a valid aging mechanism for the material/environment combination represented by the containment isolation valves and associated piping. The applicant stated that this aging mechanism is not applicable since the four systems (AC, DN, LR, NG), which provide containment isolation, are not subject to wetting from raw water. The applicant is requested to provide 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 these components are not susceptible to loss of material due to MIC.

VCSNS Response RAI 3.2-2

Other than the Reactor Building Equipment Hatch, the Reactor Building penetrations at VCSNS are located indoors. (Reference FSAR Figures for Section 1.2.) The containment isolation systems contain either treated water or air/gas. AC is a treated cooling water system. DN contains demineralized water. LR and NG contain air/gas. The in scope portions of these systems are in a sheltered environment and are not exposed to rain or raw water. Therefore, MIC is not a valid aging effect for the in scope portion of these systems.

RAI 3.2-3: For the stainless steel components, 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. Provide the basis, for all the variety of potential sheltered environments in the ESF, that stainless steel components are not susceptible to any aging effects requiring management.

VCSNS Response RAI 3.2-3

VCSNS is located well inland and is located in an area where forestry is the prime commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Samples of groundwater and rainwater show chloride and sulfate concentrations less than 10 ppm. GALL did not identify aging effects that require aging management for the stainless steel in an air environment. For example, the components included in Table 3.2-2, Item 1, are the same components listed in Table 3.2-1. Table 3.2-2, Item 1, is considered consistent with the conclusions of the efforts that resulted in the publication of GALL. GALL did not identify aging effects that require management for the external surface of stainless steel components or stainless steel in a gas environment.

RAI 3.2-4: In LRA Table 3.2-1, item 5, the applicant stated that, for the high-pressure safety injection pump mini-flow orifice, loss of material due to erosion is considered a design problem, and, therefore, does not have an identified aging management program. The applicant is requested to provide more information on this design problem, and the procedure, in place, to resolve the design problem.

VCSNS Response RAI 3.2-4

The high-pressure safety injection pump (Charging Pump) mini-flow orifices are made of stainless steel stock with a drilled hole approximately a foot long. This design is acceptable to prevent erosion from being an aging effect that requires management for the charging pump mini-flow orifices.

RAI 3.2-5: For components serviced by closed-cycle cooling system, in LRA Table 3.2-1, item 9, the applicant stated that the Chemistry Program (Appendix B.1.4) has proven effective in maintaining the system's 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 a one-time inspection, is not warranted for the components in this component group. The staff believes 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. The applicant is requested to justify that, under all circumstances, including the situation of susceptible locations in a system with potential slow and stagnant flow conditions, the Chemistry Program will be sufficient to manage the aging effects associated with the components.

VCSNS Response RAI 3.2-5

Table 3.2-1, Item 9, of the Application Table includes only the RHR heat exchanger and the RHR pump seal cooler. Both of these components have reactor coolant cooled by Component Cooling.

> **From the GALL Program Description of XI.M2 WATER CHEMISTRY:**
>
> ***"The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system."***
>
> **From the GALL Section VII.E1:**
>
> **"The effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated borated water and, therefore, are not included in this section."**

Neither GALL nor our operating experience reviews have found a reason to require one time inspection of these stainless steel RHR components in a borated water (RCS) environment. In addition, the treated water environment of these components contains corrosion inhibitors whose function is to form a passivated layer that prohibits corrosion from forming. Therefore, it is not necessary to schedule one-time inspections for the RHR Heat Exchangers and RHR Seal Coolers.

<u>RAI 3.2-6:</u> In LRA Table B-1, the applicant stated that the Bolting Integrity Program of GALL (XI.M18) 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 the 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 requests:

a. *Provide the basis for not considering loss of mechanical closure integrity an aging effect,*
b. Discuss what aging effects/mechanisms have been identified for the closure bolting, even if it is treated as a sub-component, and
c. Considering that closure bolting and its associated component may differ in their materials, environmental exposures, and potential modes of failure, 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.

VCSNS Response RAI 3.2-6

Class 1 bolted closures (as well as non-Class 1 closures for the Steam Generators and Reactor Coolant Pumps) are covered by specific ASME Section XI activities only associated with the Reactor Coolant system (i.e. harsher conditions) such that treatment of individual sub-components is warranted for license renewal.

For all other bolted closures (i.e. pressure-retaining) of components/component types subject to aging management review, the design of critical closure joint bolting involves enough redundancy to ensure joint integrity and no aging effects unique to bolting, over the components being joined/closed, require evaluation for license renewal as discussed further below.

Although identified as an aging effect in various industry references, loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for non-Class 1 component bolted closures (i.e. pressure boundary closures) within the scope of license renewal.

Mechanical components within the scope of license renewal, both Class 1 and non-Class 1, contain bolted closures that are necessary for the pressure boundary of the components being joined/closed. Examples of these bolted closures are valve bonnet to body, pump cover to casing, heat exchanger manway and channel head (end-bell), and piping flange sets. The bolted closure is comprised of two mating surfaces, a gasket, and a fastener set (studs or bolts, washers, and nuts). By themselves, the mating set, gasket, or fastener set have no component intended function. Together, the bolted closure forms an integral part of the pressure-retaining boundary of the component. Additionally, the bolted closure is exposed to the same environment(s) as the components in the plant areas where the closure is located (process fluid for internal mating surface and ambient conditions else). As such, the bolted closure (including fastener set) was considered to be a sub-component (piece-part) of the components/component types within the scope of license renewal and did not require evaluation separate from the component, except as clarified.

Loss of mechanical closure integrity can result in failure of the mechanical joint, is evidenced by leakage rather than joint failure, and can be attributed to one or more of the following effects:

- **Loss of bolt pre-load (embedment, cyclic load embedment, gasket creep, etc.),**
- **Loss of bolting material (from general and/or boric acid corrosion),**
- **Reduction of bolting material fracture toughness, and**
- **Cracking of high strength bolting material (SCC).**

For non-Class 1 bolted closures, loss of pre-load was considered to be the result of inadequate design or improper assembly (i.e. event driven) that is not related to aging and that would manifest itself during the current operating term and be corrected prior to the period of extended operation. Thus, the mechanisms associated with loss of bolting pre-load are not a license renewal concern for non-Class 1 components/component types.

It is recognized that loss of bolting material could ultimately result in the loss of a component's pressure boundary integrity and thus, requires evaluation for license renewal. However, loss of material is an aging effect requiring license renewal evaluation for carbon and alloy steel components/component types subject to aging management review. As such, no evaluation separate from the subject components/component types of which bolted closures are a part is necessary and, for carbon and alloy steel components/component types, the aging management programs credited for managing external general corrosion will also inherently address their fasteners.

Furthermore, stainless steel fasteners are immune to loss of material due to general corrosion and most bolting is normally in a dry environment and is coated with a lubricant, thus general corrosion of carbon and alloy steel bolting is not expected, nor has it been a major concern in the industry. As is the case with subject components/component types of similar material, the occurrence of general corrosion in carbon and low alloy steel fastener sets in the ambient environments is most likely in systems with operating temperatures below ambient conditions that result in condensation and in the yard environment with repeated wetting and drying from exposure to the elements.

Loss of material due to boric acid wastage (aggressive chemical attack) is the most common aging affect that has been observed in the industry for ferritic fasteners. In the concentrations used in PWR systems, boric acid is a relatively weak acid. However, under wetting and drying conditions, such as a result of leakage, boric acid may concentrate in a slurry forming a saturated solution. There appear to be no differences in the corrosion rates for the common carbon and low-alloy steel bolting materials. Stainless steel fasteners have been shown to be immune to loss of material due to boric acid wastage. The appropriate program/activities credited for management of the external aging of carbon and low-alloy steel in locations susceptible to leaking borated water, will also address carbon and low-alloy steel fasteners in that location. Additionally, note that aging management programs credited for addressing boric acid

wastage will also inherently address any general corrosion concerns for carbon or low alloy steel bolting of stainless steel components/component types.

Reduction of fracture toughness of bolting material, due to thermal/neutron effects is a license renewal concern for the fasteners of components only due to the associated elevated system operating temperatures and proximity to the reactor vessel beltline region. This is applicable to bolting of some Class 1 components and is addressed in the application. Reduction of fracture toughness for non-Class 1 bolting material is not a license renewal aging effect requiring management for the fasteners of components.

Stress corrosion cracking (SCC) of bolting materials, it is a condition in which a fastener that is statically loaded well below the material yield strength may suddenly fail. SCC bolted closure fastener failures have occurred in materials with apparently normal chemical and mechanical properties. Although there have been a few instances of cracking of bolting in the industry due to SCC, these have been attributed to high yield stress materials and contaminants, such as the use of lubricants containing MoS_2. VCSNS has not and does not use lubricants containing MoS_2. However, most bolting is normally in a dry environment and is coated with a lubricant; in general, environmental conditions that could lead to SCC of bolting are not expected to occur in non-Class 1 components. For quenched and tempered low alloy steels used for closure bolting (e.g., SA193 Grade B7), material susceptibility to SCC is minimized by having a lower yield strength. EPRI Report NP-5769 (Volume I, pg. 11-5) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified yield strength is below 150 ksi. The specification for the fabrication of nuclear piping, specifies alloy steel ASME SA 193, Class B7 bolts/studs and ASME 194 Grade 2H nuts, which have minimum yield strengths below 150 ksi (105 ksi). A minimum yield strength for bolting does not, in and of itself, preclude SCC since the actual yield strength of the bolt could be above the threshold value for SCC of low alloy steel bolting/fasteners to occur (150 ksi). However, sound maintenance bolt torquing practices can control bolting material stresses and the use of appropriate material (such as ASTM A193 Gr. B7) for bolting reduces the potential for SCC to occur. A review of industry failure databases and NRC generic communications, supports the fact that proper material selection, proper maintenance and torquing procedures, and removal of contaminants from lubricants have been effective in eliminating the potential for SCC of bolting materials. Therefore, SCC of bolting materials is not an aging effect requiring evaluation for license renewal for non-Class 1 components/component types.

3.3 AUXILIARY SYSTEMS

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 as well as other systems in the VCSNS. The applicant is requested to provide a description of the environments included in the LRA.

VCSNS Response RAI 3.3-1

The internal environments for subject components, also known as the process and/or service environment, were determined by reviewing the appropriate design and operating documentation, including system design basis documents, flow diagrams, vendor technical manuals, procedures, lesson plans, etc. General temperature ranges were not established, thus temperatures for internal environments are system specific. When performing AMR's, the temperature ranges were determined on a system-specific basis from the normal system operating ranges found on the 302 series system flow diagrams (P&ID) and are contained in the body of technical work, which is available for inspection. VCSNS is generally consistent in its internal environments with other stations and with the environments described in NUREG-1801. Internal environments for license renewal mechanical consideration for VCSNS include:

- **Air-Gas (including vacuum, compressed air, compressed gases, and exhaust gases),**
- **Borated Water (chemically treated borated water),**
- **Oil (such as fuel oil, lubricating oil, synthetic oil),**
- **Raw Water (water taken directly from a lake, reservoir or other open external source),**
- **Treated Water (including filtered and chemically treated water, condensate quality water, as well as steam),**
- **Ventilation (includes ambient building air that is contained and/or processed through ductwork, and other ventilation equipment)**

The following external environments are those to which the external portions of subject components are exposed due to the equipment location, and, therefore, require evaluation for license renewal considerations:

- **Reactor Building,**
- **Sheltered,**
- **Yard,**
- **Underground,**
- **Embedded (includes significant portions of components/component types embedded in concrete and not simply portions passing through building walls).**

For license renewal considerations, a "Sheltered" environment is considered to be the ambient conditions inside certain support buildings. These support buildings include the Auxiliary (AB), Control (CB), Intermediate (IB), Fuel Handling (FHB), Diesel Generator

(DB), Service (SB), and Turbine (TB) Buildings. A "Sheltered" environment also includes the Fire Pump House (FPH), and Service Water Pump House (SWPH).

The ambient environment for the Reactor Building and Sheltered environments are not considered to be humidity controlled. The design average maximum temperature for buildings at VCSNS is 120°F.

VCSNS is located well inland and is located in an area where forestry is the primary commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

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 the 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. However, 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 justification as to 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 sheltered environment. For carbon steel components exposed to external environments of moist air such as sheltered environment, the GALL report identified loss of material due to general, pitting, crevice corrosion and MIC as 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 sheltered environment.

VCSNS Response RAI 3.3-2

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the Service Water Pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside below the 425' elevation.

Building penetrations are inspected as part of the Maintenance Rule Structures Program (Application Section B.1.18). The VCSNS Corrective Action Program would be used disposition any groundwater in-leakage and resulting degradation.

Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

General corrosion is an aging effect for external surfaces in sheltered environments. It is managed by the Inspections for Mechanical Components.

RAI 3.3-3: This common RAI concerns the susceptibility to aging effects for stainless steel components in ambient environment in the 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. Provide your basis for determining significant concentration of contaminants 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 the reactor building or sheltered environments because of the presence of insignificant concentration of contaminants in these environments. Provide your basis for determining significant concentration of contaminants 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. Provide your basis for determining significant concentration of contaminants 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 the reactor building and sheltered environments because of the presence of insignificant concentration of contaminants in these environments. Provide your basis for determining significant concentration of contaminants and the verification/inspection activities on susceptible locations to justify this basis.

VCSNS Response RAI 3.3-3

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the service water pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside below the 425' elevation.

Building penetrations are inspected as part of the Maintenance Rule Structures Program (Application Section B.1.18). The VCSNS Corrective Actions Program would disposition any groundwater in-leakage and resulting degradation.

VCSNS is located well inland and is located in an area where forestry is the prime commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

System-specific RAIs

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 justification for your conclusion that galvanized steel ductwork in a 'yard' environment has no identified aging effects.

VCSNS Response RAI 3.3.2.4.1-1

VCSNS is located well inland and is located in an area where forestry is the primary commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

Zinc is used because of its corrosion resistance in an external environment, and by its galvanic protection of the base metal when the coating is damaged. The components in question are the air exhaust heads located on the roofs of the Control Building and the Intermediate Building. Because of the relative lack of traffic and activity in these areas, damage to the zinc coating is not expected beyond small nicks, which are protected by the self-healing properties of the zinc coating. General corrosion of galvanized steel is not an aging mechanism, because the ambient temperature in the area where these components are located is less than 140°F.

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 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.

VCSNS Response RAI 3.3.2.4.1-2

According to industry references, SCC of carbon and low alloy steel components is not considered to be an applicable aging mechanism in a treated water environment. Industry data does not exhibit widespread incidence of SCC in low strength carbon steels; however, there was a reported case suspected to be nitrate-induced SCC of carbon steel in a treated water system. VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrates are added as a corrosion inhibiter. In these closed systems there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrates are added as a corrosion inhibiter by the Chemistry Program at levels within EPRI guidelines; therefore, VCSNS maintains that the Chemistry Program adequately manages SCC of carbon steel components in a treated water environment.

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 justification, the above quoted statements in LRA Table 3.3-2, Item 11 and the AMR Query Notes "A-BS-c".

VCSNS Response RAI 3.3.2.4.3-1:

Note A-BS-c is incorrect. The note should read, "... there are *no* stainless steel components within the license renewal evaluation boundaries of the BS system ..."

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.

VCSNS Response RAI 3.3.2.4.4-1

The Above Ground Tank Inspection manages the internal surfaces of tanks. This Program inspects the tanks at the liquid / air-gas interface where degradation is more likely to be found due to alternate wetting and drying causing contaminants to concentrate. The Chemistry Program has proven effective in managing aging effects in the treated water environment. Prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

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 segments under insulation or external tank bottom surfaces are adequately managed.

VCSNS Response RAI 3.3.2.4.4-2

External surfaces of tanks will be managed by the Inspections for Mechanical Components program. Tank foundations will be inspected by the Maintenance Rule Structures program. The Chilled Water Expansion Tanks are elevated such that the tank bottoms are exposed. Also, VCSNS operating experience has identified an instance of pitting of an insulated external surface for the Chilled Water System; therefore, the Inspections for Mechanical Components program will involve removal of insulation of portions of the Chilled Water system.

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 stress corrosion cracking (SCC). The staff finds that the Chemistry Program may not be adequate to manage this aging effect because it does not contain a one-time inspection of these components at susceptible locations that verifies the absence of cracking and the effectiveness of the Chemistry Program. Justify the absence of an inspection/verification activity for the Chemistry Program.

VCSNS Response RAI 3.3.2.4.4-3

According to industry references, SCC of carbon and low alloy steel components is not considered to be an applicable aging mechanism in a treated water environment. Industry data does not exhibit widespread incidence of SCC in low strength carbon steels; however, there was a reported case suspected to be nitrate-induced SCC of carbon steel in a treated water system. VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrates are added as a

corrosion inhibiter. In these closed systems there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrates are added as a corrosion inhibiter by the Chemistry Program at levels within EPRI guidelines; therefore, VCSNS maintains that the Chemistry Program adequately manages SCC of carbon steel components in a treated water environment.

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 to not 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.

VCSNS Response RAI 3.3.2.4.6-1

Selective leaching is an applicable mechanism for copper alloys that do not contain inhibiting elements. In particular, copper-zinc alloys containing greater than 15% zinc, copper-aluminum alloys containing greater than 8% aluminum, yellow brass (30% zinc and 70% copper), and muntz metal (40% zinc and 60% copper) are all susceptible to selective leaching. Copper nickel does not fit these criteria; as such management is not needed in treated water, oil (fuel in wetted locations), or air/gas (wetted locations). In identifying aging effects for raw water environments, selective leaching was identified as an aging effect for copper-nickel in harsh raw water environments only, however, VCSNS has no in-scope copper-nickel components in a raw water environment.

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.

VCSNS Response RAI 3.3.2.4.6-2

IN 85-30 refers to MIC in internal environments. The Reactor Building environment concerns external surfaces in the ambient environment of the Reactor Building.

Plant 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 structural design of the plant is such that

any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the Service Water Pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types. The Reactor Building environment does not have these necessary conditions where MIC would be found on external surfaces. Piping, process tubing, or ductwork that penetrates the Reactor Building does not enter the building at locations where groundwater is found.

VCSNS is located well inland and is located in an area where forestry is the prime commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

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 as the applicable AMP. It should be noted that the likely material/locations determining galvanic corrosion rates depend on which specific metal/alloy is used, 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.

VCSNS Response RAI 3.3.2.4.6-3

The Chemistry Program is credited with maintaining a high purity environment that has low electrolyte conductivity by maintaining chemistry within EPRI guidelines. The Chemistry Program has proven effective in managing aging effects in the treated water environment as evidenced by the review of operating history in response to Generic Letter 89-13; however, prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

RAI 3.3.2.4.6-4: The 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.

VCSNS Response RAI 3.3.2.4.6-4

For closed, treated water systems, such as the CC system, contaminants have no way of entering the system other than corrosion of the system itself. Due to the continuous, turbulent flow through the shell of the CC heat exchanger, corrosion products will not settle out on tubes. The Chemistry Program will manage the conditions required for loss of material by maintaining chemistry within EPRI guidelines. The Chemistry Program has proven effective in managing aging effects in the treated water environment as evidenced by the review of operating history in response to Generic Letter 89-13; however, prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

3.3.2.4.7 Diesel Generator Services Systems

RAI 3.3.2.4.7-1: For the AMR results of the flexible hose and flexible coupling included in LRA Table 2.3-23, the applicant identified Table 3.3-1 Item 2 and Table 3.3-2 Item 26.

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.

LRA Table 3.3-2, Item 26 states that, internal surfaces of rubber components are not considered to be susceptible to degradation in fluid environments due to lack of excessive temperatures and to the change in material properties of elastomers being closely tied to external conditions such as ultraviolet radiation. Therefore, no aging management is required. Clarify what type of rubber is used and provide technical justification and operational history to demonstrate that internal surfaces for flexible hoses and other elastomers used in diesel generator services systems do not have aging effects of hardening, cracking, loss of strength, and wear from exposure to the process fluid.

VCSNS Response RAI 3.3.2.4.7-1

VCSNS maintains that wear is not an aging effect. The Diesel Generators are normally in standby for emergency use and are normally only started on a monthly frequency for

routine surveillance. Because of the "keep-warm" systems for the lube oil and jacket water systems in the diesel, elastomers in these DG systems will not experience extremes in temperatures; therefore, VCSNS considers that flexing of an elastomer due to thermal expansion does not contribute to any significant degradation. However, VCSNS will manage cracking due to radiation and thermal embrittlement through the Inspections for Mechanical Components, which will detect any significant degradation.

The material/environment combination for the elastomer components in the Diesel Generator Services (DG) System are: rubber/lube oil, rubber/treated water, and neoprene/air-gas.

The rubber is Buna N rubber. According to Table 5-15 of Corrosion Engineering, Third Edition, by Fontana, Buna N rubber exhibits excellent resistance to oil and good resistance to water absorption, but poor resistance to sunlight aging; therefore, external surfaces would show age related degradation before the internal surfaces. The neoprene conforms to ASTM SC 610AF. The neoprene component is in the inlet air piping, which is the same environment as ambient air; therefore, external surface conditions would be indicative of internal conditions. Thus, VCSNS maintains that the Inspections for Mechanical Components will adequately manage aging of these components during the period of extended operation.

RAI 3.3.2.4.7-2: No aging management program has been identified 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.

VCSNS Response RAI 3.3.2.4.7-2

Several programs manage galvanic corrosion for various material / environment combinations in the Diesel Generator Service Systems. These programs are the Chemistry Program, the Inspections for Mechanical Components, the Buried Piping and Tanks Inspections, and the Service Water System Reliability and In-Service Testing Program.

RAI 3.3.2.4.7-3: LRA Table 3.3-2, Item 10 indicates that exhaust piping and mufflers are thick-walled components and do not require aging management. The corrosion of carbon steel components exposed to condensation and alternate wetting and drying, such as the mufflers and piping used in the diesel exhaust, is affected by their orientation and the proper function of any installed drain traps. Does the exhaust contain any corrosive contaminants such as sulfur that may be corrosive to the material? Provide the basis for not inspecting the interiors of silencers/mufflers and exhaust piping for localized corrosion from sulfuric acid and condensation.

VCSNS Response RAI 3.3.2.4.7-3

The mufflers and muffler exhaust piping contain drain traps. The Diesel Generator Systems Inspections already includes carbon steel expansion joints exposed to engine

exhaust as components to be inspected. The scope of the program will be revised to include the mufflers and exhaust piping.

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.

VCSNS Response RAI 3.3.2.4.12-1

NRC GL 88-14 addressed the concerns of NRC IN 87-28. VCSNS responses to the recommendations of GL 88-14 resolved concerns for the quality of air supplied to safety related equipment. These responses can be found in the letter from O.S. Bradham to the Document Control Desk, NRC, dated February 2, 1989 (Reference 16): "Response to Generic Letter 88-14."

There are various carbon steel components of air systems that experience an ambient, moist air environment that are in scope for license renewal. The Service Air Systems Inspection manages the aging of these components. The Service Air System Inspection concerns these components that are not pertinent to quality of air supplied to safety-related equipment and, therefore, not pertinent to the concerns of NRC GL 88-14. The concern for these components is air as the internal environment only, not as the motive force for operation of components. These components concern the pressure boundary function of specific containment penetrations, containment hatch testing, and emergency air supply to the personnel hatches. Ambient moist air (not dried by an air dryer) is assumed to be the internal environment for these components.

3.3.2.4.14 Liquid Waste Processing System

RAI 3.3.2.4.14-1: In the LRA Table 3.3-1, Item 14 and the table entitled "Virgil C. Summer Nuclear Station Database Query", the applicant identified the aging effects on carbon steel and stainless steel heat exchanger/condenser components in the liquid waste processing system that are exposed to treated water and the corresponding aging management program. The applicant further stated that the AMR results for this group are consistent with GALL (VII.C2.2-a, C2.2.1) in material, environment, and aging effects. The applicant also stated that the Chemistry Program is considered to provide adequate management in lieu of the Closed-Cycle Cooling Water System Program that is recommended for this group by GALL. It should be noted that the AMP, Closed-Cycle Cooling Water System Program, includes preventive measures as well as surveillance testing and inspection. The applicant is requested to explain how the Chemistry Program alone (without One-Time Inspection to verify the effectiveness of the Chemistry Program) is considered to provide adequate management of the identified aging effects for these components.

VCSNS Response RAI 3.3.2.4.14-1

The heat exchanger/condenser components of the Liquid Waste Processing (WL) System are in scope for license renewal because they constitute a pressure boundary with the Component Cooling Water (CC) System. The two internal water environments for the in-scope WL components are: reactor grade, recyclable, borated water and treated water (CC System). The Liquid Waste Systems Inspections will manage aging of stainless steel components of the WL System exposed to borated water. The Chemistry Program will manage aging of the stainless steel and carbon steel components of the WL System exposed to treated water. The Chemistry Program has proven effective at managing aging degradation in the CC System as evidenced by the review of operating history in response to Generic Letter 89-13. Prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various treated water systems to demonstrate the effectiveness of the Chemistry Program.

<u>RAI 3.3.2.4.14-2:</u> The GALL report identifies 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. However, the applicant also stated that further evaluations were not recommended by GALL. Explain why the conclusion in the LRA is different from the GALL.

VCSNS Response RAI 3.3.2.4.14-2

All of the columns of Table 3.3-1 of the LRA, except for the last column, are NUREG-1801 listings. The last column is the VCSNS response. NUREG-1801 states that, for this AMR item, if there were an adequate "closed-cycle cooling water system" program then no "further evaluation is recommended." In the "Discussion" column, VCSNS discusses the adequacy of the Chemistry Program in managing aging instead of using a closed-cycle cooling water system program.

At VCSNS, cracking due to SCC is an aging effect for stainless steel components in treated water environments (i.e. heat exchangers cooled by the CC System). The Chemistry Program has proven effective at managing aging degradation in the CC System as evidenced by the review of operating history in response to Generic Letter

89-13. Prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various treated water systems to demonstrate the effectiveness of the Chemistry Program.

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 significant 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." Provide results from inspections performed that support this conclusion. If not, provide justifications for your conclusion.

VCSNS Response RAI 3.3.2.4.14-3

The subject Liquid Waste System valve is in scope because it is a boundary isolation valve for the Spent Fuel System. The piping where this valve is located is used once per refueling cycle (18 months) to pump down the refueling cavity. This small amount of cycling does not create a situation where the effects of alternate wetting and drying warrants aging management. This is demonstrated by the fact that the valve has no history of any degradation or failure.

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 performed to justify your conclusion.

VCSNS Response RAI 3.3.2.4.16-1

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the Service Water Pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork

component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside below the 425' elevation.

Building penetrations are inspected as part of the Maintenance Rule Structures Program (Application Section B.1.18). The VCSNS Corrective Action Program would disposition any groundwater in-leakage and resulting degradation.

VCSNS is located well inland and is located in an area where forestry is the prime commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

General corrosion of external surfaces of the Nuclear Sampling (SS) System will be managed by the Inspections for Mechanical Components. This program will consist of a visual inspection of surfaces for any degradation or abnormality.

<u>RAI 3.3.2.4.16-2:</u> 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.

VCSNS Response RAI 3.3.2.4.16-2

VCSNS considers boric acid corrosion to be an aging mechanism for carbon steel components in the Nuclear Sampling System. The Boric Acid Corrosion Surveillances will manage this aging mechanism. Table 2.3-30 of Section 2.3.3.16, Nuclear Sampling System, of the application refers to Table 3.3-1, Item 13 where the Boric Acid Corrosion Surveillances is discussed.

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.

VCSNS Response RAI 3.3.2.4.17-1

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the Service Water Pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside below the 425' elevation.

During the Integrated Plant Assessment for VCSNS it was deemed to be expeditious and conservative to assume that any plant system located in a sheltered environment was susceptible to MIC. This precluded the need to physically walk down or evaluate each system for this mechanism. As time permitted, systems would be evaluated or walked down to determine if indeed they were susceptible to MIC. If time did not permit, the assumption was conservative because the Maintenance Rule Structures Program looks at all walls and penetrations, and, therefore manages aging for any system at the susceptible locations. Although listed in the LRA as an aging mechanism for stainless steel pipe in the Radiation Monitoring (RM) System, the portions of the RM System in scope for license renewal are, in fact, not in locations where they would be susceptible to MIC.

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.

VCSNS Response RAI 3.3.2.4.18-1

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the service water pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside below the 425' elevation.

For the Reactor Makeup Water Supply System, only piping penetrates buildings; therefore, only piping is susceptible to MIC.

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.

VCSNS Response RAI 3.3.2.4.20-1

There are no elastomer components in the portions of the Station Service Air System that are in scope for license renewal.

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.

VCSNS Response RAI 3.3.2.4.20-2

For license renewal considerations, a "Sheltered" environment is considered to be the ambient conditions inside certain support buildings. These support buildings include the Auxiliary (AB), Control (CB), Intermediate (IB), Fuel Handling (FHB), Diesel Generator (DB), Service (SB), and Turbine (TB) Buildings. A "Sheltered" environment also includes the Fire Pump House (FPH), and Service Water Pump House (SWPH). There are some sheltered environments that do not house systems that contain borated water; therefore, in these particular sheltered environments boric acid corrosion is not an aging mechanism; however, VCSNS does consider boric acid corrosion as an aging mechanism for the Station Service Air system components in scope for license renewal. Table 2.3-34 of the LRA Section 2.3.3.20, Station Service Air system, refers to Table 3.3-1, Item 13, where the Boric Acid Corrosion Surveillances Program is discussed.

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. It should be noted that 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 and many of these factors are location-dependent. The applicant is requested to clarify whether the buried piping and tanks inspections are to be performed in areas with the highest likelihood on galvanic corrosion or are to be performed on an opportunistic basis. Provide justifications for either case.

VCSNS Response RAI 3.3.2.4.21-1

The buried portions of the Service Water (SW) System are all carbon steel; therefore, the only possible galvanic reaction would be between the wrapped/coated piping components and the soil. All buried SW components are therefore equally susceptible and the opportunistic basis of the Buried Piping and Tanks Inspection is sufficient. VCSNS coats and wraps underground components in accordance with site procedures, which are based on accepted industry standard AWWA C-203, 1973. Operating experience for the Diesel Generator Fuel Oil Storage Tanks revealed that negligible wall thinning had occurred thereby verifying that the techniques of coating and wrapping are effective.

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.

VCSNS Response RAI 3.3.2.4.21-2

For carbon steel components in the sheltered and Reactor Building environments, the Inspections for Mechanical Components will manage loss of material from galvanic and general corrosions. There are locations in sheltered environments where carbon steel components are susceptible to MIC. Loss of material due to MIC is managed for these susceptible components by the Maintenance Rule Structures Program.

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 identified components. 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 justification for this omission. If insignificant concentration of contaminants is part of the justification, provide the acceptance criterion and the verification/inspection activities on susceptible locations to justify your judgement.

VCSNS Response RAI 3.3.2.4.21-3

The susceptibility to external MIC is limited locally to the area of the interface with the pertinent wall where groundwater in-leakage can occur. Only piping, process tubing, and ductwork component types pass through building penetrations. For the stainless steel components of the Service Water (SW) System, only process tubing can meet these criteria in sheltered environments.

VCSNS is located well inland and is located in an area where forestry is the primary commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates. Because the ambient environment at VCSNS is not considered to be a corrosive environment, stress corrosion cracking, crevice corrosion, and pitting corrosion are not considered to be aging mechanisms to be managed for external surfaces of stainless steel components. There are no locations in the Reactor Building where stainless steel components of the SW System are exposed to groundwater in-leakage; therefore, these components are not susceptible to MIC. This is consistent with the operating experience reviews conducted at VCSNS.

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 for the embedded environment. If this environment involves concrete, corrosion of carbon steel components embedded in concrete through carbonation etc., is commonly known degradation process. Provide the basis for the concluding that no applicable aging effect exists for carbon steel components in this particular embedded environment.

VCSNS Response RAI 3.3.2.4.21-4

Corrosion of embedded steel is not significant if the concrete has a low water-to-cement ratio, low permeability, and designed in accordance with ACI standards (ACI-318 or ACI-349, depending on the building). The design and construction of structures at VCSNS meet these criteria; therefore, corrosion of embedded steel is not an aging effect requiring management at VCSNS.

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. However, loss of material due to MIC is not identified by the applicant as an aging effect for stainless steel components other than pipe and tubing. Provide justification as to 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.

VCSNS Response RAI 3.3.2.4.22-1

The susceptibility to external MIC is limited locally to the area of the interface with the pertinent wall where groundwater in-leakage can occur. Only piping, process tubing, and ductwork component types pass through building penetrations. For the stainless steel components of the Spent Fuel Cooling (SF) System, only pipe and pipe fitting components meet these criteria.

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.

VCSNS Response RAI 3.3.2.4.23-1

The susceptibility to external MIC is limited locally to the area of the interface with the pertinent wall where groundwater in-leakage can occur. Only piping, process tubing, and ductwork component types pass through building penetrations. For the stainless steel components of the Boron Thermal Regeneration (TR) System, only pipe and pipe fitting components meet these criteria.

SECTION 3.4: STEAM AND POWER CONVERSION SYSTEMS

RAI 3.4-1: In Section 2.3.4.10, the LRA lists turbine cycle sampling system components subject to aging management review but Section 3.4.1, which lists VCSNS steam and power conversion systems does not include the turbine cycle sampling system. Provide an explanation for this omission.

VCSNS Response RAI 3.4-1

Section 3.4.1 should indeed list the Turbine Cycle Sampling (WA) System as a steam and power conversion system. Table 2.3-46 for the WA System refers to Table 3.4-1, Items 5 and 7, which discusses the AMR results for the WA System components in scope for license renewal.

RAI 3.4-2: In Tables 2.3-38 thru 2.3-47, the LRA does not identify any steam and power conversion systems 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 steam and power conversion systems components that are managed for cumulative fatigue.

VCSNS Response RAI 3.4-2

Cumulative fatigue is considered to be a TLAA. It is discussed in Section 4 of the LRA - Time-Limiting Aging Analysis.

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 LRA 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 steam and power conversion systems 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 steam and power conversion systems 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, explain any differences.

VCSNS Response RAI 3.4-3

The NUREG-1801 Program concerns the RCS pressure boundary. For non-Class 1 components VCSNS utilized the method described in section 4.3.1.1.2 of NUREG-1800.

The methodology used at VCS includes any system or portion of system with operating temperatures greater than 220°F. The flow diagrams of non-Class 1 systems in scope for license renewal were reviewed for operating temperatures. The screened-in portions of these systems meeting the temperature threshold were further reviewed to determine the frequency with which the thermal cycles occurred. The in-scope systems that meet the temperature threshold are as follows: CS, RH, SI, SS, RC (non-class 1), AS, BD, EX, FW, MB, MS, and WA (see Table 2.2-1 of the LRA for definitions of System designators). For all of these systems except the SS System, the number of thermal cycles is related to the heat-up and cool-down of the plant (steam and primary), which, ideally, occurs once a cycle (18 months). Conservatively, if this occurred once a month for 60 years, then the total thermal cycles would only be 720, which is approximately one-tenth of the allowed 7000.

RAI 3.4-4: In Table 3.4-1, item 2, the LRA 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 LRA 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. The 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. The applicant is requested explain how operating experience will confirm that these aging effects will not occur during the period of extended operation or perform a one-time inspection of components based on severity of conditions, time of service, and lowest design margin as recommended by NUREG-1801, XI.M32, "One-Time Inspection." Also, this RAI applies to the valves in Table 3.4-2, item 5.

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

VCSNS Response RAI 3.4-4

Prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various treated water systems to demonstrate the effectiveness of the Chemistry Program for various material/environment combinations.

RAI 3.4-5: In Table 2.3-47, the LRA does not include blowdown system heat exchangers identified as within the scope of license renewal on Drawing D-302-771. Explain why these heat exchangers are not included in Table 2.3-47 and describe the aging management for these heat exchangers.

VCSNS Response RAI 3.4-5

The heat exchangers shown on Drawing D-302-771 are not the Steam Generator Blowdown System Heat Exchangers. They are the Steam Generator Blowdown Sample Coolers. These are in the Nuclear Sampling (SS) System, which is discussed in Section 2.3.3.16 and Table 2.3-30 of the LRA.

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. In Table 3.4-1, item 3, the LRA does not identify aging management of raw water exposure to AFW piping. In the discussion column, the LRA 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." 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. If any portions of the AFW system require aging management due to exposure to raw water, list the components and describe how aging will be managed.

VCSNS Response RAI 3.4-6

Although there are automatic isolation valves that isolate the Service Water (SW) System from the Emergency Feedwater (EF) System, there is a section of EF System piping (carbon steel) downstream of these automatic isolation valves that is filled from the SW System; therefore, a portion of the EF System is indeed exposed to a raw water environment. This piping is inspected under the activities described by the Service Water System Reliability and In Service Testing Program. This program will continue to effectively manage the aging effects for this section of piping for the period of extended operation.

<u>RAI 3.4-7:</u> In Table 3.4-1, item 4, the LRA states that 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. 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. Also, this RAI applies to the heat exchangers in Table 3.4-2, item 3.

VCSNS Response RAI 3.4-7

The Turbine Driven Emergency Feedwater Pump Oil Cooler is cooled by Emergency Feedwater from the discharge of the pump, such that the oil is cooled only when the pump is running. The oil temperature is therefore at ambient temperature when the pump is in its normal standby condition. Because the oil is always at or above (when the pump is running) ambient temperature, moisture does not condense out of the oil to pool in the reservoir. A review of the operating history for this reservoir revealed no degradation.

This line of reasoning is proven by the history of the Charging/SI pump oil reservoirs. When cooling was supplied continuously to the reservoir coolers, water was regularly found in the oil. When this design was changed so that cooling was supplied only when the pump was running and was stopped when the pump was in standby, water was no longer found in the reservoir.

RAI 3.4-8: In Table 2.3-46, the LRA 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.

VCSNS Response RAI 3.4-8

This system should have been included in the scope section of AMP B.2.11, "Inspections for Mechanical Components." The program will be revised to include the Turbine Cycle Sampling (WA) system components that are in scope for license renewal.

RAI 3.4-9: In Tables 2.3.38 thru 2.3.47, the LRA identifies "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. Explain why the valve bonnets are not affected by these aging effects or provide aging management for the bonnets.

VCSNS Response RAI 3.4-9

The choice of words, "valves (body only)" comes from 10 CFR 54.21(a)(1)(i). As defined in the body of technical work for the IPA (available on site for inspection) "valves (body only)" refers to body and bonnet.

RAI 3.4-10: In Table 3.411, item 6, the LRA states in the discussion column of the 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 systems piping, fitting, pump casings, and valve bodies. In Tables 2.3-38 and 2.3-40, the LRA does not identify 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 LRA does not identify aging management for Flow Accelerated Corrosion for the main steam system pump turbine (casing only). Explain why the LRA states it is consistent with NUREG-1801 but does not include the above components in the FAC program.

VCSNS Response RAI 3.4-10

EPRI considers that flow accelerated corrosion is not an aging effect requiring evaluation in Systems that are either highly oxygenated, superheated, single-phase flow below 200°F or operated less than 2% of the time.

The portion of the Auxiliary Boiler and Feedwater (AS) System in scope for license renewal only supplies steam to the evaporators and to the boric acid batch add tank. The evaporators are rarely used and the batch add tank rarely requires steam. Steam

flows less than 2% of the time in this line; therefore, loss of material due to flow accelerated corrosion is not an aging effect requiring management for the AS System.

The process fluid of the Emergency Feedwater (EF) System is less than 200°F; therefore, loss of material due to flow accelerated corrosion is not an aging effect requiring management for the EF System.

The component identified in Table 2.3-44 as the Main Steam System Pump Turbine (casing only) is the Turbine Driven Emergency Feedwater Pump casing. This pump is operated less than 2% of the time; therefore, loss of material due to flow accelerated corrosion is not an aging effect requiring management for the Turbine Driven Emergency Feedwater Pump casing.

RAI 3.4-11: The 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 LRA does not credit the Chemistry program for the following components: 1) heat exchanger tube and shell in Table 2.3-40, 2) tank (reservoir) in Table 2.3-40, and 3) 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. A one-time inspection should be used to verify the effectiveness of the Water Chemistry program if the component is not in the main steam system.

VCSNS Response RAI 3.4-11

1) & 2) The internal surfaces of the carbon steel shell and the external surfaces of the brass tubes of the Turbine Driven Emergency Feedwater Pump Oil Cooler are exposed to a lube oil environment. The oil cooler is cooled by Emergency Feedwater from the discharge of the pump, such that the oil is cooled only when the pump is running. The oil temperature is therefore at ambient temperature when the pump is in its normal standby condition. Because the oil is always at or above ambient temperature, moisture does not condense out of the oil to pool on any surfaces. This line of reasoning is proven by the history of the Charging/SI Pump Oil Reservoirs. When cooling was supplied continuously to the reservoir coolers, water was regularly found in the oil. When this design was changed so that cooling was only supplied when the pump was running and stopped when the pump was in standby, water was no longer found in the reservoir. A review of the operating history for this oil cooler and tank (reservoir) reveals no degradation. The internal surfaces of the brass tubes are exposed to a treated water environment. Table 2.3-40 refers to Table 3.4-2, Item 6, which discusses the Chemistry Program and the Heat Exchanger Inspections that will manage the aging of the tubes in the treated water environment.

3) Table 2.3-44 refers to Table 3.4-2, Item 4, which discusses the Preventive Maintenance Activities: Terry Turbine as the aging management program for the pump turbine (case

only). The turbine is normally in a standby condition; therefore, the normal environment for the turbine casing is ambient, moist air.

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 LRA 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 systems 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?

VCSNS Response RAI 3.4-12

The Turbine Driven Emergency Feedwater Pump Oil Cooler is the only steam and power conversion heat exchanger in scope for license renewal. The internal surfaces of the brass tubes are exposed to the treated water environment of the Emergency Feedwater System, which is the cooling medium. The possible aging mechanisms for this material/environment combination are crevice corrosion, galvanic corrosion, pitting corrosion, selective leaching, and stress corrosion cracking. Table 2.3-40 refers to Table 3.4-2, Item 6, which discusses the Chemistry Program and the Heat Exchanger Inspections that will manage the aging of the tubes in the treated water environment.

RAI 3.4-13: In Table 3.4-1, item 12, the LRA states that the AMP 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 may 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, explain how loss of material at the tank bottom is managed. Are periodic thickness measurements taken at the tank bottom to ensure the integrity of the tank bottom is maintained?

VCSNS Response RAI 3.4-13

The below grade foundation of the Condensate Storage Tank (CST) is comprised of a 4 feet thick slab of reinforced concrete, the top of which is 1 foot below grade. A reinforced concrete, circular ringwall that is 2 feet high and 2-½ feet thick connects to this slab and extends from the top of the slab to 1 foot above grade. The CST attaches to the top of this ringwall by a base ring flange, which is anchored to the ringwall by anchor bolts. All voids between the ringwall and base ring are grouted. The outer edge of the base ring is coated with cold plastic coal tar pitch flashing compound. Inside this ringwall, the CST sits on clean, dry sand bed (as originally poured), which extends from the top of the foundation slab to the top of the ringwall. There are four small ringwall drains penetrating the ringwall 1 foot below grade. These drains are semi-circular in shape with a 3 inch radius and are filled with clean, crushed stone to retain the sand within the ringwall.

Because of the grouting and flashing at the base ring, water intrusion to the tank bottom is not expected to occur at the base ring; however, any water intrusion would seep through the sand to the ringwall drains. The four ringwall drains also allow translation of water to and from the sand contained by the ringwall. In the unlikely event that the ground outside of the ringwall becomes moisture saturated for an extended period of time, the sand inside the ringwall could only saturate to grade level. The 1 foot of sand from grade level to the bottom of the tank would remain dry. Because the external surface of the bottom of the CST remains dry, it should experience no aging effects requiring management.

RAI 3.4-14: In Table 3.4-1, item 12, the LRA states that there is underground piping in the AFW system (emergency feedwater system at VCSNS) and that the Buried Pipe and Tanks Inspection program will manage the aging effects. Table 2.3-40 of the LRA, 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.

VCSNS Response RAI 3.4-14

Table 2.3-40 of the LRA should have included reference to Table 3.4-1, Item 12, in the AMR results for pipe. Table 3.4-1, Item 12, states that the Buried Pipe and Tanks Inspection is the credited program to manage aging for underground piping in the Emergency Feedwater System.

RAI 3.4-15: NUREG-1801 recommends a one-time inspection to verify the effectiveness of the Water Chemistry program for all components except those in the main steam system. Explain why a one-time inspection is not performed to verify the effectiveness of the Chemistry program for auxiliary boiler steam and feedwater system components in Table 2.3-38, gland sealing steam system components in Table 2.3-43, main steam dump system components in Table 2.3-45, and turbine cycle sampling system components in Table 2.3-46.

VCSNS Response RAI 3.4-15

The only license renewal function of the Gland Sealing (GS), Main Steam Dump (MB), and Turbine Cycle Sampling (WA) Systems is to provide a means of main steam isolation (when used in conjunction with components from various other Systems) for a steam line break coincident with failure of the Main Steam Isolation Valve. The components of these Systems provide this function in the Main Steam (MS) System environment; therefore, a one-time inspection is not required.

The portion of the Auxiliary Boiler and Feedwater (AS) System, in scope for license renewal, supplies steam to the evaporators and the boric acid batch add tank in the Auxiliary Building (AB). The normal supply for this steam is from the Reheat Steam (RS) System through a desuperheating valve which converts superheated steam to a high quality saturated steam for use in these components. Thus, the steam originates from

the MS System, passes through the high pressure turbine generator, then through the moisture separator reheater where moisture is removed and the steam is superheated for use in the low pressure turbines. A portion of the steam leaving the moisture separator reheater is converted by the desuperheater to saturated steam for use in the evaporators and the boric acid batch add tank. Because the steam originates as high purity steam in the MS System and is further reheated in the moisture separator reheaters, the environment of the in scope portion of the AS System should be considered as pure as the environment of the MS System; therefore, a one-time inspection is not required.

3.5 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

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.

VCSNS Response RAI 3.5-1

1) **Section 6.2 of TR00170-003 identifies electrical panels, cabinets, cable trays, etc. as being constructed of factory baked painted steel or galvanized sheet metal, both of which do not have a tendency to age with time due to general corrosion. VCSNS realized that these components are designed for outdoor service and industry operating experience has not shown a case where aging effects caused a loss of intended function. Therefore, these components in the Electrical Substation and Transformer Area were judged to have no aging effects from general corrosion due to an external environment.**

   Even though corrosion is considered unlikely, Attachment II of TR00170-003 will be revised for the external environment to include loss of material (for Cable Tray & Conduit and Electrical and Instrument Panels & Enclosures) as an aging effect which is managed by the Maintenance Rule Structures Program.

2) **The attributes of these materials (factory baked painted steel or galvanized sheet metal) were similarly deemed to provide additional protection from boric acid corrosion and thus judged to have no aging effects. However, Section 7.6 of TR00170-003, "Boric Acid Corrosion Surveillances" (Scope of Program) does include these electrical components under Boric Acid Corrosion Surveillances for**

managing aging effects (loss of material). Therefore, Attachment II of TR00170-003 will be revised for Reactor, Auxiliary, Intermediate, and Fuel Handling Buildings to include loss of material (for Cable Tray & Conduit and Electrical and Instrument Panels & Enclosures) as an aging effect which is managed by Boric Acid Corrosion Surveillances and Maintenance Rule Structures Program.

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.

Report TR00170-003, Revision 0, Attachment II also lists three (3) steel components in a concrete environment. 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. The staff position is that all accessible concrete requires aging management; this includes monitoring the condition of concrete surrounding anchorages and embedments.

AMR items 7 and 15 of LRA Table 3.5-1 indicate that for the concrete containment structure only certain aging effects require aging management. As an example, for accessible exterior concrete, only change in material properties due to leaching is identified as requiring aging management. It is the staff position that ASME Section XI, Subsection IWL should be credited for managing loss of material, cracking, and change in material properties for the concrete containment structure; and that inaccessible concrete (i.e., below grade) also requires 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

(a) verify that cracking, loss of material, and change in material properties will be managed in accordance with NUREG-1801, XI.S2, ASME XI, Subsection IWL for all accessible containment concrete components;
(b) identify the aging management programs that will manage loss of material, cracking, and change in material properties for all other concrete components in accessible areas;
(c) submit a quantitative assessment of the below-grade environment, comparing it to the specific criteria defined in GALL Volume 2;
(d) if it is non-aggressive, based on satisfaction of the specific criteria defined in GALL Volume 2, 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;

(e) if the below-grade environment does not satisfy the specific criteria defined in GALL Volume 2, describe in detail the plant-specific aging management programs for inaccessible concrete components.

VCSNS Response RAI 3.5-2

Application Section 2.1.2.2.3 states: "For concrete structures and structural components, VCSNS has used the Part 54 Process, NUREG-1801, and industry guidelines to determine those specific aging effects that are applicable and require aging management for the Extended Period of Operation (EPO). Recent positions by the NRC Staff have determined that all aging effects for concrete are credible and should be managed under the CLB programs for the EPO." -- The issue of managing all versus specific concrete aging effects for accessible areas is actually a moot point since the plant AMPs (Maintenance Rule Structures Program and Containment ISI Program - IWE/IWL) look for any concrete degradation, regardless of mechanism or effect. Therefore, the VCSNS AMPs are considered acceptable to evaluate aging of concrete elements of the Containment and other Class 1 Structures (which is the intent of the NRC Staff position).

The three steel component types identified in Attachment II of TR00170-003 [anchorage, anchorage/embedments (exposed surfaces), and embedments] are only related to aging effects for steel, and not for concrete. All accessible concrete (including that surrounding the steel anchorages and embedments) is accounted for under component type "Reinforced Concrete - Beams, Columns, Floor Slabs, Walls" which is managed under the Maintenance Rule Structures Program and Containment ISI Program - IWE/IWL.

(a) Concrete aging effects (cracking, loss of material, and change in material properties) will be managed at VCSNS in accordance with NUREG-1801, XI.S2, ASME XI, Subsection IWL (Application Section B.1.16) for all accessible containment concrete components.

(b) Concrete aging effects (cracking, loss of material, and change in material properties) will be managed at VCSNS in accordance with the Maintenance Rule Structures Program (Application Section B.1.18) for all other concrete components in accessible areas.

(c) Section 6.1 (Table 6.1-3) of TR00170-003 provides the quantitative assessment of the below-grade groundwater environment at VCSNS. These analyses results are based on samples taken in 2001 from three (3) wells in the general vicinity of plant structures. [Note that prior sample analyses for chlorides, sulfates and pH do not exist.] Groundwater chlorides (from all three wells) were determined to be < 10 ppm, which is well within the GALL defined limits of < 500 ppm. Groundwater sulfates (from all three wells) were determined to be < 10 ppm, which is well within the GALL defined limits of < 1500 ppm. Groundwater pH (from the three wells) was determined to range from 4.8 to 5.3, which marginally exceeds the GALL defined limits of 5.5. Based on these results, the VCSNS Application defines the site groundwater as non-aggressive, although mildly acidic.

(d) **Application Table 3.5-1, Item 17. specifies that periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. VCSNS Engineering Services Procedure (Inspections for Maintenance Rule - Structures) will be revised to include a chemical analysis of raw water (including groundwater) on a 5-year interval to coincide with the Maintenance Rule Structures Inspections. [Note that seasonal fluctuations are not applicable at VCSNS since the level of groundwater remains relatively constant due to the influence of Monticello Reservoir.]**

(e) **Application Table 3.5-1, Items 7 and 16, discusses aging mechanisms and effects for inaccessible concrete. Since the VCSNS below grade environment marginally exceeds the specific pH criteria defined in GALL, the concrete design was further reviewed and determined to provide protection against aggressive chemical attack. Since the below-grade structures are considered to be resistant to the mildly acidic environment, plant specific aging management programs are not required for inaccessible concrete areas.**

RAI 3.5-3: Report TR00170-003, Revision 0, Attachment II does not list O-rings for the containment airlocks and hatch or seals for fire/flood doors 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.

VCSNS Response RAI 3.5-3

1) **O-rings for the Containment equipment, personnel and escape hatches are generically included with Reactor Building Component Type "Compressible Joints and Seals" (Elastomers) in Attachment II of TR00170-003. The O-rings are inspected during each refueling outage for repair, replacement and/or lubrication, and subsequently tested under 10CFR50 Appendix J (Leak Rate Testing) as discussed in Appendix B.1.12 of the Application.**

2) **Fire/flood/pressure door seals or gaskets are subcomponents of the door and are not explicitly called out in scoping and screening, similar to mounting hardware (threshold, latches, strike plates, hinges, sills, etc.). Door gaskets are extruded closed cell sponge type, made of neoprene rubber or equal and conform to ASTM D1056, Grade 2C1 (VCSNS Specification SP-631). As evaluated in TR00170-003 Section 6.6, neoprene rubber's resistance to oils, chemical, sunlight, weathering, aging, and ozone is outstanding. It retains its properties at temperatures up to 250°F. No aging effects are expected. Although door gaskets were treated as subcomponents, fire/flood/pressure doors are managed under the Maintenance Rule Structures Program and Fire Protection Program as identified in Attachment**

II of TR00170-003. Detailed discussion on fire/flood/pressure doors is contained in Sections 7.10.2, 7.11, and 7.14 of TR00170-003.

RAI 3.5-4: Report TR00170-003, Revision 0, Attachment II identifies loss of material as the only aging effect requiring aging management, for pipe supports located in the auxiliary building; control building; intermediate building; diesel generator building; fuel handling building; reactor building; and service water structures. 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 verify (1) that loss of mechanical function is an applicable aging effect for ASME Class piping supports, and (2) that IWF is 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.

VCSNS Response RAI 3.5-4

Attachment II of TR00170-003 (for Component Type - Pipe Supports) refers to Section 6.9 for details on service-induced cracking and loss of mechanical function aging effect. Section 6.9 of TR00170-003 identifies an exception to GALL III.B1.2.2-a, whereby "loss of mechanical function" is not an aging effect, but rather a design issue. However, since the IWF program is capable of determining loss of mechanical function, TR00170-003 will be revised accordingly to:

1) **Identify loss of mechanical function as an applicable aging effect for ASME class piping supports, and**
2) **Identify ASME Section XI, Subsection IWF as the applicable aging management program.**

RAI 3.5-5: In the "Aging Management Programs" column of the Report TR00170-003, Revision 0, Attachment II Table, Technical Specification 3/4.9.10 is listed for the following component types in the fuel handling building: fuel transfer canal liner plate, spent fuel pool liner, and spent fuel storage rack; and Technical Specification 3/4.6.1 is listed for the following component types in the reactor building: personnel airlock, escape hatch, and equipment hatch. The staff requests the applicant to describe the objective, scope, and implementation procedures for each technical specification, as it relates to aging management for license renewal.

VCSNS Response RAI 3.5-5

1) **Attachment II (Reactor Building) of TR00170-003 lists Technical Specification 3/4.6.1 under AMPs for personnel airlock, escape airlock, and equipment hatch. This Technical Specification covers Limiting Conditions of Operation for Containment Air Locks, requiring periodic (6 month) tests of leakage rates. Verification of leakage rates ensures functional integrity, thereby supplementing**

Inspections conducted under the Containment ISI Program - IWE/IWL and 10CFR50 Appendix J Leak Rate Testing.

Type B LLRTs for the primary containment hatches are performed in accordance with plant procedures as identified in Section 7.2 of TR00170-003.

2) **Attachment II (Fuel Handling Building) of TR00170-003 lists Technical Specification 3/4.9.10 under AMPs for fuel transfer canal liner plate, Spent Fuel Pool liner, and spent fuel storage rack. This Technical Specification covers Limiting Conditions for Operation for water level in the Spent Fuel Pool, requiring at least 23 feet of water maintained over the top of irradiated fuel assemblies seated in the storage racks. Verification of the water level in conjunction with the Chemistry Program ensures a constant level in the pool and avoids continuous wetting/drying of steel components, thereby reducing aging effects. [Note that a recent revision to the VCSNS Technical Specifications relocated these requirements to Section 3/4.7.10.]**

RAI 3.5-6: 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:

(a) Identify whether underdrain (de-watering) systems are utilized at V. C. Summer.
(b) 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).
(c) Describe the fill concrete, including its strength, thickness, underground profile, and construction procedures. Also define the ground water level with respect to the fill concrete profile.
(d) Describe plant-specific-operating experience concerning settlement of buildings resting on fill concrete.
(e) Discuss whether fill concrete performs an intended function; and, as appropriate, submit an aging management review, including identification of credited aging management program(s)

VCSNS Response RAI 3.5-6

(a) Underdrain (de-watering) systems are not used at VCSNS.

(b) Since underdrain (de-watering) systems are not used at VCSNS, there are no monitoring or maintenance activities or functional requirements.

(c) **Fill concrete (1500 psi minimum compressive strength at 28 days) was used as a leveling mat to construct the structural foundations for the Reactor, Auxiliary and Control Buildings. The fill concrete design is not porous, rather it just has a higher water-cement ratio than the higher strength structural foundation mixes of 3000 psi and greater. The fill concrete was placed directly on clean, competent bedrock, which has a design allowable bearing capacity of 200 ksf (1389 psi). The fill concrete was placed in minimum 5' lifts on an irregular rock surface, with thickness varying from approximately 5' to 50' (depending on varying elevations of rock surface and structure foundations). The levels of fill concrete range from approximately elevation 344' to 407', while the nominal groundwater elevation is at elevation 423'. [Refer to VCSNS FSAR Sections 2.5 and 3.8.]**

(d) **There has been no operating experience at VCSNS concerning settlement of buildings resting on fill concrete. The initial design of the fill concrete determined that post-construction settlement would be practically nil, since only minimal settlement would occur from the initial construction loads.**

(e) **The fill concrete does not perform an intended function since it was designed to be equivalent to rock as an underlying base, and is not evaluated under any aging management programs.**

<u>RAI 3.5-7:</u> 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.

VCSNS Response RAI 3.5-7

1) **At VCSNS, there have been no leaks detected from the Spent Fuel Pool, thus, no operating experience.**

2) **A complete discussion related to the Chemistry Program controls for the Spent Fuel Pool is contained in Section 7.7 of TR00170-003. Refer also to RAI 3.5-5 response on monitoring Spent Fuel Pool water level.**

<u>RAI 3.5-8:</u> 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 to credit its IWF aging management program for aging management of lubrite plates, consistent with GALL.

VCSNS Response RAI 3.5-8

An extensive review of Class 1, 2, 3 supports at VCSNS discovered only two (2) Class 1 pipe hanger supports (both inside containment) which incorporated small lubrite slide plates (as discussed in Section 6.8 of TR00170-003). TR00170-003 also contains the technical basis supporting our basic conclusion that lubrite plates do not require aging management. However, all Class 1, 2, 3 supports are inspected under the ASME Section XI ISI Program - IWF (Application Section B.1.13), which provides an acceptable aging management program for aging management of lubrite plates, consistent with the GALL.

<u>RAI 3.5-9:</u> 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:

(a) 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?

(b) 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.

(c) 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.

(d) For AMR item 16, explain the reference to two (2) aging management programs that are only applicable to the containment structure.

(e) 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?

(f) 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.

(g) 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?

VCSNS Response RAI 3.5-9

(a) For containment penetration closures, the flat plate is basically no more than an extension of the containment liner plate which connects to the penetration sleeve. The containment liner plate was reviewed for fatigue analysis as originally calculated for 40 years. The calculation was subsequently revised to show that fatigue analysis was acceptable for 60 years. Application Section 4.6.1 discusses the TLAA review for the containment liner for which VCSNS utilized 10 CFR 54.21 (c) (1) Option (ii) to demonstrate that liner fatigue is adequately analyzed for the period of extended operation.

(b) As discussed in Application Table 3.5-1, Item 8, the VCSNS containment foundation is constructed directly on fill concrete over competent bedrock and is not subject to settlement; therefore, aging management is not required. [See response to RAI 3.5-6.] However, regardless if settlement is not considered as an applicable aging mechanism, existing AMPs (10 CFR 50 Appendix J General Visual Inspection, Containment ISI Program - IWE/IWL, and Maintenance Rule Structures Program) will still be used to look for concrete aging effects such as cracks and distortion.

(c) As discussed in Application Table 3.5-1, Item 15, the VCSNS containment structure does not consider freeze-thaw and reaction with aggregates as applicable aging mechanisms; therefore, aging management is not required. However, regardless if freeze-thaw and reaction with aggregates are not considered as applicable aging mechanisms, existing AMPs (10 CFR 50 Appendix J General Visual Inspection, Containment ISI Program - IWE/IWL, and Maintenance Rule Structures Program) will still be used to look for concrete aging effects such as cracks and distortion.

(d) Consistent with GALL, Application Table 3.5-1, Item 16, is applicable to all Class 1 Structures (including containment), except for Group 6 (water-control structures). Regardless of applicability of aging mechanisms, the AMPs identified are used to look for all aging effects. Therefore, as noted in the RAI, the two AMPs (Containment Coating Monitoring and Maintenance Program and Containment ISI Program – IWE/IWL) are both specific to the containment structure. The Maintenance Rule Structures Program is applicable to all Class 1 Structures, including containment.

(e) The basis of the note for steel tanks in Application Table 3.5-1, Item 24, is from a structural perspective for inspections of exterior tank surfaces, foundations, and anchorages. Steel tanks at VCSNS do not have stainless steel liners. However, SCC and crevice corrosion are monitored from a mechanical perspective, with details contained in Application Sections B.2.1 and B.2.11.

(f) For Application Table 3.5-1, Item 25, the Maintenance Rule Structures Program applies to all subcomponents for all structures (including containment), while the 10 CFR 50 Appendix J General Visual Inspection is used only as a supplementary inspection program for containment. As noted in the discussion for Item 25, cracking due to fatigue is not an aging effect requiring management for concrete components.

(g) Application Table 3.5-1., Item 28, should have credited the ASME Section XI ISI Program - IWF (Application Section B.1.13) as the primary AMP for aging management of the ASME Class 1, 2, 3 supports. The IWF AMP is credited in Attachment II of TR00170-003. The other two programs referenced are included as supplementary programs for additional inspections and are not intended as a substitute for IWF.

<u>RAI 3.5-10:</u> LRA Section 3.5.1.1 indicates that the reactor building foundation mat bears on fill concrete that extends to competent rock, and that 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 further 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 non-aggressive in LRA Table 3.5-1. It is not clear to the staff whether the retaining wall serves an intended function and is subject to an aging management review. Therefore, the applicant is requested to submit the following information related to the retaining wall:

(a) Describe in detail the primary function(s) for the retaining wall.
(b) Discuss the consequences of its failure on structures and components that serve intended functions.
(c) If the retaining wall serves an intended function, submit the aging management review for the retaining wall, including the aging management programs credited to manage aging.
(d) Otherwise, submit the technical basis for concluding that the retaining wall serves no intended function.

VCSNS Response RAI 3.5-10

(a) **The retaining wall exists along the northeast quadrant of containment (between the Intermediate and Fuel Handling Buildings) and separates the below grade portions of the containment wall from the subgrade. The design function of this wall is to provide accessibility to the exterior concrete surface of the containment structure (above the structural foundation level), primarily for access to the horizontal tendon buttress end caps.**

(b) **Hypothetical failure of the retaining wall onto the containment wall and/or horizontal tendon end caps would not result in any failures of the safety-related structures or components. The consequences of such failure would be primarily economical for repair and replacement.**

(c) **There are no intended functions for the retaining wall as defined for license renewal. It is not intended to provide shelter or protection, structural or functional support, or flood protection for safety-related equipment. Therefore, aging management and aging management programs are not required.**

<u>RAI 3.5-11:</u> 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 has addressed 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?

VCSNS Response RAI 3.5-11

Attachment II of TR00170-003 notes that there are no aging effects identified for stainless steel in air environment. However, the Maintenance Rule Structures Program does provide a complete walkdown of the interior of containment, focusing primarily on interior structural components (which does include refueling canal, sumps and screens). Therefore, the Maintenance Rule Structures Program is credited for managing aging of these components.

<u>RAI 3.5-12:</u> 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 the discussion on page 59 of the report. 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 the information presented, it is not clear to the staff which regions currently experience temperatures above 150°F; whether these are area temperatures or localized temperatures around hot piping penetrations; and how aging effects due to elevated temperatures will be managed. Therefore, the applicant is requested to provide the following information:

(a) Explain the apparent inconsistency between LRA Table 3.5-1, AMR item 10 and the information in Report TR00170-003, Rev. 0 (see above discussion).
(b) For all structures in the scope of license renewal, identify all regions that currently experience temperatures in excess of 150°F.
(c) If there are regions that currently experience temperatures in excess of 150°F, indicate whether these are area temperatures or localized temperatures around hot penetrations.
(d) If any area temperatures exceed 150°F and/or any localized temperatures exceed 200°F, how will change in material properties of concrete due to elevated temperatures be managed during the period of extended operation?

VCSNS Response RAI 3.5-12

(a) VCSNS does not believe that there is any inconsistency between Application Table 3.5-1, Item 10, and TR00170-003, Table 6.1-1 and Section 6.4. The Application states that containment <u>concrete</u> elements are not exposed to temperatures, which exceed the <u>thresholds</u> for degradation. These thresholds are consistent with the guidance provided in GALL, which defines elevated temperatures as greater than 150°F general and 200°F local (GALL II.A1.1-h). Specifically:

Table 6.1-1 of TR00170-003 lists a maximum temperature of 157°F for the area above the reactor head but below operating floor elevation 463' (which is an open area above the vessel). This specific area above the head has no direct contact or support with concrete from the surrounding primary shield walls. Therefore, the general area temperature for the concrete would actually be less than 157°F. Regardless, the reactor vessel should be considered as a large hot pipe within the penetration opening of the massive primary shield walls, which would allow a local maximum temperature of 200°F.

Section 6.4 of TR00170-003 states that the Control Rod Drive Mechanism (CRDM) is maintained at a temperature less than or equal to 170°F. The CRDM is supported by the reactor vessel head and extends upwards in an area away from surrounding concrete of the primary shield walls. For the same reason as stated above, these temperatures are considered to be localized and do not exceed the threshold value of 200°F.

(b) Temperatures for all structures in the scope of license renewal are identified in Table 6.1-1 and Section 6.4.1 of TR00170-003. Regions exceeding 150°F have been discussed in Response (a) above.

(c) As discussed above, these areas are considered to be localized temperatures.

(d) Since these temperatures fall within the industry accepted thresholds, there are no changes in material properties of concrete expected; therefore, aging management is not required.

RAI 3.5-13: 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.

It is also unclear to the staff why the non-conformance (NCN) discussed in LRA Appendix B.1.12.1 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).

Therefore, the staff requests the applicant to provide the following information:

(a) What inspections have been conducted to assess the condition of the liner embedded in the concrete base?
(b) Confirm that the nonconformance discussed in LRA Appendix B.1.12.1 was detected prior to the implementation of the B.1.16 aging management program. If not, explain why this nonconformance was not detected under the B.1.16 aging management program.
(c) Explain why this nonconformance was not detected under the B.1.11 aging management program.
(d) Clarify the scope of and interaction between all three aging management programs (B.1.11, B.1.12 and B.1.16).
(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.
(f) Since this type of degradation has already occurred, what is the technical basis for concluding that it could not occur again?
(g) Clarify whether the supplemental requirements of 10 CFR 50.55a for inaccessible areas are credited for LR aging management of the inaccessible liner plate.

VCSNS Response RAI 3.5-13

(a) Application Section B.1.12.1 states: "A non-conformance (NCN) was documented for rust found on the Reactor Building liner plate adjacent to the moisture barrier and for 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. 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." —A more in-depth inspection of the liner was not warranted as a result of this NCN, since the liner was found to have an insignificant reduction in thickness in the areas of observed rusting. [Note that additional inspections of inaccessible areas would have been warranted if any significant liner degradation had been found in these accessible areas.] Future inspections of the moisture barrier and adjacent liner will be conducted under the Containment ISI Program - IWE/IWL (B.1.16).

(b) The observed liner rusting and degradation of the moisture barrier was identified in 1999 during outage walkdowns by engineering and QC personnel. Such walkdowns have been conducted for many years and <u>preceded</u> the implementation of the Containment ISI Program - IWE/IWL. Inspection of the moisture barrier is now part of the Containment ISI Program - IWE/IWL. The NCN is discussed in Application Section B.1.12 (Appendix J Leak Rate Testing Program) since the other major containment inspection programs were not conducted during that particular outage.

(c) This NCN was identified during normal outage walkdowns and not detected under the 10 CFR 50 Appendix J General Visual Inspections (B.1.11) since they were not required during the 1999 outage. The last prior General Visual Inspection was conducted in 1997. [See Response d) below.]

(d) The 10 CF 50 Appendix J General Visual Inspection (B.1.11) is conducted two times in the 10 year period preceding the Type A ILRT. The 10 CFR 50 Appendix J Leak Rate Tests (B.1.12) (Type A, B, C) are conducted in accordance with established frequencies per regulation. The Containment ISI Program - IWE/IWL (B.1.16) was initiated in 2000 and will be conducted on a 5-year frequency.

(e) No further inspections of the liner were warranted as a result of this NCN, since the liner was found to have an insignificant reduction in thickness in the areas of observed rusting. UT examinations were conducted at various locations around containment in areas where rusting was observed, along with a control area not showing rust. The moisture barrier was removed in these areas, the rust/paint was mechanically cleaned, and UT examinations were made at floor level and several inches into the annulus gap. The UT examinations showed no significant loss of liner thickness when compared to the design thickness, with results well within the ± 10% fabrication tolerance of the liner. Consistent with the current provisions of ASME Section XI - IWE/IWL, additional inspections of inaccessible

areas would have been warranted if any significant liner degradation had been found in these accessible areas.

(f) **The moisture barrier is an elastomer, which is subject to degradation (splitting, separation, etc.) over time; therefore, this type degradation could occur again. The current AMPs (Containment ISI Program - IWE/IWL and Maintenance Rule Structures Program) have proven effective in managing the condition of the moisture barrier in order to preclude any significant degradation of the liner.**

(g) **The supplemental requirements of 10 CFR 50.55a for inaccessible areas are credited for license renewal aging management of the inaccessible liner plate. [Note that additional inspections of inaccessible areas will be warranted if any significant liner degradation is found in adjacent accessible areas during future inspections.]**

RAI 3.5-14: LRA Table 2.4-2 indicates that the aging management review results for numerous component types in 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," AMR item 13 lists four aging management programs in the "Discussion" column. These are 10 CFR 50 Appendix J General Visual Inspection; Containment Coating Monitoring and Maintenance Program; Containment ISI Program - IWE/IWL; and Maintenance Rule Structures Program. Report TR00170-003, Rev. 0, Attachment II only credits the coating monitoring and maintenance program for aging management of the containment liner plate. It is not apparent to the staff which aging management programs are being credited for which components. Therefore, the applicant is requested to clarify the following items:

(a) 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, and are the coatings managed by a coating monitoring and maintenance program?

(b) For each of the component types covered under AMR item 13, identify which of the four (4) aging management programs are credited for license renewal.

VCSNS Response RAI 3.5-14

(a) **Application Table 3.5-1, Item 13, includes all steel component types inside containment since this is the only GALL item identified for this component category. In general, all of these components are covered by all of the AMPs. The 10 CFR 50 Appendix J General Visual Inspection, Containment ISI Program - IWE/IWL, and Maintenance Rule Structures Program all look for component degradation, and since most steel components are painted, degradation of their protective coating provides an initial indication for more detailed inspections. The Containment Coating Monitoring and Maintenance Program looks for coating degradation throughout (regardless of specific component) and is therefore a subset of the other AMPs.**

(b) **The four (4) listed AMPs apply to all components in Application Table 3.5-1 (Item 13). AMPs are generic programs that look for all types of degradation throughout and do not list individual components on a specific checklist for a specific program.**

RAI 3.5-15: 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).

VCSNS Response RAI 3.5-15

1) **VCSNS based its original Tendon Surveillance Program on proposed Revision 3 of Reg. Guide 1.35 dated April 1979, although it was in a "proposed" status. The Guide remained in this status until July 1990 when the finalized Revision 3 was issued. However, on April 28, 1989, the NRC accepted the VCSNS Tendon Surveillance Program based on the proposed Revision 3 of Regulatory Guide 1.35. In March 1995, the NRC issued a new rule, 10 CFR 50.55a, which invoked the requirements of the ASME Code, Section XI, Subsections IWE and IWL, 1992 Edition and 1992 Addenda. The Reactor Building tendon prestress is monitored and programmatically controlled under plant procedures and specifications as discussed in Section 7.17 of TR00170-003.**

2) **The "partial" match was selected in Report TR00170-003 due to the fact that the GALL credits the 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda of ASME Code Section XI, Subsection IWL for managing containment post-tensioning system. Whereas, the VCSNS Tendon Surveillance Program was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda.**

RAI 3.5-16: 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 appear to 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 appear to applicable only to the welded flat plate closures, if the penetration bellows are not part of the containment pressure boundary. Therefore, the applicant is requested to provide the following information:

(a) Explain why cracking of the stainless steel penetration bellows (and the associated dissimilar metal welds) does not affect the bellows' intended function to "provide structural and/or functional support for process piping on the outboard side of containment."

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

VCSNS Response RAI 3.5-16

(a) Bellows on hot piping penetrations do not perform a pressure boundary function, but they do provide structural and/or functional support (i.e., thermal and accident movement of the process pipe). In the unlikely event that cracking of the stainless steel penetration bellows occurs, thermal and accident movement of the process pipe would not be impaired and therefore will not diminish the bellows' intended function. Loss of Material and Cracking aging effects for penetration bellows have been screened out due to the plant specific design and bellow protection features. Refer to Sections 6.2, 6.9, and 8.3 of TR00170-003 under Bellows.

(b) Stainless steel bellows are very compliant (flexible), therefore sustained high tensile stress does not exist. However, cracking of the bellows due to SCC would not impair the intended function. All containment penetrations are inspected (both inside and outside containment) as part of the Containment ISI Program - IWE/IWL.

RAI 3.5-17: 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, and 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:

(a) Verify that loss of leak tightness in a closed position, resulting from mechanical wear of locks, hinges, and closure mechanisms, is an applicable aging effect requiring management for the containment personnel airlock, escape airlock, and equipment hatch.
(b) Identify the aging management programs that are credited to manage this aging effect.
(c) Indicate whether Technical Specification 3/4.6.1, which is referenced as an aging management program for the personnel airlock, escape airlock, and equipment hatch in Report TR00170-003, Rev. 0, Attachment II, allows any deviations from the requirements specified in 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.

VCSNS Response RAI 3.5-17

(a) As stated in Application Table 3.5-1, Item 5, VCSNS does not consider mechanical wear as an applicable aging effect since locks, hinges and closure mechanisms are active components. However, leak tightness of the airlocks and hatches (in the closed position) will be managed for License Renewal for the extended period of operation via inspections conducted under maintenance surveillance activities, Containment ISI Program IWE/IWL, 10 CFR 50 Appendix J General Visual Inspection, 10 CFR 50 Appendix J Leak Rate Testing and Technical Specification 3/4.6.1 requirements.

(b) Technical Specification 3/4.6.1 does not allow any deviations from the requirements of the Containment ISI Program - IWE/IWL. It conservatively supplements the requirements for operability.

AGING MANAGEMENT PROGRAMS

<u>RAI 3.5-18:</u> 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:

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

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

VCSNS Response RAI 3.5-18

(a) ASME Section XI ISI Program - IWF (Application B.1.13) — The clarification is no more than a statement that VCSNS uses the 1989 Edition of ASME Section XI Code with no Addenda. This is consistent with GALL XI.S3 which states that the

evaluations cover the 1989 Edition through the 1995 edition with addenda through the 1996 Addenda, as approved in 10 CFR 50.55a. VCSNS has not attempted to reconcile code differences nor requested approval for later editions.

(b) Containment ISI Program - IWE/IWL (Application B.1.16) — The clarification is no more than a statement that VCSNS uses the 1992 Edition of ASME Section XI Code with 1992 Addenda. This is consistent with GALL XI.S2 which states that the evaluations cover both the 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda, as approved in 10 CFR 50.55a. VCSNS has not attempted to reconcile code differences nor requested approval for later editions.

RAI 3.5-19: 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:

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

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

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

Maintenance Rule Structures Program (B.1.18)

Service Water Pond Inspection Program (B.1.21)

Tendon Surveillance Program (B.3.3)

The aging management program descriptions in LRA Appendix A should 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. Issues related to the FSAR supplement are being addressed by the staff on a generic basis.

VCSNS Response RAI 3.5-19

1) **As stated in the Application, VCSNS maintains a 10 CFR 50 Appendix J Leak Rate Testing Program (B.1.12) using Option B, which is consistent with GALL XI.S4 and RG 1.163.**

2) **As stated in the Application, VCSNS maintains an ASME Section XI ISI Program - IWF (B.1.13), which is consistent with GALL XI.S3 and 10 CFR 50.55a. This RAI concludes that Application Section 18.2.5 is inconsistent by only monitoring loss of material and cracking. However, Section 18.2.5 represents only a very general overview (summary) of program content, which is consistent with the guidelines of NEI 95-10. VCSNS maintains an extensive IWF program in accordance with ASME Section XI, which evaluates all of the identified aging effects, plus others not cited.**

3) **As stated in the Application, VCSNS maintains an ASME Section XI - IWE/IWL Program (B.1.16), which is consistent with GALL XI.S1, XI.S2, and 10 CFR 50.55a.**

4) **As stated in the Application, VCSNS maintains a Maintenance Rule Structures Program (B.1.18), which is consistent with GALL XI.S6 and 10 CFR 50.65. Several enhancements to this program have been identified during the license renewal evaluation process and are listed in the Application (B.1.18).**

5) **As stated in the Application, VCSNS maintains a Service Water Pond Dam Inspection Program (B.1.21), which is consistent with GALL XI.S7 and RG 1.127. One enhancement to this program was identified during a NRC/FERC inspection as identified in the Application and discussed in Section 7.15 of TR00170-003.**

6) **As stated in the Application, VCSNS maintains a Tendon Surveillance Program (B.3.3), which is consistent with GALL X.S1, XI.S2, and 10 CFR 50.55a.**

7) **VCSNS does not believe that there are any further changes required for the Application Appendix A, since only summary statements are recommended by NEI 95-10. Commitment to all Regulations and Regulatory Guides are implicit in the development of each of these programs as described in Section 7 of TR00170-003.**

<u>RAI 3.5-20:</u> 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.

