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Document Control Desk
U. S. Nuclear Regulatory Commission
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Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Technical Specification Change
Addendum - VANTAGE 5 Reload

Gentlemen:

On May 20, 1988, South Carolina Electric & Gas Company (SCE&G) requested a revision to the Technical Specifications for the Virgil C. Summer Nuclear Station in support of refueling and operating the Virgil C. Summer plant with VANTAGE 5 improved fuel design. As previously provided, the attachments of the May 20, 1988 letter constituted a partial submittal for the transition to VANTAGE 5 fuel. The original analysis reduced T_{ave} to 585.5°F from 587.4°F which would have resulted in a three megawatt electrical output penalty. SCE&G elected to perform the reanalyses necessary to recover the electrical output.

This letter forwards an addendum to Attachment 1 of the original submittal which comprises the documentation necessary to complete the Technical Specifications change request with T_{ave} reanalyzed to 587.4°F. Enclosure 1 to the attached addendum contains the revised page changes to the original submittal. These should replace the pages from the original submittal on a one for one basis.

The enclosed Technical Specifications addendum has been reviewed and approved by both the Plant Safety Review Committee and the Nuclear Safety Review Committee.

Should you have any questions concerning this Technical Specifications addendum, please call at your convenience.

Very truly yours,

D. A. Nauman

MDB/DAN:lcd
Attachment

pc: See Page 2

A001
1/1

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ADDENDUM TO ATTACHMENT 1

SUPPLEMENTAL
SAFETY EVALUATION
FOR THE
VIRGIL C. SUMMER NUCLEAR STATION
TRANSITION TO WESTINGHOUSE 17 X 17 VANTAGE 5 FUEL
WITH A TAVE OF 587.4°F

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50-395

JUNE, 1988

1.0 INTRODUCTION

The safety evaluation supporting the Virgil C. Summer Nuclear Station (VCSNS) transition to VANTAGE 5 fuel was performed assuming a nominal average coolant temperature (Tave) of 585.5°F. This value was chosen in conjunction with other baseline parameters so that the RCS hot leg temperature remained at, or below, the original design value of 618.7°F with up to 15% steam generator tube plugging. This limit on reactor coolant system (RCS) hot leg temperature was originally imposed to minimize the corrosion rate of the steam generators which is sensitive to high coolant temperatures. However, full power operation with a Tave of 585.5°F, instead of the current Tave of 587.4°F, would result in a decrease in the plant electrical output of approximately 3 megawatts due to a reduction in steam pressure.

Subsequent to the selection of the baseline parameters for the VANTAGE 5 transition evaluation, SCE&G established alternative programs to manage the impact of steam generator corrosion and thus desires to maintain the current Tave of 587.4°F for Cycle 5 and beyond. This adjustment in the VANTAGE 5 baseline Tave will not only prevent the loss of 3 megawatts electrical output but will also minimize the impact of the fuel transition on station operation (i.e., Tave control, pressurizer level control, steam generator level control, etc.) and operator training.

In order to minimize the impact of the 1.9°F increase in Tave on the VANTAGE 5 safety evaluation, existing margin in thermal design flow (TDF) has been utilized to offset the decrease in the departure from nucleate boiling ratio (DNBR) which would have occurred with an increase in Tave. Specifically, TDF will be increased by approximately 2.0%, from 277800 gallons per minute (gpm) to 283500 gpm. The changes in Tave and TDF result in modifications to the VANTAGE 5 Technical Specifications revisions as previously forwarded to the Nuclear Regulatory Commission on May 20, 1988. Changes for these affected Technical Specifications are proposed herein.

The purpose of this addendum to Attachment 1 of SCE&G's licensing submittal is to (1) justify the continued applicability of the results and conclusions of the VCSNS VANTAGE 5 licensing submittal with a revised Tave of 587.4°F and (2) provide the Technical Specifications page markups, reflecting the adjustment in Tave and TDF, to replace those previously proposed.

2.0 BASELINE PARAMETER CHANGES

Two primary modifications to the VCSNS VANTAGE 5 baseline parameters are proposed:

1. Increase the 100% full power Tave from 585.5°F to 587.4°F.
2. Increase TDF from 277800 gpm to 283500 gpm.

The increase in TDF provides an offset to the resulting decrease in DNBR due to the increase in Tave. The increased TDF, however, remains adequate to allow up to 15% uniform steam generator tube plugging without the need for additional safety analysis.

