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O. W. DIXON, JR.
VICE PRESIDENT
NUCLEAR OPERATIONS

March 6, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Reactor Coolant System Flow

Dear Mr. Denton:

South Carolina Electric and Gas Company (SCE&G) hereby requests a revision to the Virgil C. Summer Nuclear Station Technical Specifications to reduce thermal design flow by 1.9%.

At SCE&G's request, Westinghouse performed an analysis to justify 1.9% reduction in thermal design flow. Using the BART code (approved by the NRC in a letter from C. O. Thomas to E. P. Rahe, Jr., dated December 21, 1983) the LOCA events were reanalyzed. Also, non-LOCA events were evaluated using methods described in the current Final Safety Analysis Report (FSAR). The results indicate the acceptability of the reduced thermal design flow.

Attachment I contains the proposed revised pages to the Technical Specifications and a safety evaluation which concludes that the proposed changes do not involve a significant hazard. Also included in Attachment II is information that will be incorporated into the FSAR sections 4.4 and 15.4.1.

These changes have been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee. Please find enclosed the application fee of one hundred fifty dollars ($150.00) required by Title 10 of the Code of Federal Regulations, Part 170.

Should you have any questions, please contact us at your convenience.

Very truly yours,

O. W. Dixon, Jr.

JAW/OWD/gj
Attachment:

cc: (See Page #2)

A001 w/check $150.00
1/40 #208305

cc: V. C. Summer
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VIRGIL C. SUMMER REDUCED RCS FLOW SAFETY EVALUATION

For a 1.9% reduction in Thermal Design Flow (TDF) and no change in flow measurement uncertainty (3.5%) the following evaluation is provided.

DNB Considerations

Thermal-hydraulic sensitivity studies indicate that 1.9 percent flow reduction results in a 3.0 percent DNBR penalty. In addition, the Rod Bow Penalty will be reduced to 2.3%, consistent with the revised Westinghouse rod bow position described in WCAP-8691, Revision 1. Based on the existing safety analysis for the Virgil C. Summer Nuclear Station, a generic DNBR margin of 9.1% is available. In order to maintain the validity of all previous DNB evaluations, 5.3% of the generic DNBR margin will be used to offset the rod bow and reduced flow penalties. Thus, the core DNB limits are unchanged, and the conclusion that the DNB design basis is met for the following FSAR transients remains valid:

- Excessive Heat Removal Due to Feedwater System Malfunction
- Excessive Load Increase
- Main Steamline Depressurization
- Main Steamline Rupture
- Loss of Load/Turbine Trip
- Partial Loss of Forced Reactor Coolant Flow
- Complete Loss of Forced Reactor Coolant Flow
- Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Bank Withdrawal at Power
- Startup of an Inactive Reactor Coolant Loop
- Inadvertent ECCS Operation at Power
- Reactor Coolant System Depressurization

Non-DNB Considerations

In addition to the DNB concern, the following evaluations are presented for those accidents which are not DNB related or for which DNBR is not the only safety criterion to be met.

Control Rod Withdrawal From a Subcritical Condition

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 15.2.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Thus, nuclear power response is primarily a function of the Doppler temperature coefficient. An increase in temperature due to reduced RCS flow would result in more Doppler feedback, thus reducing the nuclear power excursion as presented in the FSAR which partially compensates for the flow reduction.

The FSAR analysis shows that for a reactivity insertion rate of 105×10^{-5} K/sec, the peak heat flux achieved is approximately 91 percent of nominal with a resultant peak fuel average temperature of 820°F. A 1.9 percent reduction of reactor coolant flow would degrade heat transfer from the fuel by a maximum of 1.9 percent. Thus, peak fuel and clad temperatures would also increase by a maximum 1.9 percent, yielding maximum fuel and clad temperatures which are still significantly below fuel melt (4800°F) and zirconium-H_2O reaction (1800°F) limits. Therefore, the conclusions presented in the FSAR are still valid.

Boron Dilution

The results of the boron dilution transient will remain unchanged for all modes of operation due to a reduction in reactor coolant flow. The maximum dilution flow rate, RCS active volumes, and RCS boron concentrations are not impacted by a reduction in flow. Since these parameters determine the amount of time available to the operator to terminate the dilution event, the results presented in the FSAR remain unchanged.

Loss of Load

The loss of load accident is presented in section 15.2.7 of the FSAR and can result from either loss of external electrical load or a turbine trip. A loss of load results in core power which exceeds the secondary system power extraction, thus, causing an increase in core water temperature. A reduction in RCS flow will result in a more rapid pressure rise than that shown in FSAR. The effect will be minor, however, since the reactor is tripped on high pressurizer pressure. Thus, the time to trip will be decreased, which will result in a lower total energy input to the coolant. The analysis shows a peak pressurizer pressure of 2550 psia. A 1.9 percent reduction in flow will lead to a conservative increase in system pressure of 2570 psia. The pressurizer will not fill, and the maximum pressures are within the design limits. Therefore, operation at reduced flow will not violate safety limits following a loss of load accident.

Loss of Normal Feedwater/Station Blackout

This transient is analyzed to demonstrate that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that the pressurizer must not be filled with water. The effect of reducing initial core flow would result in an initial more rapid heatup of the RCS. The resultant coolant density decrease would increase the volume of water in the pressurizer. However, considerable margin to filling the pressurizer is available during the initial portion of the transient. During the long-term portion of the transient, the peak RCS temperature (and resultant peak pressurizer water volume) is reached when the heat removal capability of the auxiliary feedwater matches the core decay heat generation. If the RCS flow shortfall is due to higher than anticipated loop resistances, the natural circulation flow would be reduced by an amount proportional to the thermal design flow reduction. The slight reduction in natural circulation at the peak temperature condition would not significantly impair the heat transfer across the steam generator tubes, thus resulting in a similar peak pressurizer water volume. Therefore, the FSAR conclusions remain valid.