VCSNS Response RAI 3.5-20

(a) The 10 CFR 50 Appendix J General Visual Inspection is only one component of the total Appendix J Program. Under Appendix J, a visual examination of accessible interior and exterior surfaces of the containment system shall be conducted during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration. These inspections are conducted by Operations personnel and provide additional inspections for aging management of the containment, thus being credited for license renewal. This program is only used as a supplement, not substitute, for the Containment ISI Program - IWE/IWL.

(b) There are no elements of containment inspection (IWE/IWL) which rely solely on the 10 CFR 50 Appendix J General Visual Inspection.

(c) The AMPs listed in Attachment II (Reactor Building) of TR00170-003 are the primary programs used to manage aging. The 10 CFR 50 Appendix J General Visual Inspection is only used as supplemental information and does not substitute for any of the programs identified.

<u>RAI 3.5-21:</u> 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. In Report TR00170-003, Rev 0, Section

6.8.6, the applicant indicates that SCC is unlikely at VCSNS for the reasons identified therein, but further states "Regardless, the examination requirements of ASME Section XI ISI Program - IWF manage loss of function and cracking due to SCC for the Class 1 component supports that are exposed to the Reactor Building environment." However, IWF visual inspection would be expected to identify only relatively large cracks, as noted in GALL XI.S3. 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. Therefore, the staff requests the applicant to (1) identify all plant-specific applications of high strength bolting in Class I piping and component supports; (2) specifically describe the plant-specific operating experience related to stress corrosion cracking of high-strength bolting materials used in Class I piping and component supports; (3) describe the plant-specific resolution of the generic safety issue related to bolting integrity, including a description of any inspections/tests conducted as part of the resolution; and (4) if cracking due to SCC is an applicable aging effect, describe the inspections, in addition to IWF visual inspection, that will be credited to manage this aging effect.

VCSNS Response RAI 3.5-21

1) **High strength bolting materials are used in various Class 1 piping and component supports which are identified in Section 6.8 of TR00170-003.**

2) **There is no plant specific operating experience related to Stress Corrosion Cracking (SCC) of high strength bolts at VCSNS.**

3) **VCSNS followed resolution of the generic safety issue as an EPRI member. EPRI Report NP-5769 states that utilities with bolting materials with specified yield strengths greater than 150 ksi should review their individual applications. A review and discussion on susceptibility of high strength bolting to SCC at VCSNS is contained in TR00170-003 (Sections 6.8 and 7.3), concluding that SCC is unlikely; therefore, not considered an applicable aging effect requiring management. Regardless, the examination requirements of ASME Section XI (IWF) are in place to adequately manage loss of function and cracking due to SCC for the Class 1 supports that are exposed to the Reactor Building environment.**

4) **The intent of the discussion in Application Section B.1.13.1 is that SCC is not considered an applicable aging effect for VCSNS requiring management. See additional discussions in Section 6.8.5 of TR00170-003. Regardless, the ASME Section XI ISI Program - IWF does exist in order to manage this aging effect. VCSNS also maintains an IWA program in accordance with ASME Section XI, which evaluates corrosion effects.**

RAI 3.5-22: 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.

VCSNS Response RAI 3.5-22

(a) As noted in Application Section B.1.17, "The VCSNS Flood Barrier Inspection Program is identified for completeness since it contains individual components that have the unique [sole] function of mitigating the effects of internal flooding. All flood barrier components are managed by either the Fire Protection Program or Maintenance Rule Structures Program." Component types include: concrete curbs, designated water-tight doors, and designated penetration seals. Flood barriers are located throughout the plant. See detailed discussion on flood barriers in Section 7.11 of TR00170-003.

(b) There are currently no plant procedures written specifically for inspection of flood barriers. This program was added for license renewal to specifically capture those elements that only serve a flood protection function. There are many fire barriers (structures, doors, seals, etc.) that also serve as flood barriers and many structural components that serve as flood barriers, all of which are covered by the Fire Protection Program and Maintenance Rule Structures Program. Plant procedures for the Maintenance Rule Structures Program (B.1.18) will be enhanced to include inspections for flood barrier seals in the Control, Intermediate, and Diesel Generator Buildings in order to capture all flood barriers within the plant.

(c) Section 7.11 of TR00170-003 agrees with the Application that there are flood seals in the Intermediate Building (IB). However, Attachment II (IB) of TR00170-003 and Application Table 2.4-7 (IB) do not list flood barriers (similar to the CB and DGB). The VCSNS Drains/Sumps DBD (Section 3.8.5.4) identifies one (1) flood barrier at the IB / TB interface. Therefore, flood barriers will be added as a line item to Attachment II (IB) of TR00170-003.

(d) Flood barriers, which are not covered by the Fire Protection Program, are reviewed as part of the Maintenance Rule Structures Program which is conducted on a 5-year frequency.

RAI 3.5-23: 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:

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.

(a) 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.

(b) 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.

(c) 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."

(d) The commitment to incorporate the enhancements to this program discussed in LRA Section B.1.18 should also be included in the FSAR Supplement, Appendix A, Section 18.2.22. This section does not currently include such a commitment. Issues related to the FSAR supplement are being addressed by the staff on a generic basis.

VCSNS Response RAI 3.5-23

The Maintenance Rule Structures Program (B.1.18) includes visual inspection of concrete for all aging effects including loss of material, cracking, and change in material properties. This program requires walkdowns of all Important to Maintenance Rule Structures at VCSNS. Walkdowns are conducted by qualified engineering (structural) personnel. Plant procedures and guidelines (as described in Section 7.12 of TR00170-003) define inspection details and criteria for identifying aging mechanisms and effects.

(a) Inspection of the North Berm will be performed under plant engineering services procedures, consistent with the inspections required under the Service Water Pond Dam Inspection Program (RG 1.127), which is consistent with the attributes of GALL XI.S7.

(b) By plant design, there are no masonry walls located within safety related structures; therefore, VCSNS had no actions associated with IEB 80-11 and IN 87-67. However, masonry walls in non-safety related structures are inspected under the Maintenance Rule Structures Program, consistent with the attributes of GALL XI.S5. [Also see response to RAI 2.4.1-4.]

(c) The sentence in Application Section B.1.18.1 should read: "...including the protection and support of safety-related systems and components."

(d) Consistent with NEI 95-10, VCSNS does not see the need to include these minor enhancements into the very generic summary description of the Maintenance Rule Structures Program (Application Appendix A Section 18.2.22).

RAI 3.5-24: 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) Attachment II to TR00170-003 credits the Maintenance Rule Structures Program as the aging management program for pressure doors. However, the descriptions of the Pressure Door Inspection Program in LRA Section B.1.20 and Section 7.14 of TR00170-003 do not mention the Maintenance Rule Structures Program. Please clarify the relationship between these two programs and clarify which program is credited for LR aging management of pressure doors.

(c) 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.

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

VCSNS Response RAI 3.5-24

(a) Pressure and Fire doors are located throughout the plant. Pressure doors, which also serve as Fire doors, are inspected under the Fire Protection Program. Pressure doors which only serve a pressure function are inspected under plant maintenance procedures for quality related pressure barrier / components. See discussion in Section 7.14 of TR00170-003.

(b) Attachment II of TR00170-003 lists component types: "Flood, Pressure and Specialty Doors", all of which are in general inspected under the Maintenance Rule Structures Program. The RAI observation is correct that either the Fire Protection Program or the Pressure Door Inspection Program (both of which will be added to Attachment II of TR00170-003) inspects pressure doors in detail.

(c) **Application Section B.1.20 provides more explicit details related to inspection criteria (including excessive wear of door appurtenances), while Appendix A Section 18.2.24 provides only a very generic description of the program. The summary provided in Appendix A should not be interpreted as eliminating wear from inspections. Consistent with NEI 95-10, VCSNS does not see the need to make these two sections read the same.**

(d) **Pressure doors, which serve as Fire doors are inspected every 6 months, while Pressure doors, which only serve a pressure function, are inspected every 18 months.**

RAI 3.5-25: 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) The commitment to incorporate the enhancements to this program discussed in LRA Section B.1.21 should also be included in the FSAR Supplement, Appendix A, Section 18.2.31. This section does not currently include such a commitment. Issues related to the FSAR supplement are being addressed by the staff on a generic basis.

(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.

VCSNS Response RAI 3.5-25

(a) **Consistent with NEI 95-10, VCSNS does not see the need to include these minor enhancements into the very generic summary description of the Service Water Pond Dam Inspection Program (Application Section 18.2.31).**

(b) **The East Dam of the Service Water Pond (SWP) is the smallest and least critical (important) of the four SWP dams since it primarily caps a natural high elevation ridge line along the east side of the pond. There are no piezometers or alignment/survey monuments for this structure. The East Dam is inspected as part of the Service Water Pond Dam Inspection Program. There are no operating experience issues associated with this dam other than normal observations of minor erosion and weed growth at the edges of the riprap protection.**

RAI 3.5-26: 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.

VCSNS Response RAI 3.5-26

(a) For the Service Water Pumphouse (SWPH) and Intake Structure (SWIS) at VCSNS, the <u>primary</u> inspections for aging management are the Underwater Inspection

Program (B.1.23) and the Maintenance Rule Structures Program (B.1.18). These two programs provide a very detailed review and documentation of these structures. The Service Water Pond Dam Inspection Program (B.1.21) also incorporates walkdowns of the SWPH (at different frequencies) using the attributes of GALL XI.S7. Use of a statement that B.1.21 is supplementary does not imply that program attributes have been omitted.

(b) The program attributes discussed in Application Section B.1.23 are primarily focused on the SWIS since explicit inspection criteria have been incorporated as part of the CLB Operating License conditions for VCSNS. Detailed Engineering Services inspection procedures and acceptance criteria have also been developed. The diver's inspection of the SWPH (controlled by Plant Support procedures) serves primarily as a maintenance clean-up of the pump bays and screens, with instructions to look for any signs of degradation.

(c) The VCSNS procedures for inspection of the SWIS are focused on cracking within the concrete tunnel. Divers are also instructed to look for any structural damage such a concrete spalls or pieces on the floor. These are the primary "attributes associated with aging" which can be identified via diver inspections. The divers recover all trash/debris in the tunnel such that if concrete pieces were recovered, engineering would require additional inspections.

(d) The detailed divers inspection of the SWIS occurs every five years, and includes an underwater inspection of the SWPH. The underwater cleanup inspection of the SWPH bays and screens occurs every outage (18 months).

(e) As previously discussed, the diver's inspection of the SWPH is primarily for clean up of the pump bay and screen areas. The divers inspect for corrosion and fouling accumulations. Recovery of any unusual debris (such as pieces of concrete) would lead to additional inspections.

(f) The conclusion of Application Section B.1.23 was primarily focused on concrete inspections; however, the paragraph preceding the conclusion summarizes the effectiveness of the inspections as related to steel components in fluid environment.

g) The NRC Staff has also verbally questioned the scope of a VCSNS calculation concerning settlement of the Service Water Intake Structure as identified in Application Section 4.7.4:

1) TR00170-003 (Section 7.16) provides additional background information on the original settlement issues for the Service Water Pumphouse and Intake Structure. Additionally, Section 8.4 of TR00170-003 identified that a VCSNS Calculation was revised to account for the period of extended operation (60 years), with the results showing that the [theoretical] expected settlement is acceptable for the extended period of operation.

2) **The VCSNS Calculation was originally developed to compute settlement due to secondary consolidation of the underlying fill materials. [During the LR review process, this calculation was identified under TLAA criteria since it used a design life of these structures of 40 years.] This calculation identifies secondary compression bounding values of the underlying fill layers ranging from 0.66 - 1.12 inches per Log Time Cycle. These values were then multiplied by the Log Time Cycle of Log t_1 / t_0 = Log 40 / 1 = 1.60 (Note that t_0 was then at 1 year after construction was complete). This provides a theoretical settlement range at 40 years of 1.06" - 1.80". The calculation conclusion predicts that a further 1" - 2" settlement could occur. — For license renewal, this calculation was revised for 60 years to show a Log Time Cycle of Log 60 / 1 = 1.78. This provides a theoretical settlement range of 1.17" - 1.99". Therefore, these results are considered to be bounded by the original analysis.**

3) **Note that settlement of these structures has been monitored since construction and settlement due to secondary consolidation to date has been insignificant.**

RAI 3.5-27: 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) describe the plant-specific operating experience related to leakage and/or flooding in the tendon access gallery, and identify whether the tendon access gallery is also included in the Operations "trend logs" to prevent excessive water level; (4) indicate whether draining of the auxiliary building tendon sump area 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.

VCSNS Response RAI 3.5-27

1) **The Tendon Access Gallery (TAG) runs 360° beneath the circular containment wall and only houses the vertical tendon lower end caps. There is no structural connection of the TAG with the Auxiliary Building. The VCSNS containment structure is designed with vertical, horizontal (hoop) and dome tendons, using a three-buttress system (spaced at 120°) to anchor the horizontal tendons. One containment concrete buttress (308° azimuth) is located adjacent to (and within) the Auxiliary Building, which extends into a lower pit area (providing access to the lower horizontal tendon end caps).**

2) **As noted in Response 1), only horizontal tendon end caps are located within the Auxiliary Building sump area.**

3) **There is no operating experience at VCSNS concerning flooding in the TAG. The TAG has experienced continuing (since construction) groundwater in-leakage via cracks and construction joints along the outer wall; however, this leakage is a very slow infiltration, which is easily accommodated by the sumps and drains. [The TAG is a totally separate area within the plant and has drains, sumps and pumps to maintain the area relatively dry.] The TAG has no operating experience of any significant water accumulation; therefore, Operations trend logs are not required to prevent excessive water level accumulation.**

4) **Only one horizontal tendon (at the bottom of the 308° buttress) has been subjected to standing water; therefore, the commitment for Operations to monitor this area has not been credited for managing aging of the tendon prestressing system. Aging management will continue to be controlled via the Tendon Surveillance Program.**

5) **This lower area at the 308° buttress was not originally intended to be a sump, but rather a recessed area for access to the horizontal tendons, and does not have a drainage system installed (which would now be extremely costly to install); therefore, monitoring and pumping are significantly more cost effective. The Tendon Surveillance Program manages this particular problem without the cost of a plant modification.**

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 these calculations or analyses were evaluated against the TLAA definition provided in 10 CFR 50.3.

VCSNS Response RAI 4.1-1

The following items are those listed in NUREG-1800 that are potentially applicable to PWRs but were not addressed in Section 4 of the Application.

Metal Corrosion Allowance: VCSNS does not have any calculations or analyses that utilize a metal corrosion allowance.

Inservice flaw growth analysis that demonstrate structure stability for 40 years: VCSNS does not have any calculations or analyses that evaluate flaw growth relative to structural instability.

Inservice local metal containment corrosion analyses: VCSNS does not have any calculations or analyses that evaluate corrosion of the metal containment liner. This liner is bonded to the concrete containment wall and is protected by coatings on the interior of containment. Discussions on rusting of the liner plate at the moisture barrier are contained in the response to RAI 3.5-13.

Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding. Low-temperature overpressure protection (LTOP) analyses: VCSNS does not have any calculations or analyses that evaluates intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding. VCSNS did evaluate under clad cracking in the IPA for license renewal and found that it is not an aging effect requiring management for the reactor vessel. See the VCSNS response to RAI 3.1.2.2.7-1. Revisions to the pressure temperature curves are generated when vessel specimen are analyzed. This analysis also includes the Low-temperature overpressure protection (LTOP) analyses.

Flow-induced vibration endurance limit, transient cycle count assumptions, and ductility reduction of fracture toughness for the reactor vessel internals: VCSNS does not have any calculations or analysis that evaluates flow-induced vibration endurance limit, transient cycle count assumptions, and ductility reduction of fracture toughness for the reactor vessel internals.

Containment penetration pressurization cycles: VCSNS does not have any calculations or analyses that evaluate containment penetrations relative to pressurization cycles.

4.3 METAL FATIGUE

RAI 4.3-1: Section 4.3.1 of the LRA indicates 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:

a. 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.

b. 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 for 60 years.

c. A comparison of the thermal fatigue transients monitored to 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.

VCSNS Response RAI 4.3-1

The results of the last cycle counting data (dated 01/07/2003) are attached (Attachment Error! Reference source not found. **for Year 2002 Fatigue). This data includes all of the Class 1 thermal fatigue transient that are monitored for cycles and CUF. Westinghouse performed this Class 1 fatigue analysis using WESTEMS.**

Components monitored by WESTEMS for cycle count:

- **Reactor Vessel Outlet Nozzle Loop 1, 2, and 3**
- **Reactor Vessel Inlet Nozzle Loop 1, 2, and 3**
- **Reactor Vessel Structural Shell**
- **RCS Loop 1, 2, and 3, Hot Leg Piping**
- **RCS Loop 1, 2, and 3, Cold Leg Piping**
- **Loop 1, 2, and 3, Steam Generator Primary Side**
- **Loop 1, 2, and 3, Steam Generator Feedwater Nozzle**
- **Loop 1, 2, and 3, Reactor Coolant Pump Casing**
- **Pressurizer Upper Shell**
- **Pressurizer Spray Nozzle**
- **Auxiliary Spray Piping (ASME Code Class 1)**
- **Letdown Piping (ASME Code Class 1)**
- **Excess Letdown Piping (Section 1-1)**
- **Safety Injection Piping to Loop 1, 2, and 3, Cold Leg**
- **SI Accumulator piping to Loop 1, 2, and 3, Cold Leg**
- **Residual Heat Removal piping to Loop 1 and 3**

Components monitored by WESTEMS for CUF:

- **Normal Charging Nozzle Location 4 (Bounds cycle count components Normal Charging Piping and Normal Charging Nozzle to RCL)**
- **Alternate charging Nozzle Location 4 (Bounds cycle count components Alternate Charging Piping and Alternate Charging Nozzle to RCL)**
- **Pressurizer Lower Head Nozzle Location 2 (Bounds cycle count component Pressurizer Lower Shell)**
- **Pressurizer Surge Line to RCL Nozzle Location 19 (Bounds cycle count components Pressurizer Surge Piping Nozzle to RCL and Pressurizer Surge Piping)**

- **Pressurizer Surge Line to RCL Nozzle Location 25 (Bounds cycle count components Pressurizer Surge Piping Nozzle to RCL and Pressurizer Surge Piping)**
- **Pressurizer to Surge Line Nozzle Location 153-Mb (Bounds cycle count components Pressurizer Surge Nozzle, Pressurizer Lower Head Location 1 and Pressurizer Surge Piping)**
- **Pressurizer to Surge Line Nozzle Location 51-Mb (Bounds cycle count components Pressurizer Surge Nozzle, Pressurizer Lower Head Location 1 and Pressurizer Surge Piping)**

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's 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.

VCSNS Response RAI 4.3.1-2

The code of record for the internals is ASME Section III, Class 2, which did not specify time or cycle dependent requirements based on the current license term. (A list of the cycle counting for ASME Class 1 components is being provided as part of the response to RAI 4.3-1.)

The VCSNS aging management program B.2.4, Reactor Vessel Internal Inspection, will monitor the components listed in WCAP-14577 Table 3-3.

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's 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 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.

VCSNS Response RAI 4.3.1-3

The cast austenitic stainless steel materials used at VCSNS meets the requirements to screen out thermal aging as an effect requiring aging management.

Stress corrosion cracking is managed by a combination of material and process controls, ISI, and chemistry controls.

General corrosion, wear and stress relaxation were identified for flange bolting. In WCAP-14575A, flanges and bolting are used for the pressurizer safety valves and RTD bypass piping. At VCSNS the RTD bypass piping has been eliminated. VCSNS does use Class 1 flanged connections in the head vent piping. The head vent connections are dissembled each refueling. Each pressurizer safety valves connection is dissembled every third refueling. After these components are dissembled, the bolting is inspected, the connection reassembled and the connections are tested in accordance with ASME requirements.

The Thermal Fatigue Management Program manages fatigue issues. (A list of the cycle counting for ASME Class 1 components is being provided as part of the response to RAI 4.3-1.)

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's 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 the applicant to 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:

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

b. 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.

c. 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 Design Curves of Austenitic Stainless Steels," on the above results.

VCSNS Response RAI 4.3.1-4

Westinghouse performed our Class 1 fatigue analysis for year 2002 and the results (dated 01/07/2003) are attached (Attachment Error! Reference source not found. **for Year 2002 Fatigue). This data includes all of the class 1 thermal fatigue transient that are monitored for cycles and CUF. Westinghouse performed this Class 1 fatigue analysis using WESTEMS.**

Pressurizer components monitored by WESTEMS for cycle count:

- **Pressurizer Upper Shell**
- **Pressurizer Spray Nozzle**

Pressurizer components monitored by WESTEMS for CUF:

- **Pressurizer Lower Head Nozzle Location 2 (Bounds cycle count component Pressurizer Lower Shell)**
- **Pressurizer Surge Line to RCL Nozzle Location 19 (Bounds cycle count components Pressurizer Surge Piping Nozzle to RCL and Pressurizer Surge Piping)**
- **Pressurizer Surge Line to RCL Nozzle Location 25 (Bounds cycle count components Pressurizer Surge Piping Nozzle to RCL and Pressurizer Surge Piping)**
- **Pressurizer to Surge Line Nozzle Location 153-Mb (Bounds cycle count components Pressurizer Surge Nozzle, Pressurizer Lower Head Location 1 and Pressurizer Surge Piping)**
- **Pressurizer to Surge Line Nozzle Location 51-Mb (Bounds cycle count components Pressurizer Surge Nozzle, Pressurizer Lower Head Location 1 and Pressurizer Surge Piping)**

The present (year 2002) CUF for the pressurizer surge line is 0.38. The year 2000 CUF for the pressurizer surge line was 0.35. Changes have been made to the operating procedures to slow the accumulation of CUF on the pressurizer surge line nozzle.

For the NUREG/CR-6260 locations, VCSNS will evaluate the Fatigue Environmental Effects prior to the period of extended operation. VCSNS will evaluate the fatigue usage for components with a methodology that is approved by the Staff. The present approved methodology is to use the correlations contained in NUREG/CR-6583, for Carbon and Low-Alloy Steels and NUREG/CR-5704, for Austenitic Stainless Steels. The approved methods will be incorporated into the Thermal Fatigue Management Program prior to the period of extended operation.

RAI 4.3.1-5: Section 4.3.1 of the LRA discusses SCE&G's 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 of 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:

a. 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 Design Curves of Austenitic Stainless Steels," will be used in the evaluation.

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

c. 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.

VCSNS Response RAI 4.3.1-5

For the NUREG/CR-6260 locations, VCSNS will evaluate the Fatigue Environmental Effects prior to the period of extended operation. VCSNS will evaluate the fatigue usage for components with a methodology that is approved by the Staff. The present approved methodology is to use the correlations contained in NUREG/CR-6583, for Carbon and Low-Alloy Steels and NUREG/CR-5704, for Austenitic Stainless Steels. The approved methods will be incorporated into the Thermal Fatigue Management Program prior to the period of extended operation.

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.

VCSNS Response RAI 4.3.2-1

Only the RC Loop 'B' hot leg sampling portion of the Nuclear Sampling (SS) System could approach the 7,000 cycle limit for thermal fatigue. This tubing and piping extends from the RC System up to and including the tubing for the sample cooler for this sampling loop. Tubing and piping downstream of this cooler does not meet the system operating temperature threshold of 220°F as set forth in the EPRI Fatigue Management Handbook, TR-104534. The portion of the Nuclear Sampling System known as post-accident sampling is downstream of this cooler and, therefore, is not subject to thermal fatigue because the system operating temperature is below this threshold.

The portion of the SS System that may approach the cycle limit is composed of austenitic stainless steel tubing, austenitic stainless steel piping (delay coils), and Inconel 600 tubing (sample cooler tubing). There is no stress analysis for this system. Gilbert/Commonwealth designed the system per criteria designed to meet ASME code requirements.

The present sampling method seldom uses loop sampling. VCSNS will administratively limit the use of this loop for emergencies only, for an average of no more than once per

cycle. In so doing, the 7,000 cycle limit will not be reached during the period of extended operation. Should sampling methods change such that this loop would be required for more frequent sampling, either a detailed stress analysis would be required that increases the cycle limit to an acceptable limit or the affected components would be replaced.

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.

VCSNS Response RAI 4.5-1

a. Plots of the measured lift-off forces and trend lines, along with the minimum required values for the three sets of tendons (vertical, horizontal, and dome) are provided in Attachment XII. Guidance of IN 99-10 has also been implemented at VCSNS.

b. Based on elongation tests performed at Leheigh University for VCSNS tendon wire samples, it was found that stress relaxation of the tendons was not linearly proportional to temperature as originally projected based on manufacturer data. Therefore, stress relaxation was increased from 8.5% to 12.5% based on these tests. SCE&G is not aware of any unique characteristics of the VCSNS tendons or containment design that would cause this to occur, nor operating experience of similar behavior.

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.

VCSNS Response RAI 4.6-1

The penetration sleeve is welded to the liner via a flat plate connection, while the process pipe is welded directly to the penetration sleeve (inside containment only).

The hot piping going through the containment penetrations is classified as ASME Code Class 2. In the analysis of this Class 2 piping, the Code does not require a detailed fatigue analysis. However, an analysis is performed. In this piping, fatigue is considered in the equations used in the analysis of the piping by the use of a Stress Range Reduction Factor (f) for cyclic conditions. This factor is applied to the allowable stress limit for the calculated pipe stress due to thermal expansion. For VCSNS, this factor assumes that all Class 2 piping will be subjected to 7000 thermal cycles of heatup and cooldown from ambient to the pipe normal operating temperature. Refer to Application Section 4.3.2.

Penetrations are considered as an ASME Code pressure boundary component. They were all designed in accordance with a plant Design Specification (DSP) that provides data sheets for each penetration with temperature and cyclic information. In the Section on loadings, this DSP states, "The expansion stresses are calculated from the temperature ranges presented in the data sheets. The cyclic aspect is also presented to enable determination of the appropriate value for (f), the Stress Range Reduction

Factor." Therefore, the same analysis was performed for the penetrations as was performed for the piping.

The penetration sleeve forms a rigid anchor point embedded within the concrete containment walls, while the process pipe is welded to the end of the penetration sleeve inside containment. The penetration sleeve thus provides separation (insulation) between the process pipe and liner; therefore, there is no direct influence of the effects of the hot process pipe thermal transients and other cyclic loads on the liner.

AGING MANAGEMENT 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.

VCSNS Response RAI B.1.19-1

Application Section B.1.19 identified that wear on crane rails has been determined to not require aging management for VCSNS cranes. The basis of this position is contained in TR00170-003 (Section 6.9) which states that wear of crane rails due to rolling or sliding wheels is not expected in any measurable amount due to infrequent crane use.

Although "wear" is not considered an applicable aging mechanism at VCSNS, the Material Handling System Inspection Program (B.1.19) is capable of detecting loss of material on crane rails from corrosion and/or wear. Accordingly, plant procedures do include inspection of rails for "abnormal wear" as part of the aging management 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.

VCSNS Response RAI B.1.19-2

A detailed discussion of the Material Handling System Inspection Program AMR is contained in TR00170-003, Section 7.13, including identification of all VCSNS cranes which are within license renewal scope.

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.

VCSNS Response RAI B.1.25-1

The Preventive Maintenance Activities - Terry Turbine Program is the license renewal name for the preventive maintenance activity already being routinely performed on the component that can be credited for managing aging during the period of extended operation. The activity that inspects the Terry Turbine (Turbine Driven Emergency Feedwater Pump) is performed every third refueling outage (4.5 years).

B.1.26 Preventative Maintenance Activities - Ventilation Systems Inspections

RAI B.1.26-1: Under *Monitoring and Trending*, the LRA states "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.

VCSNS Response RAI B.1.26-1

The Preventive Maintenance Activities - Ventilation Systems Inspections is the license renewal name for the diverse preventive maintenance activities already being routinely performed on components that can be credited for managing aging during the period of extended operation. The preventive maintenance performed on the Component Cooling water pump motor cooling coils is conducted on 10-year intervals. The preventive maintenance performed on the fan coils for air handling units are conducted on semi-annual intervals. The preventive maintenance performed on the Reactor Building Cooling Units is conducted at intervals less than 5-years.

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.

VCSNS Response RAI B.1.26-2

The preventive maintenance performed on the RBCUs includes visually inspecting inlet and outlet coil faces as a step in performance testing. Performance testing involves trending temperatures for the RBCUs.

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).

VCSNS Response RAI B.2.2-1

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this aging effects identification research are contained within the body of technical work at VCSNS supporting the LRA. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS. Operating experience at VCSNS for the components managed by this program reveals no history of degradation for the internal surfaces.

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 is 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.

VCSNS Response RAI B.2.3-1

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this aging effects identification research are contained within the body of technical work at VCSNS supporting the LRA. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS.

The stainless steel ND System components managed by this program function as pressure boundary for a containment penetration. This penetration is used for the transfer of water out of the Reactor Building sump and the Incore Instrumentation Sump. These sumps contain leakage only from systems located inside the Reactor Building (i.e. high purity Systems). These components are leak tested under 10CFR50 Appendix J.

Operating experience at VCSNS for the ND components managed by this program reveals no history of degradation for the internal surfaces.

The stainless steel WL System components managed by this program either function as pressure boundary for a containment penetration or function as pressure boundary with the Component Cooling (CC) System.

The WL components that function as pressure boundary for containment isolation are leak tested under 10CFR50 Appendix J. The contents of the Reactor Coolant Drain Tank (RCDT) are transferred through this containment penetration. Because of the purity of the Systems that are the sources of the water to the RCDT, the water in the RCDT is recyclable reactor grade water. Operating experience at VCSNS for these WL components managed by this program reveals no history of degradation for the internal surfaces.

The WL components that function as pressure boundary for the CC System are the tubes for the RCDT heat exchanger and the tubes for the Waste Evaporator Concentrates Sample Cooler. The Waste Evaporator is seldom used at VCSNS. Operating experience at VCSNS for these WL components managed by this program reveals no history of degradation for the internal surfaces.

<u>RAI B.2.3-2:</u> 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.

VCSNS Response RAI B.2.3-2

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this aging effects identification research are contained within the body of technical work at VCSNS supporting the LRA. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS.

The stainless steel ND System components managed by this program function as pressure boundary for a containment penetration. This penetration is used for the transfer of water out of the Reactor Building Sump and the Incore Instrumentation Sump. These sumps contain leakage only from Systems located inside the Reactor Building (i.e. high purity systems). These components are leak tested under 10CFR50 Appendix J. Operating experience at VCSNS for the ND components managed by this program reveals no history of degradation for the internal surfaces.

The stainless steel WL System components managed by this program either function as pressure boundary for a containment penetration or function as pressure boundary with the Component Cooling (CC) System.

The WL components that function as pressure boundary for containment isolation are leak tested under 10CFR50 Appendix J. The contents of the Reactor Coolant Drain Tank (RCDT) are transferred through this containment penetration. Because of the purity of the systems that are the sources of the water to the RCDT, the water in the RCDT is recyclable reactor grade water. Operating experience at VCSNS for these WL components managed by this program reveals no history of degradation for the internal surfaces.

The WL components that function as pressure boundary for the CC System are the tubes for the RCDT Heat Exchanger and the tubes for the Waste Evaporator (WE) Concentrates Sample Cooler. The Waste Evaporator is seldom used at VCSNS. Operating experience at VCSNS for these WL components managed by this program reveals no history of degradation for the internal surfaces.

B.2.5 Reactor Building Cooling Unit Inspecton

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.

VCSNS Response RAI B.2.5-1

Visual inspections will be performed for the area below the coils. Should age-related degradation be detected there, and then volumetric examinations will be performed on the 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.

VCSNS Response RAI B.2.5-2

There is no action to trend the inspection results because it is a one-time inspection. The aim of one-time inspections is to determine if further actions are required. Element #7 of the program addresses additional inspections should degradation be detected. Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this research are contained within the body of technical work at VCSNS supporting the LR application. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS.

The Reactor Building Coolant Units (RBCUs) have the potential to concentrate contaminants from the Reactor Building atmosphere; therefore the environment is deemed to be an unmonitored water environment. A search of industry operating experience has revealed an instance of detecting boric acid crystallization in RBCU drain piping; therefore, VCSNS has conservatively deemed the environment to be unmonitored borated water. This drain piping is not pressurized nor is it likely to see much flow (less than 1 gpm) except under accident conditions. Because there is a potential for aging effects for stainless steel in this unmonitored borated water environment, VCSNS will conduct a one-time inspection prior to the end of the current operating term. According to GL 88-05, stainless steel and nickel-based alloys are not susceptible to boric acid corrosion; therefore boric acid corrosion is not considered to be an aging effect for RBCU drains. Aging effects for carbon steel components of the ventilation environment of the RBCUs are managed by Preventive Maintenance Activities - Ventilation Systems Inspections.

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.

VCSNS Response RAI B.2.5-3

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. Industry operating experience revealed an instance of detecting boric acid crystallization in RBCU drain piping; therefore, this one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS. No leaks or degradation of the stainless steel drain piping has been experienced in the RBCUs at VCSNS.

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.

VCSNS Response RAI B.2.6-1

The Service Air Systems Inspection concerns specific components that are not pertinent to quality of air supplied to safety-related equipment and, therefore, not pertinent to the concerns of NRC GL 88-14. The concern for these components is air as the internal environment only, not as the motive force for operation of the components. These components concern the pressure boundary function of specific containment penetrations, containment hatch testing, and emergency air supply to the personnel hatches. Ambient moist air (not dried by an air dryer) is assumed to be the internal environment for these components.

The Service Air Systems Inspection is a one-time inspection to verify that aging degradation has not occurred in some specific safety-related portions (containment integrity) of the Service Air (SA) System, the Instrument Air (IA) System, and the Building Services (BS) System that are not designed or required to be in a dry air environment.

The carbon steel SA and IA System components that are managed by this program only function as pressure boundaries for specific containment penetrations. These components are leak tested under 10CFR50 Appendix J. Operating experience at VCSNS for the SA and IA components managed by this program reveals no history of aging degradation for the internal surfaces.

The carbon steel BS components that are managed by this program are the hatch seal test blocks and the emergency air valves for the hatches. Operating experience at VCSNS for the BS components managed by this program reveals no history of aging degradation for the internal surfaces.

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.

VCSNS Response RAI B.2.6-2

The Service Air Systems Inspection concerns specific components that are not pertinent to quality of air supplied to safety-related equipment and, therefore, not pertinent to the concerns of NRC GL 88-14. The concern for these components is air as the internal

environment only, not as the motive force for operation of the components. These components concern the pressure boundary function of specific containment penetrations, containment hatch testing, and emergency air supply to the personnel hatches. Ambient moist air (not dried by an air dryer) is assumed to be the internal environment for these components.

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this research are contained within the body of technical work at VCSNS supporting the LR application. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found.

The Service Air Systems Inspection is a one-time inspection to verify that aging degradation has not occurred in some specific safety-related portions (containment integrity) of the Service Air (SA) System, the Instrument Air (IA) System, and the Building Services (BS) System that are not designed nor required to be in a dry air environment.

The carbon steel SA and IA System components that are managed by this program function as pressure boundaries for a containment penetration. These components are leak tested under 10CFR50 Appendix J. Operating experience at VCSNS for the SA and IA components managed by this program reveals no history of aging degradation for the internal surfaces.

The carbon steel BS components that are managed by this program are the hatch seal test blocks and the emergency air valves for the hatches. Operating experience at VCSNS for the BS components managed by this program reveals no history of aging degradation for the internal surfaces.

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).

VCSNS Response RAI B.2.8-1

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this aging effects identification research are contained within the body of technical work at VCSNS supporting the LRA. This one-time inspection was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS.

The portions of the Waste Gas (WG) System managed by the Waste Gas Systems Inspection are in scope because they are pressure boundaries for the Component Cooling Water (CC) System and the Chemical and Volume Control (CS) System.

The Recombiner Heat Exchangers are in scope for license renewal because the tube coils and tube manifolds form a pressure boundary with the CC System, which is the cooling medium for the heat exchangers. Operating experience at VCSNS for the heat exchanger components managed by this program reveals no history of aging degradation for the internal surfaces.

The WG valves and piping managed by this program allow the transfer of condensation formed in the Waste Gas Decay Tanks to the Volume Control Tank of the CS System. These components are in scope for license renewal because they form a pressure boundary with the CC System. Operating experience at VCSNS for these components managed by this program reveals no history of aging degradation for the internal surfaces.

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.

VCSNS Response RAI B.2.11-1

The Inspections for Mechanical Components does not manage loss of material due to MIC. The Maintenance Rule Structures Program, listed in Table 3.3-1, Item 5, manages loss of material due to MIC.

Plant 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 structural design of the plant is such that any groundwater intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

The VCSNS Final Safety Analysis Report [FSAR] identifies a groundwater elevation of 420' +/- 3'. Certain structures, such as the Service Water Pumphouse, are potentially exposed to a ground water level of 425'. As such, piping, process tubing, and ductwork component types were conservatively considered to be susceptible to external MIC if they either enter a building from outside or pass between buildings included in the sheltered environment below the 425' elevation. Additionally, the susceptibility to external MIC was limited locally to the area of the interface with the pertinent wall. For

building fire seal penetrations in the sheltered environment, the management of aging of the pertinent structural commodities precludes the accumulation of the necessary microbiological organisms, and thus MIC, on interfacing mechanical component types.

Therefore, loss of material due to MIC has been identified as an aging effect requiring system specific evaluation for carbon and low alloy steel in sheltered environments for piping, process tubing, and ductwork that pass between pertinent buildings through a non fire seal penetration or enters the building from outside (i.e. underground, embedded) below the 425' elevation.

Building penetrations are inspected as part of the Maintenance Rule Structures Program (Application Section B.1.18). The VCSNS Corrective Action Program would disposition any groundwater in-leakage and resulting degradation.

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.

VCSNS Response RAI B.2.11-2

Generally, the Inspections for Mechanical Components will inspect external surfaces already exposed and accessible to visual examination during normal operation. In addition, operating experience has revealed an instance of external pitting below the insulation on Chilled Water (VU) System piping; therefore, provisions for removal of insulation to permit visual inspections will be necessary for Systems where the internal fluid temperature is less than the external ambient temperature and the insulation is not tightly adhered to the components (for example, sprayed on insulation would not be considered to be susceptible to pitting under insulation). Any unacceptable degradation, whether found by these inspections or by planned maintenance activities, would be determined by engineering evaluation and dispositioned in the Corrective Action Program. Although the initial frequency for the inspections is five years, the Corrective Action Program could increase not only the frequency, but also the scope of the inspections.

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.

VCSNS Response RAI B.2.11-3

Site engineering personnel will perform visual inspections. Any degradation found during the visual inspections would be dispositioned through the VCSNS Corrective Action Program. Further inspections and qualifications required for these inspections would be determined through the Corrective Action Program. Generally, further inspections required by the Corrective Action Program would be performed by quality control personnel qualified in accordance with ASME Code and 10CFR50 Appendix B.

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?

VCSNS Response RAI B.2.11-4

These inspections will follow the same frequency as Maintenance Rule Inspections (five years) and the baseline inspection would occur within five years of obtaining the new license. Based upon the results of these inspections, or any new industry experience, the frequency may increase.

"Effective components" should 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.

VCSNS Response RAI B.2.11-5

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this research are contained within the body of technical work at VCSNS supporting the LRA. This new program was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS. VCSNS used a search of operating experience to ensure that the list of aging effects identified by industry references was complete. The particular operating experience concerning the Chilled Water System was included because it demonstrates the effectiveness of this history search. Using industry references, it was determined that because of the relatively unpolluted environment of the area, contaminants would not concentrate in sufficient quantities to cause pitting corrosion; however, the history search at VCSNS revealed that pitting has occurred under insulation in the Chilled Water System and therefore it is included as an aging effect to be managed.

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.

VCSNS Response RAI B.2.11-6

VCSNS does not use GALL Program XI.M29. The 10 elements for the Inspections for Mechanical Components are described in Section B.2.11 of the Application. The Chilled Water (VU) Expansion Tanks are elevated such that the bottoms are accessible; however, in other instances, conditions of inaccessible locations can be inferred from the external conditions of accessible locations that are closest to the subject component. Tanks are elevated, usually on elevated concrete pads, so that any accumulations on the floor around it do not affect the tank. It is expected that, should there be any external degradation of tank bottoms for the tanks on concrete pads, there would be telltale signs down the sides of the elevated pad which would be addressed by the Corrective Action Program. Any general corrosion on inaccessible tank bottoms would degrade no further than an initial oxide layer, which would provide protection from further general corrosion.

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:

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

VCSNS Response RAI B.2.11-7

For bolted closures (i.e. pressure-retaining) of components/component types subject to aging management review, the design of critical closure joint bolting involves enough redundancy to ensure joint integrity and no aging effects unique to bolting, over the components being joined/closed, require evaluation for license renewal as discussed further below. External aging degradation of carbon and low alloy steels components will be managed by the Inspections for Mechanical Components and, in locations where susceptible, the Boric Acid Corrosion Surveillances.

Although identified as an aging effect in various industry references, loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for non-Class 1 component bolted closures (i.e. pressure boundary closures) within the scope of license renewal.

Mechanical components within the scope of license renewal, both Class 1 and Non-Class 1, contain bolted closures that are necessary for the pressure boundary of the components being joined/closed. Examples of these bolted closures are valve bonnet to body, pump cover to casing, heat exchanger manway and channel head (end-bell), and piping flange sets. The bolted closure is comprised of two mating surfaces, a gasket, and a fastener set (studs or bolts, washers, and nuts). By themselves, the mating set, gasket, or fastener set have no component intended function. Together, the bolted closure forms an integral part of the pressure-retaining boundary of the component. Additionally, the bolted closure is exposed to the same environment(s) as the components in the plant areas where the closure is located (process fluid for internal mating surface and ambient conditions else). As such, the bolted closure (including fastener set) was considered to be a sub-component (piece-part) of the components/component types within the scope of license renewal and did not require evaluation separate from the component, except as clarified.

Loss of mechanical closure integrity can result in failure of the mechanical joint, is evidenced by leakage rather than joint failure, and can be attributed to one or more of the following effects:

- **Loss of bolt pre-load (embedment, cyclic load embedment, gasket creep, etc.),**
- **Loss of bolting material (from general and/or boric acid corrosion),**
- **Reduction of bolting material fracture toughness, and**
- **Cracking of high strength bolting material (SCC).**

For non-Class 1 bolted closures, loss of pre-load was considered to be the result of inadequate design or improper assembly (i.e. event driven) that is not related to aging and that would manifest itself during the current operating term and be corrected prior to the period of extended operation. Thus, the mechanisms associated with loss of bolting pre-load are not a license renewal concern for non-Class 1 components/ component types.

It is recognized that loss of bolting material could ultimately result in the loss of a component's pressure boundary integrity and thus, requires evaluation for license renewal. However, loss of material is an aging effect requiring license renewal evaluation for carbon and alloy steel components/component types subject to aging management review. As such, no evaluation separate from the subject components/component types of which bolted closures are a part is necessary and, for carbon and alloy steel components/component types, the aging management Programs credited for managing external general corrosion will also inherently address their fasteners.

Furthermore, stainless steel fasteners are immune to loss of material due to general corrosion and most bolting is normally in a dry environment and is coated with a lubricant, thus general corrosion of carbon and alloy steel bolting is not expected, nor has it been a major concern in the industry. As is the case with subject components/component types of similar material, the occurrence of general corrosion in carbon and low alloy steel fastener sets in the ambient environments is most likely in Systems with operating temperatures below ambient conditions that result in condensation and in the yard environment with repeated wetting and drying from exposure to the elements.

Loss of material due to boric acid wastage (aggressive chemical attack) is the most common aging affect that has been observed in the industry for ferritic fasteners. In the concentrations used in PWR Systems, boric acid is a relatively weak acid. However, under wetting and drying conditions, such as a result of leakage, boric acid may concentrate in a slurry forming a saturated solution. There appear to be no differences in the corrosion rates for the common carbon and low-alloy steel bolting materials. Stainless steel fasteners have been shown to be immune to loss of material due to boric acid wastage. The Boric Acid Corrosion Surveillances, credited for management of the external aging of carbon and low-alloy steel in locations susceptible to leaking borated water, will also address carbon and low-alloy steel fasteners in that location. Additionally, the Inspections for Mechanical Components will address any general

corrosion concerns for carbon or low alloy steel bolting of stainless steel components/component types.

Reduction of fracture toughness of bolting material, due to thermal/neutron effects is a license renewal concern for the fasteners of components only due to the associated elevated System operating temperatures and proximity to the reactor vessel beltline region. This is applicable to bolting of some Class 1 components and is addressed in the application. Reduction of fracture toughness for non-Class 1 bolting material is not a license renewal aging effect requiring management for the fasteners of components.

Stress corrosion cracking (SCC) of bolting materials, it is a condition in which a fastener that is statically loaded well below the material yield strength may suddenly fail. SCC bolted closure fastener failures have occurred in materials with apparently normal chemical and mechanical properties. Although there have been a few instances of cracking of bolting in the industry due to SCC, these have been attributed to high yield stress materials and contaminants, such as the use of lubricants containing MoS_2. VCSNS has not and does not use lubricants containing MoS_2. Most bolting is normally in a dry environment and is coated with a lubricant; in general, environmental conditions that could lead to SCC of bolting are not expected to occur in non-Class 1 components. For quenched and tempered low alloy steels used for closure bolting (e.g., SA193 Grade B7), material susceptibility to SCC is minimized by having a lower yield strength. EPRI Report NP-5769 (Volume I, pg. 11-5) indicates that SCC should not be a concern for closure bolting in nuclear power plant applications if the specified yield strength is below 150 ksi. The specification for the fabrication of nuclear piping, specifies alloy steel ASME SA 193, Class B7 bolts/studs and ASME 194 Grade 2H nuts, which have minimum yield strengths below 150 ksi (105 ksi). A minimum yield strength for bolting does not, in and of itself, preclude SCC since the actual yield strength of the bolt could be above the threshold value for SCC of low alloy steel bolting/fasteners to occur (150 ksi). However, sound maintenance bolt torquing practices can control bolting material stresses and the use of appropriate material (such as ASTM A193 Gr. B7) for bolting reduces the potential for SCC to occur. A review of industry failure databases and NRC generic communications, supports the fact that proper material selection, proper maintenance and torquing procedures, and removal of contaminants from lubricants have been effective in eliminating the potential for SCC of bolting materials. As documented in NRC Inspection Report No. 50-395/84-08, dated April 20, 1984, the recommended preventive measures and practices of IEB 82-02 have been incorporated into the maintenance procedures at VCSNS. Therefore, SCC of bolting materials is not an aging effect requiring evaluation for license renewal for non-Class 1 components/component types.

B.2.12 Heat Exchanger Inspection

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.

VCSNS Response RAI B.2.12-1

The Heat Exchanger Inspections (HEI) concerns certain aging effects for components in a treated water environment. The HEI is being developed to manage aging effects not already managed by other programs. For the material and environment of these components, the Chemistry program is credited to manage galvanic corrosion; therefore, the HEI is not credited to manage this aging mechanism.

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.

VCSNS Response RAI B.2.12-2

The Heat Exchanger Inspections (HEI) concerns particular aging effects for certain materials in a treated water environment. The purpose of the inspections is to manage aging effects not managed by any other program. For the components encompassed by the HEI, the Chemistry Program would not manage loss of material due to leaching and erosion-corrosion for certain materials, thus the HEI is necessary to manage those aging effects; however, because the Chemistry Program is credited with managing heat exchanger fouling, the HEI is a demonstration of the effectiveness of the Chemistry Program for that particular aging effect.

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).

VCSNS Response RAI B.2.12-3

Depending on the condition of the component as determined by engineering evaluation, subsequent inspections would be determined through the VCSNS Corrective Action Program. Should aging effects be detected that require subsequent inspections, the subsequent inspections would be at the locations previously inspected

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.

VCSNS Response RAI B.2.12-4

Any loss of material would be determined by engineering evaluation based on the design of the individual component and, where applicable, on the results of the hardness testing. Although the Chemistry Program controls heat exchanger fouling due to particulates, the HEI is an additional verification of the effectiveness of the Chemistry Program. Any heat exchanger fouling will be determined by engineering evaluation based on visual examination. Loss of material or heat exchanger fouling would be evaluated and documented in the VCSNS Corrective Action Program, with subsequent actions or inspections determined through the Corrective Action Program. Should aging effects be detected that require subsequent inspections, the subsequent inspections would be at the locations previously inspected so that rate of damage could be determined.

The Chemistry Program controls cracking. It is not an aging mechanism managed by the HEI.

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.

VCSNS Response RAI B.2.12-5

Operating experience, both site-specific and industry-wide, was researched to identify the possible aging effects for various combinations of material and environment. The results of this research are contained within the body of technical work at VCSNS supporting the LRA. This new program was developed because it was determined that the aging effects were possible, not because these aging effects were found at VCSNS. VCSNS used a search of operating experience to ensure that the list of aging effects identified by industry references was complete. At VCSNS there is no history of selective leaching, erosion-corrosion, or heat exchanger fouling occurring for the components managed by this program.

ATTACHMENT V
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Section 2.3
Accession No. ML030990546

2.3 SCOPING AND SCREENING RESULTS: MECHANICAL

Section 2.3.1 Reactor Vessel, Internals, and Reactor Coolant System

RAI 2.3.1-1: Item 5 of FSAR Section 15.4.3.2.2, which pertains to the recovery procedure after a steam generator tube rupture accident, states, "Decrease RCS pressure by use of normal pressurizer spray until the water level returns in the pressurizer and RCS pressure and the ruptured steam generator pressure are equal, or high pressurizer water level is attained, or minimum subcooling is attained."

a. Please provide the technical justification for not including the pressurizer spray head among the component types that are subject to aging management review (Table 2.3-6). Technical justification should address the requirements of Appendix R to 10 CFR 50, as well as 10 CFR 54.4(a).

b. Please indicate whether the steam generator divider plates, which separate the primary-side inlet and outlet plena in each steam generator, are considered in-scope for aging management. Please provide the technical justification for your response.

c. Please indicate whether the reactor coolant pump support structures are considered in-scope for aging management. Please provide the technical justification for your response.

VCSNS Response RAI 2.3.1-1

The recovery procedure in the FSAR is based on the Westinghouse Emergency Response Guidelines (ERG). These actions utilize available equipment whether it is credited in the accident analysis or not. VCSNS has credited the Pressurizer PORVs for depressurization of the RCS in accident conditions as well as for the Appendix R scenario. The PORVs are in scope for license renewal. VCSNS does not credit Pressurizer spray as the depressurization method in accidents or Appendix R conditions and the spray head is not in the scope of license renewal.

The Steam Generator primary side divider plate is in scope of license renewal. This item is included in the second line of ApplicationTable 2.3-7.

The Reactor Coolant Pump supports are in scope and are included in the "equipment component support" item of Application Table 2.4-2.

ATTACHMENT VI
Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station
Section 2.2
Accession No. ML030990546

2.2 PLANT LEVEL SCOPING RESULTS

RAI 2.2.2-2: During the system scoping inspection at the plant site (May 12 - 16, 2003) the team noted several inconsistencies between the system entries in the license renewal application (LRA) and the actual systems in-use at the plant. Many of these systems never existed at the plant. Provide a complete listing of systems in-use at the plant, and correct Tables 2.2-1, 2.2-2, and 2.2-3 of the LRA.

VCSNS Response RAI 2.2.2-2

The system lists in Application Table 2.2-1 and Table 2.2-3 were developed from several sources. These lists contain some system designations that were not utilized in the design and construction of the plant. The equipment database has been used for tracking of activities, tools, parts, etc. Some system designations are used only for tracking and do not contain permanent plant equipment.

"Inactive" systems contain no equipment and are not used for tracking in the plant equipment database. Twenty-three of the system designations are identified in the Application but are "inactive" in the plant equipment database.

- **Ammonia [AM]**
- **Auxiliary Steam Supply-Nuclear [AN]**
- **Chlorine [CL]**
- **Diesel Generator Fuel [DF]**
- **Emergency Power [EP]**
- **Gas Sampling [GE]**
- **Gland Sealing Water [GW]**
- **Hydrogen Vent [HM]**
- **Integrated Control [IN]**
- **Isolation Valve Seal Water [IV]**
- **Non-Nuclear Plant Vents [MV]**
- **Nitrogen Processing [NP]**
- **Oxygen [OX]**
- **Penetration Cooling (Liquid) [PC]**
- **Penetration Pressurization [PP]**
- **Condenser Priming [PR]**
- **Sump Drains [SD]**
- **Site Maintenance [SM]**
- **Spent Resin [SR]**
- **Turbine Drains [TD]**
- **Cycle Make-Up Water Treatment [WF]**
- **Chemical Cleaning [WM]**
- **Circulating Water Clarification[WN]**

The Reactor Protection Control System [RP] designation has no permanent plant hardware presently assigned. However, the reactor protection functions are performed by components in other I&C Systems.

System designations Emergency Equipment [EQ] and In-house Electric Maintenance [IE] contain no permanent plant and are used for tracking purposes only.

System designation Warehouse Electric Equipment [EW] contains no permanent plant equipment and is used for tracking purposes only. A cable, credited for an Appendix R repair, is tracked utilizing this designation. This cable is stored in the warehouse and is considered within the scope of license renewal.

ATTACHMENT TO RESPONSE; RAI 3.6-1

B.2.9 NON-EQ INSULATED CABLES AND CONNECTIONS INSPECTION PROGRAM

Program Description: 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 NUREG-1801 prior to the period of extended operation. The program will also be consistent with ISG-5, *Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal.*

This program will be applied to Non-EQ Insulated Cables and Connections including instrumentation cable with the exception that XI.E2, *Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits* may be applied to high-range-radiation and neutron flux monitoring instrumentation cables, which have high voltage, low-level signal applications that are sensitive to reduction in insulation resistance, instead of this program.

This is a new inspection program that will assess the condition of non-EQ insulated cables, connections, and in-scope, passive, non-EQ fuse holders (including the metallic fuse clip portions of these fuse holders when found to be susceptible to aging effects) in order to provide reasonable assurance that the aging effects of concern will not result in loss of the intended functions during the period of extended operation.

(1) **Scope:** The specific non-EQ insulated cables and connections that will be included in the aging management program for VCSNS include accessible non-EQ insulated cables and connections, including splices, terminal blocks, and fuse holders that are found susceptible to potential degradation in adverse thermal and radiological areas of the plant. Selection of the areas to be inspected shall include considerations for circuits with potentially significant ohmic heating. While certain areas of the Intermediate and Auxiliary Buildings will be the focus, there will be flexibility to inspect cables and connections in a variety of Environmental Zones, as determined by the responsible electrical engineering group at VCSNS. The technical basis for the location selected will be documented and will consider both thermally and radiologically adverse environments as well as considerations such as ohmic heating, vibration, mechanical stress for fuse clips, etc.

Passive, non-EQ fuse holders located outside of active devices that have been identified as being susceptible to aging effects in the Aging Management Review will be considered within the scope of this program. Equipment and components located inside an active device or panel are not within the scope of this program. An active device is characterized as an assembly or enclosure made up of parts or subcomponents built to perform a specific function. Examples of active devices include switchgear, MCCs, power supplies, inverters, battery chargers, control panels, and equipment racks.

(2) **Preventive Actions** - No actions are taken as a part of the Non-EQ Insulated Cables and Connections Inspection Program to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters to be inspected as a part of the Non-EQ Insulated Cables and Connections Inspection Program include visual evidence of cable jacket or connection surface abnormalities such as embrittlement, cracking, swelling, discoloration, surface contamination, presence of standing water or moisture, or any other visible evidence of age-related degradation, which may lead to loss of the intended function.

ATTACHMENT TO RESPONSE; RAI 3.6-1

The metallic fuse clip portion of any in-scope, passive fuse holders found to be susceptible to aging effects will be additionally monitored due to aging stressors such as vibration, thermal cycling, electrical transients, mechanical stress, fatigue, corrosion, chemical contamination, or oxidation of the connecting surfaces. In this aging management program, thermography, contact resistance testing, or other appropriate tests will be used to identify any existence of aging degradation for these fuse clips.

These parameters will be monitored or inspected on a representative sample basis. The technical basis for the sample selected will be documented.

(4) **Detection of Aging Effects** - The Non-EQ Insulated Cables and Connections Inspection Program, conducted in the thermally and radiologically severe areas of the plant containing in-scope cables and connections will serve to detect degradation of cable and connections, which could ultimately lead to electrical failure. During each inspection, visual evidence of jacket or surface abnormalities such as embrittlement, cracking, swelling, discoloration, melting, degradation of organics, radiation-induced oxidation, and moisture intrusion will be evaluated. The Inspection Program will be initially performed prior to the period of extended operation and then at 10-year intervals thereafter.

Identified fuse holders within the scope of license renewal that are located outside of an active device or panel and found to be potentially susceptible to age related degradation will likewise be inspected/tested at least once every 10 years commencing prior to the period of extended operation.

(5) **Monitoring and Trending** - Trending actions are not included as a part of this program because the ability to trend inspection results is limited. Documentation of these inspections will be available in subsequent inspections for comparison, review, and evaluation. Although not a requirement, test and inspection results that are trendable may provide additional information on the rate of degradation.

(6) **Acceptance Criteria** - The Non-EQ Insulated Cables and Connections Inspection Program consists of visual inspections for degradation of cable and connections jackets and surfaces due to aging. The accessible cables and connections are to be free from unacceptable visual indications of surface anomalies, which suggest that conductor insulation or connection degradation exists. Acceptance criteria are based on the cable and connection insulation service life. The service life evaluation of the insulation material includes consideration of the material's mechanical and electrical properties and their performance in ambient environments under plant operational conditions of temperature, radiation, and humidity as well as ohmic heating effects. The results of the Non-EQ Insulated Cables and Connections Inspection Program will serve as input into the service life evaluation of the cable and connections. An unacceptable inspection indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

The acceptance criteria for each test performed on the fuse clip portion of in-scope, passive, non-EQ fuse holders susceptible to age related degradation is defined by the specific type of test performed.

ATTACHMENT TO RESPONSE: RAI 3.6-1

(7) **Corrective Actions** - All unacceptable indication of cable jacket, connection surface, or metallic fuse clip degradation are subject to an engineering evaluation. The evaluation is to consider the age and operating environment of the component, significance of the test or inspection results, the operability of the component, the reportability of the event, the extent of the concern, the corrective actions required, and the likelihood of recurrence. Corrective actions are also necessary at any discovery of the presence of standing water or moisture on relevant cables. Corrective actions may include, but are not limited to, testing, shielding or otherwise changing the environment, relocation of the item, or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections exposed to similar aging stressors. The VCSNS Corrective Action Program is utilized as applicable to provide specific corrective and confirmatory actions. The VCSNS Corrective Action Program follows the requirements of 10 CFR 50, Appendix B.

(8) **Confirmation Process** - The VCSNS Corrective Action Program is utilized as applicable to provide specific corrective and confirmatory actions. The VCSNS Corrective Action Program follows the requirements of 10 CFR 50, Appendix B.

(9) **Administrative Controls** - The Non-EQ Insulated Cables and Connections Inspection Program will be implemented in accordance with controlled station and work processes, which are governed by 10 CFR 50, Appendix B.

B.2.9.1 Operating Experience

Industry experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections may exist next to or above (within three feet of) steam generators, pressurizer or hot process pipes such as feedwater lines. These adverse localized environments have been found to cause degradation of the insulating materials on electrical cables and connections that is visually observable, such as color changes or surface cracking. These visual observations can be used as indicators of potential degradation.

NUREG-1760 (Aging Assessment of Safety-Related Fuses Used in Low- and Medium-Voltage Applications in Nuclear Power Plants) identified fuse holders as experiencing a number of age-related failures. Aging stressors such as vibration, thermal cycling, electrical transients, mechanical stress, fatigue, corrosion, chemical contamination, or oxidation of the connecting surfaces can result in fuse holder failure. 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. Information Notices 91-78, 87-42, and 86-87 provide examples of the potential problems that can arise from age-related fuse holder failures.

The Non-EQ Insulated Cables and Connections Inspection Program is a new inspection activity for which there is no operating experience. Effective visual inspection techniques will be selected for use in performing the inspections on cables and connections. Likewise, effective inspection/testing techniques will be selected for the metallic fuse clip portion of those in-scope, passive, non-EQ fuse holders found to be susceptible to the aging stressors listed above. Lessons learned during the performance of the inspections, experience gained and shared by other utilities, and other inspection

ATTACHMENT TO RESPONSE; RAI 3.6-1

techniques developed in the industry will be considered as proposed enhancements to the program so that the effects of aging will continue to be adequately managed.

B.2.9.2 Conclusion

The Non-EQ Insulated Cables and Connections Inspection Program will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

ATTACHMENT TO RESPONSE; RAI 3.6-2

XI.E2 ELECTRICAL CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

Program Description

In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the plant design environment for those areas. Conductor insulation materials used in electrical cables may degrade more rapidly than expected in these adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

Exposure of electrical cables to adverse 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 instrument circuits.

The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are used in instrumentation circuits with sensitive, low-level signals exposed to adverse localized environments caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.

In this aging management program, calibration results or findings of surveillance testing programs are used to identify the potential existence of aging degradation. For example, when an instrumentation circuit is found to be out of calibration, additional evaluation of the circuit is performed.

This aging management program applies to non-EQ, high-range-radiation and neutron flux monitoring instrumentation cables used in high voltage, low-level signal applications that are sensitive to reduction in insulation resistance. An overlap exists between this program and program XI.E1 in that XI.E1 can be applied to all instrumentation cables, but is not applied to high-range-radiation and neutron flux monitoring instrumentation cables if the XI.E2 program is applied.

As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the electrical cables covered by this aging management program are either

ATTACHMENT TO RESPONSE; RAI 3.6-2

not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.

(1) **Scope** - This program applies to electrical cables used in circuits with sensitive, low-level signals. At VCSNS, this includes radiation monitoring and nuclear instrumentation cables.

(2) **Preventive Actions** - No actions are taken as part of this program to prevent or mitigate aging degradation.

(3) **Parameters Monitored/Inspected** - The parameters monitored are determined from the specific calibrations or surveillances performed and are based on the specific instrumentation circuit under surveillance or being calibrated, as documented in the calibration or surveillance procedures.

(4) **Detection of Aging Effects** - Review of calibration results or findings of surveillance programs can provide indication of aging effects by monitoring key parameters and providing data based on acceptance criteria related to instrumentation circuit performance. Reviews of results obtained during normal calibrations or surveillances provide reasonable assurance that severe aging degradation will be detected prior to loss of the cable intended function. The first reviews for license renewal are to be completed before the period of extended operation and every ten years thereafter. All calibrations or surveillances that fail to meet acceptance criteria will be reviewed at the time.

(5) **Monitoring and Trending** - Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement, test results that are trendable provide additional information on the rate of degradation.

(6) **Acceptance Criteria** - Calibration results or findings of surveillances are to be within the acceptance criteria, as set out in the calibration or surveillance procedures.

(7) **Corrective Actions** - Corrective actions such as recalibration and circuit trouble-shooting are implemented when calibration or surveillance results or findings of surveillances do not meet the acceptance criteria. Corrective actions are also necessary at any discovery of the presence of standing water or moisture on relevant instrumentation cables. The requirements of 10 CFR Part 50, Appendix B, will be implemented to address corrective actions.

(8) **Confirmation Process** - The requirements of 10 CFR Part 50, Appendix B, will be implemented to address the confirmation process.

(9) **Administrative Controls** - The requirements of 10 CFR Part 50, Appendix B, will be implemented to address administrative controls.

Operating Experience:

Changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential electrical cable degradation.

ATTACHMENT TO RESPONSE; RAI 3.6-2

Conclusion:

The Aging Management Program for Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

ATTACHMENT TO RESPONSE; RAI 3.6-2

Alternate to XI.E2 Program
LICENSE RENEWAL AGING MANAGEMENT PROGRAM FOR NON-EQ ELECTRICAL CABLES USED IN INSTRUMENTATION CIRCUITS

Program Description

Exposure of electrical cables to adverse localized environments caused by heat, radiation or moisture can result in reduced insulation resistance (IR). An adverse, localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified service condition for the circuit. 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 circuit.

The purpose of this aging management program is to provide reasonable assurance that the intended function of non-EQ, high voltage, low signal circuits exposed to an adverse localized environment caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation.

In this aging management program, an appropriate test, such as an insulation resistance test, will be used to identify the potential existence of a reduction in cable IR.

This program is an acceptable alternative to aging management program XI.E2.

(1) **Scope** - This program applies to electrical cables used in circuits with sensitive, low-level signals. At VCSNS, this includes radiation monitoring and nuclear instrumentation cables.

(2) **Preventive Actions** - No actions are taken as part of this program to prevent or mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters monitored include dielectric strength caused by thermal/ thermoxidative degradation of organics, moisture intrusion, or radiation-induced oxidation (radiolysis) of organics.

(4) **Detection of Aging Effects** - Cables will be tested at least once every 10 years. Testing may include insulation resistance tests, time domain reflectometry (TDR) tests, I/V testing, or other testing judged to be effective in determining cable insulation condition. Following issuance of a renewed operating license, the initial test will be completed before the end of the initial 40-year license term.

(5) **Monitoring and Trending** - Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement, test results that are trendable provide additional information on the rate of degradation.

(6) **Acceptance Criteria** - The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

ATTACHMENT TO RESPONSE; RAI 3.6-2

(7) **Corrective Actions** - An evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the cables can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. Corrective actions are also necessary at any discovery of the presence of standing water or moisture on relevant instrumentation cables. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other high voltage, low signal circuits exposed to similar adverse localized environments. The requirements of 10 CFR Part 50, Appendix B, will be implemented to address corrective actions.

(8) **Confirmation Process** - The requirements of 10 CFR Part 50, Appendix B, will be implemented to address the confirmation process.

(9) **Administrative Controls** - The requirements of 10 CFR Part 50, Appendix B, will be implemented to address administrative controls.

Operating Experience:

Operating experience has shown that anomalies found during cable testing can be caused by degradation of the instrumentation circuit cable and are a possible indication of potential cable degradation.

Conclusion

The Aging Management Program for Non-EQ Electrical Cables Used in Instrumentation Circuits will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

ATTACHMENT TO RESPONSE; RAI 3.6-3

XI.E3 AGING MANAGEMENT PROGRAM FOR INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10CFR50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Program Description

The "Aging Management Program For Inaccessible Medium-Voltage Cables Not Subject To 10 CRF 50.49 Environmental Qualification Requirements" is consistent with XI.E3, *Inaccessible Medium-Voltage Cables Not Subject To 10CFR50.49 Environmental Qualification Requirements,* of NUREG-1801.

This is a new aging management program that will assess the condition of inaccessible medium-voltage cables not subject to 10 CFR 50.49 Environmental Qualification requirements to provide assurance that the aging effects of concern will not result in loss of the intended functions during the period of extended operation.

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 purpose of the Aging Management Program described herein is to provide reasonable assurance that the intended functions of inaccessible medium-voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.

In this Aging Management Program periodic actions are taken to prevent or minimize the possibility that cables may be exposed to moisture, such as inspecting for water collection in cable manholes and conduit, and draining water, as needed. In-scope, medium-voltage cables exposed to moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The initial test performed will be determined prior to the period of extended operation for detecting deterioration of the insulation system due to wetted conditions at the time the test is performed.

As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." The electrical cables covered by this aging management program are not

ATTACHMENT TO RESPONSE; RAI 3.6-3

exposed to harsh accident conditions, hence, are not subject to the environmental qualification requirements of 10 CFR 50.49.

(1) **Scope**: The specific non-EQ medium voltage insulated cables subject to moisture and significant voltage that will be included in the aging management program for VCSNS includes the two circuits serving the Service Water Pump Motors. These circuits are inaccessible as they are routed in underground duct, except at electrical manholes or where they exit the duct bank, and are medium-voltage cables within the scope of license renewal that are potentially exposed to moisture simultaneously with significant voltage. Moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time. The moisture and voltage exposures described as significant in these definitions, which are based on operating experience and engineering judgment, are not significant for medium-voltage cables that are designed for these conditions (e.g., continuous wetting and continuous energization is not significant for submarine cables).

(2) **Preventive Actions**: Periodic actions are taken to prevent or minimize the possibility that cables may be exposed to moisture, such as inspecting for water collection in cable manholes and conduit, and draining water, as needed. Inaccessible medium-voltage cables which are in the License Renewal scope and subject to potential moisture with significant voltage are to be tested in accordance with this program since operating experience conservatively indicates that the potential exists for exposure to sufficiently prolonged moisture and voltage, which may induce or contribute to this aging mechanism.

(3) **Parameters Monitored/Inspected**: In-scope, medium-voltage cables exposed to moisture and significant voltage will be tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test. This will be a test that will not damage the cable itself.

(4) **Detection of Aging Effects**: In-scope, medium-voltage cables exposed to moisture and significant voltage are tested at least once every 10 years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. A 10-year test frequency will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed before the period of extended operation.

(5) **Monitoring and Trending**: Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement, test results that are trendable provide additional information on the rate of degradation.

(6) **Acceptance Criteria**: The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

(7) **Corrective Actions**: An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the electrical cables

ATTACHMENT TO RESPONSE; RAI 3.6-3

can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other inaccessible, in-scope, medium-voltage cables. The requirements of 10 CFR Part 50, Appendix B, will be applied to address corrective actions within the VCSNS Corrective Action Program.

(8) **Confirmation Process**: The requirements of 10 CFR Part 50, Appendix B, will be applied to address the confirmation process within the VCSNS Corrective Action Program.

(9) **Administrative Controls**: The requirements of 10 CFR Part 50, Appendix B, will be applied to address administrative controls within the VCSNS controlled station procedures and work processes.

Operating Experience

The Aging Management Program For Inaccessible Medium-Voltage Cables Not Subject To 10CRF50.49 Environmental Qualification Requirements is a new aging management program for which there is no operating experience. The relevant in-scope cables included in this program at VCSNS are 7.2kv cables with EPR insulation and a Hypalon jacket. Industry operating experience has shown that XLPE or high molecular weight polyethylene (HMWPE) insulation materials are most susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage. Treeing is much less prevalent in 4kV cables than those operated at 13 or 33kV. Also, minimizing exposure to moisture minimizes the potential for the development of water treeing. As additional operating experience is obtained both at VCSNS and in the industry, lessons learned will be considered as proposed enhancements to the program so that the effects of aging will continue to be adequately managed.

Conclusion

The Aging Management Program For Inaccessible Medium-Voltage Cables Not Subject To 10 CRF 50.49 Environmental Qualification Requirements will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053
Engineer C. BARBIER
Date 1/7/03

Project Title 2002 Yearly Review of Cycle Count from WESTEMS Tab – Page 1 of 7

PURPOSE

This TWR is to document the yearly review of the count of plant operational cycles and transients for the year 2002 as required by ES-401, revision 2 per PMTS #0209583. This annual review is required to demonstrate compliance of plant operating transients/cycles with respect to their limits specified in ASME Code required Design Specifications, FSAR (Section 5.2 and Table 5.2.2) and Technical Specifications Design Features (Section 5.7 and Table 5.7-1).

SCOPE

The monitoring of plant transients is a trending activity. It is not an exact science. As explained in the following sections of this TWR, transients are monitored via cycle count and cumulative useage factors (CUF). These variables provide a technically sound means for the trending of transients and their affect on the fatigue damage of components. Any exceedance of a specific transient limit or even CUF as identified from the output of WESTEMS does not signify that fatigue damage of the component is eminent such that the component must be removed from service. Engineering judgement may be used in evaluating any cycles or CUF that have exceeded their limits or are trending toward exceeding their limits. This judgement is required to determine if the fatigue analysis of the component affected needs to be revisited to remove all margins of conservatisms. The conclusions of such an analysis would dictate the disposition of the component. The trending activity documented by this TWR is to review and trend the accumulated cycle count and CUF determined by WESTEMS, document all engineering judgements applied to the resultant trends and to conclude if there are any components requiring further detailed analysis of fatigue damage.

Although this is only a trending activity, there is sufficient conservatism and accuracy in the methodology to justify the results of the trending can be used to demonstrate compliance with the design basis documents and the Technical Specifications in regards to transient cycle count.

REFERENCES

1. WCAP-14437, "WESTEMS Functional Specification for V. C. Summer Nuclear Station"
2. WCAP-13113, "VCS Baseline Transient and Fatigue History Evaluation Final Report"
3. Calculation DC05900-001, Revision 0
4. ES401, Revision 2
5. Westinghouse Letter CGE-01-067, dated 10/1/01
6. PMTS #0209583

ATTACHMENTS

1. "WESTEMS Cycle Count Report" Dated 1/7/2003, 21 pages.
2. "WESTEMS Fatigue Report", Dated 1/7/03, 2 pages

**ENGINEERS
TECHNICAL WORK RECORD**

Serial __CB15053__.
Engineer C. BARBIER
Date 1/7/03 .

Project Title 2002 Yearly Review of Cycle Count from WESTEMS .Tab – Page 2 of 7 .

COMPUTER PROGRAM USED

WESTEMS Integrated Diagnostics and monitoring System validated as documented in Reference (3).

BACKGROUND

This review of accumulated to date temperature and pressure cycles utilizes the Computer program WESTEMS. This program has been in development since 1995. The production version was installed on site the later part of 2001 and validation was completed early in 2002.

Every minute, data from over 200 instruments are downloaded to WESTEMS from the plant IPCS and stored. Once a day this data is run through the program to identify transients. For each instrument, if certain threshold values are reached or exceeded, the program identifies that a transient has occurred. Based on the instrument and the magnitude of exceedance over the threshold value, the program identifies the components affected by the transient. Depending upon the component affected, the computer performs one of two things:

- For thirty-five components, WESTEMS assigns a transient name to the event and increases the accumulated cycle count of the transient for each component affected.
- For five components (seven locations), WESTEMS calculates the stress caused by the transient and determines the increase in Cumulative Useage Factor (CUF) for that component.

These components bound all components in the Reactor Coolant System and also include a select few other components which have had a history of fatigue concerns based on industry experience.

PLANT OPERATIONS DURING THE YEAR

The plant operated at or near100% except for a six week duration required for RF13 in the spring. Temperature and pressure cycles do not often occur during the time the plant is at 100% power when all systems are in a steady state condition. It is during plant heatup and cooldown when significant numbers of cycles occur. Therefore, it is expected that only a small number of additional cycles have occurred from that of the 2001 accumulated total, and that most of these additional cycles occurred during the cooldown and heatup required for RF13.

REVIEW

The Attachments (1) and (2) are printouts of "ACCESS" reports of the cycle count and CUF data stored in WESTEMS. For those thirty-five components monitored by cycle count, Attachment (1) provides a list of accumulated cycles (Current_Cycles) and compares them to the design allowable cycles (Allow_Cycles). The design allowable cycles (to plant End of forty year License) are obtained from Reference (1) and are stored in the WESTEMS computer program. This comparison is done on

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053
Engineer C. BARBIER
Date 1/7/03

Project Title 2002 Yearly Review of Cycle Count from WESTEMS Tab — Page 3 of 7

a per component basis. For those five components (seven locations) monitored by CUF, Attachment (2) Provides a list of accumulated CUF (Current_Useage). This value is trended against the maximum allowable CUF of 1.0.

for all bounding components within the RCS. The following is a summary of a review of these attachments as required by ES-401:

ES-401, section 6.2.2; "The accumulated cycles and CUF for each component are reviewed to identify those components that have cycles and CUF which exceeded the allowable limits."

A review of Attachment (1) of the accumulated transient cycles for each component listed concludes that they are all below the design allowable cycles, i.e. less than 100% used

A review of Attachment (2) of the accumulated CUF for each component listed concludes that they are all below the design allowable of 1.0.

ES-401, section 6.2.3; "Another consideration is the rate at which the cycles and CUF have accumulated. If the rates exceed a prorated limit, it indicates that the component's fatigue life may not last until EOL or Plant Life Extension (PLEX)."

A review was performed of the rate of accumulated transient cycles for each component listed on Attachment (1) and (2).

To determine the acceptable rate to EOL, the total allowable cycles are prorated based on the number of years the plant has been in service from 1/83:

$$\frac{\text{(Number of years in service) X (transient/CUF Allowable)}}{\text{(Number of years to plant EOL = 40.0)}}$$

$$\frac{\text{(20.0) X (transient/CUF Allowable)}}{40}$$

.50 X transient/CUF Allowable

In other words, if the percent used of a specific transient as noted on Attachment (1) is less that 50, the accumulated cycle count is trending in an acceptable manner in regards to EOL. Also, if the Current Usage as noted on Attachment (2) is less than .50, the CUF is trending in an acceptable manner.

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053
Engineer C. BARBIER
Date 1/7/03

Project Title 2002 Yearly Review of Cycle Count from WESTEMS Tab -- Page 4 of 7

To determine the acceptable rate to PLEX, the total allowable cycles are prorated based on the number of years the plant has been in service from 1/83:

$$\frac{\text{(Number of years in service) X (transient/CUF Allowable)}}{\text{(Number of years to PLEX = 60.0)}}$$

$$\frac{\text{(20.0) X (transient/CUF Allowable)}}{60}$$

.333 X transient/CUFAllowable

In other words, if the percent used of a specific transient as noted on Attachment (1) is less that 33.3, the accumulated cycle count is trending in an acceptable manner in regards to PLEX. Also, if the Current Usage as noted on Attachment (2) is less than .333, the CUF is trending in an acceptable manner.

A review of Attachments (1) concludes that the following transients and components are not trending in an acceptable manner per the criteria notes above.

For those components monitored by Cycle Count:

Case	TRANSIENT	COMPONENT	Current Cycles	Allow Cycles	Percent Used
1	Feedwater Cycling	Loop 1 HL	1380	2000	69
		Loop1 CL	1380	2000	69
		Loop 1 RCP Casing	1380	2000	69
		Loop 2 HL	1380	2000	69
		Loop2 CL	1380	2000	69
		Loop 2 RCP Casing	1380	2000	69
		Loop 3 HL	1380	2000	69
		Loop3 CL	1380	2000	69
		Loop 3 RCP Casing	1380	2000	69
		Safety Injection Loop A CL	1380	2000	69
		Safety Injection Loop B CL	1380	2000	69
		Safety Injection Loop C CL	1380	2000	69

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053 .
Engineer C. BARBIER
Date 1/7/03 .

