SECTION 4 - TIME-LIMITING AGING ANALYSIS

4.0 TIME-LIMITED AGING ANALYSIS

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSIS

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 as those licensee calculations and analyses that meet six specific criteria. 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses be provided as part of an application for a renewed license.

10 CFR 54.3 Definitions

Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);
(2) Consider the effects of aging;
(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
(4) Were determined to be relevant by the licensee in making a safety determination;
(5) 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 §54.4(b); and
(6) Are contained or incorporated by reference in the CLB.

10 CFR 54.21 (c)

(c) An evaluation of time-limited aging analyses.

(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.

(2) A list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

4.1.1 TLAA PROCESS OVERVIEW

The process used to identify the time-limited aging analyses for Virgil C. Summer Nuclear Station (VCSNS) is consistent with the guidance provided in NEI 95-10 **[Reference 4.8.5.1]**. Calculations and analyses that meet the six criteria of 10 CFR 54.3 were identified by searching the current licensing basis, which includes the FSAR, Technical Specifications, engineering calculations, technical reports, docketed licensing correspondence, and applicable Westinghouse Topical Reports (WCAPs).

4.1.2 EXEMPTIONS TO 10 CFR 50.12

10 CFR 54.21(c) also requires that an application for a renewed license include a list of current plant-specific exemptions granted pursuant to 10 CFR 50.12 that are based on time-limited aging analyses as defined in 10 CFR 54.3.

Each exemption granted to VCSNS was evaluated. The evaluation established whether the exemption is still in effect, and if so, whether the exemption is based on a time-limited aging analysis. No current 10 CFR 50.12 exemptions based on a time-limited aging analysis as defined in 10 CFR 54.3 have been identified for VCSNS.

4.1.3 TLLA SUMMARY

The VCSNS calculations and evaluations that met all six of the criteria listed in 10 CFR 54.3 are listed in Table 4.1-1 and discussed in Sections 4.2 through 4.7 of this Application. Selected TLAAs that did not meet all six criteria for VCSNS are also listed and discussed based on their inclusion in NUREG-1801 [**Reference 4.8.4.2**] or NUREG-1800 [**Reference 4.8.4.1**].

Three options are provided in 10 CFR 54.21(c)(1) for the applicant to address each plant specific time-limited aging analyses identified. For each TLAA identified, these regulations direct that the applicant must 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.

Table 4.1-1:
VCSNS SPECIFIC - ACTUAL TLAA SUMMARY

GROUP	DESCRIPTION	SECTION	TLAA OPTION
Reactor Vessel Neutron Embrittlement	Upper Shelf Energy	4.2.1	Option ii
	Pressurized Thermal Shock	4.2.2	Option ii
	Pressure-Temperature (P-T) Limits	4.2.3	Option ii
Metal Fatigue	ASME Section III, Class I	4.3.1	Option iii
	ASME Section III, Class 2 and 3 Piping Fatigue	4.3.2	Option i
EQ	Environmental Qualification	4.4	Option iii
Concrete Containment Tendons	Concrete Containment (Reactor Building) Tendon Prestress Analysis	4.5	Option iii
Containment Liner Plate, Metal Containments, And Penetration Fatigue Analysis	Containment (Reactor Building) Liner	4.6.1	Option ii
	Metal Containments	4.6.2	Not a TLAA
	Containment (Reactor Building) Isolation Bellows	4.6.3.1	Not a TLAA
	Containment (Reactor Building) Isolation - Fracture Toughness And Effects Radiation	4.6.3.2	Not a TLAA
Other	Reactor Coolant Pump Flywheel	4.7.1	Option i
	Leak-Before-Break Analyses	4.7.2	Option ii
	Crane Load Cycle Limit	4.7.3	Option i
	Service Water Intake Structure Settlement	4.7.4	Option ii

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are contained in the following sections of 10 CFR Part 50:

10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation," requires all light water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant boundary as set forth in Appendices G and H of 10 CFR 50.60.

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," contains fracture toughness requirements for protection against pressurized thermal shock.

4.2.1 UPPER-SHELF ENERGY

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that reactor vessel beltline materials must have an initial, pre-irradiation, Charpy Upper Shelf Energy of no less than 75 ft-lb and must maintain a Charpy Upper Shelf Energy of no less than 50 ft-lb throughout the life of the reactor vessel, (unless it is demonstrated, in a manner approved by the Director, Office of Nuclear Reactor Regulation (NRR), that lower values of Charpy Upper Shelf Energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code) [**Reference 4.8.5.2**].

The VCSNS-calculated beltline fluence is one of the factors used in the determination of the decrease in Charpy Upper Shelf Energy due to radiation embrittlement and thermal aging of the reactor vessel. Since upper shelf energy is a measure of the fracture toughness of a material, a decrease in the upper shelf energy of reactor vessel materials implies a reduction in fracture toughness of the vessel.

Charpy Upper Shelf Energy values are calculated for beltline region materials using NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [**Reference 4.8.4.3**]. In response to NRC Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1, and based upon examinations of the first three VCSNS surveillance capsules, VCSNS reported [**References 4.8.2.1** and **4.8.4.4**] the end of current license (32 EFPY) upper-shelf energy and limiting beltline component to be 67.5 ft-lb for intermediate plate A9154-1 (transverse-orientation).

Revision to the analyses is required to calculate Charpy Upper-Shelf Energy for the end of the extended operating period. The two remaining VCSNS reactor surveillance capsules require additional exposure to neutron fluence in order to provide data that correlates to estimated fluence on the vessel at the end of the period of extended operation. Following adequate capsule exposure, a capsule will be withdrawn and analyzed. The Charpy Upper Shelf Energy will be recalculated for additional fast neutron fluence corresponding to the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to calculate the RPV Charpy Upper Shelf Energy for the end of the extended operating period.

4.2.2 PRESSURIZED THERMAL SHOCK

Beltline fluence is one of the factors used to determine the margin for pressurized thermal shock due to radiation embrittlement and thermal aging of the reactor vessel. The margin is the difference between the maximum nil ductility reference temperature in the limiting beltline component (RT_{PTS}) and the screening criterion established in accordance with 10 CFR 50.61. The method for calculating RT_{PTS} values is consistent with NRC Regulatory Guide 1.99, [**Reference 4.8.4.3**].

$$RT_{PTS} = RT_{NDT}\,(\text{Unirradiated}) + M + [(CF) * (FF)]$$

Where the product of the Chemistry Factor (CF) and Fluence Factor (FF) is alternatively called ΔRT_{PTS}. RT_{NDT} is the reference temperature for a reactor vessel material in the preservice or unirradiated condition. M is the margin added to account for the uncertainties in the value of RT_{NDT}.

The Chemistry Factor accounts for the effects of copper and nickel content on radiation embrittlement. If the specific material composition of the vessel beltline materials is available some conservatism may be eliminated from the RT_{PTS} determination. The calculated RT_{PTS} is refined in conjunction with analysis of each successive surveillance capsule.

The RT_{PTS} values for VCSNS were calculated for the current 40-year operating term. Per WCAP-15103, all of the beltline materials in the VCSNS reactor vessel have end-of-life (EOL) 32 EFPY and life extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds. These results show that VCSNS will not exceed the RT_{PTS} screening criteria during the current (32 EFPY) or extended (48 EFPY) operating license. The RT_{PTS} value will be recalculated when one of the two remaining VCSNS surveillance capsules is removed from the vessel. VCSNS intends to remove at least one of the surveillance capsules when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to calculate the RT_{PTS} data for the end of the extended operating period.

4.2.3 PRESSURE-TEMPERATURE (P-T) LIMITS

Beltline fluence is one of the factors used to calculate revisions to pressure-temperature limits for heatup and cooldown due to radiation embrittlement of the reactor vessel. This input is based on calculation of an Adjusted Reference Temperature (ART) using methodology of NRC Regulatory Guide 1.99, **[Reference 4.8.4.3]**. This methodology is very similar to that used to calculate RT_{PTS}. However, the calculation of ART also considers attenuation of the fast neutron fluence through the vessel wall to the depth of the postulated flaw.

The formula for calculating ART is listed below.

$$ART = IRT_{NDT} + \Delta RT_{NDT} + M$$

Where ΔRT_{NDT} is the product of the Chemistry Factor (CF) and attenuation-corrected fast neutron fluence factor. RT_{NDT} is the reference temperature for a reactor vessel material in the pre-service or unirradiated condition. M is the margin added to account for the uncertainties in the value of RT_{NDT}. IRT_{NDT} is the initial RT_{NDT}, the reference temperature for the unirradiated material as defined in paragraph NB-2332 of ASME Section III.

The Chemistry Factor accounts for the effects of copper and nickel content on radiation embrittlement. If the specific material composition data is available some conservatism may be eliminated from the ART. The calculated ART is refined in conjunction with analysis of each successive surveillance capsule. Allowable pressure-temperature curves are generated for steady state and each finite cooldown rate specified, assuming a reference flaw at the inside (most limiting) surface of the reactor vessel. A composite cooldown limit curve is constructed as the minimum of each of these curves. Similarly, allowable pressure-temperature curves are generated for steady state and each finite heatup rate specified considering each of the worst-case reference flaw locations (outside vessel surface and inside vessel surface).

The current VCSNS heatup and cooldown curves are based on calculations for the current 40-year operating term. VCSNS will revise the calculated value of ART and associated pressure-temperature limits for heatup and cooldown when one of the two remaining VCSNS surveillance capsules is removed from the vessel. VCSNS intends to remove at least one of the surveillance capsules when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to develop pressure-temperature limits for the period of extended operation.

4.3 METAL FATIGUE

4.3.1 ASME SECTION III, CLASS 1

The issue of thermal fatigue for ASME Section III Class 1 **[Reference 4.8.5.4]** components has been identified as a time-limited aging analysis for VCSNS. All six of the criteria in 10 CFR 54.3 are satisfied. Class 1 components have been designed with the transient cycle assumptions in Table 5.2-2 of the Final Safety Analysis Report **[Reference 4.8.3.2]**.

Currently, there are no components in the scope of the VCSNS Inservice Inspection (ISI) Program that contain flaws which exceed acceptance standards and require analysis to demonstrate acceptance.

The VCSNS ISI Program involves the monitoring of thermal transients. The Thermal Fatigue Management Program described in **Appendix B.3.2** of this application is equivalent to the corresponding program described and evaluated in Section X.M1 of NUREG-1801 **[Reference 4.8.4.2]**. However, enhancements to the program are warranted to incorporate the new guidance in EPRI Report MRP-47 concerning "Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," which was published in October 2001**[Reference 4.8.5.3]**.

GSI-190, Fatigue Evaluation of Metal Components for 60-year Plant life, relates to environmental effects on fatigue of reactor coolant system components for 60 years and was closed by the NRC **[Reference 4.8.2.6]**. In the closure letter, the NRC concluded that licensees should address the effects of the reactor coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The VCSNS Thermal Fatigue Management Program will be revised by the end of the current license term (40 years) to base future projections on 60 years of operation and to account for environmental effects of the reactor coolant environment on RCS components. The Thermal Fatigue Management Program as revised will meet the corresponding program described in NUREG-1801 (GALL) **[Reference 4.8.4.2]**, Section X.M.1. This program meets the requirement of 10 CFR 54.21(c)(1) by the utilization of option (iii).

4.3.2 ASME SECTION III, CLASS 2 AND 3 PIPING FATIGUE

Piping systems, designed in accordance with ASME Section III, Class 2 and Class 3 **[Reference 4.8.5.4]** or ANSI B31.1 **[Reference 4.8.5.5]**, utilize allowable stress values based on a stress reduction factor. VCSNS **FSAR Table 3.2-3** states the versions of ASME Section III and ANSI B 31.1 that apply to different plant components. Table 4.3-1 of NUREG-1800 **[Reference 4.8.4.1]** includes Class 2 and 3 piping fatigue. The stress range reduction factors range from 1 (no reduction) for 7000 cycles or less, to a stress range reduction factor of 0.5 for 100,000 cycles or more.

For all of the systems reviewed, except the Post-Accident Sampling and Nuclear Sampling Systems, the number of thermal cycles is related to the heatup and cooldown of the plant (steam and primary), which ideally, occurs once a cycle (18 months). Even if this is conservatively assumed to occur once a month for 60 years, then the total thermal cycles would only be 720, which is approximately one–tenth of the allowed 7000 cycles. Therefore, it is conservative to assume that Class 2 and 3 and ANSI B31.1 systems are adequately evaluated for fatigue for the period of extended operation.

The estimated number of thermal cycles at 60 years on the Post-Accident Sampling and Nuclear Sampling Systems system is 6668 cycles which is less than the 7000 allowed thermal cycles by a small margin. This magnitude of the estimate is due to past operational practices of obtaining frequent samples by use of the "B" RCS Loop sampling line. In order to ensure that the number of cycles remained below 7000, procedural controls will be implemented to ensure that sampling activities are discontinued on the "B" RCS Loop sampling line for all but emergency situations and/or revise the calculations to verify the acceptability of a higher safety factor.

Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (i) for this area because piping fatigue for ASME Section III, Class 2 and 3, and ANSI B31.1 is adequately analyzed for the period of extended operation.

4.4 ENVIRONMENTAL QUALIFICATION (EQ)

The environmental qualification analyses for electrical equipment included in the Environmental Qualification (EQ) Program were identified as potential time-limited aging analyses for VCSNS. Since the EQ analyses meet all six criteria for a TLAA specified in 10 CFR 54.3, EQ is considered a plant specific TLAA at VCSNS.

The EQ Program is based on the requirements of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." As part of the EQ evaluation, the electrical equipment in the program is given a quantified value for its service life in a given environment (called the 'qualified' life). This qualified life is often expressed in terms of being greater than 40 years, the original duration of the plant's license. Components in the EQ Program that have calculated lifetimes equal to or greater than 40 years are identified as time-limited aging analyses, and require re-analysis to meet the duration of the new plant operating license (60 years from initial plant start-up).

The EQ documents at VCSNS that are subject to TLAA review include the EQ binders, which are comprehensive reports prepared to support the qualification of each unique type of electrical component for the plant environments in which they are located. Each binder contains or references a vendor test report or an analysis that justifies the qualification of the equipment. Each binder contains or references either a calculation of qualified life or an evaluation to justify a qualified life. Additional documents in the EQ program subject to TLAA review include plant calculations, vendor reports, and the EQ and Regulatory Guide 1.97 design basis document. Many environmental qualification calculations of electrical equipment are identified as time-limited aging analyses for VCSNS. These calculations are considered the technical rationale that the current licensing basis will be maintained during the period of extended operation.

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI [**Reference 4.8.2.5**]. In this letter, the NRC states, "With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the Statements of Consideration is to provide a technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time." Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

The EQ TLAAs for VCSNS are listed in Table 4.4 -1. These documents were selected from the overall list of EQ binders and a review of other associated EQ documents at VCSNS. The EQ binders identified as TLAAs include all components with a qualified life of equal to or greater than 40 years. VCSNS has elected to utilize 10 CFR 54.21(c)(1) - option (iii) to demonstrate that the Environmental Qualification Program will continue to adequately manage the effects of aging of the electrical components for the period of extended operations. Components with qualified lives less than 40 years are not considered to have TLAAs and will not be evaluated further.

**Table 4.4-1:
ENVIRONMENTAL QUALIFICATION DOCUMENTS
"HARSH" EQ BINDERS**

Document No.	Title	Manufacturer
EQDP-H-CA0-R11-1	Firezone R Power, Control, and Instrument Cable	Rockbestos
EQDP-H-CA1-K03-1	600 Volt Control Cable	Kerite
EQDP-H-CA1-K03-2	Power and Control Cable	Kerite
EQDP-H-CA1-O01	600 Volt Control Cable (Armored)	Okonite
EQDP-H-CA1-R11-1	600 Volt Control Cable	Rockbestos
EQDP-H-CA2-K03	600 Volt Power Cable	Kerite
EQDP-H-CA2-O01	8 kV Power Cable	Okonite
EQDP-H-CA4-B20	Coaxial Twinax Instrument Cable	Brand-Rex
EQDP-H-CA4-O01	Special Instrument Cable	Okonite
EQDP-H-CA4-R05	Switchboard Wire	Raychem
EQDP-H-CA4-R11-1	Switchboard Wire	Rockbestos
EQDP-H-CA4-R11-2	Instrumentation Cable	Rockbestos
EQDP-H-CA4-S02	Instrument Cable	Samuel Moore (Dekoron)
EQDP-H-CA6-C06	CETS and HRCM MI Cable System / Connectors	CE/ERD
EQDP-H-CA6-K03	Thermocouple Extension Cable	Kerite
EQDP-H-CA6-O01	Heat Tracing Thermocouple Cable	Okonite
EQDP-H-CA7-O01	Splice Materials	Okonite
EQDP-H-CA7-R05-1	Raychem Splicing Products	Raychem
EQDP-H-CA7-R05-2	Motor Connection Kits	Raychem
EQDP-H-CA7-R05-3	Nuclear High Voltage Terminations	Raychem
EQDP-H-C01-A05	Electrical Cable Terminals	AMP

**Table 4.4-1:
ENVIRONMENTAL QUALIFICATION DOCUMENTS
"HARSH" EQ BINDERS**

Document No.	Title	Manufacturer
EQDP-H-C02-S04	Terminal Blocks	States
EQDP-H-CO5-C08	Electrical Penetration Assemblies	Conax Buffalo
EQDP-H-CO5-D01-1	Triax Connectors, M06 Modules	D.G. O'Brien
EQDP-H-CO5-D01-2	Penetration Modules, Plugs, and Hermetic Connectors	D.G. O'Brien
EQDP-H-CO6-C08-1	Nuclear Service Connectors	Conax Buffalo
EQDP-H-CO6-E03-1	Quick Disconnect Connectors	EGS
EQDP-H-CO6-E03-2	Grayboot Connector	EGS
EQDP-H-CO6-L02	Quick Disconnect Multipin T/C Connectors and Socket Plugboard	Litton/Veam
EQDP-H-CO6-N01	Hermetic Connectors and DC734 RTV Thread Sealant	Namco
EQDP-H-HW5-E03-1	Conduit Seal	EGS
EQDP-H-IN1-B05-1	Transmitters Model 764	Barton
EQDP-H-IN1-B05-2	Transmitters Model 763, 763A	Barton
EQDP-H-IN1-B05-3	Transmitters Model 763	Barton
EQDP-H-IN1-B05-4	Transmitters Model 764	Barton
EQDP-H-IN5-P03	Resistance Temperature Detector	Pyco
EQDP-H-IN6-G02-1	Neutron Detector Assembly	Gamma-Metrics
EQDP-H-IN6-G02-2	Neutron Flux Mon. System Amplifier Panel	Gamma-Metrics
EQDP-H-IN6-V05	HRCM, Cable Assembly Hermetic Connector	Victoreen/Hermetic Seal Corp
EQDP-H-IN7-T03	Valve Flow Monitoring System, Transient Shield, Sensor and Charge Converter	Technology for Energy/Endevco

Table 4.4-1: ENVIRONMENTAL QUALIFICATION DOCUMENTS "HARSH" EQ BINDERS

Document No.	Title	Manufacturer
EQDP-H-MO1-G03	EFW And RBS Pump Motors	GE
EQDP-H-MO1-R07-1	Class 1E Continuous Duty AC Motors	Reliance Electric
EQDP-H-MO1-W01	CCW Pump Motor	Westinghouse
EQDP-H-MO1-W10A and B	RHR & Charging Pump Motors	Westinghouse
EQDP-H-MO1-R07	Fan Motors	Reliance Electric
EQDP-H-MO2-R07	Fan Motors	Reliance
EQDP-H-SE2-E03-1	Grafoil Paste Thread Sealant	EGS
EQDP-H-SW1-T01	Transfer Switch	Terrac – Gould
EQDP-H-SW4-N01-1	Limit Switches EA-180	Namco
EQDP-H-SW4-N01-2	Limit Switches EA-740	Namco
EQDP-H-SW5-S07-1	Temperature Switch	Static-O-Ring
EQDP-H-VO4-A11	Solenoid Valves	ASCO
EQDP-H-VO4-A13	Solenoid Valves	Allied
EQDP-H-VO4-C16-1	Solenoid Valves	Chicago Fluid Power
EQDP-H-VO4-T02-1	Solenoid Valves	Target Rock Model 83GG-001
EQDP-H-VO4-T02-2	Solenoid Valves	Target Rock Model 97A-001
EQDP-H-VO4-V01	Solenoid Valves	Valcor
EQDP-H-VO5-L01	Motor-Operated Valve Actuators	Limitorque
EQDP-H-VO5-R17	Motor-Operated Valve Operators	Rotork
EQDP-H-WR3-W01	Electric Hydrogen Recombiner	Westinghouse

4.4.1 TLAA EVALUATION

The VCSNS Environmental Qualification Program ensures that the effects of aging will be adequately managed for the period of extended operation. The VCSNS EQ Program is evaluated with respect to the program attributes described in NUREG-1801 **[Reference 4.8.4.2]** in Appendix B of this application. Plant documents that address the qualified lives of the EQ components will be revised to reflect the period of extended operation.

The EQ Program at VCSNS implements the requirements of 10 CFR 50.49 and is an aging management program for license renewal. Re-analysis of an existing aging evaluation, to extend the qualification of electrical components, is performed as part of the EQ Program. These calculations are performed in response to adjusted plant environments, new vendor data, and other input which might impact the conclusions of the EQ binders.

The important attributes of re-analysis include analytical methods, data collection and reduction methods, the underlying assumptions, the acceptance criteria, and any corrective actions to components as a result of prior evaluation. These re-analysis attributes are discussed below.

4.4.1.1 Analytical EQ Re-analysis Methods

The analytical methods used in the re-analysis of an aging evaluation are the same as those applied in the prior analysis. The Arrhenius methodology is an acceptable thermal model for reaction rate and thermal aging evaluation. The analytical method used for radiation aging evaluations is to demonstrate qualification for the total integrated dose via test. The total integrated dose is the normal operational environmental dose plus the projected accident dose. For license renewal, it is acceptable to establish a 60-year normal operational dose by taking the 40-year dose previously established and multiplying this value by 1.5. The result is added to the postulated accident dose to obtain a 60-year total integrated dose. For cyclical aging evaluations, a similar methodology is acceptable. Other methods may be justified on a case-by-case basis.

