4.1 Plant Simulation Model

The District utilizes the CESEC digital computer code, References 4-1 through 4-10, to provide the simulation of the Fort Calhoun Station nuclear steam supply system. The program calculates the plant response to non-LOCA initiating events for a wide range of operating conditions. The information presented in Reference 4-9 supercedes information provided in References 4-1 through 4-8. Additional information on the model is provided in Reference 4-10. The CESEC program, which numerically integrates one dimensional mass and energy conservation equations, assumes a node/flow-path network to model the NSSS. The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the steam generators and the reactor coolant pumps. The secondary system components include the secondary side of the steam generators, the main steam system, the feedwater system and the various steam control valves. In addition, the program models some of the control and plant protection systems.

The code self initializes for any given, but constant, set of reactor power level, reactor coolant flow rate and steam generator power sharing. During the transient calculations, the time rate of change in the system pressure and enthalpy are obtained from solution of the conservation equations. These derivatives are then numerically integrated in time under the assumption of thermal equilibrium to give the system pressure and nodal enthalpies. The fluid states recognized by the code are subcooled and saturated; superheating is allowed in the pressurizer. Fluid in the reactor coolant system is assumed to be

5.1 CEA Withdrawal (Continued)

5.1.1 Definition of the Event (Continued)

Any controlled or unplanned withdrawals of the CEA's results in a positive reactivity addition which causes the core power, core average heat flux and reactor coolant system temperature and pressure to rise and in turn decrease the DNB and Linear Heat Rate (LHR) margins. The pressure increase, if large enough, activates the pressurizer sprays which mitigate the pressure rise. In the presence of a positive Moderator Temperature Coefficient (MTC) of reactivity, the temperature increase results in an additional positive reactivity addition further decreasing the margin to the DNB and LHR limits.

Withdrawal of the CEA's causes the axial power distribution to shift to the top of the core. The associated increase in the axial peak is partially compensated by the corresponding decrease in the integrated radial peaking factor. The magnitude of the 3-D peak change depends primarily on the initial CEA configuration and axial power distribution.

The withdrawal of the CEA's causes the neutron flux as measured by the excore detectors to be decalibrated due to CEA motion, i.e., rod shadowing effects. This decalibration of excore detectors, however, is partially compensated by neutron attenuation rising from moderator density changes (i.e., temperature shadowing effects).

Table 5.5.4-1

KEY PARAMETERS ASSUMED IN THE LL/1SG EVENT

Parameter	Units	Value
Initial Core Power	MW_t	700 to 1500+
Initial Core Inlet Temperature	°F	Maximum allowed+ by Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Minimum allowed+ by Tech. Specs.
Moderator Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	[ ]
Fuel Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	
Core Average H_{gap}	BTU/hr-ft²-°F	Maximum value predicted during core life.
Initial Core Mass Flow Rate	$*10^6$lbm/hr.	Best estimate flow+
Scram Reactivity Worth	$\%\Delta\rho$	Minimum predicted during core life.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

5.5 Asymmetric Steam Generator Event (Continued)

5.5.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the analysis for the LL/1SG event are discussed in Section 7 of Reference 5-2.

The results for Fort Calhoun Station are expected to be similar. The criteria of 10 CFR 50.59 are satisfied if the required overpower margin calculated for the LL/1SG event is less than the overpower margin being maintained by the current Technical Specifications.

5.6 Excess Load Incident

5.6.1 Definition of Event

An excess load transient is defined as any rapid increase in the steam generator steam flow other than a steam line break. Such a rapid increase in steam flow results in a power mismatch between the reactor core and the steam generator load demand. In addition, there is a decrease in the reactor coolant temperature and pressure. Under these conditions the negative moderator temperature coefficient reactivity causes an increase in core power.

