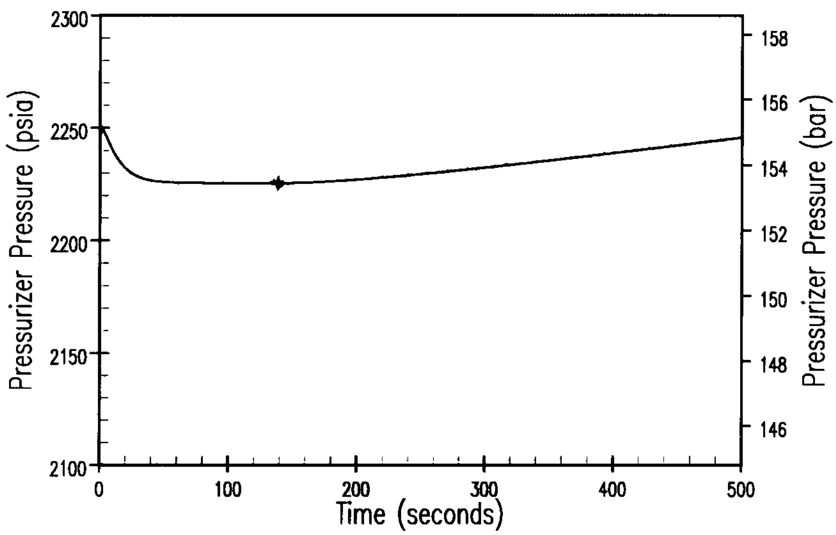

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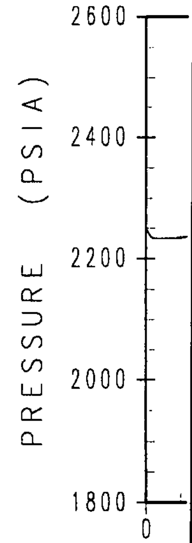

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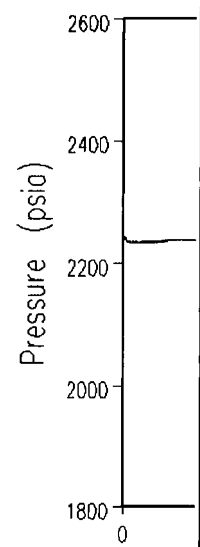

Figure 15.1.3-17

Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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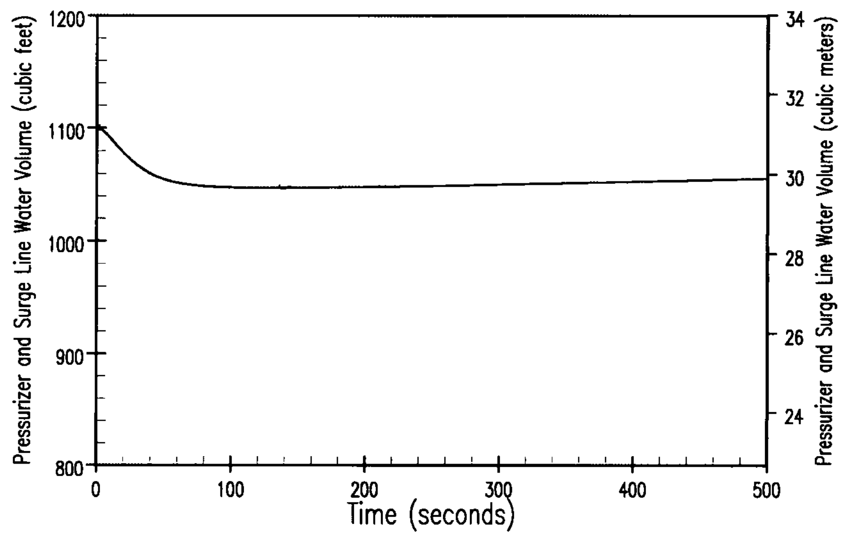

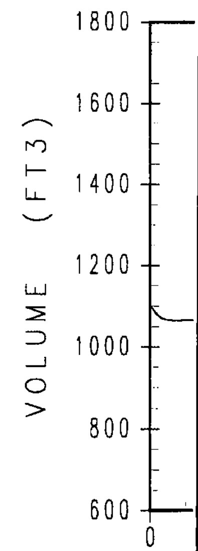

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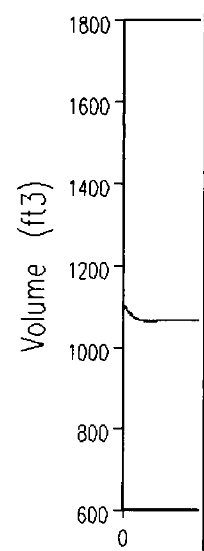

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Figure 15.1.3-18

Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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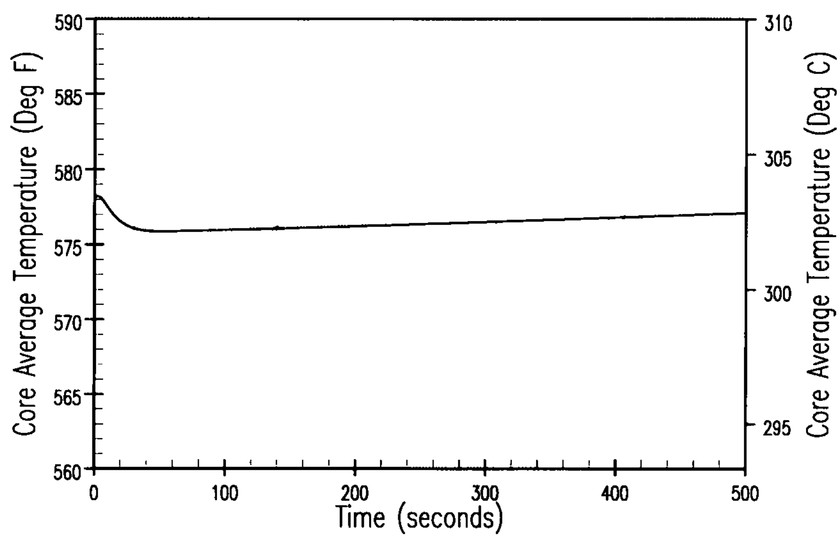

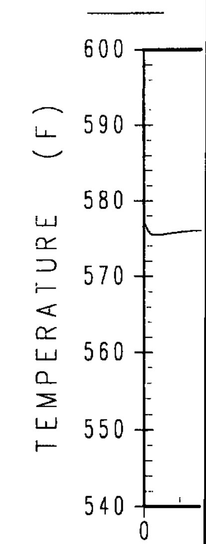

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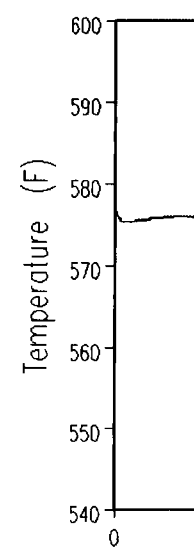