4.4.1.2 Data Collection And Reduction Methods For EQ

Reducing excess conservatism in electrical component service conditions (i.e., temperature, radiation, number of cycles) used in prior evaluations is the chief method used for re-analysis. Temperature data used in an aging analysis is typically conservative and is based on plant design temperatures. Actual plant operating temperature data is typically less than design values. Actual plant operating data may be obtained from temperature monitors specifically installed for EQ measurements, data taken by operators during rounds, or other temperature monitors in place in the plant.

4.4.1.3 Evaluation Of Underlying EQ Assumptions

EQ component aging evaluations typically contain sufficient conservatism to account for most environmental changes, which occur as a result of plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities, the affected EQ component(s) is (are) evaluated and appropriate corrective actions are taken, which may include revisions to the qualification bases and conclusions.

The re-analysis of an aging evaluation could extend the qualification of a subject component. If the qualification cannot be extended by re-analysis, the component is to be refurbished, replaced or re-qualified prior to the expiration of the current qualification. The timing of the re-analysis must permit sufficient time to refurbish, replace, or re-qualify the component if the re-analysis effort is unsuccessful.

4.4.2 EQ PROGRAM REVIEW

The VCSNS EQ Program, as described in Appendix B of this application, meets the criteria of an acceptable aging management program, as defined in Section X.E1 of NUREG-1801, Volume II **[Reference 4.8.4.2]**.

4.4.3 EQ TLAA EVALUATION CONCLUSIONS

The VCSNS Environmental Qualification Program ensures that the effects of aging will be adequately managed for the period of extended operation. The VCSNS EQ Program is evaluated with respect to the program attributes described in NUREG-1801 **[Reference 4.8.4.2]** in Appendix B of this License Renewal Application. Plant documents, which address the qualified lives of the EQ components, will be revised to reflect the period of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (iii) to demonstrate that the Environmental Qualification Program will continue to adequately manage the effects of aging of electrical components for the period of extended operation.

4.5 CONCRETE CONTAINMENT (REACTOR BUILDING) TENDON PRESTRESS ANALYSIS

The VCSNS Reactor Building was prestressed in order to have low-strain linear response at design loads and thus assure integrity of the liner. The exterior wall is post-tensioned in both vertical and hoop directions. Hoop tendons are anchored on three (3) buttresses, each spaced 120° apart along the circumference of the containment wall. On the dome, a three-way post-tensioning system is employed.

General Design Criteria 53, "Provisions for Containment Testing and Inspection," of Appendix A, requires that the reactor containment be designed to permit periodic inspection and surveillance. The guidelines to perform inservice inspections of the tendons (also referred to as "tendon surveillances") are established by NRC Regulatory Guide 1.35 [**Reference 4.8.4.5**]. These inspections are required to be performed at 1, 3, and 5-year intervals, following the Structural Acceptance Test and every 5 years thereafter. During these inspections, pre-designated tendons are examined. The examinations include tendon force measurements, visual examination of anchorages and surrounding concrete, tendon wire tensile tests of a limited number of wires, and chemical tests of the corrosion protection medium (grease).

In March 1995, the NRC issued a new rule, 10 CFR 50.55a. This rule invoked new containment inspection requirements, including the requirements of ASME Code, Section XI, Subsections IWE and IWL, 1992 Edition and 1992 Addenda [**Reference 4.8.5.2**]. This rule also imposed new requirements for future tendon surveillances.

The VCSNS tendon surveillance program is based on proposed Revision 3 of Regulatory Guide 1.35 [**Reference 4.8.4.5**]. The Regulatory Guide remained in a proposed status until July 1990 when the finalized Revision 3 was issued. The NRC accepted the VCSNS tendon surveillance program based on the proposed Revision 3 of Regulatory Guide 1.35 [**Reference 4.8.2.3** and **4.8.2.4**].

VCSNS has performed all required tendon surveillances. The Fourth Period (10th year-1990) Tendon Surveillance was used to re-tension 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. The Fifth Period (15th year - 1996) and Sixth Period (20th year - 2000) Tendon Surveillances were completed with acceptable results. 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.

Chapter X.S1, "Concrete Containment Tendon Prestress," of NUREG-1801 [**Reference 4.8.4.2**], 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) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

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

4.6.1 CONTAINMENT (REACTOR BUILDING) LINER

The Reactor Building liner provides an essentially 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 code for normal operation.

A VCSNS calculation considered Reactor Building liner fatigue for 40 years per ASME Section III, paragraph NE-3131, 1974 with all applicable addenda [**Reference 4.8.5.4**]. 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 are satisfied for 60 years. Thus VCSNS utilizes 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.

4.6.2 METAL CONTAINMENTS - N/A

4.6.3 CONTAINMENT (REACTOR BUILDING) ISOLATION

Time-Limited Aging Analyses issues related to containment isolation are Reactor Building isolation bellows fatigue, fracture toughness of the penetration materials and effects of radiation.

4.6.3.1 Containment (Reactor Building) Isolation Bellows

FSAR Section 6.2.6.2.1.1 [Reference 4.8.3.2], "Piping Penetrations and Spares," states:

"Cold penetrations are sealed by a flat plate, 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 seale on the inside of the 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."

Therefore, the credited Reactor Building isolation barrier does not utilize any flexible seals between the process pipe and Reactor Building liner. The bellows are not included in this isolation barrier and have no testing or inspection requirements. The 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 either Criterion 4 or Criterion 5 of 10 CFR 54.3 for a plant specific TLAA because they are not credited in a safety determination and do not perform a containment isolation function. Since the bellows were not required for Reactor Building isolation, they are not within the scope of license renewal (Criterion 1 of 10 CFR 54.3 for a plant specific TLAA). The bellows were not determined to be relevant in making a safety determination since they have no Reactor Building isolation function. The bellows also fail to meet Criterion 5 of 10 CFR 54.3 for a TLAA in that they do not support the capability of the containment isolation system and associated components in performing their design function of Reactor Building isolation.

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, the 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 isolation bellows and associated analyses do not qualify as a TLAA at VCSNS.

4.6.3.2 Containment (Reactor Building) Isolation - Fracture Toughness And Effects Radiation

The FSAR and the associated NRC Safety Evaluation Report, NUREG-0717, include statements involving the life of the materials used in the RB penetrations and the fracture toughness of the containment pressure boundary.

FSAR Section 6.2.4.5 [Reference 4.8.3.2], "Materials", states that the containment isolation system materials were selected to perform their design function for 40 years based on an integrated radiation dose of 10^8 rads. Materials that contact Reactor Building spray are resistant to corrosion, and isolation system components, which contain zinc and aluminum, are kept to a minimum. No time-limited aging analysis was identified for these components. The statement that components would perform their design function for 40 years was unsupported in that no analyses were identified that specifically calculated the design life.

NUREG-0717 [**Reference 4.8.3.1**] 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 NUREG-0717, it is clear that the NRC has determined that the ferritic materials used in the construction of the VCSNS containment meet the appropriate requirements of the ASME Code, comply with GDC-51, and behave in a non-brittle manner. These statements do not present a time-limited aging analysis.

No time-limited aging analyses or calculations were identified that require revision for the effects of radiation on containment isolation components.

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 REACTOR COOLANT PUMP FLYWHEEL

The reactor coolant pump (RCP) motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing, and inside the motor frame. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway from stresses due to starting the motor. Therefore, this topic is considered to be a time-limited aging analysis for license renewal.

WCAP-14535A, "Topical Report of Reactor Coolant Pump Flywheel Inspection Elimination," [**Reference 4.8.1.1**], supports the elimination of RCP flywheel inspections, based on the insignificant increase in probability of failure achieved by inspections over a 60-year service life, the relatively robust nature of flywheels with respect to detectable flaws, and the likelihood that disassembly and re-assembly for continued inspections presented the largest risk of causing flaws in the RCP flywheels. The estimated magnitude of fatigue crack growth during plant life was conservatively calculated based on an assumed initial radial crack length of 10% through the flywheel (from the keyway to the flywheel outer radius). The analysis assumed 6000 cycles of pump starts and stops for a 60-year plant life. Crack growth from postulated flaws in each flywheel was only a few mils. The existing analysis is valid for the period of extended operation. Reaching 6000 starts in 60 years would require a pump start on average every 3.7 days, which is extremely conservative. The findings of the analysis, which have been approved by the NRC, determined that the crack growth for the postulated flaw over 60 years of operation to be acceptable.

The existing analysis in WCAP-14535A is valid for the period of extended operation and demonstrates that the reactor coolant pump flywheels will continue to perform their design functions for the period of extended operation. Thus, Option (i) has been incorporated to satisfy 10 CFR 54.21 (c)(1) for the reactor coolant pump flywheels.

4.7.2 LEAK-BEFORE-BREAK ANALYSES

Leak-before-break (LBB) analyses evaluate postulated flaw growth in the primary loop piping of the Reactor Coolant System. The intent of these plant specific analyses is to allow utilities to remove the dynamic effects of primary loop pipe ruptures from the design of reactor coolant loop pipe supports. In a letter dated January 11, 1993 [**Reference 4.8.2.2**], the NRC indicated concurrence with the VCSNS LBB analysis. Since the analysis considerations could be influenced by time, LBB is a TLAA for VCSNS.

The current LBB analysis accounts for the replacement of steam generators, power up-rate, and the RCS A-hot leg repair.

The LBB analysis is based on stainless steel at 40 years and will need to be revised to account for the proposed period of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to develop a revised leak-before-break analysis for the period of extended operation.

The Thermal Fatigue Management Program will manage the Class 1 component thermal cycle count assumptions that form the foundation of this resolution, thus ensuring the continued validity of the LBB analysis, the management of thermal fatigue for these components, and the continued performance of their intended function(s) for 60 years of operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (iii) for the Thermal Fatigue Program as a whole.

4.7.3 CRANE LOAD CYCLE LIMIT

A potential TLAA issue was identified relatec › the cranes and associated crane supports that could theoretically affect irradiated fuel during refueling operations. The cranes that meet the criterion are as follows:

- Reactor Building Polar Crane
- Spent Fuel Cask Handling Crane
- Fuel Handling Machine (Spent Fuel Pit Bridge and Hoist)
- Refueling Machine (Reactor Cavity Manipulator Crane)

The Crane Manufacturers Association of America (CMAA) Specification No. 70 (CMAA 70) classifies these cranes as Class "A" cranes **[Reference 4.8.5.6]**. 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."

The current version of CMAA 70, Section 2.8, "Crane Service in Terms of Load Class and Load Cycles" states 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. Since the cranes listed above were designed to CMAA 70, the load cycle limits apply. Cranes and crane supports are considered a TLAA because they satisfy the six criteria for a TLAA defined in 10 CFR 54.3.

4.7.3.1 Reactor Building Polar Crane

The Reactor Building polar crane, including the bridge girders, end trucks and trolley, were originally designed for construction loads of 360 tons. A seismic analysis was performed for the maximum, non-construction load of 150 tons. Since construction, the polar crane was only used for capacity lifts during the VCSNS steam generator replacement project. The steam generator lifts were rated capacity lifts of 354 tons. 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) D-3 steam generator and one for each replacement (new) Delta 75 steam generator, which yields a total of six capacity lifts. Imposing an extremely conservative safety factor of five 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.

4.7.3.2 Spent Fuel Cask Handling Crane

The spent fuel cask handling crane is rated for 125 tons. The projected number of fuel cask lifts is far less than 200,000 over a 60-year period. Assuming that (1) 70 fuel bundles are replaced every 18 months for 60 years, in addition to the original 157 bundles (equals 2957 bundle lifts), (2) each bundle is loaded individually into separate casks, and (3) each cast is lifted twice; the total number of lifts is less than 10,000 lifts. This estimate is extremely conservative and the total number still does not approach the design limit for the crane. Therefore, the spent fuel cask handling crane has been analyzed and shown to be adequate for the period of extended operation.

4.7.3.3 Fuel Handling Machine And Refueling Machine

The fuel handling machines consist of a fuel handling machine (Spent Fuel Pit Bridge and Hoist) and refueling machine (Reactor Cavity Manipulator Crane).

The refueling machine and fuel handling machine lift load consists of the combination of a spent fuel or new fuel assembly and handling tool. The maximum load weighs approximately 2500 pounds [**Reference 4.8.3.3**]. The refueling machine crane is rated for 3000 pounds.

The fuel handling machine hoist is designed with a margin of two for lifts. The hoist capacity is 4000 pounds while the combined weight of the fuel assembly with a rod cluster control assembly and the spent fuel assembly handling tool is approximately 2000 pounds. The fuel handling machine structure is designed to commercial standards and also analyzed to the requirements of Section III, Appendix XVII of the ASME Boiler & Pressure Vessel Code, and has the margins included in the allowable stresses of the Code [**Reference 4.8.5.7**].

Conservatively assuming 400 lifts each refueling cycle for each machine (i.e., loading 70 new fuel assemblies, a full core offload of 157 fuel assemblies, a full core reload of 157 fuel assemblies and 16 miscellaneous fuel assembly shuffles), and 40 refueling cycles in 60 years results in approximately 16,000 cycles in 60 years.

The fuel handling cranes were analyzed for up to 200,000 cycles of maximum load based on the crane manufacturer's calculations and CMAA Specification No. 70. Since the conservative estimate of load cycles is far less than the design limit in CMAA 70, actual cycle counting is not warranted.

4.7.3.4 Summary

The VCSNS fuel handling machines (Spent Fuel Cask Handling Crane, Fuel Handling Machine, and Refueling Machine) and Reactor Building polar crane are adequately analyzed and designed for fatigue through the term of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (i) to demonstrate that the fuel handling machines and Reactor Building polar crane are adequately analyzed for the period of extended operation.

4.7.4 SERVICE WATER INTAKE STRUCTURE SETTLEMENT

The Service Water Intake Structure (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 not covered with soil is submerged within the Service Water Pond. The function of the intake structure is to draw water from the Service Water Pond into the Service Water Pump House (SWPH).

Section 3.7.2, "Seismic System and Subsystem Analysis," of NUREG-0717 [**Reference 4.8.3.1**] evaluated the VCSNS SWIS. It was noted that excessive non-uniform settlement of the intake structure occurred during construction causing considerable cracking. This settlement was analyzed in a SWPH calculation which was originally based on a plant design life of 40 years. Since this issue meets all six criteria in 10 CFR 54.3, SWIS settlement is considered an actual TLAA for VCSNS.

The VCSNS calculation 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 SWIS settlement.

4.8 REFERENCES

4.8.1 CALCULATIONS AND ANALYSES

4.8.1.1	Westinghouse WCAP-14535a, "Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination," November 1996.

4.8.2 CORRESPONDENCE

4.8.2.1	Letter, J. L. Skolds, SCE&G, to NRC Document Control Desk, dated June 30, 1992, "Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, (LTR 920001)."
4.8.2.2	Letter, G. F. Wunder, NRC, to J. L. Skolds, SCE&G, dated January 11, 1993, "Safety Evaluation of Request to Use Leak-Before-Break for Reactor Coolant System Piping, Virgil C. Summer Nuclear Station, Unit 1 (TAC No. M83971)."
4.8.2.3	Letter, J. J. Hayes, NRC, to O. S. Bradham, SCE&G, dated 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)."
4.8.2.4	Letter, O. S. Bradham, SCE&G, to NRC Document Control Desk, dated September 29, 1988, "Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Containment Structural Integrity Surveillance Requirements."
4.8.2.5	Letter, C. I. Grimes, NRC, to D. Walters, NEI, dated June 2, 1998, "Guidance on Addressing GSI 168 for License Renewal," Project 690.
4.8.2.6	Memorandum, A. Thadani, NRC NRR, to W. Travers, NRC Operations, dated December 26, 1999, "Closeout of Generic Safety Issue 190, "Fatigue Evaluations of Metal Components for 60 Year Plant Life"

4.8.3 LICENSING DOCUMENTS AND REPORTS

4.8.3.1	NUREG-0717, "Safety Evaluation Report (SER) Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1," NRC, dated February 1981 (includes Supplements).
4.8.3.2	VCSNS Final Safety Analysis Report (FSAR), through Amendment 02-01.
4.8.3.3	NRC Safety Evaluation Report, EGG-HS-6371, "Control of Heavy Loads at Nuclear Power Plants," Virgil C. Summer Nuclear Station, Unit 1, dated May 23, 1985.

4.8.4 REGULATORY DOCUMENTS

4.8.4.1	NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NRC, April 2001.
4.8.4.2	NUREG-1801, "Generic Aging Lessons Learned Report," Volumes 1 and 2, NRC, April 2001.
4.8.4.3	NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."
4.8.4.4	NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," March 6, 1992.
4.8.4.5	NRC Regulatory Guide 1.35, Proposed Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments."

4.8.5 INDUSTRY DOCUMENTS

4.8.5.1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.

4.8.5.2	ASME "American Society of Mechanical Engineers" Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1992 Edition and 1992 Addenda.
4.8.5.3	EPRI Final Report MRP-47, "Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47)" Revision 0, October 2001.
4.8.5.4	ASME "American Society of Mechanical Engineers" Boiler and Pressure Vessel Code Section III Nuclear Power Plant Components, Division I, (1971 Edition through Summer 1972).
4.8.5.5	American National Standards Institute, ANSI B31.1.0, "Power Piping Code," 1967 issue with addenda through 1972.
4.8.5.6	Crane Manufacturers Association of America (CMAA) Specification No. 70 (CMAA 70), "Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes," 1999.
4.8.5.7	ASME "American Society of Mechanical Engineers" Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I, 1974 Edition and Addenda up to Winter 1975.

APPENDIX A - FSAR CHAPTER 18

INTRODUCTION

South Carolina Electric & Gas Company (SCE&G) has prepared an Application for Renewed Operating License of the Virgil C. Summer Nuclear Station (Application). The complete application includes sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by 10 CFR 54.29.

Appendix A of the Application contains the FSAR Supplement for the Virgil C. Summer Nuclear Station.

10 CFR 54.21(d) An FSAR Supplement

The FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by paragraphs (a) and (c) of this section, respectively.

The Application contains the technical information required by 10 CFR 54.21(a) and (c). Appendix B of the Application provides descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Section 4 of the Application contains the evaluations of the time-limited aging analyses for the period of extended operation. Information contained in both of these locations of the Application has been used to prepare the program and activity descriptions that are contained in the attached FSAR Supplement.

FSAR SUPPLEMENT

South Carolina Electric & Gas Company (SCE&G) has prepared an Application for Renewed Operating License of the Virgil C. Summer Nuclear Station (Application). The complete application includes sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by 10 CFR 54.29.

Appendix A of the Application contains the FSAR Supplement for the Virgil C. Summer Nuclear Station required by 10 CFR 54.21(d).

As appropriate, station documents will be revised or established, implemented, and maintained to cover the aging management programs and activities described in Chapter 18.

Insert new FSAR Chapter 18 to read as follows:

18.0 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

18.1 INTRODUCTION

South Carolina Electric & Gas Company prepared an Application for a Renewed Operating License of the Virgil C. Summer Nuclear Station (Application) **[Reference 18.4.1]**. The application, including information provided in additional correspondence, provides sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by 10 CFR 54.29 (Final Safety Evaluation Report - Final SER) **[Reference 18.4.2]**. Pursuant to the requirements of 10 CFR 54.21(d), the FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by 10 CFR 54.21 (a) and (c), respectively.

Table 18-1 provides a summary listing of the aging management programs and activities required for license renewal. Furthermore, Table 18-2 provides a summary listing of the evaluations of time-limited aging analyses (TLAA) required for license renewal. The first column of Table 18-1 and Table 18-2 provides a listing of aging management programs/activities and TLAA evaluations respectively. The second column of each table indicates where the issue is addressed in the Application. The third column of each table identifies where the description of the program/activity or TLAA is located in the Virgil C. Summer Nuclear Station FSAR.

Section 18.2 contains summary descriptions of the aging management programs and activities that are ongoing through the duration of the operating license of the Virgil C. Summer Nuclear Station, as well as any required one-time inspections.

Section 18.3 contains summary descriptions of the evaluations of time-limited aging analyses that are applicable through the duration of the extended operating license of Virgil C. Summer Nuclear Station.

Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Programs/Activities	Application Location	FSAR Location
10 CFR 50 Appendix J General Visual Inspection	B.1.11	18.2.1
10 CFR 50 Appendix J Leak Rate Testing	B.1.12	18.2.2
Above Ground Tank Inspection	B.2.1	18.2.3
Alloy 600 Aging Management Program	B.1.1	18.2.4
ASME Section XI ISI Program - IWF	B.1.13	18.2.5
Battery Rack Inspection	B.1.14	18.2.6
Boric Acid Corrosion Surveillances	B.1.2	18.2.7
Bottom-Mounted Instrumentation Inspection	B.1.3	18.2.8
Buried Piping and Tanks Inspection	B.2.10	18.2.9
Chemistry Program	B.1.4	18.2.10
Containment Coating Monitoring and Maintenance Program	B.1.15	18.2.11
Containment ISI Program – IWE/IWL	B.1.16	18.2.12
Diesel Generator Systems Inspection	B.2.2	18.2.13
Environmental Qualification (EQ) Program	B.3.1	18.2.14
Fire Protection Program (including Mechanical, Fire Barriers and Fire Barrier Penetration Seals, and Fire Doors activities)	B.1.5	18.2.15
Flood Barrier Inspection	B.1.17	18.2.16
Flow-Accelerated Corrosion Monitoring Program	B.1.6	18.2.17
Non-EQ Insulated Cables and Connections Inspection Program	B.2.9	18.2.18

**Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES**

Programs/Activities	Application Location	FSAR Location
In-Service Inspection (ISI) Plan	B.1.7	18.2.19
Inspections for Mechanical Components	B.2.11	18.2.20
Liquid Waste System Inspection	B.2.3	18.2.21
Maintenance Rule Structures Program	B.1.18	18.2.22
Material Handling System Inspection Program	B.1.19	18.2.23
Pressure Door Inspection Program	B.1.[illegible]	18.2.24
Preventive Maintenance Activities – Ventilation Systems Inspections	B.1.26	18.2.25
Reactor Building Cooling Unit Inspection	B.2.5	18.2.26
Reactor Head Closure Studs Program	B.1.8	18.2.27
Reactor Vessel Internals Inspection	B.2.4	18.2.28
Reactor Vessel Surveillance Program	B.1.24	18.2.29
Service Air System Inspection	B.2.6	18.2.30
Service Water Pond Dam Inspection Program	B.1.21	18.2.31
Service Water Structures Survey Monitoring Program	B.1.22	18.2.32
Service Water System Reliability and In-Service Testing Program	B.1.9	18.2.33
Small Bore Class 1 Piping Inspection	B.2.7	18.2.34
Steam Generator Management Program	B.1.10	18.2.35
Tendon Surveillance Program	B.3.3	18.2.36
Thermal Fatigue Management Program	B.3.2	18.2.37
Underwater Inspection Program (SWIS) and (SWPH)	B.1.23	18.2.38

Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Programs/Activities	Application Location	FSAR Location
Waste Gas System Inspection	B.2.8	18.2.39
Heat Exchanger Inspections	B.2.12	18.2.40
Preventive Maintenance Activities - Terry Turbine	B.1.25	18.2.41

Table 18-2:
SUMMARY LISTING OF THE TLAA EVALUATIONS FOR LICENSE RENEWAL

Time Limited Aging Analyses	Application Location	FSAR Location
Crane Load Cycle Limit	4.7.3	18.3.6.1
Environmental Qualification (EQ)	4.4	18.3.3
Metal Fatigue - ASME Section III, Class 1	4.3.1	18.3.2.1
Metal Fatigue - Leak-Before-Break Analyses	4.7.2	18.3.2.2
Metal Fatigue - ASME Section III, Class 2 and 3 Piping Fatigue	4.3.2	18.3.2.3
Reactor Building Liner	4.6.1	18.3.5
Reactor Building Tendon Prestress	4.5	18.3.4
Reactor Coolant Pump Flywheel	4.7.1	18.3.6.3
Reactor Vessel Neutron Embrittlement – Upper-Shelf Energy	4.2.1	18.3.1.1
Reactor Vessel Neutron Embrittlement – Pressurized Thermal Shock	4.2.2	18.3.1.2
Reactor Vessel Neutron Embrittlement – Pressure-Temperature (P-T) Limits	4.2.3	18.3.1.3

Table 18-2:
SUMMARY LISTING OF THE TLAA EVALUATIONS FOR LICENSE RENEWAL

Time Limited Aging Analyses	Application Location	FSAR Location
Service Water Intake Structure Settlement	4.7.4	18.3.6.2

18.2 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The programs and activities described in the subsequent sections are credited for the management of aging under all current licensing basis conditions. Evaluation of the programs and activities provides reasonable assurance that subject systems, structures, and components are capable of performing their intended function(s) under all current licensing basis conditions. Aging management is provided through activities such as continued monitoring and assessment of conditions, trending and/or through control of system/structure parameters to preclude degradation. Under certain circumstances, one-time inspections are performed to ascertain plant conditions and/or confirm that degradation is not occurring.

