DUKE POWER COMPANY
MCGUIRE NUCLEAR STATION
CATAWBA NUCLEAR STATION

FSAR CHAPTER 15 SYSTEM TRANSIENT ANALYSIS METHODOLOGY

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2.0 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

2.1 Feedwater System Malfunctions That Result In A Reduction In Feedwater Temperature

A Feedwater System malfunction that results in a decrease in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Physically, as the cooler feedwater reduces the reactor coolant temperature, positive reactivity will be inserted due to the effect of a negative moderator temperature coefficient. Postulating that the Rod Control System is in automatic control, control rods would be withdrawn as RCS temperature decreased, inserting additional positive reactivity. The net effect on the RCS due to a reduction in feedwater temperature would be similar to the effect of increasing feedwater flow or increasing secondary steam flow; the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A Feedwater System malfunction that results in a decrease in feedwater temperature can be initiated from the following types of events: spurious actuation of a feedwater heater bypass valve, interruption of steam extraction flow to a feedwater heater(s), spurious startup of a single auxiliary feedwater pump, failure of a single feedwater heater drain pump or failure of all feedwater heater drain pumps. The above events are examined, with the most limiting determined to be a spurious actuation of a feedwater heater bypass valve. However, under the current Duke Power Company method of analysis, this accident is bounded by quantitative analysis of the increase in feedwater flow event or the excessive increase in secondary steam flow event. These events bound the reduction in feedwater temperature event by producing a greater RCS cooldown. The applicable acceptance criterion is that fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations.

2.2 Feedwater System Malfunction Causing an Increase in Feedwater Flow

The malfunctions considered are 1) the full opening of a single main feedwater control valve, 2) an increase in the speed of a single main feedwater pump, 3) the spurious startup of a single auxiliary feedwater pump, or 4) a malfunction which affects more than one loop. ~~The latter scenario has been identified only recently and is currently being evaluated to determine applicability to McGuire and Catawba.~~ The limiting scenario from among those listed above is evaluated to demonstrate that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations using the Statistical Core Design Methodology.

2.2.1 Nodalization

Of the events identified in the previous section, the latter, the multi-loop malfunction, is ~~expected to be~~ the most limiting, and is therefore the one that is discussed. This transient affects all loops equally and ~~would~~is therefore ~~be~~ analyzed with a single-loop NSSS system model (Reference 2, Section 3.2). ~~If the most limiting transient is not determined to be one which affects all loops equally, a multiple loop model would be selected appropriately. The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

2.2.2 Initial Conditions

<u>Core Power Level</u>
High initial power level maximizes the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

<u>Pressurizer Pressure</u>
The nominal pressure corresponding to full power operation is assumed, with the pressure initial condition uncertainty accounted for in the Statistical Core Design Methodology.

<u>Pressurizer Level</u>
Since this accident involves a reduction in RCS volume due to coolant contraction, a positive level uncertainty is applied to the nominal programmed level to minimize the initial pressurizer steam bubble volume and therefore maximize the pressure decrease due to contraction.

<u>Reactor Vessel Average Temperature</u>
The nominal temperature corresponding to full power operation is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.

<u>RCS Flow</u>
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

<u>Core Bypass Flow</u>
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

<u>Steam Generator Level</u>
A negative level uncertainty is assumed to maximize the margin to a high-high steam generator narrow range level reactor trip due to any temporary steam/feedwater flow mismatch. This maximizes the duration of the overcooling before it is ended by feedwater isolation.

Fuel Temperature
A low initial temperature is assumed to maximize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A high gap conductivity minimizes the fuel heatup and attendant negative reactivity insertion caused by the power increase. This makes the power increase more severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
In order to maximize the effects of the increased secondary system heat removal, no tube plugging is assumed.

2.2.3 Boundary Conditions

Main Feedwater Flow
A conservatively large step change in main feedwater flow to all steam generators is assumed at the initiation of the transient. **A step decrease in main feedwater temperature is assumed to account for the increased main feedwater flow rate.**

2.2.4 Control, Protection, and Safeguards Systems Modeling

Reactor Trip
The pertinent reactor trip functions are the low-low steam generator level, high flux and overpower ΔT. The safety analysis setpoint or the initial condition for the monitored parameter contains an allowance for measurement instrumentation uncertainty and setpoint setting tolerance.

Pressurizer Level Control
No credit is taken for pressurizer level control system operation to compensate for the depressurization which accompanies RCS volume shrinkage.

Rod Control
This accident will result in a decrease in RCS temperature. ~~With the Rod Control System in manual control, t~~The reduced temperature will cause a positive reactivity insertion through the negative moderator temperature coefficient. With the Rod Control System in automatic control, ~~in which~~**the control rods may insert due to the mismatch between NI power and turbine power and cause a negative reactivity insertion. However, since** the reactor vessel average temperature is maintained at a programmed value, the control rods **may withdraw in an attempt to maintain this temperature and**~~will~~ cause a positive reactivity insertion ~~as they are withdrawn in an attempt to maintain this temperature~~. Both ~~cases~~**automatic and manual control of the Rod Control System** are analyzed in order to ensure that the ~~worse one~~ **worst case** is determined.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2. In this mode any decrease in steam pressure, due for example to a shift from latent to sensible heat

transfer because of the overfeed, would be compensated for by an opening of the turbine control valves to maintain impulse chamber pressure at the programmed value.

Auxiliary Feedwater

~~AFW flow is credited, after the appropriate Technical Specification response time delay, when the safety analysis value of the low-low steam generator narrow range level setpoint is reached. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.~~

AFW flow would be credited, after the appropriate Technical Specification response time delay, when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip

Turbine trip is credited, after the appropriate Technical Specification response time delay, on high-high steam generator narrow range level **or on reactor trip**.

Feedwater Isolation

Feedwater isolation is credited, after the appropriate Technical Specification response time delay, on high-high steam generator narrow range level.

2.3 Excessive Increase in Secondary Steam Flow

The accident analyzed is a step increase in secondary steam flow of a magnitude equal to that for which the Reactor Control System is designed, 10% of licensed core thermal power. Increases of larger magnitude are discussed in Section 2.4 and in Chapter 5 of Reference 1. The accident is analyzed to demonstrate that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

2.3.1 Nodalization

The accident analyzed is an excessive increase in secondary steam flow at power. Flow increases from a zero power initial condition are evaluated in Section 2.4 and in Chapter 5 of Reference 1. Per Reference 3, Section 15.1.4, the power level analyzed for this accident should be 102% of licensed core thermal power for the number of loops initially assumed to be operating. At power, the Technical Specifications require all four loops to be operating. Therefore full power is assumed as the initial condition. An increase in steam flow to the turbine would affect all loops equally, therefore, a single-loop NSSS system model (Reference 2, Section 3.2) is used. ~~The pressurizer modeling includes~~

~~the use of the local conditions heat transfer option for the vessel conductors.~~

2.3.2 Initial Conditions

Core Power Level
Per Reference 3, Section 15.1.4, the power level analyzed for this accident should be 102% of licensed core thermal power for the number of loops initially assumed to be operating. At power, the Technical Specifications require all four loops to be operating. Therefore full power is assumed as the initial condition. The uncertainty in initial power level is accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
The nominal pressure corresponding to full power operation is assumed, with the pressure initial condition uncertainty accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Since this accident involves, particularly for the manual Rod Control System operation scenario, a reduction in RCS volume due to coolant contraction, a positive level uncertainty is assumed to minimize the initial pressurizer steam bubble volume and therefore maximize the pressure decrease due to contraction.

Reactor Vessel Average Temperature
The nominal temperature corresponding to full power operation is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.

RCS Flow
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Level
The results of this transient are not sensitive to the direction of steam generator level uncertainty as long as the transient level response is kept within the range that avoids protection or safeguards actuation.

Fuel Temperature
A low initial temperature is assumed to maximize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A high gap conductivity minimizes the fuel heatup and attendant negative reactivity insertion caused by the power increase. This makes the power increase more severe and is therefore conservative for DNB evaluation.

Steam Generator Tube Plugging
In order to maximize the effects of the increased secondary system heat removal, no tube plugging is assumed.

2.3.3 Boundary Conditions

Main Steam Flow
A step change in main steam flow to the turbine equal to 10% of full power flow is assumed at the initiation of the transient.

2.3.4 Control, Protection, and Safeguards Systems Modeling

Reactor Trip
The reactor is not expected to trip for this transient. However, reactor trip is credited, after the appropriate Technical Specification response time delay, if the safety analysis setpoint is exceeded for any reactor trip function.

Pressurizer Level Control
No credit is taken for pressurizer level control system operation to compensate for the depressurization which accompanies RCS volume shrinkage.

Steam Line PORVs and Condenser Steam Dump
While the steam line PORVs and steam dump might be a source of the increased steam flow in this postulated accident, the case analyzed assumes the increased flow exits to the turbine.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
This accident will result in a decrease in RCS temperature. With the Rod Control System in manual control, the reduced temperature will cause a positive reactivity insertion through the negative moderator temperature coefficient. With the Rod Control System in automatic control, in which the reactor vessel average temperature is maintained at a programmed value, the control rods will cause a positive reactivity insertion as they are withdrawn in an attempt to maintain this temperature. Both cases are analyzed in order to ensure that the worse one is considered.

3.0 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

3.1 Turbine Trip

The turbine trip event causes a loss of heat sink to the primary system. The mismatch between power generation in the primary system and heat removal by the secondary system causes temperature and pressure to increase in the primary and secondary until reactor trip and/or lift of the pressurizer safety valves and main steam safety valves. The transient is analyzed to ensure that both the peak Reactor Coolant System pressure and the peak Main Steam System pressure remain below the acceptance criterion of 110% of design pressure. Peak RCS pressure and peak Main Steam System pressure are analyzed separately due to the differences in assumptions required for a conservative analysis.

3.1.1 Peak RCS Pressure Analysis

3.1.1.1 Nodalization

Since the transient response of the turbine trip event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.1.1.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary-to-secondary power mismatch upon turbine trip.

Pressurizer Pressure
Positive uncertainty is applied to the initial pressurizer pressure. High initial pressure reduces the initial margin to the overpressure limit.

Pressurizer Level
High initial level minimizes the initial volume of the pressurizer steam space, which maximizes the transient primary pressure response.

Reactor Vessel Average Temperature
High initial temperature maximizes the primary coolant stored energy, which maximizes the transient primary pressure response.

RCS Flow
Low initial flow minimizes the primary-to-secondary heat transfer.

Core Bypass Flow
Core bypass flow is not an important parameter in this analysis.

Steam Generator Level
High initial level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response. Maximum secondary pressurization causes maximum secondary temperature response, which minimizes primary-to-secondary heat transfer.

Fuel Temperature
Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
A bounding high tube plugging value degrades primary-to-secondary heat transfer.

3.1.1.3 Boundary Conditions

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize the pressurizer pressure.

Steam Line Safety Valves
The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure and minimize transient primary-to-secondary heat transfer.

3.1.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
The pertinent reactor trip functions are the overtemperature ΔT (OTΔT), overpower ΔT (OPΔT), and pressurizer high pressure.