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Figure 15.1.3-19

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Core Average Temperature Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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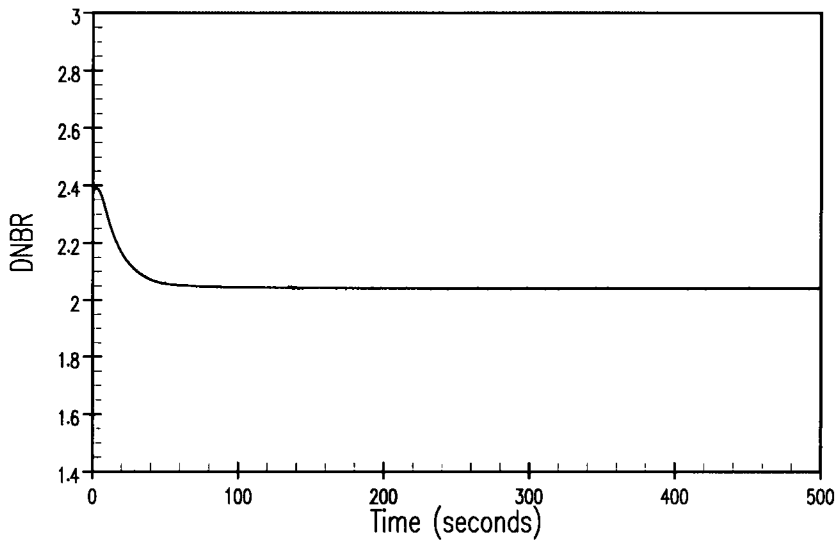

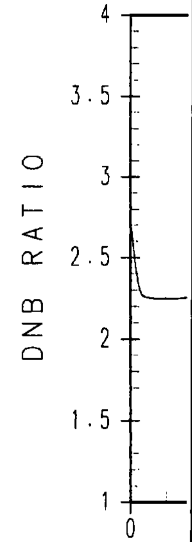

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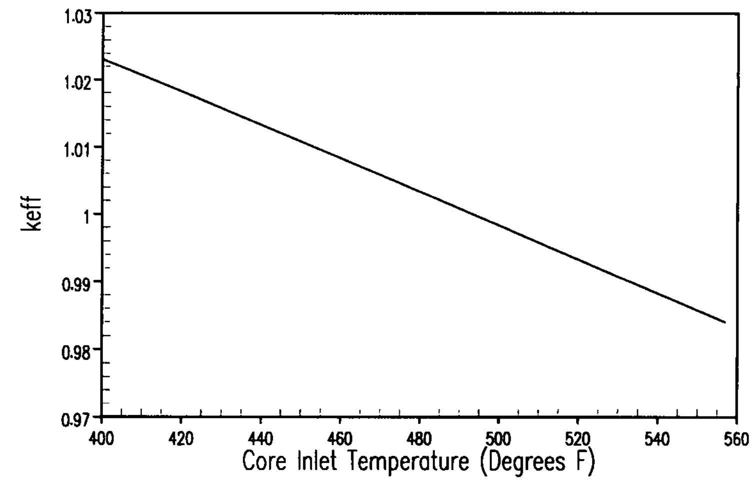
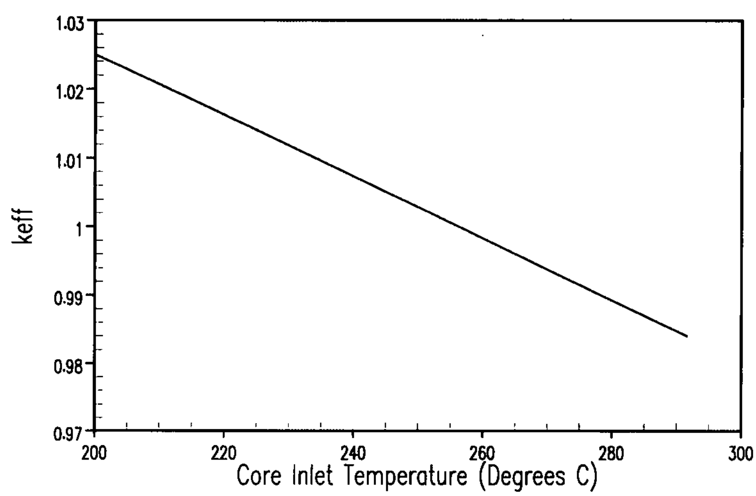
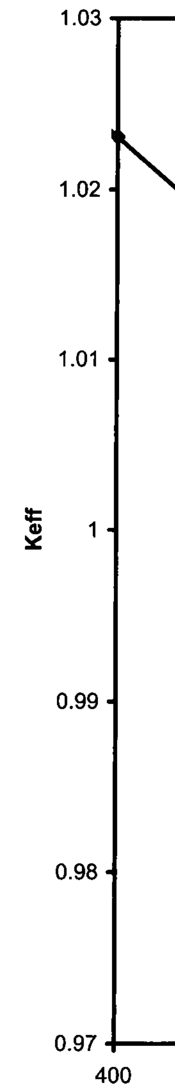
Figure 15.1.3-20

DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback







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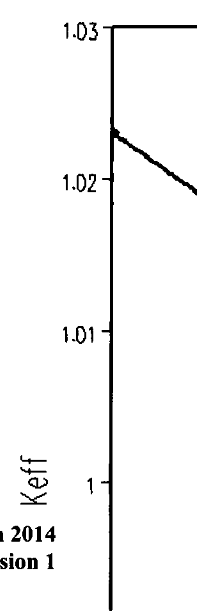
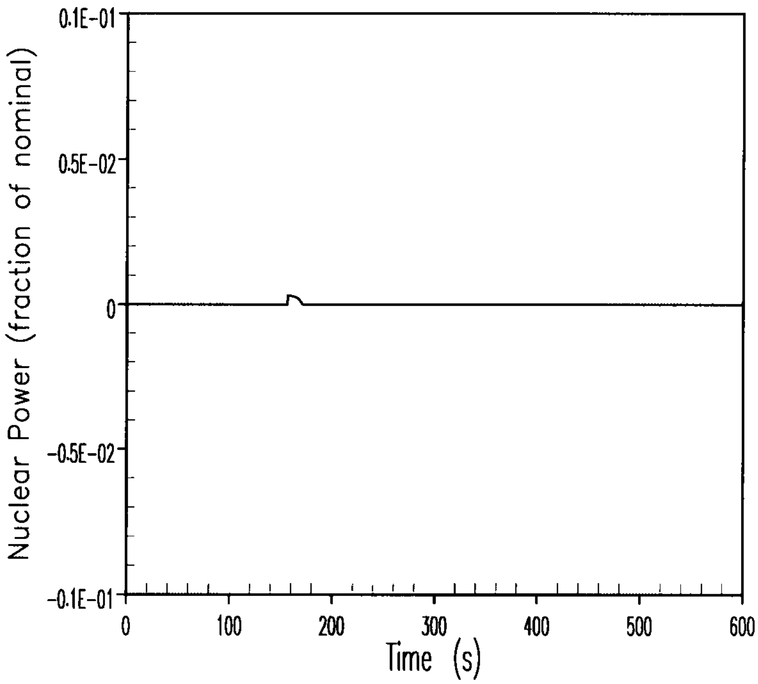


