V. C. Summer Nuclear Station, Units 2 and 3
COL Application
Part 2, FSAR

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NUCLEAR REGULATORY COMMISSION GENERAL DESIGN CRITERIA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.2.1 SEISMIC CLASSIFICATION

Add the following text to the end of DCD Subsection 3.2.1.

STD SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD.

The nonsafety-related structures, systems, and components outside the scope of the DCD are classified as non-seismic (NS).

3.2.2 AP1000 CLASSIFICATION SYSTEM

Add the following text to the end of DCD Subsection 3.2.2.

STD SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD.

3.3 WIND AND TORNADO LOADINGS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.3.1.1 Design Wind Velocity

Add the following text to the end of DCD Subsection 3.3.1.1.

VCS COL 3.3-1
VCS COL 3.5-1

The wind velocity characteristics for the VCSNS site are provided in FSAR Subsection 2.3.1.3.1. These values are bounded by the design wind velocity values given in DCD Subsection 3.3.1.1 for the AP1000 plant.

3.3.2.1 Applicable Design Parameters

Add the following text to the end of DCD Subsection 3.3.2.1.

VCS COL 3.3-1
VCS COL 3.5-1

The tornado characteristics for the VCSNS site are provided in FSAR Subsection 2.3.1.3.2. These values are bounded by the tornado design parameters given in DCD Subsection 3.3.2.1 for the AP1000 plant.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

Add the following text to the end of DCD Subsection 3.3.2.3.

STD COL 3.3-1
VCS COL 3.5-1

Consideration of the effects of wind and tornado due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single unit.

3.3.3 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.3.3.

VCS COL 3.3-1 The VCSNS site satisfies the site interface criteria for wind and tornado (see DCD Subsections 3.3.1.1, 3.3.2.1, 3.3.2.3, and 3.5.4) and will not have a tornado initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components.

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.4 WATER LEVEL (FLOOD) DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.4.1.3 Permanent Dewatering System

Add the following text to the end of DCD Subsection 3.4.1.3.

VCS COL 3.4-1 No permanent dewatering system is required because site groundwater levels are 20 feet below site grade level as described in FSAR Subsection 2.4.12.5.

3.4.3 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.4.3.

VCS COL 3.4-1 VCSNS site-specific water levels provided in FSAR Section 2.4 satisfy the AP1000 site interface requirements described in DCD Subsection 2.4.

3.5 MISSILE PROTECTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.5.1.3 Turbine Missiles

Add the following text to the end of DCD Subsection 3.5.1.3.

STD SUP 3.5-1 The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in DCD Subsection 10.2.2, the probability of generation of a turbine missile (or P1 as identified in SRP 3.5.1.3) is less than 1 x 10^{-5} per year. This missile generation probability (P1) combined with an unfavorable orientation P2xP3 conservative product value of 10^{-2} (from SRP 3.5.1.3) results in a probability of unacceptable damage from turbine missiles (or P4 value) of less than 10^{-7} per year per plant which meets the SRP 3.5.1.3 acceptance criterion and the guidance of Regulatory Guide 1.115. Thus, neither the orientation of the side-by-side AP1000 turbines nor the separation distance is pertinent to meeting the turbine missile generation acceptance criterion. In addition, the shield building and auxiliary building walls, roofs, and floors, provide further conservative, inherent protection of the safety-related SSCs from a turbine missile.

Add the following text to the end of DCD Subsection 3.5.1.3.

VCS SUP 3.5-1 The potential for a turbine missile from Unit 1 has been considered.

The Unit 1 monoblock turbine rotor design eliminates the brittle fracture mode and the probability for a turbine wheel burst and missile generation at normal operating speeds. The maximum attainable turbine shaft overspeed is 217% whereas the low pressure monoblock design overspeed capability based on material properties and rotor design is 221% for LPA and 222% for LPB. The annual probability of complete turbine control failure is in the range of 10^{-8}.

Based on this information as well as the separation distance of Unit 1 from Units 2 and 3, the protection provided by the AP1000 reinforced concrete shield building, the auxiliary building walls and roofs, the guidance of Regulatory Guide 1.115 is satisfied.

STD SUP 3.5-2 The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

3.5.1.5 Missiles Generated by Events Near the Site

Add the following text to the end of DCD Subsection 3.5.1.5.

VCS COL 3.5-1
VCS COL 3.3-1

The plant entry building, service building, guard house, warehouse, buildings and structures related to water services, diesel-driven fire pump/enclosure, and miscellaneous structures are common structures at a nuclear power plant. Therefore, any missiles resulting from a tornado initiated failure are not more energetic than tornado missiles postulated for design of the AP1000.

Postulated explosion events on or near the VCSNS site are discussed and evaluated in accordance with Regulatory Guide 1.91 in FSAR Subsection 2.2.3. The effects of these postulated events on Units 2 and 3 safety-related components are insignificant. No events were identified that had a probability of occurrence greater than 10^{-7} per year or potential consequences serious enough to affect the safety of Units 2 and 3. Additionally, the overpressure criteria of Regulatory Guide 1.91 were not exceeded; therefore, the effects of postulated missiles were not required to be considered.

3.5.1.6 Aircraft Hazards

Add the following text to the end of DCD Subsection 3.5.1.6.

VCS COL 3.5-1
VCS COL 3.3-1

Aircraft and airway hazards are discussed in FSAR Subsection 2.2.2.7.6.

3.5.4 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.5.4.

VCS COL 3.5-1

The VCS site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1, and 3.3.2.3) and will not have a tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.3.3).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.6 PROTECTION AGAINST THE DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.6.4.1 Pipe Break Hazard Analysis

Replace the last paragraph in DCD Subsection 3.6.4.1 with the following text.

STD COL 3.6-1 The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate.

A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.

The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, item 8).

This COL item is also addressed in Subsection 14.3.3.

3.6.4.4 Primary System Inspection Program for Leak-before-Break Piping

Replace the first paragraph of DCD Subsection 3.6.4.4 with the following text.

STD COL 3.6-4 Alloy 690 is not used in leak-before-break piping. No additional or augmented inspections are required beyond the inservice inspection program for leak-before-break piping. An as-built verification of the leak-before-break piping is required to verify that no change was introduced that would invalidate the conclusion reached in this subsection.

3.7 SEISMIC DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add Subsection 3.7.1.1.1 as follows:

3.7.1.1.1 Design Ground Motion Response Spectra

VCS SUP 3.7-3 A comparison of the site-specific ground motion response spectra (GMRS) to the hard rock high frequency (HRHF) spectra and Certified Seismic Design Response Spectra (CSDRS) is provided in Figures 2.0-201 and 2.0-202. The CSDRS spectra are also shown in DCD Figures 3I.1-1 and 3I.1-2.

The horizontal and vertical GMRS were developed at the top of a hypothetical outcrop of competent material at the elevation of the nuclear island basemat as described in Subsection 2.5.2.5. Bedrock at the basemat elevation has a shear wave velocity that exceeds 9,000 feet per second. Therefore, rock motion is not modified to account for effects of local soft rock or soil profiles on seismic wave propagation.

The horizontal GMRS exceeds the CSDRS at frequencies of about 15 to 80 hertz. Horizontal peak ground acceleration (PGA) at 100 hertz is approximately 0.23g. The vertical GMRS exceeds the CSDRS at frequencies of approximately 20 to 80 hertz. Vertical PGA at 100 hertz is roughly 0.22g. These high frequency exceedances are within those of the HRHF spectra.

High frequency seismic input is generally considered to be non-damaging (DCD, Appendix 3I, Reference 1). The high frequency exceedances were evaluated as acceptable as discussed in DCD Chapter 3, Appendix 3I.

3.7.2.12 Methods for Seismic Analysis of Dams

Add the following text to the end of DCD Subsection 3.7.2.12.

VCS COL 3.7-1 The evaluation of existing and new dams whose failure(s) could affect the AP1000 design flood level specified in DCD Subsection 2.4.1.2 is included in FSAR Subsection 2.4.4. This evaluation demonstrates that the VCSNS site is not subject to flooding from dam failures.

3.7.4.1 Comparison with Regulatory Guide 1.12

Add the following text to the end of DCD Subsection 3.7.4.1.

STD SUP 3.7-1 Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operation and shutdown in accordance with Regulatory Guide 1.12.

3.7.4.2.1 Triaxial Acceleration Sensors

Add the following text to the end of DCD Subsection 3.7.4.2.1.

STD COL 3.7-5 A free-field sensor will be located and installed to record the ground surface motion representative of the site. It will be located such that the effects associated with surface features, buildings, and components on the recorded ground motion will be insignificant. The trigger value is initially set at 0.01g.

3.7.4.4 Comparison of Measured and Predicted Responses

Add the following text to the end of DCD Subsection 3.7.4.4.

STD COL 3.7-2 Post-earthquake operating procedures utilize the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in Regulatory Guides 1.166 and 1.167. A response spectrum check up to 10Hz will be based on the foundation instrument. The cumulative absolute velocity will be calculated based on the recorded motions at the free field instrument. If the operating basis earthquake ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner.

In addition, the procedures address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).

3.7.4.5 Tests and Inspections

Add the following text to the end of DCD Subsection 3.7.4.5.

STD SUP 3.7-2 Installation and acceptance testing of the triaxial acceleration sensors described in DCD Subsection 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in DCD Subsection 3.7.4.2.2 is completed prior to initial startup.

3.7.5 COMBINED LICENSE INFORMATION

3.7.5.1 Seismic Analysis of Dams

Add the following text to the end of DCD Subsection 3.7.5.1.

VCS COL 3.7-1 FSAR Subsection 2.4.4 presents an analysis and evaluation of dam failures. This evaluation demonstrates that the VCSNS site is not subject to flooding from dam failures.

3.7.5.2 Post-Earthquake Procedures

STD COL 3.7-2 This COL Item is addressed in Subsection 3.7.4.4.

3.7.5.3 Seismic Interaction Review

Replace DCD Subsection 3.7.5.3 with the following text.

STD COL 3.7-3 The seismic interaction review will be updated for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is completed prior to fuel load.

3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

Replace DCD Subsection 3.7.5.4 with the following text.

STD COL 3.7-4 The seismic analyses described in DCD Subsection 3.7.2 will be reconciled for detailed design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of DCD Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. This reconciliation will be completed prior to fuel load.

3.7.5.5 Free Field Acceleration Sensor

STD COL 3.7-5 This COL Item is addressed in Subsection 3.7.4.2.1.

3.8 DESIGN OF CATEGORY I STRUCTURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.8.3.7 In-Service Testing and Inspection Requirements

Replace the existing DCD statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.4.7 Testing and In-Service Inspection Requirements

Replace the existing DCD final statement of the subsection with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.5.1 Description of the Foundations

Add the following text after paragraph one of DCD Subsection 3.8.5.1.

STD SUP 3.8-1 The depth of overburden and depth of embedment are given in Subsection 2.5.4.

VCS COL 2.5-17 A sheet type waterproofing material will be used for both the horizontal and vertical surfaces under seismic Category I structures. The material will be qualified by test, with commercial grade dedication and lab testing to achieve a minimum coefficient of friction (COF) of 0.70.

3.8.5.7 In-Service Testing and Inspection Requirements

Replace the existing DCD first statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.6.5 Structures Inspection Program

STD COL 3.8-5 This item is addressed in Subsections 3.8.3.7, 3.8.4.7, 3.8.5.7, and 17.6.

3.8.6.6 Construction Procedures Program

Add the following to the end of DCD Subsection 3.8.6.6:

STD COL 3.8-6 Construction and inspection procedures for concrete filled steel plate modules address activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in DCD Subsection 3.8.4.8. The procedures will be made available to NRC inspectors prior to use.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.9.3.1.2 Loads for Class 1 Components, Core Support, and Component Supports

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STD COL 3.9-5 Add the following after the last paragraph under DCD subheading Request 3) and prior to DCD subheading Other Applications.

PRESSURIZER SURGE LINE MONITORING

General

The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements.

Subsequent AP1000 plants (after the first AP1000 plant) confirm that the heatup and cooldown procedures are consistent with the pertinent attributes of the first AP1000 plant surge line monitoring. In addition, changes to the heatup and cooldown procedures consider the potential impact on stress and fatigue analyses consistent with the concerns of NRC Bulletin 88-11.

The pressurizer surge line monitoring activities include the following methodology and requirements:

Monitoring Method

The pressurizer surge line pipe wall is instrumented with outside mounted temperature and displacement sensors. The data from this instrumentation is supplemented by plant computer data from related process and control parameters.

Locations to be Monitored

In addition to the existing permanent plant temperature instrumentation, temperature and displacement monitoring will be included at critical locations on the surge line. The additional locations utilized for monitoring during the hot functional testing and the first fuel cycle (see Subsection 14.2.9.2.22) are selected based on the capability to provide effective monitoring.

Data Evaluation

Data evaluation is performed at the completion of the monitoring period (one fuel cycle). The evaluation includes a comparison of the data evaluation results with the thermal profiles and transient loadings defined for the pressurizer surge line, accounting for expected pipe outside wall temperatures. Interim evaluations of the data are performed during the hot functional testing period, up to the start of normal power operation, and again once three months worth of normal operating data has been collected, to identify any unexpected conditions in the pressurizer surge line.

3.9.3.4.4 Inspection, Testing, Repair, and/or Replacement of Snubbers

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Add the following text after the last paragraph of DCD Subsection 3.9.3.4.4:

STD SUP 3.9-3

a. Snubber Design and Testing

1. A list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position is included in Table 3.9-201.

2. The snubbers are tested to verify they can perform as required during the seismic events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. Production and qualification test programs for both hydraulic and mechanical snubbers are carried out by the snubber vendors in accordance with the snubber installation instruction manual required to be furnished by the snubber supplier. Acceptance criteria for compliance with ASME Section III Subsection NF, and other applicable codes, standards, and requirements, are as follows:

   - Snubber production and qualification test programs are carried out by strict adherence to the manufacturer's snubber installation and instruction manual. This manual is prepared by the snubber manufacturer and subjected to review for compliance with the applicable provisions of the ASME Pressure Vessel and Piping Code of record. The test program is periodically audited during implementation for compliance.

   - Snubbers are inspected and tested for compliance with the design drawings and functional requirements of the procurement specifications.

- Snubbers are inspected and qualification tested. No sampling methods are used in the qualification tests.

- Snubbers are load rated by testing in accordance with the snubber manufacturer's testing program and in compliance with the applicable sections of ASME QME-1-2007, Subsection QDR and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD.

- Design compliance of the snubbers per ASME Section III Paragraph NF-3128, and Subparagraphs NF-3411.3 and NF-3412.4.

- The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test. The functional parameters cited in Subparagraph NF-3412.4 are included in the snubber qualification and testing program. Other parameters in accordance with applicable ASME QME-1-2007 and the ASME OM Code will be incorporated.

- The codes and standards used for snubber qualification and production testing are as follows:

  – ASME B&PV Code Section III (Code of Record date) and Subsection NF.

  – ASME QME-1-2007, Subsection QDR and ASME OM Code, Subsection ISTD.

- Large bore hydraulic snubbers are full Service Level D load tested, including verifying bleed rates, control valve closure within the specified velocity ranges and drag forces/ breakaway forces are acceptable in accordance with ASME, QME-1-2007 and ASME OM Codes.

3. Safety-related snubbers are identified in Table 3.9-201, including the snubber identification and the associated system or component, e.g., line number. The snubbers on the list are hydraulic and constructed to ASME Section III, Subsection NF. The snubbers are used for shock loading only. None of the snubbers are dual-purpose or vibration arrestor type snubbers.

b. Snubber Installation Requirements

Installation instructions contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing that contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

STD COL 3.9-3

The description of the snubber preservice and inservice testing programs in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates the latest edition and addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

c. Snubber Preservice Examination and Testing

The preservice examination plan for applicable snubbers is prepared in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD, and the additional requirements of this Section. This examination is made after snubber installation but not more than 6 months prior to initial system preoperational testing. The preservice examination verifies the following:

1. There are no visible signs of damage or impaired operational readiness as a result of storage, handling, or installation.

2. The snubber load rating, location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3. Snubbers are not seized, frozen or jammed.

4. Adequate swing clearance is provided to allow snubber movements.

5. If applicable, fluid is to the recommended level and is not to be leaking from the snubber system.

6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial preservice examination and initial system preoperational tests exceeds 6 months, reexamination of Items 1, 4, and 5 is performed. Snubbers, which are installed incorrectly or otherwise fail to meet the above requirements, are repaired or replaced and re-examined in accordance with the above criteria.

A preservice thermal movement examination is also performed, during initial system heatup and cooldown. For systems whose design operating temperature exceeds 250ºF (121ºC), snubber thermal movement is verified.

Additionally, preservice operational readiness testing is performed on snubbers. The operational readiness test is performed to verify the parameters of ISTD 5120. Snubbers that fail the preservice operational readiness test are evaluated to determine the cause of failure, and are retested following completion of corrective action(s).

Snubbers that are installed incorrectly or otherwise fail preservice testing requirements are re-installed correctly, adjusted, modified, repaired or replaced, as required. Preservice examination and testing is re-performed on installation-corrected, adjusted, modified, repaired or replaced snubbers as required.

d. Snubber Inservice Examination and Testing

Inservice examination and testing of safety-related snubbers is conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD. Inservice examination is initially performed not less than two months after attaining 5 percent reactor power operation and is completed within 12 calendar months after attaining 5 percent reactor power. Subsequent examinations are performed at intervals defined by ISTD-4252 and Table ISTD-4252-1. Examination intervals, subsequent to the third interval, are adjusted based on the number of unacceptable snubbers identified in the current interval.

An inservice visual examination is performed on the snubbers to identify physical damage, leakage, corrosion, degradation, indication of binding, misalignment or deformation and potential defects generic to a particular design. Snubbers that do not meet visual examination requirements are evaluated to determine the root cause of the unacceptability, and appropriate corrective actions (e.g., snubber is adjusted, repaired, modified, or replaced) are taken. Snubbers evaluated as unacceptable during visual examination may be accepted for continued service by successful completion of an operational readiness test.

Snubbers are tested inservice to determine operational readiness during each fuel cycle, beginning no sooner than 60 days before the start of the refueling outage. Snubber operational readiness tests are conducted with the snubber in the as-found condition, to the extent practical, either in-place or on a test bench, to verify the test parameters of ISTD-5210. When an in-place test or bench test cannot be performed, snubber subcomponents that control the parameters to be verified are examined and tested. Preservice examinations are performed on snubbers after reinstallation when bench testing is used (ISTD-5224), or on snubbers where individual subcomponents are reinstalled after examination (ISTD-5225).

Defined test plan groups (DTPG) are established and the snubbers of each DTPG are tested according to an established sampling plan each fuel cycle. Sample plan size and composition is determined as required for the selected sample plan, with additional sampling as may be required for that sample plan based on test failures and failure modes identified. Snubbers that do not meet test requirements are evaluated to determine root cause of the failure, and are assigned to failure mode groups (FMG) based on the evaluation, unless the failure is considered unexplained or isolated. The number of unexplained snubber failures, not assigned to a FMG, determines the additional testing sample. Isolated failures do not require additional testing. For unacceptable snubbers, additional testing is conducted for the DTPG or FMG until the appropriate sample plan completion criteria are satisfied.

Unacceptable snubbers are adjusted, repaired, modified, or replaced. Replacement snubbers meet the requirements of ISTD-1600. Post-maintenance examination and testing, and examination and testing of repaired snubbers, is done to verify as acceptable the test parameters that may have been affected by the repair or maintenance activity.

Service life for snubbers is established, monitored and adjusted as required by ISTD-6000 and the guidance of ASME OM Code Nonmandatory Appendix F.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

Revise the third sentence of the third paragraph of DCD Subsection 3.9.6, and add information between the third and fourth sentences as follows:

STD COL 3.9-4 The edition and addenda to be used for the inservice testing program are administratively controlled; the description of the inservice testing program in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates the latest edition and addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

Revise the fifth sentence of the sixth paragraph of DCD Subsection 3.9.6 as follows:

STD COL 3.9-4 Alternate means of performing these tests and inspections that provide equivalent demonstration may be developed in the inservice test program ~~as described in subsection 3.9.8~~.

Revise the first two sentences of the final paragraph of DCD Subsection 3.9.6 to read as follows:

STD COL 3.9-4 A preservice test program, which identifies the required functional testing, is to be submitted to the NRC prior to performing the tests and following the start of construction. The inservice test program, which identifies requirements for functional testing, is to be submitted to the NRC prior to the anticipated date of commercial operation as described above.

Add the following text after the last paragraph of DCD Subsection 3.9.6:

Table 13.4-201 provides milestones for preservice and inservice test program implementation.

3.9.6.2.2 Valve Testing

Add the following prior to the initial paragraph of DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Valve testing uses reference values determined from the results of preservice testing or inservice testing. These tests that establish reference and IST values are performed under conditions as near as practicable to those expected during the IST. Reference values are established only when a valve is known to be operating acceptably.

Pre-conditioning of valves or their associated actuators or controls prior to IST testing undermines the purpose of IST testing and is not allowed. Pre-conditioning includes manipulation, pre-testing, maintenance, lubrication, cleaning, exercising, stroking, operating, or disturbing the valve to be tested in any way, except as may occur in an unscheduled, unplanned, and unanticipated manner during normal operation.

Add the following sentence to the end of the fourth paragraph under the heading “Manual/Power-Operated Valve Tests”:

STD COL 3.9-4 Stroke time is measured and compared to the reference value, except for valves classified as fast-acting (e.g., solenoid-operated valves with stroke time less than 2 seconds), for which a stroke time limit of 2 seconds is assigned.

Add the following paragraph after the fifth paragraph under the heading “Manual/ Power-Operated Valve Tests”:

STD COL 3.9-4 During valve exercise tests, the necessary valve obturator movement is verified while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position.

Insert new second sentence of the paragraph containing the subheading “Power-Operated Valve Operability Tests” in DCD Subsection 3.9.6.2.2 (immediately following the first sentence of the DCD paragraph) to read:

STD COL 3.9-4 The POVs include the motor-operated valves.

Add the following sentence as the last sentence of the paragraph containing the subheading “Power-Operated Valve Operability Tests” in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Table 13.4-201 provides milestones for the MOV program implementation.

Insert the following as the last sentence in the paragraph under the bulleted item titled “Risk Ranking” in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Guidance for this process is outlined in the JOG MOV PV Study, MPR-2524-A.

Insert the following text after the last paragraph under the sub-heading of “Power-Operated Valve Operability Tests” and before the sub-heading “Check Valve Tests” in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 **Active MOV Test Frequency Determination** — The ability of a valve to meet its design basis functional requirements (i.e. required capability) is verified during valve qualification testing as required by procurement specifications. Valve qualification testing measures valve actuator output capability. The actuator output capability is compared to the valve's required capability defined in procurement specifications, establishing functional margin; that is, that increment by which the MOV's actual output capability exceeds the capability required to operate the MOV under design basis conditions. DCD Subsection 5.4.8 discusses valve functional design and qualification requirements. The initial inservice test frequency is determined as required by ASME OM Code Case OMN-1, Revision 1

(Reference 202). The design basis capability testing of MOVs utilizes guidance from Generic Letter 96-05 and the JOG MOV Periodic Verification PV Program. Valve functional margin is evaluated following subsequent periodic testing to address potential time-related performance degradation, accounting for applicable uncertainties in the analysis. If the evaluation shows that the functional margin will be reduced to less than established acceptance criteria within the established test interval, the test interval is decreased to less than the time for the functional margin to decrease below acceptance criteria. If there is not sufficient data to determine test frequency as described above, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, until sufficient data exist to extend the test frequency. Appropriate justification is provided for any increased test interval, and the maximum test interval shall not exceed 10 years. This is to ensure that each MOV in the IST program will have adequate margin (including consideration for aging-related degradation, degraded voltage, control switch repeatability, and load-sensitive MOV behavior) to remain operable until the next scheduled test, regardless of its risk categorization or safety significance. Uncertainties associated with performance of these periodic verification tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) are established so as not to exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

Solenoid-operated valves (SOVs) are tested to confirm the valve moves to its energized position and is maintained in that position, and to confirm that the valve moves to the appropriate failure mode position when de-energized.

Other Power-Operated Valve Operability Tests — Power-Operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies.

Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the “baseline” performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (Reference 203 and Reference 204). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.
- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.
- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with Reference 203 and Reference 204, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.
- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when

maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.

- Guidance is included to address lessons learned from other valve programs specific to the AOV program.
- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

Insert the following paragraph as the last paragraph under the sub-heading of "Power-Operated Valve Operability Tests" (following the previously added paragraph) and just before the sub-heading "Check Valve Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Successful completion of the preservice and IST of MOVs, in addition to MOV testing as required by 10 CFR 50.55a, demonstrates that the following criteria are met for each valve tested: (i) valve fully opens and/or closes as required by its safety function; (ii) adequate margin exists and includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load-sensitive MOV behavior, and margin for degradation; and (iii) maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

Add the paragraph below as the last paragraph of FSAR Subsection 3.9.6.2.2 prior to the subheading "Check Valves Tests":

STD COL 3.9-4 The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.

Add the following new paragraph under the heading "Check Valves Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Preoperational testing is performed during the initial test program (refer to DCD Subsection 14.2) to verify that valves are installed in a configuration that allows correct operation, testing, and maintenance. Preoperational testing verifies that piping design features accommodate check valve testing requirements. Tests also verify disk movement to and from the seat and determine, without disassembly, that the valve disk positions correctly, fully opens or fully closes as expected, and remains stable in the open position under the full spectrum of system design-basis fluid flow conditions.

Add the following new last paragraphs under the subheading "Check Valve Exercise Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Acceptance criteria for this testing consider the specific system design and valve application. For example, a valve's safety function may require obturator movement in both open and closed directions. A mechanical exerciser may be used to operate a check valve for testing. Where a mechanical exerciser is used, acceptance criteria are provided for the force or torque required to move the check valve's obturator. Exercise tests also detect missing, sticking, or binding obturators.

When operating conditions, valve design, valve location, or other considerations prevent direct observation or measurements by use of conventional methods to determine adequate check valve function, diagnostic equipment and nonintrusive techniques are used to monitor internal conditions. Nonintrusive tests used are dependent on system and valve configuration, valve design and materials, and include methods such as ultrasonic (acoustic), magnetic, radiography, and use of accelerometers to measure system and valve operating parameters (e.g., fluid flow, disk position, disk movement, disk impact, and the presence or absence of cavitation and back-tapping). Nonintrusive techniques also detect valve degradation. Diagnostic equipment and techniques used for valve operability determinations are verified as effective and accurate under the PST program.

Testing is performed, to the extent practicable, under normal operation, cold shutdown, or refueling conditions applicable to each check valve. Testing includes effects created by sudden starting and stopping of pumps, if applicable, or other conditions, such as flow reversal. When maintenance that could affect valve performance is performed on a valve in the IST program, post-maintenance testing is conducted prior to returning the valve to service.

Add the following new paragraph under the heading “Other Valve Inservice Tests” following the Explosively Actuated Valves paragraph in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Industry and regulatory guidance is considered in development of the IST program for squib valves. In addition, the IST program for squib valves incorporates lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions.

3.9.6.2.3 Valve Disassembly and Inspection

Add the following paragraph as the new second paragraph of DCD Subsection 3.9.6.2.3:

STD COL 3.9-4 During the disassembly process, the full-stroke motion of the obturator is verified. Nondestructive examination is performed on the hinge pin to assess wear, and seat contact surfaces are examined to verify adequate contact. Full-stroke motion of the obturator is re-verified immediately prior to completing reassembly. At least one valve from each group is disassembled and examined at each refueling outage, and all the valves in each group are disassembled and examined at least once every eight years. Before being returned to service, valves disassembled for examination or valves that received maintenance that could affect their performance are exercised with a full- or part-stroke. Details and bases of the sampling program are documented and recorded in the test plan.

Add Subsections 3.9.6.2.4 and 3.9.6.2.5 following the last paragraph of DCD Subsection 3.9.6.2.3:

3.9.6.2.4 Valve Preservice Tests

STD COL 3.9-4 Each valve subject to inservice testing is also tested during the preservice test period. Preservice tests are conducted under conditions as near as practicable to those expected during subsequent inservice testing. Valves (or the control system) that have undergone maintenance that could affect performance, and valves that have been repaired or replaced, are re-tested to verify performance parameters that could have been affected are within acceptable limits. Safety and relief valves and nonreclosing pressure relief devices are preservice tested in accordance with the requirements of the ASME OM Code, Mandatory Appendix I.

Preservice tests for valves are performed in accordance with ASME OM, ISTC-3100.

3.9.6.2.5 Valve Replacement, Repair, and Maintenance

Testing in accordance with ASME OM, ISTC-3310 is performed after a valve is replaced, repaired, or undergoes maintenance. When a valve or its control system has been replaced, repaired, or has undergone maintenance that could affect valve performance, a new reference value is determined, or the previous value is reconfirmed by an inservice test. This test is performed before the valve is returned to service, or immediately if the valve is not removed from service. Deviations between the previous and new reference values are identified and analyzed. Verification that the new values represent acceptable operation is documented.

3.9.6.3 Relief Requests

Insert the following text after the first paragraph in DCD Subsection 3.9.6.3:

STD COL 3.9-4 The IST Program described herein utilizes Code Case OMN-1, Revision 1, “Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants” (Reference 202). Code Case OMN-1 establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves, in lieu of the requirements set forth in ASME OM Code Subsection ISTC.

OMN-1, Alternative Rules for the Preservice and Inservice Testing of Certain MOVs

Code Case OMN-1, Revision 1, “Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in Light Water Reactor Power Plants,” establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves in lieu of the requirements set forth in OM Code Subsection ISTC. However, Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” June 2003, has not yet endorsed OMN-1, Revision 1. Code Case OMN-1, Revision 0, has been determined by the NRC to provide an acceptable level of quality and safety when implemented in conjunction with the conditions imposed in Regulatory Guide 1.192. NUREG-1482, Revision 1, “Guidelines for Inservice Testing at Nuclear Power Plants,” recommends the implementation of OMN-1 by all licensees. Revision 1 to OMN-1 represents an improvement over Revision 0, as published in the ASME OM-2004 Code. OMN-1 Revision 1 incorporates the guidance on risk-informed testing of MOVs from OMN-11, “Risk-Informed Testing of Motor-Operated Valves,” and provides additional guidance on design basis verification testing and functional margin, which eliminates the need for the figures on functional margin and test intervals in Code Case OMN-1.

The IST Program implements Code Case OMN-1, Revision 1, in lieu of the stroke-time provisions specified in ISTC-5120 for MOVs, consistent with the guidelines provided in NUREG-1482, Revision 1, Section 4.2.5.

Regulatory Guide 1.192 states that licensees may use Code Case OMN-1, Revision 0, in lieu of the provisions for stroke-time testing in Subsection ISTC of the 1995 Edition up to and including the 2000 Addenda of the ASME OM Code when applied in conjunction with the provisions for leakage rate testing in ISTC-3600 (1998 Edition with the 1999 and 2000 Addenda). Licensees who choose to apply OMN-1 are required to apply all of its provisions. The IST program incorporates the following provisions from Regulatory Guide 1.192:

(1) The adequacy of the diagnostic test interval for each motor-operated valve (MOV) is evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1.

(2) The potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(3) Risk insights are applied using MOV risk ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis, consistent with the conditions in the applicable safety evaluations.

(4) Consistent with the provisions specified for Code Case OMN-11 the potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

Compliance with the above items is addressed in Section 3.9.6.2.2. Code Case OMN-1, Revision 1, is considered acceptable for use with OM Code-2001 Edition with 2003 Addenda. Finally, consistent with Regulatory Guide 1.192, the benefits of performing any particular test are balanced against the potential adverse effects placed on the valves or systems caused by this testing.

3.9.8 COMBINED LICENSE INFORMATION

3.9.8.2 Design Specifications and Reports

Add the following text after the second paragraph in DCD Subsection 3.9.8.2.

STD COL 3.9-2

Design specifications and design reports for ASME Section III piping are made available for NRC review. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in DCD Subsection 3.9.3.1.2) is completed by the COL holder after the construction of the piping systems and prior to fuel load (in accordance with DCD Tier 1 Section 2 ITAAC line items for the applicable systems).

3.9.8.3 Snubber Operability Testing

STD COL 3.9-3

This COL Item is addressed in Subsection 3.9.3.4.4.

3.9.8.4 Valve Inservice Testing

STD COL 3.9-4 This COL Item is addressed in Subsection 3.9.6.

3.9.8.5 Surge Line Thermal Monitoring

STD COL 3.9-5 This COL item is addressed in Subsection 3.9.3.1.2 and Subsection 14.2.9.2.22.

3.9.8.7 As-Designed Piping Analysis

Add the following text at the end of DCD Subsection 3.9.8.7.

STD COL 3.9-7 The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review.

This COL item is also addressed in Subsection 14.3.3.

3.9.9 REFERENCES

201. Not used.

202. ASME Code Case OMN-1, Revision 1, “Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants.”

203. Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000.

204. USNRC, Eugene V. Imbro, letter to Mr. David J. Modeen, Nuclear Energy Institute, Comments On Joint Owners' Group Air Operated Valve Program Document, dated October 8, 1999.

STD SUP 3.9-3

Table 3.9-201
Safety Related Snubbers

System	Snubber (Hanger) No.	Line #	System	Snubber (Hanger) No.	Line #
CVS	APP-CVS-PH-11Y0164	L001	RNS	APP-RNS-PH-12Y2060	L006
PXS	APP-PXS-PH-11Y0020	L021A	SGS	APP-SGS-PH-11Y0001	L003B
RCS	APP-RCS-PH-11Y0039	L215	SGS	APP-SGS-PH-11Y0002	L003B
RCS	APP-RCS-PH-11Y0067	L005B	SGS	APP-SGS-PH-11Y0004	L003B
RCS	APP-RCS-PH-11Y0080	L112	SGS	APP-SGS-PH-11Y0057	L003A
RCS	APP-RCS-PH-11Y0081	L215	SGS	APP-SGS-PH-11Y0058	L004B
RCS	APP-RCS-PH-11Y0082	L112	SGS	APP-SGS-PH-11Y0063	L003A
RCS	APP-RCS-PH-11Y0090	L118A	SGS	APP-SGS-PH-11Y0065	L005B
RCS	APP-RCS-PH-11Y0099	L022B	SGS	APP-SGS-PH-12Y0136	L015C
RCS	APP-RCS-PH-11Y0103	L003	SGS	APP-SGS-PH-12Y0137	L015C
RCS	APP-RCS-PH-11Y0105	L003	SGS	APP-SGS-PH-11Y0470	L006B
RCS	APP-RCS-PH-11Y0112	L032A	SGS	APP-SGS-PH-11Y2002	L006A
RCS	APP-RCS-PH-11Y0429	L225B	SGS	APP-SGS-PH-11Y2021	L006A
RCS	APP-RCS-PH-11Y0528	L005A	SGS	APP-SGS-PH-11Y3101	L006B
RCS	APP-RCS-PH-11Y0539	L225C	SGS	APP-SGS-PH-11Y3102	L006B
RCS	APP-RCS-PH-11Y0550	L011B	SGS	APP-SGS-PH-11Y3121	L006B
RCS	APP-RCS-PH-11Y0551	L011A	SGS	APP-SGS-PH-11Y0463	L006A
RCS	APP-RCS-PH-11Y0553	L153B	SGS	APP-SGS-PH-11Y0464	L006A
RCS	APP-RCS-PH-11Y0555	L153A	SGS	SG 1 Snubber A (1A)	(1)
RCS	APP-RCS-PH-11Y2005	L022A	SGS	SG 1 Snubber B (1B)	(1)
RCS	APP-RCS-PH-11Y2101	L032B	SGS	SG 2 Snubber A (2A)	(1)
RCS	APP-RCS-PH-11Y2117	L225A	SGS	SG 2 Snubber B (2B)	(1)

(1) These snubbers are on the upper lateral support assembly of the steam generators.

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF SEISMIC CATEGORY I MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.11.5 COMBINED LICENSE INFORMATION ITEM FOR EQUIPMENT QUALIFICATION FILE

Add the following text to the end of DCD Subsection 3.11.5.

STD COL 3.11-1 The COL holder is responsible for the maintenance of the equipment qualification file upon receipt from the reactor vendor. The documentation necessary to support the continued qualification of the equipment installed in the plant that is within the Environmental Qualification (EQ) Program scope is available in accordance with 10 CFR Part 50 Appendix A, General Design Criterion 1.

EQ files developed by the reactor vendor are maintained as applicable for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The contents of the qualification files are discussed in DCD Section 3D.7. The files are maintained for the operational life of the plant.

For equipment not located in a harsh environment, design specifications received from the reactor vendor are retained. Any plant modifications that impact the equipment use the original specifications for modification or procurement. This process is governed by applicable plant design control or configuration control procedures.

Central to the EQ Program is the EQ Master Equipment List (EQMEL). This EQMEL identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The EQMEL consists of equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal, or that is otherwise essential in preventing significant release of radioactive material to the environment. This list is developed from the equipment list provided in AP1000 DCD Table 3.11-1. The EQMEL and a summary of equipment qualification results are maintained as part of the equipment qualification file for the operational life of the plant.

Administrative programs are in place to control revision to the EQ files and the EQMEL. When adding or modifying components in the EQ Program, EQ files are generated or revised to support qualification. The EQMEL is revised to reflect these new components. To delete a component from the EQ Program, a deletion justification is prepared that demonstrates why the component can be deleted. This justification consists of an analysis of the component, an associated circuit review if appropriate, and a safety evaluation. The justification is released and/or referenced on an appropriate change document. For changes to the EQMEL, supporting documentation is completed and approved prior to issuing the changes. This documentation includes safety reviews and new or revised EQ files. Plant modifications and design basis changes are subject to change process reviews, e.g. reviews in accordance with 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52, in accordance with appropriate plant procedures. These reviews address EQ issues associated with the activity. Any changes to the EQMEL that are not the result of a modification or design basis change are subject to a separate review that is accomplished and documented in accordance with plant procedures.

Engineering change documents or maintenance documents generated to document work performed on an EQ component, which may not have an impact on the EQ file, are reviewed against the current revision of the EQ files for potential impact. Changes to EQ documentation may be due to, but not limited to, plant modifications, calculations, corrective maintenance, or other EQ concerns.

Table 13.4-201 provides milestones for EQ implementation.

APPENDIX 3A
HVAC DUCTS AND DUCT SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3B
LEAK-BEFORE-BREAK EVALUATION OF THE AP1000 PIPING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3C
REACTOR COOLANT LOOP ANALYSIS METHODS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3D
METHODOLOGY FOR QUALIFYING AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3E
HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3F
CABLE TRAYS AND CABLE TRAY SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3G
NUCLEAR ISLAND SEISMIC ANALYSES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3H
AUXILIARY AND SHIELD BUILDING CRITICAL SECTIONS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3I
EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 4
REACTOR

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CHAPTER 4
REACTOR

4.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.2 FUEL SYSTEM DESIGN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.3 NUCLEAR DESIGN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.4 THERMAL AND HYDRAULIC DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

4.4.7 COMBINED LICENSE INFORMATION

Replace the paragraph in DCD Subsection 4.4.7.2 with the following:

STD COL 4.4-2 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in DCD Subsection 7.1.6, the design limit DNBR values will be calculated. The calculations will be completed using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in DCD Section 4.4 remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty. This will be completed prior to fuel load.

4.5 REACTOR MATERIALS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.2.1.1 Compliance with 10 CFR 50.55a

Add the following text after the second sentence of the second paragraph of DCD Subsection 5.2.1.1.

STD COL 5.2-1 If a later Code edition/addenda than the Design Certification Code edition/addenda is used by the material and/or component supplier, then a code reconciliation to determine acceptability is performed as required by the ASME Code, Section III, NCA-1140. The later Code edition/addenda must be authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(a)(3). Code Cases to be used in design and construction are identified in the DCD; additional Code Cases for design and construction beyond those for the design certification are not required.

Inservice inspection of the reactor coolant pressure boundary is conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code Section XI, as described in Subsection 5.2.4. Inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code as discussed in Subsection 3.9.6 for pumps and valves, and as discussed in Subsection 3.9.3.4.4 for dynamic restraints.

5.2.3.2.1 Chemistry of Reactor Coolant

Add the following text to the end of DCD Subsection 5.2.3.2.1.

STD SUP 5.2-1 The water chemistry program is based on industry guidelines as described in EPRI TR-1002884, “Pressurized Water Reactor Primary Water Chemistry” (Reference 201). The program includes periodic monitoring and control of chemical additives and reactor coolant impurities listed in DCD Table 5.2-2. Detailed procedures implement the program requirements for sampling and analysis frequencies, and corrective actions for control of reactor water chemistry.

The frequency of sampling water chemistry varies (e.g. continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling is utilized to verify the effectiveness of these actions. When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants to within the specified range.

Chemistry procedures will provide guidance for the sampling and monitoring of primary coolant properties.

5.2.4 INSERVICE INSPECTION AND TESTING OF CLASS 1 COMPONENTS

Add the following after the first paragraph in DCD Subsection 5.2.4:

STD COL 5.2-2 The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load. Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, that are incorporated by reference in 10 CFR 50.55a(b)), subject to the limitations and modifications listed in 10 CFR 50.55a(b).

5.2.4.1 System Boundary Subject to Inspection

Add the following at the end of DCD Subsection 5.2.4.1:

STD COL 5.2-2 The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes those items within the Class 1 and Quality Group A (Equipment Class A per DCD Subsection 3.2.2 and DCD Table 3.2-3) boundary. Based on 10 CFR Part 50 and Regulatory Guide 1.26, the Class 1 boundary includes the following:

- Reactor pressure vessel;
- Portions of the Reactor System (RXS);
- Portions of the Chemical and Volume Control System (CVS);

- Portions of the Incore Instrumentation System (IIS);
- Portions of the Passive Core Cooling System (PXS);
- Portions of the Reactor Coolant System (RCS); and
- Portions of the Normal Residual Heat Removal System (RNS).

Those portions of the above systems within the Class 1 boundary are those items that are part of the reactor coolant pressure boundary as defined in Section 5.2.

Exclusions

Portions of the systems within the reactor coolant pressure boundary (RCPB), as defined above, that are excluded from the Class 1 boundary in accordance with 10 CFR Part 50, Section 50.55a, are as follows:

- Those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; or
- Components that are or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve is capable of automatic actuation and, assuming the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

The description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are defined by ASME Section XI, IWB-1220, except as modified by 10 CFR 50.55a.

The inservice inspection program is augmented for reactor vessel top head inspections by use of the ASME Code Case N-729-1, “Alternative Examination Requirements for Pressurized-Water Reactor (PWR) Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds,” as modified by the conditions specified in 10 CFR 50.55a(g)(6)(ii)(D).

Boric acid corrosion control procedures require inspection of the reactor coolant pressure boundary subject to leakage that can cause boric acid corrosion of the reactor coolant pressure boundary materials. The procedures determine the principal locations where leaks can cause degradation of the primary pressure boundary by boric acid corrosion. Potential paths of the leaking coolant are established. The boric acid corrosion control procedures also contain methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located.

The boric acid corrosion control procedures consist of:

1. Visual inspections of component surfaces that are potentially exposed to borated water leakage.
2. Discovery of leak path and removal of boric acid residue.
3. Assessment of the corrosion.
4. Follow-up inspection for adequacy of corrective actions, as appropriate.

Add the following text at the end of DCD Subsection 5.2.4.1:

STD SUP 5.2-2 The inservice inspection program, along with the boric acid corrosion control procedures, provides guidance for inspecting the integrity of bolting and threaded fasteners.

STD COL 5.3-7 The in-service inspection program is augmented to include the performance of a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld build-up acceptance standards are those provided in ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems are qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, an alternative inspection may be developed in conjunction with the voluntary consensus standards bodies (i.e., ASME) and submitted to the NRC for approval.

5.2.4.3 Examination Techniques and Procedures

Add the following at the end of DCD Subsection 5.2.4.3:

5.2.4.3.1 Examination Methods

Ultrasonic Examination of the Reactor Vessel

STD COL 5.2-2 Ultrasonic examination for the RPV is conducted in accordance with the ASME Code, Section XI. The design of the RPV considered the requirements of the ASME Code Section XI with regard to performance of preservice inspection. For the required preservice examinations, the reactor vessel meets the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment. The RPV nozzle-to-shell welds are 100% accessible for preservice inspection but might have limited areas that may not be accessible from the outer surface for inservice examination techniques. If accessibility is limited, an inservice inspection program relief request is prepared and submitted for review approval by the NRC.

Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Alternatively, nozzle inner radius examinations may be performed using enhanced visual techniques, as allowed by 10 CFR 50.55a(b)(2)(xxi).

Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided where feasible for the head and shoulders of a man within a working arm's length of the surface to be examined.

Surface Examination

Magnetic particle and liquid penetrant examination techniques are performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. Direct examination access for magnetic particle (MT) and liquid penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (see Visual Examination), except that additional access is provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision.

Volumetric Ultrasonic Direct Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

5.2.4.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII. Qualification to ASME Section XI, Appendix VIII, is in compliance with the provisions of 10 CFR 50.55a.

5.2.4.4 Inspection Intervals

Add the following after the second sentence of the first paragraph of DCD Subsection 5.2.4.4:

STD COL 5.2-2 Because 10 CFR 50.55a(g)(4) requires 120-month inspection intervals, Inspection Program B of IWB-2400 must be chosen. The inspection interval is divided into three periods. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. Each period can be extended for up to one year to enable an inspection to coincide with a plant outage. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals.

5.2.4.5 Examination Categories and Requirements

Add the following after the first sentence of DCD Subsection 5.2.4.5:

STD COL 5.2-2 Class 1 piping supports will be examined in accordance with ASME Section XI, IWF-2500.

Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5280. Components exempt from preservice examination are described in ASME Section III, NB-5283.

Add the following after the last sentence of DCD Subsection 5.2.4.5:

The preservice examination is performed once in accordance with ASME XI, IWB-2200, on all of the items selected for inservice examination, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examination categories, respectively) and the visual VT-2 examinations for category B-P.

5.2.4.6 Evaluation of Examination Results

Add the following at the end of DCD Subsection 5.2.4.6:

STD COL 5.2-2 Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWB-3132.4 or IWB-3142.4 are subjected to successive period examinations in accordance with the requirements of IWB-2420. Examinations that reveal flaws or relevant conditions exceeding Table IWB-3410-1 acceptance standards are extended to include additional examinations in accordance with the requirements of IWB-2430.

STD COL 5.2-2 Add Subsections 5.2.4.8, 5.2.4.9, and 5.2.4.10 after the last paragraph of DCD Subsection 5.2.4.7:

5.2.4.8 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the initial examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

5.2.4.9 Preservice Inspection of Class 1 Components

Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5281. Volumetric and surface examinations are performed as specified in ASME Section III, NB-

5282. Components described in ASME Section III, NB-5283 are exempt from preservice examination.

5.2.4.10 Program Implementation

The milestones for preservice and inservice inspection program implementation are identified in Table 13.4-201.

Add the following new subsection following DCD Subsection 5.2.5.3.4.

5.2.5.3.5 Response to Reactor Coolant System Leakage

STD COL 5.2-3 Operating procedures specify operator actions in response to prolonged low level unidentified reactor coolant leakage conditions that exist above normal leakage rates and below the Technical Specification (TS) limits to provide operators sufficient time to take action before the TS limit is reached. The procedures include identifying, monitoring, trending, and addressing prolonged low level leakage. The procedures for effective management of leakage, including low level leakage, are developed including the following operations related activities:

- Trends in the unidentified leakage rates are periodically analyzed. When the leakage rate increases noticeably from the baseline leakage rate, the safety significance of the leak is evaluated. The rate of increase in the leakage is determined to verify that plant actions can be taken before the plant exceeds TS limits.

- Procedures are established for responding to leakage. These procedures address the following considerations to prevent adverse safety consequence results from the leakage:

  – Plant procedures specify operator actions in response to leakage rates less than the limits set forth in the Technical Specifications. The procedures include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring, verifying the leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walkdown outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage.

  – Plant procedures specify the amount of time the leakage detection and monitoring instruments (other than those required by Technical Specifications) may be out of service to effectively monitor the leakage rate during plant operation (i.e., hot shutdown, hot standby, startup, transients, and power operation).

- The output and alarms from leakage monitoring systems are provided in the main control room. Procedures are readily available to the operators for converting the instrument output to a common leakage rate. (Alternatively, these procedures may be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring systems are conducted. The alarm(s), and associated setpoint(s), provide operators an early warning signal so that they can take corrective actions, as discussed above, i.e., before the plant exceeds TS limits.

- During maintenance and refueling outages, actions are taken to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action is taken to eliminate the condition resulting in the leakage.

The procedures described above will be available prior to fuel load.

5.2.6 COMBINED LICENSE INFORMATION ITEMS

5.2.6.1 ASME Code and Addenda

STD COL 5.2-1 This COL Item is addressed in Subsection 5.2.1.1.

5.2.6.2 Plant-Specific Inspection Program

STD COL 5.2-2 This COL Item is addressed in Subsections 5.2.4, 5.2.4.1, 5.2.4.3.1, 5.2.4.3.2, 5.2.4.4, 5.2.4.5, 5.2.4.6, 5.2.4.8, 5.2.4.9, and 5.2.4.10.

5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment

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STD COL 5.2-3 This COL item is addressed in Subsection 5.2.5.3.5.

5.2.7 REFERENCES

Add the following information at the end of DCD Subsection 5.2.7.

201. EPRI, "Pressurized Water Reactor Primary Water Chemistry Guidelines," EPRI TR-1002884, Revision 5, October 2003.

5.3 REACTOR VESSEL

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.3.2.6 Material Surveillance

Add the following information between the first and second paragraphs of DCD Subsection 5.3.2.6.

STD COL 5.3-2 Surveillance test materials are prepared from the actual materials used in fabricating the beltline region of the reactor vessel. Records are maintained of the chemical analyses, fabrication history, mechanical properties and other essential variables pertinent to the fabrication process of the shell forging and weld metal from which the surveillance test materials are prepared. The test materials are processed so that they are representative of the material in the completed reactor vessel.

Three metallurgically different materials prepared from sections of reactor vessel shell forging are used for test specimens. These include base metal, weld metal and heat affected zone (HAZ) material.

Base metal test material is manufactured from a section of ring forging, either the intermediate shell course, the lower shell course, or the transition ring of the reactor pressure vessel. Selection is based on an evaluation of initial toughness (characterized by the reference temperature (RT_{NDT}) and Upper Shelf Energy (USE)), and the predicted effect of chemical composition (nickel and residual copper) and neutron fluence on the toughness (RT_{NDT} shift and decrease in USE) during reactor operation. The ring forging with the highest predicted adjusted RT_{NDT} temperature (initial RT_{NDT} plus RT_{NDT} shift) or that with USE predicted to approach close to the minimum limit of 50 ft-lb at end-of-license (EOL) is selected as the surveillance base metal test material. The means for measuring initial toughness and for predicting irradiation induced toughness changes is consistent with applicable procedures in force at the time the material is being selected. The section of shell forging used for the base metal test block is adjacent to the test material used for fracture toughness tests.

Weld metal and HAZ test material is produced by welding together sections of the forgings from the beltline of the reactor vessel. The HAZ test material is manufactured from a section of the same shell course forging used for base metal test material. The sections of shell course forging used for weld metal and HAZ test material are adjacent to the test material used for fracture toughness tests. The heat of wire or rod and lot of flux are from the same heat and lot used in making the beltline region welds. Welding parameters duplicate those used for the beltline region welds. The procedures for inspection of the reactor vessel welds are followed for the inspection of the welds in test materials. The surveillance weld and HAZ material are heat-treated to metallurgical conditions which are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

Test Specimens are marked to identify the type of materials and the orientation with respect to the test materials. Drawings specify the identification system to be used and include plant identification, type of material, orientation of specimen and sequential number.

Baseline test specimens are provided for establishing the baseline (unirradiated) properties of the reactor vessel materials. The data from tests of these specimens provides the basis for determining the radiation induced property changes of the reactor vessel materials.

Drop weight test specimens of each of base metal, weld metal, and HAZ metal are provided for establishing the nil-ductility transition temperature (NDTT) of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination from which subsequent radiation induced changes are determined.

Standard Charpy impact test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ material are provided for developing a Charpy impact energy transition curve from fully brittle to fully ductile behavior for defining specific index temperatures for these materials. These data, together with the drop weight NDTT, are used to establish an RT_{NDT} for each material.

Tensile test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ metal are provided to permit a sufficient number of tests for accurately establishing the tensile properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and one intermediate temperature) to define the strength of the material.

The above described test specimens are to be used for determining changes in the strength and toughness of the surveillance materials resulting from neutron irradiation. Sufficient Charpy impact, compact tension and tensile test specimens are provided for establishing the changes in the properties of the surveillance materials over the lifetime of the reactor vessel. The type, quantity, and storage conditions (e.g., surveillance capsules backfilled with inert gas) of test specimens meet or exceed the minimum requirements of ASTM E-185.

Reactor materials do not begin to be affected by neutron fluence until the reactor begins critical operation. Table 13.4-201 provides milestones for reactor vessel material surveillance program implementation.

Add the following subsection after DCD Subsection 5.3.2.6.2.2.

5.3.2.6.3 Report of Test Results

STD COL 5.3-2 A summary technical report for each capsule withdrawn with the test results is submitted, as specified in 10 CFR 50.4, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

The report includes the data required by ASTM E185-82, as specified in paragraph III.B.1 of 10 CFR Part 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

If the test results indicate a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specification is provided with the report.

Add the following subsection after DCD Subsection 5.3.3.1.

5.3.3.2 Operating Procedures

STD SUP 5.3-1 Plant operating procedures are developed and maintained to prevent exceeding the pressure-temperature limits identified in reactor coolant system pressure and temperature limits report, as required by Technical Specification 5.6.6, during normal and abnormal operating conditions and system tests.

5.3.6 COMBINED LICENSE INFORMATION

5.3.6.1 Pressure-Temperature Limit Curves

Replace the text in DCD Subsection 5.3.6.1 with the following.

STD COL 5.3-1 The pressure-temperature curves shown in DCD Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. Plant-specific curves will be developed based on material composition of copper and nickel. Use of plant-specific curves will be addressed during procurement and fabrication of the reactor vessel. As noted in the bases to Technical Specification 3.4.14, use of plant-specific curves requires evaluation of the LTOP system. This includes an evaluation of the setpoint pressure for the RNS relief valve to determine if the setpoint pressure needs to be changed based on the plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressure are required prior to fuel load.

5.3.6.2 Reactor Vessel Materials Surveillance Program

STD COL 5.3-2 This COL Item is addressed in Subsections 5.3.2.6 and 5.3.2.6.3.

5.3.6.4 Reactor Vessel Materials Properties Verification

Replace the text in DCD Subsection 5.3.6.4.1 with the following.

5.3.6.4.1 Reactor Vessel Materials Properties Verification

STD COL 5.3-4 The verification of plant-specific belt line material properties consistent with the requirements in DCD Subsection 5.3.3.1 and DCD Tables 5.3-1 and 5.3-3 will be completed prior to fuel load. The verification will include a pressurized thermal shock evaluation based on as procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.

5.3.6.6 Quickloc Weld Build-up ISI

STD COL 5.3-7 This item is addressed in Subsection 5.2.4.1.

5.4 COMPONENT AND SUBSYSTEM DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.4.2.5 Steam Generator Inservice Inspection

Add the following information at the end of DCD Subsection 5.4.2.5.

STD COL 5.4-1 A steam generator tube surveillance program is implemented in accordance with the recommendations and guidance of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 201). A program for periodic monitoring of degradation of steam generator internals is also implemented in accordance with NEI 97-06. Applicable Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP) guidelines are followed as described in the NEI 97-06. The Programs are in compliance with applicable sections of ASME Section XI.

NEI 97-06 and the referenced EPRI SGMP guidelines provide recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, required actions based on findings, and tube plugging. The minimum requirements for inservice inspection of steam generators, including plugging criteria, are established in Technical Specification 5.5.4.

The tube surveillance and degradation monitoring programs include provisions to maintain the compatibility of steam generator tubing with primary and secondary coolant to limit the steam generators' susceptibility to corrosion. These provisions are in accordance with NEI 97-06.

5.4.7.1 Design Basis

Replace the second bulleted item in DCD Subsection 5.4.7.1.2.3 with the following:

VCS DEP 2.0-2

- The component cooling water system supply temperature to the normal residual heat removal system heat exchangers is based on an ambient design wet bulb temperature of no greater than 87.3°F (100 year return estimate of 2-hour duration). The 87.3°F value is assumed for normal conditions and transients that start at normal conditions.

  The steaming prevention function is evaluated assuming the ambient wet bulb temperature is at the maximum safety value for the site. During plant operation, maximum IRWST temperature is reduced below 120°F whenever necessary by circulating IRWST water through one of the RNS heat exchangers, and removing the heat through the CCS and SWS. Since the RNS heat exchangers are not being used to remove decay heat with the plant at power, at least one is available for IRWST heat removal. Only one train of CCS (pump and heat exchanger) and one train of SWS (pump, strainer, and cooling tower cell) are normally in operation with the plant at power. There is sufficient margin in CCS pump flow capacity and motor size, and in CCS heat exchanger UA, to valve in one of the RNS heat exchangers and remove IRWST heat by directing CCS flow through the heat exchanger and transferring the excess heat to the SWS cooling tower. CCS temperature rises slightly above the normal full power CCS temperature during this evolution, but does not approach the maximum allowable value of 100°F.

Prevention of IRWST steaming following high pressure heat removal operations with the Passive Residual Heat Removal (PRHR) heat exchanger is accomplished in the same manner, by lining up both RNS heat exchangers to the CCS and the IRWST. CCS is delivered to the RNS heat exchangers at a temperature consistent with the maximum safety ambient wet bulb temperature and the CCS and SWS heat duty and flow rates. Cooling is assumed to begin two hours after reactor trip, with decay heat appropriate for that time after the event. Calculations performed to determine the maximum IRWST temperature achieved following a high pressure heat removal event using the PRHR heat exchanger assumed CCS temperature is determined by use of a maximum safety ambient wet bulb temperature value of 87.4°F. The maximum predicted IRWST liquid temperature is 201°F. Therefore, it can be concluded that IRWST cooling performance (prevention of steaming) is acceptable (Reference 202).

5.4.15 COMBINED LICENSE INFORMATION ITEMS

STD COL 5.4-1 This COL Item is addressed in Subsection 5.4.2.5.

5.4.16 REFERENCES

Insert the following information at the end of DCD Subsection 5.4.16.

201. Nuclear Energy Institute, “Steam Generator Program Guidelines,” NEI 97-06, Revision 2, May 2005.

202. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010.

CHAPTER 6
ENGINEERED SAFETY FEATURES

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CHAPTER 6
ENGINEERED SAFETY FEATURES

6.0 ENGINEERED SAFETY FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.1.1.2 Fabrication Requirements

Add the following information to the end of DCD Subsection 6.1.1.2:

STD COL 6.1-1

In accordance with Appendix B to 10 CFR Part 50, the quality assurance program establishes measures to provide control of special processes. One element of control is the review and acceptance of vendor procedures that pertain to the fabrication, welding, and other quality assurance methods for safety related component to determine both code and regulatory conformance. Included in this review and acceptance process are those vendor procedures necessary to provide conformance with the requirements of Regulatory Guides 1.31 and 1.44 for engineered safety features components as discussed in DCD Section 6.1 and reactor coolant system components as discussed in DCD Subsection 5.2.3.

6.1.2.1.6 Quality Assurance Features

Replace the third paragraph under the subsection titled “Service Level I and Service Level III Coatings” within DCD Subsection 6.1.2.1.6 with the following information.

STD COL 6.1-2

During the design and construction phase, the coatings program associated with selection, procurement and application of safety related coatings is performed to applicable quality standards. The requirements for the coatings program are contained in certified drawings and/or standards and specifications controlling the coating processes of the designer (Westinghouse) (these design documents will be available prior to the procurement and application of the coating material by the constructor of the plant). Regulatory Guide 1.54 and ASTM D5144 (Reference 201) form the basis for the coating program.

During the operations phase, the coatings program is administratively controlled in accordance with the quality assurance program implemented to satisfy 10 CFR Part 50, Appendix B, and 10 CFR Part 52 requirements. The coatings program provides direction for the procurement, application, inspection, and monitoring of safety related coating systems. Prior to initial fuel loading, a consolidated plant coatings program will be in place to address procurement, application, and monitoring (maintenance) of those coating system(s) for the life of the plant.

