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SCE&G®
A SCANA COMPANY

June 30, 2005
RC-05-0097

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

ATTN: Mr. R. E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 04-3385
CORE OPERATING LIMITS REPORT - REFERENCE FOR BEST ESTIMATE
LOSS OF COOLANT ACCIDENT (BELOCA)

Pursuant to 10 CFR 50.90, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS).

The proposed change will revise Administrative Control Section 6.9.1.11, Core Operating Limits Report. The revision will permit the Westinghouse best estimate methodology to be utilized in assuring our compliance to 10 CFR 50.46.

Information contained herein provides the No Significant Hazards Determination. Attachment I provides the TS page marked up with the proposed change. Attachment II provides the retyped TS page.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed change. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91(b).

SCE&G requests approval of the proposed amendment within one year of submittal in accordance with the NRC goal for review of license amendment requests. Once approved, the amendment shall be implemented within 30 days.

There are no other TS changes in process that will affect or be affected by this change request. FSAR Chapter 15 changes will be implemented as necessary after this change is approved. There are no changes to any FPER sections.

If you have any questions or require additional information, please contact Mr. Ronald B. Clary at (803)-345-4757.

A001

I certify under penalty of perjury that the foregoing is true and correct.

6-30-05
Executed on

Jeffrey B. Archie

PAR/JBA/dr

Enclosures:
Evaluation of the proposed change
Attachment(s): 3
1. Proposed Technical Specification Change - Mark-up
2. Proposed Technical Specification Change - Retyped
3. List of Regulatory Commitments

c: N. O. Lorick
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N. S. Carns
T. G. Eppink (w/o Attachments)
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RTS (LAR 04-3385)
File (813.20)
DMS (RC-05-0097)

Subject: LICENSE AMENDMENT REQUEST - LAR 04-3385 ADMINISTRATIVE CONTROL SECTION 6.9.1.11

1.0 DESCRIPTION

South Carolina Electric & Gas Company (SCE&G) requests an amendment to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Administrative Controls. The proposed change to Section 6.9.1.11, Core Operating Limits Report, will permit utilization of the best estimate large break Loss of Coolant Accident (LOCA) methodology in assuring that the requirements of 10 CFR 50.46 remains satisfied.

2.0 PROPOSED CHANGE

Technical Specification 6.9.1.11 lists applicable references for the analytical methods used to determine core operating limits. This list of references includes the Westinghouse topical report (WCAP-10266-P-A, Rev. 2) that documents the currently approved large break LOCA analysis methodology. It is proposed that this reference would be replaced with the generically approved topical report for the Westinghouse best estimate large break LOCA analysis methodology (WCAP-12945-P-A) and reanalysis work plan (NSD-NRC-96-4746).

3.0 BACKGROUND

Westinghouse has obtained generic NRC approval of its topical report describing best-estimate large break LOCA methodology. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998). A plant specific analysis for V. C. Summer Nuclear Station has been performed using the approved methodology.

V. C. Summer Nuclear Station was one of the plants for which portions of the analysis were performed prior to the final methodology revisions that were recommended by both the NRC and the ACRS. A work plan was submitted describing the reanalysis work that would be performed based on the NRC and ACRS comments for the final best estimate methodology approval. This work plan was performed for portions of the V. C. Summer Best Estimate LOCA Analysis. Specific details of the reanalysis work plan and the implications for V. C. Summer Nuclear Station is included in Table 1.

These changes are being made to incorporate the best-estimate methodology approach as part of the licensing basis for the V. C. Summer Nuclear Station large break LOCA analysis in accordance with 10 CFR 50.46, Regulatory Guide 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance," and the Westinghouse

"Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volumes 1-5. The values of major plant parameters assumed in the best-estimate LOCA analyses will be documented in the FSAR for V. C. Summer Nuclear Station. These and other FSAR changes resulting from approval of this LAR will be made in accordance with 10 CFR 50.71(e).

4.0 TECHNICAL ANALYSIS

A best estimate large break LOCA analysis has been performed for V. C. Summer Nuclear Station using the methodology contained in WCAP-12945-P-A. All plant specific parameters used in the analysis are bounded by the models and correlations contained in the generic methodology. Therefore, the V. C. Summer Nuclear Station analysis conforms to 10 CFR 50.46 and Section II of Appendix K, and meets the intent of Regulatory Guide 1.157. The conclusions of the analysis are that there is a high level of probability that:

1. The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (maximum cladding oxidation) will not exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Table 1 presents the 95th percentile peak clad temperature (PCT), maximum cladding oxidation, maximum hydrogen generation, and cooling results for V. C. Summer Nuclear Station.