Figure 15.1.4-1

K_{eff} Versus Core Inlet Temperature
Steam Line Break Events



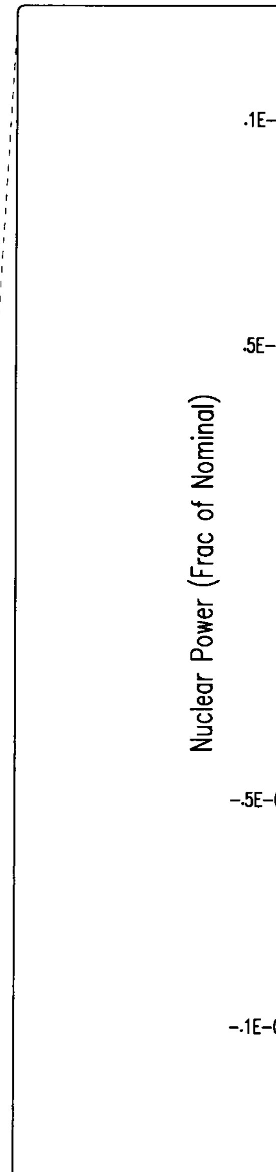


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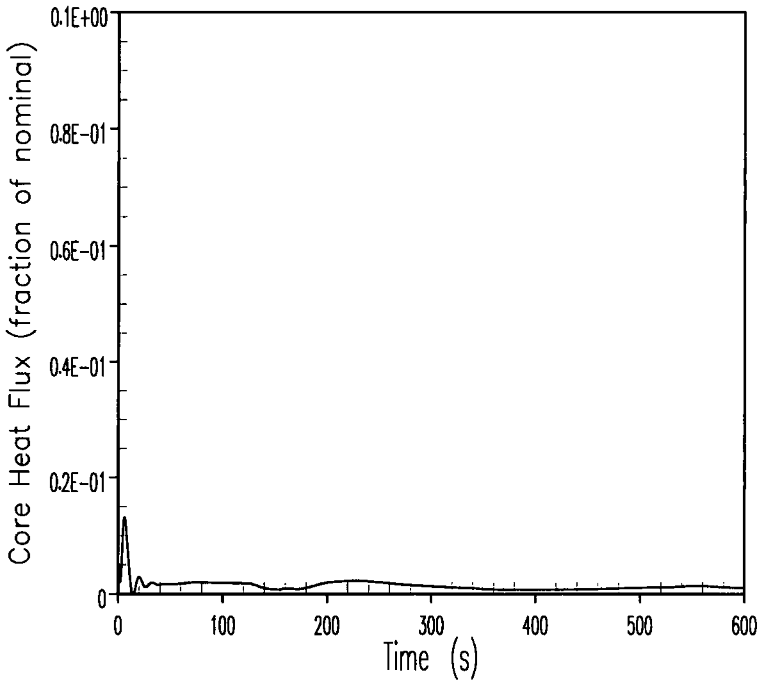
Figure 15.1.4-2

Nuclear Power Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



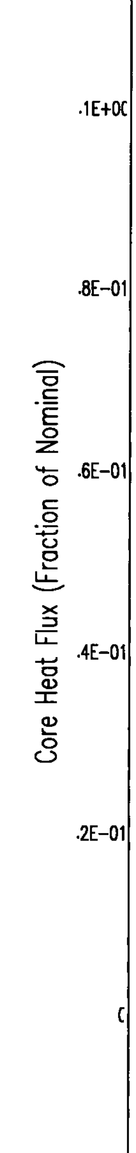


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Figure 15.1.4-3

Core Heat Flux Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

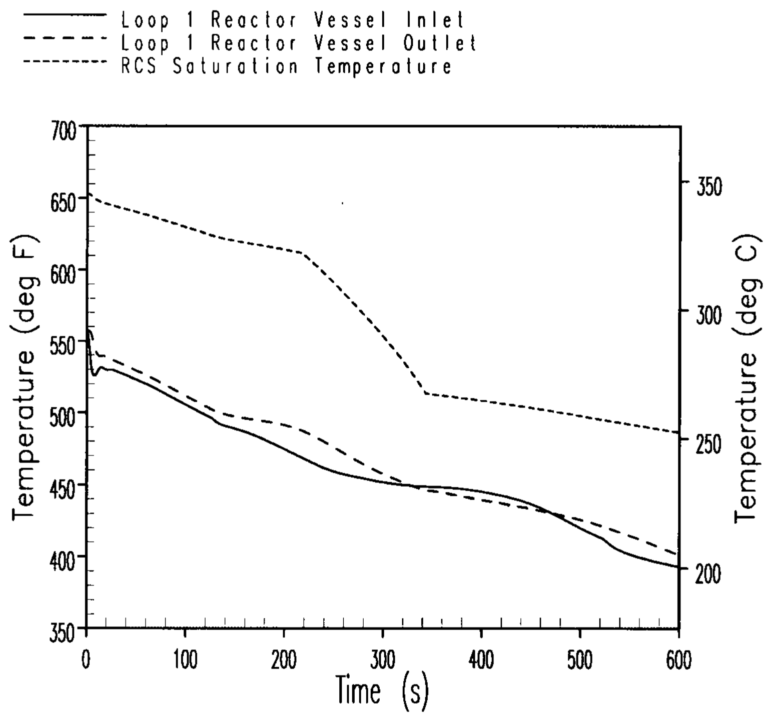

Figure 15.1.4-4

Loop 1 Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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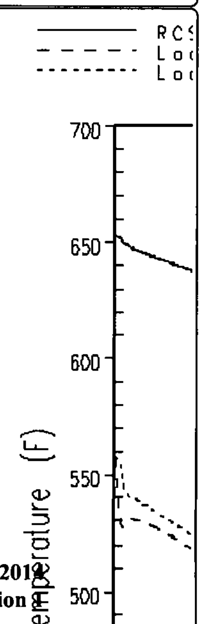


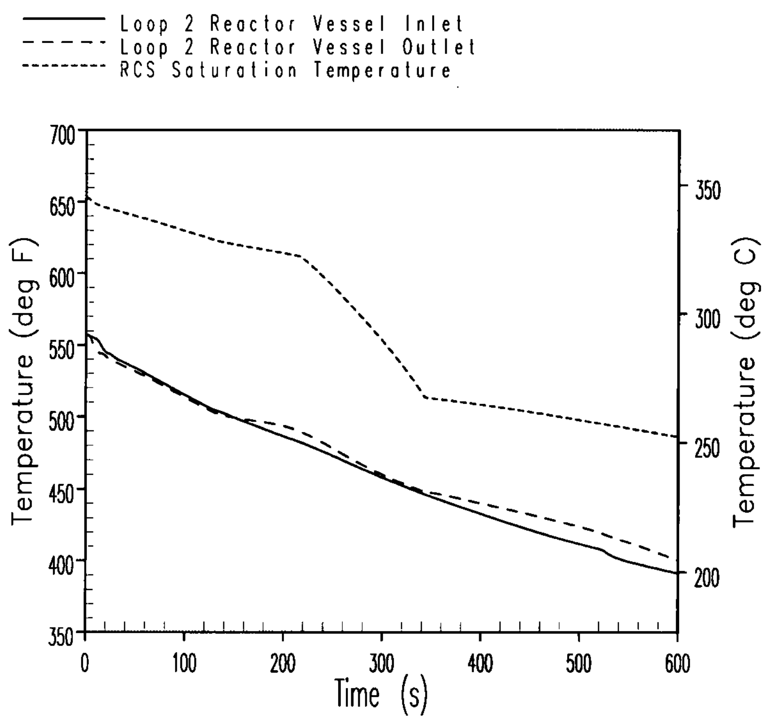
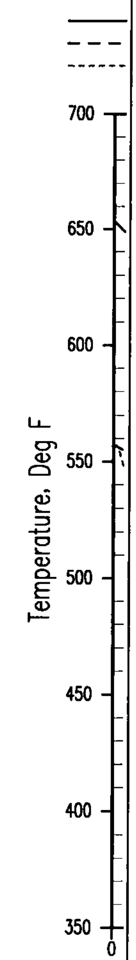


