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August 31, 2011

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

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION UNIT 1
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION BASES REVISION
UPDATED THROUGH JUNE 2011

In accordance with Technical Specification 6.8.4.i, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, submits revisions to the Technical Specification (TS) Bases in accordance with the Technical Specification Bases Control Program.

This TS Bases update includes changes to the TS Bases since the previous submittal in June 2010. These changes were made under the provisions of 10CFR50.59. Changes are annotated by vertical revision bars and the Revision Notice number at the bottom of the page.

If you have any questions or require additional information, please contact Bruce Thompson at (803) 931-5042.

Very truly yours,

Thomas D. Gatlin

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TECHNICAL SPECIFICATION BASES REVISIONS UPDATED THROUGH JUNE 2011

Revision Notice #	Date Approved	Pages Affected
BRN 10-001	07/28/10	B 3/4 6-5
BRN 11-001	01/13/11	B 3/4 4-3a B 3/4 4-3e B 3/4 4-4a B 3/4 6-1 B 3/4 7-2 B 3/4 7-6 B 3/4 9-1 B 3/4 9-2
BRN 11-002	06/22/11	B 3/4 7-3

INSTRUCTION SHEET V.C. SUMMER NUCLEAR STATION UNIT 1 TECHNICAL SPECIFICATION

Remove Pages	Insert Pages
B 3/4 4-3a	B 3/4 4-3a
B 3/4 4-3e	B 3/4 4-3e
B 3/4 4-4a	B 3/4 4-4a
B 3/4 6-1	B 3/4 6-1
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B 3/4 9-2	B 3/4 9-2

SUMMARY OF BASES CHANGES

BRN No. 10-001

Description of Change: This activity revises the Bases for Technical Specification (TS) 3/4.6.2.3 which was previously approved under License Amendment No. 69, Reactor Building Cooling System.

Reason and Basis for Change: License Amendment No. 69 was approved by the NRC to change Service Water flow from 4000 GPM to both Reactor Building Cooling Units (RBCUs) to 2000 GPM to each selected RBCU in TS 3/4.6.2.3. This change was made to the TS but not to the corresponding Bases which stated 4000 GPM to both RBCUs.

BRN No. 11-001

Description of Change: Alternative Source Term License Amendment No.183.

Reason and Basis for Change: The Bases changes were directly related to the Alternative Source Term License Amendment Request 183. This amendment implements an alternative source term application methodology for analyzing the radiological consequences for six design-basis accidents.

BRN No. 11-002

Description of Change: Revise the basis for TS 3.7.1.6 to explain when to apply the statement "The provisions of Specification 3.0.4 are not applicable."

Reason and Basis for Change: During a plant startup, the statement "The provisions of Specification 3.0.4 are not applicable" in the Limiting Condition for Operation 3.7.1.6 was mis-applied and the station entered Mode 1 with one Feedwater Isolation Valve inoperable. This change clarifies when to apply this statement.

opening of valves XVB-3107A(B)-SW, the air in the piping will act as a cushion to minimize any water hammer affects that could occur downstream of XVB-3107A(B)-SW. The opening logic of valves XVB-3107A(B)-SW has a delayed opening after valve 3106A(B)-SW begins to open. The delay allows fluid flow momentum to build to assure that additional void formation in the RBCU piping inside containment will not occur during swap over to the SW system.

To minimize the effects of the second water hammer scenario XVB-3107A(B)-SW, fast closing air operated butterfly valves, close in seven seconds upon de-energizing of the SWBPs. During times that the RBCUs are aligned with the SW system, if a LOOP were to occur, the fast valve closure will trap water in the high points above the valve and prevent void formation due to gravity drain down of the water to the SW pond. Interface logic is provided to equipment controls that tie the start of the respective SWBP to the closed position of the respective valve XVB-3107A(B)-SW. The controls prevent a SWBP start if the respective valve XVB-3107A(B)-SW failed to fully close allowing drain down of the water to the SW Pond.

The accident analysis requires the service water booster pump to be passing 2,000 gpm to each selected RBCU within 86.5 seconds. This time encompasses the driving of all necessary service water valves to the correct positions, i.e., fully opened or fully closed. Reference Technical Specification Bases B 3/4.3.1 and B 3/4.3.2 for additional details.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

The OPERABILITY of the containment filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the reactor building atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the reactor building atmosphere or pressurization of the reactor building and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits required by the safety analysis for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The accident analysis for a SGTR event accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Contaminated fluid in a ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves. To maximize its contribution to the dose releases, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be greater than or equal to the limits in LCO 3.4.8, "Reactor Coolant System, Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 50.67 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity. Refer to Action a. below.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.k and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

STEAM GENERATOR TUBE INTEGRITY (Continued)

Surveillance Requirements (Continued)

4.4.5.2 During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering MODE 4 following a SG inspection" ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. 10 CFR 50, Appendix A, GDC 19, "Control Room"
3. 10 CFR 50.67, "Accident Source Term"
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976
6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines"

OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR analysis for SGTR accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Leakage through the ruptured tube is the dominate contributor to dose releases. Since contaminated fluid in the ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant. Overall, this pathway is a small contributor to dose releases.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire 1 gpm primary-to-secondary leakage is through the effected steam generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates (including those used in demonstrating a 30 day water seal) ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservation, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the Containment Leakage Rate Testing Program

3/4.6.1.3 REACTOR BUILDING AIR LOCKS

The limitations on closure for the reactor building air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each emergency feedwater pump is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of two out of three steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

Also, each Emergency Feedwater (EFW) pump is capable of supplying 400 gpm to all 3 steam generators while the steam generators are pressurized to 1211 psig. This capacity is sufficient to ensure that the pressurizer does not overfill during a loss of normal feedwater event. The total head criteria of 3800 feet for the motor driven EFW pumps and 3140 feet for the turbine driven EFW pump includes margin that allows for a maximum EFW flow control valve leakage of 5 gpm for any one of 6 EFW flow control valves.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 11 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 50.67 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

3/4.7.10 WATER LEVEL - SPENT FUEL POOL

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.

3/4.7.11 DELETED BY AMENDMENT 183

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

3/4.9.4 DELETED BY AMENDMENT 183

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that:
1) manipulator cranes will be used for movement of control rods and fuel assemblies,
2) each crane has sufficient load capacity to lift a control rod and fuel assembly, and
3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

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.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event the steam line rupture occurs within the reactor building. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 FEEDWATER ISOLATION VALVES

The OPERABILITY of the Feedwater Isolation Valves serves to 1) limit the effects of a Steam Line rupture by minimizing the positive reactivity effects of the Reactor Coolant System Cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event of a Steam Line or Feedwater Line rupture within the reactor building.

The statement "The provisions of Specification 3.0.4 are not applicable" only applies when transitioning from mode 3 to mode 2. The provisions of Specification 3.0.4 are applicable when transitioning from mode 2 to mode 1.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on the average impact values of the steam generator material at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.