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SCE&G®
A SCANA COMPANY

November 5, 2003
RC-03-0227

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

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
COMMENTS ON THE SAFETY EVALUATION REPORT
RELATED TO THE LICENSE RENEWAL OF THE
VIRGIL C. SUMMER NUCLEAR STATION

Attachment: Safety Evaluation Report (WITH COMMENTS ANNOTATED)

Attached you will find comments and edits on the Safety Evaluation Report related to the License Renewal of the Virgil C. Summer Nuclear Station (VCSNS).

Additionally, the Safety Evaluation Report related to the License Renewal of the Virgil C. Summer Nuclear Station, October 2003, Appendix A, Commitment 20 states:

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

VCSNS originally intended to establish operating restrictions for control of vessel fluence. VCSNS has since reconsidered the use of operating restrictions and is leaving one of the two remaining capsules in the vessel for one additional cycle. During RF-15, VCSNS intends to remove the last remaining capsule and place it in storage for possible future use. Also during RF-15, VCSNS intends to install alternative dosimetry to monitor vessel fluence. These changes should be reflected in the four areas of the SER: Sections 3.1.2.3.6 (page 3-89); 4.2.1.1 (page 4-4); 4.2.3.2 (page 4-10); and Appendix A, Commitment 20 (page A-6).

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Also note that the Safety Evaluation Report related to the License Renewal of the Virgil C. Summer Nuclear Station, October 2003, Appendix A, Commitment 37 has been met. The Boraflex neutron absorbing sheets have been replaced with Boral neutron absorbing sheets.

Should you have any questions, please call Mr. Al Paglia at (803) 345-4191.

I certify under penalty of perjury that the information contained herein is true and correct.

11/5/03
Executed on

PER DIRECTION OF S. BYRNE 11/5/03
Stephen A. Byrne

AMP/SAB/mbb
w/o Attachment (unless noted)

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DMS (RC-03-0227) (w/ attachment)

# Safety Evaluation Report

## Related to the License Renewal of the Virgil C. Summer Nuclear Station

Docket No. 50-395

South Carolina Electric & Gas Company (SCE&G)

SCE+G COMMENTS / EDITS
(AFFECTED PAGES ONLY)

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555-0001

October 2003

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# ABSTRACT

This safety evaluation report (SER) documents the technical review of the Virgil C. Summer Nuclear Station (VCSNS), license renewal application (LRA) by the U.S. Nuclear Regulatory Commission staff. By letter dated August 6, 2002, South Carolina Electric & Gas Company (SCE&G or the applicant) submitted the LRA for VCSNS in accordance with Title 10 of the *Code of Federal Regulations,* Part 54 (10 CFR Part 54 or the Rule). SCE&G is requesting renewal of the operating license for VCSNS (License No. NPF-12) for a period of 20 years beyond the current license expiration of midnight, August 6, 2022.

The VCSNS plant is located in Fairfield County, in predominantly rural north-central South Carolina. It is situated on the shore ~~of the~~ Monticello Reservoir about 42 kilometers (26 miles) northwest of Columbia, the State capitol. The VCSNS unit consists of a Westinghouse pressurized-water reactor with nuclear steam supply system designed to operate at core power levels up to 2900 megawatts-thermal, or approximately 966 megawatts-electric. Details concerning the plant and the site are found in the Updated Final Safety Analysis Report (UFSAR) for VCSNS.

This SER presents the status of the staff's review of information submitted to the NRC through September 26, 2003, the cutoff date for consideration in the SER. The staff has identified open items that must be resolved before the staff can make a determination on the application. These items are summarized in Section 1.5 of this report. In order to close these items, the staff requires the additional information identified. The staff will present its final conclusion on the review on the VCSNS application in its update to this SER.

The NRC VCSNS license renewal project manager is Rajender Auluck. Dr. Auluck may be reached at (301)-415-3936. Written correspondence should be addressed to the License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rajender Auluck, Mail Stop O-11F1.

NEED POSITIVE STATEMENTS THAT THERE ARE NO OPEN ITEMS.

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# ABBREVIATIONS

| | |
|---|---|
| AC | auxiliary coolant |
| ACI | American Concrete Institute |
| ACRS | Advisory Committee on Reactor Safeguards |
| ACSR | aluminum conductor steel reinforced |
| AFW | auxiliary feedwater |
| AISC | American Institute of Steel Construction |
| AMP | aging management program |
| AMR | aging management review |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| APCSB | auxiliary and power conversion systems branch |
| ARAVS | auxiliary and radwaste area ventilation system |
| ART | adjusted reference temperature |
| AS | auxiliary system |
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society for Testing and Materials |
| ATWS | anticipated transient without scram |
| AVB | antivibration bar |
| AWWA | American Water Works Association |
| BMI | bottom-mounted instrumentation |
| BS | building services |
| BSS | building services system |
| BTP | branch technical position |
| BTRS | (boron) thermal regeneration system |
| B&W | Babcock and Wilcox Co. |
| BWR | boiling-water reactor |
| CASS | cast austenitic stainless steel |
| CBAVS | control building area ventilation systems |
| CC | component cooling |
| CCW | component cooling water |
| CCWS | component cooling water system |
| CER | condition evaluation report |
| CFR | *Code of Federal Regulations* |
| CHAMPS | component history and maintenance planning system |
| CLB | current licensing basis |
| CRD | control rod drive |
| CRDM | control rod drive mechanism |
| CREP | control room evacuation panels |
| CS | carbon steel |
| CUF | cumulative usage factor |

| | |
|---|---|
| CVCS | chemical and volume control system |
| CW | circulating water |
| DBD | design basis document |
| DBE | design basis event |
| DN | demineralized water - nuclear services |
| DOJ | U.S. Department of Justice |
| DW | demineralized water |
| ECCS | emergency core cooling system |
| ECT | eddy current testing |
| EDG | emergency diesel generator |
| EFPY | effective full power year |
| EOF | emergency offsite facility |
| EOL | end of life |
| EPR | ethylene propylene rubber |
| EPRI | Electric Power Research Institute |
| EQ | environmental qualification ~~or emergency equipment~~ |
| EQDB | equipment qualification database |
| ES | engineering services |
| ESF | engineered safety features |
| FAC | flow-accelerated corrosion |
| FERC | Federal Energy Regulatory Commission |
| FHA | fire hazard analysis |
| FHBV | fuel handling building ventilation |
| FMP | fatigue monitoring program |
| FO | fuel oil handling |
| FP | fire protection |
| FPP | fire protection program |
| FRP | fiberglass reinforced plastic |
| FSAR | final safety analysis report |
| FSER | final safety evaluation report |
| GALL | generic aging lessons learned |
| GDC | general design criterion |
| GEIS | generic environmental impact statement |
| GL | generic letter |
| gpm | gallons per minute |
| GSI | generic safety issue |
| GSSS | gland sealing steam system |
| GWPS | gaseous waste processing system |
| HAZ | heat-affected zone |

| | |
|---|---|
| HEI | heat exchanger inspections |
| HELB | high energy line break |
| HEPA | high efficiency particulate air |
| HMWPE | high molecular weight polyethylene |
| HN | hydrogen-nuclear plant use |
| HV | high voltage |
| HVAC | heating, ventilation, and air conditioning |
| IBVS | intermediate building ventilation systems |
| IA | instrument air |
| IASCC | irradiation-assisted stress corrosion cracking |
| I&C | instrumentation and controls |
| ID | inside diameter |
| IEB | Inspection and Enforcement Bulletin |
| IEEE | Institute of Electrical and Electronics Engineers |
| IGA | intergranular attack |
| IGSCC | intergranular stress corrosion cracking |
| IN | information notice |
| INPO | Institute of Nuclear Power Operations |
| IPA | integrated plant assessment |
| IR | insulation resistance |
| ISG | Interim staff guidance |
| ISI | inservice inspection |
| ITG | issues task group or Industry |
| LBB | leak before break |
| LOCA | loss-of-coolant accident |
| LR | reactor building leak rate testing |
| LR | license renewal |
| LRA | license renewal application |
| LOTP | low-temperature overpressure protection |
| LTOPS | low temperature overpressure protection system |
| LW | liquid effluent from nuclear plant to pent stock |
| LWPS | liquid waste procession system |
| MBVCS | miscellaneous building ventilation and cooling systems |
| MCC | motor control center |
| MIC | microbiologically induced or ~~induced~~ influenced corrosion |
| mm | millimeter |
| $M_oS_2$ | molybdenum disulfide |
| MRP | Materials Reliability Project |
| MS | main steam |
| MSIP | mechanical stress improvement process |
| MSIV | main steam isolation valve |
| MT | magnetic particle technique |

| | |
|---|---|
| MWe | megawatt-electric |
| NB | nuclear blowdown processing |
| NCN | nonconformance notice |
| ND | nuclear plants drains |
| NDE | nondestructive examination |
| NDT | nil ductility temperature |
| NEI | Nuclear Energy Institute |
| NFPA | National Fire Protection Association |
| NG | nitrogen blanketing |
| NN | nitrogen-nuclear plant use |
| NPS | nominal pipe size |
| NRC | Nuclear Regulatory Commission |
| NSSS | nuclear steam supply system |
| OBE | operating-basis earthquake |
| ODSCC | outer diameter stress corrosion cracking |
| ON | oxygen-nuclear plant use |
| P&ID | piping and instrumentation diagram |
| PORV | power operated relief valve |
| PTS | pressurized thermal shock |
| P-T | pressure-temperature |
| PWR | pressurized water reactor |
| PWSCC | primary water stress corrosion cracking |
| QA | quality assurance |
| RAI | request for additional information |
| RBCFS | reactor building cooling and filtering systems |
| RBCU | reactor building cooling units |
| RC | reactor coolant |
| RCCA | rod cluster control assembly |
| RCDT | reactor coolant drain tank |
| RCP | reactor coolant pump |
| RCPB | reactor coolant pressure boundary |
| RCS | reactor coolant system |
| RF | refueling outage |
| RG | regulatory guide |
| RHR | residual heat removal |
| RPV | reactor pressure vessel |
| $RT_{PTS}$ | reference temperature for pressurized thermal shock |
| $RT_{NDT}$ | reference nil ductility transition temperature |
| RV | reactor vessel |

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|---|---|
| RVI | reactor vessel internals |
| RVID | Reactor Vessel Integrity Database |
| RW | raw water |
| RWST | refueling water storage tank |
| | |
| SA | service air |
| SAS | safety assessment system |
| SBO | station blackout |
| SC | structure and component |
| SCC | stress corrosion cracking |
| SCE&G | South Carolina Electric and Gas Company |
| SE | sewer or safety evaluation |
| SER | safety evaluation report |
| SG | steam generator |
| SI | safety injection |
| SMP | structures monitoring program |
| SOC | Statements of Consideration |
| SPC | steam and power conversion |
| SPCS | steam and power conversion systems |
| SRP | Standard Review Plan |
| NO SPACE SRP-LR | Standard Review Plan — License Renewal |
| | |
| SS | stainless steel |
| SSC | structure system and component |
| SSE | safe shutdown earthquake |
| SW | service water |
| SWIS | service water intake structure |
| SWPH | service water pump/house |
| | |
| TDR | time domain reflectometry |
| TFMP | Thermal Fatigue Management Program |
| TLAA | time-limited aging analysis |
| TR | topical or technical report |
| TS | technical specification |
| | |
| UFSAR | updated final safety analysis report |
| USE | upper shelf energy |
| USI | unresolved safety issue |
| UT | ultrasonic testing |
| UV | ultraviolet |
| | |
| VCSNS | Virgil C. Summer Nuclear Station |
| VT | visual test |
| | |
| WCAP | Westinghouse Commercial Atomic Power |

| | |
|---|---|
| WD | radwaste solidification and solids handling |
| WE | waste evaporation |
| WG | waste gas |
| WL | liquid waste processing |
| WOG | Westinghouse Owner's Group |
| WPAA | west penetration access area |
| WX | excess liquid waste |
| XLPE | cross-linked polyethylene |

# 2 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW, AND IMPLEMENTATION RESULTS

This section documents the staff's review of the methodology used by the applicant to identify structures, systems, and components (SSCs) that are within the scope of the Maintenance Rule - 10 CFR 50.65 (Rule), and to identify structures and components (SCs) that are within the scope of the Rule and are subject to an aging management review (AMR). SCs subject to an AMR are those that perform an intended function, as described in 10 CFR 54.4, and meet two criteria:

1.9 They perform such functions without moving parts or without a change in configuration or properties, as set forth in 10 CFR 54.21(a)(1)(i) (denoted as "passive" SCs).

1.10 They are not subject to replacement based on a qualified life or specified time period, as set forth in 10 CFR (a)(1)(ii) (denoted as "long-lived" SCs).

The identification of the SSCs within the scope of license renewal is called "scoping." For those SSCs within the scope of license renewal, the identification of "passive," "long-lived" SCs that are subject to an AMR is called "screening."

The staff's review of the scoping and screening methodology is presented in Section 2.1 of this SER. The staff's review of the results of the implementation of the scoping and screening methodology is presented in Sections 2.2 through 2.5 of this SER.

By letter dated August 6, 2002, the applicant submitted its request and application for renewal of the operating license for V.C. Summer Nuclear Station (VCSNS). As an aid to the staff during the review, the applicant provided evaluation boundary drawings that identify the functional boundaries for systems and components within the scope of license renewal. These evaluation boundary drawings are not part of the license renewal application (LRA).

On March 28, 2003, the staff issued final requests for additional information (F-RAIs) regarding the applicant's methodology for identifying SSCs at VCSNS that are within the scope of license renewal and subject to an AMR, and the results of the applicant's scoping and screening process. By letter dated June 12, 2003, the applicant provided responses to the F-RAIs.

The staff conducted a scoping and screening inspection from June 23 – 27, 2003, to examine activities that supported the LRA, including the inspection of procedures and representative records and interviews with personnel regarding the process of scoping and screening plant equipment to select SSCs within the scope of the Rule and subject to an AMR. The results of the team inspection are contained in Inspection Report 50-395/03-07, dated June 13, 2003. On this basis, the U.S. Nuclear Regulatory Commission (NRC) staff concluded that the applicant's scoping and screening process was successful in identifying those SSCs required to be considered for aging management. In addition, for a sample of plant systems, the inspection team performed visual examinations of accessible portions of the systems to observe any effects of equipment aging. Finally, the inspection concluded that the scoping and screening portion of the applicant's license renewal activities were conducted as described in the LRA and that documentation supporting the application is in an auditable and retrievable form.

## 2.1 Scoping and Screening Methodology

### 2.1.1 Introduction

Part 54 of Title 10 of the *Code of Federal Regulations*, (CFR), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Section 54.21, "Contents of Application -- Technical Information," requires that each application for license renewal contain an integrated plant assessment (IPA). Furthermore, the IPA must list and identify those SCs that are subject to an AMR from the SSCs that are within the scope of license renewal in accordance with 10 CFR 54.4.

In Section 2.1, "Scoping and Screening Methodology," of the LRA, the applicant described the scoping and screening methodology used to identify SSCs for the VCSNS that are within the scope of license renewal and SCs that are subject to an AMR, pursuant to 10 CFR 54.21(a)(1). The staff reviewed the applicant's scoping and screening methodology to determine if it meets the scoping requirements stated in 10 CFR 54.4(a) and the screening requirements stated in 10 CFR 54.21. In developing the methodology, the applicant considered the requirements of the Rule, including the statements of consideration and the guidance presented by the Nuclear Energy Institute (NEI), "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision 3, March 2001 (NEI 95-10). In addition, the applicant also considered the NRC staff's correspondence with other applicants and with NEI in the development of this methodology.

### 2.1.2 Summary of Technical Information in the Application

In Sections 2.0 and 3.0 of the LRA, the applicant provides the technical information required by 10 CFR 54.21(a). In Section 2.1, "Scoping and Screening Methodology," the applicant describes the process used to identify the SSCs that meet the license renewal scoping criteria under 10 CFR 54.4(a), as well as the process used to identify the SCs that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

Additionally, Section 2.2, "Plant Level Scoping Results," Section 2.3, "System Scoping and Screening Results: Mechanical," Section 2.4, "Structures and Structural Components Scoping and Screening Results," and Section 2.5, "Scoping and Screening Results: Electrical and Instrumentation and Control," of the LRA amplify the process that the applicant used to identify the SCs that are subject to an AMR. Chapter 3 of the LRA, "Aging Management Review," contains the following information—Section 3.1, "Aging Management of Reactor Vessel, Internals, and Reactor Coolant System," Section 3.2, "Aging Management of Engineered Safety Features," Section 3.3, "Aging Management of Auxiliary Systems," Section 3.4, "Aging Management of Steam and Power Conversion Systems," Section 3.5, "Aging Management of Containments, Structures, and Component Supports," and Section 3.6, "Aging Management of Electrical and Instrumentation and Controls." Chapter 4, "Time-Limited Aging Analysis," contains the applicant's identification and evaluation of time-limited aging analyses (TLAAs).

#### *2.1.2.1 Scoping Methodology*

The IPA scoping process used by the applicant was performed for both plant and system level scoping. The first step was the identification of all plant systems and structures as described in

function. Technical information related to scoping activities was then incorporated into the technical report in accordance with VCSNS procedures.

Structures Scoping

Structures at VCSNS are classified as either nuclear or non-~~nuclear~~ safety-related. The safety-related structures are designed to withstand the safe shutdown earthquake (SSE) and are classified as Seismic Category I, while the non-safety-related structures, generally not designed to SSE seismic levels, are classified as non-seismic. The classification of each structure has been previously determined and documented in the UFSAR. A listing of structures within the scope of license renewal is located in Section 2.2 of the LRA. All non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii) are also within the scope of license renewal. Two types of systems and structures must be considered for inclusion within the scope of license renewal per 10 CFR 54.4(a)(2) — (1) Non-safety-related systems and structures, and non-safety-related portions of safety-related systems and structures whose physical failure could damage equipment that is performing a safety function and prevent it from performing that function and (2) non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). Structural supports that are considered to meet seismic or anti-falldown criteria or code break criteria are within the scope of license renewal. These are not included in the mechanical system scoping and screening but are treated as a structural commodity.

Electrical and Instrumentation and Control Component Scoping

Electrical components at VCSNS are classified as either Class 1E, as defined in Institute of Electrical and Electronics Engineers (IEEE) IEEE-380, "Definitions of Terms Used in IEEE Standards on Nuclear Power Generating Stations," or as non-nuclear safety. Class 1E is the safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment. These functions are the electrical equivalent to the functions specified in the scoping criteria in 10 CFR 54.4(a)(1). All electrical systems that contain equipment classified as Class 1E are considered to be safety-related and are within the scope of license renewal. Class 1E equipment is identified through a review of the VCSNS component database. The listing of electrical systems and components required for compliance with 10 CFR 54(a)(1) is found in Section 2.2 of the LRA. Electrical systems and portions of electrical systems that are non-safety-related but whose failure could prevent the satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), and (iii) are within the scope of license renewal pursuant to 10 CFR 54.4(a)(2). The electrical equipment and components that perform these functions are designated as quality-related and are identified as such in the VCSNS equipment database.

*2.1.2.2 Screening Methodology*

Following the determination of SSCs within the scope of license renewal, the applicant implemented a process for determining which SCs, contained in the SSCs which were determined to be within scope, would be subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1). In Section 2.1.2, "Screening Methodology," of the LRA,

the applicant discussed these screening activities as they relate to the in-scope SSCs. The specific screening activities for the various engineering disciplines are further described in the application in Section 2.1.2.1 for mechanical systems, Section 2.1.2.2 for structures, and Section 2.1.2.3 for electrical and instrumentation and control (I&C) components. These screening activities consisted of the identification of passive components, long-lived components, component intended functions, consumables, and component replacement based on performance or condition. The applicant relied on the guidance in Appendix B to NEI 95-10 and Chapter 2 of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," April 2001 (SRP-LR), to develop the plant-specific listing of passive components of interest during the review.

2.1.2.2.1 Mechanical Component Screening

Following component level scoping for mechanical systems, the applicant performed screening, in accordance with Section 2.1.2.1 of the LRA, to identify those mechanical components that were subject to an AMR. The applicant stated that the mechanical screening process was implemented on each of the systems that were identified during the scoping review phase to identify the passive mechanical components that support one or more of the system's intended functions. The system's intended functions, in conjunction with component information in CHAMPS, regulated event reports, and the applicable system drawings, have been used to identify the passive components within the scope of license renewal. For mechanical systems, the screening process is performed on each system identified to be within the scope of license renewal. The process includes the establishment of system evaluation boundaries, determination of components within those boundaries, the identification of component intended functions, the determination of components subject to an AMR, and the identification of commodity groups (material and environment identification). Mechanical system evaluation boundaries are established for each system within the scope of license renewal to assure that all components required to support system intended functions, which meet the requirements of 10 CFR 54.4(a)(1, 2 and/or 3), are considered for an AMR. These boundaries are determined by mapping the flow paths, including pressure boundary, that are necessary for the accomplishment of identified system intended functions onto the system flow diagrams or other drawings, such as UFSAR figures. The mechanical components found within the mapped portions of these boundary drawings comprise the complete set of mechanical components within the scope of license renewal. A menu listing all passive long-lived mechanical components or component groupings was developed based on the guidance in Appendix B to NEI 95-10. The components within the mapped areas of the license renewal evaluation boundary diagrams for each system are compared to the menu as a step in listing the components that are subject to an AMR.

A list of potential mechanical component intended functions is then developed for each grouping of the components within the mechanical evaluation boundaries. In accordance with 10 CFR 54.21(a)(1), component intended functions are those component level functions that are performed without moving parts or without a change in configuration or properties in support of identified system intended functions. The result is a list of the potential intended functions for each passive long-lived component type. Each mechanical component or component group (commodity) within the license renewal evaluation boundaries is reviewed to determine whether the potential intended functions must be performed by that component to meet the requirements that are necessary to ensure that the identified system intended

performed a screening review to determine which electrical components would be subject to an AMR. As part of this effort, the applicant relied on the requirements set forth in 10 CFR 54.21(a)(1)(i) and (ii) as supplemented by industry guidance to identify component intended functions for each electrical commodity group. All of the other electrical and I&C commodities identified are either active, subject to replacement based on a qualified life or specified time period, or do not perform any intended functions and therefore, are not subject to an AMR pursuant to 10 CFR 54.21(a)(1)(i) and industry guidance. The electrical screening results are presented in the LRA which provides a description for each of the electrical and I&C component types subject to an AMR, along with their component intended functions.

2.1.2.2.4 Commodity Groups Screening

The applicant used commodity groups as a method to evaluate certain components which share similar materials, perform the same intended functions, and operate under similar environmental conditions for many systems. For each mechanical and structural component and component type (commodity) subject to an AMR, the internal and external operating environments to which the component is subjected are established. Operating environments are established based on a review of plant design documents, the UFSAR, vendor drawings, specifications, plant drawings, and environmental data. The materials of construction for the components and component types subject to an AMR are determined based on a review of similar plant documents. Components with similar design, materials of construction, and subjected to similar environments within an individual system are evaluated as a commodity group (e.g., pipe). Commodity groups are not used for components with unique design characteristics, such as heat exchangers, pumps, and tanks, or Class 1 sub-components.

For electrical components, the intended functions for each of the electrical commodity groups and active and passive determinations, are based on the guidance in NEI 95-10. The intended functions established for each of the commodity groups are compared with the criteria of 10 CFR 54.21(a)(1)(i) and (ii). The electrical and I&C components commodity groups that perform an intended function without moving parts or without a change in configuration or properties are identified. Active and passive screening determinations are also based on the guidance in NEI 95-10. The passive electrical commodity groups that are not subject to replacement based on a qualified life or specified time period are identified as requiring an AMR.

### 2.1.3 Staff Evaluation

As part of the review of the applicant's LRA, the NRC staff evaluated the scoping and screening activities described in Section 2.1, "Scoping and Screening Methodology," to ensure that the applicant described a process for identifying SSCs that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4(a)(1-3). In addition, the NRC staff conducted a scoping and screening methodology audit at the VCSNS from January 28–31, 2003. The focus of the audit was to ensure that the applicant had developed and implemented adequate guidance to conduct the scoping and screening of SSCs in accordance with the methodologies described in the LRA and the requirements of the Rule. The audit team reviewed procedures and engineering reports which describe the scoping and screening methodology implemented by the applicant. In addition, the audit team conducted detailed discussions with the cognizant staff on the implementation and control of the program and

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emergency

reviewed administrative control documentation used by the applicant during the scoping and screening process. The team further reviewed a sample of system scoping and screening results reports for the ~~auxiliary~~ feedwater, component cooling water, main steam, and main feedwater systems to ensure the methodology outlined in the administrative controls was appropriately implemented and the results reports were consistent with the CLB, as described in the supporting design documentation.

For mechanical components, the applicant established evaluation boundaries, determined components within those boundaries, and identified component intended functions. This was accomplished by highlighting flow paths on the system drawings and verifying that the mechanical components, identified within the highlighted portions of the boundary drawings, were within the scope of license renewal. All passive, long-lived mechanical components or component groupings were developed using plant system flow diagrams, design guidelines, and the plant component database. For structures, the team verified the evaluation boundaries of structures, identified on the civil structural drawings, to be within the scope of license renewal. The evaluation boundary of structures considered within the scope of license renewal included the entire building and its foundations. The team also verified that electrical equipment within mechanical systems or structures considered within the scope of license renewal were carried forward as electrical commodity groups and then screened for long-lived passive components.

#### *2.1.3.1 Evaluation Methodology for Identifying Systems, Structures, and Components Within the Scope of License Renewal*

In Section 2.1.1 of the LRA, the applicant discussed the scoping methodology related to the safety and non-safety-related criteria and regulated events in compliance with 10 CFR 54.4(a)(1–3). The scoping process used to identify systems and structures that satisfy these requirements is performed using documents which form the CLB and other plant information sources. The CLB for the VCSNS has been defined in accordance with the definition provided in 10 CFR 54.3. The key information sources that form the CLB include the UFSAR, technical specifications, and docketed licensing correspondence. The aspects of the scoping process used to identify SSCs that satisfy the requirements of the Rule are described in Subsections 2.1.1.2, 2.1.1.3, and 2.1.1.4, respectively, of the LRA.

The staff reviewed implementation procedures and engineering reports which describe the scoping methodology implemented by the applicant. These procedures included VCSNS Technical Report (TR) TR00160-001, "Mechanical Systems Scoping for License Renewal," dated July 3, 2002; TR00170-001, "Structures Scoping for License Renewal," dated July 3, 2002; TR00150-001, "Electrical Systems Scoping for License Renewal," dated July 3, 2002; Engineering Services Procedures (ES) ES-701, "Mechanical System Scoping for License Renewal," Revision 1, dated July 31, 2000; ES-703, "Mechanical Component Aging Management Review for License Renewal," Revision 2, dated July 8, 2002; ES-704, "Electrical Systems Scoping, Screening, and Aging Management Review," Revision 2, dated February 5, 2002; ES-705, Civil/Structural Scoping, Screening, and Aging Management Review for License Renewal," Revision 2, dated September 24, 2001; and ES-706, "Identification and Evaluation of Time Limited Aging Analyses and Exemptions for License Renewal," Revision 2.

The staff found that the scoping methodology instructions were consistent with Section 2.1 of the LRA and were of sufficient detail to provide the applicant's staff with concise guidance on

Non-fluid-filled systems were included

could prevent safety-related equipment from performing their intended functions. The review of plant-specific operating experience associated with non-fluid-filled systems also did not identify any instances of such failures. As a result, no additional SSCs were brought into scope for non- fluid-filled systems. For the remaining fluid-filled systems, all were included in the supplemental review except for those systems which could not have an effect on safety-related SSCs due to their location being remote (i.e., being physically separated from) from such safety-related SSCs.

The staff reviewed the plant equipment database to identify non-safety-related components that could have an impact on the ability of nuclear safety-related SSCs to perform their required functions. In addition, the Maintenance Rule includes scoping criteria for non-safety-related SSCs which are similar to the license renewal scoping criterion. The staff reviewed several of these information sources and verified that the applicant had adequately incorporated the results of these efforts into the scoping methodology reports. The staff also discussed with the applicant the current interim staff guidance (ISG) regarding the 10 CFR 54.4(a)(2) issue, including the December 3, 2001, and March 15, 2002, letters to NEI which discussed the staff's position. The ISG discusses two types of systems and structures that must be considered for inclusion within the scope of license renewal per 10 CFR 54.4(a)(2) — (1) Non-safety-related systems and structures, and non-safety-related portions of safety-related systems and structures whose physical failure could damage equipment that is performing a safety function and prevent it from performing that function and (2) non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

The letters described the areas to be considered and options the staff expects applicants to use to determine what SSCs meet the (a)(2) criterion. The December letter provided specific examples of operating experience which identified pipe failure events (summarized in NRC Information Notice (IN) 2001-09, "Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping Inside the Containment of a Pressurized Water Reactor") and the approaches the staff considers acceptable to determine which piping systems should be included in scope. The March letter further described the staff's expectations for the evaluation of non-piping SSCs to determine which additional non safety-related SSCs are within scope. The position states that applicants should not consider hypothetical failures but rather should base their evaluation on the plant's CLB, engineering judgement and analyses, and relevant operating experience.

During the applicant's preparation of the LRA, additional guidance was developed by the NRC regarding scoping of seismic II/I piping systems and the identification and treatment of SSCs which meet 10 CFR 54.4(a)(2). To address the staff's concerns discussed in the NEI letters and the ISG, the applicant stated in Section 2.1.1.3.1 of the LRA that the review of insulation, ductwork, and piping would be provided later to the staff in a supplementary submittal. On September 12, 2002, the applicant submitted to the staff its results of previously non-analyzed insulation, piping and duct work to address the staff's concerns. The results were documented in TR00160-018, "Refined 10 CFR Part 54.4(a)(2) Criteria Evaluations for License Renewal," Revision 0, dated September 6, 2002. The reevaluation focused on AMRs of non-nuclear safety-related piping whose failure may adversely impact nuclear safety-related equipment and components due to spatial interactions (i.e., physical impact, pipe whip, jet impingement, leakage and spray). Non-fluid containing mechanical system portions, as well as ~~nonmechanical~~ SSCs, were also addressed for completeness. In this submittal the applicant

non-mechanical

stated that systems containing non-nuclear safety-related and/or quality-related components that meet the 10 CFR 54.4(a)(2) criteria were identified with respect to spatial interactions that could adversely affect the performance of a safety-related function during the period of extended operation. The results contained a list of systems having components which met the 10 CFR 54.4(a)(2) criteria. Included were 34 in-scope mechanical systems that had their scope expanded to include non-nuclear safety and quality-related portions that have a potential for adverse spatial interactions with nuclear safety-related equipment in certain designated buildings, as well as 11 additional systems added to scope. The staff's review of the applicant's scoping results and aging management evaluation of SCs in these systems is presented in Sections 2.3.5 and ~~3.05~~ 3.0.5 of this SER, respectively.

On the basis of the additional information supplied by the applicant, the staff concludes that the applicant has applied sufficient scoping criteria to demonstrate that all SSCs that meet the 10 CFR 54.4(a)(2) scoping requirements are identified. This information included expansion of the systems within the scope of license renewal, the addition of new portions of systems within scope as a result of the revised methodology, determination of the credible failures which could impact the ability of safety-related SSCs from performing their intended functions, evaluation of relevant operating experience, incorporation of identified non-safety-related SSCs into the applicant's AMPs, and the results of NRC inspection and audit activities.

10 CFR 54.4(a)(3)

In addition to those SSCs relied upon to mitigate DBEs or whose failure could prevent mitigation of such events, the systems that are credited to support compliance with NRC regulations identified in 10 CFR 54.4(a)(3) must be identified for license renewal. This requires that the applicant consider all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), EQ (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), ATWS (10 CFR 50.62), and SBO (10 CFR 50.63) to be within the scope of the license renewal.

The staff reviewed several evaluations and source documents prepared by the applicant to demonstrate compliance with each of the regulated events of interest in accordance with the regulations. The applicant's evaluations focused on identifying and verifying that specific systems or structures were relied upon in response to the particular regulated event. The applicant reviewed all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations to ensure they were included in the scoping methodology. This involved an extensive review of safety evaluation reports, the UFSAR, licensee event technical reports, licensing correspondence, the EQ list, and other design and licensing documentation.

The staff reviewed reports developed by the applicant which provided detailed design information for certain regulated events and included an RG-1.154 evaluation to verify SSCs met the pressurized thermal shock (PTS) criteria, docketed correspondence to address regulatory commitments on ATWS, including documentation to support the installation of the ATWS mitigation system actuation circuitry control system, and for SBO, the applicant developed a coping plan to address the Rule. The reports described the regulatory requirements, system descriptions, and specific equipment relied on to comply with the requirements, including components and structures. For fire protection, the staff reviewed the FPER which contained additional analyses on the essential elements of the program, including

provided a detailed evaluation of the plant with respect to the requirements of 10 CFR 54.4(a)(2).

The staff reviewed the applicant's methodology used to identify SSCs relied upon to remain functional during and following DBEs (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions – (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shut down condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11 of this chapter, as applicable. The applicant initially relied on the use of specific component information contained in CHAMPS and the EQ databases to identify safety-related components and structures credited with remaining functional during and following DBEs defined in the CLB. Several information sources were utilized for the identification of non-nuclear safety-related SSCs that meet the requirements of 10 CFR 54.4(a)(2). The plant equipment database identifies components that are not directly nuclear safety-related but that could have an impact on the ability of nuclear safety-related SSCs to perform their required functions.

Structures Scoping

The staff reviewed ES-705, which provided instructions for implementing the scoping and screening review processes for structures and structural components in accordance with the requirements of 10 CFR 54.4. The staff also reviewed TR00170-001 and TR00170-002, which provided additional guidance on structures scoping and screening, and documented the results. VCSNS developed the structural scoping process in accordance with the guidance contained in NEI 95-10 and compiled the list of structures from several document sources including the UFSAR, site facility drawings, DBDs, and plant walkdowns, and referenced such structures in the master list of structures included in TR00170-001. The structures within this list were evaluated against the 10 CFR 54.4(a) criteria to determine those structures within the scope of license renewal. For compliance with 10 CFR 54.4(a)(1), all structures were classified in the LRA according to their design function and the degree of structural integrity required to ensure the health and safety of the public. The classification of each structure has been previously determined and documented in UFSAR Table 3.2-2, "Classification of Structures." Category I structures are identified through a review of the UFSAR. Nuclear safety-related structures had been previously identified and all remained in scope.

For 10 CFR 54.4(a)(2), structures were classified as either nuclear safety-related or non-nuclear safety-related. The safety-related structures are designed to withstand the SSE and are classified as Seismic Category I, while the non-safety-related structures are generally not designed to withstand SSE seismic levels and are classified as non-seismic. Systems and components that have been seismically mounted to meet anti-falldown (seismic II/I) criteria are classified as Seismic Category II. There are no structures designated as Seismic Category II at VCSNS. Non-safety-related structures whose failure could impair the function of safety-related SSCs are designated as non-seismic but have been designed to withstand earthquake and tornado loads to the extent required for prevention of damage to Seismic Category I structures. The staff reviewed the portion of the master list which had not been identified as nuclear safety-related for potential impact on safety-related components. The applicant identified three structures which met the requirements of 10 CFR 54.4(a)(2) which were subsequently brought into scope. One structure, the north berm for flood ~~peel~~ which was not included in the Rule scoping, was scoped in for license renewal purposes and also added to the Rule. The

control

applicant also identified two structures which were brought into scope for potential structural failure due to seismic or wind.

For 10 CFR 54.4(a)(3), the staff performed a sample review of safety evaluation reports, the UFSAR, DBDs, technical reports, calculations, technical requirement packages, licensing correspondence files, and other appropriate design documents to verify that the scoping methodology demonstrated compliance with the Commission's regulations.

Electrical and Instrumentation and Control Component Scoping

The staff reviewed Sections 2.1.1.2.3 and 2.1.1.3.3 of the LRA to determine that the applicant identified the electrical components within the scope of license renewal in accordance with 10 CFR 54.4 and subject to an AMR, in accordance with the requirements of 10 CFR 54.21(a)(1). The staff discussed the methodology with applicant representatives and reviewed various documents including ES-704, TR00150-001, and TR00150-002. These documents describe the scoping and screening process used by the applicant to identify electrical components subject to an AMR and present the results of that process. The applicant assumed that all electrical systems were within the scope of license renewal unless a specific scoping evaluation was performed that demonstrated otherwise. The purpose of electrical system scoping was to identify those electrical systems which did not meet the requirements of 10CFR 54.4(a)(1-3) and, therefore, did not contain any components within the scope of license renewal. In addition, many electrical components were assigned to mechanical systems (not electrical systems). Following scoping, all electrical components were recombined into electrical commodity groups where they were reviewed as part of the commodity group and not as part of the system. The scoping evaluation described the system, component, or commodity group functions and then evaluated these functions against the scoping criteria of 10 CFR 54.4(a). Those systems which were classified as "1E" were included within the scope in accordance with 10 CFR 54.4(a)(1). The determination of which systems would be within scope in accordance with 10 CFR 54.4(a)(2) was based upon the VCSNS definition of quality-related as detailed in the system technical requirements packages and also from the Maintenance Rule system worksheets. All electrical systems relied upon to perform a function in compliance with NRC requirements for regulated events were also included within scope.

For all other scoping criteria, the applicant reviewed all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations, pursuant to 10 CFR 54.4(a)(3), to ensure they were adequately accounted for in the scoping methodology. This involved an extensive review of SERs, the FPER, the UFSAR, DBDs, licensee event technical reports, licensing correspondence, and other design and licensing documentation. To support this review, the applicant developed a set of reports which provided detailed design information for each regulated event. The reports described the regulatory requirements, the system descriptions, and specific equipment relied on to comply with the requirements including components and structures.

### *2.1.3.2 Evaluation Methodology for Identifying Structures and Components Subject to an Aging Management Review*

After the applicant identified systems and structures within the scope of license renewal and their associated intended functions, a review was performed to identify the components of each in-scope system and structure subject to an AMR. To accomplish this, the staff reviewed

This was not reviewed as part of initial screening.

implementation procedures and engineering reports which described the screening methodology implemented by the applicant. These procedures included ~~ES-427, "Program Screening," Revision 1~~; ES-702, "Mechanical Component Screening for License Renewal," Revision 1, dated July 31, 2000; ES-703, "Mechanical Component Aging Management Review for License Renewal," Revision 2, dated July 8, 2002; ES-704, ES-705, and ES-706, "Identification and Evaluation of Time Limited Aging Analyses and Exemptions for License Renewal," Revision 2. The staff also reviewed the methodology used by the applicant to identify mechanical, structural, and electrical components within the scope of license renewal that would be subject to an AMR. The applicant provided the staff with a detailed discussion of the processes used for each discipline and provided technical reports that described the screening methodology as well as a sample of the engineering analyses for a selected group of safety-related and non-safety-related systems.

2.1.3.2.1 Mechanical Component Screening

Following identification of the SSCs within the scope of license renewal, the applicant performed a screening review to determine which mechanical components would be subject to an AMR in accordance with 10 CFR 54.21(a)(1). An AMR of a mechanical component is required if the component performs an intended function without moving parts or without a change in configuration or properties (i.e., passive) and if it is not subject to replacement on the basis of a qualified life or specified time period (i.e., long-lived). The staff reviewed the screening methodology which involved the establishment of license renewal evaluation boundaries, determination of components within those boundaries, identification of mechanical components subject to an AMR, and identification of the intended functions of each component or component group. Identification of the components subject to an AMR was performed using plant system flow diagrams, equipment databases, and the guidance of Appendix B to NEI 95-10. The intended functions were determined based on the system level function which had been the basis for including the system within the scope of license renewal and the component function which is required to enable the system to perform its intended function. The staff also reviewed the methodology used by the applicant to identify and list the mechanical components subject to an AMR, as well as the applicant's technical justification for this methodology.

The staff reviewed the implementation of this methodology by reviewing a sample of the mechanical systems identified as being within the scope. The systems included safety injection, component cooling water, main feedwater, and emergency feedwater. This included a review of the evaluation boundaries drawn within those systems on the P&IDs, the resulting components determined to be within the scope of the Rule, the corresponding component level intended functions, and the resulting list of mechanical components subject to an AMR. The staff reviewed the applicant's methodology for establishment of system evaluation boundaries, reviewed applicable procedures outlining the process, verified portions of the diagrams, and held discussions with the responsible members of the applicant's LRA staff. The initial step in the component screening process was to establish the license renewal boundaries for each system within the scope of license renewal, (i.e., the physical or functional boundaries that are required to support identified system intended functions). Precise physical and functional boundaries were necessary to assure that all components and component groups required to support system intended functions were considered for inclusion. The system evaluation boundaries were established by highlighting on system flow diagrams and other pertinent drawings the flow paths that are involved with the system intended functions identified in TR00160-001 and all other portions of the system that meet the scoping criteria of

10 CFR 54.4(a)(1-3). Once the system evaluation boundaries were established, the subject components or component types (commodities) located within the evaluation boundaries were determined as described in ES-702. From the list of potential intended functions provided in ES-702, the actual intended functions of the subject components were determined by reviewing the UFSAR, system DBDs, and other appropriate design and licensing documents. Actual intended functions were those that passively support the system intended functions provided in TR00160-001. Based on this sample review of portions of the above listed systems, applicable procedures and diagrams, and discussions with the applicant, the staff determined that the screening methodology for mechanical systems was adequately implemented.

2.1.3.2.2 Structural Component Screening

Following identification of the SSCs within the scope of license renewal, the staff reviewed the applicant's screening review, in accordance with ES-705 and TR00170-002, to determine which structural components would be subject to an AMR in accordance with 10 CFR 54.21(a)(1). An AMR of a structural component is required if the component performs an intended function without moving parts or without a change in configuration or properties (i.e., passive) and if it is not subject to replacement on the basis of a qualified life or specified time period (i.e., long-lived). The applicant used industry experience and NEI 95-10 to develop a master list of component types and potential intended functions. The applicant established the structure evaluation boundaries, identified structural component types, including long-lived passive components within the evaluation boundaries, and identified potential and actual structural component intended functions and components subject to an AMR. The staff reviewed the methodology used by the applicant to identify and list the structural components and structural commodities subject to an AMR, as well as the applicant's technical justification for this methodology. The staff reviewed a sample of plant structures (auxiliary building and turbine building) identified as being within the scope, including the evaluation boundaries and resultant components, the corresponding component level intended functions, and the resulting list of structural components and structural commodity groups subject to an AMR. The staff also reviewed a sample of the structural drawing packages assembled by the applicant and discussed the process and results with the cognizant engineers who performed the review. The staff did not identify any discrepancies between the methodology documented and the implementation results.

2.1.3.2.3 Electrical and Instrumentation and Control Component Screening

After identifying the SSCs within the scope of license renewal, the staff reviewed the applicant's screening review to determine which electrical components would be subject to an AMR. As part of this effort, the applicant relied on the requirements set forth in 10 CFR 54.21(a)(1)(i) as supplemented by industry guidance in NEI 95-10 to develop a commodity evaluation approach based on a plant-level evaluation of electrical equipment. The applicant reviewed the component to determine whether the component was passive and long-lived.

The process began with a list of generic electrical commodities from Appendix B to NEI 95-10. Next the applicant applied passive screening that eliminated from the list all commodities that were active rather than passive (i.e.,components that performed an intended function without moving parts or without a change in configuration). The applicant applied long-lived screening to components that were to be replaced based on a qualified life and removed them from the license renewal scope. The remaining passive commodities included non-EQ insulated cables,

connectors, splices, penetration assemblies and terminal blocks, and high voltage electrical switchyard busses, transmission conductors, connections, and insulators. The applicant also indicated that non-EQ fuse blocks would be added to this group based upon the guidance in the corresponding NRC ISG. The applicant concluded that all electrical components included in the applicant's EQ program were short-lived and were screened out of license renewal scope. The staff also reviewed the methodology used by the applicant to identify and list the electrical components and commodities subject to an AMR, as well as the applicant's technical justification for this methodology, and discussed the methodology and results with the applicant's LRA staff. The staff also sampled several engineering analyses to verify implementation of the screening process for electrical and I&C components. Based on the above, the staff determined that the screening methodology for electrical and I&C components was adequately implemented.

### 2.1.4 Conclusions

The basis of the staff's safety determination included the review of the information presented in Section 2.1 of the LRA, the supporting information in the VCSNS UFSAR, the information presented during the staff's scoping and screening audit, NRC scoping and AMR inspections, and the applicant's responses to the staff's RAIs. The staff verified that the applicant's scoping and screening methodology, including their supplemental 10 CFR 54.4(a)(2) review, which brought additional non-safety-related piping segments and associated components into scope, was consistent with the requirements of the Rule and the staff's position on the treatment of non-safety-related SSCs. On the basis of this review, the staff finds that the applicant's methodology for identifying SSCs within the scope of license renewal, and the SCs requiring an AMR, is consistent with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

### 2.1.5 References


1. NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," April 2001

2. NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Revision 3," August 2001

3. NUREG-1801, "Generic Aging Lessons Learned Report (GALL)," April 2001

4. NRC Inspection Report No. 50-395/03-07 for the V.C. Summer Nuclear Station, dated June 13, 2003.

5. Letter from USNRC to SCE&G, "Request for Additional Information for the Review of the V.C. Summer Nuclear Station License Renewal Application," dated March 28, 2003.

6. Letter from SCE&G to USNRC, "Responses to Request for Additional Information for the Review of the License Renewal Application for the Virgil C. Summer Nuclear Station," dated June 16, 2003 (RC-03-0112).

7. ES-411, "Equipment Classifications," Revision 9.

8. ES-412, Attachment 1, "Appendix R Evaluation Phase II Composite Equipment List," dated September 30, 1998.

9. ES-427, "Program/Issue Screening," Revision 1, dated April 26, 2002.

10. ES-701, "Mechanical System Scoping for License Renewal," Revision 1, dated July 31, 2000.

11. ES-702, "Mechanical Component Screening for License Renewal," Revision 1, dated July 31, 2000.

12. ES-703, "Mechanical Component Aging Management Review for License Renewal," Revision 2, dated July 8, 2002.

13. ES-704, "Electrical Systems Scoping, Screening, and Aging Management Review," Revision 2, dated February 5, 2002.

14. ES-705, "Civil/Structural Scoping, Screening, and Aging Management Review for License Renewal," Revision 2, dated September 24, 2001.

15. ES-706, "Identification and Evaluation of Time Limited Aging Analyses and Exemptions for License Renewal," Revision 2, dated August 27, 2001.

16. DBD, "Feedwater System," Revision 8, dated August 27, 2002.

17. DBD, "Component Cooling Water System," Revision 10, dated September 28, 2002.

18. DBD, "Emergency Feedwater System," Revision 11, dated August 28, 2001.

19. DBD, "Safety Injection System," Revision 10, dated November 17, 1999.

20. TR00150-001, "Electrical Systems Scoping for License Renewal, Revision 0, dated July 3, 2002."

21. TR00150-002, "Electrical Screening for License Renewal, Revision 0, dated July 3, 2002."

22. TR00150-003, "Electrical Component Aging Management Review for License Renewal," Revision 0, dated July 3, 2002.

23. TR00160-001, "Mechanical Systems Scoping for License Renewal," Revision 0, dated July 3, 2002.

24. TR00160-003, "Mechanical Component Screening for License Renewal (Treated Water Systems," Revision 0, dated July 3, 2002.