Based on these results, V. C. Summer Nuclear Station has concluded that adopting the best estimate large break LOCA methodology for V. C. Summer Nuclear Station and making the proposed TS changes would not adversely affect the health and safety of the public.

Table 1
V. C. Summer Nuclear Station
Best Estimate Large Break LOCA Results

Description	Value	Acceptance Criteria
95th Percentile PCT (°F)*	1988	2200
Maximum Cladding Oxidation* (%)	5.0	17.0
Maximum Hydrogen Generation* (%)	0.72	1.0
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

* Calculated using the methodology in the following references:

- WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

- Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated June 13, 1996.

The Re-analysis Work Plan provisions are used for the following portions of the analysis only:

The following subset of Initial Conditions Studies used the Mod 7A predecessor code version/methodology:
- Pressurizer Pressure
- Accumulator Pressure
- Accumulator Water Volume
- SI Temperature

(All) Power Distribution Studies used the MOD7A predecessor code version/methodology.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10CFR50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

Implementation of the best-estimate large break LOCA methodology and associated TS changes is proposed to increase margin to the peak clad temperature limits defined in 10 CFR 50.46. There are no physical plant changes or changes in manner in which the plant will be operated as a result of this change. Since the plant conditions and ECCS performance assumed in the analysis are consistent with the plant's current design, the proposed change in methodology will thus have no impact on the probability of a LOCA. When applied, the best estimate methodology shows that the ECCS is more effective than previously evaluated in mitigating the consequences of a LOCA, as lower peak clad temperatures are predicted relative to current 10 CFR 50.46 Appendix K results. Since the proposed best-estimate methodology is only applicable to a large break LOCA and since the application of the proposed methodology shows there is a high probability that all of the acceptance criteria contained in 10 CFR 50.46, Paragraph b are met, the proposed change does not increase the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis remain within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Does this change involve a significant reduction in margin of safety?

Response: No.

It has been shown that the methodology used in the analysis would more realistically describe the expected behavior of V. C. Summer Nuclear Station systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties are analyzed to provide assurance that the most severe postulated loss of coolant accidents are calculated. It has been shown by analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

Pursuant to 10 CFR 50.91, the preceding analyses provide a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, requires that the ECCS system be designed to satisfy the 5 criteria listed in the regulation. These criteria are:

1. ***Peak cladding temperature***
2. ***Maximum cladding oxidation***
3. ***Maximum hydrogen generation***
4. ***Coolable geometry***
5. ***Long-term cooling.***

V. C. Summer Nuclear Station has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any general design criteria as contained in 10 CFR 50, Appendix A, differently than described in the safety analysis report. The proposed change is consistent with the requirements identified in 10 CFR 50.46 below. The conclusions of the analysis are that there is a high level of probability that:

1. The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (maximum cladding oxidation) will not exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in

the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

10 CFR 50, Appendix A, General Design Criteria, Criterion 35, Emergency Core Cooling states that the system safety function shall be to transfer heat from the reactor core following a loss of reactor coolant such that fuel and clad damage that could interfere with effective cooling is prevented and any metal water interaction is kept to negligible amounts.

The proposed change does not impact the ability of the system to perform its design function, as there are no plant hardware changes. The Emergency Core Cooling system will continue to perform to the same high reliability and availability standards.

10 CFR 50, Appendix K, ECCS Evaluation Models, lists required and acceptable features that have to be included in all evaluation models.

The BELOCA methodology has been generically approved by the NRC. NRC approval of the methodology is documented in the safety evaluation appended to the topical report (WCAP 12945-P-A, "Code Qualification Document for Best Estimate LOCA Analysis", March 1998). A plant specific analysis for the V. C. Summer Nuclear Station has been performed using the approved methodology. The BELOCA methodology relaxes some of the requirements in Appendix K without a loss in safety to the public. Regulatory Guide 1.157 was issued to provide guidance on use of the best estimate codes.

Generic Letter 97-04: Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps.

The BELOCA methodology does not alter the amount of inventory dumped into the recirculation sumps. The water level and therefore the net positive suction head are unchanged under this methodology. The values of major plant parameters assumed in the best-estimate LOCA analysis will be documented in the VCSNS Final Safety Analysis Report (FSAR).

Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment.