Coating system monitoring requirements for the containment coating systems are based on ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant," and ASTM D7167 (Reference 203), "Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality assurance requirements.

Include a new second paragraph under the subsection titled "Service Level II Coatings" within DCD Subsection 6.1.2.1.6 with the following information.

Such Service Level II coatings used inside containment are procured to the same standards as Service Level I coatings with regard to radiation tolerance and performance under design basis accident conditions as discussed below.

Replace the second sentence of the third paragraph under the subsection titled "Service Level II Coatings" within DCD Subsection 6.1.2.1.6 with the following information.

Coating system application, inspection, and monitoring requirements for the Service Level II coatings used inside containment will be performed in accordance with a program based on ASTM D5144 (Reference 201), "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," and the guidance of ASTM D5163 (Reference 202), "Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant." Any anomalies identified during coating inspection or monitoring are resolved in accordance with applicable quality requirements.

6.1.3 COMBINED LICENSE INFORMATION ITEMS

6.1.3.1 Procedure Review

STD COL 6.1-1 This COL Item is addressed in Subsection 6.1.1.2.

6.1.3.2 Coating Program

STD COL 6.1-2 This COL Item is addressed in Subsection 6.1.2.1.6.

The following information supplements the information provided in DCD Subsection 6.1.4.

6.1.4 REFERENCES

201. ASTM D5144-08, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants”

202. ASTM D5163-05a, “Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant”

203. ASTM D7167-05, “Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant”

6.2 CONTAINMENT SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.2.1.1.3 Design Evaluation

Add the following information after the fourth paragraph of DCD Subsection 6.2.1.1.3.

VCS DEP 2.0-2 The maximum safety non-coincident wet bulb temperature for VCSNS Units 2 and 3 is increased from 86.1°F to 87.3°F, however there are no impacts on the performance of the safety systems.

6.2.2.3 Safety Evaluation

Add the following information at the end of DCD Subsection 6.2.2.3.

VCS DEP 2.0-2 There are no changes to the AP1000 design required to address any safety issues associated with the VCSNS Units 2 and 3 increased maximum safety wet bulb temperature of 87.3°F. The peak containment pressure at the maximum safety wet bulb temperature of 87.3°F for the VCSNS Units 2 and 3 site is bounded by the results of the current AP1000 analysis.

The pressure decay curve for the containment utilizing the VCSNS Units 2 and 3 safety wet bulb value of 87.3°F is the same as the containment response for wet bulb temperatures equal to the standard maximum safety wet bulb value (Reference 201).

6.2.5.1 Design Basis

Add the following information at the end of DCD Subsection 6.2.5.1, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

STD COL 6.2-1 The Containment Leak Rate Test Program using 10 CFR Part 50, Appendix J Option B is established in accordance with NEI 94-01 (DCD Subsection 6.2.7, Reference 30), as modified and endorsed by the NRC in Regulatory Guide 1.163. Table 13.4-201 provides milestones for containment leak rate testing implementation.

6.2.5.2.2 System Operation

STD COL 6.2-1 Add the following information at the end of the subsection "Scheduling and Reporting of Periodic Tests" within DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, "Containment Leak Rate Test Program Description."

Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with NEI 94-01, as endorsed and modified by Regulatory Guide 1.163, and described below:

Type A Tests

A preoperational Type A test is conducted prior to initial fuel load. If initial fuel load is delayed longer than 36 months after completion of the preoperational Type A test, a second preoperational Type A test shall be performed prior to initial fuel load. The first periodic Type A test is performed within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests are performed at a frequency of at least once per 48 months, until acceptable performance is established. The interval for testing begins at initial reactor operation. Each test interval begins upon completion of a Type A test and ends at the start of the next test. The extension of the Type A test interval is determined in accordance with NEI 94-01.

Type A testing is performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as successful completion of two consecutive Type A tests where the calculated performance leakage rate was less than 1.0 L_a. A preoperational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a preoperational Type A test can be treated as a periodic test. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Type B Tests (Except Containment Airlocks)

Type B tests are performed prior to initial entry into Mode 4. Subsequent periodic Type B tests are performed at a frequency of at least once per 30 months, until acceptable performance is established. The test intervals for Type B penetrations may be increased based upon completion of two consecutive periodic as-found

Type B tests where results of each test are within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component prior to implementing Option B of 10 CFR Part 50, Appendix J. An extended test interval for Type B tests may be increased to a specific value in a range of frequencies from greater than once per 30 months up to a maximum of once per 120 months. The extension of specific test intervals for Type B penetrations is determined in accordance with NEI 94-01.

Type B Tests (Containment Airlocks)

Containment airlock(s) are tested at an internal pressure of not less than P_{ac}. (Prior to a preoperational Type A test $P_{ac}= P_a$.) Subsequent periodic tests are performed at a frequency of at least once per 30 months. In addition, equalizing valves, door seals, and penetrations with resilient seals (i.e., shaft seals, electrical penetrations, view port seals and other similar penetrations) that are testable, are tested at a frequency of once per 30 months.

For periods of multiple containment entries where the airlock doors are routinely used for access more frequently than once every seven days (e.g., shift or daily inspection tours of the containment), door seals may be tested once per 30 days during this time period.

Airlock door seals are tested prior to a preoperational Type A test. When containment integrity is required, airlock door seals are tested within seven days after each containment access.

Type C Tests

Type C tests are performed prior to initial entry into Mode 4. Subsequent periodic Type C tests are performed at a frequency of at least once per 30 months, until adequate performance has been established. Test intervals for Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests where the result of each test is within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the valve prior to implementing Option B of 10 CFR Part 50, Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 30 months up to a maximum of 60 months. Test interval extensions for Type C valves are determined in accordance with NEI 94-01.

Reporting

A post-outage report is prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The report is available on-site for NRC review. The report shows that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable.

STD COL 6.2-1 Add the following subsection at the end of DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

Acceptance Criteria

Acceptance criteria for Type A, B and C Tests are established in Technical Specification 5.5.8.

6.2.6 COMBINED LICENSE INFORMATION FOR CONTAINMENT LEAK RATE TESTING

STD COL 6.2-1 This COL item is addressed in Subsections 6.2.5.1 and 6.2.5.2.2.

6.2.7 REFERENCES

201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_ 000706, June 30, 2010.

6.3 PASSIVE CORE COOLING SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.3.8 COMBINED LICENSE INFORMATION

6.3.8.1 Containment Cleanliness Program

Insert the following information at the end of DCD Subsection 6.3.8.1:

This COL Item is addressed below.

STD COL 6.3-1 Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris left in containment following personnel entry and exits. The program is consistent with the containment cleanliness program limits discussed in DCD Subsection 6.3.8.1. The program includes, as a minimum, the following:

Responsibilities

The program defines the organizational responsibilities for implementing the program; defines personnel and material controls; and defines the inspection and reporting requirements.

Implementation

Containment Entry/Exit

- Controls to account for the quantities and types of materials introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, to ensure adequate accountability controls. This may be accomplished by the work management process. Storage of aluminum is prohibited without engineering authorization. Cardboard boxes or miscellaneous packing material is not brought into containment without approval.
- If entries are made at power, prohibited materials and limits on quantities of materials that may generate hydrogen are established.
- Controls for loose items, such as keys and pens, which could be inadvertently left in containment.

- Methods and controls for securing any items and materials left unattended in containment.
- Administrative controls for accounting for tools, equipment and other material are established.
- Administrative controls for accounting of the permanent removal of materials previously introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, that may be left unattended in containment during outages and power operation. Types of materials considered are tape, labels, plastic film, and paper and cloth products.
- Requirements and actions to be taken for unaccounted for material.
- Requirements for final containment cleanliness inspections consistent with the design bases provided in DCD Subsection 6.3.8.1.
- Record keeping requirements for entry/exit logs.

Housekeeping

Housekeeping procedures require that work areas be maintained in a clean and orderly fashion during work activities and returned to original conditions (or better) upon completion of work.

Sampling Program

A sampling program is implemented consistent with NEI Guidance Report 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology” as supplemented by the NRC in the “Safety Evaluation by The Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), ‘Pressurized Water Reactor Sump Performance Evaluation Methodology.’” Latent debris sampling is implemented before startup. The sampling is conducted after containment exit cleanliness inspections to provide reasonable assurance that the plant latent debris design bases are met. Sampling frequency and scope may be adjusted based on sampling results. Results are evaluated post-start up and any nonconforming results will be addressed in the Corrective Action Program.

6.4 HABITABILITY SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following information after the second paragraph of DCD Subsection 6.4.

VCS DEP 2.0-2 Based on system design margin of the VBS, the MCR temperature and humidity at the higher VCSNS maximum safety wet bulb temperature will remain at or below the desired design points during normal operation (Reference 201).

6.4.1.1 Main Control Room Design Basis

Add the following information after the last paragraph of DCD Subsection 6.4.1.1:

VCS DEP 2.0-2 The VBS system maintains design conditions in the MCR during all normal and accident conditions when the VBS system is operational. The LCCWS also serves the RNS and CVS pump room coolers. The nominal refrigeration capacity of each of the air-cooled chillers used in the LCCWS is 322 tons at an ambient dry bulb temperature of 115°F (Reference 201).

6.4.3 SYSTEM OPERATION

Add the following information at the end of DCD Subsection 6.4.3:

STD COL 6.4-2 Generic Issue 83 addresses the importance of maintaining control room habitability following an accidental release of external toxic or radioactive material or smoke and the capability of the control room operators to safely control the reactor. Procedures and training for control room habitability are written in accordance with Section 13.5 for control room operating procedures, and Section 13.2 for operator training. The procedures and training are verified to be consistent to the intent of Generic Issue 83.

The procedures and training address the toxic chemical events addressed in Sections 2.2 and 6.4 consistent with the guidance provided in regulatory position C.5 of Regulatory Guide 1.78, including arrangements with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals occur within five miles of the plant. The procedures include the conduct of periodic surveys of stationary and mobile sources of hazardous chemicals affecting the evaluations consistent with the guidance provided in regulatory position 2.5 of Regulatory Guide 1.196. The procedures include appropriate reviews of the configuration of the control room envelope and habitability systems consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. The procedures also include periodic assessments of the control room habitability systems' material condition, configuration controls, safety analyses, and operating and maintenance procedures consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196.

Procedures for testing and maintenance are consistent with the design requirements of the DCD including the guidance provided in regulatory position 2.7.1 of Regulatory Guide 1.196.

6.4.4 SYSTEM SAFETY EVALUATION

Insert the following information at the end of the eighth paragraph of DCD Subsection 6.4.4.

VCS COL 6.4-1
STD COL 6.4-1

Table 6.4-201 provides additional details regarding the evaluated onsite chemicals.

Insert the following subsection at the end of DCD Subsection 6.4.4.

6.4.4.1 Dual Unit Analysis

STD SUP 6.4-1

Credible events that could put the control room operators at risk from a dose standpoint at a single AP1000 unit have been evaluated and addressed in the DCD. The dose to the control room operators at an adjacent AP1000 unit due to a radiological release from another unit is bounded by the dose to control room operators on the affected unit. While it is possible that a unit may be downwind in an unfavorable location, the dose at the downwind unit would be bounded by what has already been evaluated for a single unit AP1000. Simultaneous accidents at multiple units at a common site are not considered to be a credible event.

Add the following subsection after the Subsection 6.4.4.1, at the end of DCD Subsection 6.4.4.

6.4.4.2 Toxic Chemical Habitability Analysis

VCS COL 6.4-1

Regulatory Guide 1.78 establishes the Occupational Safety and Health Association (OSHA) National Institute for Safety and Health (NIOSH) Immediately Dangerous to Life and Health (IDLH) guidelines for 30 minute exposure as the required screening criteria for airborne hazardous chemicals. Per Regulatory Guide 1.78, the NIOSH IDLH values were utilized to screen chemicals and to evaluate concentrations of hazardous chemicals to determine their effect on control room habitability. The evaluation of these hazardous materials is provided in Subsection 2.2.3.1.3.

6.4.7 COMBINED LICENSE INFORMATION

VCS COL 6.4-1
STD COL 6.4-1

This COL Item is addressed in Subsections 2.2.2.2.1.1 and 6.4.4.2.

STD COL 6.4-2

This COL Item is addressed in Subsection 6.4.3.

STD COL 6.4-1

This COL Item is addressed in Subsection 6.4.4.

6.4.8 REFERENCES

201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_ 000706, June 30, 2010.

Table 6.4-201 (Sheet 1 of 3)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals[a]

STD COL 6.4-1

A — Standard Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR intake	Evaluated Location	MCR Habitability Impact Evaluation
Hydrogen	Gas	500 scf	126.3 ft	Yard at turbine building	MCR
Hydrogen	Liquid	1500 gal	577 ft	Gas storage	MCR
Nitrogen	Liquid	3000 gal	577 ft	Gas storage	MCR
Carbon Dioxide (CO_2)	Liquid	6 tons	577 ft	Gas storage	MCR
Oxygen Scavenger [Hydrazine]	Liquid	1600 gal	203 ft	Turbine building	IH
pH Addition [Morpholine]	Liquid	1600 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	800 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	20,000 gal	436 ft	CWS area	IH
Sodium Hydroxide	Liquid	800 gal	203 ft	Turbine building	S
Sodium Hydroxide	Liquid	20,000 gal	436 ft	CWS area	S
Fuel Oil	Liquid	60,000 gal	197 ft	DG fuel oil storage tank, DG building, Annex building	IH
Corrosion Inhibitor [Sodium Molybdate]	Liquid	800 gal	203 ft	Turbine building	S
Corrosion Inhibitor [Sodium Molybdate]	Liquid	10,000 gal	436 ft	CWS area	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	800 gal	203 ft	Turbine building	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	10,000 gal	436 ft	CWS area	S

Table 6.4-201 (Sheet 2 of 3)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals[(a)]

STD COL 6.4-1

A — Standard Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR intake	Evaluated Location	MCR Habitability Impact Evaluation
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	800 gal	203 ft	Turbine building	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	10,000 gal	436 ft	CWS area	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	800 gal	203 ft	Turbine building	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	10,000 gal	436 ft	CWS area	S

Table 6.4-201 (Sheet 3 of 3)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals[(a)]

VCS COL 6.4-1

B — Site Specific Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR intake	Evaluated Location	MCR Habitability Impact Evaluation
pH Addition [Sulfuric Acid]	Liquid	10,000 gal	903 ft	CWS area	Bounded by STANDARD evaluation
Sodium Hydroxide	Not used	Not used	Not used	CWS area	Not used
Dispersant [Polymeric silt dispersant]	Liquid	800 gal	258 ft	Turbine building	S
Dispersant [Polymeric silt dispersant]]	Liquid	10,000 gal	903 ft	CWS area	S
Corrosion inhibitor [Ortho polyphosphate]	Liquid	10,000 gal	903 ft	CWS area	S
Scale inhibitor [Phosphonate]	Liquid	10,000 gal	903 ft	CWS area	S
Biocide / Disinfectant [Sodium hypochlorite]	Liquid	10,000 gal	903 ft	CWS area	Bounded by STANDARD evaluation
Algaecide [Quaternary amine]	Liquid	3,500 gal	903 ft	CWS area	NA

Notes:

STD COL 6.4-1 a) This table supplements DCD Table 6.4-1. Quantities are by largest evaluated container content for the evaluated location per unit. Quantities and distances are bounding evaluation values and may not be actual amounts and distances. Smaller quantities of a chemical at further distances from the MCR air intake are not shown on this table. Actual site locations are confirmed to be at or beyond the evaluated distance.

S - Chemicals with an Impact Evaluation designation of "S" for the MCR Habitability Impact Evaluation were evaluated and screened out based on the chemical properties, distance, and quantities.

IH - Chemicals with an Impact Evaluation designation of "IH" indicates the evaluation of this chemical considered the design detail of the main control room intake height.

MCR- Chemicals with an Impact Evaluation designation of “MCR” indicates the evaluation of this chemical considered design details of the main control room such as volume, envelope boundaries, ventilation systems, and occupancy factor.

VCS COL 6.4-1 NA - Not applicable. Chemicals with an Impact Evaluation designation of "NA" have been evaluated without consideration of main control room intake height or any additional design details of the main control room.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.6 INSERVICE INSPECTION OF CLASS 2, 3, AND MC COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following to DCD Section 6.6 ahead of Subsection 6.6.1 heading:

STD COL 6.6-1 The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load. Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in Regulatory Guide 1.147, that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b)).

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

Add the following to the end of DCD Subsection 6.6.1:

STD COL 6.6-1 Class 2 and 3 components are included in the equipment designation list and the line designation list contained in the inservice inspection program.

6.6.2 ACCESSIBILITY

Revise the first and last sentences of the third paragraph in DCD Subsection 6.6.2 to add supplemental information as follows:

STD SUP 6.6-1 Considerable experience has been drawn on in designing, locating, and supporting Quality Group B and C (ASME Class 2 and 3) and Class MC pressure-retaining components to permit pre-service and inservice inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections are used in establishing the designs. The inspection design goals are to eliminate uninspectable components, reduce occupational radiation exposure, reduce inspection times, allow state-of-the-art inspection systems, and enhance detection and the reliability of flaw characterization. There are no Quality Group B and C components or Class MC components, which require inservice inspection during reactor operation.

Add the following to the end of DCD Subsection 6.6.2:

STD COL 6.6-2 During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Modifications reviewed following design certification adhere to the same level of review as the certified design per 10 CFR Part 50, Appendix B as implemented by the Westinghouse Quality Management System (QMS). The QMS requires that changes to approved design documents, including field changes, are subject to the same review and approval process as the original design. This explicitly requires the field change process to follow the same level of review that was required during the design process. Accessibility and inspectability are key components of the design process.

Control of accessibility for inspectability and testing during post-design certification activities is provided via procedures for design control and plant modifications.

6.6.3 EXAMINATION TECHNIQUES AND PROCEDURES

Add the following Subsections 6.6.3.1, 6.6.3.2 and 6.6.3.3 to the end of DCD Subsection 6.6.3:

6.6.3.1 Examination Methods

Visual Examination

STD COL 6.6-1 Visual examination methods VT-1, VT-2 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided in accordance with Table IWA-2210-1.

Surface Examination

Magnetic particle, liquid penetrant, and eddy current examination techniques are performed in accordance with ASME Section XI, IWA-2221, IWA-2222, and IWA-2223 respectively. Direct examination access for magnetic particle (MT) and liquid penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (see Visual Examination), except that additional access is provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision.

Ultrasonic Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

6.6.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII.

6.6.3.3 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

6.6.4 INSPECTION INTERVALS

Add the following to the end of DCD Subsection 6.6.4:

STD COL 6.6-1 Because 10 CFR 50.55a(g)(4) requires 120-month inspection intervals, Inspection Program B of IWB-2400 must be chosen. The inspection interval is divided into three periods. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. The periods within each inspection interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals. It is intended that inservice examinations be performed during normal plant outages, such as refueling shutdown or maintenance shutdowns occurring during the inspection interval.

6.6.6 EVALUATION OF EXAMINATION RESULTS

Add the following new paragraph at the end of DCD Subsection 6.6.6:

STD COL 6.6-1 Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWC-3122.3 or IWC-3132.3 for Class 2 components, IWD-3000 for Class 3 components, IWE-3122.3 for Class MC components, or IWF-3112.2 or IWF-3122.2 for component supports, are subjected to successive period examinations in accordance with the requirements of IWC-2420, IWD-2420, IWE-2420, or IWF-2420, respectively. Examinations that reveal flaws or relevant conditions exceeding Table IWC-3410-1, IWD-3000, IWE-3000, or IWF-3400 acceptance standards are extended to include additional examinations in accordance with the requirements of IWC-2430, IWD-2430, or IWF-2430, respectively.

6.6.9 COMBINED LICENSE INFORMATION ITEMS

6.6.9.1 Inspection Programs

STD COL 6.6-1 This COL Item is addressed in Section 6.6 introduction, and in Subsections 6.6.1, 6.6.3.1, 6.6.3.2, 6.6.3.3, 6.6.4, and 6.6.6.

6.6.9.2 Construction Activities

STD COL 6.6-2 This COL Item is addressed in Subsection 6.6.2.

APPENDIX 6A
FISSION PRODUCT DISTRIBUTION IN THE AP1000 POST-DESIGN BASIS ACCIDENT CONTAINMENT ATMOSPHERE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 7
INSTRUMENTATION AND CONTROLS

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CHAPTER 7
INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

7.1.6.1 Setpoint Calculations for Protective Functions

STD COL 7.1-1 The Setpoint Program described in Technical Specifications Section 5.5 provides the appropriate controls for update of the instrumentation setpoints following completion of the calculation of setpoints for protective functions and the reconciliation of the setpoints against the final design.

7.2 REACTOR TRIP

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.3 ENGINEERED SAFETY FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.5 SAFETY-RELATED DISPLAY INFORMATION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

7.5.2 VARIABLE CLASSIFICATIONS AND REQUIREMENTS

Add the following paragraph at the end of DCD Subsection 7.5.2.

STD COL 7.5-1 FSAR Table 7.5-201 supplements DCD Table 7.5-1 and provides variable data shown in the DCD Table as “site specific.”

7.5.3.5 Type E Variables

Add the following paragraph at the end of DCD Subsection 7.5.3.5.

STD COL 7.5-1 FSAR Table 7.5-202 supplements DCD Table 7.5-8 and provides variable data shown in the DCD Table as “site specific.”

7.5.5 COMBINED LICENSE INFORMATION

STD COL 7.5-1
VCS COL 7.5-1 This COL item is addressed in Subsection 7.5.2 and Table 7.5-201, and in Subsection 7.5.3.5 and Table 7.5-202.

Table 7.5-201
Post-Accident Monitoring System[a]

VCS COL 7.5-1

Variable	Range/Status[b]	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication	Remarks
			Environmental	Seismic				
Boundary environs Radiation								
• Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} to 10^{-3} µCi/cc	C3, E3	None	None	No minimum number of instruments is specified. A sufficient number are provided to outfit the Emergency Planning Field Teams.	Non-1E	No	
• Radiation (portable instrumentation)	10^{-3} to 10^{4} R/hr. photons 10^{-3} to 10^{4} rads/hr. beta and low-energy photons							
• Radioactivity (portable instrumentation)	Multichannel gamma ray spectrometer							
Meteorological parameters								
• Wind speed	0–144 mph[c]	E3	None	None	2 (1 at 10 m and 1 at 60 m)	Non-1E	No	Differential temperature calculated from temperature measurements at 10 and 60 meters.
• Wind direction	0 degrees–360 degrees[d]				2 (1 at 10 m and 1 at 60 m)			
• Differential temperature	–40°F to 140°F[e]				2 (1 at 10 m and 1 at 60 m)			

a) This table supplements DCD Table 7.5-1 and provides the site specific information to address the note in the “Remarks” column of DCD Table 7.5-1.
b) These instruments conform to Regulatory Guide 1.97, Revision 3.
c) System accuracy ±0.011 mph @ 0–5 mph, ±0.11 percent @ 50 mph and ±0.11 percent @ 100 mph.
d) System accuracy ±0.22 degrees.
e) System accuracy 0.17°F (for –0.6°F to 107.7°F). Range specified is for individual temperature instruments.

Table 7.5-202
Summary of Type E Variables[a]

VCS COL 7.5-1

Function Monitored	Variable	Type/Category
Environs Radiation and Radioactivity	Plant Environs radiation levels and airborne radioactivity	E3
Meteorology	Wind speed, wind direction, and estimation of atmospheric stability (based on vertical temperature difference)	E3

a) This table supplements DCD Table 7.5-8 and provides the site specific information noted in the “Variable” column of DCD Table 7.5-8.

7.6 INTERLOCK SYSTEMS IMPORTANT TO SAFETY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.7 CONTROL AND INSTRUMENTATION SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 8
ELECTRIC POWER

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ELECTRIC POWER

8.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.1.1 UTILITY GRID DESCRIPTION

Supplement the information in DCD Subsection 8.1.1 with the following information.

VCS SUP 8.1-1 The VCSNS site consists of the existing VCSNS Unit 1 and the new VCSNS Units 2 and 3 reactors. VCSNS Unit 1 connects to the 115kV and 230kV SCE&G transmission systems via an existing 230kV switchyard and a 115kV transmission line. VCSNS Units 2 and 3 will connect to the 230kV SCE&G transmission system via a new 230kV switchyard. The SCE&G transmission system operator (TSO) is responsible for the safe and reliable operation of the electrical transmission system. The SCE&G transmission system consists of interconnected hydro plants, fossil-fueled plants, combustion turbine units and nuclear plants supplying energy to the service area at various voltages up to 230 kV. The transmission system is interconnected with neighboring utilities, and together, they form the Virginia-Carolina (VACAR) Sub region of the Southeastern Electric Reliability Council (SERC). As of January 2009, interconnected systems at 115kV and 230kV include Santee Cooper, Duke Energy, Progress Energy (East), Southeastern Power Administration (SEPA), and Southern Company.

The VCSNS Units 2 and 3 switchyard is tied to the following 230 kV transmission systems:

- SCE&G
- Santee Cooper
- Duke Energy

The switchyard is connected to each generating unit with two overhead tie-lines. One of these lines is connected to the plant main transformer circuit breaker and used for power export to the transmission system or for back feeding station loads when there is no generation. The second line is connected to the reserve auxiliary transformer circuit breaker and used when the unit auxiliary transformers are not available. Three overhead transmission lines connect the Units 2 and 3 switchyard to the Unit 1 switchyard. In addition, there are six overhead transmission lines connecting to the SCE&G transmission system, two overhead lines connecting to the Santee Cooper system, and one line connecting to the Duke Energy system.

Units 2 and 3 are served by a 230kV switchyard. The switchyard consists of ten bays, eight bays configured in a breaker-and-a-half arrangement and two bays configured in a double breaker arrangement. The Unit 2 and 3 main generator step-up transformer (GSU) and reserve auxiliary transformer (RAT) lines are connected to the east side of the switchyard and travel southeast to the plant. The three Unit 1 tie-lines, nine overhead transmission lines, and the two RAT lines are connected to both buses through the breaker-and-a-half arrangement. Two GSU lines are connected through a double breaker arrangement.

8.1.4.3 Design Criteria, Regulatory Guides, and IEEE Standards

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Add the following information between the second and third paragraphs of this subsection.

VCS SUP 8.1-2 Offsite and onsite ac power systems' conformance to Regulatory Guides and IEEE Standards identified by DCD Table 8.1-1 as site-specific is as indicated in Table 8.1-201.

VCS SUP 8.1-2

Table 8.1-201
Site-Specific Guidelines for Electric Power Systems

			Criteria	Applicability (FSAR[a] Section/Subsection) 8.2	8.3.1	8.3.2	Remarks
1.	Regulatory Guides						
	a.	RG 1.129	Maintenance, Testing, and Replacement of Vented Lead—Acid Storage Batteries for Nuclear Power Plants			G	Battery Service tests are performed in accordance with the Regulatory Guide.
	b.	RG 1.155	Station Blackout				Not applicable[b]
	c.	RG 1.204	Guidelines for Lightning Protection of Nuclear Power Plants	G	G		Implemented via IEEE 665.
	d.	RG 1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	G	G	G	
2.	Branch Technical Positions						
	a.	BTP 8-3 (BTP ICSB-11 in DCD)	Stability of Offsite Power Systems	G			Stability Analysis of the Offsite Power System is performed in accordance with the BTP.

a) "G" denotes guidelines as defined in NUREG-0800, Rev. 3, Table 8-1 (SRP). No letter denotes "Not Applicable."
b) Station Blackout and the associated guidelines were addressed as a design issue in the DCD.

8.2 OFFSITE POWER SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.2.1 SYSTEM DESCRIPTION

Replace the first, second, and sixth paragraphs of DCD Subsection 8.2.1 with the following information. The new information is placed before the fifth paragraph of DCD Subsection 8.2.1.

VCS COL 8.2-1 VCSNS Units 2 and 3 are connected into a network supplying medium-sized load centers. The VCSNS Units 2 and 3 switchyard is tied to the following 230 kV transmission systems: SCE&G, Santee Cooper, and Duke Energy. The interconnection of Units 2 and 3, the switchyard, and the 230kV transmission lines are shown on Figure 8.2-201.

There are 12 overhead transmission lines connecting the new 230kV switchyard to other substations. (Eight lines, three new lines, and five reterminated lines are required for Unit 2, and four additional lines are required for Unit 3. These include the lines listed in the table below as well as three tie lines to VCSNS Unit 1.) Each line connected to the switchyard has the capacity to feed the design house loads for both units under all design conditions without relying on the other unit's generator. Three of these lines are short tie-lines, running in an easterly direction, connecting to the Unit 1 switchyard. (Two lines are required for Unit 2 and the third line is required for Unit 3.) Each is approximately one mile long with a thermal rating of 950MVA. The remaining nine 230kV lines originate at the Units 2 and 3 switchyard and connect to various substations as shown below.

230kV Line	Existing/New	Length (miles)	Line Owner
Denny Terrace	Existing	26	SCE&G
Lake Murray 1	Existing	23	SCE&G
Lake Murray 2	New	23	SCE&G
St. George 1	New	86	SCE&G
St. George 2	New	86	SCE&G
Ward	Existing	35	SCE&G
Varnville[a]	New	40	Santee Cooper
Newberry	Existing	17	Santee Cooper
Bush River	Existing	30	Duke Energy

a) Routed via Sandy Run

The layout of transmission lines to the new and existing substations minimizes the crossing of transmission lines to the extent possible. All structures for these transmission lines are designed to meet the National Electrical Safety Code clearance requirements and SCE&G and Santee Cooper engineering standards. Each phase is designed using a conductor bundle comprised of two aluminum conductor, steel reinforced conductors. All structures are grounded with either ground rods or a counterpoise system and have provisions for two overhead ground wires.

VCS CDI A transformer area containing the GSU, the unit auxiliary transformers (UATs), and the RATs is located next to each turbine building. An area containing the GSU and RAT circuit breakers and disconnects is located approximately 150 feet to the west of the transformer area.

8.2.1.1 Transmission Switchyard

Supplement the information in DCD Subsection 8.2.1.1 with the following.

VCS COL 8.2-1 A new 230kV switchyard is used to transmit electrical power output from Units 2 and 3 to the SCE&G, the Santee Cooper, and the Duke Energy 230kV transmission systems. The switchyard is also used as a power source for plant auxiliaries when the units are in the startup or shutdown modes, or when the units are not generating.

The 230kV switchyard is air-insulated and consists of ten bays, eight in a breaker-and-a-half arrangement and two in a double breaker arrangement, located about 2000 feet northwest of Units 2 and 3. The switchyard is connected to each generating unit with two overhead tie-lines. One line per unit is connected to the GSU circuit breaker and used for power export to the transmission system or for back feeding station loads when there is no generation. The second line is connected to the RAT circuit breaker and used when the UATs are not available. Information on the switchyard connections to the transmission system is provided in Subsection 8.2.1.

Failure Analysis

VCS SUP 8.2-1 The circuit breakers in the 230kV switchyard are sized with sufficient continuous current carrying capacity and fault interrupting capability to perform their intended function. The 230kV switchyard disconnect switches are rated on the same continuous current basis as the circuit breakers. The 230kV switchyard single-line diagram is shown on Figure 8.2-202.

The switchyard configuration provides for two main buses, both normally energized. The advantages of the breaker-and-a-half and double breaker schemes include:

- High reliability and operational flexibility
- Capability of isolating any circuit breaker or either main bus for maintenance without service interruption
- A bus fault does not interrupt service
- Double feed to each circuit
- All switching can be done with circuit breakers

The breaker-and-a-half configuration has multiple connections made between the buses. Each connection between the buses is made with three circuit breakers and two circuits (referred to as a bay). This arrangement allows for breaker maintenance without interruption of service. A fault on either bus will cause no circuit interruption. A breaker failure may result in the loss of two circuits (one bay) if a common breaker fails and only one circuit if an outside breaker fails. As described in Subsection 8.2.1.2.2, the protective relay schemes are designed to provide redundancy such that adequate protection is provided given a failure of any single component of the protective relaying system.

The connections for each RAT and each generator output are connected to different bays. A fault and a subsequent breaker failure will not isolate more than one bay. The double breaker configuration for the GSU connections further isolate the GSU from line faults by allowing the GSU to connect to the switchyard on a dedicated bay.

The features of the switchyard (i.e., breaker-and-a-half/double breaker switchyard, choice of termination points for GSU and RAT, line routing, protective relaying schemes, generator breaker, and fast bus transfer) ensure a highly reliable connection to the grid.

Transmission System Provider/Operator (TSP/TSO)

VCS SUP 8.2-2 The SCE&G transmission system operator (TSO) is responsible for the safe and reliable operation of the electrical transmission system. TSO and the Operations Departments for the VCSNS nuclear plants have formal agreements and protocols to provide safe and reliable operation of the transmission system and equipment at the nuclear plants in accordance with North American Electric Reliability Corporation (NERC) Standard NUC-001-01. Elements of this agreement are implemented in accordance with the procedures of both parties.

TSO continuously monitors and evaluates grid reliability and switchyard voltages, and informs the nuclear plant operators of any grid instability or voltage inadequacies. They also work to maintain local voltage requirements as required by the nuclear plant. The nuclear plant operators review the transmission system parameters and inform TSO immediately prior to initiating any plant activities that may affect grid reliability. In addition, the nuclear plant operators inform TSO of changes in generation ramp rates and notify them of any developing problems that may impact generation.

VCS SUP 8.2-3 As set forth in NERC Reliability Standard NUC-001-1, the formal agreement between Nuclear Plant Generator Operators (described here as VCSNS Operations Department) and Transmission Entities (described here as SCE&G TSO) establishes the Nuclear Plant Interface Requirements, such as transmission system studies and analyses. TSO performs short-term grid analyses to support VCSNS plant startup and normal shutdown. Long-term grid studies, done at a minimum of every 36 months, are performed and coordinated with the VCSNS Operations Departments. Studies of future load growth and new generation additions are performed yearly in accordance with NERC and Virginia-Carolinas Reliability Council standards.

New large generating units requesting to connect to the area bulk electric system are required to complete the Large Generator Interconnection Procedure. The studies performed by TSO as part of this procedure examine the generating unit (combined turbine-generator-exciter) and the main step-up transformer(s).

VCS SUP 8.2-4 The agreement between TSO and the VCSNS Operations Departments demonstrates protocols in place for the plant to remain cognizant of grid vulnerabilities to make informed decisions regarding maintenance activities critical to the electrical system. In the operations horizon, the TSO continuously monitors real-time power flows and assesses contingency impacts through use of a state-estimator tool. Operational planning studies are also performed using offline power flow study tools to assess near-term operating conditions under varying load, generation, and transmission topology patterns.

8.2.1.2 Transformer Area

Add the following paragraph and subsections at the end of the DCD Subsection 8.2.1.2.

VCS COL 8.2-1 The transformer area for each unit contains the GSU (three single-phase transformers plus one spare), three UATs, and two RATs. An area approximately

150 feet to the west of the transformer area contains the GSU and RAT circuit breakers and disconnects. The GSU circuit breaker is connected to the 230kV switchyard with an overhead tie-line. The two RATs are also connected to the 230kV switchyard via an overhead tie-line connected to the RAT circuit breaker.

8.2.1.2.1 Switchyard Transformer Ratings

The RATs do not have intermediate supply transformers since they are fed from the 230kV system.

8.2.1.2.2 Switchyard Protection Relay Scheme

VCS COL 8.2-2 The switchyard is designed to provide high-speed fault clearing while also maintaining high reliability and operational flexibility. The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.

Under normal operating conditions, all 230kV breakers and disconnect switches are closed.

The protective relay schemes are designed to provide redundancy such that adequate protection is provided given a failure of any single component of the system. Primary protective relays are supplied with current transformer inputs, potential transformer inputs, and DC supplies that are independent of the same inputs to backup relays. The primary and backup relays trip 230kV circuit breakers via two independent trip coils supplied from two separate DC sources.

All 230kV circuit breakers are provided with a breaker failure scheme to rapidly clear faults due to a failed breaker.

Each 230kV transmission line is protected by two independent high-speed relaying schemes, each scheme using a different type of protection. The short 230kV tie-lines to Unit 1 and the tie-lines to the GSU and RAT circuit breakers also use two independent high-speed protection schemes, but each scheme may be of the same or similar type.

8.2.1.3 Switchyard Control Building

VCS COL 8.2-1 A control house is provided to serve the requirements of the switchyard. It contains the control and protective relay panels, redundant battery systems, and other panels and equipment required for switchyard operation.

All 230kV circuit breakers are controlled under the supervision of TSO.

8.2.1.4 Switchyard and Transmission Lines Testing and Inspection

An agreement between TSO and the VCSNS Operations Departments for development, maintenance, calibration, testing and modification of transmission lines, switchyards, transformer yards, and associated transmission equipment provides the procedure, policy, and organization to carry out maintenance, calibration, testing, and inspection of transmission lines and switchyards.

This agreement defines the interfaces and working relationship between TSO and VCSNS. TSO performs maintenance, calibration, and testing of transformer assets at VCSNS. TSO and the Operations Departments for the VCSNS nuclear plants are responsible for control of plant/grid interface activities. For reliability, TSO and the Operations Departments for the VCSNS nuclear plants coordinate maintenance and testing of offsite power systems. The nuclear plant operators establish communication and coordination protocols for restoration of external power supply to the nuclear plant on a priority basis.

For performance of maintenance, testing, calibration, and inspection, TSO follows its own field test manuals, vendor manuals and drawings, and industry's maintenance practices to comply with NERC reliability standards.

TSO verifies that these test results demonstrate compliance with design requirements and takes corrective actions as necessary. TSO plans and schedules maintenance activities and notifies the nuclear plant operators in advance. TSO also procures and stores necessary spare parts prior to the commencement of inspection, testing, and maintenance activities.

Transmission lines are currently inspected through an aerial inspection program twice a year. The inspection has a specific focus on right-of-way encroachments, vegetation management, conductor and line hardware condition assessment, and supporting structures. A vegetation management program is in effect to control vegetation within the boundaries of the transmission line rights-of-way. Where herbicides cannot be applied, vegetation is cut and removed. Vegetation management patrols are performed every one to three years.

The interconnecting switchyard, as well as other substation facilities, has multiple levels of inspection and maintenance. They include:

- Monthly walk-through and visual inspection including reading and recording of equipment counters and meters, site temperature and conditions, and equipment conditions.

- Quarterly oil sampling of power transformers at generating stations. Oil samples are tested for dissolved gas analysis and oil quality.

- Power circuit breakers are inspected and maintained according to the number of operations and length of time in service, in accordance with the breaker manufacturer's recommendations.

- Doble power testing on power transformers.
- Infrared and corona camera testing on bus and equipment to identify hot spots.
- Quarterly testing of battery systems including visual inspection, verification of battery voltage, and electrolyte level. Battery load testing is performed on a periodic basis not to exceed 6 years.
- Protective relay system testing including visual inspection, calibration, verification of current and potential inputs, functional trip testing, and correct operation of relay communication equipment.

8.2.2 GRID STABILITY

Add the following information at the end of DCD Subsection 8.2.2.

VCS COL 8.2-2 A transmission system study of the offsite power system is performed regularly. In order to maintain Reactor Coolant Pump (RCP) operation for three seconds following a turbine trip as specified in DCD Subsection 8.2.2, the grid voltage at the high side of the GSU and RATs must remain above the required AP1000 minimum grid voltage limit.

A forward-looking study was performed that analyzed cases for load flow, transient stability, and fault analysis. The following assumptions were made:

- Typical grid voltage is 232.3kV.
- GSU tap setting is 1.025.
- NERC Multiregional Modeling Working Group summer base cases were used for both the load flow and stability studies. The cases were modified to incorporate changes to the Southern Company and Duke Energy systems to reflect a 2015 case year as well as to incorporate changes to the SCE&G and Santee Cooper systems to reflect 2014 and 2019 case years to perform the analysis of the two Westinghouse AP1000 units.

The computer analysis was performed using the Siemens Power Technology International Software PSS/E, revision 30. The analysis examines five conditions:

- Normal running
- Normal shutdown
- Startup

- Turbine trip
- Not in service

The results of the study conclude that the transmission system remains stable preserving the grid connection, and supporting RCP operation for at least three seconds following a turbine trip under the modeled conditions.

Table 8.2-201 shows that the interface requirements for steady-state load, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and limiting under frequency value for the RCP have been satisfied.

The AP1000 generator 26kV terminal voltages met the requirement of 0.95 per unit – 1.05 per unit for steady-state conditions following transients caused by system contingencies. The results of the study conclude that the transmission system remains stable preserving the grid connection, and supporting RCP operation for at least three seconds following a turbine trip under the modeled conditions.

From 1987 to 2007, the 230kV transmission lines connecting the VCSNS site experienced 113 forced outages. The average frequency of forced line outages since 1987 is approximately 6 per year for the involved lines, with the majority being momentary outages due to lightning strikes or storm damage. The leading causes of forced outages of significant duration are equipment failures, logging and construction activities, and lightning or storm damage.

8.2.5 COMBINED LICENSE INFORMATION FOR OFFSITE ELECTRICAL POWER

VCS COL 8.2-1 This COL item 8.2-1 is addressed in Subsections 8.2.1, 8.2.1.1, 8.2.1.2, 8.2.1.3, and 8.2.1.4.

VCS COL 8.2-2 This COL item 8.2-2 is addressed in Subsections 8.2.1.2.2 and 8.2.2.

VCS COL 8.2-2

Table 8.2-201
Grid Stability Interface Evaluation

DCD Table 1.8-1 Item 8.2 Parameter	WEC AC Requirements	VCSNS 2 & 3 Value Assumed
Steady-state load	"normal running values provided as input to grid stability"	(90.00 + j58.6) MVA*
Inrush kVA for motors	56,712 kVA**	56,712 kVA**
Nominal Voltage	Not provided	1.01 pu (232.3 kV)
Allowable voltage regulation	0.95-1.05 pu steady state 0.15 pu transient dip***	0.95-1.05 pu steady state 0.15 pu transient dip***
Nominal Frequency	60 Hz	Assumed 60 Hz
Allowable frequency fluctuation	±1/2 Hz indefinite	±1/2 Hz indefinite
Maximum frequency decay rate	5 Hz/sec	5 Hz/sec

*Margin added to provide running values to account for site specific mechanical draft cooling tower loads.

**Based on the inrush of a single 10,000 HP feedwater pump assuming efficiency = 0.95, pf=0.9, and inrush =6.5x FLA.

***Applicable to Turbine Trip Only. The maximum allowable voltage dip from the pre-event steady state voltage value during the 3 second turbine trip event transient as measured at the point of connection to the high side of the generator step-up transformer and the reserve auxiliary transformer.

DCD Table 1.8-1 Item 8.2 Parameter	WEC Acceptance Criteria	VCSNS 2 & 3 Value Calculated
Limited under frequency value for RCP	≥57.7 Hz	≥57.7 Hz

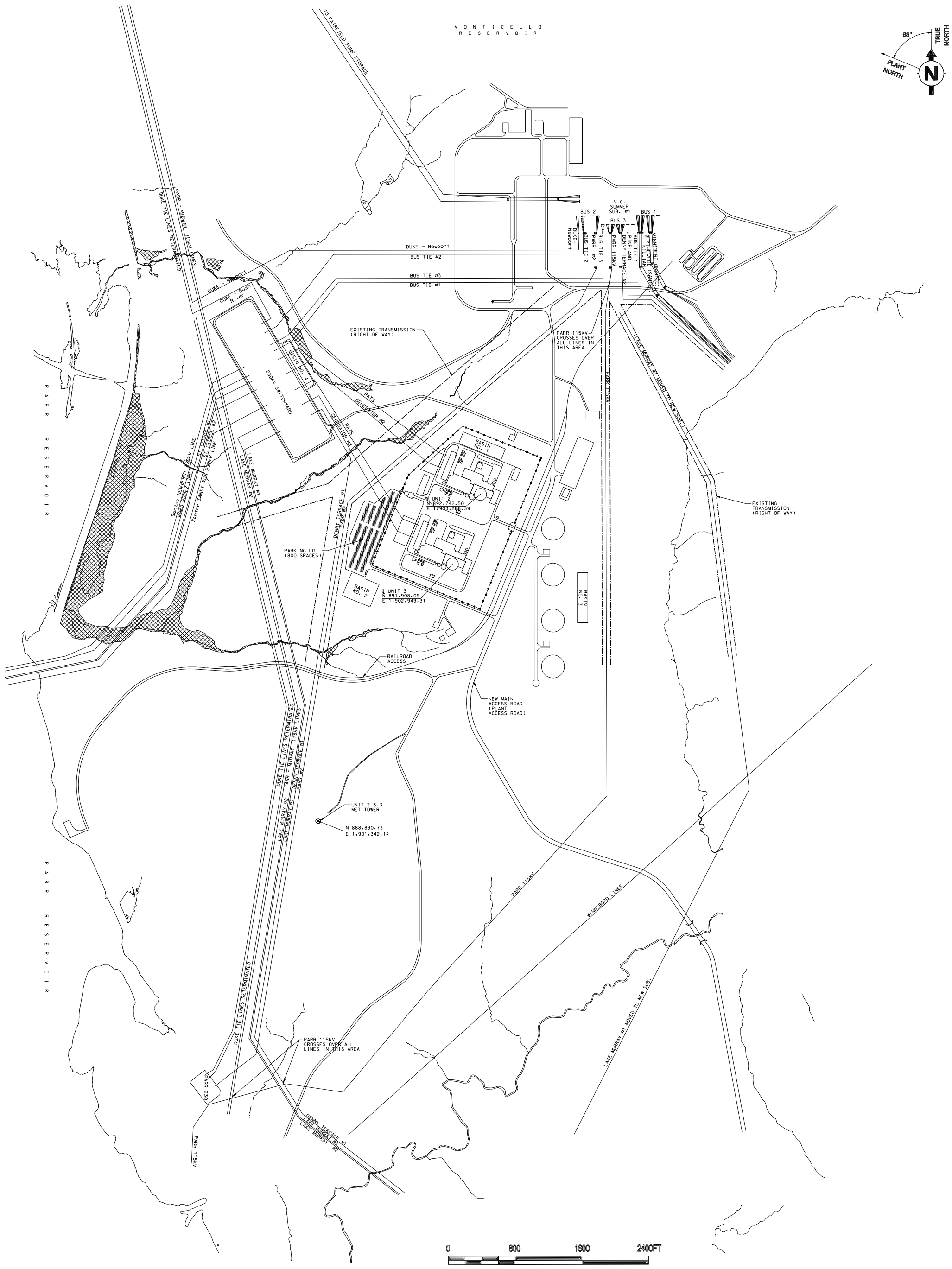

Figure 8.2-201
Site Transmission Line Map

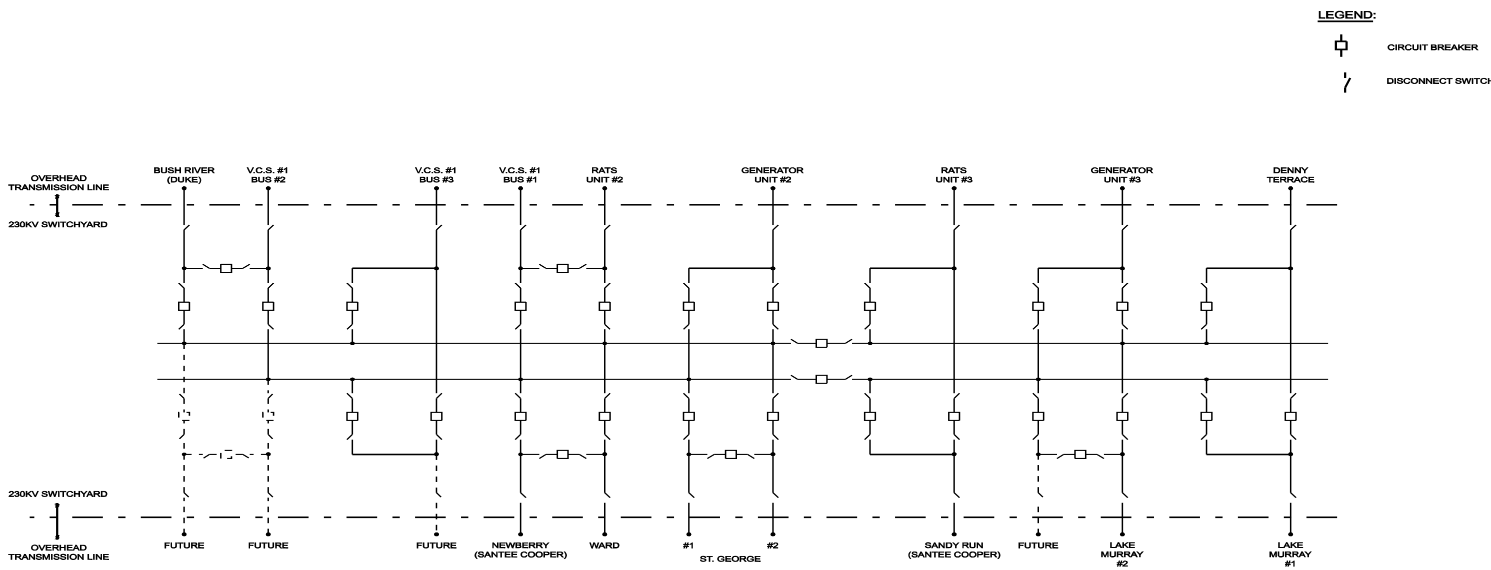

Figure 8.2-202
Switchyard Single-Line Diagram

8.3 ONSITE POWER SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.3.1.1.1 Onsite AC Power System

Add the following to the end of the fourth paragraph of DCD Subsection 8.3.1.1.1.

VCS SUP 8.3-1 The site specific 230 kV switchyard is shown on Figure 8.2-201.

8.3.1.1.2.3 Onsite Standby Power System Performance

Add the following text between the second and third paragraphs of DCD Subsection 8.3.1.1.2.3.

VCS SUP 8.3-2 The VCSNS Units 2 and 3 site conditions provided in Table 2.0-201 and Section 2.3 are bounded by the standard site conditions used to rate both the diesel engine and the associated generator in DCD Subsection 8.3.1.1.2.3.

Add the following subsection after DCD Subsection 8.3.1.1.2.3.

8.3.1.1.2.4 Operation, Inspection, and Maintenance

STD COL 8.3-2 Operation, inspection and maintenance (including preventive, corrective, and predictive maintenance) procedures consider both the diesel generator manufacturer’s recommendations and industry diesel working group recommendations.

8.3.1.1.6 Containment Building Electrical Penetrations

Add the following text at the end of DCD Subsection 8.3.1.1.6.

STD COL 8.3-2 Procedures implement periodic testing of protective devices that provide penetration overcurrent protection. A sample of each different type of overcurrent device is selected for periodic testing during refueling outages. Testing includes:

- Verification of thermal and instantaneous trip characteristics of molded case circuit breakers.
- Verification of long time, short time, and instantaneous trips of medium voltage vacuum circuit breakers.
- Verification of long time, short time, and instantaneous trips of low voltage air circuit breakers.
- Verification of Class 1E and non-Class 1E dc protective device characteristics (except fuses) per manufacturer recommendations, including testing for overcurrent interruption and/or fault current limiting.

Penetration protective devices are maintained and controlled under the plant configuration control program. A fuse control program, including a master fuse list, is established based on industry operating experience.

8.3.1.1.7 Grounding System

Replace the last paragraph of DCD Subsection 8.3.1.1.7 with the following information.

VCS COL 8.3-1 A Grounding Grid System within the Units 2 and 3 site boundary includes step and touch potentials near equipment that are within the acceptable limit for personnel safety. Actual resistivity measurements from soil samples taken at the Units 2 and 3 site were analyzed to create a soil model for the plant site. The ground grid conductor size was then determined using the methodology outlined in IEEE 80, "IEEE Guide for Safety in AC Substation Grounding" (Reference 201), and a grid configuration for the site was created. The grid configuration was modeled in conjunction with the soil model. The resulting step and touch potentials are within the acceptable limit.

8.3.1.1.8 Lightning Protection

Replace the last paragraph of DCD Subsection 8.3.1.1.8 with the following information.

VCS COL 8.3-1 In accordance with IEEE 665, "IEEE Standard for Generating Station Grounding" (DCD Section 8.3 Reference 18), a Lightning Protection Risk Assessment for the Units 2 and 3 buildings was performed based on the methodology in NFPA 780 (DCD Section 8.3 Reference 19). The tolerable lightning frequency for each of the buildings was determined to be less than the expected lightning frequency; therefore, lightning protection is required for the Units 2 and 3 buildings. The zone of protection is based on the elevations and geometry of the structures. It includes the space not intruded by a rolling sphere having a radius prescribed in Reference 19. The zone of protection method is based on the use of ground masts, air terminals, and shield wires. Either copper or aluminum is used for lightning protection. Lightning protection grounding is interconnected with the station or switchyard grounding system.

8.3.1.4 Inspection and Testing

STD SUP 8.3-4 Add the following text at the end of DCD Subsection 8.3.1.4

Procedures are established for periodic verification of proper operation of the Onsite AC Power System capability for automatic and manual transfer from the preferred power supply to the maintenance power supply and return from the maintenance power supply to the preferred power supply.

8.3.2.1.1.1 Class 1E DC Distribution

Add the following text at the end of DCD Subsection 8.3.2.1.1.1.

STD SUP 8.3-3 No site-specific non-Class 1E dc loads are connected to the Class 1E dc system.

8.3.2.1.4 Maintenance and Testing

Add the following text at the end of DCD Subsection 8.3.2.1.4.

STD COL 8.3-2 Procedures are established for inspection and maintenance of Class 1E and non-Class 1E batteries. Class 1E battery maintenance and service testing is performed in conformance with Regulatory Guide 1.129. Batteries are inspected periodically to verify proper electrolyte levels, specific gravity, cell temperature and battery float voltage. Cells are inspected in conformance with IEEE 450 and vendor recommendations.

The clearing of ground faults on the Class 1E dc system is also addressed by procedure. The battery testing procedures are written in conformance with IEEE 450 and the Technical Specifications.

Procedures are established for periodic testing of the Class 1E battery chargers and Class 1E voltage regulating transformers in accordance with the manufacturer recommendations.

- Circuit breakers in the Class 1E battery chargers and Class 1E voltage regulating transformers that are credited for an isolation function are tested through the use of breaker test equipment. This verification confirms the ability of the circuit to perform the designed coordination and corresponding isolation function between Class 1E and non-Class 1E components. Circuit breaker testing is done as part of the Maintenance Rule program and testing frequency is determined by that program.
- Fuses / fuse holders that are included in the isolation circuit are visually inspected.
- Class 1E battery chargers are tested to verify current limiting characteristic utilizing manufacturer recommendation and industry practices. Testing frequency is in accordance with that of the associated battery.

8.3.2.2 Analysis

Replace the first sentence of the third paragraph of DCD Subsection 8.3.2.2 with the following:

STD DEP 8.3-1 The Class 1E battery chargers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side, however, the voltage regulating transformers do not have active components to limit current; therefore, the Class 1E voltage regulating transformer maximum current is determined by the impedance of the transformer.

8.3.3 COMBINED LICENSE INFORMATION FOR ONSITE ELECTRICAL POWER

VCS COL 8.3-1 This COL Item is addressed in Subsections 8.3.1.1.7 and 8.3.1.1.8.

STD COL 8.3-2 This COL Item is addressed in Subsections 8.3.1.1.2.4, 8.3.1.1.6 and 8.3.2.1.4.

8.3.4 REFERENCES

201. Institute of Electrical and Electronics Engineers (IEEE), "IEEE Guide for Safety in AC Substation Grounding," IEEE Std 80-2000, August 4, 2000.

CHAPTER 9
AUXILIARY SYSTEMS

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CHAPTER 9
AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.1.3.1.3.1 Partial Core

Add the following information at the end of the third bullet in DCD Subsection 9.1.3.1.3.1.

VCS DEP 2.0-2 SFS performance following restart after a normal refueling is affected by a change in maximum safety wet bulb temperature. Calculations confirm that spent fuel pool temperature remains below 115°F with a CCS supply temperature of 97°F at the specified pool spent fuel loading condition and decay time on the fuel fraction just replaced during the previous 17 day refueling outage.

While the maximum CCS temperature expected for VCSNS Units 2 and 3 is 97.3°F, an increase of 0.3°F in CCS supply temperature will produce a similar increase in the spent fuel pool maximum temperature; therefore, the requirement to maintain spent fuel temperature below 120°F is met with margin (Reference 201).

Add the following subsection after DCD Subsection 9.1.4.3.7.

9.1.4.3.8 Radiation Monitoring

STD COL 9.1-6 Plant procedures require that an operating radiation monitor is mounted on any machine when it is handling fuel. Refer to DCD Subsection 11.5.6.4 for a discussion of augmented radiation monitoring during fuel handling operations.

9.1.4.4 Inspection and Testing Requirements

Add the following paragraph at the end of DCD Subsection 9.1.4.4.

STD COL 9.1-5 The above requirements are part of the plant inspection program for the light load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection. The light load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5 OVERHEAD HEAVY LOAD HANDLING SYSTEMS

Add the following at the end of DCD Subsection 9.1.5.

STD SUP 9.1-2 The heavy loads handling program is based on NUREG 0612 and vendor recommendations. The key elements of the program are:

- Listing of heavy loads to be lifted during operation of the plant. This list will be provided once magnitudes have been accurately formalized but no later than three (3) months prior to fuel receipt.
- Listing of heavy load handling equipment as outlined in DCD Table 9.1-5 and whose characteristics are described in Subsection 9.1.5 of the DCD.
- Heavy load handling safe load paths and routing plans including descriptions of interlocks, (automatic and manual) safety devices and procedures to assure safe load path compliance. Anticipated heavy load movements are analyzed and safe load paths defined. Safe load path considerations are based on comparison with analyzed cases, previously defined safe movement areas, and previously defined restricted areas. The analyses are in accordance with Appendix A of NUREG 0612.
- Heavy load handling equipment maintenance manuals and procedures as described in Subsection 9.1.5.5.
- Heavy load handling equipment inspection and test plans, as outlined in Subsections 9.1.5.4 and 9.1.5.5.
- Heavy load handling personnel qualifications, training, and control procedures as described in Subsection 9.1.5.5.
- QA programs to monitor, implement, and ensure compliance with the heavy load-handling procedures as described in Subsection 9.1.5.5.

A quality assurance program, consistent with Paragraph 10 of NUREG-0554, is established and implemented for the procurement, design, fabrication, installation, inspection, testing, and operation of the crane. The program, as a minimum, includes the following elements:

- design and procurement document control

- instructions, procedures, and drawings
- control of purchased material, equipment, and services
- inspection
- testing and test control
- non-conforming items
- corrective action
- records

9.1.5.3 Safety Evaluation

Add the following information at the end of DCD Subsection 9.1.5.3.

STD SUP 9.1-1 There are no planned heavy load lifts outside those already described in the DCD. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e. in support of special maintenance/ repairs). For these occasions, special procedures are generated that address, as a minimum, the following:

- The special procedure complies with NUREG-0612.
- A safe load path is determined. Mechanical and/or electrical stops are incorporated in the hardware design to prohibit travel outside the safe load path. Maximum lift heights are specified to minimize the impact of an unlikely load drop.
- Where a load drop could occur over irradiated fuel or safe shutdown equipment, the consequence of the load drop is evaluated. If the evaluation concludes that the load drop is not acceptable, an alternate path is evaluated, or the lift is prohibited.
- The lifting equipment is in compliance with applicable ANSI standards and has factors of safety that meet or exceed the requirements of the applicable standards.
- Operator training is provided prior to actual lifts.
- Inspection of crane components is performed in accordance with the manufacturer recommendations.

STD COL 9.1-6 Plant procedures require that an operating radiation monitor is mounted on any crane when it is handling fuel. Refer to DCD Subsection 11.5.6.4 for a discussion of augmented radiation monitoring during fuel handling operations.

9.1.5.4 Inservice Inspection/Inservice Testing

Add the following paragraph at the end of DCD Subsection 9.1.5.4.

STD COL 9.1-5 The above requirements are part of the plant inspection program for the overhead heavy load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection and the NUREG-0612 recommendations.