25. TR00160-013, "Mechanical Component Aging Management Review for License Renewal (Treated Water Systems)," Revision 0.

26. TR00160-018, "Refined 10 CFR Part 54.4(a)(2) Criteria Evaluations for License Renewal," Revision 0, dated September 6, 2002.

27. TR00170-001, "Structural Scoping for License Renewal, Revision 0, dated July 3, 2002."

28. TR00170-002, "Structures Screening for License Renewal, Revision 0, dated July 3, 2002."

29. TR00170-003, "Structures Aging Management Review for License Renewal," Revision 0, dated July 3, 2002.

30. SAP-605, "Application of CHAMPS", Revision 7, dated December 16, 2002.

31. SAP-630, "Procedure/Commitment Accountability Program," Revision 6, dated November 27, 2001.

32. SAP-1041, "Statement of Responsibilities, Plant Life Extension," Revision 0, dated November 11, 2002.

33. Duke Engineering & Services Task No. 7A, 8, 9, 10, 11& 12, "Project Execution Plan," Revision 0, TM/W 210-01, "Plant License Renewal Fundamentals."

34. Memorandum, "Self-Assessment Report for the License Renewal Project," dated May 24, 2002.

35. Report SA02-PX-02, "License Renewal Self Assessment Report - Aging Management Programs," dated November 11, 2002.

36. TR Package No. 2, "Fire Protection," Revision 8.

37. EQ Report No. 984, "Equipment Qualified to a Harsh Environment - Master Equipment List," dated November 13, 2002.

38. QA-SUR-200119-0, "Plant Life Extension," dated October 16, 2001.

39. ~~P&ID~~ Drawing D-302-083, "Feedwater," Revision 47.

40. ~~P&ID~~ Drawing D-302-085, "Emergency Feedwater," Revision 40.

41. ~~P&IDs~~ Drawings E-302-691, 692, and 693.

42. Drawing D-302-661, "Reactor Building Containment Spray System."

43. Drawing D-302-651, "Spent Fuel Pool Cooling."

44. Drawings D-302-611, 612, and 613, "Component Cooling System," and D-302-614, "Component Cooling System To NSSS Pumps."

45. Drawings D-302-231, "Fire Service", Sheets 1 thru 5.

46. Drawing E-302-641, "Residual Heat Removal System."

47. ~~Gilbert Commonwealth~~ Drawing B-817-026, "Control Air Signal Tubing Diagram".

## 2.2 Plant Level Scoping Results

### 2.2.1 Introduction

The statements of consideration (SOC) for the License Renewal Rule (*Federal Register*, Volume 60, No. 22478) indicate that an applicant has the flexibility to determine the set of SSCs for which an AMR is performed. In LRA Section 2.1.1, the applicant described the methodology for identifying the SSCs within the scope of license renewal. In LRA Section 2.2, the applicant uses the scoping methodology to determine which of the SSCs are required or not required to be included in the scope of license renewal. The staff reviewed the plant level scoping results to determine whether the applicant has properly identified all plant level SSCs that are relied upon to mitigate DBEs, as required by 10 CFR 54.4(a)(1), or whose failure could prevent satisfactory accomplishment of any of the safety-related functions, as required by 10 CFR 54.4(a)(2), as well as the SSCs relied on in safety analysis or plant evaluations to perform a function that is required by one of the regulations referenced in 10 CFR 54.4(a)(3).

The staff reviewed the SSCs that the applicant did not identify as being within the scope of license renewal to verify whether they have any intended functions that are within the scope of license renewal. The staff also reviewed the selected SSCs that the applicant has identified as being within the scope of license renewal to verify whether the applicant properly identified their components within the evaluation boundary that are subject to an AMR, in accordance with the requirements of 10 CFR 54.21(a)(1). To determine whether the applicant identified the SSCs that are subject to an AMR, the staff reviewed the components that the applicant had not identified as being subject to an AMR.

### 2.2.2 Summary of Technical Information in the Application

This section addresses the plant level scoping results for the license renewal. Pursuant to 10 CFR 54.21(a)(1) the applicant is required to identify and list SCs subject to an AMR. These are the passive and long-lived SCs that are within the scope of license renewal.

In LRA Table 2.2-1, the applicant lists plant level scoping results for mechanical systems, which includes all the mechanical systems both in scope and not in scope. The plant level scoping results for structures are listed in LRA Table 2.2-2, which include all the structures and buildings both in scope and not in scope. The specific mechanical systems within the scope of license renewal are described in detail in LRA Section 2.3. The specific structures and buildings within the scope of license renewal are described in detail in LRA Section 2.4. The electrical and I&C systems that support the operation of both safety- and non-safety-related systems and components are described in LRA Section 2.5. In the LRA, the electrical and I&C components are treated as commodities. In scoping the electrical systems, only the electrical commodity groups that perform a passive safety function are subject to an AMR. To verify whether the applicant has properly implemented its methodology, the staff focused its review on the implementation results to confirm that there is no omission of plant level systems and structures within the scope of license renewal.

*2.2.2.1 Systems, Structures, and Components Within the Scope of License Renewal*

In LRA Sections 2.2 through 2.5, the applicant describes the SSCs that are within the scope of license renewal, and subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1), respectively. As described in LRA Section 2.1, "Scoping and Screening Methodology," the scoping and screening of mechanical components were performed using the plant's DBDs, component databases, and flow diagrams (P&ID drawings). The applicant uses two controlled databases to perform the scoping and screening — component history and maintenance planning (CHAMPS) database and equipment qualification database (EQDB). The CHAMPS is a controlled database that contains as-built information on a component level that consists of multiple data field for each component. The EQDB is a controlled database that consists of multiple data fields for each component or subcomponent, including component identification, maintenance requirements, etc. The two databases uniquely identify most of the mechanical components at the plant and provide links to the associated systems. The applicant also identified those mechanical components in the databases not assigned with unique component numbers by evaluating design drawings and documents, and also by plant walkdowns. The items in the databases were treated as commodities for the purposes of license renewal.

LRA Table 2.2-1 provides the results of the applicant's plant-wide scoping of the mechanical systems. The table identifies which of the plant systems are within the scope of license renewal and which of them are not. The table also indicates whether the intended functions of a given system is needed to satisfactorily accomplish any of the functions identified in 10 CFR 54.4 (a)(1), (a)(2), and (a)(3). The LRA considers electrical and I&C systems as generic and treated them as groups of commodities. The scoping results for the commodity groups of electrical and I&C components are listed in LRA Table 2.2-3.

Plant structures that satisfy one or more of the criteria in 10 CFR 54.4, and contain in-scope mechanical and electrical components, are within the scope of license renewal and subject to an AMR. All seismic Class I SCs are considered safety-related and are in scope. Non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1) ~~(2), and (3)~~ are within the scope of license renewal. The applicant also evaluated the non-safety-related systems that may have spatial relationships with safety-related components such that their failure could adversely impact the performance of an intended safety function (related to seismic II/I issues). The applicant documented the seismic II/I evaluations in a technical report (i.e., RC-02-0159). The staff's review of the technical report is addressed in Section 2.3.5 of this SER.

*2.2.2.2 Systems and Structures Not Within the Scope of License Renewal*

In addition to the SSCs in scope, LRA Table 2.2-1 contains 54 mechanical systems that are not within the scope of license renewal. Also, LRA Table 2.2-2 lists 67 structures or buildings and LRA Table 2.2-3 lists 16 electrical systems that are not within the scope of license renewal. However, these tables do not provide reasons why the SSCs are not in scope (this is discussed in Section 2.2.3 of this SER).

**2.2.3 Staff Evaluation**

* (a)(2) & (a)(3) not applicable

The staff reviewed LRA Section 2.2 and Tables 2.2-1, 2.2-2, and 2.2-3 to determine whether the applicant has properly identified all plant level SSCs that are within the scope of license renewal, as required by 10 CFR 54.4. The staff's review was conducted in accordance with Section 2.2 of the SRP-LR NUREG-1800 and is described as follows.

In LRA Section 2.1, the applicant describes the process for identifying the SSCs that are within the scope of license renewal and subject to an AMR. This methodology typically consists of a review of all plant level SSCs to identify those that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4. From those in-scope SSCs, the applicant identifies and lists their components that are passive (that perform their intended functions without moving parts, or without a change in configuration or properties), and are long-lived (that are not replaced based on a qualified life or specified time period). The staff reviewed the scoping and screening methodology and provided its evaluation in Section 2.1 of this SER. The applicant documented its implementation of the methodology in LRA Sections 2.3 through 2.5. The staff's evaluation of the applicant's implementation is addressed in Sections 2.3 through 2.5 of this SER.

The staff reviewed LRA Section 2.1 to ensure that the scoping and screening methodologies described in the section were properly implemented, and that the components that are subject to an AMR were properly identified. The staff also reviewed LRA Section 2.2 and sampled the contents of VCSNS UFSAR, based on the listing of systems and structures in LRA Tables 2.2-1 and 2.2-2, to determine whether there were systems or structures that may have intended functions, as identified by 10 CFR 54.4, but were not included in the scope of license renewal.

During its review of LRA Section 2.2 and LRA Tables 2.2-1 and 2.2-2, the staff determined that additional information and/or clarification was needed to complete its review. Because the applicant did not justify the mechanical systems and structures in the LRA tables not in scope, the staff was unable to determine whether some of these mechanical systems (in Table 2.2-1) and plant structures (in Table 2.2-2) are required to be in scope. By letter dated March 28, 2003, in RAI 2.2.2-1, the staff requested the applicant to explain why the following mechanical systems and plant structures are not within the scope of license renewal:

- emergency offsite facility (~~EO~~) EOF
- emergency equipment (~~EO~~)
- liquid effluent from nuclear plant to pen stock (LW)
- radwaste solidification and solids handling (WD)
- auxiliary fire pump house
- containment access building (CAB)
- lighting masts (plant)
- radiological maintenance building

In its response, dated June 12, 2003, the applicant provided the following clarification or justification for the above systems and structures not in scope:

- The ~~EO~~ EOF is for emergency plan activities and has no direct license renewal function. Loss of system function will not result in the loss of any safety related functions.
- The ~~EQ~~ Emergency Equipment system is inactive.
- The LW was not initially included in the license renewal scope. As a result of the Criterion 2 reassessment, this non-safety-related system was added to the scope of license renewal due to its potential spatial interactions with safety-related components. The Criterion 2 supplement to the LRA was submitted to the NRC separately in a technical report (RC-02-0159) dated September 12, 2002.
- The WD was not initially included in the license renewal scope. As a result of the Criterion 2 reassessment, this non-safety-related system was added to the scope of license renewal due to its potential spatial interactions with safety-related components.
- The auxiliary fire pump house is a structure that houses the auxiliary backup fire pumps used during construction. There are no mechanical or electrical components in the structure that are within the scope of license renewal.
- The CAB was constructed to facilitate containment access during the steam generator replacement project and no longer serves a direct plant operational or access function. It is used for storage in the radiological maintenance area and performs no intended function for license renewal.
- The plant lighting masts are the high light pole structures located around the plant site. They are not used to support any of the regulated events and perform no intended functions for license renewal.
- The radiological maintenance building serves as a maintenance facility for contaminated components and tools and performs no intended functions for license renewal.

The staff reviewed the applicant's response and concurs with its decision to include the LW system and WD system in the scope of license renewal based on Criterion 2 reevaluation. The staff's review of the Criterion 2 Supplement is addressed in Section 2.3.5 of this SER. The staff also agreed with the applicant's rationale for not requiring the remaining mechanical systems and structures to be in scope, except plant lighting masts. The staff considered that the plant lighting masts have an intended function to support plant lighting. Failure of lighting pole structures may cause blackout of the plant site. Therefore, the light poles should be included in the yard structures for license renewal. In a letter, dated June 12, 2003, the applicant further explained that the high mast lighting poles should not be in scope. The applicant stated that there are seven high mast light poles located around the plant site that serve as security lighting. These high mast light poles are not used to support accident conditions or any of the regulated events and thus perform no intended functions for license renewal. In addition to these high mast light poles, exterior lighting also consists of standard height light poles and wall-mounted lights along the perimeter of each structure within the protected area of the plant. All of the exterior lighting is supplied by 480-volt, single phase power from the nearest available

480-volt load center. Because none of the exterior lights are credited for accident or any event described in 10 CFR 54.4(a)(3), the plant lighting masts are not considered to be in the scope of license renewal.

The staff reviewed the applicant's response and found its rationale acceptable because the plant site has provided redundant lighting supplied from offsite power. The plant lighting masts are not required to support plant lighting and, therefore, can be excluded from the license renewal scope. On the basis of this review, the staff did not identify any omissions.

### 2.2.4 Conclusions

The staff reviewed LRA Sections 2.2.1 and 2.2.2, supporting information in the plant's UFSAR, and the information provided in response to the staff's RAI to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. As a result of this review, the staff did not identify any omissions. On the basis of this review, the staff concludes that the applicant has appropriately identified the SSCs that are within the scope of license renewal, in accordance with the requirements of 10 CFR 54.4. The staff's detailed review of the SSCs that are subject to an AMR is provided in Section 2.3 through 2.5 of this SER.

### 2.2.5 References


1. 10 CFR 54, Requirements for Renewal of Operating License for Nuclear Power Plants, 60 FR 22461, May 8, 1995.

2. NEI 95-10 (Revision 3), Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule, August 2001.

3. VCSNS Final Safety Analysis Report (FSAR), Amendment 02–1.

4. NUREG-1800, Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants, July 2001.

5. Generic Aging Lessons Learns (GALL) Report, July 2001


## 2.3 Scoping and Screening Results: Mechanical Systems

Pursuant 10 CFR 54.21(a)(1) an applicant is required to identify and list SCs subject to an AMR. These are passive, long-lived SCs that are within the scope of license renewal. To verify that the applicant has properly implemented the scoping and screening methodology, the staff focuses its review on the implementation results. Such a focus allows the staff to confirm that the LRA has identified all the mechanical system components that would be subject to an AMR.

### 2.3.1 Reactor Vessel, Internals and Reactor Coolant System

The reactor coolant system components consist of the reactor vessel, reactor vessel internals, incore instrumentation system, pressurizer, steam generators, and associated reactor coolant system piping.

of the pressurizer relief tank, as being outside the scope of license renewal, is acceptable to the staff.

No omissions of SSCs that are within the scope of license renewal and subject to an AMR were found.

2.3.1.1.3 Conclusions

The staff reviewed Section 2.3.1.1 of the LRA and Chapter 5 of the UFSAR to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of its review, the staff concludes that the applicant has adequately identified the reactor coolant system components that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the reactor coolant system components that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.1.2 Piping, Valves and Pumps*

The applicant describes the piping, valves, and pumps in LRA Section 2.3.1.2. The mechanical component types and component intended functions for the reactor coolant system Class 1 piping and associated pressure boundary components are listed in Table 2.3-2. UFSAR Section 5.5, Component and Subsystem Design, provides additional information concerning Class 1 piping and associated pressure boundary components.

2.3.1.2.1 Summary of Technical Information in the Application

The reactor coolant system Class 1 piping and associated pressure boundary components consist of the following:

- primary loop piping interconnecting the reactor vessel, steam generators and reactor coolant pumps
- the piping (including fittings, branch connections, safe ends, thermal sleeves, flow restrictors, and thermowells) and valves leading to connecting auxiliary or support systems, up to and including the second isolation valve (from the high pressure side) on each line
- pressure boundary portion of Class 1 valves (body, bonnet, and bolting)
- pressure boundary portion of the reactor coolant pumps (casing, main closure flange, thermal barrier heat exchanger and bolting)

The primary loop piping consists of three closed reactor coolant loops interconnecting the reactor vessel, steam generators, and reactor coolant pumps. Class 1 branch piping consists of piping connected to the Class 1 primary loop piping out to and including the outermost containment isolation valve in piping which penetrates primary containment, or the second of two valves normally closed during normal reactor operation in piping which does not penetrate primary containment. Some Class 1 branch lines and instrument lines are equipped with

3/8-inch inside diameter flow restrictors. These flow restrictors limit the maximum flow from a break downstream of the flow restrictor to below the makeup capability of the charging system.

The pressure retaining portion of the Class 1 valves includes the body, bonnet, and bolting. The valves are welded into the piping, except for the ~~pressurizer relief and~~ pressurizer code safety valves, which have flanged connections. The portions of the reactor coolant pumps that perform a pressure boundary function are the pump casing, main closure flange, thermal barrier heat exchanger, and bolting. The reactor coolant pumps are vertical, single stage, centrifugal pumps, equipped with controlled leakage shaft seals. The shaft seals are excluded from AMR because they are periodically replaced.

The Class 1 portion of the reactor coolant system includes portions of the chemical and volume control system, emergency core cooling system, residual heat removal system, and safety injection system.

The component types subject to AMR and their intended functions, listed in Table 2.3-2 of the LRA, include reactor coolant pump main flange bolting materials, reactor coolant pump thermal barrier flange, main closure flange, reactor coolant pump thermal barrier piping/tubing (less than 4-inches normal pipe size (NPS), and reactor coolant pump casing. The intended functions identified for these components were pressure boundary and throttling.

2.3.1.2.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.2 and UFSAR Section 5.5 to determine whether the piping, valves and pumps and supporting structures within the scope of license renewal and subject to an AMR had been identified in accordance with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1), respectively. The staff review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

As part of the evaluation, the staff determined whether the applicant had properly identified the SSCs within the scope of license renewal and subject to an AMR, pursuant to 10 CFR 54.4(a) and 10 CFR 54.21(a)(1). The staff reviewed the relevant portions of the UFSAR for VCSNS for the piping, valves, and pumps and associated pressure boundary components, and compared the information in the UFSAR with the information in the LRA to identify those portions that the LRA did not identify as being within the scope of license renewal and subject to an AMR. The staff then reviewed the SCs that were identified as not being within the scope of license renewal to verify that these SCs do not have any of the intended functions delineated under 10 CFR 54.4(a), and for those SCs that have an applicable intended function(s), to verify that they either perform this function(s) with moving parts or a change in configuration or properties, or that they are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1).

The staff also reviewed the UFSAR for any functions delineated under 10 CFR 54.4(a) that were not identified as intended functions in the LRA, to verify that the SSCs with such functions will be adequately managed so that the functions will be maintained consistent with the CLB for the period of extended operation.

No omissions of SSCs that are within the scope of license renewal and subject to an AMR were found.

The applicant describes the reactor vessel internals in LRA Section 2.3.1.4 and provides a list of components subject to an AMR in LRA Table 2.3-4. UFSAR Section 4.2.2, Reactor Vessel Internals, provides additional information concerning the reactor vessel internals.

2.3.1.4.1 Summary of Technical Information in the Application

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure. and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation.

The coolant flows from the vessel inlet nozzles, down the annulus between the core barrel and the vessel wall, and into a plenum at the bottom of the vessel. The coolant then reverses direction and flows up through the core support and lower core plates. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the ~~vessel outlet~~ nozzles.

All reactor vessel Internals components are considered Class ~~1~~ 2 for seismic design. The effects of neutron embrittlement on materials utilized and accid~~ent loadings~~ on the internals have been considered in the design analysis.

The license renewal boundary for the reactor vessel internals consists of all components internal to the reactor vessel, excluding the reactor vessel and head, the control rod drive mechanisms, (CRDMs) and integral attachments to the reactor vessel and head.

The components of the reactor vessel internals, subject to AMR, include the following major components and their associated subcomponents:

- baffle and former assembly
- bottom mounted instrumentation columns
- clevis inserts
- core barrel and flange
- core barrel outlet nozzle
- guide tube
- lower core plate
- lower support columns
- lower support plate
- neutron panels
- radial keys
- ~~spray nozzle~~
- upper core plate
- upper instrumentation conduit and supports
- upper support column
- upper support plate assembly

The intended functions identified for the reactor vessel internals components were structure functional support, flow distribution, control rod guidance and protection, and radiation shielding.

2.3.1.4.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.4 and UFSAR Section 4.2.2, Reactor Vessel Internals, to determine whether the reactor vessel internals and supporting structures within the scope of license renewal, and subject to an AMR had been identified in accordance with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1), respectively. The staff review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

As part of the evaluation, the staff determined whether the applicant had properly identified the SSCs within the scope of license renewal and subject to an AMR, pursuant to 10 CFR 54.4(a) and 10 CFR 54.21(a)(1). The staff reviewed the relevant portions of the UFSAR for the reactor vessel internals and associated pressure boundary components and compared the information in the UFSAR with the information in the LRA to identify those portions that the LRA did not identify as being within the scope of license renewal and subject to an AMR. The staff then reviewed the SCs that were identified as not being within the scope of license renewal to verify that these SCs do not have any of the intended functions delineated under 10 CFR 54.4(a), and for those SCs that have an applicable intended function(s), to verify that they either perform this function(s) with moving parts or a change in configuration or properties, or that they are subject to replacement based on a qualified life or specified time period, as described in 10 CFR 54.21(a)(1).

The staff also reviewed the UFSAR for any function(s) delineated under 10 CFR 54.4(a) that were not identified as intended function(s) in the LRA, to verify that the SSCs with such function(s) will be adequately managed so that the function(s) will be maintained consistent with the CLB for the period of extended operation.

Many of the reactor vessel internals are identified as components that provide structural support to safety-related components. They can provide, for example, the structural support needed to maintain a coolable core geometry during a design basis loss of coolant accident (LOCA). Unlike many other long-lived, passive components, certain reactor internals are normally moved (i.e., removed and set aside) to permit the movement of fuel assemblies during refueling. This provides occasional opportunities to detect and remedy aging-related problems that might affect these reactor vessel internals. Although these particular components would have the benefit of periodic examination, they would still be included in the license renewal scope and subject to aging management requirements.

No omissions of SSCs that are within the scope of license renewal and subject to an AMR were found.

2.3.1.4.3 Conclusions

The staff reviewed the information presented in Section 2.3.1.4 of the LRA and the supporting information in UFSAR Section 4.2.2, Reactor Vessel Internals, to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. In addition, the

- pressurizer upper and lower heads
- immersion heater well assemblies
- manway cover and bolts
- nozzle safe ends and thermal sleeves
- shell barrel
- tubing (instrumentation and sample lines) and tube couplings

The intended functions identified for the pressurizer components were pressure boundary and heat transfer.

2.3.1.6.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.6 and UFSAR Section 5.5.10 to determine whether the pressurizer within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The pressurizer, a safety-related, in-scope component, contains a spray head, a non-safety-related component, which the applicant has not included in the license renewal scope.

The spray head distributes normal and auxiliary pressurizer spray water into the pressurizer steam bubble, which tends to depressurize the pressurizer, and hence the reactor coolant system. Since the normal and auxiliary pressurizer sprays are not safety systems, they cannot be relied upon to function during any of the Chapter 15 accident analyses, unless, in some postulated analysis cases, pressurizer spray could have an aggravating effect upon the transient results (e.g., by delaying a high pressurizer pressure reactor trip). Therefore, the spray function is not credited for the mitigation of any accidents addressed in the UFSAR accident analyses.

As a non-safety-related component that is wholly enclosed by the pressurizer, a safety-related component, the pressurizer spray head would be subject to the requirements of 10 CFR 54.4(a)(2), which state, "Plant systems, structures, and components within the scope of this part are .... All non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section." Paragraphs (a)(1)(i), (ii), and (iii) of this section address the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures, respectively.

Normal and auxiliary pressurizer sprays are used to reduce the primary side coolant pressure, and to end the primary to secondary side tube break flow, following a steam generator tube rupture event. If, for some reason, the spray head fails in such a way as to block all spray flow, then normal and auxiliary sprays would become unavailable for depressurization following a steam generator tube rupture event. Since there is always some spray flow into the

pressurizer, during normal operation, it is expected that such a failure would be promptly detected and rectified.

If the pressurizer spray head were to degrade and crack, and shed one or more pieces of the head, these pieces could become loose parts inside the pressurizer. During a pressurization transient, such as a loss of a normal feedwater event or a load rejection, the power-operated relief valves, or even the code safety valves, might open. A loose part, inside the pressurizer, might be drawn into the throat of a power-operated relief valve or a code safety valve, and impede the ability of the pressurizer and its pressure relieving valves to protect the integrity of the reactor coolant pressure boundary. Depending upon the position of the loose part, inside the valve throat, the loose part might prevent the valve from reseating properly, and thereby transform a pressurization event into a depressurization event.

Although loose pieces of the spray head are not likely to damage the pressurizer itself, these pieces have the potential to impair certain safety-related functions of the pressurizer, such as the power-operated relief valves or the safety valves, during pressurization transients. The possibility that such loose parts might be generated and that they might impair certain safety functions of the pressurizer is not, by itself, sufficient to require that the pressurizer spray head be included in the license renewal scope. There must be some basis, in operating experience, that such a scenario could be reasonably expected to occur sometime during the 20-year license extension, following a 40-year aging period. To date, there have been no recorded instances of this type of failure. Therefore, without an experiential basis, the requirements of 10 CFR 54.4(a)(2) would not be applicable to the pressurizer spray head.

CONSIDER INFO FROM VCSNS RESPONSE RAI 2.3.1-

By letter dated April 9, 2003, the staff requested the applicant to indicate whether the pressurizer spray head is credited in the fire protection safe shutdown analysis to satisfy 10 CFR 50.48, Appendix R requirements. Section 54.4(a)(3) of Title *10 of the Code of Federal Regulations* requires that components that are used to satisfy the requirements of 10 CFR 50.48, Appendix R, must be included within the scope of license renewal. The specific intended function of the spray head that is subject to the 10 CFR 54.4(a)(3) requirement is the spray function. The spray head does not have a pressure boundary function.

In response, the applicant stated that the pressurizer spray is not credited to depressurize the reactor coolant system in the Appendix R event. Primary system depressurization is accomplished by opening the pressurizer power-operated relief valves. Therefore, the applicant has included the power-operated relief valves, not the pressurizer spray head, within the scope of license renewal. At the staff's request, the applicant has confirmed that the pressurizer power-operated relief valves are included in the license renewal scope. They are listed in Table 2.3-2, under "Valves".

Therefore, since the spray head (1) does not perform any intended functions, (2) its failure would not prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1), and (3) it is not relied upon to depressurize the reactor coolant system in an Appendix R scenario, the staff agrees with the applicant's conclusion that the spray head need not be within the scope of license renewal.

No omissions of SSCs that are within the scope of license renewal and subject to an Aging Management Review, were found.

cooling water (AC), demineralized water — nuclear services (DN), nitrogen blanketing (NG), and reactor building leak rate testing (LR).

The auxiliary coolant/CRDM AC system is designed to remove heat from the containment air used to cool the CRDM and dissipate this heat to the atmosphere via the industrial cooler.

The DN system is designed to ~~clarify, filter, and demineralize raw water from Monticello Reservoir for distribution~~ distribute demineralized water to the nuclear steam supply system (NSSS), secondary (turbine) cycle, and other miscellaneous plant systems.

The NG system is designed to provide pressurized nitrogen to hose connections located inside containment.

The reactor building leak rate testing system is designed to permit containment leakage testing in accordance with 10 CFR Part 50, Appendix J.

2.3.2.2.2 Staff Evaluation

The staff reviewed LRA Section 2.3.2.2 and UFSAR Section 6.2.4 to determine whether the containment isolation system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1), respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of the review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted. As a result of this review, the staff did not identify any omissions.

2.3.2.2.3 Conclusions

The staff reviewed the LRA, the accompanying scoping boundary drawings, and the applicant's RAI response to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the containment isolation system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the containment isolation system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.2.3 Hydrogen Removal — Post Accident System*

2.3.2.3.1 Summary of Technical Information in the Application

The applicant describes the hydrogen removal — post accident system in LRA Section 2.3.2.3 and provides a list of components subject to an AMR in LRA Table 2.3-13. The system is further described in UFSAR Section 6.2.5, Combustible Gas Control in Reactor Building.

The hydrogen removal — post accident system is designed for control of combustible hydrogen concentrations in the reactor building following a LOCA. The system uses electric hydrogen recombiners as a primary means of reducing hydrogen concentrations, while a purge system is provided as a backup to the recombiners.

2.3.2.3.2 Staff Evaluation

The staff reviewed LRA Section 2.3.2.3 and UFSAR Section 6.2.5 to determine whether the hydrogen removal—post accident system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1), respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of the review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted. As a result of this review, the staff did not identify any omissions.

2.3.2.3.3 Conclusions

The staff reviewed the LRA and the accompanying boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the hydrogen removal—post accident system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the hydrogen removal—post accident system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.2.4 Reactor Building Spray System*

2.3.2.4.1 Summary of Technical Information in the Application

The applicant describes the reactor building spray system in LRA Section 2.3.2.4 and provides a list of components subject to an AMR in LRA Table 2.3-14. The system is further described in UFSAR Section 6.2.2, Reactor Building Heat Removal Systems.

The basic functions of the reactor building spray system are to (1) remove the thermal energy released to containment by a LOCA at a rate sufficient to limit the resulting over-pressurization to a level below the design limit, thereby maintaining containment structural integrity, and (2) to subsequently reduce the over-pressure to a level that minimizes the pressure differential which induces leakage out of containment. An additional function of the reactor building spray system is to reduce the concentration of airborne radioactive iodine in the containment atmosphere.

These functions are accomplished by spraying water containing sodium hydroxide into the containment atmosphere to absorb heat, condense steam, and remove airborne radioactive iodine from the steam-air atmosphere.

chilled water ventilation system pressure boundary only - No credit for cooling

review of SC supports is addressed in Section 2.1.2.2 of this SER. Electrical and I&C components in the HVAC systems are addressed in Section 2.1.2.3 of this SER.

The diesel generator building ventilation subsystem, service water pumphouse ventilation subsystem, and safety-related chilled water system of the MBVCS perform safety functions because loss of heat removal capability of any of these subsystems could result in failure of components credited for accident mitigation. Each of the subsystems is powered by separated Class 1E power supplies. Operation of the service water pumphouse ventilation system and the chilled water system are automatically initiated by receipt of a safety injection or loss of offsite power signal. Safety-related systems are monitored and alarms are annunciated in the control room. These subsystems are further discussed below.

Diesel Generator Building Ventilation Subsystem:

The diesel generator building ventilation subsystem is an ESF system. The main components of the system for each diesel room include two 50 percent-capacity ventilation fans to supply outside air to the diesel generator room, the diesel generator electric equipment room, and the diesel generator cable-pipe-basement area.

The fans of the system cycle and associated dampers open and close in response to room thermostats located in the diesel generator rooms and diesel generator electric equipment rooms when the diesel generators are not operating. Both fans associated with a diesel generator room start automatically and operate continuously whenever the diesel generator in that room operates. Ventilation air is drawn through roof openings which are shielded from external tornado missiles and forced into the diesel generator room by fans.

Service Water Pumphouse Ventilation Subsystem:

The service water pumphouse ventilation subsystem is an ESF system. The main components of the subsystem include two 100 percent-capacity ventilation supply fans that provide outside air to various areas of the service water pumphouse.

Either of the two supply fans operates continuously during normal operating periods. Both fans start automatically following receipt of a safety injection or loss of offsite power signal. The fans are powered from separate Class 1E power sources.

The license renewal boundaries for the MBVCS are depicted on the following P&ID drawings:

- D-912-134, Diesel Generator Areas System Flow Diagram
- D-912-155, Service Water Intake Screen/Pump House Bldg. Vent. System Flow Diagram

In LRA Section 2.3.3.1 and UFSAR Section 9.4.1, the applicant identified the following intended functions for the MBVCS:

- to provide safety-related function of heat removal capability inside the diesel generator rooms and diesel generator electric equipment rooms by maintaining these areas at acceptable ambient air temperatures between minimum and maximum levels suitable for personnel and equipment

- to provide safety-related function of heat removal capability inside the service water pump/screen room areas, related motor control center, and electrical switchgear rooms by maintaining these areas at acceptable ambient air temperatures between minimum and maximum levels suitable for personnel and equipment

In LRA Table 2.3-18, the applicant identified the component types for the MBVCS that are subject to an AMR. In LRA Tables 3.3-1 and 3.3-2, the applicant identified the component types and commodities groups (combinations of materials and environments) that are within the AMP and are evaluated in the GALL Report.

Reactor Building Cooling and Filtering Systems

The reactor building cooling and filtering systems (RBCFS) ~~has~~ have the safety functions to (1) maintain the ambient air temperature at a suitable level for continuous operation of equipment within the building under normal operating and shutdown conditions, (2) provide cleanup of the reactor building atmosphere to minimize the release of radioactivity to the environment before purging, and (3) assist other heat removal systems during a post accident conditions.

The RBCFS consists of reactor building cooling system, reactor building purge supply and exhaust system, post accident hydrogen removal and alternate reactor building purge system, reactor building charcoal cleanup system, reactor building reactor compartment and cooling system, reactor building secondary compartment cooling system, reactor building refueling water surface system and rod position indication cooling system, reactor building CRDM shroud cooling system, and reactor building elevator machine room system. These systems are further described below.

Reactor Building Cooling System:

The two cooling units powered from channel-A of the Class 1E electric system are located on the opposite side of the reactor building from the two cooling units supplied from channel-B. Also the cooling water supply and return mains to these units are physically separated as is the A and B channel wiring. Each unit can operate independently of the others and the discharge from each unit is isolated from the common air supply main by gravity operated dampers Reactor building cooling system components that must remain intact following a LOCA include four plenums and all internal components, plenum discharge ducts, common air supply main, and six vertical supply ducts from the common air supply main to the lower elevation of the reactor building. The components noted above are designed to remain intact following a LOCA.

Each plenum includes moisture separators, HEPA filters, filter bypass opening and dampers, cooling coils, and an axial flow fan driven by separate high speed and slow speed motors.

The reactor building cooling unit fans operate at high speed during normal periods, and at slow speeds during post LOCA periods and reactor building leak rate testing. The units are serviced by cooling water from the industrial cooling system during normal operation and by service water system during post LOCA or loss of offsite power conditions. For normal operation, three out of four fans operate. For LOCA, one fan in each train operates.

The units, when operating in the normal mode, are tripped upon the receipt of a safety injection or loss of offsite power signal, and are then automatically started at slow speed in accordance with the ESF actuation system and the ESF loading sequence of the re ed emergency diesel generator.

The reactor building cooling units can be manually operated from the control room at either high or slow speed. In response to an ESF loading sequence signal, the unit speed selector switches in the control room can determine which one of 2A and one of 2B electrical power channel starts. The plenum unit HEPA filter bypass damper is in the open position during normal operation and is automatically closed upon receipt of a safety injection signal.

Reactor Building Purge Supply and Exhaust System:

Containment isolation is safeguarded through the use of redundant, fail closed, butterfly valves on both the purge supply and exhaust lines. Electrical interlocks allow no more than one valve of a redundant pair of containment isolation valves to be open unless the exhaust system is operating (one valve of the pair can be open for testing purposes). Automatic closure of the four containment isolation valves of this system occurs upon receipt of a containment isolation signal or a high radiation signal. These measures, combined with administrative control of system operation, ensure that containment air is not released to the atmosphere through uncontrolled paths. The purge supply and exhaust system are not required to operate during a post accident period. The purge supply and exhaust isolation valves, as noted above, isolate the containment and are redundant safety-related equipment.

Alternate Reactor Building Purge System:

Containment isolation is assured through the use of redundant, fail closed, gate valves on both the alternate purge supply and exhaust lines. Automatic closure of the four containment isolation valves in the alternate reactor building purge system occurs upon receipt of a containment isolation signal or a high radiation signal. These measures, combined with administrative control of system operation, ensure that containment air is not released to the atmosphere through uncontrolled paths. The alternate reactor building purge system containment isolation valves and accessories are safety-related.

Reactor Building Charcoal Cleanup System:

Redundancy of the reactor building cleanup units provides iodine removal capability even if one of the units is not available. This condition extends the required cleanup time prior to purging, but does not prevent eventual completion of system function. This system is not required to operate under accident conditions and is not supplied from emergency power sources. The system is not safety-related.

Reactor Building Reactor Compartment Cooling System, Secondary Compartment Cooling System, and CRDM Shroud Cooling System:

For each of the three systems, adequate redundancy of system components is provided to ensure that sufficient cooling capacity is delivered under varying conditions of component availability. These systems are not required to operate under accident conditions and are not safety-related.

Reactor Building Elevator Machine Room System:

The reactor building elevator machine room ventilation system operates in response to the room thermostat. The system has no post-accident safety function and is not safety-related.

The license renewal boundaries for the RBCFS are depicted on the following P&ID drawings:

- D-~~808~~ 912-102, RB Cooling System Flow Diagram
- ~~D-912-~~103, RB Purge Supply and Purge Exhaust Systems Flow Diagram
- D-912-105, RB Refueling Water Surface System Flow Diagram

In LRA Section 2.3.3.1 and UFSAR Section 9.4.1, the applicant identified the following intended functions for the RBCFS:

- to maintain an average reactor building air temperature below a maximum of 120 °F during normal power operation as assumed in the accident analyses and below 100 °F during refueling operations for personnel comfort and safety

- to maintain an average reactor building air temperature above 60 °F during shutdown conditions for personnel comfort and safety

- to provide forced air cooling in sufficient capacity to remove CRDM heat and reject it to the general reactor building atmosphere

- to provide reactor building cleanup capacity to reduce airborne radioiodine levels prior to personnel entry and to minimize radioactivity released during reactor building purging

In LRA Table 2.3-18, the applicant identified the component types for the RBCFS that are subject to an AMR. In LRA Tables 3.3-1 and 3.3-2, the applicant identified the component types and commodities groups (combinations of materials and environments) that are within the AMP and are evaluated in the GALL Report.

2.3.3.1.2 Staff Evaluation

Control Building Area Ventilation System

The staff reviewed LRA Section 2.3.3.1 and UFSAR Section 9.4.1 to determine whether the CBAVS components are within the scope of license renewal in accordance with 10 CFR 54.4 and are subject to an AMR in accordance with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1), respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of this review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the license renewal Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

During its review, the staff determined that additional information was needed to complete its review. In a letter dated March 28, 2003, the staff asked the applicant in RAIs 2.3.3.1-1 and

2.3.3.7.1 Summary of Technical Information in the Application

The applicant describes the emergency diesel generators (EDGs) and their support systems in LRA Section 2.3.3.7 and provides a list of components subject to an AMR in LRA Table 2.3-23. UFSAR Sections 9.5.4 and 9.5.8 provide additional information for the diesel generator services systems.

The EDG system consists of two EDGs and their support systems. The Rule recognizes that the EDGs are active components and are excluded from the group of equipment that are subject to an AMR, as required by 10 CFR 54.21(a)(1). The following are the support systems for each EDG:

- fuel oil storage and transfer system
- cooling water system
- starting air system
- lubrication system
- combustion air intake and exhaust system

The license renewal boundaries for the EDGs and their support systems are highlighted on the following P&ID drawings:

- D-302-222, Service Water System
- D-302-351, Diesel Generator Fuel Oil
- D-302-353, Diesel Generator Miscellaneous Service
- 1MS-32-005, Sheet 2, Fuel Oil System
- 1MS-32-005, Sheet 3, Lube Oil System
- 1MS-32-005 Sheet 4, Jacket Water System
- 1MS-32-005 Sheet 5, Intercooler & Injector Cooling System
- 1MS-32-005 Sheet 6, Starting & Control Air System
- 1MS-32-005 Sheet 7, Crank Case Vac Air Intake and Exhaust System

These supporting systems are further described in the following UFSAR Sections and are summarized as below:

- 9.5.4 Diesel Generator Fuel Oil Storage and Transfer System
- 9.5.5 Diesel Generator Cooling Water System
- 9.5.6 Diesel Generator Starting Air System
- 9.5.7 Diesel Generator Lubrication System
- 9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

<u>Fuel Oil Storage and Transfer System</u>

Each EDG fuel oil storage and transfer system consists of a day tank, a fuel oil storage tank, two fuel oil transfer pumps, and its associated piping, valves, and I&Cs. Each day tank is automatically filled by its own EDG fuel oil storage tank with its own EDG fuel oil transfer pumps. A cross-tie with two normally closed valves is provided between the two EDGs at the fuel oil transfer pump suctions that allows the fuel oil transfer pumps of either EDG to fill either or both day tanks from either fuel oil storage tank.

Cooling Water System

The cooling water system consists of two subsystems — intercooler subsystem and jacket water subsystem, as described below.

Intercooler Subsystem:

The intercooler subsystem supplies cooling water to the turbocharger air intercoolers, alternator outboard bearing, and fuel injection nozzles. Circulation of cooling water is accomplished by an engine-driven centrifugal pump. Heat from the cooling water is rejected to the service water system through a thermostatically controlled heat exchanger. An expansion tank mounted on top of a standpipe is provided to serve both the intercooler subsystem and the jacket water subsystem.

Jacket Water Subsystem:

The jacket water subsystem is a closed system that cools the diesel engine. Cooling water is circulated through the cylinder liners, cylinder heads, and turbocharger cooling spaces by an engine-driven pump. Heat from the cooling water is rejected to the service water system through a thermostatically controlled heat exchanger. An electric heater and an auxiliary motor-driven pump are provided to allow "keep warm" operation under standby conditions.

Air Starting System

Each EDG is provided with two independent air starting systems, one for each bank of engine cylinders. Each bank of engine cylinders has its own engine-driven air start distributor with a connection to each cylinder. Using either or both banks can start the engine. Compressed air is supplied by two air storage tanks which are charged by two separate a-c motor driven air compressors. Because each of the air storage tanks is designed to store sufficient compressed air that permits five successive EDG starts without recharge (e.g., using both air storage tanks, 10 successive EDGs can start without recharge), those portions of the system used for charging the air storage tanks have no safety function.~~and are non-critical quality element (CQE).~~ Therefore, the air compressors and associated equipment are not highlighted in the P&ID drawings as being within the scope of license renewal.

Term not used at VCSNS

Lubrication System

The lubrication system consists of three subsystems — engine lube oil subsystem, rocker lube subsystem, and auxiliary oil subsystem. The lube oil subsystem contains an engine-driven pump which draws oil through a suction strainer from the engine sump and delivers it to a thermostatically controlled lube oil cooler and then through a strainer to the main engine lube oil header. The header supplies oil to all main bearings under pressure and, through a pressure reducing valve, to the camshaft bearings, cam followers, fuel injection pumps, and valve push rods. This subsystem also provides oil to the crank pin journals for piston cooling, as well as to accessory gearing. A separate rocker lube subsystem supplies oil to each cylinder head rocker assembly. An auxiliary oil subsystem permits continuous prelubrication of the engines at "keep warm" temperature during standby.

Combustion Air intake and Exhaust System

including 20 gallons per minute (gpm) pressure maintenance pump (jockey pump), a sprinkler system installed in the diesel generator building, and fire hose stations in various buildings, which were excluded from the scope of license renewal, are required for compliance with 10 CFR 50.48. These concerns led to the issuance of RAIs, which were sent to the applicant in a letter dated March 28, 2003. The applicant responded to the RAI in letters dated June 12 and September 2, 2003, as discussed below.

In RAI 2.3.3.8-1(1), staff requested the applicant to provide the basis for excluding the FP piping leading to the alternate fire service (AFS) pump house, turbine building, a portion of the circulating ~~chilled~~ water (CW) pump house, and the FP components (including jockey pumps, valves, piping, fittings, and diesel fuel tanks) from the scope of license renewal and subject to an AMR. In a letter dated June 12, 2003, the applicant responded that the AFS pumps are not credited for FP, because these pumps were installed for fire service needs during the construction of the station and are no longer used. However, the applicant expanded the scope to include the jockey pump (20 gpm pressure maintenance pump) and associated piping and components in the scope of license renewal. The applicant further stated that the components added by this expansion of scoping are subject to screening. If screened in, the FPP will manage the aging of these components. In a letter dated September 2, 2003, the applicant stated that it had performed further review and determined that these components are passive, long-lived, and support a license renewal intended function as a pressure boundary for fire service system. The FPP will manage the aging of these components for the period of extended operation.

The staff reviewed the applicant's response and agrees with the applicant to include the jockey pump and all the associated valves, piping, and fittings installed in the turbine building in the scope of license renewal as a part of the FP SSCs subject to an AMR. The staff further agrees that the AFS pump is not part of the fire suppression system. Therefore, the staff concurs with the applicant that the AFS pump should not be within the scope of license renewal to meet 10 CFR 50.48.

By letter dated March 28, 2003, in RAI 2.3.3.8-1(2), the staff requested the applicant to provide basis for excluding hydrants from the license renewal scope. These hydrants are in the system flow diagram D-302-231, Sht. 2, at locations H12, K8, K9, K10, K11, and K12.

In response to RAI 2.3.3.8-1(2), dated June 12, 2003, the applicant clarified that the fire hydrants in question are associated with fire hose houses 8, 9, 10, 17, 18, 19, and 20. All these fire hose houses are located outside of the protected area and are not in scope.

The staff finds the applicant's response to RAI 2.3.3.8-1(2) to be acceptable.

By letter dated March 28, 2003, in RAI 2.3.3.8-1(3), the staff requested the applicant to explain why the FP piping, fitting, valves, and fire hose stations at the reactor building (at locations E5, E7, and E8), fire hose connections in the fuel handling building (at location B4), fire hose connection in the auxiliary building (at location B13), fire hose connection in the intermediate building (at location H4), and fire hose connections in the reactor building (at location E9) are not highlighted in the system flow diagram (D-302-231, Sht. 3) as components within the scope of license renewal.

In a letter dated June 12, 2003, the applicant stated that the portion of piping in question in the reactor building (locations E5, E7, and E8 on LRA drawing D-302-231, Sht 3) is normally

and E9)

isolated per 10 CFR Part 50, Appendix A, GDC 56. The highlighted portion of this piping is in scope for containment isolation only. The fire hose connections identified by the staff on drawing D-302-231, Sht. 3, in the fuel building (at location B4), auxiliary building (at locations B13), and the intermediate building (at location H4) are included in the expanded scope for license renewal. The applicant further stated that the components added by this expansion of scope are subject to screening. In a letter dated September 2, 2003, the applicant stated that the plant had performed further review and determined that these components are passive, long-lived, and support a license renewal intended function as a pressure boundary for fire service system. The FPP will manage the aging of these components for the period of extended operation.

The staff reviewed the applicant's responses and agrees with the applicant that fire hose stations should be included in the expanded scope for license renewal. The staff also agrees with the applicant's justification for excluding piping in the reactor building (at locations E5, E7, and E8 on LRA drawing D-302-231, Sht 3) from scope of license renewal and from an AMR, since this piping does not serve any pressure boundary function for the FP system. Therefore, the staff finds the applicant's response to RAI 2.3.3.8-1(3) to be acceptable.

In RAI 2.3.3.8-1(4), the staff requested that the applicant provide the basis for excluding portions of the FP piping, fittings, valves, and fire connections from the scope of license renewal. These components are shown on the system flow diagram (D-302-231, Sht. 4) in the turbine building (at locations D6, E6, E7, ~~E8~~, E9, ~~E10~~, F7, F8, F9, and F10). The staff disagrees with the applicant's QR designation to isolate portions of the FP components by manual valves. D10

In a letter dated June 12, 2003, the applicant stated that the fire hose stations are included in the expanded scope for license renewal. The applicant further stated that the components added by this expansion of scope are subject to screening. If screened in, FPP will manage the aging of these components. In a letter dated September 2, 2003, the applicant stated that the plant had performed further review and determined that these components are passive, long-lived, and support a license renewal intended function as a pressure boundary for the fire service system. The FPP will manage the aging of these components for the period of extended operation.