Project Title 2002 Yearly Review of Cycle Count from WESTEMS .Tab -- Page 5 of 7 .

2	Inadvertent RCS Depressurization (Case II-Inadvertent Auxiliary Spray)",	Pressurizer Spray Nozzle	4	10	40

For those components monitored by CUF:

Case	Component	Current CUF	Allowable CUF
3	Normal Charging Nozzle	.46	1.0
4	Alternate Charging Nozzle	.47	1.0
5	Pressurizer Surge Line Pipe to Reactor Coolant Pipe Nozzle	.38	1.0

Technical evaluation of case 1 above:

The WESTEMS automated cycle count system was placed in operation in about 1995. At that time, it was necessary to estimate the number of transient cycles that the plant had experienced since startup. In order to do this, plant records were reviewed to ascertain how the plant had been operated. From this information, transients were estimated. These estimates were documented in Reference (2). The estimated transient count from this report was input into WESTEMS as a baseline transient cycle count. For the transients "Feedwater Cycling", Reference (2) estimated 1380 cycles had occurred from plant startup in 1982 to 1991. As can be seen from the table above, since the automated monitoring system was placed in operation in 1991, no new additional transients have occurred. Therefore, the rate of occurrence of these transients has been nil since 1991. Projecting this rate since 1991 out till end of plant life concludes that this transient will not exceed the design basis value of 2000. Therefore rate of cycle counts of case 1 are acceptable and no further review for these cases are required.

No further actions are required for this case at this time.

Technical evaluation of case 2:

For the transients "Inadvertent RCS Depressurization (Case II-Inadvertent Auxiliary Spray)", Reference (2) estimated 3 cycles had occurred from plant startup in 1982 to 1991. As can be seen from the table above, since the automated monitoring system was placed in operation in 1991, one new additional transients has occurred. Therefore, the rate of this one transient is almost within an acceptable trend.

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053
Engineer C. BARBIER
Date 1/7/03

Project Title 2002 Yearly Review of Cycle Count from WESTEMS Tab – Page 6 of 7

For the pressuriuzer spray nozzle, 17 transients are included in it's design basis. When any one of these transients occurs, it causes fatigue damage to the component. As the transients occur, fatigue damage is accumulated. However, the component is designed to withstand fatigue damage up to the point that the component has experienced all the 17 transients up to their limits. This implies, that if only one, or even several, of the 17 transients have exceeded their limits that sufficient margins to accumulated fatigue damage remain. In order for the accumulated fatigue of the component to be of a concern, the accumulated cycles for all the 17 transients would have to be near or exceed their limits. Since all the other 16 transients for the pressurizer spray nozzle are trending well below the EOL or PLEX rates, it is concluded that at EOL or PLEX accumulated fatigue damage from all 17 transients would not be of a concern.

No further actions are required for this case at this time.

Technical evaluation of case 3 through 5:

As noted previously, for the components of Normal Charging Nozzle, Alternate Charging Nozzle and Pressurizer Surge Line Pipe to Reactor Coolant Pipe Nozzle, WESTEMS calculates a cumulative useage factor (CUF). CUF represents the total accumulation of fatigue damage from all transients. All calculated CUFs that are less than 1.0 are acceptable. For the normal charging nozzle, the maximum calculated CUF is .46. This is more than .333 rate threshold for PLEX but less than the .50 threshold for EOL. For the alternate charging nozzle, the maximum calculated CUF is .47. This is more than .333 rate threshold for PLEX and less than the .50 threshold for EOL. For the Pressurizer Surge Line Pipe to Reactor Coolant Pipe Nozzle, the maximum calculated CUF is .37. This is more than .32 rate threshold for PLEX but less than the .48 threshold for EOL. In summary, this indicates that the rate of fatigue accumulation, based on total fatigue damage CUF, is acceptable to EOL for all three components, but not acceptable for PLEX.

The fatigue accumulation trend clearly shows that all three of these nozzles are probably not acceptable for PLEX and may need to be replaced prior to plant life extension. This trend will be watched closely over the next several years and does not warrant any immediate actions. In regards to the PSL to RCL pipe nozzle, several GOPs have just recently been revised to minimize insurges and outsuges to the pressurizer through the surge line during plant heatups and cooldowns. This past year which included a heatup and cooldown during an outage, the CUF increased only .01 from .37 to .38. It is hoped that in the future, this will flatten the trend in fatigue accumulation of this component.

No further actions are required for these cases at this time.

ENGINEERS
TECHNICAL WORK RECORD

Serial CB15053
Engineer C. BARBIER
Date 1/7/03

Project Title 2002 Yearly Review of Cycle Count from WESTEMS .Tab – Page 7 of 7 .

CONCLUSIONS

Transient cycle count and CUF accumulation for all components are below the allowable limits and are therefore satisfactory.

The rate of fatigue accumulation for all components are trending toward plant EOL in an acceptable manner. This indicates that the fatigue life of all components should remain acceptable through plant EOL.

The rate of fatigue accumulation for all components are trending through plant PLEX in an acceptable manner except for three components; the normal charging nozzle, alternate charging nozzle and the Pressurizer Surge Line Pipe to Reactor Coolant Pipe Nozzle. The fatigue accumulation trend clearly shows that all three of these nozzles are probably not acceptable for PLEX and may need to be replaced prior to plant life extension. This trend will be watched closely over the next several years and does not warrant any immediate actions. In regards to the PSL to RCL pipe nozzle, several GOPs have just recently been revised to minimize insurges and outsuges to the pressurizer through the surge line during plant heatups and cooldowns. This past year which included a heatup and cooldown during an outage, the CUF increased only .01 from .37 to .38. It is hoped that in the future, this will flatten the trend in fatigue accumulation of this component.

No further actions are required at this time.

V. C. Summer Nuclear Station

WESTEMS Thermal Event Cycle Counting Report

WESTEMS Thermal Event Cycle Counting Tuesday, January 07, 2003

ComponentID 101100 Reactor Vessel Outlet Nozzle Loop 1

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	255	18300	1.393443
600	Plant Unloading	201	18300	1.098361
800	Step Load Increase	47	2000	2.35
900	Step Load Decrease	30	2000	1.5
1000	Large Step Load Decrease	7	200	3.5
1110	Steady State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	7	200	3.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	52	230	22.6087
2320	Reactor Trip from Full Power (B - Cooldown, No SI)	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 101200 Reactor Vessel Inlet Nozzle Loop 1

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	1	-1	-100
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	2	200	1
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0

2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertant RCS Depressurization	1	10	10

ComponentID 101500 Reactor Vessel Structural Shell

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	200	1.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI)	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 102100 Reactor Vessel Outlet Nozzle Loop 2

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	41	200	20.5
200	Plant Cooldown	40	200	20
500	Plant Loading	245	18300	1.338798
600	Plant Unloading	189	18300	1.032787
800	Step Load Increase	46	2000	2.3
900	Step Load Decrease	30	2000	1.5
1000	Large Step Load Decrease	8	200	4
1110	Steady State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	4	80	5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	5	200	2.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	52	230	22.6087

2320	Reactor Trip from Full Power (B - Cooldown, No SI)	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 102200 Reactor Vessel Inlet Nozzle Loop 2

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	200	1.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 103100 Reactor Vessel Outlet Nozzle Loop 3

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	252	18300	1.377049
600	Plant Unloading	191	18300	1.043716
800	Step Load Increase	48	2000	2.4
900	Step Load Decrease	30	2000	1.5
1000	Large Step Load Decrease	7	200	3.5
1110	Steady State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	5	80	6.25
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	5	200	2.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	53	230	23.04348

2320	Reactor Trip from Full Power (B - Cooldown, No SI)	24	160	15
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 103200 Reactor Vessel Inlet Nozzle Loop 3

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	200	1.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10

ComponentID 111000 Loop 1 hot leg

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	9	-1	-900
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
300	Plant Loading Between 0 and 15% of Full Power	106	500	21.2
400	Plant Unloading Between 0 and 15% of Full Power	33	500	6.6
500	Plant Loading at 5% of Full Power/Minute	270	13200	2.045455
600	Plant Unloading at 5% of Full Power/Minute	198	13200	1.5
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	10	200	5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	18	70	25.71428

1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	7	80	8.75
2100	Loss of Power	3	40	7.5
2200	Partial Loss of Flow	2	80	2.5
2310	Reactor Trip from Full Power (Case A - No Cooldown	52	230	22.6087
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	4	20	20
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	5	80	6.25
2700	Inadvertent Safety Injection	5	60	8.333333

ComponentID 112000 **Loop 1 cold leg**

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	0	80	0
1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	1	80	1.25
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667

ComponentID 114100 **Loop 1 Steam Generator Primary Side**

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	13	-1	-1300
100	Plant Heatup	14	200	7

200	Plant Cooldown	12	200	6
300	Plant Loading Between 0 and 15% of Full Power	0	500	0
400	Plant Unloading Between 0 and 15% of Full Power	0	500	0
500	Plant Loading at 5% of Full Power/Minute	57	13200	0.4318182
600	Plant Unloading at 5% of Full Power/Minute	0	13200	0
1000	Large Step Load Decrease	0	200	0
1110	Stead-State Fluctuations (Random)	0	3000000	0
1111	Steady-State Fluctuations (Initial)	0	150000	0
1200	Boron Concentration Equalization	49	26400	0.1856061
1300	Feedwater Cycling	0	2000	0
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	1	70	1.428571
1600	Turbine Roll Test	0	20	0
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	0	80	0
2100	Loss of Power	0	40	0
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	0	230	0
2320	Reactor Trip from Full Power (Case B - Cooldown, N	0	160	0
2330	Reactor Trip from Full Power (Case C - Cooldown wi	0	10	0
2400	Inadvertent RCS Depressurization	0	20	0
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	0	80	0
2700	Inadvertent Safety Injection	0	60	0
2800	Excessive Feedwater Flow	0	30	0

ComponentID 114500 Loop 1 Steam Generator Feedwater Nozzle

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	487	-1	-48700
100	Plant Heatup	11	200	5.5
200	Plant Cooldown	21	200	10.5
300	Plant Loading 0% to 15%	6	500	1.2
400	Plant Unloading 0% to 15%	3	500	0.6
500	Plant Loading 5% of Full Power/Minute (>15%)	59	13200	0.4469697
600	Plant Unloading 5% of Full Power/Minute (>15%)	58	13200	0.4393939
800	Step Load Increase	0	2000	0
900	Step Load Decrease	0	2000	0
1000	Large Step Load Decrease	24	200	12
1600	Turbine Roll Test	1	20	5
1900	FW Heaters Out of Service	0	40	0
2310	Reactor Trip from Full Power (Case A)	2	230	0.8695652
2320	Reactor Trip from Full Power (Case B)	25	160	15.625

2330	Reactor Trip from Full Power (Case C)	0	10	0
2600	Control Rod Drop	0	80	0
2800	Excessive Feedwater Flow	3	30	10

ComponentID 115010 Loop 1 RCP Casing

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling at Hot Shutdown	1380	2000	69
1400	Loop Out of Service - Shutdown	0	80	0
1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	2	80	2.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown w/	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667
2800	Excessive Feedwater Flow	0	30	0

ComponentID 121000 Loop 2 hot leg

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	1	-1	-100
100	Plant Heatup	44	200	22
200	Plant Cooldown	41	200	20.5
300	Plant Loading Between 0 and 15% of Full Power	106	500	21.2
400	Plant Unloading Between 0 and 15% of Full Power	32	500	6.4
500	Plant Loading at 5% of Full Power/Minute	263	13200	1.992424
600	Plant Unloading at 5% of Full Power/Minute	194	13200	1.469697
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	10	200	5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333

1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	2	80	2.5
1410	Normal Loop Startup	15	70	21.42857
1500	Refueling	8	80	10
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	6	80	7.5
2100	Loss of Power	3	40	7.5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	2	20	10
2500	Inadvertent Startup of an Inactive Loop	1	10	10
2600	Control Rod Drop	5	80	6.25
2700	Inadvertent Safety Injection	5	60	8.333333

ComponentID 122000 **Loop 2 cold leg**

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	0	80	0
1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	1	80	1.25
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667

ComponentID 124100 **Loop 2 Steam Generator Primary Side**

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	4	-1	-400
100	Plant Heatup	16	200	8
200	Plant Cooldown	12	200	6
300	Plant Loading Between 0 and 15% of Full Power	0	500	0
400	Plant Unloading Between 0 and 15% of Full Power	0	500	0
500	Plant Loading at 5% of Full Power/Minute	49	13200	0.3712121
600	Plant Unloading at 5% of Full Power/Minute	0	13200	0
1000	Large Step Load Decrease	0	200	0
1110	Stead-State Fluctuations (Random)	0	3000000	0
1111	Steady-State Fluctuations (Initial)	0	150000	0
1200	Boron Concentration Equalization	44	26400	0.1666667
1300	Feedwater Cycling	0	2000	0
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	1	70	1.428571
1600	Turbine Roll Test	0	20	0
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	0	80	0
2100	Loss of Power	0	40	0
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	0	230	0
2320	Reactor Trip from Full Power (Case B - Cooldown, N	0	160	0
2330	Reactor Trip from Full Power (Case C - Cooldown wi	0	10	0
2400	Inadvertent RCS Depressurization	0	20	0
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	0	80	0
2700	Inadvertent Safety Injection	0	60	0
2800	Excessive Feedwater Flow	0	30	0

ComponentID 124500 **Loop 2 Steam Generator Feedwater Nozzle**

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	257	-1	-25700
100	Plant Heatup	14	200	7
200	Plant Cooldown	15	200	7.5
300	Plant Loading 0% to 15%	6	500	1.2
400	Plant Unloading 0% to 15%	1	500	0.2
500	Plant Loading 5% of Full Power/Minute (>15%)	72	13200	0.5454545
600	Plant Unloading 5% of Full Power/Minute (>15%)	33	13200	0.25
800	Step Load Increase	0	2000	0
900	Step Load Decrease	0	2000	0

1000	Large Step Load Decrease	30	200	15
1600	Turbine Roll Test	1	20	5
1900	FW Heaters Out of Service	0	40	0
2310	Reactor Trip from Full Power (Case A)	1	230	0.4347826
2320	Reactor Trip from Full Power (Case B)	29	160	18.125
2330	Reactor Trip from Full Power (Case C)	0	10	0
2600	Control Rod Drop	0	80	0
2800	Excessive Feedwater Flow	1	30	3.333333

ComponentID 125010 Loop 2 RCP Casing

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling at Hot Shutdown	1380	2000	69
1400	Loop Out of Service - Shutdown	0	80	0
1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	80	3.75
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667
2800	Excessive Feedwater Flow	0	30	0

ComponentID 131000 Loop 3 hot leg

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
300	Plant Loading Between 0 and 15% of Full Power	106	500	21.2
400	Plant Unloading Between 0 and 15% of Full Power	32	500	6.4
500	Plant Loading at 5% of Full Power/Minute	272	13200	2.060606

600	Plant Unloading at 5% of Full Power/Minute	196	13200	1.484848
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	11	200	5.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	2	80	2.5
1410	Normal Loop Startup	17	70	24.28572
1500	Refueling	7	80	8.75
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	7	80	8.75
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	24	160	15
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	3	20	15
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	5	80	6.25
2700	Inadvertent Safety Injection	5	60	8.333333

ComponentID 132000 Loop 3 cold leg

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	0	80	0
1500	Refueling	6	80	7.5
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	2	80	2.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No	51	230	22.17391

	Cooldown			
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667

ComponentID 134100 Loop 3 Steam Generator Primary Side

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	7	-1	-700
100	Plant Heatup	16	200	8
200	Plant Cooldown	12	200	6
300	Plant Loading Between 0 and 15% of Full Power	0	500	0
400	Plant Unloading Between 0 and 15% of Full Power	0	500	0
500	Plant Loading at 5% of Full Power/Minute	50	13200	0.3787879
600	Plant Unloading at 5% of Full Power/Minute	0	13200	0
1000	Large Step Load Decrease	0	200	0
1110	Stead-State Fluctuations (Random)	0	3000000	0
1111	Steady-State Fluctuations (Initial)	0	150000	0
1200	Boron Concentration Equalization	43	26400	0.1628788
1300	Feedwater Cycling	0	2000	0
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	1	70	1.428571
1600	Turbine Roll Test	0	20	0
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	0	80	0
2100	Loss of Power	0	40	0
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	0	230	0
2320	Reactor Trip from Full Power (Case B - Cooldown, N	1	160	0.625
2330	Reactor Trip from Full Power (Case C - Cooldown wi	0	10	0
2400	Inadvertent RCS Depressurization	0	20	0
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	0	80	0
2700	Inadvertent Safety Injection	0	60	0
2800	Excessive Feedwater Flow	0	30	0

ComponentID 134500 Loop 3 Steam Generator Feedwater Nozzle

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	277	-1	-27700
100	Plant Heatup	14	200	7
200	Plant Cooldown	21	200	10.5
300	Plant Loading 0% to 15%	6	500	1.2
400	Plant Unloading 0% to 15%	4	500	0.8

500	Plant Loading 5% of Full Power/Minute (>15%)	74	13200	0.5606061
600	Plant Unloading 5% of Full Power/Minute (>15%)	30	13200	0.2272727
800	Step Load Increase	0	2000	0
900	Step Load Decrease	0	2000	0
1000	Large Step Load Decrease	47	200	23.5
1600	Turbine Roll Test	1	20	5
1900	FW Heaters Out of Service	0	40	0
2310	Reactor Trip from Full Power (Case A)	1	230	0.4347826
2320	Reactor Trip from Full Power (Case B)	31	160	19.375
2330	Reactor Trip from Full Power (Case C)	0	10	0
2600	Control Rod Drop	0	80	0
2800	Excessive Feedwater Flow	1	30	3.333333

ComponentID 135010 Loop 3 RCP Casing

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	0	-1	0
100	Plant Heatup	40	200	20
200	Plant Cooldown	40	200	20
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
1000	Large Step Load Decrease	7	200	3.5
1110	Stead-State Fluctuations (Random)	670000	3000000	22.33333
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1300	Feedwater Cycling at Hot Shutdown	1380	2000	69
1400	Loop Out of Service - Shutdown	0	80	0
1500	Refueling	6	80	7.5
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	80	3.75
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	1	20	5
2700	Inadvertent Safety Injection	4	60	6.666667
2800	Excessive Feedwater Flow	0	30	0

ComponentID 151300 Pressurizer Upper Shell

Group	Tran_Name	Cycles	Allow_Cycles	Percent_Used
0	Out of Bounds	1	-1	-100
100	Plant Heatup	47	200	23.5
200	Plant Cooldown	35	200	17.5

210	Plant Cooldown from 400 to 15 psia	31	200	15.5
900	Step Load Decrease of 10% of Full Power	19	2000	0.95
1010	Large Step Load Decrease with Steam Dump	7	200	3.5
1111	Steady-State Fluctuations (Initial)	34000	150000	22.66667
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load (w/o Trip)	1	80	1.25
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (Case A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (Case B - Cooldown, N	23	160	14.375
2330	Reactor Trip from Full Power (Case C - Cooldown wi	3	10	30
2400	Inadvertent RCS Depressurization	3	30	10
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2600	Control Rod Drop	5	80	6.25
2800	Excessive Feedwater Flow	0	30	0

ComponentID 151400 Pressurizer Spray Nozzle

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	1	-1	-100
100	Plant Heatup	36	1200	3
200	Plant Cooldown to 400 psia	47	1200	3.916667
210	Plant Cooldown from 400 to 15 psia	38	1200	3.166667
500	Plant Loading at 5% of Full Power/Minute	208	13200	1.575758
600	Plant Unloading at 5% of Full Power/Minute	147	13200	1.113636
800	Step Load Increase of 10% of Full Power	10	2000	0.5
900	Step Load Decrease of 10% of Full Power	19	2000	0.95
1010	Large Step Load Decrease with Steam Dump	7	200	3.5
1200	Boron Concentration Equalization	1018	26400	3.856061
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	0	70	0
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load (w/o Trip)	1	80	1.25
2200	Partial Loss of Flow	0	80	0
2400	Inadvertent RCS Depress. (Case II - Inadv. Aux. Sp	4	10	40
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2700	Inadvertent Safety Injection	4	60	6.666667

ComponentID 153710 Auxiliary Spray Section 1-1

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	52	200	26

110	Heatup Case H1	0	200	0
120	Heatup Case H2	0	200	0
130	Heatup Case H3	0	200	0
140	Heatup Case H4	0	200	0
150	Heatup Case H5	0	200	0
160	Heatup Case H6	0	200	0
200	Plant Cooldown	31	200	15.5
210	Cooldown Case C1	0	200	0
220	Cooldown Case C2	0	200	0
230	Cooldown Case C3	0	200	0
240	Cooldown Case C4	6	200	3
250	Cooldown Case C5	3	200	1.5
260	Cooldown Case C6	0	200	0
265	Cooldown Case C6	0	200	0
270	Cooldown Case C7	3	200	1.5
275	Cooldown Case C7	0	200	0
500	Plant Loading	3	13200	2.272727E-02
600	Plant Unloading	0	13200	0
800	Step Load Increase	3	2000	0.15
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	0	200	0
1200	Boron Concentration Equalization	207	26400	0.7840909
1300	Feedwater Cycling	9	2000	0.45
1400	Normal Loop Shutdown	0	80	0
1410	Normal Loop Startup	0	70	0
2000	Loss of Load	0	80	0
2330	Reactor Trip Case C – CD with SI	0	10	0
2450	Inadvertent Aux. Spray	0	10	0
2455	Inadvertent Aux. Spray	3	10	30
2500	Inadvertent Startup of an Inactive Loop	0	10	0
2700	Inadvertent Safety Injection	0	60	0
3100	Small Steam Line Break	0	5	0
3200	Complete Loss of Flow	0	5	0
9999	Normal & Upset Spray Recovery	189	60715	0.3112904

ComponentID 212500 **Letdown Line Section 1-1**

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	16	-1	-1600
100	Plant Heatup	58	200	29
200	Plant Cooldown	50	200	25
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787

800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	20	2000	1
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	4	80	5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	7	200	3.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10
6300	Letdown Flow Shutoff and Delayed Return to Service	16	100	16

ComponentID 212600 Excess Letdown Line Section 1-1

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	9	-1	-900
100	Plant Heatup	38	200	19
200	Plant Cooldown	55	200	27.5
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	200	1.5
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip from Full Power (A - No Cooldown	51	230	22.17391
2320	Reactor Trip from Full Power (B - Cooldown, No SI	23	160	14.375
2330	Reactor Trip from Full Power (C - Cooldown with SI	3	10	30
2400	Inadvertent RCS Depressurization	1	10	10
6300	Letdown Flow Shutoff and Delayed Return to Service	24	100	24

ComponentID 312200 Safety Injection Loop A CL

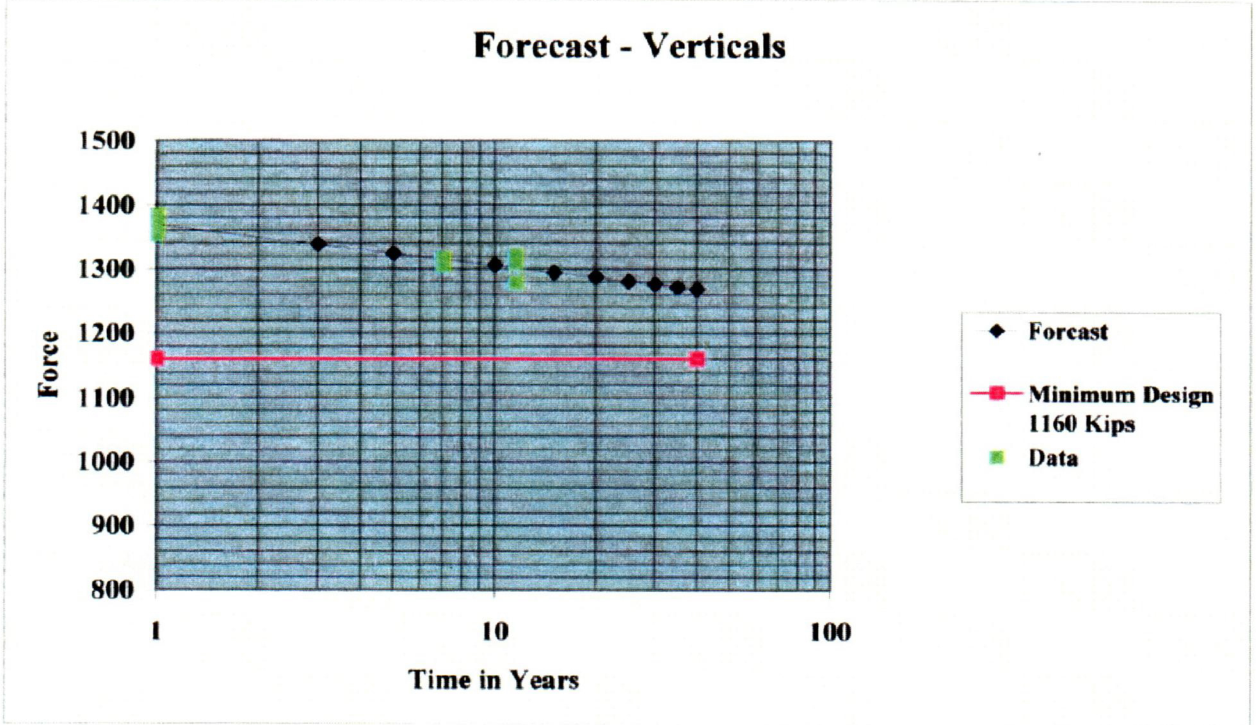
Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	0	-1	0
100	Plant Heatup	42	200	21
200	Plant Cooldown	41	200	20.5

500	Plant Loading	209	13200	1.583333
600	Plant Unloading	148	13200	1.121212
1000	Large Step Load Decrease	7	200	3.5
1300	Feedwater Cycling	1380	2000	69
1400	Normal Loop Shutdown	0	80	0
1500	Refueling	7	80	8.75
1510	Refueling - CL Group	8	80	10
1600	Turbine Roll Test	2	20	10
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	3	80	3.75
2100	Loss of Power	2	40	5
2200	Partial Loss of Flow	0	80	0
2310	Reactor Trip Case A - No CD, No SI	51	230	22.17391
2320	Reactor Trip Case B - CD, No SI	23	160	14.375
2330	Reactor Trip with CD and SI (Case C)	3	10	30
2350	Reactor Trip Case C - CD with SI - CL Group	3	10	30
2400	Inadvertent RCS Depressurization	3	20	15
2450	Inadvertent RCS Depressurization - CL Group	1	20	5
2700	Inadvertent SI Actuation	12	40	30
2710	Inadvertent SI with 70F RWST	4	20	20
2720	SI Contingency	0	8	0
2750	Inadvertent Safety Injection - A - CL Group	4	60	6.666667
3000	Small LOCA	0	5	0
3100	Small Steam Line Break	0	5	0
4000	RHR Operation	57	200	28.5
4100	Post LOCA	0	1	0
4300	Large Steam Line Break	0	1	0

ComponentID 312500 **12 Inch Accumulator Loop A CL**

Group	*Tran_Name*	*Cycles*	*Allow_Cycles*	*Percent_Used*
0	Out of Bounds	17	-1	-1700
100	Plant Heatup	46	200	23
200	Plant Cooldown	52	200	26
500	Plant Loading	208	18300	1.136612
600	Plant Unloading	147	18300	0.8032787
800	Step Load Increase	10	2000	0.5
900	Step Load Decrease	19	2000	0.95
1000	Large Step Load Decrease	7	200	3.5
1110	Steady-State Fluctuations	670000	1.683E+07	3.980986
1600	Turbine Roll Test	2	80	2.5
1900	FW Heaters Out of Service	0	40	0
2000	Loss of Load	14	200	7

20th YEAR PHYSICAL SURVEILLANCE OF CONTAINMENT BUILDING POST- TENSIONING AT THE V. C. SUMMER NUCLEAR PLANT



Years	Forcast (kips)
1	1368
3	1338
5	1324
10	1306
15	1294
20	1287
25	1280
30	1276
35	1271
40	1268

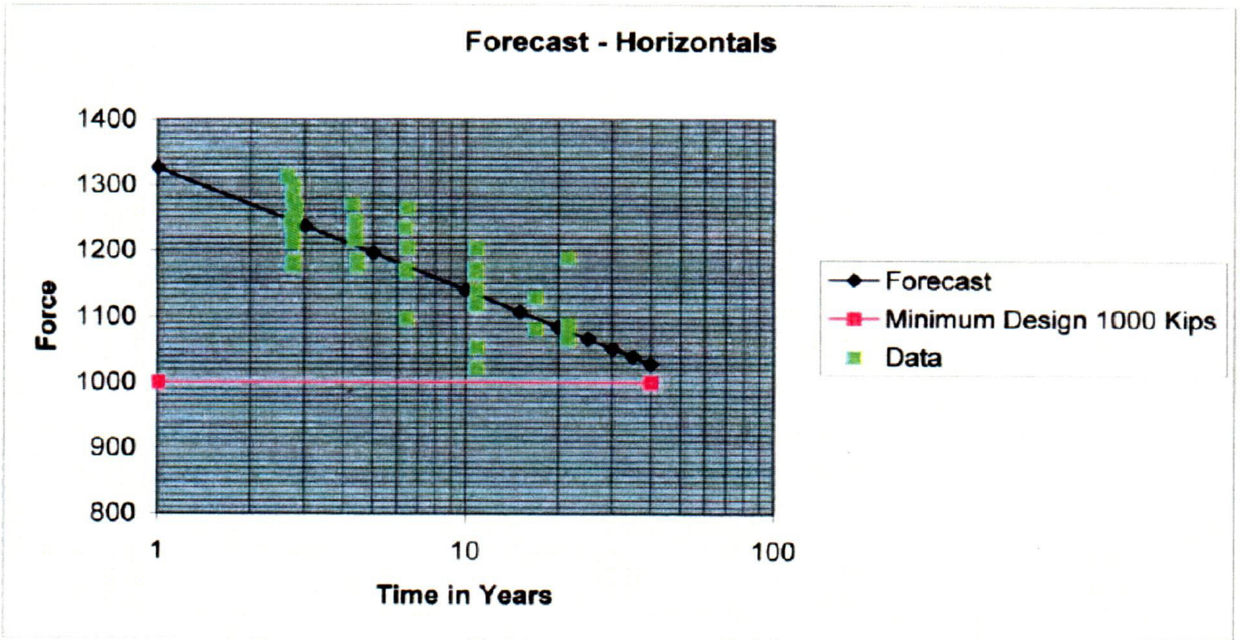
Due to retensioning conducted in 1990 the vertical timeline starts from that point. Therefore the thirty year value is effectively forty year plant life, and as expected minimum required force levels are easily achieved.

C01

Precision Surveillance Corporation

20th YEAR PHYSICAL SURVEILLANCE OF CONTAINMENT BUILDING POST- TENSIONING AT THE V. C. SUMMER NUCLEAR PLANT

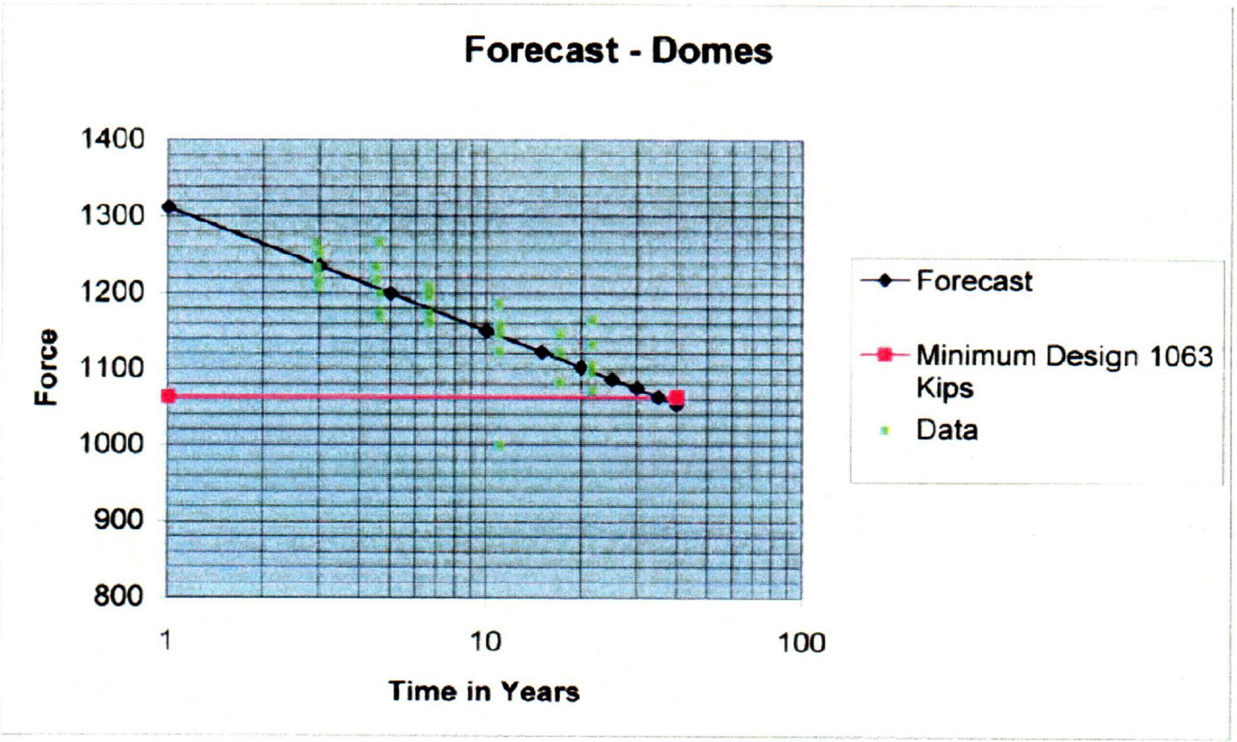
SCE&G A SCANA COMPANY



Years	Forecast (kips)
1	1326
3	1237
5	1196
10	1140
15	1107
20	1084
25	1066
30	1051
35	1039
40	1028

C02

**20th YEAR PHYSICAL SURVEILLANCE
OF CONTAINMENT BUILDING
POST- TENSIONING AT THE
V. C. SUMMER NUCLEAR PLANT**



Years	Forecast (kips)
1	1312
3	1235
5	1199
10	1150
15	1122
20	1102
25	1086
30	1074
35	1063
40	1053

C03

ATTACHMENT XII

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
ENGINEERING SERVICES TECHNICAL REPORT

TR00170-003

STRUCTURES AGING MANAGEMENT REVIEW FOR LICENSE RENEWAL

REVISION 0

Original Signed by Sing Chu — 06/27/2002
ORIGINATOR — DATE

Original Signed by Bob Whorton — 06/27/2002
REVIEWER — DATE

Original Signed by Ron Clary — 07/03/2002
APPROVAL AUTHORITY — DATE

CHANGE LETTER	EFFECTIVE DATE	CHANGE LETTER	EFFECTIVE DATE

TABLE OF CONTENTS

1.0 INTRODUCTION

The nuclear power plant license renewal rule, 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" [Reference 1], describes the license renewal process and provides the requirements for the contents of a license renewal application. In applying for a renewed operating license, this rule requires that an Integrated Plant Assessment (IPA) be completed for the Virgil C. Summer Nuclear Station (VCSNS). The IPA, as described in 10 CFR 54.21, is an assessment that demonstrates the effects of aging on specific structures and components will be adequately managed during the period of extended operation.

The VCSNS IPA has been divided along engineering discipline lines traditional to South Carolina Electric & Gas Company (Civil/Structural, Electrical and Mechanical) [Footnote 1]. Dividing the work in this way facilitated the technical reviews consistent with the current VCSNS technical information set. Procedural guidance for generating this technical report is provided by ES-705, "Civil/Structural Scoping, Screening, and Aging Management Review for Licensing Renewal" [Reference 2]. The technical information supporting the IPA for all disciplines is documented in the following VCSNS license renewal technical reports as shown in Table 1.0-1:

Table 1.0-1
Technical Reports Supporting the IPA

Technical Report Number	Technical Report Title
TR00140-001	Time-Limited Aging Analyses and Exemptions for License Renewal
TR00140-002	Fatigue Evaluations for License Renewal
TR00150-001	Electrical System Scoping for License Renewal
TR00150-002	Electrical System Screening for License Renewal
TR00150-003	Electrical System Aging Management Review for License Renewal
TR00160-001	Mechanical Systems Scoping for License Renewal
TR00160-002	Mechanical Component Screening for License Renewal (Borated Water Systems)
TR00160-003	Mechanical Component Screening for License Renewal (Treated Water Systems)
TR00160-004	Mechanical Component Screening for License

1 **Engineering Discipline Boundaries:** The discipline boundaries are as follows: if a component or parts of it carry electrical current it's Electrical; if it supports, protects or restrains the movement of a component it's Civil/Structural; everything else is Mechanical. Exceptions to this are noted where applicable.

Technical Report Number	Technical Report Title
	Renewal (Raw Water Systems)
TR00160-005	Mechanical Component Screening for License Renewal (Oil/Fuel Oil Systems)
TR00160-006	Mechanical Component Screening for License Renewal (Air/Gas Systems)
TR00160-007	Mechanical Component Screening for License Renewal (Ventilation Systems)
TR00160-010	General Aging Effects Identification for License Renewal (Mechanical)
TR00160-011	Mechanical Component Aging Management Review for License Renewal (Class 1 Components).
TR00160-012	Mechanical Component Aging Management Review for License Renewal (Borated Water Systems).
TR00160-013	Mechanical Component Aging Management Review for License Renewal (Treated Water Systems)
TR00160-014	Mechanical Component Aging Management Review for License Renewal (Raw Water Systems)
TR00160-015	Mechanical Component Aging Management Review for License Renewal (Oil/Fuel Oil Systems)
TR00160-016	Mechanical Component Aging Management Review for License Renewal (Air/Gas Systems)
TR00160-017	Mechanical Component Aging Management Review for License Renewal (Ventilation Systems)
TR00160-020	Program/Activity Evaluations for License Renewal (Mechanical)
TR00170-001	Structures Scoping for License Renewal
TR00170-002	Structures Screening for License Renewal
TR00170-003	Structures Aging Management Review for License Renewal

In addition to the IPA, the license renewal rule requires an evaluation of Time-Limited Aging Analyses (TLAA). As defined in 10 CFR 54.3, these analyses are typically the boundary conditions and assumptions within the current licensing basis specifically linked to 40 years of operation. TLAA evaluations will be performed in accordance with ES-706, "Identification and Evaluation of Time-Limited Aging Analyses & Exemptions for License Renewal" [Reference 3], and documented in the TR00140-001 technical report.

The IPA and TLAA technical reports comprise the engineering input that forms the technical bases upon which the VCSNS License Renewal Application (LRA) is built. To ensure compliance with the regulatory requirements in 10 CFR 54, the Nuclear Regulatory Commission staff will review the VCSNS LRA.

In order to meet the regulatory requirements during the processing of the VCSNS application, those plant changes that have materially affected the information reported in the application and associated clarification documents such as the SCE&G responses to the NRC requests for additional Information must be identified. Changes to the plant information could potentially invalidate conclusions drawn by the NRC in the safety evaluation report associated with the VCSNS license renewal application. This technical report will be revised (as necessary) on an annual basis to identify such exemption(s)/change(s) (if any), following application submittal until receipt of the renewed license.

Civil Structures Aging Management Review

As part of the overall IPA, this technical report provides the civil structures aging management review methodology and results used to determine structures and component types to be included in the IPA. This methodology was developed in accordance with the guidance provided in NEI 95-10 [Reference 4], the Generic Aging Lesson Learned (GALL) Report [Reference 5], and the Standard Review Plan for License Renewal [Reference 6]. The scoping and screening of structures within license renewal scope have been identified in VCSNS Technical Reports TR00170-001, Structures Scoping for License Renewal [Reference 7] and TR00170-002, Structures Screening for License Renewal [Reference 8], respectively.

This structures AMR technical report, in conjunction with the structures scoping and screening technical reports, provide the technical basis for the information submitted to the Nuclear Regulatory Commission in the Application to Renew the Operating License of the Virgil C. Summer Nuclear Station.

2.0 PURPOSE/SCOPE

The nuclear power plant license renewal rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," describes the license renewal process and provides requirements for the contents of license renewal application. 10 CFR 54.21(a)(3) of the license renewal rule requires that for each structure/component subject to an Aging Management Review (AMR), the licensee shall demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

The purpose of this report is to identify and evaluate the aging effects of concern for the Virgil C. Summer Nuclear Station structures and structural components and to identify effective aging management programs as required by the license renewal rule.

The aging management review methodology is described in Section 5.0. The aging management review includes identification of the aging effects requiring management and demonstration that the effects of aging are managed.

The identification of the aging effects requiring management for the structures and structural components for the period of extended operation is addressed in Section 6.0. The aging effects identification evaluates material/environment combinations to identify the aging effects that require programmatic management. The aging effects conclusions are then used to identify the aging effects for commodity groups based upon their material and operating environments.

Demonstration that the effects of aging are managed is addressed in Section 7.0.

A summary of applicable structures related TLAAs is addressed in Section 8.0 of this report. TLAAs that meet the criteria of 10 CFR 54.3 are dispositioned as part of the TLAA process in VCSNS Technical Report TR00140-001 [Reference 144].

Conclusions for the aging management review are provided in Section 9.0.

The aging management review process is to demonstrate that the component intended function(s) will be maintained consistent with the CLB during the period of extended operation. Therefore, the intended function(s) of components subject to an aging management review are identified to ensure the maintenance of that function in the future. To properly assess the effect of aging on components, an evaluation is performed to determine the effect aging has on the ability of a component to perform its intended function(s). For the aging effects that can degrade the ability of a component to perform its intended function in the period of extended operation, the aging effect and/or relevant conditions which cause or enable loss of component intended function are described. Existing plant programs, or portions of programs, that manage or address the aging effect and/or relevant conditions that lead to aging effects are then identified. If an existing program, or a portion of an existing program, cannot be credited with management of the applicable aging effects, program enhancements or new programs are developed. For the existing programs, or portions of programs, credited with managing applicable aging effects, a demonstration of the effectiveness of the program is provided.

This Technical Report is governed by 10CFR50, Appendix B. A 10CFR50.59 review is not required for this report.

3.0 DEFINITIONS AND ACRONYMS

The definitions and abbreviations/acronyms used in the License Renewal Project for Virgil C. Summer Nuclear Station (VCSNS) are as follows:

Aging Effect – A net change in component characteristics (due to specific processes that gradually change characteristics of a component with time or use) that could cause the component to lose its intended function prior to the end of the extended period of operation.

> Aging effects are those effects that (1) are possible for a given combination of materials and service conditions, and (2) if not managed, could result in the loss of the component intended function during the period of extended operation.
>
> An aging effect is considered to be a "potential aging effect" until the aging management review is completed. Once the aging management review determines that an aging effect is credible for a component or commodity group the aging effect is considered to be an "applicable aging effect".

Aging Management Program (AMP) – A program that adequately manages the effects of aging on structures and components (SC) within the scope of license renewal so that the intended function(s) will be maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation.

Aging Management Review (AMR) – A review of each structure and component (SC) within the scope of license renewal that demonstrates that the effects of aging on these SCs will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Commodity Group – A group of like structures or components that perform the same intended function and consist of similar materials of construction under similar environments and/or service conditions. The components determined to be passive and long lived in a given commodity group will typically be addressed under the same AMR.

Components Subject to an AMR – Structures and components subject to an aging management review shall encompass those structures and components that perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties. Additionally, these SCs are not subject to replacement based on a qualified life or specified time period.

Current Licensing Basis (CLB) – The set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR Parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and Technical Specifications. It also includes the plant-specific design basis information defined in 10 CFR 50.2 as documented in the most recent Final Safety Analysis Report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

<u>Design Basis Events (DBE)</u> – Those conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure:

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shutdown the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in §50.34(a)(1) or §100.11.

This definition of design basis events is cited by §54.4(a)(1) as articulated in §50.49(b)(1)(ii).

<u>External Events</u> – Those events arising by intent or accident from activities that ordinarily have no association with the operation or maintenance of a nuclear plant and that are initiated under human control from outside the protected area of a nuclear plant.

There is no historical record of external events of an accidental nature involving U.S. nuclear power plants in general or VCSNS in particular. U.S. nuclear power plants, including VCSNS, utilize a variety of security measures to minimize the potential for external events of an intentional nature. Deliberate misconduct by an employee is not an external event.

If an external event were to occur, the resulting challenge to plant systems would most likely be enveloped by one or more of the existing design basis accident evaluations; however, the extent of damage to plant systems could exceed the single failure criteria.

External events are not formally defined with respect to the design, operation, maintenance or licensing of nuclear power plants by either 10 CFR 54 or 10 CFR 50. Further, §50.13 excludes (external events in the form of) attacks or destructive acts committed by either an enemy of the United States or by U.S. military forces from nuclear power plant design considerations.

Integrated Plant Assessment (IPA) – A licensee assessment that demonstrates that a nuclear power plant systems, structures and components (SSCs) requiring aging management review in accordance with 10 CFR 54.21(a) for license renewal have been identified. In addition, the IPA demonstrates that the effects of aging on the functionality of such structures and components will be managed such that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Intended Functions – The intended functions that the systems, structures, and components must be shown to fulfill in 10 CFR 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs 10 CFR 54.4(a)(1) through (3).

License Renewal Application (LRA) – A plant-specific application submitted by a utility to the NRC to allow renewal of the plant's operating license beyond the initial 40 year license for a period of not more than 20 years. The application contains general and technical information. The general information is much the same as that provided with the initial operating license application. The technical information includes an Integrated Plant Assessment (IPA), the CLB changes during the NRC review of the application, a listing and evaluation of Time-Limited Aging Analyses, a supplement to the FSAR, any Technical Specifications changes or additions necessary to manage the effects of aging during the period of extended operations, and a supplement to the plant's environmental report that complies with the requirements of Subpart A of 10 CFR 51.

Long Lived – An item which meets the criteria of 10 CFR 54.21(a)(1)(ii) is considered long lived and potentially subject to aging management. The item is not subject to replacement based on a qualified life or specified time period. In addition, as per NEI 95-10 [Reference 4], structures and components with qualified lives of 40 years or longer are considered to be long lived.

Natural Phenomena – Those phenomena which are identified by General Design Criterion 2 of 10 CFR 50 Appendix A. Structures, systems and components important to safety must be designed to withstand the effects of naturally occurring phenomena (as opposed to phenomena arising from human intervention) without loss of capability to perform their intended safety functions. Natural phenomena include earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches historically reported for a nuclear plant site and/or surrounding area.

Passive – An item which meets the criteria of 10 CFR 54.21(a)(1)(I) is considered to be passive and potentially subject to aging management. As used in relation to a structure or component, the item performs an intended function without moving parts or without a change in configuration or properties.

Regulated Events – Events used for license renewal scoping that relate to the requirements of 10 CFR 54.4(a)(3). The regulated events are those NRC regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

Relevant Condition – The service conditions (e.g., temperature, oxygen concentration, stagnant flow conditions, vibration, tensile stress, etc.) that contribute to the circumstances necessary for an aging effect to occur.

Safety Related – As defined in 10 CFR 50.2 and utilized in §54.4(a)(1), a system, structure, or component is Safety Related, for the purpose of license renewal, if it is relied upon to remain functional during and following a design basis event to assure:

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shutdown the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1) or §100.11.

Safety Related equipment at VCSNS is indicated with the acronyms "NSR", "SR" or the term "Nuclear Safety Related". This classification is defined in ES-411, "Equipment Classification" [Reference 9].

Scoping – The process of identifying plant systems and structures which meet, or do not meet, the criteria of 10 CFR 54.4.

Screening – The process of identifying structures and components which are subject to an Aging Management Review per the criteria of 10 CFR 54.21(a)(1).

The License Renewal Rule ("The Rule") – The License Renewal Rule refers to 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants.

Time-Limited Aging Analysis (TLAA) – Those licensee calculations and analyses that:

- Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);
- Consider the effects of aging;

- Involve time-limited assumptions defined by the current operating term, for example, 40 years;
- Were determined to be relevant by the licensee in making a safety determination;
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CFR 54.4(b); and
- Are contained or incorporated by reference in the CLB.

ABBREVIATIONS AND ACRONYMS:

AMR	Aging Management Review
AMP	Aging Management Program
AMSAC	ATWS Mitigation System Actuation Circuitry
ATWS	Anticipated Transient Without Scram
BAW	Babcock and Wilcox
BOM	Bill of Material
BTP	Branch Technical Position
CER	Condition Evaluation Report
CFR	Code of Federal Regulations
CHAMPS	Computerized History and Maintenance Planning System
CISIP	Containment Inservice Inspection Program
CLB	Current Licensing Basis
DBD	Design Basis Document
DBE	Design Basis Event
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
EQDB	EQ Data Base
FP	Fire Protection
FPER	Fire Protection Evaluation Report
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GTR	Generic Technical Report
IB	Information Bulletin
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IPE	Individual Plant Examination
LCM	Life Cycle Management
LER	Licensee Event Report
LR	License Renewal
LRA	License Renewal Application

LT	Level Transmitter
MIC[1]	Microbiologically Induced Corrosion
NCN	Non-Conformance Notice
NEI	Nuclear Energy Institute (formerly NUMARC)
NNS[2]	Non-Nuclear Safety (Related)
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NSR[2]	Nuclear Safety Related
NSSS	Nuclear Steam Supply System
ONO	Off Normal Occurrence
ppm	parts per million
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
QR[2]	Quality Related
RG	Regulatory Guide
SBO	Station Blackout
SC	Structure and Component
SCC	Stress Corrosion Cracking
SCE&G	South Carolina Electric and Gas
SER	Safety Evaluation Report
SOC	Statements of Consideration
STTS	Surveillance Test Task Sheet
SSC	System, Structure and Component
TLAA	Time-Limited Aging Analysis
VCSNS	Virgil C. Summer Nuclear Station
WOG	Westinghouse Owners Group

1 MIC is procedurally referred to as Microbiologically Induced Corrosion at VCSNS, which is analogous to Microbiologically Influenced Corrosion in general use throughout industry.

2 This abbreviation/terminology is defined in common use at VCSNS. The definitions for these classifications are provided in ES-411, "Equipment Classification" [Reference 9]. In some generically written license renewal resources, the acronym "NSR" may indicate "Non Safety Related" instead of Nuclear Safety Related.

4.0 REFERENCES

The following references were consulted in determining structures and component groups within the scope of License Renewal (LR). The nature of the information in this technical report is such that the report will not require revision simply based on the revision of a listed document.

1. Code of Federal Regulations, Volume 10, Part 54 - Requirements for Renewal of Operating Licenses for Nuclear Power Plants, May 8, 1995.
2. VCSNS Engineering Services Procedure ES-705, Civil/Structural Scoping, Screening, and Aging Management Review for License Renewal.
3. VCSNS Engineering Services Procedure ES-706, Identification and Evaluation of Time-Limited Aging Analyses and Exemptions for License Renewal.
4. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule, Revision 3, March 2001.
5. NUREG-1801, NRC Generic Aging Lesson Learned (GALL) Report, April 2001.
6. NUREG-1800, NRC Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, April 2001.
7. VCSNS Technical Report TR00170-001, Structures Scoping for License Renewal.
8. VCSNS Technical Report TR00170-002, Structures Screening for License Renewal.
9. VCSNS Engineering Services Procedure ES-411, Equipment Classification.
10. NUREG/CR-6679, Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants, August 2000.
11. NUREG/CR-6424, Report on Aging of Nuclear Power Plant Reinforced Concrete Structures, Oak Ridge National Laboratory, March 1996.
12. VCSNS Final Safety Analysis Report (FSAR), Amendment 00-01.
13. VCSNS EQ database drawing series S-021-018.
14. VCSNS Technical Specifications, latest revision.
15. Engineer Technical Work Record, Robert Whorton (Serial 04586), Virgil C. Summer Nuclear Station Water Chemical Analysis Summary, June 14, 2001.
16. VCSNS Design Basis Document (DBD) – Nuclear Safety Related (NSR) Structures, Revision 3.

17. VCSNS Technical Report TR00160-010, General Aging Effects Identification for License Renewal (Mechanical).

18. BAW-2270, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 2.

19. NUREG/CR-4715, An Aging Assessment of Relay and Circuit Breakers and System Interactions, prepared by Franklin Research Center for Brookhaven National Laboratory, June 1987.

20. SAND 93-7069, Aging Management Guideline for Commercial Nuclear Power Plants – Motor Control Centers, Sandia National Laboratories, February 1994.

21. SAND 93-7027, Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Switchgear, Sandia National Laboratories, July 1993.

22. WCAP-14578, Aging Management Evaluation for Electrical Distribution, Control, and Monitoring Equipment, Westinghouse Energy Systems, July 1997.

23. GL 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (Generic Letter 88-05), March 17, 1988.

24. VCSNS Maintenance Rule Inspections – 2000, Assessment of In-Service Conditions of Important to Maintenance Rule (ITMR) Structures.

25. EPRI NP-5769, Volume 1, Degradation and Failure of Bolting in Nuclear Power Plants, Electric Power Research Institute, Palo Alto, California, April 1988.

26. VCSNS Technical Report TR00160-012, Mechanical Component Aging Management Review for License Renewal (Borated Water Systems).

27. VCSNS Technical Report TR00160-014, Mechanical Component Aging Management Review for License Renewal (Raw Water Systems).

28. Uhlig, M. H., Corrosion Handbook, John Wiley and Sons, 1948.

29. Leech, Miller, Renwick, and Wright, Conquering Service Water Pipe Corrosion, Pipework and Valves, January 1984.

30. Cragnolino, G., A Review of Erosion-Corrosion of Steels in High Temperature Water, Proceedings of the Third International Symposium on Environmental Degradation of Material In Nuclear Power Reactors, American Nuclear Society, LaGrange Park, Illinois, 1987, page 397.

31. EPRI NP-3482, A Study of Microbiologically Influenced Corrosion in Nuclear Power Plants and a Practical Guide for Countermeasures, Electric Power Research Institute, Palo Alto, CA, May 1986.

32. NUREG/CP-0120, Aging of Concrete Containment Structures in Nuclear Power Plants, Proceedings of the Fifth Workshop on Containment Integrity, July 1992.

33. ACI 318-71, American Concrete Institute (ACI) Building Code.

34. ACI 301-72, American Concrete Institute (ACI) Specification for Structural Concrete for Buildings.

35. ACI 201.1R-92, Guide for Making a Condition Survey of Concrete in Service, American Concrete Institute, Detroit, Michigan.

36. NUMARC Report Number 90-01, Pressurized Water Reactor Containment Structures License Renewal Industry Report, Nuclear Management and Resource Council, August 1989, Revision 1, September 1991.

37. NUMARC Report Number 90-06, Class 1 Structures License Renewal Industry Report, Nuclear Management and Resource Council, June 1990.

38. WCAP-15101, Analysis of Capsule W from the South Carolina Electric & Company V.C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program, September 1998.

39. NUREG/CR-4652, ORNL/TM-10059, Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants, Oak Ridge National Laboratory, Oak Ridge, Tenn., September 1986.

40. ORNL/NRC/LTR-90/17, Hookam, C. J., Structural Aging Assessment Methodology for Concrete Structures in Nuclear Power Plants, Contractor Report, March 1991.

41. ACI 201.2R-77, Guide to Durable Concrete, (Reapproved 1982), American Concrete Institute, Detroit, Michigan.

42. VCSNS Design Specification SP-201, Structural Concrete.

43. ACI 209R-92, Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures, American Concrete Institute, Detroit, Michigan.

44. NRC letter to T. C. Nichols dated August 24, 1981, Report No. 50-395/81-19.

45. EPRI TR-103842, Class 1 Structures License Renewal Industry Report, Revision 1, July 1994.

46. ACI 215R-74, Consideration for Design of Concrete Structures Subjected to Fatigue Loading, American Concrete Institute, Detroit Michigan.

47. NUREG CP-0100, Concrete Degradation Monitoring and Evaluation, Proceedings of the International Nuclear Power Plant Aging Symposium, Prasad, N., August 30 – September 1, 1988, U. S. Nuclear Regulatory Commission, Washington D.C.

48. NUREG-1522, Assessment of Inservice Conditions of Safety Related Nuclear Plant Structures, U. S. Nuclear Regulatory Commission, June 1995.

49. OEDB Number 96-012318 and Nuclear Network – OE8136, Problems and events associated with Masterflow 713 Non-Shrink Grout – Grand Gulf Nuclear Station, 1996.

50. NUREG/CR-3604, Bolting Applications, Brookhaven National Laboratory for U. S. Nuclear Regulatory Commission, May 1984.

51. VCSNS Individual Plant Examination for External Events (IPEEE) Report, June 1995.

52. VCSNS Containment Inservice Inspection – 2000 ASME Section XI, Subsections IWE and IWL, Responsible Engineer Evaluation Report, January 2, 2001.

53. VCSNS Technical Report TR00160-017, Mechanical Component Aging Management Review for License Renewal (Ventilation Systems).

54. Materials Handbook, McGraw-Hill, Inc., New York, New York, 1991.

55. Engineering Materials Handbook, McGraw-Hill, Inc., New York, New York, 1958.

56. SECY-96-146, Technical Assessment of Fire Barrier Penetration Seals in Nuclear Power Plants, Nuclear Regulatory Commission, dated July 1, 1996.

57. Materials Handbook, Tenth Edition - McGraw-Hill Publishing, New York, New York, 1971.

58. ASTM D-1566-95a, Standard Terminology Related to Rubber, American Standards for Testing and Materials, Philadelphia, Pennsylvania, 1995.

59. Elastomeric Materials, The International Plastics Selector, Inc., San Diego, California, 1977.

60. Concise Encyclopedia of Polymer Science and Engineering, John Wiley & Sons, 1990.

61. Engineering Materials Properties and Selection, Budinski, Kenneth G., Prentice-Hall, Inc., Englewood Cliffs, New Jersey, 1992.

62. VCSNS Design Specification SP-227, Penetration Seals.

63. Textbook of Polymer Science, Billmeyer; F. W., Second Edition, John Wiley & Sons, Inc. 1971, Table 19-3.

64. Selection Guide to Organic Materials for Nuclear Engineering, Van de Voorde; M. H. and Restat; C. CERN-72-7, European Organization for Nuclear Research, Geneva 1972.

65. Steelcote Manufacturing Company, Material Safety Data Sheet (MSDS) A04790100A01 and Product Data Sheet (PDS) No. 479, June 14, 2000 and FAX dated August 1, 2001 - One Steelcote Square - St. Louis, MO 83103-2990.

66. VCSNS Technical Requirements Package TRP-2, Passive Fire Protection, Revision 7.

67. VCSNS Design Basis Document (DBD) – Reactor Building (RB), Revision 6.

68. NUREG/CR-6154 Experimental Results from Containment Piping Bellows Subjected to Severe Accident Conditions, Volumes 1 and 2, October 1995.

69. VCSNS Design Specification SP-606, Reactor Building Piping Penetrations.

70. WCAP-14422, Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports, Revision 2-A, December 2000.

71. ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments, 1986.

72. ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit, Michigan.

73. NRC Inspection Report (IR) 50-395 / 82-54, November 23, 1982.

74. VCSNS Calculation DC00020-209, Rev. 0, Thermal Analysis Reactor Cavity.

75. VCSNS Technical Report TR00160-020, Program/Activity Evaluations for License Renewal (Mechanical).

76. ISE-4, VCSNS Containment Inservice Examination (ISE) Manual – ASME Section XI – Subsection IWE and IWL.

77. EPRI Report No. AP-4714, Inspection and Performance Evaluation of Dams, September 1986.

78. Introductory Soil Mechanics and Foundations, Third Edition, Sowers, G. B. and G. F. Sowers, The MacMillan Company, 1970.

79. Soil Mechanics in Engineering Practice, Second Edition, Terzaghi, K. and R. B. Peck, John Wiley and Sons, Inc., 1967.

80. Risk of Dam Failure in Benefit-Cost Analysis, Baecher, G. B., M. E. Pate, and R. De Neufville, Water Resources Research, vol. 16, No. 3, 1980.

81. A Database for the Evaluation of the Frequency of Random Dam Failure, Report 120-010-01, Benjamin, Jack R., and Associates, Palo Alto, California, 1982.

82. Classification of Risk, Proceedings International Congress on Large Dams, Vol. 1, Gruner, E., Madrid, 1973, pp. 55-68.

83. VCSNS Design Specification DSP-544V, Class 1 Pipe Supports.

84. VCSNS Design Specification SP-544, Fabrication of Nuclear Piping – ASME III.

85. VCSNS Design Basis Document (DBD) – Class 1 Piping, Revision 3.

86. NRC Final Rule, 52 FR 41288, October 27, 1987.

87. WCAP-13206, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Virgil C. Summer Nuclear Power Plant, April 1992.

88. NRC SER, Request to Use Leak-Before-Break Technology for Reactor Coolant System Piping – Virgil C. Summer Nuclear Station, January 11, 1993.

89. WCAP-9119, Structural Analysis of Reactor Coolant Loop for the V. C. Summer Nuclear Station Volume 1, Piping Analysis of the Reactor Coolant Loop, Rev. 4, October 1997.

90. Regulatory Guide 1.163, Performance Based Containment Leak Test Program, Revision 0, September 1995.

91. NRC Inspection Report (IR) 50-395 / 93-09, March 21, 1993.

92. NRC Inspection Report (IR) 50-395 / 94-25, November 25, 1994.

93. VCSNS NCN 99-0489, Disposition 6, April 27, 1999.

94. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition.

95. ISE-3, VCSNS Inservice Examination (ISE) Manual Document, ASME Section XI, Inservice Examination Manual for Second Inspection Interval, Revision 2, February 6, 2001.

96. WCAP-9803, Volume 4, ASME Section III Class 1 Piping Stress Analysis for Auxiliary Piping Systems, Revision 3, July 21, 2000.

97. Safety Evaluation Report (SER) for license renewal of Calvert Cliffs Nuclear Station.

98. VCSNS Design Specification SP-527, Specification for Miscellaneous Steel.

99. NRC Inspection Report (IR) 50-395 / 91-21, December 3, 1991.

100. VCSNS Fire Protection Procedure FPP-024, Fire Suppression.

101. IEEE-450, Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations, 1987 Edition, Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, New York.

102. Holtec International Report HI-20126624, Spent Fuel Storage Expansion at Virgil C. Summer for South Carolina Electric & Gas, Revision 1, Holtec Center, Marlton, NJ.

103. NRC Staff Spent Fuel Pool Action Plan - Task Action Plan for Spent Fuel Storage Pool Safety, July 26, 1996.

104. Licensing Report on High Density Fuel Racks, VCSNS Docket Number 50-395, December 28, 1983.

105. VCSNS Design Basis Document (DBD) – Spent Fuel Cooling System (SF), Revision 5.

106. Letter RC-98-0207, SCE&G response to GL 98-04.

107. Letter CGGS-25444, from A. Honardoost, Gilbert and Associates to C. A. Price, SCE&G, December 9, 1981.

108. VCSNS Engineering Services Procedure ES-438, Containment Inservice Inspection (ISI) Program.

109. Fire Protection Evaluation Report (FPER), Amendment 00-01.

110. VCSNS Fire Protection Procedure FPP-025, Fire Containment.

111. VCSNS Penetration Seal Project Plan, Revision 1, April 27, 2000.

112. NRC Inspection Report (IR) 50-395 / 93-15, June 2, 1993.

113. VCSNS Design Basis Document (DBD) – Fire Protection System (FS), Revision 2A.

114. VCSNS Fire Protection Procedure FPP-015, Shift Inspection.

115. NRC Inspection Report (IR) 50-395 / 88-04, March 4, 1988.

116. Generic Letter (GL) 88-20, Individual Plant Examinations for Severe Accident Vulnerabilities, November 23, 1988.

117. Regulatory Guide (RG) 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Revision 1, March 1978.

118. VCSNS Calculation DC03290-001, Flooding Evaluation for All AB and IB Areas Affected by FW, Revision 5, August 2, 2000.

119. VCSNS Calculation DC03290-002, Flooding Evaluation for All Areas Except Those Affected by the FW System, Revision 5, December 7,1999.

120. VCSNS Calculation DC03190-009, Mechanical Calculation Flooding in Reactor Building, Revision 3, August 3, 1995.

121. VCSNS Calculation DC03190-001, Mechanical Calculation Flooding in Reactor Building, Revision 6, April 16, 1999.

122. VCSNS Design Specification SP-631, Blast Resistant and Watertight Doors and Frames.

123. VCSNS Design Basis Document (DBD) – Drains, Sumps, and Leak Detection (ND), Revision 2.

124. VCSNS Engineering Services Procedure ES-437, Inspections for Maintenance Rule Structures.

125. Engineer Technical Work Record, Larry Bennett (Serial 15517), Review of Structures for Maintenance Rule, November 14, 1995.

126. VCSNS Technical Report TR00010-001, Maintenance Rule Summary Document.

127. ANSI B30.2, Specifications for Lifting Devices - Overhead and Gantry Cranes, 1976.

128. AISC Manual of Steel Construction - Seventh Edition, Chicago, IL 60611.

129. ANSI N45.2.15, Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants, 1981.

130. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, Revision 5, July 8, 1983

131. OSHA (CFR) 29 - Chapter XVII, Part 1910.179(j)(3).

132. Technical Evaluation Report EGG-HS-6371 attached to NUREG-0717, Safety Evaluation Report (SER) for the VCSNS Docket 50-395, 1983 including all Supplements.

133. VCSNS Technical Report TR03920-002, Control of Heavy Loads at VCSNS, Revision 9, July 1, 1998.

134. VCSNS Technical Requirements Package TRP-14, Cranes, Hoists, and Runway Systems, Revision 2.

135. NRC Inspection Report (IR) 50-395 / 82-28, April 19, 1982.

136. Engineer Technical Work Record TWR 10203, G.L. Croxton, Maintenance Rule Review of Industry Operating Experience for the MH System, June 11, 1996.

137. VCSNS Technical Requirements Package TRP-12, Pressure Barriers, Revision 4.

138. NRC Inspection Report (IR) 50-395 / 91-11, July 12, 1991.

139. NRC letter to G. J. Taylor (dated February 27, 1998), Service Water Pond Dam Safety Inspection Results for Virgil C. Summer Nuclear Station.

140. NRC letter to S. A. Byrne (dated September 19, 2000), Results of Dam Safety Inspection Related to the Category I Service Water Pond Dams at the Virgil C. Summer Nuclear Station.

141. SCE&G letter to L. M. Padovan (dated August 28, 1998), Service Water Pond Dam Safety Inspection Results.

142. VCSNS Structural Calculation Series DC0305C.

143. VCSNS Calculation DC0305C-014, Generation of revised force-time curves and normalizing factors for SP-228.

144. VCSNS Technical Report TR00140-001, Time-Limited Aging Analyses and Exemptions for License Renewal.

145. NRC Regulatory Guide 1.35, Proposed Revision 3, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments.

146. Letter to O.S. Bradham, SCE&G, from J.L. Hayes, NRC, April 28, 1989, Issuance of Amendment No. 76 to Facility Operating License No. NPF-12 - Virgil C. Summer Nuclear Station, Unit No. 1, regarding Containment Structural Integrity (TAC No. 62803).

147. VCSNS Design Specification SP-228, Surveillance of Reactor Building Post Tensioning System.

148. VCSNS Engineering Services Guideline ST-04, Reactor Building Tendon Surveillance.

149. NUREG-0717, Safety Evaluation Report (SER) Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1, February 1981 (includes Supplements).

150. VCSNS Calculation DC0306B-013, Reactor Building Penetration Assembly.

151. VCSNS Engineering Services Procedure ES-401, Evaluation of Operational Cycles and Transients.

152. VCSNS Calculation DC03650-011, Service Water Pump House (SWPH) – Secondary Consolidation.

153. CMAA 70, Crane Manufacturers Association of America (CMAA) Specification No. 70.

154. VCSNS Calculation DC03920-022, RB Polar Crane Qualification for 150 Ton Normal Lift.

155. Letter from W. E. Desmarchais to E. Wielkopolski, December 18, 1973, Polar Crane Lifts.

156. Engineer Technical Work Record, Greg Croxton (Serial 10203), PIP 01-1011: Cause Evaluation and Corrective Actions Structural Deficiencies, November 15, 2001.

157. VCSNS Annunciator Response Procedure ARP-001-XCP-609, Annunciator Response Procedure for Panel XCP-609, Spent Fuel Pool Temperature Indicators TI-7435 and TI-7437.

158. Letter RC-01-0135 to K.R. Cotton, NRC, from S.A. Byrne, SCE&G, July 24, 2001, Technical Specification Amendment Request TSP 99-0090 Spent Fuel Pool Storage Expansion.

159. VCSNS Station Administrative Procedure SAP-143, Preventive Maintenance Program.

160. VCSNS Station Administrative Procedure SAP-1131, Electronic Processing of Condition Evaluation Reports.

161. VCSNS Station Administrative Procedure SAP-134, Control of Station Surveillance Activities.

162. VCSNS Surveillance Test Procedure, STP-207.002, Inspection of Containment.

163. VCSNS Design Engineering Guideline ST-07, Containment Inservice Inspection Evaluation Criteria.

164. VCSNS General Test Procedure, GTP-315, Containment Leak Rate Testing Program.

165. EPRI Report No. TR-105714, PWR Primary Water Chemistry Guidelines, Revision 4, March 1999.

166. EPRI Report No. TR-102134, PWR Secondary Water Chemistry Guidelines, Revision 5, June 2000.

167. VCSNS Chemistry Procedure CP-618, Chemistry Specifications for Borated Systems and Tanks.

168. VCSNS Chemistry Procedure CP-602, Chemistry Reporting.

5.0 AGING MANAGEMENT REVIEW METHODOLOGY FOR STRUCTURES AND STRUCTURAL COMPONENTS

The VCSNS license renewal aging management review methodology is provided in ES-705, Civil/Structural Scoping, Screening, and Aging Management Review for License Renewal [Reference 2]. The developed methodology involves several activities and follows the guidance provided in NEI 95-10 [Reference 4]:

- Review Generic Technical Reports
- Identification and Assessment of Applicable Aging Effects / Relevant Conditions
- Review of Operating Experience
- Identification of Aging Management Programs that Manage Aging Effects
- Demonstration that Programs Manage Applicable Aging Effects

The license renewal rule requires that both the structures and components subject to an aging management review be identified. The structures within the scope of license renewal were identified in VCSNS Technical Report TR00170-001, Structures Scoping for License Renewal [Reference 7]. In order to optimize the aging management review, structures which are attached to or contained within a larger structure are reviewed with the larger structure. In addition, earthen embankments are reviewed separately from other structures because of the unique material of construction and the aging management programs. A summary listing of those structures in scope is provided below.

- Auxiliary Building [includes RWST & RMWST foundations and West Penetration Access Area (WPAA)]
- Control Building
- Intermediate Building [includes East Penetration Access Area (EPAA)]
- Diesel Generator Building
- Fuel Handling Building
- Hot Machine Shop
- Reactor Building and Internal Structures
- Turbine Building
- Service Water Pumphouse, Intake and Discharge Structures
- Yard Structures (includes Fire Service Pumphouse, CST foundation, Electrical Manhole EMH-2, Electrical Substation foundation for OCB 8892, Transformer Area foundations, and Transmission Towers and foundations from the Emergency Auxiliary Transformer to OCB 8892)
- Earthen Embankments (Includes Service Water Pond North Dam, South Dam, East Dam, West Embankment and North Berm)

5.1 REVIEW GENERIC TECHNICAL REPORTS

The following is a list of Generic Technical Reports reviewed for applicability for VCSNS AMR preparation:

NUREG-1801	Generic Aging Lessons Learned (GALL) Report, April 2001
NUREG/CR-6679	Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants, August 2000
WCAP-14422-2A	License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, December 2000
WCAP-14578	Aging Management Evaluation for Electrical Distribution, Control and Monitoring Equipment, July 1997
WCAP-14579	Aging Management Evaluation for Safety Class 1 Piping Supports, July 1997
WCAP-14580	Aging Management Evaluation for Seismic Category I Structures, February 1997
WCAP-14756-A	Aging Management Evaluation for Pressurized Water Reactor Containment Structure, May 2001
BAW-2279	Aging Effects for Structures and Structural Components (Structural Tools), December 1997
EPRI NP-5461	Component Life Estimation: LWR Structural Materials Degradation Mechanisms, September 1987
EPRI TR-103835	PWR Containment Structures License Renewal Industry Report, July 1994
EPRI TR-103842	Class 1 Structures License Renewal Industry Report, July 1994
EPRI TR-105272	Aging Management Evaluation of Reactor Coolant System Supports for Westinghouse Plants, June 1995
EPRI TR-114881	Aging Effects for Structures and Structural Components (Structural Tools), April 2000

5.2 IDENTIFICATION AND ASSESSMENT OF APPLICABLE AGING EFFECTS / RELEVANT CONDITION

The intended functions for each of the structures and structural components subject to an aging management review have been previously identified through the Structures Screening for License Renewal [Reference 8]. To properly assess the effect of aging on components, an evaluation is performed to determine the effect aging has on the ability of a component to perform its intended function(s). Therefore, the first step in the AMR process is to identify the applicable aging effects and/or relevant conditions and assess their effect on the ability of structures and structural components to perform their intended functions.

The identification and assessment of aging effects/relevant conditions includes the following:

- Identification of the aging effects for the subject components.
- Assessing the impact the aging effect has on the ability of the subject component to perform its component intended function(s), and
- Describing the aging effect(s) and/or relevant conditions which must be monitored and/or controlled in a credited aging management program to ensure that the effects of aging are managed.

5.3 REVIEW OF OPERATING EXPERIENCE

The impact of the applicable aging effects on the intended functions of structures and structural components must be determined. A review of operating experience forms part of this review. This includes a review of industry experience from Licensee Event Reports (LERs); generic documentation, such as NRC Information Notices (INs) and Information Bulletins (IBs); and VCSNS experience from Non Conformance Notices (NCNs), Condition Evaluation Reports (CERs), Off Normal Occurrences (ONOs), etc. This data search is not meant to be all-inclusive, but to provide reasonable assurance that all applicable aging effects have been identified. The review also included NRC generic communications such as Bulletins, Circulars, Generic Letters, and Information Notices for relevant industry operating experience.

A reasonable and representative dataset of VCSNS aging related operating experience was provided by Non-Conformance Notices (NCNs) documenting equipment and structural anomalies during the approximate five year period from January 1996 through July 2001. This NCN search was conducted comprehensively for multiple systems and disciplines in order to minimize duplication and to ensure that relevant operating experience was not omitted in the refinement process for a system or discipline specific search. Conservatively, no effort was made to eliminate from consideration operating experience for systems and components outside the scope of license renewal except as a final consideration for a particular NCN. The search was conducted by SCE&G Quality Control (QC) personnel knowledgeable in the NCN process and in plant structures, systems, and components, with final disposition by cognizant license renewal personnel.

Due to the potential for aging related issues that did not result in a nonconformance, a keyword search of Condition Evaluation Reports (CERs) in the PIP database was also conducted by the same SCE&G QC personnel to identify relevant events. The approximate time period of August 1998 to July 2001, which begins with the site implementation of the Database, was searched. Using 70 keywords supplied by cognizant license renewal personnel, CERs were identified as possibly related to aging. The results of this CER search did not warrant a further search of CERs prior to August 1998 with the reasonable assumption that any aging effects of significance prior to that date would have resulted in a NCN condition, and thus would be included in the representative dataset.

NUREG/CR-6679 [Reference 10] includes a discussion of NRC bulletins, information notices, and generic letters relevant to operating experience concerning age related degradation of structures and passive components. No limiting or representative time period was established for these generic communications. As such, the relevant aging effects and associated mechanisms were incorporated into the logic used to identify aging effects/mechanisms requiring consideration at VCSNS. A review of pertinent NUREG/CR-6679 information was conducted by cognizant license renewal personnel in order to identify events of particular applicability to VCSNS. For completeness, a review of pertinent NRC generic correspondence (such as information notices) was conducted by cognizant license renewal personnel for the approximate time period between the issuance of NUREG/CR-6679 and July 2001.

Lastly, a search of Licensee Event Reports (LERs) for VCSNS was conducted by cognizant license renewal personnel for the purpose of the identification of pertinent aging effects. This review was conducted for the life of the plant rather than a certain interval and served also to identify and disposition abnormal occurrences in the life of the plant that could impact component functionality and/or the conditions resulting in aging. No previously unidentified aging effects were identified for consideration at VCSNS through this search. Any abnormal occurrences during plant life were resolved such that there has been no adverse impact to the functionality of presently installed structures and structural components.

5.4 IDENTIFICATION OF AGING MANAGEMENT PROGRAMS THAT MANAGE AGING EFFECTS

The results from the identification and assessment of applicable aging effects have been summarized for each commodity group. For each aging effect identified for a subject structure or structural component, at least one plant aging management program is identified that will manage the applicable aging effect or the relevant condition during the period of extended operation.

Some VCSNS structures and components within the scope of license renewal are located in areas that are inaccessible for inspection. The VCSNS aging management reviews do not ignore any environmental conditions to which the structures and components are exposed, including those conditions in areas that may turn out to be inaccessible for inspection. For example, structures and components located below grade may be exposed to groundwater. The groundwater environment could potentially lead to aging effects different from those in an air environment. The groundwater chemistry plays a major role in the determination of the degradation of the below grade structures and components.

When determining the aging effects requiring management for below grade structures and components, the groundwater chemistry was evaluated to determine if parameters exceeded documented limits where degradation would occur. Therefore, the unique environment (groundwater) of the inaccessible structure or component was considered as part of the aging management review.

When the conditions in the inaccessible and accessible areas result in the same aging effects (i.e., concrete below grade and concrete in air environment), the aging management program of the inaccessible areas may be based on symptomatic evidence in an accessible area. For example, the aging effect due to alkali-aggregate reactions of concrete would manifest in both accessible and inaccessible areas.

