TENNESSEE VALLEY AUTHORITY

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JUN 18 1990

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket No. 50-327
Tennessee Valley Authority	)	

SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 1 CALORIMETIC

The purpose of this letter is to inform NRC of an issue recently identified during startup of SQN Unit 1 from the Cycle 4 refueling outage and how it was addressed to support continued escalation to 100 percent power. This information has been previously discussed with NRC in continuing communication with the onsite senior resident inspector and in telephone conference calls held between TVA and NRC staff on June 13 and 14, 1990. During the performance of the startup secondary and primary calorimetric, an unexplained increase in core delta T was discovered. This increased and anomalous delta T measurement resulted in a reactor coolant system (RCS) flowrate calculation less than the required technical specification (TS) value. Preliminary assessment of implemented changes to plant equipment, test data (recent and historical), and core parameters provided high confidence that the RCS flowrate had not actually degraded. However, power escalation was temporarily suspended pending confirmation of the condition and cause. Status of this issue was communicated to the senior resident inspector, and ongoing communication continued throughout the issue investigation and resolution process. To ensure that no safety concerns existed during resolution of the issue, TVA requested Westinghouse Electric Corporation to evaluate the worst case scenario of an actual reduction in RCS flowrate. The resultant justification for continued operation verified acceptability of operation at 100 percent power, and a copy is enclosed for reference.

As a result of numerous changes, which had been implemented to both primary and secondary equipment, an in-depth investigation was initiated to determine the cause of the anomalous indications. After extensive compilation and evaluation of data, TVA concluded that RCS flowrate was in fact greater than the TS value and that the earlier calculated low value had resulted from errors in indicated RCS hot leg temperature (T_H). The following summarizes the basis for this determination.

RCS flowrate, M_{RCS}, is normally calculated (inferred) from the following equation:

$$(M_{RCS}\Delta h)_{primary} = (M_{FW}\Delta h)_{secondary}$$

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The primary enthalpy change is derived from measured core delta T, i.e., T_H minus cold leg temperature. A secondary plant calorimetric is performed to establish the right side of the equation, RCS hot and cold leg temperatures are measured, and the M_{RCS} is then calculated.

As previously mentioned, a variety of plant changes had been implemented during the Cycle 4 refueling outage, a number of which were considered to have potential for impacting the primary or secondary data. Feedwater venturi's were cleaned, tested, and new calibration curves were provided; Eagle 21 protection sets were installed; RCS narrow range resistance temperature detector (RTD) bypass manifold elimination was implemented, replacing the previous manifold with thermowells and fast acting RTDs; Vantage 5H fuel was loaded to enhance fuel economy and reliability. Other key modifications such as upper head injection removal and boron injection tank deactivation were not considered to affect calorimetric data.

The following actions were taken to verify secondary plant data validity. A precision feedwater calorimetric was performed and verified with results utilizing condensate flows; calculated secondary plant power was verified consistent with both electrical output and turbine impulse pressure; the feedwater venturi calibration was checked and transmitter output verified to be consistent with raw differential pressure data; power output and data was reviewed against previous cycle data for consistency. In evaluation of primary side data, T_H RTD leads were lifted upstream of Eagle 21 processing to verify consistent input and output; power distribution was reviewed against previous cores and incore thermocouple (TC) maps; RCS elbow tap pressure drop data was reviewed against previous data; core exit TCs were compared to T_H; and RCS parameters were compared to design data and operating data for plants of similar configuration.