The staff reviewed the applicant's response and agrees with the applicant's decision to include the FP piping, fittings, valves, and fire hose stations in the expanded scope for license renewal. Therefore, the staff finds the applicant's response to RAI 2.3.3.8-1(4) to be acceptable.

In RAI 2.3.3.8-1(5), the staff requested that the applicant provide a basis for excluding FP piping, fittings, and valves from the scope of license renewal and subject to an AMR. These components are shown in system flow diagram (D-302-231, Sht. 5), in south area, El 412 (at locations J6 to J9), of the turbine building.

In a letter dated June 12, 2003, the applicant stated that the valve manifolds are included in the expanded scope for license renewal. The applicant further stated that the components added by this expansion of scope are subject to screening. If screened in, FPP will manage the aging of these components. In a letter September 2, 2003, the applicant stated that the plant had performed further review and determined that these components are passive, long lived, and

support a license renewal intended function as a pressure boundary for fire service system. The FPP will manage the aging of these components for the period of extended operation.

The staff reviewed the applicant's response and agrees with the applicant that the valve manifolds should be included in the expanded scope for license renewal. Therefore, the staff finds the applicant's response to RAI 2.3.3.8-1(5) to be acceptable.

In RAI 2.3.3.8-1(6), the staff requested that the applicant provide basis for excluding the carbon dioxide ($CO_2$) system electric control panels and the IF&S system (in P&ID drawing D-302-232) from the scope of license renewal.

In response to RAI 2.3.3.15-5, dated June 12, 2003, the applicant clarified that the $CO_2$ system electric control panels and IF&S system are not within scope because these are active components.

The staff reviewed the applicant's response and finds the applicant's response to RAI 2.3.3.8-1(6) to be acceptable.

In RAI 2.3.3.8-1(7), the staff requested that the applicant provide basis for excluding the valve station system from the scope of license renewal. The system is shown in system flow diagram (1MS-55-059) in the turbine building. These FP components perform a pressure boundary intended function with the rest of the FP water supply system that is in scope.

In a letter dated June 12, 2003, the applicant stated that the valve manifolds will be included in the expanded scope for license renewal. The applicant further states that the components added by this expansion of scope are subject to screening. If screened in, the FPP will manage the aging of these components. In a letter September 2, 2003, the applicant stated that the plant had performed further review and determined that these components are passive, long-lived, and support a license renewal intended function as a pressure boundary for the fire service system. The FPP will manage the aging of these components for the period of extended operation. [It should be noted that this is the same valve manifold as in RAI 2.3.3.8-1(5), but different drawing.]

The staff reviewed the applicant's response and agrees with the applicant's decision to include the valve manifolds in the expanded scope for license renewal. Therefore, the staff finds the applicant's response to RAI 2.3.3.8-1(7) to be acceptable.

By letter dated March 28, 2003, in RAIs 2.3.3.8-1(8) and (10), the staff requested the applicant to justify why the preaction sprinkler system should not be in scope. The system is installed in the diesel generator building and diesel fire pump room (as shown in system flow diagram 1MS-55-085, Sht. 26).

In its response dated June 12, 2003, the applicant stated that the fire suppression system for the diesel generator building and diesel fire pump on drawing 1MS-55-085, Sht. 26, should be highlighted as in scope. The system is listed as an FPER system by the plant procedures that control the requirements for the FPP. The components in this system are subject to an AMR and are encompassed by the component types listed in LRA Table 2.3.24.

The staff reviewed the applicant's response and agrees with the applicant that the fire suppression system is within the scope of license renewal. The staff, therefore, finds the applicant's response to RAIs 2.3.3.8-1(8) and (10) to be acceptable.

By letter dated March 28, 2003, in RAI 2.3.3.8-1(9), the staff requested the applicant to justify why the manual deluge sprinkler system for the charcoal filter plenum (XAA-40A-AH and XAA-40b-AH) is not in scope. The system is in the auxiliary building, as seem in system flow diagram 1MS-55-085-27-2.

In their response dated June 12, 2003, the applicant stated that the emergency safeguards feature filter system (i.e., control room emergency filter plenums and fuel handling charcoal exhaust fire suppression system) is within the scope of license renewal, but the manual deluge sprinkler system installed in charcoal filter plenums in the auxiliary building is not in scope.

The staff review NUREG-0717, and its supplements, and the CLB for fire suppression in all areas of the plant. The staff noted in NUREG-0717 (Supplement 3, August 1982) that no automatic fire suppression system is required in charcoal filter plenums located in rooms 85-01, 88-25, 97-02, 00-02, 12-11 North, and 36-18 of the auxiliary building. The staff, therefore, finds the applicant's response to RAI 2.3.3.8-1(9) to be acceptable.

2.3.3.8.3 Conclusions

On the basis of its review described above, the staff concludes that the applicant has adequately identified the FP SSCs that are within the scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively.

*2.3.3.9 Fuel Handling System*

2.3.3.9.1 Summary of Technical Information in the Application

The applicant describes the fuel handling system in LRA Section 2.3.3.9 and provides a list of components subject to an AMR in LRA Table 2.3-25. The system is further described in UFSAR Section 9.1.4, Fuel Handling System.

The fuel handling system consists of the equipment needed for transporting and handling fuel. The associated fuel handling structures may be generally divided into the (1) refueling cavity, (2) refueling canal and fuel transfer canal, which are flooded during plant shutdown for refueling, (3) spent fuel pool, which is kept full of water and is accessible to operating personnel, and (4) new fuel storage area. A fuel transfer tube connects the refueling canal and the fuel transfer canal. This tube is fitted with a blind flange on the refueling canal end and a gate valve on the fuel transfer canal end. This blind flange is always in place, except during refueling, to ensure containment integrity. The fuel transfer tube is required to maintain pressure boundary integrity.

2.3.3.9.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.9 and UFSAR Section 9.1.4 to determine whether the fuel handling system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1), respectively. The staff's

The system flow diagram drawing, E-302-742, rev. 11 (waste processing) does not identify the heat-exchanger-shell-chemical-drain piping and valve 7938A to be within the scope of license renewal. This piping and the housing of the valve provide a pressure retaining function. The staff believed that these components are long-lived with passive function and, therefore, should be within the scope of license renewal and subject to an AMR. In a letter dated March 28, 2003, in RAI 2.3.3.10-2, the staff requested the applicant to justify their exclusion of these components from the scope of license renewal and subject to an AMR.

In its response, dated June 12, 2003, the applicant stated that the piping up to and including valves 7938A and 7938B are within scope. Drawings E-302-742, 743, 744, and 745 incorrectly show the safety class as "QRG" instead of "safety class ~~2~~." The staff finds the applicant's response acceptable because the component in scope is clarified.

2b (Code Class 3)

2.3.3.10.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found except a scoping boundary drawing was not supplied with its application. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the GWPS that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the GWPS that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.11 Industrial Cooler System*

2.3.3.11.1 Summary of Technical Information in the Application

The applicant describes the industrial cooler system in LRA Section 2.3.3.11. The applicant did not identify any components of this system subject to an AMR in LRA. The system is further described in UFSAR Section 9.4.7.2.5, Industrial Cooling System.

The industrial cooler system is a closed cooling system that supplies water to the cooling coils of the reactor building cooling units during normal operation. The service water system cools the reactor building cooling units during post-accident conditions following a loss of offsite power. The activation of an ESF actuation system signal automatically transfers the source of cooling water for the reactor building cooling units.

The only license renewal intended function of the industrial cooler system is to maintain reactor building temperature monitoring capability during accident conditions. The applicant stated that there are no mechanical components or component types required for the industrial cooler system to perform its system intended function, thus requiring no AMR.

2.3.3.11.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.11 and UFSAR Section 9.4.7.2.5 to determine whether the industrial cooler system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff's review

was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of the review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted. As a result of this review, the staff did not find any omissions.

2.3.3.11.3 Conclusions

The staff reviewed the LRA to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the industrial cooler system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the industrial cooler system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.12 Instrument Air Supply System*

2.3.3.12.1 Summary of Technical Information in the Application

The applicant describes the instrument air system in LRA Section 2.3.3.12 and provides a list of components subject to an AMR in LRA Table 2.3-27. The system is further described in UFSAR Section 9.3.1, Compressed Air System.

The instrument air system, including the reactor building air system, provides clean, dry air for instruments and controls. This system is not safety-related, with the exception of the containment isolation valves for the reactor building air system and the piping between them. The containment isolation valves for the reactor building air system and the piping between them are nuclear safety-related and in scope for license renewal because they form part of the containment isolation boundary. With the exception of a few components, instruments and controls served by the instrument air system fail in a safe position after a loss of air pressure. The following valves require air pressure to be placed in a safe position for certain design basis events—the feedwater isolation valves, the control room outside air dampers, the emergency feedwater system control valves, and the turbine driven emergency feedwater pump steam isolation valve. These air-operated devices are equipped with safety-related air volume tanks or accumulators, and these components are in scope for license renewal. Also in scope for license renewal are the air accumulators and associated air components for various valves required to perform a specified manipulation for event mitigation.

2.3.3.12.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.12 and UFSAR Section 9.3.1 to determine whether the instrument air system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1), respectively. The staff's

by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the nuclear sampling system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the nuclear sampling system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.17 Radiation Monitoring System*

2.3.3.17.1 Summary of Technical Information in the Application

The applicant describes the radiation monitoring system in LRA Section 2.3.3.17 and provides a list of components subject to an AMR in LRA Table 2.3-31. UFSAR Section 11.4 and Table 11.4-1 provide additional information for the system.

As indicated in the LRA, the license renewal review boundaries are depicted on the following P&ID drawings:

- D-302-611, Component Cooling
- D-302-651, Spent Fuel Cooling
- D-302-771, Nuclear Sampling
- D-806-010, Radiation Monitoring System Diagram Area Gamma
- D-806-011, Radiation Monitoring System Diagram Area Gamma

The radiation monitoring system is designed to monitor process and effluent streams from the plant in order to record and control releases of radioactive materials generated in the plant as a result of normal operations and during postulated accidents. The system continuously monitors plant effluent discharge paths under steady-state, transient, or accident conditions. After an accident, the system provides information to aid in determining the magnitude of the accident.

The following plant systems are monitored by the radiation monitoring system:

- component cooling water system
- primary coolant letdown system
- spent fuel cooling water system
- boron recycle system

RECOMMEND DELETING SINCE VCSNS MONITORS MORE SYSTEMS THAN THIS. THIS WAS TAKEN OUT-OF-CONTEXT FROM FSAR.

The radiation monitoring system has an intended function to provide post-accident monitoring capability for the containment activities. The system control panel and alarm in the control room are part of the control instrumentation that are reviewed with the control room instrumentation. The system's monitor assemblies, detectors, effluent flow measurement, and meteorological instrumentation are the active components of the system that are not within the scope of license renewal. In LRA Table 2.3-31, the applicant lists pipe, tanks, tube and tube fittings, and valves (body only) as the components of the radiation monitoring system subject to an AMR. These components are passive and perform their intended function without moving parts or without a change in configuration or properties, and they are not subject to replacement based on a qualified life or specified time period.

2.3.3.17.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.17, UFSAR Section 11.4, and the P&ID drawings to determine whether the components of the radiation monitoring system within the scope of license renewal and subject to an AMR had been identified in accordance with the requirements of 10 CFR 54.4(a) and 54.21(a)(1), respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In performing this review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4(a) to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

During its review, the staff determined that additional information regarding some components in the system was needed to complete its review. LRA Section 2.3.3.17 indicates that one of the license renewal functions of the radiation monitoring system is to maintain system boundaries with the component cooling system, spent fuel cooling system, and chemical and volume control system (CVCS). The license renewal boundary drawings, D-302-611 (component cooling), D-302-651 (spent fuel cooling), and D-302-771 (nuclear sampling) highlight the piping and components within the scope of license renewal for these systems. However, the components of the radiation monitoring system in scope are not defined on these drawings. In a letter dated March 4, 2003, in RAI-2.3.3.17-1, the staff requested the applicant to highlight the license renewal boundaries for the radiation monitoring system in these P&ID drawings.

In its response dated April 3, 2003, the applicant stated that the only license renewal intended function for the liquid radiation monitors shown on these drawings is as pressure boundaries for the component cooling, spent fuel cooling, and nuclear sampling systems. Drawing D-806-005, which was not depicted in LRA Section 2.3.3.17, is the radiation monitoring system drawing that shows all the components of the monitors for the component cooling, spent fuel cooling, and nuclear sampling systems. The applicant stated that drawing D-806-005, rather than P&IDs D-302-611, D-302-651, and D-302-771, should have been the reference for liquid radiation monitors. In addition, the area monitors on P&IDs D-806-010 and D806-011 are not included in the LRA. Because these radiation monitors provide the required post-accident containment monitoring capability and are environmentally qualified. These monitors perform the safety function using an ion chamber probe inserted into the atmosphere of the reactor building Therefore, its intended function is being performed by instrumentation, not by mechanical components. The instrumentation performs an active function and is excluded from the AMR, according to 10 CFR 54.21(a)(1).

The staff reviewed the applicant's response and additional drawings (i.e., D-806-005, D-806-010, and D-806-011) and found its rationale acceptable for defining the radiation monitoring system license renewal boundaries. The applicant has highlighted all the components of the radiation monitors on drawing D-806-005 that are within the scope of license renewal and listed pipe, tanks, tube and tube fittings, and valve in LRA Table 2.3-31 as the components subject to an AMR. As a result of this review, the staff did not identify any omissions.

2.3.3.17.3 Conclusions

The staff reviewed the LRA and the supplied P&ID drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were

found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the radiation monitoring system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the radiation monitoring system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.18 Reactor Makeup Water Supply System*

2.3.3.18.1 Summary of Technical Information in the Application

The applicant describes the reactor makeup water supply system in LRA Section 2.3.3.18 and provides a list of components subject to an AMR in LRA Table 2.3-32. UFSAR Section 9.2.7 and Table 9.2-17 provide additional information for the system.

The license renewal boundaries are depicted in the following P&ID drawings:

- D-302-651, Spent Fuel Cooling
- D-302-675, Chemical and Volume Control
- D-302-791, Reactor Makeup

The reactor makeup water supply system provides storage for the recycled primary coolant grade water. The system is designed to perform the following intended functions:

- supply water to the chemical and volume control system
- supply makeup water to the spent fuel pool
- provide a backup water supply for spray cooling in the pressurizer relief tank
- provide a water supply for makeup to and flushing of the reactor auxiliary systems
- provide storage capacity equal to or greater than the total of 84,000 gallon capacity of the recycle holdup tanks for the recycle primary coolant grade water produced in the boron recovery system and liquid waste processing system

NOT LICENSE RENEWAL INTENDED FUNCTIONS

The reactor makeup water pumps take suction from the reactor makeup water storage tank to perform various operations in makeup and flushing throughout the system. The portion of the reactor makeup water supply system between the reactor makeup water storage tank and the CVCS and spent fuel cooling system is safety-related, and the remainder of the system is non-safety-related.

2.3.3.18.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.18 and UFSAR Section 9.2.7 to determine whether the components of the reactor makeup water supply system within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4(a) and 54.21(a)(1),

respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In performing this review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4(a) to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The staff reviewed LRA Table 3.2-32 and the P&ID drawings and did not find any omissions, except for a question regarding flow restrictors. Drawing D-302-791 highlights flow restrictors (i.e., xps-009-mu and xps-158-mu) as components of the reactor makeup water supply system within the license renewal scope. However, these components are not included in LRA Table 3.2-32. The flow restrictors are passive and long-lived and perform a pressure boundary intended function with the piping that is in scope. In RAI 2.3.3.18-2, the staff requested the applicant to clarify whether these flow restrictors should be in scope or justify their exclusion.

In its response, the applicant stated that these components are listed in Table 3.2-32 as the "orifices," that are subject to an AMR. As a result of this review, the staff did not identify any omissions.

2.3.3.18.3 Conclusions

The staff reviewed the LRA and the accompanying boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the reactor makeup water supply system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the reactor makeup water system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.19 Roof Drains System*

2.3.3.19.1 Summary of Technical Information in the Application

The applicant describes the roof drains system in LRA Section 2.3.3.19 and provides a list of components subject to an AMR in LRA Table 2.3-33. The roof drains system is not described in the UFSAR.

The roof drains system discharges water away from the demister banks and plenums of the reactor building cooling units (RBCUs). The RBCUs are capable of operation during emergency conditions with potential exposure to reactor building spray solution. The intended function of this system is to maintain the RBCU drain flow piping integrity. In LRA Table 2.3-33, the applicant lists "pipe" as component type subject to an AMR, as it serves as the pressure boundary for the roof drain system. The license renewal boundaries for the RBCU drains are depicted in P&ID drawing D-302-824.

2.3.3.19.2 Staff Evaluation

that the applicant has adequately identified the components of the spent fuel pool cooling system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.3.23 Thermal Regeneration System*

2.3.3.23.1 Summary of Technical Information in the Application

The applicant describes the thermal regeneration system (BTRS) in LRA Section 2.3.3.23 and provides a list of components subject to an AMR in LRA Table 2.3-37. The system is further described in UFSAR Section 9.3.4, Chemical and Volume Control System. The license renewal boundaries for the system are depicted in P&ID drawing E-302-676.

The LRA indicates that the load following capabilities of the (boron) thermal regeneration system were removed by plant modification MRF 21511. Now the BTRS continues to be used as the deborating demineralizers that reduce reactor coolant boron concentration towards the end of core life. The soluble neutron absorber (boric acid) concentration is controlled by the BTRS and the reactor makeup control system. The BTRS is also used to cool the letdown flow for enhanced reactor coolant pump (RCP) seal performance and to clean up the reactor coolant system (RCS) before shutting down the reactor. The letdown flow leaving the demineralizers may be directed to the BTRS. The coolant flows through the reactor coolant filter and then flows into the volume control tank through a spray nozzle on top of the tank. The BTRS is one of the subsystems of the CVCS that has an intended function to maintain a pressure boundary with the CVCS.

2.3.3.23.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.23, UFSAR Section 9.3.4, and the P&ID drawing to determine whether the components of the BTRS within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4(a) and 54.21(a)(1), respectively. The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In performing this review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

During its review, the staff determined that additional information regarding some components in the system was needed to complete its review. LRA Section 2.3.3.23 states that the BTRS is used as a deborating demineralizer to reduce reactor coolant boron concentration towards the end of core life. LRA Table 2.3-37 lists heat exchangers (channel head), heat exchangers (shell), heat exchangers (tubes), and heat exchangers (tube sheets) as the components of the BTRS subject to an AMR. LRA Table 2.3-8 lists heat exchangers as the components of the CVCS subject to an AMR. However, drawing E-302-676, which contains both the BTRS and the CVCS, shows that the heat exchangers are within the boundary of the CVCS. There are no heat exchangers in the boundary of the BTRS. In RAI 2.3.3.23-1, the staff requested the applicant to explain whether the heat exchangers in LRA Table 2.3-37 for the BTRS are those in LRA Table 2.3-8 for the CVCS and, if so, why the same heat exchangers are listed in both the tables.

In its response, the applicant stated that the letdown reheat, letdown chiller, and moderating heat exchangers are the components of the BTRS listed in LRA Table 2.3-37. The license renewal intended function for these components is to maintain a pressure boundary for the CVCS. The heat exchangers listed in LRA Table 2.3-8 are the CVCS heat exchangers for regenerative, excess letdown, seal water, and letdown. The license stated that drawings were highlighted during the screening process according to individual systems. Because there may be more than one system on a particular drawing, as in the case of E-302-676, the screening process resulted in multiple copies of a drawing showing highlighting for each system. These working copies are available on site for inspection. The drawings supplied to the NRC are composite drawings showing highlighting, in some instances, for multiple systems. Since the applicant has clarified that the heat exchangers listed in LRA Table 2.3-37 are the components of the BTRS being subject to an AMR, the staff finds the applicant's response acceptable.

The staff examined the SCs in LRA Table 2.3-37 to determine whether they are the only SCs that are subject to an AMR in accordance with 10 CFR 54.21(a)(1). On the basis of the above review, the staff did not find any omissions by the applicant.

2.3.3.23.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the BTRS that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

### 2.3.4 Steam and Power Conversion Systems

The steam and power conversion systems act as a heat sink to remove heat from the nuclear steam supply system and convert the heat generated in the reactor to the plant's electrical output.

#### *2.3.4.1 Auxiliary Boiler Steam and Feedwater System*

2.3.4.1.1 Summary of Technical Information in the Application

The applicant describes the auxiliary boiler steam and feedwater (AS) system in LRA Section 2.3.4.1 and provides a list of components subject to an AMR in LRA Table 2.3.38.

The AS system provides steam to various plant equipment, as required during all modes of operation. The system is non-safety-related and performs an intended function to isolate the section of the AS piping supplying the auxiliary building in order to prevent a high energy fluid piping rupture from affecting safety-related equipment in the auxiliary building. The license renewal boundaries of the system are depicted on the P&ID drawing, D-302-051, "Auxiliary Steam."

2.3.4.1.2 Staff Evaluation

The staff reviewed LRA Section 2.3.4.1 to determine whether the AS system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of the review, the staff selected system functions described in LRA Section 2.3.4.1 that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted. As a result of this review, the staff did not identify any omissions.

2.3.4.1.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has appropriately identified the components of the AS system that are within the scope of license renewal, as required by 10 CFR 54.4, and that the applicant has appropriately identified the components of the AS system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.3.4.2 Condensate System*

2.3.4.2.1 Summary of Technical Information in the Application

The applicant describes the condensate system in LRA Section 2.3.4.2 and provides a list of components subject to an AMR in LRA Table 2.3.39. The system is further described in UFSAR Section 10.4.7. 1, Condensate System.

The condensate system pumps condensed turbine exhaust steam from the main condenser hotwell through the low pressure feedwater heaters to maintain deaerator storage tank level for anticipated operating conditions. It also serves as a source of cooling water for the steam packing condenser and steam blowdown heat exchanger, and provides sealing water for various vacuum valves and feedwater pump seals. Except for the condensate storage tank (CST), the condensate system is non-safety-related. The CST is safety-related because it is the primary inventory source for the emergency feedwater system. The license renewal boundaries for the system are depicted on the P&ID drawings, D-302-085 and 1MS-17-125.

2.3.4.2.2 Staff Evaluation

The staff reviewed LRA Section 2.3.4.2 and UFSAR Section 10.4.7.1 to determine whether the condensate system components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1). The staff's review was conducted in accordance with Section 2.3 of the SRP-LR (NUREG-1800) and is described below.

In the performance of this review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

During its review, the staff identified that the 10-inch atmospheric vent pipe on top of the CST was not highlighted in drawing D-302-101(at location A12) as being in scope and subject to an AMR for license renewal. Also, this vent pipe was not shown on the CST in drawing D-302-085. The staff believes that the vent pipe has an intended function to provide vacuum protection for the tank and is in scope. By letter dated March 28, 2003, in RAI 2.3.4.2-1, the staff requested the applicant to explain why this 10-inch vent pipe was not within the scope of license renewal. In its response dated June 12, 2003, the applicant stated that the 10-inch vent pipe performs its function by not being a pressure boundary and plugging of this vent pipe is not a credible event. In addition, the inspection of the tank by the mechanical components program will detect any degradation of the vent pipe. inspections for

The staff reviewed the applicant's response and found it acceptable because the applicant has justified that the vent does not have a preserving pressure boundary function and is not in scope nor subject to an AMR for license renewal. The applicant further explained that this vent pipe was not shown on the CST in drawing D-302-085 because this drawing only shows emergency feedwater connections. The staff found the applicant's response acceptable because it provides the reason for not showing the vent pipe on LRA drawing D-302-085.

During its review, the staff also identified that the piping attached to the CST, and up to the first isolation valve, was not highlighted in drawing D-302-101 as components within the scope of license renewal and subject to an AMR. By letter dated March 4, 2003, in RAI 2.3.4.2-2, the staff asked the licensee to justify the exclusion of the piping attached to the CST from the scope of license renewal. By letter dated April 3, 2003, the applicant stated that the CST is designed to have a reserve volume dedicated for use by the emergency feedwater (EF) system, that the connections below this reserve volume, and only those, are designated as EF components and are, therefore, within the scope of license renewal. The applicant further explained that other tank connections are located above this reserve volume, do not affect the water supply to the EF system and, therefore, are not included in scope for license renewal. The staff found the applicant's response acceptable because it explains why some of the piping connected to the CST is not highlighted as components in the scope of license renewal. As a result of this review, the staff did not identify any ommissions.

2.3.4.2.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any SSCs within the scope of license renewal had not been identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components subject to an AMR had not been identified by the applicant. No omissions were found during the independent assessment. On the basis of this review, the staff concludes that the applicant has adequately identified the components of the condensate system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the condensate system that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

environment of the

are not induced into safety-related piping at safety-class boundaries. Code break supports protect essential equipment by extending the design requirements for safety-related piping beyond the class change until one support (at a minimum) in each of the three mutually perpendicular transverse directions (or the equivalent) is provided. Code break piping is within the scope of license renewal to preclude adverse effects on safety-related equipment and functions. The applicant evaluated the non-safety-related piping that is connected to safety-related piping to determine whether the ~~boundary of the~~ safety-related portions in scope are also applicable to the adjoining code break piping. ~~Specifically, seismic effects of non-safety-related piping are to be isolated from safety-related piping systems.~~ As a result, the applicant identified the non-safety-related piping between the non-safety-related cycle industrial cooler (CI) system and the safety-related service water (SW) system in the expended scope for license renewal to meet Criterion 2.

Non-Mechanical Component

The applicant indicated that anti-falldown requirements for various SCs are for structural supports rather than assuming the function of the supported mechanical components. The structural supports have been evaluated in LRA Sections 2.4 and 3.5. The applicant determined that no further evaluation is required per refined Criterion 2.

Insulation

The applicant assessed the insulation types (such as MIRROR, mechanically bonded glass fiber blanket, calcium/silicate, and fiberglass) for possible age-related degradation of insulation materials and their impact. The applicant did ~~not~~ identify any potential falling insulation materials on safety-related components. Therefore, ~~no~~ insulation needs to be included in the scope to meet refined Criterion 2.

INSULATION IS IN SCOPE
NO AMP REQUIRED

Ductwork

The applicant reevaluated the HVAC ductwork in the designated buildings to determine whether it is anti-falldown ductwork. The applicant ~~found that~~ those portions of the ductwork of concern ~~are already included~~ in the scope of license renewal. The existing IPA results are applicable to the anti-falldown ductwork. The applicant determined that no further evaluation is required per refined Criterion 2.

added

Pipe Failure/Rupture

Safety-related high energy piping and associated protection devices, such as restraints, barriers, and shields, were initially included in the license renewal scope and subject to an AMR. The applicant determined that no further evaluation is required per refined Criterion 2.

Analyzed High-Energy Lines

To maintain the seismic design and retain a safety margin, the applicant classified certain non-safety-related portions of several high energy lines as QR. The portions of the QR piping were initially included in the scope of license renewal. The applicant determined that no further evaluation is required per refined Criterion 2.

Unanalyzed High-Energy Lines

Portions of the piping in the steam generator blowdown (BD) system and several MS drains were not analyzed and were not initially included in the scope of license renewal. The applicant's reevaluation determined the non-safety-related BD system piping from the containment isolation valves to the turbine building/intermediate building wall to be included in the expended scope and subject to an AMR for license renewal per refined Criterion 2. The non-safety-related MS drains in the auxiliary building and intermediate building are also included in the scope of license renewal to meet refined Criterion 2.

Flow Limitation/Blockage

Certain non-safety-related portions of the mechanical systems were classified as QR to ensure that function of the system would not be inhibited by restricted flow during or after a seismic event. These QR portions were initially included in the scope of license renewal. The applicant determined that no further evaluation is required per refined Criterion 2.

Wetting (Moderate or High-Energy)

The effects of wetting on safety-related components, such as wetting from spray or leakage, are not explicitly addressed on building composite drawings. The areas identified on those drawings containing safety-related equipment are the areas where wetting due to failure of non-safety-related and/or QR fluid piping and components could adversely impact safety-related components. The applicant evaluated the non-safety-related and/or QR fluid systems for wetting considerations and ~~found that these system portions were initially~~ included in the scope of license renewal, ~~for other considerations.~~ them

Essential equipment in the reactor building is qualified for service in harsh environments, such as spray, steam, or flooding. The applicant's evaluation determined that failure of non-safety-related components will not result in the failure of safety-related components in that vicinity. Electrical equipment rooms and other unique locations are considered to be the most susceptible to spray/leakage concerns. Spray-proof enclosures are used for termination boxes, splice boxes, and for field-mounted equipment like fuse relays. Field-mounted devices, such as transmitters, limit switches, solenoid valves, and valve motor operators are also designed for spray-proof. The applicant's reevaluation found that all the safety-related components and equipment have been protected for wetting concerns.

Leakage cracks are postulated to occur in moderate-energy systems. UFSAR Section 3.6.2.1.4 provides information on the CLB of postulated moderate-energy piping leakage. The applicant evaluated all non-safety-related moderate and high-energy fluid systems in the areas of concern and found that the safety-related components and equipment have been protected from wetting due to leakage.

Flooding and Leak Detection

Flooding due to large amounts of leakage from system components into nearby areas may prevent the performance of a safety function. Systems that are credited for detection and isolation of leaks to preclude adverse effects on safety-related equipment and functions are within the scope of license renewal. The structural aspects of plant design (protective/mitigative

features) that preclude an adverse impact on safety-related components due to flooding are included in scope. The applicant reviewed current flooding analysis and plant design documents and concluded that no other SSCs needed to be included in the expanded scope for license renewal per refined Criterion 2.

As a result of this reevaluation, the applicant identified 34 systems that had their scope expanded to include non-safety-related systems and/or QR portions that have a potential for adverse spatial interactions with safety-related equipment in the designated buildings. With the exception of the interface between the safety-related SW system and non-safety-related CI system, the applicant found that these systems do not have to expand their ~~scoping~~ aging management review due to spatial effects because they are the same material and environment combination on each side of the code break. These systems were initially included in the scope of license renewal and either sides of the code break are subject to an AMR.

The interfaces between the SW system and CI system are at the supply and return valves of the RBCU. The process environment for the SW system (safety-related side of the code break) is raw water, while the CI system (non-safety-related side of the code break) is closed-cycle treated water. The SW system was included in the license renewal scope for its raw water environment, but CI piping was not selected for AMR even though the treated water is mixing with raw water. The applicant's reevaluation determined to include the CI system piping in the expended scope for license renewal and subject to AMR to meet refined Criterion 2.

Based on the above reevaluation of the plant systems, the applicant added the following non-safety-related systems to the expanded scope for license renewal due to the potential for spatial interactions with safety-related SSCs in the designated buildings:

- Condenser Air Removal (AR)
- Demineralized Water (DW)
- Fuel Oil Handling (FO)
- Hydrogen-Nuclear Plant Use (HN)
- Liquid Effluents from Nuclear Plant to Penstock (LW)
- Nuclear Blowdown Processing (NB)
- Nitrogen-Nuclear Plant Use (NN)
- Oxygen-Nuclear Plant Use (ON)
- Sewer (SE)
- RW Solidification & Solids Handling (WD)
- Excess Liquid Waste (WX)

The applicant identified the components of these systems to be subject to an AMR using a commodity approach rather than a systems approach. Systems, system portions, and components meeting only refined Criterion 2 were grouped together according to the material type and/or the environments experienced in the designated buildings. Table 1 of the technical report lists the commodities that were determined to meet refined Criterion 2 for an AMR that was not initially listed in the tables of LRA Section 2.3. Table 1 contains 17 groups of component types; each group is provided with information on material, environment, and AMP. Some of these piping systems, ventilation ductwork, and component insulations in the table were justified so that no AMP is required. These components in the table perform limited structural integrity or limited pressure boundary function instead of supporting a specific system intended function.

The staff reviewed the non-safety-related SSCs in the above specified areas to meet Criterion 2. Based on this review, the staff finds that the applicant has considered most aspects in assessing the anti-falldown components and justified the areas of concern that need no further evaluations. The reevaluation's primary focus was on piping components in the fluid systems. However, the portions of non-fluid containing mechanical system (e.g., ventilation ducts, instrument air valves, valve actuators, etc.) were not fully addressed in the report. Certain non-fluid components may not have safety functions but are spatially orientated near safety-related components, such that their failure could adversely impact the performance of an intended safety function. In a letter dated March 28, 2003, in RAI 2.3.5-1, the staff asked the applicant to explain whether any component groups that contain no fluids should be identified and reassessed to meet Criterion 2.

In its response, dated June 12, 2003, the applicant stated that piping and piping system components, ventilation ductwork, and pipe and component insulation were specifically included in the technical report. The evaluation in the technical report addresses all system piping and ductwork regardless of the internal environment (i.e., steam, treated water, raw water, gases, air, etc.). Piping and piping system components include valves, fittings, and various piping components located in the seismic portion of the piping. Ventilation ductwork includes damper housing when contained in the seismic portions of the system. Piping and component insulation was included as the portions or sections of insulation may support other sections. The applicant did ~~not~~ identify ~~any~~ non-fluid containing components that need to be added to the expanded scope per refined Criterion 2. as well as fluid-containing,

The staff reviewed the technical report and the applicant's response and found that the applicant had included all the non-safety-related SSCs with the configuration to meet NRC guidance and Criterion 2. Based on the above review, the staff concluded that the expanded scoping and additional SSCs identified in the technical report are acceptable.

*2.3.5.3 Conclusions*

The staff reviewed the information in the technical report, and its confirmation from the scoping inspection and did not find any omissions in the scoping and screening of the Criterion 2 SSCs. On the basis of its review, the staff concludes that the applicant has appropriately identified the Criterion 2 systems and components that are within the scope of license renewal and the Criterion 2 systems and components that are subject to an AMR in accordance with 10 CFR 54.4(a)(2) and 10 CFR 54.21(a)(1), respectively.

*2.3.5.4 References*


1. Technical Report RC-02-0159, "Criteria 2 Supplement to the Application for Operation License," September 12, 2002. Adams No. ML022630347.

2. NRC Letter to Nuclear Energy Institute, "License Renewal Issue: Scoping of Seismic I/I Piping Systems," December 3, 2001. Adams No. ML013380013.

3. NRC Letter to Nuclear Energy Institute, "License Renewal Issue: Guidance on the Identification and Treatment of Structures, Systems, and Components Which Meet 10 CFR 54.4(a)(2)," March 15, 2002. Adams No. ML020770026.

4. NRC Regulatory Guide 1.29, "Seismic Design Classification."

## 2.4 Scoping and Screening Results: Structures

This section addresses the structures' scoping and screening results for the VCSNS license renewal application. The structures consist of the following:

- Reactor Building (Section 2.4.1)
- Other Structures (Section 2.4.2)
- Auxiliary Building (Section 2.4.2.1)
- Control Building (Section 2.4.2.2)
- Diesel Generator Building (Section 2.4.2.3)
- Fuel Handing Building (Section 2.4.2.4) Handling
- Intermediate Building (Section 2.4.2.5)
- Turbine Building (Section 2.4.2.6)
- Service Water Pumphouse, Intake, and Discharge Structures (Section 2.4.2.7)
- Yard Structures (Section 2.4.2.8)

Pursuant to 10 CFR 54.21(a)(1) an applicant is required to identify and list SCs subject to an AMR. These are passive, long-lived SCs that are within the scope of license renewal. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results. Such a focus allows the staff to confirm that there is no omission of structural components that are subject to an AMR. If the review identifies no omission, the staff has the basis to find that the applicant has identified the structural components that are subject to an AMR.

### 2.4.1 Reactor Building

#### *2.4.1.1 Summary of Technical Information in the Application*

The applicant describes the reactor building in LRA Section 2.4.1 and provides a list of components subject to an AMR in LRA Table 2.4-2. The reactor building is described in UFSAR Section 3.8.1, Concrete Reactor Building. The reactor building is a post tensioned, reinforced concrete structure with an integral steel liner. The reactor building consists of a cylindrical wall, a shallow dome roof, and a foundation mat with a depressed incore instrumentation pit under the reactor vessel. The foundation mat bears on fill concrete that extends to competent rock. At the underside of the reactor building foundation mat, a tendon access gallery is formed into the top of the fill concrete. A retaining wall, extending approximately one quarter (1/4) of the way around the reactor building, protects the below-grade portions of the reactor building wall from the subgrade and groundwater. Adjacent buildings surround the remaining three-quarters (3/4) of the reactor building.

The reactor building shell is post-tensioned by ungrouted tendons. The cylindrical wall employs a three-buttress, 240-degree hoop tendon concept, with 115 vertical tendons and 150 hoop tendons. The dome contains a total of 99 tendons arranged in a three-way system with 33 tendons per band.

The reactor building is lined on the inside face with a carbon steel plate liner that forms an essentially leak-tight membrane sealing the entire reactor building for any postulated conditions which may be encountered throughout the operating life of the plant. At its base, in the haunch area, a truncated conical transition section tapers inward to accommodate the thickened concrete of the cylindrical shell. A dome closes the top of the cylindrical portion of the liner. The bottom of the liner consists of flat floor liner plates welded to anchors that are embedded in the mat concrete. The liner plate extends downward into the foundation mat to line the incore instrumentation pit, the reactor building sump, the incore instrumentation pit sump, the residual heat removal sumps, and the reactor building spray sumps. The incore instrumentation pit walls are lined with carbon steel plates, while the pit bottom and the walls of the incore instrumentation tunnel sump, and reactor building spray sump floors and sidewalls are lined with stainless steel plate. Small diameter circular overlay plates are welded to the liner plate to support piping, ducts, conduit, and electric cable trays. Studs or angle anchors are provided on the liner behind the attachment plates to transfer loads on the pads into the concrete shell.

All reactor building penetrations are anchored to the concrete reactor building wall or foundation mat so that loads are transferred from the penetrations to the concrete. All penetrations satisfy the requirements of 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors. Piping penetrations consist of a sleeve around the outside of the piping. The piping is joined to the sleeve inside the reactor building by an attachment plate. Outside the reactor building, piping is attached to the sleeve by an attachment plate or by a bellows assembly. Electrical penetration sleeves are provided to accommodate electrical and instrumentation cables that pass through the reactor building wall. The sleeves are welded to the reactor building inner reinforcing plates. The electrical leads are installed in the penetration assemblies that are bolted to the electrical penetration sleeve. Spare penetrations consist of sleeves passing through the reactor building wall with the liner reinforced around the sleeve. Both ends of the sleeve are sealed with butt-welded pipe caps.

A fuel transfer tube penetrates the reactor building connecting the refueling canal in the reactor building and the fuel transfer canal in the fuel handling building. This penetration consists of a pipe installed inside a sleeve. Two personnel airlocks are provided for access to the reactor building, each with two doors, one on the inside and one on the outside. Each door is sealed with double O-rings, which are tested and replaced when warranted by their condition. The O-rings are not long-lived components and therefore do not require an AMR. An equipment hatch, equipped with an inside-mounted hatch cover, is also provided for access to the reactor building. A concrete shield located outside the reactor building acts as a missile and biological shield. The hatch cover is sealed with double O-rings, which are tested and replaced when warranted by their condition. The O-rings are not long-lived and therefore do not require an AMR.

Table 2.4-2 lists 46 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended functions these structural component groups provide for:

- structural and/or functional support to safety-related equipment
- structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions

- flood protection barrier (internal and external flooding event)
- rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
- pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events
- radiation shielding
- shielding against high energy line breaks
- spray shield or curbs for directing flow
- missile barrier (internally or externally generated)
- pipe whip restraint
- shelter/protection to safety related equipment

*2.4.1.2 Staff Evaluation*

The staff reviewed LRA Section 2.4.1 and UFSAR Sections 3.8.1 and 3.8.3 to determine whether the reactor building structural components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of its review, the staff selected system functions described in the UFSAR that are set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-2 lists 46 component groups that require an AMR. These component groups are:

1. anchorage
2. achorage/embedments (exposed surfaces)
3. ~~bellows~~ (penetration)
4. cable tray and conduit
5. cable tray and conduit supports
6. checkered plate
7. compressible joints and seals
8. control board (refuel cavity crane)
9. crane rails and girders
10. electrical and instrument panels and enclosures
11. embedments
12. equipment component supports
13. equipment hatch
14. equipment pads
15. escape air lock
16. expansion anchors

17. fire barrier penetration seals
18. fire barriers (walls, ceilings and floors)
19. fire doors
20. flood curbs (concrete)
21. flood curbs (steel)
22. flood, pressure, and specialty doors
23. foundations
24. hatches (steel)
25. HVAC duct supports
26. instrument line supports
27. instrument racks and frames
28. jet barriers (concrete and steel)
29. lead shielding supports
30. liner plate
31. metal partition walls
32. metal siding
33. missile shields
34. penetrations (mechanical and electrical)
35. personnel air lock
36. pipe supports
37. pipe whip restraint
38. post-tensioning system
39. refueling canal liner plate
40. reinforced concrete – beams, columns, floor slabs, and walls
41. seismic joint filler
42. stair, platform, and grating support
43. structural steel – beams, columns, plates, and trusses
44. sump screens
45. sumps
46. tube track

The LRA states that the scoping process to identify systems and structures that satisfy the requirements of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3) is performed on systems and structures using documents which form the CLB and other information sources. The CLB for the VCSNS has been defined in accordance with the definition provided in 10 CFR 54.3. The key information sources that form the CLB include the UFSAR, technical specifications, and the docketed licensing correspondence. All safety-related structures at VCSNS are designated as Seismic Category I and are within the scope of license renewal. The classification of each structure has been previously determined and documented in UFSAR Table 3.2-2, Classification of Structures.

The LRA also states that the screening process is performed on each structure identified to be within the scope of license renewal. The process is to determine whether a structure or a structural component requires an AMR in accordance with 10 CFR 54.21(a)(1).

The LRA further states that the structural components are divided into major groupings based on materials of construction and operating environment to facilitate the AMRs. For each structural component subject to AMR, the internal and external operating environments to which the component is subjected are established. Operating environments are established based on

license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-6 lists 40 structural component groups that require an AMR. These component groups are: 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. cable tray and conduit, 4. cable tray and conduit supports, 5. caissons, 6. checkered plate, 7. compressible joints and seals, 8. crane rails and girders, 9. electrical and instrument panels and enclosures, 10. embedments, 11. equipment component supports, 12. equipment pads, 13. expansion anchors, 14. fire barrier penetration seals, 15. fire barriers (walls, ceilings and floors), 16. fire doors, 17. flood curbs (concrete), 18. foundations, 19. fuel transfer canal liner plate, 20. hatches (concrete), 21. hatches (steel), 22. HVAC duct supports, 23. instrument line supports, 24. instrument racks and frames, 25. lead shielding supports, 26. masonry block, brick walls, or knockdown walls, 27. metal siding, 28. missile shields, 29. neutron absorbing sheets in spent fuel pool—boraflex, 30. piers (concrete), 31. pipe supports, 32. reinforced concrete — beams, columns, floor slabs, and walls, 33. roof, 34. seismic joint filler, 35. spent fuel pool liner, 36. spent fuel storage rack, 37. stair, platform, and grating support, 38. structural steel - beams, columns, plates, and trusses, 39. sumps, ~~and 40.~~ tube track.

The staff has reviewed the information in LRA Section 2.4.2.~~2~~4 and the UFSAR. The staff finds that the applicant made no omissions in scoping and ~~screening~~ the fuel handling building for license renewal.

2.4.2.4.3 Conclusions

The staff reviewed the LRA to determine whether any SSCs within the scope of license renewal and subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the structural components of the fuel handling building that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the fuel handling building that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.4.2.5 Intermediate Building*

2.4.2.5.1 Summary of Technical Information in the Application

The applicant describes the intermediate building in LRA Section 2.4.2.5 and provides a list of components subject to an AMR in LRA Table 2.4-7.

The foundation system for the intermediate building consists of a reinforced concrete basement floor slab that acts in conjunction with a series of grade beams to transfer vertical loads to the reinforced concrete caissons, shear/bearing walls, and concrete piers. The shear/bearing wall foundations and reinforced concrete caissons are founded on competent bedrock. The piers are founded on fill concrete that extends beyond the reactor building and auxiliary building. Horizontal shears are transferred through the basement floor slab to the shear/bearing walls and to the control building base mat.

The intermediate building is a seismic Category I structure described in UFSAR Section 3.8.4.1.3. The superstructure is an L-shaped reinforced concrete shear wall (box type) structure containing two main floor levels above the foundation and extending up to the low roof. Above the low roof is a partial third floor of reinforced concrete and a high roof. The intermediate building is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the UFSAR.

Table 2.4-7 lists 42 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended functions these structural component groups provide for:

- structural and/or functional support to safety-related equipment
- structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions
- flood protection barrier (internal and external flooding event)
- rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
- pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events
- radiation shielding
- shielding against high energy line breaks
- spray shield or curbs for directing flow
- missile barrier (internally or externally generated)
- pipe whip restraint
- shelter/protection to safety-related equipment

2.4.2.5.2 Staff Evaluation

The staff reviewed LRA Section 2.4.2.5 and UFSAR Sections 3.8.4.1.3, 3.8.4.4.3, and 3.8.5.1.3 to determine whether the intermediate building structural components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of its review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-7 lists 42 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. battery racks, 4. blowout or blow-off panels, 5. cable tray and conduit, 6. cable tray and conduit supports, 7. caissons, 8. compressible joints and seals, 9. crane rails and girders, 10. duct banks, 11. electrical and instrument panels and enclosures, 12. embedments, 13. equipment component supports, 14. equipment pads, 15. expansion anchors, 16. fire barrier penetration seals, 17. fire barriers (walls, ceilings and floors), 18. fire doors, 19. flood curbs (concrete), 20. flood, pressure, and specialty doors, 21. foundations, 22. hatches (concrete), 23. hatches (steel), 24. HVAC duct supports, 25. instrument line supports, 26. instrument racks and frames, 27. jet barriers, 28. lead shielding supports, 29. metal siding, 30. metal spray shields, 31. missile shields, 32. piers, 33. pipe supports, 34. pipe whip restraint, 35. reinforced concrete — beams, columns, floor slabs, and walls, 36. roof slabs, 37. seismic joint filler, 38. stair, platform, and grating support, 39. structural steel—beams, columns, plates, and trusses, 40. sumps, 41. trenches, and 42. tube track.

The staff has reviewed the information in LRA Section 2.4.2.5. The staff finds that the applicant made no omissions in scoping and screening the intermediate building for license renewal.

2.4.2.5.3 Conclusion

The staff reviewed the LRA to determine whether any SSCs within the scope of license renewal and subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the structural components of the intermediate building that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the structural components of the intermediate building that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

*2.4.2.6 Turbine Building*

2.4.2.6.1 Summary of Technical Information in the Application

The applicant describes the turbine building in LRA Section 2.4.2.6 and provides a list of components subject to an AMR in LRA Table 2.4-8.

The foundation mat for the turbine building is comprised of a reinforced concrete mat supported by Zone III fill (graded crushed stone) material. The reinforced concrete pedestal foundation mats for the feedwater pumps and turbine generators are founded on fill concrete over bedrock. The turbine building is a non-seismic Category I structure as described in UFSAR Section 3.8.4.1.1. The superstructure of steel framing, metal siding, and metal roof deck is supported on a reinforced concrete substructure. The steel rigid frame structure is elastically supported at the operating floor, which acts as a diaphragm. The subsurface portion of the east, west, and south walls are reinforced concrete. The north wall is structural steel framing, with no siding, that abuts the control, intermediate, and diesel buildings. The entire building is separated from other buildings to prevent load transfer during seismic events.