Herein, the names of the programs, or activities are used only in the context of aging management during the period of extended operation and do not necessarily align with existing formalized VCSNS programs and/or procedures. The program and activity names used in this Application are intended to describe the collection of activities that are necessary to effectively manage aging.

18.2.1 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. The general visual examination satisfies Technical Specification surveillance requirement 4.6.1.6.3. The inspection manages loss of material, cracking of welds, deformed structural attachments, and surface discontinuities associated with the containment liner, deterioration of moisture barriers, and deterioration of the Reactor Building structure.

18.2.2 10 CFR 50 APPENDIX J LEAK RATE TESTING

10 CFR 50 Appendix J Leak Rate Tests are required by Technical Specifications Surveillance Requirement 4.6.1.2. Type A and Type B Leak Rate Tests are performed as described further in **FSAR Section 6.2.6**. The testing program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings and other access openings for detection of degradation of the containment pressure boundary.

18.2.3 ABOVE GROUND TANK INSPECTION

The Above Ground Tank Inspection is a new one-time inspection activity that will determine if aging management is required for the internal surfaces of certain tanks and associated components (including pipe and valves) during the period of extended operation. The Above Ground Tank Inspection will detect and characterize loss of material due to galvanic and general corrosion in locations with exposure to moist air conditions, loss of material due to general corrosion in locations with exposure to treated water in which dissolved oxygen levels are not controlled, and loss of material and/or cracking due to the corrosive effects of alternate wetting and drying of treated or borated water. The Above Ground Tank Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.4 ALLOY 600 AGING MANAGEMENT PROGRAM

The purpose of the Alloy 600 Aging Management Program is to manage primary water stress corrosion cracking (PWSCC) of nickel-based alloy (Alloy 600 and 82/182) sub-components of the reactor vessel, pressurizer, and steam generators that are exposed to borated water to ensure that the pressure boundary function is maintained during the period of extended operation. The Alloy 600 Aging Management Program includes elements of the Boric Acid Corrosion Surveillances and the ASME Section XI System Pressure Test Program which detect the presence of system leakage and the ASME Section XI Inservice Examination Program which specifies the NDE techniques and acceptance criteria applied to the evaluation of identified cracks.

18.2.5 ASME SECTION XI ISI PROGRAM - IWF

The ASME Section XI Subsection IWF Inservice Inspection (ISI) Program manages 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, as well as cracking of high strength anchorage of ASME Class 1 component supports, for the extended period of operation. The ASME Section XI ISI Program - IWF was developed to implement the applicable requirements of 10 CFR 50.55a. The subsection IWF scope of inspection for supports is based on sampling of the total support population. The inspection program includes periodic volumetric, surface, and/or visual examination of component supports for signs of degradation and provides for corrective actions.

18.2.6 BATTERY RACK INSPECTION

The regulatory basis for inspecting battery racks is found in Technical Specifications Surveillance Requirement 3/4.8.2.1.c for the Electrical DC System and a commitment in the Fire Protection Evaluation Report (FPER) **[Reference 18.4.3]** for the Fire Service System. A visual inspection for loss of material due to corrosion is conducted for the Electrical DC System, in accordance with commitments in **FSAR Section 8.3.2.2.2**. A similar examination is conducted for the Fire Service System.

18.2.7 BORIC ACID CORROSION SURVEILLANCES

The purpose of the Boric Acid Corrosion Surveillances is to manage loss of material due to boric acid corrosion of mechanical and structural components constructed of susceptible materials located in the Reactor Building and in specific areas of the Auxiliary, Intermediate, or Fuel Buildings where borated water leakage is possible. The Boric Acid Corrosion Surveillances also manage boric acid intrusion into electrical equipment located in proximity to borated water systems. Elements of the Boric Acid Corrosion Surveillances include the identification of leakage locations, procedures for locating small leaks, and corrective actions to ensure that boric acid corrosion does not lead to degradation of structures and components that could cause loss of intended function.

18.2.8 BOTTOM-MOUNTED INSTRUMENTATION INSPECTION

The purpose of the Bottom-Mounted Instrumentation Inspection is to identify loss of material due to fretting (wear) in the bottom mounted instrumentation (BMI) thimble tubes prior to leakage in order to preclude a breach of the reactor coolant pressure boundary. Suitable inspection techniques are utilized and trended. The frequency of examination is based on wear rate relationships developed from Westinghouse research data.

18.2.9 BURIED PIPING AND TANKS INSPECTION

The purpose of the Buried Piping and Tanks Inspection is to manage loss of material on the external surfaces of buried components. The conditions of coatings and wrappings are determined by visual inspection whenever buried components are excavated, such as for maintenance. Degraded coatings or wrappings are indicative of potential surface corrosion of the external piping or tank surfaces and require further evaluation.

18.2.10 CHEMISTRY PROGRAM

The Chemistry Program controls the water chemistry in plant systems to minimize contaminant concentrations and adds chemicals, such as corrosion inhibitors and biocides, to manage loss of material, cracking, and fouling. The Chemistry Program is based on Electric Power Research Institute (EPRI) guidelines for primary and secondary water chemistry. The Chemistry Program includes specifications for chemical species, limits, sampling and analysis frequencies, and corrective actions for primary, secondary, and auxiliary (borated or treated) water systems, as well as for oil and fuel oil.

18.2.11 CONTAINMENT COATING MONITORING AND MAINTENANCE PROGRAM

The Containment Coating Monitoring and Maintenance Program provides for maintenance of protective coatings inside the containment. Maintenance of protective coatings manages loss of material due to corrosion. Visual inspections and condition assessments of certain coatings inside containment are periodically conducted as part of the containment structural integrity verification, Maintenance Rule monitoring, general maintenance planning, and during recovery from refueling outages. 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 based on plant specific characteristics, industry wide experience and testing history are evaluated in lieu of actual visual inspections. Further discussions relevant to containment coatings is located in **FSAR Appendix 3A** (RG 1.54) and **FSAR Section 6.2.1.6**.

18.2.12 CONTAINMENT ISI PROGRAM - IWE/IWL

10 CFR 50.55a(g)(4) requires a detailed visual examination of the containment system for structural anomalies in accordance with ASME Section XI Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components for Light-Water Cooled Power Plants", and IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants" throughout the service life of nuclear power plants. The inspection program includes periodic volumetric, surface and visual examination of concrete and liner surfaces for signs of degradation and provides for corrective actions.

18.2.13 DIESEL GENERATOR SYSTEMS INSPECTION

The Diesel Generator Systems Inspection is a new one-time inspection activity that will determine if aging management is required for certain carbon steel diesel generator support system components during the period of extended operation. The Diesel Generator Systems Inspection will detect and characterize loss of material due to general corrosion and/or corrosive impacts of alternate wetting and drying in pertinent starting air components. The Diesel Generator Systems Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.14 ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

The NRC has established environmental qualification (EQ) requirements in 10 CFR 50.49 and Appendix A (Criterion 4) to 10 CFR Part 50. EQ component aging limits are not based on condition or performance monitoring; however, such monitoring programs are an acceptable basis for modifying aging limits. Monitoring or inspection of environmental, condition, or component parameters may be used to ensure that the component is within its qualification, or to provide a basis to modify the qualification analyses. The EQ Program quantifies the plant service conditions (i.e., the operating environments) for defined environmental zones, such that the severity of the aging effects (in comparison to other plant locations) can be determined. Further discussion of the EQ Program is contained in **FSAR Section 3.11** and Appendix 3A (RG 1.89).

18.2.15 FIRE PROTECTION PROGRAM

The Fire Protection Program utilizes the concept of defense-in-depth to achieve a high degree of fire safety as discussed in the Fire Protection Evaluation Report (FPER) [**Reference 18.4.3**]. The Fire Protection Program provides administrative requirements for ensuring the operability of equipment required to ensure safe plant shutdown. The program includes visual inspections, system flushing, and performance tests of fire barriers, fire doors, and fire suppression system components. As described in the FPER, the Fire Protection Program includes the requirements identified in Appendix A of APCSB 9.5.1 and 10 CFR 50 Appendix R, Sections III.G, III.J and III.O. Additional description of the portions of the program pertinent to the management of aging is provided below.

18.2.15.1 Mechanical

The Fire Protection Program includes the performance of flow tests and flushes to ensure that blockage of flow will not occur, performance testing of individual components to ensure they maintain their component intended function, and visual inspections to verify sprinkler and associated component condition. Fire suppression system components (e.g. piping, valves, nozzles, sprinkler heads, hydrants) are included within the scope of the mechanical inspections and tests. The normal Fire Service System pressure is monitored to provide further indication of the ability to maintain system function. Flow tests and flushes are conducted on the main distribution loops via hydrant testing. Performance testing is conducted on selected Fire Service System components based on NFPA recommendations. Visual inspections of Fire Service System components (e.g. sprinklers, hydrants, above ground piping) are periodically conducted to identify corrosion on the exterior surface, physical damage or obstructions that might impede performance of the intended functions.

In addition, disassembly/replacement of representative sprinkler heads in branch lines that do not receive flow during periodic testing is to be conducted in accordance with NFPA standards. Ultrasonic testing of a representative sample of these stagnant section of piping will be conducted at 10 year intervals.

A one-time inspection of the Fire Service System will be performed to determine if aging management is required for brass and cast iron components during the period of extended operation. The inspection activity will detect and characterize loss of material due to selective leaching. This inspection will use suitable hardness measurement techniques at the most susceptible (sample) locations.

18.2.15.2 Fire Barriers And Fire Barrier Penetration Seals

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.

18.2.15.3 Fire Doors

Fire rated door inspections are performed. 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.

18.2.16 FLOOD BARRIER INSPECTION

Periodic visual inspections are performed for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals. The Flood Barrier Inspection activity is a subset of the Maintenance Rule Structures Program and the Fire Protection Program. The inspections serve to detect cracking and loss of material prior to loss of component intended function.

18.2.17 FLOW-ACCELERATED CORROSION MONITORING PROGRAM

The purpose of the Flow-Accelerated Corrosion Monitoring Program is to manage loss of material for components located in systems within the scope of license renewal which are susceptible to flow-accelerated corrosion (FAC) (also called erosion-corrosion). This program is intended to mitigate FAC by combining the following elements: NUREG guidelines, predictive analysis, inspections, industry experience, station information gathering and communication, engineering judgement, and long-term mitigative strategies to reduce FAC wear rates.

18.2.18 NON-EQ INSULATED CABLES AND CONNECTIONS INSPECTION PROGRAM

The Non-EQ Insulated Cables and Connections Inspection Program provides for visual inspection of instrument as well as power and control cables as a means to identify age-related degradation due to localized ambient thermally and radiologically induced stress prior to significant loss of insulation resistance. The program will be performed at 10-year intervals, with the initial inspection to be performed prior to the period of extended operation. The program will involve a visual inspection of the accessible cables in selected environmental zones, to determine if the cable jackets show any signs of cracking, embrittlement, discoloration, melting, or any other visible evidence of age-related degradation which may indicate loss of insulation resistance. Guidance from the EPRI "Guideline for the Management of Adverse Localized Equipment Environments," [**Reference 18.4.4**] will be used for these inspections.

18.2.19 IN-SERVICE INSPECTION (ISI) PLAN

The In-Service Inspection (ISI) Plan implements the requirements of 10 CFR 50.55a for Class 1, 2 and 3 components is in accordance with ASME Section XI, Subsections IWB (Class 1), IWC (Class 2), and IWD (Class 3). The program consists of periodic volumetric, surface, and/or visual examination of components for signs of degradation and provides for corrective actions. The examinations are performed to the extent practicable within the limitations of design, geometry, and materials of construction of the component. The period of extended operation for VCSNS will contain the fourth, fifth, and sixth ten-year inservice inspection intervals. The program is addressed further in **FSAR Section 5.7**.

18.2.20 INSPECTIONS FOR MECHANICAL COMPONENTS

The Inspections for Mechanical Components manage loss of material and cracking for mechanical components constructed of susceptible materials and exposed to ambient conditions. The inspections involve a visual examination of the exposed external surfaces of representative mechanical components. The inspections and associated evaluations also address conditions in locations susceptible to external pitting corrosion due to the presence of insulation materials and the potential for condensation to occur.

18.2.21 LIQUID WASTE SYSTEM INSPECTION

The Liquid Waste System Inspection is a new one-time inspection activity that will determine if aging management is required for certain stainless steel pipe, valves and heat exchanger components during the period of extended operation. The Liquid Waste System Inspection will detect and characterize loss of material due to crevice and pitting corrosion, and cracking due to stress corrosion cracking in systems and components containing unmonitored and uncontrolled water. The Liquid Waste System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.22 MAINTENANCE RULE STRUCTURES PROGRAM

The Maintenance Rule Structures Program for the inspections of structures and structural components meets the regulatory requirements of 10 CFR 50.65, the Maintenance Rule. Visual inspections and condition assessments of structures and structural components are conducted in accordance with the requirements of the Maintenance Rule. The structures and structural components are visually inspected via walkdowns from accessible floors, platforms or other permanent vantage points. In cases of inaccessibility, sampling approaches are evaluated. The Maintenance Rule Structures Program includes chemical analysis of raw water (groundwater, Service Water Pond, reservoir, rainwater) per the Maintenance Rule intervals in support of condition assessment.

18.2.23 MATERIAL HANDLING SYSTEM INSPECTION PROGRAM

The Material Handling System Inspection Program manages loss of material for applicable steel rails and girders. The Material Handling System Inspection Program has been in effect for many years at VCSNS and includes Nuclear Safety Related and Quality Related (seismically restrained) material handling system components. Material handling systems steel support structures (rails, runways, monorails, girders, jib cranes, seismic restraints, and associated connections) are inspected in accordance with guidance provided by ANSI standards. Inspections are implemented in the course of routine maintenance.

18.2.24 PRESSURE DOOR INSPECTION PROGRAM

The Pressure Door Inspection Program provides examination guidelines for periodic inspections of pressure doors. VCSNS pressure doors are Nuclear Safety Related or Quality Related. Most Nuclear Safety Related pressure doors are also fire doors. Pressure door inspection attributes include freedom of movement, function (closed during normal plant operation), structural deterioration, and loss of door/door hardware material. Pressure doors are required to be operable in Plant Operating Modes 1, 2, 3, and 4. The surveillance requirements include monitoring of door position and visual inspection that the door is closed and not impaired.

.2.25 PREVENTIVE MAINTENANCE ACTIVITIES - VENTILATION SYSTEM INSPECTIONS

The Preventive Maintenance Activities - Ventilation Systems Inspections manage loss of material and fouling in susceptible components. Susceptible components include those components in the Air Handling, Local Ventilation and Component Cooling Systems that are exposed internally to moist air. Routine maintenance inspections are conducted which include detection of age-related degradation.

18.2.26 REACTOR BUILDING COOLING UNIT INSPECTION

The Reactor Building Cooling Unit Inspection is a new one-time inspection activity that will determine if aging management is required for reactor building cooling unit drain lines during the period of extended operation. The Reactor Building Cooling Unit Inspection will detect and characterize loss of material or cracking resulting from exposure to an unmonitored and uncontrolled (alternately wetted/dried) water environment. The Reactor Building Cooling Unit Inspection will use volumetric and/or visual examination techniques at the most susceptible (sample) locations in the reactor building cooling unit drain lines.

18.2.27 REACTOR HEAD CLOSURE STUDS PROGRAM

The purpose of the Reactor Head Closure Studs Program is to manage loss of mechanical closure integrity due to stress relaxation, stress corrosion cracking, and wear for the alloy steel components of the reactor vessel closure stud assembly. The In-Service Inspection (ISI) Plan portion of the Reactor Head Closure Studs Program detects cracking and loss of material through the use of surface examination (magnetic particle or liquid penetrant) and/or ultrasonic examination. Reactor vessel studs may be examined in place (under tension), when the connection is disassembled, or when the studs are removed. Further discussions of the inspection and protection of reactor vessel closure stud assemblies is contained in **FSAR Section 5.4**.

18.2.28 REACTOR VESSEL INTERNALS INSPECTION

The Reactor Vessel Internals Inspection supplements the In-Service Inspection (ISI) Plan to assess the condition of reactor vessel internals in order to ensure that the intended functions are maintained during the period of extended operation. The inspection provides examination techniques and engineering evaluations to address the aging effects listed below:

- Changes in dimensions due to irradiation creep and void swelling
- Cracking due to irradiation-assisted stress corrosion cracking
- Cracking due to primary water stress corrosion cracking (PWSCC) for nickel-based materials
- Loss of material due to wear
- Loss of preload due to stress relaxation
- Reduction of fracture toughness due to irradiation embrittlement and void swelling

For those components that are accessible or can be rendered accessible by the removal of the core and other internals for examination, a visual inspection is performed to detect the presence and extent of cracking and loss of material. For bolts and for inaccessible components, a volumetric inspection is performed to detect the presence and extent of changes in dimensions, cracking, loss of preload, and reduction of fracture toughness. With respect to changes in dimensions due to void swelling, industry activities (including WOG and EPRI) are under way to better characterize the effect and, if necessary, to develop and qualify methods for detection and management. These activities will be monitored by VCSNS and implemented, as applicable.

18.2.29 REACTOR VESSEL SURVEILLANCE PROGRAM

The purpose of the Reactor Vessel Surveillance Program is to manage reduction of fracture toughness due to irradiation embrittlement of reactor vessel beltline materials to assure that the pressure boundary function is maintained for the period of extended operation. The program includes an evaluation of radiation damage based on testing of Charpy V-notch and tensile specimens. The Reactor Vessel Surveillance Program conforms to 10 CFR 50, Appendix H. Further discussions of this program are contained in **FSAR Section 5.4.3**.

Additionally, a one-time demonstration is a necessary part of this program to ensure that materials in the upper shell/nozzle course of the reactor vessel do not become limiting during the period of extended operation, with respect to radiation damage.

18.2.30 SERVICE AIR SYSTEM INSPECTION

The Service Air System Inspection is a new one-time inspection activity that will determine if aging management is required for certain carbon steel pipe and valves during the period of extended operation. The Service Air System Inspection will detect and characterize loss of material due to general corrosion on internal surfaces of subject components resulting from exposure to moist air. The Service Air System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.31 SERVICE WATER POND DAM INSPECTION PROGRAM

The purpose of the Service Water Dam Inspection Program is to assess the condition of the Service Water Pond Dams and the West Embankment with respect to loss of material (erosion), alignment (movement), surface cracking and seepage that could result in loss of component intended function. Additionally, the submerged slope stability of the West Embankment in the vicinity of the Intake Structure is monitored as specified by VCSNS Operating License Number NPF-12, Condition 2.C.5. SCE&G conducts annual walkdowns of the Service Water Pond Dams scheduled in the course of routine maintenance.

18.2.32 SERVICE WATER STRUCTURES SURVEY MONITORING PROGRAM

Survey monitoring is required for structures that are supported by earthen fill material and which have exhibited the potential for settlement. Settlement is not considered to be adverse unless it imposes stresses on a structure that could exceed the design values. Initial settlement of the Service Water Pump House and Service Water Intake Structure was much more than the original pre-construction estimates. As a result, survey monitoring of the Service Water Pumphouse, Service Water Intake Structure, Electrical Duct Banks, and Service Water Intake Line "A" is conducted 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 and evaluate any differential in vertical and horizontal displacement in order to identify settlement issues prior to their resulting in significant degradation or loss of function.

18.2.33 SERVICE WATER SYSTEM RELIABILITY AND IN-SERVICE TESTING PROGRAM

The purpose of the Service Water System Reliability and In Service Testing Program is to manage cracking, fouling, and loss of material for susceptible materials located in systems that contain raw water from the Service Water Pond. The Service Water System Reliability and In Service Testing Program is intended to detect the presence of and assess the extent of cracking, fouling and loss of material. The program also serves to mitigate aging effects through the use of chemical additives in order to minimize fouling. Visual inspections of Service Water System piping and components are conducted on a periodic basis to monitor the extent of cracking, fouling, and loss of material. The heat transfer capabilities of heat exchangers serviced by the Service Water System are evaluated to detect the presence of fouling.