Completion of these reviews confirmed the validity of secondary plant data and that errors in core RCS temperature measurement were not being introduced by Eagle 21 implementation. The review also confirmed that RCS flowrate had not changed since initial startup as indicated by consistent elbow tap pressure drop data. The review did determine that RCS delta T had increased from previous operation values and T_H had increased from expected values as compared to core exit thermocouple data without apparent cause. Review of the previous equation shows that this indicated temperature increase thereby results in a corresponding lower calculated RCS flowrate; in fact, a significant reduction in calculated flowrate for a small increase in indicated delta T. Close review of this situation by Westinghouse's thermal hydraulics specialist confirmed TVA's previous data evaluation results and concluded that T_H was indicating erroneously high because of changes in hot leg flow streaming resulting from indicated changes in core exit temperature distribution. A similar condition had been previously observed at several other sites, although to a lesser extent. While still under evaluation by Westinghouse, it is considered to have resulted from depression of the radial power distribution at the periphery of the core. This type of profile causes colder water streaming along the bottom of the hot leg pipe. Some of this colder flow is not included in the average temperature measured by the RTDs, resulting in the erroneously high T_H indication. (T_H utilizes RTDs in three thermowells-scoops located at 0 degrees, 120 degrees, and 240 degrees from the top center of the loop piping.) Review of the SQN Unit 1 radial power profile substantiates the potential for this phenomenon. A variety of options are under consideration by Westinghouse to address this situation for affected plants. In the interim, the impact of higher indicated T_H on protection and control functions was evaluated and determined not to represent a safety issue or to adversely affect SQN analyses.

Discussions were held with the NRC staff on June 13 and 14, 1990, to provide the staff with information concerning this issue and respond to any questions that the staff may have had with regard to the present status or TVA's plans for increasing power on SQN Unit 1 to 100 percent. During the discussions, TVA provided a detailed description of the issue, investigation efforts, and resolution status. Also included in these discussions was a description of the effect of operating with an increased indicated T_H on the reactor protection set setpoints and control circuits that use RCS average temperature (Tavg) and delta T as inputs. It was concluded that all effects were in the conservative direction, and no safety concerns would be introduced by high indication.

At the conclusion of the June 14, 1990, telephone call, NRC indicated agreement with TVA's approach in resolving this issue. The staff found that power escalation to 100 percent using the secondary side calorimetric program performed by the plant process computer is acceptable. The staff did, however, express continued interest in TVA's long-term resolution. It is recognized that uncorrected, the higher T_H will result in adverse operational effects, e.g., reduced margin between 100 percent power and runback/trip setpoints and depressed actual Tavg and steam pressure. Accordingly, a number of options are being evaluated for both short and long-term resolution. In the short term, TVA rescaled indicated delta T to slightly above 100 percent power when actual power is verified by the secondary side calorimetric, to be at 100 percent. A process has been implemented to monitor delta T for further changes so that appropriate scaling changes are implemented as the streaming phenomenon is expected to dampen over core burnup. Rescaling to slightly above 100 percent provides acceptable margin to runback/trip setpoints while additionally providing margin for potential decreases in delta T prior to rescaling. Long-term actions being pursued include development of scaling correction factors based on core exit thermocouples, reprogramming of Tavg control systems, and possible testing to validate and further define the observed streaming phenomenon. Actions are being coordinated with Westinghouse and will be carefully evaluated for full assessment on safe plant operation. Preliminary results of ongoing evaluations are expected to be available by mid July. Condition Adverse to Quality Report SQP 900286 documents this issue and will be used to track short and long term corrective actions. TVA will continue to keep the senior NRC resident inspector briefed on both related changes in plant status and long term issue resolution developments.

In summary, TVA has determined both by calculation of RCS flowrate utilizing RCS elbow taps and by comprehensive review of data that the RCS TS flowrate has been satisfied. TVA has further determined that RCS T_H is indicating higher than actual bulk T_H because of fluid streaming, and that this condition does not compromise safe full power operation. Westinghouse has reviewed associated data and has concurred with these determinations. A variety of options are being evaluated by Westinghouse and TVA for long-term resolution of associated issues.

If you have any questions concerning this submittal, please telephone M. A. Cooper at (615) 843-6651.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Mark O. Medford

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Ref: TVA Contract 89NNP-75380A

Tennessee Valley Authority
Sequoyah Nuclear Plant Unit 1
Reduced RCS Flow
Justification for Continued Operations

Dear Mr. Trudel:

In response to your request, attached is a justification for continued operations of Unit 1 at a reduced RCS flow. This justification shows that a more rigorous safety evaluation would support a no significant hazards consideration pursuant to 10CFR 50.92 criteria.