The overhead heavy load handling equipment inservice inspection procedures, as a minimum, address the following:

- Identification of components to be examined
- Examination techniques
- Inspection Intervals
- Examination categories and requirements
- Evaluation of examination results

The overhead heavy load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5.5 Load Handling Procedures

STD SUP 9.1-3 Load handling operations for heavy loads that are handled over, could be handled over or are in the proximity of irradiated fuel or safe shutdown equipment are controlled by written procedures. As a minimum, procedures are used for handling loads with the spent fuel cask bridge and polar cranes, and for those loads listed in Table 3.1-1 of NUREG 0612. The procedures include and address the following elements:

- The specific equipment required to handle load (e.g., special lifting devices, slings, shackles, turnbuckles, clevises, load cells, etc.).

- Qualification and training of crane operators and riggers in accordance with chapter 2-3.1 of ASME B30.2, “Overhead and Gantry Cranes.”
- The requirements for inspection and acceptance criteria prior to load movement.
- The defined safe load path and provisions to provide visual reference to the crane operator and/or signal person of the safe load path envelope.
- Specific steps and proper sequence to be followed for handling load.
- Precautions, limitations, prerequisites, and/or initial conditions associated with movement of heavy loads.
- The testing, inspection, acceptance criteria and maintenance of overhead heavy load handling systems. These procedures are in accordance with the manufacturer recommendations and are consistent with ANSI B30.2 or with other appropriate and applicable ANSI standards.

Safe load paths are defined for movement of heavy loads to minimize the potential for a load drop on irradiated fuel in the reactor vessel, spent fuel pool or safe shutdown equipment. Paths are defined clearly in procedures and equipment layout drawings. Equipment layout drawings showing the safe load path are used to define safe load paths in load handling procedures. Deviation from defined safe load paths requires a written alternative procedure approved by a plant safety review committee.

9.1.6 COMBINED LICENSE INFORMATION FOR FUEL STORAGE AND HANDLING

STD COL 9.1-5 This COL Item is addressed in Subsections 9.1.4.4 and 9.1.5.4.

STD COL 9.1-6 This COL Item is addressed in Subsections 9.1.4.3.8 and 9.1.5.3.

STD COL 9.1-7 A spent fuel rack Metamic coupon monitoring program will be implemented when the plant is placed into commercial operation. This program will include tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and / or visual examination. The program will also include testing to monitor changes in physical properties of the absorber material, including neutron attenuation and thickness measurements.

The program will include the methodology and acceptance criteria for the tests listed and provide corrective action requirements based on vendor recommendations and industry operating experience. The program will be implemented through plant procedures.

Metamic Monitoring Acceptance Criteria:

- Verification of continued presence of the boron is performed by neutron attenuation measurement. A decrease of no more than 5% in Boron-10 content, as determined by neutron attenuation, is acceptable. This is equivalent to a requirement for no loss in boron within the accuracy of the measurement.
- Coupons are monitored for unacceptable swelling by measuring coupon thickness. An increase in coupon thickness at any point of no more than 10% of the initial thickness at that point is acceptable.

Changes in excess of either of the above two acceptance criteria are investigated under the corrective action program and may require early retrieval and measurement of one or more of the remaining coupons to provide validation that the indicated changes are real. If the deviation is determined to be real, an engineering evaluation is performed to identify further testing or any corrective action that may be necessary.

Additional parameters are examined for early indications of the potential onset of Metamic degradation that would suggest a need for further attention and possibly a change in the coupon withdrawal schedule. These include visual inspection for surface pitting, blistering, cracking, corrosion or edge deterioration, or unaccountable weight loss in excess of the measurement accuracy.

9.1.7 REFERENCES

201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature For the V.C. Summer Site, VSP_VSG_000706, June 30, 2010.

9.2 WATER SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.2.1.2.2 Component Description

Add the following paragraph at the end of DCD Subsection 9.2.1.2.2, "Component Description," subsection "Cooling Tower":

VCS SUP 9.2-3 The SWS cooling tower was evaluated for potential impacts from interference and air restriction effects due to yard equipment layout and tower operation in an adjacent unit. Based on unit spacing, yard equipment layout, and the margins inherent in the performance requirements and design conditions of the towers, no adverse impacts were determined.

9.2.2.1 Design Basis

Replace the first bullet item in the criteria for normal operation in DCD Subsection 9.2.2.1.2.1 with the following information.

VCS DEP 2.0-2
- The component cooling water supply temperature to plant components is not more than 100°F assuming a 100-year return estimate of 2-hour duration wet bulb temperature of 87.3°F for service water cooling (per Table 2.0-201).

  The most limiting component cooled by the CCS, the RCP motor cooling system, has been designed to operate for at least 6 hours continually with cooling water supplied at temperatures up to 100°F.

  The performance of the standard AP1000 CCS and SWS for single cooling water train, full power operation at a maximum safety wet bulb temperature of 87.4°F has demonstrated the highest CCS temperature achieved at these conditions is 97.4°F, for a period of less than 2 hours. As ambient wet bulb temperature decreases, the CCS temperature follows and will return to below 95°F with ambient wet bulb temperature slightly lower than 84°F, assuming nominal performance of both the CCS and SWS. Since the definition of the maximum normal wet bulb temperature value is the seasonal 1% exceedance value observed at the site, the annual total operating time for which CCS temperature could exceed 95°F is less than 30 hours per year, for periods of a few hours at most. The maximum CCS temperature of 97.3° is bounded by the maximum allowable cooling water temperature for Reactor Coolant Pumps (the most limiting component) and the increase in maximum safety wet bulb temperature is therefore acceptable on this basis (Reference 201).

9.2.5.2.1 General Description

Modify the second paragraph of DCD Subsection 9.2.5.2.1 as follows:

VCS COL 9.2-1 Potable water is supplied from a nearby water treatment facility that withdraws raw water from the Monticello Reservoir for treatment, storage, and transfer to the plant potable water system. The potable water system consists of a distribution header around the power block, hot water storage heaters, and necessary interconnecting piping and valves. The water treatment facility includes the tankage, pumps, and water treatment equipment necessary to provide potable water through a supply line to the distribution headers to meet design pressure and capacity requirements of the potable water system. Sodium hypochlorite is used as the biocide for the potable water system.

9.2.5.3 System Operation

Add the following after the first paragraph of DCD Subsection 9.2.5.3 as follows:

VCS COL 9.2-1 The site specific water source described above is considered to be the off-site water treatment facility. Filtered water described above is generated and disinfected at the off-site water treatment facility to provide a make-up source of drinking water to the Potable Water System (PWS). The location of the off-site water treatment facility is shown on FSAR Figure 1.1-202, “VCSNS Site Plan.” This facility also provides a make-up source of filtered water to the Raw Water System (RWS) to support loads described in Subsection 9.2.11 of the FSAR. The facility is depicted on FSAR Figure 9.2-201, “Raw Water System Flow Diagram.”

Add the following after the second paragraph of DCD Subsection 9.2.5.3 as follows:

VCS COL 9.2-1 The onsite water supply system described above is considered to be the off-site water treatment facility.

Add the following after the fourth paragraph of DCD Subsection 9.2.5.3 as follows:

VCS COL 9.2-1 The possibility for the PWS to become contaminated radioactively does not exist. The Raw Water System (RWS) does not have the potential to be a flowpath for radioactive fluids. Because RWS does not have the potential to be a flowpath for radioactive fluids, its filtered water make-up source from the off-site water treatment facility does not have the potential to be contaminated radioactively. Since the only association the make-up water to RWS has with the make-up water supply to PWS is the off-site water treatment facility, the possibility for PWS to become contaminated radioactively does not exist.

9.2.6.2.1 General Description

Add the following paragraph at the end of DCD Subsection 9.2.6.2.1.

VCS SUP 9.2-1 The waste treatment plant is preengineered and prefabricated, of modular construction, and includes components such as blowers; equalization, aeration, and sludge holding tanks; clarifiers; and disinfection units; which are used in a multistep process to treat sanitary waste prior to effluent discharge to the blowdown sump where it combines with other plant discharge streams for discharge to the Parr Reservoir.

9.2.7.2.4 System Operation

Add the following information at the end of the first paragraph under “Normal Operation” in DCD Subsection 9.2.7.2.4.

VCS DEP 2.0-2 The increased heat load produced by operation at the higher VCSNS maximum safety ambient wet bulb temperature of 87.3°F can be accommodated within the available capacity margin of the chiller units, without impacting the LCCWS or supporting systems design or plant operation. Cooling coil design calculations indicate that during operation at the standard plant design temperatures (115°F dry bulb, 86.1° wet bulb), the VBS air handling unit has cooling coil and system margin (Reference 201).

9.2.8 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

Modify the first paragraph of DCD Subsection 9.2.8 as follows:

VCS CDI The turbine building closed cooling water system (TCS) provides chemically treated, demineralized cooling water for the removal of heat from nonsafety-related heat exchangers in the turbine building and rejects the heat to the circulating water system (CWS). When the CWS is not in operation and only minor heat loads attributed to sporadic cooling of components under shutdown conditions or in preparation for startup, the TCS may reject heat to water supplied by the raw water system (RWS) to the CWS.

9.2.8.1.2 Power Generation Design Basis

Modify the second paragraph of DCD Subsection 9.2.8.1.2 as follows:

VCS CDI During power operation, the turbine building closed cooling water system provides a continuous supply of cooling water to turbine building equipment at a temperature of 105°F or less assuming a circulating water temperature of 100°F or less.

Modify the fourth paragraph of DCD Subsection 9.2.8.1.2 as follows:

VCS CDI The heat sink for the turbine building closed cooling water system is the CWS during power operation. The heat is transferred to the CWS through plate type heat exchangers which are components of the turbine building closed cooling water system.

9.2.8.2.1 System Description

Modify the last sentence of the first paragraph of DCD Subsection 9.2.8.2.1 as follows:

VCS CDI Heat is removed from the turbine building closed cooling water system by the CWS, or water supplied by the RWS to the CWS when applicable, via the heat exchangers.

9.2.8.2.2 Component Description

Modify the second and third sentences of the second paragraph under Heat Exchangers of DCD Subsection 9.2.8.2.2 as follows:

VCS CDI Turbine building closed cooling water circulates through one side of the heat exchanger while circulating water, or water supplied by the RWS to the CWS when applicable, flows through the other side. During system operation, the turbine building closed cooling water in the heat exchanger is maintained at a higher pressure than the circulating water, or raw water, so leakage of the circulating water, or raw water, into the closed cooling water system does not occur.

9.2.8.2.3 System Operation

Modify the first sentence of the first paragraph under Startup of DCD Subsection 9.2.8.2.3 as follows:

VCS CDI The turbine building closed cooling water system is placed in operation during the plant startup sequence after cooling water flow from the CWS, or RWS when applicable, is established but prior to the operation of systems that require turbine building closed cooling water flow.

9.2.9.2.2 Component Description

Add the following text under the Waste Water Retention Basin paragraph of DCD Subsection 9.2.9.2.2 and add Basin Transfer Pumps as follows:

VCS COL 9.2-2 The waste water retention basin is constructed using formed concrete and is a lined basin constructed such that its contents, dissolved or suspended, do not penetrate the liner and leach into the ground. Each Unit's Waste Water Retention Basin (WWRB) is located in the yard area outside of each Unit's respective Turbine Building. The WWRB is designed to allow entrained solids to settle and allow for chemical treatment of effluent concentrations required for release prior to discharge to the blowdown sump.

The configuration and size of the waste water retention basin allows settling of solids larger than 10 microns that may be suspended in the waste water stream. Waste water can be sampled prior to discharge from the waste water retention basin.

Each WWRB is divided into two separate compartments, which allows one compartment to be out of service while the other compartment is available. Each compartment discharges to a pump sump. A level transmitter located in each WWRB pump sump provides an alarm signal in the Main Control Room when the sump level(s) reach predetermined set points.

Basin Transfer Pumps

In the event of oily waste leakage into the retention basin, a recirculation line is provided to recycle the oil/water waste from the basin to the oil separator. The WWRB transfer pumps are located in pump sumps adjacent to each compartment.

The pumps are manually started and interlocked to stop based on sump level. There are two (one per sump) 100% capacity transfer pumps for each WWRB. The transfer pumps are sized to meet the maximum expected influent flow. The normal pump discharge flowpath is to the blowdown sump. Flow can also be directed to the other Unit's WWRB.

Blowdown Sump/Plant Outfall

The blowdown sump is a concrete structure and is open to the atmosphere. It is a common sump and accepts waste water from both Units' WWRBs, CWS cooling tower blowdown from both Units and sanitary waste effluent. In the absence of CWS cooling tower blowdown, RWS supplies an alternate source of dilution water. The outfall pipe is sized with adequate capacity to gravity drain the blowdown sump at the highest anticipated influent flow rate. Wastewater and blowdown effluent from the blowdown sump drains by gravity to Parr Reservoir via the plant outfall piping. Location of the plant outfall routing is shown on FSAR Figure 1.1-202.

STD DEP 1.1-1 Add the following subsection after DCD Subsection 9.2.10. DCD Subsections 9.2.11 and 9.2.12 are renumbered as Subsections 9.2.12 and 9.2.13, respectively.

9.2.11 RAW WATER SYSTEM

VCS SUP 9.2-2 The RWS supplies unfiltered water from the Monticello Reservoir for CWS cooling tower makeup and an alternate source of water makeup for the SWS cooling tower. A nearby water treatment facility provides filtered water for distribution by the Ancillary RWS Subsystem for normal supply to the demineralized water treatment system, fire protection system, normal makeup to the SWS cooling tower, and miscellaneous users. The RWS also provides water for dilution of liquid radwaste when CWS blowdown is not sufficient or available for that purpose. The RWS may also be used to provide an alternate means of cooling the turbine building closed cooling water and condenser vacuum pump seal water heat exchangers.

9.2.11.1 Design Basis

9.2.11.1.1 Safety Design Basis

The RWS serves no safety-related function and therefore has no nuclear safety design basis.

Failure of the RWS or its components will not affect the ability of safety-related systems to perform their intended function.

The RWS does not have the potential to be a flow path for radioactive fluids.

9.2.11.1.2 Power Generation Design Basis

9.2.11.1.2.1 Normal Operation

The RWS provides continuous makeup to the circulating water cooling tower basins to replace water losses due to evaporation, drift, and blowdown. The RWS also supplements or is used in place of CWS blowdown for dilution of liquid radwaste during discharge through the plant outfall.

The RWS provides filtered water from the water treatment facility for supply of the following:

- Service water system cooling tower basin makeup
- Demineralized water treatment system supply
- Primary and secondary fire water tank fill and makeup

The RWS also provides an alternate supply of unfiltered water for the following:

- Service water system alternate makeup
- Turbine building closed cooling water heat exchanger cooling
- Condenser vacuum pump seal water heat exchanger cooling

9.2.11.1.2.2 Outage Mode Operation

During plant outages, RWS provides the same continuous water supplies as during normal operation with the exception of CWS cooling tower makeup. RWS provides an alternate dilution flowpath for WLS when CWS blowdown is not available.

9.2.11.2 System Description

9.2.11.2.1 General Description

The RWS for a single unit is shown in Figure 9.2-201. Additional components and instrumentation are included as necessary for use and operation of the system. Classification of components and equipment for the RWS is given in Section 3.2.

The raw water pumps are located in individual bays of the raw water pump intake structure, a common structure for both units. Water withdrawn from the Monticello Reservoir passes through trash racks and traveling water screens before entering the pump suctions. The raw water pumps supply makeup water to the CWS at the cooling tower basins. A flow path is included to provide raw water for dilution of liquid radwaste discharged through the blowdown sump and plant outfall when CWS blowdown is not sufficient or available for dilution purposes. Flow can also be provided to the turbine building closed cooling water and condenser vacuum pump seal water heat exchangers for cooling when the CWS is not in operation. A screen wash system for the traveling water screens is provided at the intake structure.

Raw water for makeup to the service water system, fill and makeup to the fire water storage tanks, and feed to the demineralized water treatment system is provided from a water treatment facility that withdraws water from the Monticello Reservoir separate from the raw water pump intake structure. If conditions warrant, flow can also be provided for makeup to the service water system using the raw water pumps.

Provisions are included to inject chemicals into the raw water pump discharge piping to maintain a noncorrosive, nonscale-forming condition and limit biological fouling. Chemical treatment may also be performed at locations downstream of the raw water pumps and in or downstream of the water treatment facility to satisfy the supply water quality requirements of the systems to which the water is provided.

9.2.11.2.2 Component Description

Major components of the RWS are described below to provide an understanding of the operation and reliability of the system.

Intake

The raw water intake structure supports the pumps and related equipment for the RWS. The raw water pumps are located in individual bays of the raw water pump intake structure, a common structure for both units.

As discussed in FSAR Subsection 2.4.7, the minimum recorded surface water temperature in the reservoir was 37.6°F. Therefore, the potential that ice jams, frazil ice formation, or floating debris would prevent the RWS makeup to SWS is not credible.

Raw Water Pumps

Three 50% capacity raw water pumps are provided for each unit. The raw water pumps are vertical, centrifugal, constant-speed electric motor-driven pumps.

Trash Racks and Traveling Screens

Trash racks are provided for each raw water pump intake bay to prevent large debris from entering the pump intake bays.

Dual-flow traveling screens are provided in each pump bay for coarse screening of floating and suspended debris. The screens are sized to maintain a through-screen velocity of less than 0.5 feet per second to minimize the uptake of aquatic biota. A screen wash system is provided to wash off buildup on the screens.

Screen Wash Pumps

Two 100% capacity screen wash pumps are provided for each unit to draw strained water from two of each unit's raw water pump intake bays before supplying the spray water to the traveling screen through the spray wash header, which is common for all three traveling screens of each unit.

Piping

The RWS piping is designed to accommodate transient effects associated with normal operation such as starting or stopping of pumps, opening or closing of valves, or other normal operating events. The underground portions of the RWS piping are also designed to resist external loads. The system design prevents formation of voids on loss of system pumping and allows release or removal of trapped air on pump starting. Materials are selected with consideration to the effects of the external environment and internal fluid conditions. The RWS piping is designed to ASME Standard B31.1.

9.2.11.3 System Operation

For each unit, one or two raw water pumps normally operate depending on demand. The raw water pumps provide a continuous supply of water to the CWS to support normal plant operation. The raw water pumps can also provide flow for dilution of liquid radwaste when CWS blowdown is not available or sufficient for dilution for discharge through the blowdown sump discharge line and plant outfall.

Raw water for continuous makeup to the service water cooling tower basins and supply as required to the demineralized water treatment system and fire water tanks is provided from the water treatment facility. The water treatment facility includes equipment and storage capability as necessary to provide filtered and chemically treated raw water of the quality required at the interface with the various systems served and to meet the demands of those systems.

The RWS is designed to operate during all normal modes of operation and components are powered from normal ac sources. Two of each unit's raw water pumps can also be aligned to receive power from the standby diesel generators to provide makeup to the service water cooling tower basins, if necessary, following a loss of normal ac power.

9.2.11.4 Safety Evaluation

The RWS has no safety-related function and therefore requires no nuclear safety evaluation.

The RWS does not have the potential to be a flow path for radioactive fluids. The WLS discharge effluent is connected to the cooling tower blowdown pipe downstream of the RWS interface. Per DCD 11.2.3.3, the WLS effluent is released offsite through a dilution flow stream. Dilution flow is provided from the cooling tower blowdown. During normal power operation, the CWS circulating water pumps provide dilution flow to the cooling tower blowdown pipe. When CWS is not operational, RWS provides dilution flow by an interconnection with the circulating water blowdown line well upstream of the WLS connection. Contamination of the RWS is not possible since the WLS effluent gravity discharges to the blowdown pipe downstream of the RWS interface.

9.2.11.5 Tests and Inspections

Initial test requirements for the RWS are described in Subsection 14.2.9.4.24.

Performance, hydrostatic, and leakage tests associated with installation and preoperational testing are performed on the RWS. The system performance and structural and leaktight integrity of system components are demonstrated by operation of the system.

9.2.11.6 Instrumentation Applications

Pressure indication, with low and high alarms, is provided on the discharges of the raw water pumps. A low discharge pressure signal automatically starts the designated standby pump. Pressure indication, alarms, and controls for pumps included in the water treatment facility ensure the required pressure and flow of the raw water supply from that facility.

Level instrumentation on the fire water tanks automatically opens the fill valve on low tank level and closes on high level.

Instrumentation requirements for makeup to the SWS and CWS cooling tower basins are discussed in DCD Section 9.2.1 and FSAR Section 10.4.5, respectively.

STD DEP 1.1-1 9.2.12 COMBINED LICENSE INFORMATION

9.2.12.1 Potable Water

VCS COL 9.2-1 This COL item is addressed in Subsections 9.2.5.2.1 and 9.2.5.3.

9.2.12.2 Waste Water Retention Basins

VCS COL 9.2-2 This COL item is addressed in Subsection 9.2.9.2.2.

STD DEP 1.1-1 9.2.13 REFERENCES

201. Westinghouse: Evaluation of Impacts: Change to Maximum Safety Non-Coincident Ambient Wet Bulb Temperature for the V.C. Summer Site, VSP_VSG_000706, June 30, 2010.

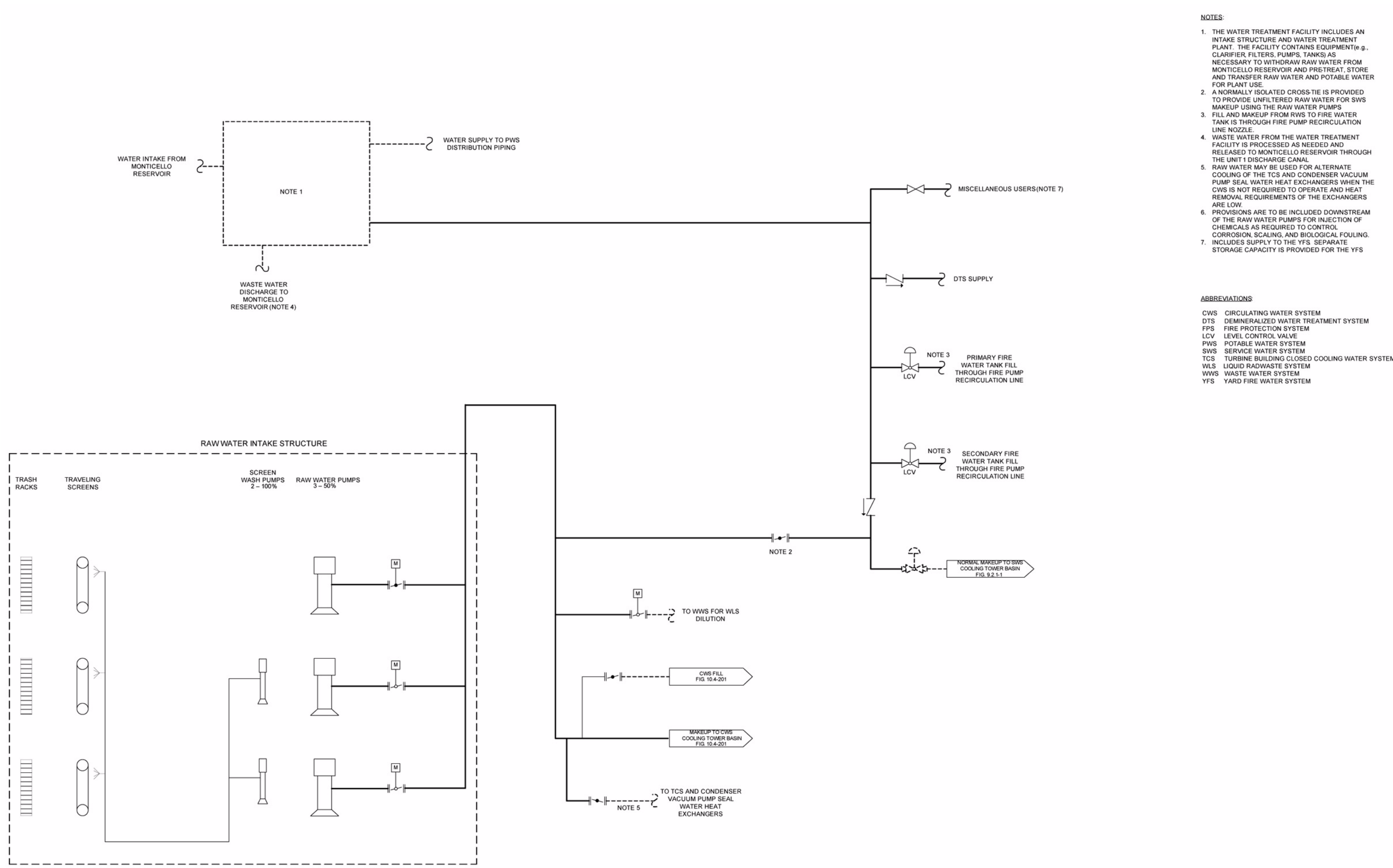

Figure 9.2-201. Raw Water System Flow Diagram

9.3 PROCESS AUXILIARIES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements:

9.3.7 COMBINED LICENSE INFORMATION

STD COL 9.3-1 This COL Item is addressed below.

Generic Issue 43, and the concerns of Generic Letter 88-14 and NUREG-1275 regarding degradation or malfunction of instrument air supply and safety-related valve failure, are addressed by the training and procedures for operations and maintenance of the instrument air subsystem and air-operated valves.

Plant systems, including the compressed and instrument air system, are maintained in accordance with procedures. Maintenance procedures are discussed in Subsection 13.5.2.2.6. The instrument air supply subsystem components are maintained and tested in accordance with manufacturers' recommendations and procedures. The safety-related air-operated valves are maintained in accordance with manufacturers' recommendations and tested in accordance with plant procedures to allow proper function on loss of air. The instrument air is periodically sampled and tested for compliance with the quality requirements of ANSI/ISA-S7.3-1981.

Operators are provided training on loss of instrument air in accordance with abnormal operating procedures. Plant systems, including the compressed and instrument air system, are operated in accordance with system operating procedures, abnormal operating procedures, and alarm response procedures which are written in accordance with Subsection 13.5.2. The training program for operations and maintenance personnel is discussed in Section 13.2.

9.4 AIR-CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.4.1.1.1 Safety Design Basis

Add the following information to the end of DCD Subsection 9.4.1.1.1.

VCS COL 9.4-1b No toxic emergencies due to onsite and offsite sources of toxic chemicals have been identified.

9.4.1.2.3.1 Main Control Room/Control Support Area HVAC Subsystem

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Add the following information to the end of DCD Subsection 9.4.1.2.3.1.

VCS COL 9.4-1b No toxic emergencies due to onsite and offsite sources of toxic chemicals have been identified.

9.4.1.4 Tests and Inspection

STD COL 9.4-1a Add the following text at the end of DCD Subsection 9.4.1.4.

The main control room / control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VBS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Duct and housing leak tests

- Airflow capacity and distribution tests
- Air-aerosol mixing uniformity test
- HEPA filter bank and adsorber bank in-place leak tests
- Duct damper bypass tests
- System bypass tests
- Air heater performance tests
- Laboratory testing of adsorbers
- Ductwork inleakage test

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.7.4 Tests and Inspections

Add the following text at the end of DCD Subsection 9.4.7.4.

STD COL 9.4-1a The exhaust subsystem of the containment air filtration system (VFS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VFS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Airflow capacity and distribution tests
- HEPA filter bank and adsorber bank in-place leak tests
- System bypass tests
- Air heater performance tests
- Laboratory testing of adsorbers
- Ductwork inleakage test

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.12 COMBINED LICENSE INFORMATION

STD COL 9.4-1a This COL Item is addressed in Subsections 9.4.1.4 and 9.4.7.4.

VCS COL 9.4-1b This COL item is discussed in Subsections 9.4.1.1.1 and 9.4.1.2.3.1.

9.4.13 REFERENCES

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201. ASME/ANSI AG-1a-2000, Addenda to ASME AG-1-1997 Code on Nuclear Air and Gas Treatment, Section HA, “Housings.”

9.5 OTHER AUXILIARY SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.5.1.2.1.3 Fire Water Supply System

STD SUP 9.5-1 Add the following paragraph at the end of DCD Subsection 9.5.1.2.1.3.

Threads compatible with those used by the off-site fire department are provided on all hydrants, hose couplings and standpipe risers, or a sufficient number of thread adapters compatible with the off-site fire department are provided.

9.5.1.6 Personnel Qualification and Training

STD COL 9.5-1 Add the following paragraph at the end of DCD Subsection 9.5.1.6.

Subsections 9.5.1.8.2 and 9.5.1.8.7 summarize the qualification and training programs that are established and implemented for the Fire Protection Program.

STD DEP 1.1-1 Insert the following subsections after DCD Subsection 9.5.1.7. DCD Subsection 9.5.1.8 is renumbered as Subsection 9.5.1.9.

9.5.1.8 Fire Protection Program

STD COL 9.5-1 The fire protection program is established such that a fire does not prevent safe shutdown of the plant and does not endanger the health and safety of the public. Fire protection at the plant uses a defense-in-depth concept that includes fire prevention, detection, control and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. These defense-in-depth principles are achieved by meeting the following objectives:

- Prevent fires from starting.
- Detect rapidly, control, and extinguish promptly those fires that do occur.

- Provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant.

- Minimize the potential for radiological releases.

9.5.1.8.1 Fire Protection Program Implementation

As indicated in Table 13.4-201, the required elements of the fire protection program are fully operational prior to receipt of new fuel for buildings storing new fuel and adjacent fire areas that could affect the fuel storage area in that reactor unit. Other required elements of the fire protection program described in this section are fully operational prior to initial fuel loading in that reactor unit.

Elements of the fire protection program are reviewed on a frequency established by procedures and updated as necessary.

9.5.1.8.1.1 Fire Protection Program Criteria

STD COL 9.5-3 STD COL 9.5-4 The fire protection program is based on the criteria of several industry and regulatory documents referenced in FSAR Subsection 9.5.5 and DCD Subsection 9.5.5, and also based on the guidance provided in Regulatory Guide 1.189. DCD Tables 9.5.1-1 and FSAR Table 9.5-201 provide a cross-reference to information addressing compliance with BTP CMEB 9.5-1. Exceptions to the National Fire Protection Association (NFPA) Standards beyond those included in DCD Table 9.5.1-3, and exceptions taken to the NFPA Standards listed in FSAR Subsection 9.5.5, are identified in FSAR Table 9.5-202.

9.5.1.8.1.2 Organization and Responsibilities

STD COL 9.5-1 The organizational structure of the fire protection personnel is discussed in Subsection 13.1.1.2.10.

The site executive in charge of the fire protection program, through the engineer in charge of fire protection, is responsible for the following:

a. Programs and periodic inspections are implemented to:

    1. Minimize the amount of combustibles in safety-related areas.

    2. Determine the effectiveness of housekeeping practices.

    3. Provide for availability and acceptability of the following:

        i. Fire protection system and components.

ii. Manual fire fighting equipment.

iii. Emergency breathing apparatus.

iv. Emergency lighting.

v. Portable communication equipment.

STD COL 9.5-8
STD COL 9.5-1

vi. Fire barriers including fire rated walls, floors and ceilings, fire rated doors, dampers, etc., fire stops and wraps, and fire retardant coating. Procedures address the administrative controls in place, including fire watches, when a fire area is breached for maintenance.

STD COL 9.5-1

4. Confirm prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence.

b. Conducting periodic maintenance and testing of fire protection systems, components, and manual fire fighting equipment, evaluating test results, and determining the acceptability of systems under test in accordance with established plant procedures.

c. Designing and selecting equipment related to fire protection.

d. Reviewing and evaluating proposed work activities to identify potential transient fire loads.

e. Managing the plant fire brigade, including:

1. Developing, implementing and administering the fire brigade training program.

2. Scheduling and conducting fire brigade drills.

3. Critiquing fire drills to determine if training objectives are met.

4. Performing a periodic review of the fire brigade roster and initiating changes as needed.

5. Maintaining the fire training program records for members of the fire brigade and other personnel.

6. Maintaining a sufficient number of qualified fire brigade personnel to respond to fire emergencies for each shift.

f. Developing and conducting the fire extinguisher training program.

g. Implementing a program for indoctrination of personnel gaining unescorted access to the protected area in appropriate procedures which implement the fire protection program, such as fire prevention and fire reporting procedures, plant emergency alarms, including evacuation.

h. Implementing a program for instruction of personnel on the proper handling of accidental events such as leaks or spills of flammable materials.

i. Preparing procedures to meet possible fire situations in the plant and for assuring assistance is available for fighting fires in radiological areas.

j. Implementing a program that utilizes a permit system that controls and documents inoperability of fire protection systems and equipment. This program initiates proper notifications and compensatory actions, such as fire watches, when inoperability of any fire protection system or component is identified.

k. Developing and implementing preventive maintenance, corrective maintenance, and surveillance test fire protection procedures.

l. Confirming that plant modifications, new procedures and revisions to procedures associated with fire protection equipment and systems that have significant impact on the fire protection program are reviewed by an individual who possesses the qualifications of a fire protection engineer.

m. Continuing evaluation of fire hazards during construction or modification of other units on the site. Special considerations, such as fire barriers, fire protection capability and administrative controls are provided as necessary to protect the operating unit(s) from construction or modification activities.

n. Establishing a fire prevention surveillance plan and training plant personnel on that plan.

o. Developing pre-fire plans and making them available to the fire brigade and control room.

VCS COL 9.5-1 The responsibilities of the engineer in charge of fire protection and his staff are discussed in Subsection 13.1.1.3.2.1.4.

STD COL 9.5-1 9.5.1.8.2 Fire Brigade

9.5.1.8.2.1 General

VCS COL 9.5-1 The organization of the fire brigade is discussed in Subsection 13.1.2.4.

STD COL 9.5-1 To qualify as a member of the fire brigade, an individual must meet the following criteria:

a. Has attended the required training sessions for the position occupied on the fire brigade.

b. Has passed an annual physical exam including demonstrating the ability for performing strenuous activity and the use of respiratory protection.

9.5.1.8.2.2 Fire Brigade Training

A training program is established so that the capability to fight fires is developed and documented. The program consists of classroom instruction supplemented with periodic classroom retraining, practice in fire fighting, and fire drills. Classroom instruction and training is conducted by qualified individuals knowledgeable in fighting the types of fires that could occur within the plant and its environs and using on-site fire fighting equipment. Individual records of training provided to each fire brigade member, including drill critiques, are maintained as part of the permanent plant files for at least three years to document that each member receives the required training.

The fire brigade leader and at least two brigade members per shift have sufficient training and knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability.

Personnel assigned as fire brigade members receive formal training prior to assuming brigade duties. The course subject matter is selected to satisfy the requirements of Regulatory Guide 1.189. Course material selection also includes guidance from NFPA 600 (Reference 204) and 1500 (Reference 210) as appropriate. Additional training may also include material selected from NFPA 1404 (Reference 208) and 1410 (Reference 209).

The minimum equipment provided for the fire brigade consists of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment and portable extinguishers. Self-contained breathing apparatus (SCBA) approved by NIOSH, using full face positive pressure masks, and providing an operating life of at least 30 minutes, are provided for selected fire brigade, emergency repair and control room personnel. At least ten masks are provided for fire brigade personnel. At least two extra air bottles, each with at least 30 minutes of operating life, are located on-site for each SCBA. An additional on-site 6-hour supply of reserve air is provided to permit quick and complete replenishment of exhausted supply air bottles. DCD Subsection 6.4.2.3 discusses the portable breathing apparatus for control room personnel. Additional SCBAs are provided near the personnel containment entrance for the exclusive use of the fire brigade. The fire brigade leader has ready access to keys for any locked fire doors.

VCS COL 9.5-1 The on-duty shift supervisor has responsibility for taking certain actions based on an assessment of the magnitude of the fire emergency. These actions include safely shutting down the plant, making recommendations for implementing the Emergency Plan, notification of emergency personnel and requesting assistance from off-duty personnel, if necessary. Emergency Plan consideration of fire emergencies includes the guidance of Regulatory Guide 1.101.

STD COL 9.5-1

9.5.1.8.2.2.1 Classroom Instruction

Fire brigade members receive classroom instruction in fire protection and fire fighting techniques prior to qualifying as members of the fire brigade. This instruction includes:

a. Identification of the types of fire hazards along with their location within the plant and its environs.

b. Identification of the types of fires that could occur within the plant and its environs.

c. Identification of the location of on-site fire fighting equipment and familiarization with the layout of the plant including ingress and egress routes to each area.

d. The proper use of on-site fire fighting equipment and the correct method of fighting various types of fires including at least the following:

   – fires involving radioactive materials

   – fires in energized electrical equipment

   – fires in cables and cable trays

   – fires involving hydrogen

   – fires involving flammable and combustible liquids or hazardous process chemicals

- – fires resulting from construction or modifications (welding)
- – fires involving record files

e. Review of each individual's responsibilities under the Fire Protection Program.

f. Proper use of communication, lighting, ventilation, and emergency breathing equipment.

g. Fire brigade leader direction and coordination of fire fighting activities.

h. Toxic and radiological characteristics of expected combustion products.

i. Proper methods of fighting fires inside buildings and confined spaces.

j. Detailed review of fire fighting strategies, procedures and procedure changes.

k. Indoctrination of the plant fire fighting plans, identification of each individual's responsibilities, and review of changes in the fire fighting plans resulting from fire protection-related plant modifications.

l. Coordination between the fire brigade and off-site fire departments that have agreed to assist during a major fire on-site is provided to establish responsibilities and duties. Educating the off-site organization in operational precautions when fighting fires on nuclear power plant sites, and awareness of special hazards and the need of radiological protection of personnel.

9.5.1.8.2.2.2 Retraining

Classroom refresher training is scheduled on a biennial basis to supplement retention of the initial training. These sessions may be concurrent with the regular planned meetings.

9.5.1.8.2.2.3 Practice

Practice sessions are held for each fire brigade and for each fire brigade member on the proper method of fighting various types of fires which might occur in the plant. These sessions are scheduled on an annual basis and provide brigade members with team experience in actual fire fighting and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting.

9.5.1.8.2.2.4 Drills

Fire brigade drills are conducted at least once per calendar quarter for each shift. Each fire brigade member participates in at least two drills annually. Drills are either announced or unannounced. At least one unannounced drill is held annually for each shift fire brigade. At least one drill is performed annually on a "back shift" for each shift's fire brigade. The drills provide for off-site fire department participation at least annually. Triennially, a randomly selected, unannounced drill shall be conducted and critiqued by qualified individuals independent of the plant staff. Training objectives are established prior to each drill and reviewed by plant management. Drills are critiqued on the following points:

a. Assessment of fire alarm effectiveness.

b. Assessment of time required to notify and assemble the fire brigade.

c. Assessment of the selection, placement and use of equipment.

d. Assessment of the fire brigade leader's effectiveness in directing the fire fighting effort.

e. Assessment of each fire brigade member's knowledge of fire fighting strategy, procedures and simulated use of equipment.

f. Assessment of the fire brigade's performance as a team.

Performance deficiencies identified, based on these assessments, are used as the basis for additional training and repeat drills. Unsatisfactory drill performance is followed by a repeat drill within 30 days.

9.5.1.8.2.2.5 Meetings

Regular planned meetings are held at least quarterly for the fire brigade members to review changes in the Fire Protection Program and other subjects as necessary.

9.5.1.8.3 Administrative Controls

Administrative controls for the Fire Protection Program are implemented through plant administrative procedures. Applicable industry publications are used as guidance in developing those procedures.

Administrative controls include procedures to:

a. Control actions to be taken by an individual discovering a fire, such as notification of the control room, attempting to extinguish the fire, and actuation of local fire suppression systems.

VCS COL 9.5-1 b. Control actions to be taken by the control room operator, such as sounding fire alarms, and notifying the shift supervisor of the type, size and location of the fire.

STD COL 9.5-1 c. Control actions to be taken by the fire brigade after notification of a fire, including location to assemble, directions given by the fire brigade leader, the responsibilities of brigade members, such as selection of fire fighting and protective equipment, and use of preplanned strategies for fighting fires in specific areas.

d. Control actions to be taken by the security force upon notification of a fire.

e. Define the strategies established for fighting fires in safety-related areas and areas presenting a hazard to safety-related equipment, including the designation of the:

1. Fire hazards in each plant area/zone covered by a fire fighting procedure (pre-fire plan). Pre-fire plans utilize the guidance of NFPA 1620 (Reference 205).

2. Fire extinguishers best suited for controlling fires with the combustible loadings of each zone and the nearest location of these extinguishers.

3. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. Access and egress routes that involve locked doors are specifically identified in the procedure with the appropriate precautions and methods for access specified.

4. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical system in the zone covered by the specific fire fighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

5. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling are designated.

6. Potential radiological and toxic hazards in fire zones.

7. Ventilation system operation that provides desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

VCS COL 9.5-1

8. Operations requiring control room and shift supervisor coordination or authorization.

STD COL 9.5-1

9. Instructions for plant operators and other plant personnel during a fire.

f. Organize the fire brigade and assign special duties according to job title so that the fire fighting functions are covered for each shift by personnel trained and qualified to perform these functions. These duties include command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishing agent to the fire, communication with the control room, and coordination with off-site fire departments.

9.5.1.8.4 Control of Combustible Materials, Hazardous Materials and Ignition Sources

The control of combustible materials is defined by administrative procedures. These procedures impose the following controls:

a. Prohibit the storage of combustible materials (including unused ion exchange resins) in areas that contain or expose safety-related equipment.

b. Govern the handling of and limit transient fire loads such as flammable liquids, wood and plastic materials in buildings containing safety-related systems or equipment.

c. Assign responsibility to the appropriate supervisor for reviewing work activities to identify transient fire loads.

d. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, grinding, brazing and soldering operations, and temporary electrical power cables. A separate permit is issued for each area where such work is done. If work continues over more than one shift, the permit is valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown. NFPA 51B (Reference 202) and 241 (Reference 203) are used as guidance.

e. Minimize waste, debris, scrap, and oil spills or other combustibles resulting from a work activity in the safety-related area while work is in progress and remove the same upon completion of the activity or at the end of each work shift.

f. Govern periodic inspections for accumulation of combustibles for continued compliance with these administrative controls.

g. Prohibit the storage of acetylene-oxygen and other compressed gasses in areas that contain or expose safety-related equipment or the fire protection system that serves those areas. A permit system is required to control the use of this equipment in safety-related areas of the plant.

h. Govern the use and storage of hazardous chemicals in areas that contain or expose safety-related equipment.

i. Control the use of specific combustibles in safety-related areas. Wood used in safety-related areas during maintenance, modification, or refueling operation (such as lay-down blocks or scaffolding) is treated with a flame retardant in accordance with NFPA 703 (Reference 207). Use of wood inside buildings containing systems or equipment important to safety is only permitted when suitable noncombustible substitutes are not available. Equipment or supplies (such as new fuel) shipped in untreated combustible packing containers are unpacked in safety-related areas if required for valid operating reasons. However, combustible materials are removed from the area immediately following unpacking. Such transient combustible material, unless stored in approved containers, is not left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material, such as wood or paper excelsior, or polyethylene sheeting, is placed in metal containers with tight-fitting self-closing metal covers. Only noncombustible panels or flame-retardant tarpaulins or approved materials of equivalent fire-retardant characteristics are used. Any other fabrics or plastic films used are certified to conform to the large-scale fire test described in NFPA 701 (Reference 206).

j. Govern the control of electrical appliances in areas that contain or expose safety-related equipment.

9.5.1.8.5 Control of Radioactive Materials

The plant is designed with provisions for sampling of liquids resulting from fire emergencies that may contain radioactivity and may be released to the environment. Plant operating procedures require such liquids to be collected, sampled, and analyzed prior to discharge. Liquid discharges are required to be below activity limits prior to discharge.

9.5.1.8.6 Testing and Inspection

Testing and inspection requirements are imposed through administrative procedures. Maintenance or modifications to the fire protection system are subject to inspection for conformation to design requirements. Procedures governing the inspection, testing, and maintenance of fire protection alarm and detection systems, and water-based suppression and supply systems, utilize the guidance of NFPA 72 (DCD Reference 9.5.5.2) and NFPA 25 (Reference 212). Installation of portions of the system where performance cannot be verified through pre-operational tests, such as penetration seals, fire retardant coatings, cable routing, and fire barriers are inspected. Inspections are performed by individuals knowledgeable of fire protection design and installation requirements. Open flame or combustion-generated smoke is not used for leak testing or similar procedures such as air flow determination. Inspection and testing procedures address the identification of items to be tested or inspected, responsible organizations for the activity, acceptance criteria, documentation requirements and sign-off requirements.

Fire protection materials subject to degradation (such as fire stops, seals and fire retardant coatings are visually inspected periodically for degradation or damage. Fire hoses are hydrostatically tested in accordance with NFPA 1962 (Reference 201). Hoses stored in outside hose stations are tested annually and interior standpipe hoses are tested every three years.

The fire protection system is periodically tested in accordance with plant procedures. Testing includes periodic operational tests and visual verification of damper and valve positions. Fire doors and their closing and latching mechanisms are also included in these procedures.

STD COL 9.5-6 The preoperational testing program describes the procedures for confirming that the as-installed configuration of fire barriers matches the tested configurations. The procedures describe the process for identifying and dispositioning deviations.

STD COL 9.5-1

9.5.1.8.7 Personnel Qualification and Training

VCS COL 9.5-1 The engineer in charge of fire protection is responsible for the formulation and implementation of the fire protection program and meets the qualification requirements listed in FSAR Subsection 13.1.1.3.2.1.4.

STD COL 9.5-1 Qualification and training of other plant personnel involved in the fire protection program is governed by plant qualification procedures and is conducted by personnel qualified by training and experience in these areas. These classifications include training personnel, maintenance personnel assigned to work on the fire protection system, and operations personnel assigned to system operation and testing.

9.5.1.8.8 Fire Doors

STD COL 9.5-3 Fire doors separating safety-related areas are self-closing or provided with closing mechanisms and are inspected semiannually to verify that the automatic hold open, release and closing mechanisms and latches are operable. Watertight and missile resistant doors are not provided with closing mechanisms. Fire doors with automatic hold open and release mechanisms are inspected daily to verify that the doorways are free of obstructions.

Fire doors separating safety-related areas are normally closed and latched. Fire doors that are locked closed are inspected weekly to verify position. Fire doors that are closed and latched are inspected daily to assure that they are in the closed position. Fire doors that are closed and electrically supervised at a continuously manned location are not inspected.

9.5.1.8.9 Emergency Planning

Emergency planning is described in Section 13.3.

STD DEP 1.1-1 9.5.1.9 Combined License Information

9.5.1.9.1 Qualification Requirements for Fire Protection Program

STD COL 9.5-1 This COL Item is addressed as follows:

Qualification requirements for individuals responsible for development of the Fire Protection Program are discussed in Subsections 9.5.1.6 and 9.5.1.8.7.

Training of firefighting personnel is discussed in Subsections 9.5.1.8, 9.5.1.8.2 and 9.5.1.8.7.

Administrative procedures and controls governing the Fire Protection Program during plant operation are discussed in Subsections 9.5.1.8.1.2, 9.5.1.8.3, 9.5.1.8.4, 9.5.1.8.5, and 9.5.1.8.6.

Fire protection system maintenance is discussed in Subsection 9.5.1.8.6.

9.5.1.9.2 Fire Protection Analysis Information

VCS COL 9.5-2 This COL Item is addressed in Subsection 9A.3.3.

9.5.1.9.3 Regulatory Conformance

STD COL 9.5-3 This COL Item is addressed in Subsections 9.5.1.8.1.1, 9.5.1.8.8, and 9.5.1.8.9 and in Table 9.5-201.

9.5.1.9.4 NFPA Exceptions

STD COL 9.5-4 This COL item is addressed in Subsection 9.5.1.8.1.1.

9.5.1.9.6 Verification of Field Installed Fire Barriers

STD COL 9.5-6 This COL Item is addressed in Subsection 9.5.1.8.6.

9.5.1.9.7 Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance

STD COL 9.5-8 This COL item is addressed in Subsection 9.5.1.8.1.2.

9.5.2.5 Combined License Information

9.5.2.5.1 Offsite Interfaces

VCS COL 9.5-9 The Emergency Notification System (ENS) and the Emergency Response Data System (ERDS) are both powered normally by the 120V-ac power system. In the event of a loss of the ac power system, the systems are automatically switched over to the diesel backed, non-Class 1E dc and uninterruptable power supply systems.

Additional information regarding emergency communication systems can be found in Part 2, Section F "Emergency Communications" of the Emergency Plan.

9.5.2.5.2 Emergency Offsite Communications

VCS COL 9.5-10 The primary system used for communication with state and county officials during an emergency is the Electric Switch System Exchange (ESSX). VCSNS employs additional backup communication systems to the ESSX system including the use of the Private Branch Exchange (PBX) telephone system, local commercial telephone system, satellite telephones, and an 800 MHz radio system. In the event of the failure of one of the primary systems, the communicator manually initiates communications using one of the backup systems as described in the

Emergency Implementing Procedures. The Implementing Procedures provide the details for the communications transfer should the primary equipment fail or otherwise be determined to be unacceptable. The 800 MHz system serves as the crisis management radio system between VCSNS onsite teams and state and county officials. Details of the primary and secondary communication systems are provided in Section F of the VCSNS Emergency Plan.

9.5.2.5.3 Security Communications

VCS COL 9.5-11 This COL Item is addressed in Section 11 “Communications” of the Physical Security Plan.

Add the following subsection after DCD Subsection 9.5.4.5.1.

9.5.4.5.2 Fuel Oil Quality

STD COL 9.5-13 The diesel fuel oil testing program requires testing both new fuel oil and stored fuel oil. High fuel oil quality is provided by specifying the use of ASTM Grade 2D fuel oil with a sulfur content as specified by the engine manufacturer.

A fuel sample is analyzed prior to addition of ASTM Grade 2D fuel oil to the storage tanks. The sample moisture content and particulate or color is verified per ASTM D4176. In addition, kinematic viscosity is tested to be within the limits specified in Table 1 of ASTM D975. The remaining critical parameters per Table 1 of ASTM D975 are verified compliant within 7 days.

Fuel oil quality is verified by sample every 92 days to meet ASTM Grade 2D fuel oil criteria. The addition of fuel stabilizers and other conditioners is based on sample results.

The fuel oil storage tanks are inspected on a monthly basis for the presence of water. Any accumulated water is to be removed.

9.5.4.7 Combined License Information

9.5.4.7.2 Fuel Degradation Protection

STD COL 9.5-13 This COL Item is addressed in Subsection 9.5.4.5.2.

9.5.5 REFERENCES

201. National Fire Protection Association, “Standard for Inspection, Care, and Use of Fire Hose, Couplings, and Nozzles and the Service Testing of Fire Hose,” NFPA 1962, 2003.

202. National Fire Protection Association, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work,” NFPA 51B, 2003.

203. National Fire Protection Association, “Standard for Safeguarding Construction, Alteration, and Demolition Operations,” NFPA 241, 2004.

204. National Fire Protection Association, “Standard on Industrial Fire Brigades,” NFPA 600, 2005.

205. National Fire Protection Association, “Recommended Practice for Pre-incident Planning,” NFPA 1620, 2003.

206. National Fire Protection Association, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films,” NFPA 701, 2004.

207. National Fire Protection Association, “Standard for Fire-Retardant Treated Wood and Fire-Retardant Coatings for Building Materials,” NFPA 703, 2006.

208. National Fire Protection Association, “Standard for Fire Service Respiratory Protection Training,” NFPA 1404, 2006.

209. National Fire Protection Association, “Standard on Training for Initial Emergency Scene Operations,” NFPA 1410, 2005.

210. National Fire Protection Association, “Standard on Fire Department Occupational Safety and Health Program,” NFPA 1500, 2007.

211. National Fire Protection Association, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” NFPA 804, 2001.

212. National Fire Protection Association, “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,” NFPA 25, 2008.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 1 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
	Fire Protection Program			
	1. Direction of fire protection program; availability of personnel.	C.1.a(1)	C	Comply. Subsections 9.5.1.8.1.2 and 13.1.1.2.10 address this requirement.
	2. Defense-in-depth concept; objective of fire protection program.	C.1.a(2)	C	Comply. Subsections 9.5.1.8 and 9.5.1.8.1 address this requirement.
VCS COL 9.5-3 VCS COL 9.5-4	3. Management responsibility for overall fire protection program; delegation of responsibility to staff.	C.1.a(3)	C	Comply. Subsections 9.5.1.8.1.2, 13.1.1.3.2.1.4 and 13.1.1.2.10
	4. The staff should be responsible for:	C.1.a(3)	C	Comply. Subsection 13.1.1.3.2.1.4 addresses this requirement.
STD COL 9.5-3 STD COL 9.5-4	a. Fire protection program requirements.			
	b. Post-fire shutdown capability.			
	c. Design, maintenance, surveillance, and quality assurance of fire protection features.			
	d. Fire prevention activities.			
	e. Fire brigade organization and training.			
	f. Prefire planning.			

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 2 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

		BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
VCS COL 9.5-3 VCS COL 9.5-4	5.	The organizational responsibilities and lines of communication pertaining to fire protection should be defined through the use of organizational charts and functional descriptions.	C.1.a(4)	C	Comply. Organization and lines of communication are addressed in Figure 13,1-201. Functional descriptions are addressed in Subsections 13.1.1.2.10 13.1.1.3.1.4, 13.1.1.3.2.1.4, and 13.1.2.4.
	6.	Personnel qualification requirements for fire protection engineer, reporting to the position responsible for formulation and implementation of the fire protection program.	C.1.a(5)(a)	C	Comply. Subsection 13.1.1.3.2.1.4 addresses this requirement.
STD COL 9.5-3 STD COL 9.5-4	7.	The fire brigade members' qualifications should include a physical examination for performing strenuous activity, and the training described in Position C.3.d.	C.1.a(5)(b)	C	Comply. Subsections 9.5.1.8.2.1 and 9.5.1.8.2.2 addresses this requirement.
	8.	The personnel responsible for the maintenance and testing of the fire protection systems should be qualified by training and experience for such work.	C.1.a(5)(c)	C	Comply. Subsection 9.5.1.8.7 addresses this requirement.
	9.	The personnel responsible for the training of the fire brigade should be qualified by training and experience for such work.	C.1.a(5)(d)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
	10.	The following NFPA publications should be used for guidance to develop the fire protection program: No. 4, No. 4A, No. 6, No. 7, No. 8, and No. 27.	C.1.a(6)	C	Alternate Compliance. The NFPA codes cited in BTP CMEB 9.5-1 are historical. Current NFPA codes are referenced for guidance for the fire protection program. Subsection 9.5.1.8.1.1 addresses this requirement.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 3 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
11.	On sites where there is an operating reactor, and construction or modification of other units is underway, the superintendent of the operating plant should have a lead responsibility for site fire protection.	C.1.a(7)	C	Comply. Subsection 13.1.1.2.10 addresses this requirement.
Fire Protection Analysis				
14.	Fires involving facilities shared between units should be considered.	C.1.b	C	Comply. The FHA demonstrates the plant's ability to perform safe shutdown functions and minimize radioactive releases to the environment. Postulated fires in shared facilities that do not contain SSCs important to safety and do not contain radioactive materials do not affect these functions.
15.	Fires due to man-made site-related events that have a reasonable probability of occurring and affecting more than one reactor unit should be considered.	C.1.b	C	Comply. Subsections 2.2.3 and 3.5 establish that these events are not credible.
Fire Suppression System Design Basis				
22.	Fire protection systems should retain their original design capability for potential man-made, site-related events that have a reasonable probability of occurring at a specific plant site.	C.1.c(4)	C	Comply. Subsections 2.2.3 and 3.5 establish that these events are not credible.
Fire Protection Program Implementation				
26.	The fire protection program for buildings storing new reactor fuel and for adjacent fire areas that could affect the fuel storage area should be fully operational before fuel is received at the site.	C.1.e(1)	C	Comply. Subsection 9.5.1.8.1 addresses this requirement.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 4 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
	27.	The fire protection program for an entire reactor unit should be fully operational prior to initial fuel loading in that unit.	C.1.e(2)	C	Comply. Subsection 9.5.1.8.1 addresses this requirement.
	28.	Special considerations for the fire protection program on reactor sites where there is an operating reactor and construction or modification of other units is under way.	C.1.e(3)	C	Comply. Subsection 9.5.1.8.1.2. m addresses this requirement.
	29.	Establishing administrative controls to maintain the performance of the fire protection system and personnel.	C.2	C	Comply. Subsection 9.5.1.8.1.2 addresses this requirement.
	Fire Brigade				
	30.	The guidance in Regulatory Guide 1.101 should be followed as applicable.	C.3.a	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
VCS COL 9.5-3 VCS COL 9.5-4	31.	Establishing site brigade: minimum number of fire brigade members on each shift; qualification of fire brigade members; competence of brigade leader.	C.3.b	C	Comply. Subsection 9.5.1.8.2.2 and 13.1.2.4 address this requirement.
STD COL 9.5-3 STD COL 9.5-4	32.	The minimum equipment provided for the brigade should consist of turnout coats, boots, gloves, hard hats, emergency communications equipment, portable ventilation equipment, and portable extinguishers.	C.3.c	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
	33.	Recommendations for breathing apparatus for fire brigade, damage control, and control room personnel.	C.3.c	C	Comply. Subsection 9.5.1.8.2.2 and DCD Subsections 6.4.2.3 and 6.4.4 address these requirements.
	34.	Recommendations for the fire brigade training program.	C.3.d	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 5 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
Quality Assurance Program				
35.	Establishing quality assurance (QA) programs by applicants and contractors for the fire protection systems for safety-related areas; identification of specific criteria for quality assurance programs.	C.4	C	Comply. DCD Subsection 9.5.1.7 and Chapter 17 address this requirement.
Building Design				
50.	Fire doors should be inspected semiannually to verify that automatic hold-open, release, and closing mechanisms and latches are operable.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.8 addresses this requirement.
51.	Alternative means for verifying that fire doors protect the door opening as required in case of fire.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.8 addresses this requirement.
52.	The fire brigade leader should have ready access to keys for any locked fire doors.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
55.	Stairwells serving as escape routes, access routes for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be enclosed in masonry or concrete towers with a minimum fire resistance rating of 2 hours and self-closing Class B fire doors.	C.5.A (6)	C	Comply. Subsection 9A.3.3 addresses this requirement for miscellaneous buildings located in the yard.
56.	Fire exit routes should be clearly marked.	C.5.a (7)	C	Comply. DCD Subsection 9.5.1.2.1.1 addresses this requirement.
71.	Water drainage from areas that may contain radioactivity should be collected, sampled, and analyzed before discharge to the environment.	C.5.a(14)	C	Comply. Capability is provided. Subsection 9.5.1.8.5 addresses this requirement.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 6 of 7) AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
Control of Combustibles				
80.	Use of compressed gases inside buildings should be controlled.	C.5.d (2)	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
Lighting and Communication				
111.	A portable radio communications system should be provided for use by the fire brigade and other operations personnel required to achieve safe plant shutdown.	C.5.g (4)	C	Comply. Subsections 9.5.1.8.1.2.a.3.v, 9.5.1.8.2.2, and DCD Subsections 9.5.2 and 9.5.2.2.1 address this requirement.
Water Sprinkler and Hose Standpipe Systems				
149.	All valves in the fire protection system should be periodically checked to verify position.	C.6.c (2)	C	Comply. Subsection 9.5.1.8.6 addresses this requirement.
157.	The fire hose should be hydrostatically tested in accordance with NFPA 1962. Hoses stored in outside hose houses should be tested annually. The interior standpipe hose should be tested every 3 years.	C.6.c (6)	C	Comply. Subsection 9.5.1.8.6 addresses this requirement.
Primary and Secondary Containment				
174.	Self-contained breathing apparatus should be provided near the containment entrances for fire fighting and damage control personnel. These units should be independent of any breathing apparatus provided for general plant activities.	C.7.a (2)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
Main Control Room Complex				
180.	Breathing apparatus for main control room operators should be readily available.	C.7.b	C	Comply. DCD Subsection 6.4.2.3 addresses this requirement.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 7 of 7)
AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
Cooling Towers				
225.	Cooling towers should be of noncombustible construction or so located and protected that a fire will not adversely affect any safety-related systems or equipment.	C.7.q	C	Comply. Subsection 9A.3.3 addresses this requirement.
Storage of Acetylene-Oxygen Fuel Gases				
228.	Gas cylinder storage locations should not be in areas that contain or expose safety-related equipment or the fire protection systems that serve those safety-related areas.	C.8.a	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
229.	A permit system should be required to use this equipment in safety-related areas of the plant.	C.8.a	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
Storage Areas for Ion Exchange Resins				
230.	Unused ion exchange resins should not be stored in areas that contain or expose safety-related equipment.	C.8.b	C	Comply. Subsection 9.5.1.8.4.a addresses this requirement.
Hazardous Chemicals				
231.	Hazardous chemicals should not be stored in areas that contain or expose safety-related equipment.	C.8.c	C	Comply. Subsection 9.5.1.8.4.h addresses this requirement.

a) This table supplements DCD Table 9.5.1-1.