The response time of each of the two ΔT trip functions is the Technical Specification value. The setpoint values of the ΔT trip functions are continuously computed from system parameters using the modeling described in Section 3.2.4.2 of Reference 2. In addition, the ΔT coefficients used in the analysis account for instrument uncertainties.

The response time of the pressurizer high pressure trip function is the Technical Specification value. Since the pressure uncertainty is accounted for in the initial pressurizer pressure, the pressurizer high pressure reactor trip setpoint is the Technical Specification value.

Pressurizer Pressure Control
Pressurizer pressure control is in manual with sprays and PORVs disabled in order to maximize primary pressure.

Pressurizer Level Control
Pressurizer level control is in ~~automatic~~ **manual with the pressurizer heaters locked on** in order to elevate primary pressure.
Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
Feedwater is isolated upon turbine trip. The addition of subcooled feedwater would tend to subcool the water in the steam generator, and reduce secondary side pressure.

Rod Control
No credit is taken for the operation of the Rod Control System. Following turbine trip, the turbine impulse chamber pressure is rapidly reduced. The corresponding reduction in the Rod Control System reference temperature would lead to control rod insertion, which would lessen the severity of the transient.

Auxiliary Feedwater
Auxiliary feedwater is disabled. The addition of subcooled auxiliary feedwater would tend to subcool the water in the steam generator, and reduce secondary side pressure.

3.1.2 Peak Main Steam System Pressure Analysis

3.1.2.1 Nodalization

Since the transient response of the turbine trip event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.1.2.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary-to-secondary power mismatch upon turbine trip.

Pressurizer Pressure
Positive uncertainty is applied to the initial pressurizer pressure. As long as a high pressurizer pressure reactor trip is avoided, maximum primary system pressure is conservative in order to delay reactor trip on OTΔT.

Pressurizer Level
High initial level minimizes the initial volume of the pressurizer steam space, which maximizes the transient primary pressure response.

Pressurizer Pressure Control
Pressurizer pressure control is in automatic with sprays and PORVs enabled in order to prevent a high pressurizer pressure reactor trip actuation prior to OTΔT trip actuation.

Pressurizer Level Control
Pressurizer level control is in ~~automatic~~ **manual with the pressurizer heaters locked on** in order to elevate primary pressure. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and condenser steam dump is unavailable in order to maximize secondary side pressurization.

Steam Generator Level Control
Feedwater is isolated upon turbine trip. The addition of subcooled feedwater would tend to subcool the water in the steam generator, and reduce secondary side pressure.

Rod Control
No credit is taken for the operation of the Rod Control System. Following turbine trip, the turbine impulse chamber pressure is rapidly reduced. The corresponding reduction in the Rod Control System reference temperature would lead to control rod insertion, which would lessen the severity of the transient.

Auxiliary Feedwater
Auxiliary feedwater is disabled. The addition of subcooled auxiliary feedwater would tend to subcool the water in the steam generator, and reduce secondary side pressure.

3.2 Loss of Non-Emergency AC Power To The Station Auxiliaries

A loss of non-emergency AC power causes the power supply to all busses not powered by the emergency diesel generators to be lost. This leads to the trip of both the main feedwater pumps and the reactor coolant pumps. A primary system heatup ensues, due to both the coastdown of the reactor coolant pumps and the loss of main feedwater heat removal. As a result of this heatup, the primary concerns for this event are **short-term core cooling capability** (DNBR), **long-term core cooling capability (natural circulation)**, and primary and secondary system overpressurization.

This transient differs from the complete loss of flow transient only in the timing of the insertion of the control rods. Both transients presume reactor coolant pump and feedwater pump trip as the initiating events. In the loss of flow event, the reactor trips when the reactor coolant pump bus undervoltage setpoint is reached and the rods begin to fall into the core after an instrumentation delay. In the loss of AC power transient, the control rods begin to fall immediately due to the loss of gripper coil voltage. Therefore, the transient core power response and consequently the ~~DNBR~~**short-term core cooling capability** result **(DNBR)** is bounded by the loss of flow event. **Long-term core cooling capability is shown by analyzing the transition from forced flow to natural circulation following a loss of non-emergency AC power.**

Similarly, the primary system temperature increase and, therefore, the peak primary system pressure is also bounded by the loss of flow event.

Secondary side pressure does not rise significantly until the turbine trip occurs and steam flow is terminated. The magnitude of this pressure increase is largely determined by the amount of heat transferred from the primary system to the secondary once the pressure increase has begun. For this event the reactor trip occurs prior to the turbine trip, such that the primary system heat generation is rapidly decreasing as secondary side pressure is increasing. Therefore, the peak secondary pressure result is bounded by the turbine trip event, in which the reactor trip occurs well after the turbine trip.

Based on the above qualitative evaluation, a quantitative analysis of this transient is not required **except for the long-term core cooling capability analysis**. Should a reanalysis become necessary, either due to plant changes, modeling changes, or other changes which invalidate any of the above arguments, the analytical methodology employed would be as follows.

Peak RCS pressure, peak Main Steam System pressure and core cooling capability (**short-term and long-term**) are each analyzed separately due to the differences in assumptions required for a conservative analysis. The **short-term** core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology. **The long-term core cooling capability analysis demonstrates that natural circulation is established.**

3.2.1 Peak RCS Pressure Analysis

3.2.1.1 Nodalization

Since the transient response of the loss of offsite power event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.2.1.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary-to-secondary power mismatch.

Pressurizer Pressure Control
Pressurizer pressure control is in manual with sprays and PORVs disabled in order to maximize primary pressure.

Pressurizer Level Control
Pressurizer level control is in automatic in order to maximize primary pressure. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on the loss of offsite power after an appropriate Technical Specification response time delay. **If applicable, a**~~A~~ purge volume of hot main feedwater is assumed to be delivered prior to the cold AFW water reaching the steam generators. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

Turbine Trip
Turbine trip occurs on the loss of offsite power.

3.2.2 Peak Main Steam System Pressure Analysis

3.2.2.1 Nodalization

Since the transient response of the loss of offsite power event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.2.2.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary-to-secondary heat transfer.

Pressurizer Pressure
Pressurizer pressure is not an important parameter in this analysis.

Pressurizer Level
Since initial level primarily affects the transient primary pressure response, it is not an important parameter in this analysis.

Reactor Vessel Average Temperature
High initial temperature maximizes the initial Main Steam System pressure and the primary coolant stored energy.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on the loss of offsite power after the appropriate Technical Specification response time delay. **If applicable, a**~~A~~ purge volume of hot main feedwater is assumed to be delivered prior to the cold AFW water reaching the steam generators. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

Turbine Trip
Turbine trip occurs on the loss of offsite power.

3.2.3 Core Cooling Capability Analysis - **Short Term**

3.2.3.1 Nodalization

Since the transient response of the loss of offsite power event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.2.3.2 Initial Conditions

Core Power Level
High initial power level maximizes the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
Nominal full power pressurizer pressure is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Low initial level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the insurge.

Reactor Vessel Average Temperature
Nominal full power vessel average temperature is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

RCS Flow
Technical Specification minimum measured Reactor Coolant System flow is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on the loss of offsite power after the appropriate Technical Specification response time delay. If applicable, a~~A~~ purge volume of hot main feedwater is assumed to be delivered prior to the cold AFW water reaching the steam generators. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

Turbine Trip
Turbine trip occurs on the loss of offsite power.

3.2.4 Core Cooling Capability Analysis - Long Term

3.2.4.1 Nodalization

Since the transient response of the loss of offsite power event is the same for all loops, the single loop model described in Section 3.2 of Reference 2 is utilized for this analysis.

3.2.4.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary system heat source.

Pressurizer Pressure
The nominal pressure corresponding to full power operation is assumed since the establishment of natural circulation is independent of initial pressurizer pressure.

Pressurizer Level
The nominal level corresponding to full power operation is assumed since the establishment of natural circulation is independent of initial pressurizer level.

Reactor Vessel Average Temperature
High initial temperature maximizes the amount of stored energy in the primary system that must be removed by the secondary system.

RCS Flow
Technical Specification minimum measured Reactor Coolant System flow is assumed since initial RCS flow has little impact on the final natural circulation flow.

Core Bypass Flow
Core bypass flow is not an important parameter in this analysis.

Steam Generator Level
High initial steam generator level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response. Maximum secondary pressurization causes maximum

secondary temperature response, which minimizes primary-to-secondary heat transfer.

Fuel Temperature
Initial fuel temperature is not an important parameter in this analysis.

Steam Generator Tube Plugging
A bounding high tube plugging value degrades primary-to-secondary heat transfer.

3.2.4.3 Boundary Conditions

Reactor Coolant Pumps
All reactor coolant pumps are assumed to trip on undervoltage at the initiation of the loss of offsite power.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

3.2.4.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
The insertion of all control and shutdown banks occurs when the power is lost to the control rod drive mechanism.

Pressurizer Pressure Control
Pressurizer sprays are lost when the reactor coolant pumps trip. Pressurizer PORVs are lost when offsite power is lost. Therefore, both are inoperable.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable since they are lost when offsite power is lost. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable due to the loss of offsite power.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on the loss of offsite power after the appropriate Technical Specification response time delay. In order to minimize post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

Turbine Trip
Turbine trip occurs on the loss of offsite power.

3.3 <u>Loss Of Normal Feedwater</u>

A loss of normal feedwater flow event could result due to the failure of both of the main feedwater pumps or a malfunction of the feedwater control valves. A primary system heatup ensues due to the degradation of the secondary heat sink. As a result of this heatup, the primary concerns for this event are DNB and primary and secondary system overpressurization.

The loss of normal feedwater transient is bounded by the turbine trip transient. Both transients involve a mismatch between primary heat source and secondary heat sink, but the mismatch is greater for the turbine trip. This is mainly due to the reactor trip and turbine trip occurring simultaneously for the loss of feedwater event, whereas reactor trip lags the turbine trip during the turbine trip transient.

Based on the above qualitative evaluation, a quantitative analysis of this transient is not required. Should a reanalysis become necessary, either due to plant changes, modeling changes, or other changes which invalidate any of the above arguments, the analytical methodology employed would be as follows.

Peak RCS pressure, peak Main Steam System pressure and core cooling capability are each analyzed separately due to the differences in assumptions required for a conservative analysis. The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

3.3.1 Peak RCS Pressure Analysis

3.3.1.1 Nodalization

Since the transient response of the loss of normal feedwater event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.3.1.2 Initial Conditions

<u>Core Power Level</u>
High initial power level and a positive power uncertainty maximize the primary-to-secondary power mismatch.

<u>Pressurizer Pressure</u>
Positive instrument uncertainty is applied to the initial pressurizer pressure. High initial pressure reduces the initial margin to the overpressure limit.

Pressurizer Level
High initial level minimizes the initial volume of the pressurizer steam space, which maximizes the transient primary pressure response.

Reactor Vessel Average Temperature
High initial temperature maximizes the initial primary coolant stored energy, which maximizes the transient primary pressure response.

RCS Flow
Low initial flow degrades the primary-to-secondary heat transfer.

Core Bypass Flow
Core bypass flow is not an important parameter in this analysis.

