Thomas D. Gatlin
Vice President, Nuclear Operations
803.345.4342

SCE&G
A SCANA COMPANY

June 12, 2014

U. S. Nuclear Regulatory Commission
Document Control Desk
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 MAY 2014

In accordance with Virgil C. Summer Nuclear Station Unit 1 Technical Specification (TS) 6.8.4.i.4, South Carolina Electric & Gas Company, acting for itself and as agent for South Carolina Public Service Authority, submits a revision to the TS Bases.

This TS Bases update includes changes to the TS Bases since the previous submittal in July 2013. TS Bases change Bases Revision Notice (BRN) 13-001 was implemented under the provision of 10 CFR 50.59 while Amendments 195, 196 and 197 were made in accordance with 10 CFR 50.90. Changes are annotated by vertical revision bars and the BRN number or license amendment 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

WCM/TDG/ts
Attachment

c: K. B. Marsh
S. A. Byrne
J. B. Archie
N. S. Carns
J. H. Hamilton
J. W. Williams
W. M. Cherry
V. M. McCree
S. A. Williams
NRC Resident Inspector
K. M. Sutton
NSRC
RTS (RR 8925, LTD 338)
File (813.20)
PRSF (RC-14-0086)

A001
NRR

TECHNICAL SPECIFICATION BASES REVISIONS
UPDATED THROUGH MAY 2014

Revision Notice #	Date Approved	Pages Affected
BRN 13-001	09/24/13	B 3/4 0-2b
AMENDMENT NO. 195	02/06/14	B 3/4 7-4f B 3/4 7-5 delete
AMENDMENT NO. 196	02/28/14	B 3/4 4-3a B 3/4 4-3c B 3/4 4-3d B 3/4 4-3e
AMENDMENT NO. 197	03/11/14	B 3/4 7-4d

SUMMARY OF BASES CHANGE

BRN No. 13-001
Description of Change: Technical Specification (TS) Bases section 4.0.3, page 3/4 0-2b, was revised replacing reference to NRC Regulatory Guide (RG) 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants," with new reference Regulatory Guide 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," as RG 1.182 was superseded by RG 1.160, Revision 3.

Reason and Basis for Change: Regulatory Guide 1.160, Revision 3, was issued in May 2012. RG 1.160, Section A, Introduction, states, "Revision 4A to NUMARC 93-01 incorporates guidance previously contained in Regulatory Guidance 1.182, Revision 0, 'Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants,' issued May 2000 (Ref 3). Therefore this revision to Regulatory Guide 1.160 supersedes Regulatory Guide 1.182, Revision 0." Stated within the Regulatory Guide was that RG 1.160 endorsed Revision 4A to NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." NUMARC 93-01, Revision 4A provides guidance acceptable to the NRC for complying with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

TS Amendment No. 195
Description of Change: The Bases for TS 3/4.7.7, "Snubbers," previously approved under License Amendment No. 195 (ML13354B643), was revised by changing TS snubber surveillance requirements. The amendment resulted in a revision to TS Bases stating that the Snubber Testing Program manages the requirement for demonstrating snubber operability. The change also ensures consistency between TS and the Inservice Inspection (ISI) program. The change also ensures the ISI program for testing snubbers is in accordance with the ASME OM Code and applicable addenda per 10 CFR 50.55a(g).

Reason and Basis for Change: License Amendment No. 195 was issued based upon completion of the third 10-year ISI interval and revision to the ISI program for the fourth 10-year interval. Per 10 CFR 50.55a(g)(5)(ii), licensees are required to apply to the NRC for an amendment to the TS if a revised ISI program conflicts with current TS. Amendment No. 195 removed the specific surveillance requirements for demonstrating snubber operability from TS by the adoption of Subsection ISTD (Inservice Testing of Dynamic Restraints), "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," of the ASME OM Code, 2004 Edition with 2005 and 2006 Addenda.

TS Amendment No. 196
Description of Change: The Bases for TS 3/4.4.5, "Steam Generator Tube Integrity," previously approved under License Amendment No. 196 (ML14010A182), was revised to reflect adoption of TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

Reason and Basis for Change: License Amendment No. 196 was issued based upon adoption of guidance provided in NRC approved TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." TSTF-510 was reviewed by SCE&G and concluded that justifications presented in TSTF-510 and the Model Safety Evaluation prepared by the NRC staff to be applicable to VCSNS Unit 1 and justified incorporation of the changes into VCSNS TS. Amendment 196 revised TSs 3/4.4.5, "Steam Generator Tube Integrity," 6.8.4.k, "Steam Generator (SG) Program," and 6.9.1.12, "Steam Generator Tube Inspection Report."