The response provided to Generic Letter 98-04 is not altered by the change in Large Break LOCA methodology. The quantities and types of debris are unchanged. The system operation is unaffected and the 10 CFR 50.46 acceptance criteria remains satisfied. The sump blockage issue is separate and is being resolved under Generic Safety Issue (GSI) 191.

Bulletin 86-03: Potential Failure of Multiple ECCS Pumps Due to a Single Failure of an Air Operated Valve in Mini Flow Recirculation Line.

Single failures and failure modes analysis were performed for the Emergency Core Cooling System (ECCS) during initial plant design. The change in Large Break (LB) LOCA analysis methods from BASH to Best Estimate has no impact on the probabilities of this or any other type of failure that could prevent the system from performing its design function.

Bulletin 93-02: Debris Plugging of Emergency Core Cooling Suction Strainers.

The change in analysis methodology will not have any impact on the quantity or types of debris generated or transported to the recirculation sumps. The sump blockage issue is separate and is being resolved under GSI-191.

Bulletin 2003-01: Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors.

GSI-191 is being used by the industry and the NRC to resolve concerns related to the potential for clogging the recirculation screens/strainers on the ECCS system. The change to the analysis methodology to a best estimate model will not have any impact on the current capability of the ECCS system. Any changes that may be made to the sump design will only affect resolution to the generic safety issue. The assumptions on available inventory in the sumps are the same for both analysis models. The best estimate methodology has the same initial conditions and assumptions as the current licensed model for VCSNS and as such the response to the Bulletin will not have any impact on the analysis model.

5.3 Design Bases (FSAR)

FSAR Section 15.4.1 Major Reactor Coolant System Pipe Ruptures.

The VCSNS FSAR will require revision as a result of this LAR.

5.4 Approved Methodologies

The proposed change is to adopt the best estimate methodology for the large break loss of coolant accident analysis. This methodology is contained in WCAP-12945-P-A and has been approved by the staff for use at other Westinghouse plants. The methodology conforms to 10 CFR 50.46 and Section II of Appendix K, and meets the intent of Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance".

The analysis performed for VCSNS utilizes parameters that are bounded by the models and correlations contained in the generic methodology.

5.5 Analysis

The parameters used in the VCSNS specific analysis performed with the BELOCA methodology are presented in Table 2. The results of the analysis are presented in Table 1 and demonstrate that the 10 CFR 50.46 criteria are satisfied.