The rapid opening of the turbine admission valves or the steam dump bypass to the condenser causes an excess load event. Turbine valves are not sized to accommodate steam flow for powers much in excess of 1500 MWt. The steam dump valves and steam bypass valves to the condenser are sized to accommodate 33% and 5%, respectively, of the steam flow at 1500 MW. Therefore, the following load increase incidents are examined:

5.6 Excess Load Incident (Continued)

5.6.1 Definition of Event (Continued)

C. (Continued)

(540°F). The maximum error that can be introduced in the referenced temperature setting is limited to 17°F since a narrow range instrument is used for this purpose. Reducing the dump valve controller reference setting from 532° to 515° would result in a partial opening of the valves but as soon as the reactor coolant temperature dropped to 518°F the valves would again be completely closed.

D. Opening the dump and bypass valves at hot standby due to steam dump controller malfunction: The most severe incident at hot standby would occur in the event the steam dump valve controller yields an incorrect signal and causes the steam dump and bypass valves to open completely. This case is considered to be much less probable than case C above but represents the most limiting event under hot standby conditions.

The possible RPS trips that might be encountered during this event are:

1. Variable high power trip (VHPT).

2. TM/LP trip.

3. Low steam generator water level trip.

5.6 Excess Load Incident (Continued)

5.6.4 Key Parameters and Analysis Assumptions

As discussed in Section 5 of Reference 5-2, sensitivity studies performed by CE have demonstrated that the maximum calculated [ ] for the excess load event occurs for the [ ] at hot full power conditions. District sensitivity studies show similar results. Therefore, only the hot full power case is analyzed. The key parameters used in the analysis of the excess load event are given in Table 5.6.4-1. The remaining assumptions are the same as those discussed in Reference 5-2.

5.6.5 Analysis Method

The steps used for determining the [ ] value and calculating the largest [ ] for all excess load events which rely on the TM/LP trip for DNBR protection are given in Section 5 of Reference 5-2. The minimum transient DNBR value for excess load events protected by the Variable High Power Trip is calculated using the procedure discussed in the same Section.

The PLHR is calculated by obtaining the core average linear heat rate at time of peak core power and multiplying it by the appropriate peaking factors and associated uncertainties.

Table 5.6.4-1

KEY PARAMETERS ASSUMED IN THE EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Value
Initial Core Power	MW_t	1500+
Initial Core Inlet Temperature	°F At Power	Maximum allowed+ by Tech. Specs.
Initial Reactor Coolant System Pressure	psia	Minimum allowed+ by Tech. Specs.
Initial Core Mass Flow Rate	$*10^6$lbm/hr.	Minimum allowed+ by Tech. Specs.
CEA Drop Time	sec.	Maximum allowed by Tech. Specs.
Scram Reactivity Worth	$\%\Delta\rho$	Minimum predicted during core life.
Moderator Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	Negative values up to the most negative value allowed by Tech. Specs.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

5.6 Excess Load Incident (Continued)

5.6.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the excess load analysis are similar to those presented in Section 5 of Reference 5-2. The criteria of 10 CFR 50.59 are met if the [ ] is less than or equal to the value used in the current TM/LP trip equation.

5.6.7 Conservatism of Results

The following points demonstrate the conservatism of the overall results for the excess load event:

1. Field measurements demonstrate that the CEA magnetic clutch decay time is less than that assumed in the analysis.

2. The actual scram worths are higher than those in the analysis.

3. Where the most negative MTC is used, the value is more negative than that measured during plant operation.

4. The actual Doppler reactivity is more negative than assumed in the analysis.

5. [ ]

5.6 Excess Load Incident (Continued)

5.6.7 Conservatism of Results (Continued)

6. Field data demonstrates that the actual CEA drop time is less than that assumed in the analysis.

7. The conservatism of the [ ] is discussed in Section 5 of Reference 5-2.

5.7 RCS Depressurization

5.7.1 Definition of Event

The RCS depressurization event is characterized by a rapid decrease in the primary system pressure caused by either the inadvertent opening of both power operated relief valves (PORVs) or the inadvertent opening of a single primary safety valve operating at rated thermal power. Following the initiation of the event, steam is discharged from the pressurizer steam space to the quench tank where it is condensed and stored. To compensate for the decreasing pressure the water in the pressurizer flashes to steam and the proportional heaters increase the heat added to the water in the pressurizer in an attempt to maintain pressure. During this time the pressurizer level also begins to decrease causing the letdown control valves to close and additional charging pumps to start so as to maintain level. As pressure continues to drop, the backup heaters energize to further assist in maintaining primary pressure. A reactor trip is initiated by the TM/LP trip to prevent exceeding the DNBR SAFDL.

