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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) [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) [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 18.1-1 provides a summary listing of the aging management programs and activities required for license renewal. Furthermore, Table 18.1-2 provides a summary listing of the evaluations of time-limited aging analyses (TLAA) required for license renewal. The first column of Table 18.1-1 and Table 18.1-2 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.

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

The implementing documents are subject to administrative controls, including a formal review and approval process, are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in the FSAR.

For all programs a condition evaluation report (CER) is generated to provide a thorough description of identified aging problems along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions. 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. The license renewal programs and inspections are implemented in accordance with station procedures and work processes controlled by the quality assurance requirements of 10 CFR Part 50 Appendix B.

TABLE 18.1-1

SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Program/Activity	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
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.1.20	18.2.24

TABLE 18.1-1 (Continued)

SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Program/Activity	Application Location	FSAR Location
Preventive Maintenance Activities – Ventilation Systems Inspections	B.1.26	18.2.25
Reactor Building Cooling Unit Inspection	B.2.5	18.2.26
Reactor Vessel 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 Fatique Management Program	B.3.2	18.2.37
Underwater Inspection Program (SWIS and SWPH)	B.1.23	18.2.38
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
Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria	B.2.13	18.2.42
Aging Management Program for Electrical Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements Used in Instrumentation Circuits		18.2.43
License Renewal Aging Management Program for Non-EQ Electrical Cables Used in Instrumentation Circuits		18.2.44
Aging Management Program for Inaccessible Medium-Voltage Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements		18.2.45

TABLE 18.1-2

SUMMARY LISTING OF THE TLAA EVALUATIONS FOR LICENSE RENEWAL

Time Limited Aging Analysis	Application Location	FSAR Location
Crane Load Cycle Limit	4.7.2	18.3.6.1
Environmental Qualification	4.4	18.3.3
Metal Fatigue - ASME Section III, Class 1	4.3.1	18.3.2.1
Metal Fatigue - Leak-Before-Break Analysis	4.3.2	18.3.2.2
Metal Fatigue - ASME Section III, Class 2 and 3 Piping Fatigue	4.3.4	18.3.2.3
Reactor Building Liner	4.5.2	18.3.5
Reactor Building Tendon Prestress	4.5.1	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 – Pressurizer 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
Service Water Intake Structure Settlement	4.7.3	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 ambient 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 In-service 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 In-service 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 for the period of extended operation. The ASME Section XI ISI Program - IWF was developed to implement the applicable requirements of 10 CFR 50.55a. This program is in accordance with ASME Section XI, Subsection IWF, 1989 Edition. 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. 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. RN 18-019

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.

The Boric Acid Corrosion Surveillances program relies on implementation of Nuclear Regulatory (NRC) Generic Letter (GL) 88-05, as well as subsequent NRC bulletins and guidance, to monitor the reactor coolant pressure boundary for borated water leakage. As a result of guidance in NUREG-1801, the program also includes monitoring for borated water leakage in all systems containing borated water.

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, specifically, EPRI TR105714 and EPRI TR102134. 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 for Class CC Components for Light-Water Cooled 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 action. This program is in accordance with ASME Section XI, Subsections IWE and IWL, 1992 Edition including 1992 Addenda.

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 and exhaust 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 Section 9.5.1. 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 Section 9.5.1, the Fire Protection Program includes the requirements 10 CFR 50.48 (c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition." Additional description of the portions of the program pertinent to the management of aging is provided below.

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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 to 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

Fires 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 judgment, 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. 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 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 these zones, to determine if the cable jackets show any signs of cracking, embrittlement, discoloration, crazing, melting, or any other visible evidence of age-related degradation which might indicate loss of insulation resistance.

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 in-service 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. This program is implemented in accordance with NUMARC 93-01, Rev. 2 and Regulatory Guide 1.160, Rev. 2. Visual inspections and condition assessments of structures and structural components are conducted in accordance with the requirements of the Maintenance Rule. These inspections include penetrations and associated piping at structural interfaces, which may be subject to degradation mechanisms (including microbiologically induced corrosion). 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 and Service Water Pond) 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.

18.2.25 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 ambient air. Routine maintenance inspections are conducted which include detection of age-related degradation.

