SCE&G -- Explanation of Technical Specification Changes for Uprate Power Operation

PAGE	Affected Section	Bar #	Description of Change	Reason for Change
1-5	1.25	1	Rated Thermal Power definition is revised to incorporate the increased power level.	This change is necessary to support the uprate power condition.
3/4 11-5	3.11.2.6	1	Revise maximum quantity of radioactivity in each gas storage tank - 160,000 curies to 131,000 curies Noble gas.	Review of calculation for offsite doses due to a gas tank rupture.
B 3/4 2-1	3/4.2	1	Discussion of the 2200°F ECCS limit is revised to reference the acceptance criteria provided in 10CFR50.46.	This change is necessary to support the Best Estimate LOCA analysis.
B 3/4 2-3	3/4.2.2 and 3/4.2.3	1	Discussion of the 2200°F ECCS limit is revised to reference the acceptance criteria provided in 10CFR50.46.	This change is necessary to support the Best Estimate LOCA analysis.
6-16a	6.9.1.11.c	1	Methodology referenced by the COLR that is used to determine the heat flux hot channel factor is changed to reference Best Estimate LOCA analyses.	This change is necessary to support the Best Estimate LOCA analysis.
OL page 4	2.C.1	1	Revising Maximum Power Level to 2900 MWt Core Power	This change is necessary to support the Uprate Power Condition.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2775 MWt.~~

2900 Mwt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of hghest reactivity worth which is assumed to be fully withdrawn.

SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

1.30 Not Used

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

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RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ~~160,000~~ 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core at or above the design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that ~~the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.~~

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The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The limits on AFD will be provided in the COLR per Technical Specification 6.9.1.11.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA ~~the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.~~

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Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on the RCS Total Flow Rate Versus R figure in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if core power is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

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The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA there is a high level of probability that the acceptance criteria of 10CFR50.46 would not be exceeded.

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CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (FQ Methodology for W(Z) surveillance requirements).)

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c. ~~WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987 (W Proprietary).~~

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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c. WCAP-12945-P, "CODE QUALIFICATION DOCUMENT FOR BEST ESTIMATE LOCA ANALYSES", (W Proprietary).

CORE OPERATING LIMITS REPORT (Continued)

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SAFETY EVALUATION
FOR REVISING THE SPECIFICATION FOR UPRATE
VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

South Carolina Electric & Gas Company (SCE&G) proposes to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) pages 1.5, 3/4.11-5, 6-16a, and Bases pages B3/4.2-1 and B3/4.2-3. These changes support the Uprate project and provide the following:

- a new definition of Rated Thermal Power (RTP) to incorporate the uprate power condition of 2900 MWt. This value represents the total heat transfer rate from the reactor core to the reactor coolant and does not include heat generated by the reactor coolant pumps.
- a revised limit for the quantity of radioactivity stored in any one gas storage tank. This new value is based on the methodology in NUREG 0133 and only affects the maximum quantity stored.
- a new reference to the Core Operating Limits Report (COLR) which is based on the Best Estimate Loss of Cooling Accident methodology, and a deletion of the reference to the BASH/BART methodology in the COLR specification.
- revised bases information to indicate that VCSNS meets the generic acceptance criteria of 10 CFR 50.46, rather than only using the ECCS acceptance criteria of 2200°F. This change is appropriate since the peak cladding temperature may not be the most limiting criteria associated with the ECCS evaluations.

Many TS changes were required to support the Steam Generator Replacement (SGR), which were approved and issued via reference 1. Many of the TS changes expected for a plant Uprate were included in the SGR submittal. Most evaluations performed for SGR utilized 2900 MWt core power as an initial condition.

This Technical Specification Change Request (TSCR) primarily revises those areas in TS which were not included in Reference 3. The primary supporting analyses performed for uprate are: Large Break Loss of Cooling Accident (LOCA) utilizing the Westinghouse Best Estimate LOCA methodology, spent fuel pool cooling capacity analysis resulting from our outage practices, and Waste Gas Decay Tank Rupture analysis resulting from a comment included in the SER for SGR (Reference 1.). Other analyses and evaluations were performed to assess the capability of other systems and components to support Uprate, with the results indicating that both the Nuclear Steam Supply System (NSSS) and the Balance of Plant systems are capable of supporting uprate power operation assuming modifications to several balance of plant systems.

Safety Evaluation

The conditions that result from uprate power are increased heat transferred from the Reactor core, increased steam flow, increased feedwater flow, and increased electrical output. The additional heat load of approximately 4.5 percent can be met with the existing capacities of all NSSS and interfacing systems.

Modifications such as Closed Cycle Cooling are being planned to improve the capability of secondary systems to meet the additional load.

The increase in the secondary mass flow rates has been evaluated and does not present any concerns. The Δ75 steam generators are rated for this condition and comply with all ASME Code requirements. The condenser, piping, and valves have all been evaluated and have adequate margin to support uprate conditions. The same is true for Feedwater and Emergency Feedwater Systems. In addition to the code requirements, chrome-moly steel has been used in feedwater piping replaced during RF-8 to reduce the effects of erosion/corrosion.

The additional heat produced will generate additional electricity. The turbine-generator has been evaluated and is capable, with a modification to the Stator Water Cooling System to adequately meet the demands of uprate.

With a RATED CORE POWER level of 2900 MWt, the calculated results (i.e., DNBR, Pressure, Peak Clad Temperature, Metal Water Reaction, Environmental Conditions Inside and Outside Containment, etc.) are acceptable and remain within applicable regulatory acceptance criteria. The results further show that the integrity of the primary/secondary/containment pressure boundary is not challenged and that the extent of fuel failures during Condition III and IV events remains bounded by assumptions within the dose analyses. The calculated radiological consequences remain well within applicable regulatory limits.