5.8 Main Steam Line Break Accident (Continued)

5.8.5 Analysis Method

The analysis of the main steam line break accident is performed using CESEC which models neutron kinetics with fuel and moderator temperature feedback, the reactor protective system, the reactor coolant system, the steam generators and the main steam and feedwater systems.

5.8.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the analysis for the Fort Calhoun steam line break event are discussed in Section 14.12 of the 1983 update of the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met if the calculated return-to-power is less than the return-to-power reported for the Cycle 1 analysis, using the current Technical Specification limit on shutdown margin and moderator temperature coefficient.

5.8.7 Conservatism of Results

Conservatism is added to the analysis by inclusion of uncertainties in moderator and fuel temperature coefficients of reactivity, by taking no credit for void reactivity feedback, by taking credit for only 1 HPSI pump and by taking no credit for the stuck CEA worth.

5.10 CEA Ejection Accident

5.10.1 Definition of Event

A CEA ejection accident is defined as a mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure would eject the CEA and the drive shaft to a fully withdr wn position. The consequences of this mechanical failure i a rapid reactivity insertion which when combined with a adverse core power distribution potentially leads to localized fuel damage. The CEA ejection accident is the most rapid reactivity insertion that can be reasonably postulated. The resultant core and thermal power excursion is limited primarily by the Doppler reactivity effect of the increased fuel temperatures and is terminated by reactor trip of the remaining CEA's activated by the high power trip or variable high power trip.

5.10.2. Analysis Criteria

The CEA ejection event is classified as a postulated accident. The design and limiting criteria are:

1. Fuel cladding and enthalpy thresholds (Reference 5-5) are:

   Clad Damage Threshold
   Total Average Enthalpy = 200 cal/gram

   Centerline Melting Threshold
   Total Centerline Enthalpy = 250 cal/gram

   Fully Molten Centerline Threshold
   Total Centerline Enthalpy = 310 cal/gram

5.10 CEA Ejection Accident (Continued)

5.10.5 Analysis Results and 10 CFR 50.59 Criteria

The results of the CEA Ejection Analysis are reported in Section 14.13 of the Fort Calhoun Station Unit No. 1 USAR. Criteria of 10 CFR 50.59 are satisfied if fuel failures are less than those assumed for input to the Radiological Consequences portion of the analysis. Cycle 10, as an example, utilized 1% fuel failure.

5.10.6 Conservatism of Results

The major area of conservatism is the calculation method used to obtain the ejected CEA worth and the ejected radial peak. The ejected worth and the ejected radial peak are calculated without any credit for Doppler or Xenon feedback. In addition, the hot full power ejected worth and ejected peak are calculated assuming the no-load temperature of 532°F. The lower temperature is more adverse since this causes a power role to the core periphery which also happens to be the location of the ejected CEA. Also, the ejected worth is calculated assuming the CEA'S are fully inserted for hot full power case regardless of PDIL. Thus, the ejected worth is conservative.

5.11 Loss of Coolant Accident

The District does not perform the Loss of Coolant Accident Analysis. The large and small break loss of coolant analyses were performed by Combustion Engineering (CE). The large break topical is mentioned in Reference 5-8. The small break analysis shows the closest approach to the 10CFR50.46 criteria for ECCS analysis. The actual difference is

5.12 Loss of Load to Both Steam Generators Event (Continued)

5.12.2 Analysis Criteria (Continued)

a) The peak RCS pressure does not exceed 2750 psia (110% of design pressure).

b) The transient minimum DNBR is greater than the 95/95 confidence interval limit for the CE-1 correction limit.

c) The Peak Linear Heat Generation Rate (PLHGR) does not exceed 22 kw/ft.