Figure 15.1.4-5

Loop 2 (Faulted Loop) Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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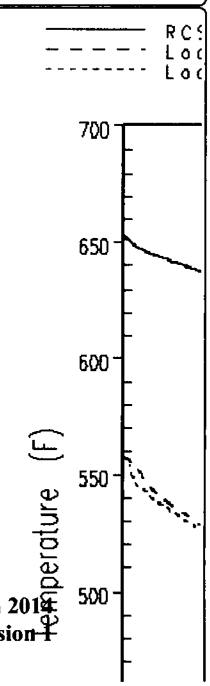

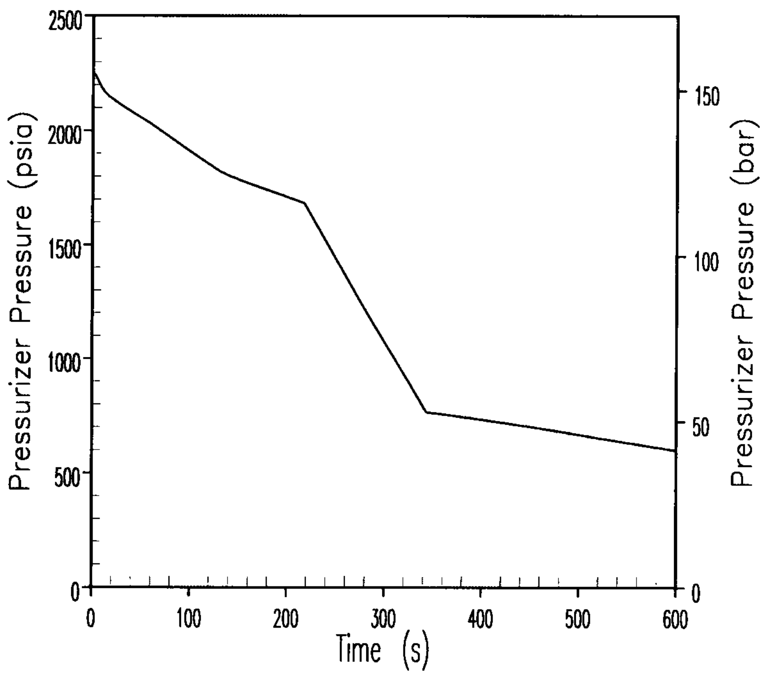

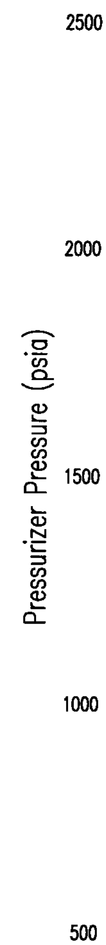

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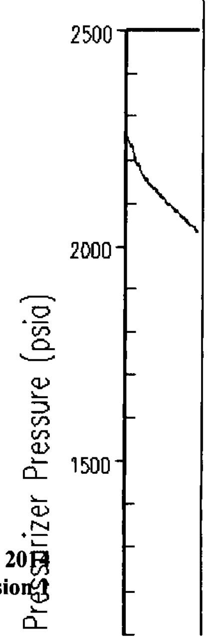
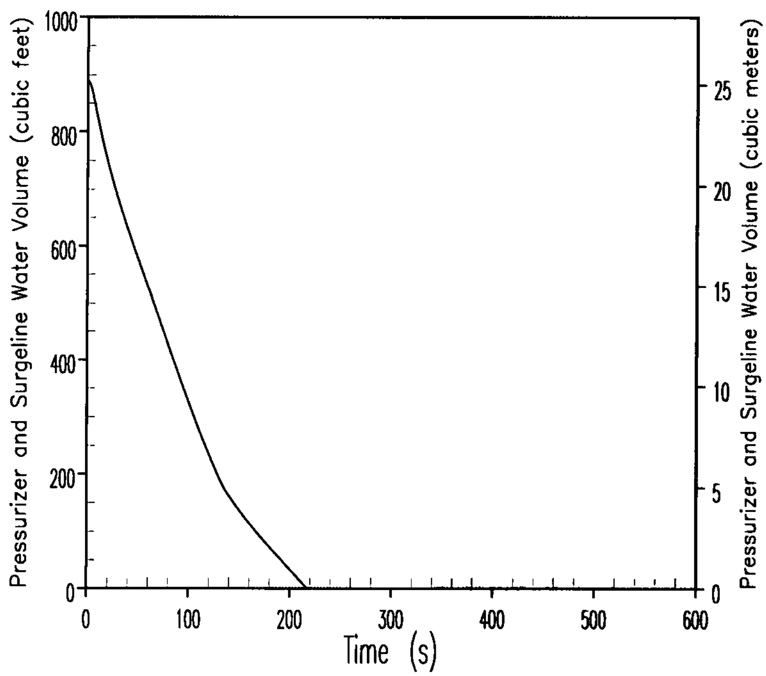


Figure 15.1.4-6

Pressurizer Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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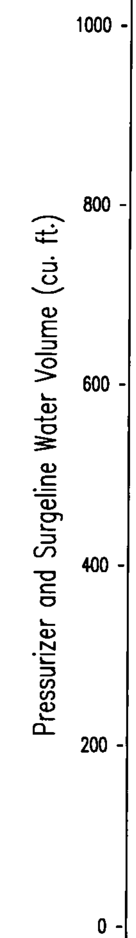


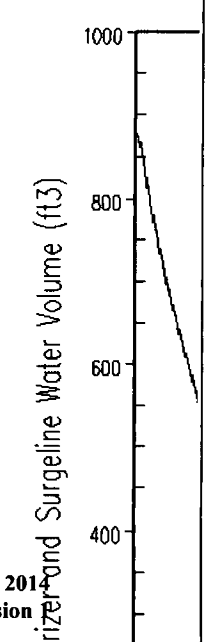
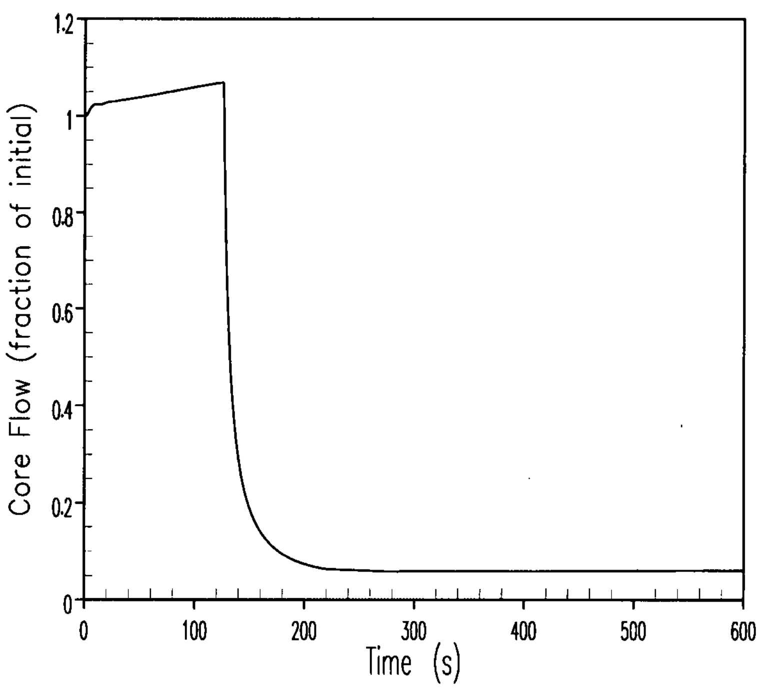


