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February 14, 1996
RC-96-0027

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

Attention: Mr. Stephen Dembek

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION CHANGE REQUEST
TSP 950001, SUPPLEMENT 2, PLANT UPRATE

Reference:
a. S. J. Furstenberg to Document Control Desk Letter RC-95-0174 dated August 18, 1995

b. G. J. Taylor to Document Control Desk Letter RC-95-0258, dated November 1, 1995

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby submits a revised Attachment II, sections 3.4.3 and 5.0, and the Safety Evaluation, Attachment IV, to the above referenced Technical Specification Change Request.

A followup review of the spent fuel cooling calculations provided by reference b. identified changes that need to be made to Section 3.4.3 of Attachment II and the Safety Evaluation, Attachment IV. In addition, a typographical error in Section 5.0 of Attachment II has been corrected. Attachments II and IV are being provided in their entirety due to the pagination process. The No Significant Hazards Evaluation is unaffected by this revision.

This supplement to the proposed change to the Virgil C. Summer Nuclear Station Technical Specifications has been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee.

These statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Should you have questions, please call Mr. Philip Rose at (803) 345-4052.

Very truly yours,

Gary J. Taylor

Gary J. Taylor

PAR/GJT/dr
Attachments (2)
c: See page 2

A001

c: J. L. Skolds (w/o Attachments)
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R. R. Mahan (w/o Attachments)
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RTS (TSP 950001)
File (813.20)

STATE OF SOUTH CAROLINA :
: TO WIT :
COUNTY OF FAIRFIELD :

I hereby certify that on the 14th day of FEB 1996, before me, the subscriber, a Notary Public of the State of South Carolina, personally appeared Gary J. Taylor, being duly sworn, and states that he is Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Notary Public

My Commission Expires July 13, 2005
Date

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TABLES

1.0 INTRODUCTION - DESCRIPTION OF LICENSE AMENDMENT REQUEST

This document contains the remaining safety analysis and evaluation results to support uprate power operation of the Virgil C. Summer Nuclear Station (VCSNS). As noted in the Replacement Steam Generator (RSG) submittals (Reference 3), SCE&G decided to analyze the plant for a range of operating conditions to provide operational flexibility. Accident analyses were performed at a core power level up to 2900 MWt (a 4.5% increase). SCE&G received a Safety Evaluation Report (SER) (Reference 1) from the NRC for the analyses and evaluations performed to support the RSG and utilized the Engineered Safeguards design rating ("uprate" power rating) of 2900 MWt core power. At that time, however, SCE&G did not seek approval for uprate power operation since additional areas needed to be addressed at the 2900 MWt core power level. There are three significant areas (not included in the RSG SER) that remain to be addressed to implement the uprate power operation at VCSNS. These areas are Large Break LOCA using the Westinghouse 1981 Evaluation Model with BASH, NSSS Fluid Systems, and Technical Specification (TS) changes. Additionally, the calculation for Waste Gas Decay Storage Tank Rupture was evaluated and determined to require a TS change, as well as reviewing the calculations for Spent Fuel Cooling capability. Uprate power will also affect the Pressure Temperature Limitation Curves due to increased neutron fluence. This results in reduced Effective Full Power Years these curves will be applicable for the Reactor Coolant System.

In order to permit more flexible plant operation, and support a coastdown at the end of the fuel cycle, a range of full power nominal T_{avg} values from a maximum value of 587.4°F to a minimum of 572.0°F was analyzed. Normal plant operation is expected to be 587.4°F (or slightly less). Thermal Design flow will be reduced to 92,600 gpm/loop, to support up to 10% steam generator tube plugging. Minimum measured flow will be 283,500 gpm. Table 2.1-1 further delineates the design performance capability parameters for VCSNS with the Δ75 SGs and uprate power operation. All other accident analysis results were submitted via Reference 3 and approved by the NRC in Reference 1. During the NRC review, the radiological consequences analysis for waste gas decay storage tank rupture was questioned, prompting SCE&G to evaluate the methodology used. As a result, the maximum quantity of radioactivity stored in any one tank is being reduced.

This amendment request reflects the impact of the design, analytical methodology, and safety analysis assumptions on the VCSNS Technical Specifications for the uprate power operation.

The proposed additional changes to the VCSNS Technical Specifications are addressed in Attachment III. These changes reflect the impact of the uprate power operation. When implemented, the proposed changes along with the previously approved Technical Specification changes (Reference 1) will preserve the design, analytical methodology, and safety analysis assumptions outlined in this amendment request.

2.0 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

The analyses and evaluations performed to support the uprate power bound a range of operating conditions for VCSNS. Four cases are presented which define a range of primary operating temperatures from 572°F to 587.4°F and a range of steam generator tube plugging levels from 0% to 10%. This will provide SCE&G with the flexibility to select the appropriate primary temperatures on a cycle-by-cycle basis necessary to achieve full megawatt electric output and to adjust the temperature as necessary to perform end-of-cycle T_{avg} coastdown.

DESIGN POWER CAPABILITY PARAMETERS

Design power capability parameters were developed for the VCSNS to encompass both the Δ75 Replacement Steam Generators (RSGs) and the uprate power level (2900 MWt Core Power). The parameters developed are bounding for the lower power level of 2787 MWt NSSS (2775 MWt Reactor Core) that is the current licensed power for VCSNS. The safety analyses presented in this submittal considered the case(s) which is most conservative for the specific analysis areas. The parameter cases are provided in Table 2.1-1 and are explained in detail below.

Cases 1 and 2, calculated for 0% and 10% Steam Generator Tube Plugging (SGTP), respectively, incorporate the conservatively low Reactor Coolant System (RCS) Thermal Design Flow (TDF) (92,600 gpm/loop), as well as the current licensed T_{avg} value of 587.4°F. The TDF of 92,600 gpm/loop was selected such that adequate margin (approximately 8%) exists between TDF and best estimate predictions of RCS flow, assuming the Δ75 steam generator with 10% SGTP. The RCS Best Estimate Flow (BEF) is based on Δ75 steam generator hydraulic characteristics, reactor coolant pump performance curves, and RCS pressure drop data. Cases 1 and 2 are used for those analyses [e.g., non-LOCA, Departure from Nucleate Boiling (DNB)-related] where high RCS temperatures and low RCS flow are bounding.