STD COL 9.5-4

Table 9.5-202[a]
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception or Clarification
NFPA 804 (Reference 211) contains requirements specific to light water reactors.	Compliance with portions of this standard is as identified within DCD Section 9.5.1 and WCAP-15871. The intake structure is non-combustible construction, does not provide any safety function, and does not contain any equipment important to safety. Automatic sprinkler protection is not warranted and is not provided.

a) This table supplements DCD Table 9.5.1-3.

APPENDIX 9A
FIRE PROTECTION ANALYSIS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9A.2 FIRE PROTECTION METHODOLOGY

9A.2.1 Fire Area Description

Add the following information at the end of the first paragraph in DCD Subsection 9A.2.1:

VCS DEP 18.8-1 Figure 9A-201 replaces DCD Figure 9A-3 (Sheet 1), to reflect the relocation of the Operations Support Center.

9A.3.3 Yard Area and Outlying Buildings

VCS COL 9.5-2 Miscellaneous yard areas, equipment, or structures that do not contain safety-related systems or components, or radioactive materials, are located so they will not present a hazard from fire or smoke to any safety-related structures, systems, or equipment located on site.

The fire protection provided for these yard areas, outlying buildings, structures, or equipment will comply with building code, fire code, and NFPA requirements. A final fire hazards analysis based on final design and purchased materials will be completed before receipt of fuel on site.

9A.3.3.1 CWS Cooling Towers

The CWS for each unit incorporates two counterflow, clustered-plume, round mechanical draft cooling towers. The structures are approximately 270 feet in diameter and 70 feet tall. The cooling towers are generally of noncombustible concrete construction, with combustible polyvinyl chloride fill and drift eliminator and fire-retardant glass reinforced polyester plenum partitions and fan shroud/stack. Fan blades are of polyester and fiberglass composite. Each tower incorporates 16 motor-driven fans, complete with gear reducers. Each fan unit is supplied with an industrial grade lubrication line, terminating outside the fan shroud/stack near the motor. The lubrication system includes an oil level sight glass and drain for maintenance.

The design and materials of construction conform to FM requirements for FM-approved towers found to be of low fire hazard not requiring automatic sprinkler protection for insurance purposes. Outdoor yard hydrants are located near the cooling towers for manual firefighting.

The cooling towers are located away from safety-related plant areas such that a tower fire or structural failure resulting in their collapse will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.2 CWS Intake Structures

The CWS for each unit incorporates an open concrete intake structure located outdoors between the two respective mechanical draft cooling towers serving the unit. The intake structure is comprised of an open flume from the cooling tower basins, removable course and fine screens for debris control, and three circulating water pump bays. Each of the three circulating water pumps is rated for 33-1/3% capacity and provides flow to the condensers.

Each intake structure incorporates a single-story service building to house switchgear and related equipment required for operation of the cooling towers, the screens, and circulating water pumps. The building is of noncombustible unprotected construction and measures approximately 1,400 ft^2 in area. The combustible loading is estimated to be less than one hour (<80,000 Btu/ft^2), the major contributor being cable insulation.

The building incorporates a fire alarm system comprised of automatic fire detection, manual pull stations, and audible alarm notification appliances in accordance with Appendix 9A.2.4 for combustible loadings up to 80,000 Btu/ft^2. The fire alarm system also produces an audible and visual alarm in the main control room and the security central alarm station.

Portable fire extinguishers located inside the building and outdoors near the circulating water pumps, and outdoor yard hydrants, provide means of manual fire suppression.

The intake structures and service buildings are located away from safety-related plant areas such that a fire will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.3 RWS Intake Structure

The RWS incorporates an open concrete intake structure located outdoors at the Monticello Reservoir. The intake structure is comprised of six intake bays and six raw water pumps; three per unit. The facility includes bar screens and dual flow traveling screens for debris control, a debris basin, and four screen wash pumps. Each of the three raw water pumps serving each unit is rated for 50% capacity.

The intake structure incorporates a single-story service building to house switchgear and related equipment required for operation of the traveling screens, the screen wash pumps, and the raw water pumps. The building is of noncombustible unprotected construction and measures approximately 2,000 ft^2

in area. The combustible loading is estimated to be less than one hour (<80,000 Btu/ft^2), the major contributor being cable insulation.

The building incorporates a fire alarm system comprised of automatic fire detection, manual pull stations, and audible alarm notification appliances in accordance with Appendix 9A.2.4 for combustible loadings up to 80,000 Btu/ft^2. The fire alarm system also produces an audible and visual alarm in the main control room and the security central alarm station.

Portable fire extinguishers located inside the building and outdoors near the raw water and screen wash pumps provide means of manual fire suppression.

The intake structure and service building are located away from safety-related plant areas such that a fire will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.4 Warehouse

A single-story warehouse is provided for storage and handling of bulk materials, of the type and quantity within acceptable limits as allowed by code. The facility is common to serve both units, and is comprised of noncombustible unprotected construction. The potential combustible loading is estimated to be greater than one hour (>80,000 Btu/ft^2); thus the facility requires detection capability and automatic and manual fire suppression in accordance with Appendix 9A.2.4.

The building incorporates full area ceiling sprinkler protection which is conservatively designed for Class IV commodities and unexpanded plastics in open rack storage up to a height of 25 feet. The automatic sprinklers satisfy the detection requirement, so supplemental detection is not required.

The building incorporates a fire alarm system comprised of manual pull stations and automatic audible alarm notification appliances. The fire alarm system also produces an audible and visual alarm in the main control room and the security central alarm station.

Portable fire extinguishers and a Class II standpipe hose system are provided throughout the building for manual firefighting. Additionally, the facility is provided with outdoor yard hydrants.

The warehouse is located away from safety-related plant areas such that a fire will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.5 Service Building

The maintenance shop and offices are located in a Service Building that is common for both units. The building is comprised of mixed use occupancy in accordance with building code criterion. The facility is noncombustible unprotected construction. The potential combustible leading is estimated to be greater than one hour (>80,000 Btu/ft^2); thus, the facility requires detection capability and automatic and manual fire suppression in accordance with Appendix 9A.2.4.

The building incorporates full area automatic sprinkler protection, with portable fire extinguishers and a Class II standpipe hose system for manual firefighting. Additionally, the facility is provided with outdoor yard hydrants.

The building incorporates a fire alarm system comprised of supplemental automatic smoke detection throughout the office areas, manual pull stations, and automatic audible alarm notification appliances. The automatic sprinklers satisfy the detection requirement in the maintenance shop area, where smoke detection may not be appropriate, so supplemental detection is not required in this area. The fire alarm system also produces an audible and visual alarm in the main control room and the security central alarm station.

The maintenance shop and offices are located away from safety-related plant areas such that a fire will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.6 Sanitary Waste Treatment

A modular prefabricated and packaged sanitary treatment facility is provided for processing domestic wastewater, and serves both units. The system incorporates blowers; pumps; equalization, aeration, and sludge holding tanks; clarifiers; and disinfection units, with all necessary piping, valves and controls. The modular system is located outdoors and is constructed of combustible and noncombustible materials.

A portable fire extinguisher is provided at an accessible location immediately adjacent to the equipment and outdoor yard hydrants are located near the unit for manual firefighting.

The sanitary waste treatment facility is located away from safety-related plant areas such that a fire involving the equipment will not damage equipment, components, or structures required for safe shutdown of the plant.

9A.3.3.7 Hydrogen Storage Tank Area

A bulk hydrogen gas storage area is provided for each unit. The storage area is located outdoors at a safe distance of greater than 100 feet from all other buildings on site. The storage area is located at a safe distance of greater than 640 feet from safety-related structures, systems and components (SSCs). Additionally, the area is located remote from overhead power lines.

The storage area is comprised of a concrete slab and valve station for piping interface with the unit. Gas storage will use mobile tube trailers, designed and constructed in accordance with DOT (U.S. Department of Transportation) requirements. Provisions are included to electrically bond the mobile trailer unit to the valve station and secure it in position to prevent movement. The storage area is enclosed by a fence to prevent unauthorized personnel from entering the area.

Electrical equipment located within a 15-foot radius of the storage area is rated Class I, Division 2, Group B in accordance with Article 501 of the NEC (National Electrical Code).

A portable fire extinguisher is provided at an accessible location immediately adjacent to the equipment and outdoor yard hydrants are located in proximity to the unit, for manual fire fighting.

The hydrogen gas storage area is located away from safety-related plant areas such that a fire involving the equipment will not damage equipment, components, or structures required for safe shutdown of the plant.

STD COL 9.5-3 Stairwells in miscellaneous buildings located in the yard serving as escape routes or access routes for firefighting are enclosed in masonry or concrete towers with a minimum fire resistance rating of 2 hours and self-closing Class B fire doors. The two-hour fire-resistance rating for the masonry or concrete material is based on testing conducted in accordance with ASTM E119 (Reference 201) and NFPA 251 (Reference 202).

9A.4 REFERENCES

201. American Society of Mechanical Engineers, “Standard Test Methods for Fire Tests of Building Construction and Materials,” ASTM E119-08a.

202. National Fire Protection Association, “Standard Methods of Tests of Fire Endurance of Building Construction and Materials,” NFPA 251, 2006.

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

VCS DEP 18.8-1 **(Note: This figure replaces DCD Figure 9A-3 Sheet 1 of 3. This replacement is necessary to support the alternate locations of the Technical Support Center and the Operations Support Center per Departure Number VCS DEP 18.8-1.)**

Figure 9A-201
[*Annex I & II Building Fire Areas Plan at Elevation 100'-0" & 107'-2"*]*

*NRC Staff approval is required prior to implementing a change to this information; see DCD Introduction Section 3.5.

CHAPTER 10
STEAM AND POWER CONVERSION

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CHAPTER 10
STEAM AND POWER CONVERSION

10.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.1.3 COMBINED LICENSE INFORMATION ON EROSION - CORROSION MONITORING

Add the following text at the end of DCD Subsection 10.1.3.

10.1.3.1 Erosion-Corrosion Monitoring

STD COL 10.1-1 The flow accelerated corrosion (FAC) monitoring program analyzes, inspects, monitors and trends those nuclear power plant components that are potentially susceptible to erosion-corrosion damage such as carbon steel components that carry wet steam. In addition, the FAC monitoring program considers the information of Generic Letter 89-08, EPRI NSAC-202L-R3, and industry operating experience. The program requires a grid layout for obtaining consistent pipe thickness measurements when using Ultrasonic Test Techniques. The FAC program obtains actual thickness measurements for highly susceptible FAC locations for new lines as defined in EPRI NSAC-202L-R3 (Reference 201). At a minimum, a CHECWORKS type Pass 1 analysis is used for low and highly susceptible FAC locations and a CHECWORKS type Pass 2 analysis is used for highly susceptible FAC locations when Pass 1 analysis results warrant. To determine wear of piping and components where operating conditions are inconsistent or unknown, the guidance provided in EPRI NSAC-202L is used to determine wear rates.

10.1.3.1.1 Analysis

An industry-sponsored program is used to identify the most susceptible components and to evaluate the rate of wall thinning for components and piping potentially susceptible to FAC. Each susceptible component is tracked in a database and is inspected, based on susceptibility. Analytical methods utilize the results of plant-specific inspection data to develop plant-specific correction factors. This correction accounts for uncertainties in plant data, and for systematic discrepancies caused by plant operation. For each piping component, the analytical method predicts the wear rate, and the estimated time until it must be re-inspected, repaired, or replaced. Carbon steel piping (ASME III and B31.1) that is used for single or multi-phase high temperature flow are the most susceptible to erosion-corrosion damage and receive the most critical analysis.

10.1.3.1.2 Industry Experience

Review and incorporation of industry experience provides a valuable supplement to plant analysis. Industry experience is used to update the program by identifying susceptible components or piping features.

10.1.3.1.3 Inspections

Wall thickness measurements establish the extent of wear in a given component, provide data to help evaluate trends, and provide data to refine the predictive model. Components are inspected for wear using ultrasonic techniques (UT), radiography techniques (RT), or by visual observation. The initial inspections are used as a baseline for later inspections. Each subsequent inspection determines the wear rate for the piping and components and the need for inspection frequency adjustment for those components.

10.1.3.1.4 Training and Engineering Judgement

The FAC program is administered by both trained and experienced personnel. Task specific training is provided for plant personnel that implement the monitoring program. Specific non-destructive examination (NDE) is carried out by personnel qualified in the given NDE method. Inspection data is analyzed by engineers or other experienced personnel to determine the overall effect on the system or component.

10.1.3.1.5 Long-Term Strategy

This strategy focuses on reducing wear rates and performing inspections on the most susceptible locations.

10.1.3.2 Procedures

10.1.3.2.1 Generic Plant Procedure

The FAC monitoring program is governed by procedure. This procedure contains the following elements:

- A requirement to monitor and control FAC.
- Identification of the tasks to be performed and associated responsibilities.
- Identification of the position that has overall responsibility for the FAC monitoring program at each plant.
- Communication requirements between the coordinator and other departments that have responsibility for performing support tasks.
- Quality Assurance requirements.

- Identification of long-term goals and strategies for reducing high FAC wear rates.
- A method for evaluating plant performance against long-term goals.

10.1.3.2.2 Implementing Procedures

The FAC implementing procedures provide guidelines for controlling the major tasks. The plant procedures for major tasks are as follows:

- Identifying susceptible systems.
- Performing FAC analysis.
- Selecting and scheduling components for initial inspection.
- Performing inspections.
- Evaluating degraded components.
- Repairing and replacing components when necessary.
- Selecting and scheduling locations for the follow-on inspections.
- Collection and storage of inspections records.

10.1.3.3 Plant Chemistry

The responsibility for system chemistry is under the purview of the plant chemistry section. The plant chemistry section specifies chemical addition in accordance with plant procedures.

Add the following after DCD Subsection 10.1.3:

10.1.4 REFERENCES

201. EPRI NSAC-202L-R3, Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R3), Electric Power Research Institute (EPRI) Technical Report 1011838, Palo Alto, CA, 2006.

10.2 TURBINE-GENERATOR

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.2.2 SYSTEM DESCRIPTION

Add the following sentence at the end of the second paragraph of DCD Subsection 10.2.2.

STD SUP 10.2-1 Subsection 3.5.1.3 addresses the probability of generation of a turbine missile for AP1000 plants in a side-by-side configuration.

Add the following statement at the end of DCD Subsection 10.2.2.

STD SUP 10.2-4 Preoperational and startup tests provide guidance to operations personnel to ensure the proper operability of the turbine generator system.

10.2.3 TURBINE ROTOR INTEGRITY

Add the following statement at the end of DCD Subsection 10.2.3.

STD SUP 10.2-5 Operations and maintenance procedures mitigate the following potential degradation mechanisms in the turbine rotor and buckets/blades: pitting, stress corrosion cracking, corrosion fatigue, low-cycle fatigue, erosion, and erosion-corrosion.

10.2.3.6 Maintenance and Inspection Program Plan

Add the following at the end of DCD Subsection 10.2.3.6.

STD SUP 10.2-3 The inservice inspection (ISI) program for the turbine assembly provides assurance that rotor flaws that lead to brittle fracture of a rotor are detected. The ISI program also coincides with the ISI schedule during shutdown, as required by the ASME Boiler and Pressure Vessel Code, Section XI, and includes complete inspection of all significant turbine components, such as couplings, coupling bolts, turbine shafts, low-pressure turbine blades, low-pressure rotors, and high-pressure rotors. This inspection consists of visual, surface, and volumetric examinations required by the code.

10.2.6 COMBINED LICENSE INFORMATION ON TURBINE MAINTENANCE AND INSPECTION

Replace the text in DCD Subsection 10.2.6 with the following:

STD COL 10.2-1 A turbine maintenance and inspection program will be submitted to the NRC staff for review prior to fuel load. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in DCD Subsection 10.2.3.6. Plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis will be available for review after fabrication of the turbine and prior to fuel load.

10.3 MAIN STEAM SUPPLY SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.3.2.2.1 Main Steam Piping

Add the following at the end of DCD Subsection 10.3.2.2.1.

STD SUP 10.3-1 Operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer, including:

- Prevention of rapid valve motion
- Process for avoiding introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Process for avoiding introduction of steam or heated water that can flash into water-filled lines and components
- Cautions for introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

10.3.5.4 Chemical Addition

Add the following at the end of DCD Subsection 10.3.5.4.

STD SUP 10.3-2 Alkaline chemistry supports maintaining iodine compounds in their nonvolatile form. When iodine is in its elemental form, it is volatile and free to react with organic compounds to create organic iodine compounds, which are not assumed to remain in solution. It is noted that no significant level of organic compounds is expected in the secondary system. The secondary water chemistry, thus, does not directly impact the radioactive iodine partition coefficients.

10.3.6.2 Material Selection and Fabrication

Add the following at the end of DCD Subsection 10.3.6.2.

STD SUP 10.3-3 Appropriate operations and maintenance procedures provide the necessary controls during operation to minimize the susceptibility of components made of stainless steel and nickel-based materials to intergranular stress-corrosion cracking by controlling chemicals that are used on system components.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.4.2.2.1 General Description

Revise the first sentence of the third paragraph of DCD Subsection 10.4.2.2.1 to remove the brackets.

VCS CDI

The circulating water system (CWS), or water supplied by the raw water system (RWS) when the CWS is not in operation, provides the cooling water for the vacuum pump seal water heat exchangers.

10.4.2.2.2 Component Description

Revise the fourth sentence of the first paragraph of DCD Subsection 10.4.2.2.2 to remove the brackets.

VCS CDI

Seal water flows through the shell side of the seal water heat exchanger and circulating water, or water supplied by the RWS when the CWS is not in operation, flows through the tube side.

Subsection 10.4.5 is modified using full text incorporation to provide site specific information to replace the DCD conceptual design information (CDI).

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Basis

10.4.5.1.1 Safety Design Basis

The circulating water system (CWS) serves no safety-related function and therefore has no nuclear safety design basis.

10.4.5.1.2 Power Generation Design Basis

VCS CDI

The CWS supplies cooling water to remove heat from the main condenser. The CWS or makeup water from the RWS supplies cooling water to the turbine building closed cooling water system (TCS) heat exchangers and the condenser vacuum pump seal water heat exchangers under varying conditions of power plant loading and design weather conditions.

DCD

10.4.5.2 System Description

10.4.5.2.1 General Description

Classification of components and equipment in the circulating water system is given in Section 3.2.

VCS COL 10.4-1

The CWS provides a heat sink for the waste heat exhausted from the steam turbine to the main condenser and dissipates this waste heat to the atmosphere using cooling towers. The CWS also provides cooling for the TCS heat exchangers and the condenser vacuum pump seal water heat exchangers. The CWS is shown in Figure 10.4-201. CWS design parameters are provided in Table 10.4-201 and Table 10.4-202.

VCS CDI

The CWS consists of three 33-1/3-percent-capacity circulating water pumps, two mechanical-draft cooling towers, and associated piping, valves, and instrumentation.

DCD

Makeup water to the CWS is provided by the raw water system (RWS). In addition, water chemistry is controlled by a local chemical feed system.

10.4.5.2.2 Component Description

Circulating Water Pumps

VCS CDI

The three circulating water pumps are vertical, wet pit, single-stage, mixed-flow pumps driven by electric motors. The pumps are mounted in an intake structure connected to the cooling tower basins by open flumes. The three pump discharge lines combine in a single main header at the intake structure. This main header with two supply lines to the turbine building forms a common header which connects to the two inlet water boxes of the condenser and may also supply cooling water to the TCS and condenser vacuum pump seal water heat exchangers. Each pump discharge line has a motor-operated butterfly valve located between the pump discharge and the main header. This permits isolation of one pump for maintenance and allows two-pump operation.

Cooling Towers

VCS COL 10.4-1 Two mechanical induced-draft, counterflow cooling towers are sized to reject a single unit's full-load waste heat to the atmosphere and cool the circulating water to less than 91°F based on an entering wet-bulb temperature of 79.4°F. Heat is rejected to the atmosphere primarily through evaporative cooling as circulating water returned from the condenser drops through the tower fill to the tower basins from which it is returned through open flumes to the CWS pump intake structure.

Cooling Tower Makeup and Blowdown

DCD The circulating water system makeup is provided by the raw water system.

VCS CDI Makeup to and blowdown from the CWS is controlled by the makeup and blowdown control valves. These valves, along with a local chemical feed system provide chemistry control in the circulating water in order to maintain a noncorrosive, nonscale-forming condition and limit biological growth in CWS components.

Makeup water may be used as a source of cooling water for the TCS and condenser vacuum pump seal water heat exchangers.

DCD **Piping and Valves**

VCS CDI The underground portions of the CWS piping are constructed of prestressed concrete pressure piping. The remainder of the piping is carbon steel and is coated internally with a corrosion-resistant compound.

VCS COL 10.4-1 Condenser water box drains allow the condenser to be drained to the cooling tower basin. Piping is routed from each water box to the condenser water box drain pump which in turn pumps the water back to the cooling tower. Each water box contains drain valves and vents so that a water box can be drained individually. Piping is sized to support an adequate drain down in the event of emergency maintenance.

DCD Motor-operated butterfly valves are provided in each of the circulating water lines at their inlet to and exit from the condenser shell to allow isolation of portions of the condenser.

VCS CDI Control valves provide regulation of cooling tower blowdown and makeup.

DCD The circulating water system is designed to withstand the maximum operating discharge pressure of the circulating water pumps.

VCS CDI Piping includes the expansion joints, butterfly valves, condenser water boxes, and tube bundles.

VCS COL 10.4-1 The design pressure of the condenser portions of the piping is identified in DCD Table 10.4.1-1. The design pressure of the remaining piping is 100 psig.

DCD **Circulating Water Chemical Injection**

Circulating water chemistry is maintained by a local chemical feed system skid at the CWS cooling tower.

VCS CDI Circulating water system chemical feed equipment injects the required chemicals into the circulating water at the CWS cooling tower basin area.

DCD This maintains a noncorrosive, nonscale-forming condition and limits the biological film formation that reduces the heat transfer rate in the condenser and the heat exchangers supplied by the circulating water system.

VCS COL 10.4-1 The specific chemicals used within the system are based on site water conditions as determined by plant chemistry.

DCD The chemicals can be divided into six categories based upon function: biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor, and a silt dispersant. The pH adjuster, corrosion inhibitor, scale inhibitor, and dispersant are metered into the system continuously or as required to maintain proper concentrations. The biocide application frequency may vary with seasons.

VCS CDI The algaecide is applied, as necessary, to control algae formation on the cooling towers.

VCS COL 10.4-1 The following chemicals are used to control circulating water chemistry:

- Biocide - Sodium hypochlorite
- Algaecide – Quaternary amine
- pH Adjuster – Sulfuric acid
- Corrosion Inhibitor – Ortho/polyphosphate
- Scale Inhibitor – Phosphonate
- Silt Dispersant – Polymeric silt dispersant

DCD Addition of biocide and water treatment chemicals is performed by local chemical feed injection metering pumps and is adjusted as required.

VCS CDI Chemical concentrations are measured through analysis of grab samples from the CWS.

DCD Residual chlorine is measured to monitor the effectiveness of the biocide treatment.

10.4.5.2.3 System Operation

VCS CDI

The three circulating water pumps take suction from the circulating water pump pit and circulate the water through the tube side of the main condenser, with smaller flows to the TCS and the condenser vacuum pump seal water heat exchangers, and back through the piping discharge network to the cooling towers. See Figure 10.4-201. The cooling towers cool the circulating water by discharging the water above the tower fill material, through which the water then falls to the basins beneath the towers and, in the process, rejects heat to the atmosphere.

Circulating water flow to the cooling towers can be diverted directly to the basins, bypassing the cooling towers' internals, by opening the bypass valves. The bypass can be used during plant startup or partial load or to maintain CWS temperatures above 40°F while operating during periods of cold weather.

VCS CDI

The raw water system supplies makeup water to the cooling tower basins to replace water losses due to evaporation, drift, and blowdown. Connections are provided to supply water from the RWS to fill the CWS piping and supply cooling for the TCS and condenser vacuum pump seal water heat exchangers when the CWS is not in operation.

DCD

A condenser tube cleaning system is installed to clean the circulating water side of the main condenser tubes.

VCS CDI

Blowdown from the CWS is taken from the discharge of the CWS pumps and is discharged to the plant outfall.

DCD

The circulating water system is used to supply cooling water to the main condenser to condense the steam exhausted from the main turbine.

VCS CDI

If the CWS malfunctions such that condenser backpressure rises above the maximum allowable value, the main condenser will no longer be able to adequately support unit operation.

DCD

Cooldown of the reactor may be accomplished by using the power-operated atmospheric steam relief valves or safety valves rather than the turbine bypass system when the condenser is not available.

Passage of condensate from the main condenser into the circulating water system through a condenser tube leak is not possible during power generation operation, since the circulating water system operates at a greater pressure than the condenser.

VCS CDI

Turbine building closed cooling water in the TCS heat exchangers is maintained at a higher pressure than the circulating water or raw water to prevent leakage of the circulating water or raw water into the closed cooling water system.

Cooling water to the condenser vacuum pump seal water heat exchangers is supplied from the circulating water or raw water system. Cooling water flow from the circulating water system is normally maintained through all four heat exchangers to facilitate placing the spare condenser vacuum pump in service.

DCD

Isolation valves are provided for the condenser vacuum pump seal water heat exchanger cooling water supply lines to facilitate maintenance.

Small circulating water system leaks in the turbine building will drain into the waste water system. Large circulating water system leaks due to pipe failures will be indicated in the control room by a loss of vacuum in the condenser shell. The effects of flooding due to a circulating water system failure, such as the rupture of an expansion joint, will not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the turbine building and the base slab of the turbine building is located at grade elevation. Water from a system rupture will run out of the building through a relief panel in the turbine building west wall before the level could rise high enough to cause damage. Site grading will carry the water away from safety-related buildings.

VCS CDI

The cooling towers are located to prevent adverse interactions with structures, systems, and components (SSCs) required for safe shutdown of the plant in case of a collapse of the towers or failure of their basins and flumes. Failure of the cooling tower basins, flumes or associated circulating water system piping will not have an adverse effect on safety-related SSCs resulting from external flooding due to the location of the cooling towers (greater than 600 feet from safety-related SSCs) in combination with site grading to direct surface water away from the nuclear island.

DCD

10.4.5.3 Safety Evaluation

The circulating water system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.5.4 Tests and Inspections

Components of the circulating water system are accessible as required for inspection during plant power generation.

VCS CDI

The circulating water pumps are tested in accordance with standards of the Hydraulic Institute.

DCD

Performance, hydrostatic, and leakage tests associated with preinstallation and preoperational testing are performed on the circulating water system. The system performance and structural and leaktight integrity of system components are demonstrated by continuous operation.

10.4.5.5 Instrumentation Applications

VCS CDI

Instrumentation provided indicates the open and closed positions of motor-operated butterfly valves in the circulating water piping. The motor-operated valve at each pump discharge is interlocked with the pump so that the pump trips if the discharge valve fails to reach the full-open position shortly after starting the pump.

Local grab samples are used to periodically test the circulating water quality to limit harmful effects to the system piping and valves due to improper water chemistry.

Pressure indication is provided on the circulating water pump discharge lines.

DCD

A differential pressure transmitter is provided between one inlet and outlet branch to the condenser. This differential pressure transmitter is used to determine the frequency of operating the condenser tube cleaning system (CES).

VCS CDI

Temperature indication is supplied on the common CWS inlet header to the TCS heat exchanger trains. This temperature is also representative of the inlet cooling water temperature to the main condenser.

A flow element is provided for the common discharge line from the TCS heat exchangers to allow monitoring of the total flow through the TCS heat exchangers. Flow measurement for the raw water makeup to the cooling towers and for CWS blowdown is also provided.

VCS CDI

Level instrumentation provided in the circulating water pump intake structure and cooling tower basins controls makeup flow from the RWS to the cooling tower basins and annunciates on low-water level at the pump intake structure and on high-water level in the cooling tower basins.

VCS COL 10.4-1

The circulating water chemistry is controlled by CWS blowdown and chemical addition to maintain the circulating water with an acceptable Langelier Index as specified in plant chemistry procedures. The system accomplishes this by regulating the blowdown valve. This regulation causes the cooling tower basin water levels to fluctuate. The fluctuation is sensed by level controllers that operate the cooling tower makeup valves.

DCD

The control approach is to allow the makeup water to concentrate naturally to its upper limit. Provisions are made to add chemicals for pH control.

VCS CDI

The cycles of concentration at which the cooling towers are operated is dependent on the quality of the cooling tower makeup water. Blowdown is directed to the waste water system (blowdown sump) and dechlorinated as needed before discharge at the plant outfall.

DCD

Monitoring of the circulating water system is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. See Chapter 7.

10.4.7.2.1 General Description

Replace the last sentence of the sixth paragraph of DCD Subsection 10.4.7.2.1 as follows.

VCS COL 10.4-2 The oxygen scavenger agent is hydrazine and the pH control agent is morpholine.

STD SUP 10.4-2 Oxygen scavenging and ammoniating agents are selected and utilized for plant secondary water chemistry optimization following the guidance of NEI-97-06, “Steam Generator Program Guidelines” (Reference 201). The EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines are followed as described in NEI 97-06.

Add new paragraph at the end of the DCD Subsection 10.4.7.2.1:

STD SUP 10.4-1 Operations and maintenance procedures include precautions, when appropriate, to minimize the potential for steam and water hammer, including:

- Prevention of rapid valve motion
- Process for avoiding introduction of voids into water-filled lines and components
- Proper filling and venting of water-filled lines and components
- Process for avoiding introduction of steam or heated water that can flash into water-filled lines and components
- Cautions for introduction of water into steam-filled lines or components
- Proper warmup of steam-filled lines
- Proper drainage of steam-filled lines
- The effects of valve alignments on line conditions

10.4.12.1 Circulating Water System

VCS COL 10.4-1 This COL Item is addressed in Subsection 10.4.5 with specific discussion of CWS configuration in Subsection 10.4.5.2.1, design pressure and cooling towers in Subsection 10.4.5.2.2, and specific chemicals and chemistry in Subsections 10.4.5.2.2 and 10.4.5.5.

10.4.12.2 Condensate, Feedwater and Auxiliary Steam System Chemistry Control.

VCS COL 10.4-2 This COL Item is addressed in Subsection 10.4.7.2.1.

10.4.12.3 Potable Water

VCS COL 10.4-3 This COL item is duplicated in the Subsection 9.2.12.1 COL Item and is addressed as stated in that subsection.

10.4.13 REFERENCES

201. Nuclear Energy Institute, “Steam Generator Program Guidelines,” NEI 97-06, Revision 2, May 2005.

Table 10.4-201
Supplemental Main Condenser Design Data

	Condenser Data	
VCS CDI	Circulating water flow	600,000 gpm

Note: This table supplements DCD Table 10.4.1-1.

VCS COL 10.4-1

Table 10.4-202
Supplemental Design Parameters For Major Circulating Water System Components

Circulating Water Pump	
Quantity	Three per unit
Flow rate (gal/min)	211,267
Mechanical Draft Cooling Tower	
Quantity	Two per unit
Approach temperature (°F)	10.7
Inlet temperature (°F)	114.8
Outlet temperature (°F)	90.1
Approximate Temperature range (°F)	24.7
Flow rate (gal/min)	309,400
Heat transfer (Btu/hr)	$3,815 \times 10^6$
Wind velocity design (mph)	110
Seismic design criteria per International Building Code	

Note: This table supplements DCD Table 10.4.5-1.

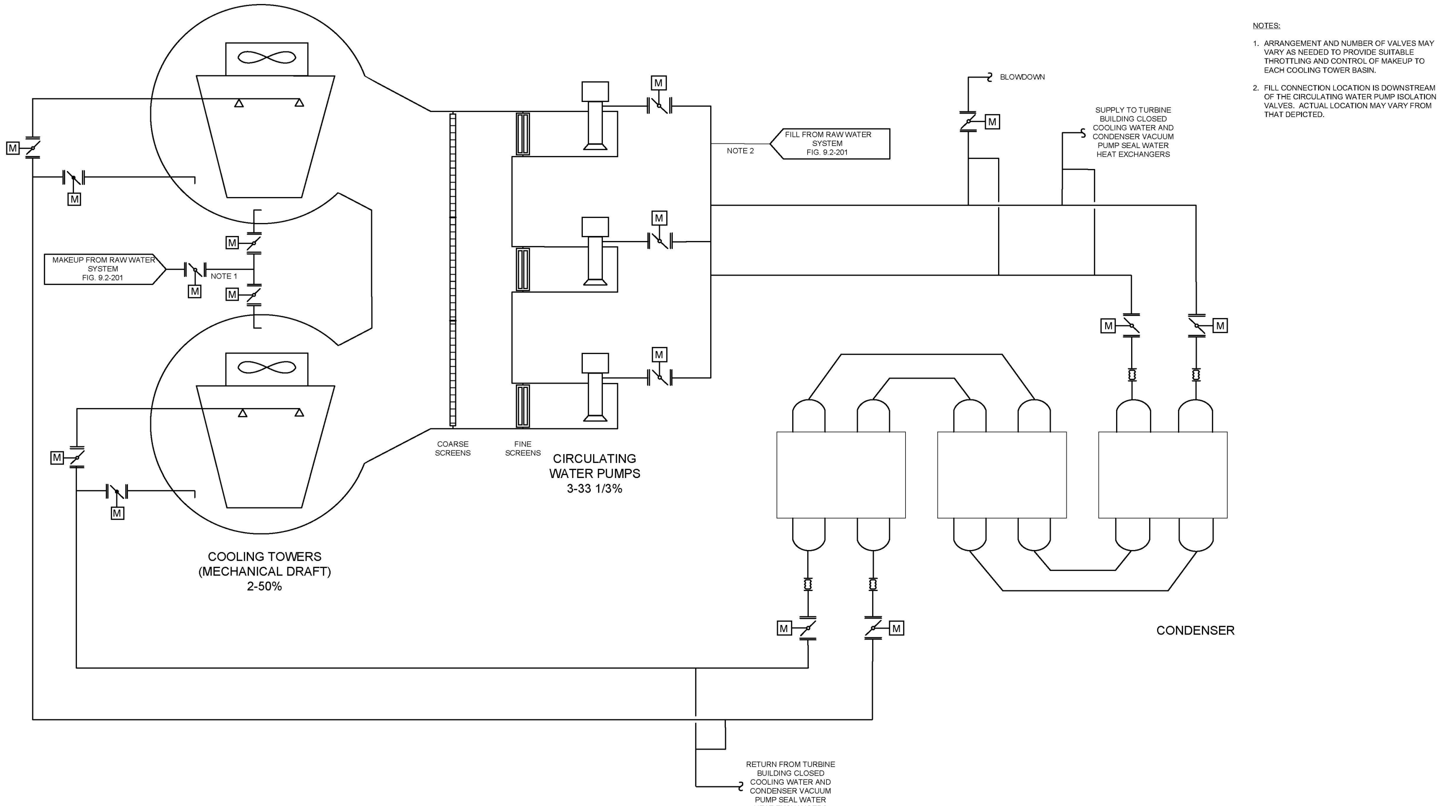

Figure 10.4-201. Circulating Water System Flow Diagram

CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

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CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.2.1.2.4 Controlled Release of Radioactivity

Add the following new paragraph at the end of DCD Subsection 11.2.1.2.4:

VCS SUP 11.2-1 The Liquid Radwaste System (WLS) discharge piping from the Units 2 and 3 Radwaste Building is stainless steel, enclosed within a guard pipe, and monitored for leakage to comply with 10 CFR 20.1406. The WLS discharge piping connects to the Waste Water System (WWS) blowdown line within the Exclusion Area Boundary for dilution to meet the release limits of 10 CFR Part 20 Appendix B, Table II, Column 2. Dilution at this point, downstream of the WWS Blowdown Sump, is primarily supplied from circulating water blowdown flow.

The WWS blowdown line to the Plant Outfall at Parr Reservoir is a buried, high density polyethylene single-walled pipe. Waste water gravity drains from the Blowdown Sump to the diffuser at the Plant Outfall. There are no valves, vacuum breakers, or pumps along the WWS blowdown line between the point where WLS connects and the Plant Outfall. Monitoring for leakage of the WWS blowdown line will be evaluated and implemented if necessary as part of the Units 2 and 3 Groundwater Monitoring Program described in NEI 08-08A (Reference 203).

11.2.1.2.5.2 Use of Mobile and Temporary Equipment

Add the following information at the end of DCD Subsection 11.2.1.2.5.2:

STD COL 11.2-1 When mobile or temporary equipment is selected to process liquid effluents, the equipment design and testing meets the applicable requirements of Regulatory Guide 1.143. When confirmed through sampling that the radioactive waste contents do not exceed the A_2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71, the liquid effluent may be processed with mobile or temporary equipment in the Radwaste Building. When the A_2 quantities are exceeded, liquid effluent is processed in the Seismic Category I auxiliary building.

Mobile and temporary equipment are designed in accordance with the applicable mobile and temporary radwaste treatment systems guidance provided in Regulatory Guide 1.143, including the codes and standards listed in Table 1 of the Regulatory Guide.

Mobile and temporary equipment have the following features:

- Level indication and alarms (high-level) on tanks.
- Screwed connections are permitted only for instrument connections beyond the first isolation valve.
- Remote operated valves are used where operations personnel would be required to frequently manipulate a valve.
- Local control panels are located away from the equipment, in low dose areas.
- Instrumentation readings are accessible from the local control panels (i.e., temperature, flow, pressure, liquid level, etc.).
- Wetted parts are 300 series stainless steel, except flexible hose and gaskets.
- Flexible hose is used only for mobile equipment within the designated “black box” locations between mobile components and at the interface with the permanent plant piping.
- The contents of tanks are capable of being mixed, either through recirculation or with a mixer.
- Grab sample points are located in tanks and upstream and downstream of the process equipment.

Inspection and testing of mobile or temporary equipment is in accordance with the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the following additions:

- After placement in the station, the mobile or temporary equipment is hydrostatically, or pneumatically, tested prior to tie-in to permanent plant piping.
- A functional test, using demineralized water, is performed. Remote operated valves are stroked (open-closed-open or closed-open-closed) under full flow conditions. The proper function of the instrumentation, including alarms, is verified. The operating procedures are verified correct during the functional test.
- Tank overflows are routed to floor drains.
- Floor drains are confirmed to be functional prior to placing mobile or temporary equipment into operation.

11.2.3.3 Dilution Factor

Add the following information at the end of DCD Subsection 11.2.3.3.

VCS COL 11.2-2 VCS COL 11.5-3 The dilution factors used for the maximum exposed individual and the population dose are calculated by the LADTAP II code in accordance with Regulatory Guide 1.113. LADTAP II input requires information on whether effluent discharge is into a river or lake, and the average flow rate.

In calculating the effluent doses, it is assumed that there is no dilution of the effluent discharge prior to entering the Broad River at the Parr Reservoir. Neglecting the blowdown flow rate of 6,000 gpm, the effluent discharge is assumed to be directly diluted by the flow rate of the Broad River. The minimum annual average flow rate of the Broad River is 1782 cfs. The Parr Reservoir retention time is four days.

The dilution factors and a summary of parameters used to calculate them are presented in Table 11.2-201.

11.2.3.5 Estimated Doses

Replace the information in DCD Subsection 11.2.3.5 with the following paragraphs and subsections.

VCS COL 11.2-2 VCS COL 11.5-3 Dose and dose rate to man was calculated using the LADTAP II computer code. This code is based on the methodology presented in Regulatory Guide 1.109. Factors common to both estimated individual dose rates and estimated population dose are addressed here. Unique data are discussed in the respective sections.

Activity pathways considered are drinking water, sport fishing, irrigated farm products, and recreational activities.

The irrigated farm products are vegetables, leafy vegetables, milk, and meat.

Drinking water from the Broad River is consumed by half the population of the city of Columbia and all the population of Fort Jackson using data from the state of South Carolina. The farm production is based on data for vegetables, leafy vegetables, milk, and meat from the state of South Carolina. The food production within the 50-mile radius of VCSNS is based on the total food production in each category multiplied by the ratio of the land area within the 50-mile radius (adjusted for nonproduction areas) to the total land area of the state. An irrigation model is used for food products. The food production rate using irrigation water is determined by multiplying the 50-mile production rates by the ratio of population using drinking water to the total population within the 50-mile radius, and the fraction of irrigated to harvested cropland using data from the state of South Carolina.

11.2.3.5.1 Estimated Individual Dose Rate

Dose rates to individuals are calculated for drinking water, sport fish consumption, irrigated farm products, and recreational activities.

Table 11.2-202 contains LADTAP II input data for dose rate calculations. Table 11.2-203 gives the maximum individual dose rates.

The total site doses due to liquid and gaseous effluents from the existing Unit 1 and Units 2 and 3 would be well within the regulatory limits of 40 CFR Part 190, as shown in Table 11.3-206. The values in this table for Unit 1 are representative based on review of the Unit 1 annual radiological operating reports (Reference 202).

11.2.3.5.2 Estimated Population Dose

The population dose is based on the fraction of the 50-mile population that will be exposed to the evaluated pathways. These pathways are drinking water, recreational activities, irrigated farm products, and sport fishing.

The sport fishing harvest is estimated using data from the state of South Carolina. The sport fishing harvest is estimated to be 3.77 x 10^5 kg/yr. Recreational activities include swimming, boating, and shoreline use. The annual usage for each of these activities is estimated to be 3.59 x 10^5, 3.59 x 10^6, and 3.59 x 10^6 person-hours, respectively.

The population doses are given in Table 11.2-204.

Table 11.2-204 shows that the total body and thyroid population doses per unit are approximately 14.6 and 6.5 person-rem per unit, respectively.

11.2.3.5.3 Liquid Radwaste Cost Benefit Analysis Methodology

STD COL 11.2-2 The application of the methodology of Regulatory Guide 1.110 was used to satisfy the cost benefit analysis requirements of 10 CFR Part 50, Appendix I, Section II.D. The parameters used in calculating the Total Annual Cost (TAC) are fixed and are given for each radwaste treatment system augment listed in Regulatory Guide 1.110, including the Annual Operating Cost (AOC) (Table A-2), Annual Maintenance Cost (AMC) (Table A-3), Direct Cost of Equipment and Materials (DCEM) (Table A-1), and Direct Labor Cost (DLC) (Table A-1). The following variable parameters were used:

- Capital Recovery Factor (CRF) — This factor is taken from Table A-6 of Regulatory Guide 1.110 and reflects the cost of money for capital expenditures. A cost-of-money value of 7 percent per year is assumed in this analysis, consistent with the “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission” (NUREG/BR-0058). A CRF of 0.0806 was obtained from Table A-6.

- Indirect Cost Factor (ICF) — This factor takes into account whether the radwaste system is unitized or shared (in the case of a multi-unit site) and is taken from Table A-5 of Regulatory Guide 1.110. It is assumed that the radwaste system for this analysis is a unitized system at a 2-unit site, which equals an ICF of 1.625.

- Labor Cost Correction Factor (LCCF) — This factor takes into account the differences in relative labor costs between geographical regions and is taken from Table A-4 of Regulatory Guide 1.110. A LCCF of 1.0 (the lowest value) is assumed in this analysis.

Appendix I to 10 CFR Part 50 prescribes a $1,000 per person-rem criterion for determining the cost benefit of actions to reduce radiation exposure.

The analysis used a conservative assumption that the respective radwaste treatment system augment is a “perfect” system that reduces the effluent and dose by 100 percent. The liquid radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for liquid radwaste treatment system augments is a 20 gpm Cartridge Filter at $11,140 per year, which yields a threshold value of 11.14 person-rem total body or thyroid dose from liquid effluents.

For AP1000 sites with population dose estimates less than 11.14 person-rem total body or thyroid dose from liquid effluents, no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I Section II.D.

11.2.3.5.4 Liquid Radwaste Cost Benefit Analysis

VCS COL 11.2-2 The population doses are given in Table 11.2-204. As discussed above, the lowest cost liquid radwaste system augment is $11,140. Assuming 100 percent efficiency of this augment, the minimum possible cost per person-rem is determined by dividing the cost of the augment by the population dose. This is $11,140/14.6 person-rem total body or $763 per person-rem total body, and $11,140/6.5 person-rem thyroid or $1,714 per person-rem thyroid. The cost per person-rem total body does not exceed the $1000 per person-rem criterion provided in Regulatory Guide 1.110, and therefore requires evaluation. The augment that requires evaluation is the 20 gpm cartridge filter. Of the 14.6 person-rem total body dose, 4.6 person-rem is due to tritium, which will not be mitigated by the 20 gpm cartridge filter. Assuming this augment completely eliminates the dose of 10 person-rem total body due to isotopes other than tritium, the cost of total body dose reduction is $11,140/10 person-rem total body or $1,114 per person-rem total body. Therefore this augment is not cost-beneficial in reducing the total body dose.

11.2.3.6 Quality Assurance

STD SUP 11.2-1 Add the following to the end of DCD Subsection 11.2.3.6:

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the liquid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.2.5 COMBINED LICENSE INFORMATION

11.2.5.1 Liquid Radwaste Processing by Mobile Equipment

STD COL 11.2-1 This COL Item is addressed in Subsection 11.2.1.2.5.2.

11.2.5.2 Cost Benefit Analysis of Population Doses

STD COL 11.2-2 This COL Item is addressed in Subsection 11.2.3.5.3.

VCS COL 11.2-2 This COL Item is addressed in Subsections 11.2.3.3, 11.2.3.5, 11.2.3.5.1, 11.2.3.5.2, and 11.2.3.5.4.

11.2.6 REFERENCES

201. Deleted.

202. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station, for the Operating Period January 1, 2005 – December 31, 2005; April 2006.

203. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

VCS COL 11.2-2

Table 11.2-201
Dilution Factor Parameters and Dilution Factors

Parameter	Average Annual Condition
Broad River Flow Rate (cfs)[a]	1782
Dilution Factor[a]	1

a) Assumed fully mixed model with annual average Broad River flow rate at Alston, SC for 1981–2008, United States Geological Survey, 2009.

VCS COL 11.2-2
VCS COL 11.5-3

Table 11.2-202 (Sheet 1 of 2)
LADTAP II Input[a]

Input Parameter	Value
Freshwater Site	Selected
Release source terms	DCD Table 11.2-7
Discharge Flow Rate	1782 ft^3 s^{-1}
Transit time to receptor	0.1, 96 hours[b]
Impoundment reconcentration model	None
50-mile population	FSAR Figures 2.1-211 and 2.1-219[c]
Shore width factor	0.2
Fish consumption	21 kg per year[d]
Drinking water consumption	730 liters per year[d]
Sport fishing harvest	3.77E+05 kg per year[e]
Commercial fishing harvest	1.21E+07 kg per year
50-mile drinking water population	299,930[f]
50-mile shoreline usage	3.59E+06 person-hours per year[g]
50-mile swimming usage	3.59E+05 person-hours per year[h]
50-mile boating usage	3.59E+06 person-hours per year[i]
Fraction of SC crops irrigated[j]	0.0696
Fraction of population using contaminated water for drinking and food production[k]	0.141
Fraction of SC agricultural products within 50 mi radius	0.258
Irrigation rate for food products[l]	110 liters per square meter per month
Fraction of contaminated water not used for feed or drinking water	0
Total production of vegetables within 50 mi radius[m]	6.86E+07 kg per year
Production rate for irrigated vegetables[n]	6.71E+05 kg per year
Total production of leafy vegetables within 50 mi radius[o]	1.80E+07 kg per year
Production rate for irrigated leafy vegetables[n]	1.76E+05 kg per year

VCS COL 11.2-2
VCS COL 11.5-3

Table 11.2-202 (Sheet 2 of 2)
LADTAP II Input[(a)]

Input Parameter	Value
Total production of milk within 50 mi radius[(p)]	6.78E+07 liters per year
Production rate for irrigated milk[(n)]	6.63E+05 liters per year
Total production of meat within 50 mi radius[(q)]	9.15E+08 kg per year
Production rate for irrigated meat[(n)]	8.96E+06 kg per year

a) Input parameters not specified use default LADTAP II values.
b) 0.1 hours assumed for maximally exposed individual (MEI) at the Parr Reservoir. 96 hours for downstream users reflecting reservoir retention time.
c) 2060 population projection.
d) Values in the table are for adult MEI. Average values of fish and water consumption of 6.9 kg and 370 liters per year, respectively, are used for population doses.
e) Boating population x 21 kg per year (adult MEI fish ingestion rate).
f) 2060 population projection.
g) Assumed same as boating usage.
h) Assumed 10% of shoreline usage.
i) Assumed 10% of boats registered in Fairfield, Lexington, Newberry, and Richland counties, 2 persons per boat, 200 hours per year.
j) USDA, National Agricultural Statistics Service, 2002 Census of Agriculture.
k) Fraction of contaminated water users (144,671) divided by the 50-mile population (1,028,075) in 2000.
l) 1 inch of water applied to the crops per week.
m) USDA, National Agricultural Statistics Service, 2005 and 2006, with apples and peaches included but leafy vegetables excluded, and projected to 2060.
n) Food product production rate multiplied by fraction of irrigated crops and fraction of contaminated water users.
o) USDA, Integrated Pest Management Center for leafy vegetables—2001, and projected to 2060.
p) *Milk Production, Disposition, and Income, 2006 Summary*, USDA, National Agricultural Statistics Service, April 2007, and projected to 2060. Density of producer milk is 1.03 kg per liter.
q) *South Carolina Agricultural Statistics, Crops, Livestock, and Poultry, 2005–2007*, USDA, National Agricultural Statistics Service. The total meat production in SC consists of broilers, turkey, commercial red meat, and young chickens. Projected to 2060.

VCS COL 11.5-3

Table 11.2-203 (Sheet 1 of 2)
Annual Individual Doses from Liquid Effluents (per Unit)

Pathway	Adult Dose (mrem/yr)							
	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish		4.5E-02	7.8E-02	5.8E-02	5.9E-03	2.7E-02	9.2E-03	6.2E-03
Drinking		1.0E-03	2.9E-02	2.9E-02	4.1E-02	2.8E-02	2.8E-02	3.5E-02
Shoreline	6.7E-05	5.7E-05	5.7E-05	5.7E-05	5.7E-05	5.7E-05	5.7E-05	5.7E-05
Irrigated Vegetables		6.1E-03	2.9E-02	2.6E-02	2.6E-02	2.4E-02	2.1E-02	5.9E-02
Irrigated Leafy Vegetables		7.7E-04	3.5E-03	3.2E-03	4.9E-03	3.0E-03	2.6E-03	7.4E-03
Irrigated Milk		4.4E-03	1.9E-02	1.7E-02	2.2E-02	1.4E-02	1.3E-02	1.3E-02
Irrigated Meat		6.3E-03	5.1E-03	5.6E-03	4.5E-03	1.6E-02	4.3E-03	3.8E-01
Total	6.7E-05	6.3E-02	1.6E-01	1.4E-01	1.0E-01	1.1E-01	7.7E-02	5.0E-01

Pathway	Teen Dose (mrem/yr)							
	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish		4.7E-02	8.0E-02	3.3E-02	5.5E-03	2.7E-02	1.1E-02	4.7E-03
Drinking		9.7E-04	2.1E-02	2.0E-02	3.1E-02	2.0E-02	2.0E-02	2.5E-02
Shoreline	3.8E-04	3.2E-04	3.2E-04	3.2E-04	3.2E-04	3.2E-04	3.2E-04	3.2E-04
Irrigated Vegetables		1.0E-02	3.9E-02	3.0E-02	3.4E-02	3.1E-02	2.6E-02	7.4E-02
Irrigated Leafy Vegetables		7.0E-04	2.6E-03	2.0E-03	3.6E-03	2.1E-03	1.7E-03	5.0E-03
Irrigated Milk		7.9E-03	2.9E-02	2.1E-02	3.2E-02	2.0E-02	1.7E-02	1.6E-02
Irrigated Meat		5.3E-03	3.2E-03	3.4E-03	2.8E-03	1.2E-02	2.6E-03	2.4E-01
Total	3.8E-04	7.2E-02	1.7E-01	1.1E-01	1.1E-01	1.1E-01	7.8E-02	3.6E-01

Table 11.2-203 (Sheet 2 of 2)
Annual Individual Doses from Liquid Effluents (per Unit)

	Child Dose (mrem/yr)							
Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish		5.8E-02	7.0E-02	1.3E-02	5.6E-03	2.3E-02	8.4E-03	2.1E-03
Drinking		2.8E-03	4.0E-02	3.8E-02	6.4E-02	3.9E-02	3.8E-02	4.2E-02
Shoreline	7.9E-05	6.7E-05	6.7E-05	6.7E-05	6.7E-05	6.7E-05	6.7E-05	6.7E-05
Irrigated Vegetables		2.4E-02	6.2E-02	4.3E-02	5.7E-02	4.9E-02	4.1E-02	7.8E-02
Irrigated Leafy Vegetables		1.2E-03	3.1E-03	2.2E-03	4.8E-03	2.5E-03	2.1E-03	4.0E-03
Irrigated Milk		1.9E-02	4.6E-02	2.8E-02	5.6E-02	3.2E-02	2.7E-02	2.5E-02
Irrigated Meat		9.9E-03	3.9E-03	4.3E-03	3.4E-03	1.6E-02	3.1E-03	1.5E-01
Total	7.9E-05	1.2E-01	2.3E-01	1.3E-01	1.9E-01	1.6E-01	1.2E-01	3.0E-01

	Infant Dose (mrem/yr)							
Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish		0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Drinking		3.0E-03	4.0E-02	3.7E-02	7.9E-02	3.8E-02	3.7E-02	4.0E-02
Shoreline		0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Total	0.0E+00	3.0E-03	4.0E-02	3.7E-02	7.9E-02	3.8E-02	3.7E-02	4.0E-02

	Maximum Dose (mrem/yr)[(a)]							
	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Dose	3.8E-04	1.2E-01	2.3E-01	1.4E-01	1.9E-01	1.6E-01	1.2E-01	5.0E-01
Age Group	Teen	Child	Child	Adult	Child	Child	Child	Adult

a) Doses meet 10 CFR 50, Appendix I limits of 3 mrem for total body and 10 mrem for any organ

VCS COL 11.2-2

Table 11.2-204
Annual Population Doses from Liquid Effluents (per Unit)

	Population Dose (person-rem/yr)							
Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Sport Fishing		9.87E-01	1.61E+00	9.87E-01	7.54E-02	5.44E-01	1.93E-01	1.03E-01
Commercial Fishing		8.97E+00	1.46E+01	8.95E+00	4.33E-01	4.93E+00	1.75E+00	8.95E-01
Drinking		1.96E-01	4.59E+00	4.49E+00	5.82E+00	4.47E+00	4.38E+00	5.26E+00
Hydrosphere Tritium		0.0E+00	7.70E-03	7.70E-03	7.70E-03	7.70E-03	7.70E-03	7.70E-03
Shoreline	2.01E-02			1.72E-02	1.72E-02			
Swimming				4.18E-05	4.18E-05			
Boating				2.09E-04	2.09E-04			
Irrigated Vegetables		1.21E-02	4.47E-02	3.71E-02	2.98E-02	3.64E-02	3.13E-02	7.60E-02
Irrigated Leafy Vegetables		2.63E-03	1.06E-02	9.28E-03	1.42E-02	8.76E-03	7.52E-03	2.08E-02
Irrigated Milk		1.66E-02	5.45E-02	4.12E-02	5.82E-02	3.87E-02	3.34E-02	3.25E-02
Irrigated Meat		1.25E-01	8.50E-02	9.30E-02	7.55E-02	2.78E-01	7.07E-02	5.80E+00
Total	2.01E-02	1.03E+01	2.10E+01	1.46E+01	6.53E+00	1.03E+01	6.47E+00	1.22E+01

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.3.3 RADIOACTIVE RELEASES

Add the following new paragraph at the end of DCD Subsection 11.3.3:

STD SUP 11.3-2 There are no gaseous effluent site interface parameters outside of the Westinghouse scope.

11.3.3.4 Estimated Doses

Add the following information at the end of DCD Subsection 11.3.3.4.

VCS COL 11.3-1 The VCSNS site-specific values are bounded by the DCD identified acceptable releases. With the annual airborne releases listed in DCD Table 11.3-3, the site
VCS COL 11.5-3 specific air doses at ground level at the site boundary are 0.71 mrad for gamma radiation and 3.0 mrad for beta radiation. These doses are based on the annual average atmospheric dispersion factor from FSAR Section 2.3. These doses are below the 10 CFR Part 50, Appendix I design objectives of 10 mrad per year for gamma radiation or 20 mrad per year for beta radiation.

Dose and dose rate to man was calculated using the GASPAR II computer code. This code is based on the methodology presented in Regulatory Guide 1.109. Factors common to both estimated individual dose rates and estimated population dose are addressed in this subsection. Unique data are discussed in the respective subsections. Activity pathways considered are plume, ground deposition, inhalation, and ingestion of vegetables, meat, and milk (both cow and goat).

Agricultural products are estimated from U.S. Department of Agriculture (USDA) National Agricultural Statistics Service. GASPAR II evenly distributes the food production over the entire 50 miles when given a total production for calculating dose.

The population doses are based on the population, projected to the year 2060, within a 50-mile radius of the centroid between Units 2 and 3. The population distribution is presented in FSAR Subsection 2.1.3 and Figures 2.1-211 and 2.1-219. Data from these figures are tabulated in Table 11.3-202.

11.3.3.4.1 Estimated Individual Doses

Dose rates to individuals are calculated for airborne decay and deposition, inhalation, and ingestion of milk (cow and goat), meat and vegetables. Dose from plume and ground deposition are calculated as affecting all age groups equally.

Table 11.3-201 contains GASPAR II input data for dose rate calculations. Information regarding the locations for the nearest resident, meat animal, milk animal, garden, and the dose evaluation periphery and power block area circle are described in Section 2.3. Table 11.3-203 contains total organ dose rates based on age group. Table 11.3-204 contains total air dose at each special location.

The total site doses due to liquid and gaseous effluents from Unit 1 and Units 2 and 3 would be well within the regulatory limits of 40 CFR Part 190, as shown in Table 11.3-206. The values in this table for Unit 1 are representative based on review of the Unit 1 annual radiological operating reports (References 202 through 206).

11.3.3.4.2 Estimated Population Dose

The population dose analysis performed to determine offsite dose from gaseous effluents is based upon the AP1000 generic site parameters included in DCD Chapter 11 and Tables 11.3-1, 11.3-2, and 11.3-4 and population data in Table 11.3-202. The population dose is shown in Table 11.3-205.

Table 11.3-205 shows that the total body and thyroid population doses per unit are approximately 2.7 and 6.4 person-rem per unit, respectively.

11.3.3.4.3 Gaseous Radwaste Cost-Benefit Analysis Methodology

STD COL 11.3-1 The guidance for performing cost-benefit analysis for the gaseous radwaste system is similar to that used and described for the liquid radwaste system in Section 11.2. The gaseous radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for gaseous radwaste treatment system augments is the Steam Generator Flash Tank Vent to Main Condenser at $6,320 per year, which yields a threshold value of 6.32 person-rem total body or thyroid from gaseous effluents.

For AP1000 sites with population dose estimates less than 6.32 person-rem total body or thyroid dose from gaseous effluents, no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I, Section II.D.

11.3.3.4.4 Gaseous Radwaste Cost-Benefit Analysis

VCS COL 11.3-1 As discussed in Subsection 11.3.3.4.3, the lowest cost gaseous radwaste system augment is $6,320. Assuming 100 percent efficiency of this augment, the minimum possible cost per person-rem is determined by dividing the cost of the augment by the population dose. This is $2,340 per person-rem total body ($6,320/2.7 person-rem) and $988 per person-rem thyroid ($6,320/6.4 person-rem thyroid). While the costs per person-rem total body reduction exceed the $1,000 per person-rem criterion, the costs per person-rem thyroid dose are below the $1,000 per person-rem and further evaluation is required.