Offsite Dose Limits will be maintained with the revision to the gas storage specification. Although this is not specifically an uprate concern, it affects the radiological consequences section in the SGR submittal (Ref. 3). The TS 3.11.2.6 limit will decrease from 160,000 curies Noble Gas to 131,000 curies Noble Gas. However, the station administrative limit of 90,000 curies Noble Gas is unchanged and has never been exceeded. These gas tanks are sampled daily when adding to the tank to assure this limit is not exceeded.

The uprate conditions will produce additional heat loads on the Spent Fuel Cooling System due to increased decay heat. Analyses indicate that the system has sufficient capacity to limit the pool temperature to less than 150°F during limiting Normal heat loads and to less than bulk boiling during limiting Abnormal heat loads. In the event of a loss of spent fuel cooling, adequate time remains available to restore spent fuel cooling to preclude the onset of boiling. For the postulated condition of an extended loss of normal cooling, various makeup water sources are available on site with sufficient capacity to match the pool boiloff rate, thus precluding fuel uncovery.

Uprate power will not adversely affect the operation of the Reactor Protection System, Engineering Safety Features, or other systems or components that are required for accident mitigation. The revised operating conditions will not affect these systems' performance or qualification for either normal operation or accident conditions. All calculated results to VCSNS FSAR Chapter 15 Analyses demonstrate that there are no challenges to the integrity of the fission product boundaries and the plant remains within the regulatory acceptance criteria applied to the VCSNS current licensing basis.

SIGNIFICANT HAZARDS EVALUATION FOR REVISING THE SPECIFICATION FOR UPRATE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

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- a new reference to the Core Operating Limits Report (COLR) which is based on the Best Estimate Loss of Cooling Accident methodology, and a deletion of the reference to the BASH/BART methodology in the COLR specification.

- revised bases information to indicate that VCSNS meets the generic acceptance criteria of 10 CFR 50.46, rather than only using the ECCS acceptance criteria of 2200°F. This change is appropriate since the peak cladding temperature may not be the most limiting criteria associated with the ECCS evaluations.

Many TS changes were required to support the Steam Generator Replacement (SGR), which were approved and issued via Reference 1. Many of the TS changes expected for a plant Uprate were included in the SGR submittal. Most evaluations performed for the SGR utilized 2900 MWt core power as an initial condition.

This Technical Specification Change Request (TSCR) primarily revises those areas in TS which were not included in Reference 3. The primary supporting analyses performed for uprate are: Large Break Loss of Cooling Accident (LOCA) utilizing the Westinghouse Best Estimate LOCA methodology, spent fuel pool cooling capacity analysis resulting from our outage practices, and Waste Gas Decay Tank Rupture analysis resulting from a comment included in the SER for SGR (Reference 1.). Other analyses and evaluations were performed to assess the capability of other systems and components to support Uprate, with the results indicating that both the Nuclear Steam Supply System (NSSS) and the Balance of Plant systems are capable of supporting uprate power operations, assuming several modifications to balance of plant systems.

Basis for No Significant Hazards Consideration Determination

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard for the following reasons:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

   Implementation of uprate power operation does not contribute to any accident evaluated in the FSAR. The NSSS Components (RV, RCPs, CRDMs, SGs, and piping) are compatible with the revised operating conditions. These components have been reanalyzed and the results show that ASME Code requirements remain satisfied and are within the current Licensing Basis.

   Interfacing Systems which are important to safety are not adversely impacted and will continue to perform their design function. Overall secondary plant performance is not significantly altered by the proposed changes.

   Therefore, since the Reactor Coolant pressure boundary integrity and system functions are not adversely impacted, the probability of occurrence of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

   An extensive analysis has been performed to evaluate the consequences of the following accident types currently evaluated in the VCSNS FSAR:

   - Non-LOCA Events
   - Large Break and Small Break LOCA
   - Steam Generator Tube Rupture

   With the Δ75 SGs and revised operating conditions, the calculated results (i.e., DNBR, Primary and Secondary System Pressure, Peak Clad Temperature, Metal Water Reaction, Challenge to Long Term Cooling, Environmental Conditions Inside and Outside containment, etc.) for the accidents are similar to those currently reported in the VCSNS FSAR and remain within applicable Regulatory Acceptance Criteria. Select results (i.e., Containment Pressure during a Steam Line Break, Minimum DNBR for Rod Withdrawal from Subcritical, etc.) are slightly more limiting than those currently reported in the FSAR due to the use of the assumed operating conditions with the Δ75 SGs and in some cases, use of an uprated core power of 2900 MWt. However, in all cases, the calculated results do not challenge the integrity of the primary/secondary/ containment pressure boundary and remain within the regulatory acceptance criteria applied to VCSNS's current licensing basis.

Given that calculated radiological consequences are not significantly higher than current FSAR results and remain well within 10CFR100 limits, it is concluded that the consequences of an accident previously evaluated in the FSAR are not significantly increased.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Uprate power operation will not introduce any new accident initiator mechanisms. Structural integrity of the RCS is maintained during all plant conditions through compliance with the ASME code. Design requirements of auxiliary systems are met with the RSGs and uprate power operation. No new failure modes or limiting single failures have been identified. Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

   Although uprate power operation will require changes to the VCSNS Technical Specifications, the proposed changes are supported by extensive LOCA, NON-LOCA and SGTR analyses. These analyses show acceptable consequences with margin to the applicable regulatory limits. All equipment required to function during accident conditions has been shown to remain qualified and thus will perform their design function, and all components remain in compliance with the codes and standards in effect when VCSNS was originally licensed (with the exception of the replacement steam generators which use the 1986 ASME Code Section III Edition). Based on the above, it is concluded that there is no significant reduction in a margin of safety.