Steam Generator Level
~~High initial level in all steam generators delays reactor trip on low-low level and maximizes the heatup of the primary system~~ .**Low initial level is assumed in order to minimize steam generator inventory at the time of reactor trip. The low-low level trip setpoint is adjusted to account for the difference between actual level and indicated level.**

Fuel Temperature
Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
A bounding high tube plugging value degrades primary-to-secondary heat transfer.

3.3.1.3 Boundary Conditions

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize the pressurizer pressure.

Steam Line Safety Valves
The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure and minimize transient primary-to-secondary heat transfer.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

3.3.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip occurs when the low-low level setpoint is reached in the steam generator.

Pressurizer Level
Pressurizer level is not an important parameter in this analysis.

Reactor Vessel Average Temperature
High initial temperature maximizes the initial Main Steam System pressure and the primary coolant stored energy.

RCS Flow
High initial flow maximizes the primary-to-secondary heat transfer.

Core Bypass Flow
Core bypass flow is not an important parameter in this analysis.

Steam Generator Level
~~High initial level in all steam generators delays reactor trip on low-low level. Also, high level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response.~~**Low initial level is assumed in order to minimize steam generator inventory at the time of reactor trip. The low-low level trip setpoint is adjusted to account for the difference between actual level and indicated level.**

Fuel Temperature
Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
Zero tube plugging is modeled to maximize primary-to-secondary heat transfer.

3.3.2.3 Boundary Conditions

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize the pressurizer pressure.

Steam Line Safety Valves
The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

3.3.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip occurs when the low-low level setpoint is reached in the steam generator.

Pressurizer Pressure Control
The results of this transient are not sensitive to the operation of pressurizer pressure control as long as the pressure is controlled to within the range that avoids protection or safeguards actuation.

Pressurizer Level Control
The results of this transient are not sensitive to the operation of pressurizer level control as long as the level is kept within the range that avoids protection or safeguards actuation.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and condenser steam dump is unavailable in order to maximize the transient secondary side pressurization.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on low-low steam generator level after the appropriate Technical Specification response time delay **If applicable, a** ~~A~~ purge volume of hot main feedwater is assumed to be delivered prior to the cold AFW water reaching the steam generators. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

3.3.3 Core Cooling Capability Analysis

3.3.3.1 Nodalization

Since the transient response of the loss of normal feedwater event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

3.3.3.2 Initial Conditions

Core Power Level
High initial power level maximizes the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
Nominal full power pressurizer pressure is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Low initial level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the insurge.

Reactor Vessel Average Temperature
Nominal full power vessel average temperature is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

RCS Flow
Minimum measured Reactor Coolant System flow is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Tube Plugging
A bounding high tube plugging level impairs the ability of the secondary side to remove primary side heat.

Fuel Temperature
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Level
~~High initial level in all steam generators delays reactor trip on low-low level and extends the primary system heatup.~~**Low initial level is assumed in order to minimize steam generator inventory at the time of reactor trip. The low-low level trip setpoint is adjusted to account for the difference between actual level and indicated level.**

3.3.3.3 Boundary Conditions

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

3.3.3.4 Control, Protection, and Safeguards System Modeling

Reactor Trip

Reactor trip occurs when the low-low level setpoint is reached in the steam generator.

Pressurizer Pressure Control

Pressurizer sprays and PORVs are assumed to be operable in order to minimize the system pressure throughout the transient.

Pressurizer Level Control

No credit is taken for pressurizer heater operation so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump

Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Rod Control

No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control

The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater

Auxiliary feedwater actuation occurs on low-low steam generator level after the appropriate Technical Specification response time delay. **If applicable, a**~~A~~ purge volume of hot main feedwater is assumed to be delivered prior to the cold AFW water reaching the steam generators. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

Turbine Trip

Turbine trip occurs on reactor trip.

3.4 Feedwater System Pipe Break

The feedwater system pipe break event postulates a rupture of the Main Feedwater System piping just upstream of the steam generator (downstream of the final feedline check valve). Following the blowdown of the faulted generator, there is a mismatch between the heat generation in the reactor and the secondary side heat removal rate. Due to the mismatch, the primary concern for this transient is the capability to effectively cool the reactor core.

Steam Generator Level
Low initial level in all steam generators decreases the long-term capability of the secondary system to remove primary system heat.

Fuel Temperature
A conservatively high initial fuel temperature is assumed in order to maximize the amount of stored energy that must be removed.

Steam Generator Tube Plugging
~~A bounding high tube plugging level impairs the ability of the secondary side to remove primary side heat.~~**Tube plugging does not significantly affect the transient results so long as the minimum Technical Specification RCS flow rate is used.**

3.4.2.3 Boundary Conditions

Break Modeling
The feedline break is modeled as a double-ended rupture of the main feedwater line just upstream of the steam generator (downstream of the check valve). A bounding flow area of the break junction is assumed in order to maximize the break flowrate. The break flowrate is determined by the Henry (subcooled) and Moody (saturated) critical flow correlations.

Reactor Coolant Pumps
~~The timing of the operator action to trip the reactor coolant pumps is investigated in a sensitivity study. Based on the results of this sensitivity study, an early pump trip time, with the corresponding natural circulation heatup, is conservative. The RCPs are tripped at 15 seconds, which is assumed to precede the time at which the pumps would be manually tripped on high high containment pressure.~~**The reactor coolant pumps are lost at the initiation of the loss of offsite power which occurs coincident with reactor trip.**

Offsite Power
Offsite power is assumed to be lost coincident with reactor trip to delay safety injection and accelerate the post-trip heatup due to the loss of the reactor coolant pumps.

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which minimize pressurizer pressure.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

3.4.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
The reactor is tripped 10 seconds into the transient. This is assumed to be after the occurrence of safety injection actuation on high containment pressure.

Pressurizer Pressure Control
Since low Reactor Coolant System pressure is conservative and the blowdown pressure of a cycling safety valve is much lower than for a cycling PORV, the PORVs are assumed inoperable. Pressurizer spray is assumed to be operable in order to minimize system pressure.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, **since the pre-trip RCS temperature change is insufficient to cause rod motion.**~~which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.~~

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Safety Injection
Safety injection actuation occurs at 10 seconds on high containment pressure. **Injection begins after the appropriate Technical Specification delay to allow for the startup of the diesel generators on the loss of offsite power**. One-train minimum injection flow, as a function of RCS pressure, is assumed to minimize the delivery of cold SI water. Injection is stopped when the emergency procedure SI termination criteria are met.

Auxiliary Feedwater
Auxiliary feedwater actuation occurs on safety injection actuation after the appropriate Technical Specification response time delay. **If applicable, a**~~A~~ purge volume of hot water is assumed to be delivered prior to the cold AFW water reaching the steam generators. Operator action to isolate AFW flow to the faulted generator occurs **with a conservative delay time to minimize the amount of cold AFW flow to the faulted generator.**~~at 120 seconds as a result of a sensitivity study~~. In order to minimize the post-trip steam generator heat removal, the minimum auxiliary feedwater flow is assumed.

MSIV Closure

~~The timing of the closure of the main steam isolation valves is the focus of a sensitivity study which shows that early MSIV closure, which initiates the overheating phase of the transient, is conservative. The valves are closed at 15 seconds, which is assumed to precede automatic closure on high-high containment pressure.~~Early MSIV closure is conservative since it accelerates the heatup portion of the transient due to the faulted SG reaching dryout sooner following MSIV closure. Main steam line isolation occurs on low steam line pressure or high-high containment pressure. Since neither of these setpoints can be reached before reactor trip, it is conservatively assumed that MSIV closure occurs coincident with turbine trip.

4.0 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

4.1 Partial Loss of Forced Reactor Coolant Flow

A partial loss of forced reactor coolant flow can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. If the reactor is at power when such a fault occurs, this could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. The necessary protection against a partial loss of coolant flow is provided by the low reactor coolant flow reactor trip signal.

The acceptance criteria for this analysis are to ensure that there is adequate core cooling capability and that the pressure in the Reactor Coolant System remains below 110% of design pressure. The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology. The peak RCS pressure criterion is met through a comparison to the peak pressure results for the more limiting locked rotor transient. In Section 4.3 of this report, the locked rotor event is shown to remain below 110% of the RCS design pressure.

4.1.1 Nodalization

This non-symmetric transient is analyzed using a two-loop model, with a single loop for the tripped reactor coolant pump and an intact triple loop. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

4.1.2 Initial Conditions

<u>Core Power Level</u>
High initial power level maximizes the primary system heat flux. The uncertainty for this parameter is incorporated in the Statistical Core Design Methodology.

<u>Pressurizer Pressure</u>
The nominal pressure corresponding to full power operation is assumed, with the uncertainty for this parameter incorporated in the Statistical Core Design Methodology.

<u>Pressurizer Level</u>
Low initial level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the insurge.

<u>Reactor Vessel Average Temperature</u>
The nominal temperature corresponding to full power operation is assumed, with the uncertainty for this parameter incorporated in the Statistical Core Design Methodology.

Pressurizer Pressure Control
Pressurizer sprays and PORVs are assumed to be operable in order to minimize the system pressure throughout the transient.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long **as** the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip
The reactor trip leads to a subsequent turbine trip.

4.2 Complete Loss Of Forced Reactor Coolant Flow

A complete loss of forced reactor coolant flow would occur if all four reactor coolant pumps tripped due to either a common mode failure or a simultaneous loss of power to the pump motors. The Reactor Protection System (RPS) senses an undervoltage condition at the pumps and initiates a reactor trip. The decrease in core flow which occurs prior to reactor trip causes a heatup of the Reactor Coolant System.

Fuel Temperature
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
For transients of such short duration, steam generator tube plugging does not have an effect on the transient results.

4.2.3 Boundary Conditions

RCP Operation
All four reactor coolant pumps are tripped at the initiation of the transient. The pump model is adjusted such that the resulting coastdown flow is conservative with respect to the flow coastdown test data.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

4.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip occurs on reactor coolant pump undervoltage, after an appropriate instrumentation delay.

Pressurizer Pressure Control
Pressurizer sprays and PORVs are assumed to be operable in order to minimize the system pressure throughout the transient.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip
The reactor trip leads to a subsequent turbine trip.

4.3 Reactor Coolant Pump Locked Rotor

The postulated accident involves the instantaneous seizure of one reactor coolant pump rotor. Coolant flow in that loop is rapidly reduced, causing the Reactor Protection System (RPS) to initiate a reactor trip on low RCS loop flow. The mismatch between power generation and heat removal capacity due to the degraded flow condition causes a heatup of the primary system.

The acceptance criteria for this analysis are to ensure that there is adequate core cooling capability and that the pressure in the Reactor Coolant System remains below 120% of design pressure. Peak RCS pressure and core cooling capability are analyzed separately due to the differences in assumptions required for a conservative analysis. The core cooling capability analysis determines to what extent fuel cladding integrity is compromised by calculating the number of fuel rods that exceed the 95/95 DNBR limit based on acceptable correlations.

4.3.1 Peak RCS Pressure Analysis

4.3.1.1 Nodalization

Due to the asymmetry of the transient, a two-loop model (Reference 2, Section 3.2), with a faulted single loop and ~~one~~ intact triple loop, is utilized for this analysis. ~~The pressurizer modeling includes the use~~

~~of the local conditions heat transfer option for the vessel conductors.~~

4.3.1.2 Initial Conditions

Core Power Level
High initial power level and a positive power uncertainty maximize the primary system heat load.