For the case where symptomatic evidence in accessible areas provides guidance for aging effects in inaccessible areas, the aging management review assures that the aging effects due to the environment in the accessible region and the aging effects due to the environment in the inaccessible region are simultaneously evaluated. This philosophy forms the successful basis for the ASME Section XI ISI Program used to programmatically deal with inaccessible locations that require assurance of integrity.

5.5 DEMONSTRATION THAT PROGRAMS MANAGE APPLICABLE AGING EFFECTS

Existing programs or portions of programs which are credited with managing the effects of aging are shown to be effective in managing aging in the structures and structural components in the program. Therefore, when an aging effect for a structure or structural component is determined, it must be demonstrated that the existing programs or portions of programs effectively manage the effects of aging so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

The program evaluation includes a review of the program to determine if it contains the attributes defined for an effective program. The primary input to the attribute definitions is from the Standard Review Plan for License Renewal [Reference 6]. The attribute definitions used to describe existing programs and activities are provided as follows:

Element	Description
1. Scope of Program	Scope of program should include the specific structures and components subject to an Aging Management Review (AMR) for license renewal.
2. Preventive Actions	Preventive actions should mitigate or prevent aging degradation. This attribute is not applicable for condition monitoring programs and performance monitoring programs.
3. Parameters Monitored or Inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure and component intended function(s).
4. Detection of Aging Effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and Trending	Monitoring and trending should provide predictability of the extent of degradation, and timely corrective or mitigative actions.
6. Acceptance Criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation.
7. Corrective Actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely. Corrective actions are completed in accordance with the VCSNS Quality Assurance Program.
8. Confirmation Process	Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective. VCSNS quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.
9. Administrative Controls	Administrative controls should provide a formal review and approval process.

Element	Description
	VCSNS quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.
10. Operating Experience	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a conclusion that the effects of aging will be adequately managed so that the structure and component intended function(s) will be maintained during the period of extended operation.
11. GALL Comparison[1]	A comparison between the plant aging management program and the GALL generic aging management program evaluation.

[1] The Generic Aging Lessons Learned (GALL) report [Reference 5] contains the staff's generic evaluation of the existing plant programs and documents the technical basis for determining where existing programs are adequate without modification and where existing programs should be augmented for the extended period of operation. The evaluation results documented in the GALL report indicate that many of the existing programs are adequate to manage the aging effects for particular structures or components for license renewal without change. The GALL report also contains recommendations on specific areas for which existing programs should be augmented for license renewal.

6.0 AGING EFFECTS EVALUATIONS

This section contains the results of the aging effects evaluation for the structures within the scope of license renewal. The Aging Management Review (AMR) process described in Section 5.0 is applied to the VCSNS structures and structural components which are within the scope of license renewal 10 CFR 54.4(a) and require an aging management review 10 CFR 54.21(a)(1).

For the structures requiring an aging management review, the aging effects evaluation is performed on commodity sets within the structures. The components are grouped into commodity sets based on the material of construction and environment. The commodity sets are:

- Steel in air environment
- Steel in fluid environment
- Concrete in air and fluid environment
- Fire Barriers
- Elastomers
- Earthen Embankments

6.1 ENVIRONMENTS

Structures and components within the scope of license renewal are exposed to several environments which are applicable to the aging management review. The in-scope VCSNS structures and components are identified in VCSNS Technical Reports TR00170-001, Structures Scoping for License Renewal [Reference 7] and TR00170-002, Structures Screening for License Renewal [Reference 8]. The environments to which these structures and components are exposed are discussed below.

- Below Grade Environment: Below grade portions of the structures are exposed to back fill and groundwater. The groundwater chemistry plays a major role in the determination of the degradation of below grade components. The results of chemical analyses of the groundwater are provided in Table 6.1-3.

- Concrete Environment: Steel components located in concrete (such as anchorage, rebar, and embedments) are protected by the alkaline environment of the concrete. The highly alkaline conditions present within the concrete (pH > 12.5) cause a passive iron oxide to form on the iron surface thus protecting it from further oxidation [References 5 and 11].

- External Environment: External surfaces of structures are exposed to the external atmospheric environment. Maximum external temperatures are projected to be 108°F, and humidity is taken to be 100% [Reference 12].

VCSNS is located in a rural environment and is not located near major industrial plants which would raise the possibility of exposure to sulfate or chloride attack.

- Internal Environment: Internal portions of the structures and their components are exposed to sheltered environments which protect them from external weather and temperature changes. These sheltered environments may be controlled, such as in the Control Room, where the temperature and humidity are relatively mild. Other areas, such as the interior of the Auxiliary Building, Intermediate Building, Turbine Building or Reactor Building, are exposed to high temperatures, humidity, and radiation (Reactor Building only). These variables play a major role in the potential degradation of the components located within this environment.

Thermal and radiation environmental data were extracted from the EQ program, combined into tables, and then sorted by location. Table 6.1-1, "Building Internal Temperature Environment," and Table 6.1-2, "Building Radiation Dose During Normal Operation," present a summary of all environmental conditions by buildings, elevations, and radiation zones so that the components can be analyzed for aging resulting from the building-specific, worst-case environment. Because temperature information was not available for some of the areas, a maximum temperature of 108° F and a minimum of -4° F, which are the recorded extreme high / low temperatures for the local climate [Reference 12, Table 2.3-49], were assumed for these areas. Local meteorology data was compiled over a 68 year period. The average daily high / low temperatures for the local climate are 92° F and 32° F.

Specific design radiation data (maximum) for normal operation was obtained from the VCSNS EQ program [Reference 13]. The EQ program manual presents dose data by radiation zones. The bounding dose for each building elevation or major area was extracted and is shown in Table 6.1-2. These values should be considered to be very conservative maximums, and the actual 40 year dose will be lower (in some cases, much lower). For all areas of the plant not listed in the EQ program manual, it was assumed that the maximum 40 year dose (gamma) was less than 1,000 rads; the actual dose in many of the areas will be negligible.

The expected normal dose for 60 years at VCSNS can be determined by multiplying the current 40 year normal dose by the ratio of 1.5 (60/40). For example, if the normal 40 year dose for a given area is 3×10^4 rads, then the 60 year dose will be 4.5×10^4 rads. Since multiplying the current 40 year normal dose by 1.5 does not change the order of magnitude of the total dose, it is reasonable to conclude that any structural material that can withstand the effects of the 40 year dose will also withstand the effects of the 60 year dose without any significant degradation in the structural properties of the material.

- Reactor Building Environment: The Reactor Building atmosphere is a moist air environment. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry. Typical systems are the Safety Injection, Main Steam, Reactor Coolant, and Feedwater Systems. The Reactor Building environment definition is analogous to the internal environment definition, it is primarily used to distinguish structural components located inside the Reactor Building.

- Raw Water Environment: Portions of the Service Water Pumphouse, Service Water Intake and Discharge Structures are exposed to raw water from the Service Water Pond. The water chemistry of Service Water Pond plays a major role in the determination of the degradation. The results of chemical analyses of VCSNS site raw water are provided in Table 6.1-3.

- Borated Water Environment: The borated water environment is associated with the Spent Fuel Pool. The spent fuel pool contains an oxygen-saturated borated water environment with a concentration of greater than or equal to 2000 ppm boron. Minimum boron concentration is set by the Technical Specifications [Reference 14], Section 3/4.9.1, Boron Concentration. The pH of the spent fuel pool ranges from 4.0 to 8.0 [Reference 12].

Table 6.1-1
Virgil C. Summer Nuclear Station
Building Temperature Environments

Building or Area	Environmental Zone from drawing S-021-018 Sheets 4-1 through 4-15	Temperature Range		Reference for Temperature Range is from drawing S-021-018
		Minimum	Maximum	
Auxiliary	AB-01 through AB-90	65°F	121°F	Sheets 3-1 through 3-96
Control - HVAC Rooms	CB-01 and CB-02	68°F	108°F	Sheets 3-97 through 3-108
Control - Cable Spreading Rooms	CB-03 and CB-03C	74°F	87°F	Sheets 3-99 and 3-102
Control - Control Access Area	CB-03A	73°F	77°F	Sheet 3-100
Control - Corridors	CB-03B, CB-04A, CB-07	73°F	97°F	Sheets 3-101, 3-104, 3-107
Control Room and Technical Support Center	CB-04	73°F	81°F	Sheet 3-103 The Control Room is normally maintained at 75°F ± 2°F, 50% relative humidity [Reference 12, Section 6.4.1]
Control	CB-05, CB-06, and CB-08	73°F	102°F	Sheets 3-105, 3-106, 3-108
Diesel Generator	DG-01 and DG-01A	79°F	121°F	Sheets 3-109 and 3-110
Diesel Generator - Engine Foundation Room	DG-01B	65°F (Assumed)	109°F	Sheet 3-111
Diesel Generator - Outside Engine Room	DG-01C and DG-01D	64°F	114°F	Sheets 3-112 and 3-113
Diesel Generator Exhaust	DG-01E	65°F (Assumed)	117°F	Sheet 3-114
Diesel Generator - Engine Room	DG-02	65°F (Assumed)	123°F	Sheet 3-115
Fuel Handling	FH-01 through FH-10	65°F	115°F	Sheets 3-116 through 3-126
Intermediate	IB-01 through IB-13	65°F	131°F	Sheets 3-127 through 3-165
West Penetration Access Area (WPAA)	PAA-01 through PAA-06	65°F	121°F	Sheets 3-166 through 3-177
East Penetration Access Area (EPAA)	PAI-01 through PAI-02	65°F	121°F	Sheets 3-178 through 3-181
Reactor - Incore Instrumentation Area	RB-01	65°F (Assumed)	120°F	Sheet 3-182
Reactor - Inside Steam Generator Cubicles	RB-02, RB-04, and RB-07	65°F (Assumed)	117°F	Sheets 3-183, 3-185, 3-188

Table 6.1-1
Virgil C. Summer Nuclear Station
Building Temperature Environments

Building or Area	Environmental Zone from drawing S-021-018 Sheets 4-1 through 4-15	Temperature Range		Reference for Temperature Range is from drawing S-021-018
		Minimum	Maximum	
Reactor - Outside Steam Generator Cubicles	RB-03, RB-05, and RB-06	65°F (Assumed)	106°F	Sheets 3-184, 3-186, 3-187
Reactor - Above Reactor Head but Below Operating Floor Elevation 463'	RB-08	65°F (Assumed)	157°F [1]	Sheet 3-189
Reactor - Inside Pressurizer Cubicle	RB-09	65°F (Assumed)	115°F	Sheet 3-190
Reactor - Above Operating Floor Elevation 463'	RB-10 and RB-11	65°F (Assumed)	114°F	Sheet 3-191
Service Water Pump House (SWPH)	SW-01	74°F	102°F	Sheet 3-193
Turbine	TB-01	65°F (Assumed)	104°F	Sheet 3-194
Fire Service Pump House (FSPH)	None Designated	65°F	95°F	SP-360, Section 2:05.1
Yard Structure – Underground Duct Banks, EMH-2	None Designated	55°F	85°F	Temperature range taken from Electrical discipline.
Yard	None Designated - Outside	-4°F	108°F	FSAR Section 2.3.1.3.12.2.2

[1] Temperature (in excess of 150 °F) exceeds the design limit for concrete temperature as defined by ACI-349-1985. High temperature is the result of poor air distribution and exhaust from CRDM Shroud Cooling System as described in CGSP-454 dated October 13, 1982. CGSP-454 states concrete temperatures are marginally acceptable. This issue was tracked as NRC Open Item 80-25-09 and closed in NRC Inspection Report 50-395 / 82-54, Item 10 [Reference 73].

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID		Maximum 40 Year Normal TID is from drawing S-021-018
		40 years (rads)	60 years (rads)	
Auxiliary - Floor Drain Tank Room	AB-01	1×10^{7}	1.5×10^{7}	Sheet 3-1
Auxiliary - Reactor Building Spray Pump Cubicles	AB-02	1×10^{6}	1.5×10^{6}	Sheet 3-2
Auxiliary - Below RWST	AB-03, AB-04, AB-04A, AB-07	5.3×10^{3}	7.95×10^{3}	Sheets 3-3, 3-4, 3-5, and 3-8
Auxiliary - Miscellaneous Waste Drain Tank Room	AB-05	2.8×10^{6}	4.2×10^{6}	Sheet 3-6
Auxiliary - Nuclear Blowdown System Rupture Reservoir	AB-06	5.3×10^{3}	7.95×10^{3}	Sheet 3-7
Auxiliary - Waste Evaporator Condensate Tank Room	AB-08 and AB-13	8.8×10^{2}	1.32×10^{3}	Sheet 3-9 and 3-14
Auxiliary - Valve Gallery	AB-09	3.5×10^{4}	5.25×10^{4}	Sheet 3-10
Auxiliary - Charging Pumps and Piping Area	AB-10, AB-11, and AB-12	4.7×10^{6}	7.05×10^{6}	Sheets 3-11, 3-12, and 3-13
Auxiliary - Recycle Holdup Tank Room	AB-14	3.9×10^{7}	5.85×10^{7}	Sheet 3-15
Auxiliary - Waste Gas Compressor Rooms	AB-15	2.1×10^{7}	3.15×10^{7}	Sheet 3-16
Auxiliary - Recycle Pump Room	AB-16	1.7×10^{7}	2.55×10^{7}	Sheet 3-17
Auxiliary - Floor Elevations 388' & 397'	AB-17 and AB-19	8.8×10^{2}	1.32×10^{3}	Sheets 3-18 and 3-20
Auxiliary - Gas Decay Tank Rooms	AB-18	1.4×10^{8}	2.1×10^{8}	Sheet 3-19
Auxiliary - Floor Elevations 408' and 412'	AB-20, AB-20A, and AB-22	8.8×10^{2}	1.32×10^{3}	Sheets 3-21, 3-22 and 3-24
Auxiliary - Moderating and Letdown Chiller Heat Exchanger Room	AB-21	7.4×10^{6}	1.11×10^{7}	Sheet 3-23
Auxiliary - Spent Resin Storage Tank Rooms	AB-23	1.0×10^{10}	1.5×10^{10}	Sheet 3-25
Auxiliary - Spent Resin Pump Room	AB-24	1.8×10^{9}	2.7×10^{9}	Sheet 3-26
Auxiliary - Waster Evaporator Concentrated Hold-Up Tank and Pump Room	AB-25	2.3×10^{7}	3.45×10^{7}	Sheet 3-27
Auxiliary - Letdown Heat Exchanger Room	AB-26	1.2×10^{7}	1.8×10^{7}	Sheet 3-28
Auxiliary - Residual Heat Removal Exchanger Room	AB-27	3.0×10^{5}	4.5×10^{5}	Sheet 3-29
Auxiliary - Sub-Tube Evaporator Room	AB-28	5.0×10^{7}	7.5×10^{7}	Sheet 3-30
Auxiliary - Recycle and Waste Evaporator Rooms	AB-29 and AB-30	5.3×10^{3}	7.95×10^{3}	Sheets 3-31 and 3-32

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID		Maximum 40 Year Normal TID is from drawing S-021-018
		40 years (rads)	60 years (rads)	
Auxiliary - Sub Tube Waste Evaporator Room	AB-31	5.0×10^7	7.5×10^7	Sheet 3-33
Auxiliary - Letdown Reheat Heat Exchanger Room	AB-32	6.4×10^6	9.6×10^6	Sheet 3-34
Auxiliary - Seal Water Heat Exchanger Room	AB-33	1.2×10^7	1.8×10^7	Sheet 3-35
Auxiliary - Valve Gallery	AB-34	5.3×10^3	7.95×10^3	Sheet 3-36
Auxiliary - Floor Elevation 412'	AB-35	8.8×10^2	1.32×10^3	Sheet 3-37
Auxiliary - Calibration Lab and Stair Tower	AB-36 and AB-37	3.5×10^2	5.25×10^2	Sheets 3-38 and 3-39
Auxiliary - Truck Access Area	AB-38	8.8×10^2	1.32×10^3	Sheet 3-40
Auxiliary - Operator Access to Filter Rooms, Boron Thermal Regenerative Demineralizer Removal Area, and Stair Tower	AB-39	8.8×10^2	1.32×10^3	Sheet 3-41
Auxiliary - Auxiliary Cooler Chiller Condenser Room	AB-39A	8.8×10^2	1.32×10^3	Sheet 3-42
Auxiliary - Operator Access to Boron Thermal Regenerative Demineralizer Removal Area	AB-40	8.8×10^2	1.32×10^3	Sheet 3-43
Auxiliary - Operator Access Corridors and Valve Gallery	AB-41and AB-41A	3.5×10^4	5.25×10^4	Sheets 3-44 and 3-45
Auxiliary - Blowdown Hold-Up Tank and Pump Room	AB-42	3.5×10^4	5.25×10^4	Sheet 3-46
Auxiliary - Blowdown Cation Demineralizer Valve Gallery	AB-42A	3.5×10^4	5.25×10^4	Sheet 3-47
Auxiliary - Nuclear Blowdown System Demineralizer Filter	AB-43	7.2×10^5	1.08×10^6	Sheet 3-48
Auxiliary - Spent Fuel Pool Pit Purification Filter Room	AB-44	3.1×10^8	4.65×10^8	Sheet 3-49
Auxiliary - Recycle Evaporator Feed Filter Room	AB-45	6.0×10^7	9.0×10^7	Sheet 3-50
Auxiliary - Spent Resin Sluice Filter Room	AB-46	6.1×10^7	9.05×10^7	Sheet 3-51
Auxiliary - Recycle Evaporator Concentration Filter Room	AB-47	2.6×10^6	3.9×10^6	Sheet 3-52
Auxiliary - Waste Gas Drain Filter Room	AB-48	5.3×10^6	1.3×10^7	Sheet 3-53
Auxiliary - Reactor Coolant Filter Room	AB-49	3.1×10^8	4.65×10^8	Sheet 3-54
Auxiliary - Seal Water Injection Filter Room	AB-50	9.9×10^7	1.44×10^8	Sheet 3-55
Auxiliary - Blowdown Mixed Bed Demineralizer Room	AB-51	1.9×10^6	2.85×10^6	Sheet 3-56

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID		Maximum 40 Year Normal TID is from drawing S-021-018
		40 years (rads)	60 years (rads)	
Auxiliary - Spent Fuel Cooling Demineralizer Room	AB-52	3.2×10^7	4.8×10^7	Sheet 3-57
Auxiliary - Cation and Waste Monitor Demineralizer Room	AB-53	1.3×10^{10}	1.95×10^{10}	Sheet 3-58
Auxiliary - Cation and Mixed Bed Demineralizer Room	AB-53	1.3×10^{10}	1.95×10^{10}	Sheet 3-58
Auxiliary - Waste Monitor Demineralizer Room	AB-54	4.0×10^7	6.0×10^7	Sheet 3-59
Auxiliary - Recycle Feed and Condensate Demineralizer Room	AB-55	1.0×10^9	1.5×10^9	Sheet 3-60
Auxiliary - Waste Evaporator Condensate Demineralizer Room	AB-56	3.6×10^5	5.4×10^5	Sheet 3-61
Auxiliary - Mixed bed Demineralizer Room	AB-57	1.1×10^{10}	1.65×10^{10}	Sheet 3-62
Auxiliary - Boron Thermal Regeneration Demineralizer Valve Gallery	AB-58	3.5×10^4	5.25×10^4	Sheet 3-63
Auxiliary - Operator Access	AB-59	8.8×10^2	1.32×10^3	Sheet 3-64
Auxiliary - Blowdown Monitor Tank Room	AB-60	3.5×10^4	5.25×10^4	Sheet 3-65
Auxiliary - Boron Thermal Regeneration Demineralizer Room	AB-61	3.2×10^7	4.8×10^7	Sheet 3-66
Auxiliary - Boric Acid Tank Room	AB-62	5.3×10^3	7.95×10^3	Sheet 3-67
Auxiliary - Boric Acid Tank Room	AB-63	1.5×10^6	2.25×10^6	Sheet 3-68
Auxiliary - Concentrated Boric Acid Demineralizer Area	AB-64	1.6×10^9	2.4×10^9	Sheet 3-69
Auxiliary - Charcoal Filter Plenum Room	AB-65	1.7×10^5	2.55×10^5	Sheet 3-70
Auxiliary - Waste Monitor Pump Room	AB-66	5.3×10^3	7.95×10^3	Sheet 3-71
Auxiliary - Waste Monitor Tank Room	AB-67	5.3×10^3	7.95×10^3	Sheet 3-72
Auxiliary - Boron Thermal Regeneration Chiller and Pump Room	AB-68	8.8×10^2	1.32×10^3	Sheet 3-73
Auxiliary - Fuel Handling Building Charcoal Filter Plenum Room	AB-69	3.5×10^4	5.25×10^4	Sheet 3-74

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID		Maximum 40 Year Normal TID is from drawing S-021-018
		40 years (rads)	60 years (rads)	
Auxiliary - Operating Floor Elevation 463' and Stair Tower and Fuel Handling Building Exhaust Fan Room	AB-70, AB-71, and AB-71A	8.8×10^2	1.32×10^3	Sheets 3-75, 3-76, and 3-77
Auxiliary - 480V Switchgear Room	AB-72	8.8×10^2	1.32×10^3	Sheet 3-78
Auxiliary - Volume Control Tank Room	AB-73	2.0×10^8	3.0×10^8	Sheet 3-79
Auxiliary - Valve Gallery on Operating Floor Elevation 463'	AB-74	3.5×10^4	5.25×10^4	Sheet 3-80
Auxiliary - 480V Switchgear Room	AB-75	8.8×10^2	1.32×10^3	Sheet 3-81
Auxiliary - Auxiliary Building Charcoal Exhaust Plenum Fan Room	AB-76	5.3×10^3	7.95×10^3	Sheet 3-82
Auxiliary - Auxiliary Building HEPA Exhaust Fan Room	AB-77	8.8×10^2	1.32×10^3	Sheet 3-83
Auxiliary - Reactor Building Exhaust Plenum Fan Room	AB-78	8.8×10^2	1.32×10^3	Sheet 3-84
Auxiliary - Labyrinth at Entrance to Charging Pump Room	AB-79	3.6×10^6	5.4×10^6	Sheet 3-85
Auxiliary - Operating Floor Elevation 400'	AB-80	5.3×10^3	7.95×10^3	Sheet 3-86
Auxiliary - Operating Floors at Elevations 374', 400', and 405'	AB-81	3.2×10^6	4.8×10^6	Sheet 3-87
Auxiliary - Above Moderating Heat Exchanger Area	AB-82	3.8×10^6	5.7×10^6	Sheet 3-88
Auxiliary - Above Residual Heat Exchanger Area	AB-83	8.8×10^2	1.32×10^3	Sheet 3-89
Auxiliary - Above Residual Heat Exchanger Area	AB-84	3.3×10^6	4.95×10^6	Sheet 3-90
Auxiliary - Above Letdown Heat Exchanger Area	AB-85	2.4×10^6	3.6×10^6	Sheet 3-91
Auxiliary - Above Valve Gallery for the Spent Fuel Resin Storage Tanks	AB-86	1.8×10^9	2.7×10^9	Sheet 3-92
Auxiliary - Above Letdown Reheat Heat Exchanger Room	AB-87	8.8×10^2	1.32×10^3	Sheet 3-93
Auxiliary - Above Operator Access Area to Boron Thermal Regeneration Demineralizer	AB-88	4.9×10^6	7.35×10^6	Sheet 3-94
Auxiliary - Above Valve Gallery to Boron Concentration Measuring Panel	AB-89	4.9×10^6	7.35×10^6	Sheet 3-95
Control - HVAC Rooms	CB-01 and CB-02	3.5×10^2	5.25×10^2	Sheets 3-97 and 3-98

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID 40 years (rads)	Maximum Normal TID 60 years (rads)	Maximum 40 Year Normal TID is from drawing S-021-018
Control - Cable Spreading Rooms	CB-03 and CB-03C	3.5×10^2	5.25×10^2	Sheets 3-99 and 3-102
Control - Control Access Area	CB-03A	3.5×10^2	5.25×10^2	Sheet 3-100
Control - Corridors	CB-03B and CB-04A	3.5×10^2	5.25×10^2	Sheets 3-101 and 3-104
Control Room and Technical Support Center	CB-04	3.5×10^2	5.25×10^2	Sheet 3-103
Control - Post Accident Sampling Panel Room	CB-05	3.5×10^4	5.25×10^4	Sheet 3-105
Control - Elevation 425'	CB-06	8.8×10^2	1.32×10^3	Sheet 3-106
Control - Post Accident Sampling System Sample Cooler Area	CB-07	2.1×10^5	3.15×10^5	Sheet 3-107
Control - Post Accident Sampling System Equipment Area	CB-08	5.3×10^3	7.95×10^3	Sheet 3-108
Auxiliary - Above Valve Gallery to Boron Concentration Measuring Panel	AB-89	4.9×10^6	7.35×10^6	Sheet 3-95
Diesel Generator - All Areas	DG-01, DG-01A, through E and DG-02	3.5×10^2	5.25×10^2	Sheets 3-109 through 3-115
Fuel Handling - Area Adjacent to Auxiliary Building	FH-01	3.5×10^4	5.25×10^4	Sheet 3-116
Fuel Handling - Stair Tower and Corridor at Elevation 412' up to Operating Floor at Elevation 463'	FH-02 and FH-02A	8.8×10^2	1.32×10^3	Sheets 3-117 and 3-118
Fuel Handling - Cask Loading Pit Area	FH-03	6.0×10^7	9.0×10^7	Sheet 3-119
Fuel Handling - Decontamination Pit Collection Tank and Excess Waste Hold-Up Tank Rooms	FH-04	1.0×10^7	1.5×10^7	Sheet 3-120
Fuel Handling - Decontamination Pit Collection Tank and Excess Waste Hold-Up Pump Rooms	FH-05	4.7×10^5	7.05×10^5	Sheet 3-121
Fuel Handling - Demineralizer Rooms	FH-06	4.0×10^7	6.0×10^7	Sheet 3-122
Fuel Handling - Demineralizer Valve Gallery and Cask Decontamination Area	FH-07	3.5×10^4	5.25×10^4	Sheet 3-123
Fuel Handling - New Fuel Storage Area	FH-08	5.3×10^3	7.95×10^3	Sheet 3-124

Table 6.1-2
Virgil C. Summer Nuclear Station
Building Radiation Dose During Normal Operation

Building or Area	Environmental Zone from drawing S-021-018, sheets 4-1 through 4-15	Maximum Normal TID		Maximum 40 Year Normal TID is from drawing S-021-018
		40 years (rads)	60 years (rads)	
Fuel Handling - Labyrinth for Reactor Building Emergency Personnel Escape Hatch	FH-09	5.3×10^3	7.95×10^3	Sheet 3-125
Fuel Handling - Hydrogen Analyzer Area	FH-10	8.8×10^2	1.32×10^3	Sheet 3-126
Intermediate	IB-01 and IB-03 through IB-13	3.5×10^2	5.25×10^2	Sheet 3-127 and Sheets IB-3-131 through 3-165
Intermediate - Above Tendon Access Gallery	IB-02	1.5×10^6	2.25×10^6	Sheet 3-129
West Penetration Access Area (WPAA)	PAA-02 through PAA-06	8.8×10^2	1.32×10^3	Sheets 3-168 through 3-176
West Penetration Access Area (WPAA) - West Penetration Access Area Supply Fan Area	PAA-01	2.4×10^6	3.6×10^6	Sheet 3-166
East Penetration Access Area (EPAA) - Penetration Access Area Supply Fan Area	PAI-01	1.5×10^6	2.25×10^6	Sheet 3-178
East Penetration Access Area (EPAA)	PAI-02	8.8×10^2	1.32×10^3	Sheet 3-180
Reactor - Incore Instrumentation Area	RB-01	2.5×10^9	3.75×10^9	Sheet 3-182
Reactor - Inside Steam Generator Cubicles	RB-02, RB-04, and RB-07	2.5×10^7	3.75×10^7	Sheets 3-183, 3-185, 3-188
Reactor - Outside Steam Generator Cubicles	RB-03 and RB-05	2.0×10^6	3.0×10^6	Sheets 3-184 and 3-186
Reactor - Seal Table Area and Area Adjacent to Equipment Hatch	RB-06 and RB-10	3.5×10^4	5.25×10^4	Sheets 3-187 and 3-191
Reactor - Upper Containment Above Operating Floor	RB-11	2.5×10^7	3.75×10^7	Sheet 3-192
Service Water Pump House (SWPH)	SW-01	3.5×10^2	5.25×10^2	Sheet 3-193
Turbine	TB-01	3.5×10^2	5.25×10^2	Sheet 3-194
Fire Service Pump House (FSPH)	None Designated	Negligible	Negligible	Assumed
Yard Structures – Underground Duct Banks	None Designated	Negligible	Negligible	Assumed
Yard	None Designated - Outside	Negligible	Negligible	Assumed

Table 6.1-3
Virgil C. Summer Nuclear Station
Chemical Analysis of Raw Water

Chemical Parameter	Groundwater [Reference 15]	Rainwater [Reference 15]	Service Water Pond [Reference 15]	Monticello Reservoir [Reference 15]
pH	4.8 - 5.3 [1]	4.8 [2]	6.8	7.2
Chlorides	< 10 ppm	< 10 ppm	< 15 ppm	< 15 ppm
Sulfates	< 10 ppm	< 10 ppm	< 10 ppm	< 10 ppm

1 Groundwater samples contain a range of pH from 4.80 - 5.32 (considered lightly acidic) which can result in a mild acid attack. EPRI Report TR-103842 [Reference 45] states that a dense concrete with low permeability and a low water-to-cement ratio may provide an acceptable degree of protection against mild acid attack. ANSI/ACI 349 defines a dense concrete with a maximum water-cement ratio of 0.50, while ACI 350R defines a dense, chemical resistant concrete with a maximum water-cement ratio of 0.45. Since the VCSNS Category I Structures use concrete which has a maximum water-cement ratio of 0.44 to 0.48, the concrete is considered to be dense and therefore provides an acceptable degree of protection against the mild acid groundwater environment.

2 The Rainwater sample contains a pH of 4.8, which is considered mildly acidic; however, rainwater results in exposure for only intermittent periods of time and therefore its aggressiveness is considered non-significant. In addition, VCSNS Category I Structures use quality dense concrete (maximum water-cement ratio of 0.44 to 0.48) which provides an acceptable degree of protection against the intermittent mild acidic rainwater environment for external concrete.

6.2 STEEL COMPONENTS IN AIR ENVIRONMENT

The design of VCSNS structural steel components complies with the provisions of AISC, "Manual of Steel Construction" [Reference 12]. Structural steel material conforms to ASTM A36-70a, ASTM A572 Grade 50, or ASTM A588 for shapes and plate. High strength structural bolt material conforms to ASTM A325 and A490. Structural grade bolt material that is not high strength conforms to ASTM A307 Grade A. Stainless steel plate conforms to ASTM A240 Grade 304. Sheet metal material used in wall blow-off panels conforms to ASTM A570 [Reference 16]. Carbon steel liner plate conforms to ASME SA516, Grade 60; stainless steel liner plate conforms to ASME SA240, Type 304; and tendon wires conform to ASTM A421 [Reference 12].

The three various air environments for structures include the following:

- Internal environment of the Reactor Building
- Sheltered environments such as the internal environment of the Auxiliary Building
- Yard environment or external environment

Aging effects for steel components located in an air environment were reviewed in VCSNS Technical Report TR00160-010, General Aging Effects Identification for License Renewal (Mechanical) [Reference 17]. Steel components exposed to the Reactor Building environment, sheltered environment, and yard environment are summarized in TR00160-010 attachments.

The conclusions of TR00160-010 provide the identification of the aging effects requiring evaluation for the specific material and environment combination, steel/air. The aging effects requiring evaluation for steel components in these three environments for structural components are:

- Loss of material due to galvanic corrosion in wetted locations when in contact with a metal higher in the galvanic series
- Loss of material due to general corrosion
- Loss of material due to boric acid corrosion (in Reactor Building, Auxiliary Building, and Intermediate Building only)
- Loss of material due to Microbiologically Induced Corrosion (MIC) – VCSNS plant specific
- Cracking and loss of preload due to stress corrosion
- Change of material properties (no aging effects has been identified for structural components)

Galvanic Corrosion

Loss of material due to galvanic corrosion is a concern for carbon and low-alloy steel when coupled with a material higher in the galvanic series. Galvanic corrosion occurs when materials with different electrochemical potentials are in contact in the presence of a corrosive environment. The material with the lower potential (higher on the galvanic series) is the anode and it sacrifices to the cathode. For example, carbon and low-alloy steels have lower potentials than stainless steels and would be preferentially attacked in a galvanic couple. Galvanic corrosion does not occur when the metals are completely dry since there is no electrolyte to electrically couple the two materials [Reference 18, BAW-2270, Appendix E, Section 3.1.2]. The effects of galvanic corrosion are precluded by proper design and the fact that components are not located in a wetted location. Components may become wet due to condensation on external surfaces in systems with normal operating temperatures below the ambient temperature. Moist air or gas in contact with dry surfaces will not result in the occurrence of galvanic corrosion. Structural components are dry surfaces and do not contain cool fluids that would result in condensation. Where dissimilar metals are used, appropriate welding techniques have been implemented. Therefore, loss of material due to galvanic corrosion is not an aging effect requiring management for structural steel components within the sheltered environments, the Reactor Buildings, and the external environment.

General Corrosion
Loss of material due to general corrosion could occur where steel components are located in moist, humid environments. All areas of the sheltered, Reactor Building, and external environments are assumed to be moist, humid environments, except for the Control Room. If left unmanaged, loss of material due to general corrosion could result in a loss of the component intended function. Therefore, loss of material due to general corrosion is an aging effect requiring management during the period of extended operation for most steel components in a sheltered environment, Reactor Building, and external environment. Exceptions for electrical panels and cabinets and cable tray are discussed below.

Metal housing systems, such as electrical panels, cabinets, etc., constructed of factory baked painted steel or galvanized sheet metal do not have a tendency to age with time [Reference 19]. Industry operating experience with metal housing systems indicates that they have performed without failure from corrosion [References 20 and 21]. Therefore, loss of material due to general corrosion is not an aging effect requiring management for electrical panels, enclosures and control boards. Likewise, the control room ceiling is located in a sheltered controlled environment which is not moist nor humid. Therefore, loss of material due to general corrosion is not an aging effect requiring management for the control room ceiling.

Cable tray is constructed of painted or galvanized sheet metal similar to metal housing; therefore, cable tray would age similarly to the metal housings. Industry operating experience does not identify any aging effects for cable tray systems [Reference 22]. Deficiencies that were identified were event driven or design/installation deficiencies. Therefore, loss of material due to general corrosion is not an aging effect requiring management for cable trays.

Boric Acid Corrosion
Boric acid corrosion (also known as boric acid wastage) has been identified as an aging mechanism for the nuclear power industry. Leaks from borated water systems contain boric acid that comes in contact with the external surfaces of other components. As the borated water evaporates from the component external surface, the boric acid is concentrated resulting in loss of material due to boric acid corrosion [Reference 23]. Structural components which may be exposed to boric acid systems are in the Reactor Building, Auxiliary Building, and Intermediate Building; therefore, loss of material due to boric acid corrosion is an aging effect requiring management for structural steel components within the Reactor Building, Auxiliary Building, and Intermediate Building.

Microbiologically Induced Corrosion (MIC)
Microbiologically Induced Corrosion (MIC) is corrosive attack accelerated by the influence of microbiological activity. Microbiological organisms disrupt the protective oxide layer and produce corrosive substances and deposit solids that accelerate the electrolytic reactions of corrosive attack, generally in the form of pitting or crevice corrosion. This aging mechanism is facilitated by stagnant conditions, fouling, crevices, contact with untreated water from a natural source, and/or contact with contaminated soils. The microbes that cause MIC are generally not assumed to be airborne contaminants. MIC, therefore, is only a potential problem where contamination from untreated water or soil may have introduced the bacteria [Reference 18, BAW-2270, Appendix D, Section 3.1.6]. For external surfaces, any wetted areas (with the exception of exposure only to humidity) should be considered potential MIC susceptible locations [Reference 18, BAW-2270, Appendix E, Section 3.1.6].

MIC has been experienced at VCSNS and some minor loss of material effects have been identified. Loss of material due to Microbiologically Induced Corrosion would required aging management and is treated as plant specific. MIC is plausible only for carbon steel and stainless steel where fluids interact with them. There have been and continue to be several instances of groundwater intrusion into buildings below the groundwater table elevation [Reference 24 and CER 98-1047]. The nominal groundwater elevation at VCSNS is 425' [Reference 12]. Most of this water is drained away by design; however, experience shows external MIC presence at the interior surfaces of external walls, where pipes pass through (RHR and Spray system isolation valve chamber guard pipes), and at construction joints between buildings. Therefore, loss of material due to MIC is an aging effect requiring management for steel components below the 425' elevation.

Stress Corrosion Cracking
Stress corrosion cracking (SCC) of steel occurs under tensile stresses either applied (external) or residual (internal) in the presence of a corrosive environment. Three parameters are required for stress corrosion cracking to occur (1) a corrosive environment, (2) a susceptible material and (3) tensile stresses. SCC is a phenomenon that occurs in sensitized stainless steels, but becomes significant only if tensile stress and a corrosive environment exist. Corrosive environments containing sodium hydroxide, seawater, nitrate solutions, sulfuric acids or aggressive groundwater (chlorides > 500 ppm, sulfates > 1500 ppm) are not present at VCSNS. The internal environments of the VCSNS structures do not contain aggressive chemicals under normal operating conditions. Therefore, the conditions necessary for SCC to occur do not exist for the steel components and with the exception of high strength bolting which is discussed below, stress corrosion cracking is not an applicable aging effect.

SCC on High Strength Bolting
Industry experience has shown that high strength bolting (i.e., those with yield strength greater than 150 ksi) installed in Class 1 component supports could be susceptible to SCC in humid environments like the Reactor Building [Reference 25]. Refer to Section 6.8 for applicability of SCC to Class 1 components supports at VCSNS.

SCC on Dissimilar Metal Welds
In bellows assemblies, stress corrosion cracking may cause aging effects, particularly if the material is not shielded from a corrosive environment due to dissimilar metal welds [Reference 5, GALL II.A3.1-d]. VCSNS Fuel Transfer Tube Material is SA-240 Type 304 Stainless Steel and is shown on Westinghouse Drawing 1MS-05-094. Bellows assembly material is SA-240 Type 304 Stainless Steel and is shown on vendor (i.e., Pathway Bellows) drawings 1MS-06 series. Table 6.2-1 below lists hot penetrations (with design temperature > 150°F) [Reference 67] and the materials for bellow sleeves and process pipe at VCSNS Penetrations.

Table 6.2-1
Virgil C. Summer Nuclear Station
Containment Hot Penetrations

PENE. No.	SYSTEM	DRAWING	MATERIAL (Process pipe to bellow sleeve)
XRP-202	Main Steam – Loop C	1MS-06-077	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-207	Main Steam – Loop B	1MS-06-036	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-428	Main Steam – Loop A	1MS-06-036	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-219	Steam Generator Blowdown – Loop C	1MS-06-066	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-224	Steam Generator Blowdown – Loop B	1MS-06-066	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-326	Steam Generator Blowdown – Loop A	1MS-06-066	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-203	Main Feedwater – Loop C	1MS-06-061	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-206	Main Feedwater – Loop B	1MS-06-061	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-306	Main Feedwater – Loop A	1MS-06-061	SA-106 Grade B Pipe to SA-516 Grade 70

Table 6.2-1
Virgil C. Summer Nuclear Station
Containment Hot Penetrations

PENE. No.	SYSTEM	DRAWING	MATERIAL (Process pipe to bellow sleeve)
XRP-423	Waste Processing	1MS-06-035	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-418	Waste Processing	1MS-06-042	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-422	Reactor Coolant – Pressurizer Relief Tank	1MS-06-045	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-420	Reactor Coolant – Pressurizer Relief Tank	1MS-06-044	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-303	Reactor Coolant – Reactor Building Spray Train B	1MS-06-064	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-401	Reactor Coolant – Reactor Building Spray Train A	1MS-06-064	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-226	Reactor Coolant – Residual Heat Removal (RHR)	1MS-06-062	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-316	Reactor Coolant – Residual Heat Removal (RHR)	1MS-06-062	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-227	Low Pressure Safety Injection	1MS-06-065	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-322	Low Pressure Safety Injection	1MS-06-065	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-325	Low Pressure Safety Injection	1MS-06-063	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-222	High Pressure Safety Injection	1MS-06-068	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-412	High Pressure Safety Injection	1MS-06-068	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-415	High Pressure Safety Injection	1MS-06-068	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-426	Safety Injection	1MS-06-068	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-409	Chemical & Volume Control – Normal Charging	1MS-06-073	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-221	Chemical & Volume Control – Seal H_2O	1MS-06-067	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-229	Chemical & Volume Control – Seal H_2O	1MS-06-067	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-408	Chemical & Volume Control – Seal H_2O	1MS-06-067	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-318	Chemical & Volume Control – Normal Letdown	1MS-06-069	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-204	Component Cooling Water – RC Pump Supply	1MS-06-070	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-320	Safety Injection – N_2 Supply to Accumulator	1MS-06-032	SA-106 Grade B Pipe to SA-516 Grade 70
XRP-323	Station Sampling – N_2 Supply to Accumulator	1MS-06-048	SA-213 Type 316 Tube to SA-240 Type 316
XRP-406B	Reactor Coolant – Pressure Sensing Line	1MS-06-050	SA-312 Type 304 Pipe to SA-240 Type 304
XRP-225	Station Sampling – Loop B Steam Generator	1MS-06-047	SA-213 Type 316 Tube to SA-240 Type 316
XRP-220	Station Sampling – Loop C Steam Generator	1MS-06-047	SA-213 Type 316 Tube to SA-240 Type 316
XRP-411	Station Sampling – Loop A Steam Generator	1MS-06-047	SA-213 Type 316 Tube to SA-240 Type 316
XRP-314	Station Sampling – Reactor Coolant Loop B	1MS-06-047	SA-213 Type 316 Tube to SA-240 Type 316
XRP-223	Station Sampling – Reactor Coolant Loop C	1MS-06-047	SA-213 Type 316 Tube to SA-240 Type 316
XRP-405	Station Sampling – from Pressurizer	1MS-06-049	SA-213 Type 316 Tube to SA-240 Type 316
XRP-321	Safety Injection – Accumulator Test	1MS-06-033	SA-312 Type 304 Pipe to SA-240 Type 304

Since all material types are similar (SA), crack initiation and growth (due to SCC for dissimilar metal welds) is not an aging effect requiring management.

SCC on Bellows
Industry experience on bellows, IN 92-20, describes an instance of containment bellows cracking, resulting in loss of leak tightness [Reference 5, GALL Item II.A3.1-d].

Industry information on bellow performance, NUREG/CR-6154 [Reference 68] documents a test program to determine the leak-tight capacity of containment penetration bellows. Several different bellows geometries, representative of actual containment bellows, were subjected to extreme deflections along with pressure and temperature loads. The bellows geometries and loading conditions are described along with the testing apparatus and procedures. A total of nineteen bellows were tested. Thirteen bellows were tested in "like-new" condition and results are reported in NUREG/CR-6154 Volume 1. The tests showed that bellows in "like-new" condition are capable of withstanding relatively large deformations, up to, or near, the point of full compression or elongation, before developing leakage. Six bellows were tested in a corroded condition and, depending on the amount of corrosion, did not perform as well. The corroded bellows test program and results are presented in NUREG/CR-6154 Volume 2.

At VCSNS, there are basically 4 types of penetration: spares, cold penetrations, hot penetrations, and specialty penetrations.

Spare penetrations are sealed by welding pipe caps to each end of the penetration sleeve. Since no resilient or flexible seals are used, these penetrations do not require Type B leakage tests.

Cold penetrations are sealed by a flat plate, of 1/2 inch or greater thickness, welded to both the sleeve and the process pipe at each end of the penetration sleeve. Since no resilient or flexible seals are used, these penetrations do not require Type B leakage tests.

Hot penetrations are sealed on the inside of containment by a flat plate in a manner similar to cold penetrations. On the outside, they are sealed by a single bellows, one end of which is attached to the penetration sleeve and the other to the process pipe. Since the containment barrier does not utilize a resilient or flexible seal, these penetrations do not require Type B leakage tests. Bellows for the Main Steam and Main Feedwater Lines are shown on Pathway Bellows Inc. drawing 1MS-06-077 and 1MS-06-061. Bellows are fabricated from SA-240 Type 304 Stainless Steel as specified in Specification SP-606 [Reference 69]. VCSNS Main Steam, Main Feedwater, and hot penetrations bellows are not tested under the Appendix J Type B LLRT Program because the "hot" penetration designs do not incorporate resilient seals, gaskets, sealant compounds, or flexible seal assemblies on the inboard side of the Reactor Building. These bellows do not perform a pressure boundary function but they do provide structural and/or functional support to safety related equipment (i.e., thermal and accident movement of the process pipe).

The specialty penetrations are No. 107, the fuel transfer tube; and No. 309, which is used to house an abandoned radiation detector and check source assembly originally assigned to radiation monitor RM-G7. Penetrations Nos. 327, 328, 329, and 425 are used for the recirculation sump line penetrations for RHR and Spray systems.

Fuel Transfer Tube - The fuel transfer tube is sealed inside containment by a flat plate welded to both the penetration sleeve and to the fuel transfer tube. The fuel transfer tube closure design provides a double barrier, and therefore, constitutes the containment boundary and is tested. This penetration has a bellows seal in the refueling cavity, which retains water within the cavity and is not a containment isolation seal. Similarly, a bellows seal is used in the transfer canal to retain water. Since no resilient or flexible seal is used for containment, these bellow seals do not require a Type B leakage test.

Recirculation Sump Line Penetrations - The lines from the recirculation sumps are sealed by a 1/2 inch thick flat plate welded to both the penetration sleeve and the process pipe. Since no resilient or flexible seal is used for containment isolation, no Type B leakage test is required. The outer end of each sleeve is attached to sump isolation valve containers which are leakage tested. The pipe to containment liner seal is leak tested with the container.

For the reasons listed above, no piping penetrations are required to be subjected to Type B leakage tests. [Reference 12, Section 6.2.6.2.1]

The internal environment of the West Penetration Access Area and East Penetration Access Area, where the Main Steam Lines and Main Feedwater Lines are located, does not contain aggressive chemicals under normal operating conditions. Stainless steel bellows are very compliant therefore sustained high tensile stress does not exist. Since a corrosive environment and a high level of sustained tensile stress do not exist for the Main Steam and Main Feedwater bellows, SCC on penetration bellows is not an aging effect requiring management.

Boric acid corrosion (also known as boric acid wastage) has been identified as an aging mechanism. Leaks from borated water systems contain boric acid that comes in contact with the external surfaces of other components. As the borated water evaporates from the component external surface, the boric acid is concentrated, resulting in loss of material due to boric acid corrosion.

Penetration bellows located at the Reactor Building external side of containment wall are protected from boric acid corrosion by a layer of insulation and carbon steel cover plates (drawings 1MS-06-077 and 1MS-06-061). Therefore boric acid corrosion is not plausible for these penetration bellows since they are protected and does not require aging management.

IGSCC
Intergranular stress corrosion cracking attack is similar to stress corrosion cracking except that a tensile stress is not required for it to occur. The ambient environment found in the Reactor Building does not contain contaminants in sufficient quantities that could be concentrated in wetted locations to provide the corrosive environment necessary for stress corrosion cracking to occur; therefore, cracking due to stress corrosion or intergranular attack has not been identified as an aging effect requiring system specific evaluation for stainless steel or carbon steel in the Reactor Building. [Reference 18, BAW-2270, App. D Section 3.2 and Figure 1].

Cracking, Loss of Material, and Lost of Preload on Tendons
As a result of the review of industry data, NRC generic communications and the GALL report (GALL Item II.A1.3), loss of material and cracking were identified as applicable aging effects for the containment tendons and anchorage. Cracking and loss of material of the tendon anchorage could ultimately lead to tendon failure. Loss of preload is also an applicable aging effect that could ultimately lead to failure of the tendon to perform its intended function. Section 8 (TLAA) provides additional details on the VCSNS Reactor Building tendons and industry experiences. The Tendon Surveillance Program is credited with managing cracking, loss of material, and loss of preload. The program is documented in Section 7.0 of this report and is evaluated to demonstrate that it adequately manages cracking, loss of material, and loss of preload of the post-tensioning system components for the period of extended operation.

6.2.1 Summary of Aging Effects for Steel in Air Environments

The previous sections discuss various aging effects and the applicability within the bounds of the specific material and environments for steel structures and components. Aging mechanisms which are deemed applicable under these conditions and the associated aging effects requiring management have been identified. A summary table of aging effects and the associated aging mechanisms for the steel components is provided in Attachment I, Table A-1.

Boric Acid Corrosion
Loss of material due to boric acid corrosion is an aging effect requiring management for steel components in the Auxiliary, Intermediate, and Reactor Buildings. Loss of material could result in the steel components being unable to perform their intended function(s). The Boric Acid Corrosion Surveillance program manages loss of material due to boric acid corrosion. This program was evaluated to demonstrate that it adequately manages loss of material for the steel components for the period of extended operation. The aging management review for each component group is summarized in Attachment II for each structure. Details of this program are contained in Section 7.0 of this report.

General Corrosion
Loss of material due to general corrosion is an aging effect requiring management for the extended period of operation for steel components in an air environment. Loss of material could result in the steel components being unable to perform their intended function(s). Loss of material is managed by the following combination of programs: Inservice Inspection Plan – IWF, Battery Rack Inspection, Material Handling System Inspection Program, Containment ISI Program – IWE/IWL, Fire Protection Program, and Maintenance Rule Structures Program. These programs were evaluated to demonstrate that they will adequately manage loss of material for the steel components for the period of extended operation. The aging management review for each component group is summarized in Attachment II for each structure. These programs are contained in Section 7.0 of this report.

Microbiologically Induced Corrosion (MIC)
Loss of material due to MIC is an aging effect requiring management for the extended period of operation for steel components in an air environment located below the 425' elevation (groundwater table elevation). Loss of material could result in the steel components being unable to perform their intended function(s). Loss of material is managed by the following combination of programs: Containment ISI Program – IWE/IWL and Maintenance Rule Structures Program. These programs were evaluated to demonstrate that they will adequately manage loss of material for the steel components for the period of extended operation. The aging management review for each component group is summarized in Attachment II for each structure. These programs are contained in Section 7.0 of this report.

Stress Corrosion
Cracking due to stress corrosion is an applicable aging effect for the ASTM A490 high strength bolting used in the Class 1 component supports, refer to Section 6.8 for more details. The ASME Section XI examination requirements for Class 1 bolting detailed in the ASME Section XI ISI Program – IWF manages loss of preload and cracking due to stress corrosion for bolting in the Class 1 portions of the reactor coolant system that are exposed to the Reactor Building environment. This program was evaluated to demonstrate that it will adequately manage loss of material for the steel components for the period of extended operation. The aging management review for each component group is summarized in Attachment II for each structure. This program is contained in Section 7.0 of this report.

6.3 STEEL COMPONENTS IN FLUID ENVIRONMENT

Steel components are exposed to fluid environments in the Fuel Handling Building structure and the Service Water structures (Service Water Pumphouse, Service Water Intake Structure, Service Water Discharge Structure). The fluid environment within the Fuel Handling Building structure is the borated water environment of the spent fuel pool. The fluid environment of the Service Water structures is the raw water environment of the Service Water Pond.

Aging effects for steel components located in fluid environments were reviewed in VCSNS Technical Reports TR00160-012, Mechanical Component Aging Management Review for License Renewal (Borated Water Systems) and TR00160-014, Mechanical Component Aging Management Review for License Renewal (Raw Water Systems) [References 26 and 27]. Steel components exposed to the borated water environment are discussed in TR00160-012 attachment section. Steel components exposed to the raw water environment are discussed in TR00160-014 attachment section.

The aging effects are further evaluated for the specific environments associated with the borated water of the spent fuel pool and raw water in the following sections.

6.3.1 Borated Water Environment

Components exposed to a borated water environment are located in the Fuel Handling Building structure within the spent fuel pool. The Fuel Handling Building structure components in fluid environments are the spent fuel pool liner, the spent fuel storage racks, structural steel and plates in the spent fuel pool, and the fuel transfer tube which spans between the spent fuel pool and the Reactor Building. These components are constructed of stainless steel. The components in the borated water environment were previously identified and documented in VCSNS Technical Report TR00170-002, Structures Screening for License Renewal [Reference 8].

The conclusions of VCSNS Technical Report TR00160-012, Mechanical Component Aging Management Review for License Renewal (Borated Water Systems) provide the identification of the aging effects requiring evaluation for the specific material and environment combination, stainless steel/borated water. The aging effects requiring evaluation for stainless steel components in a borated water environment are:

- Loss of material due to crevice corrosion in locations where the dissolved oxygen exceeds 100 ppb and chlorides exceed 150 ppb.
- Loss of material due to pitting corrosion in stagnant or low flow locations where dissolved oxygen exceeds 100 ppb and halogens exceed 150 ppb.

- Cracking due to stress corrosion cracking in locations where halogens exceed 150 ppb.
- Cracking due to stress corrosion cracking/intergranular attack in locations where dissolved oxygen exceeds 100 ppb, and temperatures are continually above 200°F with halogens below 150 ppb.
- Cracking due to intergranular attack in locations where halogens exceed 150 ppb and temperatures are greater than 200°F.
- Reduction of fracture toughness due to thermal aging where temperatures continuously exceed 482°F.
- Loss of material due to crevice and pitting corrosion when subjected to alternate wetting and drying that may concentrate contaminants.
- Cracking due to stress corrosion when subjected to alternate wetting and drying that may concentrate contaminants.

Borated Water Environment
The borated water environment for structural components is associated with the spent fuel pool. The spent fuel pool is designed with a clean-up system that removes corrosion products, fission products, and other debris. Water quality is monitored and controlled on a regular basis by checking the chemical concentrations (e.g., chloride, fluoride, boron), pH, and other parameters. The Chemistry Control Program maintains and controls the water quality. For the purposes of identifying aging effects, parameters such as chemical concentrations are assumed to exceed those required for the aging effect to be applicable. This assumption is made to ensure that no program, such as the Chemistry Control Program, is implicitly credited for eliminating the possibility of the occurrence of an aging effect. Therefore, loss of material due to crevice corrosion, loss of material due to pitting corrosion, and cracking due to stress corrosion are aging effects requiring management for the stainless steel components in the spent fuel pool.

Normal operating temperature for the spent fuel pool is less than 140°F. Spent fuel pool temperature alarm Hi setpoint is established at 120°F based on safety related Annunciator Response Procedure ARP-001-XCP-609 [Reference 157]. This temperature does not expose the components to temperatures above the limits where cracking due to stress corrosion cracking/intergranular attack, cracking due to intergranular attack, and reduction of fracture toughness would occur. Therefore, cracking due to stress corrosion/intergranular attack, cracking due to intergranular attack, and reduction of fracture toughness are not aging effects requiring management for the stainless steel components within the spent fuel pool.

Finally, components within the spent fuel pool are not subjected to alternate wetting and drying. Alternate wetting and drying occurs along a line where the component is exposed to frequent fluctuations in the water level. Without the alternate wetting and drying, contaminants would not be concentrated on the components. Alternate wetting and drying is not applicable in the spent fuel pool because the water level is maintained and monitored in accordance with VCSNS Operating License Technical Specifications, Section 3/4.9.10. Therefore, loss of material due to crevice and pitting corrosion and cracking due to stress corrosion are not aging effects requiring management. Although the fuel transfer tube is normally dry and then flooded during refueling, the tube is not subjected to frequent fluctuations along one location. Therefore, loss of material due to crevice and pitting corrosion and cracking due to stress corrosion are not aging effects requiring management.

6.3.1.1 Summary of Aging Effects for Steel Components in Borated Water Environments

Aging mechanisms which are deemed applicable under these conditions and the associated aging effects requiring management have been identified. A summary table of aging effects and the associated aging mechanisms for the steel components is provided in Attachment I, Table A-1.

Crevice and Pitting Corrosion, Stress Corrosion
Loss of material due to crevice corrosion, loss of material due to pitting corrosion, and cracking due to stress corrosion are aging effects requiring management for the stainless steel components in the spent fuel pool if proper water chemistry is not maintained. Loss of material and cracking could result in the loss of the intended functions of the components. The Chemistry Control Program manages loss of material and cracking. The results for each component group are summarized in Attachment II. The program was evaluated to demonstrate that it will adequately manage loss of material and cracking of the components for the period of extended operation and is contained in Section 7.0 of this report.

6.3.2 Raw Water Environment

Components exposed to a raw water environment are located in the Service Water Pumphouse, Service Water Intake Structure, and the Service Water Discharge Structure. The Service Water structures steel components in fluid environments are the intake screens and its associated supports which are constructed of carbon steel. The components in the raw water environment were previously identified and documented in VCSNS Technical Report TR00170-002, Structures Screening for License Renewal [Reference 8].

The conclusions of VCSNS Technical Report TR00160-014, Mechanical Component Aging Management Review for License Renewal (Raw Water Systems) provide the identification of the aging effects requiring evaluation for the specific material and environment combination, carbon steel/raw water. The aging effects requiring evaluation for steel components in a raw water environment are:

- Loss of material due to general corrosion for carbon steel.
- Loss of material due to crevice corrosion for carbon steel and stainless.
- Loss of material due to pitting corrosion for carbon steel and stainless.
- Loss of material due to erosion-corrosion for carbon steel.
- Loss of material due to Microbiologically Induced Corrosion (MIC) for carbon steel and stainless.
- Loss of material due to galvanic corrosion for carbon steel when electrolytically coupled to a material higher in the galvanic series.
- Fouling due to macro-organisms and silting for carbon steel and stainless.

General Corrosion
Carbon and low-alloy steels are susceptible to general corrosion in raw water environments. Corrosion products consisting of hydrated oxides of iron (rust) form on exposed, unprotected surfaces of the steel. If left unmanaged, loss of material due to general corrosion would result in loss of the intended function(s) of the components. Therefore, loss of material due to general corrosion is an aging effect requiring management during the period of extended operation of all carbon steel components in the Service Water structures.

Crevice Corrosion
Crevice corrosion occurs in crevices or shielded areas where conditions allow a corrosive environment to develop within the crevice. Intake trash racks, etc. are exposed to stagnant or low flow with oxygenated conditions which allow the corrosive environment to develop. The temperature of the water is at ambient or cooler temperatures. This lower temperature means that the energy required to drive the corrosion to the point where a component function could be challenged does not exist. Therefore, loss of material due to crevice corrosion is not an aging effect requiring management for carbon and low-alloy steel components in raw water.

Pitting Corrosion
Pitting corrosion can be inhibited by maintaining an adequate flow velocity, thus preventing impurities from adhering to the material surface [Reference 28]. Low flow for carbon steel in raw water is defined to be < 3 fps [Reference 29]. The carbon steel and stainless components within the Service Water structures are exposed to stagnant or low flow conditions; therefore, loss of material due to pitting corrosion is an aging effect requiring management.

Erosion-Corrosion
Erosion-Corrosion is a mechanism where fluid flow erodes or dissolves the protective oxide or passive film of a metal, causing increased corrosion and reoxidation. Flow velocities less than 6 fps do not cause erosion-corrosion of carbon and low alloy steels [Reference 30]. The carbon steel components within the Service Water structures are not exposed to flow velocities that exceed 6 fps;

therefore, loss of material due to erosion-corrosion is not an aging effect requiring management.

Microbiologically Induced Corrosion
Microbiologically Induced Corrosion (MIC) occurs under the influence of micro-organisms. MIC occurs in carbon and low-alloy steel particularly under deposits and in crevices. Buildup of slime and bacteria in crevices can lead to MIC, despite high fluid flow rates [Reference 31]. Therefore, loss of material due to MIC is an aging effect requiring management for the carbon and low-alloy steel components in the Service Water structures.

Galvanic Corrosion
Loss of material due to galvanic corrosion is a concern for carbon and low-alloy steels when coupled with a material higher in the galvanic series. Galvanic corrosion is precluded in the design phase of the structure. Where stainless steel is used such as the intake screen, the screens are attached to stainless steel angles with stainless steel bolts. Carbon and low-alloy steel are not in contact with stainless steel. Therefore, loss of material due to galvanic corrosion is not an aging effect requiring management for the carbon and low-alloy steel components in the Service Water structures.

Fouling
While fouling due to macro-organisms and silting was identified as an aging effect requiring evaluation for carbon steel and stainless components in a raw water environment, fouling is associated with mechanical components and is not associated with structural components. Therefore, fouling due to macro-organisms and silting is not an aging effect requiring management for the carbon and stainless steel components in the Service Water structures. The underwater inspection of the Service Water Intake Structure and Service Water Pump House, however, inspect and periodically remove biofouling.

6.3.2.1 Summary of Aging Effects for Steel Components In a Raw Water Environment

Aging mechanisms which are deemed applicable under these conditions and the associated aging effects requiring management have been identified. A summary table of aging effects and the associated aging mechanisms for the steel components is provided in Attachment I, Table A-1.

General Corrosion, Pitting Corrosion, and MIC
Loss of material due to general corrosion, loss of material due to pitting corrosion, and loss of material due to MIC are aging effects requiring management for the carbon steel components in a raw water environment. Loss of material due to pitting corrosion and loss of material due to MIC are aging effects requiring management for the stainless steel components in a raw water environment. Loss of material could result in the loss of the intended functions of

the components located in the Service Water structures. Loss of material is managed by the Underwater Inspection Program (Service Water Intake Structure and Service Water Pump House) and the Maintenance Rule Structures Program. The results for each component group are summarized in Attachment II. These programs were evaluated to demonstrate that they will adequately manage loss of material and cracking of the components for the period of extended operation and are contained in Section 7.0 of this report.

6.4 CONCRETE COMPONENTS IN AIR AND FLUID ENVIRONMENT

The aging effects that could result in loss of the concrete structure and structural component intended functions are [Reference 32]:

- Loss of Material
- Cracking
- Change in Material Properties

Various aging mechanisms can lead to these potential aging effects in the concrete structures and components. This section addresses the aging effects on the concrete structures and components to determine those which require aging management. Aging effects requiring management are those that are:

- Determined to be applicable for the given material and environment, and
- If unmanaged, the aging effect could progress to the point where the structure or component could lose its intended function(s) under any CLB condition during the period of extended operation.

The codes and standards used for the design and fabrication of concrete components, including applicable edition, are given in the VCSNS FSAR Section 3 [Reference 12] and Specification SP-201 [Reference 42]. The concrete design complies with the American Concrete Institute ACI 318-71 and ACI 301-72 [References 33 and 34]. The concrete parameters specified in SP-201 are:

Concrete foundations, walls, slabs, beams, and columns	3000 psi at 28 days, 5000 psi at 28 days or 90 days as specified on drawings
Concrete in cassion construction	3000 psi at 28 days
Cement	ASTM C-150 Type I or Type II
Air entraining admixtures	ASTM C-260
Water reducing densifier	ASTM C-494
Water	ASTM D-512 for chloride ion and ASTM D-516 for sulfate ion
Aggregates (fine)	ASTM C-33 and ASTM C-128
Aggregates (coarse)	ASTM C-33 and ASTM C-127
Calcium chloride or admixtures containing calcium chloride	Not used
Reinforcing Steel	ASTM A615

The following sections provide an assessment of each of the three aging effects. A review of industry experience related to concrete structures and components is also provided to validate the aging effects requiring management. The environments requiring evaluation at VCSNS are presented in Tables 6.1-1, 6.1-2, and 6.1-3. Each of the aging effects above are evaluated for these given environments in the sections that follow.

6.4.1 Loss of Material Aging Effect Assessment

Loss of material is manifested as scaling, spalling, pitting, and erosion in concrete components. Scaling, spalling, pitting, and erosion are described and illustrated in ACI 201.1, Guide for Making a Condition Survey of Concrete in Service [Reference 35].

Aging mechanisms and stressors that can lead to loss of material include:

- Freeze-Thaw
- Abrasion and Cavitation
- Elevated Temperature
- Aggressive Chemicals
- Corrosion of Embedded Steel / Rebar

The applicability of each aging mechanism and stressor is evaluated for the VCSNS concrete structures and components and is discussed in the following sections.

Freeze-Thaw
Repeated cycles of freezing and thawing can alter both the mechanical properties and physical form of the concrete, thus affecting the structural integrity of the component. Surfaces exposed to weather that can become saturated with water and freeze are vulnerable to freeze-thaw degradation [Reference 36]. Freeze-thaw damage starts at the surface and is readily detected by surface inspections. The resistance of the concrete to freeze-thaw is dependent on the amount of entrained air, permeability of the concrete to water penetration, and protection of concrete from freeze-thaw until adequate strength has developed [Reference 37].

For damage to occur by freezing of absorptive coarse aggregates, the aggregate must be saturated. Saturation can only occur when water is available from an outside source. Freeze-thaw damage typically occurs on relatively flat concrete surfaces such as pavement, where water can remain in contact with the concrete.

VCSNS concrete structures and components are designed in accordance with ACI 318-71 [Reference 33] and constructed in accordance with ACI 301-72 [Reference 34] using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete that precludes freeze-thaw. The structure design includes the use of the following [Reference 42]:

- Type I or Type II cement conforming to ASTM C150, "Specifications for Portland Cement".
- Fine and coarse aggregates conforming to ASTM C33, "Specifications for Concrete Aggregates".
- Air-entraining admixtures conforming to ASTM C260, "Specifications for Air-Entraining Admixtures for Concrete".
- Water reducing, retarding, and accelerating admixtures conforming to ASTM C494, "Specifications for Chemical Admixtures for Concrete".
- Pozzolanic admixtures conforming to ASTM C618, "Specifications for Fly Ash and Raw or Calcined Natural Pozzolans for use in Portland Cement Concrete", except that the maximum value for loss on ignition is to be 6.0 percent.

Therefore, based on the design and construction of the VCSNS structures and components, loss of material due to freeze-thaw is not an aging effect requiring management.

The concrete in structures located in the Service Water Pond may become saturated and could be susceptible to freeze-thaw. Therefore, loss of material due to freeze-thaw is an aging effect requiring management for the structures located in the Service Water Pond.

Abrasion and Cavitation
As water moves over a concrete surface, it can carry abrasive materials or it can create a negative pressure (vacuum) that can cause abrasion and cavitation. If significant amounts of concrete are removed by either of these processes, pitting or aggregate exposure occurs due to loss of cement paste. These degradations are readily detected by visual examination in accessible locations.

Loss of material due to abrasion and cavitation is not an aging effect requiring management for the concrete structures and components which are not exposed to continuously flowing water. Below-grade surfaces with exposure to ground water are also not affected since any flow of water is infinitesimal and unable to cause significant abrasion and cavitation effects. Loss of material due to abrasion and cavitation is an aging effect requiring management for concrete structures and components which are exposed to continuously flowing water. Structures located in the Service Water Pond are the only structures which are exposed to flowing water; therefore, the loss of material due to abrasion is an aging effect requiring management for these structures.

Elevated Temperature
During normal plant operation, solar heat load and equipment heat loads contribute to an increase in temperature of the internal environment of the concrete structures. Surface scaling and cracking may result from long term

exposure to high temperatures.

ACI 318 provides a maximum temperature limit of 150°F for liquid, gas, or vapor in embedded piping in structural concrete [Reference 33]. ASME Code Section III, Division 2 and ACI-349 provide limits where exposure to high temperatures could impair the concrete. ASME Code, Section III, Division 2, Subsection CC indicates that aging due to elevated temperature exposure is not significant as long as concrete temperatures do not exceed 150°F, except for local areas surrounding penetrations which are allowed to have increased temperatures not exceeding 200°F [Reference 71]. ACI-349 allows local area temperatures to reach 200°F before special provisions are required [Reference 72].

Temperatures for various locations throughout VCSNS are provided in Table 6.1-1. While most temperatures are well below the 150°F threshold, several local areas within the Reactor Buildings may experience temperatures approaching and exceeding 150°F. Containment ventilation systems control the temperature in the Containment. The Control Rod Drive Mechanism (CRDM) is maintained at a temperature of less than or equal to 170°F [Reference 12]. These temperatures are localized and do not exceed 200°F.

In 1980 during hot functional testing (HFT-1), VCSNS experienced elevated temperature concerns for three areas inside the Reactor Building. Poor air distribution was a major contributor to the RB temperature problem. VCSNS performed design modifications to rearrange air flow in the RB and provided additional ducting to the affected areas. A CRDM cooling water system was also provided during the modification process. Subsequent to the modification, test results were reviewed and the inspector identified no further problems. Inspector Followup Item 80-25-09 was closed [Reference 73].

During RF-12 a hairline crack was discovered in the weld between "A" hot leg pipe and reactor vessel nozzle, as documented in CER 00-1396. A calculation was preformed to evaluate the surrounding concrete thermal effects resulting from the blockage of the penetration area and cavity annulus and the mass release from the "A" hot leg crack within the penetration [Reference 74]. The calculation purpose was to determine the maximum steady state and transient temperature in the reactor cavity concrete. The scenario involves a conservative evaluation of the effects on the primary shield concrete during the 18 month period when the hot leg crack occurred and includes effects of reactor cavity partial blockage and crack mass release. The calculation concluded that the surrounding concrete continues to have adequate compressive strength with favorable margin of safety.

VCSNS concrete structures and components are not exposed to temperatures which exceed the thresholds for degradation; therefore, loss of material due to elevated temperature is not an aging effect requiring management for concrete structures and components.

Aggressive Chemicals
Concrete, being highly alkaline (pH > 12.5), is vulnerable to degradation by strong acids. Acid attack can increase porosity and permeability of concrete, reduce its alkaline nature at the surface of the attack, reduce strength, and render the concrete subject to further deterioration. Portland cement concrete is not acid-resistant, although varying degrees of resistance can be achieved depending on the materials used and the attention to placing, consolidating, and curing. No Portland cement concrete, regardless of its composition, will withstand exposure to highly acidic fluids for long periods.

Below grade, sulfate solutions of sodium, potassium, and magnesium (sometimes found in groundwater) may attack concrete, often in combination with chlorides. Above grade, exposed surfaces of structures located near industrial plants are vulnerable to industrial pollution from sulfur-based acid rain and are subject to deterioration. Sulfate attack produces significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss. Once established, these conditions allow further exposure to aggressive chemicals. Sulfate attack is a particular problem in arid areas, such as the northern great plains area of the United States and prairie provinces of Canada, and in parts of the Western United States [Reference 41]. At inland sites, corrosion due to aggressive chemical chloride attack has not been a problem unless exposed to brackish water, deicing salt, or other salt sources [Reference 41].

VCSNS is not located in areas exposed to sulfate or chloride attack nor located near major industrial plants whose emissions would change the environmental parameters and cause degradation to concrete.

A dense concrete with low permeability may provide an acceptable degree of protection against mild acid attack. Any factor that tends to improve the compressive strength of the concrete will have a beneficial effect on low permeability. Therefore, the better the quality of the constituent material, the less permeable the concrete. Low water-to-cement ratio, smaller aggregate, long curing period, low entrained air, and thorough consolidation all contribute to increased water tightness. Concrete thus constructed has a low permeability and effective protection against sulfate and chloride attack. The use of an appropriate cement type (e.g., ASTM C-150, Type II) and pozzolan (e.g., fly ash) also increases sulfate resistance.

Minimum degradation threshold limits for concrete have been established at pH < 5.5, chlorides > 500 ppm or sulfates > 1,500 ppm [Reference 5, GALL Items II.A.1.1 and III.A1.1]. Continued frequent or cyclic exposure to aggressive chemicals is of particular concern to foundation components which are exposed to groundwater or concrete structures exposed to lake or pond water containing chemicals beyond these threshold limits. Chemical analyses have been conducted for water samples to determine the potential external environmental influence at VCSNS. The results are presented in Table 6.1-3.

The VCSNS concrete structures and components are designed in accordance with ACI 318-71 [Reference 33] and constructed in accordance with ACI 301-72 [Reference 34] using ingredients conforming to ACI and ASTM standards which provide resistance to excessive cracking and thus to aggressive chemical attack. The VCSNS water chemical analyses results confirm that the site groundwater is considered to be non-aggressive [Reference 15]. The concrete at VCSNS is not exposed to aggressive reservoir water or aggressive groundwater. There is no environment in plant indoor air which would be considered aggressive. Therefore, loss of material due to aggressive chemicals is not an aging effect requiring management for VCSNS concrete structures and components.

Corrosion of Embedded Steel / Rebar
Corrosion is an electrochemical process that results in the formation of ferric oxide (rust) from the metallic iron. The corrosion products have a significantly greater volume than the original metal, resulting in tensile stresses and spalling in the surrounding concrete.

In good quality, well-compacted concrete, reinforcing steel with adequate cover should not be susceptible to corrosion because the highly alkaline conditions present within the concrete (pH > 12.5) cause a passive iron oxide film to form on the iron surface thus protecting it from further oxidation [References 11 and 40]. If the pH is lowered (e.g., to 10 or less), corrosion may occur [Reference 41]. The severity of corrosion is influenced by the properties and type of cement and aggregates as well as the concrete moisture content. Aggressive elements such as chlorides can be present in constituent materials of the original concrete mix (i.e., cement, aggregates, admixtures, and water), or they may be introduced environmentally.

The degree to which concrete will provide satisfactory protection for embedded steel/rebar depends in most instances on the quality of the concrete and the depth of concrete cover over the steel. The permeability of the concrete is also a major factor affecting corrosion resistance. Concrete of low permeability contains less water under a given exposure and, hence, is more likely to have lower electrical conductivity and better resistance to corrosion. Such concrete also resists absorption of salts and their penetration into the embedded steel and provides a barrier to oxygen, an essential element of the corrosion process. Low water-to-cement ratios and adequate air entrainment increase resistance to water penetration and thereby provide greater resistance to corrosion.

As described in NUREG-1557, corrosion of exterior above-grade and interior embedded steel is not significant if the steel is not exposed to an aggressive environment (concrete pH < 11.5 or chlorides > 500 ppm). If such steel is exposed to an aggressive environment, corrosion is not significant if the concrete in which the steel is embedded has a low water-to-cement ratio (0.35 – 0.45), adequate air entrainment (3-6%), low permeability, and is designed in accordance with ACI 318-63 or ACI 349-85. Therefore, if these conditions are satisfied, aging management is not required [Reference 5, GALL Items II.A.1.1 and III.A1.1].

The VCSNS concrete structures and components are designed in accordance with ACI 318-71 [Reference 33], using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete and adequate concrete cover over the reinforcing. The combined ingredient chloride content of the admixtures and mixing water did not exceed 250 ppm [Reference 12]. This chloride concentration can be absorbed by the concrete and not result in structural deterioration. The chloride concentrations of the concrete ingredients were continuously monitored to ensure that the allowable chloride content was not exceeded. The design and installation of the concrete structures and components are sufficient to preclude embedded steel/rebar corrosion. Therefore, loss of material due to corrosion of embedded steel / rebar is not an aging effect requiring management for VCSNS concrete structures and components.

6.4.2 Cracking Aging Effect Assessment

Cracking is manifested in concrete components as a complete or incomplete separation of the concrete into two or more parts. ACI 201.1, Guide of Making a Condition Survey of Concrete in Service [Reference 35] provides descriptions and illustrations of cracking. Aging mechanisms and stressors that can lead to concrete cracking are:

- Freeze-thaw
- Reaction with Aggregates
- Shrinkage
- Settlement
- Elevated Temperature
- Fatigue

The applicability of each aging mechanism and stressor is evaluated for the VCSNS concrete structures and components and is discussed in the following sections.

Freeze-Thaw
Cracking or spalling occurs when voids, already filled with water, freeze causing pressure to increase. In extreme cases of freeze-thaw damage, the cover over reinforcing steel is reduced; the reinforcing steel is eventually exposed; and is then subject to accelerated corrosion. Concrete is vulnerable to the expansive

effects of the resulting corrosion products, thereby further weakening the concrete's resistance to attack by aggressive environments.

Cracking due to freeze-thaw is not an aging effect requiring management for concrete components due to the structure and component concrete mix design and construction quality which preclude freeze-thaw (see discussion in Section 6.4.1.1). However, the concrete in structures located in the Service Water Pond may become saturated and could be susceptible to freeze-thaw. Therefore, cracking is an aging effect requiring management for the structures located in the Service Water Pond since they may become saturated and may be susceptible to freeze-thaw. Cracking due to freeze-thaw would lead to loss of material.

Reaction with Aggregates
Certain mineral constituents of all aggregates react with chemical compounds that compose Portland cement, most notably alkalis. Alkalis also may be introduced from admixtures, salt contaminated aggregates, and penetration by sea water or solutions of deicing salt. However, it is only when the expansive reaction products become extensive and cause cracking of concrete that aggregate reactivity is considered a deleterious reaction. Three principal deleterious reactions between aggregates and alkalis have been identified as alkali-aggregate, cement-aggregate, and expansive alkali-carbonate reactions.

Expansive reactions between aggregates containing active silica and alkalis derived from either cement hydration, admixtures, or external environment have caused many concrete structural failures in the past (late 1920's to early 1940's) [Reference 39]. The problem, which is generally confined to certain areas of the country, however, has been significantly reduced in recent years through proper aggregate material selection, use of low alkali cements, and addition of the pozzolanic materials. The alkali-aggregate reaction is therefore not a problem for VCSNS concrete structures and components because they were fabricated after 1960 and petrographic examination techniques were available to identify potentially reactive aggregates.

Highly siliceous aggregate materials from areas in Kansas, Nebraska, and Wyoming have produced concrete deterioration due to reaction with alkalis in cement [Reference 39]. This issue is not a problem for the VCSNS concrete structures and components because the concrete aggregates are from local quarries.

Operating history does not indicate that structural integrity is significantly affected by alkali-aggregate reactions [Reference 37]. The acceptability of the aggregate was based on VCSNS specification SP-201, Structural Concrete [Reference 42]. The concrete structure and component constituents were carefully selected to mitigate aggregate reactions; therefore, cracking due to reaction with aggregates is not an aging effect requiring management for the VCSNS concrete structures and components.