The above modifications affect other plant design parameters such as RCS hot leg and cold leg temperatures. These secondary changes are illustrated in the change pages (see Table 5.1) in Enclosure 1.

3.0 PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGES

Five modifications to the previously proposed Technical Specifications changes, supporting the transition to VANTAGE 5 fuel, are required to reflect the adjustments in Tave and TDF. These modifications are described below:

1. Core Safety Limits (TS Figure 2.1-1)

   The core safety limits, which are the loci of points of terminal power, RCS pressure and Tave, below which the calculated DNBR is no less than the design DNBR or below which the core average enthalpy at the vessel exit is less than the enthalpy of saturated liquid, are revised to reflect the increase in RCS flow. The revised limits continue to be those for the low parasitic (LOPAR) fuel which are limiting during the transition period.

2. Loop Design Flow (TS Table 2.2-1)

   The loop design flow, derived from the TDF and by the flow measurement uncertainty (2.1%) for use in the reactor protection system low flow trip, is increased from 96200 to 96500 gpm. This change insures that the consequences of a loss of flow transient would be bounded by those predicted in the VANTAGE 5 safety analysis with a nominal Tave of 587.4°F.

3. Overpower-delta-T/Overtemperature-delta-T Trip Setpoints (TS Table 2.1-1, Notes 1 & 3)

   The overpower and overtemperature delta-temperature (delta-T) trip setpoints are revised since they are a function of Tave and are derived from the core safety limits. The reference Tave at rated thermal power (RTP) is modified from 585.5°F to 587.4°F; therefore, no changes in the current values for the Overtemperature-delta-T and Overpower-delta-T trips will be required to support the transition to VANTAGE 5. In addition, a minor slope change for the change in axial flux penalty term in the Overtemperature-delta-T trip (item iii in note 1) is required due to the change in the core safety limits.

4. RCS Total Flow Rate (TS Figure 3.2-2)

The increase in system flow has been factored into the limiting conditions of operation defined by RCS total flow rate and the nuclear enthalpy rise hot channel factor within Technical Specifications Figure 3.2-2. The 100% RTP value corresponds to the minimum measured flow assumed in the VANTAGE 5 transition safety analysis.

5. DNB Parameters (TS Table 3.2-1)

The RCS Tave is modified to reflect the nominal value of 587.4°F plus the uncertainty for Tave (4.3°F) assumed in the safety analysis utilizing the Improved Thermal Design Procedure.

No other modification to the proposed Technical Specifications changes to support transition to VANTAGE 5 fuel are required. Technical Specifications markup pages for the above modification are provided in Enclosure 1.

4.0 SAFETY EVALUATION

4.1 Scope of Evaluation

The impact of the proposed changes to the VANTAGE 5 baseline parameters on the results and conclusions of the original licensing submittal dated May 20, 1988 have been assessed with the following conclusions relative to the major technical topics addressed in Attachments 1, 3, 4, 5, 6, and 7 of the original submittal.

Mechanical Evaluation (Attachment 1, Section 3.0)

The changes in baseline parameters do not impact the Mechanical Evaluation to the VCSNS license submittal since the calculations conducted to justify the mechanical integrity of the fuel assembly and core components utilized design basis parameters which are bounding for VCSNS. The results are within the allowable design limits. The Tave change also results in a negligible increase in fuel pellet temperatures and fission gas release. Therefore, the fuel rod design evaluation performed for the VCSNS licensing submittal remains valid.

Nuclear Evaluation (Attachment 1, Section 4.0)

The change in Tave from 585.5°F to 587.4°F and increase in TDF from 277800 gpm to 283500 gpm will not adversely impact Attachment 1, Section 4.0 of the VCSNS licensing submittal. The nuclear parameters analyzed are relatively insensitive to small changes in vessel temperature and flow so any changes in the parameters can be accomodated with the uncertainty that was applied during the analysis. The VCSNS licensing submittal conclusions remain valid.