If you have any comments or questions, please contact the undersigned.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

for B. J. Garry, Manager
TVA Sequoyah Project
Customer Projects Department

LVT/ee

cc: D. M. Lafever
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Tennessee Valley Authority

Sequoyah Unit 1
Reduced RCS Flow

Justification For Continued Operation

SUMMARY

This justification for continued operation of Unit 1, Cycle 5, at a reduced flow of 369,000 gpm, less 3.5 % Technical Specification uncertainties, and at 100 % power, shows that a more rigorous evaluation would support a no significant hazards consideration pursuant to 10 CFR 50.92 criteria. The analysis flow value addressed, after the uncertainty adjustment, is 356,000 gpm.

This document addresses the FSAR Chapters 6 and 15 accident analyses, which include LOCA, non-LOCA, SGTR, and Containment Integrity, and, the NSSS System components design transients. This justification is based upon the consideration that the Sequoyah Unit 1 licensing basis includes the analyses/evaluations performed to support the Cycle 5 modifications. These modifications include V5H Fuel, RTDE/Eagle/NSLB/MSS/EAM/TTD, UHI Removal and RWST Boron Concentration Increase.

JUSTIFICATION

NSSS System And Equipment

The potential impact of operation with the evaluated RCS coolant flow on reactor coolant system components was addressed. A review of the thermal design parameters for the evaluated flow condition at 5% steam generator tube plugging, indicate that the primary system temperatures remain essentially unchanged (less than one degree change from the previous 5% plugging thermal design parameters values). Thus, the original analyses of the NSSS systems and equipment influenced by primary NSSS parameters are not expected to be affected by the assumed flow condition.

For the secondary side of the steam generator, the steam pressure and temperature also show very little change for the reduced flow condition, and thus the conclusions of the structural analyses would be expected to remain unaffected. Specifically for the U-bend tube fatigue evaluation, the analysis was based on a conservative set of operating conditions which envelops the Sequoyah operating conditions by a large margin.

Even with a small reduction in steam pressure and temperature as indicated by the thermal design parameters computations, the current U-bend fatigue analysis would still bound the steam conditions coincident with the assumed reduction in primary system flow.

LOCA Accidents

Alteration of the design basis reactor coolant system (RCS) flow rate does not affect the following Loss of Coolant Accident (LOCA) related analyses: hot leg switchover to preclude boron precipitation, the post-LOCA long term core cooling subcriticality, and post-LOCA long term core cooling minimum safety injection flow. Steady-state RCS flow rates are not germane to these long term requirements. Further, the small break LOCA analysis performed with the NOTRUMP Evaluation Model in support of removing upper head injection from service and Cycle 5 operation has assumed an RCS flow rate of 355200 gpm. The small break LOCA RCS flow rate therefore remains bounding, even at the evaluated low flow condition. The above LOCA related accidents are not adversely affected at 100% power by the evaluated low flow situation, and they are conservative for lower power operation. Moreover, a reduction in the steady-state flow rates is judged to have no significant effect upon the reactor vessel and loop LOCA blowdown forcing functions.

The large break LOCA 10CFR50.46 analysis for Sequoyah 1 has also been considered. To support upper head injection removal and Cycle 5 operation, the large break LOCA limiting case was analyzed using the BASH Evaluation Model at 10% steam generator tube plugging and a RCS flow rate of 362000 gpm. At the present time the fresh fuel loaded into the core for Cycle 5 has achieved little burnup and possesses very little decay heat relative to the 100% power basis of the large break LOCA analysis. While the analysis of the freshly loaded fuel is therefore very conservative for the current situation, previous analysis of Sequoyah Unit 1 Cycle 5 has demonstrated that once-burned fuel assemblies being reinserted into the core are more limiting in calculated peak cladding temperature (PCT) than the fresh assemblies.