Cases 3 and 4 (0% and 10% SGTP) incorporate TDF and the lowest reactor vessel T_{avg} considered, 572°F. The reduced temperature conditions allow for constant operation at reduced temperatures, or end-of-cycle T_{avg} coastdown capability. These cases are used for analyses where low vessel inlet temperature is bounding (NSSS design transients for the cold leg) or where low steam pressure is bounding (e.g., consideration of pressure drop across the steam generator tubes).

Table 2.1-2 provides the Best Estimate Operating Condition Parameters. These are the parameters that will be expected following startup from RF-9. VCSNS intends to operate with T_{avg} = 587.4°F. The steam parameters have been calculated as best estimate for 100% power and will be used to predict actual performance.

TABLE 2.1-1

DESIGN PERFORMANCE CAPABILITY PARAMETERS FOR VCSNS UPRATE POWER OPERATION WITH Δ75 STEAM GENERATORS

For all parameter cases:

Parameter	
NSSS Power, MWt	2912
Non-Nuclear NSSS Power MWt*	12
Core Power, MWt	2900
Core Bypass Flow, %	8.9
RCS Design Pressure, psia	2250
Thermal Design Flow, gpm/loop	92,600
Minimum Measured Flow, gpm total	283,500
Best Estimate Flow, gpm/loop	102,600
Mechanical Design Flow, gpm/loop	107,100
Fuel Design	VANTAGE+ (V+)
Positive Moderator Temp. Coef., pcm/°F	+7

* Includes heat input from RCP and other non nuclear sources.

	High T_{avg} Cases:		Low T_{avg} Cases:	
Parameter	Case 1	Case 2	Case 3	Case 4
Coolant Temperatures, °F				
Core Outlet	627.7	627.7	613.5	613.5
Vessel Outlet	621.9	621.9	607.4	607.4
Core Average	592.8	592.8	577.1	577.1
Vessel Average	587.4	587.4	572.0	572.0
Vessel/Core Inlet	552.9	552.9	536.6	536.6
Zero Load	557.0	557.0	557.0	557.0
Steam Generator				
Feedwater Temperature, °F	440.0	440.0	440.0	440.0
Moisture Carryover, %	0.1	0.1	0.1	0.1
Steam Temperature, °F	540.4	538.4	523.7	521.7
Steam Pressure, psia	966	950	839	824
Steam Flow, million lb/hr.	12.84	12.83	12.77	12.7
Tube Plugging, %	0	10	0	10

TABLE 2.1-2

BEST ESTIMATE OPERATING CONDITION PARAMETERS
FOR VCSNS
UPRATE POWER OPERATION WITH Δ75 STEAM GENERATORS

Parameter	
Reactor Vessel Outlet Temperature (T_{HOT}), °F	618.8
Reactor Vessel Inlet Temperature (T_{COLD}), °F	556.1
Average Temperature (T_{avg}), °F	587.4
Steam Temperature, °F	544.6
Steam Pressure, psia	1000
Steam Flow, Million lb/hr	12.86

3.0 SAFETY EVALUATIONS/ANALYSES

3.1 LARGE BREAK LOCA

Introduction:

SCE&G replaced their Model D3 steam generators at the VCSNS with Delta 75 steam generators. The Δ75 SGs have been addressed previously for most of the LOCA-related accident analyses in Reference 3. Except for the Large Break LOCA analysis performed with the Westinghouse 1981 Evaluation Model with BASH, the remaining analyses (Small Break LOCA, loop/vessel hydraulic forcing functions, post-LOCA long term subcriticality/minimum flow, and hot leg switchover) addressed a core power of 2900 MWt. As such, the purpose of this licensing submittal is to address the large break LOCA analysis performed at uprate power with the Westinghouse 1981 Evaluation Model (EM) with BASH (Reference 9).

Analysis Input:

To address the VCSNS core power of 2900 MWt, the Large Break LOCA analysis was performed with the Westinghouse 1981 Evaluation Model with BASH.

Only the limiting break size at the reduced vessel average temperature of 572°F was analyzed for the LB LOCA. Previous licensing basis analyses for VCSNS have consistently shown that the DECL guillotine break with $C_D=0.4$ is much more limiting than the $C_D=0.6$ and $C_D=0.8$ DECL guillotine breaks. In addition, previous LB LOCA licensing basis analyses for VCSNS have also demonstrated that reduced vessel average temperature produces the most limiting results. The blowdown phase of the LB LOCA transient for the reduced vessel average temperature analysis is slightly longer than the nominal vessel average temperature case. This results in an increased bypass deficit for the reduced vessel average temperature analysis. Also, the accumulator mass which is lost out of the break at the end-of-blowdown is greater for the reduced vessel average temperature case and, consequently, less accumulator inventory is available to refill the downcomer. This reduction in head to reflood the core results in a higher PCT for the reduced vessel average temperature analysis for VCSNS and is typical of the results for other 3 loop plants. Therefore, only the $C_D=0.4$ case with a reduced vessel average temperature was analyzed.

In addition to the uprate power level, this analysis supports the assumptions documented in Table 3.1-1.

Results:

The limiting $C_D=0.4$ LB LOCA analysis with the Westinghouse 1981 Evaluation Model with BASH resulted in a PCT of 2099°F. Table 3.1-2 summarizes the results of the analysis and demonstrates that the 10 CFR 50.46 acceptance criteria are met.