18.2.34 SMALL BORE CLASS 1 PIPING INSPECTION

The Small Bore Class 1 Piping Inspection is a new one-time inspection activity that will determine if aging management is required for cracking due to flaw growth and stress corrosion cracking for Reactor Coolant System stainless steel piping and fittings less than four inches NPS. The Small Bore Class 1 Piping Inspection will serve to increase confidence in the current condition of small bore Reactor Coolant System piping which does not receive a volumetric examination per the ASME Code. The Small Bore Class 1 Piping Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.35 STEAM GENERATOR MANAGEMENT PROGRAM

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 program implements the requirements of Technical Specification 4.4.5. The program follows the recommendations provided by NEI and EPRI guidelines.

18.2.36 TENDON SURVEILLANCE PROGRAM

The Tendon Surveillance Program meets the requirements of the 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 the rate of tendon force loss is within predicted limits and that a minimum required tendon force level exists in the Reactor Building. Degradation of the Reactor Building post tensioning system is detected by periodic inspections of randomly selected tendons. Further discussion pertinent to tendon stresses is contained in **FSAR Appendix 3A** (1.35).

18.2.37 THERMAL FATIGUE MANAGEMENT PROGRAM

As defined for license renewal, the Thermal Fatigue Management Program seeks to preclude cracking due to low-cycle thermal fatigue by managing the thermal fatigue basis. Management of the fatigue basis is accomplished by continually showing that the severity and number of occurrences of the actual transients are enveloped (bounded) by the severity and number of occurrences of the analyzed transients. The program documents and evaluates plant operational transients/cycles using a specially designed computer program. The evaluation process includes comparison of the number of thermal cycles incurred against the design transient limit and/or calculation of cumulative usage factors (CUF) and verification that adequate margin exists. The Thermal Fatigue Management Program will incorporate applicable industry guidance once it is finalized to account for environmental effects of the reactor coolant fluid.

18.2.38 UNDERWATER INSPECTION PROGRAM (SWIS AND SWPH)

Underwater inspections of the Service Water Intake Structure are conducted and the conditions assessed in accordance with the requirements of VCSNS Operating License NPF-12, Condition 2.C.5.d. The purpose of the inspection is to monitor the condition of existing cracks in the Service Water Intake Structure that originated during construction due to greater than expected settlement of the structure. Underwater inspections of the Service Water Intake Structure and the Service Water Pump House also serve to monitor corrosion and fouling within the Service Water System.

18.2.39 WASTE GAS SYSTEM INSPECTION

The Waste Gas System Inspection is a new one-time inspection activity that will determine if aging management is required for certain stainless steel components of the Gaseous Waste Processing System during the period of extended operation. The Waste Gas System Inspection will detect and characterize loss of material due to crevice and pitting corrosion in portions of the system exposed to unmonitored and uncontrolled treated water, and cracking due to stress corrosion cracking in portions of the system containing gas. The Waste Gas System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.40 HEAT EXCHANGER INSPECTIONS

The Heat Exchanger Inspections are a new one-time inspection activity that will determine if aging management is required for certain malleable heat exchanger components during the period of extended operation. The Heat Exchanger Inspections will detect and characterize loss of material due to selective leaching and erosion-corrosion, as well as particulate fouling. The Heat Exchanger Inspections will use a combination of volumetric and visual examination and hardness measurement techniques at the most susceptible (sample) locations.

18.2.41 PREVENTIVE MAINTENANCE ACTIVITIES - TERRY TURBINE

The purpose of the Preventive Maintenance Activities - Terry Turbine is to manage loss of material in carbon steel due to crevice, general, or pitting corrosion of the turbine casing and associated components that are normally exposed to ambient conditions with periodic exposure to steam allowing moisture to accumulate. The Preventive Maintenance Activities - Terry Turbine is a condition monitoring program composed of controlled plant procedures. Routine maintenance inspections are conducted which include detection of age-related degradation and initiation of corrective actions as necessary.

18.3 TIME-LIMITED AGING ANALYSES (TLAA) EVALUATIONS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) be provided for the period of extended operation. The following TLAAs have been identified and evaluated to meet this requirement.

18.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. Analyses have been performed that address the following:

- Upper-Shelf Energy
- Pressurized Thermal Shock
- Pressure-Temperature (P-T) Limits

18.3.1.1 Upper-Shelf Energy

The requirements on reactor vessel Charpy Upper-Shelf Energy are included in 10 CFR 50, Appendix G. The Charpy Upper-Shelf Energy must be maintained at no less than 50 ft.-lb. throughout the life of the reactor vessel.

Charpy Upper-Shelf Energy values have been calculated to the end of current license (32 EFPY) to be 67.5 ft.-lb. for the limiting component of the beltline region materials. Additional analyses are required to calculate Charpy Upper-Shelf Energy for the end of the period of extended operation. The remaining capsules must incur additional exposure to neutron fluence in order to provide data that correlates to estimated fluence on the vessel at the end of the period of extended operation. Following adequate capsule exposure, a capsule will be withdrawn and analyzed. The Charpy Upper-Shelf Energy will be recalculated for additional fast neutron fluence corresponding to the end of the extended operating period. Therefore, the Charpy Upper-Shelf Energy analyses will be projected for the end of the period of extended operation.

18.3.1.2 Pressurized Thermal Shock

The requirements of 10 CFR 50.61 provide for protection against pressurized thermal shock events for pressurized water reactors. The screening criteria established by 10 CFR 50.61 are 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials. According to 10 CFR 50.61, if the calculated reference temperature (RT_{PTS}) for the limiting beltline materials is less than the specified screening criteria, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients. The regulations require updating the pressurized thermal shock assessment upon a request for a change in the expiration date of the facility operating license.

The RT_{PTS} values for VCSNS have been calculated for the end of the current 40-year license term. The RT_{PTS} values require revision in order to project the values to the end of the period of extended operation. This cannot be accomplished until the surveillance capsules are exposed to sufficient neutron fluence to provide data corresponding to the estimated fluence on the vessel wall at the end of 60 years. The analyses will be revised to project the RT_{PTS} to the end of the period of extended operation using the methods provided in 10 CFR 50.61.

18.3.1.3 Pressure-Temperature (P-T) Limits

Appendix G to 10 CFR 50 requires that heatup and cooldown of the reactor pressure vessel shall be accomplished within established pressure-temperature limits. The pressure-temperature limits are established by calculations that utilize the materials and fluence data obtained through the site reactor surveillance capsule program. Normally, the pressure-temperature limits are calculated for several years into the future and remain valid for an established period of time not to exceed the current operating license term.

Pressure-temperature limit curves have been calculated for 32 EFPY that correspond to the end of the current license term but not the period of extended operation. The pressure-temperature limit curves will be recalculated following the removal of one of the remaining surveillance capsules from the vessel. The surveillance capsule will be removed when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the period of extended operation. The Technical Specifications will be updated as required by 10 CFR 50. Therefore, the pressure-temperature limit analyses will be projected for the period of extended operation.

18.3.2 METAL FATIGUE

The thermal fatigue analyses of the station's mechanical components have been identified as time-limited aging analyses.

18.3.2.1 ASME Section III, Class 1

The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that the initiation of fatigue cracks is precluded.

Experience has shown that the transients used to analyze the ASME III requirements are often very conservative. The magnitude and frequency of the design transients are more severe than those occurring during plant operation. The magnitude and number of actual transients are monitored. This monitoring assures that the existing frequency and magnitude of transients are conservative and bounding for the period of extended operation, and that the existing ASME III equipment will perform its intended functions for the period of extended operation. A program for thermal transient cycle counting and analysis is in place and provides reasonable assurance that the actual transients are smaller in magnitude and within the number of the transients used in the design.

18.3.2.2 Leak-Before-Break Analyses

Leak-before-break analyses evaluate postulated flaw growth in the primary loop piping of the Reactor Coolant System. These analyses consider the thermal aging of the cast austenitic stainless steel material of the piping as well as the fatigue transients that drive the flaw growth over the operating life of the plant.

The leak-before-break analyses are currently valid for 40 years. The analyses require revision in order to demonstrate that the design is adequate for the extended period of operation.

18.3.2.3 ASME Section III, Class 2 and 3 Piping Fatigue

Piping systems, designed in accordance with ASME Section III, Class 2 and 3 or ANSI B31.1, utilize allowable stress values based on a stress reduction factor to account for thermal cycles during normal operation. Adequate margin is available to account for 60 years of plant operation in the current analyses for the majority of the plant systems reviewed. Restrictions on sampling activities, or reanalysis, is required on the "B" Reactor Coolant System loop sampling line in order to account for 60 years of plant operation.

Metal fatigue for the ASME Section III, Class 2 and 3 and ANSI B31.1 piping systems has been determined to be valid for the period of extended operation.

18.3.3 ENVIRONMENTAL QUALIFICATION (EQ)

The qualification analyses for some electrical equipment included in the Environmental Qualification (EQ) Program have been identified as time-limited aging analyses for license renewal. The qualification analyses for electrical equipment with a 40-year or greater qualified life have been determined to be time-limited aging analyses.

Equipment included in the VCSNS EQ Program will be evaluated to determine if existing environmental qualification analyses can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as for equipment currently qualified at VCSNS for 40 years or less. When aging analyses cannot justify a qualified life to the end of the period of extended operation then the components or parts will be replaced prior to exceeding their qualified lives in accordance with the EQ Program.

The existing EQ process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation.

18.3.4 REACTOR BUILDING TENDON PRESTRESS

The Reactor Building was prestressed in order to have low-strain linear response at design loads and thus assure integrity of the liner. The exterior wall is post-tensioned in both vertical and hoop directions. On the dome a three-way post-tensioning system is employed. The pre-stress of the containment tendons decreases over the life of the plant due to elastic deformation, creep, anchorage seating losses, tendon wire friction, stress relaxation and corrosion. Periodic inspections include examination of selected tendon parameters and provide data for prestress analyses. Tendon prestress analyses are used to determine if addition retensioning is required before the next scheduled inspection based on the state of the te 'on stress. Therefore reactor building tendon prestress is a time-limited aging analysis.

T. existing Tendon Surveillance Program will ensure that the Reactor Building tendons are analyzed for the period of extended operation.

18.3.5 REACTOR BUILDING LINER

The Reactor Building is lined on the inside face with a steel plate that provides an essentially leak-tight barrier. The liner is designed to remain within strain limits associated with serviceability in accordance with the ASME B&PV Code for normal operation.

The reactor building liner calculations evaluate liner fatigue for a 40 year period and conclude that the liner meets the criteria of ASME NB 3222.4 (d) for the suitability for cyclic condition and no fatigue analysis is required. The reactor building liner analyses has been revised for the period of extended operation.

18.3.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

18.3.6.1 Crane Load Cycle Limit

The crane load cycle limit was identified as a time-limited aging analysis for the cranes within the scope of license renewal. The cranes within the scope of license renewal are listed below.

- Reactor Building Polar Crane
- Fuel Handling Machine (Spent Fuel Pit Bridge and Hoist)
- Refueling Machine (Reactor Cavity Manipulator Crane)
- Spent Fuel Cask Handling Crane

The cranes listed above are classified as Class "A" cranes by the Crane Manufacturers Association of America Specification No. 70 (CMAA 70) which specifies a design limit for the number of load cycles for the life of a crane. The load cycles for these cranes have been evaluated for the period of extended operation. For each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed quantity of cycles. All the cranes in the scope of license renewal will continue to perform their intended function through the period of extended operation.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation.

18.3.6.2 Service Water Intake Structure Settlement

The Service Water Intake Structure 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 not covered with soil is submerged within the Service Water Pond. The function of the Service Water Intake Structure is to draw water from the Service Water Pond into the Service Water Pump House. Excessive non-uniform settlement of the intake structure occurred during construction resulting in cracking of the structure. The settlement of the structure was analyzed based on a plant design life of 40 years. Therefore, Service Water Intake Structure settlement is a time-limited aging analysis for VCSNS.

The Service Water Intake Structure settlement calculation has been revised to evaluate the settlement of the structure for the period of extended operation.

18.3.6.3 Reactor Coolant Pump Flywheel

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway from stresses due to starting the motor. An analysis has been performed to estimate the magnitude of fatigue crack growth during the plant life. The analysis assumes 6,000 cycles of pump starts and stops for a 60-year plant life.

The analysis associated with the reactor coolant pump flywheel has been evaluated and determined to remain valid for the period of extended operation.

18.4 REFERENCES

18.4.1	LRA (later)
18.4.2	SER (later)
18.4.3	VCSNS Fire Protection Evaluation Report (FPER), Amendment 02-01.
18.4.4	EPRI Report TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments," June 1999

APPENDIX B - AGING MANAGEMENT PROGRAMS AND ACTIVITIES

INTRODUCTION

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

The NRC and the industry identified ten (10) program elements that would be useful in describing an aging management program and then demonstrating its effectiveness. These program elements are described in Appendix A.1, Section A.1.2.3 of NUREG-1800 **[Reference B-1]**. NUREG-1801, Generic Aging Lessons Learned (GALL) Report **[Reference B-2]**, applies these program elements to evaluate the adequacy of generic aging management programs in managing certain aging effects.

VCSNS applied the program elements in the review of VCSNS programs and activities to demonstrate that the effects of aging will be adequately managed in accordance with the License Renewal Rule. There are two types of VCSNS programs that are evaluated in this manner. The first is a VCSNS program that is evaluated against a program in NUREG-1801. The second is a VCSNS plant-specific program.

For VCSNS aging management programs being evaluated against a program in NUREG-1801, the criteria or activities delineated in each of the 10 elements are reviewed. A conclusion is reached concerning consistency for the VCSNS program with each NUREG-1801 recommended program elements and a demonstration of overall program effectiveness is made. Any program enhancements that are required are documented. 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. Finally, an overall determination is made as to consistency with the program description in NUREG-1801.

For plant-specific aging management programs, an evaluation is performed to document how VCSNS meets the 10 generic program elements in NUREG-1800. A determination of overall program effectiveness and any required program enhancements are identified.

The VCSNS Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, "Standard Review Plan for Review of License Renewal." A description of the VCSNS Quality Assurance Program is contained in **FSAR Section 17.2**. The Quality Assurance Program addresses three of the aging management program elements: 1) Corrective Action, 2) Confirmation Process (which is an integral part of the Corrective Action), and 3) Administrative Controls. The Quality Assurance Program applies these program elements via existing corrective action and document control programs. VCSNS will employ the corrective action and document control programs to address the program elements of corrective action, confirmation process, and administrative (document) controls for both safety-related and non-safety-related structures and components that require aging management during the period of extended operation.

The VCSNS aging management programs described herein are credited for managing the effects of aging. These programs include existing programs, existing programs that have been enhanced to deal with specific aging effects, and new programs not currently defined in VCSNS administrative controls. The programs provide reasonable assurance that the effects of aging will be adequately managed so that the structures and components subject to aging management will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The demonstrations, along with the program and activity descriptions, meet the requirements of 10 CFR 54.21(a)(3). Along with the technical information contained in the body of this application, this appendix is intended to allow the NRC to make the finding required by 10 CFR 54.29(a)(1).

The results of VCSNS aging management program evaluations are documented in the following sections. Each aging management program presented in this Appendix is characterized as one of the following:

> **Existing Aging Management Program (Section B.1.0)**: A current program or activity that will continue to be implemented during the extended period of operation to manage aging. Any required enhancements to the program or activity will be implemented prior to the period of extended operation.
>
> **New Aging Management Program (Section B.2.0)**: A program or activity that does not currently exist, which will manage aging during the extended period of operation.
>
> **TLAA Support Program (Section B.3.0)**: A program or activity that supports the basis for a time-limited aging analysis during the period of extended operation.

Table B-1 presents the correlation between the programs evaluated in NUREG-1801 and the VCSNS programs credited with aging management.

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
Chapter X			
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Thermal Fatigue Management Program	**B.3.2**
X.E1	Environmental Qualification (EQ) of Electric Components	Environmental Qualification (EQ) Program	**B.3.1**
X.S1	Concrete Containment Tendon Prestress	Tendon Surveillance Program	**B.3.3**
Chapter XI			
XI.M1	ASME Section XI Inservice Inspection, Subsection IWB, IWC, IWD	In-Service Inspection (ISI) Plan	**B.1.7**
XI.M2	Water Chemistry	Chemistry Program	**B.1.4**
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs Program	**B.1.8**
XI.M4	BWR Vessel ID Attachment Welds	Not applicable, VCSNS is a PWR.	N/A
XI.M5	BWR Feedwater Nozzle	Not applicable, VCSNS is a PWR.	N/A
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable, VCSNS is a PWR.	N/A
XI.M7	BWR Stress Corrosion Cracking	Not applicable, VCSNS is a PWR.	N/A
XI.M8	BWR Penetrations	Not applicable, VCSNS is a PWR.	N/A
XI.M9	BWR Vessel Internals	Not applicable, VCSNS is a PWR.	N/A
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Surveillances	**B.1.2**

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M11	Nickel-Alloy Nozzles and Pene-trations	Alloy 600 Aging Management Program	**B.1.1**
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging manage-ment.	N/A
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging manage-ment.	N/A
XI.M14	Loose Part Monitoring	Not credited for aging manage-ment.	N/A
XI.M15	Neutron Noise Monitoring	Not credited for aging manage-ment.	N/A
XI.M16	PWR Vessel Internals	Reactor Vessel Internals Inspection	**B.2.4**
XI.M17	Flow-Accelerated Corrosion	Flow - Accelerated Corrosion Monitoring Program	**B.1.6**
XI.M18	Bolting Integrity	Not credited for aging manage-ment.	N/A
XI.M19	Steam Generator Tube Integ-rity	Steam Generator Manage-ment Program	**B.1.10**
XI.M20	Open-Cycle Cooling Water System	Service Water System Reliabil-ity and In-Service Testing Pro-gram	**B.1.9**
XI.M21	Closed-Cycle Cooling Water System	Not credited for aging manage-ment.	N/A
XI.M22	Boraflex Monitoring	Not credited for aging manage-ment.	N/A

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Material Handling System Inspection Program	**B.1.19**
XI.M24	Compressed Air Monitoring	Not credited for aging management.	N/A
XI.M25	BWR Reactor Water Cleanup System	Not applicable, VCSNS is a PWR.	N/A
XI.M26	Fire Protection	Fire Protection Program	**B.1.5**
XI.M27	Fire Water System	Fire Protection Program	**B.1.5**
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management.	N/A
XI.M29	Above ground Carbon Steel Tanks	Not credited for aging management.	N/A
XI.M30	Fuel Oil Chemistry	Chemistry Program	**B.1.4**
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program	**B.1.24**
XI.M32	One-Time Inspection	Above Ground Tank Inspection Diesel Generator Systems Inspection Liquid Waste System Inspection Reactor Building Cooling Unit Inspection Service Air System Inspection Small Bore Class 1 Piping Inspection Waste Gas System Inspection Heat Exchanger Inspections	**B.2.1** **B.2.2** **B.2.3** **B.2.5** **B.2.6** **B.2.7** **B.2.8** **B.2.12**

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M33	Selective Leaching of Materials	Fire Protection Program Heat Exchanger Inspections	**B.1.5** **B.2.12**
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection	**B.2.10**
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Non-EQ Insulated Cables and Connections Inspection Program	**B.2.9**
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Non-EQ Insulated Cables and Connections Inspection Program	**B.2.9**
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Not credited with aging management.	N/A
XI.S1	ASME Section XI, Subsection IWE	Containment ISI Program - IWE/IWL	**B.1.16**
XI.S2	ASME Section XI, Subsection IWL	Containment ISI Program - IWE/IWL	**B.1.16**
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI ISI Program – IWF	**B.1.13**
XI.S4	10 CFR 50, Appendix J	10 CFR 50 Appendix J General Visual Inspection 10 CFR 50 Appendix J Leak Rate Testing	**B.1.11** **B.1.12**
XI.S5	Masonry Wall Program	Maintenance Rule Structures Program	**B.1.18**

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.S6	Structures Monitoring Program	Maintenance Rule Structures Program	**B.1.18**
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Service Water Pond Dam Inspection Program	**B.1.21**
XI.S8	Protective Coating Monitoring and Maintenance Program	Containment Coating Monitoring and Maintenance Program	**B.1.15**
N/A	None	Bottom-Mounted Instrumentation Inspection	**B.1.3**
N/A	None	Battery Rack Inspection	**B.1.14**
N/A	None	Flood Barrier Inspection	**B.1.17**
N/A	None	Pressure Door Inspection Program	**B.1.20**
N/A	None	Service Water Structures Survey Monitoring Program	**B.1.22**
N/A	None	Underwater Inspection Program (SWIS and SWPH)	**B.1.23**
N/A	None	Inspections for Mechanical Components	**B.2.11**
N/A	None	Preventive Maintenance Activities - Ventilation Systems Inspections	**B.1.26**
N/A	None	Preventive Maintenance Activities - Terry Turbine	**B.1.25**

B.1.0 EXISTING , NG MANAGEMENT ACTIVITIES

B.1.1 ALLOY 600 AGING MANAGEMENT PROGRAM

The Alloy 600 Aging Management Program is consistent with XI.M11 *Nickel - Alloy Nozzles and Penetrations*, as identified in NUREG-1801, the enhancements specified in the following table, and with the following clarification:

- Detection of Aging Effects: The Alloy 600 Aging Management Program will not rely on an enhanced leakage detection system for detection of small leaks caused by primary water stress corrosion cracking (PWSCC) during plant operation as suggested by XI.M11. Industry operating experience indicates that PWSCC cracks can be detected by means of inspecting for signs of boric acid leakage during outages and by monitoring primary coolant leakage per Technical Specifications during plant operation prior to the structural integrity of the pressure boundary being compromised.

The following enhancements will be incorporated into the Alloy 600 Aging Management Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancements
XI.M11 Nickel - Alloy Nozzles and Penetrations	3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Changes indicated by emerging regulatory requirements and developed by the industry groups will be implemented for the applicable Attributes.