The calculated PCT for these limiting reloaded fuel assemblies is 2013°F; by considering the expected actual core peaking factors, beneficial reductions in both hot assembly and core average peaking and in initial fuel pellet temperature are achieved. Conversely, the reduced steady-state flow rate will exact a penalty in PCT on the existing analysis result. A previous sensitivity study performed for a plant similar in vessel internals to Sequoyah identified that a 1% decrease in flow causes a 17°F increase in calculated PCT for a UHI imperfect mixing case.

Although Sequoyah now operates with UHI removed, the UHI imperfect mixing transient is similar enough to a non-UHI case to apply this sensitivity for JCO purposes. The 356000 gpm flow rate represents a reduction of 1.66% from the analysis value, and over a limited range it is judged to be reasonable to extrapolate the pertinent sensitivity linearly.

The assessed penalty in PCT therefore is 26°F, giving a new net PCT value of 2041°F for the Sequoyah Unit 1 Cycle 5 large break LOCA limiting case. Since no credit has been taken for the identified beneficial aspects of the real-life core peaking factors, 2041°F is a suitably conservative value for the large break LOCA PCT at the evaluated flow, and substantial margin exists to the 2200°F regulatory limit. As concluded above for the other LOCA-related accidents, operation of Sequoyah Unit 1 with an accident analysis basis 356000 gpm RCS flow rate during Cycle 5 is judged to be acceptable.

Containment Related Analyses

Short Term Subcompartment Analyses

The short term subcompartment analysis was performed at 102% power with a thermal design flow of 365,000 GPM. This is a 2.5% decrease in RCS flow. The analyses would not be impacted by this flow change because a 2.5% reduction in RCS flow would have no significant effect on the initial system temperatures, so the initial system energy would remain unchanged. The short term mass and energy releases occur for a small blowdown period (i.e. 1 - 3 sec.). Since the initial system energy would remain the same, the mass and energy releases would remain unchanged. Therefore, the peak calculated differential pressures would remain unchanged from the current design basis subcompartment analyses for a 2.5% decrease in RCS flow rate.

Long Term Containment Analyses

LOCA Containment Integrity Analysis

The most recent analysis that was performed for long term LOCA containment integrity (CIA) was for a reduction in ice weight (Reference 2). This analysis was also performed at 102% power with a thermal design flow of 365,000 GPM. For the long term design basis LOCA containment analysis, a flow reduction of 2.5% is an insignificant change. There would be no effect on the initial average system temperature, so there would be no change to the initial system stored energy. The mass and energy releases for the long term LOCA transient have four (4) distinct phases; Blowdown, Refill, Reflood, and Post-Reflood (Froth). The initial system conditions affect the blowdown phase. Once the blowdown has been completed, the initial RCS conditions do not control the last three phases of the transient. Since the initial conditions would be the same, the mass and energy releases from the blowdown phase of the LOCA transient would remain the same as the current design basis analysis. Therefore, the peak calculated containment pressure and temperature for the LOCA analysis would remain unchanged for the current design basis analysis.

Main Steamline Break (MSLB) Containment Analysis

The long term MSLB containment analysis yields containment pressure and temperature profiles that are used to evaluate equipment qualification (EQ) for Sequoyah Units 1 & 2.

Steamline Break Outside Containment

An analysis for steamline break outside containment was recently performed as part of the Eagle-21 program. The results of this analysis were used for EQ purposes in the steamline valve vault and the auxiliary building.

SGTR Accident

The Steam Generator Tube Rupture (SGTR) accident is analyzed at a flow of 354,000 GPM. Therefore, the current SGTR FSAR analysis bounds operation at a flow of 356,000 GPM.