Pressurizer Pressure
High initial pressure yields a smaller margin to overpressurization.

Pressurizer Level
High initial level decreases the volume of the pressurizer steam space which maximizes the pressure increase resulting from the insurge.

Reactor Vessel Average Temperature
High initial temperature maximizes the initial primary coolant stored energy, which maximizes the transient primary pressure response.

RCS Flow
Low initial flow minimizes the primary-to-secondary heat transfer.

Core Bypass Flow
High core bypass flow minimizes coolant flow through the core and exacerbates heatup.

Steam Generator Level
Initial steam generator level is not an important parameter in this analysis.

Fuel Temperature
Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
For transients of such short duration, steam generator tube plugging does not have an effect on the transient results.

4. 1.3 Boundary Conditions

Reactor Coolant Pumps
The rotor of the reactor coolant pump in the faulted loop is assumed to seize at the initiation of the transient. The remaining reactor coolant pumps trip on bus undervoltage following the loss of offsite power.

Offsite Power
Offsite power is assumed to be lost coincident with the turbine trip.

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize pressurizer pressure.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

4.3.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip occurs on low Reactor Coolant System flow in the locked loop.

Pressurizer Pressure Control
In order to maximize primary system pressure, no credit is taken for pressurizer spray or PORV operation.

Pressurizer Level Control
Pressurizer heaters are assumed to be operable in order to maximize Reactor Coolant System pressure resulting from the insurge/level increase. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip

The reactor trip leads to a subsequent turbine trip.

4.3.2 Core Cooling Capability Analysis

4.3.2.1 Nodalization

Due to the asymmetry of the transient, a two-loop model (Reference 2, Section 3.2), with a single (faulted) loop and a triple (intact) loop, is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

4.3.2.2 Initial Conditions

Core Power Level

~~High initial power level and a positive power uncertainty maximize the primary system heat load.~~**High initial power level maximizes the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.**

Pressurizer Pressure

~~Low initial pressure yields a lower initial, and therefore transient, DNBR.~~**The nominal pressure corresponding to full power operation is assumed, with the pressure initial condition uncertainty accounted for the in Statistical Core Design Methodology.**

Pressurizer Level

Low initial level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the insurge.

Reactor Vessel Average Temperature

~~High initial temperature increases the stored energy in the primary system which must be removed and minimizes the transient DNBR.~~**The nominal temperature corresponding to full power operation is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.**

RCS Flow

~~Low initial flow degrades the primary-to-secondary heat transfer and minimizes the transient DNBR.~~**The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.**

Core Bypass Flow

~~High core bypass flow exacerbates heatup by minimizing coolant flow through the core and minimizes the transient DNBR.~~**The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.**

Steam Generator Level
Initial steam generator level is not an important parameter in this analysis.

Fuel Temperature
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
For transients of such short duration, steam generator tube plugging does not have an effect on the transient results.

4.3.2.3 Boundary Conditions

Reactor Coolant Pumps
The rotor of the reactor coolant pump in the faulted loop is assumed to seize at the initiation of the transient. The remaining reactor coolant pumps trip on bus undervoltage following the loss of offsite power.

Offsite Power
~~Offsite power is assumed to be lost coincident with the turbine trip.~~**Cases with offsite power maintained as well as with offsite power lost coincident with the turbine trip are analyzed.**

Pressurizer Safety Valves
The pressurizer safety valves are not challenged by this transient.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

4.3.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip occurs on low Reactor Coolant System flow in the loop with the locked rotor.

Pressurizer Pressure Control
Credit is taken for both pressurizer spray and PORV operation in order to minimize primary system pressure.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip
The reactor trip leads to a subsequent turbine trip.

4.3.2.5 Other Assumptions

The peak clad temperature calculation employs the fuel conduction model as described in Section 4.2.2 of Reference 1.

5.0 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

5.1 Uncontrolled Bank Withdrawal From a Subcritical or Low Power Startup Condition

A malfunction of the Rod Control System can result in an uncontrolled withdrawal of control rods. Beginning from a low initial power typical of Modes 2 and 3, the resulting positive reactivity addition causes a power excursion which is terminated by the high power range flux (low setpoint) **or high pressurizer pressure** RPS trip functions. Since the initial condition requires as few as three reactor coolant pumps in operation, the minimum DNBR is of concern for peak transient power levels less than full power. The peak Reactor Coolant System pressure limit of 110% of design pressure is also of concern due to the mismatch between core power and the secondary heat sink during the power excursion. Peak RCS pressure and core cooling capability are analyzed separately due to the differences in assumptions required for a conservative analysis. The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

5.1.1 Peak RCS Pressure Analysis

5.1.1.1 Nodalization

The peak RCS pressure transient is analyzed with four reactor coolant pumps in operation. Since all initial and boundary conditions are symmetric, a single-loop model or any multi-loop nodalization is appropriate. The standard model (Reference 2, Section 3.2) is used with one significant exception. Since this transient initiates at zero power, and since the duration of the transient is very short, the steam generator secondary response is not important. Rather than using the standard steam generator secondary nodalization, a single secondary volume is used. The single volume uses the bubble rise option with the local-conditions heat transfer model applied to the steam generator tube conductors. With this modeling approach the initial condition of zero power can be obtained, and the primary-to-secondary heat transfer that occurs following the power excursion can be simulated. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

5.1.1.2 Initial Conditions

Core Power Level

A minimum initial power level typical of a critical, zero power startup condition maximizes the power excursion.

5.1.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip

The pertinent reactor trip functions are the high power range flux (low setpoint) and pressurizer high pressure.

The high power range flux (low setpoint) trip includes a conservative allowance to account for calibration error, and error due to rod withdrawal effects. The response time of the high flux trip function is the Technical Specification value.

The response time of the pressurizer high pressure trip function is the Technical Specification value. Since the pressure uncertainty is accounted for in the initial pressurizer pressure, the pressurizer high pressure reactor trip setpoint is the Technical Specification value.

Pressurizer Pressure Control

Pressurizer spray and PORVs are inoperable to maximize RCS pressure during the transient.

Pressurize Level Control

Due to the short duration of this transient, heaters, makeup and letdown are unimportant.

Steam Line PORVs and Condenser Steam Dump

Steam line PORVs and steam dump to condenser are unimportant for this transient and are inoperable.

5.1.2 Core Cooling Capability Analysis

5.1.2.1 Nodalization

The core cooling capability analysis, which determines the minimum DNBR, is analyzed with three reactor coolant pumps in operation. A two-loop model with one single loop and one triple loop is utilized for this analysis. The standard model (Reference 2, Section 3.2) is used with one significant exception. Since this transient initiates at zero power, and since the duration of the transient is very short, the steam generator secondary response is not important. Rather than using the standard steam generator secondary nodalization, a single secondary volume is used. The single volume uses the bubble rise option with the local-conditions heat transfer model applied to the steam generator tube conductors. With this modeling approach the initial condition of zero power can be obtained, and the primary-to-secondary heat transfer that occurs following the power excursion can be simulated. No main or auxiliary feedwater or initial steam flow is modeled. ~~The pressurizer modeling includes the use of local conditions heat transfer option for the vessel conductors.~~

5.1.2.2 Initial Conditions

Core Power Level
A minimum initial power level typical of a critical, zero power startup condition maximizes the power excursion.

Pressurizer Pressure
Nominal pressure is assumed, with the pressure initial condition uncertainty accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Low initial pressurizer level minimizes the pressure increase following an insurge.

Reactor Vessel Average Temperature
The nominal temperature corresponding to zero power operation is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.

RCS Flow
Nominal three pump flow is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Level
Initial steam generator **level** is not an important parameter in this analysis.

Fuel Temperature
Due to the initial zero power condition, the initial fuel temperature is equal to T-ave. The fuel-clad gap conductivity is set high to maximize heat transfer from the fuel.

Steam Generator Tube Plugging
No tube plugging is assumed to maximize the RCS volume and thereby minimize the insurge into the pressurizer.

5.1.2.3 Boundary Conditions

Non-Conducting Heat Exchangers
For initialization purposes, non-conducting heat exchangers are used to remove reactor coolant pump heat since the steam generators are passive at initialization. These are turned off prior to the start of the power excursion.

RCP Operation

Since low flow is conservative for DNBR, the minimum number of reactor coolant pumps (three) required for the modes for which this transient is applicable (Modes 2 and 3) are assumed to be in operation.

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions to minimize RCS pressure during the transient.

Steam Line Safety Valves
Although not important for this transient, steam line safety valves are modeled with lift, accumulation, and blowdown assumptions to maximize primary-to-secondary heat transfer.

5.1.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
The pertinent reactor trip functions are the high power range flux (low setpoint) and pressurizer high pressure.

The high power range flux (low setpoint) trip includes a conservative allowance to account for calibration error, and error due to rod withdrawal effects. The response time of the high flux trip function is the Technical Specification value.

The response time of the pressurizer high pressure trip function is the Technical Specification value. The pressurizer high pressure reactor trip setpoint is the Technical Specification value plus an allowance which bounds the instrument uncertainty.

Pressurizer Pressure Control
Pressurizer spray and PORVs are operable to minimize RCS pressure during the transient. Heaters are not energized during the transient.

Steam Line PORVs and Condenser Steam Dump
Steam line PORVs and steam dump to condenser are unimportant for this transient and are inoperable.

5.1.2.5 Other Assumptions

Due to the potential for bottom-peaked power distributions during this transient, and due to the non-applicability of the Statistical Core Design Methodology below the mixing vane grids in the current fuel assembly designs, acceptable DNBRs are confirmed with the W-3S CHF correlation as necessary. Explicit accounting for uncertainties (i.e., non-SCD) **isare** used with the W-3S correlation.

5.2 Uncontrolled Bank Withdrawal at Power

The uncontrolled bank withdrawal at power accident is characterized by an increase in core power level that cannot be matched by the secondary heat sink. The resultant mismatch causes an increase in primary and secondary system temperatures and pressures. The increases in power and temperature, along with a change in the core power distribution, present a DNBR concern. The primary and secondary overpressure limits of 110% of design pressure are also of concern.

Peak RCS pressure~~, peak Main Steam System pressure~~ and core cooling capability are ~~each~~ analyzed separately due to the differences in assumptions required for a conservative analysis. The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

5.2.1 Peak RCS Pressure Analysis

5.2.1.1 Nodalization

Since the transient response of the uncontrolled bank withdrawal event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

5.2.1.2 Initial Conditions

Core Power Level
Initial pressurizer pressure and, thus, initial margin to the overpressurization limit ~~is~~are independent of initial power level. Due to the pressure overshoot during the reactor trip instrumentation delay, maximum pressure is achieved with the maximum pressurization rate. The maximum pressurization rate is achieved with the maximum insertion of reactivity, provided that reactor trip on high flux does not occur prior to significant system heatup. Since the initial margin to the high flux reactor trip is greatest at a low power level, this power level yields the most rapid insertion of reactivity with significant system heatup.