B.1.1.1 Operating Experience

Recent industry inspection experience documented in NRC information notices confirms that Alloy 600 PWSCC cracks may initiate and grow through-wall in vessel head penetrations with susceptible materials. NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, addressed the occurrence of circumferential cracking of reactor pressure vessel head penetration (VHP) nozzles. The SCE&G response to Bulletin 2001-01 **[Reference B-3]** concluded that VCSNS falls into the NRC category of plants considered to have low susceptibility to PWSCC of the reactor pressure vessel (RPV) top head nozzles. The August 2001 VCSNS response stated that VCSNS performed VT-3 inspections of the interior surface of the reactor vessel head in April 1999 and found no recordable indications. NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, addressed further experience with PWSCC at Davis-Besse in February of 2002. The SCE&G 15-day response to Bulletin 2002-01 **[Reference B-4]** documented the types of inspections that are performed for primary coolant leakage. SCE&G stated that based on reviews of maintenance and repair history at VCSNS that there is no boric acid build up under the insulation on the RPV head surface. Inspections of the RPV head surfaces and VHP nozzles during the April 2002 refueling outage revealed no boric acid build up other than the trace amounts of boric acid residue left behind from a previously repaired instrument penetration (conoseal). 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.

The NRC closure letter **[Reference B-14]** for SCE&G response to Generic Letter 97-01, Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations, requested that three issues be addressed in an application for license renewal of VCSNS. (1) As stated above, VCSNS falls into the NRC category of plants considered to have low susceptibility to PWSCC of the vessel head penetrations. (2) The vessel head penetrations are included within the scope of the boric acid corrosion inspection program, as confirmed by the SCE&G 15-day response to NRC Bulletin 2002-01. (3) The results of inspections completed on the vessel head penetrations are provided above, and is documented in the SCE&G response to NRC Bulletin 2001-01.

VCSNS discovered a crack on the 'A' hot leg nozzle at the beginning of Refuel Outage 12 (October 2000), when boric acid was found on the floor of the containment building. The crack in the nozzle was located in the weld between the RCS piping and the vessel nozzle, on the nozzle side of the weld. The RCS piping is SA-376, type 304 stainless steel. The vessel nozzle is SA-508 material clad with austenitic stainless steel. The weld used Inconel 182 butter (safe end) between the weld piece and the vessel nozzle. 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.

The investigation into the crack concluded that the cause was indirectly attributed to PWSCC. The weld at the nozzle was determined to be subjected to high tensile stresses as a result of extensive weld repairs performed during the original construction. A number of smaller PWSCC cracks were subsequently identified when the weld was cut out and destructively tested. A spool piece was used to replace the affected weld and was installed utilizing Inconel 52 and 152 weld materials, in effect removing the susceptible material. The welding was performed in a manner which minimized residual stresses. Further inspections of the other RCS nozzle safe end-to-pipe welds detected minor indications of cracking. An inspection of the B and C hot leg nozzles was performed in the April 2002 refueling outage. Results were provided in a letter from Stephen A. Byrne to the Document Control Desk (TAC No. MB3839) dated May 4, 2002. Future trending of these indications will be performed to assess the need for any further repair activities.

B.1.1.2 Conclusion

The Alloy 600 Aging Management Program has been demonstrated to be capable of detecting and managing cracking due to PWSCC prior to loss of component intended function based on indications of leakage. The Alloy 600 Aging Management 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 operations.

B.1.2 BORIC ACID CORROSION SURVEILLANCES

The Boric Acid Corrosion Surveillances are consistent with XI.M10, *Boric Acid Corrosion*, as identified in NUREG-1801. These surveillances additionally address boric acid corrosion of electrical connector contacts which may be exposed to borated water leakage mentioned in Chapter VI, Item A.2.1 of NUREG-1801.

B.1.2.1 Operating Experience

The Boric Acid Corrosion Surveillances were originally implemented as a result of NRC Generic Letter 88-05. In October 1999, a series of discussions involving plant personnel were initiated that resulted in procedure changes to coordinate inspection activities for boric acid leakage inside containment. The revised procedures coordinated the Generic Letter 88-05, health physics general area and ASME Section XI inspections.

The Boric Acid Corrosion Surveillances have been successful in managing 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 surveillances, 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 surveillances:

While performing visual inspections of the Reactor Building during RF-12 in October 2000, a significant quantity of boric acid deposits was discovered on the floor coming from the boot of the loop "A" RCS hot leg penetration at the bio-shield wall. Investigation revealed that the deposits originated from the leaking of reactor coolant at the welded joint between the reactor vessel nozzle and the loop "A" hot leg reactor coolant pipe.

As a result of this incident, the Boric Acid Corrosion Surveillances have been 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 task sheets for the procedures credited for the Boric Acid Corrosion Surveillances were reviewed for the results of the past five years. 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 or components in the area of any leak.

B.1.2.2 Conclusion

The Boric Acid Corrosion Surveillances have been demonstrated to be capable of identifying leaks from borated water systems, and subsequently managing the effects of boric acid corrosion. The Boric Acid Corrosion Surveillances 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.

B.1.3 BOTTOM-MOUNTED INSTRUMENTATION INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The purpose of the Bottom Mounted Instrumentation Inspection is to identify loss of material due to fretting in the bottom mounted instrumentation (BMI) thimble tubes prior to leakage. The thimble tubes are part of the Reactor Coolant pressure boundary. The Bottom Mounted Instrumentation (BMI) Inspection is a condition monitoring program.

(1) **Scope** - The Bottom Mounted Instrumentation Inspection is applicable to all BMI thimble tubes installed in the reactor vessel.

(2) **Preventive Actions** - No actions are taken as part of the Bottom Mounted Instrumentation Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The Bottom Mounted Instrumentation Inspection monitors tube wall degradation (loss of material due to fretting) of the BMI thimble tubes. Failure of the thimble tubes would result in a breach of the Reactor Coolant pressure boundary.

(4) **Detection of Aging Effects** - In accordance with information provided in Monitoring and Trending below, the Bottom Mounted Instrumentation Inspection will detect loss of material due to fretting prior to loss of component intended function.

(5) **Monitoring and Trending** - Inspection of the BMI thimble tubes is performed using eddy current testing (ECT). 100% of the thimble tubes are inspected. The frequency of examination is based on an analysis of the data obtained using wear rate relationships predicted based on Westinghouse research. The ECT results are trended, wear rates are calculated, and inspections are planned prior to the refueling outage in which thimble tube wear is predicted to exceed the acceptance criteria. This ensures that the thimble tubes continue to perform their pressure boundary intended function.

(6) **Acceptance Criteria** - The acceptance criteria for the BMI thimble tubes is in the form of an "earliest projected date". Using a wear rate formula, a calculation is performed to determine the earliest projected date for which wear on each BMI thimble tube wear exceed 75% loss of initial wall thickness.

(7) **Corrective Actions** - Thimble tubes must be capped or repositioned if projected through wall wear will exceed 75% prior to the next scheduled ECT. If measured or projected thimble tube wear exceeds 80%, then the thimble tube must be capped or replaced per the acceptance criteria. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a

disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** – The Bottom Mounted Instrumentation Inspection is implemented in accordance with station procedures and work processes.

B.1.3.1 Operating Experience

Operating Experience - Flux thimble wear was first identified as an issue when three flux thimbles developed through wall leakage in a three month period at the Salem plant in 1981. Since that time, numerous plants have detected thimble wear in varying degrees. Westinghouse has determined the cause of this wear to be flow induced vibration of the flux thimble inside of the reactor vessel lower internals support column. Wear of the thimbles is a concern because they serve as a portion of the reactor coolant system pressure boundary.

IE Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors" was issued in July 1988. The NRC requested the implementation of inspection programs for thimble tubes. Since the issuance of IE Bulletin 88-09, two inspections have been performed (RF-4 and RF-5) on thimble tubes at VCSNS. Several thimble tubes were repositioned in RF-5, but no thimble tubes have been capped or required replacement. Analysis of the wear rate data determined that ECT is not required on the thimble tubes again until RF-14 based on calculations performed in association with the inspections.

B.1.3.2 Conclusion

The Bottom Mounted Instrumentation Inspection has been demonstrated to be capable of detecting and managing loss of material in the thimble tubes. The Bottom Mounted Instrumentation Inspection 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 operations.

B.1.4 CHEMISTRY PROGRAM

The Chemistry Program is consistent with XI.M2, *Water Chemistry*, and the chemistry-related portions of XI.M30, *Fuel Oil Chemistry*, as identified in NUREG-1801 with the following clarifications:

- Detection of Aging Effects: The Chemistry Program is a mitigation program and no aging effects are detected as part of this program. The plant operating experience provides confirmation of the effectiveness of the program for managing aging during the period of extended operation. Based on this experience, VCSNS does not commit to performing one-time inspections to verify the effectiveness of the Chemistry Program as suggested by NUREG-1801 under XI.M2.

B.1.4.1 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 surfaces of components are exposed will minimize corrosion. 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.

A review of operating experience did not reveal a loss of component intended function of components that are exposed to borated water, closed cooling water, or treated water that could be attributed to an inadequacy of the Chemistry Program. This operating experience confirms the effectiveness of the Chemistry Program for borated, closed cooling (treated), and treated water to manage aging effects when continued into the period of extended operations.

Analyzing and trending the water chemistry specifications 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 Chemistry Program confirms the reasonableness and acceptability of the sampling frequency.

A review of operating experience did not reveal any instances of a loss of the component intended function of components exposed to fuel oil that could be attributed to an inadequacy of the Chemistry Program. There have been no readings found out of specification during testing of the fuel oil in the storage tanks for the Class 1E Diesel Generators. This can be credited to the fact that fuel delivered to the site is sampled/analyzed before any additions are made to the tanks.

B.1.4.2 Conclusion

The Chemistry Program has been demonstrated to be capable of managing loss of material, cracking, and fouling of components exposed to borated water, closed cooling water, treated water, or fuel oil environments. The Chemistry 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 operations.

B.1.5 FIRE PROTECTION PROGRAM

The Fire Protection Program is consistent with XI.M26, *Fire Protection*, and XI.M27, *Fire Water System*, as well as XI.M33, *Selective Leaching of Materials*, as identified in NUREG - 1801 with of the enhancements specified in the following table and with the following clarifications:

- Parameters Monitored/Inspected: VCSNS fire rated door inspections monitor holes or breaks in the door surface at a frequency of every 6 months rather than the bimonthly frequency recommended. 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. [XI.M26]
- Parameters Monitored/Inspected: Aging management of the fuel supply line for the diesel-driven fire pump at VCSNS is credited to the Chemistry Program and is not managed by the Fire Protection Program. [XI.M26]
- Parameters Monitored/Inspected: VCSNS maintains proper clearances (gap) between door, frame, and threshold in accordance with station procedures. However, VCSNS does not consider maintaining the clearances to be an aging effect for license renewal. [XI.M26]
- Detection of Aging Effects: VCSNS intends to perform ultrasonic testing of selected fire protection piping to detect aging effects in lieu of disassembly of fire protection piping for inspection or full-flow testing of stagnant portions of fire protection piping. (See program enhancements.) [XI.M27]

The following enhancements will be incorporated into the Fire Protection Program prior to the period of extended operations.

NUREG-1801 Program	Attributes	Enhancements
XI.M27, Fire Water System	4. Detection of Aging Effects	Sprinklers will either be replaced or representative samples will be submitted to a recognized laboratory for field service testing in accordance with NFPA code 25. Subsequent replacement or field service testing of representative samples will occur at 10-year intervals. Ultrasonic testing of representative portions of above ground fire protection piping that are exposed to water but do not normally experience flow will be performed before the end of the current operating term. Ultrasonic testing will occur at 10-year intervals thereafter.
XI.M33, Selective Leaching of Materials	4. Detection of Aging Effects	A new one-time inspection, to be performed just prior to the end of the current operating term, will be added to the Fire Protection Program. The inspection will include a Brinnell Hardness Test or equivalent test in order to detect and characterize a reduction of material hardness (loss of material) due to selective leaching for a representative sample of susceptible brass and cast iron components in the Fire Service System.

B.1.5.1 Operating Experience

B.1.5.1.1 Mechanical

Monthly surveillances are conducted on the fire protection system consisting of flow tests and pump start tests. Flow tests and flushes of the main distribution loops have been conducted to ensure functionality and have all met acceptance criteria with the exception described below. Working pressure and flow pressure are measured during these tests. This will indicate fouling to an unacceptable level and hence manage this aging effect. Fire hydrants and sprinklers are visually inspected for aging effects. This visual inspection looks for painted, corroded, damaged, or dirty sprinkler heads, obstruction of sprinkler heads, and proper orientation of sprinkler heads. The fire hydrants are inspected for corrosion on the exterior surface that might impede operation and standing water in the hydrant barrel that might indicate valve leakage or fouling.

A non-conformance notice (NCN) was generated in January of 1994 in association with low flow during flow testing of the main distribution loop. As part of the resolution the piping was hydrolazed to remove accumulated deposits. Additionally, engineering evaluation determined that a reduction and redistribution of sprinkler heads was permissible and would restore the required pressure at the sprinkler heads to ensure full spray pattern. The results of flow testing of the fire protection piping since this occurrence have been acceptable.

B.1.5.1.2 Fire Barriers And Fire Barriers Penetration Seals

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 non-conformance with the acceptance criteria.

NRC Inspection Report 50-395 / 98-01 **[Reference B-5]** concludes that the surveillance procedures are excellent and satisfy the requirements of Generic Letter 86-10 "Implementation of Fire Protection Requirements".

A review of past licensee event reports (LERs) and NRC Inspection Reports 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 LERs 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 LERs or NRC Inspection Reports are related to aging.

Non-conforming conditions were noted during surveillance of fire barrier penetration seals that were aging related cracks and separations. Conditions were repaired in accordance with station procedures.

No condition evaluation reports (CERs) were initiated for fire barriers or fire barrier penetration seals relevant to aging.

B.1.5.1.3 Fire Doors

VCSNS has no failures or adverse trends for fire doors. Surveillance inspections in the last five years have not identified any non-conformance relative to the acceptance criteria.

Frequency of inspections performed since the initial implementation of the Technical Specifications requirements is considered acceptable based on industry operating experience. Industry experience indicates that degradation of a door will be detected prior to loss of function.

LERs for fire doors were compiled using the licensing database. An LER was generated for a combination of fire/pressure door, but not the fire barrier function, and reported a design deficiency which is not related to aging. Two LERs were identified for missed weekly surveillances. Two LERs were identified for fire doors not fully closed.

No non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for fire doors relevant to aging.

B.1.5.2 Conclusion

The Fire Protection Program has been demonstrated to be capable of detecting and managing aging effects for the fire water system, for fire barriers and fire barrier penetrations seals, and for fire doors. The Fire Protection Plan 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 operations.

B.1.6 FLOW-ACCELERATED CORROSION MONITORING PROGRAM

The Flow-Accelerated Corrosion Monitoring Program is consistent with XI.M17, *Flow-Accelerated Corrosion*, as identified in NUREG-1801.

B.1.6.1 Operating Experience

After industry experience indicating that feedwater heaters may be subject to flow-accelerated corrosion (FAC), VCSNS conducted an inspection of feedwater heaters that revealed some degradation of the pressure boundary. A repair of the subject feedwater heater was accomplished in accordance with the requirements of the applicable code. Some degradation of the feedwater piping was found downstream of the feedwater regulating valves and the piping was replaced. The need for inspections is determined by a calculation performed in accordance with engineering procedures. If components exhibit high wear during a cycle they are replaced with more FAC-resistant material. The change to MPA (Methoxypropylamine) chemistry control after RF-9 (1996) has aided in the control of FAC at the station. Refueling summary reports for each refuel outage since RF-8 in 1994 were examined. The results demonstrate a mature well functioning FAC program at VCSNS.

B.1.6.2 Conclusion

The Flow-Accelerated Corrosion Monitoring Program has been demonstrated to be capable of detecting and managing loss of material for components susceptible to flow-accelerated corrosion. The Flow - Accelerated Corrosion Monitoring 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 operations.

B.1.7 IN-SERVICE INSPECTION (ISI) PLAN

The In-Service Inspection (ISI) Plan is consistent with XI.M1, *ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD*, as identified in NUREG-1801 with the following clarification:

- VCSNS is committed to the 1989 Edition of ASME Section XI with no addenda for the second ten-year inspection interval. VCSNS has adopted the 1995 Edition of ASME Section XI with 1996 Addenda for ultrasonic examination requirements.

B.1.7.1 Operating Experience

VCSNS has operated since October 1982, and has performed inservice inspections in accordance with relevant portions of approved editions of ASME Code Section XI throughout that period. Two specific examples of VCSNS operating experience in which the In-Service Inspection (ISI) Plan (including repair and replacement) played a role follow.

In the Reactor Coolant System, primary water stress corrosion cracking contributed to leakage that developed at the reactor vessel "A" hot leg nozzle (discovered in 2000 while Refueling Outage 12). This leakage was detected by virtue of boric acid residue, and confirmed by volumetric examination. The crack was inspected, evaluated and repaired in accordance with ASME Section XI criteria.

In the Steam Generators, inservice inspections of tubes are performed in accordance with station surveillance procedures. Degradation of steam generator tubes was noted during the first ten-year inservice inspection interval. Indications of magnetite formation and tube denting were noted as early as 1990, and damping cables were installed to reduce high-cycle vibration of steam generator tubes during RF-5. Subsequently, the steam generators were replaced in 1994.

B.1.7.2 Conclusion

The In-Service Inspection (ISI) Plan has been demonstrated to be capable of detecting and managing aging effects of ASME code components in the Reactor Coolant System. The In-Service Inspection (ISI) Plan 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 operations.

B.1.8 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is consistent with XI.M3, *Reactor Head Closure Studs*, as identified in NUREG-1801.

B.1.8.1 Operating Experience

The current aging management program for reactor head closure stud bolting is largely dependent upon the In-Service Inspection (ISI) Plan. That program has provisions regarding inspection techniques and evaluations, and other aspects relevant to monitoring the condition of reactor head closure stud bolting. The ASME Code has been shown to be effective in managing aging effects in Class 1 components, including the reactor closure head stud bolting.

VCSNS has operated since October 1982, and has performed inservice inspections in accordance with relevant portions of approved editions of ASME Code Section XI throughout that period. During this period of time no damage to the reactor head closure stud bolting materials has been detected.

B.1.8.2 Conclusion

The Reactor Head Closure Studs Program has been demonstrated to be capable of detecting and managing loss of mechanical closure integrity for the closure stud bolting. The Reactor Head Closure Studs 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 operations.

B.1.9 SERVICE WATER SYSTEM RELIABILITY AND IN-SERVICE TESTING PROGRAM

The Service Water System Reliability and In-Service Testing Program is consistent with XI.M20 *Open-Cycle Cooling Water System*, as identified in NUREG-1801.

B.1.9.1 Operating Experience

The application of measured corrosion rates has been demonstrated to provide adequate information on the rate of loss of material to predict when replacement of components might be necessary. Operating experience demonstrates that performance testing on raw water heat exchangers provides adequate predictive modeling for fouling of heat transfer surfaces to prevent loss of intended function.

Based on operating experience, the Service Water System Reliability and In-Service Testing Program is capable of detecting pin hole leaks in the Service Water System prior to loss of function. A combination of chemicals are injected into the system to form protective coatings on the internal surface of the piping, induce gradual flaking off of tubercles, and to act as a silt dispersant. VCSNS has adjusted the rate of addition of corrosion inhibitor, deposit control, and silt dispersant chemicals based on industry experience.

B.1.9.2 Conclusion

The Service Water System Reliability and In-Service Testing Program has been demonstrated to be capable of managing the effects of aging for components in raw water environments. The Service Water System Reliability and In-Service Testing 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 operations.

B.1.10 STEAM GENERATOR MANAGEMENT PROGRAM

The Steam Generator Management Program is consistent with XI.M19 *Steam Generator Tube Integrity,* as identified in NUREG-1801.

B.1.10.1 Operating Experience

Industry operating experience in the mid-1970's showed that there was a high rate of tube plugging with the dominant damage caused by tube wastage. Chemistry controls were adjusted to correct the tube wastage but led to conditions conducive to corrosion of the carbon steel support plates, which led to tubing deformation as a result of denting and cracking and an unacceptable rate of tube plugging. The industry, working through EPRI, implemented steam generator programs with aggressive improvements in control of secondary-side water chemistry and upgrades in secondary-side equipment, thus essentially eliminating wastage and denting. The industry incorporated these successful programmatic strategies in the EPRI Secondary Water Chemistry Guidelines and associated supporting documents. VCSNS meets the industry guidelines for a steam generator program and secondary water chemistry;.

All three steam generators were replaced at VCSNS during RF-8 (1994). Since then, at the recommendation of the vendor, four tubes have been plugged because they were not expanded into the tube sheet during manufacturing. A partial eddy current inspection (Steam Generators A and B) and moisture carryover modification was conducted during RF-9. Partial eddy current inspections were conducted during RF-10 (Steam Generator C) and RF-11 (Steam Generators A and B). A 100% eddy current inspection of Steam Generators A, B, and C was conducted during RF-12. Also, during RF-12, sampling at top tube sheet (TTS) and low row U bends, a full secondary side inspection, and sludge lancing of tube sheets were performed.

No significant degradation was found during these inspections.

B.1.10.2 Conclusion

The Steam Generator Management Program has been demonstrated to be capable of detecting and managing the effects of aging for the steam generator tubes and tube plugs. The Steam Generator Management 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 operations.

B.1.11 10 CFR 50 APPENDIX J GENERAL VISUAL INSPECTION

The 10 CFR 50 Appendix J General Visual Inspection is consistent with XI.S4, *10 CFR 50 Appendix J*, as identified in NUREG-1801.

B.1.11.1 Operating Experience

The most recent Type A ILRT was completed in March 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 also implemented during RF-10 and RF-12 to satisfy the additional requirements for general visual structural examination of the containment system.

No licensee event reports (LERs) were initiated subsequent to any general visual structural examination of the containment system. There were no non-conformances (NCNs) or condition evaluation reports (CERs) identified that resulted from conditions related to aging mechanisms. NRC Inspection Report 50-395 / 93-09 [**Reference B-6**] reviewed the station surveillance procedure and concluded that the procedure is acceptable to implement the general containment visual inspection prior to a Type A ILRT.

B.1.11.2 Conclusion

The 10 CFR 50 Appendix J General Visual Inspection has been demonstrated to be capable of managing the effects of aging for the containment liner, associated moisture barriers, and the Reactor Building structure. The 10 CFR 50 Appendix J General Visual Inspection 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 operations.

B.1.12 10 CFR 50 APPENDIX J LEAK RATE TESTING

The 10 CFR 50 Appendix J Leak Rate Testing is consistent with XI.S4, *10 CFR 50 Appendix J,* as identified in NUREG-1801.

B.1.12.1 Operating Experience

NRC Inspection Report 50-395 / 93-09 **[Reference B-6]** concluded that the station surveillance procedure provides proper guidance and satisfies regulatory requirements to perform a Type A ILRT.