NON-LOCA Accidents

DNB Considerations

In order to determine the effects of the evaluated flow condition on the DNB-related transients, the core thermal limits and subsequent ΔT setpoint calculations were examined. To accomodate the 2.5% decrease in the thermal design flow assumption used in the thermal hydraulic design of the fuel and the non-LOCA safety analyses, sensitivity studies as well as a Sequoyah specific evaluation have shown that 3.9% DNBR margin must be allocated to offset the Unit 1 Cycle 5 evaluated flow condition. Allocation of this margin ensures that the DNB segments of the core thermal limits will not change.

However, the vessel exit boiling limit segments of the core thermal limits do change. These core limits are used in the calculation of the Overtemperature and Overpower ΔT setpoint equation coefficients. The Overtemperature and Overpower ΔT setpoints protect against DNB and fuel centerline melting, respectively, for pressures as low as the Low Pressurizer Pressure reactor trip. The change to the vessel exit boiling segments of the core limits impacts the Overtemperature setpoint equation such that the coefficients used in the safety analyses do not remain valid at the lower flow condition. However, if the safety analysis limit for the Low Pressurizer Pressure reactor trip is increased, then the Overtemperature ΔT technical specification equation as well as the safety analysis equation will remain valid.

Sufficient margin exists between the current Technical Specification setpoint for the Low Pressurizer Pressure reactor trip and the safety analysis limit such that the vessel exit boiling segments are adequately protected and the Technical Specification setpoint does not need to be changed.

The flow condition also results in portions of the core thermal limits not being protected by the Overpower setpoint equation used in the safety analyses. Therefore, the safety analysis limit value for the K4 coefficient used in the Overpower ΔT equation was also reduced.

As with the Low Pressurizer Pressure reactor trip safety analysis setpoint change, this change is covered by existing margin (i.e., the magnitude of this change is small enough that the Technical Specification value of K4 would not be impacted).

Considering the previous discussion, it can be concluded that the DNB design basis is met for the following FSAR Chapter 15 non-LOCA safety analyses:

- Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Bank Withdrawal at Power
- RCCA Misalignment
- Partial Loss of Flow
- Startup of an Inactive Reactor Coolant Loop
- Loss of External Electrical Load/Turbine Trip
- Excessive Heat Removal Due to Feedwater System Malfunction
- Excessive Load Increase
- Accidental Depressurization of the Reactor Coolant System
- Accidental Depressurization of the Main Steam System
- Inadvertent Operation of the Safety Injection System at Power
- Complete Loss of Forced Reactor Coolant Flow
- Single RCCA Withdrawal at Power
- Main Steamline Rupture
- Locked Rotor (Rods-In-DNB)
- Steamline Break with Coincident Rod Withdrawal at Power

Non-DNB Considerations

In addition to the DNB concerns discussed earlier, the following evaluations are presented for those licensing basis events which are not DNB-related or for which DNB is not the only safety criterion to be met.

Uncontrolled Rod Withdrawal From a Subcritical Condition

An uncontrolled rod withdrawal from subcritical event results in a rapid 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 fuel (Doppler) temperature coefficient. The power excursion also causes a heatup of the moderator/coolant.

However, since the power rise is extremely rapid and short lived and reactor trip quickly terminates any additional power generation, the thermal lag of the fuel pellet limits the moderator temperature rise to a small value after reactor trip has occurred. Thus, the nuclear power response is essentially a function of the Doppler feedback.

A 2.5% reduction in RCS flow would result in a slight increase in the calculated moderator temperature rise.

However, a slightly higher moderator temperature will result in slightly more Doppler feedback due to hardening of the neutron spectrum, thereby reducing the power excursion from that calculated in the FSAR.

The FSAR analysis shows that for a reactivity insertion rate of 87 pcm/sec, the hot spot peak fuel average and clad average temperatures are conservatively calculated to be 1818 ^{o}F and 665 ^{o}F, respectively. A 2.5% flow reduction would degrade the fuel to coolant heat transfer by at most 2.5%. This in turn would result in no more than a 2.5% increase in the calculated fuel and clad temperatures when compared to the FSAR. This would yield hot spot peak fuel average and clad average temperatures which are still well below fuel melt and Zirc-H_2O reaction limits.