Shrinkage

A workable concrete mix typically contains more water than is needed to offset the effects of hydration. When concrete is exposed to air, large portions of the free water evaporate. As water evaporates, capillary tension develops in the water remaining in the concrete while the concrete dries and shrinks in volume. Should these stresses exceed the tensile strength of the concrete, a crack forms. Initial shrinkage occurs during curing and continues months after placement. Subsequent drying and shrinkage occurs in concrete that is not continuously wet or submerged. According to ACI 209R-92 [Reference 43] 91% of the shrinkage occurs during the first year, 98% in 5 years, and 100% in 20 years.

Based on the information in ACI 209R-92, shrinkage in concrete is not an issue after 20 years. The majority of VCSNS concrete structures and components were constructed more than 20 years ago. Therefore, concrete shrinkage is not an issue for the license renewal period and cracking due to shrinkage is not an aging effect requiring management for the VCSNS concrete structures and components.

VCSNS contains no Nuclear Safety Related (NSR) block walls [Reference 44]. However, in cases where access is required by personnel for maintenance purposes a structural braced knockdown wall is provided. The knockdown wall is constructed in conjunction with reinforced concrete, with more readily removable blocks for personnel access. For the purpose of license renewal, a knockdown wall may contain a section of block wall or be entirely constructed with masonry block wall.

When removable blocks are used for walls or portions of walls, the following are considered in the details of the design:

- Controlling the density of the concrete blocks at a minimum of 125 lb/ft^3 to meet shielding requirements.
- Stagger of block joints to maximize effectiveness of shielding.
- Provision of support frame to blocks to prevent seismic fall-down during an SSE event.
- Design for leak tightness against liquid or gas escape when applicable.

Knockdown walls (including those using concrete blocks) are required to sustain loads and load combinations as for normal walls. They are not however, relied upon to transmit overall shear loads through the building structure [Reference 16]. Knockdown walls are within license renewal scope since they provide intended functions such as radiation shielding, fire barrier, and Non-Safety Related / Safety Related interaction.

Shrinkage and cracking of knockdown block walls is generally small and occurs at block joints in the earlier stages of plant operation. Excessive cracking can reduce the structural strength of the wall and corrosion of embedded or reinforcing steel may occur as a result of ingress of moisture and oxygen via cracks.

Cracking is an aging effect requiring management for masonry block walls that needs to be addressed for the extended period of operation.

Settlement
All structures settle both during and after construction. The amount of settlement depends on the physical properties of the foundation material. These properties range from rock (with little or no settlement likely) to compacted soil (with some settlement expected).

The major VCSNS structures are supported on continuous rock. The seismic Category I structures supported on mats founded on rock are the Reactor, Control, and Auxiliary Buildings [Reference 12 Section 2.5.4.10.2 and Table 2.5-21]. Therefore, cracking due to settlement is not an aging effect requiring management for these structures.

VCSNS structures founded on caissons embedded in rock are the Diesel Generator, Fuel Handling, and Intermediate Buildings [Reference 12 Section 2.5.4.10.4 and Table 2.5-21]. Settlements of the caissons were computed using elastic theory. Caissons with end bearing pressures not exceeding 100 kips per square foot were estimated to settle 1/4 inch or less. Settlements are due to the elastic compression of the underlying rock mass and occur immediately as the load is applied [Reference 12 Section 2.5.4.10.4.2 and Table 2.5-21]. Therefore, cracking due to settlement is not an aging effect requiring management for these structures.

The Service Water Pumphouse, Service Water Intake Structure, and Service Water Discharge Structure are located at the West Embankment of the Service Water Pond. The foundation mats for the Service Water Pumphouse and Service Water Intake Structure are supported on soil. The base slab of the Service Water Discharge Structure is supported on decomposed rock.

During construction of the Service Water Pumphouse and Service Water Intake Structure, these structures settled more than had been originally estimated. A special settlement study was then performed for the Service Water Pumphouse and Service Water Intake Structure. In responses to additional NRC Questions on the Service Water Intake Structure, VCSNS indicated that the Service Water Pumphouse and Intake Structure will be monitored for settlement twice a year during the operating life of the plant, unless a lesser frequency can be shown to be adequate [Reference 12 Section 2.5.4.10.6.2 and Table 2.5-21]. Although no significant settlement changes have been recorded for these two structures for the past 20 years, settlement monitoring is required by Operating License Condition 2.C.5. Therefore, settlement monitoring for the Service Water Pumphouse and Service Water Intake Structure will continue under the requirements of the CLB.

For the buried Service Water piping and electrical duct banks located in the yard, no significant differential settlement is expected since they were intentionally laid and connected to the Service Water Pump House after the major initial settlement during construction had ceased. However, semiannual survey data is currently performed.

Settlement of the Service Water Discharge Structure, which is founded on decomposed rock, was determined to be negligible. Therefore, cracking due to settlement is not an aging effect requiring management for this structure.

The Condensate Storage Tank foundation mat is supported on compacted Zone III fill material (graded crushed stone). The Zone III fill materials were compacted to at least 85% of relative density as determined by ASTM Test Designation D 2049-69 [Reference 12 Section 2.5.4.10.3 and Table 2.5-21]. Total settlement of the Condensate Storage Tank was calculated prior to tank erection to be < 1/2 inch. Settlement within the Zone III fill material was estimated to be essentially nil and was neglected. The Zone III fill material is placed on a naturally occurring saprolite soil. Because the settlement of the saprolites will occur almost instantaneously upon the application of the loads, post-construction (after the tank is water tested) settlements will be negligible. The calculated settlement value was later verified after the tank was completed and filled. The measured settlement of the pad and tank since its construction has been approximately 1/2 inch. No additional settlement is expected. Therefore, cracking due to settlement is not an aging effect requiring management for the Condensate Storage Tank foundation.

The foundation mat for the Turbine Building is mostly comprised of a reinforced concrete mat supported by Zone III fill material. Since the Condensate Storage Tank has experienced minimal settlement as computed and verified, the Turbine Building was likewise expected to exhibit a similar degree of settlement. The total and differential settlement was well within tolerable limits for this type of facility. Therefore, cracking due to settlement is not an aging effect requiring management for the Turbine Building.

Elevated Temperature

VCSNS concrete structures and components are not exposed to temperatures which exceed the thresholds for degradation; therefore, cracking due to elevated temperature is not an aging effect requiring management for concrete components (see discussion in Section 6.4.1.3).

Fatigue

Fatigue is a progressive degradation problem for materials subjected to cyclic application of loads that are less than the maximum allowable static loads. For concrete components, the effects of fatigue loading initiate as internal microcracking within hardened concrete paste and at reinforcing steel boundaries. If stress repetitions are great enough, microcracks may extend to the external surface, causing possible fracture of the cover concrete. This may not cause failure of the structure, but may promote further debilitation of the exposed reinforcing steel or internal crack propagation. There are two types of fatigue that exist for structural components: (1) low-cycle fatigue, and (2) high-cycle fatigue. Low-cycle fatigue is low frequency ($<1 \times 10^5$ for steel) of high-level repeated loads due to plant start-up thermal cycles and abnormal events such as SSE loads. High-cycle fatigue is high frequency of low-level, repeated loads such as equipment vibration. Concrete structures, concrete components and reinforcing steel can be subjected to cyclic loading and therefore, can be subjected to fatigue degradation. However, concrete components have good fatigue strength properties for hundreds or thousands or cycles of below yield load application (high cycle low-level loads) [References 37 and 45]. For components that may be subjected to vibratory or cyclic loading, proper design eliminates or compensates for vibration and cyclic loading. In addition, vibration characteristically leads to cracking in a short period of time, on the order of hours to days of operation. For example, a component with 1 Hertz vibratory load will be subjected to 10^7 cycles in four months of service, so that failure, should it occur, is probable early in life for vibratory stresses above the endurance limit. Because this time period is short when compared to the overall plant operational life, any cracking would be identified and corrected to prevent recurrence long before the loss of its intended function. This type of degradation is limited to a small set of components and is corrected as discovered with inspections of similar locations and configurations to ensure the event is location specific or a one-time event.

VCSNS concrete components are designed in accordance with ACI 215R-74 [Reference 46] and have good low cycle fatigue properties. Although some concrete components are subject to high cycles of low-level repeated load, these components were designed in accordance with ACI, which limits the maximum design stress to less than 50 percent of the static stress of the concrete. The concrete fatigue strength is about 55 percent of its static strength at extremely high cycles ($>10^7$ cycles) of loading. Therefore, cracking due to fatigue is not an aging effect requiring management for concrete components.

6.4.3 Change in Material Properties Assessment

The change in material properties aging effect is manifested in concrete structures and components as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. Aging mechanisms and stressors which can lead to changes in material properties include:

- Leaching of Calcium Hydroxide
- Elevated Temperature
- Aggressive Chemical Attack
- Irradiation Embrittlement

The applicability of each aging mechanisms and stressor is discussed in the following sections.

Leaching of Calcium Hydroxide
Water (either from rain, groundwater, or melting snow) that contains small amounts of calcium ions can readily dissolve calcium compounds in concrete when it passes through cracks, inadequately prepared construction joints, or areas inadequately consolidated during placing. The most readily soluble calcium compound is calcium hydroxide (lime). The aggressiveness or affinity of water to leach calcium hydroxide depends on its dissolved salt content and its temperature. Since leaching occurs when water passes through the concrete, structures that are subject to flowing liquid, ponding, or hydraulic pressure are more susceptible to degradation by leaching than those structures that water merely passes over. When calcium hydroxide is leached away, other cementitious constituents become exposed to chemical decomposition, eventually leaving behind silica and alumina gels with little or no strength. Leaching over a long period of time increases the porosity and permeability of concrete, making it more susceptible to other forms of aggressive attack and reducing the strength of concrete. Leaching also lowers the pH of concrete and threatens the integrity of the exterior protective oxide film of rebar.

Resistance to leaching and efflorescence can be enhanced by using concrete with low permeability. A dense concrete with a suitable cement content that has been well cured is less susceptible to calcium hydroxide loss from percolating water because of its low permeability and low absorption rate. The VCSNS concrete structures and components are designed in accordance with ACI 318-71 [References 33] and constructed in accordance with ACI 301-72 [Reference 34] using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete [Reference 42].

Leaching has been identified for VCSNS in walls exposed to external environments and/or below groundwater table elevation. Therefore, change in material properties due to leaching is an aging effect requiring management for the concrete walls exposed to the external environment and/or below groundwater table elevation for the extended period of operation.

Elevated Temperature
VCSNS concrete structures and components are not exposed to temperatures which exceed the thresholds for degradation; therefore, change in material properties due to elevated temperature is not an aging effect requiring management for concrete components (see discussion in Section 6.4.1.3).

Aggressive Chemical Attack
Change in material properties due to aggressive chemical attack is not an effect requiring management for the VCSNS concrete structures and components based on the design and construction of the concrete components and the absence of an environment containing aggressive chemicals (see discussion in Section 6.4.1.4).

Irradiation Embrittlement
The VCSNS concrete structures and components could be impaired when exposed to excessive neutron or gamma radiation. Fast and slow neutrons can cause aggregate growth, decomposition of water or thermal warming of concrete. Gamma radiation affects the cement paste portion of the concrete, producing heat and causing water migration. As the temperature of concrete increases and free water within the concrete evaporates, the structural characteristics of concrete are degraded. With the water loss, concrete can experience a decrease in its compressive, tensile, and bonding strengths, and in its modulus of elasticity [Reference 37].

With high radiation doses, concrete experiences decreases in compressive and tensile strengths and in the modulus of elasticity [Reference 39]. Concrete does not begin to experience a compressive strength loss until exposure exceeds a neutron fluence of 10^{19} neutrons/cm^2. Reductions in the tensile strength of concrete also occur at 10^{19} neutrons/cm^2, the same exposure levels at which the compressive strength begins to be affected [Reference 37, p. 4-40]. The effect of irradiation on reinforcing steel is to increase the yield strength, decrease the ultimate tensile ductility and increase the ductile to brittle transition temperature [Reference 37, p. 4-53].

Irradiation embrittlement can result in degradation to the concrete and rebar; however, it requires a neutron fluence far exceeding that experienced by any components other than the Class 1 components [Reference 18]. The location in pressurized water reactors where irradiation may approach levels to initiate degradation is the reinforced concrete primary shield wall near the reactor vessel. Generic bounding estimates of neutron fluence (E>1MeV) and total integrated dose at the reactor vessel for 54 effective full power years (EFPY) were calculated for VCSNS [Reference 38]. The fluence calculated at the reactor vessel 3/4 thickness are estimated to be 9.41×10^{18} neutrons/cm^2 for VCSNS. These estimates, calculated at vessel 3/4 thickness, are below the lower fluence limit where concrete degradation would occur; therefore change in material properties due to irradiation embrittlement is not an applicable aging effect for the Reactor Building Internal Structure components located adjacent to the vessel.

For other components located throughout the VCSNS sites, the projected 60 year normal operating dose is listed in Table 6.1-2. The expected normal dose for 60 years was determined by multiplying the current 40 year normal dose by the ratio of 1.5 (60/40). For example, if the normal 40 year dose for a given area is 3×10^4 rads, then the 60 year dose will be 4.5×10^4 rads.

The projected 60 year normal operating doses listed in Table 6.1-2 are well below the lower fluence limit where degradation would occur; therefore, change in material properties due to irradiation embrittlement is not an aging effect requiring management for concrete components.

6.4.4 Industry Experience

In order to validate the set of applicable aging effects and to assure no additional aging effects beyond those discussed herein, a review of industry experience was performed. This review included a survey of Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LERs), NRC generic communications, including in some cases VCSNS specific responses, and NRC contractor research reports (NUREG/CRs). The following are the results of this investigation:

Most instances related to degradation of concrete structures in the United States occurred early in the life of the structures and have been corrected. Causes were primarily related either to improper material selection, construction/design deficiencies or environmental effects [Reference 11]. Examples of some of the problems attributed to these deficiencies include concrete cracking, concrete voids or honeycombing, and concrete compressive strength values that were low relative to design values at a specific concrete age [Reference 39]. In almost all cases, the concrete cracks were considered to be structurally insignificant or easily repaired using techniques such as epoxy injection. The voids and honeycombed areas and low-strength concrete areas were repaired or replaced. The few incidences where the structural integrity of the component was jeopardized were attributed to either design, construction, or human errors, but not to aging. Quality control/quality assurance programs at nuclear power plants generally have been very effective in ensuring that the basic factors related to the production of durable concrete are adequately addressed [Reference 11].

NUREG/CR-4652 [Reference 39] reports that three concrete Reactor Buildings at the Savannah River Plant were inspected after approximately 25 years of operation to determine suitability for an additional 20 to 30 years of operation. Except for some repairable cracking, all structures were found to be in satisfactory condition.

A study by Prasad and Orr [Reference 47] included a visit to Shippingport, Pennsylvania, while the U. S. Department of Energy was decommissioning a nuclear reactor. The prime objective was to gather data on the condition of the structure subjected to a prolonged nuclear environment. The concrete structures examined were primarily below grade and were all found to be in excellent condition. There were a few hairline shrinkage cracks on the interior faces and fine cracks on the top slab which were exposed to the outside atmosphere. The degree of cracking was not significant enough to have prevented the concrete slab from continuing to function satisfactorily.

NUREG-1522 [Reference 48] documents the inspection of six older plants (licensed prior to 1977) by the staff of Civil Engineering Geosciences Branch (ECGB) in the division of Engineering of the Office of Nuclear Reactor Regulation (NRR). During these inspections, instances of concrete cracking and spalling were identified. The Office of NRR also sponsored a survey questionnaire to obtain information on the types and location of distress in concrete structures. Twenty-nine utilities (41 reactor units) responded to the survey. The conclusion was that "deterioration has generally been minor due to the high initial quality of the original construction and the relatively young age of most plants".

The following experience is categorized by component for ease or review with associated components.

Anchorage

Potential factors related to failure or degradation of anchorage systems include design detail errors, installation errors, material defects, shear or shear-tension interaction, slip, and preload relaxation. Aging effects that could impair the ability of an anchorage to meet its performance requirements would be primarily those that result in deterioration of concrete properties, because if a failure did occur, it would most likely initiate in the concrete [Reference 11].

A LER was identified which addressed the use of "deceit bolts" on HVAC supports. A "deceit bolt" is one which has the head cut from the stem and then the head is welded onto its mounting bracket so that it appears to be a complete and properly installed bolt. This is an installation deficiency and not associated with aging.

One incident noted problems with forming and shrinkage of grouting material for anchor bolts, and when tested the material failed to meet strength requirements [Reference 49]. The deficiencies were related to material defects and not associated with aging.

NUREG/CR-3604, Bolting Applications, discussed an investigation of bolting practices specific to the nuclear industry [Reference 50]. The report covered a large spectrum of topics, e.g., bolts embedded in concrete, specifications, inspection of bolting, both at receipt and inservice. The report did not address aging.

In response to Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50-54(f)," VCSNS provided their response in the IPEEE Submittal Report [Reference 51]. As part of the seismic risk assessment, plant walkdowns were performed to verify seismic adequacy of equipment anchorage as well as other items. The review included expansion anchors, cast-in-place bolts, etc. No deficiencies were identified that would challenge the licensing basis for anchorage. Most deficiencies were related to seismic interaction concerns. No deficiencies associated with aging were noted at VCSNS.

Cassions
A search was performed to identify any incidences of degradation of cassions which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

Duct Banks
A search was performed to identify any incidences of degradation of trenches which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

For the electrical duct banks located in the yard, no significant differential settlement is expected since they were intentionally laid and connected to the Service Water Pump House after the major initial settlement at construction had ceased. However, semiannual survey data is currently performed.

Embedments
A search was performed to identify any incidences of degradation of embedments which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

Equipment Foundations/Pads
A search was performed to identify any instances of degradation of equipment foundations/pads which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

Fire Walls
A search was performed to identify any incidences of degradation of fire walls which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found (see also masonry block walls).

Flood Curbs
A search was performed to identify any incidences of degradation of flood curbs which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

Foundations
A search was performed to identify any incidences of degradation of foundations which have been recorded in NPRDS, LERs, or VCSNS OEDB. The following degradations were found.

The NRC issued Information Notice 97-11, *Cement Erosion from Containment Subfoundations at Nuclear Power Plants*, to address the unexpected erosion of the high-alumina cement used in the subfoundation of a plant. The main concern was the adequacy of the eroded porous media to transfer the containment loads to bedrock. VCSNS did not used high-alumina cement; therefore, this degradation concern is not applicable. The NRC also evaluated the concern at other nuclear power plant sites where these conditions could exist. VCSNS was not included in their investigation.

The NRC issued Information Notice 98-26, *Settlement Monitoring and Inspection of Plant Structures Affected by Degradation of Porous Concrete Subfoundations* to address settlement due to the erosion of the subfoundation. VCSNS did not use high-alumina cement; therefore, this degradation concern is not applicable.

During the construction of the Service Water Pumphouse and Service Water Intake Structure, these structures settled more than had been originally estimated. A special settlement study was then performed for the Service Water Pumphouse and Service Water Intake Structure. In responses to additional NRC Questions on the Service Water Intake Structure, VCSNS indicated that the Service Water Pumphouse and Intake Structure will be monitored for settlement twice a year during the operating life of the plant, unless a lesser frequency can be shown to be adequate [Reference 12, Section 2.5.4.10.6.2 and Table 2.5-21]. Although no significant settlement changes have been recorded for these two structures for the past 20 years, settlement monitoring is required by Operating License Condition 2.C.5. Therefore, settlement monitoring for the Service Water Pumphouse and Service Water Intake Structure will continue under the requirements of the CLB.

Hatches
A search was performed to identify any incidences of degradation of hatches which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradations were found.

Masonry Block Walls
A search was performed to identify any incidences of degradation of masonry block walls which have been recorded in NPRDS, LERs, or VCSNS OEDB. The following degradations were found.

IE Bulletin 80-11, *Masonry Wall Design*, addressed the potential for problems with the structural adequacy of hollow, unreinforced block walls. Although age-related degradation was not the primary concern of the Bulletin, it did discuss the potential for cracking due to reactions and thermal effects.

Information Notice (IN) 87-65, *Plant Operation Beyond Analyzed Conditions*, noted that high temperatures in areas can cause degradation and accelerated aging of materials. The IN identified concrete as one of the materials that could be susceptible to accelerated aging at elevated temperatures.

IN 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11*, described deficiencies discovered regarding unanalyzed conditions and faulty assumptions in the qualification of block walls.

IN 92-042, *Fraudulent Bolts in Seismically Designed Walls*, was issued to alert addressees to fraudulent anchor and through-wall bolts found in seismic structures. This was a design and installation issue and was not associated with aging.

Missile Shields
A search was performed to identify any incidences of degradation of missile shields which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

Reinforced Concrete Beams, Columns, Floor Slabs, Roof Slabs, Walls
A search was performed to identify any incidences of degradation of concrete beams, columns, slabs, floors, walls which have been recorded in NPRDS, LERs, or VCSNS OEDB. The following degradations were found:

VCSNS experience has identified concrete leaching in the Reactor Building Tendon Access Gallery [Reference 52, CER 00-0988, and CER 02-0832]. The groundwater intrusion was through identified cracks and joints. Since the groundwater at VCSNS is non-aggressive, the leaching material from the containment sub-foundation is considered structurally insignificant and it was determined it would not result in loss of function. However, due to the continuously wet environment which exists in the Tendon Access Gallery, this area has been identified for augmented inspections during future outages to ensure that the gallery does not degrade to an unacceptable structural condition. Leaching observed on the exterior containment surface is minimal and considered of no structural significance.

Due to the nature of seasonal groundwater table fluctuation and since VCSNS has experienced concrete leaching, change in material properties due to leaching is an aging effect requiring management for the concrete walls exposed to the external environment and/or below groundwater table elevation for the extended period of operation.

VCSNS experience has identified several instances of cracks in concrete. All cracks were evaluated and it was determined that these types of cracks in reinforced concrete structures would not result in loss of function. Specific instances are addressed below.

After start-up of the plant and operation of the diesel generators, damage (cracking) was observed to the fixed end pedestal of exhaust silencer XNA-7B-DG and reported in Non-Conformance Notice (NCN) No. 2172. Subsequent testing and evaluation determined that the damage was caused by higher than design shearing forces applied to the pedestal as a result of stiff insulating material which surrounded the exhaust stack, preventing free thermal expansion of the silencers. Structural Calculation DC0376B-013 provides details of the investigation and subsequent remedial action taken for the pedestal and stack insulation, as well as preventive measures for XNA-7A-DG.

During the 2000 ASME XI IWL inspection for the Reactor Building, many cracks were observed and documented, which is typical in prestressed and reinforced concrete structures. The reportable threshold for crack width was defined as >1mm for general structure and >0.3mm for proximity (within 2') of tendon anchorage baseplates. The inspection data sheets identified some cracks with width exceeding these thresholds; however, the responsible engineer review has determined that all of these reportable conditions marginally exceeded the threshold criteria and are of no structural significance. The responsible engineer inspections also identified all cracks as passive with no signs of active movement or growth. Several areas of "craze cracks" were observed and determined to be of no structural significance [Reference 52].

Sumps

A search was performed to identify any incidences of degradation of concrete sumps which have been recorded in NPRDS, LERs, or VCSNS OEDB. While there were instances associated with clogging of the sump, there were no instances of degradation associated with degradation of the concrete.

Trenches

A search was performed to identify any incidences of degradation of trenches which have been recorded in NPRDS, LERs, or VCSNS OEDB. No instances of degradation were found.

6.4.5 Summary of Aging Effects for Concrete Structures

The previous sections discuss various aging effects and the applicability within the bounds of the specific material and environments for concrete structures and components. Aging mechanisms which are deemed applicable under these conditions and the associated aging effects requiring management have been identified. A summary table of aging effects and the associated aging mechanisms for the concrete components is provided in Attachment I, Table A-2.

Freeze-Thaw
Loss of material and cracking due to freeze-thaw are aging effects requiring management for the external walls of the SWIS and SWPH. Loss of material due to abrasion and cavitation is an aging effect requiring management for concrete exposed to flowing water of the SWIS and SWPH. Loss of material and cracking in the Service Water Intake Structure is managed by the Maintenance Rule Structures Program and the Underwater Inspection Program (SWIS and SWPH).

Cracking of Block Walls
Cracking is an aging effect requiring management for masonry block walls and is managed by the Maintenance Rule Structures Program.

Settlement
Cracking due to settlement has been experienced at the SWIS and SWPH and is managed by the Service Water structures Survey Monitoring Program (SWPH, SWIS, Electrical Duct Banks and SW Intake Line "A").

Leaching
Change in material property due to leaching is an aging effect requiring management in external walls, roofs, and concrete components below site groundwater elevation of all structures. Change in material property due to leaching is managed by the Containment ISI Program – IWE/IWL for the Reactor Building and by the Maintenance Rule Structures Program for all other structures. The programs are discussed in detail in Section 7.0 and are evaluated to demonstrate that they will adequately manage aging for the period of extended operation.

6.5 FIRE BARRIERS

This section pertains to fire barriers that are installed for regulatory compliance. 10 CFR 50.48 requires that each operating nuclear power plant must have a fire protection plan that satisfies General Design Criterion (GDC) 3 of Appendix A to 10 CFR 50. Fire protection features required to satisfy GDC-3 include features to ensure that one train of systems necessary to achieve and maintain shutdown conditions is free of fire damage. As part of fire protection defense in depth, nuclear power plants are divided into separate fire areas by fire-rated walls and fire-rated floor-ceiling assemblies. These fire barriers offer reasonable assurance that a fire will not spread from one plant area to another. Openings in these fire barriers, known as fire barrier penetrations, allow such items as cables, conduits, and pipes to pass from one fire area to another. Fire barrier penetration seals are installed to seal these openings and maintain the fire-resistive integrity of the fire barriers.

This section evaluates the aging effects associated with the fire barrier penetration seals and fire barriers (walls, floors, doors, etc.). Fire barriers (walls, floors, and ceilings) are normally constructed of reinforced concrete or masonry. These materials are also evaluated for aging in Section 6.4 of this report. Some fire barriers are constructed of gypsum board. No aging effects have been identified for gypsum board that require management. Albi fire proofing is used as a coating on steel surfaces of fire barriers. Albi fire proofing is a resilient, cementitious material. No aging effects have been identified for Albi fire proofing. The Albi fire proofing is inspected as part of the fire barrier. The aging of fire doors is addressed with steel components in Section 6.2.

In order to evaluate the aging effects for a component, the materials of construction must be identified along with the environment. Section 6.5.1 provides a description of the different types of fire barrier penetrations seals and their materials of construction. Section 6.5.2 describes the environment to which the fire barriers and penetration seals are exposed. The aging effects review is provided in Section 6.5.3.

6.5.1 Fire Barrier Penetration Seals

The majority of electrical and mechanical penetration seals are silicone foam and some assemblies include fiberboard, fiber blanket and bulk fiber. Mechanical penetration fire barrier seals contain silicone foam. Electrical penetration fire barrier seals contain silicone foam with mineral fiberboard on either side of the opening. Cracks or gaps between mineral fiberboard surfaces are filled with mineral fiber bulk or blankets. No aging effects have been identified for mineral fiberboard that require management. The different types of fire barrier penetration seals are discussed in more detail as follows:

Some fire barrier penetrations are sealed with Dow Corning Sylgard 170. Sylgard 170 is a two part product which when mixed cures to a flexible elastomer with fire resistant properties. Sylgard 170 is a silicone based material. This material has been tested in accordance with the ASTM E-119 procedure for three-hour fire resistance rating.

Flexible Boot Seals, Keene Chase Foster or equivalent, are constructed of fire retardant, radiation resistant, silicone rubber with woven glass fiber reinforcing. They are installed where the seal has fire protection application and where flexibility is needed for the specific component in the penetration. This material has been tested in accordance with the ASTM E-119 procedure for three-hour fire resistance rating and is inspected under the Fire Protection Program and the Maintenance Rule Structures Program.

Silicone Elastomer Material is installed in the seismic gap between the Reactor, Auxiliary, and Intermediate Buildings at VCSNS. Cracking and change in material properties aging effects have been identified for silicone elastomer that require management. The silicone elastomer material is inspected under the Maintenance Rule Structures Program.

Grout is a cementitious filler material, used to seal around the surface of penetration items and the fire barrier. When cured, grout is considered a part of the fire barrier, rather than a penetration seal. Grout is therefore part of the wall or ceiling and is inspected on a regular basis as part of the barrier surface inspection.

Dow Corning's Brand Sealant Adhesive and putty are used to fill some small openings and as an adhesive to tape Kaowool wrap layers together. These materials are synthetic elastomers which have intumescent properties and are UL Listed for their specific applications. Caulk and putty are considered as part of the fire barrier and fire barrier penetration seals and are not evaluated separately.

Fire Wraps are installed in some cable trays and are required for separation of analyzed, redundant safe shutdown functions. There are two types of wraps in use at the VCSNS. Kaowool wrap was used to meet the Appendix A requirements. The enclosure was approved by the NRC and will remain in place to meet Appendix R requirements.

The Kaowool wrap system; as installed on cable trays, conduit, and equipment; is in accordance with ASTM E-119. Most of the Kaowool is coated with either Flamastic 77 or with a Zetex 800 aluminized cloth to preclude mechanical damage to the wrap system. A study was performed to demonstrate that the cable tray supports maintain their integrity even though they are not wrapped. These Kaowool wraps will be maintained with replacement Kaowool. No aging effects have been identified for damming materials.

Damming Material can be board type, loose type, or blanket type (Johns-Manville's Ceraboard, Cerafiber, or Cerablanket). Damming material rated fire resistant may remain after foam has cured. No aging effects have been identified for damming materials.

Fire Dampers are installed at fire barrier walls, either in HVAC ducts or in open ventilation pathways. Fire dampers are UL Listed and perform an active function. Fire damper housing located in fire barrier walls or open ventilation pathways is considered part of the wall and is managed by the Fire Protection Program and Maintenance Rule Structures Program. Fire damper housing located in-line of HVAC ducts is discussed in the mechanical AMR for ventilation systems [Reference 53].

6.5.2 Environment

Fire barrier penetration seals are located throughout the plant in a variety of environments. Located inside plant buildings, the fire barrier penetration seals are exposed to the ambient environment within each location. The internal environments of these structures may be controlled, such as in the Control Room of the Control Building, where the temperature and humidity are relatively mild. Other areas, such as the interior of the Reactor Building, are exposed to high temperatures, humidity, and radiation. These variables play a major role in the potential degradation of the components located within this environment. Thermal and radiation environmental data are included in Section 6.1 of this document.

6.5.3 Aging Effects Evaluation

Fire barrier penetration seals constructed of silicone may be susceptible to aging effects due to shrinkage. Shrinkage may lead to cracking/delamination and separation of the fire barrier material [Reference 5, GALL Section VII.G]. Therefore, cracking/delamination and separation is an aging effect requiring management for silicone fire barrier penetration seals.

Fire barrier penetration seals constructed of elastomers or polymers other than silicone-based materials may be susceptible to aging effects resulting from ultraviolet radiation, thermal exposure, and ionizing radiation. Polymers subjected to ultraviolet radiation, thermal exposure, or ionizing radiation may become brittle over time and lead to cracking [Reference 54]. Most penetration seals are constructed of silicone-based materials. Boot seals are constructed of silicone rubber. Therefore, cracking is an aging effect requiring management for boot seals. The boot seals are inspected as part of the Fire Protection Program.

6.5.4 Industry Experience

Data on failure of fire barrier penetration seals throughout the industry and NRC generic communications were reviewed to determine if there are any aging effects or mechanisms that should be considered for fire barrier and penetration seals. This review included searches of INPO Nuclear Plant Reliability Data System databases, VCSNS specific information, NRC Inspection and Enforcement Information Bulletins, Licensee Event Reports, and Generic Letters.

A utility noted a gap between fire retardant and a masonry block wall (see OEDB number 95-011383). Apparent cause of the gap was due to removal of an adjacent box for maintenance. Portions of the fire retardant adhered to the box and were torn when the box was removed. This deficiency was due to maintenance and not age-related.

A utility noted a gap in a fire barrier due to shrinkage (see OEDB number 90-003318 and Nuclear Network OE 4279). As noted in SECY-96-146, "some shrinkage is normal" and "normal shrinkage does not have a significant impact on the function and capabilities of silicone foam or elastomer as a fire barrier penetration seal material".

Several bulletins, notices, and letters have addressed the performance of Thermo-Lag material (IN 92-55, IN 94-22, IN 92-046, IN 95-032, GL 92-08, IN 91-047, IEB 92-01, etc.). These documents deal with performance issues, not an aging issue. VCSNS Thermo-Lag 330-1 resolution plan demonstrated that adequate protection is provided for the station's safe shutdown capability while eliminating the need for reliance on Thermo-Lag 330-1 fire barriers.

Information Notice 88-04 addressed inadequate qualification and documentation of fire barrier penetration seals. This is a design issue and not an aging issue.

Information Notice 88-56 addressed potential problems with silicone foam fire barrier penetration seals. The IN noted voids, gaps, and splits in the fire barrier penetration seals. These problems were associated with the installation and not aging of the fire barriers.

Information Notice 88-60, Inadequate Design and Installation of Watertight Penetration Seals, addressed water migrating between fire areas by flowing through fire barrier penetration seals. The concern was that an uncontrolled spill or possibly water discharge during fire fighting activities may damage redundant equipment required for safe shutdown. This is an installation problem and is not associated with aging.

Information Notice 93-41 addressed the one-hour fire endurance for Thermal Ceramics Kaowool, 3M Company FS-195, and 3M Company Interam E-50 fire barrier systems. This is a performance issue and is not related to aging.

Information Notice 94-28 addressed inadequate installation of penetration fire barrier seals. This was an installation issue, not an aging issue.

Information Notice 95-52 addressed fire endurance test results for 3M Company Interam Fire Barrier material. VCSNS does not use this material.

SECY-96-146, "Technical Assessment of Fire Barrier Penetration Seals in Nuclear Power Plants," reported that many fire barrier materials are resistant to thermally accelerated aging and that the material properties of silicone-based materials, which dominate the industry, are particularly age independent. Sandia National Lab (SNL) concluded that these materials are not expected to exhibit problems as they age. Moreover, on the basis of its review of operating experience and the technical literature, SNL did not find any penetration seal problems that were directly related to aging. SNL reported that it did not find information on thermal aging or radiation testing of grout, cement, and gel-type seals. SNL did not recommend an experimental aging program [Reference 56]. The report concludes that these materials are not expected to exhibit problems as they age. Moreover, Sandia National Lab performed a review of operating experience and technical literature and did not find any problems that were directly related to aging.

Previous VCSNS fire barrier inspections were also reviewed to determine if there were any aging effects which should be considered for the period of extended operation. No deficiencies associated with aging have been identified.

As a result of the review of industry, NRC data, and VCSNS specific data, no additional aging effects beyond those discussed were identified.

6.5.5 Summary of Aging Effects Requiring Management

Fire Barriers
The fire barrier walls, ceilings and floor concrete design complies with the applicable provisions of the American Concrete Institute, ACI 318-71 [Reference 33]. Aging effects requiring management for firewalls were identified in Section 6.4 with concrete components. Firewalls were evaluated for loss of material, cracking, and change in material properties. Each of these aging effects has been assessed for firewalls. In addition, a review of NRC generic communications, industry experience, and VCSNS operating experience was also performed to validate the aging effects requiring management.

Concrete cracking and spalling/freeze-thaw has been identified as aging effects requiring aging management in the GALL Section VII.G [Reference 5]. The other aging effects identified in the GALL, aggressive chemical attack, reaction with aggregates, and corrosion of embedded steel / rebar, are not applicable to VCSNS, refer to Section 6.4 for explanation. Cracking of the firewalls could result in the firewall being unable to perform its intended function of confining or retarding the spread of fire. Cracking of the firewalls is managed by the Fire Protection Program and the Maintenance Rule Structures Program. These programs were evaluated to demonstrate that they would adequately manage cracking of the firewalls for the period of extended operation.

Firewalls may perform other functions. These other functions are addressed with the aging management review for reinforced concrete walls and masonry walls.

Fire Doors

Fire doors are constructed of steel and the doors along with their frames, hardware, and attachment devices are qualified to Underwriters Laboratories specifications. All components of the doors are coated to prevent corrosion. Aging effects requiring management for fire doors were identified in Section 6.2 with steel components in an air environment. Fire doors were evaluated for loss of material, cracking, and change in material properties. Each of these aging effects has been assessed for fire doors. In addition, a review of NRC generic communications, industry experience, and VCSNS operating experience was also performed to validate the aging effects requiring management.

The fire doors are located in sheltered environments such as the Auxiliary Building. The aging effect requiring management for the steel components such as fire doors located in sheltered environments was identified as loss of material due to general corrosion. Loss of material due to corrosion of the fire doors could result in the fire door being unable to perform its intended function of confining or retarding the spread of fire. Loss of material of fire doors is managed by the Fire Protection Program. This program was evaluated to demonstrate that it will adequately manage loss of material for the fire doors for the period of extended operation.

Loss of material due to wear of the door hardware and hinges has been identified as an aging effect requiring management for license renewal [Reference 5, GALL Section VII.G]. Loss of material due to wear is not considered as an aging effect but rather a consequence of frequent or rough usage. However, excessive wear for door appurtenances such as latches, strike plates (a strike plate is a wear plate and keeper for a latch bolt), hinges, sills, and closing devices is an attribute for inspection under the Fire Protection Program.

Fire doors may perform other functions. These other functions are addressed with the aging management review for flood, pressure, and specialty doors.

Fire Barrier Penetration Seals

Fire barrier penetration seals constructed of silicone-based materials and polymer/elastomers were evaluated for aging effects requiring management. Cracking/delamination and separation were identified as aging effects requiring management for fire barrier penetration seals constructed of silicone-based

materials. Cracking was identified as an aging effect requiring management for boot seals that are constructed of elastomeric materials such as rubber.
Each of these aging effects has been assessed for fire barrier penetration seals. In addition, a review of NRC generic communications, industry experience, and VCSNS operating experience was also performed to validate the aging effects requiring management. No additional aging effects were identified from the review.

The aging effects requiring management for the fire barrier penetration seals were identified as cracking of rubber boot seals and cracking/delamination and separation of other fire barrier penetration seals. Cracking and separation of the fire barrier penetration seals could result in the barrier being unable to perform its intended function of confining or retarding the spread of fire. Cracking and separation of the fire barrier penetration seals is managed by the Fire Protection Program. This program was evaluated to demonstrate that it will adequately manage cracking and separation of the fire barrier penetration seals for the period of extended operation.

Fire Water Suppression System Equipment Support
Loss of material due to corrosion has been identified as an aging effect requiring programmatic management for equipment supports for the extended period of operation. Loss of material due to corrosion for equipment supports is managed by the Maintenance Rule Structures Program.

In addition to the Fire Protection Program, the fire water system flow tests and full valve full cycling are being credited for managing aging effects for ancillary fire protection system civil commodities not addressed by the Maintenance Rule Structures Program. By performing valve line-ups, valve tests including cycling of manual valves, and full flow system testing, reasonable assurance is obtained for identification of aging for these ancillary components. More details are given in the Mechanical AMR TR00160-020 [Reference 75].

6.6 ELASTOMERS

Elastomers are used throughout nuclear plants in applications such as joint sealants, moisture barriers, etc. The American Society for Testing and Materials [Reference 58] defines an elastomer as "rubber or polymer that has properties similar to those of rubber." Rubber is capable of recovering from large deformations quickly and elastically. Elastomers included within this review are rubber, butyl rubber, neoprene, nitrile rubbers, silicone elastomers, ethylene propylene rubber (EPR) and ethylene propylene dienyl monomer (EPDM).

Natural rubber occurs in over 200 species of plants. The "hevea brasiliensis" tree accounts for over 99% of the world's natural rubber production. Most applications for natural rubber require that the rubber be vulcanized. Unvulcanized is generally not very strong, does not maintain its shape after a large deformation, and can be very sticky. The vulcanization process, which includes the addition of sulfur and/or other additives to crude rubber, increases the retractive force of the rubber and reduces the amount of permanent deformation remaining after removal of the deforming force. Thus, vulcanization increases elasticity while it decreases plasticity. While natural rubber has higher tensile strength than synthetic varieties, the performance profiles of synthetics show some properties markedly superior to those of natural rubber, particularly heat, oil, and ozone resistance [Reference 59, page 5]. Weathering of natural rubber results in molecular chain breaking and softening.

Butyl rubber is a copolymer of isobutylene and isoprene. Butyl rubbers are characterized by high resistance to gas permeation and to ozone, water, sunlight, heat, vegetable oils, and aging. They have low resilience at room temperature or below but show a large increase in resilience with increase in temperature up to 212° F [Reference 55, page 32-2].

Neoprene is chemically and structurally similar to natural rubber, and its mechanical properties are also similar. Its resistance to oils, chemical, sunlight, weathering, aging, and ozone is outstanding. It retains its properties at temperatures up to 250°F [Reference 54, page 286]. Most neoprenes crystallize readily at temperatures near 32°F, which limits their low temperature use [Reference 55, page 32-3].

Nitrile rubber is a copolymer of butadiene and acrylonitrile. Nitrile rubbers have outstanding resistance to oils and other solvents. They also have excellent abrasion and age resistance. Low-temperature properties are poor but can be improved by the use of certain plasticizers at some sacrifice in oil resistance and a possibility of plasticizer incompatibility, producing a gradual increase in stiffness upon prolonged exposure at low temperatures [Reference 55, page 32-2].

Silicone elastomers are polymers composed basically of silicon and oxygen atoms. Silicone elastomers are the most stable group of all the elastomers. They are outstanding in resistance to high and low temperatures, oils, and chemicals. High-temperature grades have maximum continuous service temperatures up to 600°F [Reference 54, page 290]. Other important characteristics include a high degree of chemical inertness, resistance to weathering, and good dielectric strength, and low surface tension [Reference 60, page 1048].

Ethylene propylene rubber (EPR) is a copolymer of polyethylene and polypropylene. The addition of a diene monomer (polybutadiene) to the copolymer creates EPDM rubber, ethylene propylene diene monomer. Both of these materials offer exceptional weathering and aging resistance, excellent electrical properties, and good heat resistance. Like butyl rubber, they are not resistant to petroleum oils [Reference 61, page 126].

The information used to document the associated aging effects requiring management for elastomers is gathered from industry sources. The next section evaluates the aging effects of material property changes and cracking of elastomers. These aging effects are caused by exposure to ultraviolet radiation, ozone, thermal exposure, and radiation. The next section also evaluates these aging effects for the different materials and determines the aging effects requiring management for the period of extended operation.

6.6.1 Aging Effect Assessment

Specification SP-227 [Reference 62] states elastomers used for penetration seals are designed for temperatures between 50° F and 122° F. Silicone foam (Dow Corning's 3-6548 RTV), silicone elastomer (Dow Corning's Sylgard 170A and 170B), and flexible boots (Keene Chase Foster fiberglass reinforced silicone rubber) upper temperature limit is 275°F based upon Materials Handbook [Reference 57]. Engineering drawing SS-021-018 documents the maximum ambient temperatures for each room in each building. Ambient room temperatures are well below the upper temperature limit of 100°C (or 212°F), therefore aging management for thermal effects is not required.

A listing of VCSNS seal materials, safety classification, and temperature / radiation limits is as follows:

Seal Material	Elastomer	Nuclear Safety Related	Temperature [Reference 63]	Radiation [Reference 64]
Dow Corning's 3-6548 RTV (room temperature vulcanized)	Yes – Silicone Rubber	Yes	-60° C to 100° C	10^6 Rads
Dow Corning's Sylgard 170A and 170B	Yes – Silicone Elastomer	Yes	-90° C to 250° C	10^6 Rads
Keene Chase Foster fiberglass reinforced silicone rubber	Yes – Silicone Rubber	Yes	-60° C to 100° C	10^6 Rads
Serviced closed cell plastic foam filler by Grace Construction Materials Company	Yes	No	-	-
Colma Joint Sealer	No	No	-	-
Link Seals fabricated by Thunderline	Yes – Synthetic Rubber	No	-	-
Fibrated cold plastic coal tar pitch flashing compound fabricated by Pittsburgh-Des Moines	No	No	-	-
Permapol RC-2SL Polyurethane Sealant	Yes	No	-	-
Steelcote Thiocaulk	Yes – Polysulfide Elastomer	Quality Related	-60° C to 80° C up to 150° C	10^6 Rads

Engineering drawing SS-021-018 documents the ambient temperature and radiation levels for each room in each building. The temperature for each room in each building is < 275° F (upper limit for Silicone Elastomer from Materials Handbook) so aging effects due to thermal exposure do not need to be managed. The radiation levels in several areas of the Auxiliary Building, Intermediate Building, and the basement of the Fuel Handling Building exceed the limit for radiation exposure, 10^6 Rads [Reference 64]. Therefore cracking and change in material properties are aging effects requiring management for these specific areas.

6.6.2 Elastomer Application at VCSNS

Elastomers for indoor applications are given in Specification SP-227 [Reference 62]. Elastomers for indoor applications are silicone foam (Dow Corning's 3-6548 RTV (room temperature vulcanized)), silicone elastomer (Dow Corning's Sylgard 170A and 170B), and flexible boots (Keene Chase Foster fiberglass reinforced silicone rubber).

Elastomers for outdoor applications in Electrical Manhole 2 (shown on engineering drawings E-434-006 and E-434-007) are serviced closed cell plastic foam filler by Grace Construction Materials Company combined with Colma Joint Sealer (which is not an elastomer). Both of these applications are Non-Nuclear Safety (NNS), therefore are not in license renewal scope. In addition, duct banks and manholes are designed with slope and sump pits for proper rain water drainage and a non aggressive environment (see Table 6.1-3).

Circulating Water Pump House – Diesel Fire Pump Cubicle and Diesel Generator Building do use elastomers in piping penetrations. Link Seals fabricated by Thunderline are noted on engineering drawing E-303-011 and are used for Fire System (FS), Circulating Water System (CW), and Emergency Feedwater System (EF) components penetrating cubicle walls. Link-Seals are inter-locking synthetic rubber and are Non-Nuclear Safety Related. The Fire Protection Program manages those that serve a fire protection function.

A yard structure in scope is the Condensate Storage Tank (CST) shown on vendor (Pittsburgh-Des Moines) drawing 1MS-17-096. Seal material is grout and a fibrous cold plastic coal tar (pitch) flashing compound fabricated by Pittsburgh-Des Moines. Grout is considered part of the foundation and is inspected as part of the Maintenance Rule Structures Program. Coal tar pitch is not an elastomer based upon Materials Handbook [Reference 57, Page 800].

Elastomer seismic joint filler is installed between the Reactor Building foundation mat and the Fuel Handling Building, Intermediate Building, and East Penetration Access Area. Elastomer (Permapol RC-2SL Polyurethane Sealant) is 3 inch wide by 1/2 inch deep as shown on engineering drawings: E-413-067, E-413-068, and E-415-063. Permapol RC-2SL polyurethane sealant is Non-Nuclear Safety Related. Seismic joints are kept free of debris and are inspected by the Maintenance Rule Structures Program.

In the Reactor Building, the 1/2 inch separation "Moisture Barrier" between the 412' floor slab and liner plate is filled with Steelcote Thiocaulk to a depth of 1/2 inch as shown on engineering drawing E-411-517. Steelcote Thiocaulk material is a polysulfide elastomer [Reference 65]. Steelcote Thiocaulk material is classified Quality Related and is inspected by the Containment ISI Program – IWE/IWL.

For indoor applications as fire barrier penetration seals, a visual examination of each elastomer is performed as stated in Surveillance Test Procedures STP-728-series. For the Reactor Building "Moisture Barrier" between the 412' floor slab and liner plate, a visual examination is performed in accordance with ISE-4 [Reference 76]. Inspection of outdoor application is managed by the Maintenance Rule Structures Program.

6.6.3 Industry Experience

No Licensee Event Reports were found based upon Licensing Database search using elastomer as query.

NRC Inspection Report 50-395 / 98-01 identifies use of elastomers on one side of fire seals where elastomers are not part of the fire seal test reports. VCSNS addressed this issue through the Penetration Seal Project Plan where the fire barrier penetration seals were revalidated and proper documentation for as built configurations are provided. NRC tracked this issue as Follow Up Item 50-395 / 98001-05. Inspection Report 50-395 / 00-02 dated May 1, 2000 closed this Follow Up Item based upon reviewing the VCSNS Penetration Seal Project Plan implemented for 5440 penetrations, NRC concluded that the Penetration Seal Project Plan satisfied NRC Generic Letter 86-10 and future scope of the Penetration Seal Project Plan was reasonable and complied with GL 86-10. This issue was not aging related.

NRC Inspection Report 50-395 / 92-23 identifies deterioration of elastomer in Water Seal Return Valve on B Waste Gas Compressor. This issue dealt with the internal of a valve is not within scope for License renewal.

NCN 99-0489 documents rust found on the Reactor Building liner plate adjacent to the moisture barrier and a degraded moisture barrier. The nonconforming condition was identified during a visual inspection prior to a Type B LLRT as discussed in Section 7.2. In accordance with the NCN disposition, the rust on the liner plate was removed and affected portions of the moisture barrier were replaced. A visual examination and an Ultrasonic Test (UT) demonstrated that the liner plate had not degraded. The examination is documented in the Technical Work Record (TWR) Disposition to NCN 99-0489. This NCN reports normal surface life exposure and is not aging related.

CER 00-1617 discusses minor cracks and separation of the moisture barrier at elevation 412 feet in the Reactor Building. This condition was identified during the initial ASME Section XI Subsection IWE/IWL inspections as discussed in Section 7.9. The moisture barrier was repaired during RF-12. During RF-13, CER 02-1111 identified minor separation of the moisture barrier and the affected areas were subsequently repaired.

6.6.4 Summary of Aging Effects Requiring Management

Cracking and change in material properties are aging effects requiring management for elastomers. They include flood seals, seismic joints, and the "Moisture Barrier" between the floor slab and liner plate located in the Reactor Building.

Silicone foam Dow Corning's 3-6548 RTV (room temperature vulcanized), silicone elastomer Dow Corning's Sylgard 170A and 170B, flexible boots (Keene Chase Foster fiberglass reinforced silicone rubber), and Thiocaulk polysulfide elastomer require aging management for effects of radiation. Silicone foam Dow Corning's 3-6548 RTV (room temperature vulcanized), silicone elastomer Dow Corning's Sylgard 170A and 170B, and flexible boots (Keene Chase Foster fiberglass reinforced silicone rubber) are managed under STP-728 series procedures since these elastomers perform a fire barrier penetration seal function. Thiocaulk polysulfide elastomer is managed under ISE-4 [Reference 76] since this elastomer is a moisture barrier inside the Reactor Building.

6.7 EARTHEN EMBANKMENTS

This section provides an overview of the aging effects and aging mechanisms that are applicable to the Service Water Pond (SWP) Dams (North, South and East), West Embankment and North Berm. The aging effects/mechanisms are important to the overall structural integrity of these earthen embankments. Soil and rock do not age significantly over the life of the facility; however, aging effects do exist that could degrade the overall form and/or function of an embankment comprised of soil and rock.

The aging effects that could potentially result in a loss of intended function(s) for these earthen embankments are:

- Loss of material
- Cracking
- Change in Material Property

Different aging mechanisms can lead to these aging effects in the earthen structures. This section addresses the aging effects on the earthen structures to determine those that require management and need to be addressed by aging management programs. An aging effect will be classified an aging effect requiring management if the following criteria are met:

1. The aging effect is applicable for the given material and environment, and

2. If undetected, the aging effect could progress to the point of loss of the intended function(s) under any current licensing basis condition during the period of extended operation.

The following sections provide an assessment of each of the three aging effects.

6.7.1 Loss of Material Aging Effect Assessment

Loss of material in earthen structures is caused by erosion. The aging effect of erosion is usually the result of one or more of the following mechanisms:

- Wind
- Rain surface runoff
- Wave action
- Subsurface seepage flow

Wind
During periods of drought, wind can erode loose surface soil from earthen structures. Wind erosion is generally limited to earthen structures that are not protected by good ground cover and/or riprap. The Service Water Pond Dams,

West Embankment and North Berm have sufficient ground cover in the form of vegetation and riprap to protect them against any loss of material owing to wind. Therefore, loss of material due to wind erosion is not an aging effect requiring management for these earthen structures.

Rain and Surface Runoff
The energy of raindrops impacting soil and the subsequent surface runoff can loosen soil particles and scour exposed surfaces. Flow erosion occurs when the loosened soil is carried away by surface runoff. Topography is a major factor in the erosion process. Steeper topography increases the flow of water and results in more erosion. Loss of material due to runoff is minimized by good design practices such as limiting embankment slopes to minimize overland flow velocities and by providing ground cover vegetation or riprap in areas with high fluid velocities and hydraulic jumps. Although the SWP Dams, West Embankment, and North Berm are protected by adequate vegetation and riprap to protect them against the loss of material caused by rain and surface runoff, flow erosion is an aging effect that requires management for the extended period of operation.

Wave Action
The shoreline of earthen structures may experience loss of material due to wave action. This effect is primarily caused by wind blowing across the surface of the water or by the flow of the water. This effect could cause the loss of slope stability owing to the undercutting erosion action of the waves. Riprap is provided on the slopes of earthen structures to dissipate the wave action. Therefore, erosion due to wave action is not an aging effect that requires management for the extended period of operation.

Subsurface Seepage Flow
Subsurface flow is caused by excess groundwater seeping through the soil mass and forming subterranean soil tubes. This phenomenon is known as piping. Piping results in the loss of soil through internal erosion, which can lead to rapid deterioration and/or failure of an embankment. Seepage is generally only a problem during the initial filling of a reservoir or water control structure. However, foundations of all earthen structures will have some seepage under prolonged storage conditions. It is important that foundation seepage be controlled or kept within tolerable limits in all areas to maintain stability. Indications of subsurface flow include sudden unexplained water level drops, surface cracks, sloughing, and unexplained settlement. Loss of material due to seepage and piping is an aging effect requiring management for the SWP Dams, West Embankment and North Berm.

6.7.2 Cracking Aging Effect Assessment

Cracking of earthen structures may result from one or more of the following mechanisms:

- Settlement
- Frost heave

Settlement
All earthen structures experience some degree of settlement. As the subgrade soil is loaded, it settles under the pressure of the overburden. The majority of settlement commonly occurs during construction or within the first year following construction. Consolidation occurs predominantly in fine-grained soils as water is pressed out of the voids in the soil and the void ratio is reduced. Consolidation can occur over many years depending on the thickness of the stratum. Slowly decreasing settlement due to consolidation is not a serious problem unless the crest camber is lost and the loss of freeboard is reduced below a permissible limit [Reference 77]. Cracking within the embankments may be caused by differential settlements. Therefore, cracking due to settlement is an aging effect requiring management for the SWP Dams, West Embankment and North Berm.

Frost Heave
Water in soil begins to freeze when the daily mean temperature remains below 32° F for an extended period (three or more days) of time. As groundwater continues to freeze, earthen structures may experience deformation caused by a phenomenon called frost heave. Frost heave is caused by water in the soil that expands approximately 10% when it freezes. Over an extended period of subfreezing temperatures, frost heave may cause permanent deformation and cracking to an earthen structure. However, this effect only occurs in fine grain soils in areas with a deep frost line. VCSNS is not located in a geographic region where there is a deep frost line [Reference 78]. Therefore, loss of form due to frost heave is not an aging effect requiring management for the SWP Dams, West Embankment and North Berm.

6.7.3 Change in Material Property Aging Effect Assessment

The process of desiccation is the primary reason for changes to the material properties of earthen structures. Desiccation is the process of completely drying out the soil by thoroughly removing all of the moisture. This may occur when soil is exposed to the air for extended periods of time. Water not ionically combined with the soil is drawn toward the surface where it evaporates [Reference 79]. Highly plastic preloaded clay is especially susceptible owing to its potential for shrinkage and loss in pliability. This shrinkage and loss in pliability in turn may cause the exposed surface of the clay to become brittle and crack or flake off. Desiccation combined with surface flow, wind, and/or wave action may accelerate the effects of erosion.

Adequate ground cover tends to hold moisture in the soil, and can control or minimize the effects of desiccation. The Service Water Pond Dams, West Embankment and North Berm have sufficient ground cover in the form of vegetation and riprap to prevent desiccation. Therefore, desiccation is not an aging effect requiring management for these earthen structures.

6.7.4 Industry Experience

In order to substantiate the applicable aforementioned aging effects and to assure that there are no additional aging effects beyond those discussed herein, a review of the industry experience was performed.

This review included a search of Licensee Event Reports (LERs), NRC generic communications and NRC contractor research reports (NUREG/CRs), and information on dams in general. The following summaries provide information from this investigation.

Previous investigations into dams indicate that the frequency of failure decreases with later years of construction [Reference 80]. This is generally attributed to improvements in the methods of design and construction over time. The age of the dam is another factor that has been identified as having an effect on the rate of dam failure. Approximately half the dam failures occur during the first 5 years of operation [Reference 81].

Earthen structures have extremely low failure rates. In fact, statistics on dam (earthen and concrete) failures, based on the sum of operation years of a regional group of dams, show a frequency of one failure every 1500 to 1800 dam years [Reference 82]. These statistics indicate that earthen structures have a natural resistance to aging.

The Executive Committee of the United States Committee on Large Dams (USCOLD) authorized the Dam Safety Committee's Subcommittee on Dam Incidents and Accidents to compile a list of dam incidents from 1972 through 1986. The compilation of data was documented in "Lessons from Dam Incidents", American Society of Civil Engineers, New York, New York, 1988. There were 164 incidents documented for earthen dams over fifty feet in height. Six of the incidents were major failure of an operating dam that resulted in complete abandonment of the dam. Seven of the incidents were the failure of an operating dam which permitted the damage to be successfully repaired and the dam again placed in operation. The majority of the incidents (eighty-four) were identified as repairs that were required because of deterioration or to update certain features. The causal factors for these incidents were loss of material due to piping and sliding; overtopping; or deficiencies in the design or construction of the dam/foundation/spillway.

6.7.5 Summary of Aging Effects for Earthen Embankments

Loss of material due to surface erosion and/or subsurface seepage, and cracking due to settlement are aging effects requiring management for the SWP Dams, West Embankment and North Berm. Loss of material and cracking can result in the loss of intended function(s) of these earthen structures. The Service Water Pond Dams (North Dam, South Dam, East Dam and West Embankment) are managed by the Service Water Pond Dam Inspection Program and the North Berm is managed by the Maintenance Rule Structures Program. These programs are discussed in detail in Section 7.0 and are evaluated to demonstrate that they will adequately manage aging for the period of extended operation.

6.8 CLASS 1 COMPONENT SUPPORTS

Class 1 component supports are those supports for Class 1 major equipment and Class 1 piping that are subject to aging management review. Class 1 component supports subject to an aging management review include: Class 1 piping supports and major equipment supports (pressurizer base flange and upper lateral supports; reactor vessel supports; steam generator vertical, lower lateral, and upper lateral supports; and reactor coolant pump lateral and vertical support assemblies).

The design of Class 1 piping supports is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF 1971 Edition with Addenda up to an including Winter 1973 [Reference 83, Section 1:01]. The fabrication of Class 1 piping supports is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF 1971 Edition with Addenda up to an including Winter 1973 [Reference 83, Section 2:11.1.1].

Design and fabrication of the Class 1 major equipment supports are in accordance with the AISC specifications for the "Design, Fabrication, and Erection of Structural Steel for Buildings," 1969 Edition, and are evaluated under applicable portions of the ASME Boiler and Pressure Vessel Code Subsection NF 1971 Edition with Addenda up to an including Winter 1973 [Reference 12 Section 5.5.14.3].

The Reactor Building internal structure reinforced concrete used at these supports is designed in accordance with the American Concrete Institute ACI-318, 1971 Edition [Reference 12 Section 3.8.3.2].

Each of the different Class 1 supports is discussed in more detail below.

Class 1 Piping Supports

The piping supports described herein include those required to support and restrain Class 1 piping. All Class 1 piping supports are designed and analyzed for design, normal, upset and faulted conditions in accordance with ASME Code, Section III Subsection NF 1971 Edition with Addenda up to an including Winter 1973 [Reference 12 Section 5.2.1.10.7 and design specification DSP-544V Section 1:01]. The properties of the materials used for the supports are given in Appendix I to ASME Code, Section III 1974 Edition [Reference 85, Table 4.2.3-1 and 83]. Hilti Kwik bolts are used for Class 1 pipe support attachments to building structures [Reference engineering drawings; S-321-601 series, S-321-641 series, S-321-671 series, and S-321-691 series].

Protection of structures, systems, and components important to safety from the dynamic effects of piping failure is provided in accordance with the requirements of 10 CFR 50, Appendix A, General Design Criterion GDC-4. Subsequent to the

GDC-4 final rule change [Reference 86], postulated breaks in the reactor coolant loop piping, except for branch line connections, were eliminated for VCSNS. The dynamic effects of the postulated breaks at six terminal ends in the cold, hot, and crossover legs, the steam generator inlet elbow, and the loop closure weld in the crossover leg were eliminated from the structural design basis by application of leak-before-break methodology, as presented in WCAP-13206 [Reference 87]. Approval of the elimination of the VCSNS reactor coolant loop piping breaks is given in a NRC Safety Evaluation Report [Reference 88]. To provide the high margins of safety required by GDC-4, the non-mechanistic pipe rupture design basis is maintained for containment design and ECCS analysis, while the environmental effects subsequent to postulated pipe ruptures are retained for electrical and mechanical equipment environmental qualification [Reference 12, Section 3.6].

The only set of LOCA inputs re-analyzed to support steam generator replacement and their associated revised operating conditions were the large auxiliary lines, RHR line/primary coolant loop connection, accumulator line/primary coolant loop connection, and pressurizer surge line/primary coolant loop connection. Documentation of the reactor coolant loop analysis as affected by steam generator replacement is furnished in stress report WCAP-9119 [Reference 89]. The conclusions of the report indicate that the impact of steam generator replacement will not result in reactor coolant loop piping stresses and fatigue usage that exceeds the ASME Code requirements for normal, upset, emergency, and test conditions. The structural integrity and Safety Related design requirements of the reactor coolant loop will be maintained during all conditions of the design specification [Reference 12, Section 5.2.1.10.3.1].

The extent of all restraints includes the components attached to the piping, support structure or vessel, the main hanger components, and all necessary rods and turnbuckles. Structural steel and concrete embedments are considered structural. Exceptions to the extent of restraints would include attachment lugs, plates, etc., which are designed and fabricated with vessels or containment in accordance with ASME Section III.

The snubbers provided for VCSNS are the hydraulic type and are designed and manufactured to requirements and criteria which are equivalent to those of ASME Section III, Subsection NF. Additionally, all snubbers are subjected to a thorough testing program which verifies their capability to function properly before, during, and after upset and faulted condition loadings [Reference 12, Section 5.2.1.10.4]. Although snubbers are excluded from aging management review by 54.21(a)(1)(i) because they are active, the items that mount the snubber to the pipe and to the structure are included within the scope of license renewal and require aging management review.

Lubrite plate has been used in a few Class 1 pipe hanger supports at VCSNS (drawing S-321-671, Sheets CSH-219 and CSH-934). Lubrite is the trade name for a low friction lubricant material used in applications where relative motion (sliding) is desired. The Lubrite proprietary lubricant is a custom compound mixture of metals, metal oxides, minerals, and other lubricating materials combined with a lubricating binder. At VCSNS, the intended function of the lubrite plates is to facilitate relative motion (sliding) during RCS heat-up and cool-down. As described in an engineering brief supplied by Lubrite vendor Jackson-Wheeler Metals Services, Inc., Lubrite material resists deformation, has a low coefficient of friction, resists softening at elevated temperatures, absorbs grit and abrasive particles, is not susceptible to corrosion, withstands high intensities of radiation, and will not score or mar. From additional literature on Lubrite provided by Lubrite Technologies (formerly Merriman), Lubrite products are solid, permanent, completely self lubricating, and require no maintenance for the design life of the product. The lubrite lubricants used in nuclear applications are designed for the environments to which they are exposed. An extensive search of industry and VCSNS plant specific operating experience did not identify any instances of lubrite plate degradation or failure to perform its intended function. Consequently, there are no known aging effects that would lead to a loss of intended function.

Pressurizer Supports

The pressurizer supports consist of a lower skirt ring support (circular ring bearing plate) and upper lateral supports near the center of gravity of the pressurizer. The pressurizer supports are shown on FSAR Figure 5.5-10 [Reference 12 and 70] and drawing 1MS-07-069.

The upper lateral supports consist of wide flange sections and members constructed from structural plates and beams. These supports are attached to embedments anchored to the secondary shield walls (drawing 1MS-07-069).

The pressurizer lower support consists of a steel ring bearing plate which mates the pressurizer skirt support to the anchor bolts embedded in the foundation pedestal (drawings E-411-234 and 1MS-07-069).

Reactor Vessel Supports

The reactor vessel supports are individual shoe and welded structural steel assemblies at each reactor vessel nozzle support point. They are located beneath the three cold leg nozzles and three hot leg nozzles. A typical Westinghouse schematic of the reactor vessel supports is shown on FSAR Figure 5.5-7. These supports are constructed from steel plate sections and anchored/embedded within the primary shield wall (drawing 1MS-07-057). Only the top support plates are accessible for inspection.

Provisions are made for the reactor cavity cooling fans to provide cooling for the reactor vessel support structure and adjacent concrete to prevent operating temperatures from exceeding 200°F [Reference 67 and 74]. Specific VCSNS operating experience exceptions are addressed and acceptably resolved in Design Calculation DC00020-209 [Reference 74]. The temperature of the vessel nozzle shoes, which contact the support structure, is assumed to be 600°F for design purposes.

Steam Generator Supports

Each steam generator is supported by four vertical columns (pinned at both ends using spherical bushings) attached to the steam generator support lugs, lower lateral supports including compression bumpers, and an upper lateral support consisting of a structural ring band with compression snubbers. [Reference 12, Figure 5.5-8 and 70].

The steam generator support columns provide vertical support for the steam generators. These columns are constructed from large pipe sections with clevises attached to each end. The support columns are attached with base plate supports and anchor bolts which are embedded and anchored within the base slab (drawing 1MS-07-072).

The steam generator lower lateral supports are mounted around each steam generator and consist of large wide flange sections and structural plates. Each support is attached to embedments anchored to the secondary shield walls (drawing 1MS-07-058).

Each steam generator upper lateral support consists of a ring girder restraint and two "A-frame" tie-rods supporting the ring girder from the steam generator trunnions. The horizontal supports (exterior to the ring girder) consist of large flanged sections constructed from structural plates and attached to embedments anchored to the secondary shield walls (drawing 1MS-07-066).

Reactor Coolant Pump Supports

The reactor coolant pump supports consist of three vertical columns (pinned at both ends using spherical bushings) and lateral steel supports. The reactor coolant pump supports are shown on FSAR Figure 5.5-9 [Reference 12 and 70].

The reactor coolant pump support columns provide vertical support for each reactor coolant pump. These columns are constructed from large pipe sections with clevises attached to each end. The support columns are attached with base plate supports with anchor bolts which are embedded and anchored within the base slab (drawing 1MS-07-072).

The reactor coolant pump lateral supports consists of three tie-rods attached to the pump (at the same location as the columns) which are attached by embedments anchored to the secondary shield walls (drawing 1MS-07-058).

6.8.1 Intended Functions

The intended function of the component supports is to provide structural and/or functional support to the safety related equipment.

6.8.2 Materials of Construction

The Class 1 major equipment supports and pipe supports are constructed of carbon and low alloy steel materials. References for the materials used for the construction of the component supports are summarized in Table 6.8-1, "Class 1 Component Supports – Aging Management Review Results". Structural steel materials for the Class 1 component supports generally conform to ASTM A36, ASTM A572 Grade 50, and ASTM A588. High strength structural bolt material conforms to ASTM A490.

Class 1 pipe supports are generally constructed of a standard support, a structural frame, or some combination of the two. A standard support is an assembly consisting of one or more units usually referred to as a catalogue item and generally mass-produced. The jurisdictional boundary of the pipe support extends from the attachment to the pipe back to the attachment to the supporting structure. Jurisdictional boundaries are illustrated on design specification DSP-544V, Figure 1.

6.8.3 Items Not Subject to Aging Management Review

No Class 1 component supports have been identified that are not subject to aging management review. It should be noted that the reactor vessel supports are anchored/embedded within the concrete primary shield wall, with only the top support plates accessible for inspection.

Snubbers are excluded from aging management review by 10 CFR 54.21(a)(1)(i) because they are active; however, the items that mount the snubber to the pipe and to the structure are included within the scope of license renewal and require aging management review.

6.8.4 Internal and External Environment

All Class 1 component supports are located inside the Reactor Building in a moist, humid environment. The Reactor Building temperature environment and radiation doses during normal operation are identified in Tables 6.1-1 and 6.1-2, respectively.

6.8.5 Aging Effects Identification

Aging effects for Class 1 component supports inside the Reactor Building are consistent with the aging effects for steel components in air environment as described in Section 6.2 as follows:

General Corrosion

Loss of material due to general corrosion could occur where steel components are located in moist, humid environments, which includes all areas of the Reactor Building. If left unmanaged, significant (relevant) loss of material due to general corrosion could result in a loss of the component intended function. Therefore, loss of material is an aging effect requiring management during the period of extended operation for most steel components inside the Reactor Building.

A review of VCSNS operating experience identified one NCN written on Class 1 component supports. NCN 00-1517 (10/23/2000) describes surface corrosion (non-relevant) on a lateral support and anchor bolts for the Loop "B" reactor coolant pump. The cause of the corrosion was moisture from a leaking flange above the support. The leaking flange was repaired, while the support and bolts were cleaned and reworked to prevent further corrosion.

Boric Acid Corrosion

Boric acid corrosion (also known as boric acid wastage) has been identified as an aging mechanism for the nuclear power industry. Leaks from borated water systems contain boric acid that comes in contact with the external surfaces of other components. As the borated water evaporates from the component external surface, the boric acid is concentrated, resulting in loss of material due to boric acid corrosion (refer to Section 6.2 of this report). Class 1 component supports which may be exposed to boric acid systems are in the Reactor Building; therefore, loss of material due to boric acid corrosion is an aging effect requiring management for Class 1 component supports within the Reactor Building.

Stress Corrosion Cracking

Stress Corrosion Cracking (SCC) of steel occurs under a high level of sustained tensile stress either applied (external) or residual (internal) in a corrosive environment. Three parameters are required for stress corrosion cracking to occur: (1) a corrosive environment, (2) a susceptible material and (3) a high level of sustained tensile stress; as stated in the SER to WCAP-14422 Section 3.3.1.1 [Reference 70]. SCC is a phenomenon that primarily occurs in sensitized stainless steels, but may be evident in carbon and low alloy steel if a high level of sustained tensile stress and a corrosive environment exist. Corrosive environments containing sodium hydroxide, seawater, nitrate solutions, sulfuric acids or aggressive groundwater (chlorides > 500 ppm, sulfates > 1500 ppm) are not present at VCSNS. The internal environment of the Reactor Building does not contain aggressive chemicals under normal operating conditions. Therefore, the conditions necessary for SCC to occur do not exist for the majority of steel components, and with the exception of high strength bolting which is discussed below, SCC is not an aging effect requiring management.

Industry experience has shown that high strength bolts (bolts with tensile strength greater than 150 ksi) installed in Class 1 component supports could be susceptible to SCC in humid environments like the Reactor Building. The key factors necessary for SCC include high-strength materials, moist environments, and a high level of sustained tensile stress. Operating experience also shows that improperly heat-treated anchor bolts have been susceptible to SCC, especially when under a high preload (full preload of 70% of ultimate strength). Anchor bolts are also exposed to concrete where chlorides can leach-out and attack the intergranular structure of the bolts over time. Therefore based on industry experience, stress corrosion cracking is a potential aging effect for the ASTM A490 high strength anchor bolts used in the Class 1 component supports at VCSNS.

However, SCC of high strength anchor bolts should also be considered as a negligible aging effect if any of the following conditions apply:

- ASTM A490 anchor bolt material is properly heat-treated by conforming to ASTM Specification A490 through a certified mill test report.

- Anchor bolts are tightened snug-tight as defined by AISC; therefore, for bolts greater than 1" in diameter, a significant preload (in the order of 70% of ultimate strength) is not practical to develop.

- Anchor bolts do not have a high level of sustained tensile stress as evidenced by lower faulted condition design loads due to elimination of dynamic effects subsequent to postulated High Energy Line Break (HELB) of the reactor coolant system primary coolant piping.

6.8.6 Summary of Aging Effects for Class 1 Component Supports

Table 6.8-1 provides a summary of the aging management review results for the Class 1 component supports:

1) Loss of material due to general corrosion is an aging effect requiring management for the extended period of operation for Class 1 component supports. This loss of material could result in the steel components being unable to perform their intended function(s). Loss of material for the Class 1 component supports is primarily managed by the ASME Section XI ISI Program – IWF. The Maintenance Rule Structures Program also provides

a supporting level of inspection. These programs were evaluated to demonstrate that they will adequately manage loss of material for these steel components for the period of extended operation. Details of these programs are contained in Section 7.0 of this report.

2) Loss of material due to boric acid corrosion is an aging effect requiring management for Class 1component supports in the Reactor Building. This loss of material could result in the steel components from being unable to perform their intended function(s). The Boric Acid Corrosion Surveillance program manages loss of material due to boric acid corrosion. This program was evaluated to demonstrate that it adequately manages loss of material for these steel components for the period of extended operation. Details of this program are contained in Section 7.0 of this report.

3) Cracking due to SCC is a potential aging effect for ASTM A490 high strength anchor bolts used in the Class 1 major equipment supports at VCSNS. SCC is unlikely for ASTM A490 high strength anchor bolts given the following:

   - Anchor bolts are certified as quenched and tempered in accordance with ASTM A490.
   - Anchor bolts are installed snug-tight, thus no high preload.
   - Anchor bolts do not have a high level of sustained tensile stress as evidenced by lower faulted condition design loads due to elimination of dynamic effects subsequent to postulated High Energy Line Break (HELB) of the reactor coolant system primary coolant piping.

   Regardless, the examination requirements in the ASME Section XI ISI Program – IWF manage loss of function and cracking due to SCC for the Class 1 component supports that are exposed to the Reactor Building environment. This program was evaluated to demonstrate that it will adequately manage cracking for the A490 Class 1 component support bolts for the period of extended operation. Under the IWF Program, a visual inspection of the bolt heads is used to determine if loss of integrity or cracking (SCC) is evident; otherwise the intended function of the bolt is considered acceptable. Details of this program are contained in Section 7.0 of this report.

Generic Aging Lessons Learned (GALL) Comparison

The aging effects and aging management programs described above for Class 1 component supports are consistent with those described in Section III.B1 (Class 1 supports for ASME piping and components) of the GALL.

TABLE 6.8-1 AGING MANAGEMENT REVIEW RESULTS

CLASS 1 COMPONENT SUPPORTS

Component Type	Comp. Function	Material	References	External Env.	Aging Effect	Aging Mechanism	Aging Management Program
Class 1 Pipe Supports Hangers Restraints	Support	Carbon Steel Low Alloy Steel	FSAR Sections 3.6, 5.2 ASME Section III, Appendix I Class 1 Piping DBD Drawings: 1MS-07-055 1MS-07-056 1MS-07-060 1MS-07-065	Reactor Building Interior	Loss of Material	General Corrosion	ASME Section XI ISI Program – IWF [Section 7.3]
					Loss of Material	Boric Acid Wastage	Boric Acid Corrosion Surveillances [Section 7.6]
					Cracking	Stress Corrosion Cracking (SCC)	ASME Section XI ISI Program – IWF, Visual inspection of bolts [Section 7.3]
Pressurizer Upper Lateral Support Lower Support Skirt Ring	Support	Carbon Steel Low Alloy Steel	FSAR Section 5.5 FSAR Figure 5.5-10 Class 1 Piping DBD WCAP-14422, Rev 2-A Drawings: E-411-234 1MS-07-069	Reactor Building Interior	Loss of Material	General Corrosion	ASME Section XI ISI Program – IWF [Section 7.3]
					Loss of Material	Boric Acid Wastage	Boric Acid Corrosion Surveillances [Section 7.6]
					Cracking	Stress Corrosion Cracking (SCC)	ASME Section XI ISI Program – IWF, Visual inspection of bolts [Section 7.3]

Component Type	Comp. Function	Material	References	External Env.	Aging Effect	Aging Mechanism	Aging Management Program
Reactor Vessel Supports	Support	Carbon Steel Low Alloy Steel	FSAR Section 5.5 FSAR Figure 5.5-7 Class 1 Piping DBD WCAP-14422, Rev 2-A Specification SP-625 Drawings: E-423-046 E-511-200 Series 1MS-07-057	Reactor Building Interior	Loss of Material	General Corrosion	ASME Section XI ISI Program - IWF [Section 7.3]
					Loss of Material	Boric Acid Wastage	Boric Acid Corrosion Surveillances [Section 7.6]
					Cracking	Stress Corrosion Cracking (SCC)	ASME Section XI ISI Program - IWF, Visual inspection of bolts [Section 7.3]
Steam Generator Column Support Upper Lateral Support Lower Lateral Support	Support	Carbon Steel Low Alloy Steel	FSAR Section 5.5 FSAR Figure 5.5-8 Class 1 Piping DBD WCAP-14422, Rev 2-A Drawings: 1MS-07-056 1MS-07-058 1MS-07-059 1MS-07-061 1MS-07-066 1MS-07-067 1MS-07-068 1MS-07-072	Reactor Building Interior	Loss of Material	General Corrosion	ASME Section XI ISI Program - IWF [Section 7.3]
					Loss of Material	Boric Acid Wastage	Boric Acid Corrosion Surveillances [Section 7.6]
					Cracking	Stress Corrosion Cracking (SCC)	ASME Section XI ISI Program - IWF, Visual inspection of bolts [Section 7.3]

Component Type	Comp. Function	Material	References	External Env.	Aging Effect	Aging Mechanism	Aging Management Program
Reactor Coolant Pump Column Support Lateral Support	Support	Carbon Steel Low Alloy Steel	FSAR Section 5.5 FSAR Figure 5.5-9 Class 1 Piping DBD WCAP-14422, Rev 2-A Drawings: 1MS-07-056 1MS-07-058 1MS-07-059 1MS-07-061 1MS-07-062 1MS-07-072	Reactor Building Interior	Loss of Material	General Corrosion	ASME Section XI ISI Program – IWF [Section 7.3]
					Loss of Material	Boric Acid Wastage	Boric Acid Corrosion Surveillances [Section 7.6]
					Cracking	Stress Corrosion Cracking (SCC)	ASME Section XI ISI Program – IWF, Visual inspection of bolts [Section 7.3]

6.9 SUMMARY OF AGING EFFECT EVALUATIONS DIFFERENT FROM THOSE DESCRIBED IN THE GALL REPORT

GALL Item No.	GALL Aging Effect/ Mechanism Component Types	VCSNS Discussion
II.A1.3-a	Loss of material / Corrosion and Loss of Prestress Prestressing system: Tendons; anchorage components	The GALL does not recommend further evaluation if within the scope of the applicant's ASME Section XI, Subsection IWL program. The VCSNS Tendon Surveillance Program was originally established using the requirements of Regulatory Guide (RG) 1.35, proposed Revision 3, dated April 1979. In March 1995, the NRC issued a new rule, 10 CFR 50.55a, which invoked the requirements of the ASME Code, Section XI, Subsections IWE and IWL, 1992 Edition and 1992 Addenda. The present Tendon Surveillance Program adequately addresses the new requirements.
II.A3.1-c	Cracking / Cyclic loading Penetration sleeves; penetration bellows	The GALL indicates that if the CLB fatigue analysis does not exist, then further evaluation is recommended for detection of aging effects. The GALL identifies ASME Section XI, Subsection IWE and 10 CFR 50, Appendix J as aging management programs. It also noted that VT- 3 visual inspection may not detect fine cracks. Cracking due to cyclic loading on bellows is not a concern since the bellow's cyclic life exceeds the plant's normal operation cycle estimates by order of magnitude. VCSNS Main Steam, Main Feedwater, and hot penetrations bellows are not tested under the Appendix J Type B LLRT Program since the "hot" penetration designs do not incorporate resilient seals, gaskets, sealant compounds, or flexible seal assemblies on the inboard side of Reactor Building [Reference 12]. These bellows do not perform a pressure boundary function, but they do provide structural and/or functional support to safety related equipment (i.e., thermal and accident movement of the process pipe). Therefore, in the unlikely event that a crack develops in the bellow, the intended function of allowing process pipe movement would not be affected.