Pressurizer Pressure
Initial pressurizer pressure is the nominal value, and the uncertainty in pressure is accounted for in the high pressure reactor trip setpoint.

Pressurizer Level
High initial level minimizes the initial volume of the pressurizer steam space, which maximizes the transient primary pressure response.

Reactor Vessel Average Temperature
Initial temperature is not an important parameter in this analysis.

RCS Flow
Initial RCS flowrate is not an important parameter in this analysis.

Core Bypass Flow
Core bypass flow is not an important parameter in this analysis.

Steam Generator Level

~~High initial level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response. Maximum secondary pressurization causes maximum secondary temperature response, which minimizes primary-to-secondary heat transfer.~~**Initial steam generator level is not an important parameter in this analysis.**

Fuel Temperature

Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging

A bounding high tube plugging value degrades primary-to-secondary heat transfer.

5.2.1.3 Boundary Conditions

Pressurizer Safety Valves

The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize the pressurizer pressure.

Steam Line Safety Valves

The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure and minimize transient primary-to-secondary heat transfer.

5.2.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip

The pertinent reactor trip functions are the overtemperature ΔT (OTΔT), overpower ΔT (OPΔT), pressurizer high pressure and power range high flux (high setpoint).

The response time of each of the two ΔT trip functions is the Technical Specification value. The setpoint values of the ΔT trip functions are continuously computed from system parameters using the modeling described in Section 3.2 of Reference 2. In addition, the ΔT coefficients used in the analysis account for instrument uncertainties.

The response time of the pressurizer high pressure trip function is the Technical Specification value. The pressurizer high pressure reactor trip setpoint is the Technical Specification value plus an allowance which bounds the instrument uncertainty.

The response time of the power range high flux trip function is the Technical Specification value. The power range high flux trip high setpoint is the Technical Specification value plus an allowance which bounds the instrument uncertainty. The high flux signal is adjusted to account for the effects of bank withdrawal.

Pressurizer Pressure Control
In order to maximize primary system pressure, no credit is taken for pressurizer spray or PORV operation.

Pressurizer Level Control
Pressurizer level control system operation has negligible impact on the results of this analysis.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
Feedwater control is in automatic to prevent steam generator low-low level reactor trip.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
Auxiliary feedwater is disabled. The addition of subcooled auxiliary feedwater would tend to subcool the water in the steam generator, and provide better heat removal capability.

Turbine Trip
Turbine trip upon reactor trip is modeled in order to minimize the post-trip primary-to-secondary heat transfer.

5.2.2 Core Cooling Capability Analysis

5.2.2.1 Nodalization

Since the transient response of the uncontrolled bank withdrawal event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

5.2.2.2 Initial Conditions

Core Power Level
The uncontrolled bank withdrawal event is analyzed with a spectrum of initial power levels which range from low power to full power. Uncertainties in initial power level are accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
Initial pressurizer pressure is the nominal value, and the uncertainty in pressure is accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Initial pressurizer level is the nominal value which corresponds to the initial power level, and uncertainties are accounted for in the initial value. Low initial level maximizes the initial volume of the pressurizer steam space, which minimizes the transient primary pressure response.

Reactor Vessel Average Temperature
The nominal temperature corresponding to the initial power level is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.

RCS Flow
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Level
~~Initial steam generator level is the nominal value which corresponds to the initial power level, and uncertainties are accounted for in the initial value. High initial level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response. Maximum secondary pressurization causes maximum secondary temperature response, which minimizes primary-to-secondary heat transfer.~~ **Initial steam generator level is not an important parameter in this analysis.**

Fuel Temperature
Initial fuel temperature is the value which corresponds to the initial power level. Low fuel temperature maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
~~A bounding high tube plugging value degrades primary-to-secondary heat transfer.~~ **The bounding tube plugging assumption (high or low) varies depending on other initial and boundary conditions.**

5.2.2.3 Boundary Conditions

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which minimize the pressurizer pressure.

Steam Line Safety Valves
The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure and minimize transient primary-to-secondary heat transfer.

The acceptance criterion for this event is to ensure that there is adequate core cooling capability. The core cooling capability analysis determines to what extent fuel cladding integrity is compromised by calculating the number of fuel rods that exceed the 95/95 DNBR limit based on acceptable correlations.

5.4.1 Nodalization

Since the transient response of the single rod withdrawal event is the same for all loops, the single-loop model described in Section 3.2 of Reference 2 is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

5.4.2 Initial Conditions

Core Power Level
Initial power is the nominal full power value. Uncertainty in power level is accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
Initial pressurizer pressure is the nominal value. Uncertainty in pressure is accounted for in the Statistical Core Design Methodology.

Pressurizer Level
High initial level minimizes the initial volume of the pressurizer steam space, which maximizes the transient primary pressure response. Up to the limit of the ability of the pressurizer sprays to control pressure, maximum pressure is conservative in order to delay reactor trip on OTΔT.

Reactor Vessel Average Temperature
Initial temperature is the full power nominal value. Uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

RCS Flow
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Level
~~High initial level minimizes the initial volume of the steam generator steam space, which maximizes the transient secondary pressure response. Maximum secondary pressurization causes maximum secondary temperature response, which minimizes primary-to-secondary heat transfer.~~ Initial

steam generator level is not an important parameter in this analysis.

Fuel Temperature
Low fuel temperature, associated with high gap conductivity, maximizes the transient heat transfer from the fuel to the coolant.

Steam Generator Tube Plugging
~~A bounding high tube plugging value degrades primary-to-secondary heat transfer.~~**Steam generator tube plugging is not an important parameter in this analysis.**

5.4.3 Boundary Conditions

Pressurizer Safety Valves
The pressurizer safety valves are modeled with lift, accumulation, and blowdown assumptions which minimize the pressurizer pressure.

Steam Line Safety Valves
The steam line safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize transient secondary side pressure and minimize transient primary-to-secondary heat transfer.

5.4.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
The pertinent reactor trip functions are the overtemperature ΔT (OTΔT), overpower ΔT (OPΔT), pressurizer high pressure and power range high flux (high setpoint).

The response time of each of the two ΔT trip functions is the Technical Specification value. The setpoint values of the ΔT trip functions are continuously computed from system parameters using the modeling described in Section 3.2 of Reference 2. In addition, the ΔT coefficients used in the analysis account for instrument uncertainties.

The response time of the pressurizer high pressure trip function is the Technical Specification value. The pressurizer high pressure reactor trip setpoint is the Technical Specification value plus an allowance which bounds the instrument uncertainty.

The response time of the power range high flux trip function is the Technical Specification value. The power range high flux trip high setpoint is the Technical Specification value plus an allowance which bounds the instrument uncertainty. The high flux signal is adjusted to account for the effects of rod withdrawal.

Pressurizer Pressure Control
Pressurizer pressure control is in automatic with sprays enabled and PORVs disabled in order to delay reactor trip on OTΔT and delay reactor trip on high pressurizer pressure.

Pressurizer Level Control
Pressurizer level control is in manual with the pressurizer heaters disabled in order to delay reactor trip on high pressurizer pressure. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
Feedwater control is in automatic to prevent steam generator low-low level reactor trip.

Auxiliary Feedwater
Auxiliary feedwater is disabled. The addition of subcooled auxiliary feedwater would tend to subcool the water in the steam generator, and reduce secondary side pressure.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Turbine Trip
Turbine trip upon reactor trip is modeled in order to minimize the post-trip primary-to-secondary heat transfer.

5.5 Startup Of An Inactive Reactor Coolant Pump At An Incorrect Temperature

The McGuire and Catawba plant Technical Specifications currently require that all four RCPs be running at power operation. Furthermore, low flow in any RCS loop, coincident with reactor power above the P-8 interlock (currently at 48% of rated thermal power) will cause a reactor trip. Therefore, the only situation in which the subject accident is possible is a trip of one RCP below P-8. For this situation the operator might choose, during allowable at power outage time for the fourth RCP, to attempt a restart of the tripped pump. The accident is analyzed from the most conservative condition allowed by the Reactor Protection System, even though operator error is required for the analyzed scenario to occur. The acceptance criterion is that fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains the above the 95/95 DNBR limit based on acceptable correlations.

5.5.1 Nodalization

Because of the loop asymmetry between the inactive single loop and the three active loops, the double-loop RCS model described in Section 3.2 of Reference 2 is used. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

5.5.2 Initial Conditions

Core Power

Since the positive reactivity insertion due to the colder moderator average temperature causes a power increase, the initial indicated power level must be sufficiently less than the P-8 setpoint such that, by the time the indicated power level reaches the P-8 setpoint, the indicated flow in the loop containing the restarted pump is greater than the low flow reactor trip setpoint. This delays or prevents reactor trip and is therefore conservative for DNBR evaluation.

Pressurizer Pressure

A pressure initial condition uncertainty is applied to minimize pressure during the transient since this is conservative for DNBR evaluation.

Pressurizer Level

The heatup of the colder water and the increase in core power will cause an expansion of the reactor coolant and an increase in pressurizer level. A negative level uncertainty is used in order to maximize the size of the pressurizer steam bubble to be compressed, which minimizes the transient pressure response.

Reactor Vessel Average Temperature

A positive temperature uncertainty is used to minimize the margin to DNB.

RCS Flow

In order to minimize core flow, and therefore the margin to DNB, the three pump equivalent of the Technical Specification minimum measured flow is adjusted by a negative flow uncertainty.

Core Bypass Flow

A positive flow uncertainty is used to minimize the margin to DNB.

Steam Generator Level

The results of this transient are not sensitive to the direction of steam generator level uncertainty as long as the transient level response is kept within the range that avoids protection or safeguards actuation.

Fuel Temperature

A low initial temperature is assumed to maximize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A high gap conductivity minimizes the fuel heatup and attendant negative reactivity insertion caused by the power increase. This makes the power increase more severe and is therefore conservative for DNB evaluation.

Steam Generator Tube Plugging

Steam generator tube plugging is not an important parameter in this analysis.

6.0 INCREASE IN REACTOR COOLANT INVENTORY

6.1 Inadvertent Operation Of ECCS During Power Operation

The inadvertent operation of the Emergency Core Cooling System could be caused by either operator error or a spurious electrical actuation signal. Upon receipt of the actuation signal, the centrifugal charging pumps begin delivering highly borated refueling water storage tank water to the Reactor Coolant System. The resultant negative reactivity insertion causes a decrease in core power and, consequently, a decrease in temperature. **Initially,** ~~C~~coolant shrinkage causes a reduction in both pressurizer water level and pressure. **Core cooling capability** (DNB) is the primary concern ~~for this transient~~ **during this time period** due to ~~this~~**the** decrease in system pressure. **Following the initial depressurization, the increase in reactor coolant inventory causes pressurizer level to increase and pressurization to occur. Pressurizer level might increase sufficiently to overfill the pressurizer and cause water relief through the pressurizer safety valves (PSVs). Water relief through the PSVs could degrade valve operability and lead to a Condition III event.**

The magnitude of the pressure decrease for this transient is no more severe than that for the inadvertent opening of a pressurizer safety or relief valve transient, which also trips the reactor on low pressurizer pressure. Furthermore, the opening of a safety valve does not introduce the core power and Reactor Coolant System temperature decreases that are characteristic of the inadvertent ECCS actuation. Neither event involves any reduction in the Reactor Coolant System flowrate, since the reactor coolant pumps are not tripped. Therefore, the DNB results of this transient are bounded by the inadvertent opening of a pressurizer safety or relief valve transient.