Thermal Hydraulic Evaluation (Attachment 1, Section 5.0)

For the Thermal Hydraulic Evaluation, Attachment 1, Section 5.0 of the VCSNS licensing submittal, the effects of a Tave change from 585.5°F to 587.4°F and a TDF change from 277800 gpm to 283500 gpm were evaluated. The net change resulted in a maximum DNBR penalty of 1%. There is sufficient margin between the safety analysis and design DNBR values to accomodate this 1% DNBR penalty.

Accident Evaluation (Attachment 1, Section 6.0 and Attachments 3 and 4)

The proposed changes in baseline parameters are key initial conditions to the accident analyses. Each accident analysis has been assessed for impact and a detailed overview is provided in Section 4.2. Overall, all conclusions in the VCSNS licensing submittal relative to accident analysis were confirmed to remain valid with no degradation in margin to safety.

Thimble Plug Removal (Attachment 5)

The Thimble Plug Removal Evaluation, Attachment 5 of the VCSNS licensing submittal is not affected by the baseline parameter changes, because the reactor flow resistances do not change.

Significant Hazards Evaluation (Attachment 6)

The engineering evaluation and additional accident analyses performed show that the technical bases for the Significant Hazards Evaluation is maintained. The proposed changes in baseline parameters do not invalidate the conclusions of the VCSNS licensing submittal, including those in the Significant Hazards Evaluation. With changes to Table 1 of Attachment 6 (see Enclosure 1) to reflect changes in baseline parameters, it is concluded that the Significant Hazards Evaluation is appropriate for the VCSNS VANTAGE 5 Technical Specifications as modified by this addendum.

Radiological Impact Assessment (Attachment 7)

The discussion of Radiological Impact Assessment provided in Attachment 7 of the VCSNS licensing submittal addresses the impact of extended fuel burnup on core fission product inventories and on the radiological consequences of accidents. The increase in Tave from 585.5°F to 587.4°F does not affect this discussion.

4.2 Accident Evaluation

The impact of the proposed changes in baseline parameters has been assessed for each of the non-loss-of-coolant-accident (LOCA) and LOCA safety analyses reported in Attachments 1, 3, and 4 of the VCSNS licensing submittal. A summary of this input assessment is provided below. Each section title provides a cross-reference to the corresponding section of the previous licensing submittal.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (Attachment 3, Section 15.2.1)

An increase in the nominal at power Tave from 585.5°F to 587.4°F will not have an effect on this transient. This transient is analyzed under hot zero power conditions and is not impacted by changes in the at power reactor coolant conditions. In addition, a 2% increase in flow would have a beneficial impact on the calculated minimum DNBR. Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (Attachment 3, Section 15.2.2)

This analysis is performed at various power levels, assuming minimum and maximum feedback, for a range of reactivity insertion rates. Protection is provided by the overtemperature delta-T and high neutron flux-high setting reactor trips. Revised overtemperature delta-T setpoints have been calculated based on a nominal Tave of 587.4°F and a minimum measured flow of 289500 gpm. These setpoints ensure that departure from nucleate boiling (DNB) will not occur over the range of reactivity insertion rates analyzed.

Proper calibration of the excore neutron flux detectors at the revised RCS average temperature will ensure that the high neutron flux trip function will provide adequate protection. For those reactivity insertion rates which rely upon the high neutron flux trip for protection, the DNB penalty associated with the temperature increase is offset by the increase in RCS flow.

Thus, the revised overtemperature delta-T setpoints ensure that the conclusions in the VCSNS licensing submittal remain valid.

Rod Cluster Control Assembly Misoperation (Attachment 3, Section 15.2.3)

Three rod cluster control assembly (RCCA) misalignment accidents are presented in VCSNS licensing submittal Section 15.2.3: one or more dropped RCCAs within the same group, a dropped bank, and a statically misaligned RCCA. The analysis presented in the VCSNS licensing submittal demonstrates that for all cases, the DNB design basis is met. Evaluations have shown that, with an increase in Tave to 587.4°F and an increase in the minimum measured flow to 289500 gpm, the DNB design basis is still met for each of these events.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Uncontrolled Boron Dilution (Attachment 3, Section 15.2.4)

The VCSNS licensing submittal presents the results of analyses performed in Modes 1 through 6. The analyses demonstrate that during a dilution event, adequate operator action time is available to detect and terminate the dilution before shutdown margin is lost. Verification that an increase in Tave to 587.4°F will not impact the core physics characteristics has been made. Since the reactor coolant flow has no impact upon this safety analysis, the behavior of the reactor during the dilution event and the time at which shutdown margin is lost has not been impacted.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Partial Loss of Forced Coolant Flow (Attachment 3, Section 15.2.5)

The evaluation for the partial loss of forced coolant flow analysis presented in Attachment 3, Section 15.2.5 is discussed with the complete loss of flow evaluation.