GALL Item No.	GALL Aging Effect/ Mechanism Component Types	VCSNS Discussion
II.A3.1-d	Crack initiation and growth / Stress corrosion cracking Penetration sleeves; penetration bellows	Stress Corrosion Cracking (SCC) is a concern for dissimilar metal welds. In the case of bellows assemblies, SCC may cause aging effects particularly if the material is not shielded from a corrosive environment. Subsection IWE covers inspection of these items under examination categories E-B, E-F, and E-P (10 CFR 50, Appendix J pressure tests). 10 CFR 50.55a identifies examination categories E-B and E-F as optional during the current term of operation. For the extended period of operation, examination categories E-B & E-F, and additional appropriate examinations to detect SCC in bellow assemblies and dissimilar metal welds are warranted to address this issue. VCSNS Main Steam, Main Feedwater, and hot penetrations bellows are not tested under the Appendix J Type B LLRT Program since the "hot" penetration designs do not incorporate resilient seals, gaskets, sealant compounds, or flexible seal assemblies on the inboard side of Reactor Building [Reference 12]. These bellows do not perform a pressure boundary function, but they do provide structural and/or functional support to safety related equipment (i.e., thermal and accident movement of the process pipe). A corrosive environment and a high level of sustained tensile stress do not exist for penetration bellows, therefore SCC on penetration bellows is not an aging effect requiring management. See Section 6.2 for aging evaluation on bellows. Penetration bellows located at the Reactor Building external side of containment wall are protected from boric acid corrosion by a layer of insulation and carbon steel cover plates (drawings 1MS-06-077 and 1MS-06-061). Therefore boric acid corrosion is not plausible for these penetration bellows since they are protected.

GALL Item No.	GALL Aging Effect/ Mechanism Component Types	VCSNS Discussion
II.A3.2-b	Loss of leak tightness in closed position / Mechanical wear of locks, hinges and closure mechanisms Personnel airlock; equipment hatch: Locks, hinges, and closure mechanisms	The personnel airlocks and equipment hatch contain operating mechanisms, which include gears, latches, hinges, linkages, etc. which operate to open and close the doors of the hatch to allow passage into and out of containment. These operating mechanisms perform their intended function with moving parts and with a change of configuration. Operation of the hatches is governed by Technical Specifications, Section ¾.6.1. Both the personnel hatch and the emergency hatch are required to be operable in Modes 1, 2, 3, and 4 (as defined by Technical Specifications). Surveillance requirements are also included in Technical Specifications. Actions are required to be taken up to and including plant shutdown in the event one or more of the hatch doors become inoperable. Plant experience did not identify any fretting or seal degradation for the personnel airlocks and equipment hatch. Mechanical wear of locks, hinges and closure mechanisms is not considered as an aging effect. The GALL does not recommend further evaluation if within the scope of the applicant's 10 CFR 50, Appendix J and plant Technical Specifications. The personnel airlocks and equipment hatch are within VCSNS 10 CFR 50 Appendix J Leak Rate Testing Program. VCSNS Technical Specifications inspection and maintenance requirements for these locks, hinges and closure mechanisms are considered as alternative AMP.
II.A3.3-a	Loss of sealing; leakage through containment / Deterioration of Seals, gaskets, and moisture barriers (caulking, flashing, and other sealants) Seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	Loss of sealing is not considered as an aging effect but rather a consequence of elastomer degradation. This effect may be caused by cracking and/or change in material properties for elastomeric material. VCSNS seals, gaskets, and moisture barriers aging effects are managed by Containment ISI Program – IWE / IWL and leak tightness is monitored by 10 CFR 50 Appendix J Leak Rate Testing Program

GALL Item No.	GALL Aging Effect/ Mechanism / Component Types	VCSNS Discussion
III.B1.2.2-a	Loss of mechanical function / Corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; elastomer hardening Constant and variable load spring hangers; guides; stops; sliding surfaces; design clearances; vibration isolators.	Loss of mechanical function, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; and elastomer hardening are not considered as aging effects but rather a design issue. Loss of material due to corrosion is covered by III.B1.2.1-a and is managed by the ASME Section XI ISI Program – IWF. Cracks in steel elements of component supports caused by vibratory stresses above the material endurance limit would develop in a matter of hours or days. This time frame is not consistent with the requirements of the license renewal rule, which address slow aging processes affected by extended operation. In addition, the potential for cracking induced by other cyclic loads, such as thermal cycling of the supported system, is implicitly considered in structural steel design through the specification of conservative design allowable stresses that account for a minimum of 10^5 load cycles. The GALL does not recommend further evaluation if within the scope of the applicant's ASME Section XI, Subsection IWF Program. Class 2 and 3 constant and variable load spring hangers; guides; stops; sliding surfaces are within VCSNS ASME Section XI ISI Program – IWF.

GALL Item No.	GALL Aging Effect/ Mechanism Component Types	VCSNS Discussion
III.B1.1.4-a III.B1.2.3-a III.B2.2-a III.B3.2-a III.B4.3-a III.B5.2-a	Reduction in concrete anchor capacity due to local concrete degradation / Service-induced cracking or other concrete aging mechanisms Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	Concrete structures and concrete components can be subjected to cyclic loading and therefore, can be subjected to fatigue degradation. However, concrete components have good fatigue strength properties for hundreds or thousands or cycles of below yield load application (high cycle low-level loads) [Reference 37]. For components that may be subjected to vibratory or cyclic loading, proper design eliminates or compensates for vibration and cyclic loading. In addition, vibration characteristically leads to cracking in a short period of time, on the order of hours to days of operation. For example, a component with 1 Hertz vibratory load will be subjected to 10^7 cycles in four months of service, so that failure, should it occur, is probable early in life for vibratory stresses above the endurance limit. Because this time period is short when compared to the overall plant operational life, any cracking would be identified and corrected to prevent recurrence long before the period of extended operation. This type of degradation is limited to a small set of components and is corrected as discovered with inspections of similar locations and configurations to ensure the event is location specific or a one-time event. The potential for cracking induced by other cyclic loads, such as thermal cycling of the supported system, is implicitly considered in structural steel design through the specification of conservative design allowable stresses that account for a minimum of 10^5 load cycles. VCSNS concrete components are designed in accordance with ACI standards and have good low cycle fatigue properties. Plant experience did not identify any concrete degradation due to service-induced loads. Therefore, cracking due to fatigue is not an aging effect requiring management for concrete components. The GALL does not recommend further evaluation if within the scope of the applicant's structures monitoring program. Concrete equipment pads are within VCSNS Maintenance Rule Structures Program.
VII.A1.1-a VII.A1.1.1	Loss of material / General, pitting, and crevice corrosion New fuel rack New fuel rack assembly	The new fuel rack was determined to support no license renewal intended functions as delineated by NEI 95-10 and Table 2.1-4 of NUREG-1800. The new fuel rack is therefore screened out. The new fuel rack at VCSNS is located in a mild dry air environment inside the Fuel Handling building.

GALL Item No.	GALL Aging Effect/ Mechanism ――― Component Types	VCSNS Discussion
VII.B.2.1	Loss of Material / Wear ――― Rail System	Wear of crane rails due to rolling or sliding wheels is not expected in any measurable amount due to the infrequent crane use. Review of past inspection reports at VCSNS indicates that those cranes within License Renewal scope are in good working condition.
VII.G.1.3 VII.G.2.3 VII.G.3.3 VII.G.4.3	Loss of material / Wear ――― Fire rated doors	Fire doors are passive features to seal passageways through fire rated barriers. Fire doors are to comply with NFPA 80 and satisfy UL or FM standards for approved fire ratings, again unless otherwise exempted. Loss of material due to wear is not considered as an aging effect but rather a consequence of frequent or rough usage. However, excessive wear for door appurtenances such as latches, strike plates (a strike plate is a wear plate and keeper for a latch bolt), hinges, sills, and closing devices is an attribute for inspection under the Fire Protection Program. In addition, review of past inspection reports at VCSNS indicates that fire doors are in good working condition.

7.0 AGING MANAGEMENT PROGRAMS

The aging management review is performed to assess whether the aging effects for the structures and structural components can be adequately managed by existing programs when the programs are continued into the period of extended operation. Further demonstration (or objective evidence) of the effectiveness of each current program in managing the aging effects is provided as a part of the review in order to offer reasonable assurance that the aging effects are being managed.

If an aging effect requiring management cannot be adequately managed by an existing program when the program is continued into the period of extended operation (either because a current program is judged to be inadequate or perhaps does not exist), then program attribute changes and additions shall be offered by the license renewal applicant to establish program adequacy for the period of extended operation. Demonstration of the effectiveness of the new or revised program requires two separate efforts. The first effort at the time of application will involve description of the methodology to implement the new or revised program with its defined attributes. The second effort at some future committed time will involve providing results, i.e., objective evidence, of the effectiveness of the implemented (new or revised) program in managing the aging effects in order to offer reasonable assurance that the aging effects are being managed.

The following sections provide descriptions of the aging management programs for earthen structures, concrete structures and structural components, steel structures and structural components, and fire barriers. The programs are evaluated in accordance with the guidance provided in NEI 95-10 [Reference 4]. Additional objective evidence of the effectiveness of the programs is provided to demonstrate that the effects of aging are managed so that the intended functions of the structures and structural components will be maintained consistent with the CLB for the period of extended operation.

The programs that are credited with managing the aging of structures and structural components delineated in Attachment II for the period of extended operation are:

Section	Title	Classification: Prevention	Classification: Mitigation	Classification: Condition Monitoring	Classification: Performance Monitoring
7.1	10 CFR 50 Appendix J General Visual Inspection			X	
7.2	10 CFR 50 Appendix J Leak Rate Testing				X
7.3	ASME Section XI ISI Program – IWF			X	
7.4	Battery Rack Inspection			X	
7.5	Boraflex Monitoring Program				X
7.6	Boric Acid Corrosion Surveillances		X	X	
7.7	Chemistry Program	X	X		
7.8	Containment Coating Monitoring and Maintenance Program	X		X	
7.9	Containment ISI Program – IWE/IWL			X	
7.10	Fire Protection Program			X	
7.11	Flood Barrier Inspection			X	
7.12	Maintenance Rule Structures Program			X	
7.13	Material Handling System Inspection Program			X	
7.14	Pressure Door Inspection Program			X	
7.15	Service Water Pond Dam Inspection Program (North, South & East Dams and West Embankment)			X	
7.16	Service Water Structures Survey Monitoring Program (SWIS, SWPH, Electrical Duct Banks & SW Intake Line A)			X	
7.17	Tendon Surveillance Program				X
7.18	Underwater Inspection Program (SWIS and SWPH)			X	

7.1 10 CFR 50 Appendix J General Visual Inspection

Prior to conducting a 10 CFR 50 Appendix J Type A Integrated Leak Rate Test (ILRT), a general visual structural examination of the containment system is conducted as required by Regulatory Guide (RG) 1.163 [Reference 90], Section C.3. The general visual examination identifies problems that may affect the ILRT results and establishes an adequate baseline for implementing the ILRT. The general visual examination satisfies Technical Specifications, Surveillance Requirement 4.6.1.6.3. General containment structural examination prior to 10 CFR 50 Appendix J Leak Rate Testing is a condition monitoring program. The general visual examination is for the attributes delineated in Surveillance Test Procedure, STP-207.002, "Inspection of Containment" [Reference 162]. These attributes include aging effects that are described in Design Engineering Guideline ST-07, "Containment Inservice Inspection Evaluation Criteria" [Reference 163]. The aging effects associated with the Steel Containment Vessel (SCV) or the containment liner are loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities. The aging effects related to moisture barriers are cracking, embrittlement, erosion, separation from attaching surface, or other deterioration. The aging effects associated with the Reactor Building structure are concrete spalling, cracks, delamination, or other deterioration.

Scope of Program – Accessible surfaces of the ASME Code Class MC Steel Containment Vessel (SCV) as defined in ASME Section XI, Table IWE-2500-1, Item E1.11 are in scope. Moisture barriers at concrete to SCV embedment zones and peripheries of attachments to the SCV (as defined in ASME Section XI, Table IWE-2500-1, Item E5.30) are in scope. The Reactor Building exterior and interior concrete delineated in STP-207.002 are in scope. Inspection areas for inside the Reactor Building are shown on engineering drawing series E-411-600. The general visual examination satisfies Technical Specifications Surveillance Requirement 4.6.1.6.3.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The 10 CFR 50 Appendix J General Visual Inspection is a condition monitoring program.

Parameters Monitored or Inspected – Aging effects are described in STP-207.002 as:

- For Steel Containment Vessel (liner), aging effects credited for license renewal are loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities.
- For moisture barriers, aging effects credited for license renewal are cracking, embrittlement, erosion, separation from attaching surface, or other deterioration.

- For the Reactor Building structure, aging effects credited for license renewal are concrete spalling, cracks, delamination, or other deterioration.

Detection of Aging Effects – The 10 CFR 50 Appendix J General Visual Inspection program conducted in accordance with STP-207.002 detects the following aging effects:

For the Steel Containment Vessel (liner) – Loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities.

For moisture barriers – Cracking, embrittlement, erosion, separation from attaching surface, or other deterioration.

For the Reactor Building structure – concrete spalling, cracks, delamination, or other deterioration.

Monitoring and Trending – Prior to conducting 10 CFR 50 Appendix J, Option B Type A ILRT, a general visual structural examination is performed to comply with 10 CFR 50 Appendix J as required by Regulatory Guide 1.163, Section C.3. The general visual structural examination implements Technical Specifications Surveillance Requirement 4.6.1.6.3. Since a ten year interval (10 CFR 50 Appendix J, Option B) for Type A ILRT is invoked, Regulatory Guide 1.163 requires a general visual structural examination of the containment system during two other refueling outages before the next Type A ILRT. General visual structural examinations of the Steel Containment Vessel (liner), moisture barriers, and Reactor Building concrete are in accordance with STP-207.002. No actions are taken as part of this program to trend inspection or test results.

Acceptance Criteria – Acceptance criteria for aging effects are in accordance with STP-207.002. Acceptance criteria are based on ASME Section XI, Subsection IWE-3510.1 for the SCV liner surface, Subsection IWL-3211 for Reactor Building concrete structure, and ASME Section XI, Subsection IWE-3513.1 for the moisture barriers. Acceptance criteria are delineated in ST-07.

Corrective Actions – A Condition Evaluation Report (CER) is initiated for any structure or component that does not meet the acceptance criteria after an engineering evaluation is performed in accordance with STP-207.002. The CER process is discussed in Station Administrative Procedure, SAP-1131 [Reference 160]. Repair / replacement of any structure or component is in accordance with ASME Section XI, Subsections IWE-3122.2 or IWE-3122.3. Specific corrective actions are implemented in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – Prior to conducting 10 CFR 50 Appendix J, Option B Type A ILRT, a general visual structural examination of the containment system is conducted as stated in Regulatory Guide 1.163. Since a ten year interval (10 CFR 50 Appendix J, Option B) for Type A ILRT is invoked, Regulatory Guide 1.163 requires a general visual structural examination of the containment system during two other refueling outages before the next Type A ILRT. The general visual structural examination implements Technical Specifications Surveillance Requirement 4.6.1.6.3 and is implemented by STP-207.002.

Operating Experience –The most recent Type A ILRT was completed on March 11, 1993 during RF-8, with a general visual structural examination of the containment system also implemented during RF-8. General visual structural examinations of the containment system were implemented during RF-10 and RF-12 to satisfy the requirement for a general visual structural examination of the containment system during two other refueling outages before the next Type A ILRT.

No Licensee Event Reports (LERs) were initiated subsequent to any general visual structural examination of the containment system. NRC Inspection Report 93-09 [Reference 91] reviewed STP-207.002 and concluded that this procedure is acceptable to implement the general containment visual inspection prior to a Type A ILRT.

Based on operating experience, the continued implementation of the VCSNS general structural inspection prior to a Type A ILRT (established by Technical Specifications Surveillance Requirement 4.6.1.6.3 and implemented by STP-207.002) manages the identified effects of aging. Continued implementation of the VCSNS general structural inspection prior to a Type A ILRT provides reasonable assurance that the aging effects will be managed and that the intended function(s) of containment will be maintained for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS general visual structural inspection prior to a Type A ILRT has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding a general visual structural inspection prior to a Type A ILRT is consistent with the VCSNS program. The VCSNS 10 CFR 50 Appendix J General Visual Inspection program contains those attributes delineated in GALL Chapter XI.S4 which have been determined by the NRC to provide an acceptable aging management program.

7.2 10 CFR 50 Appendix J Leak Rate Testing

The 10 CFR 50 Appendix J Leak Rate Tests are performed to ensure primary containment can withstand an internal pressure load and to measure the containment overall integrated leakage rate. 10 CFR 50 Appendix J Leak Rate Tests are required by Technical Specifications Surveillance Requirement 4.6.1.2. The Steel Containment Vessel (liner) aging effects are loss of material due to corrosion, cracking of welds, deformed structural attachments, gouges, dents, or other surface discontinuities. For moisture barriers the aging effects are cracking, embrittlement, separation from attaching surface, or other deterioration. Reactor Building structure aging effects are concrete spalling, cracks, etc. The 10 CFR 50 Appendix J Type A and Type B Leak Rate Tests are performance monitoring programs.

Scope of Program – Two types of tests are implemented, the Type A test is performed to measure the overall primary containment integrated leakage rate, which is obtained by summing leakage through all potential leakage paths, including containment welds, valves, fittings, and components that penetrate containment. Type B tests are performed to measure local leakage rates across each pressure-retaining or leakage-limiting boundary for containment penetrations and containment hatches. For VCSNS, Type A and Type B tests are defined in General Test Procedure, GTP-315, "Containment Leak Rate Testing Program" [Reference 164].

Accessible surfaces of the ASME Code Class MC Steel Containment Vessel (liner) are defined in ASME Section XI, Table IWE-2500-1, Item E1.11. Moisture barriers at concrete to Steel Containment Vessel (liner) embedment zones, and peripheries of attachments to the liner are defined in ASME Section XI, Table IWE-2500-1, Item E5.30. The Reactor Building exterior and interior concrete delineated in Surveillance Test Procedure, STP-207.002, "Inspection of Containment" is in scope. Inspection areas for inside the Reactor Building are shown on engineering drawing series E-411-600. Type B LLRT penetrations in scope are listed in FSAR [Reference 12], Table 6.2-53.

Primary containment hatches are in scope and include the Reactor Building Personnel Access Hatch (XRA-001), Emergency Escape Hatch (XRA-002) and Equipment Hatch (XRA-003).

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The 10 CFR 50 Appendix J Leak Rate Testing is a performance monitoring program.

Parameters Monitored or Inspected – The Type A ILRT and Type B LLRT tests detect or measure leakage across pressure retaining or leakage limiting boundaries other than valves. The Type A test is performed in accordance with STP-206.001, "Integrated Leak Rate Test" to measure the overall primary containment integrated leakage rate, which is obtained by summing leakage through all potential leakage paths, including containment welds, valves, fittings, and components that penetrate containment. Type B LLRT of primary containment penetrations is performed in accordance with STP-215.006, "Penetration Type B Leakage Testing". Type B LLRT for the primary containment hatches is performed in accordance with STP-215.001A, "Reactor Building Personnel Airlock test"; STP-215.001B, "Reactor Building Personnel Escape Airlock test" and STP-215.001C, "Reactor Building Equipment hatch Test". For VCSNS Type B LLRT tests include:

- Containment penetrations listed in FSAR Table 6.2-53
- Seals including door operating mechanism penetrations which are part of the primary containment
- Doors (except for seal welded doors) and hatches with resilient seals or gaskets

Detection of Aging Effects – Type A ILRT and Type B LLRT programs detect aging effects prior to loss of component intended function.

Monitoring and Trending – The 10 CFR 50 Appendix J, Option B LRT program is performed in compliance with Technical Specifications Surveillance Requirement 4.6.1.2. Type A ILRT and Type B LLRT are performed using technical requirements delineated in ANSI/ANS 56.8-1994 as stated in GTP-315.

VCSNS 10 CFR 50 Appendix J Leak Rate Testing is effective in detecting degradation of containment shells, liners, and components that compromise the containment pressure boundary, including seals and gaskets. While the calculation of leakage rates demonstrates the leak-tightness and structural integrity of the containment, it does not by itself provide information that would indicate that aging degradation has initiated or that the capacity of the containment may have been reduced for other types of loads such as seismic loading. Because a LRT program is repeated throughout the operating period, the entire pressure boundary is monitored over time. Type A ILRT is performed every 10 years from the previous Type A ILRT in accordance with 10 CFR 50 Appendix J, Option B. VCSNS is in the second Type A ILRT interval. The most recent Type A ILRT was performed during RF-8 in 1993. A Type B LLRT is performed every 30 months in accordance with 10 CFR 50 Appendix J, Option B. Procedure STP-215.006 states that a Type B LLRT is performed each refueling outage. Procedure GTP-315 states that the testing intervals shall be 30 months. Regulatory Guide 1.163, Section C.2 permits extended intervals greater than 60 months or 3 refueling cycles (whichever interval is shorter) under 10 CFR 50 Appendix J, Option B Type B LLRT. Each hatch is subject to a Type B LLRT at least every 30 months as stated in GTP-315. Outage Penetrations (XRP-600, XRP-602, XRP-505) and Fuel Transfer Tube Penetration (XRP-107) are tested every refueling outage. No actions are taken as part of this program to trend inspection or test results.

Acceptance Criteria – The acceptance criteria for the Type A ILRT that is in accordance with GTP-315 are as follows:

- As-found primary containment leakage rates less than or equal to 1.0 L_a, where L_a equals 0.20 % weight of the containment air per 24 hours at P_a equal to 45.1 psig.
- Prior to entering a mode where containment integrity is required, the as-left leakage rate shall be less than or equal to 0.75 L_a.

Acceptance criteria for the Type B LLRT for containment penetrations are in accordance with GTP-315. Acceptance criteria for the Type B LLRT for containment penetrations is as follows:

- Prior to entering a mode where containment integrity is required, the as-left leakage rate shall be less than 0.6 L_a.

Acceptance criteria for the Type B LLRT for containment hatches are in accordance with GTP-315. Acceptance criteria for the Type B LLRT for containment hatches is as follows:

- Overall airlock leakage rates less than or equal to 0.10 L_a at P_a equal to 45.1 psig.

Corrective Actions – A Condition Evaluation Report is initiated for any structure or component that does not meet the acceptance criteria. The CER initiates an engineering evaluation in accordance with GTP-315. The CER process is discussed in Station Administrative Procedure, SAP-1131 [Reference 160], "Electronic Processing of Condition Evaluation Reports". Repair / replacement of any structure or component is in accordance with GTP-315. Specific corrective actions are implemented in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The 10 CFR 50 Appendix J, Option B LRT program (in compliance with Technical Specifications Surveillance Requirement 4.6.1.2) is effective in preventing unacceptable leakage through the containment pressure boundary. Results of the LRT program are documented as described in 10 CFR 50, Appendix J to demonstrate that the acceptance criteria for leakage is satisfied. The test results that exceed the performance criteria must be assessed under 10 CFR 50.72 and 10 CFR 50.73. Type A ILRT is implemented by STP-206.001. Type B LLRT for penetrations is implemented by STP-215.006. LRT of the Reactor Building Personnel Access Hatch (XRA-001) is performed in accordance with STP-215.001A. LRT of the Reactor Building Emergency Escape Hatch (XRA-002) is performed in accordance with STP-215.001B. LRT of the Reactor Building Equipment Access Hatch (XRA-003) is performed in accordance with STP-215.001C.

Operating Experience – NRC Inspection Report 50-395 / 93-09 [Reference 91] concluded that STP-206.001 provides proper guidance and satisfies regulatory requirements to perform a Type A ILRT. Successful completion of the Type A ILRT during RF-8 (in accordance with STP-206.001) is documented in Surveillance Test Task Sheet (STTS) 0046167 and satisfies the commitment of Technical Specifications Surveillance Requirement 4.6.1.2.

NRC Inspection Report 50-395 / 94-25 [Reference 92] observed implementation of STP-215.006 to perform a Type B LLRT and verified that all Type B LLRT tests were performed punctually and adequately. Over three refueling cycles (most recently RF-10, RF-11, and RF-12), Type B penetrations delineated in STP-215.006 were tested and the results were satisfactory.

NCN 99-0489 documents 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. Visual examination and an Ultrasonic Test (UT) demonstrated that the liner plate had not degraded [Reference 93]. This NCN reports normal surface life exposure and is not aging related.

No Condition Evaluation Reports were initiated subsequent to the most recent Type B Containment Leak Rate Tests:

- Leak Rate Tests (LRT) for the Reactor Building Personnel Access Hatch (XRA-001) in accordance with STP-215.001A are satisfactory.
- LRT for the Reactor Building Emergency Escape Hatch (XRA-002) in accordance with STP-215.001B are satisfactory.
- LRT for the Reactor Building Equipment Hatch (XRA-003) in accordance with STP-215.001C are satisfactory.

LER 84-012 was initiated for not performing a LLRT on the Reactor Building Emergency Escape Hatch (XRA-002) within 72 hours of opening the hatch. LER 83-134 was initiated because the Personnel Air Lock did not close properly. Neither LER is an aging related issue.

No NCNs or CERs were initiated subsequent to the most recent Type B Leak Rate Test for containment hatches:

- The Reactor Building Personnel Access Hatch (XRA-001) seal inspections in accordance with Mechanical Maintenance Procedure MMP-515.002 are satisfactory.
- The Reactor Building Emergency Escape Hatch (XRA-002) seal inspections in accordance with MMP-515.004 are satisfactory.
- The Reactor Building Equipment Hatch (XRA-003) seal inspections in accordance with MMP-515.003 are satisfactory.

No NCNs or CERs were initiated subsequent to inspections of hatch seals.

Based on operating experience, continued implementation of the VCSNS 10 CFR 50 Appendix J Leak Rate Testing Program (established by Technical Specification Surveillance Requirement 4.6.1.2 and implemented by plant procedures) manages the identified effects of aging. The VCSNS Procedures are: Type A ILRT implemented by STP-206.001, Type B LLRT for penetrations implemented by STP-215.006, and Type B LLRT for the Containment Hatches implemented by STP-215.001A, STP-215.001B, and STP-215.001C. Continued implementation of the VCSNS 10 CFR 50 Appendix J Leak Rate Testing Program throughout the period of extended operation maintains the intended function of containment.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS 10 CFR 50 Appendix J Leak Rate Testing Program has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding 10 CFR 50 Appendix J Leak Rate Testing is consistent with the VCSNS program. The VCSNS 10 CFR 50 Appendix J Leak Rate Testing Program contains those attributes delineated in GALL Chapter XI.S4 which have been determined by the NRC to provide an acceptable aging management program.

7.3 ASME Section XI ISI Program – IWF

The ASME Section XI [Reference 94], Subsection IWF Inservice Inspection (ISI) program is credited with managing potential loss of material for ASME Class 1, 2, and 3 piping supports (not including shock suppressors) and ASME Class 1, 2, and 3 major equipment supports. Loss of material due to corrosion is an aging effect requiring programmatic management for ASME Class 1, 2, and 3 component supports for the extended period of operation. Cracking due to stress corrosion is an aging effect requiring programmatic management for high strength anchorage of ASME Class 1 component supports for the extended period of operation. The ASME Section XI, Subsection IWF ISI program is described in VCSNS Inservice Examination Program Plan, ISE-3 [Reference 95], Section 6.0. ASME Section XI, Subsection IWF ISI program is credited with managing potential loss of material for the ASME Class 1, 2, and 3 piping supports and ASME Class 1, 2, and 3 major equipment supports. ASME Section XI, Subsection IWF ISI program is credited with managing potential cracking due to stress corrosion of high strength anchorage for Class 1 component supports.

10 CFR 50.55a imposes the ISI requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 component supports. Inservice inspection of supports for ASME components is addressed in ASME Section XI, Subsection IWF. This evaluation covers the 1989 Edition as approved in 10 CFR 50.55a. ASME Section XI, Subsection IWF constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, and 3 component supports for license renewal.

Subsection IWF scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Discovery of support deficiencies during scheduled inspections triggers an increase of the inspection scope, in order to ensure that the full extent of deficiencies is identified. Degradation that potentially compromises support function or load capacity is identified for evaluation. IWF specifies acceptance criteria and corrective actions. Supports requiring corrective actions are re-examined during the next inspection period.

The ASME Section XI ISI Program – IWF for Class 1, 2, and 3 component supports, and support anchorage is evaluated in accordance with the guidance provided in NEI 95-10 [Reference 4] and the Generic Aging Lessons Learned (GALL) Report [Reference 5]. ASME Section XI, Subsection IWF ISI program for Class 1, 2, and 3 component supports and support anchorage provides reasonable assurance that the effects of aging are adequately managed for the components so that their intended functions are maintained consistent with the CLB for the period of extended operation.

Scope of Program – ISE-3, Section 6.0 discusses inservice inspection of ASME Code Class 1, 2, and 3 component supports. ISE-3, Section 6.1 discusses inservice inspection of ASME Code Class 1, 2, and 3 pipe supports. ISE-3, Section 6.2 discusses inservice inspection of ASME Code Class 1, 2, and 3 major equipment supports.

Scope for Piping Supports:

ASME CODE CLASS	PIPING SUPPORT POPULATION INSPECTED DURING SECOND INTERVAL
1	86 (25% of 341)VCSNS nonexempt Class 1 piping supports shall be visually examined
2	113 (15% of 745) VCSNS nonexempt Class 2 piping supports shall be visually examined
3	62 (10% of 619) VCSNS nonexempt Class 3 piping supports shall be visually examined

Scope for Major Equipment Supports:

ASME CODE CLASS	MAJOR EQUIPMENT SUPPORT POPULATION INSPECTED DURING SECOND INTERVAL
1	Three (3) supports shall be visually examined from a total of three Reactor Coolant Pump A,B,C supports; three Steam Generator A,B,C supports and Pressurizer support with welded attachment
2	Three (3) supports shall be visually examined from a total of Three Charging Pump Supports (XPP-043A, B & C), two RHR Pump Supports (XPP-031A & B), and two Reactor Building Spray Pump Supports (XPP-038A & B)
3	Thirteen (13) supports shall be visually examined from Total of 29 supports delineated in ISE-3, Section 6.2

The reactor vessel supports (XQR-001) shown on drawing 1MS-07-153 (sheet 1) are not in the ISI Program per exemption noted in ASME Section XI Subsection IWF-1230. The only parts of the reactor vessel supports that are visible are the very top piece, the bearing plate. The remainder of each reactor vessel support is encased in concrete.

Class 1 major equipment support anchor bolts are fabricated from ASTM A490 material based upon drawing 1MS-07-153 (sheet 1, note 9). Minimum ultimate strength of ASTM A490 is 150,000 psi. Anchor bolts for Class 1 major equipment supports are as follows:

CLASS 1 MAJOR EQUIPMENT SUPPORT	ANCHOR BOLTS ARE SHOWN ON 1MS-07-153
Steam Generator Upper Lateral Supports (XQG-002)	2 inch diameter from Sheet 8 Detail S
Steam Generator Upper Lateral Supports (XQG-002)	2 inch diameter from Sheet 9 Detail I and Sheet 10 Section M-M
Steam Generator Lower Lateral Supports (XQG-002)	1-1/2 inch diameter from Sheet 6 Detail 8 or 9
Steam Generator Column Base (XQG-002)	4 inch diameter from Sheet 2
Reactor Coolant Pump Tie Rods (XQG-002)	2-1/2 inch diameter from Sheet 7 View C-C
Reactor Coolant Pump Column Base (XQG-002)	4 inch diameter from Sheet 2
Upper Pressurizer Support (XQQ-001)	1-1/2 inch diameter from Sheet 11 Detail J
Lower Pressurizer Support (XQQ-001)	1-1/2 inch diameter from Sheet 11 Section C-C
Crossover Leg Supports (XPH-001)	1-1/2 inch diameter from Sheet 14 Detail A and Section G-G
Crossover Leg Supports (XPH-001)	2 inch diameter from Sheet 13 Details 1 and 2

ISE-3 lists Class 1 systems as Reactor Coolant (RC), Safety Injection (SI), and Chemical & Volume Control (CS). WCAP-9803 [Reference 96] Section 11 lists Class 1 pipe supports for auxiliary piping associated with Class 1 systems. Engineering drawing series S-321-601, S-321-641, S-321-671, and S-321-691 indicate that Hilti Kwik bolts are used to anchor Class 1 piping supports for auxiliary piping associated with Class 1 systems.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The ASME Section XI ISI Program – IWF is a condition monitoring program.

Parameters Monitored or Inspected – IWF specifies visual examination (VT-3) of supports. The parameters monitored or inspected include evidence of corrosion; indication of deformation; indication of misalignment; proper clearances; proper spring settings; indication of damage to close tolerance machined or sliding surfaces; missing, detached, or loose support items; and cracks. Table IWF-2500-1 (1989 edition) specifies examination of the following:

- (F1.10) Mechanical connections to pressure-retaining components and building structure
- (F1.20) Weld connections to building structure
- (F1.30) Weld and mechanical connections for intermediate joints in multi-connected integral and non-integral supports
- (F1.40) Clearances of guides and stops, alignment of supports, and proper assembly
- (F1.50) Spring supports and constant load supports

- (F1.60) Sliding surfaces
- (F1.70) Hot or cold position of spring supports and constant load support

Anchor bolts are monitored using visual inspections. Anchorage for Class 1 component supports are inspected for "loose or missing" parts and for "functional adequacy".

Detection of Aging Effects – VT-3 visual examination is specified in Table IWF-2500-1. Complete scope is repeated every 10 year inspection interval. The qualified VT-3 inspector uses judgment in assessing general corrosion and observed degradation is documented if loss of structural capacity is suspected. Implementing inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 piping and major equipment supports detects aging effects prior to loss of intended function.

Monitoring and Trending – 10 CFR 50.55a imposes the ISI requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 piping and major equipment supports. The Scope for the ASME Section XI Subsection IWF ISI Program is performed during the VCSNS second ISI interval (January 1, 1994 through December 31, 2003). There is no requirement to monitor or report progressive, time-dependent degradation. Unacceptable conditions, according to IWF-3400, are noted for repair or further evaluation.

Acceptance Criteria – The acceptance standards for visual examination are specified in ASME Subsection IWF-3400. IWF-3410 (b)(5), stating "roughness or general corrosion which does not reduce the load bearing capacity of the support" is given as an example of a "non-relevant condition," that requires no further action.

IWF-3410 (a) identifies the following conditions as unacceptable:

- Deformation or structural degradation of fasteners, springs, clamps, or other items
- Missing, detached, or loose items
- Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on finely machined or sliding surfaces
- Improper hot or cold positions of spring supports and constant load supports
- Misalignment of supports
- Improper clearances of guides and stops

The acceptance criteria for anchor bolts are no cracking or failure. Cracking or loss of bolt head is sufficient to indicate stress corrosion cracking in anchor bolts.

Corrective Actions – In accordance with IWF-3122, supports containing unacceptable conditions are evaluated or tested, or corrected before returning to service. Corrective actions are delineated in IWF-3122.2. IWF-3122.3 provides an alternative for evaluation or testing, to substantiate structural integrity and/or functionality. Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – 10 CFR 50.55a imposes the ISI requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 piping and major equipment supports. Applying requirements of 10 CFR 50, Appendix B, is acceptable to address administrative controls.

Operating Experience – ASME Class 1, 2, and 3 component support inspections over the past five years identified one case where acceptance criteria was not complied with. NCN 97-1147 documents a gap at the top of a pipe support in excess of 0.125 inch. This condition is not aging related. Operating experience at other nuclear facilities shows that improperly heat-treated anchor bolts are susceptible to stress corrosion cracking [Reference 97]. At VCSNS, ASTM A490 anchor bolt material is properly heat-treated by conforming to ASTM Specification A490 through a Certified Material Test Report (CMTR) in accordance with Specification SP-527 [Reference 98]. The Safety Evaluation Report for WCAP-14422 [Reference 70] Section 3.3.1.1 states "In the absence of a high level of sustained tensile stress, stress corrosion cracking is not likely to occur". ASTM A490 anchor bolts are not highly pre-loaded and do not have a high level of sustained tensile loads due to lower LOCA applied loads as a result of the elimination of the dynamic effects of postulated High Energy Line Break (HELB) of the Reactor Coolant System - Primary Coolant Piping. Therefore stress corrosion cracking is not a significant aging effect for ASTM A490 anchor bolts for major equipment supports. Class 1 pipe supports use Hilti Kwik bolts as anchors. Hilti Kwik bolts are not susceptible to stress corrosion cracking since the tensile strength is 125,000 psi as stated in SP-527, which is below the threshold tensile strength level of 150,000 psi where SCC is a concern.

IWF sampling inspections are effective in managing aging effects for ASME Class 1, 2, and 3 supports. There is reasonable assurance that the IWF inspection program will be effective through the period of extended operation.

Licensee Event Reports (LER) were initiated for shock suppressors (snubbers). Snubbers are not within license renewal scope as stated in NEI-95-10, Appendix B, Item 25. See Section 6.8.3 of this report.

NRC Inspection Report (IR) 50-395 / 91-21 [Reference 99] cites VCSNS for a violation for not removing insulation on pipe supports prior to implementing ISI. VCSNS denied this violation on January 7, 1992 and appealed to the American Society of Mechanical Engineers (ASME) on January 16, 1992 for a Code interpretation. ASME stated removal of insulation on pipe supports prior to implementing ISI is not required provided the support either carries the weight of the component or serves as a structural restraint in compression even though the support may also be designed for thermal loads, seismic loads, and transient loads. ASME concurred with the VCSNS interpretation. The NRC withdrew this violation in IR 50-395 / 91-21 on September 3, 1992 subsequent to the ASME code interpretation. The issue in NRC IR 50-395 / 91-21 is not aging related.

Non-Conformance Notices (NCN) are as follows:

NCN	EQUIPMENT	SUBJECT	DISPOSITION
97-1147	CSH-0108	Gap at the top of support exceeds 0.125 inch	Shim gap at the top of support to meet acceptance criteria.
00-1517	XPP-030B	Surface corrosion on lateral support for the Loop B Reactor Coolant Pump due to leaking flange.	Lateral support for the Loop B Reactor Coolant Pump cleaned and painted. Surface corrosion in lieu of flaking or pitting is not damaging to structural integrity per Technical Work Record disposition of NCN 00-1517 dated October 29, 2000.
00-1870	XHE-0009	Rust on bare carbon steel anchor bolt for the Excess Letdown Heat Exchanger.	Rust on bare carbon steel anchor bolt for the Excess Letdown Heat Exchanger is considered normal. The carbon steel anchor bolt for the Excess Letdown Heat Exchanger is cleaned and painted in accordance with site procedures.

The above Non-Conformance Notice conditions are not aging related.

No Condition Evaluation Reports were initiated subsequent to ASME Class 1, 2, and 3 component support inspections.

10 CFR 50.55a imposes the ISI requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 piping and major equipment supports. Based on operating experience, the continued implementation of the ASME Class 1, 2 and 3 component supports examinations established by ASME Section XI, Subsection IWF and implemented by ISE-3 manages the effects of aging. Continued implementation of the ASME Class 1, 2 and 3 component supports examinations throughout the period of extended operation maintains the intended functions of the ASME Class 1, 2 and 3 component supports.

Generic Aging Lessons Learned (GALL) Comparison – VCSNS ASME Section XI – IWF program for ASME Class 1, 2 and 3 component supports has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding ASME Section XI, Subsection IWF inspection program is consistent with the VCSNS program. The VCSNS ASME Section XI, Subsection IWF Inspection Program for ASME Class 1, 2 and 3 component supports contains those attributes delineated in GALL Chapters III.B1.1.1 and III.B1.2.1 which have been determined by the NRC to provide an acceptable aging management program.

7.4 Battery Rack Inspection

Loss of material due to corrosion is an aging effect requiring programmatic management for steel battery racks for the extended period of operation. The following battery rack inspections are credited with managing loss of material due to corrosion that could impact the intended function of structural support.

- Electrical DC (ED) System – 125 VDC Instrumentation & Control (I & C) Battery (XBA-1A-ED and XBA-1B-ED) Vital Battery and Terminal Post Inspection
- Fire Service (FS) System – Diesel Fire Service Pump Battery (XPP-134B-FS) steel racks inspection

The regulatory basis for inspecting battery racks is found in the VCSNS Technical Specifications Surveillance Requirement 4.8.2.1.c for the ED System and commitment in Fire Protection Procedure, FPP-024 [Reference 100] for the FS System. Battery rack inspection is a condition monitoring program. Aging Management Programs are evaluated in accordance with the guidance provided in NEI 95-10 [Reference 4]. The aging management programs credited provide reasonable assurance that effects of aging are adequately managed for the battery racks so their intended function is maintained consistent with the CLB for the period of extended operation.

Scope of Program – The scope of the Battery Rack Inspection includes the battery racks for the following systems:

- ED System (Vital Batteries)
- FS System (Diesel Fire Service Pump Battery)

The regulatory basis for inspecting battery racks for the ED System is found in the VCSNS Technical Specifications Surveillance Requirement 4.8.2.1.c, while the regulatory basis for inspecting battery racks for the FS System is the commitment in FPP-024.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Battery Rack Inspection is a conditioning monitoring program.

Parameters Monitored or Inspected – The ED System and FS System battery racks specific examination guidelines are given in IEEE-450 [Reference 101]. For the ED System and FS System, battery racks are inspected for loss of material due to corrosion. Although not credited for license renewal, the battery racks are also inspected for physical damage. For the ED System, loss of material due to corrosion is a specific inspection attribute in Surveillance Test Procedures STP-501.003 and STP-501.004. For the FS System, loss of material due to corrosion is a specific inspection attribute in STP-505.003.

Detection of Aging Effects – The Battery Rack Inspection Program detects structural damage or degradation (including loss of material due to corrosion) prior to loss of structure intended function.

Monitoring and Trending – For the ED System, a visual examination is performed every 18 months in accordance with commitments in FSAR Section 8.3.2.2.2 and Technical Specifications Surveillance Requirement 4.8.2.1.c. Visual examination for structural integrity of racks (that includes an attribute for loss of material due to corrosion) is implemented in accordance with STP-501.003. Alternatively, as stated in Surveillance Test Procedure STP-501.004, visual examination may be performed every five years for structural integrity of the racks which includes loss of material due to corrosion. Visual examination for structural integrity of the racks is in accordance with Surveillance Test Procedure STP-501.004 when the five year option is implemented.

For the FS System, visual examination is performed every 18 months in accordance with a commitment in FPP-024. Visual examination for structural integrity of racks (that includes loss of material due to corrosion) is implemented in accordance with STP-505.003.

Results of 18 month battery rack inspections are retained in sufficient detail to permit adequate confirmation of the inspection program. In particular these records identify inspectors, results of the inspections, note discrepancies with the cause, and prescribe corrective action. No actions are taken as part of this program to trend inspection or test results.

Acceptance Criteria – For the ED System, the acceptance criterion is "no visual indication of loss of material due to corrosion" as stated in STP-501.003. For the FS System, acceptance criterion is "no visual indication of loss of material due to corrosion" as stated in STP-505.003.

Corrective Actions – For the ED System, STP-501.003 provides guidance when abnormalities (UNSAT conditions) are observed. Repair / replacement of unacceptable batteries or racks is in accordance with Electrical Maintenance Procedure EMP-115.005. For the FS System, STP-505.003 provide guidance when abnormalities (UNSAT conditions) are observed.

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process –Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – For the ED System, visual examination for structural integrity of racks, that includes an attribute for loss of material due to corrosion, is implemented in accordance with STP-501.003. Visual examination of the ED System is performed to comply with commitments in FSAR Section 8.3.2.2.2 and Technical Specifications Surveillance Requirement 4.8.2.1.c. For the FS System, visual examination for structural integrity of racks, that includes an attribute for loss of material due to corrosion, is implemented in accordance with STP-505.003. Visual examination of the FS System is performed to comply with the commitment in FPP-024.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – Visual inspection of the battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade performance. The presence of physical damage or deterioration does not necessarily represent a failure, provided an evaluation determines that the physical damage or deterioration does not affect the ability of the battery to perform its function. Review of work orders for the past five years did not identify any instance where abnormal deterioration of battery racks occurred.

Licensee Event Reports (LER) associated with batteries were reviewed. LERs document missed weekly surveillances or electrical test deficiencies that were not aging related. Inspections were performed punctually and were satisfactory. No Non-Conformance Notices or Condition Evaluation Reports were initiated subsequent to inspections for battery racks.

Operating experience provides reasonable assurance that the continued implementation of the Battery Rack Inspection Program manages the loss of material throughout the period of extended operation. Battery rack inspection is established for the ED System by commitments in FSAR Section 8.3.2.2.2 and Technical Specifications Surveillance Requirement 4.8.2.1.c and for the FS System by commitment in FPP-024. Inspections for the ED System racks are implemented by STP-501.003. Inspections for the FS System racks are implemented by STP-505.003 to maintain intended function of battery racks.

Generic Aging Lessons Learned (GALL) Comparison – The Battery Rack Inspection Program is not evaluated in the GALL Report [Reference 5] and is, therefore, plant specific to VCSNS.

7.5 Boraflex Monitoring Program

VCSNS Technical Specifications amendment request TSP 99-0090 has been submitted to the NRC in correspondence RC-01-0135 dated July 24, 2001 [Reference 158] to increase the spent fuel pool storage capacity by replacing the existing eleven high density storage rack modules with twelve high density storage racks. Holtec International Report HI-20126624 [Reference 102] documents the design adequacy of twelve high-density storage racks, and replaces Boraflex with Boral to improve neutron absorbing material longevity. The Holtec International Report states Boral does not degrade, as a result of long-term exposure to radiation, and Boral is stable, durable, and corrosion resistant. The NRC Staff Action Plan for Spent Fuel Pool Safety Issues [Reference 103] concludes degradation of neutron absorption performance has not been observed in materials other than Boraflex. Therefore the existing Boraflex Monitoring Program ceases with the implementation of Technical Specifications amendment request TSP 99-0090; however, further discussion is provided for completeness of the current program.

Sample coupon tests of Boraflex neutron absorber material comply with the long term surveillance requirements delineated in the licensing report on the current Spent Fuel Racks [Reference 104]. The sample coupon test for Boraflex neutron absorber material is a performance monitoring program. Sample coupon tests of Boraflex neutron absorber material monitor physical integrity (excessive swelling and hardness) and neutron absorption properties (boron density). Aging management programs are evaluated in accordance with the guidance provided in NEI 95-10 [Reference 4]. The aging management programs credited for license renewal provide reasonable assurance that effects of aging are adequately managed for Boraflex neutron absorber material in the spent fuel racks so their intended function is maintained consistent with the CLB for the period of extended operation.

Scope of Program – Sample coupon tests of Boraflex neutron absorber material are performed to comply with long term surveillance requirements delineated in the licensing report on the current Spent Fuel Racks. High density spent fuel racks are discussed in the Design Basis Document for the Spent Fuel Cooling (SF) System [Reference 105]. Spent fuel pool regions are discussed in the licensing report on the current Spent Fuel Racks. Spent fuel pool regions 1 and 2 are shown in the licensing report Figures 4.3 and 4.4, respectively. Spent fuel pool region 1 is designated for storage of new or freshly discharged fuel assemblies with a maximum initial enrichment of 5 weight percent U-235. Spent fuel pool region 2 is designated for fuel assemblies with a maximum initial enrichment of 2.5 weight percent U-235 with no burnup and up to 5 weight percent U-235 with a minimum burnup (21,600 MWD per MTU). High density spent fuel racks use Boraflex as a neutron absorber material. Boraflex neutron absorber material 0.082 inches thick (XNF-047A) is placed in region 1 in a Type A Module. Boraflex neutron absorber material 0.032 inches thick (XNF-047B) is placed in region 2 in a Type B Module. Boraflex neutron absorber material is shown on vendor (Joseph Oates) drawing 1MS-15-148 identified as "specimen". Each assembly contains 18 coupons (specimens). Two assemblies are required for region 1 and two assemblies are required for region 2. There are 72 coupons (specimens) of Boraflex neutron absorber material.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. Sample coupon tests for Boraflex neutron absorber material represent a performance monitoring program.

Parameters Monitored or Inspected – Boraflex neutron absorber material is tested to ensure excessive swelling does not occur. Excessive swelling is detected by dimensional measurements. Percent change in length, width, thickness, hardness, and density is determined. Boraflex neutron absorber material is tested to maintain adequate neutron attenuation since Boraflex neutron absorber material becomes very brittle subsequent to neutron irradiation. Testing of Boraflex neutron absorber material is in accordance with Reactor Engineering Procedure, REP-108.001.

Detection of Aging Effects – Testing of Boraflex neutron absorber material will detect excessive swelling and neutron attenuation deterioration prior to loss of intended function.

Monitoring and Trending – Long Term Surveillance per commitment in the licensing report on the current Spent Fuel Racks is delineated in REP-108.001. The long-term surveillance requirement is to test two coupons every five years. The test schedule is in the licensing report [Reference 104]. No actions are taken as part of this program to trend inspection or test results.

Acceptance Criteria – Acceptance criteria is in accordance with REP-108.001. Dimensional changes should not exceed 5%. Density should not change by more than 20%. Hardness should be greater than 90% of the original value. Neutron attenuation is measured as stated in REP-108.001. Minimum areal density of boron should be greater than or equal to 0.0220 g per cm^2 for region 1 and greater than or equal to 0.0015 g per cm^2 for region 2.

Corrective Actions – REP-108.001 provides guidance when unsatisfactory (UNSAT) test results occur. If excessive swelling occurs, Plant Test Procedure PTP-130.003 is invoked for the current Spent Fuel Racks to ensure dimensional changes are not induced in the racks to cause excessive drag forces.

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – Long Term Surveillance Requirements are implemented by REP-108.001. Completing the Surveillance Test Task Sheets (STTS) documents Boraflex coupon tests.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – The results of previous tests are:

STTS	COUPON	RESULTS OF TEST
S0000089	XNF-047A	During RF-11 – Specimens 5 and 6 – All Acceptance Criteria Satisfied - 3- 19-01
S0051538	XNF-047B	During RF-8 – Specimens 33 and 34 – All Acceptance Criteria Satisfied - 2-21-95
S0055599	XNF-047A	During RF-8 – Specimens 3 and 4 – All Acceptance Criteria Satisfied - 2-21-95
S0050027	XNF-047B	During RF-7 – Specimens 31 and 32 – All Acceptance Criteria Satisfied - 10-15-93
S0039953	XNF-047B	During RF-6 – Specimens 29 and 30 – All Acceptance Criteria Satisfied - 2-14-92
S0034500	XNF-047B	During RF-5 – Specimens 27 and 28 – All Acceptance Criteria Satisfied - 8-4-90
S0034499	XNF-047A	During RF-5 – Specimens 1 and 2 – All Acceptance Criteria Satisfied - 8-4-90

NRC Inspection Report 50-395 / 93-25 documents review of 14 specimens and concluded that the acceptance criteria are satisfied.

No NCNs or CERs were initiated for the Boraflex Monitoring Program based upon a Document Management System (DMS) search.

The parameters monitored include physical conditions of Boraflex panels, such as gap formation and decreased boron areal density. Concentration of silica in the spent fuel pool is not part of the test. These are conditions directly related to degradation of the Boraflex material. When Boraflex is subjected to gamma radiation and long term exposure to the spent fuel pool environment, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released into the spent fuel pool water. As indicated in Information Notice (IN) 95-38 and Generic Letter (GL) 96-04, the loss of boron carbide (washout) from

Boraflex is characterized by slow dissolution of silica from the surface of the Boraflex and a gradual thinning of the material. Because Boraflex contains about 25% silica, 25% polydimethyl siloxane polymer, and 50% boron carbide, sampling and analysis of the presence of silica in the spent fuel pool provides an indication of depletion of boron carbide from Boraflex; however, the degree to which Boraflex has degraded is ascertained through measurement of the boron areal density.

Based on Boraflex coupon tests, continued reliance on long term surveillance requirements manages Boraflex neutron absorber material aging effects and maintains its intended function throughout the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison –The VCSNS Boraflex Monitoring Program has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in GALL Chapter XI.M22 is consistent with the VCSNS program with the following exception.

VCSNS does perform sampling and analysis of the presence of silica in the spent fuel pool which provides an indication of depletion of boron carbide from Boraflex; however, VCSNS does not correlate boron areal density to silica level. VCSNS Boraflex panels are not susceptible to flow corrosion since the panels are enclosed, thereby limits flow through the panels. Based on VCSNS experience and results of past coupon testing, Boraflex degradation has not been an issue.

7.6 Boric Acid Corrosion Surveillances

The Boric Acid Corrosion Surveillances conducted at VCSNS are comprised into two separate programs. The first is the program established to satisfy the requirements of Generic Letter (GL) 88-05 to monitor the condition of the Reactor Coolant System (RCS) pressure boundary for borated coolant leakage. Each time the reactor is shutdown for refueling a visual inspection of the RCS is performed prior to RCS cooldown and depressurization, and before removal of insulation in accordance with Health Physics Procedure, HPP-402, "Radiological Survey Requirements and Controls for Reactor Building and Incore Pit Entries". After RCS cooldown (< 200°F) and depressurization and removal of insulation, but prior to decontamination of the piping/component to be inspected, a visual inspection is conducted in accordance with Surveillance Test Procedure, STP-250.001A, "Reactor Coolant System Leak Test". These inspections are performed to detect any signs of boric acid deposits or residue that would indicate a leak.

The second program demonstrates the integrity of systems, which contain boron, when performing an ASME Code, Section XI system pressure test at normal pressure per Technical Specifications Surveillance Requirement 4.0.5. The following list of surveillance test procedures provide specific guidance to look at bolted connections and joints (where borated water leakage is most likely to occur) for signs of boric acid induced corrosion, although they are performed more to satisfy the ASME Code requirements of system integrity.

- STP-250.001A, Reactor Coolant System Leakage/Pressure Test
- STP-250.002, Safety Injection/Chemical and Volume Control Leak Test
- STP-250.003A, Safety Injection Accumulator Leak Test
- STP-250.004A, Residual Heat Removal System Plant Cooldown Piping Leak
- STP-250.006A, Reactor Building Spray System Leak Test
- STP-250.015, Spent Fuel Cooling System Leak Test

The Boric Acid Corrosion Surveillances for VCSNS were evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provided reasonable assurances that the effects of boric acid induced corrosion are adequately managed so that there is no degradation of the leakage source or adjacent structures and components during the extended period of operation. The Boric Acid Corrosion Surveillances are condition monitoring and mitigation programs. Minimizing leakage from systems that contain boron by periodic monitoring of the locations where potential leakage may occur, and timely repair if leakage is detected, prevents and/or mitigates boric acid corrosion.

Scope of Program – Boric Acid Corrosion Surveillances are credited with managing the aging effects (i.e., loss of material) due to aggressive chemical attack of carbon steel, low-alloy steel, and other susceptible materials. The programs encompass structural components including bolting and electrical structural components (e.g., cable trays, conduit, connectors, enclosures) that are located in the Reactor Building, Auxiliary Building, Intermediate Building, and Fuel Handling Building where exposure to leakage from borated water systems is possible.

Preventive Actions – Programmatic oversight of boric acid leakage is accomplished through systematic visual inspections and corrective actions to ensure that boric acid corrosion does not lead to degradation of the leakage source or adjacent components, supports, or structures. The removal of concentrated boric acid and boric acid residue and the elimination of boric acid leakage mitigate corrosion by minimizing the exposure of the susceptible material to the corrosive environment.

Parameters Monitored or Inspected – Typical plant systems using borated water are the Reactor Coolant System, Safety Injection System, Spent Fuel Cooling System, and Residual Heat Removal System. Components in these systems, including adjacent components and structures are visually inspected to detect conditions leading to boric acid corrosion, such as boron crystal buildup and evidence of moisture.

Detection of Aging Effects – The aging effect (i.e., corrosion) associated with boric acid leakage is minimized by periodic visual inspection of the systems containing boron. Conditions leading to boric acid corrosion, such as white crystal buildup and the presence of moisture are readily detectable by visual examination. The Boric Acid Corrosion Surveillances at VCSNS assure that degradation of the leakage source or adjacent components or structures due to boric acid induced corrosion will be detected prior to the loss of any intended function(s) by locating small leaks, conducting examinations, and performing engineering evaluations.

Monitoring and Trending – A visual inspection of the RCS pressure boundary is performed each refueling outage in accordance with HPP-402 and STP-250.001A to satisfy the requirements delineated in GL 88-05. Other systems containing boron are visually inspected for signs of boric acid leakage every 3-1/3 years in accordance with STP-250.002, STP-250.003A, STP-250.004A, STP-250.006A, and STP-250.015. Inspection results are documented and an engineering evaluation is conducted for any unsatisfactory condition.

An overall plant leakage reduction program to reduce the number of leaks and the amount of leakage to as low as practical is addressed in General Maintenance Procedure, GMP-114.002. This program ensures leakage is identified, tracked and trended, prioritized, and corrected in a timely manner.

Periodic assessment of leaks, which do not warrant repair at the time of initial identification, is in accordance with GMP-100.020.

Acceptance Criteria – External surfaces of components and structures that contain borated water, including their surroundings (e.g., adjacent components or structures, including electrical), are expected to be free from pitting and corrosion, abnormal discoloration or accumulated residues that may be evidence of leakage from nearby borated water systems. Acceptance criteria and inspection guidelines for the examination and evaluation of components and structures that contain borated water for evidence of boric acid leakage are delineated in the following procedures:

- STP-250.001A
- STP-250.002
- STP-250.015

Corrective Actions – Leaks involving boric acid residues are documented and examined to determine the extent of degradation, if any, that may have been caused by boric acid induced corrosion on the leakage source or adjacent structures and components. If it is determined that an unsatisfactory condition exists, a Condition Evaluation Report (CER) is initiated in accordance with Station Administrative Procedure, SAP-1131, "Electronic Processing of Condition Evaluation Reports".

For leaks at bolted connections, a CER is generated and an engineering evaluation is performed to determine the susceptibility of the bolting to corrosion and potential failure. All leaks discovered during the visual inspection conducted in accordance with HPP-402 are documented on a CER. The CER and engineering evaluation processes will determine whether the component / condition is acceptable for continued service or repair or replacement is required. Additional preventive measures may include evaluations and trending of leaks, which may lead to design modifications (1) to reduce the probability of leaks at locations where they may cause corrosion damage, (2) which use suitable corrosion resistant materials, or (3) for application of protective coatings.

Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The Boric Acid Corrosion Surveillances include recommendations of GL 88-05 and are implemented through the following procedures for the RCS pressure boundary:

- HPP-402, Radiological Survey Requirements and Controls for Reactor Building and Incore Pit Entries
- STP-250.001A, Reactor Coolant System Leakage/Pressure Test

The procedures listed below are credited with detecting boric acid leakage from systems containing boron, other than the RCS. Although they are generally performed to satisfy the ASME Code requirement of system integrity, there is specific guidance to look at bolted connections and joints where boric acid leakage is most likely to occur:

- STP-250.002, Safety Injection/Chemical and Volume Control Leak Test
- STP-250.003A, Safety Injection Accumulator Leak Test
- STP-250.004A, Residual Heat Removal System Plant Cooldown Piping Leak
- STP-250.006A Reactor Building Spray System Leak Test
- STP-250.015, Spent Fuel Cooling System Leak Test

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – The Boric Acid Corrosion Surveillances were originally implemented as a result of GL 88-05. The VCSNS initial program for boric acid inspections was contained in Preventive Test Procedure, PTP-151.001, "Inspection for Boric Acid Corrosion". However, in October 1999 a series of discussions involving plant personnel was initiated in order to reduce dose rates and optimize efficiency of visual inspections inside containment for boric acid leakage. In addition to the inspections conducted in accordance with PTP-151.001, health physics general area and AMSE Section XI inspections were also being performed causing duplication and overlap. As a result of these discussions the following procedural changes were made:

- PTP-151.001 was canceled 5-16-00 and later replaced by PTP-151.001A on 4-15-02.
- HPP-402, Revision 10, was issued 6-20-00 and STP-250.001A, Revision 0, was issued 3-5-02 to incorporate the commitment to GL 88-05 for the requirement to perform boric acid leakage inspections.
- HPP-402 was issued containing no specific inspection locations, but as a general area inspection procedure. STP-250.001A contained the same specific locations for inspection as was contained in PTP-151.001A.

The present Boric Acid Corrosion Surveillances adequately address the GL 88-05 requirements, including:

- Detection of the principle locations where leaks that are smaller than the allowable Technical Specifications limits could cause degradation.
- Methods for conducting examinations that are integrated into ASME Code VT-2 inspections conducted during systems pressure tests.
- Engineering evaluations and corrective actions to correct and prevent recurrences of this type of leakage.

The programs have been successful in managing the loss of material due to boric acid induced corrosion. It has provided for timely identification of leakage and implementation of corrective actions. Since establishing the program, there have been no instances of boric acid corrosion that have impacted components, structures, or systems from performing their intended functions. The following example illustrates the capability of the program:

> While performing visual inspections of the Reactor Building in accordance with HPP-402 during RF-12 in October 2000, a significant quantity of boric acid deposits was discovered on the floor coming from the air boot of the loop "A" RCS hot leg penetration to the bio-shield wall. Consequently, Condition Evaluation Report 00-1324 was initiated to address the problem. Further investigation revealed that the deposits originated from the leaking of reactor coolant through the loop "A" hot leg reactor vessel nozzle-to-reactor coolant pipe welded joint. (Reference CER 00-1392 and NCN 00-1396).

As a result of this incident, SCE&G committed (Reference CER 00-1676 and letter RC-00-0377) to enhance the procedures for boric acid inspections, both to satisfy GL 88-05 and ASME Code, Section XI. VCSNS has now enhanced the boric acid inspection programs 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.

The surveillance test task sheets from the past five years (for the STP-250 series procedures credited for Boric Acid Corrosion Surveillances) have been reviewed. The majority of the leaks were identified as small or showing signs of previous leakage. All leaks were documented, cleaned, visually examined, and evaluated by engineering for continued service, repair, or replacement. No significant loss of material has been found on leaking components or on adjacent structures and components in the area of any leak.

The overall effectiveness of the Boric Acid Corrosion Surveillances is supported by the excellent operating experience for systems, structures, and components that are encompassed by the program. A review of the Boric Acid Corrosion

Surveillances conducted at VCSNS confirms the reasonableness and acceptability of the inspection frequency in that degradation of the components / structures will be detected prior to loss of function. The Boric Acid Corrosion Surveillances provide reasonable assurances that the aging effects associated with boric acid induced corrosion are managed so that the leakage source or adjacent structures and components will continue to perform their intended function(s) consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Boric Acid Corrosion Surveillances described in the SCE&G response to GL 88-05, have been compared to the "Boric Acid Corrosion" program documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding boric acid corrosion is consistent with the VCSNS program. The VCSNS Boric Acid Corrosion Surveillances contain those attributes delineated in GALL Chapter XI.M10, which have been determined by the NRC to provide an acceptable aging management program.

7.7 Chemistry Program

The VCSNS Chemistry Program manages the water chemistry in plant systems and structures to prevent and minimize loss of material, cracking, and fouling aging effects for primary and secondary systems, structures, and components. The Chemistry Program is based on Electric Power Research Institute (EPRI) guidelines TR-105714 [Reference 165] for primary water chemistry and TR-102134 [Reference 166] for secondary water chemistry. The aging effects are prevented or minimized by controlling the chemical species (e.g., chlorides, flourides, sulfides) or biological species (e.g., Corbicula sp., and Mytilus sp.) that cause the underlying mechanism(s) that result in these aging effects. Alternatively, chemical agents, such as corrosion inhibitors and biocides, are introduced to prevent certain aging mechanisms. The program includes sampling activities and analysis for primary, secondary, and borated treated water systems. The program provides assurance that an elevated level of contaminants and oxygen do not exist in the systems, structures, and components covered by the program, and thus, precludes the occurrences of aging effects and mitigates aging degradation.

The Spent Fuel Pool (SFP) is the only structure that is covered by the Chemistry Program. The SFP and the structural components within the SFP are constantly exposed to borated water. Boron plays a critical role as a chemical shim moderator. Water samples are drawn and analyzed as specified in Chemistry Procedure, CP-618, "Chemistry Specifications for Borated Systems and Tanks" [Reference 167]. Chemical specifications for the SFP water are listed on CP-618, Attachment I. The results of the analysis are documented in accordance with CP-602, "Chemistry Reporting" [Reference 168]. The parameters monitored by the Chemistry Program for the purpose of aging management are chloride, fluoride, and sulfate.

The Chemistry Program for the SFP has been evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provided reasonable assurance that the effects of aging are adequately managed for the SFP so that its intended function will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation. Sampling the water chemistry of the SFP (in accordance with CP-618) is a prevention and mitigation program.

Scope of Program – The portion of the Chemistry Program that is applicable to the SFP is credited with managing the aging effects (i.e., general corrosion, pitting and stress corrosion cracking) for the internal surfaces and structural components within the SFP. The SFP water is normally sampled once a week; however, during fuel movement or when the core is off-loaded into the SFP, sampling is required daily for boron concentration. Samples are drawn and analyzed in accordance with CP-618 and documented on Attachment VII of CP-602.

To preclude the aging effects associated with an alternate wetting and drying environment, the SFP water level is maintained and monitored in accordance with Technical Specifications Section 3/4.9.10.

Preventive Actions – Maintaining the SFP water chemistry specifications below the maximum acceptable limits listed on Attachment I of CP-618 will prevent or minimize the aging related effects (i.e., general corrosion, pitting and stress corrosion cracking) that are associated with the SFP. The chemistry specification limits for the SFP are based on acceptable industry standards and operating experience.

Parameters Monitored or Inspected – Water samples from the SFP are analyzed for the chemical specifications listed on Attachment I of CP-618. The parameters monitored for the purpose of preventing or minimizing the effects of aging for the SFP and structural components within the SFP are chloride, fluoride and sulfate. Makeup water added to the SFP is analyzed prior to being added. This includes the Refueling Water Storage Tank (RWST) and the Recycle Holdup Tank (RHT).

Detection of Aging Effects – The aging effects of concern for the SFP (i.e., general corrosion, pitting and stress corrosion cracking) are minimized by controlling the chemicals that cause the underlying aging mechanism (corrosion) that results in these aging effects. This is accomplished by weekly sampling of the SFP water chemistry to verify that the specifications listed on Attachment I of CP-618 are below the maximum limits. The water chemistry specifications for the SFP are measured utilizing standard proven industry techniques.

Monitoring and Trending – Weekly sampling of the SFP water chemistry in accordance with CP-618 ensures that the aging effects (i.e., general corrosion, pitting and stress corrosion cracking) associated with the SFP are prevented or minimized. Makeup water added to the SFP is analyzed prior to being added. This includes the RWST and the RHT.

SFP water samples are drawn and analyzed weekly (every 24 hour ± 4 hours for boron during fuel movement or when the core is off-loaded into the SFP). The results are documented on Attachment VII of CP-602, and entered into the "Chemistry Data Management System" computer database. Chemistry records are generated to provide a history file and for trend analysis.

Acceptance Criteria – The chemistry specification acceptance criteria for the SFP borated water are specified in CP-618, Attachment I. Makeup water added to the SFP is analyzed prior to being added.

Corrective Actions – Any Out-Of-Specification (OOS) condition for the SFP chemistry parameters are immediately reported to the Control Room and handled in accordance with CP-612, "Out-Of-Specification Handling and Reporting".

If it is determined that a Condition Evaluation Report (CER) is required to document the OOS condition, it will be generated in accordance with Station Administrative Procedure SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

If the OOS condition is caused by a plant design problem, an Engineering Change Request (ECR) will be initiated in accordance with SAP-133, "Design Control/Implementation and Interface".

If the OOS condition is caused by an equipment malfunction, a Maintenance Work Request (MWR) will be initiated per SAP-300, "Conduct of Maintenance".

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – SFP water samples are drawn and analyzed as specified in CP-618. The results are documented on Attachment VII of CP-602, and entered into the "Chemistry Data Management System" computer database. Any OOS condition for the SFP chemistry parameters are handled in accordance with CP-612.

Chemistry records are generated to provide a history file and for trend analysis. These records are retained and maintained in accordance with SAP-1340, "Transmittal and Maintenance of Records".

Operating Experience – The VCSNS Chemistry Program is an ongoing program that incorporates the best practices of industry organizations, vendors, utilities, and water treatment experts. The program provides assurance that the fluid environment to which the internal surfaces and structural components within the SFP are exposed will minimize corrosion. This is accomplished through effective monitoring of key parameters at established frequencies with well-defined acceptance criteria. The chemistry analyses are governed by the plant Quality Control Program to assure accurate results. Chemistry data is monitored for trends that might be indicative of an underlying operational problem.

A review of the SFP weekly sampling data for the past five years shows that the maximum acceptable limits for chloride (0.15 ppm), fluoride (0.15 ppm) and sulfate (0.100 ppm) have not been exceeded. Typically, chloride and fluoride concentrations are less than 10 ppb and sulfate ranges from less than 0.5 to 70 ppb, which are considerably lower than the maximum specification limits.

The Chemistry Program incorporates EPRI and Institute of Nuclear Power Operations (INPO) guideline documents as well as the "lessons learned" from South Carolina Electric and Gas (SCE&G) and external industry operating experience. The program has been subject to periodic internal and external assessment activities that help to maintain highly effective chemistry control, and facilitate continuous improvement. The overall effectiveness of the chemistry program is supported by the excellent operating experience for systems, structures, and components that are influenced by the program. No chemistry-related degradation associated with aging has resulted in the loss of the SFP intended function(s).

Analyzing and trending the water chemistry specifications for the SFP has been in effect since the initial implementation of the facility operating license at VCSNS and is considered acceptable based on industry operating experience. A review of the Chemical Program for the SFP conducted at VCSNS confirms the reasonableness and acceptability of the sampling frequency in that degradation of the structures / components will be detected prior to loss of function. The Chemistry Program provides reasonable assurances that the aging effects associated with corrosion (i.e., general corrosion, pitting and stress corrosion cracking) are managed so that the SFP will continue to perform its intended function(s) consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Chemistry Program for the Spent Fuel Pool has been compared to the "Water Chemistry" program documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding "Water Chemistry" is consistent with the VCSNS program. The VCSNS Chemistry Program for the Spent Fuel Pool contains those attributes delineated in GALL Chapter XI.M2, which have been determined by the NRC to provide an acceptable aging management program.

7.8 Containment Coating Monitoring and Maintenance Program

The Containment Coating Monitoring and Maintenance Program described in the SCE&G response to Generic Letter (GL) 98-04 is appropriate in meeting the regulatory requirements of Regulatory Guide (RG) 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants", Revision 0, dated June 1973.

Proper maintenance of protective coatings inside the containment, defined as Service Level I in RG 1.54, is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. This has been addressed in GL 98-04.

Maintenance of Service Level I coatings applied to carbon steel surfaces inside reactor containment (e.g., containment steel liner, containment penetrations, hatches) also serves to prevent or minimize loss of material due to corrosion.

Visual inspections of the Service Level I coatings inside containment are conducted periodically in accordance with ES-437, "Inspections for Maintenance Rule – Structures", ES-438, "Containment Inservice Inspection Program", QSP-506, "IWE and IWL Visual Examination", general maintenance planning and, to a limited degree, during recovery and restoration from refueling outages. The Containment Coating Monitoring and Maintenance Program for Service Level I coatings used inside the Reactor Building is a prevention and condition monitoring program.

The protective coatings used inside the Reactor Building are evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for Service Level I coatings so that their intended function(s) will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation.

Scope of Program – The scope of the containment coating inspections conducted in accordance with ES-437, ES-438 and QSP-506, includes the Service Level I protective coatings inside the Reactor Building, as defined in RG 1.54.

Preventive Actions – The periodic visual inspection of coatings inside the containment ensures that the aging related effect (i.e., loss of material due to corrosion) of carbon steel components is prevented or minimized from occurring. Prevention also includes ensuring that coatings inside the containment do not detach from their substratum and interfere with the operation of post-accident fluid systems and thereby impair safe shutdown.

Parameters Monitored or Inspected – Paint or coatings inside containment are inspected in accordance with ES-437, ES-438 and QSP-506 for the following visible defects:

- Blistering
- Peeling
- Cracking
- Flaking
- Discoloration
- Rusting

Detection of Aging Effects – VCSNS periodically conducts inspections and condition assessments of Service Level I coatings inside containment. Inspections are conducted as part of containment structural integrity verification, Maintenance Rule monitoring, general maintenance planning and, to a limited degree, during recovery and restoration from refueling outages. Coatings condition assessments are conducted on an as needed basis as part of nonconformance corrective actions. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled, usually in the following outage, for repair or replacement, as necessary. The periodic inspections, condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level I coatings that may be susceptible to detachment from the substrate during a design basis LOCA is minimized and therefore not interfere with the operation of the Emergency Core Cooling System (ECCS) and the Safety Related Containment Spray System (CSS).

Monitoring and Trending – Degradation of containment coatings are detected by periodic visual inspections conducted in accordance with the requirements of ES-437, ES-438 and QSP-506. Containment coatings are visually inspected via walkdowns from accessible floors, platforms or other permanent vantage points. The degree of examination depends on many factors such as accessibility, environmental and radiological conditions, and safety. In cases of inaccessibility, sampling approaches such as plant specific characteristics, industry wide experience and testing history may be evaluated instead of actual visual inspection in areas of similar environmental and/or service conditions.

Maintenance Rule inspection results are documented by the evaluating structural engineer(s) utilizing ES-437, Attachment II, which are assembled into a technical report. These results are compared and trended to previous inspection data.

Containment Inservice Inspection (CISI) results are documented utilizing QSP-506, Attachment I and entered into the CISI database, which is an electronic data management tool for tracking and scheduling of inspections required by ASME Section XI, Subsections IWE and IWL as modified by 10 CFR 50.55a. This tool allows the user to list components associated with the CISI program, schedule the components by outage within an interval and period, record and track examination results and create reports for activities such as 90 day outage reporting. A summary report generated by engineering provides an evaluation assessment of the CISI ASME Section XI, Subsections IWE and IWL inspections.

Acceptance Criteria – The acceptance criteria and guidelines for the inspection of containment coatings are specified in ES-437, ES-438 and QSP-506. Design Engineering Guideline ST-07, "Containment Inservice Inspection Evaluation Criteria", provides the acceptance criteria used for determining containment coating failures and degraded conditions identified during the IWE and IWL containment examinations.

Corrective Actions – Any degraded conditions of containment coatings that require more detailed evaluations, repair, replacement, or augmented inspections are documented under the plant CER and/or NCN programs. Specific corrective actions are documented in accordance with the VCSNS corrective action program and implemented through the applicable work management program. Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – Periodic visual inspections and condition assessments of Service Level I coatings inside containment are conducted in accordance with ES-437, ES-438 and QSP-506. Design Engineering Guideline ST-07 provides the inspection criteria and guidance used to identify, document, review, and evaluate degraded conditions identified during the IWE and IWL containment examinations.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – GL 98-04 describes industry experience pertaining to coatings degradation inside containment and the consequential clogging of sump strainers. Monitoring and maintenance of Service level I coatings (conducted in accordance with RG 1.54) are effective programs for managing degradation of

Service Level I coatings, and consequently an effective means to manage loss of material due to corrosion of carbon steel inside containment.

Protective coatings are used extensively throughout the inside of the Reactor Building at VCSNS. The primary purposes of these protective coatings are to provide corrosion protection, ease of decontamination, and to prevent dust migration from concrete surfaces. The only Nuclear Safety Related function for these coatings is to remain intact and adhere to their applied surfaces.

The SCE&G response to GL 98-04 [Reference 106] stated that SCE&G has implemented controls for the procurement, application, and maintenance of Service Level I protective coatings used inside the containment in a manner that is consistent with the licensing basis and regulatory requirements applicable to VCSNS. The requirements of 10 CFR 50 Appendix B are implemented through specification of appropriate technical and quality requirements for a Service Level I coatings program that includes ongoing maintenance activities.