Criteria b. and c. are not of major concern since DNBR increases during the event and the PLHGR margin required is much less limiting than other AOO's. Therefore, criterion a. is the main concern in analyzing this event. The loss of load to both steam generators event is the limiting AOO event with respect to peak RCS pressure.

5.12.3 Objectives of the Analysis

The objective of the analysis is to demonstrate, for modifications to the plant which potentially degrade RCS heat removal capability (including steam generator plugging) that the peak RCS pressure stays within 110% of the design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia.

Table 5.12.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF LOAD TO BOTH STEAM GENERATORS ANALYSIS

Parameter	Units	Value
Initial Core Power Level	MW_t	1530 (102%)
Initial Core Inlet Temperature	°F	Maximum allowed by Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed by Tech. Specs
Initial Steam Generator Pressure	psia	Minimum value corresponding to core inlet temperature operating range.
Initial Core Mass Flow Rate	$*10^6$ lbm/hr.	Minimum allowed by Tech. Specs.
Moderator Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	Most positive allowed by Tech. Specs.
Fuel Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	Least negative predicted during core life.
Fuel Temperature Coefficient Multiplier		0.85
CEA Drop Time	sec.	Maximum allowed by Tech. Specs.
Scram Reactivity Worth		Minimum predicted during core lifetime
Scram Reactivity Curve		Consistent with most positive axial shape (bottom peaked) allowed by Tech. Specs.
Core Average H_{gap}	BTU/hr-Ft²-°F	Maximum predicted during core lifetime.
Kinetics Parameters		EOC parameters (minimum absolute β).
RPS Response Time	sec.	1.4

5.12 Loss of Load to Both Steam Generators Event (Continued)

5.12.7 Conservatism of Results (Continued)

6. The maximum pressurizer safety valve capacities are assumed to be 90% of the ASME rated values.

7. A one percent pressure uncertainty is applied to the primary and secondary safety valve setpoints, i.e., a 1.01 multiplier.

5.13 Loss of Feedwater Flow Event

5.13.1 Definition of Event

A total loss of main feedwater flow event is defined as a loss of feedwater flow when operating at power without a corresponding reduction in steam flow from the steam generators. The most likely causes for this event are the loss of all feedwater or condensate pumps or the inadvertent closure of either the main feedwater regulating valves or the feedwater isolation valves due to a feedwater controller malfunction or manual positioning by the operator. The result of this mismatch in which turbine demand remains at 100%, is a reduction of the steam generator liquid inventories and a degrading RCS heat removal capability. As the heat removal capability is lost, through decreasing steam generator inventories (i.e., levels) the RCS temperatures and pressure increase. Normally the event would be terminated by a reactor trip on low steam generator level. Since no credit is taken in the analysis for the steam generator low level trip, a high pressurizer trip eventually results.

5.13 Loss of Feedwater Flow Event (Continued)

5.13.1 Definition of Event (Continued)

Automatic actuation of the auxiliary feedwater (AFW) system will also eventually occur (after reactor trip) if either main feedwater is not restored or manual actuation of the AFW system is not performed by the operator. The AFW system actuation ensures the maintenance of a secondary heat sink.

5.13.2 Analysis Criteria

The loss of feedwater flow event is classified as an Anticipated Operational Occurrence (AOO) for which the following criteria must be met:

a. The peak RCS pressure does not exceed 2750 psia (110% of design pressure).

b. The transient minimum DNBR is greater than the 95/95 confidence interval limit for the CE-1 correlation limit.

c. The Peak Linear Heat Generation Rate (PLHGR) does not exceed 22 kw/ft.

Criteria b. and c. are not of major concern because DNBR does not decrease below the initial steady state value and the PLHGR margin required is much less limiting than other AOO's. Therefore, only criterion a. requires reevaluation should plant modifications (such as steam generator tube

5.13 Loss of Feedwater Flow Event (Continued)

5.13.2 Analysis Criteria (Continued)

plugging) be made which result in degraded secondary heat transfer capability beyond that of this event. For Fort Calhoun Station, this event is bounded by the loss of load incident.