Startup of an Inactive Reactor Coolant Loop (Attachment 3, Section 15.2.6)

The analysis presented in the VCSNS licensing submittal is initiated from 60% power. The initial Tave corresponding to this power is 574.1°F (without uncertainties) for the analysis presented in the VCSNS licensing submittal. An increase in nominal Tave to 587.4°F would represent a Tave of 575.2°F at 60% power, a 1.1°F increase from the VCSNS licensing submittal case.

The transient is terminated by a reactor trip on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint which is conservatively assumed to be 74% power. The nuclear power transient is driven by the effect of the moderator temperature coefficient and the reduction in the RCS temperature caused by the forced flow of colder water (cold leg temperature) to the core from the inactive loop.

A 2% increase in the TDF to 283500 gpm will result in a decrease in the hot leg to cold leg delta-T. Thus, the driving force of the nuclear power transient discussed above will be slightly diminished, resulting in a slower transient. However, reactor trip will still occur upon reaching the P-8 setpoint. The small increase in the initial average temperature would not have a significant effect on the transient behavior of this event.

An examination was made of the available DNB margin. Due to the large margin available, a 1.1°F increase in Tave would not have a significant adverse impact upon the available DNB margin. In addition, a 2% increase in flow would have a beneficial impact upon the minimum DNBR.

Additionally, an examination was made of the available margin to the overpressurization limit of 110% design pressure. The maximum RCS pressure calculated in the VCSNS licensing submittal is 2247 psia with a limit of 2750 psia. On the secondary side, the maximum steam pressure was calculated to be 1084 psia with a limit of 1320 psia. Due to the margins available, the changes in RCS average temperature and flow will not have a significant adverse impact on the values presented in the VCSNS licensing submittal.

Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Loss of External Electrical Load and/or Turbine Trip (Attachment 3, Section 15.2.7)

A reanalysis of the loss of load/turbine trip event presented in the VCSNS licensing submittal has been performed. The revised analysis assumes initial conditions consistent with a nominal Tave of 587.4°F; however, no credit was taken for any increase in RCS flow over the value assumed in the VCSNS licensing submittal analysis. This is a conservative assumption for this event. The analysis demonstrates that the DNB design basis is met and that overpressurization of the primary and secondary systems does not occur.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Loss of Normal Feedwater (Attachment 3, Section 15.2.8)

A reanalysis of the loss of normal feedwater event presented in the VCSNS licensing submittal has been performed. The revised analysis assumes initial conditions consistent with a nominal Tave of 587.4°F. However, no credit was taken for any increase in RCS flow over the value assumed in the VCSNS licensing submittal analysis. This is a conservative assumption for this event. The analysis demonstrates that the core is not adversly affected and that no water relief occurs through the pressurizer safety or relief valves.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Loss of Offsite Power to the Station Auxiliaries (Station Blackout) (Attachment 3, Section 15.2.9)

The results of this event which could be impacted by the subject increases in Tave and minimum measured flow are bounded by the results for the loss of normal feedwater evaluation discussed above.

Excessive Heat Removal Due to Feedwater System Malfunctions (Attachment 3, Section 15.2.10)

The analysis presented in the VCSNS licensing submittal simulates the opening of one feedwater control valve. Two cases are analyzed: one with the reactor just critical at zero power with a moderator density coefficient characteristic of end-of-life conditions and second with the reactor in manual control at full power.

An increase in the nominal Tave to 587.4°F will have no impact on the zero power case. Thus, the maximum reactivity insertion rate analyzed in Section 15.2.1 of the VCSNS licensing submittal, Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition, will still bound that calculated for the feedwater system malfunction at zero power. In addition, a 2% increase in the minimum measured flow will provide a DNB benefit for this analysis.