VCSNS is in compliance with the recommendations of RG 1.54, Revision 0 with the following clarifications and exceptions:

For the Westinghouse scope of supply (NSSS equipment and components), an alternate methodology for meeting the requirements of RG 1.54 was employed. The VCSNS FSAR states:

> *"For the Westinghouse scope of supply, Westinghouse employs process specifications and the Westinghouse Quality Assurance Program, including quality assurance surveillance and auditing, to provide adequate confidence that coating work within Westinghouse scope will perform satisfactorily in service."*
>
> *An alternate method of compliance with this regulatory guide has been submitted to the NRC (via letter NS-CE-1352, dated February 1, 1977, to MR. C. J. Heltemes, Jr., Quality Assurance Branch, NRC, from Mr. C. Eicheldinger, Westinghouse PWRSD, Nuclear Safety Department) and accepted (via letter, dated April 27, 1977, to Mr. C. Eicheldinger from MR. C. J. Heltemes, Jr.).*

Additionally, a small amount of unqualified coatings was identified inside containment. The FSAR states:

> *"The estimated quantity of unqualified paint inside containment is 0.18 cubic feet. This is based upon conservative estimates of 0.25 percent of all painted surfaces inside the Reactor Building including both the concrete and steel. For an average dry film thickness of 8 mils, this represents an area of 270 square feet."*

This quantity of unqualified coatings was confirmed in a survey conducted prior to power operations in November 1981. [Reference 107].

The ASME Section XI, Subsections IWE and IWL inspections conducted in 2000 are considered as the baseline examination. All previous inspections conducted for other programs (e.g., Maintenance Rule and Appendix J) had not identified any areas inside containment with surface areas likely to experience accelerated degradation or aging. Therefore, there were no areas designated for augmented examination prior to this baseline inspection.

The IWE inspection of the containment liner conducted during the 2000 Containment Inservice Inspection, revealed several areas of containment liner coating degradation. Minor flaking and/or split separation of the liner top coat in the vicinity of the spray rings was noted. One indication of a small split in the top coat (approximately 6 inches long) and several indications of partial delamination (flaking) of the top coat were observed with no exposure of the primer coat. These conditions are documented on CER 00-1388 and NCN 00-1388.

The IWE conditions and other areas of containment coating degradation were documented in the "Maintenance Rule Inspections – 2000 Assessment of In-Service Conditions of Important to Maintenance Rule (ITMR) Structures" report. None of the degraded conditions have an immediate adverse effect on the ability of the coatings to perform their intended function(s). CER 01-1011 was also originated which requires the PSE Building Services Engineer to evaluate these conditions for corrective actions. Each item/area will be tracked via supplemental CERs [Reference 156].

In accordance with the disposition of NCN 00-1388, most of the areas with identified conditions were reworked per appropriate Civil Maintenance Procedures during RF-12. Although the non-repaired conditions are considered minimal at this time and the coating is judged to be capable of performing its intended function of protection of the liner without significant failures, these conditions could deteriorate to an unacceptable condition if not evaluated with more frequent monitoring. Therefore, these areas will be monitored by periodic (each outage) Civil Maintenance walk-downs and by augmented ASME Section XI IWE inspections for any changes in conditions that may suggest a loss of integrity or function. NCN 00-1388 was closed on April 18, 2001. Augmented inspections conducted during RF-13 (2002) determined that the previously identified conditions remained acceptable.

The VCSNS Containment Coating Monitoring and Maintenance Program is considered acceptable based on industry standards and operating experience. A review of the Containment Coating Monitoring and Maintenance Program provides reasonable assurances that the aging effects for containment coatings will be managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Containment Coating Monitoring and Maintenance Program, described in the SCE&G response to GL 98-04, has been compared to the "Protective Coating Monitoring and Maintenance Program" documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding the monitoring and maintenance of Service Level I protective coatings used inside containment is consistent with the VCSNS program. The VCSNS Containment Coating Monitoring and Maintenance Program contains those attributes delineated in GALL Chapter XI.S8, which have been determined by the NRC to provide an acceptable aging management program.

7.9 Containment ISI Program –IWE/IWL

10 CFR 50.55a requires a detailed visual examination of the containment system for structural anomalies in accordance with ASME Section XI, Subsections IWE and IWL which is a condition monitoring program. The detailed visual examination is for attributes delineated in Quality Systems Procedure, QSP-506. For the Steel Containment Vessel (SCV) (liner), aging effects are loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities. For moisture barriers, aging effects are cracking, embrittlement, erosion, separation from attaching surface, or other deterioration. For the Reactor Building structure, aging effects are concrete spalling, cracks, delamination, or other deterioration.

Scope of Program – 10 CFR 50.55a requires detailed visual examination of the containment system for structural anomalies in accordance with ASME Section XI Subsections IWE and IWL. Accessible surfaces of ASME Section XI, Class MC SCV (liner) as defined in ASME Section XI, Table IWE-2500-1, Item E1.11 are in scope. Accessible surfaces of the ASME Section XI, Subsection IWL, Code Class CC concrete containment as defined in ASME Section XI, Table IWL-2500-1, Item L1.11 are in scope. Moisture barriers at concrete to SCV embedment zones and peripheries of attachments to the SCV as defined in ASME Section XI, Table IWE-2500-1, Item E5.30 are in scope. The Reactor Building exterior concrete delineated in QSP-506 is in scope. Inspection areas for inside the Reactor Building are shown on engineering drawing series E-411-600.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. Detailed visual examination of the containment system for structural anomalies (in accordance with ASME Section XI Subsections IWE and IWL) is a condition monitoring program.

Parameters Monitored or Inspected – The SCV (liner) aging effects are loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities. The aging effects related to moisture barriers are cracking, embrittlement, erosion, separation from attaching surface, or other deterioration. The aging effects associated with the Reactor Building structure are concrete spalling, cracks, delamination, or other deterioration. Aging effects are described in QSP-506 for ASME Section XI, Subsection IWE, Code Class MC and ASME Section XI, Subsection IWL, Code Class CC.

Detection of Aging Effects – The SCV (liner) visual examination detects loss of material due to corrosion, cracking of welds, deformed structural attachments, pitting, gouges, dents, or other surface discontinuities. Visual examination of moisture barriers detects cracking, embrittlement, erosion, separation from attaching surface, or other deterioration. The Reactor Building structure visual examination detects concrete spalling, cracks, delamination, or other deterioration. Aging effects are described in QSP-506, for ASME Section XI, Subsection IWE, Code Class MC and ASME Section XI, Subsection IWL, Code Class CC.

Monitoring and Trending – Visual examination (VT-3) is in compliance with ASME Section XI, Subsections IWE and IWL. Detailed visual examination of the SCV (liner) and Reactor Building concrete is in accordance with QSP-506. The VCSNS Containment Inservice Examination (ISE) program plan document, ISE-4 [Reference 76], Section 4.1.2.4 provides the ISE schedule. The first ISE Interval is from January 1, 1997 through December 31, 2006 for ASME Section XI, Subsections IWE and IWL. Examinations for the first period of first interval were performed during RF-12 as stated in VCSNS Engineering Services Procedure, ES-438 [Reference 108]. The baseline inspection will be used to trend future inspection results.

Acceptance Criteria – Acceptance criteria for aging effects are delineated in Design Engineering Guideline ST-07. The acceptance criteria is based on ASME Section XI, Subsection IWE-3510.1 for the SCV and Reactor Building structure surfaces, and ASME Section XI, Subsection IWE-3513.1 for moisture barriers.

Corrective Actions – When areas of degradation are identified, an evaluation is performed to determine whether repair or replacement is necessary. Subsequent to an evaluation that concludes repair or replacement is necessary, Subsection IWE requires verifying effectiveness of completed action(s). Subsection IWE repairs and reexaminations are to comply with the requirements of IWA-4000. Reexaminations are conducted in accordance with IWA-2200 and the results must demonstrate that the repair meets the acceptance standards delineated in Table IWE-3410-1.

A Condition Evaluation Report is initiated for any structure, or component that does not meet the acceptance criteria in accordance with Engineering Procedure ES-438. The CER process is discussed in Station Administrative Procedure SAP-1131 [Reference 160]. Repair / replacement of any structure or component is in accordance with ES-438. Specific corrective actions are implemented in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – 10 CFR 50.55a require detailed visual examination of the containment system for structural anomalies in accordance with ASME Section XI Subsections IWE and IWL. IWA-6000 provides guidance for the preparation and retention of reports. Detailed visual examination of the containment system for structural anomalies is implemented by QSP-506. The VCSNS Containment Inservice Examination (ISE) database is an electronic data management tool for tracking and scheduling inspections required by ASME Section XI Subsections IWE and IWL, as modified by 10 CFR 50.55a.

Operating Experience – Examinations for first period of first interval were performed during RF-12 as stated in Engineering Services Procedure ES-438. Examinations for first period of first interval in accordance with QSP-506 are satisfactory. Detailed discussion of the examination is provided in Responsible Engineer Evaluation Report dated January 2, 2001.

There are no Licensee Event Reports (LER) based upon examinations performed during RF-12 for first period of the first interval (as stated in Responsible Engineer Evaluation Report dated January 2, 2001).

One Non-Conformance Notice based upon initial examinations performed during RF-12 for first period of the first interval (as stated in Responsible Engineer Evaluation Report dated January 2, 2001) is as follows:

NCN	SUBJECT	DISPOSITION
00-1388	Containment Liner Coating Degradation. Several areas, Items 1 through 19 are noted	Four areas, Items 1 to 4 are cleaned and painted as stated in Technical Work Record (TWR) RW04586 dated October 14, 2000 Disposition 2. All other areas, Items 5 through 19, are painted as stated in TWR RW04586 dated October 14, 2000 Disposition 3. Discoloration is not a nonconforming condition as stated in TWR Disposition 4 for NCN 00-1388 dated October 14, 2000. Discoloration is not an aging effect as stated in GALL Report Chapter XI.S8.

This Non-Conformance Notice does describe an aging effect specified in the GALL Report [Reference 5] Chapter XI.S8.

CER's for the examinations performed during RF-12 (2000) and RF-13 (2002) are:

CER	SUBJECT	DISPOSITION
00-1617	Cracks and Separation of Moisture Barrier at Elevation 412'	Repair or replace during RF-12
02-1111	Cracks and Separation of Moisture Barrier at Elevation 412'	Repair or replace during RF-13

CER	SUBJECT	DISPOSITION
98-1047	Groundwater in-leakage at RHR and RB Spray Line Penetrations at AB Elevation 397'	• Remove waterproof material around guard pipes to perform visual and UT Inspection. UT Inspection results confirm that minimum wall thickness for guard pipes meets acceptance criteria. Waterproof material is no longer installed around the guard pipes to keep annulus area dry. Augmented inspections for guard pipes are required during each refueling outage to prevent corrosion.
02-0916		• Augmented inspection during RF-13.
00-0988	Leaching of minerals and Corrosion at one location of exterior wall in Tendon Access Gallery	• Leaching determined to be structurally insignificant but a chemical analysis is required of the leached material and groundwater to determine corrosive environment. Chemical analysis of leached material indicates primarily calcium carbonate, with the groundwater properties pH > 12.5, chlorides < 5 ppm, and sulfides < 1 ppm, which is acceptable based upon EPRI Report TR-114881. Therefore groundwater is non-aggressive.
02-0832		• Augmented inspection during RF-13.

Based on operating experience, continued implementation of the VCSNS ASME Section XI, Subsection IWE and IWL inspection program (established by 10 CFR 50.55a and implemented by ISE-4) manages the identified effects of aging. Continued implementation of the VCSNS ASME Section XI, Subsection IWE and IWL inspection program throughout the period of extended operation will maintain the intended function of containment.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS ASME Section XI, Subsection IWE and IWL inspection program has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding ASME Section XI, Subsections IWE and IWL inspection programs is consistent with the VCSNS program. The VCSNS ASME Section XI, Subsections IWE and IWL inspection programs contain those attributes delineated in GALL Chapters XI.S1 and XI.S2 which have been determined by the NRC to provide an acceptable aging management program.

7.10 Fire Protection Program

The VCSNS Fire Protection Program utilizes the concept of defense-in-depth to achieve a high degree of fire safety as discussed in the FPER [Reference 109]. Defense-in-depth design basis consists of preventing fires from starting, detecting and suppressing fires quickly to limit their damage, and designing the plant safety systems so that in the unlikely event of a major fire, the capability to safely shutdown the reactor is maintained.

The VCSNS Fire Protection Program contains many activities to achieve defense-in-depth and minimize the impact of a potential fire. One activity is fire containment as discussed in FPP-025 [Reference 110]. Fire containment utilizes compartmentation as discussed in the Penetration Seal Project Plan [Reference 111], consisting of fire rated assemblies that form passive barriers to divide buildings and rooms into separate fire areas or zones. Fire rated assemblies are defined in TRP-2 [Reference 66] as fire barriers, fire barrier penetrations, and fire barrier penetration sealing devices. Inspections of fire rated assemblies are required by Branch Technical Position APCSB 9.5-1, Appendix A based on commitments in the FPER Section 5.0.C.6. Surveillance requirements are established in FPP-025. Periodic inspections are conducted for fire barriers (walls, floors, ceilings, and cable tray enclosures), fire barrier penetration seals, and insulating material (Kaowool cable wrap, Albi structural steel proofing). The Fire Protection Program is a condition monitoring program.

Fire rated assemblies are defined in TRP-2 as walls, floors, ceilings, dampers, doors, penetration seals, and cable wrap which have different inspection criteria, different acceptance criteria, and different inspection frequency based on FPP-025.

- Fire barriers are walls, floors, ceilings, and cable tray enclosures
- Fire barrier penetrations are doors, pipe penetrations, electrical penetrations, building service (duct) penetrations, and cable tray enclosure penetrations
- Fire barrier penetration sealing devices are penetration seals (silicone foam, flexible boot seals) and insulating materials (Kaowool cable wrap, M-Board, and Albi structural steel fireproofing)

Engineering inspection requirements for doors, mechanical piping penetration seals and electrical cable penetration seals are given in FPP-025.

Fire barrier inspections are credited with managing aging effects covering the period of extended operation. Aging effects for fire barrier penetration sealing devices are cracking, separation from walls or components, separation of material layers, rupture or puncture of seals, shrinkage, and voids as specified in FPP-025. Aging effects for fire barriers are cracking, spalling, and loss of material in fire barrier walls, ceilings, and floors as specified in FPP-025.

The regulatory basis for inspection of the controlled fire barriers is found in 10 CFR 50 Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979". The Fire Protection Rule, 10 CFR 50.48, requires each nuclear power plant to have in place a fire protection plan that satisfies 10 CFR 50, Appendix A, General Design Criterion 3. General Design Criterion 3 adopts Appendix A to Branch Technical Position APCSB 9.5-1. The VCSNS Fire Protection Program has been in effect since the initial licensing of the station. Fire barrier inspections are implemented by Surveillance Test Procedures, STP-728.027 through STP-728.050.

When the License Renewal Rule was issued, the NRC concluded that "it is appropriate to allow license renewal applicants to take credit for the fire protection plan as an existing program that manages the detrimental effects of aging of fire protection components that perform active functions". Similarly, it is reasonable to expect that the fire protection plan could also be effective at managing the aging effects of components that are not considered to be active. Technical Specifications governing the operating and surveillance of fire protection equipment were initially implemented at VCSNS in 1984 and compliance was the subject of NRC Inspection Report 50-395 / 84-08 [Reference 112]. Cracking and separation of the fire barrier penetration seals and cracking, loss of material, and spalling of fire barrier walls have been identified as applicable aging effects requiring programmatic management for the period of extended operation. The functional integrity of the fire barrier penetration seals and walls ensures that fires are confined or adequately retarded from spreading to adjacent portions of the facility. The fire protection, fire barrier and fire barrier sealing device inspection is credited for managing cracking and separation for the period of extended operation.

The Fire Protection Program uses the guidance provided in NEI 95-10 [Reference 4] to provide reasonable assurance that the effects of aging are adequately managed such that the fire barrier penetration intended function is maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation.

The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, and fire barrier walls, ceilings, and floors to ensure that their operability is maintained.

7.10.1 Fire Barriers and Fire Barriers Penetration Seals

Scope of Program – The VCSNS fire barrier and fire barrier penetration sealing device inspections are required in order to comply with FPP-025. Fire barriers (walls, floors, ceilings, and cable tray enclosures) are identified using engineering drawings E-023-001 through E-023-023 and Architectural Drawings D-101-011 through D-101-028, D-105-011, D-105-012, D-128-001, and D-128-002. Fire barriers (walls, floors, ceilings, and cable tray enclosures) and fire barrier penetration seals are discussed in Fire Protection System (FS) Design Basis Document [Reference 113] Sections 4.6.1 and 4.6.2 respectively. Electrical penetrations requiring fire barrier penetration seals and insulating materials are listed by penetration number on engineering drawing E-224-531. Fire barrier penetration seal details are shown on engineering drawings E-201-240, sheets 1 and 2. Electrical, mechanical, and building service penetrations are numbered and shown on engineering drawing series E-400. The total population of fire barriers and fire barrier penetration seals are from the database generated during Phase II of the Penetration Seal Project Plan [Reference 111] and are illustrated as attachments to STP-728.027 through STP-728.050 by building, room number, and location.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Fire Barrier and Fire Barrier Seal Inspections are condition monitoring programs.

Parameters Monitored or Inspected – Aging effects for fire barrier walls, ceilings, and floors are cracking (excluding hairline cracks), spalling, and loss of material. Aging effects are specified in TRP-2 and inspections to manage these effects are implemented by STP-728.027 through STP-728.050. Aging effects for fire barrier penetration seals are cracking, separation from walls or components, separation of material layers, rupture or puncture of seal, shrinkage, and voids as specified in TRP-2, and inspections to manage these effects are implemented by STP-728.027 through STP-728.050.

Detection of Aging Effects – Fire barrier and fire barrier penetration sealing device inspections detect structural damage or degradation for fire barriers and fire barrier penetration sealing devices prior to loss of intended function.

Monitoring and Trending – Visual examinations of each fire barrier and each fire barrier penetration sealing device are performed every 18 months as stated in STP-728.027 through STP-728.050. Applicable attachments in STP-728.027 through STP-728.050 are completed as a means of documenting inspections. No actions are taken as part of this program to trend inspections or test results.

Acceptance Criteria – Acceptance criteria for fire barriers is specified in TRP-2 and FPP-025 and is included in STP-728.027 through STP-728.050. The acceptance criteria for fire barriers are no visual indication of through barrier holes for walls, ceilings, and floors, no visible cracks (excluding hairline cracks), no surface gouges, no unsealed core drills, and no separation between mating surfaces in the required thickness of the barrier.

For fire barrier penetration sealing devices using silicone foam and elastomers, acceptance criteria are from TRP-2 implemented by STP-728.027 through STP-728.050 as follows:

- No visual cracks with a depth or width less than 1/2 inch
- No separation of layers of material with a depth or width in excess of 1/2 inch
- No through-wall holes with a depth or width exceeding 1/2 inch for walls, ceilings, and floors
- No gouges with a depth or width greater than 1/2 inch

For fire barrier penetration sealing devices using M-Board, acceptance criteria are from TRP-2 and implemented by STP-728.027 through STP-728.050 as follows:

- No visual cracks with a depth or width greater than 1/4 inch
- No separation from wall or separation of layers of material greater than 1/4 inch
- No through-wall holes exceeding 1/4 inch for walls, ceilings, and floors,
- No rips or tears in cable wrap in excess of 1/4 inch long
- No rupture or puncture of seal exceeding 1/4 inch deep
- No surface gouges with a depth or width greater than 1/4 inch deep

Corrective Actions – For fire barriers, the Non-Conformance Notice (NCN) process is initiated for items that do not meet the acceptance criteria in compliance with the FPER, Section 5.0.C.7. For fire barrier penetration sealing devices, the NCN process is initiated for items that do not meet the acceptance criteria as specified in TRP-2 in compliance with the FPER, Section 5.0.C.7. Repair / replacement of fire barrier penetration seals is in accordance with Civil Maintenance Procedure, CMP-600.002.

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – Fire barrier and fire barrier penetration seal inspections are required to comply with FPP-025. Fire barrier and fire barrier penetration seal inspections are implemented by STP-728.027 through STP-728.050. Component History and Maintenance Planning System (CHAMPS) schedules implementation of STP-728.027 through STP-728.050 using work orders and maintains records of the inspections in the form of completed attachments from STP-728.027 through STP-728.050.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – Fire barrier and fire barrier penetration seal inspections in the past five years do not indicate any fire barrier or fire barrier penetration seal that is in nonconformance with the acceptance criteria.

NRC Inspection Report 50-395 / 98-01 concludes that STP-728.027 through STP-728.050 are excellent and satisfy the requirements of Generic Letter 86-10 "Implementation of Fire Protection Requirements".

Licensee Event Reports (LER) for fire barriers are compiled using the licensing database. A review of past LER's and NRC Inspection Reports (IR's) associated with fire barriers indicated design deficiencies such as lack of qualifying documentation, or installation deficiencies such as not sealing core drills or damaging Kaowool wrap and seals during maintenance. Other LER's indicate fire watches not initiated after a fire barrier is inoperable, and effects of electrical storms on the fire protection system. Based on this review, none of the LER's or IR's are related to aging.

Non-Conformance Notice NCN-01-1882 was initiated subsequent to implementation of STP-728.035 in WPAA under STTS 0112490. The nonconformance is aging related and details are as follows:

PENETRATION	NONCONFORMING CONDITION	DISPOSITION 3
Trace 4 PAA-12-01 North Wall	4 inch Horizontal Crack	Repaired Satisfactorily
Trace 72 PAA-36-01 North Wall	½ inch Separation - 7 inches Deep	Repaired Satisfactorily
Trace 99 AB-85-08 Floor	½ inch Separation - 7 inches Deep	Repaired Satisfactorily

No Condition Evaluation Reports were initiated for fire barriers or fire barrier penetration seals relevant to aging.

Based on operating experience, continued implementation of the fire barrier and fire barrier penetration sealing device inspections (required by FPP-025) manages effects of aging throughout the period of extended operation. Therefore, the Intended function(s) of the fire barriers and fire barrier penetration seals are maintained.

Generic Aging Lessons Learned (GALL) Comparison – Fire barrier and fire barrier penetration seals inspection programs have been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding fire barriers and fire barrier penetration seals inspection programs is consistent with VCSNS with the following exception:

- GALL mandates 10% of the population of fire barrier penetration seals to be inspected for aging effects every refueling outage. VCSNS conservatively inspects all fire barrier penetration seals every 18 months.

The fire barriers and fire barrier penetration seals inspection programs contain those attributes delineated in GALL Chapter XI.M26 which have been determined by the NRC to provide an acceptable aging management program.

7.10.2 Fire Doors

Fires rated door inspections are performed to comply with FPP-025, [Reference 110]. Fire rated door inspections represents a condition monitoring program. Examination guidelines and results of periodic inspections of fire rated doors are provided. Inspections are credited with managing loss of material of doors and door hardware for the period of extended operation. Excessive wear for latches, strike plates (a strike plate is a wear plate and keeper for a latch bolt), hinges, sills, and closing devices, and maintaining proper clearances (gaps) between the door, frame and threshold are also inspected under FPP-025, but these attributes are not credited for license renewal. The inspection procedure is evaluated in accordance with the guidance provided in NEI 95-10 to provide reasonable assurance that the effects of aging are adequately managed so that the fire doors intended function is maintained consistent with the CLB for the period of extended operation.

Scope of Program – Fires rated door inspections are performed to comply with FPP-025. Fire rated doors that are equipped with automatic or self-closing devices and doors that are manually closed are in scope. Fire doors are labeled with the fire rating shown on engineering drawings E-023-001 through E-023-023. Doors are located and numbered on architectural drawing series D-101 for Intermediate Building, Auxiliary Building, Reactor Building, Turbine Building, WPAA, EPAA, Fuel Handling Building, and Diesel Generator Building. Doors are located and numbered on architectural drawings D-105-011 through D-105-014 for the Control Building. Doors are located and numbered on architectural drawing D-128-001 for the Fire Service Pump House and on architectural drawing D-128-002 for the Service Water Pump House. The door schedule is listed on architectural drawings D-108-011, D-108-012, D-108-013, and D-108-018. The door schedule identifies (under label column) each fire door by label A, B, or C. The door schedule for license renewal scope structures (as delineated in FPP-025) identifies the doors that are rated. Fire doors are discussed in Design Basis Document Fire Protection System (FS) [Reference 113, Section 4.6.4].

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Fire Door Inspection is a condition monitoring program.

Parameters Monitored of Inspected – TRP-2 [Reference 66] specifies monitoring for loss of material of doors and door hardware as aging effects. Excessive wear for door appurtenances such as latches, strike plates, hinges, sills, and closing devices and maintaining proper clearances (gaps) between the door, frame, and threshold are additional attributes for inspection, but are not credited for license renewal. Fire door mechanisms and latches are visually inspected and tested every six months as stated in FPP-025. The position of each interior closed fire door is verified twice daily in accordance with FPP-015

[Reference 114]. Each locked closed fire door is verified closed once a week as stated in FPP-025.

Detection of Aging Effects – The fire door inspection program detects structural damage or degradation prior to loss of intended function.

Monitoring and Trending – Fire rated door inspections are performed in order to comply with FPP-025. Visual examination of the door and frame and functional testing for closure are performed every six months as stated in FPP-025, in accordance with Surveillance Test Procedure STP-128.019. Visual examination for maintaining proper clearances (gaps) between door, frame, and threshold is performed every 18 months in accordance with STP-728.027 through STP-728.050. No actions are taken as part of this program to trend inspections or test results.

Acceptance Criteria – Fire door acceptance criteria are provided in TRP-2 and implemented by STP-128.019. The acceptance criteria are based on FPP-025. Self-closing doors are visually inspected to verify that hinges are intact, with all screws tight, pins in good condition, and each door closes. Double self-closing doors are visually inspected to verify that bolts are in good condition, the astragal (metal molding strip) is in good condition, and the door closes. Automatic closing doors are visually inspected to verify that they are in good operating condition and the door closes. Hollow fire doors are visually inspected for holes and damage in the skin of the door and the frame. Acceptance criteria for maintaining proper clearances (gap) between door, frame, and threshold are in accordance with STP-728.027 through STP-728.050.

Corrective Actions – The NCN process is initiated for doors that do not meet the acceptance criteria as stated in TRP-2 in compliance with FPER Amendment 98-01. Minor abnormalities (loose knobs, loose latches or other loose appurtenances) do not require initiating a NCN. Maintenance work requests are initiated to repair any minor abnormality. Fire doors are repaired in accordance with CMP-100.008.

For conditions other than a minor abnormality, a CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – Fires rated door inspections are performed to comply with FPP-025. Visual examination of the door and frame and functional testing for closure are in accordance with Surveillance Test Procedure STP-128.019. Visual examination for maintaining proper clearances (gaps) between door, frame, and threshold is in accordance with STP-728.027 through STP-728.050.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – VCSNS has no failures or adverse trends for fire doors. STP inspections in the last five years have not identified any nonconformances relative to the acceptance criteria.

Frequency of inspections (in effect since the initial implementation of the Technical Specifications requirements in 1984) is considered acceptable based on industry operating experience in that degradation of a door is detected prior to loss of function.

If the results of the visual inspection indicate that repairs are required, then specific repairs are made in accordance with CMP-100.008. The fire barrier inspections are implemented by plant procedures and controlled by the SCE&G Quality Assurance Program.

Fire dampers are addressed in VCSNS Technical Report TR00160-017 [Reference 53].

NRC Inspection Report 88-04 [Reference 115] concluded that the fire door surveillance procedure, STP-128.019 met the requirements of FPP-025 and concluded that the procedure followed NFPA, industry, and NRC guidelines.

LERs for fire doors were compiled using the licensing database. LER 88-008 is related to a combination fire/pressure door, but not the fire barrier function, and reports a design deficiency which is not related to aging. LER 82-009 reports a missed weekly surveillance. LER 84-033 and LER 87-030 report fire doors not fully closed. LER 83-080 reported that the fire door surveillance during an outage was not performed.

No Non-Conformance Notices or Condition Evaluation Reports were initiated for fire doors relevant to aging.

Based on operating experience, the continued implementation of the fire barrier inspections (in compliance with 10 CFR 50.48) manages identified effects of aging throughout the period of extended operation such that the intended functions of the fire barriers are maintained.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS fire door inspection program has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in GALL Chapter XI.M26 is consistent with the VCSNS program with the following exceptions:

- VCSNS fire rated door inspections do not monitor holes or breaks in the door surface bimonthly, instead the VCSNS fire rated door inspections monitor holes or breaks in the door surface every 6 months. Based on VCSNS and industry operating experience the inspection frequency provides reasonable assurance that degradation of a door is detected prior to loss of function.

- VCSNS does not consider maintaining proper clearances (gaps) between door, frame, and threshold as an aging effect. However, maintaining proper clearances (gap) between door, frame, and threshold are performed in accordance with STP-728.027 through STP-728.050.

7.11 Flood Barrier Inspection

Flood evaluations are required by Generic Letter 88-20 [Reference 116]. Flood barrier inspections are performed to comply with Regulatory Guide (RG) 1.127 [Reference 117].

The VCSNS Flood Barrier Inspection Program provides periodic inspections for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals.

Flood barrier inspection is a subset of the Maintenance Rule Structures Program and the Fire Protection Program which are credited with managing aging effects for the period of extended operation for flood barriers, flood doors, and flood barrier penetration seals.

These inspection programs have been evaluated in this report using the guidance provided in NEI 95-10 [Reference 4]. Inspections provide reasonable assurance that the effects of aging are adequately managed such that flood barrier intended functions are maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation.

Barriers for internal flood are identified in Nuclear Safety Related (NSR) Structures Design Basis Document [Reference 16] Sections 3.1.5, 3.2.5, and 3.3.5 as follows:

- Protective barriers, curbing, and equipment pedestals
- Designated flood doors
- Flood barrier penetration seals

Letters RC-93-0170 dated June 18, 1993 and RC-96-0068 dated March 20, 1996 provide the VCSNS response to GL 88-20. Letter RC-93-0170 provides the Individual Plant Examination (IPE) required by GL 88-20. Letter RC-96-0068 describes mitigation of internal floods subsequent to a NRC Request for Additional Information (RAI) dated January 11, 1996. Internal flood levels for the Intermediate Building, Auxiliary Building, Diesel Generator Building, Fuel Handling Building, Control Building, Turbine Building, and Service Water Pump House are documented in calculations DC03290-001 [Reference 118] and DC03290-002 [Reference 119]. Internal flood levels for the Reactor Building are documented in calculations DC03190-009 [Reference 120] and DC03190-001 [Reference 121].

Flood doors are monitored for:

- Structural damage (broken lever handles, broken pins, or broken hinges)

- Damage (through holes, gouges etc.) to door leafs or appurtenances
- Damage or deterioration of bolting material
- Missing or damaged weather stripping and missing hardware

Flood seals are required for penetrations in structural barriers delineated in Drains, Sumps, and Leak Detection (ND) Design Basis Document Section 3.8.5.4. Flood seals are monitored for:

- Tears
- Punctures
- Tight fit

Flood barriers (walls, curbs, equipment pedestals, penetration seals) outside containment are identified in SCE&G correspondence to NRC letter RC-96-0068. Letter RC-96-0068 describes mitigation of an internal flood subsequent to a NRC RAI dated January 11, 1996. A watertight design for interfacing walls up to elevation 426'-9" isolates Nuclear Safety Related buildings from an internal flood from the Turbine Building as stated in the ND DBD. Curbs are provided at entrances to cubicles housing Safety Related equipment as stated in FSAR [Reference 12] Section 6.3.2.2.7.

Flood barriers (walls, curbs, equipment pedestals, penetration seals) inside the Reactor Building are not credited, based on the Reactor Building DBD [Reference 67] Section 3.7 and letter RC-96-0068, Section 1.2.1.

Scope of Program – Flood barrier inspections are performed to comply with RG 1.127, Section C.2. Nuclear safety-related flood barriers are credited in Drains, Sumps, and Leak Detection (ND) Design Basis Document, Section 3.6.3 to mitigate the effects of internal flood. Nuclear safety-related flood barriers consist of curbs at entrances to cubicles housing safety grade equipment as stated in FSAR Section 6.3.2.2.7 as identified in RC-96-006B.

Designated flood doors (watertight doors) are specified in Specification SP-631 [Reference 122] and listed on architectural drawing D-108-018. Ten doors are designated flood doors (watertight doors).

Penetrations requiring Nuclear Safety Related flood seals are specified in Drains, Sumps, and Leak Detection (ND) Design Basis Document [Reference 123] Section 3.8.5.4. Penetrations requiring Nuclear Safety Related flood seals are shown on engineering drawings E-413-081 and E-413-083 for Intermediate Building, engineering drawing E-414-081 for Control Building, and engineering drawing E-417-081 for the Diesel Generator Building. Eleven penetrations require Nuclear Safety Related flood seals.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Flood Barrier Inspection is a condition monitoring program.

Parameters Monitored or Inspected – Aging effects for flood barriers are cracks, exposed reinforcing steel, corrosion, scaling, popouts, surface pitting, and spalling. Aging effects are listed in Engineering Services Procedure, ES-437 [Reference 124]. Aging effects are the same for Nuclear Safety Related or Quality Related flood barriers.

The aging effects for flood barrier penetration seals are similar to aging effects for fire barrier penetration seals from TRP-2 and are:

- Indication of cracking with a width or depth exceeding 1/4 inch where length of crack may be any dimension
- Indication of fraying
- Indication of separation from penetration with a width or depth exceeding 1/4 inch where length of separation may be any dimension
- Indication of through-wall holes

Detection of Aging Effects – The Flood Barrier Inspection Program detects aging effects prior to loss of structure intended function.

Monitoring and Trending – Nuclear safety-related flood barrier and flood barrier penetration seal inspections are performed to comply with RG 1.127, Section C.2. Visual examination of concrete structures is performed as stated in ES-437. Visual examination of the flood barrier penetration seals that are also fire barrier penetration seals is performed as stated in Surveillance Test Procedures STP-728.045 and STP-728.046 for the Control Building, STP-728.027 for the Diesel Generator Building, and STP-728.049 for the Intermediate Building. Visual examination of nuclear safety-related flood barriers is performed once every five years commencing with year 2000 as stated in Engineering Services Procedure ES-437. Visual examination of Nuclear Safety Related flood barrier penetration seals that are also fire barrier penetration seals is performed once every 18 months as stated in Surveillance Test Procedures STP-728.045 and STP-728.046 for the Control Building, STP-728.027 for the Diesel Generator Building, and STP-728.049 for the Intermediate Building. No actions are taken as part of this program to trend inspection or test results.

Acceptance Criteria – Flood barrier and flood barrier penetration seal examination acceptance criteria are provided in ES-437 for flood barriers that are not fire barriers and in TRP-2 for flood barriers that are also fire barriers. Acceptance criteria are no cracks, no exposed reinforcing steel, no corrosion, no scaling, no popouts, no surface pitting, and no spalling. Satisfying examination acceptance criteria for flood barriers that are not fire barriers is documented by completing Engineering Services Procedure ES-437, Attachment II (Important To

Maintenance Rule (ITMR) Structures Deficiency Identification) and Attachment I (Important To Maintenance Rule (ITMR) Structures Inspection Checklist). Satisfying the examination acceptance criteria for flood barriers that function also as fire barriers is documented by completing attachments to STP-728.027 through STP-728.050. Acceptance criteria are the same for Nuclear Safety Related or Quality Related flood barriers.

Flood barrier penetration seal examination acceptance criteria are the same as for fire barrier penetration seals and are provided in Technical Requirements Package TRP-2 [Reference 66]. TRP-2 is implemented by STP-728.045 and STP-728.046 for the Control Building, STP-728.027 for the Diesel Generator Building, and STP-728.049 for the Intermediate Building. Acceptance criteria are as follows:

- Indication of cracking with a width or depth less than 1/4 inch but length of crack may be any dimension
- Separation between surfaces at penetration of a width or depth less than 1/4 inch where length of separation may be any dimension
- No through-wall holes

The flood barrier penetration seal examination acceptance criteria is satisfied and documented by inspections performed under ES-437. Examination acceptance criteria for flood barrier penetration seals (that function also as fire barrier penetration seals) is satisfied and documented by completing attachments to STP-728.027 through STP-728.050. Acceptance criteria are the same for Nuclear Safety Related or Quality Related flood barrier penetration seals.

Corrective Actions – Maintenance work requests are initiated to repair abnormalities. The NCN process is initiated for flood barrier penetration seals that do not meet the acceptance criteria (as specified in TRP-2) in compliance with the Fire Protection Evaluation Report (FPER) [Reference 109, Section 5.0.C.7].

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The flood barriers inspections are implemented through Engineering Service Procedure ES-437 and Surveillance Test Procedures STP-728 described above.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – Since many flood barriers are also fire barriers, inspection attributes delineated in STP-728.027, STP-728.045, STP-728.046, and STP-728.049 are the same for fire barriers and flood barriers and thus satisfy the same acceptance criteria. Therefore if fire barrier and fire barrier penetration seal inspections are satisfactory, then flood barrier and flood barrier penetration seal inspections are satisfactory. All flood doors are also fire doors. Fire door inspection attributes are the same as flood door inspection attributes. Flood doors are adequate if the fire door inspections are adequate.

No Licensee Event Reports, Non-Conformance Notices or Condition Evaluation Reports were initiated for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals relevant to aging.

Based on operating experience, continued implementation of Nuclear Safety Related flood barrier inspections (in compliance with RG 1.127, Section C.2) manages identified effects of aging throughout the period of extended operation. The intended functions of flood barriers are maintained.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Flood Barrier Inspection Program is not evaluated in the GALL Report [Reference 5] and is, therefore, plant specific to VCSNS.

7.12 Maintenance Rule Structures Program

The VCSNS Maintenance Rule Structures Program for the inspections of Important to Maintenance Rule (ITMR) structures and structural components is appropriate in meeting the regulatory requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", here after referred to as the Maintenance Rule (MR). The MR requires that the performance condition of structures/components be monitored in a manner sufficient to provide reasonable assurance that the structures/components are capable of fulfilling their intended function(s). The MR program closely follows the guidance set forth in Regulatory Guide (RG) 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" and NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants".

The License Renewal Rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants", requires that for each structure/component subject to an Aging Management Review (AMR), the licensee shall demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. The Maintenance Rule program for structures/components has been credited with managing the attributes associated with the aging of structures and structural components. Aging attributes identified within the scope of the Maintenance Rule such as, loss of material due to corrosion, cracking, and change in material properties, are detected by visual inspection of external surfaces prior to the loss of the structural or structural component intended function(s).

NUREG-1526 states that "certain structures such as the primary containment can be monitored through the performance of established testing requirements. However, other structures such as the Reactor Building, Auxiliary Building, and Cooling Tower, may be more amenable to condition monitoring". The VCSNS Safety Related and non-Safety Related ITMR structures that are included in the MR program are listed in Engineering Service Procedure, ES-437, "Inspections for Maintenance Rule – Structures" [Reference 124]. This list includes several structures that are outside the scope of license renewal because it addresses those structures that are specifically identified within the scope of the Maintenance Rule. Maintenance Rule inspections are conducted every five years in accordance with ES-437. The Maintenance Rule Structures Program is a condition monitoring program.

The MR inspections of structures and structural components are evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for these structures/components so that their intended functions will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation.

Scope of Program – A detailed review of the structures affected by the Maintenance Rule was completed by SCE&G Plant Support Engineering and is documented in Technical Work Record (TRW) 15517 [Reference 125]. This TRW documents the basis for selection of those ITMR structures and the maintenance procedures that are currently performed. Based on the Plant Support Engineering and License Renewal evaluations, the following Safety Related and non-Safety Related structures are identified within the scope of the Maintenance Rule Structures Program:

- Auxiliary Building (AB)
- Auxiliary Boiler House (ABH) *
- Control Building (CB)
- Closed Cycle Cooling Tower and Pumphouse (CCCT) *
- Circulating Water Discharge Structures (CWDS) *
- Circulating Water Intake Structures (CWIS) *
- Diesel Generator Building (DGB)
- Electrical Manhole #2 (EMH-2)
- Fuel Handling Building (FHB)
- Fire Service Pumphouse (FSPH)
- Hot Machine Shop (HMS)
- Intermediate Building (IB)
- North Berm
- Reactor Building (RB)
- Service Water Discharge Structures (SWDS)
- Service Water Intake Structures (SWIS)
- Service Water Pond Dams (SWP)
- Service Water Pumphouse (SWPH)
- Switchyard Foundation for OCB 8892
- Switchyard Relay House (SYRH) *
- Transformer Area Foundations
- Transmission Towers and Foundations from Emergency Auxiliary Transformer to OCB 8892
- Turbine Building
- Large Tank Foundations (CST, RMWST, RWST)

The structures marked with an asterisk (*) are not within the scope of licensing renewal. They are all classified as non-seismic, Non-Safety Related whose functions are not related to nuclear safety and whose failure under design basis event conditions will not impair the integrity of Nuclear Safety Related systems, structures, and components. It was determined that these structures did not meet the license renewal scoping criteria provided in 10 CFR 54.4. [Reference 1].

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation. The Maintenance Rule Structures Program is a condition monitoring program.

Parameters Monitored or Inspected – Reinforced concrete surfaces are inspected for the following:

- Cracks (other than hairline / temperature)
- Exposed reinforcing steel
- Corrosion staining on surfaces
- Popouts / scaling
- Surface pitting / spalling / delamination / honeycombs
- Peeling or cracked paint
- Leaching (mineral bleeding via water intrusion)
- Dissolution (via chemical or galvanic processes)
- Weather (exposure of aggregates and rebar)
- Radiation / temperature damage (scorching, embrittlement, etc.)
- General signs of settlement / movement / distortion

Structural steel and liners are inspected for the following:

- Rusting or scaling metal
- Peeling or flaking paint
- Corrosion (via pitting, galvanic processes, microbiologically induced corrosion, etc.)
- Beam/column twisting / bowing / alignment / deflection
- Connections (loose or missing bolts/nuts, cracked welds, etc.)
- Loose, missing, damaged, or rusted anchors / fasteners
- Degraded base plates, grout or anchorage
- Liner bulging / deformation / degradation
- Liner gouges / dents / wear

Seismic Gaps are inspected for the following:

- Deterioration or loss of gap filler material
- Foreign objects or debris (which limit structural movement)
- Closure of gap spacing due to settlement / movement / alignment

Masonry block walls are inspected for the following:

- Cracks (other than hairline)

- Missing or broken blocks
- General signs of settlement / movement / distortion
- Deteriorated / displaced penetrations

Large tanks are inspected for the following:

- Signs of leakage
- Significant rusting
- Loose, missing, damaged, or rusted anchors
- Cracked welds
- Settlement / movement (deformation in the tank body)
- Damage, settlement, or movement of the tank foundation pad

Visual inspections of fire and pressure barriers/components are conducted periodically in accordance with the STP-728-series (fire barrier inspections) and CMP-700.009 (pressure barrier inspections). These inspections include a visual inspection of the walls and floors. Visual inspections for the integrity of fire doors are conducted semi annually in accordance with STP-128.019. These inspections include damage, alignment, closure, seals, locks, etc. which may indicate potential structural damage in the vicinity. See Sections 7.10 and 7.11 of this report for more details.

Visual inspections of roofs in the protected area are conducted semi annually in accordance with CMP-700.005. Roofs are inspected in general for the following:

- Inspect roof underside for sagging / water entry / leaking tar/asphalt / etc.
- Inspect roof top surface for damage / debris/contaminants on roof / water ponding / plugged drains / bare spots in ballast / blistering / splits / flashing deterioration / expansion joint deterioration / delamination etc.

Detection of Aging Effects – Attributes associated with the aging of structures identified within the scope of the Maintenance Rule (loss of material, cracking, and change in material properties), are detected by visual inspection of external surfaces prior to the loss of structural or structural component intended function(s).

Monitoring and Trending – Aging effects are detected by visual inspection of structures and structural components conducted every five years in accordance with the requirements of the Maintenance Rule. The ITMR structures and structural components are visually inspected via walkdowns from accessible floors, platforms or other permanent vantage points. Procedure ES-437 provides the guidance and process for performing and documenting these inspections. The degree of examination depends on many factors such as accessibility, environmental and radiological conditions, and safety. In cases of inaccessibility, sampling approaches such as plant specific characteristics, industry wide experience and testing history may be evaluated instead of actual visual inspection in areas of similar environmental and/or service conditions.

Inspection results are documented by the evaluating structural engineer(s) utilizing ES-437, Attachments I and II, which are assembled into a technical report in accordance with ES-437. These results are compared and trended to previous inspection data. Relevant industry experience is periodically obtained via industry network communications / notifications which will be used to assess the effectiveness of the MR program and to gain insights on specific areas of concern which may need to be incorporated or enhanced.

Acceptance Criteria – The acceptance criteria and guidelines for the inspection of ITMR structures and structural components are specified in ES-437. "Acceptable" structures or structural components are those that are free of deficiencies or degradation which could lead to possible failure and are capable of performing their intended function(s), including protection or support of ITMR systems and components until the next scheduled inspection, and are considered to meet the criterion associated with 10 CFR 50.65(a)(2). "Acceptable with Deficiencies" structures or structural components are those that are capable of performing their intended function(s), including the protection or support of ITMR systems and components, but are degraded or have deficiencies which could deteriorate to an unacceptable condition if not evaluated or corrected prior to the next scheduled inspection. "Unacceptable" structures or structural components are those that are damaged or degraded to the extent that they are no longer capable of performing their intended function(s), including the protection or support of ITMR systems or components.

Corrective Actions – Structures or structural components that are identified as "Acceptable with Deficiencies" or "Unacceptable" are evaluated by the Plant Support Engineering (PSE) Building Service Engineer to determine the appropriate corrective action(s).

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner. During subsequent inspections, engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

Administrative Controls – The Maintenance Rule inspections and assessments of ITMR structures and structural components are conducted and implemented through ES-437, ES-514 and SAP-1252. Visual inspections of roofs in the protected area are conducted semi annually in accordance with CMP-700.005.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – An initial baseline position was established at VCSNS for the acceptability of the ITMR structures to remain capable of providing their ITMR functions over the life of the plant in accordance with the Maintenance Rule. This baseline position documented numerous periodic inspections and surveillances that were performed on certain ITMR structures in accordance with existing regulatory or licensing commitments at VCSNS during the period of 1993 to 1996. This baseline position, "Assessment of In-Service Conditions of Important to Maintenance Rule (ITMR) Structures", dated June 11, 1996 (revised on September 19, 1996), is documented in Enclosure 4 of VCSNS Technical Report TR00010-001, "V.C. Summer Nuclear Station Maintenance Rule Summary Document". [Reference 126]

VCSNS Technical Reports TR00010-002 and TR00010-03 document the overall periodic assessments of the VCSNS Maintenance Rule program as required by 10 CFR 50.65 (a)(3), ES-514 and SAP-1252. These reports cover the periods from July 10, 1996 (effective date for Maintenance Rule) through the end of RF-10 (November 7, 1997) and from the end of RF-10 to the end of RF-11 (May 11, 1999), respectively.

The 1996 baseline assessment concluded that the ITMR structures and structural components were acceptable and were free of deficiencies or degradation, which could lead to possible failure. Therefore, these structures were determined to be capable of performing their structural functions, including the protection and support of Safety Related systems and components.

The MR inspection report completed in 2000 noted that in general, most of the ITMR structures and structural components were evaluated to be "acceptable" with regards to continued function. However, 9 items/areas were identified as "Acceptable with Deficiencies" that exhibited a trend of accelerated aging, with increased degradation mechanisms that could deteriorate to an unacceptable condition if not corrected prior to the next scheduled inspection in 2005. These conditions mostly deal with rust/corrosion due to weathering, water in-leakage and ponding. None of the conditions have an immediate adverse effect on the ability of the structures or components to perform their intended function(s). CER 01-1011 was originated which requires the PSE Building Services Engineer to evaluate these conditions for corrective actions. Each item/area will be tracked via supplemental CERs [Reference 156].

Unless more frequent inspections are specified, the ITMR structures and structural components will subsequently be inspected every five years. Beyond the periodic inspections and surveillances currently performed on certain ITMR structures, the next complete Maintenance Rule inspection is scheduled to be completed in 2005.

The inspection requirements in support of the Maintenance Rule have been in effect since 1996 and have proven effective at maintaining structures and structural components. Visual inspections of ITMR structures and structural components conducted under the VCSNS Maintenance Rule Structures Program is considered acceptable based on industry standards and operating experience. A review of the MR program provides reasonable assurances that the aging effects for the ITMR structures and structural components will be managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Maintenance Rule Structures Program has been compared to the "Structures Monitoring Program" documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding "Structures Monitoring" is consistent with the VCSNS program. The VCSNS Maintenance Rule Structures Program contains those attributes delineated in GALL Chapter XI.S6, which have been determined by the NRC to provide an acceptable aging management program.

7.13 Material Handling System Inspection Program

Loss of material due to corrosion is identified as an aging effect requiring programmatic management for crane rails and girders for the period of extended operation. The Material Handling System Inspection Program is credited with managing loss of material for the steel rails and girders within the scope of license renewal. The Material Handling System Inspection Program has been in effect for many years at VCSNS and is based on guidance contained in ANSI B30.2 [Reference 127] for overhead and gantry cranes. The Material Handling System Inspection Program includes Nuclear Safety Related and Quality Related material handling systems and is a condition monitoring program.

Material handling systems steel support structures (rails, runways, monorails, girders, jib cranes, seismic restraints, and associated connections) are designed in accordance with AISC [Reference 128] and inspected in accordance with guidance provided by ANSI N45.2.15 [Reference 129]. Steel support structures are inspected for attributes delineated in Engineering Services Procedure, ES-437 [Reference 124] and these attributes address the ANSI N45.2.15 requirements.

The regulatory basis for periodically inspecting Nuclear Safety Related and Quality Related cranes is found in NUREG-0612 [Reference 130, Article 5.1.1] and OSHA 29 CFR Chapter XVII [Reference 131, Part 1910.179(j)(3)]. The Material Handling Systems Inspection Program is evaluated in accordance with the guidance provided in NEI 95-10 [Reference 4]. The review provides reasonable assurance that the effects of aging are adequately managed for VCSNS Nuclear Safety Related and Quality Related cranes, rails, and girders and their intended function is consistent with the CLB for the period of extended operation.

Scope of Program – Material handling systems for the Reactor Building, Auxiliary Building, Fuel Handling Building, Service Water Pump House, Intermediate Building, and Diesel Generator Building (subject to NUREG-0612) are listed in the Technical Evaluation Report EGG-HS-6371 [Reference 132, Tables 2.1, 2.2, and 3.1].

Nuclear Safety Related cranes are listed in FSAR [Reference 12], Table 3.2-1 as Safety Class 1 and are listed as follows:

NSR CRANE	BUILDING	DESCRIPTION OF CRANE
XCR-2-FH	Fuel Handling	Spent Fuel Pit Bridge Crane (Fuel Handling Machine) by Dwight-Foote @ 2 Ton
XCR-3-FH	Fuel Handling	Fuel Handling Building Crane by Whiting @ 125 Ton
XCR-4-FH	Reactor	Reactor Building Polar Crane by Whiting @ 360 Ton

Cranes and hoists were identified using engineering drawing E-520-010, VCSNS Technical Report TR03920-002 [Reference 133] and Technical Evaluation Report EGG-HS-6371, Tables 2.1, 2.2 and 3.1. Cranes and hoists are designated "Not in License Renewal Scope" when engineering drawing E-520-010 indicates that all parts (hoist, trolley, and runway structure) for a particular material handling system are Non-Nuclear Safety Related (NNS). A NNS designation for the entire material handling system means that no part of the system is "anti-fall down".

Other cranes and hoists in safety related buildings are Quality Related (anti-fall down), yet may or may not be classified as heavy load (loads > 2500 lbs.) material handling equipment as defined by NUREG-0612. Cranes and hoists in safety related buildings that are Quality Related (anti-fall down) are as follows:

BUILDING	CRANE	DESCRIPTION OF CRANE
Auxiliary	XCR-21A-AB and XCR-21B-AB	Two Containment Spray Pump Monorail Hoists & Trolleys @ 5 Ton Each
Auxiliary	XCR-54A-AB, XCR-54B-AB, and XCR-54C-AB	Three Charging Pump Monorail Hoists & Trolleys @ 5 Ton Each
Auxiliary	XCR-23A-AB and XCR-23B-AB	Two Equipment Monorail Hoists & Trolleys above Equipment Hatches @ 2 Ton Each
Auxiliary	XCR-20A-AB and XCR-20B-AB	Two Residual Heat Removal Pump Monorail Hoists & Trolleys @ 5 Ton Each
Diesel Generator	XCR-29A-DG and XCR-29B-DG	Two Manual Hoists and Trolleys for Diesel Generator @ 2 Ton Each
Intermediate	XCR-40A-IB, XCR-40B-IB, and XCR-40C-IB	Three Main Steam Isolation Valve Hoists & Trolleys @ 10 Ton Each
Intermediate	XCR-39-IB	Main Steam Isolation Valve Hoist & Trolley by American Chain & Cable Co. @ 5 Ton
Intermediate	XCR-33-IB	One Turbine Driven Emergency Feedwater Pump Hoist @ 2 Ton shown on E-303-220
Reactor	XCR-55-RB, XCR-56-RB, XCR-57-RB and	Three Reactor Coolant Pump Jib Cranes by Wright @ 1 Ton Each
Reactor	XCR-58-RB	Upper Storage Stand Jib Crane by Wright @ 1 Ton
Reactor	XCR-1-FH	Reactor Cavity Manipulator Crane by Stearns & Rogers at 2 Ton (Refueling Machine)
Service Water Pump House	XCR-50-SW	Travelling Screen Hoist by American Chain & Cable Co. @ 10 Ton

The regulatory basis for periodically inspecting Nuclear Safety Related and Quality Related cranes is found in NUREG-0612, Article 5.1.1 and OSHA 29 CFR Chapter XVII, Part 1910.179(j)(3).

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation. The Material Handling System Inspection Program is a condition monitoring program.

Parameters Monitored or Inspected – The inspection attribute credited for license renewal is to ensure no indication of loss of material due to corrosion of rails, rail supports, and structural supports. Although not credited for license renewal, other attributes such as member misalignment, loose bolts, cracks, loose members, and elongation of holes are inspected for Nuclear Safety Related or Quality Related (anti-fall down) material handling systems. Parameters inspected for Quality Related material handling systems are in accordance with TRP-14 [Reference 134]. Specific attributes are provided in maintenance procedures for specific material handling equipment. Examination guidelines are specified in ANSI B30.2 and implemented by the appropriate maintenance procedure for the particular crane.

Inspection of overhead and gantry cranes are implemented by the following maintenance procedures:

1. Mechanical Maintenance Procedures:

   - MMP-165.007 – Polar Crane (XCR-4-FH) and Spent Fuel Cask Handling Crane (XCR-3-FH)
   - MMP-165.010 – Spent Fuel Pit Bridge Crane (a.k.a. Fuel Handling Machine) (XCR-2-FH) Maintenance
   - MMP-165.013 – Polar Crane (XCR-4-FH) Maintenance / Inspection
   - MMP-165.006 – Periodic and Annual Inspection of Miscellaneous Hoists and Cranes

2. General Maintenance Procedures:

   - GMP-100.011 – Crane Operation in Reactor Building
   - GMP-100.012 – Crane Operation in Fuel Handling Building
   - GMP-102.001 – Control Use and Inspection of Rigging Equipment

3. Electrical Maintenance Procedures:

   - EMP-165.001 – Polar Crane Preventive Maintenance
   - EMP-165.002 – Fuel Building Cranes Preventive Maintenance
   - EMP-165.004 – Overhead Cranes Preventive Maintenance

Steel support structures are inspected for attributes delineated in Engineering Services Procedure ES-437, and these attributes address ANSI N45.2.15 requirements and aging effects.

Detection of Aging Effects – The Material Handling System Inspection Program detects loss of material due to corrosion prior to loss of structure or component intended function.

Monitoring and Trending – Aging effects for material handling systems are detected by visual inspection of crane structural components and steel support structures.

The regulatory basis for periodically inspecting Nuclear Safety Related and Quality Related cranes is found in NUREG-0612, Article 5.1.1 and OSHA 29 CFR Chapter XVII, Part 1910.179(j)(3).

Nuclear Safety Related cranes are listed with the following inspection frequency:

CRANE	DESCRIPTION OF CRANE	FREQUENCY
XCR-2-FH	Spent Fuel Pit Bridge Crane by Dwight-Foote @ 2 Ton	Prior to use and every 18 months
XCR-3-FH	Fuel Handling Building Crane by Whiting @ 125 Ton	Prior to use and every 18 months
XCR-4-FH	Reactor Building Polar Crane by Whiting @ 360 Ton	Prior to use and every 18 months

Results of material handling inspections are retained in sufficient detail to permit adequate confirmation of the inspection program. In particular these records identify inspectors, results of the inspection, discrepancies with the cause, and corrective action. No actions are taken as part of the Material Handling Inspection Program to trend inspection results.

Acceptance Criteria – The acceptance criteria for cranes and crane rails are no excessive deformation of primary load members and no visual indication of any loss of material due to corrosion as stated in VCSNS maintenance procedures for the specific material handling equipment. Roughness or general corrosion that does not reduce the load bearing capacity (intended function of the member) is an example of a non-relevant condition.

Corrective Actions – Structures and components that do not meet the acceptance criteria are evaluated by engineering for continued service.

A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls –Inspections of cranes are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – Through monitoring effectiveness of maintenance at nuclear power plants there has been no corrosion-related degradation that has impaired cranes. Cranes have not operated beyond their design lifetime so there are no significant fatigue-related structural failures. [Reference 5]

NRC Inspection Report 50-395 / 82-28 [Reference 135] documents review and approval of the VCSNS material handling system maintenance procedures and concludes that the procedures comply with NUREG-0612 requirements.

VCSNS Reactor Building (RB) Design Basis Document [Reference 67] discusses 10 CFR 21 violations associated with the design of the Reactor Building polar crane (XCR-4-FH) prior to 1996. NCN 4036 identified overstressed bolted connections in the trolley. The polar crane vendor (Whiting) identified this nonconformance as a design deficiency. Disposition of NCN 4036 replaced the overstressed bolts with high strength ASTM A325 bolts during refueling outage RF7. Structural Calculation DC03920-012 provided the technical justification for the disposition of NCN 4036. The nonconformance was a design deficiency identified by the manufacturer and not related to aging.

While evaluating the polar crane for handling the replacement steam generators, polar crane vendor (Whiting) identified a design deficiency on several areas of the trolley and bridge girders in accordance with 10 CFR 21. NCN 4905 was initiated identifying overstressed areas of the trolley and bridge girders. Structural Calculations DC03920-014 and DC03920-016 provided the technical justification for the disposition of NCN 4905. Repairs are delineated on the disposition to NCN 4905. Repairs were completed prior to refueling outage RF-8. The nonconformance was a design deficiency identified by the manufacturer (Whiting) and not related to aging.

VCSNS Engineering Technical Work Record (TWR) 10203 [Reference 136], Maintenance Rule review of industry operating experience for the MH system, documents a review of industry operating experience for material handling (MH) systems. A total of 15 events are reported and 8 events are relevant to VCSNS. This TWR concludes that system design and operating procedures anticipate these events. Based on the TWR, no licensee event reports were initiated for the material handling system structures at VCSNS. This is confirmed using the Licensing Database, using search terms ("cranes", "monorails", "rails", "girders").

A Non-Conformance Notice (NCN) within the period of review is as follows:

CRANE	NCN	SUBJECT	DISPOSITION
XCR-2-FH	00-0086	Catastrophic Failure of Roller Guide Bearing due to age related stress corrosion cracking	Replaced Four (4) Roller Guide Bearing Assemblies and recommended replacing Four (4) Roller Guide Bearing Assemblies every Tenth Outage. Probable cause for catastrophic failure was long period of inactivity in humid environment as stated in Technical Work Record (TWR) Disposition for NCN 00-0086 dated March 2, 2000.

This Nonconformance Notice condition does not affect structural integrity or function of the Spent Fuel Bridge Crane (XCR-2-FH) as stated in Disposition 1 for NCN 00-0086 dated March 2, 2000. Therefore this condition is not an aging effect since the intended function (crane structural integrity) is maintained.

No Condition Evaluation Reports were identified for material handling systems relevant to aging.

The VCSNS Material Handling System Inspection Program includes the key elements of an effective program (as identified in NEI 95-10) which are necessary to ensure that cranes, rails, and girders continue to perform their intended functions for the period of extended operation consistent with the CLB. The Material Handling System Inspection Program demonstrates the capability to detect and manage loss of material. Based on operating experience, it is reasonable to expect that the continued implementation of the Material Handling System Inspection Program in compliance with NUREG-0612, Article 5.1.1 will manage aging effects. Therefore, the intended functions of cranes, rails, and girders are maintained consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Material Handling System Inspection Program has been compared to the program documented in the GALL Report [Reference 5]. Based on a review of the system, components, material, environment, applicable aging effects, and operating experience, the information in the GALL Report regarding the Material Handling System Inspection Program is consistent with the VCSNS program. The VCSNS Material Handling System Inspection Program contains those attributes delineated in GALL Chapter XI.M23 which have been determined by the NRC to provide an acceptable aging management program for the bridge and trolley structural members and the rails. The polar crane girders and brackets are covered by the Maintenance Rule Structures Program and match attributes in GALL Chapter XI.S6.

7.14 Pressure Door Inspection Program

Examination guidelines and results of periodic inspections are presented for pressure doors. VCSNS pressure doors are Nuclear Safety Related and Quality Related. Pressure door inspections are a condition monitoring program. Most Nuclear Safety Related pressure doors are also fire doors and are inspected for attributes delineated in TRP-2 [Reference 66]. For Quality Related pressure doors, TRP-12 [Reference 137] specifies aging effects as loss of material of doors and loss door hardware. Excessive wear for door appurtenances such as latches, gaskets, hinges, sills, and closing devices are inspection attributes but are not credited for license renewal. Quality Related pressure doors inspection attributes are described in TRP-12 as follows:

- Freedom of movement
- Function (closed during normal plant operation)
- Structural deterioration

Pressure doors are required to be operable in Plant Operating Modes 1, 2, 3, and 4 (as defined by General Test Procedure, GTP-702). The surveillance requirement for Quality Related pressure doors is in TRP-12. The surveillance requirement for Nuclear Safety Related pressure doors (fire doors) is established in FPP-025 for fire doors. The surveillance requirements include monitoring of door position and visual inspection that the door is closed and not impaired.

Scope of Program – The need to maintain pressure barriers (which also serve as fire barriers) is contained in Branch Technical Position (BTP) APCSB 9.5-1, Appendix A per the commitments in the FPER [Reference 109]. The surveillance requirement is established in FPP-025. Nuclear Safety Related pressure resistant doors are specified in SP-631 [Reference 122]. There are 34 doors that are Nuclear Safety Related pressure resistant doors. Quality Related pressure doors are listed in TRP-12. Thirteen (13) doors are Quality Related pressure doors. There are 47 doors that are rated as pressure resistant. The door schedule is listed on architectural drawings D-108-011, D-108-012, D-108-013, and D-108-018. The vendor (Sonicbar Division of Rysdon Products) manual is 1MS-94B-808.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation. The Pressure Door Inspection Program is a condition monitoring program.

Parameters Monitored or Inspected – Parameters monitored for Nuclear Safety Related pressure doors are loss of material of doors and door hardware from TRP-2. Parameters monitored for Quality Related pressure doors are loss of material of doors and door hardware from TRP-12. Excessive wear for door appurtenances such as latches, gaskets, hinges, sills, and closing devices are additional attributes in each TRP, but are not credited for license renewal.

Detection of Aging Effects – The pressure door inspection program detects structural damage or degradation, including loss of material due to corrosion prior to loss of structure intended function.

Monitoring and Trending – Aging effects for Quality Related pressure doors are detected by a visual examination of the door and frame and functional testing for closure as stated in Civil Maintenance Procedure CMP-700.009 every 18 months. Aging effects for Nuclear Safety Related pressure doors are detected by visual examination every six months of the door and frame and functional testing for closure as stated in STP-128.019. No actions are taken as part of this program to trend inspections or test results.

Acceptance Criteria – Quality Related pressure door acceptance criteria is provided in TRP-12. Nuclear Safety Related pressure door acceptance criteria is provided in STP-128.019. Acceptance criteria for self-closing doors are that hinges are intact with all screws tight, pins in good condition, and the door closes. Acceptance criteria for double self-closing doors are that bolts are in good condition, the astragal (metal molding strip) is in good condition, and the door closes. Automatic closing doors are checked to be in good operating condition and the door closes. Acceptance criteria for hollow pressure doors are no holes and no damage in the skin of the door or the frame.

Corrective Actions – The Condition Evaluation Report (CER) process is initiated for pressure doors that do not meet the acceptance criteria as stated in CMP-100.008. A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Minor abnormalities (loose knobs, latches or other appurtenances) are repaired using guidance provided by the vendor (Sonicbar Division of Rysdon Products) manual 1MS-94B-808 and do not require initiating a CER as stated in CMP-100.008.

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The need to maintain pressure barriers (which also serve as fire barriers) is contained in Branch Technical Position (BTP) APCSB 9.5-1, Appendix A per the commitments in FPER, Section 5.0.C.6. The surveillance requirements are established in FPP-025 [Reference 110].

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – VCSNS has no failures or adverse trends for Nuclear Safety Related or Quality Related pressure doors. Inspections in the last five years do not identify any nonconformances relative to the acceptance criteria.

LER 88-008 discusses steam propagation into sensitive rooms through fire door DRIB-407. Initial corrective action reinforced and sealed fire door DRIB-407 to prevent steam propagation. Final corrective action replaced DRIB-407 (implemented by MRF-32976) with a Quality Related pressure resistant / fire door and added a Quality Related pressure resistant door DRAB-515 (implemented by MRF-32995 during RF-5). LER 88-008 reports a design deficiency that is not related to aging. LER 88-008 initiated NCN # 2976 that identified a design deficiency. NCN # 2995, NCN # 3000, and NCN # 3348 provided the corrective actions. LER 88-008 corrective actions were reviewed by the NRC as part of Inspection Report 91-11 [Reference 138] Item 2e and NRC concurred with the corrective actions.

Review of each pressure door LER and Inspection Report did not identify any effects related to aging.

No Non-Conformance Notices or Condition Evaluation Reports were initiated for pressure doors relevant to aging.

The frequency of inspections in effect (since the initial implementation of the Technical Specifications requirements in 1984) is acceptable based on industry operating experience. A review of pressure door inspections confirms the reasonableness and acceptability of this inspection frequency such that any degradation of a door is detected prior to loss of function.

If the results of the visual inspection indicate that repairs are required, then specific repairs are made in accordance with CMP-100.008. The pressure door inspections are implemented by plant procedures and controlled by the SCE&G Quality Assurance Program.

Based on operating experience, continued implementation of Nuclear Safety Related pressure door (fire door) inspections in compliance with FPP-025, manages identified effects of aging throughout the period of extended operation.

Based on operating experience, continued implementation of Quality Related pressure door inspections in compliance with TRP-12 manages identified effects of aging throughout the period of extended operation. The intended functions of pressure doors are maintained.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS pressure door inspection program which monitors Nuclear Safety Related and Quality Related pressure doors is not evaluated in the GALL Report [Reference 5] and is, therefore, plant specific to VCSNS.

7.15 Service Water Pond Dam Inspection Program (North Dam, South Dam, East Dam and West Embankment)

Loss of material (erosion), alignment (movement), surface cracking and seepage have been identified as aging effects for the Service Water Pond (SWP) Dams and West Embankment, which require programmatic management for the extended period of operation. Inspections are conducted to satisfy the requirements of Regulatory Guide (RG) 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants". Additionally, the submerged slope stability of the West Embankment in the vicinity of the Intake Structure requires monitoring as specified by VCSNS Operating License Number NPF-12, Condition 2.C.5. The VCSNS five year visual inspection and survey monitoring of the SWP Dams and West Embankment (conducted in accordance with Engineering Service Procedure, ES-400, "Service Water Pond Structure and Dam Inspections" and Civil Maintenance Procedure, CMP-700.001, "Survey Monitoring", respectively) are credited with managing these aging effects. Survey monitoring data is reviewed in accordance with Design Engineering Guideline CV-01, "Survey Monitoring Data Review".

In addition to the RG 1.127 inspections conducted every five years, SCE&G has conducted annual walkdowns of the SWP Dams since 1999. These inspections are scheduled for the fall of each year by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159]. Furthermore, the NRC/FERC has initiated inspections of the SWP Dams and West Embankment every 2 to 3 years. The RG 1.127 inspections and the elevation/alignment/slope surveys are documented in Structural Calculations DC02210-001 and DC02210-002. The SCE&G "Engineers Technical Work Record" program is used to document the results of the annual inspection.

The five year RG 1.127 inspection and the PMTS annual visual inspection are evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for the SWP Dams and West Embankment so that their intended functions will be maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation. The visual inspections and survey monitoring of the SWP Dams conducted in accordance with ES-400 and CMP-700.001 respectively, are condition monitoring programs.

Scope of Program – The scope of the Service Water Pond Dam Inspection Program includes the North Dam, South Dam, East Dam and West Embankment. These earthen structures are shown on the E-726-400 series drawings.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation. The RG 1.127 inspection of the Service Water Pond Dams and West Embankment is a condition monitoring program.

Parameters Monitored or Inspected – Guidelines for visually inspecting the SWP Dams and West Embankment as specified in ES-400 are:

1. Inspect earthen structures for:
   - Alignment (movement)
   - Surface cracks
   - Sloughing
   - Erosion

2. Inspect riprap for:
   - Sloughing
   - Movement
   - Boundaries at ends of structures for erosion undercutting and movement
   - Excessive growth of weeds, grass, brush or trees

3. Inspect surrounding (natural terrain) for:
   - Excessive growth of brush and trees within 100 feet of riprap areas
   - Areas of erosion that may effect the dams

4. Review of the elevation, alignment and slope survey monitoring data from CMP-700.001 in accordance with ES-400 and CV-01.

Elevation and alignment monitoring of the SWP North Dam, South Dam and West Embankment and slope survey monitoring of the West Embankment are conducted in accordance with CMP-700.001 per the survey frequencies specified in ES-400. The survey monitoring data is reviewed in accordance with CV-01 to ensure that elevation / alignment changes remain within established criteria.

Detection of Aging Effects – Attributes associated with aging for the Service Water Pond Dams and West Embankment are detected by the RG 1.127 inspections (conducted on a five year frequency), with the next inspection scheduled for 2005. These RG 1.127 inspections include survey monitoring of the elevation/alignment monuments of the North Dam, South Dam and West Embankment, along with slope surveys of the West Embankment. In addition, a NRC/FERC Dam inspection (conducted every 2 to 3 years with the last inspection conducted July 25, 2001) also detects attributes associated with aging. Furthermore, an annual visual inspection of the SWP Dams is scheduled through the PMTS program.

Inspection of the Service Water Earthen Embankments will also be performed following the occurrence of an operating basis earthquake or any other significant natural phenomena and if draining (planned or unplanned) of the Monticello Reservoir should occur.

Monitoring and Trending – Inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The elevation/alignment/slope surveys and the RG 1.127 visual inspection documentation are filed in Structural Calculations DC02210-001 and DC02210-002, respectively. In particular these records identify the inspection team, review of previous inspection results, the results of the current inspection, whether or not the results were acceptable, discrepancies and their cause and any corrective action resulting from these inspections. Any problems or concerns observed during the inspections are documented in the applicable calculation.

Acceptance Criteria – The acceptance criteria for the SWP Dams and West Embankment inspections are specified in procedure ES-400 and guideline CV-01.

The acceptance criteria for the RG 1.127 visual inspection of the SWP Dams and West Embankment (as specified in ES-400) is that any problems or concerns observed during the inspections should be documented using the following guidelines:

- CER with action to Plant Facilities for weed, brush or tree removal
- An engineering evaluation may conclude that an increased frequency of inspection is required to monitor potential problems
- Determine if NRC notification is necessary
- Generate a CER for corrective actions (other than cosmetic touch-up) of dams or riprap, and in cases where survey monument(s) are damaged

The acceptance criteria for the survey monitoring of the North Dam, South Dam and West Embankment as specified in CV-01 is that the survey data should be evaluated for any trends in movement. Any vertical or horizontal changes that exceed 1/2 inch (0.04 feet) since the previous survey should be identified and further evaluations performed.

The acceptance criteria for the underwater slope survey monitoring of the West Embankment as specified in CV-01 is that the survey data should be evaluated for any trends in movement. Any changes in 3 adjacent survey points of more than 2.0 feet should be identified and further evaluations performed.

Corrective Actions – Each inspection notes recommendations concerning repairs or studies. Engineering prepares a plan to address these recommendations. Structures and components that do not meet the acceptance criteria are evaluated by engineering personnel for continued service and repair as required. Structures and components that are deemed unacceptable are documented under the VCSNS corrective action program. A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The SWP Earthen Embankments Inspections are governed by RG 1.127 and VCSNS Operating License NPF-12, Condition 2.C.5, and implemented through Engineering Service Procedure, ES-400; Civil Maintenance Procedure, CMP-700.001; and Design Engineering Guideline, CV-01.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – During each inspection of the SWP Dams and West Embankment a review of the previous inspection's observations/recommendations is performed and the current status (such as repairs implemented or continued monitoring) is documented. Previous abutment erosion control modifications completed in 1989 significantly reduced earlier erosion problems overall, as noted in the 1990 and 1995 RG 1.127 inspections. Additional grading of diversion trenches/berms to direct rainwater away from the dams has further controlled erosion. There are currently no erosion areas that have a direct impact on any of the earthen structures. Weed, brush and sapling growth are controlled via cutting or annual spraying of herbicides conducted in accordance with Facilities Procedure, FP-111.002, "Application of Herbicide".

Structural Calculation DC02210-001 documents the results of the survey monitoring data for the SWP North, South Dam and West Embankment. It provides a review of the vertical and horizontal displacements of the SWP North Dam and South Dam since 1977. It also provides a review of the vertical displacement of the West Embankment since 1978 and the horizontal displacement of the West Embankment since 1983. For the 2000 survey, all vertical and horizontal displacements were within the acceptance criteria as compared to the previous survey and found to be acceptable. In addition to the information stated above, Structural Calculation DC02210-001 provides a review of the slope survey of the West Embankment since 1983. For the 2000 survey, all of the measurements were within the acceptance criteria as compared to the previous survey and found to be acceptable. No further evaluations were required and no unusual trends were noted.

In addition to the NRC required five year inspection of the SWP Dams, FERC conducted inspections of the SWP Dams on February 27, 1997; July 27, 1999 and July 25, 2001. The conclusions reached by these inspections were that no significant conditions were observed that were considered detrimental to the safety of the Dams. The FERC inspection reports were submitted to VCSNS via the following NRC correspondence:

- 1997 inspection –NRC letter to G. J. Taylor (dated February 27, 1998) subject "Service Water Pond Dam Safety Inspection Results for Virgil C. Summer Nuclear Station" [Reference 139]

- 1999 inspection –NRC letter to S. A. Byme (dated September 19, 2000) subject "Results of Dam Safety Inspection Related to the Category I Service Water Pond Dams at the Virgil C. Summer Nuclear Station" [Reference 140]

- 2001 Inspection – (submittal of report to VCSNS pending)

The 1997 FERC Dam safety inspection report [Reference 139] recommended that SCE&G visually inspect the SWP Dams and West Embankment annually and test the accessible piezometers. SCE&G agreed to conduct and document these inspections [Reference 141]. The annual visual inspection is scheduled through the PMTS program for the fall of each year. The first annual visual inspection and testing of the accessible piezometers was conducted in November 1999. Three accessible piezometers located along the crest of the North Dam were tested and found to be functional. The results confirmed the expected equilibrium water level in the North Dam as a medium between the Monticello Reservoir and the Service Water Pond elevations. SCE&G will continue to monitor piezometer levels in accordance with ES-400.

Visual inspections of the SWP Dams and survey monitoring of the SWP North Dam, South Dam and West Embankment have been in effect since the initial implementation of the facility operating license at VCSNS and are considered acceptable based on industry operating experience. A review of the SWP Dams inspection and monitoring programs confirms the reasonableness and acceptability of these inspection frequencies in that degradation of the earthen structures is detected prior to loss of function. The program provides reasonable assurances that the aging effects for the earthen structures will be managed so that their intended functions will be maintained consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Service Water Pond Dam Inspection Program (North Dam, South Dam, East Dam and West Embankment) has been compared to the "Inspection of Water-Control Structures Associated with Nuclear Power Plants" program documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding the inspection of water-control structures at nuclear power plants is consistent with the VCSNS program. The VCSNS Service Water Pond Dam Inspection Program contains those attributes delineated in GALL Chapter XI.S7, which have been determined by the NRC to provide an acceptable aging management program.

7.16 Service Water Structures Survey Monitoring Program (SWPH, SWIS, Electrical Duct Banks and SW Intake Line "A")

Survey monitoring is required for structures that are supported by earthen fill material and that have exhibited the potential for settlement. Settlement is not considered adverse unless it imposes stresses on structures that may exceed their design capacities. Initial settlement of the SWPH and SWIS was much more than the original pre-construction estimates. As a result, semi-annual survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are conducted in accordance with Engineering Service Procedure, ES-400, "Service Water Pond Structure and Dam Inspections" to satisfy the requirements specified by Operating License Condition 2.C.5 and FSAR Section 2.5.4.10.6.2. The purpose of the surveys is to monitor any differential in vertical and horizontal displacement. The surveys are conducted in accordance with Civil Maintenance Procedure, CMP-700.001, "Survey Monitoring" twice a year for the life of the plant, typically during January and July (± 30 days). Attachments I through IV of CMP-700.001 are used to document the survey data and transmit this data to Design Engineering. The survey data is reviewed per ES-400 using Design Engineering Guideline CV-01, "Survey Monitoring Data Review" and documented in Structural Calculations DC03650-004 and DC0360B-006.

The survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 - The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for these structures so that their intended functions will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation. Survey settlement monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" conducted in accordance with ES-400 and CMP-700.001 is a condition monitoring program.