A top skewed power shape is not expected to cause more limiting results than the chopped cosine used in the analysis. This type of power distribution typically impacts the results of plants where the PCT occurs at the higher elevations (e.g., >8.0 ft) late in the reflood phase of the large break transient. For VCSNS, the LB LOCA analysis with a chopped cosine power shape resulted in a PCT which occurs at the 6.25 ft burst node elevation at 61.7 seconds. In addition, the highest cladding temperature which occurs at the higher elevations for VCSNS is much less than the PCT and occurs significantly later in the transient. Therefore, the skewed power shape will not cause more limiting results to occur at the higher node elevations in the VCSNS LB LOCA analysis.

Conclusions:

The VCSNS uprate core power level of 2900 MWt with the Δ75 SGs has been evaluated for the LB LOCA analysis indicated above. It was determined that the acceptance criteria of 10 CFR 50.46 would not be exceeded in the event of a LB LOCA.

TABLE 3.1-1

KEY LARGE BREAK LOCA ASSUMPTIONS FOR VCSNS
STRETCH POWER OPERATION WITH
Δ75 REPLACEMENT STEAM GENERATORS

License Core Power (MWt)	2900[1]
Vessel Average Temperature (°F)	566.7[2]
Vessel Inlet Temperature (°F)	530.43[2]
Vessel Outlet Temperature (°F)	602.97[2]
Pressurizer pressure, maximum (psia)	2300
Thermal Design Flow (gpm/loop)	92600
Steam Generator Tube Plugging Level (%)	10
Peak Linear Power on the Hot Rod (KW/ft)	14.5105
Total Peaking Factor, FQ	2.5
Axial Peaking Factor, Fz	1.47
Hot Channel Enthalpy Rise Factor, FDH	1.70
Hot Assembly Average Power Factor (PHA)	1.514
Power Shape	Chopped Cosine
Fuel Assembly Array	17x17 Vantage +
Accumulator Water Volume, minimum (ft^3/accumulator)	1000[3]
Accumulator Gas Pressure, minimum (psia)	600
Accumulator Temperature, maximum (°F)	110
Reactor Trip Setpoint (psia)	1845
Safety Injection Signal Setpoint (psia)	1715
Safety Injection Delay Time (sec)	27.0
RWST Temperature, min./nom. (°F)	40 / 80

1. A calorimetric uncertainty of 2% is added to this value.

2. Values are based on design power capability parameters but differ slightly due to bounding assumptions used in the LB LOCA analysis.

3. This value does not include the accumulator line volume which was modeled in the analysis.

TABLE 3.1-2

SUMMARY OF RESULTS FOR LIMITING LARGE BREAK LOCA ANALYSIS
USING
WESTINGHOUSE 1981 EVALUATION MODEL WITH BASH

TIME SEQUENCE OF EVENTS	SECONDS
Start	0.0
Reactor Trip Signal	0.371
Safety Injection Signal	0.9
Accumulator Injection Begins	15.8
End-of-Bypass	33.2
End-of-Blowdown	33.2
Pump Injection Begins	27.9
Bottom of Core Recovery	45.7
Accumulator Empty	52.8

RESULTS

Peak Clad Temperature (°F)	2099
Peak Clad Temperature Location (ft)	6.25
Peak Clad Temperature Time (sec)	61.7
Local Zr/H_2O Reaction Maximum (%)	7.9
Local Zr/H_2O Reaction Location (ft)	6.25
Total (avg) Zr/H_2O Reaction (%)	< 1.0
Hot Assembly Burst Time (sec)	45.0
Hot Assembly Burst Location (ft)	6.25
Assembly Channel Blockage (%)	26.6

3.2 RCS HEATUP AND COOLDOWN EVALUATION

The increase in core power will have an associated increase in the neutron fluence which interacts with the reactor vessel. Increased neutron fluence resulting from uprate core conditions has an effect on the reactor vessel Pressure Temperature Curves. Their applicability will change from 14 Effective Full Power Years (EFPY) to 13 EFPY with no other changes at this time. However, these curves will be reviewed after the next surveillance specimen capsule is analyzed.

3.3 FLUID AND AUXILIARY SYSTEMS EVALUATIONS

3.3.1 Introduction

The impact of the RSG/Uprate Power program upon the Nuclear Steam Supply System (NSSS) Fluid Systems, NSSS auxiliary equipment and the NSSS/Balance-of-Plant interface systems was performed for the 4 bounding plant operation cases presented in Table 2.1-1. The NSSS Fluid Systems are comprised of the Residual Heat Removal System (RHR) and Chemical and Volume Control System (CVCS). For the initial licensing submittal regarding the RSG program (Reference 3), the auxiliary equipment evaluation was performed for an NSSS power level of 2912 MWt.

3.3.2 Discussion of Evaluations Performed

3.3.2.1 Fluid Systems Evaluations

Residual Heat Removal System

The RHR System is designed to remove residual and sensible heat from the core during the second phase of plant cooldown. The RHR System was designed to reduce the temperature of the Reactor Coolant System from 350°F to 140°F within 16 hours assuming two RHR heat exchangers and two RHR pumps are in service. The cooldown calculation performed for 2775 MWt demonstrated that the RHR System could achieve this cooldown in 15 hours. The calculation performed for 2900 MWt indicated the RHR System required 21 hours to cool the Reactor Coolant System to 140°F. This increase in cooldown time is an operational issue (ie, outage scheduling) but does not impact any licensing bases for the plant.

The VCSNS Technical Specifications contain action statements which require plant cooldown from Hot Standby to Cold Shutdown within 30 hours following a reactor trip from full power. The cooldown calculation performed for 2900 MWt demonstrated that this cooldown could be achieved with two RHR heat exchangers and two RHR pumps available.