Based on the above qualitative evaluation, a quantitative **core cooling capability** analysis of this transient is not required. Should a reanalysis become necessary, either due to plant changes, modeling changes, or other changes which invalidate any of the above arguments, the analytical methodology employed would be as follows.

The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

The concern in the pressurizer overfill analysis is that water relief through the PSVs will degrade valve operability and lead to a Condition III event. However, even if water relief occurs, valve operability is not degraded provided that the temperature of the pressurizer water is sufficiently high. Therefore, the acceptance criterion for this analysis is the minimum water relief temperature to assure PSV operability.

6.1.1 Core Cooling Capability Analysis

6.1.1.1 Nodalization

Since the inadvertent ECCS operation transient is symmetrical with respect to the four reactor coolant loops, a single-loop model (Reference 2, Section 3.2) is utilized for this analysis. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

6.1.1.2 Initial Conditions

<u>Core Power Level</u>
High initial power level maximizes the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

<u>Pressurizer Pressure</u>
Nominal full power pressurizer pressure is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

<u>Pressurizer Level</u>
High initial level minimizes the volume of the pressurizer steam space which maximizes the pressure decrease resulting from the outsurge.

<u>Reactor Vessel Average Temperature</u>
Nominal full power vessel average temperature is assumed. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

<u>RCS Flow</u>
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

<u>Core Bypass Flow</u>
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

<u>Steam Generator Level</u>
Steam generator level is not an important parameter in this analysis.

<u>Fuel Temperature</u>
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
Steam generator tube plugging is not an important parameter in this analysis.

6.1.1.3 Boundary Conditions

ECCS Flow
A maximum safety injection flowrate along with a conservatively high boron concentration yields the most limiting transient response. In order to minimize the delay in the delivery of the borated injection water, no credit is taken for the purge volume of unborated water in the injection lines.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

6.1.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip is assumed to occur on low pressurizer pressure, after an appropriate instrumentation delay.

Pressurizer Pressure Control
Pressurizer sprays and PORVs are assumed to be operable in order to minimize the system pressure throughout the transient.

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. Charging/letdown has negligible impact.

Steam Line PORVs and Condenser Steam Dump
Secondary steam relief via the steam line PORVs and the condenser steam dump is unavailable in order to maximize secondary side pressurization and minimize transient primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long **as** the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in ~~an~~ decrease in RCS temperature. With the Rod Control System in automatic, the control rods would cause a positive reactivity addition as they are withdrawn in an attempt to maintain RCS

temperature at its nominal value. The resultant power increase would retard the system depressurization.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip
The reactor trip leads to a subsequent turbine trip.

6.1.2 Pressurizer Overfill Analysis

6.1.2.1 Initial Conditions

Core Power
Zero power is assumed in this analysis. Reference 3 states that the acceptable initial power for the analysis is the licensed core thermal power, i.e., full power. However, lower power is more limiting in order to minimize the initial NC system temperature. If overfill occurs at lower initial power, then the water relief temperature is more likely to be less than the acceptance criterion.

Pressurizer Pressure
Actual system response to an safety injection (SI) would be an initial pressure drop then subsequent pressurization above initial pressure. During the depressurization phase, SI flow would increase above the initial flowrate, and during the pressurization phase, SI flow would decrease below initial flowrate. Initial pressure is assumed conservatively low to determine the SI flow during the event.

Reactor Vessel Average Temperature
Low initial temperature is conservative in order to minimize pressurizer water temperature.

Steam Generator Tube Plugging
High steam generator tube plugging is assumed in order to decrease the volume of the initial RCS water, which will minimize the RCS water temperature as it mixes with the cold SI water.

6.1.2.2 Boundary Conditions

RCP Operation
For Modes 1-3, the Technical Specifications require at least one reactor coolant pump be operating.

6.1.2.3 Control, Protection, and Safeguards System Modeling

Pressurizer Level Control
The pressurizer heaters are assumed to be in manual and off since heater operation would increase the temperature of the pressurizer water. Normal makeup is isolated upon SI, and credit is not taken for letdown.

ECCS Flow
A maximum safety injection flowrate from both centrifugal charging pumps is assumed. RCS pressure remains above the shutoff head of the intermediate head and low head safety injection pumps for the duration of the event.

ECCS Temperature
Minimum injection temperature is conservative in order to minimize relief temperature.

7.0 DECREASES IN REACTOR COOLANT INVENTORY

7.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The loss of inventory through the open valve causes a depressurization of the RCS. Since the core power, flow, and temperature are relatively unaffected prior to reactor trip by this depressurization, the reduction in pressure causes a reduction in DNB margin. The applicable acceptance criterion is that fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains the above the 95/95 DNBR limit based on acceptable correlations. The minimum DNBR is determined using the Statistical Core Design Methodology.

7.1.1 Nodalization

Since the valve opening is in the pressurizer, it affects all RCS loops identically. Therefore a single-loop RCS system model is used. ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

7.1.2 Initial Conditions

Power Level
Full power is assumed in order to maximize the primary system heat flux. The uncertainty in this parameter is accounted for in the Statistical Core Design Methodology.

Pressurizer Pressure
Nominal pressure is assumed, with the pressure initial condition uncertainty accounted for in the Statistical Core Design Methodology.

Pressurizer Level
Since this accident involves a reduction in RCS volume due to inventory loss, a positive level uncertainty is assumed to minimize the initial pressurizer steam bubble volume and therefore maximize the pressure decrease due to inventory loss

Reactor Vessel Average Temperature
The nominal temperature corresponding to full power operation is assumed, with the temperature initial condition uncertainty accounted for in the Statistical Core Design Methodology.

RCS Flow
The Technical Specification minimum measured flow for power operation is assumed since low flow is conservative for DNBR evaluation. The flow initial condition uncertainty is accounted for in the Statistical Core Design Methodology.

Core Bypass Flow
The nominal calculated flow is assumed, with the flow uncertainty accounted for in the Statistical Core Design Methodology.

Steam Generator Level
The results of this transient are not sensitive to the direction of steam generator level uncertainty as long as the transient level response is kept within the range that avoids protection or safeguards actuation.

Fuel Temperature
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease due to moderator density. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
The results of this analysis are not sensitive to the amount of steam generator tube plugging.

7.1.3 Boundary Conditions

Steam Line Safety Valves
The steam line safety valves are modeled with setpoint drift, accumulation, and blowdown assumptions which maximize the transient secondary pressure and therefore minimize secondary side heat removal.

7.1.4 Control, Protection, and Safeguards Systems Modeling

Reactor Trip
Reactor trip is on either low pressurizer pressure or overtemperature ΔT. The Technical Specification response times are used and the safety analysis setpoints include the effects of uncertainty in the monitored parameter and in the setpoint.

Pressurizer Pressure Control
No credit is taken for pressurizer heater operation to compensate for the decrease in pressurizer pressure which occurs due to the inventory loss. This results in a lower post-trip pressurizer pressure, which is conservative for DNBR evaluation.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Steam Generator Level Control

The results of this transient are not sensitive to the mode of steam generator level control as long as the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control

The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control

A penalty is taken for automatic rod control to insert positive reactivity to increase power and reactor vessel average temperature. These parameters would otherwise decrease in response to the negative reactivity inserted by the moderator density reduction.

Turbine Control

The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater

AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip

The turbine is tripped on reactor trip. No time delay is assumed since this assumption minimizes post-trip primary-to-secondary heat removal.

7.2 Steam Generator Tube Rupture

The steam generator tube rupture analyzed is a double ended guillotine break of a single tube. This transient is evaluated in two parts; first to evaluate minimum DNBR, and secondly to provide offsite dose input data for a separate evaluation to determine whether the fission product release to the environment is within the established dose acceptance criteria.

The DNBR analysis for this transient is modeled as a complete loss of coolant flow event initiated from an off-normal condition, using the Statistical Core Design methodology. The loss of flow is assumed to occur subsequent to the OTΔT reactor trip caused by the steam generator tube rupture depressurization.

The initiating event for the offsite dose input analysis is the double-ended guillotine break of a single steam generator tube. This analysis generates the offsite steam release boundary condition for the dose evaluation. The single failure identified for maximizing offsite dose is the failure of the PORV on the ruptured steam generator to close. In this analysis, this valve remains open until operator action is taken to

Fuel Temperature
A high initial temperature is assumed to minimize the gap conductivity calculated for steady-state conditions and used for the subsequent transient. A low gap conductivity minimizes the transient change in fuel rod surface heat flux associated with a power decrease. This makes the power decrease less severe and is therefore conservative for DNBR evaluation.

Steam Generator Tube Plugging
For transients of such short duration, steam generator tube plugging does not have an effect on the transient results.

7.2.1.3 Boundary Conditions

RCP Operation
All four reactor coolant pumps are tripped on the loss of offsite power. The pump model is adjusted such that the resulting coastdown flow is conservative with respect to the flow coastdown test data.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which maximize secondary side pressure and minimize primary-to-secondary heat transfer.

Offsite Power
Offsite power is assumed to be lost coincident with turbine trip **in order to minimize RCS flow following reactor trip.** ~~This isolates steam flow to the condenser, thereby maximizing the atmospheric steam releases.~~

7.2.1.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
Reactor trip is assumed to occur on overtemperature ΔT, after an appropriate instrumentation delay.

Pressurizer Pressure Control
~~Pressurizer sprays and PORVs are assumed to be operable in order to minimize the system pressure throughout the transient.~~ **Following the tube rupture, RCS pressure continuously decreases through the time at which minimum DNBR occurs. Thus, pressurizer sprays are not activated nor are the pressurizer PORVs challenged during the transient.**

Pressurizer Level Control
Pressurizer heaters are assumed to be inoperable so that Reactor Coolant System pressure is minimized. ~~Charging/letdown has negligible impact.~~ **Charging and letdown are assumed to be balanced at all times during the event with no action taken to increase charging flow due to RCS pressure and pressurizer level decreasing. This will maximize the RCS depressurization rate.**

Steam Line PORVs and Condenser Steam Dump
The main steam PORVs and condenser dumps valves are assumed to be unavailable during this transient. This maximizes the secondary side pressure and temperature and therefore reduces primary-to-secondary heat transfer.

Steam Generator Level Control
The results of this transient are not sensitive to the mode of steam generator level control as long **as** the level is kept within the range that avoids protection or safeguards actuation.

MFW Pump Speed Control
The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoids protection or safeguards actuation.

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in an increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control
The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2.

Auxiliary Feedwater
AFW flow would be credited when the safety analysis value of the low-low steam generator level setpoint is reached. However, the parameter of interest for this transient has reached its limiting value before the appropriate Technical Specification response time delay has elapsed. Therefore, no AFW is actually delivered to the steam generators.

Turbine Trip
The reactor trip leads to a subsequent turbine trip.