Over three refueling cycles (most recently RF-10, RF-11, and RF-12), Type B penetrations delineated in the station surveillance procedure were tested with satisfactory results.

A non-conformance (NCN) was documented for rust found on the Reactor Building liner plate adjacent to the moisture barrier and a degraded moisture barrier. The disposition was to clean-up the rust on the Reactor Building liner plate adjacent to the moisture barrier and to replace affected portions of the moisture barrier. 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.

B.1.12.2 Conclusion

The 10 CFR 50 Appendix J Leak Rate Testing Program has been demonstrated to be capable of detecting and managing the effects of aging for the components forming the containment pressure boundary. The 10 CFR 50 Appendix J Leak Rate Testing 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 operations.

B.1.13 ASME SECTION XI ISI PROGRAM - IWF

The ASME Section XI ISI Program - IWF is consistent with XI.S3, *ASME Section XI, Subsection IWF*, as identified in NUREG-1801 with the following clarification:

- VCSNS uses 1989 edition of ASME Section XI with no Addenda

B.1.13.1 Operating Experience

A review of ASME Class 1, 2, and 3 component support inspections for the past five years identified one case where acceptance criteria was not met. The gap at the top of a pipe support exceeded the acceptance criteria, however this is not aging related. Operating experience at other nuclear facilities shows that improperly heat-treated anchor bolts are susceptible to stress corrosion cracking. At VCSNS, ASTM A490 anchor bolt material is properly heat-treated by conforming to ASTM Specification A490 through a Certified Material Test Report in accordance with station specifications. Section 3.3.1.1 of the Safety Evaluation Report for WCAP-14422 [**Reference B-7**] 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 at VCSNS 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 specified to be 125,000 psi, which is below the threshold tensile strength level of 150,000 psi where stress corrosion cracking 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.

Two non-conformance notices (NCNs) were identified for instances of minor surface corrosion on supports and anchor bolting. The intended functions were not affected and corrective actions were performed in accordance with site procedures.

No condition evaluation reports (CERs) were initiated subsequent to ASME Class 1, 2, and 3 component support inspections.

B.1.13.2 Conclusion

The ASME Section XI ISI Program - IWF has been demonstrated to be capable of detecting and managing the effects of aging for ASME code supports. The ASME Section XI ISI Program - IWF for Class 1, 2, and 3 component supports, and support anchorage 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 operations.

B.1.14 BATTERY RACK INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

(1) **Scope** - The scope of the Battery Rack Inspection includes the battery racks for the following systems:

- Electrical DC (ED) System (Vital Batteries)
- Fire Service (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 3.8.2.1, while the regulatory basis for inspecting battery racks for the FS System is the commitment in the fire protection procedure.

(2) **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.

(3) **Parameters Monitored or Inspected** - The ED System and FS System battery racks specific examination guidelines are provided in IEEE-450. 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.

(4) **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.

(5) **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.

For the FS System, visual examination is performed every 18 months in accordance with a commitment in the fire protection procedure.

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.

(6) **Acceptance Criteria** - For the ED System, the acceptance criterion is "no visual indication of loss of material due to corrosion" as stated in a surveillance test procedure. For the FS System, acceptance criterion is "no visual indication of loss of material due to corrosion" as stated in a surveillance test procedure.

(7) **Corrective Actions** - For the ED and FS Systems, the surveillance procedure provides guidance when abnormalities are observed. Repair / replacement of unacceptable batteries or racks is in accordance with an electrical maintenance procedure. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) **Administrative Controls** - For the ED and FS Systems, visual examination for structural integrity of racks, that includes an attribute for loss of material due to corrosion, is implemented in accordance with a surveillance procedure. 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. Visual examination of the FS System is performed to comply with the commitment in the fire protection procedure.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedures.

B.1.14.1 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 rack 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 (LERs) 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

(NCNs) or condition evaluation reports (CERs) were initiated subsequent to inspections for battery racks.

B.1.14.2 Conclusion

The Battery Rack Inspection Program has been demonstrated to be capable of managing loss of material for steel battery racks. The Battery Rack Inspection 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 operations.

B.1.15 CONTAINMENT COATING MONITORING AND MAINTENANCE PROGRAM

The Containment Coating Monitoring and Maintenance Program is consistent with XI.S8, *Protective Coating Monitoring and Maintenance,* as identified in NUREG-1801with the following clarifications:

- Scope of Program: The VCSNS aging management program is based on RG 1.54 Revision 0 which is acceptable per the XI.S8 program description discussion.
- Scope of Program: For the Westinghouse scope of supply (NSSS components and equipment) an alternative methodology for meeting the requirements of RG 1.54 was employed and was accepted by the NRC, as documented in Section 3A of the VCSNS FSAR.

B.1.15.1 Operating Experience

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 were documented in the non-conformance (NCN) program.

The IWE inspection conditions and other areas of containment coating degradation were also documented in the report for maintenance rule inspections performed in 2000. None of the degraded conditions have an immediate adverse effect on the ability of the coatings to perform their intended function(s). Requirements were established for engineering to evaluate the identified conditions for corrective actions. These conditions were documented in the condition evaluation report (CER) program.

In accordance with the dispositions of the NCN and CER documents, most of the areas with identified conditions were reworked per appropriate civil maintenance procedures during RF-12. Conditions that have not been repaired 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 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. Augmented inspections were conducted during the April 2002 refueling outage with no observable changes in the condition of the identified areas.

Additionally, GSI-191, Assessment of Debris Accumulation on PWR Sump Performance, addresses operating experience which identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings, that could potentially impact the performance of the emergency core cooling system. However, degradation of coatings is an issue under the CLB and is not specifically related to the 40-year term of the current operating license, and therefore is not a TLAA. As such, the issue is not specifically a license renewal concern, but has been accounted for during the period of extended operation as described below.

Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, describes the 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, and therefore an effective means to manage loss of material due to corrosion of carbon steel inside containment.

The SCE&G response to Generic Letter 98-04 [**Reference B-8**] states that the plant 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.

B.1.15.2 Conclusion

The Containment Coating Monitoring and Maintenance Program has been demonstrated to be capable of maintaining the integrity of the protective coatings inside the Reactor Building. The Containment Coating Monitoring and Maintenance 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 operations.

B.1.16 CONTAINMENT ISI PROGRAM - IWE/IWL

The Containment ISI Program - IWE/IWL is consistent with XI.S1 *ASME Section XI, Subsection IWE* and XI.S2 *ASME Section XI, Subsection IWL,* as identified in NUREG-1801with the following clarification:

- VCSNS uses the 1992 Edition of ASME XI with 1992 Addenda

B.1.16.1 Operating Experience

Examinations for the first period of first interval were performed during RF-12 with satisfactory results. There were no licensee event reports (LERs) based on these examinations.

Non-conformance notices (NCNs) and/or condition evaluation reports (CERs) were originated and dispositioned for the following conditions identified during these examinations:

- Containment Liner Coating Degradation (NCN) - Several areas of top coat were identified as degraded; however, the primer coat was intact with no signs of deterioration. The affected areas were cleaned and re-coated. Two areas of top coat in the dome were identified with initial signs of degradation. These areas have been identified for augmented inspections during future refueling outages.
- RHR and Spray Guard Pipe (CER) - Groundwater leakage identified at penetrations in the Auxiliary Building resulted in degradation (corrosion) of guard pipes. Subsequent evaluations determined that the guard pipe wall thickness remained acceptable. These areas have been identified for augmented inspections during future refueling outages.
- Concrete Leaching (CER) - Concrete leaching in the Tendon Access Gallery has been attributed to groundwater seepage through cracks and construction joints within the surrounding fill concrete. One specific location was also identified with a minor corrosion build-up on the outer wall. Chemical analysis has determined that the groundwater is not aggressive. These areas have been identified for augmented inspections during future refueling outages.
- Moisture Barrier (CER) - Minor cracking and separation of the moisture barrier was identified at a few locations. These areas were repaired an/or replaced.

Augmented inspections were conducted during the April 2002 refueling outage for the above conditions. Additional CERs were originated for follow-up repair and/or replacement.

B.1.16.2 Conclusion

The Containment ISI Program - IWE/IWL has been demonstrated to be capable of detecting and managing the effects of aging for the liner, associated moisture barriers, and the Reactor Building structure. The Containment ISI Program - IWE/IWL 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 operations.

B.1.17 FLOOD BARRIER INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The VCSNS Flood Barrier Inspection Program is identified for completeness since it contains individual components that have the unique 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.

(1) **Scope** – Nuclear safety-related flood barriers are credited with mitigating the effects of internal flood. Nuclear safety-related flood barriers include curbs at entrances to cubicles housing safety grade equipment as stated in **FSAR Section 6.3.2.2.7**.

Designated flood doors (watertight doors) are identified in plant specifications and are listed on architectural drawings. Ten doors are designated as flood doors (watertight doors).

Penetrations requiring Nuclear Safety Related flood seals are specified in design basis documents. Penetrations requiring Nuclear Safety Related flood seals are shown on engineering drawings for the Intermediate Building, the Control Building, and the Diesel Generator Building. Eleven penetrations require Nuclear Safety Related flood seals.

(2) **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.

(3) **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 an engineering procedure. 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 and include cracking, fraying, separation from penetration, and through-wall holes.

(4) **Detection of Aging Effects** – The Flood Barrier Inspection Program detects aging effects prior to loss of intended function.

(5) **Monitoring and Trending** – Visual examination of concrete structures is performed as stated in an engineering procedure. Visual examination of the flood barrier penetration seals that are also fire barrier penetration seals is performed

as stated in Surveillance Test Procedures. 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. No actions are taken as part of this program to trend inspection or test results.

(6) **Acceptance Criteria** – Flood barrier and flood barrier penetration seal examination acceptance criteria are provided in an engineering services procedure for flood barriers that are not fire barriers. Acceptance criteria are no cracks, no exposed reinforcing steel, no corrosion, no scaling, no popouts, no surface pitting, and no spalling. 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 a technical requirements package. Acceptance criteria are provided for indication of cracking, separation between surfaces at penetration, and no through-wall holes.

(7) **Corrective Actions** – Maintenance work requests are initiated to repair abnormalities. The non-conformance (NCN) process is initiated for flood barrier penetration seals that do not meet the acceptance criteria in compliance with the Fire Protection Evaluation Report. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) **Administrative Controls** – The flood barriers inspections are implemented through an engineering procedure and surveillance test procedures described above. Insp·ctions are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure.

B.1.17.1 Operating Experience

Since the majority of f d barriers are also fire barriers, inspection attributes delineated in surveillance test procedures 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 (LERs), non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals relevant to aging.

B.1.17.2 Conclusion

The Flood Barrier Inspection Program has been demonstrated to be capable of detecting and managing the effects of aging for flood barrier components. The Flood Barrier Inspection 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 operations.

B.1.18 MAINTENANCE RULE STRUCTURES PROGRAM

The Maintenance Rule Structures Program is consistent with XI.S6, *Structures Monitoring Program*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Maintenance Rule Structures Program prior to the period of extended operation:

NUREG-1801 Program	Attributes	Enhancement
XI.S6, Structures Monitoring Program	1.) Scope	Future inspections will add: North Berm, Electrical Manhole EMH-2 interior inspection, Inaccessible Areas when exposed by excavation, Flood Barrier Seals for Control and Diesel Generator Buildings, Portions of the power path from the power circuit breaker (PCB) in the substation to the safety-related buses, and Groundwater chemical analyses.
	3.) Parameters Monitored / Inspected	Groundwater chemical analyses will include: pH, Sulfates and Chlorides.
	5.) Monitoring and Trending	Groundwater chemical analyses will be used to monitor changes in aggressiveness of the below grade environment.

B.1.18.1 Operating Experience

An initial baseline position was established at VCSNS for the acceptability of the maintenance rule structures to remain capable of providing their maintenance rule 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 maintenance rule structures in accordance with existing regulatory or licensing commitments at VCSNS during the period of 1993 to 1996.

The 1996 baseline assessment concluded that the maintenance rule 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 0 systems and components.

The maintenance rule inspection report completed in 2000 noted that in general, most of the maintenance rule structures and structural components were evaluated to be "acceptable" with regards to continued function. However, nine items/areas were identified as "Acceptable with Deficiencies" that exhibited a trend of aging. 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). These items were entered into the plant corrective action program for resolution. The next inspection is scheduled in 2005.

B.1.18.2 Conclusion

The Maintenance Rule Structures Program has been demonstrated to be capable of detecting and managing the effects of aging for structures and structural components. The Maintenance Rule Structures 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 operations.

B.1.19 MATERIAL HANDLING SYSTEM INSPECTION PROGRAM

The Material Handling System Inspection Program is consistent with XI.M23, *Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems*, for the bridge and trolley structural members and rails as identified in NUREG-1801 with the following clarifications:

- **Scope:** The polar crane girders and brackets are addressed under the Maintenance Rule Structures Program.
- **Scope:** Wear on the crane rails has been determined to not require aging management for the VCSNS cranes.
- **Parameters Monitored and Inspected:** The number and magnitude of lifts made by the cranes are addressed as a TLAA under **Section 4.7.2** of this Application.

B.1.19.1 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.

NRC Inspection Report 50-395 / 82-28 [**Reference B-9**] documents review and approval of the VCSNS material handling system maintenance procedures and concludes that the procedures comply with NUREG-0612 requirements.

Prior to 1996, a condition was identified of overstressed bolted connections in the trolley of the polar crane. The vendor (Whiting) identified this non-conformance as a design deficiency. The overstressed bolts were replaced with high strength ASTM A325 bolts during refueling outage RF-7.

While evaluating the polar crane for handling the replacement steam generators, the vendor (Whiting) identified a design deficiency on overstressed areas of the trolley and bridge girders. Modifications were completed prior to refueling outage RF-8.

Industry operating experience for the Material Handling System was reviewed in association with the maintenance rule. A total of 15 events were identified of which eight are relevant to VCSNS. VCSNS concluded that system design and operating procedures would anticipate these events.

In 2000 a non-conformance (NCN) was identified for catastrophic failure of the spent fuel bridge crane roller guide bearing due to age related stress corrosion cracking. The probable cause for catastrophic failure was determined to be a long period of inactivity in a humid environment. The NCN evaluation determined that this condition does not affect structural integrity or function of the spent fuel bridge crane. Therefore, this condition is not an aging effect since the intended function of the crane is maintained.

B.1.19.2 Conclusion

The Material Handling System Inspection Program has been demonstrated to be capable of managing loss of material for crane rails, rail supports, and structural supports. The Material Handling System Inspection 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 operations.

B.1.20 PRESSURE DOOR INSPECTION PROGRAM

There is no NUREG-1801 item addressing this program. This is a plant specific program.

Pressure doors at VCSNS are used to separate critical equipment from high energy pipe breaks; and are designed, procured and installed to specific specifications.

(1) **Scope** – The need to maintain pressure barriers (which also serve as fire barriers) is required by VCSNS Technical Specification 4.7.6.e.3. There are 34 doors that are Nuclear Safety Related pressure resistant doors. Thirteen (13) doors are Quality Related pressure doors. There are 47 doors that are rated as pressure resistant.

(2) **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.

(3) **Parameters Monitored or Inspected** – Parameters monitored for Nuclear Safety Related pressure doors are loss of material of doors and door hardware. Parameters monitored for Quality Related pressure doors are loss of material of doors and door hardware. 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.

(4) **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 intended function.

(5) **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. Aging effects for Nuclear Safety Related pressure doors are detected by visual examination. No actions are taken as part of this program to trend inspections or test results.

(6) **Acceptance Criteria** – Quality Related pressure door acceptance criteria is provided in technical requirement packages. Nuclear Safety Related pressure door acceptance criteria is provided in surveillance test procedures. 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.

(7) **Corrective Actions** – Minor abnormalities (loose knobs, latches or other appurtenances) are repaired using guidance provided by the vendor. The condition evaluation report (CER) process is initiated for pressure doors that do not meet the acceptance criteria. A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) **Administrative Controls** – The need to maintain pressure barriers (which also serve as fire barriers) is required by VCSNS Technical Specification 4.7.6.e.3. The surveillance requirements are established in fire protection procedures. Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.20.1 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 non-conformances (NCNs) relative to the acceptance criteria.

An occurrence of steam propagation into sensitive rooms through fire doors was identified. One door was replaced with a Quality Related pressure resistant / fire door, while a Quality Related pressure resistant door was added at another location.

No non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for pressure doors relevant to aging.

The frequency of inspections performed since the 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 plant procedures. The pressure door inspections are implemented by plant procedures and controlled by the SCE&G Quality Assurance Program.

B.1.20.2 Conclusion

The Pressure Door Inspection Program has been demonstrated to be capable of detecting and managing the effects of aging for pressure doors. The Pressure Door Inspection 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 operations.

B.1.21 SERVICE WATER POND DAM INSPECTION PROGRAM

The Service Water Pond Dam Inspection Program is consistent with XI.S7, *RG 1.127 Inspection of Water-Control Structures Associated With Nuclear Power Plants*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Service Water Pond Dam Inspection Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancement
XI.S7, RG 1.127 Inspection of Water Control Structures Associated with Nuclear Power Plants	1.) Scope	North Dam piezometers will be added.
	3.) Parameters Monitored / Inspected	Water level.
	5.) Monitoring and Trending	Inspections will be made every 5-years concurrent with the RG 1.127 inspections.
	6.) Acceptance Criteria	Nominal elevation of adjacent Service Water Pond and Monticello Reservoir.

B.1.21.1 Operating Experience

During each inspection of the Service Water Pond 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 by inspections performed in 1990 and 1995. 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 spraying of herbicides conducted in accordance with plant procedures.

Structural calculations document the results of the survey monitoring data for the SWP North, South Dam and West Embankment. The calculations provide a review of the vertical and horizontal displacements of the Service Water Pond North Dam and South Dam since

1977. The calculations also provide 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. Structural calculations also provide 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 five year inspection of the Service Water Pond Dams required by the NRC, FERC conducted inspections of the Service Water Pond Dams in February 1997, July 1999, and July 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 1997 FERC Dam safety inspection report **[Reference B-13]** recommended that SCE&G visually inspect the Service Water Pond Dams and West Embankment annually and test the accessible piezometers. The annual visual inspection is scheduled 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 with acceptable results.

B.1.21.2 Conclusion

The Service Water Pond Dam Inspection Program has been demonstrated to be capable of detecting and managing trends in movement and the effects of aging for the service water dams. The Service Water Pond Dam Inspection 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 operations.

B.1.22 SERVICE WATER STRUCTURES SURVEY MONITORING PROGRAM

There is no NUREG-1801 item addressing this program. This is a plant specific program.

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 Service Water Pump House (SWPH) and the Service Water Intake Structure (SWIS) was much more than the original pre-construction estimates. As a result, survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is conducted to monitor any differential in vertical and horizontal displacement. This monitoring is conducted to satisfy the requirements specified by Operating License condition 2.C.5 and **FSAR Section 2.5.4.10.6.2**.

(1) **Scope** – Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is conducted in accordance with plant procedures.

(2) **Preventive Actions** – No actions are taken, as part of the survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A", which prevent aging effects or mitigate aging degradation. Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is a condition monitoring program.

(3) **Parameters Monitored or Inspected** – Survey monitoring is conducted to detect any vertical and/or horizontal movement associated with settlement of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A". The survey monitoring data is reviewed by Design Engineering to ensure that settlements remain within established criteria.

In addition to survey monitoring, the structures are visually inspected in accordance with engineering services procedures 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

(4) **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 survey monitoring. The survey results are reviewed and evaluated for trends in movement associated with settlement that exceeds the acceptance criteria. This review and the visual inspection of the structures will detect any adverse horizontal or vertical displacements prior to the loss of structure intended function(s).

(5) **Monitoring and Trending** – Aging effects associated with settlement for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are detected by survey monitoring in accordance with Current Licensing Basis (CLB) requirements. 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. 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.

(6) **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 design engineering guidelines. Survey results are evaluated for adverse trends in vertical displacement. The measurements are compared to the previous survey results and to acceptance criteria defined in engineering guidelines. The SWIS is also monitored for differential displacement between the middle and ends of the tunnel. If the acceptance criterion for the differential displacement is reached or exceeded, further engineering evaluations are required.

(7) **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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) Administrative Controls – Periodic survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" is performed in accordance with Current Licensing Basis (CLB) requirements. Survey data is collected and documented. Design Engineering reviews the survey data in accordance with procedures and documents the results of the evaluation in structural calculation series. Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line "A" are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.22.1 Operating Experience

Initial settlement of the SWPH and SWIS was much more than the original pre-construction estimates. The degree and manner of settlement caused cracking to occur in the SWIS, which was subsequently repaired (grouted). A special settlement study was performed for the SWPH and SWIS. There has been no significant settlement of the SWPH or SWIS 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. 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. Considering survey process inaccuracy and seasonal fluctuations affecting data collection, the total differential was not considered significant enough to warrant further evaluation. Survey results from 1977 to the present are documented in structural calculations.

Survey monitoring for differential settlement (middle to ends) of the SWIS has been conducted since February 1985. Between the July 1985 survey and February 1986 survey of the SWIS there was a sudden increase in the recorded differential displacement 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. 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.

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. Survey results are documented in structural calculations.

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 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 are documented in structural calculations.

B.1.22.2 Conclusion

The Service Water Structures Survey Monitoring Program has been demonstrated to be capable of detecting and managing trends in movement associated with settlement of the service water structures. The Service Water Structures Survey Monitoring 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 operations.

B.1.23 UNDERWATER INSPECTION PROGRAM (SWIS AND SWPH)

There is no NUREG-1801 item addressing this program. This is a plant specific program.

(1) **Scope** – The scope of the SWIS underwater inspection, conducted in accordance with engineering services procedures, includes visual inspection of the interior length of the intake tunnel, survey monitoring masts, trash racks, access ladder and east end wing walls. The scope of the SWPH underwater inspection, conducted in accordance with engineering services procedures, includes a visual inspection of the intake tunnel, traveling screens/bays and service water pump bays.

(2) **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.

(3) **Parameters Monitored or Inspected** – Guidelines for the underwater inspection of the SWIS and SWPH are specified in engineering services procedures. The main reason for inspecting the SWIS 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 specific areas that are inspected (and for which the condition is documented) are the access ladder, trash racks, survey monitoring masts, and concrete wing walls at the intake end of the SWIS.

Underwater inspections of the SWIS and SWPH monitor 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. 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.

