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Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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Docket No. 50-395
License No. NPF-12

DOMINION ENERGY SOUTH CAROLINA (DESC)
VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
TECHNICAL SPECIFICATION BASES CHANGES
UPDATED THROUGH NOVEMBER 2022

In accordance with Virgil C. Summer Nuclear Station (VCSNS) Unit 1 Technical Specifications (TS) 6.8.4.i.4, Dominion Energy South Carolina (DESC), acting for itself and as agent for South Carolina Public Service Authority, submits changes to the TS Bases.

This update includes changes to the TS Bases since the previous submittal in December 2021. The enclosed changes were revised by License Amendment 222. Changes are annotated by vertical revision bars and the amendment number at the bottom of the affected TS Bases pages.

Should you have any questions, please call Michael S. Moore at (803) 345-4752.

Sincerely,

Robin R. Haselden
Director, Nuclear Station Safety and Licensing
V.C. Summer Nuclear Station

Commitments contained in this letter: None

Enclosure 1: Summary of TS Bases Changes Through November 2022
Enclosure 2: Technical Specification Bases Changes Updated Through November 2022

cc: G. J. Lindamood – Santee Cooper
L. Dudes – NRC Region II
G.E. Miller – NRC Project Manager
NRC Resident Inspector

Enclosure 1
Summary of TS Bases Changes Through November 2022

License Amendment No. 222

Description of Change:

Technical Specification (TS) Bases were updated to address the relocation of the affected TS surveillance frequencies to the Surveillance Frequency Control Program (SFCP) following the adoption of Technical Specification Task Force (TSTF)-425, Revision 3 (ML090850642). Specifically, the following two insertions were implemented as indicated in V. C. Summer Nuclear Station's (VCSNS) application dated April 8, 2021 (ML21102A127), as supplemented by letter dated January 20, 2022 (ML22020A226).

> Insert 1
> in accordance with the Surveillance Frequency Control Program
>
> Insert 2
> The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Reason and Basis for Change:

The NRC published a notice in the Federal Register on July 6, 2009 (74 FR 31996) announcing the availability of TSTF-425, Revision 3, for adoption by licensees and provided a model safety evaluation (SE) for the NRC staff to use to more efficiently process License Amendment Requests to adopt TSTF-425, Revision 3.

When implemented, TSTF-425, Revision 3, relocates most periodic frequencies of TS surveillance requirements to the SFCP, and provides requirements for the new SFCP in the Administrative Controls section of the TS.

The requirements for the SFCP were added to the administrative controls section 6.8.4.o, "Surveillance Frequency Control Program" under TS Amendment 222. The SFCP allows VCSNS to make changes to the relocated surveillance frequencies in accordance with the guidance in NEI 04-10, Revision 1.

Enclosure 2
Technical Specification Bases Changes
Updated Through November 2022

Amendment #	Pages Affected
222	B 3/4 1-4
	B 3/4 2-5
	B 3/4 3-1
	B 3/4 3-4
	B 3/4 3-5
	B 3/4 4-1a
	B 3/4 4-2
	B 3/4 4-4h
	B 3/4 4-5
	B 3/4 4-6
	B 3/4 4-14a
	B 3/4 5-2a
	B 3/4 6-3a
	B 3/4 7-4e
	B 3/4 8-5
	B 3/4 9-3

MOVABLE CONTROL ASSEMBLIES (Continued)

For purposes of determining compliance with Technical Specification 3.1.3.1, any inoperability of full length control rod(s), due to being immovable, invokes ACTION statement "a".

The intent of Technical Specification 3.1.3.1 ACTION Statement "a" is to ensure that before leaving ACTION Statement "a" and utilizing ACTION Statement "c" that the rod urgent failure alarm is illuminated or that an obvious electrical problem is detected in the rod control system by minimal electrical troubleshooting techniques. Expeditious action will be taken to determine if rod immovability is due to an electrical problem in the rod control system.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experience3d during a reactor trip at operating conditions.

Control rod position and OPERABILITY of the rod position indicators are required to be verified in accordance with the Surveillance Frequency Control Program with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$ value.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt power ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or the PDMS are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum of DNBR in the core at or above the design limit throughout each analyzed transient. The maximum indicated T_{avg} limit of 589.2°F and the minimum indicated pressure limit of 2206 psig correspond to analytical limits of 591.4°F and 2185 psig respectively, read from control board indications.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed in accordance with the Surveillance Frequency Control Program are sufficient to demonstrate this capability. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Maintenance outage times have been determined in accordance with WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and Westinghouse letter CGE-05-46. Specified surveillance intervals and RTB outage times have been determined in accordance with WCAP-15376-P-A, Rev. 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," dated March 2003. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

Consistent with the requirement in Regulatory Guide 1.177 to include Tier 2 insights into the decision-making process before taking equipment out of service, restrictions on concurrent removal of certain equipment when a logic train is inoperable for maintenance are included (note that these restrictions do not apply when a logic train is being tested under the 4-hour bypass Note). Entry into Actions 12, 14, 21, or 25 is not a typical, pre-planned evolution during power operation, other than for surveillance testing. Since Actions 12, 14, 21, or 25 are typically entered due to equipment failure, it follows that some of the following restrictions may not be met at the time of entry into Actions 12, 14, 21, or 25. If this situation were to occur during the 24-hour AOT of Actions 12, 14, 21, or 25, the configuration risk assessment procedure will assess the emergent condition and direct activities to restore the inoperable logic train and exit Actions 12, 14, 21, or 25, or fully implement these restrictions, or perform a unit shutdown, as appropriate from a risk management perspective. The following restrictions will be observed:

- To preserve ATWS mitigation capability, activities that degrade the availability of the emergency feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic train is inoperable for maintenance.