Note that in addition to the impact on the fuel/clad temperatures, an uncontrolled rod withdrawal from subcritical will result in pressurization of the RCS due to the primary to secondary power mismatch. However, the pressurization which results from this event is bounded by the pressurization experienced during the loss of load event discussed later.

Therefore, a 2.5% reduction in RCS flow would not result in the violation of the fuel/clad temperature or peak RCS pressure criterion for the control rod withdrawal from subcritical event.

Boron Dilution

The results of the boron dilution analysis would 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 2.5% reduction in RCS flow. Since these parameters determine the amount of time available to the operator to terminate the dilution event, the results presented in the FSAR will remain unchanged.

Therefore, a 2.5% reduction in RCS flow would not result in the violation of the licensing basis criteria following a boron dilution event.

Loss of Load

The loss of load event presented in Section 15.2.7 of the FSAR may result from either a loss of external electrical load or a turbine trip. The result of a loss of load is a rapid decrease in the secondary side heat removal, causing a rapid primary side heatup and pressurization. Four case are analyzed, beginning and end-of-life core physics characteristics, with and without pressure control. Of the four cases analyzed, one case (beginning-of-life, with pressure control) trips on low-low SG level, while the remaining three cases trip on high pressurizer pressure.

A reduction in the RCS flow will result in a more rapid primary side heatup and pressurization than that shown in the FSAR. However, the effect will be minor. For those cases which trip on high pressurizer pressure, the time to trip will be slightly reduced which will result in less total energy input to the RCS. For the case which tripped on low-low SG level, the reduction in RCS flow would not be expected to change the time at which the low-low SG level setpoint is reached. In all four cases there is substantial margin to the primary/secondary side pressure limits as well as significant margin to the minimum DNBR.

An additional concern during the loss of load event is ensuring that the pressurizer will not fill as pressurization of the RCS results in an insurge into the pressurizer. An RCS flow reduction leading to a more rapid pressurization may result in a greater pressurizer insurge. However, as shown in the FSAR there is approximately 400 cubic feet (~ 22% of total pressurizer volume) of margin to filling. Thus, there is more than sufficient margin to pressurizer filling to accommodate a 2.5% RCS

flow reduction. It is worth noting, that for the 3 cases which trip on high pressurizer pressure, a more rapid trip may actually reduce the surge into the pressurizer since there is less energy input to the RCS.

Thus, a 2.5% reduction in RCS flow would not result in the violation of the licensing basis criteria following a loss of load event.

Loss of Normal Feedwater/Loss of AC Power to the Station Auxiliaries

The loss of normal feedwater analysis in Section 15.2.8 of the FSAR presents the consequences of a complete loss of normal feedwater flow simultaneous to all four steam generators. The loss of AC power event is similar except that the loss of offsite power also results in all four reactor coolant pumps (RCPs) coasting down. These transients are analyzed to demonstrate that neither the primary or secondary sides are overpressurized, that the core is not adversely affected, and the pressurizer does not fill.

Following the loss of normal feedwater, the reactor continues to operate until, due to the rapid loss of steam generator inventory and the continued heat transfer to the secondary side, it is tripped on a low-low steam generator level signal. It is anticipated that a 2.5% reduction in RCS flow would have little or no impact on the time of trip on low-low steam generator level. The effect of reducing the RCS flow would be an increase in the heatup of the RCS during the inital phase of the transient. The increased heatup results in a decrease in the coolant density which in turn would increase the pressurizer insurge during this heatup. However, considerable margin exists to filling the pressurizer during this initial portion of the transient so that filling would not occur.

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 system matches the core decay heat generation. If the assumed RCS flow reduction is due to higher than anticipated loop flow resistances, the natural circulation flow will be reduced by an amount proportional to the 2.5% thermal design flow reduction. This slight reduction in natural circulation flow at the peak RCS temperature condition would not significantly impair the heat transfer across the steam generator tubes, thus, resulting in a similar hot leg temperature and peak pressurizer water volume.