(4) **Detection of Aging Effects** – Attributes associated with aging for the SWIS and SWPH are detected by the underwater inspections. Additionally, survey monitoring of the SWIS and SWPH will detect any horizontal or vertical movement associated with settlement.

(5) **Monitoring and Trending** – The underwater inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The SWIS inspection documentation and reviews/evaluations are filed in structural calculations. In particular these records include the subcontractor's underwater

inspection report, 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.

(6) **Acceptance Criteria** – The acceptance criteria for the underwater inspection of the SWIS is that the inspection data is reviewed by engineering.

Cracks (old and new) are documented and mapped on an engineering procedure attachment. Crack width is measured using wire gauges on a "Go – No/Go" basis by inserting the wire into the crack.

(7) **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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) **Administrative Controls** – The SWIS crack inspections are governed by VCSNS Operating License, Condition 2.C.5.d and implemented through an engineering procedure. The inspection of the SWIS and SWPH to monitor and control corrosion and fouling within the service water system is governed by the SCE&G response to Generic Letter 89-13 **[Reference B-10]** and implemented through another engineering procedure. Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.23.1 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 were grouted with a high strength epoxy grout in 1978 prior to filling the Service Water Pond. Underwater 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 and four cracks with a maximum width greater than the minimum criteria. These cracks were evaluated and documented in a structural calculation. The 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 the minimum criteria. 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.

After filling the Service Water Pond, visual inspection and cleaning of the SWIS and SWPH was performed once each refueling cycle within the preventive maintenance program. In response to Generic Letter 89-13, a new engineering procedure was developed to direct the SWIS and SWPH 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) is recorded and subsequently removed by divers using an eductor.

B.1.23.2 Conclusion

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. The Underwater Inspection Program of the Service Water Intake Structure (SWIS) and Service Water Pump House (SWPH) 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 operations.

B.1.24 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is consistent with XI.M31, *Reactor Vessel Surveillance*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Reactor Vessel Surveillance Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancement
XI.M31, Reactor Vessel Surveillance	1.) Scope	Perform a one-time analysis to demonstrate that the materials in the inlet and outlet nozzles and upper shell course will not become controlling during the period of extended operations. Successful demonstration will preclude the addition of such materials to the material surveillance program for the period of extended operation.
	4.) Detection of Aging Effects	Remove both remaining surveillance capsules during RF-14, have one capsule analyzed and place the other capsule in storage, in accordance with the recommendations of Item 6 of the December 3, 1999 Christopher Grimes (NRC) letter to Douglas Walters(NEI) **[Reference B-11]**.

B.1.24.1 Operating Experience

Industry experience to date supports the conclusion that a Reactor Vessel Surveillance Program compliant with regulatory requirements provides assurance that fracture toughness requirements are met.

Direct measurement of material properties of the irradiated samples recovered from the surveillance capsules provides objective evidence of the effects of radiation embrittlement on reactor vessel materials. Analysis of the VCSNS surveillance capsules removed to date demonstrates that changes to the material properties of the vessel beltline materials are well known and will not result in brittle failure. The VCSNS vessel beltline materials are currently demonstrated by Westinghouse analysis to be below the RTPTS screening criteria for 48 effective full power years (EFPY). Further evaluation to 54 EFPY will be performed prior to the period of extended operations.

The NRC Staff response to License Renewal Issue 98-0085 [**Reference B-11**] confirms that existing radiation surveillance programs provide a suitable means of monitoring vessel fracture toughness to neutron fluence levels associated with the end of the extended operating period, provided that surveillance capsules are available or can be reconstituted.

Industry experience with Inservice Inspections suggests that such programs are capable of detecting flaws before those flaws grow larger than one quarter of vessel wall thickness.

The fuel loading program was revised to implement a low-leakage pattern that reduced the fast neutron flux escaping the core. This change effectively reduced the neutron flux on, and resulting embrittlement of the reactor vessel.

B.1.24.2 Conclusion

The Reactor Vessel Surveillance Program has been demonstrated to be capable of managing reduction of fracture toughness for the reactor vessel beltline materials. The Reactor Vessel Surveillance 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 operations.

B.1.25 PREVENTIVE MAINTENANCE ACTIVITIES - TERRY TURBINE

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The Preventive Maintenance Activities - Terry Turbine is a condition monitoring program that manages loss of material due to general corrosion of carbon steel. This program is composed of controlled plant procedures.

(1) **Scope** - The Preventive Maintenance Activities – Terry Turbine is applicable to the turbine casing and components exposed to an air environment with periodic exposure to steam.

(2) **Preventive Actions** - No actions are taken as part of the Preventive Maintenance Activities – Terry Turbine to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - Parameters monitored or inspected as part of the Preventive Maintenance Activities - Terry Turbine include visible evidence of corrosion on internal surfaces to indicate potential loss of material.

(4) **Detection of Aging Effects** - The Preventive Maintenance Activities - Terry Turbine will detect the presence and extent of aging effects on internal surfaces by visual inspection prior to a loss of component intended function. These effects are loss of material due to general corrosion.

(5) **Monitoring and Trending** - Routine periodic visual inspections are conducted as part of the Preventive Maintenance Activities - Terry Turbine in order to detect age-related degradation and to initiate corrective actions as necessary.

No actions are taken as part of this program to trend inspection results.

(6) **Acceptance Criteria** - The acceptance criteria for the Preventive Maintenance Activities - Terry Turbine is no unacceptable visible indication of loss of material. Indications of loss of material are evaluated by engineering to determine if the condition could result in a loss of the component intended function(s).

(7) **Corrective Actions** - If the results of an inspection are not acceptable an engineering evaluation(s) is performed to assess the material condition and to determine whether the component intended function is affected. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Preventive Maintenance Activities - Terry Turbine are implemented in accordance with controlled station procedures and work processes.

B.1.25.1 Operating Experience

A review of work histories, including non-conformance notices (NCNs), condition evaluation reports (CERs), and problem reports, for the past ten years reveals that no age-related degradation has been detected for the subject components. The inspection activities incorporate vendor recommendations. Based on vendor recommendations and inspection results, continuation of the periodic inspections will identify any age-related degradation prior to a loss of component intended function.

B.1.25.2 Conclusion

The Preventive Maintenance Activities - Terry Turbine has been demonstrated to be capable of detecting and managing loss of material in carbon steel components of the Terry Turbine. The Preventive Maintenance Activities - Terry Turbine 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 operations.

B.1.26 Preventive Maintenance Activities - Ventilation Systems Inspections

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The Preventive Maintenance Activities - Ventilation Systems Inspections is a condition monitoring program that will manage loss of material due to boric acid corrosion, galvanic corrosion and general corrosion in carbon steel, galvanized steel, and copper components, and fouling due to particulates in aluminum, copper, and copper-nickel heat exchanger components.

(1) **Scope** - The Preventive Maintenance Activities – Ventilation Systems Inspections is applicable to the following systems and components exposed to a ventilation environment:

Air Handling (HVAC) System:
- Carbon steel air handling units, air plenums, and fan and plenum housings.
- Galvanized steel air handling units and heat exchanger tubesheets.
- Copper heat exchanger fins and tubes.

Component Cooling System (pump motor coolers):
- Aluminum heat exchanger fins.
- Copper-nickel heat exchanger tubes.
- Carbon steel heat exchanger tubesheets.

Local Ventilation and Cooling System:
- Galvanized steel air handling units.
- Copper heat exchanger fins and tubes.

(2) **Preventive Actions** - No actions are taken as part of the Preventive Maintenance Activities – Ventilation Systems Inspections to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - Parameters monitored or inspected as part of the Preventive Maintenance Activities – Ventilation Systems Inspections include visible evidence of corrosion, including pitting and discoloration, to indicate possible loss of material, and accumulation of dust and particulates on fins and tubes to indicate possible fouling. For those components located in the Reactor Building, visible evidence of boron precipitation may indicate loss of material due to boric acid corrosion.

(4) **Detection of Aging Effects** - In accordance with the information provided in Monitoring and Trending below, the Preventive Maintenance Activities – Ventilation

Systems Inspections will detect the presence and extent of the following aging effects prior to a loss of component intended function:

- Loss of material due to boric acid corrosion, galvanic corrosion and general corrosion.
- Fouling due to particulates (in the ventilation environment).

(5) **Monitoring and Trending -** Routine periodic inspections are conducted as part of the Preventive Maintenance Activities – Ventilation Systems Inspections in order to detect age-related degradation and to initiate corrective actions, as necessary.

Except for the Reactor Building Cooling Units (RBCUs), no actions are taken as part of this program to trend inspection results. For the RBCUs, an engineering procedure requires recording of temperature monitoring data annually or at least once per refueling cycle.

(6) **Acceptance Criteria -** The acceptance criteria for the Preventive Maintenance Activities – Ventilation Systems Inspections is no unacceptable loss of material or fouling of subject components that could result in a loss of the component intended function(s), as determined by engineering evaluation. The engineering procedure contains specific acceptance criteria for the RBCUs.

(7) **Corrective Actions -** If the results of a inspection are not acceptable, an engineering evaluation is performed to assess material condition and to determine whether the component intended function is affected. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability, The corrective action 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.

(9) **Administrative Controls -** The Preventive Maintenance Activities - Ventilation Systems Inspections are implemented in accordance with controlled station procedures and work processes.

B.1.26.1 Operating Experience

A review of work histories, including non-conformance notices (NCNs), condition evaluation reports (CERs), and problem reports, for the past ten years reveals that no age-related degradation has been detected for the subject components.

B.1.26.2 Conclusion

The Preventive Maintenance Activities - Ventilation Systems Inspections has been demonstrated to be capable of detecting and managing the effects of aging for components exposed to a ventilation environment. The Preventive Maintenance Activities - Ventilation Systems Inspections 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 operations.

B.2.0 NEW AGING MANAGEMENT ACTIVTIES

B.2.1 ABOVE GROUND TANK INSPECTION

The Above Ground Tank Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Above Ground Tank Inspection is a new one-time inspection that will detect and characterize loss of material due to galvanic and general corrosion in an internal air space environment, and loss of material and cracking to the due to the corrosive effects of alternate wetting and drying in treated or borated water environments. The Above Ground Tank Inspection will also detect and characterize loss of material due to general corrosion in a treated water (with uncontrolled oxygen levels) environment. The Above Ground Tank Inspection will be performed prior to the period of extended operation.

(1) **Scope -** The Above Ground Tank Inspection is applicable to the internal surfaces of the following components:
- Carbon steel tanks exposed to an internal air space environment in the Condensate, Component Cooling, and Chilled Water Systems;
- Carbon steel pipe and valves exposed to an internal air space environment in the Component Cooling System;
- Carbon steel and stainless steel tanks exposed to a treated water environment in the Condensate, Component Cooling, Reactor Makeup Water Supply and Chilled Water Systems;
- Carbon steel tanks, pipe and valves exposed to treated water having uncontrolled oxygen levels in the Sodium Hydroxide Storage Tank in the Reactor Building Spray System;
- Stainless steel tanks exposed to a borated water environment in the Refueling Water System (Refueling Water Storage Tank).

(2) **Preventive Actions -** No actions are taken as part of the Above Ground Tank Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored/Inspected -** The parameters inspected by the Above Ground Tank Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) **Detection of Aging Effects -** The Above Ground Tank Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation.The results will be applied to the remainder of the components within the scope of the inspection activity. For components exposed to borated and treated

water environments, the sample population should include locations near the air-water interface within the stainless steel Refueling Water Storage Tank (RWST), and near the air-water interface within one of the following carbon steel tanks: the Condensate Storage Tank, the Component Cooling Surge Tank, or one of the Chilled Water Expansion Tanks. It is expected that an engineering evaluation will confirm that the borated water environment of the RWST is more likely to concentrate contaminants at the air-water interface than the treated water environment of the Reactor Makeup Water Supply Tank.

- For components exposed to treated water with uncontrolled oxygen levels, the sample population should include the submerged portions of the Sodium Hydroxide Tank.

For components exposed to an internal air space environment, the sample population should include locations within the air space of one of the following carbon steel tanks: the Condensate Storage Tank, the Component Cooling Surge Tank, or one of the Chilled Water Expansion Tanks. If possible, to simplify the inspection, the same tank chosen to inspect for corrosive impacts of alternate wetting and drying should be selected here.

The Above Ground Tank Inspection will detect the presence and extent of loss of material and cracking on internal surfaces prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Above Ground Tank Inspection to trend inspection results. This is a new one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Above Ground Tank Inspection is no unacceptable loss of material or cracking of subject components that could result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic over-

sight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Above Ground Tank Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.1.1 Operating Experience

The Above Ground Tank Inspection is a new one-time inspection activity for which there is no operating experience.

B.2.1.2 Conclusion

Implementation of the Above Ground Tank Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.2 DIESEL GENERATOR SYSTEMS INSPECTION

The Diesel Generator Systems Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Diesel Generator Systems Inspection is a new one-time inspection that will detect and characterize loss of material due to general corrosion and the corrosive impacts of alternate wetting and drying in air-gas environments. The Diesel Generator Systems Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Diesel Generator Systems Inspection is applicable to the following components in the Diesel Generator Services System:
- Carbon steel expansion joints normally exposed to moist air, and exposed to exhaust air during engine operation, both of which are air-gas internal environments.
- Carbon steel tanks and associated tubing components exposed to starting and control air, an air-gas internal environment.

(2) **Preventive Actions** - No actions are taken as part of the Diesel Generator Systems Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected as part of the Diesel Generator Systems Inspection include wall thickness and/or visible evidence of corrosion, including pitting and discoloration, to indicate possible loss of material for the carbon steel components.

(4) **Detection of Aging Effects** - The Diesel Generator Systems Inspection will use a combination of proven volumetric and/or visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity.

The Diesel Generator Systems Inspection will detect the presence and extent of any loss of material due to general corrosion and the corrosive impacts of alternate wetting and drying for the subject components prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Diesel Generator Systems Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Diesel Generator Systems Inspection is no unacceptable loss of material of subject components that could

result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspections for completeness and acceptability and the corrective action 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.

(9) **Administrative Controls** - The Diesel Generator Systems Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.2.1 Operating Experience

The Diesel Generator Systems Inspection is a new one-time inspection program for which there is no operating experience.

B.2.2.2 Conclusion

Implementation of the Diesel Generator Systems Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.3 LIQUID WASTE SYSTEM INSPECTION

The Liquid Waste System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Liquid Waste System Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion, and cracking due to stress corrosion cracking (SCC) in unmonitored and uncontrolled borated water environments. The Liquid Waste System Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Liquid Waste System Inspection is applicable to stainless steel components exposed to unmonitored and uncontrolled borated water in the following systems:

- Nuclear Plant Drains (ND) – pipe and valve bodies.
- Liquid Waste Processing (WL) – pipe, valve bodies and heat exchanger components.

The unmonitored and uncontrolled borated water environment consists of the following:

- Contents of the Reactor Building or Incore Instrumentation Sumps being discharged through a containment penetration (ND System).
- Contents of the Reactor Coolant Drain Tank (RCDT) on the tube-side of the RCDT Heat Exchanger and passing through a containment penetration (WL System).
- Concentrates from the Waste Evaporator Package on the tube-side of the Concentrates Sample Cooler (WL System).

(2) **Preventive Actions** - No actions are taken as part of the Liquid Waste System Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected by the Liquid Waste System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking, or other age-related degradation.

(4) **Detection of Aging Effects** - The Liquid Waste System Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity. The sample population will consist of susceptible locations within the boundaries of either of the two affected containment penetrations, as well as the internal tube surfaces of the affected heat exchangers.

The Liquid Waste System Inspection is a new one-time inspection that will detect the presence and extent of any loss of material and cracking prior to a loss of component intended function.

(5) **Monitoring and Trending -** No actions are taken as part of the Liquid Waste System Inspection to trend inspection results. This is a new one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** The acceptance criteria for the Liquid Waste System Inspection is no unacceptable loss of material or cracking of subject components that could result in a loss of the component intended function(s), as determined by engineering evaluation.

(7) **Corrective Actions -** If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability, The corrective action 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.

(9) **Administrative Controls -** The Liquid Waste System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.3.1 Operating Experience

The Liquid Waste System Inspection is a new one-time inspection for which there is no operating experience.

B.2.3.2 Conclusion

Implementation of the Liquid Waste System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.4 REACTOR VESSEL INTERNALS INSPECTION

The Reactor Vessel Internals Inspection will be consistent with XI.M16, *PWR Vessel Internals,* as identified in NUREG-1801 prior to the period of extended operation, with the following clarification:

- **Detection of Aging Effects:** The VCSNS resolution criterion for the enhanced VT-1 examination is expected to be less than specified in the GALL program.

The reactor vessel internals inspection is a new inspection, supplementing the In-Service Inspection (ISI) Plan, that will 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.

(1) Scope - The Reactor Vessel Internals Inspection is applicable to the following stainless steel reactor vessel internals components, that are exposed to the borated water environment of the Reactor Coolant System:

- Baffle and Former Assembly (and Bolts)
- Core Barrel (including Flange and Outer Nozzle)
- Fuel Alignment Pins
- Guide Tubes (including Bolts and Support Pins)
- Head and Vessel Alignment Pins
- Hold-down Springs
- Lower Core Plate
- Lower Support Columns (and Bolts)
- Lower Support Plate
- Neutron Panels
- Radial Keys
- Secondary Core Support
- Spray Nozzles
- Upper Core Plate (and Alignment Pins)
- Upper Instrumentation Column
- Upper Support Column (and Bolts)
- Upper Support Plate Assembly
-

The Reactor Vessel Internals Inspection is also applicable to the following nickel-based reactor vessel internals component that are exposed to the borated water environment of the Reactor Coolant System:

- Clevis Inserts and Clevis Insert Bolts

(2) **Preventive Actions** - No actions are taken as part of the Reactor Vessel Internals Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored/Inspected** - The parameters inspected by the Reactor Vessel Internals Inspection include visual evidence of cracking (enhanced by reduction of fracture toughness) and loss of material, and volumetric measurements to indicate possible changes in dimensions, cracking, reduction of fracture toughness and loss of preload.

(4) **Detection of Aging Effects** - In accordance with the information provided in Monitoring and Trending below, the Reactor Vessel Internal Inspection will detect the following aging effects prior to a loss of component intended function:

- Changes in dimensions due to irradiation creep and void swelling
- Cracking due to irradiation-assisted stress corrosion cracking (IASCC)
- Cracking due to primary water stress corrosion cracking (PWSCC) in nickel-based materials
- Loss of material due to wear
- Loss of preload due to stress relaxation
- Reduction of fracture toughness due to irradiation embrittlement and void swelling

(5) **Monitoring and Trending** - 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 dimensions due to void swelling, industry activities 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.

(6) **Acceptance Criteria** - The Reactor Vessel Internals Inspection includes 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.

(7) **Corrective Actions** - If the results of the Reactor Vessel Internals Inspection are not acceptable, based on acceptance criteria to be determined by analysis, then actions will be taken to repair or replace the affected item, or to determine by analysis the acceptability of the item. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Reactor Vessel Internals Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.4.1 Operating Experience

The Reactor Vessel Internals Inspection is a new inspection for which there is no operating experience.

B.2.4.2 Conclusion

The Reactor Vessel Internals Inspection 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.

B.2.5 REACTOR BUILDING COOLING UNIT INSPECTION

The Reactor Building Cooling Unit Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Reactor Building Cooling Unit Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion and cracking due to stress corrosion cracking resulting from exposure to an unmonitored and uncontrolled borated water environment. The borated water environment results from condensation out of the Reactor Building atmosphere, onto Reactor Building Cooling Unit cooling coils, and then into the associated drain lines. The Reactor Building Cooling Unit Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Reactor Building Cooling Unit Inspection is applicable to stainless steel pipe exposed to an unmonitored borated water environment in the reactor building cooling unit drain lines that are part of the VCSNS Roof Drain System.

(2) **Preventive Actions** - No actions are taken as part of the Reactor Building Cooling Unit Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected by the Reactor Building Cooling Unit Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) **Detection of Aging Effects** - The Reactor Building Cooling Unit Inspection will use a combination of proven volumetric and visual examination techniques at sample locations in the drain lines determined by engineering evaluation to be most susceptible to the applicable aging effects. 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 piping.

    The Reactor Building Cooling Unit Inspection will detect the presence and extent of any loss of material and cracking prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Reactor Building Cooling Unit Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Reactor Building Cooling Unit Inspection is no unacceptable loss of material or cracking of subject compo-

nents that could result in a loss of the component intended function(s), as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Reactor Building Cooling Unit Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.5.1 Operating Experience

The Reactor Building Cooling Unit Inspection is a new one-time inspection for which there is no operating experience.

B.2.5.2 Conclusion

Implementation of the Reactor Building Cooling Unit Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.6 SERVICE AIR SYSTEM INSPECTION

The Service Air System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operations.

The Service Air System Inspection is a new one-time inspection that will detect and characterize loss of material due to general corrosion resulting from exposure to an internal moist air environment. The Service Air System Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Service Air System Inspection is applicable to carbon steel pipe, tubing, and valve body components exposed to internal moist air environment that perform the function of maintaining pressure boundary for containment integrity i the following locations:

- Service Air System components in the supply line to the Reactor Building where the line penetrates the containment.
- Service Air and Building Services Systems components used for the leak testing of the Personnel Hatch, Equipment Hatch, and Emergency Personnel Hatch seals.
- Building Services System components used to supply emergency air to the Personnel Hatch and Emergency Personnel Hatch.
- Instrument Air System components in the air intake piping upstream of the instrument air dryers where the piping penetrates the containment.

(2) **Preventive Actions** - No actions are taken as part of the Service Air System Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected by the Service Air System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material or other age-related degradation

(4) **Detection of Aging Effects** - The Service Air System Inspection will use a combination of proven volumetric and visual examination techniques to inspect for general corrosion at selected sample locations to be determined by engineering evaluation.

The Service Air System Inspection will detect the presence and extent of any loss of material prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Service Air System Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Service Air System Inspection is no unacceptable loss of material in subject components that could result in a loss of the component intended function, as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Service Air System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.6.1 Operating Experience

The Service Air System Inspection is a new one-time inspection for which there is no operating experience.

B.2.6.2 Conclusion

Implementation of the Service Air System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.7 SMALL BORE CLASS 1 PIPING INSPECTION

The Small Bore Class 1 Piping Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will increase confidence in the current condition of small bore Reactor Coolant System piping that does not receive a volumetric examination. The Small Bore Class 1 Piping Inspection will be scheduled at or near the end of the second period of the fourth ISI interval.