Table 2
V. C. SUMMER NUCLEAR STATION
BEST ESTIMATE LARGE BREAK LOCA
MAJOR PLANT PARAMETER ASSUMPTIONS

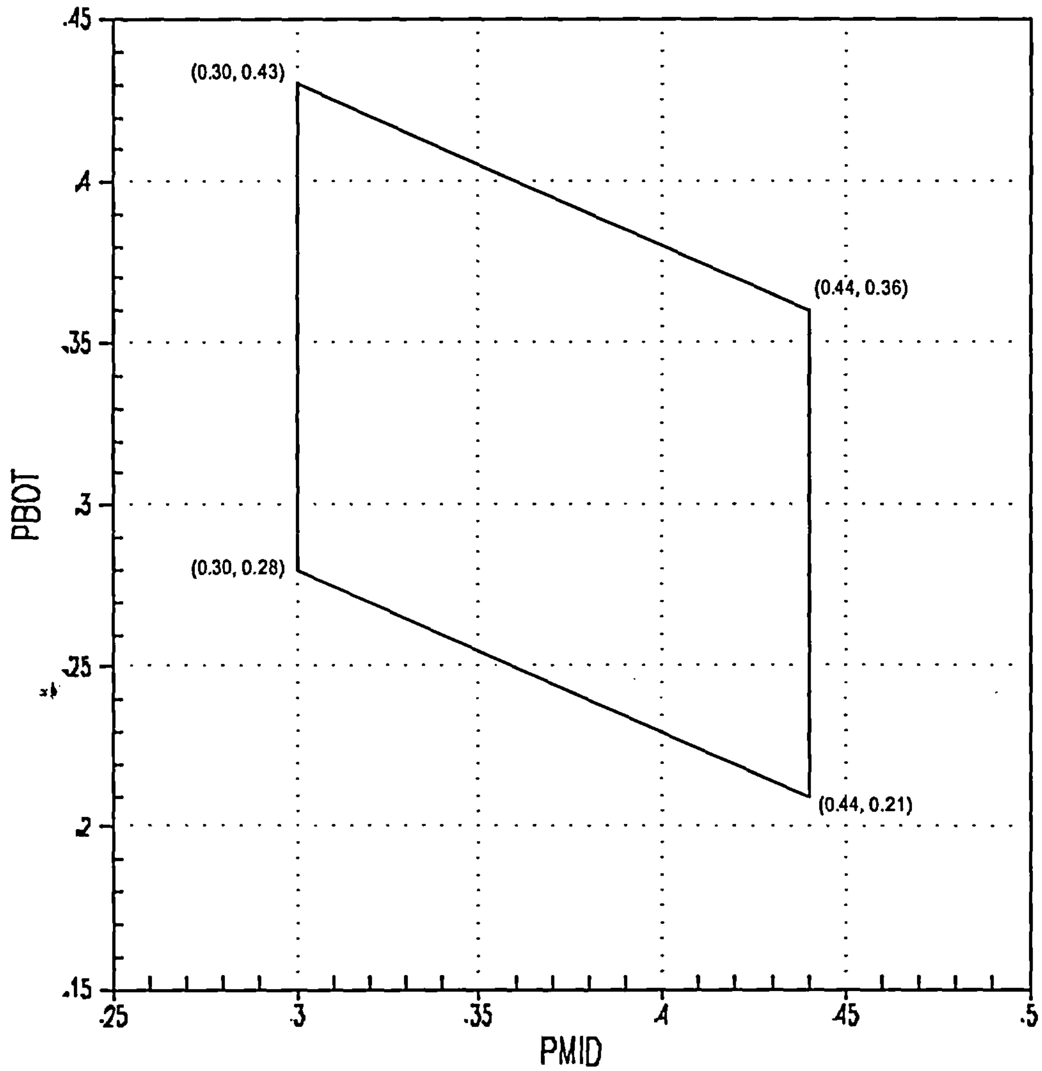
Parameter	Value
Plant Physical Description	
Steam Generator Tube Plugging	$\leq$ 10%
Plant Initial Operating Conditions	
Reactor Power	$\leq$ 102% of 2900 MWt w/2% Calorimetric uncertainty
Peaking Factors	F_Q=2.5, $F\Delta H$=1.7
Axial Power Distribution	Figure 1
Fluids Conditions	
RCS T_{avg}	$572.0 \pm 5.3 \leq T_{avg} \leq 587.4 \pm 5.3$ °F
Pressurizer Pressure	2250 ± 100 psia
Reactor Coolant Flow	$\geq$ 92,600 gpm/loop
Accumulator Temperature	$85 \leq T_{acc} \leq 115$ °F
Accumulator Pressure	$570 \leq P_{acc} \leq 686$ psig
Accumulator Water Volume	$994 \leq V_{acc} \leq 1034$ ft^3
Accident Boundary Conditions	
Single Failure Assumptions	1 Train of SI Pumps (WCOBRA/TRAC Studies)
	No Failures (Containment Pressure Study)
Safety Injection Flow	Table 3
Safety Injection Temperature	$55 \leq T_{SI} \leq 95$ °F
Safety Injection Initiation Delay Time	$\leq$ 22 sec No-Loss Of Offsite Power (LOOP)
	$\leq$ 32 sec LOOP
Containment Pressure	Figure 2

Table 3

V. C. SUMMER NUCLEAR STATION
BEST ESTIMATE LARGE BREAK LOCA
TOTAL MINIMUM INJECTED SI FLOW
TOTAL CHG/SI AND LHSI INTO 2 INTACT LOOPS

RCS Pressure (psig)	Flow Rate (gpm)
0	2709
15	2561
20	2467
30	2270
40	2064
50	1848
60	1612
70	1356
80	1071
90	699
100	305
200	295
300	285
400	274