The turbine building is designed to withstand the various combinations of dead and live loads, seismic loads, wind loads, tornado loads, and other generic design criteria loads as defined in the UFSAR. However, for earthquake loads and tornado wind loads, the turbine building is only

designed to the extent required to prevent damage to seismic Category I structures. The primary function of the turbine building is to house the turbine generators. The functiona requirement of the building in the event of an earthquake or tornado is that no portion of the building collapses and results in damage to seismic Category I structures.

Table 2.4-8 lists 34 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended functions these structural component groups provide for:

- structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions
- flood protection barrier (internal and external flooding event)
- rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
- pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events
- spray shield or curbs for directing flow
- missile barrier (internally or externally generated)
- shelter/protection to safety-related equipment
- source of cooling water

2.4.2.6.2 Staff Evaluation

The staff reviewed LRA Section 2.4.2.~~7~~ 6 and UFSAR Section 3.8.4.1.1 to determine whether the turbine building structural components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of its review, the staff selected system functions described in the UFSAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-8 lists 34 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. cable tray and conduit, 4. cable tray and conduit supports, 5. compressible joints and seals, 6. crane rails and girders, 7. duct banks, 8. electrical and instrument panels and enclosures, 9. embedments, 10. equipment component supports, 11. equipment pads, 12. expansion anchors, 13. fire barrier penetration seals, 14. fire barriers (walls, ceilings and floors), 15. fire doors, 16. flood curbs (concrete), 17. flood, pressure, and specialty doors, 18. foundations, 19. grating, 20. hatches (concrete), 21. hatches (steel), 22. HVAC duct supports, 23. instrument line supports, 24. instrument racks and frames, 25. masonry block, brick walls, or knockdown walls, 26. metal siding, 27. pipe supports, 28. reinforced concrete — beams, columns, floor slabs, and

walls, 29. roof, 30. seismic joint filler, 31. stair, platform, and grating support, 32. structural steel — beams, columns, plates, and trusses, 33. sumps, and 34. trenches.

The staff has reviewed the information in LRA Section 2.4.2.6 and the UFSAR. The staff finds that the applicant made no omissions in scoping and screening the turbine building for license renewal.

2.4.2.6.3 Conclusions

The staff reviewed the LRA to determine whether any SSCs within the scope of license renewal and subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified that the structural components of the turbine building that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the turbine building that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.4.2.7 Service Water Pumphouse, Intake, and Discharge Structures*

2.4.2.7.1 Summary of Technical Information in the Application

The applicant describes the service water pumphouse, intake, and discharge structures in LRA Section 2.4.2.7 and provides a list of components subject to an AMR in LRA Table 2.4-9.

Service Water Pumphouse

The foundation for the service water pumphouse consists of a reinforced concrete structural mat. The discharge pipe pits on the south side and the control areas on the west side of the service water pumphouse are supported by buried reinforced concrete columns, which extend to the supporting foundation mat. The entire structural mat is supported on compact fill that is in turn supported on a layer of in-situ soils (saprolite), then decomposed rock down to competent rock. The service water pumphouse is a seismic Category I structure described in UFSAR Section 3.8.4.1.7. The superstructure is a reinforced concrete building separated from the service water intake structure and from buried connecting pipes and electrical duct banks by flexible joints, which accommodate relative settlement and seismic movement.

The service water pumphouse is designed to withstand the various combinations of dead and live loads, OBE and SSE seismic loads, wind loads, tornado loads, and other generic design criteria loads as defined in the UFSAR. The primary function of the service water pumphouse is to house the service water pumps that pump water from the service water pond to supply the service water system. The service water pumphouse is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the UFSAR.

Service Water Intake And Discharge Structures

Service Water Intake Structure:

The foundation for the service water intake structure consists of a reinforced concrete mat supported by compacted fill material, except for a portion of the inlet end, which rests on in-situ soils.

The service water intake structure is a seismic Category I structure as described in UFSAR Section 3.8.4.1.8. The structure is a reinforced concrete rectangular box culvert with two reinforced concrete wing walls at the intake end. The foundation mat forms the floor of the structure. An expansion joint separates the service water intake structure from the service water pumphouse, which accommodates relative settlement and seismic movement. The structure extends into the service water pond and is mostly buried in the west embankment except for the intake end, which is submerged within the pond.

The service water intake structure is designed to withstand the various combinations of dead loads, OBE and SSE seismic loads, and other generic design criteria loads as defined in the UFSAR. The primary function of the service water intake structure is to extend the point at which water is drawn from the service water pond into the service water pumphouse. The functional requirement of the service water intake structure during and following a design basis event is that it does not collapse and result in a loss of supply water from the service water pond to the service water pumphouse.

Service Water Discharge Structure: (underline)

The foundation for the service water discharge structure consists of a reinforced concrete mat that bears partly on decomposed rock and partly on fill concrete that extends to the decomposed rock. The service water discharge structure is a seismic Category I structure as described in UFSAR Section 3.8.4.1.9. The structure is a reinforced concrete rectangular basin mostly buried in the west embankment of the service water pond. The foundation mat forms the floor of the basin. A 15-foot high abutment wall forms the west end of the basin, and a 3-foot high sill wall forms the east end. Wing walls form the north and south sides of the basin. Two 30-inch diameter service water pipes terminate at the abutment wall and are connected to the service water discharge structure by flexible connections.

The service water discharge structure is designed to withstand the various combinations of dead loads, OBE and SSE seismic loads, and other generic design criteria loads as defined in the UFSAR. The primary function of the service water discharge structure is to release service water into the service water pond. The functional requirement of the service water discharge structure during and following a design basis event is that it does not collapse and result in an interruption of service water discharge.

Table 2.4-9 lists 34 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended functions these structural component groups provide for:

- structural and/or functional support to safety-related equipment
- structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions
- flood protection barrier (internal and external flooding event)

- rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
- spray shield or curbs for directing flow
- missile barrier (internally or externally generated)

2.4.2.7.2 Staff Evaluation

The staff reviewed LRA Section 2.4.2.~~8~~7 and UFSAR Sections 3.8.4.1.7, 3.8.4.1.8, 3.8.4.1.9, 3.8.4.4.7, 3.8.4.4.8, 3.8.4.4.9, ~~3.8.5.1.7~~, 3.8.5.1.8, and 3.8.5.1.9 to determine whether the components of the service water pumphouse, intake, and discharge structures within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-9 lists 34 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. cable tray and conduit, 4. cable tray and conduit supports, 5. checkered plate, 6. compressible joints and seals, 7. crane rails and girders, 8. duct banks, 9. electrical and instrument panels and enclosures, 10. embedments, 11. equipment component supports, 12. equipment pads, 13. expansion anchors, 14. fire barrier penetration seals, 15. fire barriers (walls, ceilings and floors), 16. fire doors, 17. flood curbs (concrete), 18. flood, pressure, and specialty doors, 19. foundations, 20. grating, 21. hatches (concrete), 22. HVAC duct supports, 23. instrument line supports, 24. instrument racks and frames, 25. intake bays or canals, 26. intake screens, 27. missile shields, 28. pipe supports, 29. reinforced concrete — beams, columns, floor slabs, and walls, 30. roof slab, 31. seismic joint filler, 32. stair, platform, and grating support, 33. structural steel — beams, columns, plates, and trusses, and 34. sumps.

The staff has reviewed the information in LRA Section 2.4.2.~~2~~7 and the UFSAR. The staff finds that the applicant made no omissions in scoping and ~~screening~~ the service water pumphouse, intake and discharge structures for license renewal.

2.4.2.7.3 Conclusions

The staff reviewed the LRA to determine whether any SSCs pumphouse, that should be within the scope of license renewal and subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the structural components of the service water pumphouse, intake and discharge structures are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the service water pumphouse, intake and discharge structures that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

*2.4.2.8 Yard Structures*

2.4.2.8.1 Summary of Technical Information in the Application

The applicant describes the yard structures in LRA Section 2.4.2.8 and provides a list of components subject to an AMR in LRA Table 2.4-10.

The following structures are included in yard structures:

- Condensate Storage Tank Foundation
- Fire Service Pumphouse
- Electrical Manhole MH-2
- Earthen Embankments (Service Water Pond Dams, West Embankment, North Berm)
- Electrical Substation and ~~Relay House~~ Transformer Area

Condensate Storage Tank Foundation

The foundation for the condensate storage tank is designed to satisfy seismic Category I requirements as defined in UFSAR Sections 2.5.4.10.3 and 9.2.6. The foundation consists of a reinforced concrete mat supported by Zone III (graded crushed stone) fill material and an integral reinforced concrete ring wall that extends above the top of the foundation mat. The condensate storage tank is secured to the foundation by anchor bolts embedded in the ring wall. The interior area of the ring wall is filled with clean dry sand to form a sand mat beneath the tank. A reinforced concrete valve pit for the condensate storage tank drainpipe is integrated into the south side of the foundation.

The primary function of the condensate storage tank foundation is to support the nuclear safety-related condensate storage tank. The functional requirement of the foundation during and following a design basis event is that its failure would not result in a loss of the condensate storage tank contents.

Table 2.4-10 lists 11 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended functions provided for by these structural component groups:

- structural and/or functional support to safety-related equipment

- structural support to non nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions

Fire Service Pumphouse

The fire service pumphouse is a concrete block building described in the FPER Section 4.10. The building is founded upon the reinforced concrete circulating water intake structure. Hollow concrete blocks are used to form the exterior and interior walls of the building, and solid concrete blocks are used under steel framing members. The composite roof is a built-up insulated roof with gravel over steel decking and metal roof trusses. A reinforced concrete slab, located on the east side of the fire service pumphouse and founded upon the circulating water intake structure, is the foundation for the diesel engine-driven fire service pump fuel oil tank. The tank is secured to the foundation by embedded anchor bolts. The primary function of the fire service pumphouse is to house one electric motor-driven fire pump and one diesel engine-driven fire pump.

Table 2.4-14 lists 10 structural component groups requiring an AMR, provides a reference to the results of the AMR for each component group, and identifies the following intended function provide these structural component groups:

- structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions

2.4.2.8.2 Staff Evaluation

The staff reviewed LRA Section 2.4.2.8 and UFSAR Section 3.8.4 to determine whether the yard structures components within the scope of license renewal and subject to an AMR had been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

Table 2.4-10 lists 11 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. checkered plate, 4. expansion anchors, 5. foundation dowels, 6. foundations, 7. instrument line supports, 8. instrument racks and frames, 9. pipe supports,10. reinforced concrete — beams, columns, floor slabs, and walls, and 11. stair, platform, and grating support.

Table 2.4-11 lists 25 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. battery racks, 4. cable tray and conduit, 5. cable tray and conduit supports, 6. electrical and instrument panels and enclosures, 7. embedments, 8. equipment component supports, 9. equipment pads, 10. expansion anchors, 11. fire barrier penetration seals, 12. fire barriers (walls, ceilings and floors), 13. fire doors, 14. flood curbs (concrete), 15. foundations, 16. hatches (steel), 17. HVAC duct supports, 18. instrument line supports, 19. instrument racks and frames, 20. masonry block, brick walls, or knockdown walls, 21. pipe supports, 22. reinforced concrete — beams, columns, floor slabs, and walls, 23. structural steel—beams, columns, plates, and trusses, 24. sumps, and 25. trenches.

Table 2.4-12 lists 5 structural component groups that require an AMR. These component groups are 1. foundations, 2. manhole covers, 3. manholes, 4. missile shields, and 5. reinforced concrete — beams, columns, floor slabs, and walls.

Table 2.4-13 lists 2 structural component groups that require an AMR. These component groups are 1. service water pond dams (north dam, south dam, and east dam) and west embankment, and 2. north berm.

Table 2.4-14 lists 10 structural component groups that require an AMR. These component groups are 1. anchorage, 2. anchorage/embedments (exposed surfaces), 3. cable tray and conduit, 4. cable tray and conduit supports, 5. electrical and instrument panels and enclosures, 6. embedments, 7. equipment component supports, 8. equipment pads (buslines, PCBs, and transformers), 9. reinforced concrete — foundations and walls, 10. structural steel—beams, columns, plates, and trusses (transmission towers).

The staff has reviewed the information in LRA Section 2.4.2.~~2~~8 and the UFSAR. The staff finds that the applicant made no omissions in scoping and ~~screening~~ the yard structure for license renewal.

2.4.2.8.3 Conclusions

The staff reviewed the LRA to determine whether any SSCs within the scope of license renewal and subject to an AMR had not been identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that the applicant has adequately identified the structural components of the yard structures that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the yard structures that are subject to an AMR, as required by 10 CFR 54.21(a)(1).

## 2.5 Scoping and Screening Results: Electrical and Instrumentation and Control

The applicant identified electrical and I&C component commodity groups subject to an AMR in Section 2.5, "Scoping and Screening Results: Electrical and Instrumentation and Control," of the LRA. The staff reviewed this section of the LRA to determine that all electrical component commodity groups, which are subject to an AMR as required by 10 CFR 54.21(a)(3), have been identified as required by 10 CFR 54.4(a) and 10 CFR Part 54.21(a)(1).

### 2.5.1 Summary of Technical Information in the Application

The applicant developed a listing of electrical and I&C component commodity groups for systems and structures within the scope of license renewal as well as active/passive determinations following the guidance of NEI 95-10, Appendix B. No commodity groups beyond those listed in Appendix B to NEI 95-10, were identified by the applicant for VCSNS.

The applicant reviewed these electrical component commodity groups (determined to be passive) to identify those that are not subject to replacement based on a limited qualified life or specified time period.

Based on its review, the applicant determined that the following electrical and I&C component commodity groups are subject to an AMR:

- insulated cables, connectors, splices, electrical penetration assemblies, and terminal blocks that are not covered by the VCSNS 10 CFR 50.49 EQ program
- high voltage electrical switchyard bus
- high voltage transmission conductors and connections
- high voltage insulators.

All other electrical and I&C component commodity groups are either (a) active (active/passive screening), (b) subject to replacement based on a qualified life or specified time period (long lived screening), or (c) not subject to an AMR because they do not perform any intended functions (scoping).

### 2.5.2 Staff Evaluation

Section 2.1 of the LRA, Scoping and Screening Methodology, discussed the scoping methodology as it related to the safety-related criteria pursuant to 10 CFR 54.4(a)(1), non-safety-related criteria pursuant to 10 CFR 54.4(a)(2), and regulated events pursuant to 10 CFR 54.4(a)(3). Following the determination of the systems and structures within the scope of license renewal, the applicant implemented a process for determining which components, among those systems and structures that were determined to be within scope of license renewal, would be subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

*2.5.2.1 Identification of Passive Components*

The applicant developed a listing of passive electrical and I&C component commodity groups for systems and structures within the scope of license renewal following the guidance of NEI 95-10 (Revision 3), Appendix B. No commodity groups, beyond those listed in Appendix B to NEI 95-10 (Revision 3), were identified by the applicant for VCSNS.

Guidance of NEI-95-10, Appendix B, utilized by the applicant for active/passive screening determinations, identifies the following passive electrical and I&C component commodity groups from typical nuclear plant systems and structures:

- cables and connections, bus, electrical portions of electrical and i&c penetration assemblies (e.g., electrical penetration assembly cables and connections, connectors, electrical splices, terminal blocks, power cables, control cables, instrument cables, insulated cables, communication cables, uninsulated ground conductors, transmission conductors, isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, switchyard bus)

- elements, resistance temperature detectors (RTD), sensors, thermocouples, transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors, watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, variable transducers, vibration transducers, voltage transducers) [passive for a pressure boundary, if applicable]

- high-voltage insulators (e.g., porcelain switchyard insulators, transmission line insulators)

Passive components (for which aging degradation is not readily monitored) are those that perform an intended function without moving parts or without a change in configuration or properties. As examples of passive components, 10 CFR 54.21(a)(1)(i) provides a list including, but not limited to, electrical penetrations, cables, and connections; and excluding, but not limited to, motors, diesel generators, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies.

The staff reviewed the above identified component commodity groups to verify that the applicant had not omitted any passive component commodity groups and the groups identified met the above defined passive screening criteria and/or examples provided in 10 CFR 54.21(a)(1)(i). The staff concluded that the above identified component commodity groups are consistent with the examples of passive components listed in 10 CFR 54.21(a)(1)(i), and are therefore considered acceptable. In addition, these component commodity groups are the same as the passive determinations described in NEI 95-10 (Revision 3), Appendix B, for component commodity groups typically found in nuclear plants in the electrical category. The staff has reviewed these NEI determinations and concluded (1) that each component commodity group identified performs its intended function without moving parts or without a change in configuration or properties, and its aging degradation is not readily monitored and (2) that these component commodity groups acceptably identify passive components pursuant to 10 CFR 54.21(a)(1)(i). Therefore, the staff agrees that the above identified subgroup of electrical component commodity groups represents the passive electrical component commodity groups that would be required to be included in an AMR if they also met scoping and long-lived screening criteria. and (3) fuse blocks / fuse clips will be added as a part of the cable and connections commodity group.

*2.5.2.2 Identification of Components that are Passive but Not Long-Lived*

From the above electrical and I&C component commodity groups determined to be passive, the applicant identified the following component commodity groups as not meeting long-lived screening criteria and thus not subject to an AMR:

- insulated cables and connections and terminal blocks that are included in the VCSNS 10 CFR 50.49 EQ program

- electrical portions of electrical and I&C penetration assemblies that are included in the VCSNS 10 CFR 50.49 EQ program

A component that is not replaced either (1) on a specified interval based on the qualified life of the component or (2) periodically in accordance with a specified time period, is deemed to be "long-lived," and therefore subject to an AMR.

Components subject to EQ aging requirements pursuant to 10 CFR 50.49(e)(5) are required to be replaced or refurbished at the end of their designated life. These components, pursuant to 10 CFR 50.49(e)(5), are subject to replacement based on a qualified life or specified time period. The applicant in the LRA indicated that the above identified components are included in its 10 CFR 50.49 EQ program and subject to aging requirements of 10 CFR 50.49(e)(5). The staff, therefore, agrees that the above identified components do not meet long-lived screening criteria and are thus not subject to an AMR.

*2.5.2.3 Identification of Components Not Within the Scope of License Renewal*

In its review, the staff noted that the applicant had not identified the following passive component commodity groups as within the scope of 10 CFR 54.4(a):

- uninsulated ground conductors

- isolated-phase bus, nonsegregated-phase bus, segregated-phase bus

- elements, RTDs, sensors, thermocouples, and transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors, watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, variable transducers, vibration transducers, voltage transducers)[passive for a pressure boundary only, if applicable]

As part of its review, the staff requested the applicant to explain how each of these passive component commodity groups were found not to meet any of the scoping criteria of 10 CFR 54.4(a).

Elements, RTDs, Sensors, Thermocouples, and Transducers — Section 2.5 of the LRA indicates that the passive electrical component commodity group of elements, RTDs, sensors, thermocouples, and transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors, watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, variable transducers, vibration transducers, voltage transducers) that are passive because of their pressure boundary function were found not to meet any of the scoping criteria of 10 CFR 54.4(a). Consequently, Section 2.5 of the LRA indicated that this commodity group is considered outside the scope of license renewal. In a followup question, the staff requested that the response to RAI 2.5-1 (requested by letter dated March 28, 2003) be expanded to explain why this commodity group was found not to meet any of the scoping criteria of 10 CFR 54.4(a). In its response dated June 12, 2003, the applicant stated that from an electrical standpoint, the "Elements" commodity group is active, and from a pressure boundary standpoint, these elements are not pressure boundary at VCSNS, and were, thus, screened out of consideration.

Based on its review, the staff concludes that there is no omission of electrical components (or elements) at VCSNS that could maintain a pressure boundary; therefore, the screening of this "Elements" commodity group from the scope of license renewal is considered acceptable.

Isolated-phase bus, nonsegregated-phase bus, segregated-phase bus — Section 2.5 of the LRA indicates that the passive electrical component commodity group of isolated-phase bus, nonsegregated-phase bus, and segregated-phase bus were found not to meet any of the scoping criteria of 10 CFR 54.4(a). Consequently, Section 2.5 of the LRA indicated that this "Bus" commodity group is considered outside the scope of license renewal. By letter dated March 28, 2003, the staff requested, in RAI 2.5-1, the applicant to explain why this "Bus" commodity group was found not to meet any of the scoping criteria of 10 CFR 54.4(a). In its response dated June 12, 2003, the applicant stated the following:

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

In addition, in its response to RAI 2.5-4 dated June 12, 2003, the applicant stated the following:

The EG system provides for the transmission of power from the site. The handling of plant loads, which are in the LR scope, is provided by one of the two preferred paths of offsite power, which do not include system EG [reference FSAR 8.1]. The Main Generator bus is not used by either of the two preferred sources of offsite power and is isolated by the associated substation 230 KV circuit breaker OCB-8892. The main electrical generator bus is not subject to aging management because it does not meet any of the criteria in 10 CFR 54.4(a). The ~~main~~ main transformer is in the same category, and system EG is not relied upon for any in-scope electrical back feed in response to an SBO event. The system is therefore not in the scope of license renewal consideration.

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

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

Based on this response, the staff concludes that this "Bus" commodity group was screened out from the scope of license renewal pursuant to 10 CFR 54.4(a) as part of applicant's electrical systems scoping review. Based on its review, the staff concludes that there is no omission of electrical bus at VCSNS. The screening of this "Bus" commodity group from the scope of license renewal pursuant to 10 CFR 54.4(a) is considered acceptable.

Uninsulated ground conductors — Section 2.5 of the LRA indicates that the passive electrical component commodity group of uninsulated ground conductors was found not to meet any of the scoping criteria of 10 CFR 54.4(a). Consequently, this commodity group was considered outside the scope of license renewal. After a series of RAIs and responses thereto, the staff found that uninsulated ground conductors are part of the VCSNS CLB. In a letter dated September 2, 2003, the applicant clarified that the uninsulated ground conductors within the EC system are considered part of the CLB for VCSNS.

However, the staff's conclusion on this matter, based on the plant's conformance with single failure criteria, is that no credible uninsulated ground conductor failure mode or mechanism would prevent satisfactory accomplishment of any of the safety-related functions identified in 10 CFR 54.4(a)(1)(i),(ii), or (iii). Although the unavailability or failure of the uninsulated ground conductor may increase the damage/impact to one train if a single failure occurs, uninsulated ground conductors do not meet the non-safety-related scoping criterion of 10 CFR 54.4(a)(2). Therefore, the passive electrical commodity of uninsulated ground conductor is not within the scope of license renewal.

### 2.5.3 Conclusions

Based on its review, the staff did not find any omissions and, therefore, concludes that the applicant has identified component commodity groups of the electrical and I&C systems that are within the scope of license renewal pursuant to 10 CFR 54.21(a), and subject to an AMR pursuant to passive screening criterion 10 CFR 54.21(a)(1)(i) and the long-lived screening criterion 10 CFR 54.21(a)(1)(ii).

The staff reviewed the information in Section B.1.2 of Appendix B to the LRA, the summary description of the program in the FSAR supplement (Section 18.2.7 of Appendix A to the LRA), and the applicant's responses to the staff's request for additional information (RAIs). The 10 program elements in GALL AMP XI.M10, "Boric Acid Corrosion," provide detailed programmatic characteristics and criteria that the staff considers to be necessary to manage the aging effects in components. In LRA Section B.1.2, the applicant stated that the program elements for the BACS program are consistent with those specified in AMP XI.M10 of the GALL report except for enhancements related to dissimilar metal weld inspections.

[Operating Experience] In LRA Section B.1.2, the applicant stated that the BACS Program was enhanced following the incident of a weld cracking between the hot leg and RPV nozzle at VCSNS on October 7, 2000. The enhancements included provisions to ensure that all dissimilar metal welds were included in the population of components that are visually inspected at refueling outages or when appropriate plant conditions permit access. By letter dated March 28, 2003, the staff requested, in RAI B.1.2-1, the applicant to clarify the post-GALL VCSNS operating history and to discuss how the systems outside of containment will be inspected under the enhanced BACS Program.

In its response dated June 12, 2003, the applicant stated that the current BACS Program focuses on GL 88-05 requirements. The applicant also noted that GALL is driving the industry to make enhancements to the surveillances (i.e., to inspect systems outside of containment that contain boric acid solutions). In addition, recent industry events are also driving the industry to perform additional inspections. These events are described in NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," and Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs." The applicant stated that it intends to enhance the surveillance test procedures, required by technical specifications, for leakage of primary coolant sources outside containment (i.e., boron recycle, liquid waste, nuclear sampling, chemical and volume control, residual heat removal, and RB spray systems). In addition, the applicant stated that it also intends to enhance the leak tests performed for the SI accumulators and the spent fuel pool cooling system. These enhanced leak tests would specify inspections for boric acid crystallization on the system being tested and, in the cases when boric acid is found, also on the surrounding systems. These enhancements will be noted on the procedures and maintained as license renewal commitments. The applicant finally stated that the development of an overall Boric Acid Corrosion Program will incorporate GL 88-05 requirements, license renewal commitments, and the additional inspections that result from the NRC Bulletins. As documented in a telecommunications discussion on July 9, 2003, these enhancements are considered commitments. Applicant has agreed that this is a license renewal commitment and this commitment is documented in Appendix A of this SER.

NEED TO REPLACE WITH ATTACHED REVISED PROGRAM SUBSEQUENT TO RAI

Based on the applicant's responses to NRC Bulletins 2002-01 and 2002-02, its response to the RAI, and the discussion of enhancements to this program, the staff finds the applicant response adequate in addressing the concerns related to the detection of cracking in dissimilar metal welds. Therefore, RAI B.1.2-1 is considered closed.

By letter dated March 28, 2003, the staff requested, in RAI B.1.2-2, the applicant to list the location of the other dissimilar metal welds exposed to borated coolant to be included within the scope of the BACS Program in light of recent events. In its response dated June 12, 2003, the applicant listed the welds provided in Attachment IX to the letter from Stephen A. Byrne to the

NRC Document Control Desk, dated January 24, 2003, entitled, "Response for Additional Information Regarding 60 Day Response to NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

The following dissimilar welds are included within the scope of the BACS Program:

- "A" hot leg weld to reactor vessel nozzle
- "A" cold leg weld to reactor vessel nozzle
- "B" hot leg weld to reactor vessel nozzle
- "B" cold leg weld to reactor vessel nozzle
- "C" hot Leg weld to reactor vessel nozzle
- "C" cold leg weld to reactor vessel nozzle
- Pressurizer surge line weld to pressurizer nozzle
- Pressurizer nozzle weld to "A" pressurizer safety valve
- Pressurizer nozzle weld to "B" pressurizer safety valve
- Pressurizer nozzle weld to "C" pressurizer safety valve
- Pressurizer nozzle weld to PORVs
- Pressurizer nozzle weld to spray piping
- "A" hot leg weld to steam generator nozzle
- "A" crossover weld to steam generator nozzle
- "B" hot leg weld to steam generator nozzle
- "B" crossover weld to steam generator nozzle
- "C" hot leg weld to steam generator nozzle
- "C" crossover weld to steam generator nozzle

Based on this response, the scope of this surveillance program includes the dissimilar welds that may be susceptible to cracking as discussed in the recent NRC Bulletins. Therefore, the staff finds the response satisfactory and considers RAI B.1.2-2 closed.

The LRA credits the BACS Program for managing loss of material due to boric acid corrosion of the pressurizer, CS and LAS components (e.g., shell, upper and lower heads, nozzles, integral support, and manway cover and bolts), the external surfaces of CS components in the RCS pressure boundary (LRA Table 3.1-1, AMR Item 26), and the steam generator (SG) elliptical head and channel head (LRA Table 2.3-7). By letter dated March 28, 2003, the staff requested, in RAIs B.1.2-3 and B.1.2-4, the applicant to discuss how the BACS Program sufficiently manages the corrosive effects of boric acid leakage on the base metal of insulated components during the extended period of operation (e.g., leakage from the pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials). In addition, the staff requested the applicant to discuss how the BACS Program would manage VCSNS steam generator external surfaces in light of Bulletin 2002-01, and GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants."

In its response dated June 12, 2003, the applicant stated that BACS Program will evaluate all boric acid leaks, continue to remove insulation and inspect all joints for leakage during each refueling outage, and apply corrective actions for boric acid leaks, as required, for the source and the adjacent components, supports, or structures. The applicant also referenced the response to Bulletin 2002-01 from Stephen A. Byrne of VCSNS to the NRC Document Control Desk dated January 24, 2003, as a source of further information. The staff reviewed this document and finds the detailed information provided on the inspection techniques, scope,

**Change to the Boric Acid Corrosion Surveillances Program:** It was stated in the response to RAI B.1.2-1 that VCSNS intends to enhance the leakage assessment tests for the following systems: boron recycle, liquid waste, nuclear sampling, chemical and volume control, residual heat removal, and RB spray. Subsequent to the RAI response, it was discovered that the leakage assessments for the chemical and volume control, residual heat removal, and RB spray systems were limited to only portions of the systems. It was decided that the Boric Acid Corrosion Surveillances program should credit the leak tests for these systems instead of the leakage assessments. This should be reflected in the SER by changing the text on page 3-7.

## INSERT - SER PAGE 3-7

"The applicant stated that it intends to enhance three of the surveillance test procedures required by Technical Specifications for leakage of primary coolant sources outside containment (i.e., boron recycle, liquid waste, and nuclear sampling systems). In addition, the applicant stated that they intend to enhance the surveillance test procedures that are used to perform leak tests for safety injection / chemical and volume control, SI accumulators, residual heat removal, reactor building spray, and spent fuel pool cooling systems."

extent of coverage, frequency of inspections, personnel qualifications, and degree of insulation removal is adequate in addressing the staff concerns. Therefore, RAIs B.1.2-3 and B.1.2-4 are considered closed because the January 24, 2003 document describes how the BACS Program would manage the corrosive effects of boric acid leakage on the base metal of insulated components and steam generator external surfaces.

The staff reviewed the criteria 2 supplemental information in Section B.1.2, "Boric Acid Surveillances," in which the applicant credited this AMP for managing components located in the Auxiliary, Intermediate, and Fuel Handling buildings. These components are constructed of carbon steel, low-alloy steel, and other susceptible materials to loss of material due to boric acid corrosion. The applicant concluded that revisions or clarifications to the previous evaluation of this program is not needed to ensure management of these components.

The staff concurs with the applicant's conclusion because the materials of construction for these components is similar to components within the scope of this AMP. The staff notes that the scope of this AMP has been increased to include these components and finds that this AMP is adequate in managing these components for loss of material due to boric acid corrosion.

Section 18.2.7 of Appendix A to the LRA contains the applicant's FSAR supplement for the Boric Acid Corrosion Surveillances (BACS) Program. The staff reviewed this section and finds the program description consistent with the material contained in Section B.1.2 of Appendix B to the LRA, except for the reference to GL 88-05 and the enhancements to the BACS Program discussed in Section 3.0.3.1.2 of this SER. By letter dated September 2, 2003, the applicant revised the FSAR supplement to include reliance of this program on the implementation of GL 88-05, as well as subsequent NRC bulletins and guidance, to monitor the reactor coolant pressure boundary for borated water leakage. In addition, the program also includes monitoring of borated water leakage in all systems containing borated water. Based on this revision, the staff finds that the FSAR supplement provides an adequate summary of the program activities are required by 10 CFR 54.21(d).

3.0.3.1.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

*3.0.3.2 Chemistry Program*

The Chemistry Program is described in Section B.1.4 of Appendix B in the LRA. The LRA credits the Chemistry Program with managing loss of material, cracking, and fouling of components exposed to borated water, closed cooling water, treated water, or fuel oil environments for the Virgil C. Summer Nuclear Station (VCSNS). The staff reviewed the LRA to determine whether the applicant has demonstrated that the Chemistry Program will

adequately manage the applicable aging effects for the components that credit this program throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.0.3.2.1 Summary of Technical Information in the Application

The applicant's Chemistry Program is discussed in LRA Section B.1.4, "Chemistry Program." The applicant stated that the program is consistent with GALL AMP XI.M2, "Water Chemistry," and the chemistry-related portions of XI.M30, "Fuel Oil Chemistry," with the following clarifications concerning the detection of aging effects. The applicant indicated that the Chemistry Program is a mitigation program; therefore, no aging effects are detected as part of this program. In addition, plant operating experience confirms the effectiveness of the program for managing aging during the period of extended operation. The applicant stated that based on this experience, VCSNS does not commit to performing one-time inspections to verify the effectiveness of the Chemistry Program as recommended by the GALL AMP XI.M2.

In LRA Section B.1.4 and FSAR Supplement 18.2.10, the applicant stated that aging effects will be managed by the Chemistry Program such that the components subject to aging management review (AMR) will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operations. The applicant stated that the Chemistry Program is an ongoing program that incorporates the best practices of industry organizations, vendors, utilities, and water treatment experts. This aging management program (AMP) 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 applicant noted that the Chemistry Program is based on EPRI guidelines for primary and secondary water chemistry. Analyzing and trending the water chemistry specifications has been in effect since the initial implementation at VCSNS and is considered acceptable based on industry operating experience. 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.

II/I

By letter dated September 12, 3002, SCE&G supplemented the license renewal application for VCSNS. The letter provided the results of the additional reviews based on the NRC staff positions on scoping of seismic 11/1 piping sustems in letters dated December 3, 2003, and March 15, 2002. As a result, VCSNS added several additional SSC's into the scope of several aging management programs including Boric Acid Surveillances program, chemestry program and Flow-accelerated Corrosion Monitoring program. The staff evaluation is provided below.

By letter dated September 12, 2002, SCE&G supplemented the license renewal application for VCSNS. The letter provided the results of the additional reviews based on the NRC staff positions on scoping of seismic II/I piping systems in letters dated December 3, 2001, and March 15, 2002. As a result, VCSNS added several additional SSC's into the scope of license renewal and expanded the program description of several aging management programs including chemistry program. The staff evaluation is provided below.

3.0.3.2.2 Staff Evaluation

In LRA Section B.1.4, "Chemistry Program," the applicant described its program to manage the aging effects of components exposed to borated water, closed cooling water, or treated water.

The LRA states that this program is consistent with GALL AMPs XI.M2, "Water Chemistry," and the chemistry related portions of XI.M30, "Fuel Oil Chemistry." The staff confirmed the applicant's claim of consistency during the AMR Audit on July 16 - 17, 2003. The staff verified that the Chemistry Program, as described, is consistent with GALL AMPs XI.M2 and XI.M30. Based on the consistency of this program with the GALL Report, the staff focused its review on the operating history program element supporting the effectiveness of this program.

[Operating Experience] The applicant stated that a review of the operating experience did not reveal a loss of intended function for components that are exposed to borated water, closed cooling water, or treated water that could be attributed to an inadequacy of the Chemistry Program. Therefore, the applicant stated that no special one-time inspection will be performed for the purpose of verifying the effectiveness of the Chemistry Program. This position deviates from the recommendation in the GALL report for a one-time inspection in low-flow and/or stagnant areas.

By letter dated March 28, 2003, the staff requested, in RAI B.1.4-1, the applicant to clarify from operating history, recent surveillances, and inspections that cracking and crevice, general, pitting, and galvanic corrosion are adequately managed for carbon steel (CS) and stainless steel (SS) components, and cited examples from the AMR Tables. In addition, the applicant was asked to clarify if there is any inspection of the most susceptible locations (e.g., low-flow or stagnant areas) for the aging effects of loss of material, cracking, and fouling. In its response dated June 12, 2003, the applicant stated that the LRA lists the component-aging effect combination where the Chemistry Program alone is credited for aging management and presented evidence that such inspections are not required because a review of VCSNS operating experience did not reveal a loss of intended function of components that are exposed to borated water. In addition, the effects of pitting and crevice corrosion on SS components are not significant in chemically treated borated water. The staff determined that the applicant had not satisfactorily justified the effectiveness of the Chemistry Program in lieu of the one-time inspection for loss of material for CS components and requested the applicant to further discuss why the one-time inspection for low flow or stagnant locations is not needed. With respect to SS non-Class 1 RCS components, the staff notes that these components are internally exposed to chemically treated borated water and are subject to crack initiation and growth due to stress corrosion cracking (SCC). Thus, the staff found that the applicant had not adequately justified the management of cracking of non-Class 1 SS components and requested the applicant to further discuss the aging management of these components.

In subsequent correspondence dated September 2, 2003, the applicant stated that one-time inspections will be performed in low flow areas of the different chemistry regimes prior to the period of extended operation. The various chemistry regimes to be verified are found in the feedwater (FW) system, the condensate (CO) system, the emergency feedwater (EF) system, the component cooling (CC) system, the chilled water (VU) system, the local ventilation (VL) system, the air handling (AH) system, and the diesel generator services (DG) system. The FW, CO, and EF systems share one chemistry regime. The VU, VL, AH, and DG systems share another chemistry regime. Therefore, the applicant concluded that an inspection of one system per chemistry regime should be representative of the other systems. The applicant further stated that any abnormalities resulting from the visual inspection of the low flow areas will be dispositioned through engineering evaluation and addressed in site's Corrective Actions Program. If further inspections are needed, quality control inspectors will perform volumetric inspections at representative sites for the chemistry regime of the VU, VL, AH, and DG



The CC system alone has chromates.

systems. With respect to SCC of non-Class 1 SS piping, the applicant stated that Table 3.1-2, AMR Item 6, lists the aging management of both SS Class 1 and non-Class 1 components susceptible to SCC. In addition, Table 3.1-1, AMR Item 6, lists the aging management of SS Class 1 piping with the Small Bore Class 1 Piping Inspection. This inspection activity will be representative of the conditions for SS piping and components (Class 1 and non-Class 1) in borated water service.

Based on the discussion above, the staff finds the applicant's commitment to complete a one-time inspection of low flow areas of the different chemistry regimes satisfactory because it provides a method of verifying the program's effectiveness as recommended in the GALL report. With respect to the aging management of SS non-Class 1 components, the staff reviewed Table 3.1-1, AMR Item 24, which manages the aging effects of management of Class 1 SS components through the Chemistry Program and the In-Service Inspection (ISI) Plan. This AMR Item bounds the management of large bore non-Class 1 SS components. In addition, the management of non-Class 1 SS small bore piping is bounded by the Small Bore Class 1 Piping Inspection discussed in Table 3.1-1, AMR Item 6. Thus, the staff finds that the applicant will adequately manage the aging effects of SS non-Class 1 components through a combination of chemistry control and inspection. Therefore, RAI B.1.4-1 is considered closed.

The staff notes that the applicant appears to have combined the aspects of several GALL programs into its Chemistry Program. By letter dated March 28, 2003, the staff requested, in RAI B.1.4-2, the applicant to clarify to what extent the Chemistry Program relies on the GALL AMPs XI.M20, "Open-Cycle Cooling Water System," and XI.M21, "Closed-Cycle Cooling Water System." In addition, the staff requested a discussion on how the features of these GALL programs are incorporated into the VCSNS Chemistry Program.

In its response dated June 12, 2003, the applicant stated that the Service Water System Reliability and In Service Testing Program, not the Chemistry Program, is credited for meeting the requirements of GALL AMP XI.M20, "Open-Cycle Cooling Water System." The applicant stated that this program meets the intent of GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment." As a response to Recommended Action #2 of GL 89-13, VCSNS evaluated its component cooling water, chemical volume and control, residual heat removal, spent fuel cooling and chilled water systems. The results of the evaluation indicated that the corrosion protection of these systems had not been compromised. This conclusion was based on a review of the historical maintenance work requests from the time of adopting the CHAMPS computer software (to track condition reports and work orders). VCSNS maintains the chemical concentrations of its closed cycle cooling systems within the guidelines of EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines." The applicant stated that, prior to the period of extended operation, one-time inspections will be conducted in low flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

The applicant's response indicates that the requirements for GALL AMP XI.M20 are credited in the Service Water System Reliability and In-Service Testing Program. Therefore, the components managed by the open-cycle cooling water system, as defined in the GALL report, are discussed and evaluated in Section 3.3.2.3.1 of this SER. With respect to the GALL program requirements for the closed-cycle cooling water system, the staff finds that the applicant appropriately applied the scoping requirements in the GALL report by treating the aforementioned systems as open-cycle cooing water systems. This action is required in

carbon (TOC). The staff finds the increased activities to monitor and trend the constituents of this system adequate and appropriate for mitigating the aging effects through maintenance of water quality. The staff's evaluation of the AMP is found in Section 3.0.3.2 of this SER.

Section 18.2.10 of Appendix A to the LRA contains the applicant's FSAR supplement for the Chemistry Program at VCSNS. The staff reviewed this section and finds that the information provided in the FSAR supplement for the aging management of systems and components discussed above is equivalent to the information in the GALL report, and therefore, provides an adequate summary of the program activities as required by 10 CFR 54.21. Although the applicant noted that the Chemistry Program is based on EPRI guidelines for primary and secondary water chemistry, the staff requested in RAI B.1.4-2 that the FSAR supplement reference the specific EPRI documents that are consistent with the SRP-LR. By letter dated September 2, 2003, the applicant revised the FSAR supplement to include the primary and secondary water chemistry guidelines (i.e., EPRI TR-105714 and EPRI TR-102134). Based on this revision, the staff finds that the FSAR supplement provides an adequate summary of the program activities are required by 10 CFR 54.21(d).

3.0.3.2.3 Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the exceptions to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

*3.0.3.3 Fire Protection Program*

The applicant described its Fire Protection Program (FPP) in Section B.1.5 of Appendix B to the LRA, "Fire Protection Program." The applicant credits this program with managing the aging of FP system components that are within the scope of license renewal and subject to an AMR. The staff reviewed LRA Section 3 and Section B.1.5 to determine whether the applicant has demonstrated that the program will adequately manage the applicable effects of aging during the period of extended operation, as required by 10 CFR 54.21(a)(3).

The applicant's AMR identifies one or more AMPs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801, and were evaluated for consistency with NUREG-1801 programs that are relied on for nuclear power plant license renewal. The results are documented and discussed in LRA Section 3, Tables 3.3-1 and 3.3-2, using the format suggested by NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants".

3.0.3.3.1 Summary of Technical Information in the Application

Section B.1.5 of Appendix B to the LRA states that the FPP is consistent with XI.M26, "Fire Protection," and XI.M27, "Fire Water System," as well as XI.M23, "Selective Leaching of Materials," as identified in NUREG-1801 with the following enhancements that will be made to the current plant program. The applicant's fire door inspections monitor holes or breaks in the door surface at a frequency of every 6 months rather than the recommended bimonthly frequency. Aging management of the fuel supply line for the diesel-driven fire pump at the plant is credited to the chemistry program and is not managed by the FPP. The applicant maintains proper clearances (gap) between doors, frame, and threshold in accordance with station procedures. However, the applicant does not consider maintaining the clearances to be an aging effect for license renewal. The applicant intends to perform ultrasonic testing of selected FP piping to detect aging effects in lieu of disassembly of FP piping for inspection or full-flow testing of stagnant portions of FP piping.

For operating experience, LRA Section B.1.5 states that the fire barrier and fire barrier penetration seal inspection 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. Non-conforming conditions that were aging related cracks and separations were noted during surveillance of fire barrier penetration seals. Conditions were repaired in accordance with station procedures. No condition evaluations reports (CERs) were initiated for fire barriers of fire barriers penetrations seals relevant to aging. Furthermore, LRA Table 3.3-1, Item 19, for the commodity groups of doors and barrier penetration seals and concrete structures in fire protection, provides the following discussions for the AMP:

> The plant's aging management programs for this group are generally consistent with those reviewed and approved in NUREG-1801. The plant's fire protection program (Appendix B.1.5) contains many activities to achieve defense-in-depth and minimize the impact of a potential fire.
>
> The fire barrier and fire barrier seal inspections detect structural damage or degradation of fire barriers and fire barrier penetration sealing devices. Fire barriers include walls, ceilings and floors. The corresponding aging effects are cracking, separation from walls or components, separation of material layers, rupture or puncture of seals, shrinkage and voids.
>
> The fire door inspections detect structural damage or degradation of fire rated doors. Inspections are credited with managing loss of material of doors and door hardware for the period of extended operation. Excessive wear for door appurtenances such as latches, strike plates, hinges, sills and closing devices, and maintaining proper clearances (gaps) between the door, frame and threshold are also inspected, but these attributes are not credited for license renewal. Loss of material due to wear of the door hardware and hinges is not considered an aging effect but rather a consequence of frequent or rough usage.

According to LRA Section B.1.5, the plant 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. No non-conformance notices (NCNs) or CERs were initiated for fire doors relevant to aging.

The LRA states that monthly surveillance are conducted on the FP system consisting of ~~flow tests and~~ pump start tests. Flow tests and flushes of the main distribution loops have been conducted to ensure their functionality and have all met acceptance criteria. 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

* Flow tests are done every 36 months.

intended functions will be maintained consistent with the CLB for the period of extended operation as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

*3.0.3.4 Maintenance Rule Structures Program*

The applicant described its Maintenance Rule Structures Program in Section B.1.18 of Appendix B to the LRA. The applicant credits this program with the capability of detecting and managing the effects of aging for structures and structural components at VCSNS. The staff reviewed the LRA to determine whether the applicant has demonstrated that the Maintenance Rule Structures Program will adequately manage the applicable aging effects for the components that credit this program throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.0.3.4.1 Summary of Technical Information in the Application

In LRA Section B.1.18, the applicant states that the Maintenance Rule Structures Program is consistent with XI.S6, Structures Monitoring Program, as identified in NUREG-1801. The applicant further states that the following enhancements will be incorporated into the Maintenance Rule Structures Program prior to the period of extended operation:

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
- groundwater chemical analyses

Groundwater chemical analyses will include:

- ph
- Sulfates
- Chlorides

Groundwater chemical analyses will be used to monitor changes in aggressiveness of the below grade environment.

The Maintenance Rule Structures Program is included in the discussion column of LRA Table 3.5-1. The structures and structural components that credit this program for license renewal are identified in Report TR00170-003, Rev 0, Attachment II.

In 1996, a baseline assessment concluded that the maintenance rule structures and structural components were acceptable and were free of deficiencies or degradation that could lead to

possible failure. Therefore, these structures were determined to be capable of performing their structural functions, including the protection and support of systems and components.

The maintenance rule inspection report completed in 2000 noted that 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. The applicant determined that 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.

The applicant states that the Maintenance Rule Structures Program provides reasonable assurance that the aging effects for structures and structural components 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.

3.0.3.4.2 Staff Evaluation

In LRA Section B.1.18, "Maintenance Rule Structures Program," the applicant described its AMP to manage aging in structures and structural components. The LRA stated that this AMP is consistent with GALL AMP XI.S6, "Structures Monitoring Program," with several enhancements described in SER Section 3.0.3.4.1. The staff reviewed the enhancements to determine whether the AMP, with the enhancements, remains adequate to manage the aging effects for which it is credited, and reviewed the FSAR supplement to determine whether it provides an adequate description of the revised program. The staff audit on July, 16-17, 2003 confirmed the applicant's claim of consistency.