For the full power case, excessive feedwater flow results in an increase in core heat flux, pressurizer pressure and RCS delta-T and a decrease in Tave. Reactor trip occurs on high-high steam generator level. An increase in Tave to 587.4°F will not have an adverse impact on the transient behavior. The time of reactor trip would be essentially unchanged since this is primarily a function of the feedwater flow increase. In addition, a 2% increase in flow would provide a DNB benefit.

Thus, the conclusions in the VCSNS licensing submittal will remain valid.

Excessive Load Increase Incident (Attachment 3, Section 15.2.11)

The analysis presented in the VCSNS licensing submittal demonstrates plant behavior following a 10% step load increase from full power. Four cases are analyzed: at the beginning and end-of-core life, each considered with the reactor in manual and automatic control. In all four cases, reactor trip does not occur and the plant reaches a stabilized condition following the load increase.

An increase in Tave to 587.4°F will not have a significant adverse impact upon the transient behavior calculated in the VCSNS licensing submittal. As can be seen in the VCSNS licensing submittal, the excessive load increase events without automatic control result in a reduction in Tave due to increased heat extraction by the secondary side. Should these events be initiated with a 1.9°F increase in Tave, the rate and magnitude of the Tave reduction would be unchanged. The result would merely be a 1.9°F increase in the stable temperature when it is finally reached.

Conversely, for those cases in which automatic control is assumed to be available, Tave is essentially maintained at the initial value throughout the transient. Thus, a 1.9°F increase in the initial Tave will result in a 1.9°F increase in the stable temperature calculated in the VCSNS licensing submittal.

For all of the above cases, a 2% increase in the RCS flow would tend to reduce the time required for the primary system to achieve a new stable condition. This is due to the improved heat transfer characteristics which result from the higher flow. In addition, the higher RCS flow would serve to increase the margin to DNB calculated in the VCSNS licensing submittal.

Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Accidental Depressurization of the Reactor Coolant System (Attachment 3, Section 15.2.12)

The analysis presented in the VCSNS licensing submittal simulates the inadvertent opening of a pressurizer safety valve. This results in a rapid depressurization of the RCS until saturation occurs in the hot leg at which time depressurizations continues, but at a slower rate. The reactor is tripped on an overtemperature delta-T signal.

An increase in the nominal Tave to 587.4°F and a minimum measured flow of 289500 gpm will have no significant impact upon the transient. The transient behavior (e.g., the RCS depressurization) is driven by the safety valve relief rate which will not be significantly impacted by the change in RCS flow or the small increase in temperature.

As discussed above, revised overtemperature delta-T setpoints have been calcuated based on a nominal Tave of 587.4°F and a minimum measured RCS flow of 289500 gpm. These setpoints will maintain the initial margin to trip and provide a reactor trip signal with the same margin to DNB as in the VCSNS licensing submittal analysis.

Thus, the revised overtemperature delta-T setpoints will ensure that the DNB design basis is met and the conclusions in the VCSNS licensing submittal remain valid.

Accidental Depressurization of the Main Steam System (Attachment 3, Section 15.2.13)

An increase in the nominal at power Tave from 585.5°F to 587.4°F will not have an effect on this transient. This transient is analyzed under hot zero power conditions and is not impacted by changes in the at power reactor coolant conditions. In addition, sensitivity studies (WCAP 9227) show that a 2% increase in flow would have essentially no impact on the calculated minimum DNBR. Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Spurious Operation of the Safety Injection System at Power (Attachment 3, Section 15.2.14)

Two cases are discussed in the VCSNS licensing submittal. In Case A the reactor trip signal occurs at the same time as the spurious safety injection begins. For this case the operator would terminate safety injection and bring the plant to a cold shutdown condition. The results for this case are unchanged for an increase in the nominal Tave to 587.4°F and an increase in the minimum measured flow to 289500 gpm.

In Case B, reactor trip is not initiated by a safety injection signal, but occurs later on low pressurizer pressure. Throughout the Case B transient, the average temperature, nuclear power, pressurizer water level and pressure decrease due to the negative reactivity provided by the injection of borated water. The minimum DNBR is never less than the initial value.