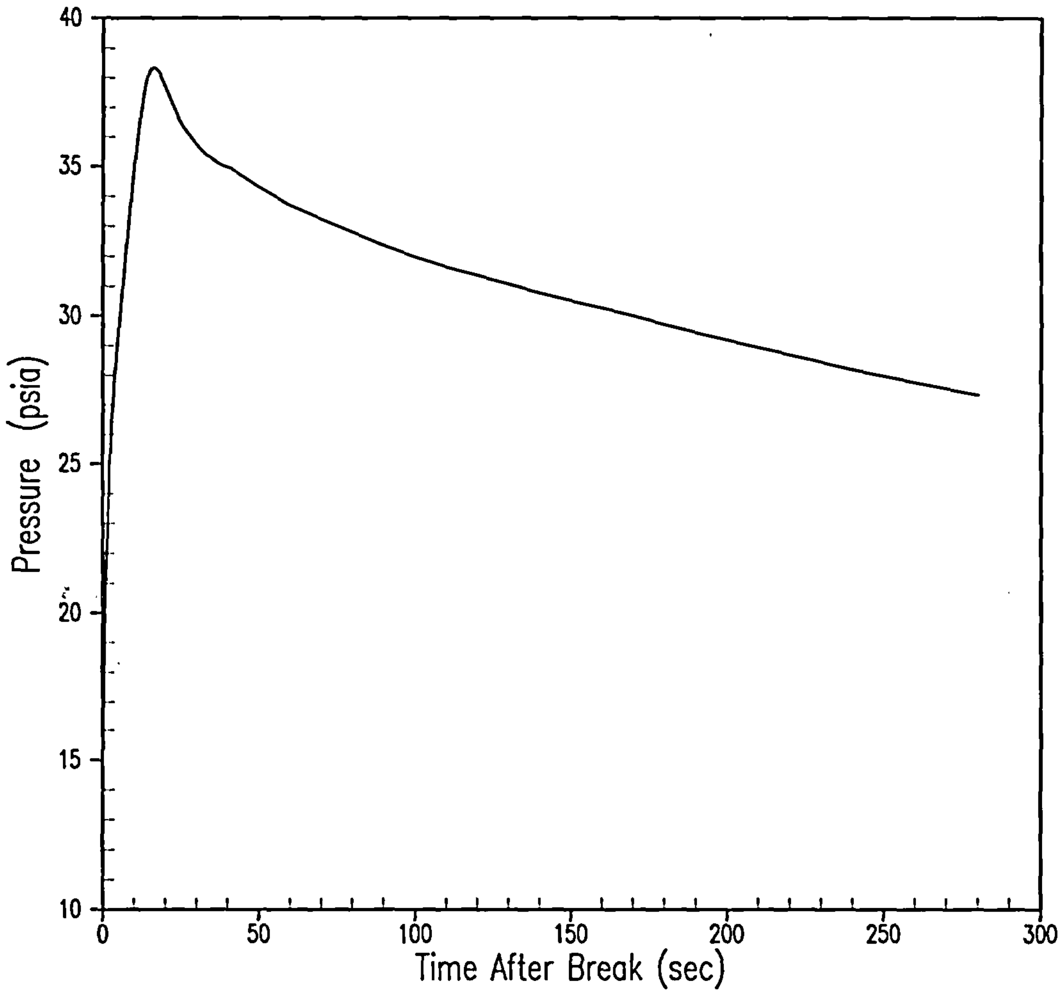
Figure 1
Virgil Summer Best Estimate Analysis Axial Power Distribution Operating Space



Legend:

PBOT- Fraction of Power in lower third of the core
PMID - Fraction of Power in mid region of the core

Figure 2
Virgil Summer Best Estimate Analysis Containment Backpressure



5.6 <u>Conclusion</u>

The proposed change to the VCSNS TS is to adopt the BELOCA methodology.

6.0 ENVIRONMENTAL CONSIDERATION

SCE&G has determined that the proposed amendment would not change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. The proposed amendment is a methodology change for TS 3.2.3, Heat Flux Hot Channel Factor. The methodology to be used has been generically approved for use by the NRC (WCAP 12945-P-A).

SCE&G has evaluated the proposed change and has determined that the change does not involve, (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed changes do not involve a significant hazards consideration. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51, specifically 10 CFR 51.22(c)(9). Therefore, pursuant 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGE (MARK-UP)

Attachment to License Amendment No. 04-3385
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Pages	Insert Pages
6-16a	6-16a

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
6-16a	6.9.1.11	1	Revising the reference for Large Break LOCA analysis methodology from BASH to Best Estimate.	Implementing the Best Estimate Loss of Coolant Accident analysis methodology.

Insert for Section 6.9.1.11

c. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated 6/13/1996.

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q 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 (F_Q Methodology for W(z) surveillance requirements).)

insert

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," ~~March 1987~~; Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS," May 1988, (W Proprietary).

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

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGE (RETYPED)

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q 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 (F_Q Methodology for W(z) surveillance requirements).)

c. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated 6/13/1996.

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

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

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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.

ATTACHMENT III
LIST OF REGULATORY COMMITMENTS

There are no regulatory commitments created due to this License Amendment Request.