The staff noted several inconsistencies between the FSAR Supplement summary descriptions of the aging management programs in LRA Appendix A and the scope of the aging management programs identified in LRA Appendix B as "consistent with GALL." In RAI 3.5-19, the staff requested the applicant to verify that the complete scope of the aging management program, as described in NUREG-1801, GALL Volume 2, is being credited for license renewal aging management. If this is not the case, the applicant was requested to identify and document the justification for each exception. In response to RAI 3.5-19, the applicant stated the following:

> As stated in the LRA, VCSNS maintains a Maintenance Rule Structures Program (B.1.18), which is consistent with GALL XI.S6 and 10 CFR 50.65. Several enhancements to this program have been identified during the license renewal evaluation process and are listed in the Application (B.1.18).
>
> VCSNS does not believe that there are any further changes required for the Application Appendix A, since only summary statements are recommended by NEI 95-10. Commitment to all Regulations and Regulatory Guides are implicit in the development of each of these programs as described in Section 7 of TR00170-003.

LRA Section B.1.18 states that the Maintenance Rule Structures Program is consistent with GALL XI.S6 with several listed enhancements that will be incorporated into the program prior to

potential surface corrosion of the external piping or tank surfaces and will require further evaluation as discussed in the FSAR supplement (Section 18.2.9 of Appendix A to the LRA).

Within the auxiliary system, the following major components and systems will be monitored by this aging management program: carbon steel (CS) pipes and couplings in the service water system; ductile iron pipe and cast iron hydrants and valve bodies the in fire service system; CS pipes in the emergency feedwater system; and CS fuel oil pipes, fittings, and tanks in the diesel generator service systems. ~~Within the steam and power conversion systems, this inspection program will also monitor orifices in the emergency feedwater system.~~ *

3.0.3.6.2 Staff Evaluation

* THERE ARE NO BURIED ORIFICES IN THE EFW.

In LRA Section B.2.10, "Buried Piping and Tanks Inspection Program," the applicant described its AMP to manage the loss of material of buried components. The staff's evaluation of the Buried Piping and Tanks Inspection focuses on how the program detects and characterizes aging effects through the effective incorporation of the ten elements described in Branch Technical Position RLSB-1 in Appendix A-1 of the SRP-LR.

Since the applicant claimed consistency with GALL AMP XI.M34, this AMP was cross-referenced in the staff's review. The 10 program elements in this GALL AMP define programmatic characteristics and criteria to manage buried components except for the program elements/attributes of detection of aging effects (regarding inspection frequency) and operating experience. Thus, the staff further evaluates an applicant's inspection frequency and operating experience with buried components. The LRA indicates that the corrective actions and confirmation process are implemented through the site corrective actions process, while the administrative controls are implemented through the site procedures. The staff's evaluation of the corrective actions, confirmation process, and administrative controls is contained in Section 3.0.4, "Quality Assurance Program," of this SER. The remaining elements are evaluated below.

[Program Scope] The staff finds that the systems and components that will be monitored by this program, as listed in the LRA, are within the scope of license renewal and identified in Section 2.3 of the LRA. The staff finds that the scope of the program is acceptable since it includes the buried components within the scope of license renewal exposed to an underground environment.

[Preventive Actions] The applicant stated that underground components are coated and wrapped during installation to prevent direct contact with the soil environment. Otherwise, no actions will be taken as part of the buried piping and tanks inspection to prevent aging effects or mitigate age-related degradation. By letter dated March 28, 2003, the staff requested, in RAI B.2.10-1, the applicant to discuss the adequacy of coating techniques. In its response dated June 12, 2003, the applicant stated that VCSNS coats and wraps underground components in accordance with site procedures, available onsite for inspection. These procedures are based on accepted industry standard American Water Works Association (AWWA) C-203, 1973. In addition, operating experience for the diesel generator fuel oil storage tanks revealed negligible wall thinning thereby verifying that the coating and wrapping techniques implemented are effective. The staff subsequently requested the applicant to supply a copy of industry standard AWWA C-203 or its equivalent for review and comparison with the industry standards referenced in the GALL report. During the AMR audit conducted on July 16 - 17, 2003, the staff

received the mechanical maintenance procedure for applying coating on embedded piping. Based on a review of this document, the staff finds this procedure meets the intent of recommended practices of referenced in GALL AMP XI.M34 for surface preparation, application, and inspection of coatings on embedded piping. Therefore, RAI B.2.10-1 is considered closed.

[Parameters Monitored or Inspected] The applicant stated that the condition of coatings and wrappings will be determined by visual inspection whenever buried components are excavated for maintenance or for other reasons. The applicant later cited operating experience with buried piping and tanks, which used the ultrasonic inspection technique (UT). By letter dated March 28, 2003, the staff requested, in RAI B.2.10-2, the applicant to discuss if UT will supplement or replace visual inspection, and the criteria used to determine the applicability of the technique used. In its response dated June 12, 2003, the applicant stated that a visual inspection of the wrapping and coating will be performed and evaluated upon initial excavation of the component. If the wrapping or coating is damaged or removed as part of the maintenance activity, then the underlying metal will be visually inspected for degradation. Depending on the condition of the underlying metal, subsequent inspections and the types of inspections will be determined through the VCSNS Corrective Action Program. Based on the applicant's response, the staff finds that this program will appropriately monitor the parameters directly related to the integrity of the external surface of buried carbon steel piping and tanks. Thus, RAI B.2.10-2 is considered closed.

[Detection of Aging Effects] The applicant claimed that the rate of wall thinning for components within this program is very slow (or negligible). In addition, since the process of excavation itself can damage protective coatings and wrappings, a specific inspection frequency for buried components is not warranted. Instead, if buried components are excavated for maintenance or for other reasons, the integrity of the coatings and wrappings will be evaluated. If the coatings or wrappings are damaged or removed as part of the maintenance activity, the underlying metal will be visually inspected for degradation. By letter dated March 28, 2003, the staff requested, in RAI B.2.10-3, the applicant to discuss why periodic inspection of the most susceptible locations is not needed especially in areas with the highest likelihood of corrosion and/or a history of corrosion problems. In its responses dated June 12, 2003, the applicant stated that GALL AMP XI.M34 allows the inspection frequency to be whenever underground piping is excavated for maintenance depending on operating experience. In addition, VCSNS operating experience has shown no history of corrosion problems for buried piping and tanks, as evidenced by the negligible wall thinning of the diesel fuel oil storage tanks. Therefore, based on this operating experience, the applicant concluded that an inspection frequency based upon scheduled maintenance is justified. The staff finds that the applicant has not adequately demonstrated that periodic inspection, at the most susceptible locations, is unnecessary. In addition, the staff notes that the GALL Report states that the inspection frequency is plant specific and depends on the plant operating experience. Therefore, the staff requested a summary of the most recent excavations, including information about any age-related degradation of systems and components within the scope of this program. In subsequent correspondence dated September 2, 2003, the applicant stated that modification on the Fire Service System piping in 1997 and 1998 required excavation and revealed no external degradation. Based on this most recent operating history and the negligible wall thinning of the diesel fuel oil storage tanks, the staff finds the inspection of buried components during maintenance activities is acceptable. Therefore RAI B.2.10-3 is considered closed.

examinations are conducted on an opportunistic basis with external surfaces already exposed and accessible to visual examination during normal operation, or if the examinations include external surfaces at susceptible locations that are exposed to visual examination due to targeted planned actions that may or may not involve suspension of normal operation. The staff requested that the applicant provide the technical basis for determining which additional component external surfaces are to be inspected if unacceptable degradation is observed.

In its response dated June 12, 2003, the applicant stated that the Inspections for Mechanical Components program will generally examine external surfaces already exposed and accessible to visual examination during normal operation.

The applicant also stated that operating experience revealed an instance of external pitting below the insulation on chilled water (VU) system piping. Consequently, loose insulation removal is necessary to permit visual inspection of systems for which the internal fluid temperature is less than the external ambient temperature. The applicant stated that any unacceptable degradation, whether found by these inspections or by planned maintenance activities, would be determined by engineering evaluation and dispositioned in the Corrective Action Program. The applicant concluded that, although the initial frequency for the inspections is 5 years, the Corrective Action Program could increase not only the frequency, but also the scope of the inspections.

The staff required a clarification as to the extent of component surfaces inspected. During a telecommunication on July 14, 2003, the applicant identified that a walkdown is made of all accessible components and any degradation is thoroughly addressed by the Corrective Action Program. By letter dated September 2, 2003, the applicant clarified that the Inspections for Mechanical Components program will inspect external surfaces exposed and accessible to visual inspection during normal operation in addition to removal of insulation to permit visual examinations for systems where the internal fluid temperature is less than the ambient temperature and the insulation is not tightly adhered to the components. The staff finds that the applicant's response satisfactorily addresses the staff's concerns and RAI B.2.11-2 is considered closed.

[Detection of Aging Effects] The applicant stated that, in accordance with guidance in Element 5, "Detection of Aging Effects" for AMPs, the AMP will detect loss of material and cracking prior to loss of component intended function. The applicant further stated that pitting is a concern in locations where components are insulated and internal system fluid temperatures are below the ambient temperature conditions. The staff finds that these inspection techniques are sufficient to provide reasonable assurance that the aging effects for the components managed by the Inspections for Mechanical Components program will be detected and evaluated before a component has lost its intended function.

[Monitoring and Trending] The applicant stated that the inspections will be performed and documented in accordance with station procedures and, following baseline inspection, the frequency of inspections will be determined based on inspection results and industry experience. By letter dated March 28, 2003, the staff requested, in RAI B.2.11-4, that the applicant provide the schedule for the baseline inspection. In its response dated June 12, 2003, the applicant stated that inspections follow the same frequency as maintenance rule structures inspections (5 years) and the baseline inspection would occur within 5 years of obtaining the new license. Based upon the results of these inspections, or any new industry

experience, the frequency may increase. The applicant also confirmed that "effective components," as written in Element 7, "Monitoring and Trending" for AMPs, should be corrected to "affected components." The staff finds that the applicant described and justified the inspection frequency. Thus, the staff finds that the applicant's response satisfactorily addresses its concerns and RAI B.2.11-4 is considered closed. The staff finds that the overall monitoring and trending proposed by the applicant is acceptable because periodic inspections performed in accordance with station procedures will effect timely corrective actions.

[Acceptance Criteria] The applicant stated that the acceptance criterion is that no unacceptable visible indications of loss of material or cracking exist. The applicant further stated that 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. The staff considers the acceptance criteria to be adequate to assure that the intended functions for components in the Inspections for Mechanical Components program will be maintained under all CLB design conditions during the period of extended operation.

By letter dated March 28, 2003, the staff stated, in RAI B.2.11-3, that the SRP-LR Section A.1.2.3.6 indicates that qualitative inspections should be performed to some predetermined criteria as quantitative inspections by personnel in accordance with American Society of Mechanical Engineers (ASME) Code and through site-specific programs. The staff therefore requested the applicant to stipulate the qualifications of inspection personnel conducting the "visual examination of the exposed external surfaces of mechanical components for loss of material or cracking." In its response dated June 12, 2003, the applicant stated that site engineering personnel will perform the visual inspections and that any degradation found during the visual inspections would be dispositioned through the VCSNS Corrective Action Program. The applicant stated that further inspections and qualifications required for these inspections would be determined through the Corrective Action Program, which generally requires inspection by quality control personnel qualified in accordance with ASME Code and 10 CFR Part 50 Appendix B. This response did not identify the qualifications of the personnel performing the initial inspection. During a telecommunication on July 14, 2003, the applicant identified that actual system engineers perform the initial walkdowns who observe and report any degradation or abnormality. By letter dated September 2, 2003, the applicant clarified that site engineering personnel ~~(rather than system engineers)~~ will perform visual inspections to specific developed criteria. The staff finds that the applicant's response satisfactorily addresses the staff's concerns and RAI B.2.11-3 is considered closed.

[Operating Experience] The applicant stated that the Inspections for Mechanical Components program is a new inspection activity. The applicant also described relevant operating experience with the identification of pitting below the insulation in the chilled water system, which were detected and repaired under existing inspection activities, and several instances of leakage in the chilled water system, which were identified by surveillance procedures. By letter dated March 28, 2003, the staff requested, in RAI B.2.11-5, that the applicant discuss any additional operating experience relevant to the systems within scope, or provide confirmation that this is the only system in the scope of this program with observed degraded conditions. In its response dated June 12, 2003, the applicant stated that Inspections for Mechanical Components program were developed because it was determined that the aging effects were possible—not because they were found at VCSNS. The particular industry operating experience concerning the chilled water system was included because it demonstrates the

applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.1.2.3.2 Bottom-Mounted Instrumentation Inspection Program

The applicant described its Bottom-mounted Instrumentation Inspection Program in LRA Appendix B.1.3. This is an existing, plant-specific program. The applicant credits this AMP for managing loss of material due to wear of the thimble tubes. There is no corresponding AMP in GALL, but GALL suggests a program based on recommendations of NRC I&E Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors."

Summary of Technical Information in the Application

The objective of the program is to identify loss of material, (i.e., tube wall thinning) due to fretting wear in the bottom-mounted instrumentation (BMI) thimble tubes, prior to loss of their intended function through leakage and loss of pressure boundary. The applicant stated that the program is a condition monitoring program. The program includes inspection of all VCSNS BMI thimble tubes using eddy current testing (ECT). The ECT data are trended, wear rates are calculated, and inspections are planned prior to the refueling outage at which thimble tube wear is predicted to exceed the acceptance criteria of 75 percent loss of initial wall thickness. The corrective actions include capping, repositioning, or replacing a thimble tube if predicted tube wear exceeds the acceptance criteria.

The applicant summarized its operating experience related to thimble tube wear by briefly describing its response to NRC IEB 88-09. Since issuance of the bulletin, the applicant has performed four inspections (RF-4 ,R-5, R-6, and RF-13) of BMI thimble tubes at VCSNS and repositioned several of them. Based on the calculations performed using the results of these inspections, the applicant determined that the next ECT is not required on the thimble tubes until RF-~~14~~. 17 SEE RAI B.1.3-3

Staff Evaluation

The staff reviewed the applicant's description of the program in LRA Appendix B.1.3 to determine whether the applicant demonstrated that it will adequately manage the applicable aging effects at VCSNS during the period of extended operation, as required by 10 CFR 54.21(a)(3).

[Program Scope] The objective of the subject program is to monitor tube wall degradation, (i.e., loss of wall due to fretting wear) of all thimble tubes installed in the VCSNS reactor pressure vessel. The staff finds that the scope of the subject AMP is adequate because it includes inspection of all thimble tubes that are susceptible to wall thinning due to fretting wear caused by flow-induced vibrations.

[Preventive or Mitigative Actions] The subject program is a condition monitoring program. There are no preventive or mitigative attributes associated with the program, nor did the staff identify a need for such.

[Parameters Monitored/Inspected] The subject program monitors BMI thimble tube wall degradation (loss of material due to fretting wear). The staff finds this acceptable because tube wall degradation directly relates to the thimble tube capacity to perform its intended function of maintaining the integrity of reactor coolant pressure boundary.

[Detection of Aging Effects] The subject program monitors tube wall degradation in 100 percent of the BMI thimble tubes using ECT. The staff issued RAI B.1.3-1, requesting the applicant to submit information about whether the entire length of each thimble tube is inspected, and if not, to present the technical basis for not doing so. In response to RAI B.1.3-1, in a letter dated June 12, 2003, the applicant stated that the full length of each BMI thimble tube is inspected. The applicant also stated that the eddy current inspection performed during RF-4 detected wear occurring at the core plate or fuel assembly bottom nozzle area. The staff finds the response acceptable because these wear locations are consistent with the wear locations reported in the NRC IEB 88-09 and NRC IN 87-44, "Thimble Tube Thinning in Westinghouse Reactors."

The applicant stated that the frequency of ECT examination is based on an analysis of data obtained using the wear rate relationships developed based on Westinghouse research. The staff issued RAI B.1.3-2, requesting the applicant to submit an explanation for the wear rate relationships and describe the Westinghouse research. This RAI was discussed during a June 22, 2003, conference call. As a result of the conference call, the applicant provided the following additional information in response to RAI B.1.3-2.

Research was performed for the WOG and is documented in WCAP-12866. WCAP-12866 includes an evaluation of a large amount of operating experience from multiple plants. Data from multiple thimble tubes at these plants were evaluated for wear. The wear was typically evaluated over one operating cycle, but two, and even three, cycles of wear data were used in the research. Hot cell examination of worn thimbles was performed and its results were compared with eddy current data. The comparison determined that eddy current data conservatively predict the extent of loss of material due to wear. The staff finds the use of ECT data from the Westinghouse research for developing wear rate relationship acceptable because the wear data are obtained from thimble tubes in several plants, they cover one to three operating cycles, and they conservatively predict the extent of loss of material due to wear.

[Monitoring and Trending] The applicant stated that 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. Regarding the predicted wear rate, the IEB 88-09 states that, based on the available data, it is not possible to accurately predict thimble tube wear rates because several plant-specific factors affect the wear rate including the gap distance from the lower core plate to the fuel assembly instrument tube, the amount of clearance between the thimble tube and the guide tube, the axial component of the local fluid velocity, the thickness of the thimble tube, and the moment of inertia of the thimble tube. In describing its operating experience, the applicant stated that, based on the analysis of the wear rate data derived from the eddy current inspections performed at RF-4 and RF-5, the next inspection of the thimble tubes is not required until RF-14. The staff issued RAI B.1.3-3, requesting the applicant to explain and justify the use of this extrapolation of the limited inspection results over nine refueling cycles for scheduling the next inspection of the thimble tubes.

In response to RAI B.1.3-3, in a letter dated June 12, 2003, the applicant submitted the following information. VCSNS now has four sets of data for wear of its thimble tubes. Data have been gathered in RF-4, RF-5, RF-6 and RF-13. The highest recorded measurement in RF-4 and RF-13, respectively, was 38 percent and 57 percent of the initial wall thickness. The applicant used the wear rate relationship developed by Westinghouse to predict the wear damage based on RF-13 measurements. The projections for wear at RF-17 are all below 75 percent of the initial wall thickness and the highest wear predicted for RF-18 is between 75 percent and 80 percent of the initial wall thickness. The acceptance criterion for wear damage is 75 percent loss of initial wall thickness. VCSNS plans to perform the next inspection of thimble tubes in RF-17. The staff finds the VCSNS monitoring and trending activities acceptable because the extrapolation is based on inspection results from four refueling cycles. The relationship developed by Westinghouse conservatively predicts the extent of loss of material due to wear for one to three operating cycles. The staff finds this acceptable because the Westinghouse relationship will be periodically evaluated by the applicant

[Acceptance Criteria] The subject program uses 75 percent loss of initial wall thickness as an acceptance criterion. The staff issued RAI B.1.3-4, requesting the applicant to submit the technical justification for this criterion and explain how the allowances for such items as inspection methodology and wear scar geometry uncertainties, which were identified in IEB 88-09, are included in the criterion. In response to RAI B.1.3-4, in a letter dated June 12, 2003, the applicant stated that the wear relationship developed by Westinghouse makes allowances for the uncertainties. The Westinghouse methodology has an acceptance criterion of 80 percent, whereas VCSNS uses 75 percent for additional conservatism. The staff finds the response acceptable because the acceptance criterion adopted by VCSNS is more conservative than the one recommended by Westinghouse and it allows for the uncertainties as identified by IEB 88-09.

[Operating Experience] Since the issuance of IEB 88-09, the applicant has performed four inspections (RF-4, -5, -6, and -13) of thimble tubes at VCSNS. The applicant reported that several thimble tubes were repositioned in RF-5, but no thimble tubes have been capped or required replacement.

The FSAR Supplement for this program is presented in LRA Appendix A, Section 18.2.8. The staff concludes that the applicant's FSAR Supplement provides an adequate description of the program credited with managing this aging effect, as required by 10 CFR 54.21(d).

Conclusions

On the basis of its review of the applicant's program, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.1.2.3.3 In-Service Inspection Plan (ISI)

The applicant described its In-Service Inspection (ISI) Plan in LRA Appendix B.1.7. The plan is based on the ASME Code Section XI in-service inspection requirements. Throughout the

service life of nuclear power plants, Class 1 components and associated Class 1 supports must meet the requirements set forth in Section XI of the ASME Code and Addenda that are incorporated by reference in 10 CFR 50.55a(b).

Inservice examinations and system pressure tests conducted during successive 120-month inspection intervals, following the initial 120-month ISI interval, must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month inspection interval, subject to the limitations and modifications, such as code editions and addenda, as listed in paragraph 10 CFR 50.55a(b)(2)(i).

The period of extended operation will contain the fifth and sixth ISI interval. The ISI plan for each interval of the renewed license period of extended operation for VCSNS will comply with 10 CFR 50.55a(g)(4)(ii) except that if an examination required by the Code or Addenda is determined to be impractical, then the applicant will submit a relief request to the Commission in accordance with the requirements contained in 10 CFR 50.55a(g)(5)(iii) and (iv), for Commission evaluation, as required by 10 CFR 50.55a(g)(6)(i).

Summary of Technical Information in the Application

The Inservice Inspection Plan is an existing program. The applicant states that the program is consistent with GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD," 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. In addition, VCSNS has adopted the 1995 Edition of ASME Section XI with 1996 Addenda for ultrasonic examination requirements, which includes mandatory Appendices VII and VIII. VCSNS has performed Inservice inspections in accordance with the relevant portions of approved editions of ASME Code Section XI from the beginning of its operation in 1982.

As part of the operating experience, the applicant mentions the primary water SCC of the reactor vessel "A" hot leg nozzle that resulted in leakage, which was discovered in 2000 during RF-12. The applicant states that the 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.

Staff Evaluation

In LRA Appendix B.1.7, "In-Service Inspection (ISI) Plan," the applicant describes its AMP for detecting and managing aging effects of ASME code components in the reactor coolant system. The LRA states that this AMP is consistent with GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD," with no deviation. The staff confirmed the applicant's claim of consistency during the AMR inspection. In addition, for VCSNS, the staff determined whether the applicant properly applied the GALL program to its facility.

The plant operating experience, described in the LRA, has indicated that the VCSNS ISI plan has been effective in detecting and managing aging effects in ASME code components in the reactor coolant system identified in Tables 3.1-1 and 3.1-2 of the LRA for which the ISI plan is identified as an AMP. The staff, therefore, has determined that the applicant's ISI plan will

* SEE COVER LETTER RC-03-0227

The applicant stated that its program is consistent with GALL AMP XI.M31. The recommendations of GALL AMP XI.M31 are similar to those of the December 3, 1999, C. Grimes letter to D. Walters (NEI). The VCSNS Reactor Vessel Surveillance Program consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years.

The applicant indicated that it will remove the two remaining surveillance capsules during RF-14. As a result, no surveillance capsules will be left in the vessel during the extended period of operation. Therefore, the staff identified in RAI B.1.24-1 that the applicant needs to confirm whether the operating restrictions will be established at the end of RF-14 to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and that the exposure conditions of the reactor vessel will be monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license.

In addition, the applicant did not make any commitments for installing an alternative dosimetry for monitoring neutron fluence during the period of extended operation. GALL AMP Chapter XI.M31, "Reactor Vessel Surveillance," recommends the use of alternative dosimetry for applicants without in-vessel capsules. In response to RAI B.1.24-1, the applicant stated that a program will be established at the end of RF-14 to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and that the exposure conditions of the reactor vessel will be monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. The applicant further states that this program may be supplemented or revised by using alternative dosimetry or other effective neutron monitoring techniques during the period of extended operation. The applicant has agreed that this is a licensee commitment and this commitment is documented in Appendix A of this SER. The staff finds these responses acceptable because they follow the recommendations of GALL AMP XI.M31. *

By RAI B.1.24-3, the staff requested that the applicant describe the analysis for demonstrating that the materials in the inlet and outlet nozzles and upper shell course will not become limiting materials during the period of extended operation. In response, the applicant stated that it has performed an analysis for such demonstration. Since no information about the copper and nickel contents for the nozzle forgings was found in the material test reports for the vessel, the applicant used the values of 0.35 percent copper and 1.00 percent nickel, which are recommended in 10 CFR 50.61 when the values are not available. The highest temperature for the unirradiated reference temperature is 0 °F for one of the inlet nozzles. The applicant used this reference temperature in its analysis. For the nozzle, a distance of 8 feet from the core midplane to the edge of the nozzle was used for estimating the fluence value at the nozzle. Using these data, the applicant conservatively projected that the $RT_{PTS}$ for the nozzle material at the 54 EFPY end of life (EOL) value is 145.2 °F. Therefore, the staff agrees with the applicant that the vessel nozzles do not become limiting for a 60-year plant life because the highest projected value for the vessel nozzles is below the limiting beltline plate material of 158.1 °F. A detailed discussion of the pressurized thermal shock (PTS) is provided in Section 4.2.2 of this SER.

The FSAR Supplement for this program is presented in LRA Appendix A, Section 18.2.29. The staff concludes that the applicant's FSAR Supplement provides an adequate description of the programs credited with managing this aging effect, as required by 10 CFR 54.21(d).

Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.1.2.3.7 Reactor Vessel Internals Inspection Program

The applicant described its Reactor Vessel Internals Inspection Program in LRA Appendix B.2.4. This is a new program and the applicant stated that the program is consistent with GALL AMP XI.M16, "PWR Vessel Internals," with clarifications.

The staff reviewed the applicant's description of the program in LRA Appendix B, Section B.2.4 to determine whether the applicant demonstrated that it will adequately manage the applicable aging effects at VCSNS during the period of extended operation, as required by 10 CFR 54.21(a)(3).

Summary of Technical Information in the Application

The applicant's Reactor Vessel Internals Inspection Program is discussed in LRA Appendix B, Section B.2.4, and in Appendix A, FSAR Section 18.2.28. This is a new program, and the applicant stated that the program will be consistent with GALL AMP XI.M16, "PWR Vessel Internals." However, the applicant added the clarification that the VCSNS resolution criterion for the enhanced VT-1 inspection is expected to be less than that specified in the GALL program.

This new AMP is credited with managing the following aging effects:

- loss of fracture toughness in baffle/former bolts and other reactor vessel internals
- changes in dimension due to void swelling in reactor vessel internals, crack initiation and growth in baffle/former bolts, and other reactor vessel internals
- loss of preload in baffle/former bolts and other reactor internals and
- loss of material due to wear in reactor vessel internals

The applicant stated that this new inspection program will supplement the existing ISI Plan to assess the condition of RV internals. The applicant has identified the components that will be inspected under this program. 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 will be performed to detect the presence and extent of cracking and loss of material. For bolts and other inaccessible components, a volumetric inspection will be performed to detect the

In LRA Tables 3.1-1 and 3.1-2, the applicant lists the reactor vessel and its appurtenances within the scope of the license renewal with their material groups and environment. The intended functions of these components are listed in LRA Table 2.3-3. In the LRA tables, the applicant also identifies the aging effects requiring management and the plant-specific AMPs required to manage these aging effects during the period of extended operation. The components within the scope of license renewal are grouped in accordance with their component types, and these groups are listed in these tables.

In LRA Section 4.0, the applicant identifies the following two TLAAs applicable to the reactor vessel and its appurtenances:

- reactor vessel neutron embrittlement (Section 4.2)
- metal fatigue (Section 4.3)

Aging Effects:

In accordance with LRA Section 3.1, the applicant has performed a review of industry experience and NRC generic communications relative to the reactor vessel and its appurtenances to present reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for VCSNS. This also included the plant-specific operating experience at VCSNS.

The materials of construction for the reactor pressure vessel and its appurtenances subject to an AMR are low-alloy steel for vessel shell, closure head, bottom head, and flange (including stainless steel cladding); carbon steel or low-alloy steel for vessel support; stainless steel for CRDM housings; Alloy 600 for head penetrations and reactor vessel core support pads; and high-strength low-alloy steel for reactor vessel closure studs. The inside surface of reactor vessel components are exposed to borated water and the external surfaces of reactor vessel components are exposed to air. In LRA Table 3.1-1, the applicant identifies the following aging effects applicable to the reactor vessel and its appurtenances requiring an AMR:

- loss of fracture toughness for low-alloy steel reactor vessel beltline shell and welds internally exposed to chemically treated borated coolant
- cracking of stainless steel reactor vessel nozzle safe ends and CRD housings, and cladding on low-alloy steel vessel shell, heads and nozzles exposed to chemically treated borated coolant
- cracking of Alloy 600 closure head and bottom head penetrations, and core support pads exposed to chemically treated borated coolant
- loss of closure integrity of high strength low alloy reactor vessel closure studs exposed to containment environment (i.e., air)
- loss of material due to wear of reactor vessel flange, closure studs and core support pads

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**loss of material in low alloy steel reactor vessel shell and heads, carbon steel vessel support and closure studs exposed to leaking borated coolant**

The applicant states that the identification of the above aging effects in LRA Table 3.1-1 is consistent with the GALL report. In LRA Table 3.1-1, AMR Item 7, the applicant states that the vessel shell materials at VCSNS do not include ASME SA-508, Class 2 material, which is susceptible to underclad cracking if the cladding was deposited on it with a high-heat input welding process. Since the VCSNS vessel shell does not include ASME SA-508, Class 2 material, it is not susceptible to underclad cracking. However, as discussed in Section 3.1.2.2.7 of this SER, the vessel flange and nozzle forgings are made of ASME SA 508, Cl2 material. The underclad cracking is not an applicable aging effect for these forgings because the high-heat input welding processes affecting underclad cracking were not used for application of cladding to these components. The staff accepts that underclad cracking is not an applicable aging effect for the VCSNS vessel components made of ASME SA-508, Class 2 material because the high-heat input welding processes were not used for application of cladding to these components.

LRA Table 3.1-1, AMR Item 9, represents the AMR results for various Ni alloy components except CRD nozzles exposed to the chemically treated borated coolant. The applicant states that at VCSNS, only, the core support pads and bottom head penetration tubes are included in this item. The applicant identifies crack initiation and growth due to PWSCC as an applicable aging effect for these components. The staff finds this identification acceptable because it is consistent with industry experience.

In LRA Table 3.1-1, AMR Item 18, the applicant states that for closure studs, the aging effect requiring management is loss of closure integrity rather than cracking. The staff has evaluated the management of aging effects for reactor vessel closure studs in Section 3.1.2.3.4 and determined that management of loss of closure integrity includes the management of cracking of closure studs.

In LRA Table 3.1-1, AMR Item 22, the applicant states that loss of material due to wear is not considered a valid aging effect for the control rod drive flange bolting requiring management. This statement implies that VCSNS has installed control rod drive flange bolting. However, Section 5.4.2 of the VCSNS UFSAR states that the upper ends of the CRD nozzles have a welded flexible canopy seal and not bolting. The staff issued RAI 3.1.2.4.3-2, requesting the applicant explain this discrepancy. In response to RAI 3.1.2.4.3-2, in a letter dated June 12, 2003, the applicant states that the VCSNS CRD nozzles are seal welded to the CRDMs. Therefore, the pressure boundary is not a bolted connection. The applicant further states that bolts are used for the magnetic housings; however, they do not constitute pressure boundary and are not in scope. Therefore, the staff finds the applicant's clarification acceptable because it is consistent with the design of the Westinghouse plants where the upper end of the CRD nozzles are welded to the flexible canopy seal.

LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 28, for the AMR results for the reactor vessel closure studs assembly. However, LRA Table 3.1-1, AMR Item 28, presents the AMR results for vessel and vessel closure head flanges and not for closure studs assembly. The

staff issued RAI 3.1.2.4.3-3, requesting the applicant to explain this discrepancy. In response to RAI 3.1.2.4.3-3, in a letter dated June 12, 2003, the applicant states that the reactor vessel closure studs are not included in Table 3.1-1, Item 28. The staff finds the response acceptable because it is consistent with GALL.

The staff has reviewed NUREG-1801 Chapter IV.A2, Reactor Vessel, and confirmed that the applicant's identification of the aging effects in Table 3.1-1 for the reactor vessel and its appurtenances is consistent with the GALL report, except for the discrepancies noted in SER Table 3.1-1, and therefore acceptable.

LRA Table 3.1-2, AMR Items 1 and 2, state that the stainless steel CRD housings, Alloy 600 vessel closure head and bottom head penetrations, and Alloy 82/182 welds between the vessel nozzle safe ends and main coolant loop piping are exposed to moist air environment. These components are not susceptible to any aging effects requiring management. This is acceptable because the stainless steel and Ni-alloy based components are resistant to general corrosion and the ambient environment at VCSNS does not contain contaminants of sufficient concentration to cause an aging effect requiring management.

In Table 3.1-2, AMR Item 7, the applicant identifies loss of material as an aging effect for stainless steel and Ni-alloy components attached to the reactor vessel.

The stainless steel components are inherently tough and resistant to general corrosion; however, loss of material due to crevice and pitting corrosion may be an applicable aging effect for these components under wet conditions, especially if the components have creviced areas that may be exposed to the fluids. Therefore, the applicant's identification of loss of material as an aging effect for stainless steel components internally exposed to chemically treated borated coolant is acceptable.

LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 23, and LRA Table 3.1-2, AMR Item 11, for AMR results for Alloy 600 reactor vessel closure head penetration tubes. Both AMR Items address cracking as an aging effect for these tubes. AMR Item 23 proposes the Alloy 600 aging management program whereas AMR Item 11 proposes the chemistry program. The staff issued RAI 3.1.2.4.3-5, requesting the applicant to explain this discrepancy. In response to RAI 3.1.2.4.3-5, in a letter dated June 12, 2003, the applicant states that LRA Table 3.1-2, Item 11, should not be referenced in LRA Table 2.3-3 for RV closure head penetration tubes. The staff finds this explanation acceptable because it is consistent with the AMR results presented in the LRA.

The austenitic stainless steel and Ni-alloy based reactor vessel appurtenances (i.e., CRD housings, vessel head penetrations, and Alloy 82/182 welds) are susceptible to stress corrosion cracking at the external surface if they come in contact with halogens that may be present in the thermal insulation. The applicant has not identified cracking as an aging effect at the external surface of these components. The staff issued RAI 3.1.2.4.3-4, requesting the applicant to submit a description of all insulation used on austenitic stainless steel RCS piping to ensure that the reactor vessel appurtenances are not susceptible to stress-corrosion cracking from halogens. In response to RAI 3.1.2.4.3-4, in a letter dated June 12, 2003, the applicant states that stainless steel reflective insulation is the most commonly used insulation type on stainless steel piping and components. Various other types of insulation used on stainless steel are encapsulated in stainless steel. The applicant further states that unlike

The Bulletins were written for internal surfaces.

fibrous insulation, stainless steel does not need controls for halogens. The staff agrees with the applicant that the external surface of the stainless steel and Ni-alloy based reactor vessel appurtenances is not susceptible to cracking due to halogen induced SCC, because VCSNS uses stainless steel insulation, which does not contain halogens. However, the external surfaces of these materials have cracked due to SCC and are discussed in Bulletins 2001-1, 2002-1, and 2002-2. Cracking due to SCC of these materials is managed by the Alloy 600 aging management program.

The AMR results for the PWR reactor vessel leak detection line (GALL Item IV.A2.1-f) are presented in Table 1 of NUREG-1801. Therefore, AMR Item 9 in LRA Table 3.1-1 should also include these AMR results. The staff issued RAI 3.1.2.4.3-6, requesting the applicant to confirm whether the AMR results for the reactor vessel leak detection line are included in LRA Table 3.1-1, AMR Item 9. In response to RAI 3.1.2.4.3-6, in a letter dated June 12, 2003, the applicant states that the reactor vessel flange leak detection line components are classified as Code Class 2 components and the corresponding AMR results for these components are presented in LRA Table 3.1-2, Items 1, 5, and 6. The applicable aging effects are loss of material and cracking at the inside surface of the line. The staff finds the response acceptable because stainless steel components are susceptible to loss of material due to crevice and pitting corrosion and cracking due to SCC when exposed to chemically treated borated coolant.

The aging effects identified in the LRA for the reactor vessel and its appurtenances are consistent with industry operating experience for the materials and environments listed. The staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified.

Aging Management Programs:

In LRA Tables 3.1-1 and 3.1-2, the applicant has identified the following six AMPs for managing the aging effects associated with reactor vessel and its appurtenances:

- boric acid corrosion surveillance program
- chemistry program
- Inservice inspection plan
- reactor vessel surveillance program
- alloy 600 aging management program
- reactor head closure studs program

The boric acid corrosion surveillance program (LRA Section B.1.2) was developed by the applicant in response to NRC Generic Letter 88-05. Inspections are performed to present reasonable assurance that borated water leakage from the reactor coolant pressure boundary does not lead to undetected loss of material on the external surface of carbon steel or low alloy steel bolting. The staff has evaluated this common AMP and found it to be acceptable for managing the aging effect of loss of material identified for the RCS pressure boundary closure bolting. The staff's evaluation of this AMP is documented in Section 3.0.3.1 of this SER. The evaluation of this AMP, as it is applied for managing PWSCC cracking in the reactor vessel closure head penetrations, is presented in Section 3.1.2.3.1 of this SER.

The applicant credits the Inservice inspection plan (LRA Section B.1.7), the chemistry program (LRA Section B.1.4), and the Alloy 600 aging management program (LRA Section B.1.1) for

- limit fuel assembly movement
- maintain alignment between fuel assemblies and control rod drive mechanisms
- direct coolant flow past the fuel elements
- direct coolant flow to the pressure vessel head
- provide gamma and neutron shielding
- provide guides for the in-core instrumentation

The coolant flows from the vessel inlet nozzles, down the annulus between the core barrel and the vessel wall, and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

Summary of Technical Information in the Application

The description of the reactor vessel internals can be found in LRA Section 2.3.1.4. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-4. The components, aging effects, and aging management programs are discussed in Section 3.1 of the LRA, and are listed in LRA Tables 3.1-1 and 3.1-2. The staff reviewed Section 3.1 of the LRA to determine whether the applicant had demonstrated that the effects of aging on the RV internals will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

All of the major RV internals are fabricated from Type 304 stainless steel except for (a) the bolts and dowel pins, which are fabricated from Type 316 stainless steel; (b) the radial support key bolts, which are fabricated from Alloy X-750; and (c) the radial support clevis inserts and clevis insert bolts, which are fabricated from Alloy 600. There are no cast austenitic stainless steel (CASS) RV internal components within the scope of license renewal.

The RV internals that are within the scope of license renewal are exposed to borated reactor coolant water at approximately 315.6 °C (600 °F) and 15.41 MPa (2235 psig). These components are all located within the reactor pressure vessel.

Aging Effects:

In LRA Tables 3.1-1 and 3.1-2, the applicant identifies the following applicable aging effects for the RV internal components subject to an AMR:

- loss of fracture toughness
- changes in dimension
- crack initiation and growth
- loss of preload
- loss of material

As previously noted in Section 3.1.2.2.7 of this SER, the applicant states that, 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. The applicant proposes to monitor these activities and implement the resulting methods, as necessary.

Aging Management Programs:

In LRA Tables 3.1-1 and 3.1-2, the applicant identifies the following two AMPs to manage the aging effects associated with RV internals:

- reactor vessel internals inspection program
- chemistry program

The reactor vessel internals inspection program is a new program developed by the applicant to manage the aging effects impacting the RV internals. It supplements the applicant's existing in-service inspection plan. The chemistry program is credited with managing the aging effects of several components in different structures and systems and is, therefore, considered a common aging management program. The applicant concluded that these two AMPs will manage the effects of aging such that the intended function of the reactor vessel internal components will be maintained consistent with the CLB under all design loading conditions throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

LRA Section 4.0 and Table 3.1.1 identify metal fatigue of Class 1 components as the only TLAA applicable to RV internals.

INTERNALS ARE CLASS 2
(SEE PAGE 4-13)

Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.1, and in LRA Tables 3.1-1 and 3.1-2 and pertinent sections of the LRA Appendices A and B, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended function(s) of the reactor vessel internal components will be maintained consistent with the CLB throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

In LRA Tables 3.1-1 and 3.1-2, the applicant lists the reactor vessel internal components within the scope of the license renewal with their material groups and environment. The intended functions of these components are listed in LRA Table 2.3-4. LRA Tables 3.1-1 and 3.1-2 identify the aging effects requiring management and the plant-specific AMPs required for managing these aging effects during the period of extended operation. The list of components within the scope of license renewal is grouped in accordance with their component types.

In LRA Table 3.1-1, the applicant identifies metal fatigue for ASME Class 1 components as the TLAA that is applicable to the reactor vessel internals. The staff's evaluation of this TLAA is presented in Section 4.3 of this SER.

Aging Effects:

In accordance with LRA Section 3.1, the applicant has performed a review of industry experience and NRC generic communications, relative to the reactor vessel internal components, to provide reasonable assurance that the aging effects that require management

staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified.

Aging Management Programs:

In LRA Tables 3.1-1 and 3.1-2, the applicant has identified the following three AMPs for managing the aging effects associated with in-core instrumentation system components:

- bottom-mounted instrumentation inspection program
- chemistry program
- Inservice inspection plan

The bottom-mounted instrumentation inspection program (LRA Section B.1.3) is credited for managing loss of material due to fretting wear of the BMI thimble tubes. The staff has evaluated this AMP in Section 3.1.2.3.6 of this SER.

The Chemistry Program (LRA Section B.1.4) is credited for managing the aging effect of loss of material and cracking on the outside surface of the BMI thimble tubes and the pressure retaining portion of the in-core thermocouples, as well as the inside surface of the guide tubes supporting the thimble tubes between the seal table and vessel lower head. The use of chemistry program alone may not be adequate for managing loss of material and cracking of thimble tubes and guide tubes for the following reason: According to LRA Drawing 1MS-44-014-1, the seal table elevation is the same as the vessel flange elevation. Since the reactor coolant is exposed to containment environment during refueling, the stagnant reactor coolant near the seal table may be oxygenated because of the high elevation. As a result, the stagnant coolant in the guide tubes would be more aggressive than the normal RCS coolant. Therefore, the applicant needs to provide an aging management program to ensure that loss of material and cracking are not taking place at the inside surface of the guide tube and the outside surface of the thimble tube surrounded by the guide tube.

The staff issued RAI B.1.4-1, requesting the applicant to address its concern about potentially accumulated oxygen in the guide tube near the seal table. The response to RAI B.1.4-1, was discussed in the June 22, 2003, conference call between the staff and the applicant. In response to an action item based on this conference call in a letter dated September 2, 2003, the applicant states that buildup of oxygen in the guide tube is not a significant concern. The area between the outside surface of the BMI tubes and the inside surface of the guide tubes supporting the thimble tubes between the seal table and vessel lower head remains filled during refueling and is not opened and drained. If during refueling there is a leakage while the thimbles are withdrawn or inserted, coolant will leak out but air will not leak into the region of concern. The staff does not agree with the applicant that oxygen buildup may not take place in upper portion of the guide tubes. High levels of oxygen may be introduced into the primary system during shutdown operations as a result of exposing the reactor coolant system to the outside air. This oxygen may accumulate in the upper portion of the guide tubes because the elevation of the seal table, which is same as the vessel flange elevation. The applicant, accumulated oxygen is not of concern because except very near the vessel the temperature of the thimble tube and guide tube is at ambient air temperature. At the seal table the temperature is less than 49°C (120°F), which significantly reduces the potential of loss of material and cracking of the fitting. Therefore, the staff accepts the applicant's position that the chemistry program is adequate for managing loss of material and cracking of the thimble tube

and guide tube between the vessel lower head and seal table except very near the vessel where temperature are higher than 93°C (200°F).

In a conference call on September 16, 2003, the staff informed the applicant that their September 2, 2003 response did not provide aging management for cracking due to SCC of the stainless steel guide tube in close proximity to the reactor vessel. In a response, by letter dated September 24, 2003, the applicant states that the stainless steel in-core neutron detector conduits (guide tubes) are welded to the nickel-based alloy bottom head penetration tubes. The staff reviews aging management of bottom head penetrations, as proposed by the applicant, in Section 3.1.2.2.9 of this SER and finds it adequate. The applicant credits the Alloy 600 aging management program (LRA Appendix B.1.1) in addition to the chemistry program (LRA Appendix B.1.4) for managing cracking in the bottom head penetration tubes. The Alloy 600 aging management program provides for inspecting the signs of boric acid leakage from the bottom head penetration, including its weld with the guide tube, during outages and monitoring primary coolant leakage per Technical Specifications during plant operation.

The applicant further states that the bottom head penetrations extend over eight inches from the bottom surface of the vessel where they are welded to in-core neutron detector conduits. The configuration of the conduits tends to allow a significant temperature reduction from the RCS temperature. In addition, chemistry controls of the reactor coolant significantly reduce the source of contaminates and measures were taken to prevent sensitization during fabrication. The applicant concludes that the combination of these factors reduces the likelihood of stress corrosion cracking of the conduit and, therefore, the chemistry program alone is adequate for managing cracking due to SCC of the conduit and a one-time inspection is not required. The staff finds the use of the chemistry program alone for managing cracking due to SCC in the in-core neutron detector conduit that is in close proximity of the vessel bottom head acceptable because the weld between the conduit and the bottom head penetration will be inspected as part of the aging management of the bottom head penetrations. In addition, this weld is the bounding location for the guide tube as far as cracking due to SCC is concerned because the temperature at the weld will be higher than at any other locations on the conduit. This closes the RAI B.1.4-1.

The applicant credits the ISI plan (LRA Section B.1.7) for managing loss of closure integrity of the closure bolting. The staff issued RAI 3.1.2.4.5-1b, requesting the applicant to explain how ISI plan would manage loss of closure integrity, i.e., loss of material due to wear and loss of preload, and ensure that the pressure boundary of the bolted joint would be maintained during the extended period of operation. In response to RAI 3.1.2.4.5-1b, in a letter dated June 12, 2003, the applicant states that this bolting is disassembled and assembled at each refueling. The ISI plan requires surface examination of all bolting when it is disassembled, i.e., at each refueling. The surface examination will detect any loss of material due to wear. Retorquing to the desired preload during each refueling will remove the loss of preload that might have taken place during the preceding fuel cycle. The staff accepts the applicant response because the applicant is adequately managing the applicable aging effects for the closure bolting for in-core thermocouple seals by managing loss of closure integrity of this bolting.

The staff has evaluated this common AMP and, found it acceptable for managing the aging effects for the in-core instrumentation system components. The staff's evaluation of this AMP is presented in Section 3.0.3 of this SER.

On the basis of its review, the staff finds that the AMPs credited in the LRA for the in-core instrumentation system components will effectively manage or monitor the aging effects identified in the LRA.

Conclusions

On the basis of its review, the staff finds the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.4.6 Pressurizer

The VCSNS pressurizer is a vertical, cylindrical vessel with a hemispherical top and bottom heads constructed of low alloy steel (SA533 Grade A Class 2), with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. A stainless steel liner, i.e., thermal sleeve, is used ~~in lieu of cladding~~ in some nozzles. The surge line nozzle and removable electric heaters are installed in the bottom head. A thermal sleeve is provided to minimize stresses in the surge line nozzle. Spray line nozzles, relief, and safety valve connections are located in the top head of the pressurizer. The skirt type support is attached to the lower head and extends for a full 360 degrees around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes securing to its foundation. The VCSNS pressurizer is designed and constructed in accordance with the ASME Code, Section III.

The VCSNS pressurizer instrumentation includes 2 temperature detectors, one in the steam phase and 1 in the water phase, ~~3~~ 6 pressure transmitters, and ~~3~~ 5 liquid level transmitters.

During an outsurge from the pressurizer, flashing of water to steam, and the generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the RCS, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set point of the power operated relief valves for normal operating transients. Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

UFSAR Sections 5.5.10 and 5.6 describe the VCSNS pressurizer design and LRA Section 2.3.1.6 identifies the pressurizer components that are within the scope of license renewal and their intended functions.