Safety Injection System

The minimum emergency core cooling flow requirement for the Charging/Safety Injection System is dictated by Small Break LOCA (SB LOCA) requirements. The SB LOCA safety analyses for the uprating used the original emergency core cooling flow rates based on a minimum delivered flow of 321 gpm through two of three Changing/Safety Injection branch lines and a maximum pump runout flow rate of 680 gpm. Acceptable SB LOCA analysis results were obtained with these minimum core cooling flow rates. The NRC has reviewed and accepted the SB LOCA analysis, as listed in the NRC SER (Reference 1). As part of the uprating program, the Technical Specification limits were changed to require a minimum flow rate of 338 gpm through two of three branch lines and a maximum pump runout flow rate of 688 gpm. This provides margin with respect to the analysis limits.

3.3.2.2 NSSS Auxiliary Equipment Evaluation

As stated in Section 3.3.1, the evaluation of the auxiliary equipment supporting the RSG program was performed assuming core thermal power of 2900 MWt and NSSS power of 2912 MWt. These plant conditions were determined not to have an adverse effect upon these systems, and therefore, no further evaluation of the NSSS auxiliary equipment is necessary.

3.4 NSSS/BALANCE-OF-PLANT INTERFACE

The interfacing systems are relied on for heat rejection to the Ultimate Heat Sink (Service Water Pond). These systems are Component Cooling Water System (CCW) and Service Water System (SW). The equipment evaluation was performed for an NSSS power level of 2912 MWt.

3.4.1 Component Cooling Water System

This system's major function is to transfer heat from the Residual Heat Removal System (RHR) to the Service Water System. The heat rejection load from RHR will increase with the plant uprate for cooling requirements at 4 and 20 hours after plant shutdown. A secondary function is to remove decay heat from the Spent Fuel Pool Cooling Systems during both power and refueling operations. The projected heat loads for both concerns have been evaluated and are within the capacity of the existing system. Along with the increases in required heat removal rates, we are changing the Design Basis requirements for the heat exchanger to reflect conservative regulatory requirements for cooldown time frames rather than the component suppliers' contractual requirements. This may result in changes to minimum required flow rates for the CCW and/or SW systems. Changes as necessary will be reflected in design modification packages.

3.4.2 Service Water System

This system's major function is to transfer heat from the various heat exchangers connected to the Ultimate Heat Sink (Service Water Pond). For plant uprate, the heat load of primary concern is the CCW Heat Exchanger, with the increase in heat removal requirements which cascade from the RHR system. Other heat loads on the SW System do not significantly change from current design requirements. Monitoring of the performance of this heat exchanger will ensure that design basis capabilities are maintained. Calculations indicate that sufficient margin remains for the SW System.

3.4.3 Spent Fuel Pool Cooling

Due to the proposed increase in RTP (2775 to 2900 MWt), decay heat levels will increase, which will impact the Spent Fuel Cooling System (SFCS). This section outlines the efforts made and provides results of evaluations that were performed to address this increased heat load.

The current Spent Fuel Pool (SFP) heat-up analyses for VCSNS, that was approved under the SER issued with the licensing Amendment 27, dated September 27, 1984, was performed using a transient fuel off-load model. The model assumed that fuel movement began at 144 hours after shutdown and that the fuel was off-loaded at certain rates. The fuel shuffle case resulted in a peak heat load of 16.605 MBTU/HR and a peak SFP temperature of 140°F with the credit of one SFP heat exchanger. The full core off-load case resulted in a peak heat load of 31.647 MBTU/HR and a peak SFP temperature of 139°F with the credit of two SFP heat exchangers.

In order to bound the expected range of temperatures and heat loads on the VCSNS SFCS, the following cases are presented for up-rated (2900 MWt) conditions:

NOTE: The following cases assume that the reactor has been subcritical for at least 100 hours prior to movement of irradiated fuel.

SFP Results for VCSNS (2900 MWt)

Case #	Case Description	PARAMETERS		1 HX	2 HX
1	• SFP contains 1119 burnt fuel assemblies prior to assumed off-load (full pool minus one core) • 72 assemblies (i.e., a core shuffle) are instantaneously placed in the pool with 150 hours of decay time	PEAK HEAT LOAD (MBTU/HR)	21.23	/////////	/////////
		PEAK BULK TEMP (°F)	///////////	150.2	127.8
		TIME TO BOILING UPON LOSS OF HX(S) (HOURS)	///////////	7.1	9.5
		BOIL-OFF RATE (GPM)	45	/////////	/////////
2	• SFP contains 1119 burnt fuel assemblies prior to assumed off-load • 157 assemblies (i.e, a full core off-load) are instantaneously placed in the pool with 150 hours of decay time	PEAK HEAT LOAD (MBTU/HR)	38.211	/////////	/////////
		PEAK BULK TEMP (°F)	///////////	186.1	145.7
		TIME TO BOILING UPON LOSS OF HX(S) (HOURS)	///////////	1.6	4
		BOIL-OFF RATE (GPM)	81	/////////	/////////
3	• SFP contains 1119 burnt fuel assemblies prior to assumed off-load • 157 assemblies (i.e, a full core off-load) are instantaneously placed in the pool with 150 hours of decay time. This full core off-load is assumed to occur 36 days after the prior shut-down during which 72 assemblies were placed in the pool.	PEAK HEAT LOAD (MBTU/HR)	44.76	/////////	/////////
		PEAK BULK TEMP (°F)	///////////	199.9	152.6
		TIME TO BOILING UPON LOSS OF HX(S) (HOURS)	///////////	0.65	3
		BOIL-OFF RATE (GPM)	95	/////////	/////////

A full core off-load with two SFCS loops in operation is a typical practice (Case 2 with 2 heat exchangers) for VCSNS. The acceptance criteria in the Standard Review Plan (SRP), NUREG 0800, for full core offload required no bulk boiling of the pool; single failure need not be considered. The SRP, Section 9.1.3, limit of 140°F is exceeded in all cases where the failure of one cooling loop is considered; however, the peak bulk temperatures presented above are acceptable based upon the evaluation of the:

1. Structural integrity of the SFP and the SFP liner
2. SFCS Pipe Stresses
3. SFCS Components
4. SFP Ventilation System
5. Margin to localized boiling
6. Adequacy of Net Positive Suction Head Available ($NPSH_A$) for the SFCS Pumps

Also, based on the indicated time to boiling, sufficient time exists to restore the SFCS or the various make-up water sources available on-site to reasonably conclude that fuel would not be uncovered if boiling were to occur.