Since the estimated thyroid dose of 6.4 person-rem exceeds the 6.32 person-rem threshold value, those system augments listed in Regulatory Guide 1.110 with a Total Annual Cost less than $6,400 are evaluated to determine if they would be cost beneficial. The only system augment with a Total Annual Cost less than $6,400 is the lowest-cost option for gaseous radwaste treatment system augments, the Steam Generator Flash Tank Vent to Main Condenser. It is noted that this augment would not mitigate the dose contribution from noble gases. Of the 6.4 person-rem thyroid dose given in FSAR Section 11.3.3.4.2, 1.2 person-rem is due to noble gases. Assuming this system augment completely eliminates the dose of the remaining 5.2 person-rem thyroid due to isotopes other than noble gases, the cost of the thyroid dose reduction would be $6,320/5.2 person-rem thyroid, or $1,215 per person-rem thyroid. This cost per person-rem reduction exceeds the $1,000 per person-rem criterion prescribed in Appendix I to 10 CFR Part 50 and this system augment is therefore not cost beneficial.

Due to the low VCSNS population doses, there are no other system augments from those listed in Regulatory Guide 1.110 which would be considered cost beneficial.

11.3.3.6 Quality Assurance

STD SUP 11.3-1 Add the following to the end of DCD Subsection 11.3.3.6:

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation, and testing provisions of the gaseous radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.3.5 COMBINED LICENSE INFORMATION

11.3.5.1 Cost Benefit Analysis of Population Doses

STD COL 11.3-1 This COL Item is addressed in Subsection 11.3.3.4.3.

VCS COL 11.3-1 This COL Item is addressed in Subsections 11.3.3.4, 11.3.3.4.1, 11.3.3.4.2, and 11.3.3.4.4.

VCS COL 11.5-3 This COL Item is addressed in Subsection 11.3.3.4.

11.3.6 REFERENCES

201. Deleted

202. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station, for the Operating Period January 1, 2005 – December 31, 2005; April 2006.

203. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station, for the Operating Period January 1, 2003 – December 31, 2003; April 2004.

204. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station for the Operating Period January 1, 2004 – December 31, 2004; April 2005.

205. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station, for the Operating Period January 1, 2006 – December 31, 2006; April 2007.

206. Annual Effluent and Waste Disposal Report, Virgil C. Summer Nuclear Station, for the Operating Period January 1, 2007 – December 31, 2007; April 2008.

VCS COL 11.3-1

VCS COL 11.5-3

Table 11.3-201
GASPAR II Input[a]

Input Parameter	Value
Number of Source Terms	1
Read Met data from XOQDOQ-generated file	Selected
Distance from site to NE Corner of the US	1129 mi
Source Term	DCD Table 11.3-3
Population Data	Table 11.3-202
Fraction of the year leafy vegetables are grown	0.583
Fraction of the year milk cows are on pasture	0.75
Fraction of maximally exposed individual's vegetable intake from own garden	0.76
Fraction of milk-cow feed intake from pasture while on pasture	1
Fraction of the year goats are on pasture	0.83
Fraction of goat feed intake from pasture while on pasture	1
Fraction of the year beef cattle are on pasture	0.75
Fraction of beef-cattle feed intake from pasture while on pasture	1
Total Production Rate for the 50-mile area	
-Vegetables	8.66E+07 kg per year
-Milk	6.78E+07 liters per year
-Meat	9.15E+08 kg per year
Special Location Data	Section 2.3

a) Input parameters not specified use default GASPAR II values.

VCS COL 11.3-1
VCS COL 11.5-3

Table 11.3-202
Population Input for Population Dose Rates

	Distance (mi)									
Direction	1	2	3	4	5	10	20	30	40	50
N	0	0	0	0	10	346	873	5086	9609	56103
NNE	0	0	0	10	73	491	651	9504	14976	214038
NE	0	0	115	25	83	155	2060	3485	12585	77448
ENE	0	51	0	19	0	793	12225	1477	2634	19934
E	0	19	147	0	0	915	4637	8552	31951	43930
ESE	117	4	12	133	22	321	6820	106337	19823	10765
SE	0	29	57	0	156	394	48768	343866	58718	14087
SSE	0	0	0	0	0	3242	118703	210614	59842	16596
S	0	6	0	117	102	3020	35109	57548	29388	15465
SSW	0	0	12	44	92	3907	18332	32814	14385	15326
SW	0	0	47	9	57	1576	5334	4697	10615	26568
WSW	0	36	17	0	168	1000	6268	3601	5059	9065
W	0	0	9	24	62	701	23548	2522	7991	79542
WNW	0	18	0	6	54	865	2800	4997	33560	44593
NW	0	0	0	9	0	639	721	4774	5727	20941
NNW	35	0	9	225	23	415	434	2812	23936	15182
Total	152	163	425	621	902	18780	287283	802686	340799	679583
									Grand Total	2131394

VCS COL 11.3-1
VCS COL 11.5-3

Table 11.3-203 (Sheet 1 of 2) Annual Individual Doses from Gaseous Effluents per Unit (mrem)

Pathway	Age Group	Nearest Site Boundary (0.50 mi SE, 0.50 mi ENE/NE)[a]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume	All	4.4E-01	4.4E-01	4.4E-01	4.4E-01	4.4E-01	4.4E-01	4.7E-01	2.2E+00
Ground	All	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.4E-01	1.6E-01
Inhalation	Adult	4.8E-02	4.8E-02	7.6E-03	4.9E-02	5.0E-02	4.5E-01	6.2E-02	4.6E-02
	Teen	4.8E-02	4.9E-02	9.1E-03	5.0E-02	5.1E-02	5.6E-01	7.1E-02	4.7E-02
	Child	4.3E-02	4.2E-02	1.1E-02	4.5E-02	4.6E-02	6.6E-01	6.1E-02	4.1E-02
	Infant	2.5E-02	2.4E-02	5.6E-03	2.7E-02	2.7E-02	5.9E-01	3.8E-02	2.4E-02

Pathway	Age Group	Nearest Residence (1.68 mi SE, 1.3 mi ENE)[a]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume	All	5.8E-02	5.8E-02	5.8E-02	5.8E-02	5.8E-02	5.8E-02	6.2E-02	3.1E-01
Ground	All	2.9E-02	2.9E-02	2.9E-02	2.9E-02	2.9E-02	2.9E-02	2.9E-02	3.4E-02
Inhalation	Adult	7.1E-03	7.2E-03	1.1E-03	7.3E-03	7.4E-03	6.4E-02	9.1E-03	6.9E-03
	Teen	7.2E-03	7.3E-03	1.3E-03	7.5E-03	7.7E-03	8.0E-02	1.0E-02	7.0E-03
	Child	6.4E-03	6.3E-03	1.5E-03	6.7E-03	6.8E-03	9.4E-02	8.9E-03	6.2E-03
	Infant	3.7E-03	3.6E-03	7.8E-04	4.0E-03	4.0E-03	8.4E-02	5.5E-03	3.6E-03

Pathway	Age Group	Nearest Garden (1.68 mi SE, 1.3 mi ENE)[a]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Vegetable	Adult	5.4E-02	5.5E-02	2.8E-01	5.4E-02	5.1E-02	6.4E-01	4.7E-02	4.6E-02
	Teen	8.0E-02	8.1E-02	4.3E-01	8.3E-02	7.9E-02	8.6E-01	7.2E-02	7.1E-02
	Child	1.7E-01	1.7E-01	1.0E+00	1.8E-01	1.7E-01	1.7E+00	1.6E-01	1.6E-01

Pathway	Age Group	Nearest Meat Animal (1.68 mi SE, 1.3 mi ENE)[a]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Meat	Adult	1.6E-02	2.0E-02	6.9E-02	1.6E-02	1.6E-02	3.9E-02	1.5E-02	1.5E-02
	Teen	1.3E-02	1.5E-02	5.9E-02	1.3E-02	1.3E-02	2.9E-02	1.3E-02	1.3E-02
	Child	2.3E-02	2.4E-02	1.1E-01	2.4E-02	2.3E-02	4.8E-02	2.3E-02	2.3E-02

VCS COL 11.3-1
VCS COL 11.5-3

Table 11.3-203 (Sheet 2 of 2)
Annual Individual Doses from Gaseous Effluents per Unit (mrem)

Pathway	Age Group	Nearest Milk Cow (1.68 mi SE, 1.3 mi ENE)[(a)]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Milk	Adult	2.4E-02	2.0E-02	8.3E-02	2.6E-02	2.4E-02	6.7E-01	2.0E-02	1.9E-02
	Teen	3.8E-02	3.4E-02	1.5E-01	4.5E-02	4.2E-02	1.1E+00	3.4E-02	3.3E-02
	Child	8.2E-02	7.7E-02	3.7E-01	9.6E-02	9.1E-02	2.1E+00	7.7E-02	7.6E-02
	Infant	1.6E-01	1.5E-01	7.1E-01	2.0E-01	1.8E-01	5.2E+00	1.6E-01	1.5E-01

Pathway	Age Group	Nearest Milk Goat (1.68 mi SE, 1.3 mi ENE)[(a)]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Milk	Adult	3.6E-02	2.5E-02	9.6E-02	4.2E-02	3.3E-02	9.0E-01	2.5E-02	2.3E-02
	Teen	5.2E-02	4.1E-02	1.7E-01	7.0E-02	5.6E-02	1.4E+00	4.2E-02	3.9E-02
	Child	9.8E-02	8.6E-02	4.2E-01	1.4E-01	1.1E-01	2.8E+00	9.0E-02	8.5E-02
	Infant	1.8E-01	1.7E-01	7.8E-01	2.7E-01	2.1E-01	6.8E+00	1.8E-01	1.7E-01

Pathway	Age Group	Maximally Exposed Individual (1.68 mi SE, 1.3 mi ENE)[(a)]							
		Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
All	Adult	2.0E-01	1.9E-01	5.3E-01	2.1E-01	1.9E-01	1.7E+00	1.9E-01	4.4E-01
	Teen	2.4E-01	2.3E-01	7.5E-01	2.6E-01	2.4E-01	2.5E+00	2.3E-01	4.8E-01
	Child	3.9E-01	3.7E-01	1.6E+00	4.4E-01	4.0E-01	4.7E+00	3.8E-01	6.2E-01
	Infant	2.7E-01	2.6E-01	8.7E-01	3.6E-01	3.0E-01	7.0E+00	2.7E-01	5.2E-01

a) The distances and directions are for the maximum applicable X/Q and the maximum D/Q respectively.

VCS COL 11.3-1
VCS COL 11.5-3

Table 11.3-204
Comparison of Gaseous Effluent Doses to 10 CFR 50 Appendix I Limits

Type of Dose	Location	Annual Dose per Unit	
		Unit 2 or 3	Limit
Gaseous Effluent	EAB		
Gamma Air (mrad)		0.71	10
Beta Air (mrad)		3.0	20
Total Body (mrem)		0.58	5
Skin (mrem)		2.4	15
Iodines and Particulates in Gaseous Effluent—Thyroid (mrem)	MEI[(a)]	7.0	15

a) Maximum X/Q occurs at 1.68 mi SE while the maximum D/Q occurs at 1.3 mi ENE. The MEI dose is based on this combination of X/Q and D/Q.

VCS COL 11.3-1
VCS COL 11.5-3

Table 11.3-205
Annual Population Doses from Gaseous Effluents (per Unit)

Pathway	Dose (person-rem/yr)							
	Total Body	GI-Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.4E+00	1.1E+01
Ground	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.1E-01	2.5E-01
Inhalation	3.1E-01	3.1E-01	3.5E-02	3.1E-01	3.2E-01	2.4E+00	3.7E-01	3.0E-01
Vegetable	2.9E-01	2.9E-01	1.3E+00	2.9E-01	2.8E-01	3.0E-01	2.8E-01	2.8E-01
Cow Milk	1.5E-01	1.4E-01	6.1E-01	1.6E-01	1.5E-01	1.4E+00	1.4E-01	1.4E-01
Meat	6.2E-01	6.5E-01	2.8E+00	6.2E-01	6.1E-01	9.1E-01	6.1E-01	6.1E-01
Total	2.7E+00	2.8E+00	6.1E+00	2.8E+00	2.7E+00	6.4E+00	3.0E+00	1.3E+01

Table 11.3-206
Comparison of Maximally Exposed Individual Doses with 40 CFR Part 190 Criteria

	Dose (mrem/yr)					
	Units 2 and 3			Unit 1[c]	Site Total	Regulatory Limit
	Liq[a]	Gas[b]	Total			
Total Body	0.28	0.78	1.1	1.2	2.2	25
Thyroid	0.38	14	14	0.043	14	75
Other Organ - Bone	0.23	3.2	3.5	0.043	3.5	25

a) Doses from Table 11.2-203 are doubled for two units.
b) Maximum doses (by age group) from Table 11.3-203 are doubled for two units.
c) Unit 1 doses are based on annual effluent reports.

11.4 SOLID WASTE MANAGEMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.4.5 QUALITY ASSURANCE

Add the following to the end of DCD Subsection 11.4.5:

STD SUP 11.4-1 Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the solid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.4.6 COMBINED LICENSE INFORMATION FOR SOLID WASTE MANAGEMENT SYSTEM PROCESS CONTROL PROGRAM

Add the following information to the end of DCD Subsection 11.4.6.

This COL Item is addressed below.

STD COL 11.4-1 A Process Control Program (PCP) is developed and implemented in accordance with the recommendations and guidance of NEI 07-10A (Reference 201). The PCP describes the administrative and operational controls used for the solidification of liquid or wet solid waste and the dewatering of wet solid waste. Its purpose is to provide the necessary controls such that the final disposal waste product meets applicable federal regulations (10 CFR Parts 20, 50, 61, 71, and 49 CFR Part 173), state regulations, and disposal site waste form requirements for burial at a low level waste (LLW) disposal site that is licensed in accordance with 10 CFR Part 61.

Waste processing (solidification or dewatering) equipment and services may be provided by the plant or by third-party vendors. Each process used meets the applicable requirements of the PCP.

No additional onsite radwaste storage is required beyond that described in the DCD.

Table 13.4-201 provides milestones for PCP implementation.

11.4.6.1 Procedures

STD SUP 11.4-1 Operating procedures specify the processes to be followed to ship waste that complies with the waste acceptance criteria (WAC) of the disposal site, 10 CFR 61.55 and 61.56, and the requirements of third party waste processors.

Each waste stream process is controlled by procedures that specify the process for packaging, shipment, material properties, destination (for disposal or further processing), testing to verify compliance, the process to address non-conforming materials, and required documentation.

Where materials are to be disposed of as non-radioactive waste (as described in DCD Subsection 11.4.2.3.3), final measurements of each package are performed to verify there has not been an accumulation of licensed material resulting from a buildup of multiple, non-detectable quantities. These measurements are obtained using sensitive scintillation detectors, or instruments of equal sensitivity, in a low-background area.

Procedures document maintenance activities, spill abatement, upset condition recovery, and training.

Procedures document the periodic review and revision, as necessary, of the PCP based on changes to the disposal site, WAC regulations, and third party PCPs.

11.4.6.2 Third Party Vendors

Third party equipment suppliers and/or waste processors are required to supply approved PCPs. Third party vendor PCPs describe compliance with Regulatory Guide 1.143, Generic Letter 80-09, and Generic Letter 81-39. Third party vendor PCPs are referenced appropriately in the plant PCP before commencement of waste processing.

11.4.7 REFERENCES

201. NEI 07-10A, “Generic FSAR Template Guidance for Process Control Program (PCP),” Revision 0, March 2009 (ML091460627).

11.5 RADIATION MONITORING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.5.1.2 Power Generation Design Basis

Revise the fourth bullet in DCD Subsection 11.5.1.2 as follows.

STD COL 11.5-2
- Data collection and data storage to support compliance reporting for the applicable NRC requirements and guidelines, such as General Design Criterion 64 and Regulatory Guide 1.21 and Regulatory Guide 4.15, Revision 1.

11.5.2.4 Inservice Inspection, Calibration, and Maintenance

Add the following information at the end of DCD Subsection 11.5.2.4:

STD COL 11.5-2 Daily checks of effluent monitoring system operability are made by observing channel behavior. Detector response is routinely observed with a remotely-positioned check source in accordance with plant procedures. Instrument background count rate is also observed to determine proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely-positioned check source can have its response checked with a portable check source. A record is maintained showing the background radiation level and the detector response.

Calibration of the continuous radiation monitors is done with commercial radionuclide standards that have been standardized using a measurement system traceable to the National Institute of Standards and Technology.

11.5.3 EFFLUENT MONITORING AND SAMPLING

Add the following information at the end of DCD Subsection 11.5.3.

VCS COL 11.5-2 SCE&G is extending the existing VCSNS Unit 1 program for quality assurance of radiological effluent and environmental monitoring that is based on Regulatory Guide 4.15, Revision 1, to apply to Units 2 and 3. Regulatory Guide 4.15, Revision 1, is a proven methodology for quality assurance of radiological effluent and environmental monitoring programs that is acceptable to the NRC staff as a method for demonstrating compliance with applicable requirements of 10 CFR Parts 20, 50, 52, 61, and 72. Use of Revision 2 of Regulatory Guide 4.15 would necessitate conducting two separate programs involving the use of common staff, facilities and equipment, which will create an undue burden and may lead to an increased possibility for human error. Therefore, SCE&G commits to use Regulatory Guide 4.15, Revision 1, methodology for Units 2 and 3 for optimal consistency, efficiency and practicality.

11.5.4 PROCESS AND AIRBORNE MONITORING AND SAMPLING

STD COL 11.5-2 Add the following information at the end of the first paragraph in DCD Subsection 11.5.4.

The sampling program for liquid and gaseous effluents will conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1AA).

Add the following information at the end of DCD Subsection 11.5.4:

11.5.4.1 Effluent Sampling

STD COL 11.5-2 Effluent sampling of potential radioactive liquid and gaseous effluent paths is conducted on a periodic basis to verify effluent processing meets the discharge limits to offsite areas. The effluent sampling program provides the information for the effluent measuring and reporting required by 10 CFR 50.36a and 10 CFR Part 20 and implemented through the Offsite Dose Calculation Manual (ODCM) and plant procedures. The frequency of the periodic sampling and analyses described herein are nominal and may be increased as permitted by procedure. Tables 11.5-201 and 11.5-202 summarize the sample and analysis schedules and sensitivities, respectively. The information contained in Tables 11.5-201 and 11.5-202 are derived from Regulatory Guide 1.21.

Laboratory isotopic analyses are performed on continuous and batch effluent releases in accordance with the ODCM. Results of these analyses are compiled and appropriate portions are utilized to produce the Radioactive Effluent Release Report.

11.5.4.2 Representative Sampling

Representative samples are obtained from well-mixed streams or volumes of effluent liquid through the use of proper sampling equipment, proper location of sampling points, and the development and use of sampling procedures. The recommendations of ANSI N42.18 (Reference 203) are considered for the selection of instrumentation specific to the continuous monitoring of radioactivity in liquid effluents.

Sampling of effluent liquids is consistent with guidance in Regulatory Guide 1.21. When practical, effluent releases are batch-controlled, and prior to sampling, large volumes of liquid waste are mixed, in as short a time span as practicable, so that solid particulates are uniformly distributed in the liquid volume. Sampling and analysis is performed, and release conditions set, before release. Sample points are located to minimize flow disturbance due to fittings and other characteristics of equipment and components. Sample lines are flushed consistent with plant procedures to remove sediment deposits.

Representative sampling of process effluents is attained through sample and monitor locations and methods and criteria detailed in plant procedures.

Composite sampling is employed to analyze for hard to measure radionuclides and to monitor effluent streams that normally are not expected to contain significant amounts of radioactive contamination. Composite liquid samples are collected in proportion to the volume of each batch of effluent release. The composite is thoroughly mixed prior to analysis. Collection periods for composites are as short as practicable and periodic checks are performed to identify changes in composite samples. When grab samples are collected instead of composite samples, the time of the sample, location, and frequency are considered to provide a representative sample of the radioactive materials.

The pressure head of the fluid, if available, is used for taking samples. If sufficient pressure head is not available to take samples, then sample pumps are used to draw the sample from the process fluid to the detector panels and back to the process.

Testing and obtaining representative samples using the radiation monitors described in DCD Subsection 11.5 will be performed in accordance with ANSI N13.1 (Reference 201).

For obtaining representative samples in unfiltered ducts, isokinetic probes are tested and used in accordance with ANSI N13.1 (Reference 201).

Analytical Procedures

Typically, samples of process and effluent gases and liquids are analyzed in the station laboratory or by an outside laboratory via the following techniques:

- Gross alpha/beta counting
- Gamma spectrometry
- Liquid scintillation counting

"Available" instrumentation and counting techniques change as other instruments and techniques become available. For this reason, the frequency of sampling and the analysis of samples are generalized in this subsection.

Gross alpha/beta analysis may be performed directly on unprocessed samples (e.g., air filters) or on processed samples (e.g., evaporated liquid samples). Sample volume, counting geometry, and counting time are chosen to match measurement capability with sample activity. Correction factors for sample-detector geometry, self-absorption and counter resolving time are applied to provide the required accuracy.

Liquid effluent samples are prepared for alpha/beta counting by evaporation onto steel planchets. Gamma analysis may be done on any type of sample (gas, solid or liquid) in a gamma spectrometer.

Tritiated water vapor samples are collected by condensation or adsorption, and the resultant liquid is analyzed by liquid scintillation counting techniques.

Radiochemical separations are used for the routine analysis of Sr-89 and Sr-90.

Liquid samples are collected in polyethylene bottles to minimize absorption of nuclides onto container walls.

11.5.6.5 Quality Assurance

Add the following information at the end of DCD Subsection 11.5.6.5.

STD COL 11.5-2 The sampling program and the associated monitors conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1AA).

11.5.8 COMBINED LICENSE INFORMATION

STD COL 11.5-1 An Offsite Dose Calculation Manual (ODCM) is developed and implemented in accordance with the recommendations and guidance of NEI 07-09A (Reference 202). The ODCM contains the methodology and parameters used for calculating doses resulting from liquid and gaseous effluents. The ODCM addresses operational setpoints, including planned discharge rates, for radiation monitors and monitoring programs (process and effluent monitoring and environmental monitoring) for the control and assessment of the release of radioactive material to the environment. The ODCM provides the limitations on operation of the radwaste systems, including functional capability of monitoring instruments, concentrations of effluents, sampling, analysis, 10 CFR Part 50, Appendix I dose and dose commitments, and reporting. The ODCM will be finalized prior to fuel load with site-specific information.

Table 13.4-201 provides milestones for ODCM implementation.

STD COL 11.5-2 This COL Item is addressed in Subsections 11.5.1.2, 11.5.2.4, 11.5.4, 11.5.4.1, 11.5.4.2, and 11.5.6.5.

VCS COL 11.5-2 This COL Item is addressed in Subsection 11.5.3.

VCS COL 11.5-3 This COL Item is addressed in Subsection 11.2.3.5 and 11.3.3.4 for liquid and gaseous effluents, respectively.

Add the following subsection after DCD Subsection 11.5.8.

11.5.9 REFERENCES

201. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

202. NEI 07-09A, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," Revision 0, March 2009 (ML091050234).

203. ANSI N42.18-2004, "Specification and Performance of On-Site Instrumentation for Continuous Monitoring Radioactivity in Effluents."

STD COL 11.5-2

Table 11.5-201 (Sheet 1 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Gaseous	Continuous Release	A sample is taken within one month of initial criticality, and at least weekly thereafter to determine the identity and quantity for principal nuclides being released. A similar analysis of samples is performed following each refueling, process change, or other occurrence that could alter the mixture of radionuclides. When continuous monitoring shows an unexplained variance from an established norm. Monthly for tritium.
	Batch Release	Prior to release to determine the identity and quantity of the principal radionuclides (including tritium).
	Filters (particulates)	Weekly. Quarterly for Sr-89 and Sr-90. Monthly for gross alpha.

STD COL 11.5-2

Table 11.5-201 (Sheet 2 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Liquid	Continuous Releases	Weekly for principal gamma-emitting radionuclides.
		Monthly, a composite sample for tritium and gross alpha.
		Monthly, a representative sample for dissolved and entrained fission and activation gases.
		Quarterly, a composite sample for Sr-89, Sr-90, and Fe-55.
	Batch Releases	Prior to release for principal gamma-emitting radionuclides.
		Monthly, a composite sample for tritium and gross alpha.
		Monthly, a representative sample from at least one representative batch for dissolved and entrained fission and activation gases.
		Quarterly, a composite sample for Sr-89, Sr-90 and Fe-55.

STD COL 11.5-2

Table 11.5-202
Minimum Sensitivities

Stream	Nuclide	Sensitivity
Gaseous	Fission & Activation Gases	1.0E-04 μCi/cc
	Tritium	1.0E-06 μCi/cc
	Iodines & Particulates	Sufficient to permit measurement of a small fraction of the activity that would result in annual exposures of 15 mrem to thyroid for iodines, and 15 mrem to any organ for particulates, to an individual in an unrestricted area.
	Gross Radioactivity	Sufficient to permit measurement of a small fraction of the activity that would result in annual air dose of 1) 10 mrad due to gamma, and 2) 20 mrad of beta at any location near ground level at or beyond the site boundary.
Liquid	Gross Radioactivity	1.0E-07 μCi/ml
	Gamma-emitters	5.0E-07 μCi/ml
	Dissolved & Entrained Gases	1.0E-05 μCi/ml
	Gross Alpha	1.0E-07 μCi/ml
	Tritium	1.0E-05 μCi/ml
	Sr-89 & Sr-90	5.0E-08 μCi/ml
	Fe-55	1.0E-06 μCi/ml

CHAPTER 12
RADIATION PROTECTION

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CHAPTER 12
RADIATION PROTECTION

12.1 ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY ACHIEVABLE (ALARA)

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 12.1-1 This section incorporates by reference NEI 07-08A, Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA), Revision 0. See Table 1.6-201. ALARA practices are developed in a phased milestone approach as part of the procedures necessary to support the Radiation Protection Program. Table 13.4-201 describes the major milestones for ALARA procedures development and implementation.

Revise the last sentence of NEI 07-08A Subsection 12.1.2 to read:

STD COL 12.1-1 ALARA procedures are established, implemented, maintained and reviewed consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description, which is discussed in Section 17.5.

Add the following information at the end of DCD Subsection 12.1.2.4:

12.1.2.4.3 Equipment Layout

STD SUP 12.1-1 A video record of the equipment layout in areas where radiation fields are expected to be high following operations may be used to assist in ALARA planning and to facilitate decommissioning.

12.1.3 COMBINED LICENSE INFORMATION

STD COL 12.1-1 This COL item is addressed in NEI 07-08A and Appendix 12AA.

12.2 RADIATION SOURCES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.2.1.1.10 Miscellaneous Sources

Add the following information at the end of DCD Subsection 12.2.1.1.10:

STD COL 12.2-1 Licensed sources containing byproduct, source, and special nuclear material that warrant shielding design consideration meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50, and 70.

There are byproduct and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively.

These sources include but are not limited to:

- Sources in field monitoring equipment.
- Sources in radiation monitors to maintain a threshold sensitivity.
- Sources used for radiographic operations.
- Depleted uranium slabs used to determine beta response and correction factors for portable monitoring instrumentation.
- Sources used to calibrate and response check field monitoring equipment (portable and fixed).
- Liquid standards and liquids or gases used to calibrate and verify calibration of laboratory counting and analyzing equipment.
- Radioactive waste generated by the use of radioactive sources.

Specific details of these sources are maintained in a database on-site following procurement. This database, at a minimum, contains the following information:

- Isotopic composition
- Location in the plant
- Source strength

- Source geometry

Written procedures are established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. These procedures are developed in accordance with the radiation protection program to comply with 10 CFR Parts 19 and 20. A supplementary warning symbol is used in the presence of large sources of ionizing radiation consistent with the guidance in Regulatory Issue Summary (RIS) 2007-03.

Sources maintained on-site for instrument calibration purposes are shielded while in storage to keep personnel exposure ALARA. Sources used to service or calibrate plant instrumentation are also routinely brought on-site by contractors. Radiography is performed by the licensed utility group or licensed contractors. These sources are maintained and used in accordance with the provisions of the utility group's or contractor’s license. Additional requirements and restrictions may apply depending on the type of source, use, and intended location of use. If the utility group or contractor source must be stored on-site, designated plant personnel must approve the storage location, and identify appropriate measures for maintaining security and personnel protection.

During the period prior to the implementation of the Emergency Plan (in preparation for the initial fuel loading following the 52.103(g) finding), no specific materials related emergency plan will be necessary because:

a) No byproduct material will be received, possessed, or used in a physical form that is "in unsealed form, on foils or plated sources, or sealed in glass," that exceeds the quantities in Schedule C in 10 CFR 30.72, and

b) No 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during this period.

The following radioactive sources will be used for the Radiation Monitoring System and laboratory/portable monitoring instrumentation:[1]

Radioactive Licensee Material (Element and Mass Number)[1]	Chemical and/or Physical Form[1]	Maximum Quantity That Licensee May Possess at Any One Time[1]
• Any byproduct material with atomic numbers 1 through 93 inclusive	Sealed Sources[2]	No single source to exceed 100 millicuries 5 Curies total
• Americium-241	Sealed Sources[2]	No single source to exceed 300 millicuries 500 millicuries total

Notes:

1. This information remains in effect between the issuance of the COL and the Commission's 52.103(g) finding for each unit, and will be designated historical information after that time.
2. Includes calibration and reference sources.

12.2.3 COMBINED LICENSE INFORMATION

STD COL 12.2-1 This COL item is addressed in Subsection 12.2.1.1.10.

12.3 RADIATION PROTECTION DESIGN FEATURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.3.1 FACILITY DESIGN FEATURES

12.3.1.2 Radiation Zoning and Access Control

Add the following information at the end of the second paragraph of DCD Subsection 12.3.1.2.

VCS DEP 18.8-1

Figure 12.3-201, Figure 12.3-202, and Figure 12.3-203 replace DCD Figure 12.3-1 (Sheet 11), DCD Figure 12.3-2 (Sheet 11), and DCD Figure 12.3-3 (Sheet 11), respectively, to reflect the relocation of the Operations Support Center.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Add the following text to the end of DCD Subsection 12.3.4.

STD COL 12.3-2

Procedures detail the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in Regulatory Guides 1.21-Appendix A, 8.2, 8.8, and 8.10. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Appendix 12AA.

Surveillance requirements are determined by the functional manager in charge of radiation protection based on actual or potential radiological conditions encountered by personnel and the need to identify and control radiation, contamination, and airborne radioactivity. These requirements are consistent with the operational philosophy in Regulatory Guide 8.10. Frequency of scheduled surveillance may be altered by permission of the functional manager in charge of radiation protection or their designee. Radiation Protection periodically provides cognizant personnel with survey data that identifies radiation exposure gradients in area resulting from identified components. This data includes recent reports, with survey data, location and component information.

The following are typical criteria for frequencies and types of surveys:

Job Coverage Surveys

- Radiation, contamination, and/or airborne surveys are performed and documented to support job coverage.

- Radiation surveys are sufficient in detail for Radiation Protection to assess the radiological hazards associated with the work area and the intended/specified work scope.

- Surveys are performed commensurate with radiological hazard, nature and location of work being conducted.

- Job coverage activities may require surveys to be conducted on a daily basis where conditions are likely to change.

Radiation Surveys

- Radiation surveys are performed at least monthly in any radiological controlled area (RCA) where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.

- Radiation surveys are performed prior to or during entry into known or suspected high radiation areas for which up to date survey data does not exist.

- Radiation surveys are performed prior to work involving highly contaminated or activated materials or equipment.

- Radiation surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.

- Radiation surveys are performed to support movement of highly radioactive material.

- Neutron radiation surveys are performed when personnel may be exposed to neutron emitting sources.

Contamination Surveys

- Contamination surveys are performed at least monthly in any RCA where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.

- Contamination surveys are performed during initial entry into known or suspected contamination area(s) for which up to date survey data does not exist.

- Contamination surveys are performed at least daily at access points, change areas, and high traffic walkways in RCAs that contain contaminated areas. Area access points to a High Radiation Area or Very High Radiation Area are surveyed prior to or upon access by plant personnel or if access has occurred.

- Contamination surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.

- A routine surveillance is conducted in areas designated by the functional manager in charge of radiation protection or their designee likely to indicate alpha radioactivity. If alpha contamination is identified, frequency and scope of the routine surveillance is increased.

<u>Airborne Radioactivity Surveys</u>

- Airborne radioactivity surveys are performed during any work or operation in the RCA known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping, use of compressed air, using volatiles on contaminated material, waste processing, or insulation).

- Airborne radioactivity surveys are performed during a breach of a radioactive system, which contains or is suspected of containing significant levels of contamination.

- Airborne radioactivity surveys are performed during initial entry (and periodically thereafter) into any known or suspected airborne radioactivity area.

- Airborne radioactivity surveys are performed immediately following the discovery of a significant radioactive spill or spread of radioactive contamination, as determined by the functional manager in charge of radiation protection.

- Airborne radioactivity surveys are performed daily in occupied radiological controlled areas where the potential for airborne radioactivity exists, including containment.

- Airborne radioactivity surveys are performed any time respiratory protection devices, alternative tracking methods such as derived air concentration-hour (DAC-hr), and/or engineering controls are used to control internal exposure.

- Airborne radioactivity surveys are performed using continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalations of radioactivity by workers. Determination of air flow patterns are considered for locating air samplers.

- Airborne radioactivity surveys are performed prior to use and monthly during use on plant service air systems used to supply air for respiratory protection to verify the air is free of radioactivity.

- Tritium sampling is performed near the spent fuel pit when irradiated fuel is in the pit and other areas of the plant where primary system leaks occur and tritium is suspected.

Appropriate counting equipment is used based on the sample type and the suspected identity of the radionuclides for which the sample is being done. Survey results are documented, retrievable, and processed per site document control and records requirements consistent with Regulatory Guide 8.2. Completion of survey documentation includes the update of room/area posting maps and revising area or room postings and barricades as needed.

Air samples indicating activity levels greater than a procedure specified percentage of DAC are forwarded to the radiochemistry laboratory for isotopic analysis. Samples which cannot be analyzed on-site are forwarded to an off-site laboratory or a contractor for analysis; or, the DAC percentage may be hand calculated using appropriate values from 10 CFR Part 20, Appendix B.

The responsible radiation protection personnel review survey documentation to evaluate if surveys are appropriate and obtained when required, records are complete and accurate, and adverse trends are identified and addressed.

An in-plant radiation monitoring program maintains the capability to accurately determine the airborne iodine concentration in areas within the facility where personnel may be present under accident conditions. This program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment consistent with Regulatory Guides 1.21 (Appendix A) and 8.8. Training and personnel qualifications are discussed in Appendix 12AA.

A portable monitor system meeting the requirements of NUREG-0737, Item III.D.3.3, is available. The system uses a silver zeolite or charcoal iodine sample cartridge and a single-channel analyzer. The use of this portable monitor is incorporated in the emergency plan implementing procedures. The portable monitor is part of the in-plant radiation monitoring program. It is used to determine the airborne iodine concentration in areas where plant personnel may be present during an accident. Accident monitoring instrumentation complies with applicable parts of 10 CFR Part 50, Appendix A.

Sampling cartridges can be removed to a low background area for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.3.5.1 Administrative Controls for Radiological Protection

STD COL 12.3-1 This COL Item is addressed in Subsection 12.5.4 and Appendix 12AA.

12.3.5.2 Criteria and Methods for Radiological Protection

STD COL 12.3-2 This COL Item is addressed in Subsection 12.3.4.

12.3.5.3 Groundwater Monitoring Program

STD COL 12.3-3 This COL Item is addressed in Appendix 12AA.

12.3.5.4 Record of Operational Events of Interest for Decommissioning

STD COL 12.3-4 This COL Item is addressed in Appendix 12AA.

Security-Related Information — Withheld Under 10 CFR 2.390(d) **(See Part 9 of this COL Application)**

VCS DEP 18.8-1 **(Note: This figure replaces DCD Figure 12.3-1 Sheet 11 of 16. This replacement is necessary to support the alternate locations of the Technical Support Center and the Operations Support Center per Departure Number VCS DEP 18.8-1.)**

Figure 12.3-201
Radiation Zones, Normal Operations/Shutdown Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

VCS DEP 18.8-1 **(Note: This figure replaces DCD Figure 12.3-2 Sheet 11 of 15. This replacement is necessary to support the alternate locations of the Technical Support Center and the Operations Support Center per Departure Number VCS DEP 18.8-1.)**

Figure 12.3-202
Radiation Zones, Post-Accident Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

VCS DEP 18.8-1 **(Note: This figure replaces DCD Figure 12.3-3 Sheet 11 of 16. This replacement is necessary to support the alternate locations of the Technical Support Center and the Operations Support Center per Departure Number VCS DEP 18.8-1.)**

Figure 12.3-203
Radiological Access Controls, Normal Operations/Shutdown Annex Building, Elevation 100'-0" & 107'-2"

12.4 DOSE ASSESSMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following new subsections after DCD Subsection 12.4.1.8:

VCS SUP 12.4-1

12.4.1.9 Dose to Construction Workers

This section evaluates the potential radiological dose impacts to construction workers at VCSNS resulting from the operating unit(s). Construction workers at Units 2 and 3 may be exposed to direct radiation and gaseous radioactive effluents from Unit 1. Since a portion of the Unit 3 construction period overlaps operation of Unit 2, construction workers at Unit 3 may be exposed to direct radiation and gaseous radioactive effluents from Unit 2.

12.4.1.9.1 Site Layout

The VCSNS power block areas are shown on FSAR Figure 2.1-203. Separation will be provided such that construction activity for Unit 3 is outside the protected area for Units1 and 2 but inside the owner controlled area.

12.4.1.9.2 Radiation Sources

Workers constructing Units 2 and 3 may be exposed to direct radiation and gaseous radioactive effluents emanating from the routine operation of Unit 1. Construction workers at Unit 3 are not exposed to any radiation sources from Unit 2 until it becomes operational. Workers constructing Unit 3 may be exposed to direct radiation and gaseous radioactive effluents emanating from the routine operation of Unit 2. Radiation doses to construction workers are from direct radiation and from airborne effluents from Unit 2 and from background radiation.

Direct radiation from the Unit 1 containment and other plant buildings is negligible. Routine operational thermo-luminescent dosimeter (TLD) measurements at the site boundary for Unit 1 show that the annual doses are comparable to the preoperational annual dose rates. For conservatism, the direct dose rate from Unit 1 in the construction area for Units 2 and 3 is assumed to be 1 mrem per year. Small quantities of monitored airborne effluents are normally released from the Unit 1 waste gas decay tank, reactor building purges, and oil incineration. The construction workers are assumed to be exposed to the gaseous and liquid doses from routine operation of Unit 1.

Routine operational TLD measurements at the old steam generator recycling facility show that the annual doses are comparable to the preoperational annual dose rates. Therefore, the direct dose from this facility is negligible.

For Unit 2, the radiation exposure at the site boundary is considered in DCD Section 12.4.2. As stated in that section, direct radiation from the containment and other plant buildings is negligible. Additionally, there is no contribution from refueling water since the refueling water is stored inside the containment instead of in an outside storage tank. For conservatism, the dose rate from Unit 2 in the construction area for Unit 3 is assumed to be 1 mrem per year.

For Unit 2, small quantities of monitored airborne effluents are normally released through the plant vent or the turbine building vent. The plant vent provides the release path for containment venting releases, auxiliary building ventilation releases, annex building releases, radwaste building releases, and gaseous radwaste system discharge. The turbine building vents provide the release path for the condenser air removal system, gland seal condenser exhaust and the turbine building ventilation releases. The ventilation system is described in DCD Section 9.4. The expected radiation sources (nuclides and activities) in the gaseous effluents are listed in DCD Table 11.3-3.

Exposure of Unit 3 construction workers to radioactive liquid effluents is evaluated for conservatism. Although the construction workers would not be exposed to the liquid exposure pathways at the Unit 3 construction site, it is conservatively assumed that the workers receive the same doses as the maximally exposed member of the public offsite.

While Unit 2 is operating and Unit 3 is under construction, workers may be exposed to liquid effluents from Unit 2 while performing Unit 3 liquid waste effluent discharge piping connections. However, this work will be performed by trained radiation workers, not general site construction workers. Hence, this activity is not considered a contributor to construction worker doses.

12.4.1.9.3 Construction Worker Dose Estimates

For liquid effluent doses from Unit 1, the determination of construction worker dose due to Unit 1 operation is assumed to be equal to the calculated liquid effluent dose for routine operation per the Unit 1 Off-Site Dose Calculation Manual (ODCM). For liquid effluent doses from Unit 2, the determination of construction worker dose due to Unit 2 operation depends on the liquid effluent release and the transport to worker location. The liquid dilution and transport to the maximally exposed person used the guidance in Regulatory Guide 1.113 for the average flow rate for the Broad River.

The construction worker doses are conservatively estimated using the following information:

- The workers receive the same doses as the maximally exposed member of the public offsite in accordance with Section 11.2.3.5
- A construction worker exposure time of 2000 hours per year

- A peak loading of 3600 construction workers per year for Unit 3 construction

The use of 2000 hours assumes that the worker works 40 hours per week for 50 weeks per year.

The methodology used to calculate the doses to construction workers from normal effluent releases complies with the guidance provided in Regulatory Guide 1.109. Construction worker doses were estimated by use of LADTAP II computer code (NUREG/CR-4013). The total effective dose equivalent (TEDE), which is the sum of the deep dose equivalent (DDE) and the committed effective dose equivalent (CEDE), was determined based on the LADTAP II results. The annual TEDE dose was corrected for the actual time the construction workers are onsite by multiplying by the ratio of hours worked per year to hours in a year.

For airborne doses, the determination of construction worker dose due to Unit 2 operation depends on the airborne effluent release and the atmospheric transport to the worker location. The atmospheric dispersion calculation used the guidance provided in Regulatory Guide 1.111, meteorological data for the years 2007 and 2008, and downwind distances to the construction worker locations. The XOQDOQ computer code (NUREG/CR-2919) was used to determine the χ/Q and D/Q values for the nearest location along the Unit 2 protected area fence in each direction as well as the nearest point of the Unit 3 construction area.

Construction worker doses are conservatively estimated using the following information:

- The estimated maximum dose rate for each pathway.
  - External exposure to contaminated ground.
  - External exposure to noble gas radionuclides in the airborne plume.
  - Inhalation of air.
- A construction worker exposure time of 2000 hours per year.
- A peak loading of 3600 construction workers per year for Unit 3 construction.

The use of 2000 hours assumes the worker works 40 hours per week for 50 weeks per year.

The methodology used to calculate the doses to construction workers from normal effluent releases complies with the guidance provided in Regulatory Guide 1.109. Construction worker doses were estimated by use of GASPAR computer code (NUREG/CR-4653). The TEDE, which is the sum of the DDE and the CEDE, was determined based on the GASPAR results. The annual TEDE dose was corrected for the actual time the construction workers are onsite by multiplying by a ratio of hours worked per year to hours in a year.

When adjusted for an occupancy time of 2000 hours per year, the direct, gaseous, and liquid doses from Unit 1 are 0.23, 0.27, and 0.0020 mrem TEDE, respectively, yielding a total annual dose of 0.50 mrem TEDE. The occupancy-adjusted direct, gaseous, and liquid doses from Unit 2 are 0.23, 0.48, and 0.067 mrem TEDE, respectively, resulting in a total of 0.77 mrem TEDE. Therefore, the total annual dose to the Unit 3 construction worker from Units 1 and 2 is 1.3 mrem TEDE.

12.4.1.9.4 Compliance with Dose Regulations

VCSNS Units 2 and 3 construction workers are, for the purposes of radiation protection, members of the general public. This means that the dose to the individual does not exceed 100 mrem per year, the limit for a member of the public. The construction workers do not deal with radiation sources.

Dose limits to the public are provided in 10 CFR 20.1301 and 10 CFR 20.1302. Because the construction workers are considered members of the public, the requirements of 10 CFR 20.1201 through 20.1204 do not apply.

The 10 CFR 20.1301 limits annual doses from licensed operations to individual members of the public to 100 mrem TEDE. In addition, the dose from external sources to unrestricted areas must be less than 2 mrem in any one hour. This applies to the public both outside and inside access controlled areas. The maximum dose rates are given in Table 12.4-201. For an occupational year, dose at the Unit 3 construction area is 1.3 mrem TEDE. This value is less than the limits specified for members of the public. Therefore, construction workers can be considered to be members of the general public and do not require radiation monitoring.

12.4.1.9.5 Collective Doses to VCSNS Units 2 and 3 Workers

The collective dose is the sum of all doses received by all workers. It is a measure of population risk. The total worker collective maximum annual dose is 4.6 person-rem. This estimate is based upon the construction workforce of 3600 and assumes 2000 hours per year for each worker.

12.4.1.9.6 Operating Unit Radiological Surveys

STD SUP 12.4-1 The operating unit conducts radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluents discharged to unrestricted and controlled areas in implementing 10 CFR 20.1302. These surveys demonstrate compliance with the dose limits of 10 CFR 20.1301 for construction workers.

VCS SUP 12.4-1

Table 12.4-201
Construction Worker Dose Comparison to 10 CFR 20.1301 Criteria

Type of Dose	Dose Limits[1] (TEDE)	Estimated Dose[2]
Annual total effective dose equivalent	100 mrem	1.3 mrem
Maximum dose in any hour	2 mrem	6.5 E-04 mrem

Notes:

1. 10 CFR 20.1301 criteria.
2. Estimated dose is at Unit 3 construction area. Total body dose calculated using the methodology in Regulatory Guide 1.109.

12.5 HEALTH PHYSICS FACILITIES DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.5.2.2 Facilities

Revise the first sentence of DCD Subsection 12.5.2.2 to read:

VCS DEP 18.8-1 The ALARA briefing room is located off the main corridor immediately beyond the main entry to the annex building.

12.5.3.2 Job Planning Facilities

Revise the last sentence of DCD Subsection 12.5.3.2 to read:

VCS DEP 18.8-1 The ALARA briefing room in the annex building is an example of such a facility where job planning and ALARA briefing and debriefing activities can take place.

12.5.4 CONTROLLING ACCESS AND STAY TIME

Add the following text to the end of DCD Subsection 12.5.4.

STD COL 12.3-1 A closed circuit television system may be installed in high radiation areas to allow remote monitoring of individuals entering high radiation areas by personnel qualified in radiation protection procedures.

12.5.5 COMBINED LICENSE INFORMATION

STD COL 12.5-1 This COL Item is addressed in Appendix 12AA.

Add the following Appendix after Section 12.5 of the DCD.

APPENDIX 12AA RADIATION PROTECTION PROGRAM DESCRIPTION

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

This appendix incorporates by reference NEI 07-03A, Generic FSAR Template Guidance for Radiation Protection Program Description. See Table 1.6-201. The numbering of NEI 07-03A is revised from 12.5# to 12AA.5# through the document, with the following revisions and additions as indicated by strikethroughs and underlines. Table 13.4-201 provides milestones for radiation protection program implementation.

Revise bullet number 3 of NEI 07-03A Section 12.5 as follows:

3. Prior to initial loading of fuel in the reactor, all of the radiation program functional areas described in Appendix 12AA~~Section12.5~~ will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, the position of radiation protection manager (as described in ~~s~~Section 13.1 ~~12.5.2.3~~) will be filled and at least one (1) radiation protection technician for each operating shift, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor.

Revise the first paragraph of NEI 07-03A Subsection 12.5.2 as follows:

Qualification and training criteria for site personnel are consistent with the guidance in Regulatory Guide 1.8 and are described in FSAR Chapter 13. Specific radiation protection responsibilities for key positions within the plant organization are described in Section 13.1 ~~below~~.

Subsections 12.5.2.1 through 12.5.2.5 of NEI 07-03A are not incorporated into Appendix 12AA.

Subsection 12.5.3.1 of NEI 07-03A is not incorporated into Appendix 12AA. Facilities are described in DCD Subsection 12.5.2.2.

Add the following text after the first paragraph of NEI 07-03A Subsection 12.5.3.3.

If circumstances arise in which NIOSH tested and certified respiratory equipment is not used, compliance with 10 CFR 20.1703(b) and 20.1705 is maintained.

The following headings (and associated material) in Subsection 12.5.4.2 of NEI 07-03A are described in DCD Subsection 12.5.3, and are therefore not incorporated into Appendix 12AA:

- Radwaste Handling
- Spent Fuel Handling
- Normal Operation
- Sampling

Add the following text after the second paragraph of NEI 07-03A Subsection 12.5.4.4.

STD COL 12.3-1 Table 12AA-201 identifies plant areas designated as Very High Radiation Areas (VHRAs), lists corresponding plant layout drawings showing the VHRA in DCD Section 12.3, specifies the condition under which the area is designated VHRA, identifies the primary source of the VHRA, and summarizes the frequency of access and reason for access. VHRAs are listed as Radiation Zone IX, which corresponds to a dose rate greater than 500 rad/hr.

In each of the VHRAs, with the exception of the Reactor Vessel Cavity and Delay-Bed / Guard-Bed Compartment, the primary radioactive source is transient (such as fuel passing through the transfer tube), removable (such as resin in the demineralizers), or can be relocated. When the primary source is removed, the dose rate in each of these areas will be less than Zone IX and, in effect, the area will no longer be a VHRA. With planning, the need for human entrance to a VHRA when the primary source is present can be largely or entirely avoided.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates:

- Sign(s) conspicuously posted stating GRAVE DANGER, VERY HIGH RADIATION AREA.
- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the functional manager in charge of radiation protection as described in Section 13.1.
- Plant Manager’s (or designee) approval required for entry.
- Radiation Protection personnel shall accompany person(s) making the entry. Radiation Protection personnel shall assess the radiation exposure conditions at the time of the entry.

A verification walk down will be performed with the purpose of verifying barriers to the Very High Radiation Areas in the final design of the facility are consistent with Regulatory Guide 8.38 guidance as part of the implementation of the Radiation Protection and ALARA programs on the schedule identified in Table 13.4-201.

Revise the third paragraph of NEI 07-03A Subsection 12.5.4.7 as follows.

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

As described in Sections 12.1, ~~12.5.1~~Appendix 12AA and ~~12.5.2~~ 13.1, management policy is established, and organizational responsibilities and authorities are assigned to implement an effective program for maintaining occupational radiation exposures ALARA. Procedures are established and implemented that are in accordance with 10 CFR 20.1101 and consistent with the guidance in Regulatory Guides 8.8 and 8.10. Examples of such procedures include the following:

Add the following text after the last bullet of NEI 07-03A Subsection 12.5.4.8.

STD COL 12.5-1

This subsection adopts NEI 08-08A (Reference 201), for a description of the operational and programmatic elements and controls that minimize contamination of the facility, site, and the environment, to meet the requirements of 10 CFR 20.1406.

Revise the first paragraph of Subsection 12.5.4.12 of NEI 07-03A to read:

STD COL 12.5-1

The radiation protection program and procedures are established, implemented, maintained, and reviewed consistent with the 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description described in Section 17.5.

Add the following Subsection to the information incorporated from NEI 07-03A.

STD COL 12.3-3

12AA.5.4.14 Groundwater Monitoring Program

A groundwater monitoring program beyond the normal radioactive effluent monitoring program is developed. If necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are (all directions based on plant standard):

- West of the auxiliary building in the area of the fuel transfer canal.

- West and south of the radwaste building.
- East of the auxiliary building rail bay and the radwaste building truck doors.

This subsection adopts NEI 08-08A (Reference 201) for the Groundwater Monitoring Program description.

Add the following Subsection to the information incorporated from NEI 07-03A.

STD COL 12.3-4 12AA.5.4.15 Record of Operational Events of Interest for Decommissioning

This subsection adopts NEI 08-08A (Reference 201) for discussion of record keeping practices important to decommissioning.

Revise the REFERENCES section of NEI 07-03A, Reference 8, to read as follows:

Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."~~4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."~~

Add the following reference to the NEI 07-03A REFERENCES.

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

STD COL 12.3-1

Table 12AA-201 (Sheet 1 of 2)
Very High Radiation Areas (VHRA)

Room Number	VHRA Location	DCD Figure 12.3-1, Sheet No.	Primary Source(s)	VHRA Conditional Notes	Frequency of Access to VHRA Areas While VHRA Conditions Exist
11105	Reactor Vessel Cavity	3, 4, 5	Neutron activation of the material in and around the cavity during reactor operations, such as the concrete shield walls and the reactor insulation	Note 1	None Required
12151	Spent Fuel Pool Cooling System / Liquid Radwaste System Demineralizer/ Filter room (Inside Wall)	3	Resin in vessels	Notes 6, 8	None Required
12153	Delay-Bed/ Guard-Bed Compartment	3	Activated carbon holding radioactive gases	Note 10	None Required
12371	Filter-Storage Area	6, 7	Spent filter cartridges	Notes 4, 6, 7	None required
12372	Resin Transfer Pump/ Valve Room	6	Spent resin in lines	Note 6	None required
12373	Spent-Resin Tank Room	6	Spent resin in tanks	Note 6	None Required
12374	Waste Disposal Container Area	6	Spent resin in vault	Note 6	None Required
12463	Cask Loading Pit	6	Spent fuel	Notes 2, 6	None Required
12563	Spent Fuel Pit	5, 6	Spent fuel	Note 6	None Required
Fuel Transfer Areas					
12564	Fuel Transfer Tube	6	Fuel in transit	Notes 2, 5, 9	None Required
11205	Reactor Vessel Nozzle Area	5	Fuel in transit	Notes 2, 3, 9	None Required
11504	Refueling Cavity	6	Fuel in transit	Notes 2, 3, 9	None Required

STD COL 12.3-1

Table 12AA-201 (Sheet 2 of 2)
Very High Radiation Areas (VHRA)

Notes

1. VHRA during full power operation; less than 10 Rem/hr 24 hours after plant shutdown.
2. During underwater spent fuel transfer operations, this area can be as high as VHRA.
3. During underwater reactor internals transfers/ storage, this area can be as high as VHRA.
4. During spent resin waste disposal container transfer or loading, this area can be as high as VHRA. The contact dose rate of spent resin containers can be greater than 1000 Rem/hr.
5. Discussion about the Spent Fuel Transfer Canal and Tube Shielding is provided in DCD Subsection 12.3.2.2.9.
6. Source is transient, removable, or can be relocated.
7. VHRA when hatch is removed during spent resin container handling operation.
8. In the event that the room does need to be accessed for maintenance or other reasons, temporary shielding is put in place and the resin is removed from the vessels. These measures reduce exposure rates in the room, such that this room is no longer a VHRA. Remote handling is used for any tasks that require the opening of the access hatch in the ceiling of this room when media is present.
9. These areas have no planned reasons for entry and are only classified as VHRAs during periods of fuel movement. In the event that these rooms do need to be accessed to repair the Fuel-Transfer System, Fuel Transfer Tube Gate Valve, or other components, it is done during a non-fuel movement time. This keeps the dose received by the worker as low as reasonably achievable.
10. Inspection of the equipment in this room, when required, is done using remote viewing equipment. Two plugs between Room 12153 and 12155 contain instruments and the plugs are expected to be removed every 12 to 18 months for performance of maintenance. Administrative procedures are implemented to protect workers pursuant to Regulatory Guide 8.38.

CHAPTER 13
CONDUCT OF OPERATIONS

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CHAPTER 13
CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD DEP 1.1-1 DCD Subsection 13.1.1, Combined License Information, is renumbered in this FSAR section to 13.1.4.

VCS COL 13.1-1 This section describes organizational positions for VCSNS Units 2 and 3 SCE&G and associated functions and responsibilities. The position titles used in the text may be generic and describe the function of the position. Table 13.1-201, Generic Position/Site Specific Position Cross Reference, provides a cross-reference to identify the corresponding site-specific position titles.

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

SCE&G has over 35 years of experience in the design, construction, and operation of nuclear generating stations. SCE&G has designed, constructed, and operates V.C. Summer Nuclear Station (VCSNS) Unit 1.

13.1.1.1 Design, Construction, and Operating Responsibilities

The President and Chief Operating Officer has overall responsibility for functions involving design, construction, and operation. Line responsibilities for those functions are assigned to the Executive Vice President-Generation (EVPG) through the Senior Vice President Nuclear Operations (SVPNO) via the Vice President, New Nuclear Deployment (VPNND) (Figure 13AA-201) for the design and construction of new nuclear plants. At the appropriate time after construction, direct control of nuclear plant operation is assigned to the site executive in charge of VCSNS, the Vice President Nuclear Operations, (VPNO), and his direct reports. The first priority and responsibility of each member of the nuclear staff throughout the life of the plant is nuclear safety. Decision-making for station activities is performed in a conservative manner with expectations of this core value regularly communicated to appropriate personnel by management interface, training, and station directives.

Lines of authority, decision-making, and communication are clearly and unambiguously established to enable the understanding of the various project members, including contractors, that utility management is in charge and directs the project. Key executive and corporate management positions, functions, and responsibilities are discussed in Subsection 13.1.1.3.1. The corporate organization is shown in Figure 13.1-203. The management and technical support organization for design, construction, and preoperational activities is addressed in Appendix 13AA.

13.1.1.2 Provisions for Technical Support Functions

Before beginning preoperational testing, the VPNND and the VPNO will establish the organization of managers, functional managers, supervisors, and staff sufficient to perform required functions for support of safe plant operation. These functions include the following:

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and material, and instrumentation and controls engineering
- Safety review
- Quality assurance, audit and surveillance
- Plant chemistry
- Radiation protection
- Fueling and refueling operations support
- Training
- Maintenance support
- Operations support
- Fire protection
- Emergency response organization
- Outside contractual assistance

In the event that station personnel are not qualified to deal with a specific problem, the services of qualified individuals from other functions within the company or an outside consultant are engaged. For example, major contractors, such as the reactor technology vendor or turbine generator manufacturer, provide technical support when equipment modifications or special maintenance problems are considered. Special studies, such as environmental monitoring, may be contracted to qualified consultants. Figure 13.1-201 illustrates the management and technical support organizations supporting operation of the plant. See Section 13.1.1.3.2 for description of responsibilities and authorities of management positions for organizations providing technical support. Table 13.1-201 shows the estimated number of positions required for each function.

Physical separation of units helps to minimize wrong-unit activities. In addition station procedures and programs include such features as tagging programs, procedure adherence requirements, and training to provide operating staff with methods to minimize human error.

13.1.1.2.1 Engineering

The engineering department consists of plant support (system) engineering, design engineering, and materials and procurement engineering. These groups are responsible for performing the classical design activities as well as providing engineering expertise in other areas.

Each of the engineering groups has a functional manager who reports to the General Manager, Engineering Services (GMES).

The engineering department is responsible for:

- Supporting plant operations in the engineering areas of mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, electronic, instrument and control, and fire protection. Priorities for support activities are established based on input from the plant manager with emphasis on issues affecting safe operation of the plant.
- Engineering programs.
- Supporting procurement, chemical and environmental analysis and maintenance activities in the plant as requested by the plant manager.
- Performing design engineering of plant modifications.
- Maintaining the design basis by updating the record copy of design documents as necessary to reflect the actual as-built configuration of the plant.
- Accident and transient analyses.
- Human Factors Engineering design process.
- Audit, surveillance, and evaluation of nuclear division suppliers.
- Procurement and materials storage.

Reactor engineering, part of design engineering, provides technical assistance in the areas of core design, core operations, core thermal limits, and core thermal hydraulics.

Engineering work may be contracted to and performed by outside companies in accordance with the quality assurance (QA) program.

Engineering resources are shared between units. A single management organization oversees the engineering work associated with the station units.

13.1.1.2.2 Safety Review

Review and audit activities are addressed in Chapter 17. Oversight of safety review of station programs, procedures, and activities is performed by a plant safety review committee and a nuclear safety review committee.

Personnel who perform safety reviews are shared between units.

13.1.1.2.3 Quality Assurance

Safety-related activities associated with the operation of the plant are governed by QA direction established in Chapter 17 of the FSAR and the Quality Assurance Program Description (QAPD). The requirements and commitments contained in the QAPD apply to activities associated with structures, systems, and components that are safety-related and are mandatory and must be implemented, enforced, and adhered to by individuals and organizations. QA requirements are implemented through the use of approved procedures, policies, directives, instructions, or other documents that provide written guidance for the control of quality-related activities and provide for the development of documentation to provide objective evidence of compliance. The QA function includes:

- Maintaining the QAPD.
- Coordinating the development of audit schedules.
- Supporting general QA indoctrination and training for the nuclear station personnel.

The QA organization is independent of the station management line organization. Quality control (QC) inspection/testing activities to support plant operation, maintenance, and outages are independent of the station management line organization.