Figure 15.1.4-7

Pressurizer and Surgeline Water Volume Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



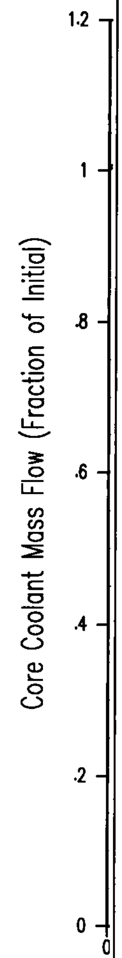


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Figure 15.1.4-8

Core Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

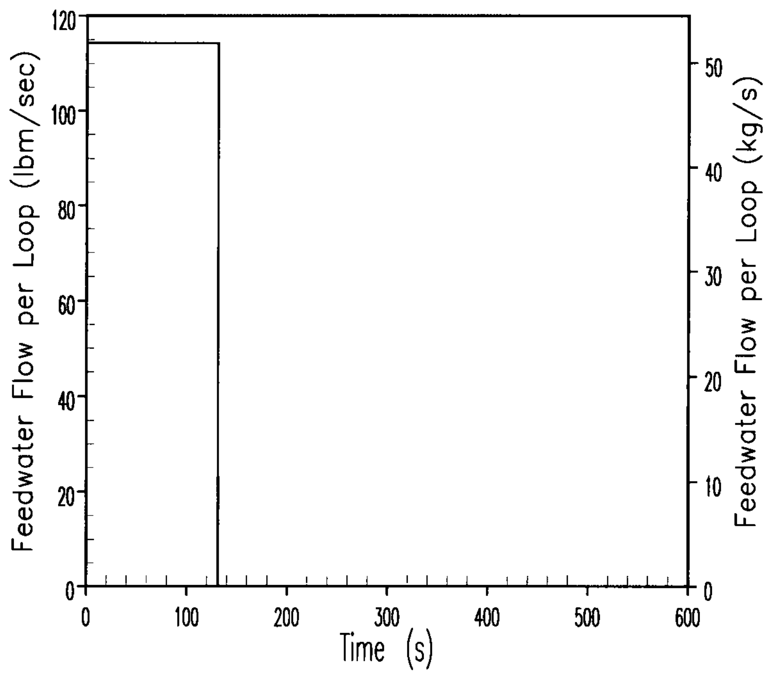

Figure 15.1.4-9

Feedwater Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

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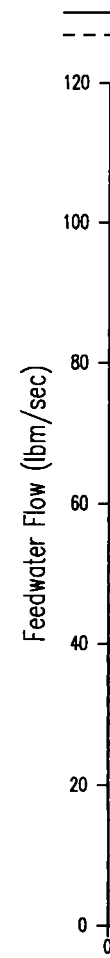