7.2.2 Offsite Dose Calculation Input Analysis

7.2.2.1 Nodalization

Due to the asymmetry of this transient a ~~two~~**three**-loop model, with ~~a~~**two** single loop**s** and a ~~triple~~**double** loop, is utilized for this analysis. The boundary conditions for the two intact steam generators **with operable steam line PORVs** are symmetric.~~symmetric, enabling the use of the two loop model.~~ **The loop with the tube rupture requires separate modeling, as does the loop with the inoperable steam line PORV.** ~~The pressurizer modeling includes the use of the local conditions heat transfer option for the vessel conductors.~~

7.2.2.2 Initial Conditions

Core Power Level
High initial core power and a positive uncertainty maximize the primary system heat load.

Pressurizer Pressure
High initial pressure ~~and~~ **with** a positive uncertainty delays the time of reactor trip. This retards the primary system cooldown, extending primary-to-secondary leakage, and therefore maximizing the offsite dose.

Pressurizer Level
High initial level with a positive uncertainty maximizes primary-to-secondary leakage and maximizes pressurizer heater operation.

Reactor Vessel Average Temperature
Nominal vessel average temperature with a negative uncertainty is used to minimize the initial steam generator steam pressure. This maximizes the initial differential pressure across the steam generator tubes and therefore maximizes the initial primary-to-secondary leakage. **A lower vessel average temperature also maximizes the initial mass flow rate through the ruptured tube.**

RCS Flow
Nominal primary system loop flow with a negative uncertainty is assumed. Low forced circulation flow results in lower natural circulation flow during the post-trip cooldown. This reduces primary-to-secondary heat transfer and extends plant cooldown. **Frictional and form losses will also be smaller throughout the RCS, resulting in a higher primary pressure at the break location. This maximizes primary to secondary leakage.**

Core Bypass Flow
Core bypass flow is not an important parameter for this transient.

Steam Generator Level
Minimum steam generator level reduces the initial secondary inventory available to mix with and dilute the primary-to-secondary leakage. **This also minimizes the secondary side static head at the break location, thus maximizing primary to secondary leakage.**

Fuel Temperature
High initial fuel temperature maximizes the stored energy which must be removed during the post-trip natural circulation cooldown.

Steam Generator Tube Plugging
~~A bounding high tube plugging level minimizes the initial steam generator steam pressure and therefore maximizes the pressure differential between the primary and secondary systems.~~ **Steam generator tube plugging is not an important parameter in this analysis.**

7.2.2.3 Boundary Conditions

Single Failure
The single failure identified for maximizing offsite dose is the failure of the PORV on the ruptured steam generator to close. In this analysis, this valve remains open until operator action is taken to isolate the PORV. Per Reference 4, page 5-7, "The most limiting failure would be the loss of air supply or power which prevents actuation of the (PORVs) from the main control room. The valves could be operated (locally) by manual action to correct for this single failure." This failure is incorporated into the analysis as it prolongs the transient, maximizing the primary-to-secondary leakage.

Pressurizer Safety Valves
The pressurizer code safety valves are not challenged during the course of this transient.

Steam Line Safety Valves
The main steam code safety valves are modeled with lift, accumulation, and blowdown assumptions which ~~minimize secondary pressure and maximize atmospheric steam releases~~**maximize secondary pressure. This delays operator identification of the failed open steam line PORV.**

Steam Line PORVs
Only two of the three steam line PORVs on the intact steam generators are assumed to be operable. This lengthens the cooldown time, thereby maximizing the atmospheric steam releases. A negative bias is applied to the ruptured steam generator PORV control signal. This results in an earlier opening time which maximizes atmospheric releases and delays operator identification of the failed open steam line PORV. A positive bias is applied to the intact SG PORV control signals to maximize secondary side post-trip pressurization. This delays operator identification of the failed open steam line PORV.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

Offsite Power
Offsite power is assumed to be lost coincident with turbine trip. This isolates steam flow to the condenser, thereby maximizing the atmospheric steam releases.

Break Model
The break is assumed to be a double-ended guillotine break of a single steam generator tube at the tubesheet surface on the steam generator outlet plenum. This location maximizes the mass flow through the break.

RCP Operation
The reactor coolant pumps are assumed to operate normally until offsite power is lost coincident with turbine trip.

ECCS Injection
SI actuation is assumed to occur on low pressurizer pressure at a setpoint with an applied positive uncertainty **or on manual operator action**. Maximum ECCS injection flow is assumed to maximize the primary-to-secondary leakage.

Main Feedwater
Main feedwater flow is assumed to terminate coincident with the loss of offsite power to minimize the secondary inventory available to mix with and dilute primary-to-secondary leakage.

Charging Flow
A conservatively high charging flow capacity is modeled to delay reactor trip and maximize total primary-to-secondary leakage.

Manual Actions

- Immediate action to maximize charging flow (penalty).

- Immediate action to energize pressurizer heater banks (penalty).

- **Operators identify the abnormal condition of the RCS at 20 minutes and manually trip the reactor if not already tripped.**

- Identify and isolate ruptured steam generator consistent with assumptions in WCAP-10698 (Reference 5), ~~10~~**15** minute minimum delay (credit).

- **Isolate failed open steam line drains upstream of the main steam isolation valves. This action occurs 10 minutes after the ruptured steam generator is identified.**

- Isolate the steam supply to the turbine-driven auxiliary feedwater pump from the ruptured steam generator after identification of the ruptured steam generator. An operator action delay time of ~~5~~**30** minutes is assumed (credit).

- Isolate failed open steam line PORV on the ruptured steam generator with an operator action delay time from when it should have closed normally. The delay times assumed are ~~5~~**10** minutes for control room and ~~15~~**30** minutes for local operation (credit).

- Manually control auxiliary feedwater to maintain zero power steam generator levels (nominal).

- Using the steam line PORVs, initiate natural circulation cooldown of the primary system after identification of the ruptured steam generator. Operator action delay times of ~~5~~**15** minutes for control room action and ~~10~~**45** minutes for local action are assumed (credit).

- Initiate depressurization of the primary system using the pressurizer PORVs to terminate break flow ~~2~~**10** minutes after the primary system is **20°F** subcooled at the ruptured steam generator pressure~~, or 2 minutes after the cooldown has been completed~~ (credit).

— ~~Terminate safety injection flow when pressurizer level recovers with a conservative delay (penalty).~~

— ~~Manually control charging flow after safety injection termination to maintain the zero power pressurizer level (nominal).~~

7.2.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
A reactor trip occurs on either low pressurizer pressure or **manual operator action at 20 minutes. A negative uncertainty is applied to the low pressurizer pressure trip setpoint to delay reactor trip. The overtemperature ΔT trip function is not credited.** ~~overtemperature ΔT. The Technical Specification response times are used and the safety analysis setpoints include the effects of uncertainty in the monitored parameter and in the setpoint~~

Pressurizer Pressure Control
This control system is assumed to be in manual and therefore is not modeled. Operator action is assumed to energize the pressurizer heaters and control the PORVs. Pressurizer spray is not available for the duration of this transient.

Pressurizer Level Control
This control system is assumed to be in manual and therefore is not modeled. Operator action is assumed to maximize charging flow.

~~Steam Line PORVs and~~ Condenser Steam Dump
~~The main steam PORVs are assumed to be operable for this transient with a positive bias applied to the control signal. This assumption minimizes secondary pressure and maximizes atmospheric steam releases.~~ The condenser steam dump valves are not assumed to be operable. Condenser steam dump would nonconservatively minimize offsite doses.

Steam Generator Level Control
This control system is assumed to operate to maintain the initial steam generator level prior to reactor trip.

Main Feedwater Pump Speed Control
~~This control system is assumed to operate to maintain the initial steam generator level prior to reactor trip.~~**The results of this transient are not sensitive to the mode of MFW pump speed control as long as the steam generator level is kept within the range that avoid protection or safeguards actuation.**

Rod Control
No credit is taken for the operation of the Rod Control System for this transient, which results in a slight increase in RCS temperature. With the Rod Control System in automatic, the control rods would cause a negative reactivity addition as they are inserted in an attempt to maintain RCS temperature at its nominal value.

Turbine Control

The turbine is modeled in the load control mode, which is described in Section 3.2.5.1 of Reference 2. **Turbine trip on reactor trip is delayed by 0.3 seconds to maximize primary to secondary leakage.**

Auxiliary Feedwater

Auxiliary feedwater initiation occurs after the loss of offsite power with a delay, consistent with Technical Specifications. **If applicable, a**~~A~~ purge volume of hot water is assumed to be delivered before cold feedwater reaches the steam generators. Minimum flow rates are assumed to minimize primary-to-secondary heat transfer.

MSIV Closure

Automatic MSIV closure is assumed using both a dynamically compensated and a static steam line pressure signal. Early closure maximizes the primary leakage released to the atmosphere through the failed open steam line PORV.

Table 8-1
Accident Analysis Assumptions

FSAR Section	15.1.2	15.1.3	15.2.3	15.2.3	15.2.6	15.2.6	15.2.6	15.2.6	15.2.7	15.2.7
Report Section	2.2	2.3	3.1.1	3.1.2	3.2.1	3.2.2	3.2.3	**3.2.4**	3.3.1	3.3.2
Power	Nominal	Nominal	High	High	High	High	Nominal	**High**	High	High
Pzr Pressure	Nominal	Nominal	High	High	High	**	Nominal	**	High	**
Pzr Level	High	High	High	High	High	**	Low	**	High	**
RCS Temp	Nominal	Nominal	High	High	High	High	Nominal	**High**	High	High
RCS Flow	Nominal	Nominal	Low	High	Low	High	Nominal	**	Low	High
Bypass Flow	Nominal	Nominal	**	**	**	**	Nominal	**	**	**
SG Level	Low	**	High	High	**	High	**	**High**	~~High~~Low	~~High~~Low
Fuel Temp	Low	Low	Low	Low	Low	Low	High	**	Low	Low
SG Tube Plugging	None	None	High	None	High	None	**	**High**	High	None
Pzr Spray	-	-	Off	Auto	Off	**	Auto	**Off**	Off	**
Pzr Heaters	Off	Off	~~Auto~~On	~~Auto~~On	Auto	**	Off	**Off**	Auto	**
Pzr PORVs	-	-	Closed	Auto	Closed	**	Auto	**Closed**	Closed	**
SM PORVs	-	-	Closed	Closed	Closed	Closed	Closed	**Closed**	Closed	Closed
Steam Dump	-	-	Closed	Closed	Closed	Closed	Closed	**Closed**	Closed	Closed
SG Level	-	**	*	*	-	-	-	-	-	-
MFW Pump Speed	-	**	*	*	-	-	-	-	-	-
Rod Control	*	*	Manual	Manual	-	-	-	-	Manual	Manual
Turbine Control	Auto	Auto	-	-	**	**	**	**	Auto	Auto
SI Signal	-	-	-	-	-	-	-	-	-	-
SI Flow	-	-	-	-	-	-	-	-	-	-
SI Delay	-	-	-	-	-	-	-	-	-	-
AFW Signal	~~SG Lvl~~**	**	*	*	LOSP	LOSP	LOSP	**LOSP**	SG Lvl	SG Lvl
AFW Flow	~~Min~~**	**	*	*	Min	Min	Min	**Min**	Min	Min
AFW Delay	~~TS~~**	**	*	*	TS	TS	TS	**TS**	TS	TS
Turb Trip Signal	SG Lvl	-	-	-	LOSP	LOSP	LOSP	**LOSP**	Rx Trip	Rx Trip
Turb Trip Delay	TS	-	None	None	None	None	None	**None**	None	None
Stm Line Isol Signal	-	-	-	-	-	-	-	-	-	-
Stm Line Isol Delay	-	-	-	-	-	-	-	-	-	-
MFW Isol Signal	SG Lvl	-	*	*	-	-	-	-	-	-
MFW Isol Delay	TS	-	-	-	-	-	-	-	-	-

Notes:
- * Refer to the text discussion of this transient.
- ** Results of the transient are insensitive to the choice about this parameter.
- \- Not applicable, either because the transient does not challenge that contol system or because the malfunction of that system might be the cause of the transient.