5.13.3 Objectives of the Analysis

The objective of this analysis is to demonstrate, for plant modifications which potentially degrade RCS heat removal capability (including setam generator tube plugging), that the peak RCS pressure stays within 110% of the design pressure in accordance with Section III of the ASME Pressure Vessel Code. This objective is achieved if the peak RCS pressure does not exceed 2750 psia.

5.13.4 Key Parameters and Analysis Assumptions

The key parameters used in the loss of feedwater flow event are given in Table 5.13.4-1. Assumptions in the analysis to maximize heat up of the RCS and consequently the peak RCS pressure include:

1. The event is initiated by an instantaneous loss of main feedwater. No credit is taken for the low steam generator level trip.

2. The steam dump and bypass system is assumed to be in MANUAL (i.e., inoperative).

5.13 Loss of Feedwater Flow Event (Continued)

5.13.4 Key Parameters and Analysis Assumptions (Continued)

3. The pressurizer pressure control system is in MANUAL (i.e., PORV's and sprays are inoperable).

4. The pressurizer level control system is in MANUAL with maximum charging and zero letdown flows.

5. The rod block system is assumed to prevent rod motion (other than scram) during the transient.

5.13.5 Analysis Method

The analysis methods used by the District to analyze a loss of main feedwater flow event consists of using the CESEC computer code to simulate the event, utilizing the analysis assumption listed in Section 5.13.4 (above) as input, and extracting the peak RCS pressure for comparison with the 2750 psia upset limit.

5.13.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the loss of feedwater flow are contained in the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met, if the peak RCS pressure is less than the value reported in Section 14.10.1 of the USAR.

Table 5.13.4-1

KEY PARAMETERS ASSUMED IN THE LOSS OF FEEDWATER FLOW ANALYSIS

Parameter	Units	Value
Initial Core Power Level	MW_t	1530 (102%)
Initial Core Inlet Temperature	°F	Maximum allowed by Tech. Specs.
Initial RCS Pressure	psia	Minimum allowed by Tech. Specs
Initial Steam Generator Pressure	psia	Minimum value corresponding to core inlet temperature operating range.
Initial Core Mass Flow Rate	$*10^6$ lbm/hr.	Minimum allowed by Tech. Specs.
Moderator Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	Most positive allowed by Tech. Specs.
Fuel Temperature Coefficient	$*10^{-4}\Delta\rho/°F$	Least negative predicted during core life.
Fuel Temperature Coefficient Multiplier		0.85
CEA Drop Time	sec.	Maximum allowed by Tech. Specs.
Scram Reactivity Worth		Minimum predicted during core lifetime
Scram Reactivity Curve		Consistent with most positive axial shape (bottom peaked) allowed by Tech. Specs.
Core Average H_{gap}	BTU/hr-Ft²-°F	Maximum predicted during core lifetime.
Kinetics Parameters		EOC parameters (minimum absolute β).
RPS Response Time	sec.	1.4

5.13 Loss of Feedwater Flow Event (Continued)

5.13.6 Analysis Results and 10 CFR 50.59 Criteria

The results of the loss of feedwater flow are contained in the Fort Calhoun Station Unit No. 1 USAR. The criteria of 10 CFR 50.59 are met, if the peak RCS pressure is less than the value reported in Section 14.10.1 of the USAR.

5.13.7 Conservatisms of Results

1. Field measurements demonstrate that the CEA magnetic clutch decay time is less than that assumed in the analysis.

2. The actual scram worths are greater than those assumed in the analysis.

3. The actual MTC is more negative during power operation than assumed in the analysis.

4. The steam dump and bypass system and the pressurizer pressure control system (PORV's and sprays) are operated in the AUTO mode rather than the MANUAL mode as assumed in the analysis.

5. Actual secondary pressure is higher which results in earlier secondary safety valve opening and earlier alleviation of the primary system temperature and pressure rises.

6. No credit is taken for a steam generator low level trip.