Scope of Program – Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" conducted in accordance with ES-400, includes four settlement monitoring points for the SWPH, 3 monitoring masts for the SWIS, 2 differential measurement points for each of the three Electrical Duct Banks entering the SWPH, and 7 survey rods attached to SW Intake Line "A" that extend to the yard surface. These survey points are shown and documented on Attachments I through IV, of CMP-700.001, respectively. Equations for calculating the vertical displacement of the SWPH and the vertical and differential displacements of the SWIS can be found in Design Engineering Guideline CV-01.

Preventive Actions – No actions are taken, as part of the settlement survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A", which prevent aging effects or mitigate aging degradation. Survey settlement monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is a condition monitoring program.

Parameters Monitored or Inspected – Survey monitoring (to detect any vertical and/or horizontal movement associated with settlement) of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A", is conducted in accordance with CMP-700.001 per the survey frequencies specified in ES-400. The survey monitoring data is reviewed by Design Engineering in accordance with CV-01 to ensure that settlements remain within established criteria.

In addition to survey monitoring, the structures are visually inspected in accordance with ES-400 for the following:

SWPH	movement, alignment or sloughing, cracking, settlement, and structural degradation
SWIS	cracking (per underwater diver's inspection)
SW Electrical Duct Bank	differential movement and integrity of the expansion joint material
SW Intake Line "A"	ground above is inspected for settlement, sloughing, surface cracking, and erosion

Detection of Aging Effects – Attributes associated with aging for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are detected by the semi-annual settlement survey monitoring conducted in accordance with ES-400. The survey results are reviewed and evaluated for trends in movement associated with settlement that exceeds the acceptance criteria specified in CV-01. This review and the visual inspection of the structures per ES-400 will detect any adverse horizontal or vertical displacements prior to the loss of structure intended function(s).

Monitoring and Trending – Aging effects associated with settlement for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are detected by semi-annual survey monitoring per the requirements specified by Operating License Condition 2.C.5 and FSAR Section 2.5.4.10.6.2. Survey monitoring data is retained in sufficient detail to permit adequate confirmation of the inspection program. The survey data reports and reviews/evaluations are filed in Structural Calculations DC03650-004 and DC0360B-006. In particular these records identify the person(s) performing the survey, the structure/component and points surveyed, the person(s) reviewing/evaluating the survey data, whether or not the results are acceptable, discrepancies and their causes, and any corrective action(s) taken as a result. Trending is accomplished by comparing the current survey data to the previous survey data and evaluating for trends in movement that exceed the acceptance criteria thru data plots for the SWPH and Service Water Intake Line "A".

Acceptance Criteria – The acceptance criteria and guidelines for reviewing the survey settlement data for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are specified in CV-01. Survey results are evaluated for adverse trends in vertical displacement. The measurements are compared to the previous survey results and any change in movement that is equal to or greater than 1/4 inch is evaluated by engineering to determine the cause. The review of the SWPH and SWIS also compares survey data from Point "A" on the SWPH with the survey data of monitoring mast MM-3 on the SWIS because of their proximity to one another. The SWIS is also monitored for differential displacement between the middle and ends of the tunnel. The acceptance criterion for the differential displacement is 0.083 foot (1 inch). If the 0.083 foot differential is reached or exceeded, further engineering evaluations are required.

Corrective Actions – Any settlement of structures or components that exceeds the established acceptance criteria is evaluated for adverse trends to determine whether or not there is a potential problem. Corrective actions may include increased frequency of inspection or further engineering evaluations to ascertain an exact cause for the movement. NRC notification is required for the following conditions:

- Acceptance criteria for the survey results are exceeded and a potential problem has been identified by additional engineering evaluation(s).
- Review of survey data shows significant movement(s) from baseline data that are unexplainable.
- Survey data or visual inspection identifies unusual deterioration of structures or components.

In these cases, a CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner. During succeeding surveys, Design Engineering reviews and compares the current survey data to preceding survey data to monitor any abnormalities and trends in movement. This review would also ensure the implementation and effectiveness of previously recommended corrective actions.

Administrative Controls – Periodic survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is governed by Operating License Condition 2.C.5 and FSAR Section 2.5.4.10.6.2 and implemented through ES-400. Survey data is collected in accordance with CMP-700.001and documented on Attachments I through IV. Design Engineering reviews the survey data in accordance with CV-01 and documents the results of the evaluation in Structural Calculation Series DC-0365.

Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – Initial settlement of the SWPH and SWIS was much more than the original pre-construction estimates. As a result, a licensing commitment was made and documented in FSAR Section 2.5.4.10.6.2 and Operating License Condition 2.C.5 to survey monitor the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" for settlement twice a year during the operating life of the plant.

Estimates of the SWPH and SWIS settlement prior to construction were 3 to 4 inches and 1-1/2 to 2 inches respectively. However, during construction these structures settled much more than had been estimated. The actual settlement of the SWPH and SWIS after construction was complete was in the region of 13 inches and 10 inches respectively. Additionally, the degree and manner of settlement also caused cracking to occur in the SWIS, however these cracks were subsequently repaired (grouted). A special settlement study was performed for the SWPH and SWIS. This study is documented in "Service Water Intake Structure Settlement Effects and Related Work", prepared by Gilbert Associates Incorporated and Woodward-Clyde Consultants in December 1977 [Reference12, Section 2.5.4.10.6.2]. There has been no significant settlement of the SWPH since December 1978, subsequent to filling the Service Water Pond in February 1978.

Since 1991, there have been two instances where movement of the SWPH exceeded the acceptance criteria of a measured difference of 1/4 inch from the previous survey. The first instance was in February 1991, a re-survey was conducted in March 1991 and it was determined that the initial survey data was in error. In the second instance (July 1994), the acceptance criterion was minimally exceeded by 0.002 inch. Considering survey process inaccuracy and seasonal fluctuations affecting data collection, the 0.252 inch total differential was not considered significant enough to warrant further evaluation. Survey results from 1977 to the present are documented in Structural Calculation DC03650-004.

There have been several instances since 1991 where the SWIS settlement acceptance criteria (of a measured difference of 1/4 inch) was exceeded from one survey to the next. Survey data from February 1991 was considered to be in error, as was the case with the SWPH survey data. Deviations noted between January 1994 and June 1997 consistently reflect seasonal changes between summer and winter surveys. Survey results since 1997 have shown that the average displacements from previous surveys are within the acceptance criteria. The net displacements for the SWIS survey points have remained relatively constant over the years and periodically reflect seasonal changes between surveys. Survey results from 1978 to the present are documented in Structural Calculation DC03650-004.

Survey monitoring for differential settlement (middle to ends) of the SWIS has been conducted since February 1985 with an acceptance criteria of 0.083 feet. Between the July 1985 survey and February 1986 survey of the SWIS there was an unexplained sudden increase in the recorded differential displacement from 0.0366 feet to 0.0640 feet (0.768 inches) for which no ready explanation could be found. As a result of this sudden change, the survey monitoring frequency was increased to monthly for a period of eight months and the results showed the differential movement remained steady at close to 0.064 feet. Consequently, the frequency of monitoring was returned back to semi annually. No further significant increase in differential movement has been recorded since February 1986 and the total settlement to date is within the acceptance limit of 0.083 feet (1.00 inch). Structural Calculation DC03650-004 documents the survey results from 1985 to the present.

No significant differential settlement was expected between the SWPH and incoming buried services as these were intentionally laid and connected to the SWPH after the major initial settlement during construction and the effects of filling the Service Water Pond in February 1978 had ceased. However, semi-annual survey data is recorded and evaluated.

Settlement of the Electrical Duct Banks is measured from inside the SWPH where the duct banks terminate on the inside face of the west wall of the SWPH. Historically, gap measurements have not undergone any significant changes since monitoring began, with any differential measurements well within the established acceptance criteria of a measured difference of 1/4 inch from the previous survey. Survey results from 1991 to the present are documented in Structural Calculation DC03650-004. Results prior to 1991 are documented in Structural Calculation DC0360B-006.

Service Water Intake Line "A" settlement has been monitored since January 1983. Since then there has been no appreciable movement or trend based on data reviews. However, there have been three occasions, one each in 1996, 1999 and 2000, when the acceptance criteria of a measured difference of 1/4 inch from the previous survey was minimally exceeded. These conditions are considered acceptable since the overall measurements remain within the general bounds of the long-term trend of data. These minor fluctuations may well be attributed to survey process imprecision, seasonal changes between summer and winter surveys, or ground water fluctuations. Survey results from 1991 to the present are documented in Structural Calculation DC03650-004. Results prior to 1991 are documented in Structural Calculation DC0360B-006.

Monitoring and trending the settlement of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" has been in effect since the initial implementation of the facility operating license at VCSNS and is considered acceptable based on industry operating experience. A review of the Service Water Structures Survey Monitoring Program (SWPH, SWIS, Electrical Duct Banks and SW Intake Line "A") conducted at VCSNS confirms the reasonableness and acceptability of the survey frequencies in that degradation of the structures / components will be detected prior to loss of function. The Service Water Structures Survey Monitoring Program provides reasonable assurances that the aging effects associated with settlement are managed so that the intended function(s) of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" will be maintained consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The Service Water Pumphouse (SWPH), Service Water Intake Structure (SWIS), Electrical Duct Banks and Service Water Intake Line "A" survey monitoring is not evaluated in the GALL Report [Reference 5] and is, therefore, plant specific to VCSNS.

7.17 Tendon Surveillance Program

The Tendon Surveillance Program described in Engineering Service Procedure, ES-438, "Containment Inservice Inspection Program" (CISIP), specification SP-228, "Surveillance of Reactor Building Post Tension System" and Surveillance Test Procedure, STP-160.001, "Containment Tendon Test" is appropriate in meeting the requirements of the 1992 Addenda of ASME Code, Section XI, Subsection IWL, as supplemented by the requirements of 10 CFR 55.55a(b)(2)(viii).

The tendon lift-off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required tendon force level exists in the Reactor Building. In order to assess the rate of force loss, the average lift-off force for a tendon is compared with 95% of the predicted force. The predicted force is calculated by subtracting the initial, time-dependent, and other losses where applicable from the original stressing forces, consistent with the recommendations of Regulatory Guide (RG) 1.35.1, Revision 3, dated July 1990 [Reference 142].

The measurement of Reactor Building tendon lift-off force, the tensile tests of the tendon wire, the visual examination of tendons, anchorage and exposed interior and exterior surfaces of the Reactor Building are sufficient to demonstrate the structural integrity of the Reactor Building.

The periodic surveillance testing of the Reactor Building tendons is a performance monitoring program.

The Tendon Surveillance Program is evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for the Reactor Building tendons so that their intended function(s) will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation.

Scope of Program – The scope of the Tendon Surveillance Program conducted in accordance with ES-438, SP-228, and STP-160.001 includes periodic testing of the Reactor Building dome, vertical, and hoop tendons to ensure that they are maintained above the minimum required prestressing forces.

Preventive Actions – Periodic testing of the Reactor Building tendons (i.e., dome, vertical and hoop) to maintain their prestress forces above the minimum required prestressing forces, also called minimum required value, will ensure that the structural and functional adequacy of the Reactor Building are maintained.

The tendon lift-off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) the minimum required tendon force level exists in the Reactor Building. In order to assess the rate of force loss, the average lift-off force for a tendon is compared with 95% of the predicted force. The predicted force is calculated by subtracting the initial, time-dependent, and other losses where applicable from the original stressing forces, consistent with the recommendations of RG 1.35.1, Revision 3, dated July 1990 [Reference 142].

Parameters Monitored or Inspected – The Reactor Building post tensioning system is inspected in accordance with ES-438, SP-228 and STP-160.001 for the following:

- Tendon group average lift-off force
- Individual tendon lift-off force
- Tendon wire physical damage
- Tendon ultimate tensile strength
- Tendon elongation
- Grease leakage evidence on Reactor Building surface
- Grease cap integrity
- Sheathing filler grease condition
- Tendon anchorage hardware integrity
- Tendon anchorage hardware corrosion
- Concrete condition near bearing plate

Detection of Aging Effects – The loss of prestressing forces in the Reactor Building post tensioning system is detected by periodic testing of the Reactor Building tendons. Currently a total of nine tendons (three hoop, three vertical, and three dome) are randomly selected for each inspection period. SP-228, Table 1 specifies the tendons that have been selected and the corresponding inspection period for the next four (7 through 10) inspection periods. One common tendon of each type (dome, vertical, and hoop) has been selected from the first year inspection sample and is examined during each succeeding inspection period for the remaining life of the plant. Individual and group average lift-off forces are measured for the selected tendons. In addition, at least one sample tendon of each type is detensioned and a single wire is removed for laboratory examination and testing. The tendon anchorage assemblies and associated hardware (e.g., bearing plates, stressing washers, stressing shims, button heads) of all selected tendons are visually inspected for signs of corrosion, cracked button heads, cracks in anchorage hardware, missing wires, and broken or protruding wires. All accessible grease caps of tendons are inspected for grease leakage and deformation, and the concrete surface within two feet of the edge of the bearing plate of the tendon anchorage area is inspected for cracks.

Monitoring and Trending – Degradation of the Reactor Building post tensioning system is detected by periodic inspections conducted in accordance with the requirements of ES-438, SP-228, and STP-160.001. SP-228 specifies which randomly selected tendons are to be inspected and the corresponding surveillance period for the life of the plant. In addition to the random sampling used for tendon examinations, one tendon of each type (dome, vertical, and hoop) has been selected from the first year inspection sample and designated as a common tendon. Each common tendon is examined during each subsequent inspection to provide monitoring and trending information over the life of the plant.

Tendon prestressing forces are measured by lift-off tests and compared with acceptance standards based on the predicted force for each type of tendon over its life. If the projections for the trend of tendon force versus time indicate that the prestress force for an individual tendon or for the average for any group of tendons could fall below the minimum level required prior to the next regularly scheduled surveillance, a report shall be submitted to the NRC within 30 days of completion of the surveillance.

Acceptance Criteria – The acceptance criteria and guidelines for the inspection of the Reactor Building tendons are specified in ES-438, SP-228 and STP-160.001. Design Engineering Guideline ST-04, "Reactor Building Tendon System Surveillance", provides a detailed practical guidance for engineering to capture and document the surveillance testing experiences and lessons learned.

The acceptance criterion addresses the following test parameters:

Tendon group average lift-off force:

- Equal to or greater than the minimum required average tendon force at the anchorage for that group (i.e., 1160 kips for vertical tendons, 1063 kips for dome tendons, and 1000 kips for hoop tendons).

Individual tendon lift-off force:

- The measured force is not less than 95 percent of the predicted force.

Tendon wire conditions:

- Free of physical damage.
- Laboratory analysis test results show that the ultimate strength and elongation are not less than minimum specified values.

Broken tendon wires:

- Broken or unseated wires and detached buttonheads.

Grease leakage evidence on containment surface:

- No excessive grease leakage as determined by engineering.

Grease cap integrity:

- No excessive grease leakage as determined by engineering.
- Grease cap deformation.

Sheathing filler grease condition:

- Maximum 10 % by weight chemically combined water content.
- Maximum 10 parts per million (ppm) water-soluble chlorides.
- Maximum 10 ppm water soluble nitrates.
- Maximum 10 ppm water soluble sulfides.
- Reserve alkalinity is a minimum 50 % of the installed value or no less than zero when the installed value is 5 or less.
- No grease voids, which are determined when the absolute difference between the amount of grease removed and the amount replaced is less than or equal to 10 % of the tendon net duct volume.

Tendon anchorage areas:

- No evidence of cracking in anchor heads, shims, or bearing plates.
- No evidence of active corrosion.
- Condition of concrete within two feet of the edge of the bearing plate is in acceptable condition.

Corrective Actions – Design Engineering reviews the Surveillance Test Procedure data sheets generated by the vendor. Any abnormal conditions identified during the Reactor Building Tendon Surveillance are documented and evaluated as test deficiencies per SAP-134, "Control of Station Surveillance Activities" [Reference 161]. A Nonconformance Notice (NCN) may be generated depending on the extent or significance of the abnormal condition. NCNs are reviewed by engineering personnel and required corrective actions are specified to ensure that the design adequacy of the containment is maintained.

Any significant degradation that seriously challenges the structural integrity of the Reactor Building is reported to the NRC. Specific corrective actions are documented in accordance with the VCSNS corrective action program and implemented through the applicable work management program. Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160] or the Nonconformance program.

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner.

Administrative Controls – The Reactor Building tendon prestress is monitored and programmatically controlled in accordance with ES-438. Design Engineering Guideline ST-07, "Containment Inservice Inspection Evaluation Criteria", provides the inspection criteria and guidance used to identify, document, review, and evaluate degraded conditions identified during the IWL containment examinations. Reactor Building tendon surveillance testing is conducted in accordance with SP-228 and STP-160.001. Design Engineering Guideline ST-04, "Reactor Building Tendon System Surveillance", provides detailed practical guidance for engineering and other personnel involved in the implementation of the Tendon Surveillance Program.

Any abnormal conditions identified during the Reactor Building Tendon Surveillance are documented and evaluated as test deficiencies per SAP-134, "Control of Station Surveillance Activities" [Reference 161]. A Nonconformance Notice (NCN) may be generated depending on the extent or significance of the abnormal condition. NCNs are reviewed by engineering personnel and required corrective actions are specified to ensure that the design adequacy of the containment is maintained.

In order to make tendon installation and surveillance data history more readily accessible, the database "TENDON" was developed by Gilbert/Commonwealth Incorporated for SCE&G. This database incorporates all the key information from the contractor and engineering reports back to original installation data. This is a "live" document and is updated after each surveillance or any other activities affecting the database records.

Operating Experience – The VCSNS Tendon Surveillance Program was originally established using the requirements of RG 1.35, proposed Revision 3, dated April 1979. In March 1995, the NRC issued a new rule, 10 CFR 50.55a, which invoked the requirements of the ASME Code, Section XI, Subsections IWE and IWL, 1992 Edition and 1992 Addenda. The present program adequately addresses the new requirements.

A brief history of the Tendon Surveillance Program is as follows:

- Reactor Building concrete placement was completed In August 1976.
- Vertical tendon stressing was completed In March 1979.
- The Structural Integrity Test (SIT) was performed in January 1981.
- The first tendon surveillance was performed during March and April 1982.
- The plant operating license was issued in August 1982.
- The second tendon surveillance was performed during October – December 1983.
- The third tendon surveillance was performed during November and December 1985.
- The test results from the first three surveillance's indicated that the wire relaxation force losses in the tendon system were greater than that which were predicted during design. Consequently in June 1988, the predicted wire relaxation force losses were increased from 8.5% to 12.8% [Reference 143].
- The fourth period (10th year) tendon surveillance was performed during January – April 1990. In addition 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.
- The fifth period (15th year) tendon surveillance was performed during March – April 1996.
- The sixth period (20th year) tendon surveillance was performed during September – November 2000.

For a more detailed history of the Tendon Surveillance Program refer to the VCSNS Reactor Building (RB) Design Basis Document (DBD).

A review of the NCN's 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.

NCN 00-0193 was written to address the collection of water in-leakage into the AB 08-02 tendon sump area to a depth that submerged tendon H-AC01. The following actions were taken to resolve this nonconforming condition:

- CER 00-0193 was initiated to reduce the water level in the pit 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.
- In order to ensure that the tendon is not submerged in the future, Operations added the Auxiliary Building tendon sump area to their trend logs (OAP

106.1) and will request facilities to drain the area if the water level in the area approaches the level of the tendon end cover.

The vendor surveillance reports for the past three surveillance periods [4th (1990), 5th (1995) and 6th (2000)] have each concluded that no abnormal degradation of the post tensioning system has occurred at VCSNS.

The VCSNS Tendon Surveillance Program is considered acceptable based on industry standards and operating experience. A review of the Tendon Surveillance Program provides reasonable assurances that the aging effects for the Reactor Building tendons will be managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – The VCSNS Tendon Surveillance Program has been compared to the "Concrete Reactor Building Tendon Prestress" program documented in the GALL Report [Reference 5]. Based on a review of the 10 aging management program elements, the information in the GALL Report regarding "Containment Tendon Prestress" is consistent with the VCSNS program. The VCSNS Tendon Surveillance Program contains those attributes delineated in GALL Chapter X.S2, which have been determined by the NRC to provide an acceptable aging management program.

7.18 Underwater Inspection Program (Service Water Intake Structure and Service Water Pump House)

The loss of material due to the corrosion of steel components and the loss of material and changes in material properties of concrete components in fluid environments have been identified as applicable aging effects for the SWPH and SWIS, which require programmatic management for the extended period of operation. Underwater inspection of the SWIS is conducted every five years to satisfy the requirement specified by VCSNS Operating License NPF-12, Condition 2.C.5.d. The diver's inspection is conducted in accordance with Engineering Service Procedure, ES-400, "Service Water Pond Structure and Dam Inspections", Attachment III. The purpose of the inspection is to monitor the condition of cracks (old and new) in the SWIS that originated due to earlier settlement. To comply with the SCE&G response to Generic Letter (GL) 89-13, underwater inspections of the SWIS and SWPH are conducted, once per refueling cycle, in accordance with Engineering Service Procedure, ES-505, "Service Water System Corrosion Monitoring and Control Program", to monitor and control corrosion and fouling within the service water system. Additionally, semi-annual survey monitoring of the SWPH and SWIS (to satisfy the requirements specified in FSAR Section 2.5.4.10.6.2) is conducted in accordance with Civil Maintenance Procedure, CMP-700.001, "Survey Monitoring". The survey data is reviewed in accordance with Design Engineering Guideline CV-01, "Survey Monitoring Data Review" and ES-400.

The SWIS diver's inspections conducted in accordance with ES-400 are documented in Structural Calculation Series DC0360B and DC03650. Attachment I of ES-505 is used to document the underwater corrosion control and fouling inspections of the SWIS and SWPH and are included as a part of the work documentation package.

The underwater inspections and the survey monitoring of the SWPH and SWIS are evaluated for the aging management program elements identified in NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule". This provides reasonable assurance that the effects of aging are adequately managed for these structures so that their intended functions will be maintained consistent with the Current Licensing Basis (CLB) for the extended period of operation. The underwater inspections of the SWIS and SWPH (conducted in accordance with ES-400 and ES-505) are condition monitoring programs.

Scope of Program – The scope of the SWIS diver's inspection, conducted in accordance with ES-400, includes a visual inspection of the interior length of the intake tunnel, survey monitoring masts MM-1, MM-2 and MM-3, trash racks, access ladder and east end wing walls. The scope of the underwater inspection conducted in accordance with ES-505 includes the interior of the SWIS and the

SWPH forebay area, traveling screen bays, service water pump bays, trash racks and weir. These structures are shown on the E-426-700 and E-726-200 series drawings.

Preventive Actions – No actions are taken, as part of the SWIS and SWPH underwater inspections, which prevent aging effects or mitigate aging degradation. The Underwater Inspection Program (SWIS and SWPH) is a condition monitoring program.

Parameters Monitored or Inspected – Guidelines for the underwater diver's inspection of the SWIS and SWPH are specified in ES-400 and ES-505. The main reason for inspecting the SWIS in accordance with ES-400 is to measure/monitor cracks (old and new) in the concrete structure that originated due to earlier settlement. Additionally, a general inspection of the structure is made to document the as-found condition, noting any unusual observations. The following specific areas are also inspected and their condition documented; (a) access ladder, (b) trash racks, (c) survey monitoring masts, and (d) concrete wing walls at the intake end of the SWIS.

The main purpose for the inspection of the SWIS and SWPH (conducted in accordance with ES-505) is to monitor and control corrosion and fouling within the service water system. The SWIS and the SWPH forebay area, traveling screen bays and service water pump bays are inspected for fouling (clam and silt) accumulations. The density of the accumulation is documented and subsequently removed. Additionally, the submerged trash racks, traveling screen components, service water pump components and other structural components are inspected for corrosion. Any corrosion observed is documented in the inspection report.

Elevation and alignment monitoring of the SWIS and SWPH are conducted in accordance with CMP-700.001 per the survey frequencies specified in ES-400. The survey monitoring data is reviewed in accordance with CV-01 to ensure that settlement remains within established criteria.

Detection of Aging Effects – Attributes associated with aging for the SWIS and SWPH are detected by the underwater inspections conducted and documented in accordance with ES-400 (conducted on a five year frequency), and ES-505 (conducted each refueling cycle). Additionally, semi-annual survey monitoring of the SWIS and SWPH (conducted in accordance with CMP-700.001) will detect any horizontal or vertical movement associated with settlement.

Monitoring and Trending – The underwater diver's inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The SWIS diver's inspection documentation and reviews/evaluations are filed in Structural Calculation Series DC0360B and DC03650. In particular these records include the subcontractor's underwater inspection report, VCSNS

Design Engineering review and evaluation of the results, comparison with previous inspection results, and whether or not the results are acceptable. Discrepancies and their cause and any corrective action resulting from these inspections are also documented in the calculations.

Acceptance Criteria – The acceptance criteria for the underwater diver's inspection of the SWIS (conducted in accordance with ES-400) is that the diver's inspection data is reviewed by Design Engineering. The following guidelines are used by the evaluating engineer to determine the adequacy of the diver's inspection and to determine any remedial actions:

Cracks (old and new) are documented and mapped on ES-400, Attachment III, "SWIS Diver's Inspection Crack Documentation Form". Crack width is measured using wire gauges (ranging from 0.015 to 0.050 inches) on a "Go – No/Go" basis by inserting the wire approximately 1/4 inch into the crack. If the crack is greater than 0.050 inches, its actual width should be measured to the nearest 1/16 inch.

All cracks identified with a maximum width of 0.015 inch or greater are grouted to eliminate/reduce the potential for corrosion of the reinforcing steel. Any changes in length or width to existing (old) cracks and any new cracks are reported by SCE&G to the NRC in accordance with Operating License Condition 2.C.5.d.

The acceptance criteria for the underwater inspection of the SWIS and SWPH (conducted in accordance with ES-505) is that the diver's inspection data is reviewed by Plant Support Engineering. Any accumulation of biofouling (silt or clams) is removed. Engineering evaluates any corrosion on the traveling screen components, service water pump components or any other structural component that was noted on the inspection checklist. Any corrective action(s) based on the results of this evaluation are initiated by engineering.

Corrective Actions – Any problems or concerns observed during the underwater inspections of the SWIS or SWPH that exhibit attributes associated with aging are evaluated by engineering for continued service or repair as required and documented. A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). Tracking and status of corrective action implementation is through the Condition Evaluation Report (CER) Primary Identification Program (PIP) computer application process described in SAP-1131, "Electronic Processing of Condition Evaluation Reports" [Reference 160].

The aging aspects are addressed by increased frequency of inspection or repaired in accordance with a work order [i.e., Maintenance Work Request (MWR) or Preventive Maintenance Task Sheet (PMTS)]. In either case a technical assessment is performed to demonstrate that the structures can maintain their intended function.

Specific corrective actions are completed in accordance with the SCE&G Quality Assurance Program.

Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the CER/NCN processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

Administrative Controls – The SWIS crack inspections are governed by VCSNS Operating License, Condition 2.C.5.d and implemented through ES-400. The inspection of the SWIS and SWPH to monitor and control corrosion and fouling within the service water system is governed by the VCSNS response to GL 89-13 and implemented through ES-505.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure, SAP-143, "Preventive Maintenance Program" [Reference 159].

Operating Experience – VCSNS Operating License Condition 2.C.5.d requires SCE&G to perform an inspection of the SWIS every five years to monitor and measure the cracks in the reinforced concrete tunnel which originated due to settlement problems during construction.

Cracks in the SWIS (tunnel) which were identified during construction with widths greater than 0.012 inches were grouted with a high strength epoxy grout in 1978 prior to filling the Service Water Pond. Diver's inspections were initiated in 1983 and have been performed every five years. The inspections of 1983 and 1988 identified very little change in the existing grouted and ungrouted cracks along with a few new hairline cracks. An improved method of marking old cracks was implemented during 1988, with additional improvements made during the 1993 inspection, which allows better distinction of old versus new ungrouted cracks.

The 1993 inspection also identified nine existing cracks that had widened to greater than 0.050 inch and four cracks with a maximum width greater than 0.015 inch. An evaluation was performed by Gilbert Commonwealth (G/C) and is documented in Structural Calculation DC03650-001. Additionally, NCN 4850 was initiated to document this condition. In accordance with the recommendation of the G/C evaluation and the disposition of NCN 4850, all of these cracks were grouted in 1994 with a flexible urethane grout in order to eliminate/reduce the potential for corrosion of the reinforcing steel.

No new cracks were identified during the 1998 inspection and all cracks that had any visible gap were measured to be less than 0.015 inches wide. The 1998 inspection data for each crack was compared to the results of the 1993 inspection to ensure consistency and no significant differences were noted between the two inspection reports. The next underwater diver's inspection of the SWIS is scheduled for 2003.

Prior to the development of ES-505, visual inspection and dredging (as required) of the SWIS was performed once each refueling cycle within the preventive maintenance program. In response to GL 89-13, ES-505 was developed to direct the SWIS inspections. A review of the inspection data for the past five years shows that no corrosion has been discovered on the trash racks, foot section of each traveling screen, endbell of each service water pump and/or other submerged structural components. The location and density of fouling accumulations (e.g., silt and clams) was recorded and removed by divers using an eductor.

Monitoring and trending of the cracks in the SWIS has been in effect since the initial implementation of the facility operating license at VCSNS and is considered acceptable based on industry operating experience. A review of the SWIS and SWPH underwater diver's inspection and monitoring program conducted at VCSNS confirms the reasonableness and acceptability of the inspection frequencies in that degradation of the SWIS and SWPH will be detected prior to loss of function. The inspection programs provide reasonable assurances that the aging effects for the SWIS and SWPH are managed so that their intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Generic Aging Lessons Learned (GALL) Comparison – Underwater Inspections of the Service Water Intake Structure (SWIS) and Service Water Pumphouse (SWPH) are not evaluated in the GALL Report [Reference 5] and are, therefore, plant specific to VCSNS.

8.0 TIME-LIMITED AGING ANALYSES (TLAA)

The license renewal rule requires that the Time-Limited Aging Analyses (TLAA) as defined in 10 CFR 54.3 must be identified and evaluated. These analyses are typically the boundary conditions and assumptions within the current licensing basis specifically linked to 40 years of operation. The license renewal criteria that are used to evaluate the TLAA are provided in 10 CFR 54.21(c)(1) [Reference 1]. These criteria are as follows:

§54.21(c) (1)

A list of time-limited aging analyses, as defined in §54.3 must be provided. The applicant shall demonstrate that-

- *(i) The analyses remain valid for the period of extended operation;*
- *(ii) The analyses have been projected to the end of the period of extended operation; or*
- *(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.*

The methodology used to identify potential structural TLAA and the results when this methodology is applied to VCSNS are provided in Technical Report TR00140-001, Time-Limited Aging Analyses and Exemptions for License Renewal [Reference 144]. The following potential TLAAs were identified for structural components:

- Containment Liner Fatigue
- Reactor Building Tendons
- Fatigue of Bellows and Penetrations
- Service Water Intake Structure Settlement
- Cranes
- Water Impoundment Structure - Dams
- Boraflex Neutron Absorber

These potential TLAAs are summarized in the following sections. If they were determined to be an actual TLAA, the resolution is also provided.

8.1 Containment Liner Fatigue

The containment liner provides a leak tight membrane on the inside face of the prestressed concrete Reactor Building that can contain airborne radioactive particles and gases due to postulated accidents such as a LOCA. However, the liner is required to remain within certain strain limits associated with serviceability that are set by the ASME B&PV for normal operation. These requirements were reviewed and it was determined that the liner requires re-evaluation for 60 years.

VCSNS calculation DC0306A-010 "Reactor Building Liner" considered Reactor Building liner fatigue for 40 years per ASME Section III, paragraph NE-3131, 1974 with all applicable addenda. This calculation performed comparisons based on 40 years and concluded that the liner (both stainless base and carbon sidewalls) met the criteria of NB 3222.4 (d) for the suitability for cyclic condition and no fatigue analysis was required.

This calculation was revised and concluded that the design criteria for the Reactor Building liner were satisfied for 60 years. Thus VCSNS elects to utilize 10 CFR 54.21 (c) (1) option (ii) to demonstrate that the Reactor Building liner fatigue is adequately analyzed for the period of extended operation.

8.2 Reactor Building Tendons

The VCSNS Reactor Building (RB) was prestressed in order to have low-strain linear response at design loads and thus assure integrity of the liner. The wall is post-tensioned in both vertical and hoop directions. Hoop tendons are anchored on three (3) buttresses, each spaced at 120° apart along the circumference of the containment wall. On the dome, a three-way post-tensioning system is employed [Reference 67].

VCSNS based its tendon surveillance program on proposed Revision 3 of Reg. Guide 1.35 [Reference 145], although it was in a "proposed" status. The Guide remained in this status until July 1990 when the finalized Revision 3 was issued. Nevertheless, on April 28, 1989, the NRC accepted the VCSNS tendon surveillance program based on the proposed Revision 3 of Regulatory Guide 1.35 [Reference 146]. The Reactor Building tendon prestress is monitored and programmatically controlled under ES-438, "Containment Inservice Inspection Program" [Reference 108] and SP-228, "Surveillance of Reactor Building Post Tensioning System," [Reference 147].

VCSNS has performed all required tendon surveillances as previously discussed. The Fourth Period (10th year – 1990) Tendon Surveillance was used to retension the vertical tendons, since the tendon force losses were projected to reach a level where the minimum required force could not be demonstrated by the next surveillance period. This surveillance also indicated the need for potential retensioning of the dome and hoop tendons by the year 2015 [Reference 67]. The Fifth Period (15th year – 1996) and Sixth Period (20th year – 2000) Tendon Surveillances were completed with acceptable results. Engineering Services Design Guideline, ST-04, "Reactor Building Tendons," was also generated to provide a guide for performing future Reactor Building tendon surveillances [Reference 148]. Based on trending data and results from previous surveillances, VCSNS does not expect the tendons to provide adequate prestress for 60 years without re-tensioning various members.

NUREG-1801, Generic Actions Lessons Learned (GALL), Chapter X.S1, "Concrete Containment Tendon Prestress," applies to those facilities that adopt 10 CFR 54.21 (c) (1) option (iii) for containment tendon prestress. This option credits the Containment Tendon Program (or equivalent) with managing the effects of aging. NUREG-1801 Chapter XI.S2, "ASME Section XI, Subsection IWL" presents a generic Reactor Building Tendon Program.

Programmatic controls are used to ensure that the Reactor Building tendons are capable of performing their design function. Therefore the Reactor Building tendons are a TLAA and VCSNS will utilize 10 CFR 54.21 (c) (1) option (iii).

Aging management of Concrete Containment Tendon Prestress is managed by the Tendon Surveillance Program. This program, evaluated in Section 7.0, provides reasonable assurances that the aging effects for the tendons will be managed so that their intended functions will be maintained consistent with the CLB for the period of extended operation.

8.3 Fatigue of Bellows and Penetrations

Penetration Bellows

FSAR [Reference 12] Section 6.2.6.2.1.1, Piping Penetrations and Spares, states that:

> "Hot penetrations are sealed on the inside of containment by a flat plate in a manner similar to cold penetrations. ... Since the containment barrier does not utilize a resilient or flexible seal, these penetrations do not require Type B leakage tests...
>
> For the reasons listed above, no piping penetrations are required to be subjected to Type B leakage tests."

The credited Reactor Building isolation barrier does not utilize any flexible seals between the process pipe and the Reactor Building liner. The bellows are not included in this isolation barrier and have no testing or inspection requirements. The Reactor Building piping penetration bellows were designed to form a barrier without limiting thermal expansion. The absence of a leak tight bellows does not affect the thermal expansion.

The pipe penetration bellows do not meet Criterion 4 and 5 of 10 CFR 54.3 for a Plant Specific TLAA.

The Fuel Transfer Tube bellows are not required for Reactor Building Isolation. The refueling canal is sealed and tested as required for Reactor Building integrity. This seal involves installing a blind flange that seals the Fuel Transfer Tube, which has a double gasketed seal. The collar on the transfer tube that mates with the flange is drilled with passageways that allow pressurization between the gaskets. Therefore, Fuel Transfer Tube bellows are not utilized to provide part of the Reactor Building isolation barrier. The Fuel Transfer Tube is included in the VCSNS 10 CFR 50 Appendix J, Type B Testing program.

The Reactor Building piping penetration bellows and fuel transfer canal bellows do not meet all six criteria for a plant specific TLAA provided in 10 CFR 54.3. Therefore the penetration bellows and associated analyses do not qualify for a TLAA at VCSNS.

Penetration

NUREG-0717 [Reference 149] states: "ferritic materials of the containment pressure boundary which were considered in our (NRC) assessment are those which have been used in the fabrication of the equipment hatch, personnel air lock, penetrations and fluid system components, including the valves required to isolate the system." Taken in the context of the NRC SER (initial issue), it is clear that the NRC has determined that the ferritic materials, used in the construction of the VCSNS containment, meets the appropriate requirements of the ASME code, complies with GDC-51, and behaves in a non-brittle manner. These statements do not present a Time-Limited Aging Analysis. The aging effects on containment isolation materials, steel in air environment, are discussed in Section 6.2 of this report. Therefore no TLAA analysis was identified for Reactor Building isolation materials.

8.4 Service Water Intake Structure Settlement

The Service Water Intake Structure (SWIS) is described in FSAR Section 3.8.4.1.8 [Reference 12]. The SWIS is a reinforced concrete rectangular box culvert with two reinforced concrete wing walls at the intake end. The structure is mostly buried within the West Embankment. The portion which is not covered with soil is submerged within the Service Water Pond. The function of the SWIS is to draw water from the Service Water Pond into the Service Water Pump House.

NRC NUREG-0717 (SER) Supplement 1, Section 3.7.2 "Seismic System and Subsystem Analysis," evaluated the VCSNS Service Water Intake Structure. It was noted that "Excessive non-uniform settlement of the intake structure occurred during construction causing considerable cracking" [Reference 149]. This settlement was analyzed in Service Water Pump House (SWPH) Calculation DC03650-011, "SWPH – Secondary Consolidation" [Reference 152]. This calculation was originally based on a plant design life of 40 years. Since this issue meets all six criteria in 10 CFR 54.3, Service Water Intake Structure settlement is considered an actual TLAA for VCSNS.

To support the VCSNS LRA, Calculation DC03650-011 was revised to account for the period of extended operation (60 years). This revision demonstrates that the expected settlement is acceptable for the period of extended operation. Therefore, VCSNS incorporated Option (ii) to satisfy 10 CFR 54.21(c)(1) for the Service Water Intake Structure settlement. In addition, the Service Water Structures Survey Monitoring Program, as evaluated in Section 7.0, provides reasonable assurances that the aging effects associated with settlement are managed so that the intended function(s) of the Service Water structures will be maintained consistent with the CLB for the period of extended operation.

8.5 Cranes

A potential TLAA issue was considered related to the cranes and associated crane supports that could theoretically impact irradiated fuel during refueling operations. The cranes that meet these criteria are as follows:

Reactor Building Polar Crane (XCR-4-FH)
Spent Fuel Cask Handling Crane (XCR-3-FH)
Fuel Handling Machine (XCR-2-FH - Spent Fuel Pit Bridge and Hoist)
Refueling Machine (XCR-1-FH - Reactor Cavity Manipulator Crane)

These cranes are classified as Class "A" cranes by the Crane Manufacturers Association of America (CMAA) Specification No. 70 (CMAA 70) [Reference 153]. CMAA 70 Class A is defined in paragraph 2.2 as "cranes which may be used in installations such as powerhouses, public utilities, turbine rooms, motor rooms and transformer stations where precise handling of equipment at low speeds with long idle periods between lifts are required. Capacity loads may be handled for initial installation of equipment or infrequent maintenance."

Current versions of CMAA 70 Section 2.8, "Crane Service in Terms of Load Class and Load Cycles" state that Class A Cranes should be designed for 20,000 to 100,000 load cycles. However, the version of CMAA 70 in effect during VCSNS construction lists 20,000 to 200,000 load cycles instead. Since the cranes listed above were built to CMAA 70, the load cycle limits apply. Cranes and supports are considered a TLAA because they satisfy the six criteria for a TLAA defined in 10 CFR 54.3.

The Reactor Building Polar Crane, including the bridge girders, end trucks and trolley, were originally design for construction loads of 360 tons. Seismic analysis was performed for the maximum, non-construction load of 150 tons [Reference 154]. Since construction, the polar crane was only used for capacity lifts during the steam generator replacement project. The steam generator lifts were rated capacity lifts of 354 tons [Reference 155]. Lifts of the lower internals (135 tons), vessel head (125 tons), upper internals (52 tons), reactor coolant pumps, missile shields and other routine refueling operation lifts are commonly done during an outage and do not exceed the seismic load limit of 150 tons. Lifts of 150 tons or less do not qualify as capacity lifts since they are far less than the crane's rated capacity of 360 tons.

The number of lifts was based on one lift for each replaced (old) steam generator and one for each replacement (new) steam generator, which yields a total of six (6) capacity lifts. Imposing an extremely conservative factor of safety of five (5) yields 30 lifts. Assuming a similar number of lifts during initial construction yields an estimate of 60 lifts. In addition, the crane lifted the reactor (330 tons) during construction. This conservative estimate of 61 lifts is exponentially less than the CMAA 70 limit of 200,000 cycles. Therefore, the crane is adequately analyzed and designed for fatigue through the term of extended operation. VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (i) to demonstrate that the polar crane is adequately analyzed for the period of extended operation.

Similar conservative calculations were performed on the Spent Fuel Cask Handling Crane, the Fuel Handing Machine, and the Refueling Machine. The estimate of load cycles is far less than the design limit in CMAA 70. Therefore these cranes are adequately analyzed for the period of extended operation. In addition, the Material Handling System Inspection Program ensures that the cranes are capable of performing their design functions. Refer to TR00140-001 [Reference 144] for more details.

In addition, the Material Handling System Inspection Program, as evaluated in Section 7.0, provides reasonable assurances that the aging effects associated with loss of material are managed so that the intended function(s) of the cranes will be maintained consistent with the CLB for the period of extended operation.

8.6 Water Impoundment Structures - Dams

FSAR Section 2.5.6.8.1, "Settlement and Alignment Monitoring Instrumentation" states that "settlement and alignment of the dams will be monitored throughout the operating life of the plant" [Reference 12]. However no Time-Limited Aging Analyses or calculations were identified that involve time-limited assumptions defined by the current operating term, for example, 40 years. Therefore, settlement issues that are related to water impoundment structures do not qualify as plant specific TLAAs.

VCSNS Engineering Services Procedure ES-400, "Service Water Pond Structure and Dam Inspections" control water impoundment structure inspections. These structures are not time-limited but require aging management to maintain the structures capable of performing their design functions. Aging management of water impoundment structures is managed by the Service Water Pond Dam Inspection Program (North Dam, South Dam, East Dam and West Embankment). This program, evaluated in Section 7.0, provides reasonable assurances that the aging effects for the earthen structures will be managed so that their intended functions will be maintained consistent with the CLB for the period of extended operation.

8.7 Boraflex Neutron Absorber

A review of the FSAR revealed possible TLAAs concerning aging of the neutron absorber spent fuel rack. VCSNS previously utilized Boraflex panels to absorb neutrons in the spent fuel pool. VCSNS will replace the Boraflex panels with Boral during Cycle 13. VCSNS Technical Specification Amendment Request, TSP 99-0090, was submitted to the NRC in document RC-01-0135 dated July 24, 2001 [Reference 158]. This request was made in order to increase the spent fuel pool storage capacity by replacing the existing eleven high-density storage rack modules with twelve high-density storage racks. Holtec International's Report [Reference 102] documents the design adequacy of twelve high-density storage racks, and replaces Boraflex with Boral to improve neutron absorbing material longevity. The Holtec International Report states that Boral does not degrade because of long-term exposure to radiation and that Boral is stable, durable, and corrosion resistant. The "NRC Staff Action Plan for Spent Fuel Pool Safety Issues" concludes that degradation of neutron absorption performance has not been observed in materials other than Boraflex [Reference 103].

Since the existing Boraflex panels will be replaced and no longer utilized, they will no longer be within the scope of license renewal. The new Boral panels will perform their design function for the period of extended operation.

VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to demonstrate that the new spent fuel pool neutron absorbers (Boral) will perform their design function for the period of extended operation.

9.0 CONCLUSIONS

The aging management review for VCSNS structures and structural components has been performed in accordance with §54.21(a). The aging management review was performed for the following structures:

- Auxiliary Building [includes RWST & RMWST foundations and West Penetration Access Area (WPAA)]
- Control Building
- Intermediate Building [includes East Penetration Access Area (EPAA)]
- Diesel Generator Building
- Fuel Handling Building
- Reactor Building and Internal Structures
- Turbine Building
- Service Water Pumphouse, Intake and Discharge Structures
- Yard Structures (includes Fire Service Pumphouse, CST foundation, Electrical Manhole EMH-2, Electrical Substation foundation for OCB 8892, Transformer Area foundations, and Transmission Towers and foundations from the Emergency Auxiliary Transformer to OCB 8892)
- Earthen Embankments (Includes Service Water Pond North Dam, South Dam, East Dam, West Embankment and North Berm)

For these structures and the components included within the structures that are within the scope of the LR Rule, the license renewal intended functions will be maintained consistent with the CLB by the programs, as described in Section 7.0 of this report, for the period of extended operation. The programs that have been credited for managing the aging effects of the VCSNS structures and structural components are:

Section	Title
7.1	10 CFR 50 Appendix J General Visual Inspection
7.2	10 CFR 50 Appendix J Leak Rate Testing
7.3	ASME Section XI ISI Program – IWF
7.4	Battery Rack Inspection
7.5	Boraflex Monitoring Program
7.6	Boric Acid Corrosion Surveillances
7.7	Chemistry Program
7.8	Containment Coating Monitoring and Maintenance Program
7.9	Containment ISI Program – IWE/IWL
7.10	Fire Protection Program
7.11	Flood Barrier Inspection
7.12	Maintenance Rule Structures Program
7.13	Material Handling System Inspection Program
7.14	Pressure Door Inspection Program
7.15	Service Water Pond Dam Inspection Program (North, South & East Dams and West Embankment)

Section	Title
7.16	Service Water Structures Survey Monitoring Program (SWIS, SWPH, Electrical Duct Banks & SW Intake Line A)
7.17	Tendon Surveillance Program
7.18	Underwater Inspection Program (SWIS and SWPH)

Aging Management Review summary tables are provided in Attachment II of this report for each VCSNS structure within license renewal scope. The summary tables identify the components within the structures, the aging effects requiring management for those components, and the programs which are credited with managing the aging effects. VCSNS plant environment parameters and aging effect evaluations are presented in Section 6.0 of this report. The credited programs evaluated include the key elements of an effective program as identified in NEI 95-10 and comparisons were made against the ten program attributes from the Standard Review Plan and the GALL report. The programs were evaluated in Section 7.0 of this report to demonstrate that they will provide reasonable assurance that the aging effects for VCSNS structures and structural components will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation.

ATTACHMENT I: Aging Effects Evaluation Summary

ORGANIZATION OF ATTACHMENT I

TABLE A-1

SUMMARY OF AGING EFFECTS FOR STEEL STRUCTURAL COMPONENTS

Environment	Component Material	Loss of Material	Cracking	Change in Material Properties
Borated Water	Stainless Steel	Crevice Corrosion, Pitting Corrosion	Stress Corrosion Cracking (SCC)	None Identified
Raw Water	Carbon and Low Alloy Steel	General Corrosion, Pitting Corrosion, MIC	None Identified	None Identified
Reactor Building	Carbon and Low Alloy Steel	Boric Acid Corrosion, General Corrosion	None Identified	None Identified
Reactor Building	Stainless Steel	None Identified	Stress Corrosion Cracking (SCC) is a potential aging effect for high strength bolting	None Identified
Internal	Carbon and Low Alloy Steel	Boric Acid Corrosion, General Corrosion, MIC (Below groundwater table elevation)	None Identified	None Identified
Internal	Stainless Steel	None Identified	None Identified	None Identified
External	Carbon and Low Alloy Steel	General Corrosion	None Identified	None Identified
External	Stainless Steel	None Identified	None Identified	None Identified

TABLE A-2
SUMMARY OF AGING EFFECTS FOR CONCRETE COMPONENTS

Component	Loss of Material					Cracking						Material Property Change			
	Freeze-Thaw	Abrasion & Cavitation	Elevated Temperature	Aggressive Chemicals	Corrosion of Embedded Steel / Rebar	Freeze-Thaw	Reaction with Aggregates	Shrinkage	Settlement	Elevated Temperature	Fatigue	Leaching	Elevated Temperature	Aggressive Chemicals	Irradiation Embrittlement
Anchorage	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Embedments	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Equipment Pads	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Fire Walls	NA	NA	NA	NA	NA	NA	NA	✔	NA	NA	NA	NA	NA	NA	NA
Flood Curbs	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Foundation	NA	NA	NA	NA	NA	NA	NA	NA	✔	NA	NA	NA	NA	NA	NA
Foundation Dowels	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Hatches	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Masonry Block Walls	NA	NA	NA	NA	NA	NA	NA	✔	NA	NA	NA	NA	NA	NA	NA
Missile Shields	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Pipe Piles	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA

TABLE A-2
SUMMARY OF AGING EFFECTS FOR CONCRETE COMPONENTS

Component	Loss of Material					Cracking						Material Property Change			
	Freeze-Thaw	Abrasion & Cavitation	Elevated Temperature	Aggressive Chemicals	Corrosion of Embedded Steel / Rebar	Freeze-Thaw	Reaction with Aggregates	Shrinkage	Settlement	Elevated Temperature	Fatigue	Leaching	Elevated Temperature	Aggressive Chemicals	Irradiation Embrittlement
Reinforced Concrete beams, columns, floor slabs, walls	✔[1]	✔[2]	NA	NA	NA	✔[1]	NA	NA	NA	NA	NA	✔[3]	NA	NA	NA
Roof slabs	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	✔[3]	NA	NA	NA
Sumps	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
Trenches	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA

Key:
NA – Aging effect due to this mechanism does not require management for this set of components.
✔ - Aging effect due to this mechanism requires management for this set of components.

Notes:
1. Loss of material and cracking due to freeze-thaw are aging effects requiring management for the structures and components located in lake or pond water.
2. Loss of material due to abrasion is an aging effect requiring management for the structures and components located in lakes and ponds and exposed to continuously flowing water.
3. Change in material property due to leaching is an aging effect requiring management for the walls and roofs exposed to the external environment and concrete components below site groundwater elevation.
4. Cracking due to settlement has been experienced at the SWIS and SWPH and is managed by the Service Water Structures Survey Monitoring Program (SWPH, SWIS, Electrical Duct Banks and SW Intake Line "A"). Refer to Operating License Condition 2.C.5.

ATTACHMENT II: Aging Management Review Results for Structures and Structural Components

ORGANIZATION OF ATTACHMENT II

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
			External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B5.1-a	No	Y	
Bellows (RHR and Spray system isolation valve chambers and guard pipes)	3,5	Steel (Stainless)	Internal	Loss of Material Fatigue / Cracking	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing See TLAA Section 8.0	II.A3.1-a II.A3.1-b II.A3.1-c II.A3.1-d	Yes	Partial	Specialty penetrations. See TLAA Section 8.3 under Bellows.
Blowout or Blow-off Panels	5	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Compressible Joints & Seals	4,5	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints and flood seals.
Crane Rails & Girders	3,10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b III.B2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.3.2	No	Y	
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.3.3	No	Y	See Section 6.9 on wear of fire door appurtenances.
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	2,4,5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Hatches (Concrete)	1,2,4,5,8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b III.B3.1-b	No	Y	
Jet Barriers	9	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Lead Shielding Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Liner Plate (RHR and Spray system isolation valve chambers and guard pipes)	2,3,5,7	Steel	Internal	Loss of Material	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing Boric Acid Corrosion Surveillances	II.A1.2-a II.A3.1-a	No	Y	Specialty penetration, extension of containment boundary.
				MIC	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	See Section 6.2 for MIC on guard pipes.
Masonry Block, Brick Walls, or Knockdown Walls	1,2,3,7,10	Masonry Block	Internal	Cracking	Maintenance Rule Structures Program	III.A3.3-a	No	Partial	Maintenance Rule Structures Program instead of Masonry Wall Program. No Safety Related block walls at VCSNS.
Metal Spray Shields	6	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Missile Shields	8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Pipe Supports	3,10	Steel	Internal	Loss of Material	Boric Acid Corrosion Surveillances	III.B1.2.1-b III.B2.1-b	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
					Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Pipe Whip Restraint	3,9,10,11	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,5,7,8, 9,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal (below groundwater level)	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Roof Slabs	1,2,3,5,7,8, 10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
AUXILIARY BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Seismic Joint Filler	3,10	Elastomer	Internal	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Structural Steel – beams, columns, plates, trusses	2,3,8,9,10,11	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	
Sumps	3,4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Tube Track	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete Internal	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	N/A No	N/A Y	See Section 6.4.
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Checkered Plate	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Compressible Joints & Seals	4,5	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints and flood seals.
Control Boards	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Control Room Ceiling	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Crane Rails & Girders	10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	No specific GALL item, used VII.G.3-a.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.3.2	No	Y	No specific GALL item, used VII.G.3.2

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.3.3	No	Y	See Section 6.9 on wear of fire door appurtenances. No specific GALL item, used VII.G.3.3.
Flood Barriers	4	Elastomer	Internal / Below Grade	Cracking / Shrinkage	Maintenance Rule Structures Program	N/A	N/A	N/A	
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	2,4,5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Hatches (Concrete)	1,2,4	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
Lead Shielding Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Masonry Block, Brick Walls, or Knockdown Walls	1,2,3,7,10	Masonry Block	Internal	Cracking	Maintenance Rule Structures Program	III.A1.3-a	No	Partial	Maintenance Rule Structures Program instead of Masonry Wall Program. No Safety Related block walls at VCSNS.
Metal Partition Walls	1	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Missile Shields	8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Pipe Supports	3,10	Steel	Internal	Loss of Material	ASME Section XI ISI Program – IWF Maintenance Rule Structures Program	III.B1.2.1-a III.B2.1-a	No No	Y Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,7,8, 10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal (below groundwater level)	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Roof Slabs	2,3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Seismic Joint Filler	3,10	Elastomer	Internal	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Structural Steel – beams, columns, plates, trusses	2,3,8,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A1.2-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
CONTROL BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Tube Track	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
			External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B5.1-a	No	Y	
Battery Racks	3	Steel	Internal	Loss of Material	Battery Rack Inspection Maintenance Rule Structures Program	III.B3.1-a	No	Y	Rack Insp. Plus MR
Blowout or Blow-off Panels	3	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Caissons	3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Compressible Joints & Seals	4,5,7	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints and flood seals.
Crane Rails & Girders	10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.9.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.2.1-b III.B2.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	No specific GALL item, used VII.G.3-a.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.3.2	No	Y	No specific GALL item, used VII.G.3.2

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.3.3	No	Y	See Section 6.9 on wear of fire door appurtenances. No specific GALL item, used VII.G.3.3.
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	2,4,5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Hatches (Concrete)	1,2,4,5,8	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Hatches (Steel)	1,2,4,5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B3.1-a III.B2.1-b III.B3.1-b	No No	Y Y	
Jet Barriers	9	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Lead Shielding Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Metal Siding	2	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Metal Spray Shields	6	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Missile Shields	8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Piers	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Pipe Supports	3,10	Steel	Internal	Loss of Material	Boric Acid Corrosion Surveillances	III.B1.2.1-b III.B2.1-b	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
					Maintenance Rule Structures Program	III.B2.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Pipe Whip Restraint	3,9,10,11	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,5,7,8, 10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal (below groundwater level)	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Roof Slabs	1,2,3,5,7,8, 10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Seismic Joint Filler	3,10	Elastomer	Internal	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
INTERMEDIATE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Structural Steel – beams, columns, plates, trusses	2,3,8,9,10,11	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	No specific GALL item, used III.A3.2-a.
Sumps	3,4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Trenches	2,4	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Tube Track	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
DIESEL GENERATOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete Internal	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	N/A No	N/A Y	See Section 6.4.
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Caissons	3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Compressible Joints & Seals	4	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints and flood seals.
Crane Rails & Girders	10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
DIESEL GENERATOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.4.1	No	Y	
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.4.2	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
DIESEL GENERATOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.4.3	No	Y	See Section 6.9 on wear of fire door appurtenances.
Flood Barriers	4	Elastomer	Internal / Below Grade	Cracking / Shrinkage	Maintenance Rule Structures Program	N/A	N/A	N/A	
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	2,4	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Grating	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Hatches (Steel)	1,2,8	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
Metal Partition Walls	1	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
DIESEL GENERATOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Missile Shields	8	Concrete	External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Pipe Supports	3,10	Steel	Internal	Loss of Material	ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
					Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,8,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal (below groundwater level)	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Roof Slabs	1,2,8,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
DIESEL GENERATOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Seismic Joint Filler	3,10	Elastomer / Styrofoam	Internal / External	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Structural Steel – beams, columns, plates, trusses	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	
Sumps	3,4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel / (Stainless)	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel / (Stainless)	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.1.1-b	No	Y	
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	
Caissons	3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Checkered Plate	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1b	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Compressible Joints & Seals	4,5,7	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints and flood seals.
Crane Rails & Girders	3	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	See Section 8.5 for crane TLAA.
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel / (Stainless)	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a III.B1.2.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances ASME Section XI ISI Program – IWF	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a III.B1.2.1-b III.B2.1-b III.B1.2.1-a	No No No	Y Y Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	No specific GALL item, used VII.G.3-a.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.3.2	No	Y	No specific GALL item, used VII.G.3.2
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.3.3	No	Y	See Section 6.9 on wear of fire door appurtenances. No specific GALL item, used VII.G.3.3.

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Fuel Transfer Canal Liner Plate	5	Steel (Stainless)	Internal / Borated Water	Loss of Material / Cracking (SCC)	Chemistry Control Program Technical Specification ¾.9.10	III.A5.2-b	No	Y	The Fuel Transfer Canal by the spent fuel pool is normally flooded during plant operation but can be drained for work on equipment.
Hatches (Concrete)	1,2,4,5,8	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Hatches (Steel)	1,2,4,5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B3.1-a III.B2.1-b III.B3.1-b	No No	Y Y	
Lead Shielding Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Masonry Block, Brick Walls, or Knockdown Walls	1,2,3,7,10	Masonry Block	Internal	Cracking	Maintenance Rule Structures Program	III.A5.3-a	No	Partial	Maintenance Rule Structures Program instead of Masonry Wall Program. No Safety Related block walls at VCSNS.
Metal Siding	2	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Missile Shields	8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Neutron absorbing sheets – Boraflex	3	Boraflex Sheets	Borated Water	Boraflex Degradation	Boraflex Monitoring Program	VII.A2.1-a	No	Y	Boraflex to be replaced with Boral per Tech Spec Amendment request TSP 99-0090. See Section 8.7.
Piers	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Pipe Supports	3,10	Steel	Internal	Loss of Material	Boric Acid Corrosion Surveillances	III.B1.2.1-b III.B2.1-b	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
					ASME Section XI ISI Program – IWF	III.B1.2.1-a	No	Y	
					Maintenance Rule Structures Program	III.B2.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
FUEL HANDLING BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,5,7,8, 10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal (below groundwater level)	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Roof	2,3,10	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.A5.2-a	No	Y	
Seismic Joint Filler	3,10	Elastomer	Internal	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	
Spent Fuel Pool Liner	5	Steel (Stainless)	Internal / Borated Water	Loss of Material / Cracking (SCC)	Chemistry Control Program Technical Specification 3 /4.9.10	III.A5.2-b	No	Y	
Spent Fuel Storage Rack	3	Steel (Stainless)	Borated Water	Loss of Material / Cracking (SCC)	Chemistry Control Program Technical Specification 3 /4.9.10	VII.A2.1-c	No	Y	

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Structural Steel – beams, columns, plates, trusses	2,3,8,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A5.2-a	No	Y	
Sumps	3,4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Tube Track	2,3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.1.1-b III.B1.2.1-b	No	Y	
				Cracking – SCC (high Strength bolts)	ASME Section XI ISI Program – IWF	III.B1.1.1-a III.B1.2.1-a	No	Y	See Section 6.8 on Class 1 component supports.
Bellows (Penetration)	3	Steel (Stainless)	Reactor Building	Loss of Material and Fatigue / Cracking	See Sections 6.2, 6.9, and TLAA Section 8.3 under Bellows.	II.A3.1-a II.A3.1-b II.A3.1-c II.A3.1-d	Yes	N	Loss of Material and Fatigue / Cracking aging effects screened out See Sections 6.2, 6.9, and 8.3 under Bellows.
Cable Tray & Conduit	2,3,10	Steel	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Checkered Plate	10	Steel (Stainless)	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Compressible Joints & Seals	4	Elastomer	Reactor Building	Cracking Change in Material Properties	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing Maintenance Rule Structures Program	II.A3.3-a	No	Y	Reactor Building moisture barrier. See also "Personnel airlock, escape airlock, and equipment hatch" Component Type for seals associated with these components.
Control Boards (Refuel Cavity Crane)	2	Steel	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.2.
Crane Rails & Girders	3	Steel	Reactor Building	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	See Section 8.5 for crane TLAA.
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Component Supports	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.1.1-b III.B.1.2.1-b	No	Y	
				Cracking – SCC (high Strength bolts)	ASME Section XI ISI Program – IWF	III.B1.1.1-a III.B1.2.1-a	No	Y	See also Table 6.8-1 for more details on specific Class 1 component supports.
Equipment Pads	3,10	Concrete	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B1.1.1-b	No	Y	
					ASME Section XI ISI Program – IWF	III.B1.1.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Fire Barrier Penetration Seals	1	Elastomer	Reactor Building	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	For fire stops between floors. No specific GALL item, used VII.G.3-a.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete	Reactor Building	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.5.2	No	Y	
Fire Doors	1	Steel	Reactor Building / External	Loss of Material	Fire Protection Program	VII.G.5.3	No	Y	See Section 6.9 on wear of fire door appurtenances.
Flood Curbs (Concrete)	4,6	Concrete	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood Curbs (Steel)	4,6	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.A4.2-a	No	Y	
Foundations	3,7,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Hatches (Steel)	1,2,4,5,7,8,9	Steel	Reactor Building / External	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
HVAC Duct Supports	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Instrument Line Supports	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B2.1-b	No No	Y Y	
Instrument Racks & Frames	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B2.1-a III.B3.1-a III.B2.1-b III.B3.1-b	No No	Y Y	
Jet Barriers	3,6,9,10	Concrete	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Jet Barriers	3,6,9,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Lead Shielding Supports	3,10	Steel / Lead Brick	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Liner Plate	3,5,7,10	Steel	Reactor Building	Loss of Material	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing Containment Coating Monitoring and Maintenance Program	II.A1.2-a II.A3.1-a	No, if corrosion is not significant for inaccessible areas.	Y	
				Fatigue	Time-Limited Aging Analysis	II.A3.1-b	Yes, if CLB fatique analysis exist.	Y	See Section 8.1, Containment Liner Fatigue TLAA.
Metal Partition Walls	2	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B5.1-b	No	Y	
Metal Siding	2	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Missile Shields	2,3,6,8,9,10	Concrete	Reactor Building / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N; N/A, Partial)	Notes
Penetrations (Mechanical and Electrical)	2,5	Steel	Reactor Building	Loss of Material Fatigue / Cracking	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing See TLAA Section 8.0	II.A3.1	No	Y	Mechanical penetration includes fuel transfer tube.
Personnel airlock, escape airlock, and equipment hatch	1,5,7	Steel	Reactor Building	Loss of Material	Containment ISI Program – IWE/IWL 10 CFR 50 Appendix J Leak Rate Testing Technical Specifications 3/4.6.1	II.A3.2-a II.A3.2-b	No	Y	
Pipe Supports	3,10	Steel	Reactor Building	Loss of Material	Boric Acid Corrosion Surveillances ASME Section XI ISI Program – IWF Maintenance Rule Structures Program	III.B1.1.1-b III.B1.2.1-b III.B1.1.1-a III.B1.2.1-a III.B2.1-a	No No No	Y Y Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
Pipe Whip Restraint	3,9,10,11	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Post-Tensioning System	3	Steel	Concrete / Reactor Building / External	Loss of Material	Tendon Surveillance Program	II.A1.3-a	No	Partial	See Section 6.9.
				Loss of Prestress		II.A1.3-b	Y, see TLAA Section 8.2	Partial	See Section 8.2 for Tendon TLAA.
Refueling Canal Liner Plate	5	Steel (Stainless)	Reactor Building / Borated Water	None	None Required	N/A	N/A	N/A	See Section 6.2. Although no aging effects have been identified for stainless steel in air environment, the Maintenance Rule Structures Program inspects it. Borated water environment occurs during refueling and is infrequent.
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,5,6,7, 8,9,10,11	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Reactor Building (below groundwater level)	Change in Material Properties due to leaching	Containment ISI Program – IWE/IWL	Plant specific	N/A	N/A	Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
			External	Change in Material Properties due to leaching	Containment ISI Program – IWE/IWL	II.A.1.1-b	No	Y	Leaching has been experienced for external concrete in Tendon Access Gallery. See Section 7.9.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Seismic Joint Filler	1,3,4	Elastomer	Internal	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Seismic joint between RB and IB, FB, and EPAA.
Stair, Platform, & Grating Support	3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program Boric Acid Corrosion Surveillances	III.B5.1-a III.B5.1-b	No No	Y Y	
Structural Steel – beams, columns, plates, trusses	2,3,8,9,10,11	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.A4.2-a	No	Y	
Sump Screens	3,4	Steel (Stainless)	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.2. Although no aging effects have been identified for stainless steel in air environment, the Maintenance Rule Structures Program inspects it.
Sumps	3,4,5,7	Steel lined (Stainless)	Reactor Building	None	None Required	N/A	N/A	N/A	See Section 6.2. Although no aging effects have been identified for stainless steel in air environment, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Tube Track	2,3,10	Steel	Reactor Building	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
					Boric Acid Corrosion Surveillances	III.B2.1-b	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
TURBINE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	10	Steel	Concrete Internal	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	N/A No	N/A Y	See Section 6.4.
Cable Tray & Conduit	10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Compressible Joints & Seals	4	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints, penetration and flood seals.
Crane Rails & Girders	10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Electrical and Instrument Panels & Enclosures	10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
TURBINE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Component Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads	10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.2-a	No	Y	
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.2.2	No	Y	
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.2.3	No	Y	See Section 6.9 on wear of fire door appurtenances.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
TURBINE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	5	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Grating	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Hatches (Concrete)	10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Hatches (Steel)	10	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
HVAC Duct Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Line Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Racks & Frames	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
TURBINE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Masonry Block, Brick Walls, or Knockdown Walls	1,10	Masonry Block	Internal / External	Cracking	Maintenance Rule Structures Program	III.A3.3-a	No	Partial	Maintenance Rule Structures Program instead of Masonry Wall Program. No Safety Related block walls at VCSNS.
Metal Siding	10	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Pipe Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,4,8,10	Concrete	Below Grade Internal (below groundwater level)	None Change in Material Properties due to leaching	None Required Maintenance Rule Structures Program	N/A Plant specific	N/A N/A	N/A N/A	See Section 6.4. Leaching has been experienced at VCSNS for internal concrete below groundwater elevation.
Roof	10	Steel	Internal / External	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	
Seismic Joint Filler	10	Elastomer / Styrofoam	Internal / External	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	
Stair, Platform, & Grating Support	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
TURBINE BUILDING

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Structural Steel – beams, columns, plates, trusses	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	
Sumps	4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Trenches	4	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete Internal	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	N/A No	N/A Y	See Section 6.4.
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Checkered Plate	2	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Compressible Joints & Seals	4	Elastomer	Internal / Below Grade	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	Inspected items include seismic joints, penetration and flood seals.
Crane Rails & Girders	10	Steel	Internal	Loss of Material	Material Handling System Inspection Program Maintenance Rule Structures Program	VII.B.1.b VII.B.1.1	No No	Y Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Electrical and Instrument Panels & Enclosures	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Equipment Component Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads	3,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.1-a	No	Y	
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete / Drywall	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.1.2	No	Y	
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.1.3	No	Y	See Section 6.9 on wear of fire door appurtenances.

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Flood, Pressure and Specialty Doors	2,4	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	N/A	N/A	N/A	Doors rated for fire protection are also inspected via the Fire Protection Program.
Foundations	3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	
				Settlement	Service Water Structures Survey Monitoring Program for SWPH/SWIS/Electrical Duct Banks/SW Intake Line "A"	III.A6.4-a	No	N	Service Water Structures Survey Monitoring Program is a plant specific AMP for detecting identified aging effects.
Grating	2	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Hatches (Concrete)	1,2,8	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
HVAC Duct Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Instrument Line Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Racks & Frames	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
Intake Bays or Canals	3,13	Concrete	Raw Water	Loss of Material, Cracking	Underwater Inspection Program (SWIS and SWPH)	III.A6.1-a III.A6.1-h	No	N	Underwater Inspection Program (SWIS and SWPH) is a plant specific AMP for detecting identified aging effects.
				Settlement	Service Water Structures Survey Monitoring Program for SWPH/SWIS/Electrical Duct Banks/SW Intake Line "A"	III.A6.4-a	No	N	Service Water Structures Survey Monitoring Program is a plant specific AMP for detecting identified aging effects. See Section 8.4 SWIS Settlement TLAA.
Intake Screens	3,13	Steel	Raw Water	Loss of Material	Underwater Inspection Program (SWIS and SWPH)	III.A6.2-a	No	N	Underwater Inspection Program (SWIS and SWPH) is a plant specific AMP for detecting identified aging effects.
Missile Shields	8	Concrete	External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Pipe Supports	3,10	Steel	Internal	Loss of Material	ASME Section XI ISI Program – IWF Maintenance Rule Structures Program	III.B1.2.1-a III.B2.1-a	No No	Y Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,10,13	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	None	None Required	N/A	N/A	N/A	
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete at building structures.
			Raw Water	Loss of Material and Cracking due to Freeze-Thaw	Underwater Inspection of Service Water Intake Structure and Service Water Pump House	III.A.6.1-a	No	Y	
Roof Slabs	1,2,8,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4.
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete.
Seismic Joint Filler	3,10	Elastomer / Styrofoam	Internal / External	Cracking Change in Material Properties	Maintenance Rule Structures Program	Plant Specific	N/A	N/A	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Stair, Platform, & Grating Support	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Structural Steel – beams, columns, plates, trusses	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A6.2-a	No	Y	
Sumps	3,4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (CONDENSATE STORAGE TANK FOUNDATION)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3,10	Steel	Concrete External	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a	N/A No	N/A Y	See Section 6.4.
Checkered Plate	3,10	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
Expansion Anchors	3	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Foundations	3	Concrete	Below Grade External	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.A8.1-a	N/A No	N/A Y	See Section 6.4.
Instrument Line Supports	3	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Racks & Frames	3	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
Pipe Supports	3	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (CONDENSATE STORAGE TANK FOUNDATION)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	3	Concrete	External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete at building structures.
Stair, Platform, & Grating Support	3,10	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
			External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Battery Racks	10	Steel	Internal	Loss of Material	Battery Rack Inspection Maintenance Rule Structures Program	III.B3.1-a	No	Y	Rack Insp. Plus MR
Cable Tray & Conduit	10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Electrical and Instrument Panels & Enclosures	10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Equipment Component Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads	10	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Expansion Anchors	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Fire Barrier Penetration Seals	1	Elastomer	Internal	Cracking / Shrinkage	Fire Protection Inspections	VII.G.3-a	No	Y	No specific GALL item, used VII.G.3-a.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete	Internal	Cracking	Fire Protection Inspections Maintenance Rule Structures Program	VII.G.3.2	No	Y	No specific GALL item for fire pump house, used VII.G.3.2.
Fire Doors	1	Steel	Internal	Loss of Material	Fire Protection Program	VII.G.3.3	No	Y	See Section 6.9 on wear of fire door appurtenances. No specific GALL item, used VII.G.3.3.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Flood Curbs (Concrete)	4,6	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Foundations	10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Hatches (Steel)	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	
HVAC Duct Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Line Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Instrument Racks & Frames	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a	No	Y	
Masonry Block, Brick Walls, or Knockdown Walls	1,10	Masonry Block / Brick	Internal / External	Cracking	Maintenance Rule Structures Program	III.A3.3-a	No	Partial	Maintenance Rule Structures Program instead of Masonry Wall Program. No Safety Related block walls at VCSNS.
Pipe Supports	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	See Section 6.9 for details on Service-induced cracking and Loss of Mechanical Function aging effect.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,4,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	None	None Required	N/A	N/A	N/A	
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete at building structures.
Structural Steel – beams, columns, plates, trusses	10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	No specific GALL item, used III.A3.2-a.
Sumps	4,10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Trenches	10	Concrete	Internal	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

Note No.	Description
FSPH-General	Intended function number 3, number 10, "provides structural support to Non-Safety Related components whose failure could prevent satisfactory accomplishment of any of the required Safety Related functions" was used when identifying structures and components supporting equipment which are relied upon to demonstrates compliance with regulated events. For example, equipment pads are in-scope since they provide structural support to the fire pumps. While the equipment within the Fire Service Pumphouse is Non Safety Related, the fire pumps are required to be operable in order to demonstrate compliance with the Fire Protection (FP) 10 CFR 50.48 regulated event.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (ELECTRICAL MANHOLE, EMH-2)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	3	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	3	Steel	Concrete Internal (below grade)	None Loss of Material	None Required Maintenance Rule Structures Program	N/A III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	N/A No	N/A Y	See Section 6.4.
Cable Tray & Conduit	2,3,10	Steel	Internal	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	3,10	Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Duct Banks	1,2,3,10	Concrete	Internal / Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Embedments	3,10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Fire Barriers (Walls, Ceilings and Floors)	1	Concrete	Internal	Cracking	Maintenance Rule Structures Program	VII.G.3.2	No	Partial	Missile barrier hatch and manhole internal concrete which function also as fire barriers are managed by Maintenance Rule Structures Program.
Foundations	3	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
Manhole Covers	3	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B5.1-a	No	Y	

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (ELECTRICAL MANHOLE, EMH-2)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Manholes	1,2,3,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Missile Shields	2,3,8	Concrete	Internal / External	None	None Required	N/A	N/A	N/A	See Section 6.4. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	1,2,3,4,10	Concrete	Below Grade	None	None Required	N/A	N/A	N/A	See Section 6.4.
			Internal	None	None Required	N/A	N/A	N/A	
			External	Change in Material Properties due to leaching	Maintenance Rule Structures Program	Plant specific	N/A	N/A	Minor Leaching has been experienced for external concrete at building structures.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (ELECTRICAL SUBSTATION AND TRANSFORMER AREA)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Anchorage	10	Galv. Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Anchorage / Embedments (exposed surfaces)	10	Galv. Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	See Section 6.4.
Cable Tray & Conduit	10	Steel	External	None	None Required	N/A	N/A	N/A	See Section 6.2.
Cable Tray & Conduit Supports	10	Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a	No	Y	
Electrical and Instrument Panels & Enclosures	10	Steel	External	None	None Required	N/A	N/A	N/A	See Section 6.2.
Embedments	10	Steel	Concrete	None	None Required	N/A	N/A	N/A	See Section 6.4.
Equipment Component Supports	10	Galv. Steel	External	Loss of Material	Maintenance Rule Structures Program	III.B2.1-a III.B3.1-a III.B4.1-a III.B5.1-a	No	Y	
Equipment Pads (Buslines, PCBs, transformers)	10	Concrete	External	None	None Required	N/A	N/A	N/A	See Section 6.4 and 6.9. Although no aging effects have been identified for this component type, the Maintenance Rule Structures Program inspects it.

SCREENING RESULTS FOR VIRGIL C. SUMMER NUCLEAR STATION
YARD STRUCTURES (ELECTRICAL SUBSTATION AND TRANSFORMER AREA)

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
Reinforced Concrete (transmission tower foundation, transformer fire wall)	10	Concrete	Below Grade External	None Change in Material Properties due to leaching	None Required Maintenance Rule Structures Program	N/A Plant specific	N/A N/A	N/A N/A	See Section 6.4.
Structural Steel – beams, columns, plates, trusses (Transmission Towers)	10	Galv. Steel	Internal	Loss of Material	Maintenance Rule Structures Program	III.A3.2-a	No	Y	

AGING MANAGEMENT REVIEW FOR VIRGIL C. SUMMER NUCLEAR STATION
EARTHEN EMBANKMENTS

Prepared by: Sing Chu
Reviewed by: Robert Whorton

AGING MANAGEMENT REVIEW SUMMARY TABLE

Component Type Within Structure Boundary	Component Intended Function(s) For VCSNS (See Key)	Materials	Environment	Aging Effect	Aging Management Programs	GALL Item Number	GALL Recommendation (Further Evaluation)	Match with GALL (Y, N, N/A, Partial)	Notes
North Berm	4	Earthen	External	Loss of Material / erosion Cracking / settlement	Maintenance Rule Structures Program	III.A6.4-a	No	Partial	The Maintenance Rule Structures Program inspection attributes and acceptance criteria are consistent with RG 1.127 requirements.
Service Water Pond Dams (North Dam, South Dam, East Dam, and West Embankment)	4,13,15	Earthen	External / Below Grade / Raw Water	Loss of Material / erosion Cracking / settlement	Service Water Pond Dam Inspection Program (North Dam, South Dam, East Dam, and West Embankment)	III.A6.4-a	No	Y	See Section 8.6 Water Impoundment Structures – Dams TLAA.

Key to Component Intended Functions

1. Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
2. Provide shelter/protection to safety-related components.
3. Provide structural and/or functional support to safety-related equipment.
4. Provide flood protection barrier (internal and external flooding event).
5. Provide pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
6. Provide spray shield or curbs for directing flow.
7. Provide shielding against radiation.
8. Provide missile barrier (internally or externally generated).
9. Provide shielding against high energy line breaks.
10. Provide structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.
11. Provide pipe whip restraint.
12. Provide path for release of filtered and unfiltered gaseous discharge.
13. Provide source of cooling water for plant shutdown.
14. Provide heat sink during SBO or design basis accidents.
15. Impound water for ultimate heat sink during loss of Monticello Reservoir.