Table 8-1 (continued)
Accident Analysis Assumptions

FSAR Section	15.2.7	15.2.8	15.2.8	15.3.1	15.3.2	15.3.3	15.3.3	15.4.1	15.4.1	15.4.2
Section	3.3.3	3.4.1	3.4.2	4.1	4.2	4.3.1	4.3.2	5.1.1	5.1.2	5.2.1
Power	Nominal	Nominal	High	Nominal	Nominal	High	~~High~~**Nominal**	0	~~Nominal~~0	*
Pzr Pressure	Nominal	Nominal	Low	Nominal	Nominal	High	~~Low~~**Nominal**	High	Nominal	High
Pzr Level	Low	Low	Low	Low	Low	High	Low	High	Low	High
RCS Temp	Nominal	Nominal	High	Nominal	Nominal	High	~~High~~**Nominal**	**	Nominal	**
RCS Flow	Nominal	Nominal	Low	Nominal	Nominal	Low	~~Low~~**Nominal**	**	Nominal	**
Bypass Flow	Nominal	Nominal	**	Nominal	Nominal	High	~~High~~**Nominal**	**	Nominal	**
SG Level	~~High~~**Low**	**	Low	**	**	**	**	**	**	~~High~~**
Fuel Temp	High	High	High	High	High	Low	High	Low	Low	Low
SG Tube Plugging	High	**	~~High~~**	**	**	**	**	High	None	High
Pzr Spray	Auto	Auto	Auto	Auto	Auto	Off	Auto	Off	Auto	Off
Pzr Heaters	Off	Off	Off	Off	Off	Auto	Off	**	Off	**
Pzr PORVs	Auto	Auto	Closed	Auto	Auto	Closed	Auto	Closed	Auto	Closed
SM PORVs	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
Steam Dump	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
SG Level	-	-	-	**	**	**	**	-	-	Auto
MFW Pump Speed	-	-	-	**	**	**	**	-	-	**
Rod Control	Manual	Manual	Manual	Manual	Manual	Manual	Manual	-	-	-
Turbine Control	Auto	Auto	Auto	Auto	Auto	Auto	Auto	-	-	Auto
SI Signal	-	-	*	-	-	-	-	-	-	-
SI Flow	-	-	Min	-	-	-	-	-	-	-
SI Delay	-	-	TS	-	-	-	-	-	-	-
AFW Signal	SG Lvl	**	SI	**	**	**	**	**	**	*
AFW Flow	Min	**	Min	**	**	**	**	**	**	*
AFW Delay	TS	**	TS	**	**	**	**	**	**	*
Turb Trip Signal	Rx Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	Rx Trip	-	-	Rx Trip
Turb Trip Delay	None	None	None	None	None	None	None	-	-	None
Stm Line Isol Signal	-	-	*	-	-	-	-	-	-	-
Stm Line Isol Delay	-	-	*	-	-	-	-	-	-	-
MFW Isol Signal	-	**	-	**	**	**	**	-	-	**
MFW Isol Delay	-	**	-	**	**	**	**	-	-	**

Notes:

- * Refer to the text discussion of this transient.
- ** Results of the transient are insensitive to the choice about this parameter.
- \- Not applicable, either because the transient does not challenge that contol system or because the malfunction of that system might be the cause of the transient.

Table 8-1 (continued)
Accident Analysis Assumptions

FSAR Section	15.4.2	15.4.3c	15.4.3d	15.4.4	15.4.6	15.4.7	15.5.1	**15.5.1**	15.6.1	15.6.3	15.6.3
Section	5.2.2	5.3	5.4	5.5	5.6	5.7	6.1.1	**6.1.2**	7.1	7.2.1	7.2.2
Power	Nominal	-	Nominal	*	-	-	Nominal	**0**	Nominal	Nominal	High
Pzr Pressure	Nominal	-	Nominal	Low	-	-	Nominal	**Low**	Nominal	Nominal	High
Pzr Level	Low	-	High	Low	-	-	High	**Low**	High	Low	High
RCS Temp	Nominal	-	Nominal	High	-	-	Nominal	**Low**	Nominal	Nominal	Low
RCS Flow	Nominal	-	Nominal	Low	-	-	Nominal	-	Nominal	Nominal	Low
Bypass Flow	Nominal	-	Nominal	High	-	-	Nominal	-	Nominal	Nominal	**
SG Level	~~High~~**	-	~~High~~**	**	-	-	**	-	**	**	Low
Fuel Temp	Low	-	Low	Low	-	-	High	-	High	High	High
SG Tube Plugging	~~High~~*	-	~~High~~**	**	~~-~~**High**	-	**	**High**	**	**	~~High~~**
Pzr Spray	*	-	Auto	Auto	-	-	Auto	-	-	~~Auto~~ -	Off
Pzr Heaters	Off	-	Off	Off	-	-	Off	**Off**	Off	Off	Manual
Pzr PORVs	*	-	Closed	Auto	-	-	Auto	-	-	~~Auto~~ -	Manual
SM PORVs	Closed	-	Closed	-	-	-	Closed	-	Closed	Closed	*
Steam Dump	Closed	-	Closed	-	-	-	Closed	-	Closed	Closed	Closed
SG Level	Auto	-	Auto	Auto	-	-	**	-	**	**	Auto
MFW Pump Speed	**	-	**	Auto	-	-	**	-	**	**	~~Auto~~**
Rod Control	-	-	-	Auto	-	-	Manual	-	Auto	Manual	~~**~~Manual
Turbine Control	Auto	-	Auto	Manual	-	-	Auto	-	Auto	Auto	Auto
								-			
SI Signal	-	-	-	-	-	-	-	-	**	-	~~Low Pzr Press~~*
SI Flow	-	-	-	-	-	-	Max	**Max**	**	-	Max
SI Delay	-	-	-	-	-	-	-	-	**	-	None
AFW Signal	*	-	*	**	-	-	**	-	**	**	LOSP
AFW Flow	*	-	*	**	-	-	**	-	**	**	Min
AFW Delay	*	-	*	**	-	-	**	-	**	**	TS
Turb Trip Signal	Rx Trip	-	Rx Trip	-	-	-	Rx Trip	-	Rx Trip	Rx Trip	Rx Trip
Turb Trip Delay	None	-	None	-	-	-	None	-	None	None	~~None~~*
Stm Line Isol Signal	-	-	-	-	-	-	-	-	-	-	~~-~~*
Stm Line Isol Delay	-	-	-	-	-	-	-	-	-	-	~~-~~*
MFW Isol Signal	**	-	**	-	-	-	-	-	**	**	LOSP
MFW Isol Delay	**	-	**	-	-	-	-	-	**	**	None

Notes:
- * Refer to the text discussion of this transient.
- ** Results of the transient are insensitive to the choice about this parameter.
- \- Not applicable, either because the transient does not challenge that contol system or because the malfunction of that system might be the cause of the transient.

Table 8-1 (continued)
Accident Analysis Assumptions

FSAR Section	15.4.2	15.4.3c	15.4.3d	15.4.4	15.4.6	15.4.7	15.5.1	**15.5.1**	15.6.1	15.6.3	15.6.3
Section	5.2.2	5.3	5.4	5.5	5.6	5.7	6.1.1	**6.1.2**	7.1	7.2.1	7.2.2
Power	Nominal	-	Nominal	*	-	-	Nominal	**0**	Nominal	Nominal	High
Pzr Pressure	Nominal	-	Nominal	Low	-	-	Nominal	**Low**	Nominal	Nominal	High
Pzr Level	Low	-	High	Low	-	-	High	**Low**	High	Low	High
RCS Temp	Nominal	-	Nominal	High	-	-	Nominal	**Low**	Nominal	Nominal	Low
RCS Flow	Nominal	-	Nominal	Low	-	-	Nominal	-	Nominal	Nominal	Low
Bypass Flow	Nominal	-	Nominal	High	-	-	Nominal	-	Nominal	Nominal	**
SG Level	~~High**~~	-	~~High**~~	**	-	-	**	-	**	**	Low
Fuel Temp	Low	-	Low	Low	-	-	High	-	High	High	High
SG Tube Plugging	~~High*~~	-	~~High**~~	**	~~-~~**High**	-	**	**High**	**	**	~~High**~~
Pzr Spray	*	-	Auto	Auto	-	-	Auto	-	-	~~Auto~~ -	Off
Pzr Heaters	Off	-	Off	Off	-	-	Off	**Off**	Off	Off	Manual
Pzr PORVs	*	-	Closed	Auto	-	-	Auto	-	-	~~Auto~~ -	Manual
SM PORVs	Closed	-	Closed	-	-	-	Closed	-	Closed	Closed	*
Steam Dump	Closed	-	Closed	-	-	-	Closed	-	Closed	Closed	Closed
SG Level	Auto	-	Auto	Auto	-	-	**	-	**	**	Auto
MFW Pump Speed	**	-	**	Auto	-	-	**	-	**	**	~~Auto**~~
Rod Control	-	-	-	Auto	-	-	Manual	-	Auto	Manual	~~**~~Manual
Turbine Control	Auto	-	Auto	Manual	-	-	Auto	-	Auto	Auto	Auto
								-			
SI Signal	-	-	-	-	-	-	-	-	**	-	~~Low Pzr Press*~~
SI Flow	-	-	-	-	-	-	Max	**Max**	**	-	Max
SI Delay	-	-	-	-	-	-	-	-	**	-	None
AFW Signal	*	-	*	**	-	-	**	-	**	**	LOSP
AFW Flow	*	-	*	**	-	-	**	-	**	**	Min
AFW Delay	*	-	*	**	-	-	**	-	**	**	TS
Turb Trip Signal	Rx Trip	-	Rx Trip	-	-	-	Rx Trip	-	Rx Trip	Rx Trip	Rx Trip
Turb Trip Delay	None	-	None	-	-	-	None	-	None	None	~~None*~~
Stm Line Isol Signal	-	-	-	-	-	-	-	-	-	-	—*
Stm Line Isol Delay	-	-	-	-	-	-	-	-	-	-	—*
MFW Isol Signal	**	-	**	-	-	-	-	-	**	**	LOSP
MFW Isol Delay	**	-	**	-	-	-	-	-	**	**	None

Notes:
* Refer to the text discussion of this transient.
** Results of the transient are insensitive to the choice about this parameter.
\- Not applicable, either because the transient does not challenge that contol system or because the malfunction of that system might be the cause of the transient.