Finally, it should be noted that the SER issued with the licensing Amendment 116, dated August 23, 1994, provided approval of the recent SFP Rack Criticality Analysis. The racks have been approved for 5 weight percent enriched uranium fuel, and the results remain bounding for the up-rated conditions of 2900 MWt.

3.5 GAS STORAGE TANKS

To support proposed technical specification changes for VCSNS's replacement Steam Generators, the offsite dose calculations for Chapter 15 of the FSAR were performed using revised source terms which reflected plant operation at the Engineered Safeguards Power Level. The licensing submittal (Reference 3) included an evaluation on the consequence of a Waste Gas Decay Tank rupture.

NRC approval of the technical specifications supporting SG replacement was issued via Reference 1. With regard to the dose resulting from the rupture of a Waste Gas Decay Tank, the NRC Staff's Safety Evaluation was as follows:

> "The licensee reevaluated the consequences of a waste gas decay tank rupture. The licensee's submittal stated that the analysis was performed not because of the RSG or due to changes in the design power capability, but rather to reflect TS limits on decay tank radioactivity. The licensee assumed the release of 160,000 Ci of ^{133}Xe. Gamma and beta doses were calculated at the Exclusion Area Boundary and the Low Population Zone. The staff independently assessed the potential consequences of the release of the contents of a waste gas decay tank. The acceptance criterion for the release of the contents of a waste gas decay tank is 0.5 rem total body. Based upon this criterion, the staff determined that the allowable waste gas tank

inventory would be approximately 131,000 Ci of ^{133}Xe. While this particular issue is not associated with the replacement of the D3 steam generators, the licensee should reevaluate the determination of the allowable TS quantity of ^{133}Xe in the waste gas decay tank."

The recommended reevaluation has lead to this proposed change in the Limiting Condition For Operation for Specification 3.11.2.6. It should be noted, however, that the VCSNS has never exceeded an administrative limit of 90,000 Ci of ^{133}Xe in a gas storage tank.

3.6 BALANCE OF PLANT SYSTEMS

3.6.1 Main Steam System

The Main Steam System has been evaluated relative to the Replacement Steam Generator conditions of higher normal steam pressures and the increased steam flow conditions required for uprate when compared to the initial thermal design for the plant. The ASME Design Specifications for the system identify the "Normal" operating pressure for the system to be 1185 psig which bounds the increase in thermal design pressure from 940 to 1000 psia.

Note: The "thermal design pressure" is the pressure identified in the thermal performance kits prepared by the turbine manufacturer.

The mass flow rates for the system remain within acceptable fluid velocities for the piping systems. No load steam conditions remain constant for the NSSS uprate conditions at 2912 MWt.

The capacity of the Main Steam Code Safety valves has been evaluated and adequate margin exists with the existing setpoints and tolerances as identified in the Technical Specifications for the VCSNS.

The rapid (7 seconds or less) closure requirements for Main Steam Isolation Valves (MSIVs) will be retained for the RSG/Uprate power program. The rapid closure of these valves during a large downstream steam line break causes a significant differential pressure across the valve seats and a thrust load on the main steam piping supports in the area of the Main Steam Isolation Valves. During the evaluation of the RSG impact upon NSSS/BOP systems, it was determined that the worst case loading on the MSIVs is generated by rapid closure from no load conditions in response to a Steam Line Break. Since no load temperature and pressure are unaffected by normal operation at a core power of 2900 MWt, the MSIV loadings will not be affected.

The Turbine bypass system is designed to bypass main steam to the main condenser and/or atmosphere, to provide an artificial steam load for the steam generators. The system was originally designed to accommodate a load reduction from 100 percent of rated turbine power to plant auxiliary load without a reactor trip. The capacity of the system was ~85 percent of main steam flow at full load temperature and pressure. With the uprate increase in steam flow of approximately 650,000 lbm/hr, the steam dump capacity of 85 percent is not maintained. The projected capacity at uprate conditions is ~81% of uprate steam flow out of the steam generators. This reduction in capacity would indicate that a large load reduction would result in a reactor trip. The response of the system is not changed due to the plant uprate, only a slight reduction in capacity.

3.6.2 Feedwater System

3.6.2.1 Pumps

The Feedwater system was evaluated relative to the overall capability to provide adequate Feedwater not only for normal steady state operation, but to also operate through expected transients without causing a plant trip. The capability of the pumping systems via the Feedwater Pumps and Feedwater Booster Pumps has been found adequate to provide reliable uprated Feedwater flow. The flow requirements utilized to evaluate these components is normal 100% uprate flow plus, 1% blowdown from the steam generators plus, a 5% flow margin for surges. This results in a required capacity of 106% of normal uprate flow which is within the capability of the pumping components.

3.6.2.2 Feedwater Isolation Valve Closure Piping Loads

Actual Feedwater Isolation Valve closure times are dependent on the pressure difference across the valve as it closes. The power uprate results in higher Feedwater flows thereby changing the pressure differential across the valves.

These new conditions were evaluated via computer model and resulted in acceptable closure times and piping loads. The evaluation was performed by comparing the rate of decrease in feedwater flowrate at uprate conditions to the previous evaluation. Three different cases were run, with all values bounded by previous analysis results.

3.6.2.3 Feedwater Heaters and Drains

Feedwater heaters and drains were evaluated for the uprate conditions. The Feedwater heaters are of adequate size and have the capacity to support plant uprate conditions in an efficient manner. Several of the feedwater heater drain valves have been found to be somewhat undersized and will require minor modifications to increase their capacity to an acceptable value with appropriate margins.