TS Amendment No. 197

Description of Change: The Bases for TS 3/4.7.6, "Control Room Emergency Filtration System (CREFS)," previously approved under License Amendment No. 197 (ML14030A374), was revised to add an exception statement to Specification 3.0.4 in TS 3/4.7.6, "Control Room Emergency Filtration System (CREFS)." The exception statement was inadvertently deleted under License Amendment No. 180.

Reason and Basis for Change: License Amendment No. 197 was approved to restore the statement "The provisions of Specification 3.0.4 are not applicable" to Action b. of TS 3/4.7.6. Action b. is applicable in Modes 5 and 6 and contains 2 action statements. The first Action statement 3.7.6.b.1 addresses the condition of one CREFS train inoperable for reasons other than an inoperable control room envelope (CRE) boundary. The second Action statement 3.7.6.b.2 addresses the condition of both CREFS trains inoperable or one or more CREFS trains inoperable due to an inoperable CRE boundary. The change to TS Bases reflects addition of a third Action statement, exception statement 3.7.6.b.3, "The provisions of Specification 3.0.4 are not applicable," to CREFS TS Action b.

SR 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of MODE changes imposed by required Actions.

Failure to comply with specified frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility, which is not intended to be used as an operational convenience to extend surveillance intervals.

While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementing guidance, NRC Regulatory Guide 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Allowed Outage Time (AOT) for the required Action for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is considered outside the specified limits and the AOT of the required Action for the applicable LCO begin immediately upon the failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the AOT of the Action, restores compliance with SR 4.0.1.

4.0.4 This specification establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Action 3.7.6.a.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

1. FSAR, Section 9.4.
2. FSAR, Chapter 15.
3. FSAR, Section 6.4.
4. FSAR, Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

3/4.7.7 SNUBBERS

All snubbers on systems required for safe shutdown/accident mitigation shall be OPERABLE. This includes safety and non-safety related snubbers on systems used to protect the code boundary and to ensure the structural integrity of these systems under dynamic loads.

The "Snubber Testing Program" manages the requirement for demonstrating snubber operability (examination, testing and service life monitoring) as reflected in section 6.8.4.n thereby ensuring the TS remains consistent with the ISI program. The program for ISI and testing of snubbers in accordance with ASME OM Code and the applicable addenda as required by 10 CFR 50.55a(g) is required to include evaluation of supported components/systems when snubbers are found to be unacceptable.

Bases Page 3/4 7-5 deleted

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 plugging criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging 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)

Applicability

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In Modes 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

Actions

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the required ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the required ACTIONS may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated required ACTIONS.

a. The Condition applies if it is discovered that one or more SG tubes examined in an Inservice Inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, LCO 3.4.5 Action b. applies.

   A completion time of seven days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

   If the evaluation determines that the affected tube(s) have tube integrity, the ACTION statement allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

STEAM GENERATOR TUBE INTEGRITY (Continued)

ACTIONS (Continued)

b. If the required actions and associated completion times of LCO 3.4.5 Action a. are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours.

The allowed completion times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements (SR)

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

A condition monitoring assessment of the SG tubes is performed during SG inspections. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the method used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 6.8.4.k until subsequent inspections support extending the inspection interval.

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 plugging criteria is removed from service by plugging. The tube plugging 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 plugging 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 plugging 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"

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

ACTIONS (Continued)

3.7.6.a.3

If both CREFS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., LCO ACTION 3.7.6.a.2), the CREFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

3.7.6.b.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREFS train cannot be restored to OPERABLE status within the required AOT, action must be taken to immediately place the OPERABLE CREFS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action 3.7.6.b.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

3.7.6.b.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREFS trains inoperable or with one or more CREFS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

3.7.6.b.3

The provisions of Specification 3.0.4 are not applicable for ACTIONS 3.7.6.b. (1 and 2). Entry into MODE 5 or 6 is permitted when the LCO is not met and while relying on ACTION 3.7.6.b. (1 or 2). This is acceptable because these ACTIONS provide an adequate level of safety to permit continued operation in the MODES of Applicability for an unlimited time.

SURVEILLANCE REQUIREMENTS

SR 4.7.6.a

The control room temperature should be checked periodically to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by the CREFS and that the control room will remain habitable for operations personnel during and following all credible accident conditions.