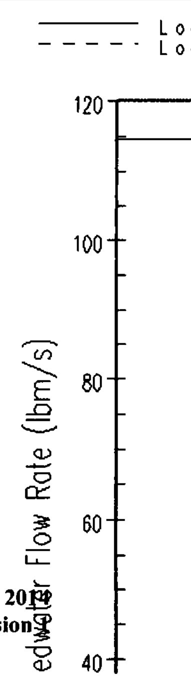

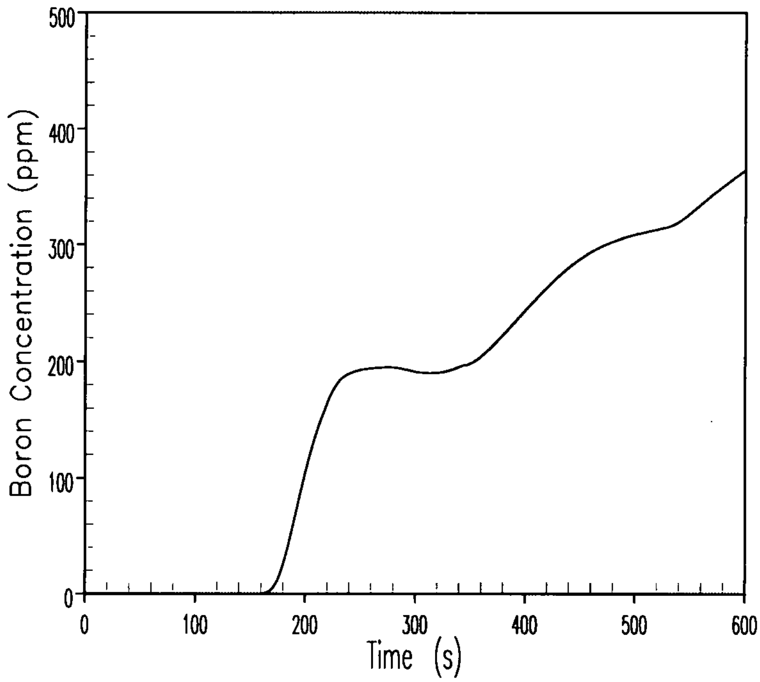

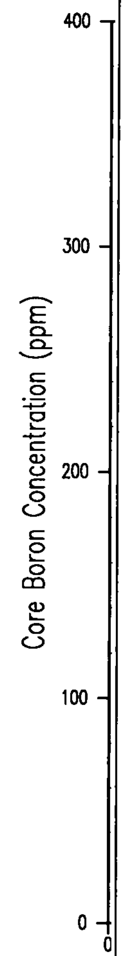

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Figure 15.1.4-10

Core Boron Concentration Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

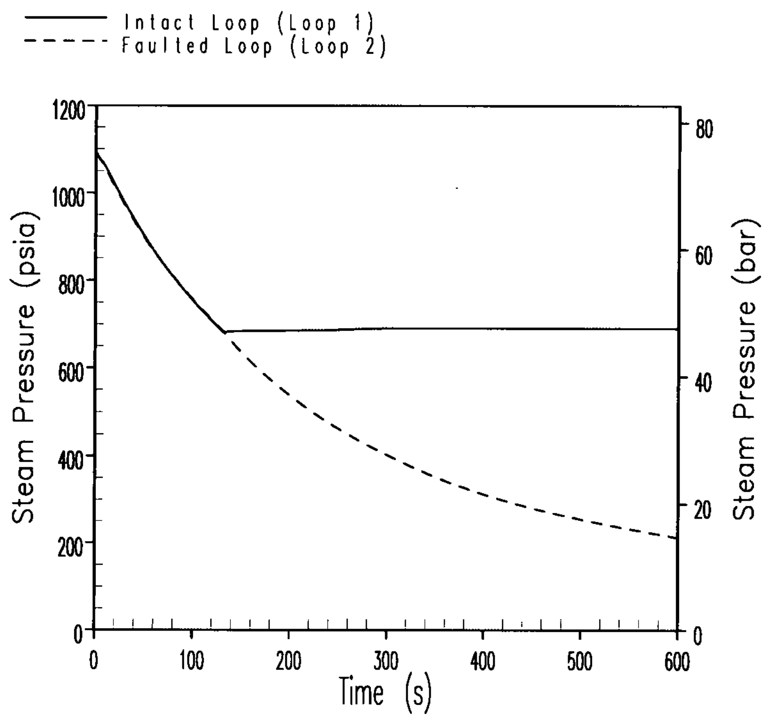
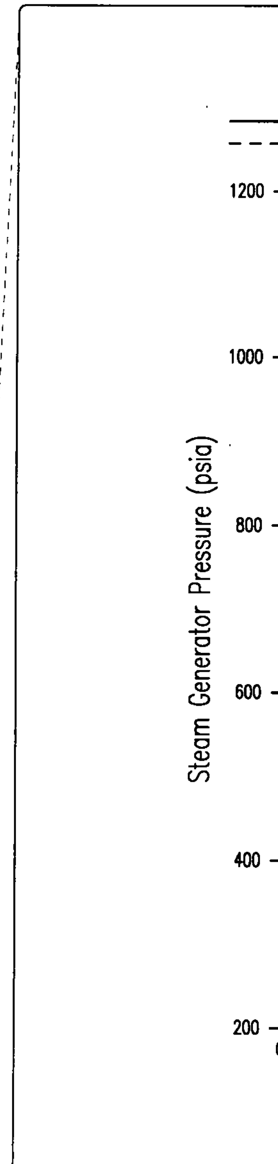


Figure 15.1.4-11

Steam Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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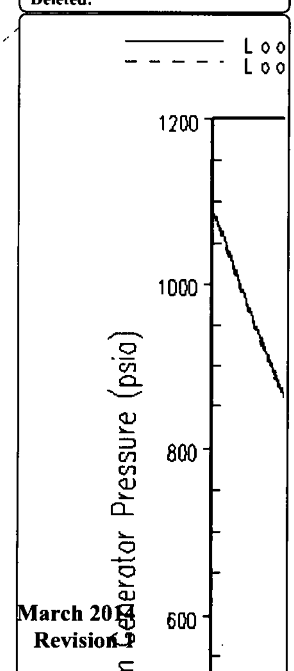


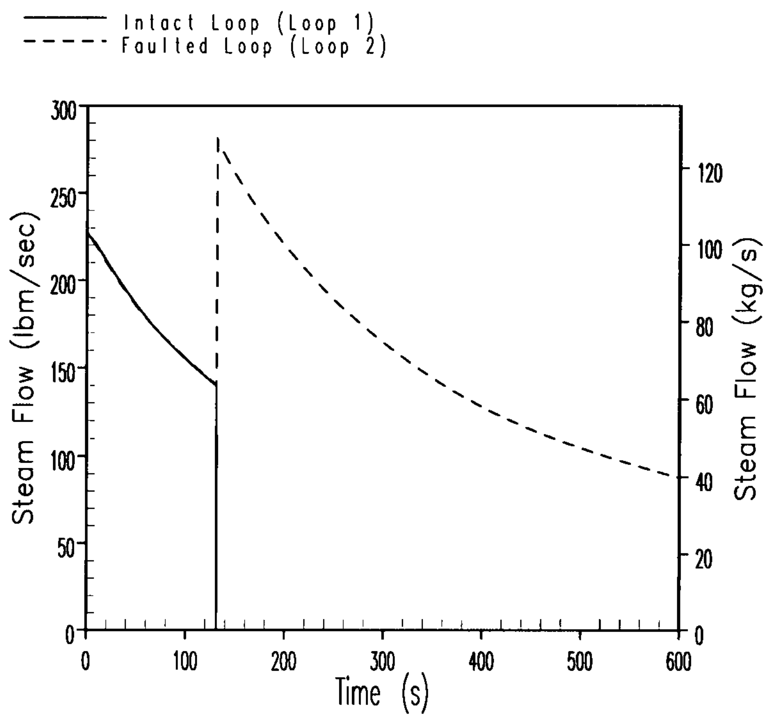
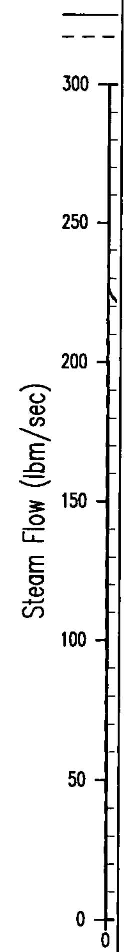


Figure 15.1.4-12

Steam Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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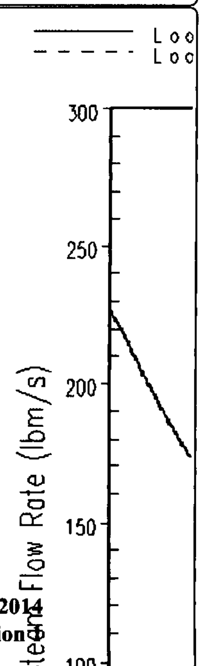

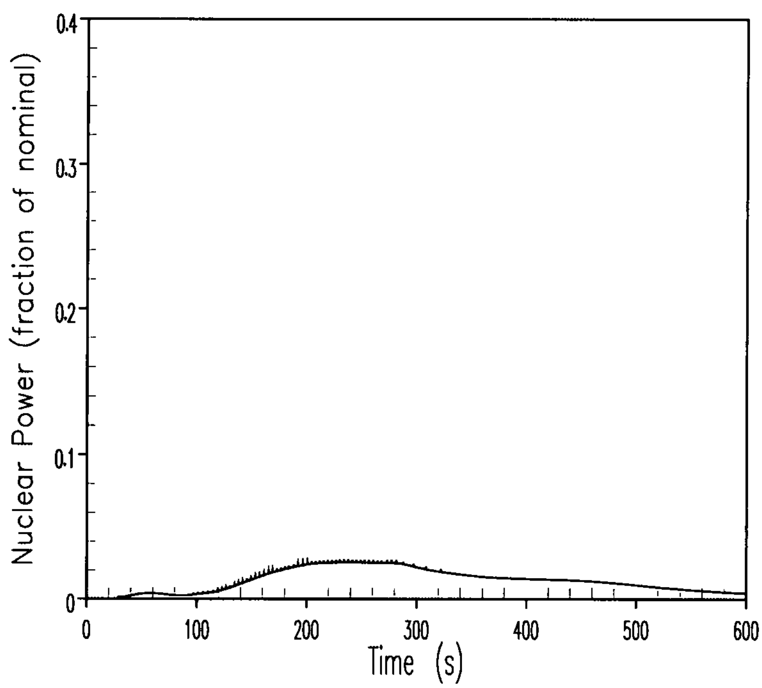

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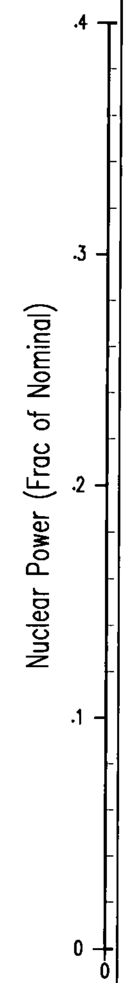