3.6.3 Condensate

The Condensate System was evaluated relative to the capability to pass the required flow, condense uprate steam flows, and fulfill design requirements under transient conditions. The system will fulfill the necessary requirements for uprate conditions with a minor modification. A summary of the evaluation for key components and a description of the modification is included.

3.6.3.1 Condensate Pumps

The system has the pumping capacity to support uprate operations. Typical 100% power operation requires two condensate pumps to supply flow to the Feedwater System. These pumps had an original design capacity of 9360 gpm @ 561 ft TDH. The pumps supplied capacity is 10,015 gpm @ 590 ft TDH. The uprate flow requirements have increased to ~9635 gpm per pump and a total flow of 19268 gpm. The performance of the pumps has been evaluated and found adequate to support uprate power operation. For Enhanced System Reliability, two of the pump motors have been rewound with a higher class of insulation to preclude any potential for excessive condensate motor stator temperatures during high ambient temperature conditions. The third motor will be rewound during RF-9 and will be available prior to plant startup following the outage.

3.6.3.2 Condenser

The Main and Auxiliary Condensers have been evaluated relative to their capability to support uprate steam flows, maintain adequate back pressures for turbine performance, and protect maximum exit temperatures based on approximate Circulating Water flows. Uprate steam rejection conditions to the condenser have been evaluated based on input from the turbine vendor. While the steam mass flow rates through the Low Pressure Turbine rotors increase, the exit/exhaust enthalpies will be lower due to higher turbine efficiencies. This results in an net increased heat load to the condenser of ~4%. This heat load increase was evaluated and found to have minimal affect on the back pressures of the condensers. Exit temperatures of the Circulating Water through the condensers are projected to increase ~0.8°F above current operating conditions, which has been found to be acceptable for the uprate conditions. The temperature effects and environmental concerns will be discussed further in the submittal.

3.6.3.3 Condensate System Modifications

One modification to the Condensate System has been identified as necessary to support uprate flows. The Condensate pumps supply water to the Deaerator through two, parallel cage trim ball valves. Through evaluation and analysis, the capacity of this configuration was determined to be marginal. The configuration will be modified by replacing one of the valves with a higher capacity valve. The valves are sequentially opened, thus the summation of the capacity is the important component. This modification will have minimal if any apparent impact on the operation of the system other than the increased capacity of the flow loop.

3.6.4 Circulating Water System

The Circulating Water System (CW) was evaluated relative to the ability to remove heat from not only the Main Condenser, but also from the Auxiliary Condensers and the various turbine auxiliary systems cooled by this system. A major part of the CW System evaluation included a system flow measurement, utilizing a fluorescent dye dilution method. From this SCE&G confirmed that the design flow of the CW System was still appropriate, and that the flow rates to the Main Condensers were slightly higher than design which would negate some of the effects of uprate heat loads. These higher CW flow rates through the condenser are within design allowables and present an advantage in heat removal capabilities relative to projected uprate heat loads.

3.6.4.1 Main Condenser

The Main Condensers were evaluated relative to the uprate steam conditions and found acceptable. Heat rejection to the CW System is projected to increase ~4% above existing thermal design. This results in a projected increase in the differential temperature across the condenser of ~1.0°F. With the Main Condensers being the major heat load on the system, the effects on maximum lake return temperatures were a major concern. The original and current discharge temperature limit of 113°F to Monticello lake was based on a maximum projected lake supply temperature of 88°F, plus a temperature rise of 25°F across the condensers. With a higher projected temperature rise at uprate conditions, the maximum allowable inlet temperature will be ~86°F which should be acceptable for operation at uprate power conditions. The exit temperature limit of 113°F currently in place is not being changed. Operational strategies have been developed in the event that we challenge the exit temperature limit.

3.6.4.2 Auxiliary Condensers

The Auxiliary Condensers were evaluated and found acceptable. The condensers were initially designed for 50% load (i.e. two out of three FW Pumps operating at 100% power). The condensers have inherent margin in that at 100% power, three Feedwater pumps are operated at approximately 33 1/3% each.

3.6.4.3 Turbine Auxiliary Systems

Initial design of the plant had various turbine auxiliary systems cooled by an Open Cycle Cooling Water System supplied by the Circulating Water system. These auxiliary systems exhibited accelerated corrosion problems due to lower flow velocities and fouling. In order to support the plant uprate operation and to arrest the problem with corrosion and fouling, this open cycle system will be converted to a closed system cooled with a modular forced draft cooling tower.

Not only will this modification solve the fouling problem, enhance performance, and increase reliability, but it will also take a heat/flow load off of the CW System. This will remove a heat load of ~54.087 MBTU/hr along with a reduction in flow demand from the CW System of ~10,000 gpm. This heat load and flow reduction should lessen the impact of the uprate loads on the Main Condensers. This reduction in flow and heat load has not been included in evaluations of the condensers which maintains the original conservatism designed in the plant.

3.6.5 Turbine Generator

While not directly tied to nuclear safety, an important aspect of the uprate project is the turbine/generator systems. The LP Turbine rotors are being replaced due to stress corrosion cracking issues and to incorporate technological improvements in steam path design which will increase the efficiency of the turbine and increase the output of the plant. The new LP Turbine rotors are of a monoblock design which eliminates the shrunk on wheels.

3.6.5.1 Missile Elimination

The existing LP Turbine rotors are of a design which incorporates a wheel which is "shrunk" onto the shaft of the turbine rotor. These wheels then had the turbine blades attached to the wheels. This design had several disadvantages in that at any point where a machined joint between two pieces exists, a susceptibility to stress corrosion cracking also exists. Cracking was identified on these rotors in the area of the keyways on the respective wheels and also in the dovetail blade attachment areas of the wheels. Another negative aspect of this design is that the rotor/wheels could reach a critical speed such that a wheel could burst (due to centrifugal stresses). This condition met the criteria requiring consideration of missile generation. A detailed analysis was required to evaluate the probability of missile generation based on crack growth rates identified at inspection intervals prescribed by the manufacturer.