Personnel resources of the QA and QC organization are shared between units. A single management organization oversees the QA group for the station units.

13.1.1.2.4 Chemistry

A chemistry program is established to monitor and control the chemistry of various plant systems such that corrosion of components and piping is minimized and radiation from corrosion byproducts is kept to levels that allow operations and maintenance with radiation doses as low as reasonably achievable.

The functional manager in charge of chemistry is responsible to the General Manager, Nuclear Support Services (GMNSS) for maintaining chemistry programs and for monitoring and maintaining the water chemistry of plant systems. The staff of the chemistry department consists of laboratory technicians, support personnel, and supervisors who report to the functional manager in charge of chemistry.

Personnel resources of the chemistry organization are shared between units. A single management organization oversees the chemistry group for the station units.

13.1.1.2.5 Radiation Protection

A radiation protection (RP) program is established to protect the health and safety of the surrounding public and personnel working at the plant. The RP program is described in Chapter 12 of the FSAR. The program includes:

- Respiratory Protection
- Personnel Dosimetry
- Bioassay
- Survey Instrument Calibration and Maintenance
- Radioactive Source Control
- Effluents and Environmental Monitoring and Assessment
- Radioactive Waste Shipping
- Radiation Work Permits
- Job Coverage
- Radiation Monitoring and Surveys

The Health Physics/Safety (RP) department is staffed by radiation protection technicians, support personnel, and supervisors who report to the functional manager in charge of radiation protection. To provide sufficient organizational freedom from operating pressures, the manager in charge of radiation protection reports directly to the GMNSS.

Personnel resources of the RP organization are shared between units. A single management organization oversees the RP group for the station units.

13.1.1.2.6 Fueling and Refueling Support

The function of fueling and refueling is performed by a combination of personnel from various departments including operations, maintenance, radiation protection, engineering, and reactor technology vendor or other contractor staff. Initial fueling and refueling operations are a function of the Planning/Outage support organization. The manager in charge of Planning/Outage support is responsible for planning and scheduling outages and for refueling support and reports to the General Manager Nuclear Plant Operations (GMNPO).

Personnel resources of the outage support organization are shared between units. A single management organization oversees outage support work associated with Units 2 and 3.

13.1.1.2.7 Training

The training department is responsible for providing training programs that are established, maintained, and implemented in accordance with applicable plant administrative directives, regulatory requirements, and company operating policies so that station personnel can meet the performance requirements of their jobs in operations, maintenance, technical support, and emergency response. The objective of training programs is to provide qualified personnel to operate and maintain the plant in a safe and efficient manner and to provide compliance with the license, technical specifications, and applicable regulations. The training department's responsibilities encompass operator initial license training, requalification training, and plant staff training as well as the plant access training (general employee training) course and radworker training. The functional manager of nuclear training is independent of the operating line organization to provide for independence from operating pressures. Nuclear plant training programs are described in Section 13.2 of the FSAR.

Personnel resources of the training department are shared between units. A single management organization provides oversight of station training activities.

13.1.1.2.8 Maintenance Support

In support of maintenance activities, planners, schedulers, and parts specialists prepare work packages, acquire proper parts, and develop procedures that provide for the successful completion of maintenance tasks. Maintenance tasks are integrated into the station schedule for evaluation of operating or safe shutdown risk elements and to provide for efficient and safe performance. The manager in charge of maintenance reports to the GMNPO.

Personnel resources of the maintenance support organization are shared between units. A single management organization oversees the function of maintenance support for Units 2 and 3.

13.1.1.2.9 Operations Support

The operations support function is provided under the direction of the manager in charge of operations. Operations support includes the following programs:

- Operations procedures
- Operations surveillances

- Equipment tagging
- Fire protection testing and surveillance

13.1.1.2.10 Fire Protection

VCS COL 13.1-1
VCS COL 9.5-1

The station is committed to maintaining a fire protection program as described in Subsection 9.5.1.8.

The VPNO has overall responsibility for fire protection. Assigning the responsibilities at that level provides the authority to obtain the resources and assistance necessary to meet fire protection program objectives, resolve conflicts, and delegate appropriate responsibility to fire protection staff. The relationship of the VPNO to other plant staff personnel with fire protection responsibilities is shown on Figure 13.1-201. Fire protection for the facility is organized and administered through the fire protection program staff by the engineer in charge of fire protection. The fire protection program staff is made up of members from operations, design engineering, plant support engineering, licensing, and nuclear training. The engineer in charge of fire protection reports to the GMES. The GMES reports directly to the VPNO.

Inspections of fire protection systems and functions, the operations-related fire protection program activities and development and implementation of the fire protection program including development of fire protection procedures are the responsibility of the manager in charge of operations who reports to the GMNPO.

Site personnel and the fire brigade training is the responsibility of the Manager, Nuclear Training. The Manager, Nuclear Training reports to the GMNSS. The GMNSS reports directly to the VPNO.

Personnel resources of the fire protection organization are shared between units. A single management organization oversees the fire protection group for the station units.

13.1.1.2.11 Emergency Response Organization

VCS COL 13.1-1

The emergency response organization is a matrixed organization composed of personnel who have the experience, training, knowledge, and ability necessary to implement actions to protect the public in the case of emergencies. Managers and station personnel assigned positions in the emergency organization are responsible for supporting the emergency preparedness organization and emergency plan as required. The staff members of the emergency planning organization administrate and orchestrate drills and training to maintain qualification of station staff members and develop procedures to guide and direct the emergency organization during an emergency. The functional manager in charge of emergency preparedness reports to the GMNSS. The VCSNS emergency response organization is described in the Emergency Plan.

Resources of the emergency planning group are shared between units. A single management organization oversees the emergency planning group for the station units.

13.1.1.2.12 Outside Contractual Assistance

Contract assistance with vendors and suppliers of services not available from organizations established as part of utility staff is provided by the Business and Financial Services (BFS) organization. Personnel in the BFS organization perform the necessary functions to contract vendors of special services to perform tasks for which utility staff does not have the experience or equipment required. The functional manager in BFS reports to the VPNO.

Resources of the BFS organization are shared between units. A single management organization oversees the BFS group for the station units.

13.1.1.3 Organizational Arrangement

13.1.1.3.1 Executive Management Organization

Executive management is ultimately responsible for executing activities and functions for the nuclear generating plants owned by the utility. Executive management establishes expectations such that a high level of quality, safety, and efficiency is achieved in aspects of plant operations and support activities through an effective management control system and an organization selected and trained to meet the above objectives. A high-level chart of the utility headquarters organization is illustrated in Figure 13.1-203. Executives and management with direct line of authority for activities associated with operation of the station are shown in Figure 13.1-201.

13.1.1.3.1.1 Chief Executive Officer

The Chief Executive Officer (CEO) has the ultimate responsibility for the safe and reliable operation of each nuclear unit owned and/or operated by SCE&G. The CEO is responsible for the overall direction and management of the corporation, and the execution of the company policies, activities, and affairs. The CEO is assisted by the EVPG (also the chief nuclear officer), and other executive staff in the nuclear division of the corporation.

13.1.1.3.1.2 President and Chief Operating Officer

The President and Chief Operating Officer (COO) is responsible for directing SCE&G’s core operational business including the fossil, hydroelectric, and nuclear generation. The COO reports to the CEO.

13.1.1.3.1.3 Executive Vice President - Generation (EVPG)/Chief Nuclear Officer

The EVPG reports to the CEO through the COO. The EVPG/Chief Nuclear Officer (CNO) is responsible for electric generation and overall plant nuclear safety and takes the measures needed to provide acceptable performance of the staff in operating, maintaining, and providing technical support to the nuclear site. The EVPG/CNO delegates authority and responsibility for the operation and support of the site through the SVPNO. It is the responsibility of the EVPG/CNO to provide guidance and direction such that safety-related activities, including engineering, construction, operations, operations support, maintenance, and planning, are performed following the guidelines of the quality assurance program.

The EVPG/CNO is responsible for new nuclear plant licensing, design, and construction through the SVPNO.

13.1.1.3.1.4 Senior Vice President Nuclear Operations (SVPNO)

The SVPNO reports to the EVPG. The SVPNO is responsible for the safe operation of all current nuclear plant operations along with the design, licensing, and construction of new nuclear plants. The SVPNO delegates authority and responsibility for the operation and support of the operating nuclear plants through the VPNO. The SVPNO is responsible for new nuclear plant licensing, design, and construction via the VPNND who maintains control of nuclear plant construction through construction completion.

13.1.1.3.1.5 Vice President Nuclear Operations (VPNO)

The VPNO reports to the SVPNO. The VPNO is directly responsible for management and direction of activities associated with the efficient, safe, and reliable operation of the nuclear station. The VPNO is assisted in management and technical support activities by the GMNPO, GMES, GMNSS, and the General Manager Organizational Effectiveness (GMOE). The VPNO is responsible for the site fire protection program through the engineer in charge of fire protection. See Subsection 13.1.1.2.10.

13.1.1.3.2 Site Support Organization

13.1.1.3.2.1 General Manager Engineering Services (GMES)

The GMES is the onsite lead position for engineering and reports to the VPNO. The GMES is responsible for engineering activities related to the operation or maintenance of the plant and design change implementation support activities and other functions described in Subsection 13.1.1.2.1.

The GMES directs functional managers responsible for plant support (system) engineering, design engineering, and materials and procurement engineering.

13.1.1.3.2.1.1 Functional Manager In Charge of Plant Support (System) Engineering

The functional manager in charge of plant support engineering reports to the GMES and supervises a technical staff of engineers and other engineering specialists and coordinates their work with that of other groups. The functional manager in charge of plant support engineering is responsible for providing direction and guidance to system engineers as follows:

- Monitoring the efficiency and proper operation of balance of plant and reactor systems.
- Planning programs for improving equipment performance, reliability, or work practices.
- Overseeing operational tests and analyzing the results.
- Maintaining engineering programs such as ISI/IST, valve testing, maintenance rule, piping erosion/corrosion, and equipment reliability.

13.1.1.3.2.1.2 Functional Manager In Charge of Design Engineering

The functional manager in charge of design engineering reports to the GMES and is responsible for:

- Resolving design issues.
- Onsite development of design-related change packages and plant modifications.
- Implementing effective project management methods and procedures, including cost controls, for implementation of modifications and construction activities.
- Managing contractors who may perform modification or construction activities.
- Maintaining configuration control program.
- Reactor engineering and core design as discussed in Subsection 13.1.1.2.1.
- Developing and maintaining accident analysis activities and programs.

13.1.1.3.2.1.3 Functional Manager In Charge of Materials and Procurement Engineering (M&PE)

The functional manager in charge of M&PE is responsible for providing sufficient and proper materials to support the material needs of the plant and performing related activities including:

- Procedure development
- Procurement and Materials storage
- Supply system database maintenance
- Meeting quality assurance and external audit requirements

The functional manager in charge of M&PE is also responsible for site purchasing. The functional manager in charge of M&PE reports to the GMES.

13.1.1.3.2.1.4 Engineer in Charge of Fire Protection

VCS COL 13.1-1
VCS COL 9.5-1

The engineer in charge of fire protection is responsible for the following:

- Fire protection program requirements, including consideration of potential hazards associated with postulated fires, knowledge of building layout, and system design.
- Post-fire shutdown capability.
- Design, maintenance, surveillance, and QA of fire protection features (e.g., detection systems, suppression systems, barriers, dampers, doors, penetration seals and fire brigade equipment.
- Oversight of fire prevention activities (administrative controls and training).
- Oversight of fire brigade organization and training.
- Pre-fire planning including review and updating of pre-fire plans at least every two years.

The engineer in charge of fire protection reports to the GMES who has ultimate responsibility for the fire protection program of the plant. Additionally, the engineer in charge of fire protection works with the operations support supervisors to coordinate activities and program requirements with the operations department. In accordance with Regulatory Guide 1.189, the engineer in charge of fire protection is a graduate of an engineering curriculum of accepted standing and has completed not less than six years of engineering experience, three of which were in a responsible position in charge of fire protection engineering work. The engineer in charge of fire protection has training and experience in nuclear plant safety or has personnel available to assist him who have training and experience in nuclear plant safety.

VCS COL 13.1-1 13.1.1.3.2.2 General Manager Nuclear Support Services (GMNSS)

The GMNSS is responsible for support functions including training, chemistry, radiation protection, emergency preparedness, and licensing. The GMNSS delegates authority and responsibility through managers in charge of each of these support functions. The GMNSS reports to the VPNO.

13.1.1.3.2.2.1 Functional Manager In Charge of Training (Nuclear Training)

VCS COL 13.1-1
VCS COL 18.10-1
The functional manager in charge of nuclear training is responsible for training programs at the site required for the safe and proper operation and maintenance of the plant including:

- Operations training programs
- Plant staff training programs
- Plant access training
- Radiation worker training
- Fire brigade training

The functional manager in charge of nuclear training may seek assistance from other departments within the company or outside specialists such as educators and manufacturers. The functional manager in charge of training ensures individuals providing fire brigade training are qualified by knowledge, suitable training, and experience for such work, and that coordination with the engineer in charge of fire protection is maintained. The functional manager in charge of nuclear training supervises a staff of training supervisors who coordinate the development, preparation, and presentation of training programs for nuclear plant personnel and reports directly to the GMNSS.

VCS COL 13.1-1 13.1.1.3.2.2.2 Functional Manager In Charge of Plant Licensing (Nuclear Licensing)

The functional manager in charge of nuclear licensing is responsible for providing a coordinated focus for interface with the NRC, and for technical direction and administrative guidance to the licensing staff for licensing activities including the following:

- Developing licensee event reports and responding to notices of violations.
- Writing/submitting operating license and technical specification amendments and updating the FSAR.
- Tracking commitments and answering generic letters.
- Monitoring industry issues.
- Preparing station for special NRC inspections, interfacing with NRC inspectors, and interpreting NRC regulations.
- Maintaining the licensing basis.
- Probabilistic risk assessment studies.

The functional manager in charge of nuclear licensing reports to the GMNSS.

13.1.1.3.2.2.3 Functional Manager In Charge of RP (Health Physics/ Safety - HPS)

The functional manager in charge of HPS has the direct responsibility for providing adequate protection of the health and safety of personnel working at the plant and members of the public during activities covered within the scope and extent of the license. RP responsibilities of the functional manager in charge of HPS are consistent with the guidance in Regulatory Guide 8.8 and Regulatory Guide 8.10. They include:

- Managing the RP organization.
- Establishing, implementing, and enforcing the RP program.
- Providing RP input to facility design and work planning.
- Tracking and analyzing trends in radiation work performance and taking necessary actions to correct adverse trends.
- Supporting the plant emergency preparedness program and assigning emergency duties and responsibilities within the RP organization.
- Delegating authority to appropriate RP staff to stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety.

The functional manager in charge of HPS reports to the GMNSS and is assisted by the supervisors in charge of RP.

13.1.1.3.2.2.3.1 Supervisor of Radwaste Operations

The supervisor of radwaste operations is responsible for developing, implementing, directing, and coordinating the radwaste program. The supervisor of radwaste operations reports to the manager in charge of HPS. The supervisor of radwaste operations supervises radwaste operators assigned to the radwaste area.

13.1.1.3.2.2.3.2 Supervisor In Charge of RP

The supervisors in charge of RP are responsible for carrying out the day-to-day operations and programs of the RP department as listed in Subsection 13.1.1.2.5.

Supervisors in charge of RP report to the functional manager in charge of HPS.

13.1.1.3.2.2.3.3 RP Technicians

RP technicians directly carry out responsibilities defined in the RP program and procedures. In accordance with technical specifications, an RP technician is on site whenever there is fuel in the vessel. See Table 13.1-202.

The following are some of the duties and responsibilities of the RP technicians:

- As delegated authority by the manager in charge of HPS, stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety.
- Provide coverage and monitor radiation conditions for jobs potentially involving significant radiation exposure.
- Conduct surveys, assess radiation conditions, and establish RP requirements for access to and work within restricted, radiation, high radiation, very high radiation, airborne radioactivity areas, and areas containing radioactive materials.
- Provide control over the receipt, storage, movement, use, and shipment of licensed radioactive materials.
- Review work packages, proposed design modifications, and operations and maintenance procedures to facilitate integration of adequate radiation protection controls and dose-reduction measures.
- Review and oversee implementation of plans for the use of process or other engineering controls to limit the concentrations of radioactive materials in the air.

- Provide personnel monitoring and bioassay services.
- Maintain, prescribe, and oversee the use of respiratory protection equipment.
- Perform assigned emergency response duties.

13.1.1.3.2.2.4 Functional Manager In Charge of Chemistry

The functional manager in charge of chemistry is responsible for developing, implementing, directing, and coordinating the chemistry, radiochemistry and nonradiological environmental monitoring programs. This area includes overall operation of the hot lab, cold lab, and nonradiological environmental monitoring. The functional manager in charge of chemistry is responsible for developing, administering, and implementing procedures and programs that provide for effective compliance with environmental regulations. The functional manager in charge of chemistry reports to the GMNSS and directly supervises the chemistry supervisors and chemistry technicians as assigned.

13.1.1.3.2.2.5 Functional Manager In Charge of Emergency Services

The functional manager in charge of emergency services is responsible for:

- Coordinating and implementing the plant emergency response plan with state and local emergency plans.
- Developing, planning, and executing emergency drills and exercises.
- Emergency action level development.
- NRC reporting associated with 10 CFR 50.54(q).

The functional manager in charge of emergency services reports to the GMNSS.

13.1.1.3.2.3 General Manager, Organizational Effectiveness (GMOE)

The GMOE reports to the VPNO and is responsible for support functions including quality services, nuclear protection services (security), and organizational development and performance.

13.1.1.3.2.3.1 Functional Manager In Charge of Security (Nuclear Protection Services)

The functional manager in charge of nuclear protection services is responsible for:

- Implementing and enforcing security directives, procedures, and instructions received from appropriate authorities.

- Day-to-day supervision of the security guard force.
- Administration of the security program.

The functional manager in charge of nuclear protection services reports directly to the GMOE.

13.1.1.3.2.3.2 Functional Manager In Charge of Organizational Development and Performance (OD&P)

The responsibilities of the functional manager in charge of OD&P includes establishing processes and procedures to facilitate identification and correction of conditions adverse to quality and implement corrective actions. The functional manager in charge of OD&P also manages the Operating Experience and Human Performance programs. The functional manager in charge of OD&P reports to the GMOE.

13.1.1.3.2.3.3 Functional Manager In Charge of Quality Assurance (Quality Systems)

The functional manager in charge of quality systems is responsible for those functions described in Subsection 13.1.1.2.3 and reports to the GMOE. Responsibilities of the functional manager in charge of quality systems are fulfilled through the supervisors and staff of the quality systems organization.

13.1.1.3.2.4 Manager In Charge of Site Business (BFS)

The manager in charge of site business is responsible for business and financial services and project management activities and reports to the VPNO.

13.1.1.4 Qualifications of Technical Support Personnel

VCS COL 13.1-1
VCS COL 18.6-1

The qualifications of managers and supervisors of the technical support organization meet the qualification requirements in education and experience for those described in ANSI/ANS-3.1-1993 (Reference 201) as endorsed and amended by Regulatory Guide 1.8.

13.1.2 OPERATING ORGANIZATION

VCS COL 13.1-1

13.1.2.1 Plant Organization

The plant management, technical support, and plant operating organizations are shown in Figure 13.1-201. The on-shift operating organization is presented in Figure 13.1-202, which shows those positions requiring NRC licenses. Additional personnel are required to augment normal staff during outages.

Nuclear plant employees are responsible for reporting problems with plant equipment and facilities. They are required to identify and document equipment problems in accordance with the QA program. QA program requirements as they apply to the operating organization are described in Chapter 17. Administrative procedures or standing orders include:

- Establishing a QA program for the operational phase.
- Preparing procedures necessary to carry out an effective QA program. See Section 13.5 for description of the station procedure program.
- A program for review and audit of activities affecting plant safety. See Section 17.5 for description of station review and audit programs.
- Programs and procedures for rules of practice as described in Section 5.2 of N18.7-1976/ANS-3.2 (Reference 203).

Managers and supervisors within the plant operating organization are responsible for establishing goals and expectations for their organization and to reinforce behaviors that promote radiation protection. Specifically, managers and supervisors are responsible for the following, as applicable to their position within the plant organization:

- Interface directly with RP staff to integrate RP measures into plant procedures and design documents and into the planning, scheduling, conduct, and assessment of operations and work.
- Notify RP personnel promptly when RP problems occur or are identified, take corrective actions, and resolve deficiencies associated with operations, procedures, systems, equipment, and work practices.
- Ensure department personnel receive training on RP and periodic retraining, in accordance with 10 CFR Part 19 so that they are properly instructed and briefed for entry into restricted areas.
- Periodically observe and correct, as necessary, radiation worker practices.
- Support RP management in implementing the RP program.
- Maintain exposures to site personnel ALARA.

13.1.2.1.1 General Manager, Nuclear Plant Operations (GMNPO/Plant Manager)

The GMNPO reports to the VPNO, is responsible for overall safe operation of the plant, and has control over those onsite activities necessary for safe operation and maintenance of the plant including the following:

- Operations

- Maintenance and modification
- Planning/outage management

Additionally, the GMNPO has overall responsibility for occupational and public radiation safety. RP responsibilities of the GMNPO are consistent with the guidance in Regulatory Guide 8.8 and Regulatory Guide 8.10 including the following:

- Provide management RP policy throughout the VCSNS Units 2 and 3 organization.
- Provide an overall commitment to RP by the VCSNS Units 2 and 3 organization.
- Interact with and support the manager in charge of RP on implementation of the RP program.
- Support identification and implementation of cost-effective modifications to plant equipment, facilities, procedures, and processes to improve RP controls and reduce exposures.
- Establish plant goals and objectives for RP.
- Maintain exposures to site personnel ALARA.
- Support timely identification, analysis, and resolution of RP problems (e.g., through the plant corrective action program).
- Provide for training to site personnel on RP in accordance with 10 CFR Part 19.
- Establish an ALARA Committee with delegated authority from the plant manager that includes, at a minimum, the managers in charge of operations, maintenance, RP, and representatives from engineering to help provide for effective implementation of line organization responsibilities for maintaining worker doses ALARA.

The line of succession of authority and responsibility for overall operations in the event of unexpected events of a temporary nature is:

a. Manager in charge of operations

b. Manager in charge of plant maintenance

c. Supervisor in charge of operations

As described in Subsection 13.1.2.1.1.3.2.1, the manager in charge on-shift (the shift supervisor) is the GMNPO's direct representative for the conduct of operations. The succession of authority includes the authority to issue standing or special orders as required.

13.1.2.1.1.1 Manager In Charge of Maintenance

Maintenance of the plant is performed by the maintenance department mechanical, electrical, and instrumentation and control disciplines. Planning, scheduling, and work package preparation are performed by the planning/outage group. The functions of the maintenance department are to perform preventive and corrective maintenance, equipment testing, and implement modifications as necessary.

The manager in charge of plant maintenance is responsible for the performance of preventive and corrective maintenance and modification activities required to support operations, including compliance with applicable standards, codes, specifications, and procedures. The manager in charge of plant maintenance reports to the GMNPO and provides direction and guidance to the maintenance discipline supervisors and maintenance support staff.

13.1.2.1.1.1.1 Maintenance Discipline Supervisors

The supervisors of each maintenance discipline (mechanical, electrical, instrumentation and control, and support) are responsible for maintenance activities within their discipline including plant modifications. They provide guidance in maintenance planning and craft supervision. They establish the necessary manpower levels and equipment requirements to perform both routine and emergency-type maintenance activities, seeking the services of others in performing work beyond the capabilities of the plant maintenance group. Each discipline supervisor is responsible for liaison with other plant staff organizations to facilitate safe operation of the station. These supervisors report to the manager in charge of maintenance.

13.1.2.1.1.1.1.1 Maintenance Discipline Foremen

The maintenance discipline foremen (mechanical, electrical, and instrumentation and control) supervise maintenance activities, assist in the planning of future maintenance efforts, and guide the efforts of the craft within their discipline. The maintenance discipline foremen report to the appropriate maintenance discipline supervisor.

13.1.2.1.1.2 Manager in Charge of Planning/Outage

The manager in charge of planning/outage support is responsible for:

- Planning and scheduling refueling, maintenance, and forced outages.
- Providing direction and guidance to staff members in establishing outage activities.

- Minimizing shutdown risk during outages with proper planning and preparation.
- Directing activities during outages to provide safe, efficient, and effective outages.
- Preparing work packages.

The manager in charge of planning/outage reports to the GMNPO. See Subsection 13.1.1.2.6.

13.1.2.1.1.3 Operations Department

Operations activities are conducted with safety of personnel, the public, and equipment as the overriding priority. The operations department is responsible for:

- Operation of station equipment.
- Monitoring and surveillance of safety and nonsafety-related equipment.
- Fuel loading.
- Providing the nucleus of emergency and fire-fighting teams.

The operations department maintains sufficient licensed and senior licensed operators to staff the control room continuously using a crew rotation system. The operations department is under the authority of the manager in charge of operations, who through the operations supervisor, directs the day-to-day operation of the plant.

Specific duties, functions, and responsibilities of key shift members are discussed in Subsections 13.1.2.1.1.3.2.1 through 13.1.2.1.1.3.2.1.4 and in plant administrative procedures and the technical specifications. The minimum shift manning requirements are shown in Table 13.1-202.

Some resources of the operations organization are shared between units. Administrative and support personnel perform their duties on either unit. Additional operations staff is required to fill the on-shift staffing requirements of the additional units. To operate or supervise the operation of more than one unit, an operator (senior reactor operator [SRO] or reactor operator [RO]) must hold an appropriate, current license for each unit. A single management organization oversees the operations group for Units 2 and 3. See Table 13.1-201 for estimated number of staff in the operations department for single or multiple units.

The operations support section is staffed with sufficient personnel to provide support activities for the operating shifts and overall operations department. The following is an overview of the operations organization.

Manager In Charge of Operations

The manager in charge of operations has overall responsibility for the day-to-day operation of the plant. The manager in charge of operations reports to the GMNPO and is assisted by the operations supervisor and operations support supervisor. The manager in charge of operations or the operations supervisor is SRO licensed.

13.1.2.1.1.3.1 Operations Support Supervisor

The operations support supervisor, under the direction of the manager in charge of operations, is responsible for:

- Directing and guiding plant operations support activities in accordance with the operating license, technical specifications, and written procedures.
- Providing supervision of operating support personnel, for operations support activities, and coordination of support activities.
- Coordinating operations-related fire protection program activities with the engineer in charge of fire protection.

The operations support supervisor is assisted by the operations procedures group, operations scheduling, and other support personnel. In the absence of the manager in charge of operations or operations supervisor, the operations support supervisor may assume the duties and responsibilities of either of these positions.

13.1.2.1.1.3.2 Operations Supervisor

The operations supervisor, under the direction of the manager in charge of operations, is responsible for:

- Shift plant operations in accordance with the operating license, technical specifications, and written procedures.
- Providing supervision of operating shift personnel for operational shift activities including those of emergency and firefighting teams.
- Coordinating with the operations support supervisor and other plant staff sections.
- Verifying that nuclear plant operating records and logs are properly prepared, reviewed, and evaluated.

The operations supervisor is assisted in these areas by the shift supervisors who direct the operating shift personnel. The operations supervisor reports to the manager in charge of operations and in the absence of the manager in charge of operations or operations support supervisor, may assume the duties and responsibilities of either of these positions.

13.1.2.1.1.3.2.1 Shift Supervisor

The shift supervisor is a licensed SRO responsible for the control room command function, and is the GMNPO's direct management representative for the conduct of operations. As such, the shift supervisor has the responsibility and authority to direct the activities and personnel onsite as required to:

- Protect the health and safety of the public, the environment, and personnel on the plant site.
- Protect the physical security of the plant.
- Prevent damage to site equipment and structures.
- Comply with the operating license.

The shift supervisor retains this responsibility and authority until formally relieved of operating responsibilities by a licensed SRO. Additional responsibilities of the shift supervisor include:

- Directing nuclear plant employees to report to the plant for response to potential and real emergencies.
- Seeking the advice and guidance of the shift technical advisor and others in executing the duties of the shift supervisor whenever in doubt as to the proper course of action.
- Promptly informing responsible supervisors of significant actions affecting their responsibilities.
- Participating in operator training, retraining, and requalification activities from the standpoint of providing guidance, direction, and instruction to shift personnel.

The shift supervisor is assisted in carrying out the above duties by the control room supervisors and the operating shift personnel. The shift supervisor reports to the operations supervisor.

13.1.2.1.1.3.2.1.1 Control Room Supervisor

The control room supervisor is a licensed SRO. The primary function of the control room supervisor is to administratively support the shift supervisor such that the "command function" is not overburdened with administrative duties and to supervise the licensed and non-licensed operators in carrying out the activities directed by the shift supervisor. Other duties include:

- Being aware of maintenance and testing performed during the shift.
- Shutting down the reactor if conditions warrant this action.
- Informing the shift supervisor and other station management in a timely manner of conditions that may affect public safety, plant personnel safety, plant capacity or reliability, or cause a hazard to equipment.
- Initiating immediate corrective action as directed by the shift supervisor in any upset situation until assistance, if required, arrives.
- Participating in operator training, retraining, and requalification activities from the standpoint of providing guidance, direction, and instruction to shift personnel.

The control room supervisor reports directly to the shift supervisor.

13.1.2.1.1.3.2.1.2 Reactor Operator

The ROs are licensed reactor operators and normally report to the control room supervisor or shift supervisor. They are responsible for routine plant operations and performance of major evolutions at the direction of the supervisor in charge on-shift. The RO duties include:

- Monitoring control room instrumentation.
- Responding to plant or equipment abnormalities in accordance with approved plant procedures.
- Directing the activities of non-licensed operators.
- Documenting operational activities, plant events, and plant data in shift logs.
- Initiating plant shutdowns or scrams or other compensatory actions when observation of plant conditions indicates a nuclear safety hazard exists or when approved procedures so direct.

Whenever there is fuel in the reactor vessel, at least one RO is in the control room monitoring the status of the unit at the main control panel. The RO assigned to the main control panel is designated the “operator at the controls” and conducts monitoring and operating activities in accordance with the guidance set forth in Regulatory Guide 1.114, which is further described in Subsection 13.1.2.2, Conduct of Operations.

13.1.2.1.1.3.2.1.3 Non-Licensed Operator (Auxiliary Operator)

The non-licensed operators perform routine duties outside the control room as necessary for continuous, safe plant operation including:

- Assisting in plant startup, shutdown, surveillance, and emergency response by manually or remotely changing equipment operating conditions, placing equipment in service, or securing equipment from service at the direction of the reactor operator.

- Performing assigned tasks in procedures and checklists such as valve manipulations for plant startup or data sheets on routine equipment checks, and making accurate entries according to the applicable procedure, data sheet, or checklist.

- Assisting in training of new employees and for improvement and upgrading of their own performance by participating in the applicable sections of the training program.

13.1.2.1.1.3.2.1.4 Shift Technical Advisor

The station is committed to meeting NUREG-0737 TMI Action Plan item I.A.1.1 for shift technical advisors. The shift technical advisor (STA) reports directly to the shift supervisor and provides advanced technical assistance to the operating shift complement during normal and abnormal operating conditions. The STA's responsibilities are detailed in plant administrative procedures as required by TMI Action Plan I.A.1.1 and NUREG-0737 Appendix C. These responsibilities include:

- Activities to monitor core power distribution and critical parameters.

- Activities to assist the operating shift with technical expertise during normal and emergency conditions.

- Evaluation of technical specifications, special reports, and procedural issues.

The STA is to primarily contribute to maximizing safety of operations by independently observing plant status and advising shift supervision of conditions that could compromise plant safety. During transients or accident situations, the STA independently assesses plant conditions and provides technical assistance and advice to mitigate the incident and minimize the effect on personnel, the environment, and plant equipment.

An SRO on shift who meets the qualifications for the combined SRO/STA position specified for Option 1 of Generic Letter 86-04 (Reference 202) may also serve as the STA. If this option is used for a shift, then the separate STA position may be eliminated for that shift.

13.1.2.2 Conduct of Operations

Station operations are controlled and/or coordinated through the control room. Maintenance activities, surveillances, and removal from/return to service of structures, systems, and components affecting the operation of the plant may not commence without the approval of senior control room personnel. The rules of practice for control room activities, as described by administrative procedures, which are based on Regulatory Guide 1.114, address the following:

- Position/placement of operator at the controls workstation and the expected area of the control room where the majority of the time of the control room supervisor should be spent.
- Definition and outline of “surveillance area” and requirement for continuous surveillance by the operator at the controls.
- Relief requirements for operator at the controls and the control room supervisor/shift supervisor in charge on shift.

In accordance with 10 CFR 50.54:

- Reactivity controls may be manipulated only by licensed operators and senior operators except as allowed for training under 10 CFR Part 55.
- Apparatus and mechanisms other than controls which may affect reactivity or power level of the reactor shall be operated only with the consent of the operator at the controls or the control room supervisor/shift supervisor.
- During operation of the facility in modes other than cold shutdown or refueling, a senior operator shall be in the control room and a licensed operator or senior operator shall be present at the controls.

13.1.2.3 Operating Shift Crews

Plant administrative procedures implement the required shift staffing. These procedures establish crews with sufficient qualified plant personnel to staff the operational shifts and be readily available in the event of an abnormal or emergency situation. The objective is to operate the plant with the required staff and to develop work schedules that minimize overtime for plant staff members who perform safety-related functions. Work hour limitations and shift staffing requirements defined by TMI Action Plan I.A.1.3 are retained in station procedures. When overtime is necessary, the provisions in the technical specifications and the plant administrative procedures apply. Shift crew staffing plans may be modified during refueling outages to accommodate safe and efficient completion of outage work in accordance with the proceduralized work hour limitations.

The minimum composition of the operating shift crew is contingent on the unit operating status. Position titles, license requirements, and minimum shift manning for various modes of operation are contained in Technical Specifications, administrative procedures, and Table 13.1-202, and illustrated in Figure 13.1-202.

13.1.2.4 Fire Brigade

The station is designed and the fire brigade organized to be dedicated when necessary with respect to firefighting activities. The fire brigade is organized to deal with fires and related emergencies that could occur. It consists of a fire brigade leader and a sufficient number of team members to be consistent with the equipment that must be put in service during a fire emergency. A sufficient number of trained and physically qualified fire brigade members are available on site during each shift. The fire brigade consists of at least five members on each shift. Members of the fire brigade are knowledgeable of building layout and system design. The assigned fire brigade members for any shift do not include the shift supervisor or any other members of the minimum shift operating crew necessary for safe shutdown of the unit. It does not include any other personnel required for other essential functions during a fire emergency. Fire brigade members for a shift are designated in accordance with established procedures at the beginning of the shift.

13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

VCS COL 18.6-1

13.1.3.1 Qualification Requirements

VCS COL 13.1-1 Qualifications of managers, supervisors, operators, and technicians of the operating organization meet the qualification requirements in education and experience for those described in ANSI/ANS-3.1-1993 (Reference 201), as endorsed and amended by Regulatory Guide 1.8.

13.1.3.2 Qualifications of Plant Personnel

Résumés and/or other documentation of qualification and experience of initial appointees to appropriate management and supervisory positions are available for review by regulators upon request after position vacancies are filled.

STD DEP 1.1-1

13.1.4 COMBINED LICENSE INFORMATION ITEM

VCS COL 13.1-1 This COL item is addressed in Subsections 13.1.1 through 13.1.3 and Appendix 13AA.

13.1.5 REFERENCES

201. American Nuclear Society, “American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” ANSI/ANS -3.1-1993.

202. U.S. Nuclear Regulatory Commission, “Generic Letter 86-04, Policy Letter, Engineering Expertise on Shift.”

203. American Nuclear Society, “American National Standard for Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants,” N18.7-1976/ANS-3.2.

VCS COL 18.6-1
VCS COL 13.1-1

Table 13.1-201 (Sheet 1 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
Executive management	chief executive officer		Chief Executive Officer	1*	-
	chief operating officer		President and Chief Operating Officer	1*	-
	chief nuclear officer		Executive Vice President, Generation/ CNO	1*	-
	executive, nuclear generation		Senior Vice President, Nuclear Operations	1*	-
			Vice President, Nuclear Plant Operations	1*	-
Nuclear support	executive, operations support		General Manager Nuclear Support Services	1*	-
			General Manager Organizational Effectiveness	1	-
Plant management	plant manager	4.2.1	General Manager Nuclear Plant Operations	1	-
Engineering	executive/manager	4.2.4	General Manager Engineering Services	1*	-
system engineering	functional manager	4.3.9	Manager, Plant Support Engineering	1*	-
	system engineer		System Engineer	16	12
engineering programs	functional manager/	4.3.9			-
	programs engineer		Programs Engineer	3	2
safety and engineering analysis	functional manager/ analysis engineer	4.3.9	Analysis Engineer	1	-
reactor engineering	functional manager	4.3.9	Supervisor, Design Engineering	1	-

VCS COL 18.6-1
VCS COL 13.1-1

Table 13.1-201 (Sheet 2 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
	reactor engineer		Reactor Engineer	1	1
design engineering	functional manager	4.3.9	Manager, Design Engineering	1*	-
	design engineer		Design Engineer	11	6
Fire protection	supervisor	4.4	Fire Protection Engineer	1*	-
Maintenance	manager	4.2.3	Manager, Maintenance Services	1	-
instrumentation and control	functional manager	4.3.4	I&C/Plant Support Supervisor	1	-
	supervisor	4.4.7	Supervisor, Maintenance	5	-
	technician	4.5.3.3	Instrumentation and Control Technician	20	15
mechanical	functional manager	4.3.6	Supervisor Maintenance, Mechanical	1	-
	supervisor	4.4.9	Supervisor, Maintenance	5	-
	technician	4.5.7.2	Mechanic	30	15
electrical	functional manager	4.3.5	Supervisor Maintenance, Electrical	1	-
	supervisor	4.4.8	Supervisor, Maintenance	5	-
	technician	4.5.7.1	Electrician	20	10
support	functional manager	4.3	Supervisor, Maintenance	1	-
Operations	manager	4.2.2	Manager, Operations	1	-
operations, plant	functional manager	4.3.8	Operations Supervisor	1	1
operations, admin	functional manager	4.3.8	Operations Support Supervisor	1	-
operations, (on-shift)	functional manager	4.4.1	Shift Supervisor	5	5
	supervisor	4.4.2	Control Room Supervisor	5	5
	licensed operator	4.5.1	Reactor Operator	15	15

VCS COL 18.6-1
VCS COL 13.1-1

Table 13.1-201 (Sheet 3 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
	non-licensed operator	4.5.2	Auxiliary Operator	25	25
	shift technical advisor	4.6.2	Shift Technical Advisor	5	5
Radiation protection	functional manager	4.3.3	Manager, HP and Safety Services	1*	-
	supervisor	4.4.6	Health Physics Supervisor	6	-
	technician	4.5.3.2	Health Physics Specialist	20	13
	ALARA specialist		Health Physics Specialist	3	2
Operations - rad waste	supervisor	4.4	Rad Waste Supervisor	1	-
Chemistry	functional manager	4.3.2	Manager Chemistry	1*	-
	supervisor	4.4.5	Chemistry Supervisor	2	-
	technician	4.5.3.1	Chemistry Specialist	12	8
Nuclear licensing	manager/functional manager	4.3	Manager, Nuclear Licensing	1*	-
	supervisor		Supervisor, Nuclear Licensing	2*	-
	licensing engineer		Licensing Engineer	1	1
Corrective action	functional manager	4.3	Supervisor, Corrective Action	1	-
	corrective action specialist		Corrective Action Specialist	4	1
Emergency preparedness	functional manager	4.3	Manager, Emergency Services	1*	-
	EP planner		Emergency Planning Specialist	2	-
Training	functional manager	4.3.1	Manager, Nuclear Training	1*	-
	supervisor ops trng	4.4.4	Operations Training Supervisor	1	1
	ops training instructor		Nuclear Training Instructor	6	2

VCS COL 18.6-1
VCS COL 13.1-1

Table 13.1-201 (Sheet 4 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
	supervisor tech staff/ maint trng		Nuclear Craft/Technical Supervisor	1	-
	tech staff/maint instructors		Nuclear Technical Instructor	6	3
Purchasing, and contracts	functional manager	4.3	Manager, Business and Financial Services	1*	-
Security	functional manager	4.3	Manager, Nuclear Protection Services	1*	-
Planning and scheduling	functional manager	4.3	Manager, Planning/Outage	1	-
	supervisor	4.4	Supervisor Planning and Scheduling	1	-
Quality assurance	functional manager	4.3.7	Manager, Quality Systems	1*	-
	supervisor	4.4.13	Quality Assurance Supervisor	1	-
	QA auditor		Surveillance Specialist	3	3
	supervisor	4.4.13	Supervisor, Quality Control	1	-
	QC inspector		Inspector	4	2
Startup testing	supervisor	4.4.11	Startup Testing Supervisor	1	-
	startup test engineer		Startup Test Engineer	20	-
	supervisor	4.4.12	PT&O Support Supervisor	1	-
	preop test engineer		PT&O Engineer	6	-

*The number indicated is the total for the nuclear organization.

VCS COL 18.6-1
VCS COL 13.1-1

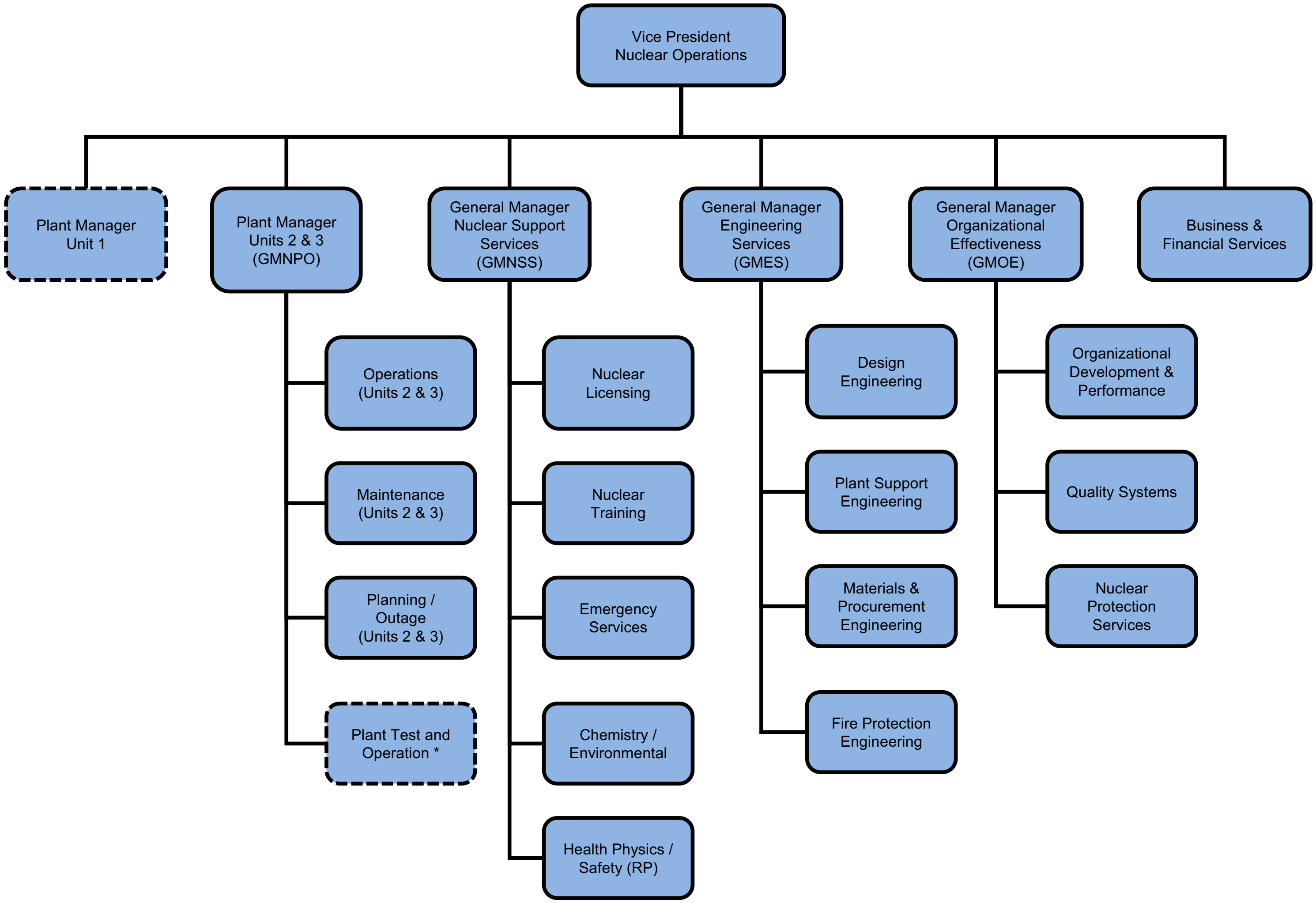
Table 13.1-202
Minimum On-Duty Operations Shift Organization for Two-Unit Plant

Units Operating	Two Units Two Control Rooms
All Units Shutdown	1 shift supervisor (SRO) 2 RO 3 nonlicensed operator
One Unit Operating[a]	1 shift supervisor (SRO) 1 SRO 3 RO 3 nonlicensed operator
Two Units Operating[a]	1 shift supervisor (SRO) 2 SRO 4 RO 4 nonlicensed operator
SRO – Licensed Senior Reactor Operator	RO – Licensed Reactor Operator

(a) Operating modes other than cold shutdown or refueling.

Notes:

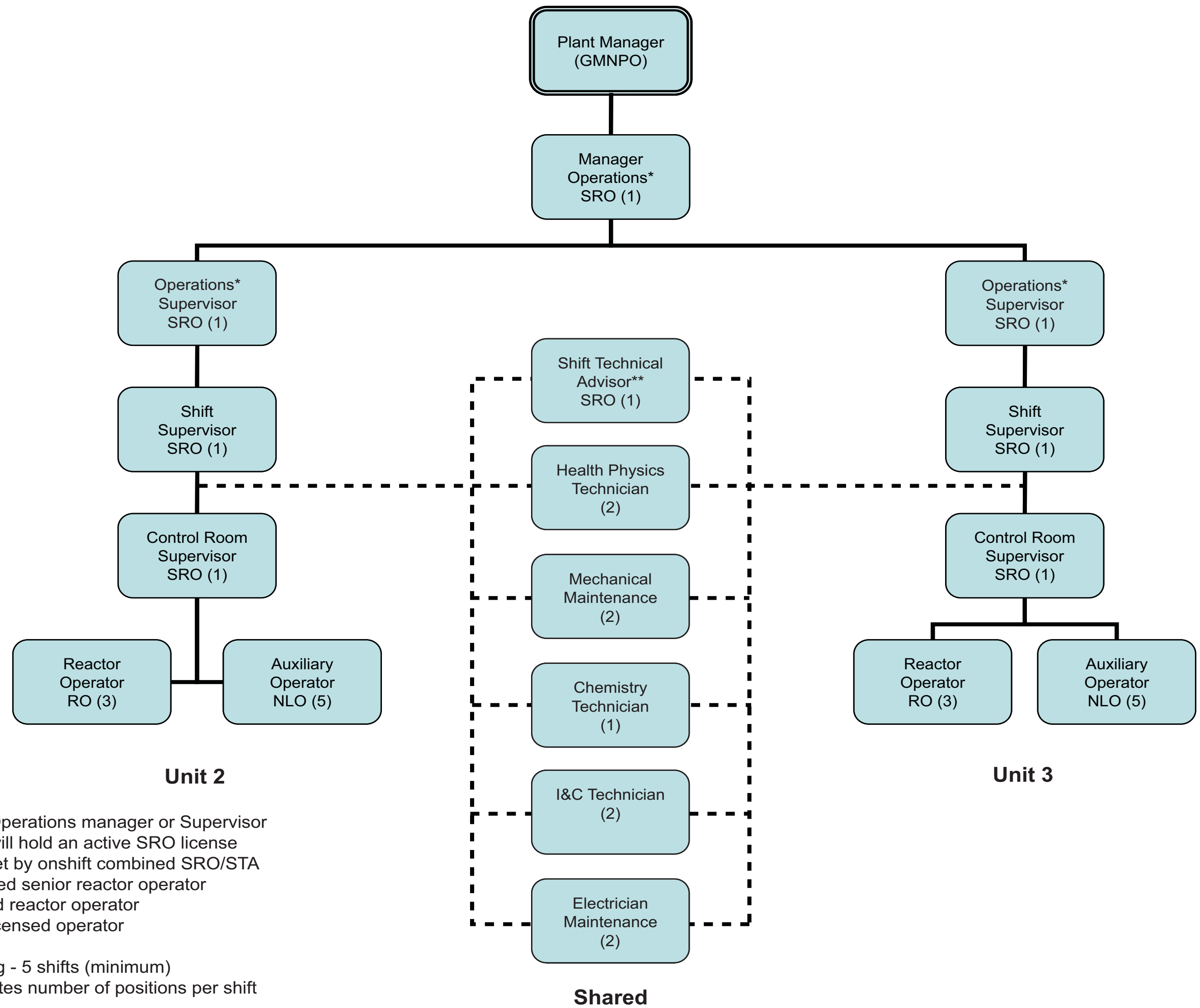
1. In addition, one STA is assigned per shift during plant operation. A shift supervisor or another SRO on shift, who meets the qualifications for the combined Senior Reactor Operator/Shift Technical Advisor position, as specified for option 1 of Generic Letter 86-04, (Reference 202) the commission's policy statement on engineering expertise on shift, may also serve as the STA. If this option is used for a shift, then the separate STA position may be eliminated for that shift.
2. In addition to the minimum shift organization above, during refueling, a licensed SRO or SRO limited (fuel handling only) is required to directly supervise any core alteration activity.
3. A shift supervisor (SRO licensed for each unit that is fueled), shall be on site at all times when at least one unit is loaded with fuel.
4. An RP technician shall be on site at all times when there is fuel in a reactor.
5. A chemistry technician shall be on site during plant operation in modes other than cold shutdown or refueling.
6. To operate, or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each unit.



*During construction the functional manager, PT&O, reports to the VPNND. As the organization transitions into the operational phase the functional manager, PT&O reports to the Plant Manager

VCS COL 13.1-1

Figure 13.1-201
Plant Management Organization

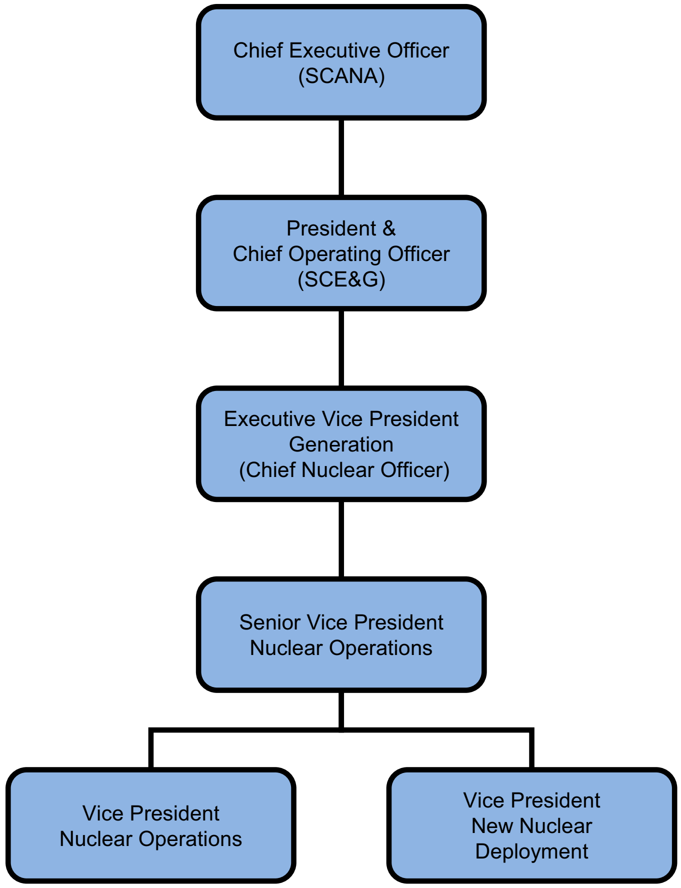


* either the Operations manager or Supervisor Operations will hold an active SRO license
** may be met by onshift combined SRO/STA
SRO - licensed senior reactor operator
RO - licensed reactor operator
NLO - non-licensed operator

Shift Manning - 5 shifts (minimum)
(No.) - indicates number of positions per shift

VCS COL 13.1-1

Figure 13.1-202
Shift Operations



VCS COL 13.1-1

Figure 13.1-203
Corporate Organization

13.2 TRAINING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 13.2-1 This section incorporates by reference NEI 06-13A, Template for an Industry Training Program Description. See Table 1.6-201.

Table 13.4-201 provides milestones for training implementation.

STD COL 18.10-1 Operators involved in the Human Factors Engineering Verification and Validation (V&V) Program receive additional training specific to the task of performing V&V. A systematic approach to training is incorporated in developing this training program along with input from WCAP-14655, Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel (Reference 201).

13.2.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.2-1 This COL Item is addressed in Section 13.2.

13.2.2 REFERENCES

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201. Westinghouse, "Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel," WCAP-14655, Revision 1, August 1996.

13.3 EMERGENCY PLANNING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 13.3-1 The emergency planning information is submitted to the Nuclear Regulatory Commission as a separate licensing document and is incorporated by reference (see Table 1.6-201).

Post-72 hour support actions, as discussed in DCD Subsections 1.9.5.4 and 6.3.4, are addressed in DCD Subsections 6.2.2, 8.3, and 9.1.3. Provisions for establishing post-72 hour ventilation for the main control room, instrumentation and control rooms, and dc equipment rooms are established in operating procedures.

STD COL 13.3-2 The emergency plan describes the plans for coping with emergency situations, including communications interfaces and staffing of the emergency operations facility.

STD SUP 13.3-1 Table 13.4-201 provides milestones for emergency planning implementation.

13.3.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.3-1 This COL Item is addressed in Section 13.3.

STD COL 13.3-2 This COL Item is addressed in Section 13.3 and in the Emergency Plan.

13.4 OPERATIONAL PROGRAMS

This section of the referenced DCD is incorporated by reference with the following departures and /or supplements.

STD COL 13.4-1 Operational programs are specific programs that are required by regulations. Table 13.4-201 lists each operational program, the regulatory source for the program, the section of the FSAR in which the operational program is described, and the associated implementation milestone(s).

13.4.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.4-1 This COL Item is addressed in Section 13.4.

13.4.2 REFERENCES

201. ASME Boiler and Pressure Vessel Code (B&PVC), “Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components.”

202. ASME “OM Code for the Operation and Maintenance of Nuclear Power Plants.”

STD COL 13.4-1

Table 13.4-201 (Sheet 1 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
1.	Inservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	Prior to Commercial service	10 CFR 50.55a(g): ASME XI IWA 2430(b) (Reference 201)
2.	Inservice Testing Program	10 CFR 50.55a(f); 10 CFR Part 50, Appendix A	3.9.6, 5.2.4	After generator online on nuclear heat[a]	10 CFR 50.55a(f), ASME OM Code (Reference 202)
3.	Environmental Qualification Program	10 CFR 50.49(a)	3.11	Prior to initial fuel load	License Condition
4.	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	Completion prior to initial plant start-up	10 CFR 50.55a(g); ASME XI IWB-2200(a) (Reference 201)
5.	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR Part 50, Appendix H	5.3.2.6	Prior to initial criticality	License Condition
6.	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	Prior to initial fuel load	License Condition
7.	Containment Leakage Rate Testing Program	10 CFR 50.54(o); 10 CFR 50, Appendix A (GDC 52); 10 CFR 50, Appendix J	6.2.5.1	Prior to initial fuel load	License Condition
8.	Fire Protection Program	10 CFR 50.48	9.5.1.8	Prior to receipt of fuel onsite Prior to initial fuel load	License Condition
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		Prior to initial receipt of byproduct, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)

STD COL 13.4-1

Table 13.4-201 (Sheet 2 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
9.	Process and Effluent Monitoring and Sampling Program:				
	Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls	10 CFR 20.1301 and 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I, Section II and IV	11.5	Prior to initial fuel load	License Condition
	Offsite Dose Calculation Manual	Same as above	11.5	Prior to initial fuel load	License Condition
	Radiological Environmental Monitoring Program	Same as above	11.5	Prior to initial fuel load	License Condition
	Process Control Program	Same as above	11.4	Prior to initial fuel load	License Condition
10.	Radiation Protection Program (including ALARA principle)	10 CFR 20.1101 10 CFR 20.1406	12.1 12.5		License Condition
	• Radioactive Source Control (assignment of RP Supervisor) • Assignment of RP Supervisor • Minimization of Contamination			1. Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	
	• Personal Dosimetry • Radiation Monitoring and Surveys • Radiation Work Permits			2. Prior to receipt of fuel onsite	

STD COL 13.4-1

Table 13.4-201 (Sheet 3 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	• Assignment of RP Manager • Respiratory Protection • Bioassay • Effluents and Environmental Monitoring and Assessment • Job Coverage			3. Prior to initial fuel load	
	• Radioactive Waste Shipping			4. Prior to first shipment of radioactive waste	
11.	Non Licensed Plant Staff Training Program	10 CFR 50.120	13.2	18 months prior to scheduled date of initial fuel load	10 CFR 50.120(b)
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		Prior to initial receipt of byproduct, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)
12.	Reactor Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2	18 months prior to scheduled date of initial fuel load	License Condition
13.	Reactor Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(i); 10 CFR 55.59	13.2	Within 3 months after the date the Commission makes the finding under 10 CFR 52.103(g)	10 CFR 50.54 (i-1)

STD COL 13.4-1

Table 13.4-201 (Sheet 4 of 7) Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
14.	Emergency Planning	10 CFR 50.47; 10 CFR 50, Appendix E	13.3	Full participation exercise conducted within 2 years of scheduled date for initial loading of fuel.	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				Onsite exercise conducted within 1 year before the schedule date for initial loading of fuel	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				Applicant's detailed implementing procedures for its emergency plan submitted at least 180 days prior to scheduled date for initial loading of fuel	10 CFR Part 50, Appendix E, Section V
VCS SUP 13.4-1	Emergency Response Data System (ERDS) Implementation Program	10 CFR 50, Appendix E		Applicant's ERDS Implementation program submitted at least 180 days prior to scheduled date for initial loading of fuel	10 CFR Part 50, Appendix E, Section VI.4.a
15.	Security Program:				
	Physical Protection Program (applicable to protection of special nuclear material prior to the protected area being delcared operational)	10 CFR 73.1 10 CFR 73.67	13.5.2.2.8 13.6	Prior to initial receipt of special nuclear material	10 CFR 73.1(a) 10 CFR 73.67
	Physical Security Program	10 CFR 73.55(b); 10 CFR 73.55(c)(3); 10 CFR 73.56; 10 CFR 73.57;	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)

STD COL 13.4-1

Table 13.4-201 (Sheet 5 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	Safeguards Contingency Program	10 CFR 73.55(c)(5); 10 CFR 73.55(k); 10 CFR Part 73, Appendix C	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
	Training and Qualification Program	10 CFR 73.55(c)(4); 10 CFR 73.55(d)(3); 10 CFR Part 73, Appendix B	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
16.	Quality Assurance Program – Operation	10 CFR 50.54(a); 10 CFR Part 50, Appendix A (GDC 1); 10 CFR Part 50, Appendix B	17.5	COL issuance	10 CFR 50.54(a)(1)
17.	Maintenance Rule	10 CFR 50.65	17.6	Prior to fuel load authorization per 10 CFR 52.103(g)	10 CFR 50.65(a)(1)
18.	Motor-Operated Valve Testing	10 CFR 50.55a(b)(3)(ii)	3.9.6.2.2	Prior to initial fuel load	License Condition
19.	Initial Test Program	10 CFR 50.34; 10 CFR 52.79(a)(28)	14.2	Prior to the first construction test being conducted for the Construction Test Program Prior to the first preoperational test for the Preoperational Test Program Prior to initial fuel load for the Startup Test Program	License Condition

STD COL 13.4-1

Table 13.4-201 (Sheet 6 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
20.	Fitness for Duty (FFD) Program for Construction (workers and first-line supervisors)	10 CFR 26.4(f)	13.7	Prior to initiating 10 CFR Part 26 construction activities	10 CFR Part 26, Subpart K
	FFD Program for Construction (management and oversight personnel)	10 CFR 26.4(e)	13.7	Prior to initiating 10 CFR Part 26 construction activities	10 CFR Part 26, Subparts A - H, N, and O
	FFD Program for Security Personnel	10 CFR 26.4(e)(1)	13.7	Prior to initiating 10 CFR Part 26 construction activities	10 CFR Part 26, Subparts A - H, N, and O
		10 CFR 26.4(a)(5) or 26.4(e)(1)		Prior to the earlier of: A. Licensee's receipt of SNM in the form of fuel assemblies, or B. Establishment of a protected area, or C. The 10 CFR 52.103(g) finding	10 CFR Part 26, Subparts A - I, N, and O
	FFD Program for FFD Program personnel	10 CFR 26.4(g)	13.7	Prior to initiating 10 CFR Part 26 construction activities	10 CFR Part 26, Subparts A, B, D - H, N, O, and C per licensee’s discretion
	FFD Program for persons required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF)	10 CFR 26.4(c)	13.7	Prior to the conduct of the first full-participation emergency preparedness exercise under 10 CFR Part 50, App. E, Section F.2.a	10 CFR Part 26, Subparts A - I, N, and O, except for §§ 26.205 – 209

STD COL 13.4-1

Table 13.4-201 (Sheet 7 of 7)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	FFD Program for Operation	10 CFR 26.4(a) and (b)	13.7	Prior to the earlier of: A. Establishment of a protected area, or B. The 10 CFR 52.103(g) finding	10 CFR Part 26, Subparts A - I,N, and O, except for individuals listed in § 26.4(b), who are not subject to §§ 26.205 – 209
21.	Cyber Security Program	10 CFR 73.54(b); 10 CFR 73.55(b)(8); 10 CFR 73.55(c)(6)	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
22.	SNM Material Control and Accounting Program	10 CFR 74, Subpart B (§§ 74.11 – 74.19, excl. § 74.17)	13.5.2.2.9	Prior to receipt of special nuclear material	License Condition

(a) Inservice Testing Program will be fully implemented by generator on line on nuclear heat. Appropriate portions of the program are implemented as necessary to support the system operability requirements of the technical specifications.