18.2.26 REACTOR BUILDING COOLING UNITS INSPECTION

The Reactor Building Cooling Units 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 VESSEL 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 was performed to ensure that materials in the upper shell/nozzle course of the Reactor Vessel would 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. This program is based on the recommendations of Regulatory Guide 1.127, Rev. 1. 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 also 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 Pumphouse 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 to ensure that cracking and loss of material of nickel-based alloy steam generator tubes and tube plugs are identified and corrected prior to exceeding allowable limits. The program implements the requirements of Technical Specification 4.4.5 for tube inspections. Other components, in addition to tubes, are inspected under this program. The program follows the recommendations provided by NEI 97-06 and associated EPRI guidelines for Steam Generator component inspections and chemistry controls.

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 50.55a(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 due to greater than expected settlement of the structure. Underwater inspections of the Service Water Intake Structure and the Service Water Pumphouse 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 INSPECTION

The Heat Exchanger Inspection is 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 Inspection will detect and characterize loss of material due to selective leaching and erosion-corrosion, as well as particulate fouling. The Heat Exchanger Inspection 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 general corrosion of the turbine casing and associated components in the steam inlet and exhaust paths 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.

18.2.42 AREA BASED INSPECTIONS FOR REFINED 10 CFR 54.4(A)(2) CRITERIA

The Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria is a new one-time inspection activity that will determine if aging management is required for components that have an anti falldown requirement during the period of extended operation. The Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria will detect and characterize loss of material or cracking resulting from exposure to an unmonitored and uncontrolled water environment. The Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria will use volumetric and/or visual examination techniques at the most susceptible (sample) locations in the piping in carbon steel piping anti falldown systems with uncontrolled water.

18.2.43 AGING MANAGEMENT PROGRAM FOR ELECTRICAL CABLES NOT SUBJECT TO 10CFR50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

The *"Aging Management Program for Electrical Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements Used in Instrumentation Circuits"* provides for reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are used in instrumentation circuits with sensitive, low-level signals exposed to adverse localized environments caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation. In this aging management program, calibration results or findings of surveillance testing programs are used to identify the potential existence of aging degradation. This program applies to the in-scope instrumentation cables that are included in the circuit during loop calibrations.

This program will be implemented as appropriate at least once every 10 years. The initial implementation will be completed prior to the period of extended operation.

18.2.44 LICENSE RENEWAL AGING MANAGEMENT PROGRAM FOR NON-EQ ELECTRICAL CABLES USED IN INSTRUMENTATION CIRCUITS

High-range-radiation and neutron flux monitoring instrumentation cables used in high voltage, low-level signal applications that are sensitive to reduction in insulation resistance for which calibrations are performed without the cable in the "loop calibration" circuit will be subject to the *"License Renewal Aging Management Program for Non-EQ Electrical Cables Used in Instrumentation Circuits"*. In this program, an appropriate test, such as an insulation resistance test, will be used to identify the potential existence of a reduction in cable insulation resistance (IR). This program is an acceptable alternative to aging management program 18.2.43.

This program will be implemented at least once every 10 years. The initial implementation will be completed prior to the period of extended operation.

18.2.45 AGING MANAGEMENT PROGRAM FOR INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10CFR50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The purpose of this aging management program is to provide reasonable assurance that the intended functions of inaccessible medium-voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of extended operation.

In this aging management program periodic actions are taken to prevent or minimize the possibility that cables may be exposed to moisture, such as inspecting for water collection in cable manholes and conduit, and draining water, as needed. In-scope, medium-voltage cables exposed to moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The initial test performed will be determined and completed prior to the period of extended operation, and is to be a proven test for detecting deterioration of the insulation system. In-scope, medium-voltage cables exposed to moisture and significant voltage are tested at least once every 10 years.

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.