Technical Information in the Application

The applicant describes its AMR of the pressurizer components in LRA Section 3.1, "Aging Management of Pressurizer." The staff reviewed this section of the LRA to determine whether the applicant has demonstrated that the effects of aging on the pressurizer will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

Ten component groups associated with the pressurizer are listed in LRA Tables 3.1-1 and 3.1-2. They include shell components, nozzles, head penetrations, and CRDM housings. The

intended function of all of these components is to provide a pressure boundary. The reactor vessel core support pads support the reactor vessel internals.

Aging Effects:

In LRA Tables 3.1-1 and 3.1-2, the applicant identifies the following aging effects for the pressurizer components that are subject to an AMR:

- cracking
- loss of material
- loss of closure integrity

Aging Management Programs:

In LRA Tables 3.1-1 and 3.1-2, the applicant identifies the following existing AMPs to manage the aging effects associated with the pressurizer:

- alloy 600 aging management program
- chemistry program
- Inservice inspection plan
- boric acid corrosion surveillance program

The applicant concluded that these AMPs will manage the effects of aging such that the intended functions of the pressurizer components will be maintained consistent with the CLB under all design loading conditions throughout the period of extended operation, as required by 10 CFR 54.21(a)(3). The applicant identifies metal fatigue (Section 4.3 of the LRA) as a TLAA in Section 3.1 of the LRA that is applicable to pressurizer components.

Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in LRA Section 3.1, Tables 3.1-1 and 3.1-2, and pertinent sections of LRA Appendices A and B, regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended function(s) of the pressurizer will be maintained consistent with the CLB throughout the period of extended operation, as required by 10 CFR 54.21(a)(3).

In LRA Tables 3.1-1 and 3.1-2, the applicant lists the pressurizer components within the scope of the license renewal with their material groups and environment. The intended functions of these components are listed in LRA Table 2.3-6. The tables also identify the aging effects requiring management and the plant-specific AMPs required managing these aging effects during the period of extended operation. The components within the scope of license renewal are grouped in accordance with their component types, and these groups are listed in these tables. In LRA Section 4.3, the applicant identifies metal fatigue as a TLAA that is applicable to pressurizer components. The staff's evaluation of this TLAA is presented in Section 4.3 of this SER.

Aging Effects:

In accordance with LRA Section 3.1, the applicant has performed a review of industry experience and NRC generic communications, relative to the pressurizer components, to provide reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for VCSNS. This also included the VCSNS plant-specific operating experience.

The material of construction for the pressurizer components subject to an AMR are low-alloy steels (SA533 Grade A, Cl 2 and SA508 Cl 2A) for the pressure retaining components including shell and heads; austenitic stainless steels for nozzle safe ends, thermal sleeves, and heater wells; stainless steel weld metal for cladding and buttering; SA193 Gr B-7 for closure bolting; and Alloy 82/182 weld metal for bimetallic welds and partial attachment welds. In LRA Table 3.1-1, the applicant identifies the following aging effects applicable to the pressurizer components requiring an AMR:

- cracking of Alloy 82/182 welds between pressurizer nozzles and safe ends, between thermal sleeves and safe ends, and between heater sleeves and the pressurizer lower head, exposed to chemically treated borated coolant
- cracking of stainless steel instrument and sample lines, heater sleeves, thermal sleeves, and safe ends exposed to chemically treated borated coolant
- cracking of pressurizer shells, heads, and nozzle cladding with stainless steel and internally exposed to chemically treated borated coolant
- cracking of pressurizer integral supports

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**loss of material in low-alloy steel pressurizer shells and heads, and closure bolting externally exposed to leaking borated coolant**

The applicant states that the identification of the above aging effects in LRA Table 3.1-1 is consistent with the GALL report. However, the GALL report presents an AMR for six additional pressurizer components (pressurizer seismic lugs, heater elements (heater sheaths), manway pad gasket seating surface, safety valves, relief valves, and spray nozzles) that are not addressed in the LRA. The staff issued RAI 3.1.2.4.6-1, requesting that the applicant submit an explanation for not presenting an AMR for the following pressurizer components: seismic lugs, heater elements, manway pad gasket seating surface, safety valves, and relief valves. In response to RAI 3.1.2.4.6-1, in a letter dated June 12, 2003, the applicant submitted the following explanation for the first three components. The pressurizer seismic lugs are included with the pressurizer shell and are not included as a separate component. Immersion heater well assemblies is the component name used at VCSNS for the heater sheaths and they are included in LRA Table 3.1-1, Item 24, and LRA Table 3.1-2 Items 1 and 7. The manway pad gasket-seating surface is the stainless steel clad mating surface of the manway nozzle and is not called out as a separate component. The staff finds the explanation acceptable because the AMR results for the three components are included in the LRA. In response to RAI 3.1.2.4.6-1, the applicant further states that the AMR results for the pressurizer safety and relief valves are included in LRA Table 3.1-1, Items 19 and 24, and LRA Table 3.1-2, Items 1 and 5.

The staff finds the response acceptable because the applicant includes these valves within the scope of license renewal. The staff's evaluation of the AMR results for these valves is presented in Section 3.1.2.4.2 of this SER. As discussed in Section 2.3.1.6.2 of this SER, pressurizer spray head is not within the scope of license renewal.

According to LRA Table 2.3-6, the applicant presents the AMR results for the pressurizer nozzles and safe ends. However, it is not clear to the staff about which specific nozzles are addressed by the LRA. The staff issued RAI 3.1.2.4.6-2, requesting the applicant to confirm whether the following five pressurizer nozzles and safe ends are included within the scope of the LRA: surge nozzles, spray nozzles, safety nozzles, relief nozzles, and their safe ends, and instrument nozzles. In response to RAI 3.1.2.4.6-2, in a letter dated June 12, 2003, the applicant states that the five nozzles and safe ends are within the scope of the LRA. The staff finds the response acceptable because the same nozzles and safe ends are identified as the components within the scope of license renewal by the Westinghouse report WCAP 14574-A.

In LRA Table 2.3-6, the applicant presents the AMR results of the pressurizer manway cover (Row 4), manway cover (Row 6) and manway forgings (Row 7) exposed to chemically treated borated coolant. The staff issued RAI 3.1.2.4.6-3, requesting that the applicant explain the difference between the manway cover and the manway forgings. In response to RAI 3.1.2.4.6-3, in a letter dated June 12, 2003, the applicant explains that the manway cover listed in Row 4 of LRA Table 2.3-6 is a non-structural stainless steel insert. The staff finds the response related to the manway cover (Row 4) acceptable because it is consistent with the corresponding AMR results presented in the LRA. However, the staff determined that the remaining part of the applicant's explanation was not clear. Therefore, the staff further discussed this RAI with the applicant during the June 22, 2003, conference call which is discussed below.

Based on the conference call, the applicant submitted the following additional information in response to RAI 3.1.2.4.6-3. The major components of the pressurizer manway include the manway forgings, manway covers, and manway cover inserts. The manway covers and forgings are made of low-alloy steel, whereas the manway cover insert is made of stainless steel, as mentioned earlier. The manway forging is welded to the pressurizer shell. The applicant further indicated that the manway cover is a pressure-retaining component that is bolted to the manway forging. The stainless steel insert prevents the manway cover from being in direct contact with the reactor coolant. The aging effect that the applicant identified for the manway covers and forgings is loss of material due to boric acid corrosion. An additional aging effect for the manway forgings is cumulative fatigue damage. The aging effects for the manway cover inserts are loss of material due to pitting and crevice corrosion and cracking due to stress corrosion cracking. The staff finds the information provided by the applicant in its response to the staff acceptable because the identification of aging effects is consistent with GALL.

The staff reviewed NUREG-1801 Chapter IV.C2, Reactor Coolant System and Connected Lines, and confirmed that the applicant's identification of the aging effects in Table 3.1-1 for the pressurizer components is consistent with the GALL report, and, therefore, acceptable.

In Table 3.1-2, AMR Items 7 and 11, the applicant identifies the following two aging effects for the pressurizer components:

likely to be susceptible to corrosion and, therefore, will not likely to experience ligament cracking.

In LRA Table 3.1-1, AMR Item 21, the applicant addresses potential loss of material (wall thinning) due to flow-assisted corrosion (FAC) for the steam generator steam and feedwater nozzles and safe ends. The applicant states that this effect is not an applicable aging effect for these components at VCSNS and aging management is therefore not required, but provides no justification for this conclusion. The staff issued RAI 3.1.2.4.7-4, requesting the applicant to provide the technical basis for determining that loss of material caused by FAC is not an applicable aging effect for the steam generator nozzles and safe ends. In its response to RAI 3.1.2.4.7-4, in a letter dated June 12, 2003, the applicant states that the main steam exiting the steam generators is dry (less than 0.1% moisture), and dry steam is not a concern for flow-accelerated corrosion. In addition, according to the Westinghouse report, "Westinghouse Delta 75 Steam Generator Design and Fabrication information for the Virgil C. Summer Nuclear Station," WCAP-13480, Rev. 1, October 1993, the main steam and feedwater nozzles are fabricated from low-alloy steel and, therefore, not susceptible to FAC. Also there is no safe end for feedwater nozzle at VCSNS. The staff accepts the applicant's conclusion that loss of material due to FAC is not an applicable aging effect for the steam generator main steam nozzle and safe end, and feedwater nozzle because these components are made of low alloy steel and main steam exiting steam generator is dry.

In LRA Table 3.1-1, AMR Item 22, the applicant states that for steam generator bolting, loss of closure integrity rather than loss of preload or cracking is the aging effect requiring management. The staff issued RAI 3.1.2.4.7-5, requesting the applicant to explain how does the aging effect of loss of closure integrity in steam generator bolting differ from the effects of loss of material, loss of preload, and cracking. The staff had issued similar RAIs (RAI 3.1.2.4.2-1 and RAI 3.1.2.4.6-7) for bolting closures for Class 1 piping and the pressurizer. In its response to RAI 3.1.2.4.7-5, the applicant stated that loss of mechanical closure integrity can result in failure of the mechanical joint and its evidenced by leakage rather than joint failure. The applicant further states that this failure of mechanical joint can be attributed to loss of bolt preload, loss of bolting material by water, and cracking of high strength bolting material. Therefore, loss of closure integrity includes the effects of loss of preload, loss of material, and cracking of bolting materials. The staff concludes that the applicant has identified appropriate aging effect for the steam generator bolting and that management of loss of closure integrity includes management of three additional aging effects: loss of material, loss of preload, and cracking.

In LRA Table 3.1-1, AMR Item 32, the applicant identifies the aging effect of crack initiation and growth due to SCC and PWSCC for the channel head divider plate in the VCSNS steam generators. The staff issued RAI 3.1.2.4.7-6, requesting the applicant to confirm whether the weld on the divider plate is an Alloy 82/182 weld. In its response to RAI 3.1.2.4.7-6, in a letter dated June 12, 2003, the applicant states that the divider plate is welded with Alloy 82/182; however, the final pass was made with Alloy 52/152 so the weld does not have Alloy 82/182 exposed to borated water. The applicant further states that VCSNS has no evidence of cracking of the 52/152 welds since the installation of the replacement steam generators in the 1994 outage. The staff finds the applicant's response acceptable because the absence of cracking in these welds is consistent with the use of Alloy 52/152 weld metal for which industry experience has not revealed any cracking.

The aging effects identified in the LRA for the steam generator components in LRA Table 3.1-1 are consistent with GALL IV.D1, "Steam Generator (Recirculating)." The staff finds that the aging effects were identified.

Review of Aging Effects on Steam Generator-Related Items in LRA Table 3.1-2

In LRA Table 3.1-2, the applicant identifies the following additional aging effects for steam generator components exposed to secondary treated water. These aging effects are considered to be different from or are not addressed in GALL but are identified as a result of the applicant's license renewal review.

- 11. loss of material in carbon steel components exposed to the treated secondary water/steam environment due to crevice corrosion, general corrosion, pitting corrosion, and galvanic corrosion
- 12. loss of material in various steam generator pressure boundary components due to crevice corrosion and pitting corrosion
- 13. loss of material in secondary-side thermal sleeves and the steam outlet nozzle flow limiter due to crevice corrosion and pitting corrosion
- 14. crack initiation and growth in secondary-side thermal sleeves and the steam outlet nozzle flow limiter due to stress corrosion cracking and flaw growth at welds (Item 9)
- 15. loss of material in the feedwater distribution pipe and fittings due to crevice and pitting corrosion
- 16. crack initiation and growth in the feedwater distribution pipe and fittings due to stress corrosion cracking
- 17. loss of material in stainless steel and nickel-based alloy components exposed to treated secondary water due to crevice and pitting corrosion; crack initiation and growth in these components due to SCC

In LRA Table 3.1-2, AMR Item 2, the applicant lists primary side nozzle safe ends that are made of austenitic stainless steel and exposed to an air-gas (moist air) environment. The applicant states that no aging effect or mechanism is identified for these components, since the ambient environment at VCSNS does not contain contaminants of sufficient concentration to cause aging effects requiring management. The staff agrees with the applicant's conclusion that these materials are not expected to undergo any aging degradation in this environment.

In LRA Table 3.1-2, AMR Item 3, the applicant identifies loss of material as an applicable aging effect for carbon steel components (other than shell) exposed to secondary treated water. In addition, in LRA Table 3.1-2, AMR Items 8, 9, and 10, the applicant identifies loss of material and cracking as aging effects for stainless steel and low alloy steel components exposed to secondary treated water. The staff finds this identification of applicable aging effects acceptable because it is consistent with the similar identification in GALL IV.D1. For example, GALL Item IV.D1.1-c, identifies loss of material as an aging effect for carbon steel and low alloy steel components exposed to secondary treated water.

cracking from thermal and radiation embrittlement. Exposure of hypalon, rubber, and neoprene components to a ventilation air environment results in aging effects.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the air handling and local ventilation and cooling systems:

- Chemistry Program (B.1.4)
- Boric Acid Corrosion Surveillances Program (B.1.2)
- Service Water System Reliability and In-Service Testing Program (B.1.9)
- Preventive Maintenance Activities — Ventilation Systems Inspections (B.1.26)
- Inspections for Mechanical Components Program (B.2.11)
- Maintenance Rule Structures Program (B.1.18)
- Heat Exchanger Inspections Program (B.2.12)

A description of these AMPs is provided in Appendix B of the LRA. The applicant concluded that the effects of aging associated with the components of the air handling and local ventilation and cooling systems will be adequately managed by these AMPs during the period of extended operation.

Staff Evaluation

Aging Effects:

The staff reviewed the information in Section 2.3.3.1 and Tables 2.3-18, 3.3-1, and 3.3-2 in the LRA, as well as in the supplementary table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query." During its review, the staff determined that additional information was needed.

Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1, pertaining to this issue of the plant-specific characteristics of the environment. The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER, and is characterized as resolved.

For components in this system listed in the supplemental table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query," the applicant indicated that galvanized steel ductwork in a "yard" environment has no identified aging effects and does not require an AMP. The staff finds that this conclusion may not be justified because of factors associated with corrosive agents in the local environment and rainfall. By letter dated March 28, 2003, the staff requested, in RAI 3.3.2.4.1-1, the applicant to provide justification for the conclusion that galvanized steel ductwork in a "yard" environment has no identified aging effects.

In its response dated June 12, 2003, the applicant stated that VCSNS is located well inland and is located in an area where forestry is the primary commercial activity. VCSNS does not see salt or other corrosive materials in the air from agriculture or industry. Crevice and pitting corrosion are not considered to be aging effects for external surfaces because the ambient environment

does not contain contaminants of sufficient quantity to concentrate on external surfaces such that pitting or crevice corrosion would occur. Rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates.

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

On the basis of its review, the staff finds the applicant's response to RAI 3.3.2.4.1-1 acceptable because (1) VCSNS does not see salt or other corrosive materials in the air from agriculture or industry, (2) rainwater analyses reveal a concentration of less than 10 ppm for chlorides and sulfates, and (3) the ambient temperature in the area where these components are located is less than 140 °F.

For components in this system list in the supplemental table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query," the applicant stated that carbon steel cooling coil headers in a treated water environment are subject to SCC. However, no AMP has been provided to address this aging effect. By letter dated March 28, 2003, the staff requested, in RAI 3.3.2.4.1-2, the applicant to explain why no AMP has been provided to address this aging effect.

In its response dated June 13, 2003, the applicant stated that according to industry references, SCC of carbon and low-alloy steel components is not considered to be an applicable aging mechanism in a treated water environment. Industry data do not exhibit widespread incidence of SCC in low-strength carbon steels; however, there was a reported case suspected to be nitrate-induced SCC of carbon steel in a treated water system. VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrates are added as a corrosion inhibiter. In these closed systems, there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrates are added as a corrosion inhibitor by the Chemistry Program at levels within EPRI guidelines; therefore, VCSNS maintains that the Chemistry Program adequately manages SCC of carbon steel components in a treated water environment.

On the basis of its review, the staff finds the applicant's response acceptable because the applicant has properly identified that the Chemistry Program will be used to manage SCC of carbon steel components in a treated water environment. However, the staff questioned whether a one-time inspection is used to verify the effectiveness of the Chemistry Program. The staff notes that the response to RAI 3.3.2.4.4-1, clarified that a one-time inspection will be conducted in low-flow areas of various closed, treated water systems to demonstrate the effectiveness of the Chemistry Program.

In its response dated September 2, 2003, the applicant stated that VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrites are added as a corrosion inhibiter. Nitrites do not cause SCC of carbon and low-alloy steel components;

however, nitrites can convert to nitrates in the presence of microorganisms. Nitrate levels in these systems are typically in the range of 300 ppm. According to EPRI guidelines, nitrate-induced SCC occurs at levels above 10,000 ppm. In these closed systems, there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrites are added as a corrosion inhibiter by the Chemistry Program at levels within EPRI guidelines; therefore, VCSNS maintains that the Chemistry Program adequately manages SCC of carbon steel components in a treated water environment. The applicant also stated that one-time inspections will be performed in low-flow areas prior to the period of extended operation to verify the effectiveness of the Chemistry Program to manage aging in the various chemistry regimes within the scope for license renewal.

On the basis of its review, the staff finds the applicant's response acceptable because that the applicant has committed to performing one-time inspections in low-flow areas prior to the period of extended operation to verify the effectiveness of the Chemistry Program to manage aging in the various chemistry regimes within the scope for license renewal.

The aging effects identified in the LRA for the components in the air handling and local ventilation and cooling systems are consistent with industry operating experience for the materials and environments listed. The staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified.

Aging Management Programs:

The applicant credited the following AMPs for managing the aging effects in the air handling and local ventilation and cooling systems:

- Chemistry Program (3.0.3.2)
- Boric Acid Corrosion Surveillances Program (3.0.3.1)
- Service Water System Reliability and In-Service Testing Program (3.3.2.3.1)
- Preventive Maintenance Activities — Ventilation Systems Inspections Program (3.3.2.3.3)
- Inspections for Mechanical Components Program (3.0.3.7)
- Maintenance Rule Structures Program (3.0.3.4)
- Heat Exchanger Inspections Program (3.0.3.8)

The Chemistry Program, Boric Acid Corrosion Surveillances Program, Inspections for Mechanical Components Program, Maintenance Rule Structures Program, and Heat Exchanger Inspections Program are credited with managing the aging effects of several components in different structures and systems and are, therefore, considered common AMPs. The staff has evaluated these common AMPs and has found them to be acceptable for managing the aging effects identified for this system. The staff's evaluation of these AMPs is documented in Sections 3.0.3.1, 3.0.3.2 , 3.0.3.4, 3.0.3.7, and 3.0.3.8 of this SER.

The staff evaluated the system-specific AMPs of Service Water System Reliability and In-Service Testing Program and Preventive Maintenance Activities — Ventilation Systems Inspections Program and finds them to be acceptable for managing the aging effects identified for this system. The staff's evaluation of these AMPs is documented in Sections 3.3.2.3.1 and 3.3.2.3.3 of this SER.

After evaluating the applicant's AMR for each of the components in the air handling and local ventilation and cooling systems, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects for this system. For those components identified in Table 3.3-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.3-2, the staff verified that the applicant credited AMPs that are appropriate for the identified aging effects.

On the basis of its review, the staff finds that the AMPs credited in the LRA for the components in the air handling and local ventilation and cooling systems will effectively manage or monitor the aging effects identified in the LRA.

Conclusions

On the basis of its review, the staff finds the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.2.4.2 Boron Recycle System

Summary of Technical Information in the Application

The description of the boron recycle system can be found in Section 2.3.3.2 of this SER. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-19. The components, aging effects, and AMPs are provided in LRA Tables 3.3-1 and 3.3-2.

Aging Effects:

Components of the boron recycle system are described in Section 2.3.3.2 of the LRA as being within the scope of license renewal and subject to an AMR. Tables 2.3-19, 3.3-1, and 3.3-2 of the LRA, and the supplementary table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query," list system components and component group.

The component groups in this category of the boron recycle system listed by the applicant in the VCSNS LRA include condensers (XEV0008-CN1, XEV0008-CN2) recycle evaporator-channel head; condensers (XEV0008-CN1, XEV0008-CN2) recycle evaporator-channel head (Nozzles); condensers (XEV0008-CN1, XEV0008-CN2) recycle evaporator-tubes; condensers (XEV0008-CN1, XEV0008-CN2) recycle evaporator-tubesheet; heat exchanger (XEV0008-HE2) recycle evaporator-shell; heat exchanger (XEV0008-HE2) recycle evaporator-shell (Nozzles); heat exchanger (XEV0008-HE2), recycle evaporator-tubes; heat exchanger (XEV0008-HE2) recycle evaporator-tubesheet; heat exchanger (XHE0021) recycle evaporate concentrates sample-manifolds; heat exchanger (XHE0021) recycle evaporate concentrates sample-shell; heat exchanger (XHE0021) recycle evaporate concentrates sample-tubes; and valves (body only).

Carbon steel and stainless steel components exposed to borated water, treated water, or sheltered environments are subject to the aging effects of loss of material from general (for carbon steel only), pitting, and crevice corrosion, boric acid corrosion, and galvanic corrosion

(for carbon steel only) and cracking from SCC (for stainless steel only). Stainless steel components exposed to a sheltered environment in the boron recycle system experience no aging effects. Carbon steel exposed to a shelted environment may experience loss of material due to general corrosion.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the boron recycle system:

- Chemistry Program (B.1.4)
- Boric Acid Corrosion Surveillances Program (B.1.2)
- Inspections for Mechanical Components (B.2.11)

A description of these AMPs is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the boron recycle system will be adequately managed by these AMPs during the period of extended operation.

Staff Evaluation

Aging Effects:

The staff reviewed the information in Section 2.3.3.2 and Tables 2.3-19, 3.3-1, and 3.3-2 in the LRA, as well as in the tables entitled, "Virgil C. Summer Nuclear Station Database AMR Query" and "Virgil C. Summer Nuclear Station Database AMR Query Notes," in the supplement and finds the applicant's identification of the applicable aging effects of carbon steel and stainless steel components acceptable. The applicant's conclusion that the stainless steel valves (body only) in the sheltered environment experience no aging effects is also acceptable.

Aging Management Programs:

The applicant credited the following AMPs for managing the aging effects in the boron recycle system:

- Chemistry Program (3.0.3.2)
- Boric Acid Corrosion Surveillances Program ( 3.0.3.1)
- Inspections for Mechanical Components (3.0.3.7)

The Chemistry Program and the Boric Acid Corrosion Surveillances Program are credited with managing the aging effects of several components in different structures and systems and are, therefore, considered common AMPs. The staff has evaluated these common AMPs and has found them to be acceptable for managing the aging effects identified for this system. The staff's evaluation of these AMPs is documented in Sections 3.0.3.1 and 3.0.3.2, respectively, of this SER.

After evaluating the applicant's AMR for each of the components in the boron recycle system, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects for this system. For those components identified in Table 3.3-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.3-2 , the staff verified that the applicant credited AMPs that are appropriate for the identified aging effects.

On the basis of its review, the staff finds that the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with the boron recycle system. In addition, the staff finds the associated program descriptions in the FSAR Supplement to be acceptable to satisfy 10 CFR 54.21(d).

Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.3.2.4.3 Building Services System

Summary of Technical Information in the Application

The description of the building services system can be found in Section 2.3.3.3 of this SER. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-20. The components, aging effects, and AMPs are provided in LRA Tables 3.3-1 and 3.3-2.

Aging Effects:

Components in the building services system are described in Section 2.3.3.3 of the LRA as being within the scope of license renewal and subject to an AMR. Tables 2.3-20, 3.3-1, and 3.3-2 of the LRA, and the supplementary table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query," list system components and component group.

The component groups in this category in the building services system listed by the applicant in the VCSNS LRA include pipe and fittings, tube and tube fittings, and valves (body only). The applicant stated that the stainless steel components exposed to air-gas, reactor building, ventilation, and sheltered environments in building services experience no aging effect. Carbon steel components in the air-gas and sheltered environments are subject to the aging effects of loss of material due to galvanic corrosion and general corrosion. Carbon steel components exposed to a sheltered environment are subject to the aging effect of loss of material due to boric acid corrosion.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the building services system:

- Boric Acid Corrosion Surveillances Program (B.1.2)
- Service Air System Inspection Program (B.2.6)
- Inspections for Mechanical Components Program (B.2.11)

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

On the basis of its review of the above information, the staff further requested the applicant to clarify whether any aging management activity is used to verify the absence of cracking and the effectiveness of the Chemistry Program and if so, what AMP is used. Otherwise, the applicant was requested to provide the justification for not verifying the effectiveness of the Chemistry Program.

In its response by letter dated September 2, 2003, the applicant stated that VCSNS has conservatively listed SCC as a possible aging mechanism in certain closed systems where nitrites are added as a corrosion inhibitor. Nitrites do not cause SCC of carbon and low-alloy steel components; however, nitrites can convert to nitrates in the presence of microorganisms. Nitrate levels in these systems are typically in the range of 300 ppm. According to EPRI guidelines, nitrate-induced SCC occurs at levels above 10,000 ppm. In these closed systems, there is no other pathway for the introduction of contaminants beyond the corrosion products of the system itself. Nitrites are added as a corrosion inhibitor by the Chemistry Program at levels within EPRI guidelines. In addition, the applicant stated that one-time inspections will be performed in low-flow areas prior to the period of extended operation to verify the effectiveness of the Chemistry Program to manage aging in the various chemistry regimes within the scope for license renewal.

On the basis of its review, the staff finds that the applicant's response dated June 12, 2003, as well as the applicant's supplemental response dated September 2, 2003, acceptable because the applicant has committed to perform one-time inspections in low-flow areas prior to the period of extended operation to verify the effectiveness of the Chemistry Program to manage aging in the various chemistry regimes within the scope for license renewal. All issues associated with RAI 3.3.2.4.4-3, are considered resolved.

On the basis of its review of the information provided in the LRA and the additional information included in the applicant's response to the above RAIs, the staff finds that the aging effects that result from contact of the chilled water system SSCs to the environments described in Tables 2.3-21, 3.3-1, and 3.3-2 are consistent with industry experience for these combinations of materials and environments. Therefore, the staff finds that the applicant has identified the appropriate aging effects for the materials and environments associated with the components in the chilled water system.

Aging Management Programs:

The applicant credited the following AMPs for managing the aging effects in the chilled water system:

- Chemistry Program Program (3.0.3.2)
- Inspections for Mechanical Components Program (3.0.3.7)
- Service Water System Reliability and In-Service Testing Program (3.3.2.3.1)
- Above Ground Tank Inspection Program (3.0.3.5)
- Heat Exchanger Inspection Program (3.3.0.8)
- Maintenance Rule Structures Program (3.0.3.4)

The Chemistry Program, Inspections for Mechanical Components Program, Above Ground Tank Inspection Program, Heat Exchanger Inspections Program, and Maintenance Rule Structures Program are credited with managing the aging effects of several components in different structures and systems and are, therefore, considered common AMPs. The staff has evaluated these common AMPs and found them to be acceptable for managing the aging effects identified for this system. The staff's evaluation of these AMPs is documented in Sections 3.0.3.2, 3.0.3.4, 3.0.3.5, 3.0.3.7, and 3.3.0.8, respectively, of this SER.

The staff has evaluated the system-specific Service Water System Reliability and In-Service Testing Program and finds it to be acceptable for managing the aging effects identified for this system. The staff's evaluation of this AMP is documented in Section 3.3.2.3.1 of this SER.

After evaluating the applicant's AMR for each of the components in the chilled water system, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects for this system. For those components identified in Table 3.3-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.3-2, the staff verified that the applicant credited AMPs that are appropriate for the identified aging effects.

On the basis of its review, the staff finds that the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with the chilled water system. In addition, the staff finds the associated program descriptions in the FSAR Supplement to be acceptable to satisfy 10 CFR 54.21(d).

Conclusions

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.3.2.4.5 Circulating Water System

As described in Section 2.3.3.5 of the LRA, the applicant's scoping and screening review concluded that there are no mechanical components/component types required for the circulating water system to perform its system intended function; therefore, no AMR is required. The staff's evaluation of the scoping and screening process is documented in Section 2.3.3.5 of this SER.

An internal environment of alternating wet-dry air-gas causes the aging effect of loss of material from corrosive impacts in carbon steel components.

An internal environment of raw water causes the aging effects of loss of material from crevice and pitting corrosion, MIC, and erosion, and heat exchanger fouling from biological materials and particulates in stainless steel and brass components.

An internal environment of treated water causes the aging effect of loss of material from crevice and pitting corrosion and cracking from SCC in carbon steel, stainless steel, and brass components. For carbon steel components, the internal environment of treated water causes the aging effect of loss of material from galvanic corrosion and general corrosion. An internal environment of treated water causes the aging effect of loss of material from erosion or erosion-corrosion for brass and stainless steel. The same internal environment of treated water causes the aging effect of heat exchanger fouling due to particulates for brass components. An internal environment of treated water also causes the aging effect of loss of material from crevice corrosion, erosion-corrosion, galvanic corrosion, and pitting corrosion in copper components.

An internal environment of fuel oil causes the aging effect of loss of material from MIC for copper, brass, and carbon steel components. For carbon steel components, an internal environment of fuel oil also causes the aging effect of loss of material due to crevice and pitting corrosion, galvanic corrosion, and general corrosion. No aging effect is identified for any components exposed to an internal environment of dry air-gas.

Loss of material is identified as aging effect from MIC, crevice and pitting corrosion, galvanic corrosion, and general corrosion for carbon steel components exposed to an underground environment. Loss of material is identified as an aging effect from general and galvanic corrosion for carbon steel and cast iron components exposed to a sheltered environment. No aging effect is identified for stainless steel, brass, aluminum, and copper components exposed to a sheltered environment. Loss of material is identified as an aging effect from general and galvanic corrosion for carbon steel components exposed to a yard environment.

Cracking is identified as an aging effect from radiation and thermal embrittlement for rubber components exposed to a sheltered environment.

No aging effect is identified for rubber components exposed to fuel oil or treated water environments.

No aging effect is identified for aluminum components in either an air-gas or a sheltered environment. No aging effect is identified for ductile iron components in an oil environment. Loss of material due to general corrosion is identified as an aging effect for ductile components in a sheltered environment.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the diesel generator service systems:

- 1~~2~~ Inspections for Mechanical Components Program (B.2.11)

- Diesel Generator Systems Inspection Program (B.2.2)
- Chemistry program Program (B.1.4)
- Service Water System Reliability and In-Service Test Program (B.1.9)
- Buried piping and Tanks Inspection Program (B.2.10)
- Heat Exchanger Inspections Program (B.2.12)

A description of the AMPs is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the diesel generator service systems will be adequately managed by the AMPs during the period of extended operation.

Staff Evaluation

Aging Effects:

The staff reviewed the information in Section 2.3.3.7 and Tables 2.3-23, 3.3-1, and 3.3-2 in the LRA, as well as in the tables entitled, "Virgil C. Summer Nuclear Station Database AMR Query" and "Virgil C. Summer Nuclear Station Database AMR Query Notes," in the supplement. During its review, the staff determined that additional information was needed.

Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1, pertaining to this issue of the plant specific characteristics of the environment. The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER and is characterized as resolved.

The applicant identified the flexible coupling in the diesel generator service systems as subject to AMR. The applicant stated that component/component type AMR results for VCSNS are consistent with NUREG-1801 in material and environment, and partially consistent in aging effects. The VCSNS plant-specific program credited for managing aging effects is B.2.11 Inspections for Mechanical Components Program. This AMP inspects component external surfaces for signs of degradation.

In the GALL Report, elastomer-based components in warm, moist air have the aging effects of hardening, cracking, and loss of strength due to elastomer degradation. The associated AMP is plant-specific. In the VCSNS LRA (Table 3.3-1), elastomer-based components in an air or gas (indoor) environment have the aging effects of hardening, cracking, and loss of strength due to elastomer degradation and the AMP credited is B.2.11, "Inspections for Mechanical Components Program."

For flexible hose and flexible couplings included in LRA Table 2.3-23, the applicant identified Table 3.3-1, Item 2, and Table 3.3-2, Item 26. LRA Table 3.3-1, Item 2 states that loss of material due to wear is not considered an aging effect because mechanical components must perform their license renewal intended functions without moving parts. Wear that occurs on nonmoving components is considered to be caused by improper design and should be corrected by normal maintenance activities. The staff disagrees with the applicant's explanation that wear is caused by improper design in the nonmoving components. The staff believes that wear of elastomer may be attributed to many conditions, such as relative movement due to thermal expansion. By letter dated March 28, 2003, the staff requested, in

for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.3.2.4.8 Fire Service System

Summary of Technical Information in the Application

The description of the fire service system can be found in Section 2.3.3.8 of this SER. The passive, long-lived components of this system that are subject to an AMR are identified in LRA Table 2.3-24. The components, aging effects, and AMPs are provided in LRA Tables 3.3-1 and 3.3-2. FSAR Section 9.5.1, "Fire Protection System," provides additional information concerning the interior/exterior fire protection system.

Aging Effects:

LRA Table 2.3-24 lists individual system components that are within the scope of license renewal and subject to an AMR. These components include bolting, piping, tubing, fittings, valves, nozzles, fire hydrant and pump casings, components in the water-based fire suppression system, components in the $CO_2$ fire suppression system, components in the diesel fire system, doors, barrier penetration seals, and concrete structures (fire barrier walls, ceilings, and floors).

The LRA identified that carbon steel, galvanized steel, cast iron, and copper in air are subject to loss of material due to general external corrosion, and carbon steel and low-alloy steel in dripping boric acid are subject to loss of material due to boric acid corrosion. The LRA also identifies that stainless steel in treated water is subject to loss of material due to crevice and pitting corrosion and cracking due to SCC. The LRA identifies that components in water-based fire suppression systems are subject to loss of material due to general pitting, crevice, and galvanic corrosion, MIC, and biofouling. Fire barriers, walls, ceilings, floors, doors, and penetration seals are subject to loss of material due ~~to water~~, hardening, and shrinkage caused by ~~to~~ weathering, concrete cracking, and spalling from ~~to~~ freeze thaw, aggressive chemical attack, and reaction with aggregates, and loss of material due to corrosion of embedded steel. Stainless steel in oil (reactor coolant pump oil collection system) is subject to loss of material due to galvanic, general, pitting, and crevice corrosion. Buried piping and fittings are subject to loss of material due to general, pitting, and crevice corrosion and MIC.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the fire service system:

- Boric Acid Corrosion Surveillances Program (B.1.2)
- Fire Protection Program (B.1.5)
- ~~Selective Leaching of Materials Program (B.1.5)~~
- ~~Flow-Accelerated Corrosion Monitoring Program (B.1.6)~~
- Structures Monitoring Program (B.1.18)
- Buried Piping and Tanks Inspection Program (B.2.10)
- Inspections for Mechanical Components (B.2.11)

A description of these AMPs is provided in LRA Appendix B. The applicant indicated that the effects of aging associated with the components of the fire protection system will be adequately managed by these AMPs during the period of extended operation.

Staff Evaluation

The staff reviewed the applicant's fire protection system in the LRA to determine whether the applicant had demonstrated that the effects of aging for the fire protection system will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff's review was conducted in accordance with Section 3.3 of the SRP-LR (NUREG-1800) and is described below.

Aging Effects:

The staff reviewed the information in LRA Tables 2.3.24, 3.3-1, and 3.3-2 for the fire protection system. During its review, the staff determined that additional information was needed. By letter dated March 28, 2003, in RAI 3.3.2.4.8-1(1), pertaining to the one-time inspection of the components in the reactor coolant pump oil collection system, the staff questioned why these components should not be inspected periodically for managing aging.

In its response dated June 12, 2003, the applicant stated that NUREG-1801 recommends a one-time inspection for the components of the reactor coolant pump oil collection system that are composed of carbon steel, copper, and brass. The reactor coolant pump oil collection system components at the plant are composed of stainless steel. The staff finds this response reasonable and acceptable since none of the component types of the reactor coolant pump oil collection system will collect water in low spots, all are subject to high ambient condition which would cause evaporation of any moisture minimizing corrosion.

In RAI 3.3.2.4.8-1(2), the staff asked why Item 18 of LRA Table 3.3-2 does not identify any aging effects or mechanism to be evaluated for the fire service system nozzles, piping, and fire hydrants.

In its response dated June 12, 2003, the applicant stated that LRA Table 3.3-2 concerns auxiliary system components identified in NUREG-1801. Item 18 of this table addresses components that are normally in a standby mode where air is the predominant internal environment. The plant's external environments for these components are addressed in LRA Table 3.3.1, Items 5 and 20. The staff reviewed Item 20 of LRA Table 3.3-1 which addresses the AMPs for components in the water-based fire protection system. Therefore, the staff finds the applicant's response acceptable.

On the basis of its review of the information provided in the LRA and the additional information in the applicant's response to the staff's RAIs, the staff finds that the aging effects identified for the fire protection system components described in LRA Tables 2.3.24, 3.3-1, and 3.3-2 are consistent with industry experience for these combinations of materials and environments. Therefore, the staff finds that the applicant has identified the appropriate aging effects for the materials and environments associated with the components in the fire protection system

Aging Management Programs:

The applicant credited the following AMPs for managing the aging effects in the fire service systems:

- Boric Acid Corrosion Surveillances Program (B.1.2)
- Fire Protection Program (B.1.5)
- ~~Selective Leaching of Materials Program (B.1.5)~~
- ~~Flow-Accelerated Corrosion Monitoring Program (B.1.6)~~
- Structures Monitoring Program (B.1.18)
- Buried Piping and Tanks Inspections Program (B.2.10)
- Inspections for Mechanical Components (B.2.11)

These AMPs are credited for managing the aging effects of components in several structures and systems that are considered as common AMPs. The staff has evaluated these common AMPs and found them to be acceptable for managing the effects of the components in several structures and systems and, therefore, are considered common AMPs.

On the basis of its review of the information provided in the LRA, the staff concluded that the above identified AMPs will effectively manage the aging effects of the fire protection system.

On the basis of its review, the staff finds that the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with fire protection system.

Conclusion

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.3.2.4.9 Fuel Handling System

Summary of Technical Information in the Application

The description of the fuel handling system can be found in Section 2.3.3.9 of this SER. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-25. The components, aging effects, and AMPs are provided in LRA Table 3.3-2.

Aging Effects:

Components of the fuel handling system are described in Section 2.3.3.9 of the LRA as being within the scope of license renewal and subject to an AMR. Tables 2.3-25 and 3.3-2 of the LRA, and the table entitled, "Virgil C. Summer Nuclear Station Database AMR Query," in the supplement lists the system component which consists of fuel and transfer tubes.

The applicant identified no aging effects for the stainless steel and carbon steel components that are embedded in concrete. In addition, the applicant identified no aging effects for the stainless steel exposed to a sheltered or ventilation environment.

Aging Management Programs:

The applicant identified no aging effects for the components of the fuel handling system. Therefore, no AMPs are required for this system.

Staff Evaluation

Aging Effects:

The staff reviewed the information in Section 2.3.3.9 and Tables 2.3-25, 3.3-1, and 3.3-2 in the LRA, as well as in the tables entitled, "Virgil C. Summer Nuclear Station Database AMR Query," and "Virgil C. Summer Nuclear Station Database AMR Query Notes," in the supplement. During its review, the staff determined that additional information was needed.

Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1, pertaining to this issue of the plant-specific characteristics of the environment. The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER and is characterized as resolved.

On the basis of its review of the information provided in the LRA, the staff finds that the absence of aging effects that result from contact of the fuel handling system SSCs to the environments described in Tables 2.3-23, 3.3-1, and 3.3-2 is consistent with industry experience for these combinations of materials and environments and is, therefore, acceptable.

Aging Management Programs:

Based on the review of the information provided in the LRA, the staff concurs with the applicant's conclusion that no AMPs are required for the fuel handling system because there are no applicable aging effects for the components of this system.

Conclusions

On the basis of its review, the staff concludes that the applicant has justified that no AMPs are required because there are no applicable aging effects for components in the fuel handling system. In addition, there is reasonable assurance that the component intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.3.2.4.10 Gaseous Waste Processing System

Summary of Technical Information in the Application

In its response dated June 12, 2003, the applicant stated that all of the columns of Table 3.3-1 of the LRA, except for the last column, are NUREG-1801 listings. The last column is the VCSNS response. NUREG-1801 states that, for this AMR item, if there were an adequate "closed-cycle cooling water system" program, then no further evaluation is recommended. In the "Discussion" column, VCSNS discusses the adequacy of the Chemistry Program in managing aging instead of using a closed-cycle cooling water system program.

The applicant further stated that at VCSNS, cracking due to SCC is an aging effect for stainless steel components in treated water environments (i.e., heat exchangers cooled by the component cooling water system). The Chemistry Program has proven effective at managing aging degradation in the component cooling water System as evidenced by the review of operating history in response to GL 89-13. Finally, the applicant stated that prior to the period of extended operation, one-time inspections will be conducted in low-flow areas of various treated water systems to demonstrate the effectiveness of the Chemistry Program.

On the basis of its review, the staff finds the applicant's response acceptable because the applicant has properly identified the basis of the difference between GALL and the LRA. In addition, prior to the period of extended operation, the applicant has committed to conduct one-time inspections in low-flow areas of various treated water systems to demonstrate the effectiveness of the Chemistry Program.

On the basis of its review of the information provided in the LRA and the additional information included in the applicant's response to the above RAIs, the staff finds that the aging effects that result from contact of the liquid waste processing system SSCs to the environments described in LRA Tables 2.3-9, 3.3-1, and 3.3-2 are consistent with industry experience for these combinations of materials and environments. Therefore, the staff finds that the applicant has identified the appropriate aging effects for the materials and environments associated with the components in the liquid waste processing system.

Aging Management Programs:

The applicant credited the following AMPs for managing the aging effects in the liquid waste processing system:

- Boric Acid Corrosion Surveillances Program (3.0.3.1)
- Chemistry Program (3.0.3.2)
- Maintenance Rule Structures Program (3.0.3.4)
- Liquid Waste System Inspection Program (3.3.2.3.14)
- Inspections for Mechanical Components (3.0.3.7)

The Boric Acid Corrosion Surveillances Program, the Chemistry Program, and Maintenance Rule Structures Program are credited with managing the aging effects of several components in different structures and systems and are, therefore, considered common AMPs. The staff has evaluated these common AMPs and has found them to be acceptable for managing the aging effects identified for this system. The staff's evaluation of these AMPs is documented in Sections 3.0.3.1, 3.0.3.2, and 3.0.3.4, respectively, of this SER.

The staff has evaluated the system-specific Liquid Waste System Inspection AMP and has found it to be acceptable for managing the aging effects identified for this system. The staff's evaluation of this AMP is documented in Section 3.3.2.3.14 of this SER.

After evaluating the applicant's AMR for each of the components in the liquid waste processing system, the staff evaluated the AMPs listed above to determine if they are appropriate for managing the identified aging effects for this system. For those components identified in Table 3.3-1 of the LRA, the staff verified that the applicant credited the AMPs recommended by the GALL Report. For the components identified in LRA Table 3.3-2 , the staff verified that the applicant credited AMPs that are appropriate for the identified aging effects.

On the basis of its review, the staff finds that the applicant has credited the appropriate AMPs to manage the aging effects for the materials and environments associated with liquid waste processing system. In addition, the staff finds the associated program descriptions in the FSAR Supplement to be acceptable to satisfy 10 CFR 54.21(d).

Conclusions:

On the basis of its review and audit of the applicant's program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. Since the GALL program is acceptable to the staff, the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.3.2.4.15 Nuclear and Nonnuclear Plant Drains System

Summary of Technical Information in the Application

The description of the nuclear and nonnuclear plant drains can be found in Section 2.3.3.15 of this SER. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-29. The components, aging effects, and aging management programs are provided in the LRA Table 3.3-2.

Aging Effects:

The components in this group category described in Section 2.3.3.15 of the LRA are identified as being within the scope of license renewal and subject to an AMR. Tables 2.3-29 and 3.3-2 of the LRA, and the table entitled, "Virgil C. Summer Nuclear Station Database AMR Query," in the LRA supplement documents list individual components of the system.

The component groups of nuclear plant drains listed by the applicant in the VCSNS LRA include valves (body only) and pipe and fittings. Stainless steel components exposed to borated water are identified as subject to loss of material due to crevice corrosion, pitting corrosion, and cracking from SCC. Stainless steel components exposed to sheltered, reactor building, ventilation, and embedded in concrete environments are identified as having no aging effects.

The applicant identified the intended function of the nonnuclear plant drains system to be providing the circulating water pump trip function to prevent flooding in the control and

Aging Effects:

Components of the nuclear sampling system are described in Section 2.3.3.16 of the LRA as being within the scope of license renewal and subject to AMR. Table 2.3-30, 3.3-1, and 3.3-2 of the LRA, and the supplementary table and notes entitled, "Virgil C. Summer Nuclear Station Database AMR Query," list individual components of the system including heat exchanger (shell and tubes), pipe, pumps (casing only), tanks, tube and tube fittings, and valves (bodies only).

Loss of material is identified as an aging effect due to pitting, crevice, galvanic, and general corrosion for carbon steel components exposed to an internal environment of treated water. Loss of material from pitting and crevice corrosion, corrosion impact from alternate wetting and drying (for stainless steel), and cracking from SCC are identified as aging effects for stainless steel and nickel-based alloy components exposed to an internal environment of treated water. Loss of material is identified as an aging effect from pitting and crevice corrosion and cracking from SCC (for nickel-based alloy) for stainless steel and nickel-based alloy components exposed to an internal environment of borated water.

Loss of material is identified as an aging effect from general corrosion (for carbon steel) and MIC (for stainless steel) for carbon steel and stainless steel components exposed to an external ~~reactor building or~~ sheltered environment. No aging effect is identified for stainless steel components exposed to an external ventilation environment.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the nuclear sampling system:

- Inspections for Mechanical Components Program (B.2.11)
- Chemistry Program (B.1.4)
- Above Ground Tank Inspection Program (B.2.1)
- Maintenance Rule Structures Program (B.1.18)

A description of the AMP is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the nuclear sampling system will be adequately managed by the identified AMPs during the period of extended operation.

<u>Staff Evaluation</u>

Aging Effect:

The staff reviewed the information in Section 2.3.3.16 and Tables 2.3-30, 3.3-1, and 3.3-2 in the LRA, as well as in the supplementary table and notes entitled, "Virgil C. Summer Nuclear Station Database AMR Query." During its review, the staff determined that additional information was needed.

Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1, pertaining to this issue of the plant-specific characteristics of the environment. The staff's

evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER and is characterized as resolved.

For carbon steel components exposed to external environments of moist air, such as reactor building or sheltered, the GALL Report identifies that loss of material is an aging effect that is caused by general, pitting, and crevice corrosion and MIC. The VCSNS LRA identifies loss of material as an aging effect due to general corrosion only. By letter dated March 28, 2003, the staff requested, in RAI 3.3.2.4.16-1, the applicant to justify why pitting or crevice corrosion or MIC does not occur for the carbon steel components exposed to external environments of moist air, such as reactor building or sheltered. If an insignificant concentration of contaminants is part of the technical basis, the staff also requested the applicant to provide the acceptance criterion used and the verification/inspection activities performed to justify its conclusion.

In its response dated June 12, 2003, the applicant stated that plant operating experience has identified the accumulation of microbiological organisms on the external surfaces of some piping components at building wall penetrations as a result of ground water intrusion effects. The structural design of the plant is such that any ground water intrusion in the sheltered environment is directed to sumps and away from equipment within the scope of license renewal. It is the residual presence of microbiological organisms that is of concern for subject mechanical components.

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

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

The applicant further stated that building penetrations are inspected as part of the Maintenance Rule Structures Program (Application Section B.1.18). The VCSNS Corrective Action Program would disposition any ground water in-leakage and resulting degradation.

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

The staff has evaluated this AMP and has found it to be acceptable for managing the aging effects identified for this system. The staff's evaluation of this AMP is documented in Section 3.3.2.3.6 of the SER.

After evaluating the applicant's AMR for each of the components in the roof drains system , the staff evaluated the AMP listed above to determine if it is appropriate for managing the identified aging effects for this system. For the components identified in LRA Table 3.3-2, the staff verified that the applicant credited the aging management program that is appropriate for the identified aging effects.

In the table entitled "Virgil C. Summer Nuclear Station Database AMR Query", the applicant stated that stainless steel piping and fitting component of the roof drains system is subjected to the aging effect of cracking from stress corrosion cracking in a borated water environment. LRA Table 3.3-2 Item 22 identifies that the stainless steel drain lines are less than 140°F and are not susceptible to SCC but are susceptible to crevice or pitting corrosion. AMP B.2.5 is actually credited with managing SCC in addition to crevice and pitting corrosion which is acceptable to the staff. 140°F

On the basis of its review, the staff finds the applicant has credited the appropriate AMP to manage the aging effects for the materials and environments associated with the roof drains system . In addition, the staff finds the associated program description in the FSAR Supplement to be acceptable to satisfy 10 CFR 54.21(d).

Conclusions

On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects, and AMPs credited for managing the aging effects, for components in the roof drains system, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

The staff also reviewed the applicable FSAR Supplement program descriptions and concludes that the FSAR Supplement provides an adequate program description of the AMP credited for managing aging in the roof drains system as required by 10 CFR 54.21(d).

3.3.2.4.20 Station Service Air System

Summary of Technical Information in the Application

The description of the station service air system can be found in Section 2.3.3.20 of this SER. The passive, long-lived components in this system that are subject to an AMR are identified in LRA Table 2.3-34. The components, aging effects, and aging management programs are provided in LRA Tables 3.3-1 and 3.3-2.

Aging Effects:

Components and component group of the station service air system are described in Section 2.3.3.20 of the submittal as being within the scope of license renewal and subject to an AMR. Tables 2.3-34, 3.3-1, and 3.3-2 of the LRA and the table entitled Virgil C. Summer Nuclear

Station Database AMR Query in the supplement list system components and components group.

The component groups in this category in the station service air system listed by the applicant in the VCSNS LRA include pipe and fittings, tube and tube fittings, and valves (body only). The applicant states that carbon steel components exposed to air-gas, reactor building, and sheltered environments are subject to aging effect of loss of material due to general corrosion. Stainless steel components of this system experience no aging effects while in air-gas, reactor building, and sheltered environments.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the station service air system :

- Boric Acid Corrosion Surveillances (B.1.2)
- Service Air System Inspection (B.2.6)
- Inspections for Mechanical Components (B.2.11)

A description of these aging management programs is provided in Appendix B of the LRA. The applicant states that the effect of aging associated with the components of the station service air system will be adequately managed by these aging management programs during the period of extended operation.

Staff Evaluation

Aging Effects:

The staff reviewed the information in Section 2.3.3.20 and Tables 2.3-34, 3.3-1, and 3.3-2 in the LRA, as well as in the tables entitled Virgil C. Summer Nuclear Station Database AMR Query and Virgil C. Summer Nuclear Station Database AMR Query Notes in the supplement. During its review, the staff found that additional information was needed to complete its review.

Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1, pertaining to this issue of the plant- specific characteristics of the environment . The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER, and is characterized as resolved.

By letter dated March 28, 2003, the staff issued RAI 3.3-3, pertaining to this issue of the susceptibility to aging effects for stainless steel components in ambient environment. The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.3 of this SER and is characterized as resolved.

Normally station service air system may contain elastomer materials in hose connection seals, duct seals, flexible collars between ducts and fans, rubber boots, etc. For some plant designs, elastomer components are used as vibration isolators to prevent transmission of vibration and dynamic loading to the rest of the system. The aging effects on those elastomer components are hardening and loss of material. However, no elastomer component associated with the

Loss of material is identified as aging effect for carbon steel and stainless steel components exposed to internal environment of treated water. Loss of material and cracking are identified as aging effects for stainless steel refueling water storage tank (RWST) exposed to internal environment of borated water due to alternate wet and dry at borated water surface. Loss of material due to crevice and pitting corrosion is identified as aging effects for stainless steel components exposed to internal environment of borated water other than RWST. No aging effect is identified for stainless steel components exposed to internal ventilation (i.e., moisture air) environment. Cracking is not identified as an aging effect for components exposed to borated water or treated water because the system is normally operated well below 140° F.

Loss of material due to boric acid corrosion is identified as aging effect for carbon steel ~~and stainless steel~~ components exposed to sheltered environment. Loss of material due to micro biologically influenced corrosion is identified as an aging effect for vulnerable stainless steel components including pipe and tubing exposed to sheltered environment. Loss of material due to micro biologically influenced corrosion is not identified as aging effect for stainless steel components other than pipe and tubing exposed to sheltered environment. No aging effect is identified for stainless steel components exposed to yard environment.

Aging Management Programs:

The following AMPs are utilized to manage aging effects in the spent fuel cooling system:

- Chemistry program (B.1.4)
- Boric Acid Corrosion Surveillance (B.1.4)
- Maintenance Rule Structures Program (B.1.18)
- Above Ground Tank Inspection (B.2.1)

A description of the aging management program is provided in Appendix B of the LRA. The applicant states that the effect of aging associated with the components of the spent fuel cooling system will be adequately managed by the aging management program during the period of extended operation.

Staff Evaluation

Aging Effect:

The staff reviewed the information in Section 2.3.3.22 and Tables 2.3-36, 3.3-1 and 3.3-2 in the LRA; as well as in the supplementary table and notes, entitled "Virgil C. Summer Nuclear Station Database AMR Query." The applicant has stated that cracking is not identified as an aging effect for components exposed to borated water or treated water because the system is normally operated well below 140° F. The staff agrees that cracking is not an aging effect for spent fuel cooling system components exposed to borated water or treated water because temperature of the borated water or treated water is below 140°F. Below 140°F, stress corrosion cracking is not an aging effect requiring aging management.
However, during its review, the staff determined that additional information was needed to complete its review.
Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. By letter dated March 28, 2003, the staff issued RAI 3.3-1,

pertaining to this issue of the plant- specific characteristics of the environment . The staff's evaluation of the applicant's response is documented in Section 3.3.2.5.1 of this SER, and is characterized as resolved.

On page 211 of the VCSNS Database AMR Query Notes, the applicant states that loss of material due to MIC is identified as an aging effect for vulnerable stainless steel components including pipe and tubing of the spent fuel cooling system exposed to sheltered environment. However, loss of material due to MIC is not identified by the applicant as an aging effect for stainless steel components other than pipe and tubing. By letter dated March 28, 2003, the staff requested , in RAI 3.3.2.4.22-1, to provide justification as to why loss of material due to MIC is identified as an aging effect only for stainless steel pipe and tubing components and not for other stainless steel components such as heat exchangers, orifices, pumps, and valves.

In its response dated June 13, 2003, the applicant stated that the susceptibility to external MIC is limited locally to the area of the interface with the pertinent wall where groundwater in-leakage can occur. Only piping, process tubing, and ductwork component types pass through building penetrations. Finally the applicant stated that for the stainless steel components of the spent fuel cooling system, only pipe and pipe fitting components meet these criteria.

On the basis of its review, the staff finds that the applicant's response acceptable because that the applicant has properly identified that the susceptibility to external MIC is limited locally to the area of the interface with the pertinent wall where groundwater in-leakage can occur and that for the stainless steel components of the boron thermal regeneration system, only pipe and pipe fitting components meet these criteria. However, the staff questioned whether there are other types of water (such as water from condensation) other than ground water from intrusion present in the sheltered environment such that loss of material from MIC may become an applicable aging effect for the external surfaces of some of the applicable components of this system. The applicant was requested to provide the justification for not considering MIC from other types of water, including operating experience.

In its response dated September 2, 2003, the applicant stated that the ambient environment does not contain nutrients necessary to promote external MIC in other types of water, such as water from condensation and that because external MIC has not been found at locations other than at building penetrations, VCSNS does not specifically credit the Inspections for Mechanical Components for aging management for this aging effect; however, the applicant further stated that the Inspections for Mechanical Components will inspect for any abnormalities on external surfaces.

On the basis of its review, the staff finds that the applicant's response acceptable because the applicant has properly identified that the ambient environment does not contain nutrients necessary to promote external MIC in other types of water, such as water from condensation and that the applicant has committed to use the Inspections for Mechanical Components program to inspect for any abnormalities on external surfaces. All issues associated with this RAI 3.3.2.4.22-1, are considered resolved.

On the basis of its review of the information provided in the LRA and the additional information included in the applicant's response to the above RAIs, the staff finds that the aging effects that result from contact of the spent fuel cooling system SSCs to the environments described in Tables 2.3-36, 3.3-1 and 3.3-2 are consistent with industry experience for these combinations

| Table 3.4-1: Staff Evaluation for VCSNS Steam and Power Conversion Systems Components Evaluated in the GALL Report | | | | |
|---|---|---|---|---|
| **Component Group** | **Aging Effect/Mechanism** | **AMP in GALL Report** | **AMP in LRA** | **Staff Evaluation** |
| (11) External surface of above ground condensate storage tank | Loss of material due to general (carbon steel only), pitting, and crevice corrosion | Above-Ground Carbon Steel Tanks | ~~Not applicable~~ Inspection for mechanical components | See staff evaluation in Section 3.4.2.4.2. |
| (12) External surface of buried condensate storage tank and AFW piping | Loss of material due to general, pitting, and crevice corrosion and MIC | Buried Piping and Tanks Surveillance or Buried Piping and Tanks Inspection | Buried Piping and Tanks Inspection Program | GALL recommends further evaluation (See staff evaluation in Section 3.4.2.2.5). |
| (13) External surface of carbon steel components | Loss of material due to boric acid corrosion | Boric Acid Corrosion | Boric Acid Corrosion Surveillances Program | Consistent with GALL (See staff evaluation in Section 3.4.2.1). |

### *3.4.2.1 Aging Management Evaluations in the GALL Report That Are Relied On For License Renewal, Which Do Not Require Further Evaluation*

For component groups evaluated in the GALL Report for which the applicant has claimed consistency with GALL, and for which the GALL Report does not recommend further evaluation, the staff sampled components in these groups to determine whether the plant-specific components contained in these GALL Report component groups were bounded by the GALL evaluation. The staff also sampled component groups to determine whether the applicant had properly identified those component groups in the GALL Report that were not applicable to its plant. The staff also identified three areas where additional information or clarification was needed. The staff's evaluation of the applicant's responses to those RAIs is included in Sections 3.4.2.4.2 (RAI 3.4-13), 3.4.2.4.3 (RAI 3.4-12), and 3.4.2.4.13 (RAI 3.4-10) of this SER.

Table 3.4-1 of this SER contains a summary of the AMPs for SPC systems evaluated in Chapter VIII of the GALL Report. The GALL Report identifies specific component, material, environment, and aging effect/mechanism combinations that are managed by the GALL Report AMPs; therefore, VCSNS AMPs that are consistent with the GALL Report are only applicable to these specific material, environment, and aging effect/mechanism combinations. In addition to those component, material, environment, and aging effect/mechanism identified in the GALL Report, the applicant identified the following materials and aging mechanisms as being managed by the VCSNS AMPs that are consistent with the GALL Report AMPs:

- In Table 3.4-1, Item 6 of the SER, the applicant identified low-alloy steel components as being managed for wall thinning by the Flow-Accelerated Corrosion Program.

- In Table 3.4-1, Item 7 of the SER, the applicant identified the aging mechanisms of general corrosion and galvanic corrosion as being managed for loss of material by the Chemistry Program.

- In Table 3.4-1, Item 13 of the SER, the applicant identified cast iron as being managed for loss of material by the Boric Acid Corrosion Surveillances Program.

The staff finds the materials and aging mechanisms identified above as being adequately managed by GALL Report AMPs; therefore the staff finds that the applicant's aging management of these materials and aging mechanisms is acceptable.

On the basis of its review, the staff has verified the applicant's claim of consistency with the GALL report. The staff finds that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

*3.4.2.2 Aging Management Evaluations in the GALL Report That Are Relied On For License Renewal, For Which GALL Recommends Further Evaluation*

For component groups evaluated in the GALL Report for which the applicant has claimed consistency with GALL, and for which the GALL Report recommends further evaluation, the staff reviewed the applicant's evaluation to determine whether it adequately addressed the issues for which the GALL Report recommended further evaluation. In addition, the staff sampled components in these groups during the review to determine whether the plant-specific components contained in these GALL Report component groups were bounded by the GALL evaluation.

The GALL Report indicates that further evaluation should be performed for the aging effects discussed in the following sections:

3.4.2.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviewed the evaluation of this TLAA in Section 4.3 of this SER, following the guidance in Section 4.3 of the SRP-LR. The staff issued RAI 3.4-2 and RAI 3.4-3, to clarify aging management of SPC systems components for fatigue.

In Tables 2.3-38 through 2.3-47 of the LRA, the applicant does not identify any SPC systems components that are managed for cumulative fatigue. The GALL Report recommends aging management of cumulative fatigue for piping and fittings in the main steam, feedwater, and auxiliary feedwater systems. The staff issued RAI 3.4-2, requesting the applicant to explain why Tables 2.3-38 thru 2.3-47 do not identify any SPC systems components that are managed for cumulative fatigue.

In its response by letter dated June 12, 2003, the applicant stated that cumulative fatigue is considered to be a TLAA. It is discussed in Section 4 of the LRA entitled, "Time-Limiting Aging Analysis." The staff finds the applicant's response to RAI 3.4-2, reasonable and acceptable because it explains that fatigue for SPC systems is discussed in Section 4 of the LRA.

- The VCSNS Above Ground Tank Inspections program will perform inspections of the condensate storage tank interior to verify the effectiveness of the Chemistry Program and to manage the corrosive effects of alternate wetting and drying.
- In addition to the aging mechanisms identified in the GALL Report, VCSNS credits the Chemistry Program for managing galvanic corrosion, SCC. ~~and the corrosive effects of alternate wetting and drying.~~
- In addition to the materials identified in the GALL Report, VCSNS credits the Chemistry Program for managing aging effects for low-alloy steel and nickel based metal.
- In addition to the components identified in the GALL Report, VCSNS includes similar components from the nuclear sampling system in a treated water environment.

The staff finds the components, materials, and aging mechanisms identified above as being adequately managed by the VCSNS Chemistry Program; therefore the staff finds that the applicant's aging management of these components, materials, and aging mechanisms is acceptable.

On the basis of its review, the staff finds that the applicant appropriately evaluated AMR results involving management of the loss of material due to general, pitting, and crevice corrosion for components in the SPC systems, as recommended in the GALL Report. Since the applicant's AMR results are otherwise consistent with the GALL report, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.4.2.2.3 Loss of Material Due to General, Pitting, and Crevice Corrosion, Microbiologically Influenced Corrosion, and Biofouling

The SRP-LR recommends further evaluation of programs to manage loss of material due to general corrosion, pitting, and crevice corrosion, MIC, and biofouling for carbon steel piping and fittings for untreated water from the backup water supply in the auxiliary feedwater system. The staff reviewed the applicant's proposed program to ensure that an adequate program will be in place for the management of this aging effect.

Loss of material due to general corrosion, pitting and crevice corrosion, MIC, and biofouling could occur in carbon steel piping and fittings for untreated water from the backup water supply in the auxiliary feedwater system. In Table 3.4-1, Item 3 of the LRA, the applicant does not identify aging management of raw water exposure to AFW piping. In the "discussion" column, the LRA states that:

> AFW piping at VCSNS is not exposed to untreated water. The service water system provides emergency backup to the emergency feedwater system through automatic isolation valves that normally provide boundary isolation between the treated water of the emergency feedwater system and the untreated water of the service water system.

The staff issued RAI 3.4-6, requesting verification that the AFW piping has not been exposed to raw water. If any portion of the auxiliary feedwater system requires aging management due to

exposure to raw water, the applicant was requested to list the components and describe how aging will be managed.

In its response by letter dated June 12, 2003, the applicant stated that although there are automatic isolation valves that isolate the service water system from the emergency feedwater system, there is a section of emergency feedwater system piping (carbon steel) downstream of these automatic isolation valves that is filled from the service water system. Therefore, a portion of the emergency feedwater system is indeed exposed to a raw water environment. This piping is inspected under the activities described by the Service Water System Reliability and In-Service Testing Program and will continue to effectively manage the aging effects for this section of piping for the period of extended operation.

The staff finds the applicant's response to RAI 3.4-6, reasonable and acceptable because it provides an explanation that a section of the emergency feedwater system piping exposed to a raw water environment is managed for aging effects by the Service Water System Reliability and In-Service Testing Program.

On the basis of its review, the staff finds that the applicant appropriately evaluated the AMR results involving management of the loss of material due to general corrosion, pitting and crevice corrosion, MIC, and biofouling for auxiliary feedwater system components exposed to a raw water environment, as recommended in the GALL Report. Since the applicant's AMR results are otherwise consistent with the GALL report, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.4.2.2.4 Loss of Material Due to General Corrosion

The GALL Report recommends further evaluation of programs to manage loss of material due to general corrosion for external surfaces of all carbon SCs, including closure bolting, exposed to operating temperatures less than 212 °F. Such corrosion may be due to air, moisture, or humidity. The applicant credits the Inspections of Mechanical Components Program to manage corrosion in ambient, moist air for loss of material due to general corrosion and galvanic corrosion. The applicant credits the Maintenance Rule Structures Program to manage loss of material due to MIC on external surfaces in contact with ground water. The staff reviewed the applicant's Inspections of Mechanical Components Program in Section 3.0.3.7 of this SER and the Maintenance Rule Structures Program in Section 3.0.3.4 of this SER to ensure that these programs adequately manage this aging effect.

In addition to carbon steel components identified in Table 3.4-1 of the LRA, the applicant included low-alloy steel and cast iron components to be managed for loss of material on external surfaces by the Inspections of Mechanical Components Program and the Maintenance Rule Structures Program. The staff considers it acceptable for the applicant to credit these programs to manage low-alloy steel and cast iron components for loss of material on external surfaces.

On the basis of its review, the staff finds that the applicant appropriately evaluated AMR results involving management of the loss of material due to general corrosion for components in the SPC systems, as recommended in the GALL Report. Since the applicant's AMR results are

- loss of material due to general, pitting, and crevice corrosion and MIC of carbon steel components in soil and ground water environments

Aging Management Programs:

The following AMPs are utilized to manage aging effects to the auxiliary boiler steam and feedwater system:

- Chemistry Program
- Inspections for Mechanical Components Program
- Maintenance Rule Structures Program
- Boric Acid Corrosion Surveillances Program
- ~~Buried Pipe and Tanks Inspection Program~~

A description of these AMPs is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the auxiliary boiler steam and feedwater system will be adequately managed by these AMPs such that there is reasonable assurance that the intended functions will be maintained consistent with the CLB during the period of extended operation.

Staff Evaluation

In addition to Section 3.4 of the LRA, the staff reviewed the pertinent information provided in Section 2.3.4, "Steam and Power Conversion Systems," and the applicable AMP descriptions provided in Appendix B of the LRA to determine whether the aging effects for the auxiliary boiler steam and feedwater system components have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of the auxiliary boiler steam and feedwater system components at VCSNS. The staff's evaluation includes a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the auxiliary boiler steam and feedwater system components.

Aging Effects:

The component groups identified in LRA Table 2.3.38 for the auxiliary boiler steam and feedwater system are pipe and valves. The staff reviewed the aging effects identified in LRA Tables 3.4-1 and 3.4-2 for these component groups and finds the applicant properly identified the aging effects for these component groups. The aging effects are listed in SER Section 3.4.2.4.1.

The aging effects identified in the LRA for the auxiliary boiler steam and feedwater system are consistent with industry operating experience for the materials and environments listed. The staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified.

* THESE PARAGRAPHS APPLY TO SECTION 3.4.2.4.3 "EMERGENCY FEEDWATER SYSTEM"

Aging Management Programs:

The following AMPs are utilized to manage aging effects to the auxiliary boiler steam and feedwater system.

- Chemistry Program
- Inspections for Mechanical Components Program
- Maintenance Rule Structures Program
- Boric Acid Corrosion Surveillances Program
- ~~Buried Pipe and Tanks Inspection Program~~

Each of the above AMPs is credited with managing the aging of several components in different structures and systems and are, therefore, considered common AMPs. The staff review of the common AMPs is in Section 3.0.3 of this SER.

In Table 3.4-1, Item 12 of the LRA, the applicant stated that there is underground piping in the auxiliary feedwater system. The Buried Pipe and Tanks Inspection Program will manage the aging effects for this underground piping. Table 2.3-40 of the LRA, for the emergency feedwater system, only identifies orifices as subject to aging management by the Buried Pipe and Tanks Inspection Program. The staff issued RAI 3.4-14, requesting the applicant to explain why the auxiliary feedwater system piping in Table 2.3-40 does not refer to the Buried Pipe and Tanks Inspection Program and how the underground piping in the auxiliary feedwater system is managed for aging.

*

In its response by letter dated June 12, 2003, the applicant stated that Table 2.3-40 of the LRA should have included reference to Table 3.4-1, Item 12 in the AMR results for pipe. Table 3.4-1, Item 12, states that the Buried Pipe and Tanks Inspection Program is the credited program to manage aging for underground piping in the emergency feedwater system.

The staff finds the applicant's response to RAI 3.4-14, reasonable and acceptable because it provides an explanation that the underground auxiliary feedwater system piping is managed against aging effects by the Buried Pipe and Tanks Inspection Program.

On the basis of its review, the staff finds that the AMPs credited in the LRA for the auxiliary boiler steam and feedwater system components will effectively manage or monitor the aging effects identified in the LRA.

Conclusions

On the basis of its review, the staff finds the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.4.2.4.2 Condensate System

Summary of Technical Information in the Application

The AMR results for the condensate system are presented in Tables 3.4-1 and 3.4-2 of the LRA. The applicant used the GALL Report format to present its AMR of condensate system components in LRA Table 3.4-1. In LRA Table 3.4-2, the applicant identified the component group designation along with its (1) material, (2) environment, (3) aging effects, and (4) AMPs.

As described in Section 2.3.4.2, the condensate system is designed to pump exhaust steam from the main condenser hotwell through the low pressure feedwater heaters to maintain deaerator storage tank level for anticipated operating conditions. It also serves as a source of cooling water for the steam packing condenser and steam generator blowdown heat exchanger, and provides sealing water for various vacuum valves and feedwater pump seals.

Except for the CST, the condensate system is nonnuclear, safety-related. The CST is safety-related since it is the primary inventory source for the emergency feedwater system. Makeup water to the CST is demineralized water, admitted through the condenser and condenser storage subsystem.

Aging Effects:

LRA Tables 3.4-1 and 3.4-2 identify the following applicable aging effects for the condensate system:

- loss of material due to general (carbon steel only), pitting, and crevice corrosion of carbon and stainless steel components in treated water and steam environments
- loss of material due to general (carbon steel only), pitting, and crevice corrosion in sun, weather, humidity, and moisture environments

Aging Management Programs:

The following AMPs are utilized to manage aging effects to the condensate system:

- Chemistry Program
- Inspections for Mechanical Components Program
- Maintenance Rule Structures Program

A description of these AMPs is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the condensate system will be adequately managed by these AMPs such that there is reasonable assurance that the intended functions will be maintained consistent with the CLB during the period of extended operation.

Staff Evaluation

In addition to Section 3.4 of the LRA, the staff reviewed the pertinent information provided in Section 2.3.4, "Steam and Power Conversion Systems," and the applicable AMP descriptions provided in Appendix B of the LRA to determine whether the aging effects for the condensate system components have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of the condensate system components at VCSNS. The staff's evaluation includes a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the condensate system components.

Aging Effects:

The component group identified in LRA Table 2.3.26 for the condensate system is the condensate storage tank.

In Table 3.4-1, Item 11 of the LRA, the applicant stated that the Inspections of Mechanical Components Program is used to monitor the external surfaces of the aboveground CST for loss of material. For tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations, such as the tank bottom. The staff issued RAI 3.4-13, requesting the applicant to explain if the bottom of the CST is located on an earthen or concrete foundation, and if so, to provide justification for not managing aging effects on the exterior, bottom portion of the tank.

In its response by letters dated June 12, 2003, and September 2, 2003, the applicant stated that the below grade foundation of the CST is comprised of a 4 foot thick slab of reinforced concrete, the top of which is 1 foot below grade. A reinforced concrete, circular ringwall that is 2 feet high and 2½ feet thick connects to this slab and extends from the top of the slab to 1 foot above grade. The CST attaches to the top of this ringwall by a base ring flange, which is anchored to the ringwall by anchor bolts. All voids between the ringwall and base ring are grouted. The outer edge of the base ring is coated with cold plastic coal tar pitch flashing compound. Inside this ringwall, the CST sits on a clean, dry sand bed (as originally poured), which extends from the top of the foundation slab to the top of the ringwall. There are four small ringwall drains penetrating the ringwall 1 foot below grade. These drains are semicircular in shape with a 3-inch radius and are filled with clean, crushed stone to retain the sand within the ringwall. Because of the grouting and flashing at the base ring, water intrusion to the tank bottom is not expected to occur at the base ring; however, any water intrusion would seep through the sand to the ringwall drains. The four ringwall drains also allow translation of water to and from the sand contained by the ringwall. In the unlikely event that the ground outside of the ringwall becomes moisture saturated for an extended period of time, the sand inside the ringwall could only saturate to grade level. The 1 foot of sand from grade level to the bottom of the tank would remain dry. Because the external surface of the bottom of the CST remains dry, it should experience no aging effects requiring management. Because of the grouting and flashing at the base ring, water intrusion to the tank bottom is not expected to occur at the base ring; however, if it did occur, any water intrusion would seep through the sand to the ringwall drains, therefore, water would not pool at the bottom external surface of the tank. The four ringwall drains also allow translation of water to and from the sand contained by the ringwall. In the unlikely event that the ground outside of the ringwall becomes moisture saturated for an extended period of time, the sand inside the ringwall could only saturate to grade level. The 1 foot of sand from grade level to the bottom of the tank would remain dry. Because the external surface of the bottom of the CST remains dry, it should experience no aging effects requiring management; however, should the tank bottom experience any moisture it is unlikely that the tank would experience any significant degradation. Carbon steel exposed to the ambient, moist

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On the basis of its review, the staff finds the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.4.2.4.4 Extraction Steam System

Summary of Technical Information in the Application

The AMR results for the extraction steam system are presented in Tables 3.4-1 and 3.4-2 of the LRA. The applicant used the GALL Report format to present its AMR of extraction steam system components in LRA Table 3.4-1. In LRA Table 3.4-2, the applicant identified the component group designation along with its (1) material, (2) environment, (3) aging effects, and (4) AMPs.

As described in Section 2.3.4.4, the extraction steam system supplies steam for heating the condensate and feedwater and for maintaining the auxiliary boilers in a hot standby condition. The mechanical license renewal function of this system is to provide a means of main steam isolation (when used in conjunction with components from various other systems) for a steam line break coincident with failure of a main steam isolation valve.

Aging Effects:

LRA Tables 3.4-1 and 3.4-2 identify the following applicable aging effects for the extraction steam system:

- loss of material due to general (carbon steel only), pitting, and crevice corrosion of carbon and stainless steel components in treated water and steam environments
- loss of material due to general corrosion of carbon and low-alloy steel components (external surfaces) in air, moisture, and humidity environments
- wall thinning due to flow-accelerated corrosion of carbon steel components in steam and treated water environments
- loss of material due to boric acid corrosion of carbon steel components (external surfaces) in air, leaking, and dripping chemically treated borated water environments

Aging Management Programs:

The following AMPs are utilized to manage aging effects to the extraction steam system:

- Chemistry Program
- Inspections for Mechanical Components Program
- ~~Maintenance Rule Structures Program~~
- Flow-Accelerated Corrosion Monitoring Program
- Boric Acid Corrosion Surveillances Program
- ~~Heat Exchanger Inspections Program~~

A description of these AMPs is provided in Appendix B of the LRA. The applicant indicated that the effects of aging associated with the components of the extraction steam system will be adequately managed by these AMPs such that there is reasonable assurance that the intended functions will be maintained consistent with the CLB during the period of extended operation.

Staff Evaluation

In addition to Section 3.4 of the LRA, the staff reviewed the pertinent information provided in Section 2.3.4, "Steam and Power Conversion Systems," and the applicable AMP descriptions provided in Appendix B of the LRA to determine whether the aging effects for the extraction steam system components have been properly identified and will be adequately managed during the period of extended operation, as required by 10 CFR 54.21(a)(3).

This section of the SER provides the staff's evaluation of the applicant's AMR for the aging effects and the appropriateness of the programs credited for the aging management of the extraction steam system components at VCSNS. The staff's evaluation includes a review of the aging effects considered and the basis for the applicant's elimination of certain aging effects. In addition, the staff has evaluated the appropriateness of the AMPs that are credited for managing the identified aging effects for the extraction steam system components.

Aging Effects:

The component groups identified in LRA Table 2.3.41 for the extraction steam system are piping and valve bodies. The staff reviewed the aging effects identified in LRA Tables 3.4-1 and 3.4-2 for these component groups and finds the applicant properly identified the aging effects for these component groups. The aging effects are listed in SER Section 3.4.2.4.4.

The aging effects identified in the LRA for the extraction steam system are consistent with industry operating experience for the materials and environments listed. The staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified.

Aging Management Programs:

The following AMPs are utilized to manage aging effects to the extraction steam system:

- Chemistry Program
- Inspections for Mechanical Components Program
- ~~Maintenance Rule Structures Program~~
- Flow-Accelerated Corrosion Monitoring Program
- Boric Acid Corrosion Surveillances Program
- ~~Heat Exchanger Inspections Program~~

Each of the above AMPs (except the Flow-Accelerated Corrosion Monitoring AMP) is credited with managing the aging of several components in different structures and systems and are, therefore, considered common AMPs. The staff review of the common AMPs is presented in Section 3.0.3 of this SER. The Flow-Accelerated Corrosion Monitoring Program is credited with managing aging effects in the SPC systems only and is, therefore, considered a plant-specific

corrosion of embedded steel are significant. Possible aging effects for containment concrete structural components due to these three aging mechanisms are cracking, change in material properties, and loss of material.

The AMP recommended by the GALL Report for managing the above aging effects for containment concrete components in accessible portions of the containment structures is the ASME Section XI, Subsection IWL (XI.S2) Program. The staff's evaluation of the applicant's ASME Section XI, Subsection IWL AMP is in Section 3.5.2.3.6 of this SER.

Subsection IWL exempts from examination those portions of the concrete containment that are inaccessible (e.g., foundation, below-grade exterior walls, concrete covered by liner). For inaccessible portions of the containment structure, 10 CFR 50.55a(b)(2)(ix) requires that the licensee evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas.

The applicant addressed the specific criteria defined in the GALL Report, regarding the need for further evaluation to manage the potential aging of containment concrete structural components in inaccessible areas in LRA Table 3.5-1. The GALL Report recommends further evaluation for containment concrete in inaccessible areas if the aging mechanism's (1) leaching of calcium hydroxide, (2) aggressive chemical attack, or (3) corrosion of embedded steel are significant. In LRA Table 3.5-1, AMR item 7, the applicant stated that —

> The VCSNS containment structure is not exposed to flowing water and designed in accordance with ACI-318 and constructed in accordance with ACI-301 and ASTM Standards, which provide a good quality, dense, low permeability concrete.
>
> The water chemical analysis results confirm that the site groundwater is mildly acidic but considered to be non-aggressive.

Further, the applicant concluded that —

> Inaccessible areas at VCSNS do not require a plant-specific aging management for leaching of calcium hydroxide, aggressive chemical attack or corrosion of embedded steel.

The staff position is that inaccessible concrete components (i.e., below grade) require aging management unless specific criteria defined in NUREG-1801, GALL Volume 2, are satisfied to demonstrate a nonaggressive below-grade environment. As part of RAI 3.5-2, the staff requested the following information:

(c) Submit a quantitative assessment of the below-grade environment, comparing it to the specific criteria defined in GALL Volume 2.

(d) If it is nonaggressive, based on satisfaction of the specific criteria defined in GALL Volume 2, describe the groundwater monitoring program that will be implemented to verify that the below-grade environment remains nonaggressive, including monitoring frequency and consideration of seasonal fluctuations.

(e) If the below-grade environment does not satisfy the specific criteria defined in GALL Volume 2, describe in detail the plant-specific AMPs for inaccessible concrete components.

In its initial response to RAI 3.5-2, parts (c), (d), and (e), the applicant stated the following:

> (c) Section 6.1 (Table 6.1-3) of TR00170-003 provides the quantitative assessment of the below-grade groundwater environment at VCSNS. These analyses results are based on samples taken in 2001 from three (3) wells in the general vicinity of plant structures. [Note that prior sample analyses for chlorides, sulfates and pH do not exist.] Groundwater chlorides (from all three wells) were determined to be < 10 ppm, which is well within the GALL defined limits of < 500 ppm. Groundwater sulfates (from all three wells) were determined to be < 10 ppm, which is well within the GALL defined limits of < 1500 ppm. Groundwater pH (from the three wells) was determined to range from 4.8 to 5.3, which marginally exceeds the GALL defined limits of 5.5. Based on these results, the VCSNS Application defines the site groundwater as non-aggressive, although mildly acidic.
>
> (d) Application Table 3.5-1, Item 17 specifies that periodic monitoring of the below-grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. VCSNS Engineering Services Procedure (Inspections for Maintenance Rule - Structures) will be revised to include a chemical analysis of raw water (including groundwater) on a 5-year interval to coincide with the Maintenance Rule Structures Inspections. [Note that seasonal fluctuations are not applicable at VCSNS since the level of groundwater remains relatively constant due to the influence of Monticello Reservoir.]
>
> (e) Application Table 3.5-1, Items 7 and 16, discusses aging mechanisms and effects for inaccessible concrete. Since the VCSNS below grade environment marginally exceeds the specific pH criteria defined in GALL, the concrete design was further reviewed and determined to provide protection against aggressive chemical attack. Since the below-grade structures are considered to be resistant to the mildly acidic environment, plant specific aging management programs are not required for inaccessible concrete areas.

The staff position is that any deviation from the specific criteria defined in GALL Volume 2 constitutes an aggressive environment, and aging management of inaccessible concrete is necessary.

In its supplemental response to RAI 3.5-2, the applicant committed to a plant-specific program to manage aging of inaccessible concrete:

> The NRC Staff position is that the VCSNS groundwater is considered to be aggressive since it has a pH < 5.5. In order to satisfy this concern, the following provisions will be incorporated as part of existing plant programs and procedures:
>
> SEE REVISED RESPONSE FOR RAI 3.5-2 (c)
>
> 3. The site excavation and backfill procedure will be revised to include a concrete surface inspection by engineering personnel if soil is removed adjacent to any concrete structure surfaces at or below the nominal groundwater elevation of 423'.
>
> 4. As noted in response to RAI 3.5-2(d), chemical analysis of groundwater will be conducted on a 5-year interval to coincide with the Maintenance Rule Structures Inspection Program. This analysis will also include a water sample from the Service Water Pond.
>
> 5. Underwater diver's inspections of the Service Water Intake Structure (tunnel) will continue as described in response to RAI 3.5-26. These inspections will provide additional assurance of the integrity of concrete structures exposed to below water conditions.

Since the applicant's program is consistent with programs previously accepted by the staff to address this issue, the staff finds it acceptable.

On the basis of its review, the staff finds that the applicant appropriately evaluated AMR results involving management of aging of inaccessible concrete areas for containment, as recommended in the GALL report. Since the applicant's AMR results are otherwise consistent with the GALL report, the staff finds that the applicant has demonstrated that the effects of

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 5-year inspection of the service water pond dams required by the NRC, Federal Energy Regulatory Commission (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 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.

The applicant concluded that 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 applicant further concluded that the Service Water Pond Dam Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to AMR will continue to perform their intended functions consistent with the CLB for the period of extended operations.

Staff Evaluation

In LRA Section B.1.21, "Service Water Pond Inspection Program," the applicant described its AMP to manage trends in movement and the effects of aging for the service water dams. The LRA stated that this AMP is consistent with GALL AMP XI.S7, "RG 1.127 Inspection of Water-Control Structures Associated With Nuclear Power Plants," with several enhancements described in Section 3.5.2.3.9. The staff ~~will confirm~~ confirmed the applicant's claim of consistency during the AMR inspection. Furthermore, the staff reviewed the enhancements to determine whether the AMP, with the enhancements, remains adequate to manage the aging effects for which it is credited, and reviewed the FSAR supplement to determine whether it provides an adequate description of the revised program.

The staff noted several inconsistencies between the FSAR supplement summary descriptions of the AMPs in LRA Appendix A and the scope of the AMPs identified in LRA Appendix B as "consistent with GALL." In RAI 3.5-19, the staff requested the applicant to verify that the complete scope of the AMP, as described in NUREG-1801, GALL Volume 2, is being credited to manage aging effects for license renewal. If this is not the case, the applicant was requested to identify and document the justification for each exception. In response to RAI 3.5-19, the applicant stated the following:

> As stated in the Application, VCSNS maintains a Service Water Pond Dam Inspection Program (B.1.21), which is consistent with GALL XI.S7 and RG 1.127. One enhancement to this program was identified during a NRC/FERC inspection as identified in the Application and discussed in Section 7.15 of TR00170-003.

VCSNS does not believe that there are any further changes required for the Application Appendix A, since only summary statements are recommended by NEI 95-10. Commitment to all Regulations and Regulatory Guides are implicit in the development of each of these programs as described in Section 7 of TR00170-C03.

LRA Section B.1.21 states that the Service Water Pond Dam Inspection Program is consistent with GALL XI.S7 with several listed enhancements that will be incorporated into the program prior to the period of extended operation. In RAI 3.5-25, the staff requested that the applicant provide the following information regarding this program:

1. The commitment to incorporate the enhancements to this program discussed in LRA Section B.1.21 should also be included in the FSAR supplement, Appendix A, Section 18.2.31. This section does not currently include such a commitment. Issues related to the FSAR supplement are being addressed by the staff on a generic basis.

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

In response to RAI 3.5-25, the applicant stated the following:

(a) Consistent with NEI 95-10, VCSNS does not see the need to include these minor enhancements into the very generic summary description of the Service Water Pond Dam Inspection Program (Application Section 18.2.31).

(b) The East Dam of the Service Water Pond (SWP) is the smallest and least critical (important) of the four SWP dams since it primarily caps a natural high elevation ridge line along the east side of the pond. There are no piezometers or alignment/survey monuments for this structure. The East Dam is inspected as part of the Service Water Pond Dam Inspection Program. There are no operating experience issues associated with this dam other than normal observations of minor erosion and weed growth at the edges of the riprap protection.

The staff finds the applicant's discussion on operating experience for the East Dam to be acceptable.

Conclusions

On the basis of its review and audit of the applicants program, the staff finds that those portions of the program for which the applicant claims consistency with the GALL program are consistent with the GALL program. In addition, the staff has reviewed the enhancements to the GALL program and finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the FSAR supplement for this AMP and finds that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

3.5.2.3.10 Service Water Structures Survey Monitoring Program

Summary of Technical Information in the Application

In LRA Section B.1.22, the applicant stated that the Service Water Structures Survey Monitoring Program is a plant specific program that is not addressed in GALL. The applicant further stated that survey monitoring is required for structures that are supported by earthen fill

building structures and structural components is contained in Report TR00170-003, Revision 0, Attachment II, pages 2 through 51.

The staff notes that Report TR00170-003, Revision 0, Attachment II, page 34, identifies "Neutron absorbing sheets - Boraflex" and a 'Boraflex Monitoring Program." However, in the "Notes" column, the applicant indicates that Boraflex will be replaced with Boral. The applicant's AMR for Boral is presented in LRA Table 3.3-1, AMR Items 9 and 12, the applicant did not identify any aging effects requiring management for license renewal.

A brief description of the other building structures is provided in LRA Section 2.4.2, "Other Structures." The materials of construction for the building structures and structural components are carbon steel, stainless steel, concrete, elastomers, masonry block, drywall, Boral, and styrofoam. These materials are exposed to one or more of the following environments — outdoor, indoor, borated water, below-grade.

Aging Effects:

Report TR00170-003, Revision 0, Attachment II identifies the following applicable aging effects for other building structures and structural components:

- loss of material and MIC for carbon steel components
- change in material properties, cracking, and loss of material for concrete components
- cracking of masonry block
- cracking, shrinkage, and change in material properties for elastomers
- cumulative fatigue and cracking for stainless steel components
- degradation of styrofoam and drywall

Aging Management Programs:

Report TR00170-003, Revision 0, Attachment II credits the following AMPs with managing the identified aging effects for other building structures and structural components:

- Chemistry Control Program
- Fire Protection Program
- Containment ISI Program — IWE/IWL
- 10 CFR Part 50 Appendix J Leak Rate Testing Program
- Maintenance Rule Structures Program
- Boric Acid Corrosion Surveillances Program
- Battery Rack Inspection Program
- ASME Section XI ISI Program — IWF
- Material Handling Systems Inspection Program

A description of these AMPs is provided in LRA Appendix B. The applicant concluded that the effects of aging associated with other building structures and structural components will be adequately managed by these AMPs such that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

Staff Evaluation

The staff reviewed the information in Sections 2.4, 3.3, and 3.5 of the LRA; Report TR00170-003, Revision 0, Attachment II; the applicant's responses to the staff's RAIs; and the applicable AMP descriptions in Appendix B of the LRA, to determine whether the applicant has demonstrated that the aging effects associated with the other building structures and structural components will be adequately managed during the period of extended operation.

In the initial review of the applicant's AMR for the other building structures and structural components, the staff identified several issues in need of resolution.

In Report TR00170-003, Revision 0, Attachment II: Aging Management Review Results for Structures and Structural Components, cable trays, conduit, and electrical and instrument panels and enclosures are identified as component types within most of the buildings and structures. These components are identified as steel in an internal environment, except for the electrical substation and transformer area, where the environment is external. In all cases, no aging effect requiring aging management is identified.

The staff believes that these components located in the reactor, auxiliary, intermediate, and fuel handling buildings are susceptible to boric acid corrosion and that these components located in an external environment are susceptible to environmental corrosion. Therefore, in both cases loss of material is an applicable aging effect requiring aging management. In RAI 3.5-1, the staff requested the applicant to identify and describe the AMPs, which will manage loss of material for these components located in the reactor, auxiliary, intermediate, and fuel handling buildings, and in an external environment.

In its response to RAI 3.5-1, the applicant stated the following:

> 1) Section 6.2 of TR00170-003 identifies electrical panels, cabinets, cable trays, etc. as being constructed of factory baked painted steel or galvanized sheet metal, both of which do not have a tendency to age with time due to general corrosion. VCSNS realized that these components are designed for outdoor service and industry operating experience has not shown a case where aging effects caused a loss of intended function. Therefore, these components in the Electrical Substation and Transformer Area were judged to have no aging effects from general corrosion due to an external environment.
>
> Even though corrosion is considered unlikely, Attachment II of TR00170-003 will be revised for the external environment to include loss of material (for Cable Tray & Conduit and Electrical and Instrument Panels & Enclosures) as an aging effect which is managed by the Maintenance Rule Structures Program.
>
> 2) The attributes of these materials (factory baked painted steel or galvanized sheet metal) were similarly deemed to provide additional protection from boric acid corrosion and thus judged to have no aging effects. However, Section 7.6 of TR00170-003, "Boric Acid Corrosion Surveillances" (Scope of Program) does include these electrical components under Boric Acid Corrosion Surveillances for managing aging effects (loss of material). Therefore, Attachment II of TR00170-003 will be revised for Reactor, Auxiliary, Intermediate, and Fuel Handling Buildings to include loss of material (for Cable Tray & Conduit and Electrical and Instrument Panels & Enclosures) as an aging effect which is managed by Boric Acid Corrosion Surveillances and Maintenance Rule Structures Program.

The staff finds that the part of the applicant's response to RAI 3.5-1, pertaining to a borated environment, is acceptable because the applicant has committed to manage aging of cable trays, conduit, and electrical and instrument panels and enclosures in a borated water environment as part of the Boric Acid Corrosion Surveillances Program and the Maintenance Rule Structures Program.

In the initial review of the applicant's AMR for earthen embankments, the staff determined that the applicant has identified the appropriate material and aging effects, and has credited appropriate AMPs to manage aging. The staff did not issue any RAIs related to earthen embankments. The detailed staff review of the Maintenance Rule Structures Program is in Section 3.0.3.4 of this SER. The detailed staff review of the Service Water Pond Dam Inspection Program is in Section 3.5.2.3.9 of this SER.

The aging effects identified in the LRA for the earthen embankments are consistent with industry operating experience for the materials and environments listed. The staff finds that all the plausible aging effects were identified and that the aging effects listed are appropriate for the combination of materials and environments specified. On the basis of its review, the staff finds that the AMPs credited in the LRA for the earthen embankments will effectively manage or monitor the aging effects identified in the LRA.

Conclusions

On the basis of its review, the staff finds the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

## 3.6 Electrical and Instrumentation and Controls

The applicant described its AMR of electrical and instrumentation and controls components requiring AMR in Section 3.6 of the LRA. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the effect of aging on the electrical and instrumentation and controls components will be adequately managed during the period of extended operation, as required by 10 CFR 54.2(a)(3).