Figure 15.1.5-1

Nuclear Power Transient Steam System Piping Failure

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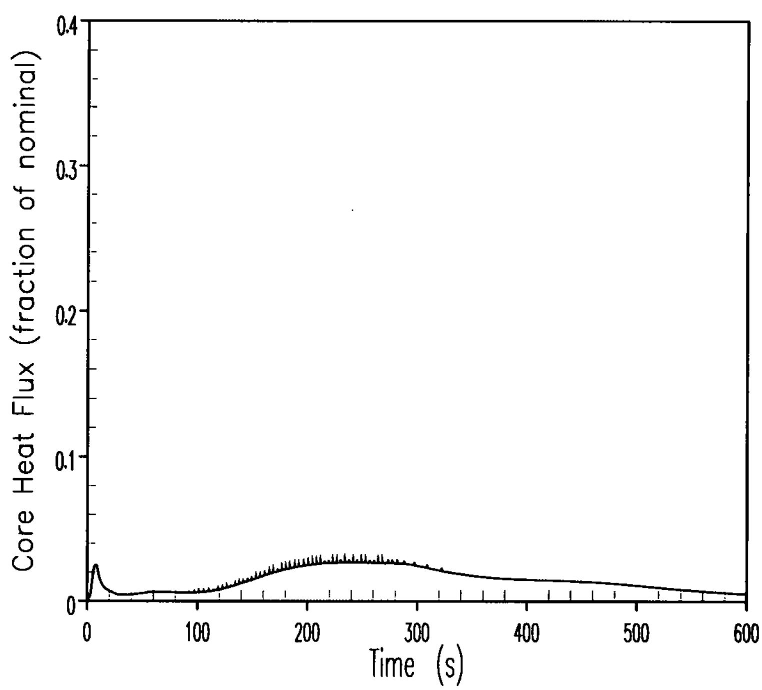


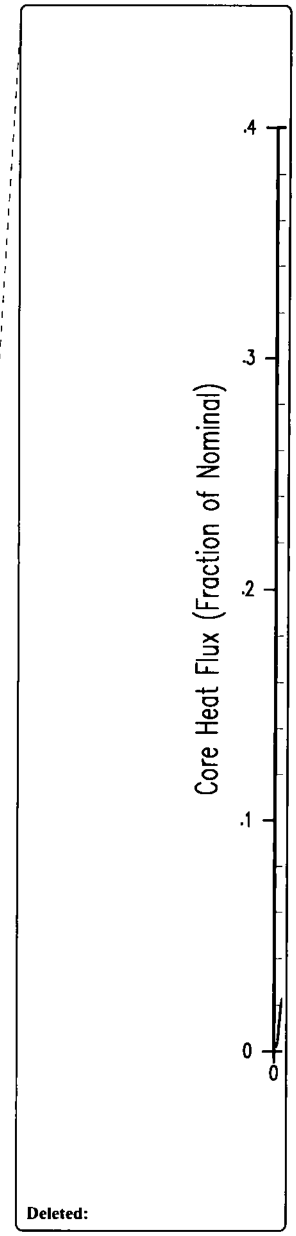


Figure 15.1.5-2

Core Heat Flux Transient Steam System Piping Failure

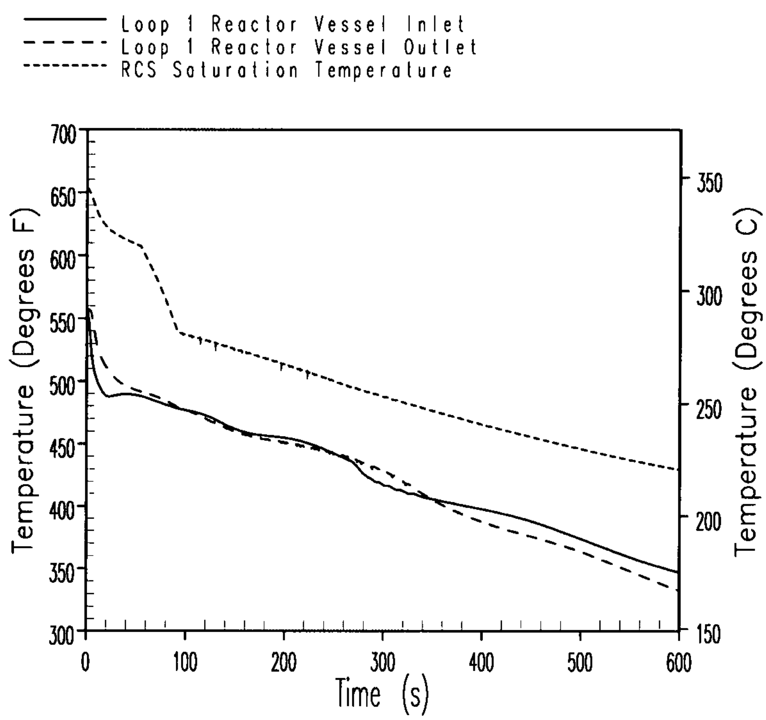
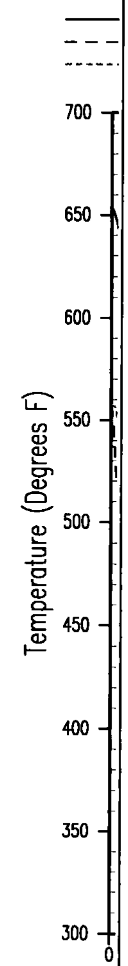


Figure 15.1.5-3

Loop 1 Reactor Coolant Temperatures
Steam System Piping Failure



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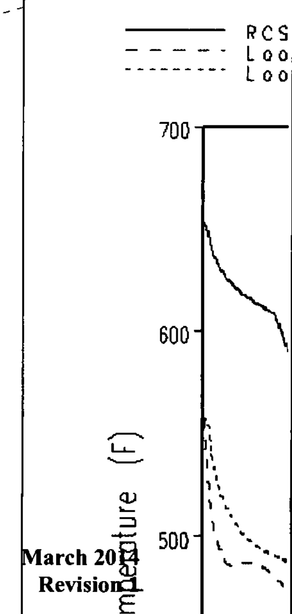