Should this event be initiated with a higher Tave of 587.4°F and a 2% increase in minimum measured flow, the time of reactor trip would remain essentially unchanged since the depressurization of the RCS is driven by the negative reactivity provided by the safety injection flow. Thus, the behavior of this transient would be unchanged and the minimum DNBR will increase throughout the transient.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Minor Secondary System Pipe Breaks (Attachment 3, Section 15.3.2)

An increase in the nominal at power Tave from 585.5°F to 587.4°F will not have an effect on this transient. This transient is analyzed under hot zero power conditions and is not impacted by changes in the at power reactor coolant conditions. In addition, sensitivity studies (WCAP 9227) show that a 2% increase in flow would have essentially no impact on the calculated minimum DNBR. Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Inadvertent Loading of a Fuel Assembly Into An Improper Position (Attachment 3, Section 15.3.3)

An evaluation has been performed for this event. An increase in the nominal Tave to 587.4°F, as well as an increase in the RCS flow rate will not have an adverse impact upon the analysis presented in the VCSNS licensing submittal. Thus, the conclusions in the VCSNS licensing submittal remain valid.

Complete Loss of Forced Reactor Coolant Flow (Attachment 3, Section 15.3.4)

A reanalysis of the most limiting of the 1 of 3 loops loss of flow (partial loss of flow) and 3 of 3 loops loss of flow (complete loss of flow) cases has been performed. This analysis assumed nominal conditions consistent with a vessel average temperature of 587.4°F and a minimum measured flow of 289500 gpm. All other assumptions were identical to those made in the VCSNS licensing submittal analysis. The results demonstrate that the DNB design basis is met for both the partial and complete loss of flow events.

Thus, the conclusions in sections 15.2.5 and 15.3.4 of Attachment 3 of the VCSNS licensing submittal remain valid.

Single Rod Cluster Control Assembly Withdrawal at Full Power (Attachment 3, Section 15.3.6)

An evaluation has been performed for this event. An increase in the nominal Tave to 587.4°F, as well as an increase in the RCS flow rate will not have an adverse impact upon the analysis presented in the VCSNS licensing submittal. Thus, the conclusions in the VCSNS licensing submittal remain valid.

Major Secondary System Pipe Ruptures (Attachment 3, Section 15.4.2)

Two analyses are presented in Attachment 3, Section 15.4.2: rupture of a main steam line and a major rupture of a main feedwater pipe.

Rupture of a Main Steam Line (Attachment 3, Section 15.4.2.1)

An increase in the nominal at power Tave from 585.5°F to 587.4°F would not have an effect on this transient. This transient is analyzed under hot zero power conditions and is not impacted by changes in the at power reactor coolant conditions. In addition, sensitivity studies (WCAP 9227) show that a 2% increase in flow would have essentially no impact on the calculated minimum DNBR.

Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Major Rupture of a Main Feedwater Pipe (Attachment 3, Section 15.4.2.2)

A reanalysis of the feedline break event has been performed. The revised analysis was performed at engineered safeguards design power assuming initial conditions consistent with a nominal Tave of 587.4°F. However, no credit was taken for any increase in RCS flow over the value assumed in the VCSNS licensing submittal analysis, which is a conservative assumption for this event. The analysis demonstrates that the core remains covered and that overpressurization of the RCS does not occur.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Single Reactor Coolant Pump Locked Rotor (Attachment 3, Section 15.4.4)

A reanalysis of the locked rotor event has been performed. This analysis assumed nominal conditions consistent with a Tave of 587.4°F and a 2% increase in the minimum measured flow, all other assumptions were identical to those made in the VCSNS licensing submittal analysis. The maximum RCS pressure was calculated to be 2605 psia, a reduction of 1 psi from the analysis presented in the VCSNS licensing submittal. The maximum clad temperature at the hot spot was calculated to be 1974°F, 9°F less than that presented in the VCSNS licensing submittal, and the amount of zirconium-water reacting at the hot spot is 0.55% by weight. The results also demonstrate that no more than 15% of the fuel rods will experience DNB.

Thus, the conclusions in the VCSNS licensing submittal remain valid.