Therefore, a 2.5% reduction in RCS flow would not result in the violation of the licensing basis criteria following a loss of normal feedwater or loss of AC power event. The same discussion and conclusions apply to the part-power loss of normal feedwater analyses completed for the Trip Time Delay.

Rupture of a Main Feedwater Line

The analysis in Section 15.4.2.2 of the FSAR presents the consequences of a double-ended-rupture of a main feedline at full power. Initially, the RCS is cooled as the faulted steam generator blows down removing heat from the corresponding RCS loop. However, after the faulted steam generator empties, the reduction in secondary side inventory results in inadequate heat removal from the primary which in turn, increases primary system temperatures and pressurizes the RCS. Two cases are examined: with and without offsite power available. The case with offsite power is more limiting since the operating RCPs increase the energy addition into the primary system.

The FSAR analysis demonstrates that sufficient auxiliary feedwater (AFW) is available to prevent overpressurization of the primary and secondary systems and to ensure that the core remains intact and in a coolable geometry. This latter criteria is assured by showing that bulk boiling does not occur in the RCS hot leg prior to AFW turnaround.

A 2.5% reduction in the RCS flow would result in a slightly more rapid heatup of the RCS following the initial steam generator blowdown. The lower RCS flow would also result in a slightly higher hot leg temperature. However, the FSAR analysis shows that there is considerable margin ($\geq 25^oF$) to hot leg saturation throughout the transient. A 2.5% flow reduction would not significantly degrade the heat transfer across the steam generator tubes; thus, the long term RCS heatup calculated in the FSAR will not be significantly impacted. Therefore, any increase in the hot leg outlet temperatures due to a 2.5% flow reduction is expected to be negligible with respect to the available margin.

Furthermore, the FSAR analysis demonstrates that there is a significant amount of margin to overpressurization of the primary and secondary systems. A 2.5% RCS flow reduction should have no impact on ability of the secondary side safety valves to relieve the pressure transient. On the primary side, the PORVs were conservatively modeled to minimize the margin to hot leg boiling. The analysis shows that the PORV capacity is more than sufficient to limit the RCS pressure transient. A 2.5% flow reduction would not compromise this ability. Thus, neither the primary or secondary systems would overpressurize.

Therefore, a 2.5% reduction in RCS flow will not result in the violation of the licensing basis criteria following a feedline break event. The same discussion and conclusions apply to the part-power feedline break analyses completed for the Trip Time Delay.

Locked Rotor

Section 15.4.4 of the FSAR presents the results of an instantaneous seizure of an RCP rotor at full power with four RCPs operating. Following the seizure of a rotor, flow in the affected loop rapidly falls and the RCS temperature rises. Reactor trip is promptly initiated on a low loop flow signal. Analyses are done to predict the peak RCS pressure as well as the maximum metal-to-water reaction and peak clad temperature.

Since the low flow setpoint is a fraction of initial loop flow, a 2.5% reduction in the RCS flow will not impact the time of trip, and thus, the nuclear power and heat flux transients are unchanged. However, the lower RCS flow will result in slightly higher system pressures than those calculated in the FSAR. The peak RCS pressure has been calculated to be 2603 psia. This value is well below the pressure at which the vessel stress limits are exceeded. A 2.5% flow reduction would not significantly reduce the available margin.

The peak clad temperature analysis performed for the locked rotor event calculates a value of 2026 °F. This analysis conservatively assumes that DNB occurs upon the initiation of the event. This assumption maximizes the calculated PCT and minimizes the impact of a flow reduction since fuel-to-coolant heat transfer is already substantially degraded. The calculated PCT of 2026 °F is well below the 2700 °F limit, and shows that a slight increase in the temperature due to a 2.5% RCS flow reduction can be accommodated. Thus, the peak clad temperature during a locked rotor event will not exceed 2700 °F due to a 2.5% reduction in the RCS flow. Therefore, a 2.5% reduction in RCS flow will not result in the violation of the licensing basis criteria following a locked rotor event.

Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

The RCCA ejection analysis is analyzed at four conditions: beginning and end-of-life core physics characteristics, at hot zero power and full power (see Section 15.4.6 of the FSAR). The analysis demonstrates that gross fuel damage will not occur, that the core will remain in a coolable geometry, and that the RCS will remain intact. In order to demonstrate that these criteria are met Westinghouse applies the following, more restrictive, criteria:

1) The average fuel pellet enthalpy at tha hot spot is less than 200 cal/gm (360 Btu/lbm).
2) Fuel melt at the hot spot is limited to less than the innermost 10% of the fuel pellet.
3) Peak RCS pressure is less than that which would cause stresses to exceed the Faulted Condition Stress Limits.

The rod ejection event is characterized by a rapid power excursion terminated by Doppler feedback. The reactor is tripped on high neutron flux (low setting for the zero power cases, high setting for the full power cases). A reduction in RCS flow will result in a reduction in the fuel rod-to-coolant heat transfer. This may result in an increase in the calculated fuel and clad temperatures as well as the fuel stored energy during an RCCA ejection.

As shown in the FSAR, the full power cases result in the highest fuel pellet temperatures and approach criteria 1 and 2 with the least amount of margin. Examination of these cases reveals that, due to the rapid power and fuel temperature rise coupled with the thermal lag in the fuel pellet itself, the time at which the maximum pellet enthalpy and fuel melt are calculated to occur is before any significant amount of heat has reached the coolant. Thus, a reduction in the fuel-to-coolant heat transfer due to a 2.5% flow reduction should not impact the maximum pellet enthalpy and fuel melt calculated in the FSAR. However, it may be noted that, should the time at which the peak fuel temperatures occur increase, it is expected that sufficient margin is available to accommodate a 2.5% flow reduction.

The analysis of the peak pressure transient for the RCCA ejection event is discussed in WCAP-7588, Rev. 1. A reduction in RCS flow could increase the primary side pressurization by reducing the primary-to-secondary side heat transfer. However, due to the rapid nature of this event it is anticipated that any secondary side heat removal will lag well behind the heat addition to the primary side. Thus, it is judged that a 2.5% flow reduction will have a minimal impact on the primary side peak pressure. However, in WCAP-7588, several cases are presented which calculate the peak RCS pressure.

The most detailed of these cases calculates a peak pressurizer pressure of 2600 psia. This is more than sufficient margin to the Faulted Condition Stress Limits to accommodate a 2.5% reduction in the RCS flow.

Therefore, a 2.5% reduction in RCS flow would not result in the violation of the licensing basis criteria following a RCCA ejection event.

Steamline Break Mass\Energy Release Inside Containment

Generic sensitivity studies have shown that four major factors influence the release of mass and energy following a steamline break:

1. Steam generator inventory
2. Protection system operation
3. State of the secondary fluid blowdown
4. Primary to secondary heat transfer

A 2.5% reduction in RCS flow would not affect the first two factors and would have an insignificant impact on the last two factors. A decrease in RCS flow would tend to reduce the primary to secondary heat transfer, thereby reducing the steam pressure and temperature during normal operation. Any reduction in the secondary side temperature and pressure would tend to minimize the mass and energy released during a steamline break event. As a result, a 2.5% reduction in RCS flow would not adversely affect the steamline break mass/energy releases provided in Chapter 6.2.1.3.11 of the Sequoyah FSAR.

Steamline Break Mass/Energy Release Outside Containment

In order to address NRC concerns over the effect of superheated steam release on the environmental qualification of equipment located outside containment, steamline break mass/energy releases for breaks outside containment were provided for Sequoyah in WCAP-10961. The impact of a 2.5% reduction in RCS flow on these mass/energy releases would be insignificant for the same reasons as cited in the previous section for steamline breaks inside containment. The movement of the break location from inside to outside containment does not invalidate any of the arguments made above.