The upper shelf energy (USE) for VCSNS reactor vessel materials have been estimated for the end of the 60 year period of extended operation, 56 Effective Full Power Years (EFPY). The estimates were made using Figure 2 from Regulatory Guide 1.99, Revision 2, the estimated fluence and chemistry of the materials and the Charpy test data from surveillance capsules. The fluence projections were obtained from WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the SCE&G V. C. Summer Radiation Surveillance Program", August 2004. The USE values are developed in WCAP-16306-NP, Revision 0, "Evaluation of Pressurized Thermal Shock for V. C. Summer", August 2004, at ¼ the reactor vessel wall thickness. The calculated USE values for the limiting reactor vessel material is above 50 ft-lb for 56 EFPY. RN 09-008

18.3.1.2 Pressurized Thermal Shock

The requirements of 10 CFR 50.61 provide rules 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 60 year period of extended operation, 56 EFPY. The projected RT_{PTS} values are based upon surveillance capsule data. The last capsule removed from the reactor vessel was exposed to a neutron fluence estimated to be equivalent to that exposed to the vessel wall at the end of 60 years. The RT_{PTS} fracture toughness of the reactor vessel beltline region has been evaluated according to the requirements of the Pressurized Thermal Shock (PTS) rules of 10 CFR 50.61. Using the PTS rules and surveillance capsule data, the evaluation is documented in WCAP-16306-NP, Revision 0, "Evaluation of Pressurized Thermal Shock for V. C. Summer ", August 2004. This evaluation shows maximum calculated RT_{PTS} values at 56 EFPY for the most limiting plate/axial weld and for the most limiting circumferential weld to be well below the screening criteria of 270° F and 300° F, respectively, as specified in the PTS rule. RN 09-008

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 developed for 56 EFPY in WCAP-16305-NP, Revision 0, "V. C. Summer Heatup and Cooldown Limit Curves for Normal Operation", August 2004, that corresponds to the end of the 60 year extended operational period. These curves are based upon data from the last capsule removed from the reactor vessel which was exposed to a neutron fluence estimated to be equivalent to that exposed to the vessel wall at the end of 60 years. The Technical Specification has been updated to include these curves as required by 10 CFR 50. Therefore, the curves that are now in the Technical Specifications are valid for the period of extended operation. RN 09-008

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 Boiler and Pressure Vessel Code, Section III, Class 1

The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and to 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 their 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 transients used in the design.

Prior to the end of year 40, the fatigue monitoring program will be revised to account for environmentally assisted fatigue using a Staff approved methodology. The present Staff approved methodology re-computes the CUF for the locations listed in NUREG/CR-6260 using the interim fatigue curves found in NUREG/CR-6583 (for carbon and low alloy steels) and NUREG/CR-5704 (for stainless steel).

Component CUF will be maintained below 1.0. If predictions show CUF could exceed 1.0, further action is required for management of environmental fatigue of the components for the period of extended operation. Fatigue will be managed using one or more of the following options:

1. CUF below 1.0.
2. Further refinement of the fatigue analyses to maintain the environmentally assisted fatigue CUF below 1.0.
3. Repair of the affected locations.
4. Replacement of the affected locations.
5. Manage the effects of fatigue through the use of an augmented in-service inspection program that has been reviewed and approved by the NRC. If this option is selected, the scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to the CUF exceeding 1.0.

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 Boiler and Pressure Vessel Code, Section III, Class 2 and 3

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 are required on the "B" Reactor Coolant System loop sampling line in order to account for 60 years of plant operation.

18.3.3 ENVIRONMENTAL QUALIFICATION

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 as time-limited aging analyses.

Equipment included in the Virgil C. Summer 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 Virgil C. Summer 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 additional retensioning is required before the next scheduled inspection based on the state of the tendon stress. Therefore reactor building tendon prestress is a time-limited aging analysis.

The 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 liner 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 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 have 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 (CMAA) 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 Virgil C. Summer.

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

1. RC-02-0123, Letter from SCE&G (S. A. Byrne) to NRC (Document Control Desk) dated August 6, 2002, "APPLICATION FOR RENEWED OPERATING LICENSE".

2. NUREG-1787 Safety Evaluation Report Related to the License Renewal of the Virgil C. Summer Nuclear Station Docket No. 50-395.

3. Fire Protection (FP) DBD RN 18-019

4. RC-02-0159, Letter from SCE&G (S. A. Byrne) to NRC (Document Control Desk) dated September 12, 2002, "Criteria 2 Supplement to the Application for Renewed Operating License".