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (Attachment 3, Section 15.4.6)

Four cases are analyzed in the VCSNS licensing submittal, at the beginning and end-of-life, each considered under hot zero power and full power conditions. As can be seen in the VCSNS licensing submittal, those analyses performed under hot zero power conditions approach the peak clad temperature limit of 2700°F, while the hot full power cases approach the fuel melt limit of 10% at the hot spot.

An increase in the nominal at power Tave from 585.5°F to 587.4°F will not have an effect on the hot zero power transient cases. These cases are analyzed under hot zero power conditions and are not impacted by changes in the at power reactor coolant conditions. In addition, a 2% increase in the reactor coolant flow would provide improved fuel to moderator heat transfer, serving to reduce the calculated peak clad temperatures presented in the VCSNS licensing submittal.

The hot full power analyses will not be adversely impacted by an increase in Tave to 587.4°F. Due to the inherent thermal lag in the fuel pellet, fuel melt occurs prior to effective heat tranfer to the clad and coolant. Thus, the amount of energy deposition in the pellet following the ejection of a control assembly and the degree of fuel melt is not sensitive to Tave. In addition, the analysis presented in the VCSNS licensing submittal conservatively assumes that DNB occurs immediately following ejection of the control assembly. This conservatively degrades heat transfer capability to the coolant, thereby reducing any heat transfer temperature dependency. Note that a 1.9°F increase in Tave would slightly increase the initial pellet average temperature. However, this effect is very small and will not significantly impact the results. Similarly, due to the thermal lag in the pellet, a 2% increase in the reactor coolant flow would have no significant beneficial impact on the degree of fuel melt.

Thus, for this event, the conclusions in the VCSNS licensing submittal remain valid.

Steam Line Break Mass & Energy Releases (Attachment 1, Section 6.1.4)

An examination was made of the steam line break mass and energy (M/E) release analyses performed for the VCSNS plant with standard fuel as well as VANTAGE 5 fuel. Both of these analyses conservatively assume a Tave of 587.4°F and a TDF of 294000 gpm. An increase in Tave from 585.5°F (as assumed in the VCSNS licensing submittal) to 587.4°F will not have an adverse impact on the steam line break M/E release analysis since this condition is already accommodated.

In addition, the revised TDF of 283500 gpm is lower than the value of 294000 gpm assumed in the M/E analysis. Reduced RCS flows reduce the primary to secondary heat transfer characteristics which will impede the ability to transfer heat to the effluent. Thus, the enthalpy of the fluid released to containment will be reduced.

Therefore, the steam line break M/E releases previously calculated for V. C. Summer, will remain bounding for VANTAGE 5 fuel with a Tave of 587.4°F and a TDF of 283500 gpm.

Large Break LOCA (Attachment 4, Section 15.4.1)

The licensing basis for the large break LOCA analysis for VCSNS was performed to bound a range of operating temperature conditions. Sensitivities to the range of temperature conditions determined that the higher operating temperature conditions yielded higher peak clad temperatures. Thus the break spectrum was analyzed at the higher operating temperature conditions. The Tave used in the LOCA analysis for the upper range of plant temperatures bounds the revised Tave of 587.4°F. Thus it can be concluded that the revised Tave for VCSNS will not have an adverse effect on the large break LOCA results.

Small Break LOCA (Attachment 4, Section 15.3.1)

The small break LOCA licensing analysis for VCSNS included calculations to investigate the influence of loop operating temperatures on peak clad temperature results. The results of these calculations showed that peak clad temperatures were predicted to decrease with decreasing initial loop temperatures in the range of several degrees from nominal. Based upon this behavior the actual licensing calculations were performed at the higher, nominal operating temperatures. The Tave used in the LOCA analysis for the upper range of plant temperatures bounds the revised Tave of 587.4°F; therefore, the calculations reflect the proper conditions for the licensing basis. Thus, it can be concluded that the revised Tave for VCSNS will not have an adverse effect on the small break LOCA results.

Mass and Energy Releases (Attachment 1, Section 6.2.4)

Evaluations were performed for VCSNS for reduced temperature and TDF. These evaluations were performed to support the transition to VANTAGE 5 fuel. Since VCSNS requested a return to a Tave of 587.4°F, a discussion of the effects on the long-term M/E releases follows.