(1) **Scope -** The Small Bore Class 1 Piping Inspection is applicable to Reactor Coolant System piping and fittings less than 4 inches NPS which are subject to cracking due to flaw growth and stress corrosion cracking (SCC). This inspection is elective in that a sample of piping less than 4 inches NPS that is not required to be examined volumetrically per ASME Section XI code requirements will be examined.

(2) **Preventive Actions -** No actions are taken as part of the Small Bore Class 1 Piping Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected -** The parameter inspected by the Small Bore Class 1 Piping Inspection is evidence of cracking as determined by visual and surface examination that indicates age-related degradation.

(4) **Detection of Aging Effects -** Table IWB 2500-1 of ASME Section XI contains the requirements for in-service inspection of Class 1 piping. The examination categories for piping require volumetric and surface examinations for piping greater than or equal to 4 inches NPS and surface examinations for piping greater than 1 inch NPS but less than 4 inches NPS. Piping less than or equal to 1 inch NPS is exempt from volumetric and surface examinations and receives visual examination during pressure testing.

    Cracking due to flaw growth and SCC are aging mechanisms that originate from the inside diameter of the piping. Volumetric examination of small diameter piping by current ultrasonic or radiographic methods is often ineffective and provides poor results. Engineered removal and replacement of representative sections of small bore piping for destructive testing will provide more reliable inspection results. Destructive examination of the sample permits examination of the inside of the piping and allows for visual and surface examination of the interior surfaces without outage schedule pressure.

    Inspection locations will be selected by engineering using risk-based approaches. Locations most susceptible to cracking will be identified based on engineering evaluation, operating experience, current code requirements, and industry initiatives. Actual inspection locations will be selected based on physical accessibility, exposure levels, and scheduling requirements as well as the results of a review of failure consequences. One of the sample locations should include a butt-weld.

The Small Bore Class 1 Piping Inspection will detect cracking due to flaw growth and SCC in stainless steel components prior to loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Small Bore Class 1Piping Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - No established acceptance criteria exists for destructive examinations or for volumetric examination of small bore Class 1 piping. Acceptance standards for a destructive examination will be determined by VCSNS if not established by industry initiative prior to the time of the inspection.

(7) **Corrective Actions** - Since destructive examination is the currently preferred technique (in the absence of reliable NDE methods) and since piping will be replaced in accordance with ASME Section XI, no corrective actions need be defined prior to the inspection. The results of the inspection may be used as a baseline inspection if results indicate a longer term monitoring program is warranted. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Small Bore Class 1 Piping Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.7.1 Operating Experience

The Small Bore Class 1 Piping Inspection is a new inspection for which there is no operating experience. NRC Information Notice 97-46, Unisolable Crack in High-Pressure Injection Piping, contains industry experience related to cracking of small bore piping.

B.2.7.2 Conclusion

Implementation of the Small Bore Class 1 Piping Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.8 WASTE GAS SYSTEM INSPECTION

The Waste Gas System Inspection will be consistent with XI.M32, *One-Time Inspection,* as identified in NUREG-1801 prior to the period of extended operation.

The Waste Gas System Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion and cracking due to stress corrosion cracking in unmonitored and uncontrolled treated water, and cracking due to stress corrosion cracking in gas environments. The Waste Gas System Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Waste Gas System Inspection is applicable to the following Gaseous Waste Processing System components:

- Stainless steel pipe and valve bodies exposed to an unmonitored and uncontrolled treated water environment.
- Stainless steel tube coils and manifolds in the Hydrogen Recombiner Cooler Condenser exposed to a gas environment.

The unmonitored and uncontrolled treated water environment consists of condensation that forms within the Waste Gas Decay Tanks and is periodically pumped to the Volume Control Tank in the Chemical and Volume Control System.

The gas environment is a mostly nitrogen, with trace amounts of hydrogen, oxygen, and fission product gases, and water vapor from the recombination of hydrogen and oxygen.

(2) **Preventive Actions** - No actions are taken as part of the Waste Gas System Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected by the Waste Gas System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) **Detection of Aging Effects** - The Waste Gas System Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity. The sample population should consist of at least one susceptible location in the stainless steel Waste Gas Decay Tank drain piping (preferably at a low point), and at least one susceptible location in the stainless steel tube-side inlet piping to the Hydrogen Recombiner Cooler Condenser.

The Waste Gas System Inspection will detect the presence and extent of any loss of material or cracking prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Waste Gas System Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Waste Gas System Inspection is no unacceptable loss of material or cracking of subject components that could result in a loss of the component intended function(s), as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Waste Gas System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.8.1 Operating Experience

The Waste Gas System Inspection is a new one-time inspection activity for which there is no operating experience.

B.2.8.2 Conclusion

Implementation of the Waste Gas System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.9 NON-EQ INSULATED CABLES AND CONNECTIONS INSPECTION PROGRAM

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.

This same program will be applied to XI.E2, *Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits*, and will have the following clarification:

- The calibration of instrument circuits as a means to detect age-related degradation of cable insulation as identified in XI.E2 is not included in the VCSNS program. The visual inspection of instrument as well as power and control cables is considered a better means to identify age-related degradation due to localized ambient thermally and radiologically induced stress prior to loss of intended function. The cables addressed by XI.E2 are therefore bounded by the XI.E1 cable aging management program.

This is a new inspection program that will assess the condition of non-EQ insulated cables and connections to provide 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 and terminal blocks, that are subject to degradation in the more 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. The program will involve a visual inspection of the accessible cables in these zones, to determine if the cable jackets or connections show any signs of cracking, embrittlement, discoloration, melting, or any other visible evidence of age-related degradation, which may lead to loss of the intended function.

(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. The program provides for component inspection only.

Issues of component degradation will be addressed within other program attributes.

(3) **Parameters Monitored or Inspected** – The parameters inspected as a part of the Non-EQ Insulated Cables and Connections Inspection Program include visual evidence of cable jacket or connection embrittlement, cracking, swelling, discoloration, melting, loss of dielectric strength leading to reduced insulation resistance and/or electrical failure.

(4) **Detection of Aging Effects** – The Non-EQ Insulated Cables and Connections Inspection Program, conducted in the more thermally and radiologically severe areas of the plant containing in-scope cables and connections will provide input into Monitoring and Trending (see below). These inspections will serve to detect degradation of cable and connections, which could ultimately lead to electrical failure. During each inspection, and visual evidence of embrittlement, cracking, swelling, discoloration, melting, degradation of organics, radiation-induced oxidation, and moisture intrusion will be evaluated.

(5) **Monitoring and Trending** - The Non-EQ Insulated Cables and Connections Inspection Program will be initially performed prior to the period of extended operation and then at 10-year intervals thereafter. Documentation of these inspections will be available in subsequent inspections for comparison, review, and evaluation such that an increase in the degradation of the cable and connections seen by visual means may be monitored, trended as appropriate, and evaluated as input to make a determination of remaining service life.

(6) **Acceptance Criteria** - The Non-EQ Insulated Cables and Connections Inspection Program consists of visual inspections for degradation of cable jackets and connections due to aging. 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 as well as the rate of visually detectable degradation through monitoring and trending from the baseline taken prior to the period of extended operation.

(7) **Corrective Actions** – The VCSNS Corrective Action Program is utilized as applicable to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded condi-

tions 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.

(9) **Administrative Controls** – The Non-EQ Insulated Cables and Connections Inspection Program will be implemented in accordance with controlled station and work processes.

B.2.9.1 Operating Experience

The Non-EQ Insulated Cables and Connections Inspection Program is a new inspection activity for which there is no operating experience. Effective and proven visual inspection techniques will be selected for use in performing the inspections. Lessons learned during the performance of the inspections, experience gained and shared by other utilities, and other inspection 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.

B.2.10 BURIED PIPING AND TANKS INSPECTION

The Buried Piping and Tanks Inspection will be consistent with XI.M34, *Buried Piping and Tanks Inspection*, as identified in NUREG –1801 prior to the period of extended operation.

The Buried Piping and Tanks Inspection is a new inspection activity that will manage loss of material due to crevice, galvanic, general, pitting, and microbiologically influenced corrosion (MIC) on the external surfaces of components exposed to an underground environment. The Buried Piping and Tanks Inspection contains elements of a condition-monitoring program and a prevention program.

(1) **Scope** - The Buried Piping and Tanks Inspection is applicable to carbon steel, cast iron and ductile iron components exposed to an underground environment in the following systems:
- Diesel Generator Services – carbon steel fuel oil pipe and tanks
- Emergency Feedwater - carbon steel pipe
- Fire Service – ductile iron pipe, and cast iron hydrants and valve bodies
- Service Water – carbon steel pipe and couplings

(2) **Preventive Actions** – In accordance with standard industry practice, underground components were coated and wrapped during installation to prevent them from directly contacting the soil environment. Otherwise, no actions are taken as part of the Buried Piping and Tanks Inspection to prevent aging effects or mitigate aging degradation.

(3) **Parameters Monitored or Inspected** – As part of the Buried Piping and Tanks Inspection, the condition of coatings and wrappings will be determined by visual inspection whenever buried components are excavated for maintenance or for other reasons.

(4) **Detection of Aging Effects** – The results of previous inspections, as discussed below under Operating Experience, indicate a very slow (or negligible) rate of wall thinning. Since the process of excavation itself can damage protective coatings and wrappings, a specific inspection frequency for buried components is not warranted. If buried components are excavated for maintenance or for other reasons, the integrity of their coatings and wrappings will be evaluated. If the coatings or wrappings are damaged of removed as part of the maintenance activity the underlying metal will be visually inspected for degradation.

The Buried Piping and Tanks Inspection will detect loss of material prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Buried Piping and Tanks Inspection to trend inspection results. However, the results of an inspection may indicate the need for additional inspections to be performed.

(6) **Acceptance Criteria** - The acceptance criteria for the Buried Piping and Tanks Inspection is 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.

(7) **Corrective Actions** – If the results of the Buried Piping and Tanks Inspection are not acceptable, as determined by engineering evaluation, then corrective actions are taken to repair or replace the affected items. The corrective action includes a determination of whether additional inspections or programmatic oversights are required. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** – The Buried Piping and Tanks Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.10.1 Operating Experience

The Buried Piping and Tanks Inspection is a new inspection activity. There is some operating experience with buried piping and tanks at VCSNS. During the EDSFI evaluation, the Cathodic Protection System was found to provide inadequate protection to the diesel generator fuel oil storage tanks and associated underground piping. As a result, an ultrasonic examination of the fuel oil storage tanks and associated piping was performed. Each tank was inspected at 102 locations, evenly distributed over the entire surface area, and no significant corrosion or age-related degradation was found. The tank inspection indicated a very slow (or negligible) rate of wall thinning. Approximately 35 feet of fuel oil piping was inspected and found to be in good condition with no corrosion identified.

B.2.10.2 Conclusion

The Buried Piping and Tanks Inspection 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.

B.2.11 INSPECTIONS FOR MECHANICAL COMPONENTS

There is no NUREG-1801 item addressing this program. This is a plant specific program.

Inspections for Mechanical Components is a new inspection activity that will manage loss of material due to galvanic, general, and pitting corrosion and cracking due to radiation and thermal embrittlement for the external surfaces of mechanical components within the scope of license renewal that are exposed to ambient conditions. Inspections for Mechanical Components is a condition monitoring program.

(1) Scope - Inspections for Mechanical Components will manage the relevant aging effects for mechanical components constructed of carbon steel, low alloy steel, and other susceptible materials. All or portions of the following mechanical systems contain components/component types subject to the aging effects managed by this program:

- Air Handling (HVAC)
- Auxiliary Boiler Steam and Feedwater
- Auxiliary Coolant (Closed Loop) / CRDM Cooling Water
- Boron Recycle
- Building Services
- Chemical and Volume Control
- Chilled Water
- Component Cooling
- Condensate
- Demineralized Water – Nuclear Service
- Diesel Generator Services
- Emergency Feedwater
- Extraction Steam
- Feedwater
- Fire Service
- Gaseous Waste Processing
- Gland Sealing Steam
- Hydrogen Removal
- Instrument Air Supply
- Liquid Waste Processing
- Local Ventilation and Cooling
- Main Steam
- Main Steam Dump
- Nitrogen Blanketing
- Nuclear Sampling
- Radiation Monitoring
- Reactor Building Leak Rate Testing
- Reactor Building Spray
- Reactor Coolant
- Reactor Makeup Water Supply
- Residual Heat Removal
- Safety Injection
- Service Water
- Spent Fuel Cooling
- Station Service Air
- Steam Generator Blowdown
- Thermal Regeneration

(2) Preventive Actions - No actions are taken as part of Inspections for Mechanical Components to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - Inspections for Mechanical Components involves a visual examination of the exposed external surfaces of mechanical components for loss of material or cracking.

(4) **Detection of Aging Effects** - In accordance with information provided in Monitoring and Trending below, Inspections for Mechanical Components will detect loss of material and cracking prior to loss of component intended function. Pitting is a concern in locations where components are insulated and internal system fluid temperatures are below the ambient temperature conditions.

(5) **Monitoring and Trending** - Inspections of material surface condition will be performed and documented in accordance with station procedures. Following a baseline inspection, the frequency of the inspections will be determined based on inspection results and industry experience.

(6) **Acceptance Criteria** - The acceptance criteria for Inspections for Mechanical Components are no unacceptable visible indication of loss of material or cracking. An indication of a rate of deterioration due to loss of material or cracking that could cause the component to fail its intended function prior to its next scheduled inspection, as determined by engineering evaluation, is considered unacceptable.

(7) **Corrective Actions** - If the results of the inspections for Mechanical Components are not acceptable, as determined by engineering evaluation, then corrective actions are taken to repair or replace the effective components. The corrective action includes determination of whether additional inspections or programmatic oversights are required.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Inspections for Mechanical Components will be implemented in accordance with controlled station procedures and work processes.

B.2.11.1 Operating Experience

The Inspections for Mechanical Components is a new inspection activity. There is VCSNS operating experience with the detection of aging effects on exposed external surfaces of components. An instance of pitting below insulation in the Chilled Water System was identified and repaired. Several instances of leakage in the Chilled Water System have been identified by surveillance procedures. The leakage was evaluated and repaired under the Corrective Action Program.

B.2.11.2 Conclusion

Inspections for Mechanical Components 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.

B.2.12 HEAT EXCHANGER INSPECTIONS

The Heat Exchanger Inspections will be consistent with XI.M32, *One-Time Inspection*, and XI.M33, Selective Leaching of Materials, as identified in NUREG-1801 prior to the period of extended operation.

The Heat Exchanger Inspections is a new one-time inspection activity that will detect and characterize 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. The Heat Exchanger Inspections will be performed prior to the period of extended operation.

(1) **Scope** - The Heat Exchanger Inspections is applicable to copper, copper-nickel, and brass heat exchanger components (as well as brass thermowells) exposed to a treated water environment in various systems. These systems are the Air Handling System, the Component Cooling System, the Chemical and Volume Control System, the Diesel Generator System, the Emergency Feedwater System, the Chilled Water System, and the Local Ventilation and Cooling System.

(2) **Preventive Actions** - No actions are taken as part of the Heat Exchanger Inspections to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected** - The parameters inspected by the Heat Exchanger Inspections are wall thickness as a measure of loss of material, material hardness as a measure of selective leaching, and visual evidence of loss of material, heat exchanger fouling or other age-related degradation.

(4) **Detection of Aging Effects** - The Heat Exchanger Inspections will use a combination of proven volumetric and visual examination techniques at sample locations in the various heat exchangers determined by engineering evaluation to be most susceptible to the applicable aging effects. 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. The inspection will include a Brinnell Hardness Test or an equivalent test on a sample of susceptible components in order to characterize a reduction of material hardness (loss of material) due to selective leaching.

The Heat Exchanger Inspections will detect the presence and extent of any loss of material and heat exchanger fouling prior to a loss of component intended function.

Inspection locations for heat exchange fouling should focus on heat exchanger components having an intended function of heat transfer and which are normally in a standby condition with no flow.

(5) **Monitoring and Trending** - No actions are taken as part of the Heat Exchanger Inspections to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria** - The acceptance criteria for the Heat Exchanger Inspections is no unacceptable loss of material or heat exchanger fouling of subject components that could result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) **Corrective Actions** - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** - Engineering reviews the inspection results for completeness and acceptability. The corrective action 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.

(9) **Administrative Controls** - The Heat Exchanger Inspections will be implemented in accordance with controlled station procedures and work processes.

B.2.12.1 Operating Experience

The Heat Exchanger Inspections is a new one-time inspection for which there is no operating experience.

B.2.12.2 Conclusion

Implementation of the Heat Exchanger Inspections will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.3.0 TLAA SUPPORT ACTIVITIES

B.3.1 ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

The Environmental Qualification (EQ) Program is consistent with X.E1, *Environmental Qualification (EQ) of Electric Components*, as identified in NUREG-1801.

Prior to the period of extended operation, the equipment subject to the provisions of 10 CFR 50.49 will be re-evaluated for 60 years of installation. Parameters to be evaluated include:

- Thermal Considerations - The component qualification temperatures will be evaluated for 60 years using the Arrhenius method. Components not meeting a 60 year qualified life will be replaced prior to expiration of qualified life.
- Radiation Considerations - The bounding 60 year radiation dose qualification values for all EQ components will be compared to the dose values typically established by manufacturers in qualification testing. Any EQ components qualification dose level not enveloping the 60 year radiation environment will be evaluated by available means, shielded, or replaced prior to expiration of its radiation qualified life.
- Wear Cycle Aging Considerations - Wear cycle aging is a factor for some equipment in the VCSNS EQ Program. In cases where wear cycle aging is a credible aging mechanism, wear cycles will be evaluated and/or controlled through the end of the period of extended operation.

B.3.1.1 Operating Experience

The VCSNS EQ Program includes consideration of operating experience, both at VCSNS and in the industry, to modify the qualification bases and conclusions. This includes data on specific components and materials, data on aging limits, and new test data from manufacturers, industry groups, or the NRC. VCSNS is actively involved in the nuclear industry technical groups on EQ, such as NEI, NUGEQ, and various EPRI programs, including the License Renewal Electrical Working Group (under NEI).

B.3.1.2 Conclusion

The EQ Program has been demonstrated to be capable of maintaining the qualification of components within the scope of 10 CFR 50.49. 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.

B.3.2 THERMAL FATIGUE MANAGEMENT PROGRAM

The Thermal Fatigue Management Program is consistent with X.M1, *Metal Fatigue of Reactor Coolant Pressure Boundary*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Thermal Fatigue Management Program prior to the period of extended operation:

NUREG-1801 Program	Attributes	Enhancement
X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary	1.) Scope	Incorporate the new guidance in EPRI Report MRP-47 **[Reference B-12]** for the environmental effects of the reactor coolant environment on fatigue into the VCSNS program.
	6.) Acceptance Criteria	Revise the acceptance criteria to account for the environmental effects on fatigue.

B.3.2.1 Operating Experience

The Thermal Fatigue Management Program includes consideration of operating experience, both at VCSNS and in the industry. Thermal fatigue transients have been tracked since operation began at VCSNS. Operating experience has demonstrated that the program continues to monitor plant transients and track the accumulation of those transients consistent with the requirements in VCSNS Technical Specification 5.7. Industry issues identified after plant start-up have been incorporated into the Thermal Fatigue Management Program.

B.3.2.2 Conclusion

The Thermal Fatigue Management Program has been demonstrated to be capable of managing the thermal fatigue basis to preclude cracking due to thermal fatigue. The Thermal Fatigue Management 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.

B.3.3 TENDON SURVEILLANCE PROGRAM

The Tendon Surveillance Program is consistent with X.S1, *Concrete Containment Tendon Prestress*, as identified in NUREG-1801.

B.3.3.1 Operating Experience

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%.
- 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.

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.

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 surveillance reports for the past three surveillance periods [Fourth (1990), Fifth (1996) and Sixth (2000)] have each concluded that no abnormal degradation of the post tensioning system has occurred at VCSNS.

B.3.3.2 Conclusion

The Tendon Surveillance Program has been demonstrated to be capable of maintaining the Reactor Building dome, vertical, and hoop tendons above the minimum required prestressing forces. The Tendon Surveillance 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.

B.4.0 REFERENCES

B-1	NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, April, 2001.
B-2	NUREG-1801, Generic Aging Lessons Learned (GALL) Report, April 2001.
B-3	SCE&G Letter RC-01-0155, S.A. Byrne to USNRC Document Control Desk, Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, August 31, 2001.
B-4	SCE&G Letter RC-02-0055, S.A. Byrne to USNRC Document Control Desk, Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, April 3, 2002.
B-5	NRC Inspection Report (IR) 50-395 / 98-01, March 20, 1998 for VCSNS.
B-6	NRC Inspection Report (IR) 50-395 / 93-09, April 21, 1993 for VCSNS.
B-7	Westinghouse Topical Report WCAP-14422, Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports, Revision 2-A, December 2000.
B-8	SCE&G Letter RC-98-0207, G.J. Taylor to USNRC Document Control Desk, Response to Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, November 11, 1998.
B-9	NRC Inspection Report (IR) 50-395 / 82-28, April 19, 1982.
B-10	SCE&G Letter dated January 31, 1990, Response to Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment.
B-11	Letter, Christopher Grimes (NRC) to Douglas Walters(NEI), License Renewal Issue No. 98-0085, "Reactor Vessel Surveillance Program", December 3, 1999.
B-12	EPRI Final Report MRP-47, Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47), Revision 0, October 2001.
B-13	NRC letter to G. J. Taylor (dated February 27, 1998), Service Water Pond Dam Safety Inspection Results for Virgil C. Summer Nuclear Station.

B-14	USNRC Letter NL-356-99, K.R. Cotton to G.J. Taylor, Generic Letter 97-01, "Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations": Review of the Responses for the Virgil C. Summer Nuclear Station, December 17, 1999.

APPENDIX C - COMMODITY GROUPS (OPTIONAL)

Appendix C is not being used in the Application to Renew the Operating License of the Virgil C. Summer Nuclear Station.

APPENDIX D - TECHNICAL SPECIFICATION CHANGES

10 CFR 54.22 requires that an application for license renewal include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation. No changes to the Virgil C. Summer Nuclear Station Technical Specifications are necessary in that regard.

APPENDIX E - ENVIRONMENTAL REPORT

The Environmental Report for Virgil C. Summer Nuclear Station is contained in a separate document entitled "**Environmental Report for License Renewal, Virgil C. Summer Nuclear Station**."