Steamline Break

The steamline break transient is analyzed at hot zero power, end-of-life conditions for the following cases:

- Inadvertent opening of a steam dump, safety, or relief valve (Section 15.2.13 of the FSAR)

- Main Steam pipe rupture with and without offsite power available (Section 15.4.2 of the FSAR)

A steamline break results in a rapid depressurization of the steam generators and primary side cooldown. This causes a large reactivity insertion due to the presence of a negative moderator temperature coefficient. A reduction in reactor coolant flow will result in a reduction in heat transfer from the fuel to the coolant. Therefore, the reactivity insertion and return to power in the double-ended rupture case for reduced flow conditions would be less limiting than the cases presented in the FSAR. For the double-ended rupture case, the time of safety injection actuation is unaffected by reduced coolant flow. This, coupled with a slower return to power would result in a significant reduction in peak average power from the FSAR results. The main steam depressurization case is bounded by the double-ended rupture. Since the return to power is less severe and the DNB evaluations remain valid as previously stated, the conclusions presented in the FSAR are still valid for a 1.9 percent reduction in reactor coolant flow.

Rupture of a Main Feedwater Line

This transient is analyzed to demonstrate that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that bulk voiding does not occur at the outlet of the core. The effect of reducing initial core flow would result in an initial more rapid heatup of the reactor coolant system (RCS). However, a considerable margin to hot leg saturation exists during the initial portion of the transient.

For the case analyzed with offsite power available, the slight reduction in RCS flow would not significantly degrade the heat transfer across the steam generator tubes, hence resulting in a similar long term RCS heatup. If the RCS flow shortfall is due to higher than the anticipated loop resistances, the natural circulation flow would be reduced by an amount proportional to the thermal design flow reduction for the case analyzed without offsite power available. The slight reduction in natural circulation flow would not significantly degrade the heat transfer from primary to secondary, thus resulting in a similar peak hot leg temperature at the time at which the heat removal capability of the auxiliary feedwater matches the core generated decay heat.

Therefore, the FSAR conclusions remain valid.

Locked Rotor

The FSAR (Section 15.4.4) shows that the most severe locked rotor is an instantaneous seizure of a reactor coolant pump rotor at 100 percent power with three loops operating. Following the incident, reactor coolant system temperature rises until shortly after reactor trip. A reduction in RCS flow will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. The flow reduction in the affected loop is so rapid that the time of reactor trip, on low flow, does not change due to the 1.9 percent reduction in reactor coolant flow. Therefore, the nuclear power and heat flux transients will not change from those presented in the FSAR. However, the reduction in flow will result in slightly higher system pressures and clad temperatures. The peak RCS pressure reported in the FSAR was 2726 psia. A 1.9 percent reduction in reactor coolant flow would cause a conservative increase in pressure of 11 psia to 2737 psia, which is still below the pressure at which vessel stress limits are exceeded. The peak clad temperature reported in the FSAR is 1960°F, well below the limit of 2700°F (the temperature at which clad embrittlement may be expected), and shows that a slight increase in this parameter due to reduced RCS flow can be easily accommodated. Therefore, the FSAR conclusions are still valid.

Control Rod Ejection

The rod ejection transient is analyzed at full power and hot standby for both beginning and end-of-life conditions (Section 15.4.6 of the FSAR). A reduction in core flow will result in a reduction in heat transfer to the coolant, which will increase peak clad and fuel temperatures and peak fuel stored energy. However, all cases have margin to fuel failure limits. The effect of reducing the reactor coolant flow is to increase the peak clad temperatures. The analysis shows that, for the worst case, there is sufficient conservatism in the analysis assumptions and margin in the results such that the peak clad temperature limit (2700°F) is not violated with the reduced flow. The fuel temperatures and peak fuel stored energy will also increase slightly due to the 1.9 percent decrease in reactor coolant flow. However, there is sufficient margin between the analysis results and the limits to accommodate the effects of the reduced flow. Therefore, the conclusions presented in the FSAR are still valid.

Reduction in Power for Reduced Flow

The no significant hazards analysis for the Amendment No. 37 addition of Region III to Technical Specifiction Figure 3.2-3 was reviewed, and the basis for a no significant hazards consideration remains the same. A 2% power reduction for every 1% flow reduction ensures a sufficient margin of safety remains.

ANALYSIS OF SIGNFICANT HAZARDS CONSIDERATION

The proposed reduction in thermal design flow will not involve a significant increase in the probability or consequences of an accident previously evaluated because the evaluation of those accidents which are DNB related indicates that the conclusion that the DNB design basis is met for FSAR transients is still valid and the core DNB limits are unchanged. The evaluation of those accidents which are not DNB related or for which DNBR is not the only safety criterion to be met indicate that the FSAR conclusions remain valid.

The propoced reduction in thermal design flow will not create the possibility of a new or different kind of accident from any previously evaluated because the plant systems and design remain the same.

The proposed reduction in thermal design flow will not involve a significant reduction in margin of safety because the generic DNBR margin is used to offset the rod bow and reduced flow penalties and the core DNB limits are unchanged.

Therefore, it is concluded that the proposed revision does not constitute a significant hazards consideration.