The LP Rotors will be replaced in Refueling 9 (RF-9) due to reduced service life from stress corrosion cracking previously identified. The new rotor design incorporates enhancements to minimize stress corrosion cracking. The new rotors' monoblock design incorporates integral wheels and couplings as part of a solid shaft to eliminate the material surfaces that promote the initiation of Stress Corrosion Cracking. The monoblock rotor design decreases the probability for missile generation/wheel burst to an annual probability in the range of 10^{-8}, as evaluated by the turbine manufacturer.

The rotor replacement is being implemented under the design change program, with all necessary analysis and calculation changes being made under these programs.

3.6.5.2 Turbine Stop/Control Valves

Another change as a result of the LP Rotor replacement is the relaxation of the testing frequency of the Turbine Stop/Control valves. The proposed change is to extend the valve testing frequency to quarterly. Since the capability to generate a missile is reduced to less than the annual guideline of 10^{-5} with the new rotor design, control system reliability is no longer a significant input to missile probability analysis.

The turbine manufacturer recommends extensions to the testing frequency interval based on experience with the current control system design, no reportable inservice failures of this design, and the annual probability of a complete control system failure being less than 10^{-8}. Post monoblock rotor installation valve testing will be performed for system reliability only.

3.6.5.3 Generator Stator Water Cooling

Uprate evaluations have concluded that the generator stator cooling system capacity will need to increase to ensure adequate heat removal. The stator cooling flow rates are being increased to fulfill this requirement and changes are being made to instrumentation in the system to enhance operability of the system. The generator stator was rewound in Refuel 8 due to increasing Stator Bar cooling gas inleakage and to enhance performance and margin capacity at uprated power. As a result, the generator capacity rating was upgraded to 1,162,300 KVA.

4.0 IMPACT ON THE INDIVIDUAL PLANT EXAMINATION (IPE)

The Individual Plant Examination (IPE) was based on a core power level of 2785 MWt. The effect of increasing core power to 2900 MWt was investigated to determine if any changes in the results of the original Probabilistic Safety Assessment (PSA) would occur. Additionally, the review "sought out" areas of the PSA where a decrease in margin in equipment or human performance might occur. The analysis of the impact of uprate on the PSA serves to enhance the more traditional deterministic analysis described in the body of this document.

The evaluation methodology was based on the following criteria which was applied to the individual success criteria and then to the overall IPE results:

- Will the power uprating impact any of the numbers that directly link to the core damage frequency, the frequency of fission product release categories, or change any of the top 100 core damage sequences reported in the IPE?
- Does the power uprating cause a change to any of the assumptions or models used in the IPE analysis, but does not impact the IPE results reported in the submittal report?
- Does the power uprating cause a significant loss of margin in the success criteria such that the success criteria may change if other modeling sensitivities are considered?
- Is there an issue related to the original IPE success criteria that was identified during this review?

During the review, several issues were identified as being impacted by the power uprate. For example, the time available for successful operator action in areas dependent on core power, such as the time to steam generator dry out during a loss of heat sink event, have been slightly effected. The various issues were reviewed for impact on the IPE results and conclusions. In all cases, the effects were determined to be minor and bounded by the original IPE analysis. The impact of the uprate on plant response will be incorporated into the "living" PSA program.

5.0 ENVIRONMENTAL EFFECTS

The plant uprate effects on the environment are considered to be minor. The power uprate does increase the heat rejection rate to the environment ~4% via the Main Condenser to the Circulating Water System. Original environmental assessments identified that the system had a capability to transfer ~6670 MBTU/hr based on a total Circulating Water system flow rate of ~534,000 gpm with a temperature rise of ~25°F as noted in the Operating License Environmental Report (OLER). The conversion of the open and closed cycle auxiliary system heat loads will remove ~54.087 MBTU/hr from the system. The following summarizes the design, current, and projected uprate heat loads:

HEAT SOURCE	DESIGN DUTY MBTU	CURRENT MBTU	UPRATE MBTU
Main Condensers	6000.00	6006.3	6240.0
Auxiliary Condensers	249.75*	161.1	164.4
Other Auxiliary Loads	54.10	54.1	---
TOTALS	6303.85	6221.5	6404.4

The uprate heat load is:

~1.6% >Design
~2.9% >Current
~96% of System Capability

* The load in the OLER reflects only the design duty of one condenser.

The heat rejection increase is relatively insignificant and is within the system capability values presented in the Operating License Environmental Report.

Slight changes to overall system flow will be made when the Auxiliary System loads are removed from the CW system. This will result in a corresponding reduction of flow demand on the system of ~7000 to 10,000 gpm. A majority of this flow will re-distribute itself within the Main and Auxiliary Condensers, but the overall CW system flow rate should decrease to something less than the 534,000 gpm identified in the OLER. Therefore, the velocities and flow rates identified in the OLER will remain bounding for the uprate conditions.

A key aspect of this effort, is that the existing discharge temperature limit for the Circulating Water return stream to the lake will not change from the existing 113°F limit.

This information has been presented to the South Carolina Department of Health and Environmental Control (SCDHEC) and has been found acceptable.

Environmental monitoring will continue along with trending of the CW system discharge temperatures to ensure that the environmental impact is benign.

CLOSED CYCLE COOLING SYSTEM MODIFICATION

The removal of the closed and open cycle cooling systems from the CW system are being performed under the modification program. Environmental assessments of the heat loads, atmospheric affects, and operational considerations are coordinated within the design control program. Submittals for environmental considerations have been coordinated through the SCDHEC. Environmental monitoring is maintained by SCE&G within prescribed requirements.