### 3.6.1 Summary of Technical Information in the Application

The applicant has performed an AMR on the following electrical and I&C commodity groups:

- non-EQ insulated cables
- non-EQ connectors
- non-EQ spices
- non-EQ electrical penetration assemblies
- non-EQ terminal blocks
- high voltage electrical switchyard bus
- high voltage transmission conductors and connections
- high voltage insulators

The AMR methodology for the electrical discipline for VCSNS is summarized in the following points:

- evaluation of the electrical component commodity groups (subject to AMR) to identify the organic materials subject to age-related degradation

- identification and evaluation of the 60-year service-limiting environmental parameters for these organic materials
- identification and evaluation of the aging mechanisms and effects to determine which require review
- identification and evaluation of the service conditions (i.e., the operating environments and locations) for the electrical component commodity groups
- evaluation of the industry and plant-specific operating experience for the electrical component commodity groups
- aging management program evaluation (following NUREG-1801)
- demonstration of aging management

The review of the VCSNS electrical component commodity groups with respect to aging mechanisms and effects was performed based upon the guidance of various industry documents, primarily SAND 96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants — Electrical Cable and Terminations." This document provides detailed materials analysis for cable and termination materials exposed to nuclear power plant environments. It also provides guidance for performing AMRs pursuant to 10 CFR Part 54.

The methodology used for the AMR of the electrical commodity groups employs the "Plant Spaces" approach in which the plant is segregated into areas (or spaces) where common bounding environmental parameters can be assigned. The VCSNS plant operating environments are delineated as "Environmental Zones." Each bounding environmental zone is evaluated against the material of the commodity groups most susceptible to aging to determine if the components will be able to maintain their intended function through the period of extended operation. With respect to the electrical components, the environmental parameters of interest are temperature, radiation, and moisture.

The intended functions of the electrical component commodity groups under review are as follows:

- to electrically connect or insulate two sections of an electrical circuit and/or to provide for continuity or insulation of electrical circuits
- to provide a leak-tight barrier for containment isolation (this is evaluated in Section 2.4.1.3 of the LRA)

The applicant's AMRs included an evaluation of plant-specific and industry operating experience. The plant-specific evaluation included reviews of condition reports and discussions with appropriate site personnel to identify aging effects that require management. These reviews concluded that no additional aging effects requiring management were identified beyond those identified using the methods described in Section 3.6.2.1 of the LRA.

The applicant's review of industry operating experience included an evaluation of industry operating experience since the publication of NUREG-1801 to identify any additional aging

On the basis of its review, the staff has verified the applicant's claim of consistency with the GALL Report. The staff finds that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

*3.6.2.2 Aging Management Evaluations in the GALL Report That Are Relied On For License Renewal, For Which GALL Recommends Further Evaluation*

For component groups evaluated in GALL for which the applicant has claimed consistency with GALL, and for which GALL recommends further evaluation, the staff reviewed the applicant's evaluation to determine whether it adequately addressed the issues for which GALL recommended further evaluation. In addition, the staff sampled components in these groups to determine whether the plant-specific components contained in these GALL component groups were bounded by the GALL evaluation.

3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

Environment qualification is a TLLA as defined in 10 CFR 54.3. TLLAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviewed the evaluation of this TLAA separately in Section 4.4 of this SER, following the guidance in Section 4.4 of the SRP-LR.

*3.6.2.3 Aging Management Programs for Electrical and Instrumentation and Controls Components*

In SER Section 3.6.2.1, the staff determined that the applicant's AMRs and associated AMPs will adequately manage component aging in electrical and I&C systems. The staff then reviewed specific electrical and I&C components to ensure that they were properly evaluated in the applicant's AMR.

To perform this review, the staff reviewed the components listed in LRA Tables 2.5-1, 2.5-2, and 2.5-3 to determine whether the applicant has properly identified the applicable AMRs and AMPs needed to adequately manage the aging effects for the components. This portion of the staff review involved identification of the aging effects for each component, ensuring that each aging effect was evaluated using the appropriate AMR in Section 3, and that management of the aging effect was captured in the appropriate AMP. The results of the staff's review are provided below.

The staff also reviewed the FSAR supplement for the AMPs credited with managing aging in electrical and I&C system components to determine whether program descriptions adequately describe the programs.

The applicant credits two AMPs to manage the aging effects associated with electrical and I&C components. One of these AMPs is credited to mange aging for components in other system group (common AMPs), while the other AMP is credited with managing aging only for electrical and I&C components. The staff's evaluation of the common AMP credited with managing aging in electrical and I&C components is provided in Section 3.0.3 of this SER. The common AMP is Boric Acid Corrosion Surveillance, SER Section 3.0.3.1.

The staff's evaluation of the electrical and I&C component system AMP is provided here.

3.6.2.3.1 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Summary of Technical Information in the Application

Non-EQ Insulated Cables:

The applicant stated that non-EQ insulated cables include power cables, control cables, and instrument cables. For VCSNS, the applicant defines these applications to be at the following voltage levels:

- ~~1.12~~ low voltage cables — 480 VAC, 240/120 VAC, 125 VDC (and less)
- medium voltage cables — 7.2 kV
- high voltage cables — greater than 7.2 kV (none in scope)

In order to facilitate the review of the cables at VCSNS, the applicant places cables into two categories — power cable and I&C cable. The power cable category includes all 7.2 kV cables and the 480 VAC power cables. The I&C category includes the 480 VAC control cable, all 240/120 VAC cable, and all DC cables (125 VDC and less). Depending upon their application, cables utilized as switchboard wire are placed into one of these two categories, typically as I&C cable. The applicant indicated that VCSNS purchased nearly all of its electric power cable, control cable, and instrument cable (with the exception of certain communication cables, cables ordered for specific non-safety applications, and special cables ordered subsequently for specific modifications) to 10 CFR 50.49 harsh EQ standards.

The worst case cable insulation possible in application used in license renewal is polyethylene with a 60 year service limiting temperature of 131 °F. The non-EQ insulated cables will be subject to an AMP as described in Table 3.6.1.

Non-EQ Electrical Connectors:

The applicant stated that cable connections are used to connect the cable conductors with other cables or with a variety of electrical devices (e.g., instruments, motors, etc.). The various types of insulated cable connections (or terminations) are identified in the Cable Aging Management Guideline (AMG). The Cable AMG describes the cable termination grouping as follows:

- compression connectors
- fusion connectors
- plug-in/multi-pin connectors

The applicant reviewed a variety of plant documents to identify electrical connectors in use at VCSNS, including procurement records, plant drawings, EQ binders, and plant maintenance documents. This review provided reasonable assurance that all types of connectors have been identified and that the bounding materials for the connectors at VCSNS have also been identified. Connectors are included in the Non-EQ Insulated Cables and Connections Inspection Program.

3.6.2.4.1 Non-EQ Electrical Penetration Assemblies

Summary of Technical Information in the Application

The applicant stated that electrical penetration assemblies are utilized to carry electrical circuits through the reactor building containment wall while maintaining pressure-tight integrity. They provide the electrical continuity of the circuit and the pressure boundary for containment integrity. The scope of the review in this report applies only to the electrical function of the penetration assemblies. The pressure-retaining function of the penetration assemblies is addressed in Section 2.4 of this application for the reactor building. All the electrical penetrations at VCSNS have been listed in the VCSNS EQ Program, whether or not they carry Class 1E circuits. The non-Class 1E electrical penetrations are classified as category "B1, B2" components with respect to EQ (i.e., they must not fail and prevent the accomplishment of a safety-related function) and are administratively included in the EQ Program in order to credit the portion of the EQ testing which justifies the pressure-retaining function of the penetrations. VCSNS utilizes D.G. O'Brien electrical penetration for its non-Class 1E applications. The D.G. O'Brien electrical penetration assemblies are subject to AMR. This review provides for their identification and also for the listing of the organic materials found during the review. Because there are D.G. O'Brien electrical penetration assemblies that are part of the VCSNS EQ Program and have been evaluated in detail for that purpose, there is reasonable assurance that all their organic materials have been identified and properly evaluated with respect to aging for the non-EQ installations. An additional review has shown non-EQ electrical penetrations at VCSNS to be located in areas inside and outside of the reactor building which have less severe environments, that are clearly enveloped by material properties and for which aging testing and evaluation has been done through the manufacturer. The non-EQ electrical penetrations at VCSNS are not included in the Non-EQ Insulated Cables and Connections Inspection Program. The component type, material, environment, and aging effects are identified in Table 3.6-2 of the LRA. The evaluation of the non-EQ electrical penetrations at VCSNS is further documented in Table 3.6-2 Item 2 of the LRA.

Aging Effects:

The LRA identified the following aging effects for the non-EQ electrical penetrations:

- 13. embrittlement
- 14. cracking
- 15. melting
- 16. discoloration
- 17. swelling
- 18. loss of dielectric strength leading to reduced insulation resistance
- 19. electrical failure caused by thermal/thermoxidative degradation of organic
- 20. radiolysis and photolysis (ultraviolet sensitive materials only) of organic
- 21. radiation-induced oxidation
- 22. moisture intrusion

Aging Management Programs:

No AMP is required for non-EQ electrical penetration. The applicant states that a review has shown non-EQ electrical penetrations at VCSNS to be located in areas inside and outside of

the reactor building which have less severe environments, that are clearly enveloped by material properties and for which aging testing and evaluation has been done through the manufacturer. Non-EQ electrical penetrations at VCSNS are not included in the Non-EQ Insulated Cables and Connections Inspection Program.

Staff Evaluation

This section provides the results of the staff's evaluation of the applicant's AMR for the aging effects and the AMPs credited for managing the aging effects in non-EQ electrical penetrations at VCSNS. The staff also reviewed the applicable FSAR supplement for the AMPs to ensure that the program description adequately describe the AMPs.

Aging Effects and Aging Management Programs:

The applicant identified embrittlement, cracking, melting, discoloration, swelling, loss of dielectric strength leading to reduced insulation resistance, electrical failure caused by thermal/thermoxidative degradation of organics, radiolysis and photolysis (ultraviolet sensitive materials only) of organic, radiation-induced oxidation, and moisture intrusion are the aging effects/mechanism of non-EQ electrical penetrations. The staff agrees with the scope of aging effects identified by the applicant. These aging effects are consistent with the aging effects identified by the staff in the GALL Report.

The applicant stated that its review has shown non-EQ electrical penetrations at VCSNS to be located in areas inside and outside of the reactor building which have less severe environments, that are clearly enveloped by material properties and aging testing and evaluation done through the manufacturer. Non-EQ electrical penetrations at VCSNS are not included in the Non-EQ Insulated Cables and Connections Inspection Program.

The staff was not convinced that there are no aging effects for non-EQ electrical penetration because these penetrations are located in a less severe environment and are covered by evaluation done by manufacturer. In most areas within a nuclear power plant, the actual ambient environments are less severe than the nominal plant environment. However, in a limited number of localized areas, the actual environments may be more severe than the nominal plant environment. Insulation materials used in non-EQ electrical penetration assemblies may degrade more rapidly than expected in these adverse localized environments. The staff requested the applicant to provide a description of an AMP for non-EQ electrical penetration exposed to localized environment caused by heat, radiation, or moisture, or provide a technical justification of why an AMP is not necessary (RAI 3.6-4).

In response to the staff's request, in a letter dated June 12, 2003, the applicant stated that all VCSNS electrical penetrations are included within the VCSNS Harsh EQ Program and meet the requirements of 10 CFR 50.49. The non-Class 1E as well as the Class 1E electrical penetrations are considered subject to a TLAA and will be reanalyzed for a 60-year life under the EQ Program. All electrical penetrations have a definitive long-lived qualified life assigned within the EQ database, "HARSH EQ Maintenance Manual", the same as all harsh EQ related equipment. Non-Class 1E electrical penetrations were previously conservatively listed as requiring an AMR because of their non-Class 1E status [reference LRA 3.6.1.4]. The AMR is not required as these electrical penetrations are to receive a TLAA for consideration of a 60-year life. There will be no AMP for electrical penetrations as these electrical penetrations have

* SEE COVER LETTER RC-03-0227

*4.2.1.1 Summary of Technical Information in the Application*

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that reactor vessel beltline materials must have an initial, pre-irradiated, Charpy Upper Shelf Energy (USE) of no less than 75 ft-lbs. and must maintain a Charpy USE of no less than 50 ft-lbs. throughout the life of the reactor vessel.

VCSNS calculated the beltline fluence for the determination of the decrease in Charpy USE due to radiation embrittlement and thermal aging of the reactor vessel. VCSNS then calculated the Charpy USE values for the beltline region materials using Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." In response to the U.S. Nuclear Regulatory Commission's (NRC) Generic Letter 92-01, Revision 2, "Reactor Vessel Structural Integrity," Revision 1, VCSNS reported the end of current license 32 effective full-power years (EFPYs) USE for the limiting beltline material to be 67.5 ft-lbs. for the intermediate plate A9154-1. The response to NRC Generic Letter 92-01 was based upon examinations of the first three VCSNS surveillance capsules.

VCSNS has two surveillance capsules remaining in the vessel. These capsules will be kept in the vessel until they receive sufficient additional exposure to neutron fluence in order to provide data that correlates to the estimated fluence on the vessel at the end of the extended period of operation. VCSNS will then withdraw these two capsules and analyze one and place the other one in storage. The Charpy USE will then be recalculated for additional fast neutron fluence corresponding to the end of the extended operating period. Therefore, as discussed above, VCSNS is utilizing 10 CFR 54.21 (c)(1)(ii) to calculate the reactor pressure vessel (RPV) Charpy USE to the end of the extended operating period. *

*4.2.1.2 Staff Evaluation*

The staff reviewed the USE evaluations contained in Section 4.2.1 of the LRA and Section 18.3.1.1 of Appendix A to the LRA. The staff issued RAI 4.2.2.1-1, in which it requested that the applicant submit 60-year end-of-life (EOL) USE values for each of the beltline materials and requested that the applicant address how surveillance capsule results were evaluated in its determination of the USE values. In a letter dated June 12, 2003, in response to RAI 4.2.2.1-1, the applicant indicated that (1) the EOL for VCSNS is 54 EFPY, and (2) the 54-EFPY fluence (E > 1.0 MeV) values may be found in Table 6-14 of Westinghouse Commercial Atomic Power report (WCAP-15101, "Analysis of Capsule W from the South Carolina Electric & Gas Company V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," dated September 1998.)

The WCAP report was attached to a letter from G. J. Taylor, South Carolina Electric & Gas Co. (SCE&G), to NRC Document Control Desk, dated October 9, 1998. This WCAP report contains the data from the test results of capsule W that was removed after 10.8 EFPYs with a lead factor of 3.40. The applicant states that the intermediate shell plate has 0.10 wt% Cu and is the limiting plate material. The highest percent of copper in the weld material is 0.05 wt%. The staff confirmed the data provided by the applicant by reviewing the data for the VCSNS reactor vessel materials against the data in the NRC Reactor Vessel Integrity Database (RVID). The staff determined that the intermediate shell plate, A9154-1, is the limiting plate material and its Cu and Ni contents are 0.10 wt% and 0.51 wt%, respectively. The staff noted that the VCSNS reactor vessel beltline materials have welds of only one heat, 4P4784, and their Cu and Ni contents are 0.05 wt% and 0.91 wt%, respectively.

The surveillance plate has Charpy test results from longitudinally oriented specimens, but does not have Charpy data from transversely oriented specimens. Therefore, to determine whether the plate material is in compliance with 10 CFR Part 50, Appendix G, the staff and the applicant must estimate the transverse properties from the longitudinal properties. The applicant states that the unirradiated USE value for the limiting plate is 132 ft-lbs. in the longitudinal direction, and 91 ft-lbs for the limiting weld material. The applicant also states that the unirradiated USE value for the limiting plate in the transverse direction is 75 ft-lbs., which is 56.8 percent of the USE value in the longitudinal direction. The staff estimated the unirradiated USE value for the plate material in transverse direction according to the guidance provided in Section 5.3.2 of the NRC report, "Standard Review Plan for the Review of Safety Analysis Reports for the Nuclear Power Plants," NUREG-0800, 1987. According to this NUREG report, the USE of the plate specimens in the transverse direction is 65 percent of that in the longitudinal direction and, therefore, is equal to 85.8 (0.65 x 132) ft-lbs., which is greater than the one reported by the applicant. The staff finds the unirradiated USE value of 75 ft-lbs. for the limiting plate material in the transverse direction acceptable for two reasons: (1) the unirradiated USE value is less than the one estimated using the guidance provided in NUREG-0800 and (2) the ratio of unirradiated USE value in the transverse to the longitudinal direction bounds the corresponding ratio of the measured USE values for the irradiated surveillance specimens as discussed in the next paragraph.

As reported in WCAP-15101, the measured USE at a fluence of 4.664E+19 n/cm$^2$ for the limiting plate is 126 ft-lbs. in the longitudinal direction and 74 ft-lbs. in the transverse direction. In other words, the measured USE value in the transverse direction is 58.7 percent of the one along the longitudinal direction. The measured USE at a fluence of 4.664E+19 n/cm$^2$ for the limiting weld material is 87 ft-lbs. The staff made independent estimates of corresponding USE values using the curves in Figure 2 of RG 1.99, Revision 2. The staff estimated that at a fluence of 4.664E+19 n/cm$^2$, the percentage drop in the USE value for the limiting plate and weld materials is 16 percent. The corresponding USE value for the limiting plate is 111 (132 x 0.84) lbs. in the longitudinal direction, which is less than the measured value. The staff estimate is based on an unirradiated USE of 84 ft-lbs, which is the value reported in the RVID. Similarly, the estimated USE value for the limiting weld is 71 (84 x 0.84) ft. lbs., which is less than the measured value. In other words, the estimated USE values, using RG 1.99, Revision 2 methodology, for both the limiting plate and weld materials are lower than the measured values. Since the values using RG 1.99, Revision 2 are lower than the measured values, the RG predicts conservative values.

The applicant provides the following information about the EOL USE values. The highest 54-EFPY fluence value listed in Table 6-14 of the WCAP report is 6.40 x $10^{19}$ n/cm$^2$, which is the calculated value at the vessel ID surface. The highest 54-EFPY fluence value at 1/4T is 4.29 x $10^{19}$ n/cm$^2$. Using curves in Figure 2 of RG 1.99, Rev. 2, the applicant estimates the predicted decrease in USE to be 31 percent for the limiting beltline plate material with 0.10 wt% Cu and EOL fluence of 6.40 x $10^{19}$ n/cm$^2$. This would reduce the USE for the limiting plate material from the unirradiated values of 132 ft-lbs. in the longitudinal direction and 75 ft-lbs. in the transverse direction to EOL values of 91 ft-lbs. and 51.75 ft-lbs., respectively. The applicant uses 91 ft-lbs. as an unirradiated USE value for the weld material. This value is higher than the one reported in the RVID (i.e., 84 ft-lbs.). For the weld material, the drop in USE reduces the unirradiated value of 91 ft-lbs. to EOL value of 62 ft-lbs.

* SEE COVER LETTER RC-03-0227

reactor surveillance capsules in order to obtain data that correlates to estimated fluence on the vessel at the end of extended operation. The Technical Specifications will be updated as required by 10 CFR Part 50. Therefore, the P-T limit analyses will be projected for the period of extended operation. This is acceptable because the staff will evaluate the recalculated ART values and associated P-T curves for the VCSNS reactor vessel beltline materials in accordance with the P-T limits requirements of 10 CFR Part 50, Appendix G, when the applicant submits them for an approval pursuant to the license amendment requirements of 10 CFR 50.90. *

The applicant has not provided any information about the maximum allowable low-temperature overpressure protection (LTOP) system power-operated relief valve (PORV) set points that are applicable for current 40-year operating period. The staff issued RAI 4.2.2.3-1 requesting the applicant to identify LTOP as part of the reactor vessel neutron embrittlement TLAA and commit to develop LTOP values for the period of extended operation, as was done for the P-T limits. In response to RAI 4.2.2.3-1, in a letter dated June 12, 2003, the applicant states that at VCSNS, the LTOP analysis is part of the calculation that develops the heatup and cooldown curves from analysis of the reactor vessel surveillance specimens. The applicant further states that the LTOP analysis will be done as part of the recalculation of the P-T curves when one of the two remaining surveillance capsules is removed from the vessel and analyzed. The staff finds this response acceptable because the applicant will submit the LTOP analysis along with the recalculated ART values and associated P-T curves as mentioned in the preceding paragraph for staff approval.

Pursuant to the requirements of 10 CFR 54.21(d), the applicant provided the FSAR Supplement description of the TLAA for the P-T limits in Section 18.3.1.3 of Appendix A, FSAR Chapter 18, to the LRA. The applicant states that the P-T limit curves for the period of extended operation will be constructed after the removal of the remaining two capsules. The remaining capsules must incur additional exposure to neutron fluence in order to provide data that correlates to the estimated fluence on the vessel at the end of the period of extended operation. Since the NRC staff will review the revised P-T limit curves along with the LTOP limits and approve them, the staff finds the applicant's FSAR Supplement statement to be acceptable.

*4.2.3.3 Conclusions*

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii) that, the analyses will be projected to the end of the period of extended operation for the P-T limits as part of the reactor vessel neutron embrittlement TLAA. The staff also concludes that the FSAR Supplement contains an appropriate summary description of the P-T limits as part of the reactor vessel neutron embrittlement TLAA evaluation for the period of extended operation. Therefore, the staff has reasonable assurance that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

## 4.3 Metal Fatigue

A metal component subjected to cyclic loading at loads less than the static design load may fail due to fatigue. Metal fatigue of components may have been evaluated based on an assumed

number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

### 4.3.1 Summary of Technical Information in the Application

The reactor vessel and major reactor coolant system (RCS) components were designed to the ASME Boiler and Pressure Vessel Code, Section III requirements for Class 1 components.

The applicant indicated that Class 1 components have been designed using the transient cycle assumptions in Table 5.2-2 of the FSAR. The applicant indicated that the VCSNS Inservice Inspection Program involves monitoring of thermal transients. The applicant uses the Thermal Fatigue Monitoring Program (TFMP) to track thermal transients. The TFMP is discussed in Section B.3.2 of the LRA. The applicant indicated that enhancements to the program are warranted to incorporate the new guidance in Electric Power Research Institute (EPRI) Report MRP-47, "Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." The applicant made a commitment to revise the TFMP to account for the effects of the reactor coolant environment, in accordance with NUREG-1801, Section X.M.1, prior to the period of extended operation.

The applicant discussed the evaluation of ASME Class 2 and 3 and American National Standards Institute (ANSI) B31.1 components in Section 4.3.2 of the LRA. ASME Class 2 and 3 and ANSI B31.1 require that a stress reduction factor be applied to the allowable thermal bending stress range if the number of full range cycles exceeds 7,000. The applicant indicated that most piping systems within the scope of license renewal are only subject to occasional cyclic operation and, consequently, the analyses will remain valid during the period of extended operation. However, the applicant did indicate that the RCS loop sampling line could exceed the 7,000 cyclic limit during the period of extended operation. The applicant indicated that either procedural controls would be implemented to assure the number of cycles remains below the 7,000 cycle or the calculation would be revised to verify the acceptability of the number of actual cycles.

### 4.3.2 Staff Evaluation

As discussed in the previous section, components of the RCS at VCSNS were designed to the Class 1 requirements of the ASME Code. These requirements contain explicit criteria for the fatigue analysis of components. Consequently, the applicant identified the fatigue analysis of these components as TLAAs. The staff reviewed the applicant's evaluation of the RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for fatigue analysis of RCS components involves calculating the cumulative usage factor (CUF). The fatigue damage in the component caused by each thermal or pressure transient depends on the magnitude of the stresses caused by the transient. The CUF sums the fatigue damage resulting from each transient. The design criterion requires that the CUF not exceed 1.0. The applicant indicated that the Thermal Fatigue Monitoring Program monitors the design transients at VCSNS. In RAI 4.3-1, the staff requested that the applicant provide the following information for each of the transients monitored at VCSNS:

states that the NRC accepted the VCSNS Tendon Surveillance Program based on the proposed Rev. 3 of RG 1.35.

LRA Section 4.5 indicates that programmatic controls are used to ensure that the reactor building tendons are capable of performing their design function. The LRA states that the reactor building tendons are a TLAA, and VCSNS will utilize 10 CFR 54.21(c)(1)—Option (iii) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The LRA also states that Chapter X.S1, "Concrete Containment Tendon Prestress", of NUREG-1801, applies to these facilities that adopt 10 CFR 54.21(c)(1)—Option (iii) for containment tendon prestress.

LRA Appendix B.3.3, Tendon Surveillance Program, states that the Tendon Surveillance Program is consistent with X.S1, "Concrete Containment Tendon Prestress", in NUREG-1801. A brief history of the Tendon Surveillance Program is provided which describes when the various tendon surveillance were performed and some of the significant observations. Several important observations noted in Appendix B.3.3 and Section 4.5 of the LRA are: (1) tendon wire relaxation force losses greater than predicted during design, (2) retensioning of vertical tendons were required because the tendon forces would be below the technical specifications minimum values prior to the following period surveillance, (3) VCSNS expects that future retensioning will be needed before 60 years of operation, and (4) substantial amount of water in-leakage into the auxiliary building tendon sump area has occurred.

### 4.5.2 Staff Evaluation

As reported in Appendix B.3.3.1 of the LRA, test results from the first three surveillances indicated that the wire relaxation force losses in the tendon system were greater than the force losses predicted during design (resulting in lower measured prestressing forces). Therefore, in June 1988, the predicted wire relaxation force losses were increased from 8.5 percent to 12.5 percent. Then in the fourth period (10th year) tendon surveillance, the vertical tendons were retensioned because the previous surveillance data indicated that the vertical tendon forces would be below the technical specifications minimum prior to the fifth period surveillance. Although the fifth period (15th year) and sixth period (20th year) tendon surveillance have been completed, no information was provided regarding the comparison of the measured tendon forces to the predicted lower limit at the 15th and 20th year tendon surveillance. LRA Section 4.5 indicates that, based on trending data and results from previous surveillance, VCSNS does not currently expect the tendons to provide adequate prestress for 60 years without future retensioning of various members.

In order to make a reasonable assessment regarding the effectiveness of the TLAA, the staff requested, in RAI 4.5-1, that the applicant provide the following information:

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

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

In its response to RAI 4.5-1, the applicant stated –

> (a) Plots of the measured lift-off forces and trend lines, along with the minimum required values for the three sets of tendons (vertical, horizontal, and dome) are provided in Attachment XII. Guidance of IN 99-10 has also been implemented at VCSNS.
>
> (b) Based on elongation tests performed at Leheigh University for VCSNS tendon wire samples, it was found that stress relaxation of the tendons was not linearly proportional to temperature as originally projected based on manufacturer data. Therefore, stress relaxation was increased from 8.5% to 12.5% based on these tests. SCE&G is not aware of any unique characteristics of the VCSNS tendons or containment design that would cause this to occur, nor operating experience of similar behavior.

The staff review of the plots of the measured lift-off forces and trend lines, and the comparison to the minimum required values for the vertical, horizontal, and dome tendons, demonstrate that the approach being used is consistent with the TLAA AMP X.S1, Tendon Surveillance Program, identified in NUREG-1801. On this basis, the TLAA for concrete containment tendon prestress at VCSNS is in accordance with 10 CFR 54.21(c)(1)—Option (iii). Conformance to the guidance of IN 99-10 has been confirmed by VCSNS.

A description was provided by VCSNS which explains why the tendon wire relaxation values, greater than those used in design, occurred at VCSNS. Regardless of the cause, the staff concludes that this aging effect will be adequately managed by implementation of the VCSNS Tendon Surveillance Program.

### 4.5.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation for the concrete containment tendon prestress TLAA. Therefore the staff has reasonable assurance that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

UNDERLINE PROBLEM

## 4.6 Containment (Reactor Building) Liner Plate, Metal Containment, and Penetration Fatigue Analysis

### 4.6.1 Containment (reactor building) Liner

#### *4.6.1.1 Summary of Technical Information in the Application*

The applicant stated that the reactor building liner at VCSNS provides an essentially leak-tight membrane on the inside face of the prestressed concrete reactor building that is designed to contain airborne radioactive particles and gases due to postulated accidents, such as a LOCA.

The staff concurs that SWIS settlement is a TLAA for VCSNS. In addition, the staff finds it acceptable for the applicant to address SWIS settlement using Option (ii) of 10 CFR 54.21(c)(1) by revising existing calculations to account for the period of extended operation to 60 years. The staff notes that no description of the analytical methodology or summary of the results utilized in the TLAA calculation have been provided in the LRA. During the AMR inspection (August 18-22, 2003; IR 50-395/03-08, dated September 29, 2003), the staff reviewed numerical calculation demonstrating that changing from a 40-year operating life to a 60-year operating life has no impact on the conclusions reached in the original calculation, namely that maximum predicted sublayer fill compaction will be about 2 inches.

VCSNS has committed in the LRA to a Service Water Structures Survey Monitoring Program and an Under Water Inspection Program (SWIS and SWPH). The Service Water Structures Survey Monitoring Program is an AMP which monitors any vertical or horizontal movement associated with settlement of the SWIS, SWPH, electrical duct banks, and service water intake line "A." The survey monitoring data is reviewed by VCSNS Design Engineering to ensure that settlements remain within established criteria. The Underwater Inspection Program (SWIS and SWPH) is an AMP which visually inspects the interior length of the intake tunnel, survey monitoring masts, trash racks, access ladder and east end wing walls. 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. The staff evaluations of the Service Water Structures Survey Monitoring Program and the Under water Inspection Program (SWIS and SWPH) as AMPs are presented in SER Sections 3.5.2.3.10 and 3.5.2.3.11, respectively.

The staff considers the VCSNS Service Water Intake Structure Settlement TLAA performed in accordance with Option (ii) of 10 CFR 54.21(c)(1) to be acceptable.

*4.7.4.3 Conclusions*

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), that, for the Service Water Intake Structure Settlement TLAA, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the FSAR Supplement contains an appropriate summary description of the Service Water Intake Structure Settlement TLAA evaluation for the period of extended operation, as reflected in the license condition. Therefore, the staff has reasonable assurance that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1).

## 4.8 Conclusion for Time-Limited Aging Analyses

On the basis of its review of the TLAAs, the staff concludes that actions have been identified and have been or will be taken with respect to TLAAs that have been identified to require review under 10 CFR 54.21(c) such that there is reasonable assurance that the activities authorized by a renewed license will continue to be conducted in accordance with the CLB, as required by 10 CFR 54.29(a).

# 5 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) will review the 10 CFR Part 54 portion of the V.C. Summer license renewal application. The ACRS Subcommittee on Plant License Renewal will continue its detailed review of the LRA after this report is issued. South Carolina Electric and Gas Company (SCE&G), and the staff will meet with the subcommittee committee to discuss issues associated with the review of the LRA. After the ACRS completes its review of the V.C. Summer LRA and SER, the full committee will issue a report discussing the results of its review. This report will be included in an update to this SER. The staff will address any issues and concerns identified in that report.

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# APPENDIX A: COMMITMENTS FOR LICENSE RENEWAL

systems

During the review of the VCSNS LRA by the NRC staff, the applicant made commitments related to aging management programs (AMPs) to manage aging effects of structures and components (~~SCs~~) prior to the period of extended operation. The following table lists these commitments, along with the implementation schedule and the source of the commitment

SSCs

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
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| **No.** | **Commitment** | **FSAR Supp. Location (LRA App. A)** | **Implementation Schedule** | **Source** |
| 8 | The following enhancement will be incorporated into the Fire Protection Program prior to the period of extended operations. 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. | 18.2.15.1, Mechanical | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.5; Response to RAI B.1.5-1 of extended operations. Sprink ? |
| 9 | The following enhancement will be incorporated into the Fire Protection Program prior to the period of extended operations. 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 of a representative sample of these stagnant sections of piping will be conducted at 10-year intervals thereafter. | 18.2.15.1, Mechanical | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.5; Response to RAI B.1.5-1 |
| 10 | 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. | 18.2.15.1, Mechanical | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.5 |
| 11 | 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.In addition to the visual inspection of in-scope, passive fuse holders on a 10-year periodicity for indication of age related degradation, the metallic fuse clip portion of the in-scope, passive fuse holders that are found to be susceptible to age related degradation, will receive a continuity check or will undergo thermography or other appropriate test on a representative sample basis to assure the metallic fuse clip is still making a good connection. | 18.2.18, Non-EQ Insulated Cables and Connections Inspection Program | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.9 Response to RAI 3.6-1 |
| 12 | The Aging Management Programs for cracking of the Core Support Pads and Bottom Head Penetrations include the Alloy 600 Aging Management Program, Chemistry Program, as well as the In-Service (ISI) Plan. ISI inspections are done in accordance with the ASME code requirements. VCSNS is active in industry groups specifically EPRI and WOG. New developments will be reviewed and if deemed appropriate incorporated into the aging management of the Core Support Pads and Bottom Head Penetrations. | 18.2.19, In-Service Inspection (ISI) Plan | Prior to the end of the current operating license term. | Response to RAI 3.1.2.2.9-2 |

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
|---|---|---|---|---|
| No. | Commitment | FSAR Supp. Location (LRA App. A) | Implementation Schedule | Source |
| 13 | Inspections for Mechanical Components will manage the relevant aging effects for mechanical components constructed of carbon steel, low alloy steel, and other susceptible materials. These inspections will follow the same frequency as Maintenace Rule Inspections (five years) and the baseline inspection would occur within five years of obtaining the new license. Based upon the results of these inspections, or any new industry experience, the frequency may increase. | 18.2.20 Inspections for Mechanical Components | Every five years with baseline inspection within five years of obtaining the new license | LRA Appendix B Section B.2.11, Response to RAI B.2.11-4 |
| 14 | 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 will be performed prior to the period of extended operation. | 18.2.21, Liquid Waste System Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.3 |
| 15 | The following enhancements will be incorporated into the Maintenance Rule Structures Program prior to the period of extended operation. (1) - Future inspections will add: North Berm, Electrical Manhole EMH-2 interior inspection, Inaccessible Areas when exposed by excavation, Flood Barrier Seals for Control, Intermediate, 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. (2) - Groundwater chemical analyses will include: pH, Sulfates and Chlorides. Groundwater chemical analyses will be used to monitor changes in aggressiveness of the below grade environment. | 18.2.22, Maintenance Rule Structures Program | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.18; Response to RAI 3.5-22 |
| 16 | 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 will be performed prior to the period of extended operation. | 18.2.26, Reactor Building Cooling Unit Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.5 |

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
|---|---|---|---|---|
| No. | Commitment | FSAR Supp. Location (LRA App. A) | Implementation Schedule | Source |
| 17 | The Reactor Vessel Internals Inspection will be consistent with XI.M16, *PWR Vessel Internals*, as identified in NUREG-1801. The program details have not been developed. VCSNS will follow industry initiatives and will have the program in place prior to the period of extended operation.<br><br>With respect to cracking due to SCC & IASCC, staff approved recommendations of the industry initiatives applicable to inspection of vessel internals will be implemented. It is the intent of VCSNS to follow staff approved industry initiatives for these inspections.<br><br>With respect to IASCC, and loss of fracture toughness due to neutron irradiaation embrittement of the RV Interval components, VCSNS will follow industry initiatives develop a reactor vessel internals inspection program which will be in place prior to the period of extended operation. It is the intent of VCSNS to follow industry initiatives for this inspections. | 18.2.28, Reactor Vessel Internals Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.4; Response to RAI B.2.4-2, B.2.4-4, 3.1.2.2.12-1 |
| 18 | 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. It is the intent of VCSNS to follow industry initiatives for these inspections. | 18.2.28, Reactor Vessel Internals Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.4; Response to RAIs 3.1.2.2.4-1, 3.1.2.4.4-1, 3.1.2.4.4-2, B.2.4-2, and B.2.4-4 |
| 19 | The following enhancement will be incorporated into the Reactor Vessel Surveillance Program prior to the period of extended operation. 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. | 18.2.29, Reactor Vessel Surveillance Program | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.24 |
| 20 | A program will be established at the end of RF-14 to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the Reactor Vessel will be monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. This program may be supplemented or revised by using alternative dosimetry or other effective neutron fluence monitoring techniques during the period of extended operation. | 18.2.29, Reactor Vessel Surveillance Program | RF-14 | Response to RAI B.1.24-1 |

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* SEE COVER LETTER RC-03-0227

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
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| No. | Commitment | FSAR Supp. Location (LRA App. A) | Implementation Schedule | Source |
| 21 | The Service Air System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801. The Service Air System Inspection will be performed prior to the period of extended operation. | 18.2.30, Service Air System Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.6 |
| 22 | The following enhancements will be incorporated into the Service Water Pond Dam Inspection Program. (1) - Scope - North Dam piezometers will be added. (2) - Parameters Monitored / Inspected - Water level. (3) - Monitoring and Trending - Inspections will be made every 5-years concurrent with the RG 1.127 inspections. (4) - Acceptance Criteria - Nominal elevation of adjacent Service Water Pond and Monticello Reservoir. | 18.2.31, Service Water Pond Dam Inspection Program | Prior to the end of the current operating license term. | LRA Appendix B, Section B.1.21 |
| 23 | 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 will be scheduled at or near the end of the second period of the fourth ISI interval.<br><br>VCSNS will evaluate the small-bore class 1 piping with a methodology that is approved by the Staff. The present approved methodology is to perform destructive examinations of small-bore piping. The approved method will be incorporated into the Small Bore Class 1 Piping Inspection. | 18.2.34, Small Bore Class 1 Piping Inspection | Before the end of the second period of the fourth ISI interval. | LRA Appendix B, Section B.2.7; Response to RAI B.2.1-1 |
| 24 | The Waste Gas System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801. The Waste Gas System Inspection will be performed prior to the period of extended operation. | 18.2.39, Waste Gas System Inspection | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.8 |
| 25 | 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. The Heat Exchanger Inspections will be performed prior to the period of extended operation. | 18.2.40, Heat Exchanger Inspections | Prior to the end of the current operating license term. | LRA Appendix B, Section B.2.12 |
| 26 | The Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria Commodities is a new one-time inspection that will detect and characterize loss of material due to general, crevice, and pitting corrosion resulting from exposure to an unmonitored and uncontrolled water environment. The Area Based Inspections for Refined 10 CFR 54.4(a)(2) Criteria commodities will be performed prior to the period of extended operation. | 18.2.42. | Prior to the end of the current operating license term. | App. B Sec. B.2.13 (Supplement to LRA) |
| 27 | This is a new program. A summary description of the X1.E2 GALL type program was provided. In this program, calibration results or fidings of surveillance testing programs will be used to identify the potential exixtence of aging degradation. This program applies to the in-scope instrumentation cables that are included in the circuit during loop calibrations. | 18.2.43. | Prior to the end of the current operating license term. | Response to RAI 3.6-2 |

findings

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
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| **No.** | **Commitment** | **FSAR Supp. Location (LRA App. A)** | **Implementation Schedule** | **Source** |
| 28 | VCSNS will establish a GALL type program for relevant, non-EQ, in-scope I&C cables with sensitive, low-level signals for the NI and RM systems. Implementation of an alternate program will be considered, when appropriate, for low signal level NI and RM circuit cables without loop calibrations, after the industry finalizes the approach. | 18.2.44. | Prior to the end of the current operating license term. | Response to RAI 3.6-2 |
| 29 | VCSNS recognizes the potential uncertainties involved with water treeing, even with ducts that are sloped to preclude moisture accumulation, and will create a program consistent with NUREG-1801 section XI.E3. The VCSNS program described herein will result in a 10-year test interval by an appropriate industry approved testing method selected to validate the satisfactory condition of the cable insulation and to give some assurance of the remaining life of the cable, while not damaging the cable itself. The specific type of test performed will be determined prior to the initial test. The 10-year interval will commence prior to the start of the period of extended operation. | 18.2.45. | Prior to the end of the current operating license term. | Response to RAI 3.6-3 |
| 30 | Additional analyses are required to calculate Charpy Upper-Shelf Energy for 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. The capsule will be tested and will provide bounding data for the EOL fluence of 54 EFPY. | 18.3.1.1, Upper-Shelf Energy | Prior to the end of the current operating license term. | LRA Section 4.2.1; Response to RAI 4.2.2.1-1 |
| 31 | 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 50.61. The LTOP analysis will be done as part of this calculation revision. | 18.3.1.3, Pressure-Temperature (P-T) Limits | Prior to the end of the current operating license term. | LRA Section 4.2.3; Response to RAI 4.2.2.3-1 |

| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
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| No. | Commitment | FSAR Supp. Location (LRA App. A) | Implementation Schedule | Source |
| 32 | 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.<br><br>For the NUREG/CR-6260 locations, VCSNS will evaluate the Fatigue Environmental Effects prior to the period of extended operation. VCSNS will evaluate the fatigue usage for components with a methodology that is approved by the Staff. The present approved methodology is to use the correlations contained in NUREG/CR-6583, for Carbon and Low-Alloy Steels and NUREGICR-5704, for Austenitic Stainless Steels. Component CUF will be maintained below 1.0. | 18.3.2.1, ASME Section III, Class 1 | Prior to the end of the current operating license term. | LRA Section 4.3.1; Response to RAI 4.3.1-4 and 4.3.1-5 |
| 33 | 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. | 18.3.2.2, Leak-Before-Break Analyses | Prior to the end of the current operating license term. | LRA Section 4.7.2 |
| 34 | [RC Loop 'B' hot leg sampling portion of SS.] The present sampling method seldom uses loop sampling. VCSNS will administratively limit of activities on the "B" RCS loop sampling line in order to account for 60 years of plant operation. | 18.3.2.3, ASME Section III, Class 2 and 3 Piping Fatigue | Prior to the end of the current operating license term. | LRA Section 4.3.2; Response to RAI 4.3.2-1 |
| 35 | 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. Components not meeting a 60 year qualified life will be replaced prior to expiration of qualified life. | 18.3.3, Environmental Qualification (EQ) | Prior to the end of the current operating license term. | LRA Appendix B, Section B.3.1 |
| 36 | As appropriate, station documents will be revised or established, implemented, and maintained to cover the aging management programs and activities described in Chapter 18. | Appendix A Prefix, FSAR Supplement | Varies by program and activity | LRA Appendix A, page 1. Various RAI responses |
| 37 | At VCSNS, the Boraflex neutron absorbing sheets will be replaced with Boral neutron absorbing sheets prior to the Refueling Outage 14 (September 2003). | N/A - Aging management is not required for the new components. | ~~RF-14, September 2003~~ Completed April, 2003 | LRA Table 3.3-1, AMR items 9, 12 |

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| Appendix A - VCSNS Commitment List Associated with Renewal of the Operating License | | | | |
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| No. | Commitment | FSAR Supp. Location (LRA App. A) | Implementation Schedule | Source |
| 38 | VCSNS is developing a process which will be implemented to capture the LRA methodology and guidance for use during the period of extended operation to satisfy the requirements of 10 CFR 54.35. Existing plant programs and procedures (associated with aging management) will be revised and/or enhanced to identify those commitments (governed by the license / CLB) which cannot be altered without prior review against the LRA criteria. New ""one-time inspection"" aging management programs will be developed in accordance with the LRA, incorporating the commitment process identified above. | N/A - Implementation activity. | Prior to the end of the current operating license term. | Response to RAI 2.1-2 |
| 39 | Plant procedures which impact "control of facility changes", including modifications and documentation, will be reviewed to determine an acceptable screening review process against the 10 CFR 54 requirements to ensure consistency with the LRA methodology and guidance. | N/A | Prior to the end of the current operating license term. | Response to RAI 2.1-2 |
| 40 | To support Items 50 and 51 above, a License Renewal DBD will be developed as a guidance document which can be used for all future plant procedure, documentation and modification changes to ensure consistency with 10 CFR 54. | N/A | Prior to the end of the current operating license term. | Response to RAI 2.1-2 |
| 41 | All Technical Reports, which have been developed to substantiate the LRA submittal, are filed as permanent records and will be available for future reference and/or update. | N/A | Prior to the end of the current operating license term. | Response to RAI 2.1-2 |

38 and 39

# APPENDIX B: CHRONOLOGY

This appendix provides a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and South Carolina Electric & Gas Company (SCE&G) and other correspondence regarding the NRC staff's review of the Virgil C. Summer Nuclear Station (VCSNS), for license renewal application (LRA) (Docket Nos. 50-395).

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| August 6, 2002 | In a letter (signed by S.A. Byrne), SCE&G submitted its LRA for Virgil C. Summer Nuclear Station to the NRC. |
| August 20, 2002 | In a letter (signed by G.A. Suber), NRC confirmed a telephone conversation concerning the maintenance of reference material for the Virgil C. Summer Nuclear Station LRA. |
| August 26, 2002 | In a letter (signed by P. Kuo), NRC informed SCE&G that the NRC received the Virgil C. Summer Nuclear Station license renewal application on August 6, 2002, and enclosed a copy of the notice related to the application that was sent to the Office of Federal Register for publication. |
| August 27, 2002 | NRC announced the availability of License Renewal Application for the Virgil C. Summer Nuclear Station. |
| September 12, 2002 | In a letter, R. Auluck noticed a meeting to provide the NRC staff an overview of the Virgil C. Summer Nuclear Station LRA, on September 24, 2003. |
| September 12, 2002 | In a letter SCE&G provided the NRC additional information on Section 2 of the Virgil C. Summer Nuclear Station, LRA. |
| September 27,2002 | In a letter (signed by P. Kuo), the NRC informed the SCE&G that tha NRC staff had determinated that SCE&G had submitted sufficient information that was complete and acceptable for docketing, proposed review schedule, and opportunity for hearing. |
| October 23,2002 | In a letter (signed by P. Kuo), the NRC informed of its intent to prepare an Environmental Impact Statement and conduct the scoping process for the LRA of VCSNS. |
| October 24, 2002 | In a meeting summary (signed by R. Auluck), NRC summarized the September 24, 2002, meeting with SCE&G regarding the VCSNS license renewal application. |
| November 27, 2002 | NRC announced to hold a public meeting on December 11, 2002, on Summer Nuclear Station License Renewal. |

RG 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

**Westinghouse Topical Reports (WCAP)**

WCAP-10456, *The Effects of Thermal Aging on Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems, November 1983.*

WCAP-12866, *Bottom Mounted Instrumentation Flux Thimble Wear*, 1991.

WCAP-14535A, *Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination*, November 1996.

WCAP-14574, *License Renewal Evaluation: Aging Management Evaluation for Pressurizers.*

WCAP-14574-A, *License Renewal Application: Aging Management Evaluation for Pressurizers*, December 2000.

WCAP-14575-A, *Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components*, August 1996.

WCAP-14577, Revision 1-A, *License Renewal Application: Aging Management Evaluation for Reactor Internals*, March 2001.

WCAP-14901, Revision 0, *Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group*, July 1997.

# APPENDIX D: PRINCIPAL CONTRIBUTORS

| NAME | RESPONSIBILITY |
|---|---|
| R. Auluck | Sr. Project Manager |
| F. Akstulewicz | Management Supervision |
| T. Chan | Management Supervision |
| P. Y. Chen | Technical Support |
| S. Coffin | Management Supervision |
| K. Corp | Program Support |
| K. Coyne | Quality Assurance |
| Z. Cruz | Technical Support |
| B. Elliot | Materials Engineering |
| J. Fair | Mechanical Engineering |
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| A. Hodgdon | Legal Counsel |
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| ~~S. Jones~~ | Plant Systems |
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| ~~M. Khanna~~ | Materials Engineering |
| J. Knox | Electrical Engineering |
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| A. Lee | Mechanical Engineering |
| S. Lee | Management Supervision |
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| J. Pulsipher | Plant Systems |
| J. Rajan | Civil Engineering |
| J. Raval | Plant Systems |
| D. Shum | Plant Systems |
| J. Strnisha | Mechanical Engineering |
| R. Subbaratnam | Project Manager |
| D. Terao | Management Supervision |
| D. Thatcher | Management Supervision |

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|---|---|
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| S. Weerakkady | Management Supervision |
| C. Wu | Mechanical Engineering |
| H. Wang | Civil Engineering |

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