13.5 PLANT PROCEDURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD DEP 1.1-1 DCD Subsection 13.5.1, Combined License Information, is renumbered in this FSAR section to 13.5.3.

STD COL 13.5-1 This section of the FSAR describes the administrative and other procedures which are not described in the DCD that the operating organization (plant staff) uses to conduct the routine operating, abnormal, and emergency activities in a safe manner.

The Quality Assurance Program Description (QAPD), as discussed in Section 17.5, describes procedural document control, record retention, adherence, assignment of responsibilities, and changes.

Procedures are identified in this section by topic, type, or classification in lieu of the specific title and represent general areas of procedural coverage.

Procedures are issued prior to fuel load to allow sufficient time for plant staff familiarization and to develop operator licensing examinations.

The format and content of procedures are controlled by the applicable AP1000 Writer’s Guideline.

Each procedure is sufficiently detailed for an individual to perform the required function without direct supervision, but does not provide a complete description of the system or plant process. The level of detail contained in the procedure is commensurate with the qualifications of the individual normally performing the function.

Procedures are developed consistent with guidance described in DCD Section 18.9, “Procedure Development” and with input from the human factors engineering process and evaluations.

13.5.1 ADMINISTRATIVE PROCEDURES

This section describes administrative procedures that provide administrative control over activities that are important to safety for the operation of the facility.

Procedures outline the essential elements of the administrative programs and controls as described in ANSI/ANS 3.2-1988 (Reference 201) and in Section 17.5. These procedures are organized such that the program elements are prescribed in documents normally referred to as administrative procedures. Regulatory and industry guidance for the appropriate format, content and typical activities delineated in written procedures is implemented as appropriate.

Administrative procedures contain adequate programmatic controls to provide effective interface between organizational elements. This includes contractor and owner organizations providing support to the station operating organization.

A Writer's Guideline promotes the standardization and application of human factors engineering principles to procedures. The Writer's Guideline establishes the process for developing procedures that are complete, accurate, consistent, and easy to understand and follow. The Writer's Guideline provides objective criteria so that procedures are consistent in organization, style, and content. The Writer's Guideline includes criteria for procedure content and format including the writing of action steps and the specification of acceptable acronym lists and acceptable terms to be used.

Procedure maintenance and control of procedure updates are performed in accordance with the QAPD, as discussed in Section 17.5.

The administrative programs and associated procedures developed in the pre-COL phase are described in Table 13.5-201 (for future designation as historical information).

The plant administrative procedures provide procedural instructions for the following:

- Procedures review and approval.
- Equipment control procedures - These procedures provide for control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment.
- Control of maintenance and modifications.
- Crane Operation Procedures - Crane operators who operate cranes over fuel pools are qualified and conduct themselves in accordance with ANSI B30.2 (Chapter 2-3), "Overhead and Gantry Cranes" (Reference 202).
- Temporary changes to procedures.
- Temporary procedure issuance and control.
- Special orders of a temporary or self-canceling nature.

VCS SUP 13.5-1
- Standing orders to shift personnel including the authority and responsibility of the shift supervisor, licensed senior reactor operator in the control room, control room operator and shift technical advisor.

STD COL 13.5-1

- Manipulation of controls and assignment of shift personnel to duty stations per the requirements of 10 CFR 50.54 (i), (j), (k), (l), and (m) including delineation of the space designated for the "At the Controls" area of the control room.
- Shift relief and turnover procedures.
- Fitness for Duty.
- Control Room access.
- Working hour limitations.
- Feedback of design, construction, and applicable important industry and operating experience.

VCS SUP 13.5-2

- Shift Supervisor administrative duties.

STD COL 13.5-1

- Verification of correct performance of operational activities.
- A vendor interface program that provides vendor information for safety related components is incorporated into plant documentation.
- Fire protection program implementation.
- A process for implementing the safety/security interface requirements of 10 CFR 73.58.

13.5.2 OPERATING AND MAINTENANCE PROCEDURES

13.5.2.1 Operating and Emergency Operating Procedures

This information is addressed in the DCD.

13.5.2.2 Maintenance and Other Operating Procedures

The QAPD, as described in Section 17.5, provides guidance for procedural adherence. Regulatory and industry guidance for the appropriate format, content, and typical activities delineated in written procedures is implemented as appropriate.

13.5.2.2.1 Plant Radiation Protection Procedures

The plant radiation protection program is contained in procedures. Procedures are developed and implemented for such things as: maintaining personnel exposures, plant contamination levels, and plant effluents ALARA; monitoring both external and internal exposures of workers, considering industry-accepted techniques; routine radiation surveys; environmental monitoring in the vicinity of the plant; radiation monitoring of maintenance and special work activities; evaluation of radiation protection implications of proposed modifications; establishing quality assurance requirements applicable to the radiation protection program; and maintaining radiation exposure records of workers and others.

13.5.2.2.2 Emergency Preparedness Procedures

A discussion of emergency preparedness procedures can be found in the Emergency Plan.

13.5.2.2.3 Instrument Calibration and Test Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for instrumentation calibration and testing.

13.5.2.2.4 Chemistry Procedures

Procedures provided for chemical and radiochemical control activities include the nature and frequency of sampling and analyses; instructions for maintaining fluid quality within prescribed limits; the use of control and diagnostic parameters; and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces or become sources of radiation hazards due to activation.

Procedures are also provided for the control, treatment, and management of radioactive wastes and control of radioactive calibration sources.

13.5.2.2.5 Radioactive Waste Management Procedures

Procedures for the operation of the radwaste processing systems provide for the control, treatment, and management of on-site radioactive wastes. Procedural controls are in place for radiological releases.

13.5.2.2.6 Maintenance, Inspection, Surveillance, and Modification Procedures

13.5.2.2.6.1 Maintenance Procedures

Maintenance procedures describe maintenance planning and preparation activities. Maintenance procedures are developed considering the potential impact on the safety of the plant, license limits, availability of equipment required to be operable, and possible safety consequences of concurrent or sequential maintenance, testing or operating activities.

Maintenance procedures contain sufficient detail to permit the maintenance work to be performed correctly and safely. Procedures include provisions for conducting and recording results of required tests and inspections, if not performed and documented under separate test and inspection procedures. References are made to vendor manuals, plant procedures, drawings, and other sources as applicable.

Instructions are included, or referenced, for returning the equipment to its normal operating status. Testing is commensurate with the maintenance that has been performed. Testing may be included in the maintenance procedure or be covered in a separate procedure.

The preventive maintenance program, including preventive and predictive procedures, as appropriate for structures, systems and components, prescribes the frequency and type of maintenance to be performed. An initial program based on service conditions, experience with comparable equipment and vendor recommendations is developed prior to fuel loading. The program is revised and updated as experience is gained with the equipment. To facilitate this, equipment history files are created and kept current. The files are organized to provide complete and easily retrievable equipment history.

13.5.2.2.6.2 Inspection Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for inspections.

13.5.2.2.6.3 Modification Procedures

Plant modifications and changes to setpoints are developed in accordance with approved procedures. These procedures control necessary activities associated with the modifications such that they are carried out in a planned, controlled, and orderly manner. For each modification, design documents such as drawings, equipment and material specifications, and appropriate design analyses are developed or the as-built design documents are utilized. Separate reviews are conducted by individuals knowledgeable in both technical and QA requirements to verify the adequacy of the design effort.

Proposed modification(s) which involve a license amendment or a change to Technical Specifications are processed as proposed license amendment request(s).

Plant procedures impacted by modifications are changed prior to declaring the system operable to reflect revised plant conditions; and cognizant personnel who are responsible for operating and maintaining the modified equipment are adequately trained.

13.5.2.2.7 Material Control Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for material control.

13.5.2.2.8 Security Procedures

A discussion of security procedures is provided in the Security Plan.

The Special Nuclear Material (SNM) Physical Protection Program describes the 10 CFR Part 70 required protection program in effect for the period of time during which new fuel as SNM is received and stored in a controlled access area (CAA), in accordance with the requirements of 10 CFR 73.67.

The New Fuel Shipping Plan addresses the applicable 10 CFR 73.67 requirements in the event that unirradiated new fuel assemblies or components are returned to the supplying fuel manufacturer(s) facility.

13.5.2.2.9 Special Nuclear Material (SNM) Material Control and Accounting Procedures

A material control and accounting system consisting of special nuclear material accounting procedures is utilized to delineate the requirements, responsibilities, and methods of special nuclear material control from the time special nuclear material is received until it is shipped from the plant. These procedures provide detailed steps for SNM shipping and receiving, inventory, accounting, and preparing records and reports. The Special Nuclear Material (SNM) Material Control and Accounting (MC&A) Program description is submitted to the Nuclear Regulatory Commission as a separate licensing basis document.

STD DEP 1.1-1 13.5.3 COMBINED LICENSE INFORMATION ITEM

STD COL 13.5-1 Information for this COL item is addressed in 13.5.

13.5.4 REFERENCES

201. ANSI/ANS 3.2-1988, “Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants.”

202. ANSI B30.2 (Chapter 2-3), “Overhead and Gantry Cranes.”

Table 13.5-201
Pre-COL Phase Administrative Programs and Procedures

STD COL 13.5-1 (This table is included for future designation as historical information.)

- Design/Construction Quality Assurance Program
- Reporting of Defects and Noncompliance, 10 CFR Part 21 Program
- Design Reliability Assurance Program

13.6 SECURITY

This section of the referenced DCD is incorporated by reference with the following departures and /or supplements.

STD COL 13.6-1 The Security Plan consists of the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan. The Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document in order to fulfill the requirements of 10 CFR 52.79(a)(35) and 52.79(a)(36) and is incorporated by reference (see Table 1.6-201). The Security Plan meets the requirements contained in 10 CFR Part 26 and 10 CFR Part 73 and will be maintained in accordance with the requirements of 10 CFR 52.98. The Plan is categorized as Security Safeguards Information and is withheld from public disclosure pursuant to 10 CFR 73.21.

The Cyber Security Plan is submitted to the Nuclear Regulatory Commission as a separate licensing document to fulfill the requirements contained in 10 CFR 52.79(a)(36) and 10 CFR 73.54 and is incorporated by reference (see Table 1.6-201). The Cyber Security Plan will be maintained in accordance with the requirements of 10 CFR 52.98. The Plan is withheld from public disclosure pursuant to 10 CFR 2.390.

Table 13.4-201 provides milestones for security program and cyber security program implementation.

13.6.1 COMBINED LICENSE INFORMATION ITEMS

STD COL 13.6-1 Information for the Security Plan portion of this COL Item is addressed in Section 13.6.

Information for the Physical Security ITAAC portion of this COL item is addressed in Section 14.3.2.3.2.

STD COL 13.6-5 Information for the cyber security program portion of this COL item is addressed in Section 13.6.

13.6.2 REFERENCES

201. Not used.

202. Not used.

STD DEP 1.1-1 DCD Section 13.7 is redistributed to include DCD Section 13.7 references 7, 8, and 10 with COLA FSAR Subsection 13.5.4 and DCD Section 13.7 references 2, 3, and 4 with COLA FSAR Subsection 13.6.2.

Add the following new section after DCD Section 13.6.

13.7 FITNESS FOR DUTY

STD SUP 13.7-1 The Fitness for Duty Program (FFD) is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site. In general, two different FFD programs will be implemented: a construction FFD program and an operations FFD program. The construction and operations phase programs are illustrated in Table 13.4-201.

The construction FFD program is consistent with NEI 06-06 (Reference 201). NEI 06-06 applies to persons constructing or directing the construction of safety- and security-related structures, systems, or components performed onsite where the new reactor will be installed and operated. Management and oversight personnel, as further described in NEI 06-06, and security personnel prior to the receipt of special nuclear material in the form of fuel assemblies (with certain exceptions) will be subject to the operations FFD program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and O. At the establishment of a protected area, all persons who are granted unescorted access will meet the requirements of an operations FFD program. Prior to issuance of a Combined License, the construction FFD program at a new reactor construction site for those subject to Subpart K will be reviewed and revised as necessary should substantial revisions occur to either NEI 06-06 following NRC endorsement or the requirements of 10 CFR Part 26.

VCS SUP 13.7-1 The following site-specific information is provided:

- The FFD program for the construction site, as defined in NEI 06-06, will be administered under a VCSNS-approved Shaw Stone & Webster (Shaw) program. The 10 CFR Part 26 requirements will be implemented for the construction site area based on the descriptions provided in Table 13.4-201.
- Construction Workers & First Line Supervisors (Shaw employees and subcontractors) will be covered by a VCSNS-approved Shaw FFD Program (elements Subpart K).
- SCE&G employees and SCE&G subcontractor's construction management and oversight personnel will be covered by the VCSNS Unit 1 Operations FFD Program (elements Subpart A - H, N and O) and Shaw's

employees and Shaw's subcontractors construction management and oversight personnel will be covered by a VCSNS-approved Shaw FFD Program (elements Subpart A - H, N and O).

- VCSNS security personnel will be covered by the VCSNS Unit 1 FFD Operations Program (elements Subpart A - H, N and O) and Shaw's security personnel will be covered by the VCSNS-approved Shaw FFD Program (elements Subpart A - H, N and O). This coverage is applicable from the start of construction activities to the earlier of (1) the receipt of SNM in the form of fuel assemblies, (2) the establishment of a protected area, or (3) the 10 CFR 52.103(g) finding.

- VCSNS FFD Program personnel will be covered by the VCSNS Unit 1 Operations FFD Program and Shaw's FFD Program personnel will be covered by the VCSNS-approved Shaw FFD Program (elements Subpart A, B, D-H, N, O, and C per Licensee discretion).

- VCSNS security personnel protecting fuel assemblies will be covered by the VCSNS Unit 1 Operations FFD Program (elements Subpart A - I, N and O).

- Personnel required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF) by Emergency Plans and procedures when that requirement is in effect will be covered by the VCSNS Unit 1 Operations FFD Program, except for subsections 26.205-209.

STD SUP 13.7-1 The operations phase FFD program is consistent with the applicable subparts of 10 CFR Part 26.

13.7.1 REFERENCES

201. Nuclear Energy Institute, “Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites,” NEI 06-06, Revision 5, August 2009 (ML092430016).

Add the following new appendix at the end of DCD Chapter 13.

VCS COL 13.1-1

APPENDIX 13AA CONSTRUCTION-RELATED ORGANIZATION

The information in this appendix is included for future designation as historical information. Paragraphs are numbered to be subsequent to Subsection 13.1.1.1.

13AA.1.1.1.1 Design and Construction Activities

Westinghouse was selected to design, fabricate, deliver, and install the AP1000 advanced light water pressurized water reactors (PWR) and to provide technical direction for installation and startup of this equipment. DCD Subsection 1.4.1 provides detailed information regarding Westinghouse past experience in design, development, and manufacturing of nuclear power facilities. Operating experience from design, construction, and operation of earlier Westinghouse PWRs is applied in the design, construction, and operation of the AP1000 as described in numerous locations throughout the DCD (e.g., DCD Subsections 3.6.4.4, 3.9.4.2.1, 4.2.3.1.3).

A construction architect-engineer provides the construction of the plant and additional design engineering for selected site-specific portions of the plant. The architect-engineer is selected based on experience and proven technical capability in nuclear construction projects or projects of similar scope and complexity.

Other design and construction activities are generally contracted to qualified suppliers of such services. Implementation or delegation of design and construction responsibilities is described in the subsections below. QA aspects of these activities are described in Chapter 17.

13AA.1.1.1.1.1 Principal Site-Related Engineering Work

The principal site engineering activities accomplished towards the construction and operation of the plant are:

a. Meteorology

Information concerning local (site) meteorological parameters is developed and applied by station and contract personnel to assess the impact of the station on local meteorological conditions. An onsite meteorological measurements program is employed by station personnel to produce data for the purpose of making atmospheric dispersion estimates for postulated accidental and expected routine airborne releases of effluents. A maintenance program is established for surveillance, calibration, and repair of instruments. More information regarding the study and meteorological program is found in Section 2.3.

b. Geology

Information relating to site and regional geotechnical conditions is developed and evaluated by utility and contract personnel to determine if geologic conditions could present a challenge to safety of the plant. Items of interest include geologic structure, seismicity, geological history, and groundwater conditions. During construction, foundations within the power block area are mapped or visually inspected and photographed. Section 2.5 provides details of these investigations.

c. Seismology

Information relating to seismological conditions is developed and evaluated by the utility and contract personnel to determine if the site location and area surrounding the site is appropriate from a safety standpoint for the construction and operation of a nuclear power plant. Information regarding tectonics, seismicity, correlation of seismicity with tectonic structure, characterization of seismic sources, and ground motion are assessed to estimate the potential for strong earthquake ground motions or surface deformation at the site. Section 2.5 provides details of these investigations.

d. Hydrology

Information relating to hydrological conditions at the plant site and the surrounding area is developed and evaluated by the utility and contract personnel. The study includes hydrologic characteristics of streams, lakes, shore regions, the regional and local groundwater environments, and existing or proposed water control structures that could influence flood control and plant safety. Section 2.4 includes more detailed information regarding this subject.

e. Demography

Information relating to local and surrounding area population distribution is developed and evaluated by utility and contract personnel. The data is used to determine if requirements are met for establishing exclusion area, low population zone, and population center distance. Section 2.1 includes more detailed information regarding population around the plant site.

f. Environmental Effects

Monitoring programs are developed to enable the collection of data necessary to determine possible impact on the environment due to construction, startup, and operational activities and to establish a baseline from which to evaluate future environmental monitoring.

13AA.1.1.1.1.2 Design of Plant and Ancillary Systems

Responsibility for design and construction of systems outside the power block such as circulating water, service water, switchyard, and secondary fire protection systems are delegated to qualified contractors.

13AA.1.1.1.1.3 Review and Approval of Plant Design Features

Design engineering review and approval is performed in accordance with the reactor technology vendor QA program and Section 17.1. The reactor technology vendor is responsible for design control of the power block. Verification is performed by competent individuals or groups other than those who performed the original design. Design issues arising during construction are addressed and implemented with notification and communication of changes to the manager in charge of design engineering for review. As systems are tested and approved for turnover and operation, control of design is turned over to plant staff. The manager in charge of design engineering, along with functional managers and staff, assumes responsibility for review and approval of modifications, additions, or deletions in plant design features, as well as control of design documentation, in accordance with the Operational QA Program. Design control becomes the responsibility of the manager in charge of design engineering prior to loading fuel. During construction, startup, and operation, changes to human-system interfaces of control room design are approved using a human factors engineering evaluation addressed within Chapter 18. See Organization Charts, Figure 13.1-201 and 13AA-201 for reporting relationships.

13AA.1.1.1.1.4 Site Layout With Respect to Environmental Effects and Security Provisions

Site layout was considered when determining the expected environmental effects from construction.

The Physical Security Plan is designed with provisions that meet the applicable NRC regulations. Site layout was considered when developing the Security Plan.

13AA.1.1.1.1.5 Development of Safety Analysis Reports

Information regarding the development of the Final Safety Analysis Report is found in Chapter 1.

13AA.1.1.1.1.6 Review and Approval of Material and Component Specifications

Safety-related material and component specifications of structures, systems, and components designed by the reactor technology vendor are reviewed and approved in accordance with the reactor technology vendor QA program and Section 17.1. Review and approval of items not designed by the reactor vendor are controlled for review and approval by Section 17.5 and the QA Program Document.

13AA.1.1.1.1.7 Procurement of Materials and Equipment

Procurement of materials during the construction phase is the responsibility of the reactor technology vendor and constructor. The process is controlled by the construction QA programs of these organizations. Oversight of the inspection and receipt of materials process is the responsibility of the supervisor in charge of QA.

13AA.1.1.1.1.8 Management and Review of Construction Activities

Overall management and responsibility for construction activities is assigned to the VPNND. The project director of the engineering, procurement, and construction (EPC) contractor is accountable to the VPNND for construction activities. See Organization Chart Figure 13AA-201. Monitoring and review of construction activities by utility personnel is a continuous process at the plant site. Contractor performance is monitored to provide objective data to utility management in order to identify problems early and develop solutions. Monitoring of construction activities verifies that contractors are in compliance with contractual obligations for quality, schedule, and cost. Monitoring and review of construction activities is divided functionally across the various disciplines of the utility construction staff, e.g., electrical, mechanical, instrument and control, etc., and tracked by schedule based on system and major plant components/areas.

After each system is turned over to plant staff, the construction organization relinquishes responsibility for that system. At that time they will be responsible for completion of construction activities as directed by plant staff and available to provide support for preoperational and startup testing as necessary.

Periodic assessment involving both the construction and operations organizations continues to identify SSCs that could reasonably be expected to be impacted by scheduled construction activities. Appropriate administrative and managerial controls are then established as necessary. Specific hazards, impacted SSCs, and managerial and administrative controls are reviewed on a recurring basis and, if necessary, controls are revised/developed and implemented and maintained current as work progresses on site. For example, prior to construction activities that involve the use of large construction equipment such as cranes, managerial and administrative controls are in place to prevent adverse impacts on any operating unit(s) overhead power lines, switchyard, security boundary, etc., by providing the necessary restrictions on the use of large construction equipment.

13AA.1.1.1.2 Preoperational Activities

The PT&O manager reports to the GMNPO. The plant manager, with the aid of those managers that report directly to the plant manager, (see Figure 13.1-201) is responsible for the activities required to transition the unit from the construction phase to the operational phase. These activities include turnover of systems from construction, preoperational testing, schedule management, procedure development for tests, fuel load, integrated startup testing, and turnover of systems to plant staff.

13AA.1.1.1.2.1 Development of Human Factors Engineering Design Objectives and Design Phase Review of Proposed Control Room Layouts

Human factors engineering (HFE) design objectives are initially developed by the reactor technology vendor in accordance with Chapter 18 of the FSAR and the Design Control Document (DCD). As a collaborative team, personnel from the reactor technology vendor design staff and personnel, including, licensed operators, engineers, and instrumentation and control technicians from owner and other organizations in the nuclear industry, assess the design of the control room and man-machine interfaces to attain safe and efficient operation of the plant. See Section 18.2 for additional details of HFE program management.

Modifications to the certified design of the control room or man-machine interface described in the DCD are reviewed in accordance with engineering and site support procedures, as required by Section 18.2, to evaluate the impact to plant safety. The manager in charge of design engineering is responsible for the HFE design process and for the design commitment to HFE during construction and throughout the life of the plant as noted in Subsection 13.1.1.2.1. The HFE program is established in accordance with the description and commitments in Chapter 18.

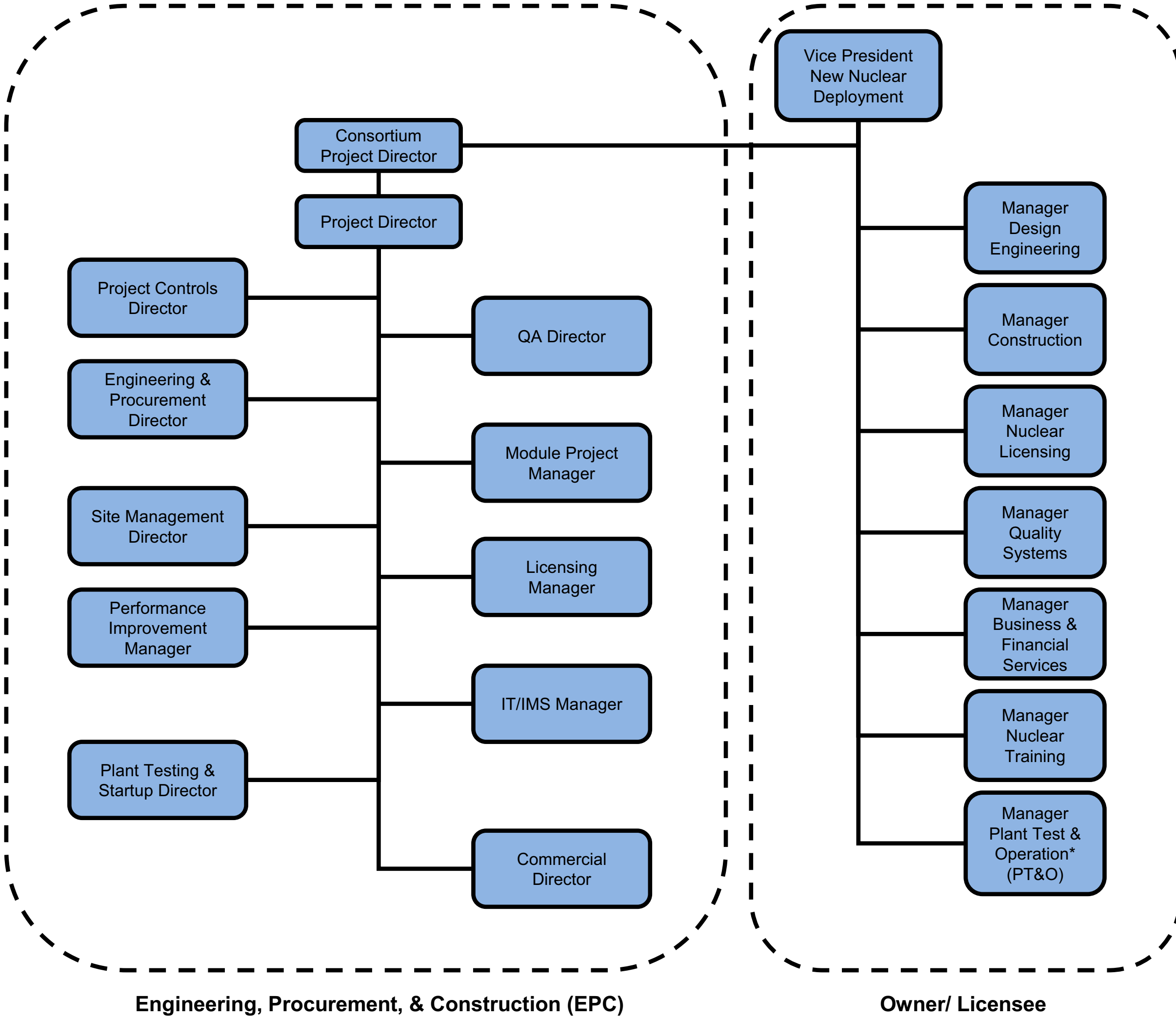
13AA.1.1.1.2.2 Preoperational and Startup Testing

Preoperational and startup testing is conducted by the plant test and operations (PT&O) organization. The plant test and operations organization, PT&O functions, and responsibilities are addressed in Section 14.2. Sufficient numbers of personnel are assigned to perform preoperational and startup testing to facilitate safe and efficient implementation of the testing program. Plant-specific training provides instruction on the administrative controls of the test program. To improve operational experience, operations and technical staff are used as support in conducting the test program and in reviewing test results.

13AA.1.1.1.2.3 Development and Implementation of Staff Recruiting and Training Programs

Staffing plans are developed based on operating plant experience with input from the reactor technology vendor for safe operation of the plant as determined by HFE. See Section 18.6. These plans are developed under the direction and guidance of the VPNND and VPNO. Staffing plans are completed and manager level positions are filled prior to start of preoperational testing. Personnel selected to be licensed ROs and SROs along with other staff necessary to support the safe operation of the plant are hired with sufficient time available to complete appropriate training programs, and become qualified, and licensed, if required, prior to fuel being loaded in the reactor vessel. See Figure 13AA-202 for an estimated timeline of hiring requirements for operator and technical staff relative to fuel load.

Because of the dynamic nature of the staffing plans and changes that occur over time, it is expected that specific numbers of personnel on site will change; however, Table 13.1-201 includes the initial estimated number of staff for selected positions and the estimated number of additional positions required for a second unit. Recruiting of personnel to fill positions is the shared responsibility of the functional manager in charge of human resources and the various heads of departments. The training program is described in Section 13.2.



*During construction the functional manager, PT&O, reports to the VPNND. As the organization transitions into the operational phase the functional manager, PT&O reports to the Plant Manager (Figure 13.1-201)

VCS COL 13.1-1

Figure 13AA-201
Construction Management Organization

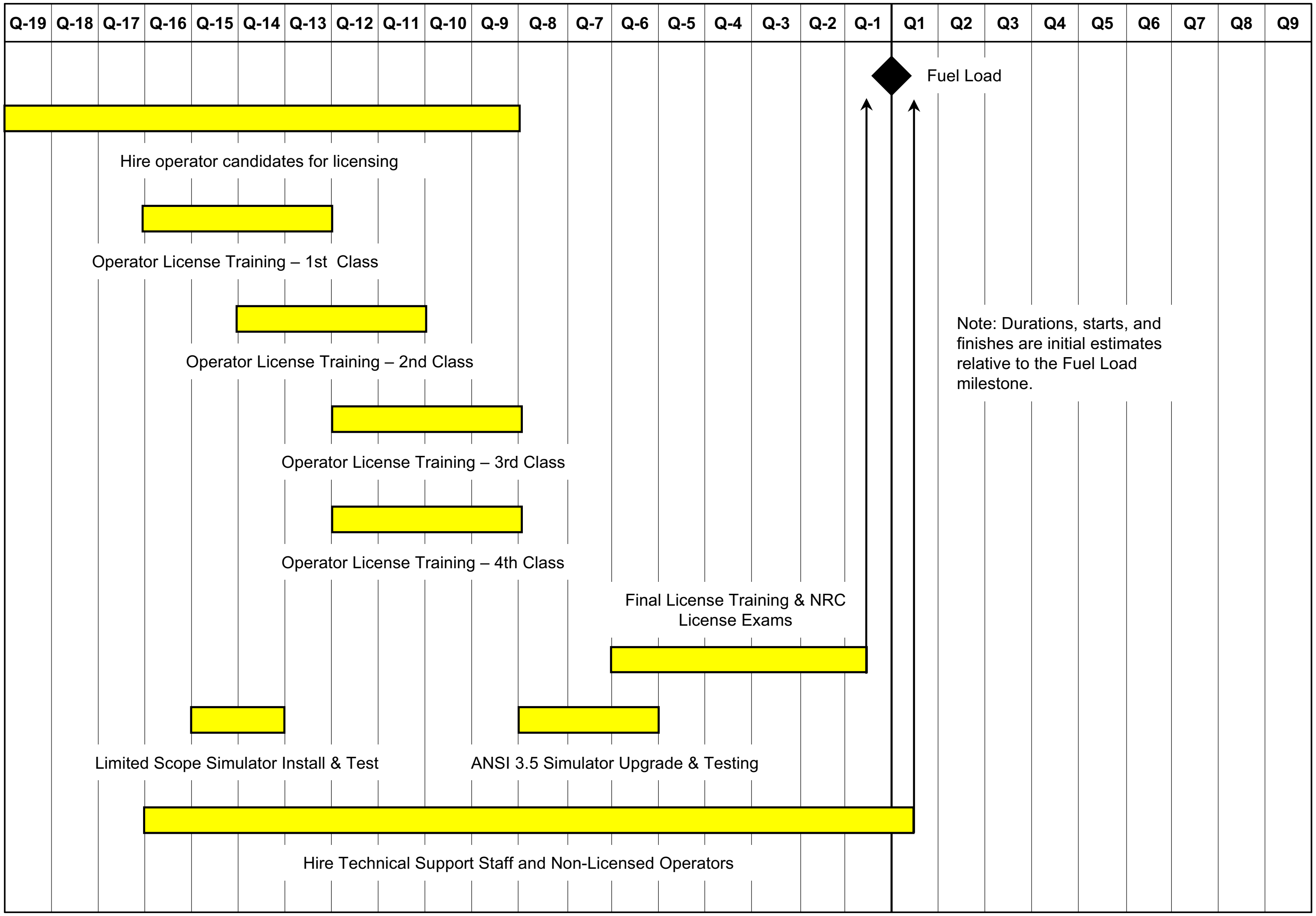


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Figure 13AA-202
Hiring Schedule for Plant Staff

CHAPTER 14
INITIAL TEST PROGRAM

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INITIAL TEST PROGRAM

14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PRELIMINARY/FINAL SAFETY ANALYSIS REPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

14.2 SPECIFIC INFORMATION TO BE INCLUDED IN STANDARD SAFETY ANALYSIS REPORTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements

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14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

Add the following subsection at the end of DCD Subsection 14.2.1:

STD COL 14.4-3 FSAR Section 14.2 provides the requirements to be included in the Startup Administrative Manual (Procedures), as discussed in DCD Subsection 14.4.3. The information referenced in this section meets the Initial Test Program (ITP) criteria of NUREG-0800 and is formatted to follow Regulatory Guide 1.206, Part I, Section C.I.14.2.

The ITP is applied to structures, systems, and components that perform the functions described in the Regulatory Guide 1.68 evaluation in FSAR Section 1.9. The ITP is also applied to other structures, systems and components. The Startup Administrative Manual includes a list of the AP1000 structures, systems and components to which the ITP is applied.

Add the following Subsections after DCD Subsection 14.2.1.3

STD COL 14.4-3

14.2.1.4 Testing of First of a Kind Design Features

First of a kind (FOAK) testing may occur in any of the phases, depending on the nature of the testing and required sequencing of the tests. When testing FOAK design features, applicable operating experience from previous test performance on other AP1000 plants is reviewed, where available, and the ITP modified as needed based on those lessons learned.

14.2.1.5 Credit for Previously Performed Testing of First of a Kind Design Features

In some cases, FOAK testing is required only for the first of a new design or for the first few plants of a standard design. In such cases, credit may be taken for the previously performed tests. A discussion is included in the startup test reports of the results of those tests that are credited.

14.2.2 ORGANIZATION, STAFFING, AND RESPONSIBILITIES

Replace the existing information in DCD Subsection 14.2.2 with the following new paragraph and subsections.

STD COL 14.4-1 The AP1000 plant test and operations (PT&O) organization is described in Subsection 14.2.2.1. The organization for operating and maintaining the AP1000 plant is described in Section 13.1.

The PT&O organization structure (organizational chart) is included in the Startup Administrative Manual.

Table 13.4-201 provides milestones for initial test program implementation.

14.2.2.1 PT&O Organization

The Initial Test Program (ITP) is the responsibility of the PT&O Organization. The ITP includes three phases of testing:

- Construction and Installation Testing
- Preoperational Testing
- Startup Testing

14.2.2.1.1 Manager In Charge of PT&O

The manager in charge of PT&O reports directly to the plant manager. The manager in charge of PT&O manages the ITP. The manager in charge of PT&O is responsible for:

- Staffing the PT&O Organization.
- Developing, reviewing, and approving the administrative and technical procedures associated with the preoperational and startup phases.
- Managing the ITP and personnel.
- Implementing the ITP schedule.
- Managing contracts associated with the ITP.

14.2.2.1.2 Functional Manager In Charge of PT&O Support

The functional manager in charge of PT&O support reports directly to the manager in charge of PT&O. The functional manager in charge of PT&O support plans and schedules procedure development to support startup. The functional manager in charge of PT&O support verifies that the test documents conform to the approved project procedures.

The functional manager in charge of PT&O support reviews and approves test procedures. These procedures are used to demonstrate that a system and its components meet the design and performance criteria.

14.2.2.1.3 PT&O Engineers

The PT&O engineers report directly to the functional manager in charge of PT&O support. The PT&O engineers are responsible for developing system test procedures.

14.2.2.1.4 Functional Manager In Charge of Startup

The functional manager in charge of startup reports directly to the manager in charge of PT&O. The functional manager in charge of startup manages the preoperational and startup testing. The functional manager in charge of startup is responsible for:

- Participating in the Joint Test Working Group (JTWG) and ensuring that the JTWG reviews and approves administrative and test procedures. The JTWG structure and responsibilities are defined in Subsection 14.2.2.3.
- Preparing a detailed preoperational and startup testing schedule.
- Coordinating construction turnover to the PT&O organization.
- Informing the functional manager in charge of PT&O when vendor support essential to preoperational and startup testing is required, and coordinating vendor participation.
- Supervising and directing the startup engineers.
- Involving operations personnel in testing activities. Utilizing operations personnel, to the extent practical, as test witnesses or test performers to provide the operations personnel with experience and knowledge.
- Developing and implementing administrative controls to address system and equipment configuration control.
- Maintaining the startup schedule.
- Maintaining a daily startup log and issuing periodic progress reports that identify overall progress and potential challenges.

14.2.2.1.5 Startup Engineers

The startup engineers report directly to the functional manager in charge of startup.

The startup engineers are responsible for:

- Complying with administrative controls.
- Identifying any special or temporary equipment or services needed to support testing.
- Coordinating testing with involved groups.
- Reviewing and evaluating test results.

14.2.2.2 PT&O Organization Personnel Qualifications and Training

Procedures are prepared to confirm that test personnel have adequate training, qualification and certification. Records are kept for extent of experience, involvement in procedure and test development, training programs, and level of qualification. The training organization qualifies Test Personnel as applicable, in accordance with the requirements of the applicable Quality Assurance Program. Training is performed as agreed between Westinghouse and the Licensee. Westinghouse test personnel training is per certified design.

Acceptable qualifications of non-supervisory test engineers follow the guidance provided in Regulatory Guide 1.28 as discussed in Appendix 1AA, i.e., ASME NQA-1-1994, Appendix 2A-1, Nonmandatory Guidance on the Qualification of Inspection and Test Personnel.

The training program/procedures shall include:

- The education, training, experience, and qualification requirements of supervisory personnel, test personnel, and other major participating organizations responsible for managing, developing, or conducting each test phase, or development of testing, operating, and emergency procedures.
- The establishment of a training program for each organizational unit, with regard to the scheduled preoperational and initial startup testing. This training program provides meaningful technical information beyond that obtained in the normal startup test program and provide supplemental operator training. This program also satisfies the criteria described in TMI Action Plan Item I.G.1 of NUREG-0660 and NUREG-0737.

The Startup Administrative Manual (Procedure) shall include:

- The implementation of measures to verify that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested. This provision does not preclude members of the design organization from participating in test activities. This description also includes considerations of staffing effects that could result from overlapping initial test programs at multi-unit sites.

14.2.2.3 Joint Test Working Group

The Joint Test Working Group (JTWG) consists of an organizational group of authorized representative personnel from the Plant's operations and support group functions, Westinghouse Electric Company (WEC), Architect Engineer and other test support groups as identified below.

The Licensee has the overall responsibility for conduct of the Startup Test Program. The Westinghouse Startup Manager may be assigned overall responsibility and authority for technical direction of the Startup Test Program and may act as the JTWG Chairman.

The JTWG Chairman reports to the Chairman of the Plant Owner's Operations Review Committee (PORC) or qualified designee for matters of Startup test authority and acceptance.

The JTWG provides the following administrative oversight activities associated with the Startup Test Program:

- Review, evaluate and approve Startup Test Program administrative and test procedures.

- Oversee the implementation of the Preoperational Test Program and the Startup Test Program, including planning, scheduling and performance of Preoperational and Startup testing.

- Review and evaluate Construction, Preoperational and Startup test results and test turnover packages.

At a minimum, the JTWG is composed of qualified representatives provided from the following organizations:

- Licensee's Operations Group

- Licensee's Maintenance Group

- Site Preoperational Test Group

- Site Startup Test Group

- Licensee's Engineering Group

- Licensee's Corrective Action Organization
- Westinghouse Site Engineering Group
- Licensee's Health Physics/Chemistry Group
- Licensee's Quality Assurance Group

The following are additional generic details of the key responsibilities, authorities and interfaces of the Licensee Organizations delineated above:

- Operations Group

  The Operations Group has the overall responsibility for Plant Operations, including administrative control and tag-outs subsequent to system turnover. Their primary interfaces are with the Licensee Engineering and Technical Support organizations as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Maintenance Group

  The Maintenance Group has the overall responsibility for the Maintenance of Plant systems and components subsequent to System Turnover. They are key participants and maintainers of system maintenance control and tag-outs. Their primary interfaces are with the Licensee Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Corrective Action Organization

  The Corrective Action Organization may be an organization specific to itself or may be a part of the Performance Assessment organization, the Quality Organization or another organization. This organization, together with every other site organization, is responsible for the administration and management of the corrective action program, as well as the identification of conditions adverse to quality. This organization interfaces with site organizations and identifies and documents conditions which need to be documented in the corrective action program.

- Engineering Group

  This group has the primary responsibility for site engineering and design oversight of the plant components and systems, as well as interfacing with the vendor engineering organization. This organization primarily interfaces with the Operations Group as well as the Westinghouse Site Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group. The responsibility for training the testing personnel in accordance with applicable Quality Assurance Program is delegated and implemented as agreed to by Westinghouse. Westinghouse test personnel training is per certified design.

- Health Physics/Chemistry Group

  This Technical Support organization has the responsibility and authority to maintain Health Physics and system chemistry conditions at the plant, particularly after system turnover. This organization primarily interfaces with the Licensee Operations Group, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Quality Assurance Group

  This group has the responsibility to verify that the applicable site Quality commitments are met within the scope of work performed at the site. This includes meeting the Criteria of 10 CFR 50 Appendix B. The primary interfaces for this group are the Licensee Operations Group and Technical Support organizations, including Quality Control and other quality organizations, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Site Preoperational Test Group

  This group has the primary responsibility for the development, maintenance and performance of the site preoperational procedures at the site. The primary interfaces for this group are the Licensee Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Startup Testing Teams and the Construction Services Group. Additional specific information regarding this organization's responsibilities and interfaces is described in Subsection 14.2.2.5, below. Once preoperational testing is complete, this group turns systems over to the Startup Group.

- Site Startup Test Group

  This group has the primary responsibility for the development, maintenance and performance of the site startup procedures at the site. The primary interfaces for this group are the Licensee's Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Preoperational Testing Team and the Construction Services Group. Additional specific information regarding this organization's responsibilities and interfaces is described in Subsection 14.2.2.6, below. The Startup Test Group turns over systems to the licensee when testing is complete.

- Westinghouse Site Engineering Group

  This group has the primary responsibility for the vendor interface between the site and the vendor's home offices, as well as the design authority for the primary vendor's components and systems. The various Westinghouse site leads for specific disciplines are a part of this organization. This organization primarily interfaces with Licensee Operations Group, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group. The responsibility for training the testing personnel in accordance with the applicable Quality Assurance Program is delegated and implemented as agreed to by Westinghouse and the Licensee. Westinghouse test personnel training is per certified design.

14.2.2.4 Site Construction Group (Architect Engineer)

The Site Construction Group consists of the following, as necessary to support the Site Startup Test Program:

- Construction Group

  The Construction group has the primary responsibility for the construction and construction testing of the Balance of Plant (BOP) engineering systems and components. During Construction and Construction Testing, this group has authority over administrative control and tagouts of these systems. Their main interface is with the System Preoperational and Startup Testing Groups, as well as the Licensee Operations Group. The Construction Group is responsible for addressing open items in the system turnover punch lists to address turnover acceptability of the system.

- Construction Services Group

  The Construction Services Group primarily supports the Construction Group with activities necessary to support construction of systems and testing of the BOP systems and components, including the construction of scaffolding, installation and removal of insulation, and similar activities. With agreement between the necessary parties, this group may also support the Westinghouse Site Engineering Group with similar activities on the primary side. The primary interfaces of this group are the Construction Group and the organizations of the JTWG.

- Construction Services Procurement Group

  The Construction Services Procurement Group is responsible for the quality procurement of components and equipment necessary to support plant construction and testing. The primary interfaces of this group include the Construction Services Group and the Construction Services Quality Group.

- Construction Services Quality Group

  The Construction Services Quality Group is responsible for the oversight of the Quality Program during Construction Activities, including those pertinent to 10 CFR 50 Appendix B and the disposition of Significant Construction Deficiencies, 10 CFR 50.55(e) reports as necessary. This group primarily interfaces with the Construction and Services groups as well as the Westinghouse Site Engineering group and the JTWG.

- Construction Services Training Group

  This group is primarily responsible for the training and qualification of Site Construction Personnel in accordance with the applicable Quality Assurance Program. Their primary interface is with the qualified Construction personnel.

The Site Construction Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Construction Installation and Testing, including management of construction testing documentation.
- Construction and Installation activities required to support Preoperational and Startup Test Programs.
- Vendor interface and procurement associated with supporting testing activities.
- Provide staffing as needed to support the testing activities.
- Turnover of Construction and Installation tested equipment, systems, and testing documentation to the Site Preoperational Test Group.

14.2.2.5 Site Preoperational Test Group

The Site Preoperational Test Group consists of the following, as necessary to support the Site Startup Test Program:

- Engineering Leads
- Preoperational Test Teams

The Site Preoperational Test Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Coordinate tagging and maintenance prior to turnover to the Licensee to support system acceptance testing.
- Accept systems for turnover from the installation organization.

- Plan, scope and schedule plant systems for test to support the plant Preoperational Test Program.
- Manage and oversee the testing of plant systems to support the Plant Hot-Functional Test Program.
- Resolve open items and exceptions identified during implementation of the Preoperational Test Program.
- Accept and turn over Preoperational Test Packages to the Site Licensee.
- Support completion of Hot-Functional Test Program.
- Coordinate other support tasks required during Startup Testing activities with responsible groups (e.g., Licensee's Organization).

14.2.2.6 Site Startup Test Group

The Site Startup Test Group consists of the following, as necessary to support the Site Startup Test Program:

- Engineering Leads
- Startup Test Teams

The Site Startup Test Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Coordinate tagging and maintenance as required to support system and equipment acceptance testing.
- Accept systems, structures and components from the Licensee for integrated testing.
- Plan, scope and schedule plant systems, structures and components for testing, to support Plant Startup.
- Manage and oversee the testing of plant systems, structures and components to support the Plant Power Ascension Test Program.
- Resolve open items and exceptions identified during implementation of the Startup Test Program.
- Accept and turn over Startup Test Packages to the Site Licensee.
- Coordinate other support tasks required during Startup Testing activities with responsible groups (e.g., Licensee's Organization).

14.2.3 TEST SPECIFICATIONS AND TEST PROCEDURES

Add the following text at the end of DCD Subsection 14.2.3:

STD COL 14.4-3 The Startup Administrative Manual shall include the following controls:

- Controls to provide test procedures that include appropriate prerequisites, objectives, safety precautions, initial test conditions, methods to direct and control test performance, and acceptance criteria by which the test is evaluated.
- Controls for the format of individual test procedures to provide consistency with the guidance contained in RG 1.68; or provide justifications for any exceptions.
- Controls to provide for participation of the principal design organizations in establishing test objectives, test acceptance criteria, and related performance requirements during the development of detailed test procedures. Each test procedure should include acceptance criteria that account for the uncertainties used in transient and accident analyses. The participating system designers should include the nuclear steam supply system vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable.
- Controls to provide for personnel with appropriate technical backgrounds and experience to develop and review test procedures. Persons filling designated management positions should perform final procedure review and approval.
- Controls to make the approved test procedures for satisfying FSAR testing commitments are made available to the NRC inspectors approximately 60 days prior to their intended use.

14.2.3.1 Conduct of Test Program

Add the following text and Subsection at the end of DCD Subsection 14.2.3.1:

STD COL 14.4-3 The Startup Administrative Manual (procedure) governs the initial testing and is issued no later than 60 days prior to the beginning of the pre-operational phase. Testing during all phases of the test program is conducted using approved test procedures.

14.2.3.1.1 Procedure Verification

Since procedures may be approved for implementation weeks or months in advance of the scheduled test date, a review of the approved test procedure is required before commencement of testing. The test engineer is responsible for verifying:

- Drawing and document revision numbers listed in the reference section of the test procedure agree with the latest revisions.
- The procedure text reflects any design and licensing (i.e., FSAR and Technical Specifications) changes made since the procedure was originally approved for implementation.
- Any new (since preparation of the procedure) Operating Experience lessons learned are incorporated into individual test procedures.

Procedures require signoff verification for prerequisites and instruction steps. This signoff includes identification of the person doing the signoff and the date and time of completion.

Test engineers maintain chronological logs of test status to facilitate turnover and aid in maintaining operational configuration control. These logs become part of the test documentation.

There is a documented turnover process to make known the test status and equipment configuration when personnel transfer responsibilities, such as during a shift change.

Test briefings are conducted for each test in accordance with administrative procedures. When a shift change occurs before test completion, another briefing occurs before resumption or continuation of the test.

Data collected is marked or identified with test, date, and person collecting data. This data becomes part of the test documentation.

The plant corrective action program is used to document deficiencies, discrepancies, exceptions, non-conformances and failures (collectively known as test exceptions) identified in the ITP. The corrective action documentation becomes part of the test documentation. WEC and/or other design organizations participate in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria.

The plant manager approves proceeding from one test phase to the next during the ITP. Approvals are documented in an overall ITP governance document.

Administrative procedures detail the test documentation review and approval. Review and approval of test documentation includes the test engineer, testing supervisor, Startup Group manager, WEC site representative or appropriate vendor, and JTWG. Final approval is by the plant manager.

Plant readiness reviews are conducted to assure that the plant staff and equipment are ready to proceed to the next test phase or plateau.

14.2.3.1.2 Work Control

STD SUP 14.2-5 The Startup Group is responsible for preparing work requests when assistance is required from the Construction organization. Work requests are issued in accordance with a site specific procedures governing the work management process. The plant staff, upon identifying a need for Construction organization assistance, coordinates their requirements through the appropriate Startup Test Engineer.

Activities requiring Construction organization work efforts are performed under the plant tagging procedures. Tagging requests are governed by a site-specific procedure for equipment clearance. Tagging procedures shall be used for protection of personnel and equipment and for jurisdictional or custodial conditions that have been turned over in accordance with the turnover procedure.

The Startup Group is responsible for supervising minor repairs and modifications, changing equipment settings, and disconnecting and reconnecting electrical terminations as stipulated in a specific test procedure. Startup Test Engineers may perform independent verification of changes made in accordance with approved test procedures.

14.2.3.1.3 System Turnover

STD SUP 14.2-6 During the construction phase, systems, subsystems, and equipment are completed and turned over in an orderly and well-coordinated manner. Guidelines are established to define the boundary and interface between related system/subsystem and are used to generate boundary scope documents; for example, marked-up piping and instrument diagrams (P&IDs) and electrical schematic diagrams are provided for scheduling and subsequent development of component and system turnover packages. The system turnover process includes requirements for the following:

- Documenting inspections performed by the construction organization (e.g., highlighted drawings showing areas inspected).
- Documenting results of construction testing.

- Determining the construction-related inspections and tests that need to be completed before preoperational testing begins. Any open items are evaluated for acceptability of commencing preoperational testing.
- Developing and implementing plans for correcting adverse conditions and open items, and means for tracking such conditions and items.
- Verifying completeness of construction and documentation of incomplete items.

14.2.3.1.4 Conduct of Modifications During the Initial Test Program

STD SUP 14.2-7 Temporary alterations may be required to conduct certain tests. These alterations are documented in the test procedures. The test procedures contain restoration steps and retesting necessary to confirm satisfactory restoration to the required configuration. Modifications may be performed by the Construction organization or the plant staff processes prior to NRC issuance of the 10 CFR 52.103(g) finding. If the modification invalidates a previously completed ITAAC, then that ITAAC is re-performed. Each modification is reviewed to determine the scope of post-modification testing that is to be performed. Testing is conducted and documented to maintain the validity of preoperational testing and ITAAC. Alterations made following NRC issuance of the 10 CFR 52.103(g) finding are in accordance with plant processes and meet license conditions. Modifications that require changes to ITAAC require NRC approval of the ITAAC change.

14.2.3.1.5 Conduct of Maintenance During the Initial Test Program

STD SUP 14.2-8 Corrective or preventive maintenance activities are reviewed to determine the scope of postmaintenance testing to be performed. Prior to NRC issuance of the 10 CFR 52.103(g) finding, post-maintenance testing is conducted and documented to maintain validity of associated preoperational testing and ITAAC remain valid. Maintenance performed following NRC issuance of the 10 CFR 52.103(g) finding is in accordance with plant staff processes and meets license conditions.

14.2.3.2 Review of Test Results

Add the following Subsections at the end of DCD Subsection 14.2.3.2:

14.2.3.2.1 Review and Approval Responsibilities

STD COL 14.4-4 Upon completion of a test, the startup engineer is responsible for:

- Reviewing the test data.
- Evaluating the test results.
- Verifying that the acceptance criteria are met.
- Verifying that the test results that do not meet acceptance criteria are entered into the corrective action program.
- Verifying that the results of retesting do not invalidate ITAAC acceptance criteria.

Test results are reviewed and approved by the JTWG. Review and approval of test results are kept current such that succeeding tests are not dependent on systems or components that have not been adequately tested. Test exceptions which do not meet acceptance criteria are identified to the affected and responsible design organizations and entered into the corrective action program. Implementation of corrective actions and retests are performed as required.

Prior to initial fuel load, the results of the preoperational test phase are comprehensively reviewed by the PT&O organization and the JTWG to verify the results indicate that the required plant structures, systems, and components are capable of supporting the initial fuel load and subsequent startup testing. The plant manager approves fuel loading.

Each area of startup testing is reviewed and evaluated by the PT&O organization and the JTWG. The test results at each power ascension testing power plateau are reviewed and evaluated by the PT&O organization and the JTWG and approved by the plant manager before proceeding to the next plateau. Startup test reports are prepared in accordance with the guidance in position C.9 of Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

The reactor vendor is responsible for reviewing and approving the results of the tests of supplied equipment. Architect Engineer representatives review and approve the results of the tests of supplied equipment. Other vendors' representatives review and approve the results of the tests of supplied equipment. Final approval of individual test completion is by the plant manager after approval by the Joint Test Working Group (JTWG).

14.2.3.2.2 Technical Evaluation

Each completed test package is reviewed by technically qualified personnel to confirm satisfactory demonstration of plant, system or component performance and compliance with design and license criteria.

14.2.3.3 Test Records

Add the following subsection at the end of DCD Subsection 14.2.3.3:

14.2.3.3.1 Startup Test Reports

STD COL 14.4-4 Startup test reports are generated describing and summarizing the completion of tests performed during the ITP. A startup report is submitted at the earliest of:

1) 9 months following initial criticality,

2) 90 days after completion of the ITP, or

3) 90 days after start of commercial operations. If one report does not cover all three events, then supplemental reports are submitted every three months until all three events are completed. These reports:

- Address each ITP test described in the FSAR.
- Provide a general description of measured values of operating conditions or characteristics obtained from the ITP as compared to design or specification values.
- Describe any corrective actions that were required to achieve satisfactory operation.
- Include any other information required by license conditions.

Add the following subsections after DCD Subsection 14.2.5:

Utilization of Operating Experience

STD SUP 14.2-4 Administrative procedures provide methodologies for evaluating and initiating action for operating experience information (OE). DCD Subsection 14.2.5 describes the general use of operating experience by WEC in the development of the test program.

14.2.5.1 Use of OE During Test Procedure Preparation

Administrative procedures require review of recent internal and external operating experience when preparing test procedures.

14.2.5.2 Sources and Types of Information Reviewed for ITP Development

Multiple sources of operating experience were reviewed to develop this description of the ITP administration program. These included INPO Reports, INPO Guidelines, INPO Significant Event Reports, INPO Significant Operating Experience Reports and NRC Regulatory Guide 1.68.

14.2.5.3 Conclusions from Review

The following conclusions are a result of the OE review conducted to develop this ITP administration program description:

- The test procedures should provide guidance as to the expected plant response and instructions concerning what conditions warrant aborting the test. Errors and problems with the procedures should be anticipated. A means for prompt but controlled approval of changes to test procedures is needed. Critical test procedures should provide specific criteria for test termination and specific steps to conduct termination is conducted in a safe and orderly manner. Providing procedural guidance for aborting the test could prevent delays in plant restoration. Conservative guidance for actions to be taken should be included in the procedures.

- Plant simulators may prove useful in preparing for special tests and verifying procedures.

- Appropriate component/system operability should be verified prior to critical tests.

- The need to perform physics tests that can produce severe power tilts should be evaluated, particularly if tests at other similar reactors have provided sufficient data to verify the adequacy of the nuclear physics analysis.

- Compensatory measures should be implemented in accordance with guidance for infrequently performed tests or evolutions, where appropriate.

14.2.5.4 Summary of Test Program Features Influenced by the Review

The conclusions from the preceding section were incorporated in Section 14.2.

14.2.5.5 Use of OE during Conduct of ITP

Administrative procedures require discussion of operating experience when performing pre-job briefs immediately prior to the conduct of a test.

14.2.6 USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

Add the following text and Subsection to the end of DCD Subsection 14.2.6:

STD COL 14.4-3 These procedures are used extensively in the Human-Machine Interface Testing, which is integrated as a part of the Control Room Design finalization. Additionally, the AP1000 plant operating and emergency procedures are developed to support the following design finalization activities:

- Human Factors Engineering
- Operational Task Analysis
- Training Simulator Development
- Verification and Validation of the Procedures and the Training Material

The AP1000 emergency, abnormal and some normal operating procedures, along with some Alarm Response Procedures and surveillance procedures, are exercised and verified in the processes delineated above and in the Control Room design finalization process.

In addition, the AP1000 Preoperational Testing and Startup Test procedures are verified and validated during the design finalization process, which helps prevent human factors issues with the development of these procedures. In addition, the plant operators use the Normal Operating Procedures while preoperational and startup tests are performed, which adds to their validity and the plant operators training.

14.2.6.1 Operator Training and Participation during Certain Initial Tests (TMI Action Plan Item I.G.1, NUREG-0737)

The objective of operator participation is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and offnormal events is conducted.

Operators are trained on the specifics of the ITP schedule, administrative requirements and tests. Specific Just In Time training is conducted for selected startup tests.

The ITP may result in the discovery of an acceptable plant or system response that differs from the expected response. Test results are reviewed to identify these differences and the training for operators is changed to reflect them. Training is conducted as soon as is practicable in accordance with training procedures.

14.2.8. TEST PROGRAM SCHEDULE

Add the following text and subsection at the end of DCD Subsection 14.2.8:

STD SUP 14.2-1 A site-specific initial test program schedule will be provided to the NRC after issuance of the COL. This schedule will address each major phase of the test program (including tests that are required to be completed before fuel load), as well as the organizational impact of any overlap of first unit initial testing with initial testing of the second unit.

The sequential schedule for individual startup tests should establish that testing is completed in accordance with plant technical specification requirements for structures, systems and components (SSC) operability before changing plant modes. Additionally, the schedule establishes that the safety of the plant is not dependent on the performance of untested SSCs. Guidance provided in Regulatory Guide 1.68 is used for development of the schedule.