For the long-term M/E release calculations, an evaluation was performed for the effects of a Tave of 587.4°F. This evaluation shows that the results of the current FSAR analysis would remain valid and unaffected. The FSAR containment integrity analysis would still be applicable since the M/E releases remain unchanged.

Steam Generator Tube Rupture - (Attachment 1, Section 6.2.5)

For the steam generator tube rupture (SGTR) event, an analysis was performed to evaluate the radiological consequences of this accident. The major factors that affect the radiological doses of an SGTR event are the amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube after reactor trip and the steam released from the ruptured steam generator to the atmosphere. The impact on these factors of operating with nominal Tave of 587.4°F, 15% steam generator tube plugging, and VANTAGE 5 fuel has been determined.

A sensitivity analysis was completed to assess the effect of these parameters using the original VCSNS SGTR design basis analysis assumptions. The results of the sensitivity analysis which incorporated a nominal Tave of 587.4°F, 15% steam generator tube plugging and VANTAGE 5 fuel parameters indicate that the primary to secondary break flow will increase by less than 1% while the steam released via the ruptured steam generator will decrease by 4.9%. In addition, the offsite radiation doses considering the net effect of the above parameters have been calculated to be less than that reported for the VCSNS SGTR analysis. Therefore, the SGTR radiological consequences reported in the VCSNS FSAR are valid for this scenario and remain well within the 10CFR100 limits.

LOCA Hydraulic Forces (Attachment 1, Section 6.2.6)

The LOCA hydraulic forcing functions are considered as part of Attachment 1, Section 6.2.6 of the VCSNS licensing submittal. For LOCA hydraulic forces considerations, the RCS fluid temperatures are used as analysis input. Two of the major factors in determining the magnitude of the calculated LOCA hydraulic forcing functions from a postulated LOCA is the RCS temperature and pressure. It has been determined that increasing the Tave from 585.5°F to 587.4°F decreases the magnitude of the peak LOCA hydraulic forcing functions. Since, Tave is being increased from 585.5°F to 587.4°F no design or regulatory limits will be violated with respect to the LOCA hydraulic forcing functions.

Post-LOCA Long-Term Core Cooling (Attachment 1, Section 6.2.7)

Post-LOCA long-term core cooling is discussed in Attachment 1, Section 6.2.7 of the VCSNS licensing submittal. The Westinghouse licensing position for satisfying the requirements of 10CFR50.46(b)(5) "Long Term Cooling" concludes that the reactor will remain shutdown by borated emergency core cooling system (ECCS) water residing in the RCS/containment sump after a LOCA. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the refueling water storage tank (RWST) must have a boron concentration that, when mixed with other water sources, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the containment sump post-LOCA. It has been determined that increasing Tave from 585.5°F to 587.4°F will result in small beneficial increases in the RCS/containment sump boron concentration. On this basis, an evaluation of the post-LOCA long-term core cooling calculation has been demonstrated that there will be no decrease in the calculated RCS/containment sump boron concentration after a postulated LOCA for VCSNS with no violations of design or regulatory limits.

Hot Leg Switchover To Prevent Boron Precipitation (Attachment 1, Section 6.2.7)

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, and the RCS, RWST and accumulator water volumes and boron concentrations. Since the revised Tave for VCSNS will not affect the maximum boron concentrations or volumes associated with the RCS, RWST and accumulators, there is no effect on the post-LOCA hot leg switchover time.

5.0 SUMMARY AND CONCLUSION

Revisions to the VANTAGE 5 baseline parameters to reflect an increase in nominal Tave to 587.4°F and in TDF to 283500 gpm have been shown by engineering evaluation and additional safety analysis to have a minor impact on the results of SCE&G's licensing submittal supporting the transition to VANTAGE 5 fuel. All conclusions in the licensing submittal, including those in the Significant Hazards Evaluation, remain applicable for the proposed changes in baseline parameters and resulting Technical Specifications changes. Thus, it is concluded that VCSNS can be safety operated at 100% full power while maintaining the current design Tave of 587.4°F during the transition through mixed fuel cores to an all VANTAGE 5 core.

Enclosure 1 contains change pages which reflect the impact of the baseline parameter changes on Attachment 1, 2, 5, 6, and 7. Changes to Attachment 3 and 4 are not deemed necessary because of minor impact on the calculated results of the accident analysis.