6.0 EQUIPMENT ENVIRONMENTAL QUALIFICATION AND HIGH ENERGY LINE BREAK (HELB) IMPACT

The Equipment Environmental Qualifications were evaluated under the Replacement Steam Generator Project which was implemented in Refueling 8. Due to power uprate and changes in mass inventory of the Steam Generators, the environmental consequences during Loss-of-Coolant Accidents and Steam Line Breaks, both inside and outside containment, were evaluated. The results of these evaluations were then used to evaluate the effects on the respective Equipment Qualifications and other components as applicable. The uprate conditions were utilized to perform the evaluations to avoid repeating the effort. All analysis was completed under the Replacement Steam Generator Project and were submitted for review as appropriate.

7.0 OTHER SYSTEMS EVALUATED BUT NOT AFFECTED BY UPRATE

Various systems were evaluated and found not effected by the uprate project over the course of several review phases. Those systems were evaluated for respective capacities, heat removal capabilities, and in many cases no direct connection to plant uprating was found. The following is a summary listing of major plant systems that were not affected by the uprate:

Auxiliary Steam
Condenser Air Removal
Chemical Feed Systems
Emergency Diesel Generators and Auxiliaries
Solid and Liquid Waste Systems
Fire Service
Station / Instrument Air
Reactor Building Cooling
Generator Gas and Vents
Non-Nuclear Drains
Plant Waste
RB Spray
Demineralized Water System
Nuclear and Secondary Sampling
HVAC

8.0 CONCLUSIONS

The power uprate of the VCSNS will increase the RATED THERMAL POWER of the reactor core from a nominal 2775 MWt to 2900 MWt, with a corresponding increase in net electrical output of approximately 64 MWe.

Amendment No. 119 to the Facility Operating License No. NPF-12 for the VCSNS was recently issued by the Nuclear Regulatory Commission via Reference 1. This amendment changed the Technical Specification in response to SCE&G's requests in Reference 3 to support VCSNS operation with replacement steam generators at its current RATED THERMAL POWER level of 2775 MWt. Where possible, the analyses and evaluations within Reference 3 (see Attachment I) were, however, performed at a core power level up to 2900 MWt, which corresponds to the plant's original Engineered Safeguards design rating.

This amendment request seeks approval to operate the VCSNS at a RATED THERMAL POWER level of 2900 MWt with replacement steam generators. In support, additional analyses, evaluations, and Technical Specifications changes are provided to supplement those outlined in Attachment 1. When implemented, the proposed changes along with those previously approved via Reference 1 will preserve the design, analytical methodology, and safety analysis assumptions outlined in this amendment request.

The safety analyses, evaluations and supporting documentation provided in or referenced by this submittal demonstrate that the VCSNS can be safety operated at a core power level of up to 2900 MWt without undue risk to the health and safety of the public.

9.0 REFERENCES

1. USNRC Safety Evaluation Report (SER) Related to Amendment No. 119 to Facility Operating License NPF-12, South Carolina Electric and Gas Company, South Carolina Public Service Authority, Virgil C. Summer Nuclear Station, Unit No. 1, Docket No. 50/395, November 18, 1994.

2. USNRC Safety Evaluation Report (SER) Related to Amendment No. 27 to Facility Operating License NPF-12, South Carolina Electric and Gas Company, South Carolina Public Service Authority, Virgil C. Summer Nuclear Station, Unit No. 1, Docket No. 50/395, September 27, 1994.

3. SCE&G Letter from J. L. Skolds to the USNRC Document Control Desk, "Virgil C. Summer Nuclear Station, Docket 50/395, Operating Licensing No. NPF-12, Steam Generator Replacement Technical Specification Change Request (REM 6000-8, TSP 930015)", October 29, 1993. (Additional submittals in support of this were made September 3, 1992; March 12, 1993; April 30, 1993; March 11, 1994; and October 20, 1994.)

4. SCE&G Letter from O. W. Dixon to the USNRC Document Control Desk, "Virgil C. Summer Nuclear Station, Docket 50/395, Operating Licensing No. NPF-12, High Density Spent Fuel Storage Racks", January 23, 1984.

5. Virgil C. Summer Nuclear Station Final Safety Analysis Report

6. Virgil C. Summer Nuclear Station Technical Specifications

7. SECL-93-036 (dated October 1993) and SECL-93-036, Rev. 1 (dated February 1994): Licensing Submittal to Support Replacement Steam Generator Technical Specification Changes for the VCSNS.

8. WCAP-13714 (Proprietary Class 2C), "Virgil C. Summer Nuclear Station Replacement Steam Generator/Uprating Engineering Report", June 1994.

9. WCAP-12266-P-A, Rev. 2 (Proprietary), WCAP-11524-NP-A, Rev. 2 (Non-proprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987; Including Addendum 2-A, "BASH Methodology Improvements and Reliability Enhancements," May 1988.

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 following Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) pages: 1-5, 3/4 4-31, 3/4 4-32, 3/4.11-5, and 6-16a. 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 BASH/BART methodology for Large Break Loss of Coolant Accident analysis.
- revision to the Pressure Temperature Limitations Curves due to effects of increased neutron fluence at 2900 MWt.

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 1981 Evaluation Model with BASH, 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.

Increased neutron fluence resulting from uprate core conditions has an effect on the reactor vessel Pressure Temperature Curves. Their applicability will change from 14 Effective Full Power Years (EFPY) to 13 EFPY with no other changes at this time.

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 153°F with two Spent Fuel Cooling loops operating and to less than bulk boiling during limiting heat loads with one Spent Fuel Cooling loop operating. 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.

The Pressure Temperature Limitations Curves are derived using NRC Approved Methodology to comply with 10 CFR 50, Appendix G. These curves provide an acceptable range of operating temperatures and pressures for heatup, cooldown, low temperature overpressure, criticality, and inservice leak and hydrostatic testing conditions. The reduction in applicability for these curves has no effect on the curves themselves. Only the amount of time between the next scheduled specimen capsule analysis and the next revision to these curves will be effected.

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. The calculated results to VCSNS FSAR Chapter 15 Analyses demonstrate that there are no challenges to the integrity of the primary/secondary/containment pressure boundaries and that the plant remains within the regulatory acceptance criteria applied to the VCSNS current licensing basis.