The Startup Administrative Manual shall include the following controls:

- Test Procedure Development Schedule:
    - Controls to establish a schedule for the development of detailed testing, plant operating, and emergency procedures. These procedures, to the extent practical, are trial-tested and corrected during the initial test program prior to fuel loading in order to establish their adequacy.
    - Controls to confirm that approved test procedures are in a form suitable for review by NRC inspectors at least 60 days prior to their intended use, or at least 60 days prior to fuel loading for fuel loading and startup test procedures.
    - Controls to provide timely notification to the NRC of changes in approved test procedures previously available for NRC review.
- Initial Test Program Schedule:
    - Controls to establish a schedule to conduct the major phases of the initial test program, relative to the expected fuel loading date. This is covered in License Conditions in Part 10 of the COL Application.
    - Controls to allow at least 9 months for conducting preoperational testing.
    - Controls to allow at least 3 months for conducting startup testing, including fuel loading, low-power tests, and power-ascension tests.

- Controls to overlap test program schedules (for multi-unit sites) such that they do not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

- Controls to sequence the schedule for individual startup tests, insofar as is practicable, such that testing is completed prior to exceeding 25 percent power for the plant SSCs that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. The schedule should establish that, insofar as is practicable, testing is accomplished as early in the test program as is feasible and that the safety of the plant is not dependent on the performance of untested SSCs.

The milestone schedule for developing plant operating procedures is presented in Table 13.4-201. The operating and emergency procedures are available prior to start of licensed operator training and, therefore, are available for use during the ITP. Required or desired procedure changes may be identified during their use. Administrative procedures describe the process for revising plant operating procedures.

14.2.9 PREOPERATIONAL TEST DESCRIPTIONS

Add the following subsection at the beginning of DCD Subsection 14.2.9

STD SUP 14.2-2 During preoperational testing, it may be necessary to return system control to Construction organization to repair or modify the system or to correct new problems. Administrative procedures include direction for:

- Means of releasing control of systems and or components to construction.
- Methods used for documenting actual work performed and determining impact on testing.
- Identification of required testing to restore the system to operability/ functionality/availability status, and to identify tests to be re-performed based on the impact of the work performed.
- Authorizing and tracking operability and unavailability determinations.
- Verifying retests stay in compliance with ITAAC.

14.2.9.2.22 Pressurizer Surge Line Testing (First Plant Only)

STD COL 3.9-5 **Purpose**

The purpose of the pressurizer surge line testing is: a) to obtain data to verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line, and b) to obtain Reactor Coolant System piping displacement measurements for baseline data, as described in DCD subsections 3.9.3, 14.2.5, and 14.2.9.1.7 item (d).

Prerequisites

The construction tests for the individual components associated with the Reactor Coolant System have been completed. The testing and calibration of the required test instrumentation has been completed. The temporary sensors and instrumentation lead wires required for monitoring thermal stratification, cycling, and striping have been installed. The calibration of the transducers and the operability of the data acquisition equipment have been verified. Prior to testing of the piping system, a pretest walk-down shall be performed to verify that the anticipated piping movement is not obstructed by objects not designed to restrain the motion of the system (including instrumentation and branch lines). The system walk-down shall also verify that supports are set in accordance with the design.

General Test Methods and Acceptance Criteria

The performance of the Reactor Coolant System is observed and recorded during a series of individual tests that characterize the various modes of system operation. This testing verifies that the temperature sensors operate as described in DCD subsection 3.9.3 and in appropriate design specifications.

a) Verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line.

b) Record sensor data at specified intervals throughout hot functional testing of the RCS system, including during the drawing and collapsing of the bubble in the pressurizer.

c) Retain the following plant parameters time history for the same data recording period:

- Hot leg temperature
- Reactor Coolant System pressure
- Reactor coolant pump status
- Pressurizer level
- Pressurizer temperature (liquid and steam)

- Pressurizer spray temperature
- Pressurizer spray and auxiliary spray flow
- Normal residual heat removal system flow rate
- Passive core cooling system – passive residual heat removal flow rate.

d) Monitor pressurizer surge line and pressurizer spray line for valve leakage.

e) Remove the transducers and associated hardware after the completion of testing.

f) Proper operation of the temperature sensors in the pressurizer surge and spray lines is verified.

14.2.9.4.15 Seismic Monitoring System Testing

Add the following text at the beginning of DCD Subsection 14.2.9.4.15:

STD COL 14.4-5 The seismic monitoring system testing described in this section of the DCD also applies to site-specific seismic sensors.

Add the following subsections after DCD Subsection 14.2.9.4.21:

14.2.9.4.22 Storm Drains

STD COL 14.4-5 **Purpose**

Storm drain system testing verifies that the drains prevent plant flooding by diverting storm water away from the plant, as described in Section 2.4.

Prerequisites

Construction of the storm drain system is completed, and the system is operational.

General Test Methods and Acceptance Criteria

The storm drain system is visually inspected to verify the flow path is unobstructed. The system is observed under simulated or actual precipitation events to verify that the runoff from roof drains and the plant site and adjacent areas does not result in unacceptable soil erosion adjacent to, or flooding of, Seismic Category I structures.

14.2.9.4.23 Off-site AC Power Systems

Purpose

Off-site alternating current (ac) power system testing demonstrates the energization and proper operation of the as-installed switchyard components, as described in Section 8.2.

Prerequisites

Construction testing of plant off-site ac power systems, supporting systems, and components is completed. The components are operational and the switchyard equipment is ready to be energized. The required test instrumentation is properly calibrated and operational. The off-site grid connection is complete and available.

General Test Methods and Acceptance Criteria

The plant off-site ac power system components undergo a series of individual component and integrated system tests to verify that the off-site ac power system performs in accordance with the associated component design specifications. The individual component and integrated tests include:

a. Availability of ac and direct current (dc) power to the switchyard equipment is verified.

b. Operation of high voltage (HV) circuit breakers is verified.

c. Operation of HV disconnect switches and ground switches is verified.

d. Operation of substation transformers is verified.

e. Operation of current transformers, voltage transformers, and protective relays is verified.

f. Operation of switchyard equipment controls, metering, interlocks, and alarms that affect plant off-site ac power system performance is verified.

g. Design limits of switchyard voltages and stability are verified.

h. Under simulated fault conditions, proper function of alarms and protective relaying circuits is verified.

i. Operation of instrumentation and control alarms used to monitor switchyard equipment status.

j. Proper operation and load carrying capability of breakers, switch gear, transformers, and cables, and verification of these items by a non-testing means such as a QC nameplate check of as-built equipment where testing would not be practical or feasible.

k. Verification of proper operation of the automatic transfer capability of the preferred power supply to the maintenance power supply through the reserve auxiliary transformer.

l. Switchyard interface agreement and protocols are verified.

The test results confirm that the off-site ac power systems meet the technical and operational requirements described in Section 8.2.

14.2.9.4.24 Raw Water System

Purpose

Raw water system testing verifies that the as-installed components supply raw water to the circulating water cooling tower basin, service water system cooling tower basin, fire protection water storage tanks, and other systems, as described in Subsection 9.2.11.

Prerequisites

Construction testing of the raw water system is completed. The components are operational and the storage tanks and cooling tower basins are able to accept water. Required support systems, electrical power supplies, and control circuits are operational.

General Test Methods and Acceptance Criteria

The raw water system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.11 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of the system pumps, traveling screens, automatic strainers, and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, alarms, and interlocks is verified.

c. Operation of heat tracing on system piping is verified.

14.2.9.4.25 Sanitary Drainage System

Purpose

Sanitary drainage system testing verifies that the as-installed components properly collect and discharge sanitary waste, as described in DCD Subsection 9.2.6.

Prerequisites

Construction testing of the sanitary drainage system is completed. Required support systems, electrical power supplies, and control circuits are operational.

General Test Methods and Acceptance Criteria

The sanitary drainage system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.6.2.1 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of lift stations and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.26 Fire Brigade Support Equipment

Purpose

Fire brigade support equipment testing verifies that the equipment operates and is available when needed to perform the fire brigade functions, as described in Section 9.5.

Prerequisites

Equipment is ready and available for testing.

General Test Methods and Acceptance Criteria

The fire brigade support equipment undergoes a series of inspections to verify availability and operability. Equipment is available for selection and use, based on the hazard. Fire brigade support equipment tests include:

a. Location of portable extinguishers is verified; portable extinguishers are verified fully charged.

b. Operation of portable ventilation equipment is verified.

c. Operation of portable communication equipment is verified.

d. Operation of portable lighting is verified.

e. Operation of self-contained breathing apparatus and face masks is verified.

f. Operation of keys to open locked fire area doors is verified.

g. Turnout gear functionality and availability is verified.

h. Compatibility of threads for hydrants, hose couplings, and standpipe risers with the local fire department equipment is verified, or alternatively, an adequate supply of readily available hose thread adaptors is verified.

14.2.9.4.27 Portable Personnel Monitors and Radiation Survey Instruments

Purpose

Portable personnel monitors and radiation survey instruments testing verifies that the devices operate in accordance with their intended function in support of the radiation protection program, as described in Chapter 12.

Prerequisites

Portable personnel monitors, radiation survey instruments, and appropriate certified test sources are on site.

General Test Method and Acceptance Criteria

The portable personnel monitors and radiation survey instruments are source checked, tested, maintained, and calibrated in accordance with the manufacturers' recommendations. The portable monitors and instruments tests include:

a. Proper function of the monitors and instruments to respond to radiation is verified, as required.

b. Proper operation of instrumentation controls, battery, and alarms, if applicable.

14.2.10 STARTUP TEST PROCEDURES

Add the following at the beginning of DCD Subsection 14.2.10:

STD SUP 14.2-3 The startup testing program is based on increasing power in discrete steps. Major testing is performed at discrete power levels as described in DCD Subsection

14.2.7. The first tests during Power Ascension Testing that verify movements and expansion of equipment are in accordance with design, and are conducted at a power level as low as practical (approximately 5 percent).

The governing Power Ascension Test Plan requires the following operations to be performed at appropriate steps in the power-ascension test phase:

- Conduct any tests that are scheduled at the test condition or power plateau.
- Confirm core performance parameters (core power distribution) are within expectations.
- Determine reactor power by heat balance, calibrate nuclear instruments accordingly, and confirm the existence of adequate instrumentation overlap between the startup range and power range detectors.
- Reset high-flux trips just prior to ascending to the next level to a value no greater than 20 percent beyond the power of the next level unless Technical Specification limits are more restrictive.
- Perform general surveys of plant systems and equipment to confirm that they are operating within expected values.
- Check for unexpected radioactivity in process systems and effluents.
- Perform reactor coolant leak checks.
- Review the completed testing program at each plateau; perform preliminary evaluations, including extrapolation core performance parameters for the next power level; and obtain the required management approvals before ascending to the next power level or test condition.

Upon completion of a given test, a preliminary evaluation is performed that confirms acceptability for continued testing. Smaller transient changes are performed initially, gradually increasing to larger transient changes. Test results at lower powers are extrapolated to higher power levels to determine acceptability of performing the test at higher powers. This extrapolation is included in the analysis section of the lower power procedure.

Surveillance test procedures may be used to document portions of tests, and ITP tests or portions of tests may be used to satisfy Technical Specifications surveillance requirements in accordance with administrative procedures. At Startup Test Program completion, a plant capacity warranty test is performed to satisfy the contract warranty and to confirm safe and stable plant operation.

Add the following subsection after DCD Subsection 14.2.10.4.28:

14.2.10.4.29 Cooling Tower(s)

STD COL 14.4-5 **Objectives**

- Verify proper cooling tower(s) function. Provide thermal acceptance testing of the cooling tower's heat removal capabilities.

Prerequisites

- The cooling tower(s) is structurally complete and in good operating condition.
- Circulating water system testing is complete.
- Required support systems, electrical power supplies, and control circuits are operational.

Test Method

Thermal performance of the cooling towers is tested and verified using established industry test standards.

Performance Criteria

The cooling tower(s) perform as described in Subsection 10.4.5 and in appropriate design specifications.

14.3 CERTIFIED DESIGN MATERIAL

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following subsections after DCD Subsection 14.3.2.2.

14.3.2.3 Site-Specific ITAAC (SS-ITAAC)

STD SUP 14.3-1 A table of inspections, tests, analyses, and acceptance criteria (ITAAC) entries is provided for each site-specific system described in this FSAR that meets the selection criteria, and that is not included in the certified design. The intent of these ITAAC is to define activities that are undertaken to verify the as-built system conforms with the design features and characteristics defined in the system design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspection, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has associated inspections, tests, or analyses (ITA) requirements specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

SS-ITAAC do not address ancillary buildings and structures on the site, such as administrative buildings, parking lots, warehouses, training facilities, etc.

Selection Criteria - The following are considered when determining what information is included in the SS-ITAAC:

- In determining those structures, systems, or components for which ITAAC must be prepared, the following questions are considered for each structure, system, or component:
  - Are any features or functions classified as Class A, B, or C?
  - Are any defense-in-depth features or functions provided?
  - For nonsafety-related systems, are any features or functions credited for mitigation of design basis events?
  - For nonsafety-related systems, are there any features or functions that have been identified in DCD Section 16.3 as candidates for additional regulatory oversight?

If the answer to any of the above questions is yes, then ITAAC are prepared.

- The scope and content of the ITAAC correspond to the scope and content of the site-specific system design description.
- One inspection, test, or analysis may verify one or more provisions in the system design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the system design description. There is not necessarily a one-to-one correspondence between the ITAAC and the system design descriptions.
- As required by 10 CFR 52.103, the inspections, tests, and analyses are completed (and the acceptance criteria satisfied) prior to initial fuel loading.
- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the system design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the system design descriptions and preparing an ITAAC table entry for each design description criterion that satisfies the selection criteria. A close correlation exists between the left-hand column of the ITAAC and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis) and the acceptance criteria for the as-built feature.

The approach used to perform the tests, inspections, or analyses is similar to that described in DCD Subsection 14.3.2.2.

14.3.2.3.1 Emergency Planning ITAAC (EP-ITAAC)

EP-ITAAC have been developed to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-B of Regulatory Guide 1.206. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

14.3.2.3.2 Physical Security ITAAC (PS-ITAAC)

STD COL 13.6-1 Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI). These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.

14.3.2.3.3 Other Site-Specific Systems

STD SUP 14.3-1 One additional site-specific system has been determined to meet the ITAAC selection criteria, and ITAAC have been included for the Transmission Switchyard and Offsite Power System (ZBS) as indicated in Table 14.3-201. Such systems are subject to the normal functional testing to verify that newly designed and installed systems, structures, or components perform as designed.

VCS SUP 14.3-2 A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-201. This table supplements DCD Table 14.3-1.

14.3.3 CDM SECTION 3.0, NON-SYSTEM BASED DESIGN DESCRIPTIONS AND ITAAC

Add the following new subsection after the first paragraph in DCD Subsection 14.3.3

14.3.3.1 Pipe Rupture Hazard Analysis ITAAC

STD COL 3.6-1 A pipe rupture hazard analysis is part of the piping design. The analyses will document that structures, systems, and components (SSCs) which are required to be functional during and following a design basis event have adequate high-energy and moderate-energy pipe break mitigation features. The locations of postulated ruptures and essential targets will be established and required pipe whip restraint and jet shield designs will be included. The as-designed pipe rupture hazards analysis will be based on the as-designed piping analysis and will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The report of the pipe rupture hazard analysis shall conclude that, for each postulated piping failure, the systems, structures, and components that are required to be functional during and following a design basis event are protected.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 are covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built pipe rupture hazards mitigation features reflect the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The as-designed pipe rupture hazard analysis completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

14.3.3.2 Piping Design ITAAC

STD COL 3.9-7 The piping design ITAAC consists of the piping analysis for safety-related ASME Code piping. The piping design is completed on a package-by-package basis for applicable systems. In order to support closure of the piping design ITAAC, information consisting of the as-designed piping analysis for piping lines chosen to demonstrate all aspects of the piping design will be made available for NRC review, inspection, and/or audit. This information will consist of a design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping. The piping packages to be analyzed are identified the in DCD.

The ASME Code prescribes certain procedures and requirements that are to be followed for completing the piping design. The piping design ITAAC includes a verification of the ASME Code design report to ensure that the appropriate code design requirements for each system's safety class have been implemented.

A reconciliation of the applicable safety-related as-built piping systems is covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built piping reflects the design, is reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The piping design completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

Table 14.3-201
ITAAC Screening Summary

	Structure/ System Acronym	Structure/System Description	Selected for ITAAC
VCS SUP 14.3-2	DRS	Storm Drain System	XX
	MES	Meteorological and Environmental Monitoring System	XX
	OWS	Offsite Water Treatment System(a)	XX
	RWS	Raw Water System	XX
	TVS	Closed Circuit TV System	XX
	VPS	Pump House Building Ventilation System	NA
	YFS	Yard Fire Water System	XX
	ZBS	Transmission Switchyard and Offsite Power System	XX
	ZRS	Offsite Retail Power System	NA

Legend: XX = Site-specific system selected for ITAAC – title only, no entry for COLA

XX = Selected for ITAAC

NA = System is not part of VCS design

a) OWS is outside the scope of the standard AP1000 but will be on the VCSNS property.

14.4 COMBINED LICENSE APPLICANT RESPONSIBILITIES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

14.4.1 ORGANIZATION AND STAFFING

STD COL 14.4-1 This COL Item is addressed in Section 14.2.

14.4.2 TEST SPECIFICATIONS AND PROCEDURES

STD COL 14.4-2 Preoperational and startup test specifications and procedures are provided to the NRC in accordance with the requirements of DCD Subsection 14.2.3. The controls for development of test specifications and procedures are also described in Subsection 14.2.3.

A cross reference list is provided between ITAACs and test procedures and/or sections of test procedures.

14.4.3 CONDUCT OF TEST PROGRAM

STD COL 14.4-3 A site-specific startup administration manual (procedure), which contains the administration procedures and requirements that govern the activities associated with the plant initial test program, FSAR Section 14.2, is provided.

14.4.4 REVIEW AND EVALUATION OF TEST RESULTS

STD COL 14.4-4 Review and evaluation of individual test results, as well as final review of overall test results and selected milestones or hold points are addressed in Subsection 14.2.3.2. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed.

14.4.5 INTERFACE REQUIREMENTS

STD COL 14.4-5 This COL Item is addressed in Subsections 14.2.9.4.15, 14.2.9.4.22 through 14.2.9.4.27, 14.2.10.4.29, and in the Physical Security Plan.

14.4.6 FIRST-PLANT-ONLY AND THREE-PLANT-ONLY TESTS

STD COL 14.4-6 First-plant-only and first-three-plant-only tests either are performed in accordance with DCD Section 14.2.5 or a justification is provided that the results of the first-plant-only and first-three-plant-only tests are applicable to a subsequent plant. If the tests are not performed, the justification is provided prior to preoperational testing.

APPENDIX 14A
DESIGN ACCEPTANCE CRITERIA/ITAAC CLOSURE PROCESS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 15
ACCIDENT ANALYSES

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CHAPTER 15
ACCIDENT ANALYSES

15.0 ACCIDENT ANALYSES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.0.3.2 Initial Conditions

Add the following paragraph at the end of DCD Subsection 15.0.3.2.

STD COL 15.0-1 The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System (Reference 201), which will be calibrated (in a certified laboratory using a piping configuration representative of the plant piping design) prior to installation and will be tested after installation in the plant in accordance with the LEFM CheckPlus commissioning procedure. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. The calculated calorimetric is done in accordance with a previously accepted Westinghouse methodology (Reference 202). Administrative controls implement maintenance and contingency activities related to the power calorimetric instrumentation.

15.0.15 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 15.0.15.1.

STD COL 15.0-1 This COL item is addressed in FSAR Subsection 15.0.3.2.

15.0.16 REFERENCES

Add the following text to the end of DCD Subsection 15.0.16.

201. Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, “Caldon Ultrasonics Engineering Report ER-

157P, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus^{TM} System'," (TAC No. ME1321). August 16, 2010. ADAMS Accession No. ML102160694.

202. Final Safety Evaluation for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) — Issuance of Amendment re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves. September 24, 2001, ADAMS Accession No. ML012490569.

15.1 INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.6.5.3.7.3 Atmospheric Dispersion Factors

Add the following paragraph at the end of DCD Subsection 15.6.5.3.7.3.

VCS COL 2.3-4 Site-specific χ/Q values provided in Subsection 2.3.4 are bounded by the values given in DCD Tables 15A-5 and 15A-6.

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.7.6 COMBINED LICENSE INFORMATION

VCS COL 15.7-1 This COL item is addressed in Subsection 2.4.13.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 15A
EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15A.3.3 Atmospheric Dispersion Factors

Replace the third paragraph in DCD Subsection 15A.3.3 with the following:

VCS COL 2.3-4 Site-specific χ/Q values provided in Subsection 2.3.4 are bounded by the values given in DCD Tables 15A-5 and 15A-6.

APPENDIX 15B
REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 16
TECHNICAL SPECIFICATIONS

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CHAPTER 16
TECHNICAL SPECIFICATIONS

16.1 TECHNICAL SPECIFICATIONS

Subsections 16.1.1 and 16.1.2 of the DCD are incorporated by reference with no departures or supplements. The generic technical specifications and bases in Chapter 16 of the DCD are not considered Tier 2 information; therefore they are not incorporated by reference within this FSAR. However, the generic technical specifications and bases provided with Chapter 16 of the DCD are incorporated directly into the plant-specific technical specifications and bases provided in Part 4 of this COL application.

16.1.1 INTRODUCTION TO TECHNICAL SPECIFICATIONS

Combined License Information

VCS COL 16.1-1 This COL Item (i.e., information addressing each of the remaining brackets [ ] in the AP1000 generic technical specifications) is addressed in Part 4 of the COLA.

16.2 DESIGN RELIABILITY ASSURANCE PROGRAM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

16.3 INVESTMENT PROTECTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

16.3.1 INVESTMENT PROTECTION SHORT-TERM AVAILABILITY CONTROLS

Add the following paragraph after the bulleted items at the end of the second paragraph of DCD Subsection 16.3.1:

STD COL 16.3-1 Station procedures govern and control the operability of investment protection systems, structures, and components, in accordance with Table 16.3-2 of the DCD, and provide the operating staff with instruction for implementing required actions when operability requirements are not met. Procedure development is addressed in FSAR Section 13.5.

16.3.2 COMBINED LICENSE INFORMATION

STD COL 16.3-1 This COL Item is addressed in Subsection 16.3.1.

CHAPTER 17
QUALITY ASSURANCE

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CHAPTER 17
QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Replace the information in DCD Section 17.1 with the following information.

VCS COL 17.5-1 South Carolina Electric & Gas (SCE&G) is responsible for the establishment and execution of quality assurance program requirements during the design and construction phases of V.C. Summer Nuclear Station Units 2 and 3. SCE&G may delegate and has delegated to others, such as NuStart Energy Development, LLC, Westinghouse Electric Company, and Bechtel Power Corporation, the work of establishing and executing the quality assurance program, or any parts thereof, but retains responsibility for the quality assurance program.

Effective during COL application development, through and until COL issuance, the NuStart Energy Development, LLC (NuStart) Quality Assurance Plan (Reference 201), the Westinghouse Electric Company Quality Management System (DCD Section 17.6) and Bechtel Power Corporation Nuclear Quality Assurance Manual (Reference 205) define the QA program requirements for design activities. Safety-related construction activities at SCE&G are not planned before the COL is issued.

NuStart was created by multiple utilities and, as such, is comprised of multiple member utilities to include SCE&G. NuStart's purpose is two-fold of which the first is demonstrating the licensing process defined by 10 CFR Part 52. The second purpose is to work with reactor vendors in completing the engineering work for the standardized plant designs.

Bechtel Power Corporation was contracted by SCE&G to develop the V. C. Summer Nuclear Station Units 2 and 3 COL application. This included the process of collection, review and analysis of specific data for site characterization and was controlled under the Bechtel Power Corporation Quality Assurance Manual (Reference 205).

SCE&G maintains oversight of the COL application development as well as design and construction activities under its existing 10 CFR Part 50, Appendix B, program as described in the NRC approved SCE&G V.C. Summer Nuclear Station Unit 1 "Operational Quality Assurance Plan" (Reference 206). The Unit 1 Quality Assurance Program complies with the requirements of Regulatory Guide 1.28, Revision 0 – "Quality Assurance Program Requirements (Design and Construction)". The "Operational Quality Assurance Plan" (Reference 206) is supplemented, in part, by the SCE&G "New Nuclear Deployment Quality

Assurance Plan" (Reference 204). The "New Nuclear Deployment Quality Assurance Plan (Reference 204) serves as an interfacing document between the work activities of the New Nuclear Deployment organization and the "Operational Quality Assurance Plan" (Reference 206). It assures that the proper administrative controls and the quality of activities related to the procurement of services, equipment, oversight of construction/manufacturing, and licensing activities being performed within the New Nuclear Deployment organization conform to the applicable requirements of 10 CFR 50, Appendix B. These plans provide the necessary quality assurance guidance for oversight of site characterization activities and COL application content providers. SCE&G maintains this oversight through the review and approval of the NuStart Quality Assurance Plan (Reference 201) and industry standard COL application sections as well as conducting audits/surveillances of Bechtel activities, and providing input to the COL application development, including, but not limited to, review of COL application content.

Implementation of the applicable portions of the "Quality Assurance Program Description" (QAPD) discussed in Section 17.5 begins at COL issuance. The program establishes the quality assurance program requirements for the remaining portion of the design and construction phases and for operations; full implementation of the operations related requirements will be no later than as indicated in FSAR Table 13.4-201.

17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

17.3 QUALITY ASSURANCE DURING DESIGN, PROCUREMENT, FABRICATION, INSPECTION, AND/OR TESTING OF NUCLEAR POWER PLANT ITEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

17.4 DESIGN RELIABILITY ASSURANCE PROGRAM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD SUP 17.4-1 The quality assurance requirements for non-safety related SSCs within the scope of D-RAP is in accordance with the Quality Assurance Program Description (QAPD), Part III.

STD DEP 1.1-1

17.5 QUALITY ASSURANCE PROGRAM DESCRIPTION - NEW LICENSE APPLICANTS

VCS COL 17.5-1
STD COL 17.5-2
STD COL 17.5-4
STD COL 17.5-8

The Quality Assurance Program in place during the design, construction, and operations phases is described in the QAPD, which is maintained as a separate document. This QAPD is incorporated by reference (see Table 1.6-201). This QAPD is based on NEI 06-14A, "Quality Assurance Program Description," (Reference 207).

Conformance statements for QA-related Regulatory Guides (including Regulatory Guides 1.28, 1.30, 1.33, 1.38, 1.39, 1.94, and 1.116) are provided in Appendix 1AA. While many Regulatory Guide positions can be identified as applicable to the scope of work identified and addressed by the DCD and others can be identified as applicable to the scope of work identified and addressed by the COLA, some QA guidance related positions could be accomplished by either scope of work and thus be addressed in either the DCD or the COLA. These positions are primarily dependent on who performs the work. The DCD conformance statement indicates an exception to apply NQA-1. The COLA identifies an exception to apply NQA-1. Per DCD Section 17.3, WEC work performed up to March 15, 2007 applied a 1991 version of the standard. A 1994 version of the standard is applied for work performed after that date by WEC. If the work is performed under the applicant's COL program, the 1994 version of NQA-1 identified in the COLA QAPD is applied. Thus, DCD scope (identified in DCD Appendix 1A) and "remaining scope" differentiate the application of the guidance identified in these Regulatory Guides.

VCS COL 17.5-1

The QAPD is the SCE&G VCSNS Units 2 and 3 Quality Assurance Program Description.

STD COL 17.5-4

Table 13.4-201 provides milestones for operational quality assurance program implementation.

VCS COL 17.5-1

The Quality Assurance Program in place prior to implementation of the QAPD is described in Section 17.1.

STD DEP 1.1-1

17.6 MAINTENANCE RULE PROGRAM

STD SUP 17.6-1
STD COL 3.8-5

This section incorporates by reference NEI 07-02A, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52,” (Reference 208) with the following supplemental information. See Table 1.6-201.

Table 13.4-201 provides milestones for maintenance rule program implementation.

The text of the template provided in NEI 07-02A is generically numbered as “17.X.” When the template is incorporated by reference into this FSAR, section numbering is changed from “17.X” to “17.6.”

STD SUP 17.6-1

Descriptions of the programs listed in Subsection 17.6.3 of NEI 07-02A are provided in the following FSAR chapters/sections:

The maintenance rule program (Section 17.6)

The quality assurance program (Section 17.5)

Inservice inspection program (Sections 5.2 and 6.6)

Inservice testing program (Section 3.9)

The technical specifications surveillance test program (Chapter 16)

STD SUP 17.6-2

Condition monitoring of underground or inaccessible cables is incorporated into the maintenance rule program. The cable condition monitoring program incorporates lessons learned from industry operating experience, addresses regulatory guidance, and utilizes information from detailed design and procurement documents to determine the appropriate inspections, tests and monitoring criteria for underground and inaccessible cables within the scope of the maintenance rule (i.e., 10 CFR 50.65). The program takes into consideration Generic Letter 2007-01.

STD DEP 1.1-1

17.7 COMBINED LICENSE INFORMATION ITEMS

Section 17.5 of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VCS COL 17.5-1 This COL Item is addressed in Sections 17.1 and 17.5.

STD COL 17.5-2 This COL Item is addressed in Section 17.5.

STD COL 17.5-4 This COL Item is addressed in Section 17.5.

STD COL 17.5-8 This COL Item is addressed in Section 17.5.

STD DEP 1.1-1

17.8 REFERENCES

Section 17.6 of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VCS SUP 17.8-1

201. NuStart Energy, LLC., “NuStart Energy Project Instruction – Quality Assurance Plan,” PI-009.

202. “NRC Audit Report for the South Carolina Electric and Gas (SCE&G) VC Summer Nuclear Plant Combined License Application Review,” J. W. Chung to A. M. Monroe, November 16, 2007 (ML073100387).

203. Not used.

204. South Carolina Electric & Gas, “New Nuclear Deployment Quality Assurance Plan,” Rev. 2, June 4, 2009.

205. Bechtel Power Corporation, “Nuclear Quality Assurance Manual,” Rev. 4, 11/01/02.

206. South Carolina Electric & Gas, V.C. Summer Nuclear Station Unit 1 “Operational Quality Assurance Plan,” Rev. 27, February 21, 2005.

207. Nuclear Energy Institute, Technical Report NEI 06-14A, “Quality Assurance Program Description,” Revision 7, July 2009.

208. Nuclear Energy Institute, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52,” NEI 07-02A, Revision 0, March 2008 (ML080910149).

CHAPTER 18
HUMAN FACTORS ENGINEERING

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CHAPTER 18
HUMAN FACTORS ENGINEERING

18.1 OVERVIEW

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.2 HUMAN FACTORS ENGINEERING PROGRAM MANAGEMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

18.2.1.3 Applicable Facilities

Add the following information at the end of DCD Subsection 18.2.1.3:

VCS COL 18.2-2 The EOF and TSC communications strategies, as well as the EOF and TSC Human Factors attributes, are described in the Emergency Plan.

18.2.6 COMBINED LICENSE INFORMATION

18.2.6.2 Emergency Operations Facility

VCS COL 18.2-2 This COL item is addressed in Section 18.2.1.3.

18.3 OPERATING EXPERIENCE REVIEW

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.4 FUNCTIONAL REQUIREMENTS ANALYSIS AND ALLOCATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.5 AP1000 TASK ANALYSIS IMPLEMENTATION PLAN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.6 STAFFING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 18.6-1 Replace the DCD paragraph in Section 18.6 with the following information.

Table 13.1-201 contains the estimated staffing levels for those categories of personnel that are addressed by the Human Factors Engineering program per NUREG-0711, “Human Factors Engineering Program Review Model” (Reference 201), as follows:

- Licensed operators
- Shift Supervisors
- Non-licensed operators
- Shift technical advisors
- Instrumentation and control technicians
- Mechanical maintenance technicians
- Electrical maintenance technicians
- Radiation protection technicians
- Chemistry technicians
- Engineering support

The minimum level of staffing for control room personnel who directly monitor and control the plant is stated in Table 13.1-202 and meets the requirements of 10 CFR 50.54(m). Information about the staffing levels of security personnel is contained in the separately submitted physical security plan.

Qualification requirements of plant personnel listed above are discussed in Subsections 13.1.1.4, Qualifications of Technical Support Personnel, and 13.1.3, Qualification Requirements of Nuclear Plant Personnel, and, for security personnel, in the physical security plan.

The baseline level of staffing for the categories of personnel discussed above is derived from experience in current operating nuclear power plants. The number of personnel in operating plants has evolved over many years to a level that is safe and efficient and provides adequate personnel to operate the plant under all conditions, including abnormal and emergency, meets regulatory requirements, and supports individual training and personal needs.

Iterative adjustments are implemented to the level of staffing, as necessary, based on findings and input from periodic reviews and staffing analysis. Input to this analysis includes information derived from the other elements of the human factors engineering program, particularly operating experience review, functional requirements analysis and function allocation, task analysis, human reliability analysis, human-system interface design, procedure development, and training program development.

In addition to the regulatory requirements referenced, input to the analyses and the level of staffing is provided by WCAP-14694, “Designer’s Input to Determination of the AP600 Main Control Room Staffing Level” (DCD Section 18.6, Reference 1), AP1000 Combined License Technical Report APP-GW-GLR-010, “AP1000 Main Control Room Staff Roles and Responsibilities” (Reference 202), and EPRI Technical Report 1011717, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants” (Reference 203).

18.6.1 COMBINED LICENSE INFORMATION ITEM

STD COL 18.6-1 This COL Item is addressed in Section 18.6.

18.6.2 REFERENCES

201. United States Nuclear Regulatory Commission, “Human Factors Engineering Program Review Model,” NUREG-0711, Revision 2, February 2004.

202. Westinghouse, “AP1000 Main Control Room Staff Roles and Responsibilities,” APP-GW-GLR-010, Rev. 2, June 2007.

203. EPRI, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants,” Technical Report 1011717, Final Report, August 2005.

18.7 INTEGRATION OF HUMAN RELIABILITY ANALYSIS WITH HUMAN FACTORS ENGINEERING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.8 HUMAN SYSTEM INTERFACE DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

18.8.3.5 Technical Support Center Mission and Major Tasks

Replace the last sentence of the first paragraph with the following:

VCS DEP 18.8-1 The Technical Support Center (TSC) location is described in the Emergency Plan.

18.8.3.6 Operations Support Center Mission and Major Tasks

Replace the last sentence of the first paragraph with the following:

VCS DEP 18.8-1 The Operations Support Center (OSC) location is described in the Emergency Plan.

18.9 PROCEDURE DEVELOPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.10 TRAINING PROGRAM DEVELOPMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following paragraphs at the end of DCD subsection 18.10:

STD COL 18.10-1 Information regarding training program development is located in Section 13.2, Training. The training organization and roles and responsibilities of training personnel are discussed in Section 13.1, Organizational Structure of Applicant.

18.10.1 COMBINED LICENSE INFORMATION

STD COL 18.10-1 This COL Item is addressed in Section 18.10, 13.1 and 13.2.

18.11 HUMAN FACTORS ENGINEERING VERIFICATION AND VALIDATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.12 INVENTORY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.13 DESIGN IMPLEMENTATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.14 HUMAN PERFORMANCE MONITORING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Replace the DCD paragraph with the following text.

STD COL 18.14-1 Human performance monitoring applies after the plant is placed in operation. The human performance monitoring process implements the guidance and methods as described in DCD Section 18.14 Reference 1.

The human performance monitoring process provides reasonable assurance that:

- The design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centers.
- Changes made to the human system interface(s), procedures, and training do not have adverse effects on personnel performance, (e.g., a change does not interfere with previously trained skills).
- Human actions can be accomplished within time and performance criteria.
- The acceptable level of performance established during the design integrated system validation is maintained.

The human performance monitoring process is structured such that:

- Human actions are monitored commensurate with their safety importance.
- Feedback of information and corrective actions are accomplished in a timely manner.
- Degradation in performance can be detected and corrected before plant safety is compromised (e.g., by use of the plant simulator during training exercises).

The human performance monitoring process for risk-informed changes is integrated into the corrective action program, training program and other programs as appropriate. Identified human performance conditions/issues are evaluated for human factors engineering applicability.

Human factors engineering conditions are assigned specific human factors cause determination codes, trended for indications of degraded performance or potential human performance failures and have specific corrective actions identified.

The cause investigation:

- Identifies the cause of the failure or degraded performance to the extent that corrective action can be taken consistent with the corrective action program requirements.
- Addresses failure significance which includes the circumstances surrounding the failure or degraded performance, the characteristics of the failure, and whether the failure is isolated or has generic or common cause implications.
- Identifies and establishes corrective actions necessary to preclude the recurrence of unacceptable failures or degraded performance in the case of a significant condition adverse to quality.

When appropriate, design changes are integrated into training exercises to monitor for degradation in performance and allow for early detection and corrective actions before plant safety is challenged (e.g., by use of the plant simulator during training exercises).

Plant or personnel performance under actual design conditions may not be readily measurable. When actual conditions cannot be simulated, monitored, or measured, the available information that most closely approximates performance data in actual conditions should be used.

Monitoring strategies for human performance trending after the implementation of design changes is capable of demonstrating that performance is consistent with that assumed in the various analyses conducted to justify the change.

Risk-informed changes are screened commensurate with their safety importance to determine if the change requires monitoring of actions. For changes which require monitoring, the appropriate monitoring requirements are determined and implemented in the training program or other program as appropriate.

CHAPTER 19
PROBABILISTIC RISK ASSESSMENT

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CHAPTER 19
PROBABILISTIC RISK ASSESSMENT

19.1 INTRODUCTION

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19.2 INTERNAL INITIATING EVENTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.3 MODELING OF SPECIAL INITIATORS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.4 EVENT TREE MODELS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.5 SUPPORT SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.6 SUCCESS CRITERIA ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.7 FAULT TREE GUIDELINES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.8 PASSIVE CORE COOLING SYSTEM - PASSIVE RESIDUAL HEAT REMOVAL

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.9 PASSIVE CORE COOLING SYSTEM - CORE MAKEUP TANKS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.10 PASSIVE CORE COOLING SYSTEM - ACCUMULATOR

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.11 PASSIVE CORE COOLING SYSTEM - AUTOMATIC DEPRESSURIZATION SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.12 PASSIVE CORE COOLING SYSTEM - IN-CONTAINMENT REFUELING WATER STORAGE TANK

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.13 PASSIVE CONTAINMENT COOLING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.14 MAIN AND STARTUP FEEDWATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.15 CHEMICAL AND VOLUME CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.16 CONTAINMENT HYDROGEN CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.17 NORMAL RESIDUAL HEAT REMOVAL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.18 COMPONENT COOLING WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.19 SERVICE WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.20 CENTRAL CHILLED WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.21 AC POWER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.22 CLASS 1E DC & UPS SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.23 NON-CLASS 1E DC & UPS SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.24 CONTAINMENT ISOLATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.25 COMPRESSED AND INSTRUMENT AIR SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.26 PROTECTION AND SAFETY MONITORING SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.27 DIVERSE ACTUATION SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.28 PLANT CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.29 COMMON CAUSE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.30 HUMAN RELIABILITY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.31 OTHER EVENT TREE NODE PROBABILITIES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.32 DATA ANALYSIS AND MASTER DATA BANK

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.33 FAULT TREE AND CORE DAMAGE QUANTIFICATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.34 SEVERE ACCIDENT PHENOMENA TREATMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.35 CONTAINMENT EVENT TREE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.36 REACTOR COOLANT SYSTEM DEPRESSURIZATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.37 CONTAINMENT ISOLATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.38 REACTOR VESSEL REFLOODING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.39 IN-VESSEL RETENTION OF MOLTEN CORE DEBRIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.40 PASSIVE CONTAINMENT COOLING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.41 HYDROGEN MIXING AND COMBUSTION ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.42 CONDITIONAL CONTAINMENT FAILURE PROBABILITY DISTRIBUTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.43 RELEASE FREQUENCY QUANTIFICATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.44 MAAP4.0 CODE DESCRIPTION AND AP1000 MODELING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.45 FISSION PRODUCT SOURCE TERMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.46 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.47 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.48 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.49 OFFSITE DOSE EVALUATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.50 IMPORTANCE AND SENSITIVITY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.51 UNCERTAINTY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.52 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.53 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.54 LOW POWER AND SHUTDOWN PRA ASSESSMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.55 SEISMIC MARGIN ANALYSIS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.55.6.3 Site Specific Seismic Margin Analysis

VCS COL 19.59.10-6

The VCSNS site seismic demand based on the site-specific Ground Motion Response Spectra (GMRS) is enveloped by a seismic demand which combines both the Certified Seismic Design Response Spectra (CSDRS) and Hard Rock High Frequency (HRHF) design response spectra as defined by the Tier 1 criteria for SSE. Therefore, it can be concluded that the Seismic Margin Assessment analysis documented in FSAR Section 19.55 is applicable to the VCSNS Units 2 and 3 site.

The VCSNS Nuclear Island (NI) is founded on hard (sound) rock which eliminates any potential for site specific effects such as seismically induced liquefaction settlements, slope stability, foundation failure or relative displacements which would lower the HCLPF values calculated for the certified design. For non-safety related structures and foundations adjacent to the NI, these site specific effects are evaluated in FSAR Section 2.5.4 and shown to have no effect on the NI; therefore, having no potential to lower the HCLPF values calculated for the certified design.

19.56 PRA INTERNAL FLOODING ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.57 INTERNAL FIRE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.58 WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.58.3 CONCLUSION

Add the following information at the end of DCD Subsection 19.58.3:

VCS SUP 19.58-1 Table 19.58-201 documents the site specific external events evaluation that has been performed to VCSNS Units 2 and 3. This table provides a general explanation of the evaluation and resultant conclusions and provides a reference to applicable sections of the FSAR where more supporting information (including data used, methods and key assumptions) regarding the specific event is located. Based upon this evaluation, it is concluded that the VCSNS Units 2 and 3 site is bounded by the High Winds, Floods and Other External Events analysis documented in DCD Section 19.58 and APP-GW-GLR-101 (Reference 201) and no further evaluations are required at the COL application stage.

19.58.4 REFERENCES

201. Westinghouse Electric Company LLC, "AP1000 Probabilistic Risk Assessment Site-Specific Considerations," Document Number APP-GW-GLR-101, Revision 1, October 2007.

Table 19.58-201 (Sheet 1 of 3)
External Event Frequencies for VCSNS Units 2 and 3

Category	Event	Applicable to site? (Y/N)[a]	Explanation of Applicability Evaluation	Event Frequency
High Winds	EF0 Tornado	Y	Tornado activity in the surrounding counties of the VCSNS Units 2 and 3 site is provided in FSAR Table 2.3-227 from 1950 through August 2003. Due to the relative proximity of Laurens County to the other surrounding counties, activity in this area was also included within the evaluation. The event frequency was determined for each tornado category using a point probability method [**PS=n(a/A)**]. First, the average impacted area (**a**) was calculated by averaging the area of each category of tornado activity (events with an area of zero value were conservatively disregarded in determining the average area). Second, the tornado frequency (**n**) was calculated by dividing the total count of tornado events in each category including those with zero area by the measured duration (54 years). Third, the point probability of a tornado impacting a square mile (site area estimated as 1 mi^2.) is calculated by taking the product of the average impacted area and the average tornado frequency and dividing by the total area of the surrounding counties (**A**). This computation assumes that tornadoes with a zero path length have an area equal to the average area of the category.	1.17E-05
	EF1 Tornado	Y		1.26E-05
	EF2 Tornado	Y		8.38E-05
	EF3 Tornado	Y		7.34E-05
	EF4 Tornado	Y		3.91E-05
	EF5 Tornado	Y		No Recorded Events
	Cat. 1 Hurricane	Y	Historical data for tropical weather is archived by the National Coastal Services Center and covers from 1851 to 2006. FSAR Subsection 2.3.1.3.3 summarizes the frequencies of occurrence of the various categories of hurricanes that have tracked within approximately 100 nautical miles of the VCSNS site. This data was used to analyze the event frequency of hurricane activity (in an extremely conservative manner since the site is located greater than 100 miles inland from the coast) traveling in the vicinity of the VCSNS site. The storms were sorted to remove duplicate values. The event frequency is determined by dividing the number of occurrences of tropical weather by the measured duration (155 years).	4.52E-02
	Cat. 2 Hurricane	Y		1.94E-02
	Cat. 3 Hurricane	Y		6.45E-03
	Cat. 4 Hurricane	Y		6.45E-03
	Cat. 5 Hurricane	Y		No Recorded Events
	Extratropical Cyclones	Y	The "Extratropical Cyclone" subcategory of storms, used in APP-GW-GLR-101, was assigned an initiating event frequency of 3E-02 events per year. However, if an evaluation indicates a CDF less than 1.0E-08 events per year, then no detailed PRA is necessary. Initially, a 25 mile radius around the site was evaluated for extratropical storms. 5 storms were observed. When obtaining weather data for a radius of 100 nautical miles, the observed number of storms is 31. Utilizing the 31 events, the incident event frequency (IEF) increases from 3.22E-02 to 2.0E-01. The CCDP used in APP-GW-GLR-101 for the Case 1 Loss of Offsite Power (LOOP) scenario is 9.81E-09. Even with the increased event frequency, the core damage frequency (CDF) remains less than 1E-08 at 1.9E-09. Therefore, no detailed PRA is necessary. As documented in COLA FSAR Table 2.0-201, the VCSNS site characteristic tornado wind loadings are equal to the AP1000 DCD site characteristic tornado wind loadings. The VCSNS site characteristic operating basis wind speed (102 mph) is below the DCD site characteristic operating basis wind speed of 145 mph. Therefore, it is concluded that the safety features of the AP1000 are unaffected and the resultant CDFs given in APP-GW-GLR-101 Table 3.0-1 for these events are applicable to VCSNS Units 2 and 3.	2.0E-01

Table 19.58-201 (Sheet 2 of 3)
External Event Frequencies for VCSNS Units 2 and 3

Category	Event	Applicable to site? (Y/N)[(a)]	Explanation of Applicability Evaluation	Event Frequency
External Flood	External Flood	N	As discussed in COLA FSAR Subsections 2.4.1.1 and 2.4.10 the site grade of 400 ft NAVD88 (which corresponds to DCD grade elevation 100 ft.) is about 150 ft above the Broad River flood plain. Additionally, as discussed in COLA FSAR Subsections 2.4.2.2 and 2.4.2.3, the maximum water level in the power block area due to any local PMP flood event is below the entrance and openings to safety related structures. Therefore, no external flood protection measures are required for VCSNS Units 2 and3. Subsections 2.4.3 and 2.4.4 also discuss other natural and man-made (dams) flooding scenarios which further reinforce the VCSNS site is not susceptible to any external floods which would adversely impact safe operation of VCSNS Units 2 and 3.	N/A
Transportation and Nearby Facility Accidents	Aviation (commercial/ general/military)	N	Subsections 2.2.2.7 and 2.2.2.7.6 provide the detailed evaluation that confirms the probability of an aviation accident is less than 10E-07 and therefor requires no further evaluation. Therefore, it is concluded that the PRA remains applicable.	N/A
	Marine (ship/barge)	N	As discussed in FSAR Subsection 2.2.2.4, since neither the Broad River, Parr Reservoir, nor the Monticello Reservoir is used as commercial transport waterways, the potential safety effect to the site is regarded as being insignificant. Thus, no further analysis is necessary.	N/A
	Pipeline (gas/oil)	N	As stated in FSAR Subsection 2.2.2.3.1, the only pipeline in the general vicinity of the site is a 12 inch natural gas buried pipeline located greater than a mile from VCSNS Units 2 and 3. This pipeline is bounded by the evaluation performed in APP-QW-GLR-101, and therefore no further evaluation is necessary.	N/A
	Railroad	N	Potential explosion and flammable vapor cloud hazards to VCNS Units 2 and 3 resulting from railroad accidents are discussed in FSAR Subsection 2.2.3.1.1.3. The results of this evaluation concluded that no adverse impacts to VCSNS Units 2 and 3 are expected. Based upon the quantitative consequence evaluations performed, no risk-important events related to rail transportation have been identified for VCSNS Units 2 and 3. Therefore, the potential for hazards from these sources are minimal and will not adversely affect safe operation of VCSNS Units 2 and 3.	N/A
	Truck	N	Potential hazards resulting from trucks were discussed in FSAR 2.2.2.5. The evaluation that was performed to address the explosion of a tanker truck on site as it filled on-site storage tanks was considered bounding for any highway accident and therefore no additional evaluation was required. The evaluations to address these onsite truck hazards are described in FSAR Subsections 2.2.3.1.1.1 and 2.2.3.1.2.1, and the results of these evaluations concluded that the hazards do not result in any significant damage to the plant.	N/A

Table 19.58-201 (Sheet 3 of 3)
External Event Frequencies for VCSNS Units 2 and 3

Category	Event	Applicable to site? (Y/N)[a]	Explanation of Applicability Evaluation	Event Frequency
Other Events	A number of external events beyond those evaluated in DCD Subsection 19.58 were evaluated for the VCSNS site. These events are discussed below.		Based on the evaluations below, these events do not pose a credible threat to the safe operation of VCSNS Units 2 and 3. Thus, these events are not considered to be risk-important and it can be concluded that the VCSNS Units 2 and 3 site is within the bounds of the Floods and Other External Events analysis documented in DCD Tier 2 Section 19.58.	
	Additional events at nearby facilities	N	Based on the discussions in FSAR Subsections 2.2.3.1.1, 2.2.3.1.2 and 2.2.3.1.3, the effects of explosions, flammable vapor clouds and toxic chemicals at the Parr Combustion Turbines and VCSNS Unit 1 were evaluated and determined to meet the safe distance requirements and toxicity limits of Regulatory Guides 1.91 and 1.78. Therefore, because no risk significant consequences were identified for these events, the potential safety effect to the site is regarded as being insignificant. Thus, no further analysis is necessary.	N/A
	External fires	N	As stated in FSAR Subsection 2.2.3.1.4, for an assumed wildfire in the vegetation surrounding the site, given the low incident heat flux calculated, the long separation distances to safety-related structures, and the various conservatisms assumed in the analysis, a wildfire would not affect the safe operation or shutdown of Units 2 and 3. In addition, as described in Section 2.2.2, due to the lack of other facilities with hazardous materials that could create nonflammable gases or chemical bearing clouds as a result of a forest fire located within 5 miles of the site, these clouds are not considered to be a concern. Therefore. no further evaluation is necessary for these external fire events.	N/A

a) An event is applicable (Y) to the VCSNS site if the initiating event frequency is greater than 1E-07, or if a quantitative consequence evaluation has demonstrated that there are site specific parameters that exceed the parameters used in APP-GW-GLR-101. An event is not applicable (N) to the VCSNS site if the initiating event frequency is less than 1E-07 or if the quantitative consequence evaluation has demonstrated that the event will not adversely impact the safe operation of VCSNS Units 2 and 3.

19.59 PRA RESULTS AND INSIGHTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.59.10.5 Combined License Information

STD COL 19.59.10-1
STD COL 19.59.10-6

A review of the differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis will be completed prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design. A comparison of the as-built SSC high confidence, low probability of failures (HCLPFs) to those assumed in the AP1000 seismic margin evaluation will be performed prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis will be evaluated to determine if vulnerabilities have been introduced.

The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:

1. Specific minimum seismic requirements consistent with those used to define the AP1000 DCD Table 19.55-1 HCLPF values.

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of failure is expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the seismic margin insights of the site-specific PRA. The range of frequency response that is required for the equipment with its structural support is defined.

2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.

STD COL 19.59.10-2

A review of the differences between the as-built plant and the design used as the basis for the AP1000 PRA and DCD Table 19.59-18 will be completed prior to fuel load. The plant specific PRA-based insight differences will be evaluated and the plant specific PRA model modified as necessary to account for plant-specific design and any design changes or departures from the design certification PRA.

As discussed in Section 19.58.3, it has been confirmed that the Winds, Floods, and Other External Events analysis documented in DCD Section 19.58 is applicable to the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Section 19.58 of the AP1000 DCD is applicable to this design.

STD COL 19.59.10-3

A review of the differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analyses will be completed prior to fuel load. Plant specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design.

STD COL 19.59.10-4

The AP1000 Severe Accident Management Guidance (SAMG) from APP-GW-GLR-070, Reference 1 to DCD Section 19.59, is implemented on a site-specific basis. Key elements of the implementation include:

- SAMG based on APP-GW-GLR-070 is provided to Emergency Response Organization (ERO) personnel in assessing plant damage, planning and prioritizing response actions and implementing strategies that delineate actions inside and outside the control room.
- Severe accident management strategies and guidance are interfaced with the Emergency Operating Procedures (EOP's) and Emergency Plan.
- Responsibilities for authorizing and implementing accident management strategies are delineated as part of the Emergency Plan.
- SAMG training is provided for ERO personnel commensurate with their responsibilities defined in the Emergency Plan.

STD COL 19.59.10-5

A thermal lag assessment of the as-built equipment required to mitigate severe accidents (hydrogen igniters and containment penetrations) will be performed to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment will be performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The ability of the as-built equipment to perform during severe accident hydrogen burns will be assessed using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354 (DCD Section 19.59, Reference 3).

STD COL 19.59.10-6 VCS COL 19.59.10-6

As discussed in Subsection 19.55.6.3, it has been confirmed that the Seismic Margin Analysis (SMA) documented in DCD Section 19.55 is applicable to the site. The site-specific effects (i.e., soil-related failure modes, etc.) have been evaluated and it was concluded that the plant-specific plant-level HCLPF value is equal to or greater than 1.67 times the site-specific GMRS peak ground acceleration.

Add the following new information after DCD Subsection 19.59.10.5:

STD SUP 19.59-1

19.59.10.6 PRA Configuration Controls

PRA configuration controls contain the following key elements:

- A process for monitoring PRA inputs and collecting new information.
- A process that maintains and updates the PRA to be reasonably consistent with the as-built, as operated plant.
- A process that considers the cumulative impact of pending changes when applying the PRA.
- A process that evaluates the impact of changes on currently implemented risk-informed decisions that have used the PRA.
- A process that maintains configuration control of computer codes used to support PRA quantification.
- A process for upgrading the PRA to meet PRA standards that the NRC has endorsed.
- Documentation of the PRA.

PRA configuration controls are consistent with the regulatory positions on maintenance and upgrades in Regulatory Guide 1.200.

<u>Schedule for Maintenance and Upgrades of the PRA</u>

The PRA update process is a means to reasonably reflect the as designed and as operated plant configurations in the PRA models. The PRA upgrade process includes an update of the PRA plus a general review of the entire PRA model, and as applicable, the application of new software that implements a different methodology, implementation of new modeling techniques, as well as a comprehensive documentation effort.

- During construction, the PRA is upgraded prior to fuel load to cover those initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to the scheduled date of the initial fuel load for a Level 1 and Level 2 PRA.

- Prior to license renewal the PRA is upgraded to include all modes of operation.

- During operation, PRA updates are completed as part of the upgrade process at least once every four years.

- A screening process is used to determine whether a PRA update should be performed more frequently based upon the nature of the changes in design or procedures. The screening process considers whether the changes affect the PRA insights. Changes that do not meet the threshold for immediate update are tracked for the next regulatory scheduled update. If the screening process determines that the changes do warrant a PRA update, the update is made as soon as practicable consistent with the required change importance and the applications being used.

PRA upgrades are performed in accordance with 10 CFR 50.71(h).

Process for Maintenance and Upgrades of the PRA

Various information sources are monitored to determine changes or new information that affects the model assumptions or quantification. Plant specific design, procedure, and operational changes are reviewed for risk impact. Information sources include applicable operating experience, plant modifications, engineering calculation revisions, procedure changes, industry studies, and NRC information.

The PRA upgrade includes initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade.

This PRA maintenance and update incorporates the appropriate new information including significant modeling errors discovered during routine use of the PRA.

Once the PRA model elements requiring change are identified, the PRA computer models are modified and appropriate documents revised. Documentation of modifications to the PRA model include the changes as well as the upgraded portions clearly indicating what has been changed. The impact on the risk insights is clearly indicated.

PRA Quality Assurance

Maintenance and upgrades of the PRA are subject to the following quality assurance provisions:

Procedures identify the qualifications of personnel who perform the maintenance and upgrade of the PRA.

Procedures provide for the control of PRA documentation, including revisions.

For updates of the PRA, procedures provide for independent review, or checking of the calculations and information.

Procedures provide for an independent review of the model after an upgrade is completed. Additionally, after the PRA is upgraded, the PRA is reviewed by outside PRA experts such as industry peer review teams and the comments incorporated to maintain the PRA current with industry practices. Peer review findings are entered into a tracking system. PRA upgrades receive a peer review for those aspects of the PRA that are upgraded.

PRA models and applications are documented in a manner that facilitates peer review as well as future updates and applications of the PRA by describing the processes that were used, and provide details of the assumptions made and their bases. PRA documentation is developed such that traceability and reproducibility is maintained. PRA documentation is maintained in accordance with Regulatory Position 1.3 of Regulatory Guide 1.200.

Procedures provide for appropriate attention or corrective actions if assumptions, analyses, or information used previously are changed or determined to be in error. Potential impacts to the PRA model (i.e., design change notices, calculations, and procedure changes) are tracked. Errors found in the PRA model between periodic updates are tracked using the site tracking system.

PRA-Related Input to Other Programs and Processes

The PRA provides input to various programs and processes, such as the Maintenance Rule implementation, reactor oversight process, the RAP, and the RTNSS program. The use of the PRA in these programs is discussed below, or cross-references to the appropriate FSAR sections are provided.

PRA Input to Design Programs and Processes

The PRA insights identified during the design development are discussed in DCD Subsection 19.59.10.4 and summarized in DCD Table 19.59-18. DCD Section 14.3 summarizes the design material contained in AP1000 that has been incorporated into the Tier 1 information from the PRA. A discussion of the plant features important to reducing risk is provided in DCD Subsection 19.59.9.

PRA Input to the Maintenance Rule Implementation

The PRA is used as an input in determining the safety significance classification and bases of in-scope SSCs. SSCs identified as risk-significant via the Reliability Assurance Program for the design phase (DRAP, Section 17.4) are included within the initial Maintenance Rule scope as high safety significance SSCs.

For risk-significant SSCs identified via DRAP, performance criteria are established, by the Maintenance Rule expert panel using input from the reliability and availability assumptions used in the PRA, to monitor the effectiveness of the maintenance performed on the SSCs.

The Maintenance Rule implementation is discussed in Section 17.6.

PRA Input to the Reactor Oversight Process

The mitigating systems performance indicators (MSPI) are evaluated based on the indicators and methodologies defined in NEI 99-02 (Reference 201).

The Significance Determination Process (SDP) uses risk insights, where appropriate, to determine the safety significance of inspection findings.

PRA Input to the Reliability Assurance Program

The PRA input to the Reliability Assurance Program is discussed in DCD Subsection 19.59.10.1.

PRA Input to the Regulatory Treatment of Nonsafety-Related Systems Programs

The importance of nonsafety-related SSCs in the AP1000 has been evaluated using PRA insights to identify SSCs that are important in protecting the utility's investment and for preventing and mitigating severe accidents. These investment protection systems, structures and components are included in the D-RAP/MR Program (refer to Subsection 17.4), which provides confidence that availability and reliability are designed into the plant and that availability and reliability are maintained throughout plant life through the Maintenance Rule. Technical Specifications are not required for these SSCs because they do not meet the selection criteria applied to the AP1000 (refer to Subsection 16.1.1).

MOV Program

The MOV Program includes provisions to accommodate the use of risk-informed inservice testing of MOVs (Subsection 3.9.6).

19.59.11 REFERENCES

Add the following text to the end of DCD Subsection 19.59.11:

201. NEI 99-02, Nuclear Energy Institute, "Regulatory Assessment Performance Indicator Guideline," Technical Report NEI 99-02, Revision 5, July 2007.

APPENDIX 19A
THERMAL HYDRAULIC ANALYSIS TO SUPPORT SUCCESS CRITERIA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19B
EX-VESSEL SEVERE ACCIDENT PHENOMENA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19C
ADDITIONAL ASSESSMENT OF AP1000 DESIGN FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19D
EQUIPMENT SURVIVABILITY ASSESSMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19E
SHUTDOWN EVALUATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19F
MALEVOLENT AIRCRAFT IMPACT

This section of the referenced DCD is incorporated by reference with no departures or supplements.