3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.11 POWER DISTRIBUTION MONITORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are in accordance with the Surveillance Frequency Control Program. Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.
2. Assure sufficiently accurate calibration of these sensors.
3. Assure an adequate calibration data base regarding the number of data sets.
4. Assure the overall accuracy of the calibration.

POWER DISTRIBUTION MONITORING SYSTEM (PDMS) (Continued)

The minimum number of required plant and core condition inputs includes the following:

1. Control Bank Positions.
2. At least 50% of the cold leg temperatures.
3. At least 75% of the signals from the Power Range excore detector channels (comprised of a top and bottom detector section).
4. Reactor Power Level.
5. A minimum number and distribution of operable core exit thermocouples.
6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of items 1., 2., 3., and 4. above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing factors which cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)

RHR system piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the RHR loops and may also prevent water hammer, pump cavitation, and pumping of non-condensable gas into the reactor vessel.

Selection of RHR System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrument drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

Surveillance Requirement 4.4.1.3.4 is modified by a Note that states the Surveillance Requirement is not required to be performed until 12 hours after entering MODE 4. In a rapid shutdown, there may be insufficient time to verify all susceptible locations prior to entering MODE 4.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operating relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will be performed in accordance with the provisions of the ASME OM Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady sate envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES (PORVs)

The pressurizer power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. The PORVs and block valves may be used to depressurize the RCS when normal pressurizer spray is unavailable. Operation of the air operated PORVs minimizes the undesirable opening of the spring loaded pressurizer code safety valves. Each PORV has a remotely controlled motor-operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The series arrangement of the PORV and its associated block valve permit surveillance while at power.

OPERATIONAL LEAKAGE (Continued)

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and containment sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System, Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

4.4.6.2.2 This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

Those PIVs listed in TS Table 3.4-1 will be demonstrated OPERABLE by verifying their leakage rates are within TS allowable leakage limits via testing performed in accordance with the Surveillance Frequency Control Program.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 2. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cool down are limited by curves developed using the methodology from Westinghouse Topical Report, WCAP-14040-NP-A, updated to include the requirements of the 1998 ASME Boiler and Pressure Vessel Code, Section XI, through the 2000 Addenda, Appendix G, along with ASME Code Case N-641.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHRSRVs or an RCS vent opening of at least 2.7 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than equal to 300°F. Either RHRSRV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stream generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of an HPSI pump and its injection into a water solid RCS.

The limitation for a maximum of one charging pump to be capable of injecting into the RCS, and the Surveillance Requirement to verify at least two charging pumps are demonstrated to be INOPERABLE in accordance with the Surveillance Frequency Control Program, while the RCS is below 300°F, provides assurance that a mass addition transient can be mitigated by a single RHR suction relief valve.

ECCS SUBSYSTEMS (Continued)

(e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

Surveillance Requirement 4.5.2.b.1) is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent path who is in continuous communication with the operators in the control room. The individual will have a method to rapidly close the system vent flow path if directed.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA, a steamline break or inadvertent RCS depressurization. The limits of RWST minimum volume and boron concentration ensure 1) that sufficient water is available within containment to permit recirculation cooling flow to the core, 2) that the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, Spray Additive Tank (SAT), containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) that the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area ≥ 3.0 sq. ft.) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (Post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

REACTOR BUILDING SPRAY SYSTEM (Continued)

Containment Spray System flow path piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the containment spray trains and may also prevent a water hammer and pump cavitation.

Selection of Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrument drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Surveillance Requirement 4.6.2.1.a.1) is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent path who is in continuous communication with the operators in the control room. The individual will have a method to rapidly close the system vent flow path if directed.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.7.6.b

Standby systems should be checked periodically to ensure that they function properly. In accordance with the Surveillance Frequency Control Program provide an adequate check of this system. The VCSNS CREFS does not have heaters and each train need only be operated for a minimum of 15 minutes to demonstrate the function of the system. The Surveillance Frequency is controlled by the Surveillance Frequency Control Program.

SR 4.7.6.c

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP implementing procedures.

SR 4.7.6.d

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is controlled by the Surveillance Frequency Control Program.

SR 4.7.6.e

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, ACTION 3.7.6.a.2 must be entered. Action 3.7.6.a.2 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 5)

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to the medium voltage (7.2 kV) circuit breakers provide assurance of breaker reliability by testing 10% of the circuit breakers and their associated relays and control circuits in accordance with the Surveillance Frequency Control Program on a rotating basis. The only medium voltage conductors that penetrate the containment are for the three RCP motors. The breakers associated with these conductors are listed on FSAR Figure 8G-2. A failure of any portion of the integrated system (relays, control circuit and circuit breaker) results in an inoperable circuit breaker. A retest in accordance with Surveillance Requirement 4.8.4.1.a.1.(c) of an additional representative sample of at least 10% of all circuit breakers equates to a retest of one circuit breaker and its associated relays and control circuit. Retests are performed until no more failures are found or all breakers have best tested.

The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The surveillance requirements of the circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that, if faulted, could cause failure in both adjacent, redundant Class 1E cables ensures that the integrity of Class 1E cables is not compromised by the failure of protection devices to operate in the non-Class 1E cables.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

The Surveillance Frequency is controlled by the Surveillance Frequency Control Program.

3/4.9.8 DELETED BY AMENDMENT 183

3/4.9.9 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99.5% of the assumed 16% I-131 and 10% other halogens gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.