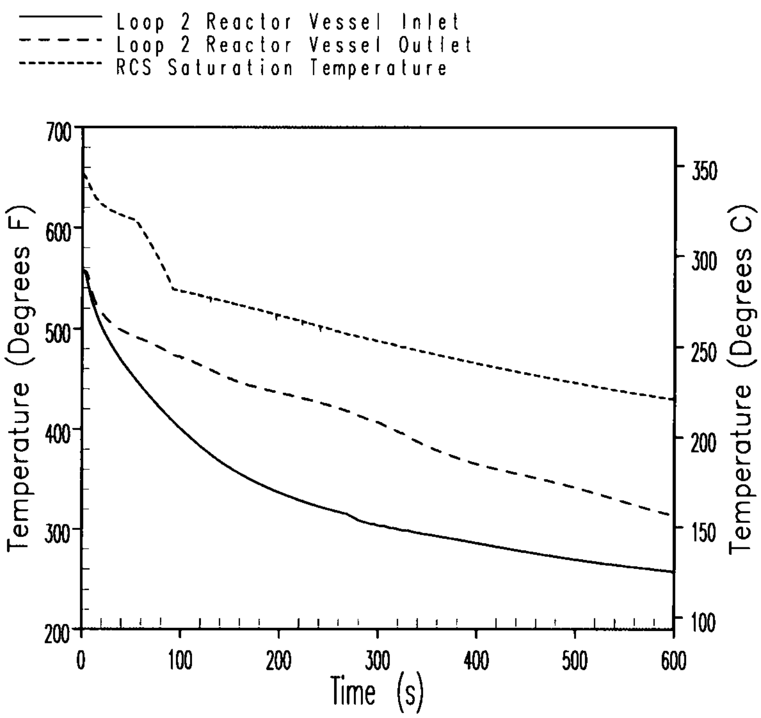

Figure 15.1.5-4

Loop 2 Reactor Coolant Temperatures
Steam System Piping Failure

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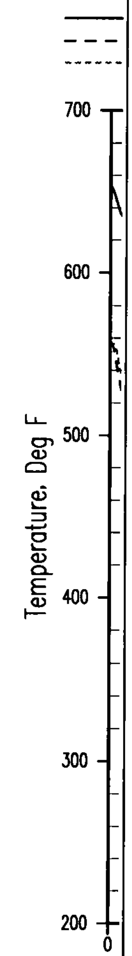
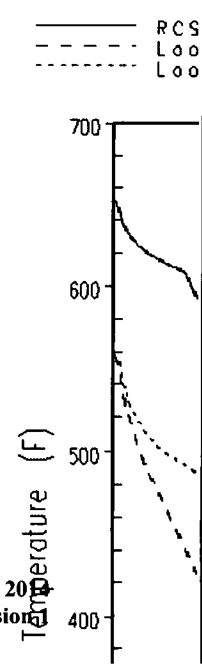




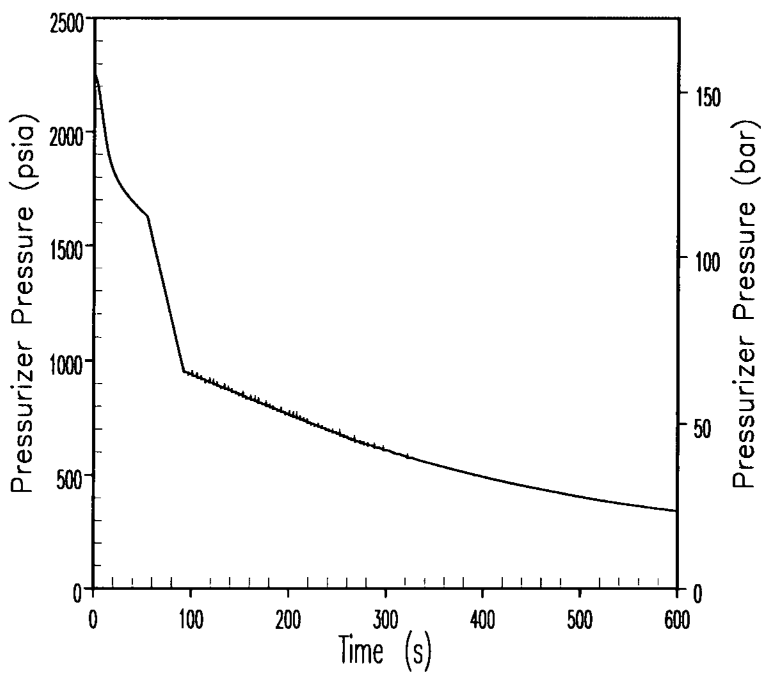
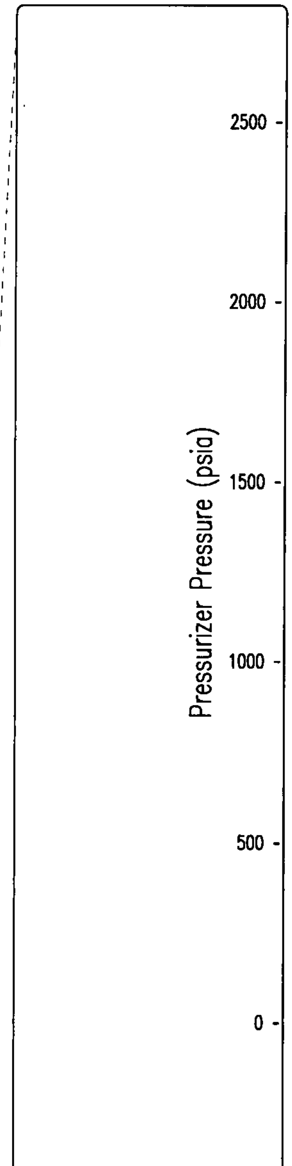


Figure 15.1.5-5

Pressurizer Pressure Transient
Steam System Piping Failure



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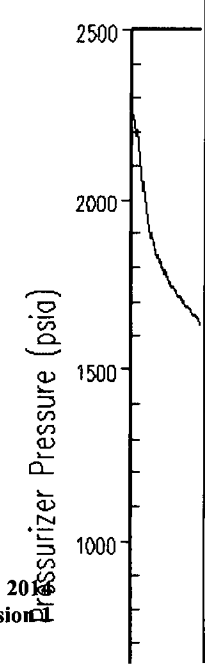

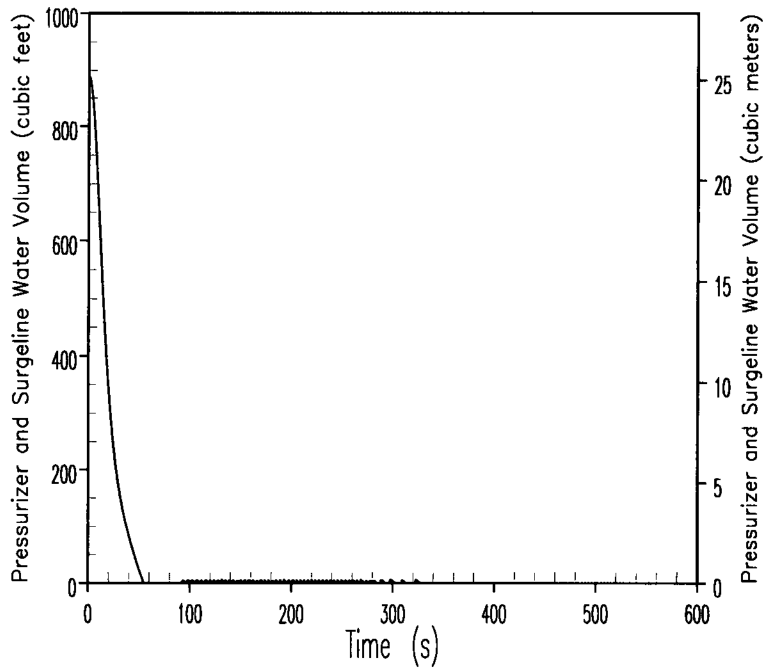

Figure 15.1.5-6

Pressurizer and Surgeline Water Volume Transient
Steam System Piping Failure

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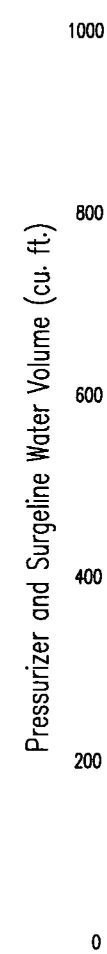

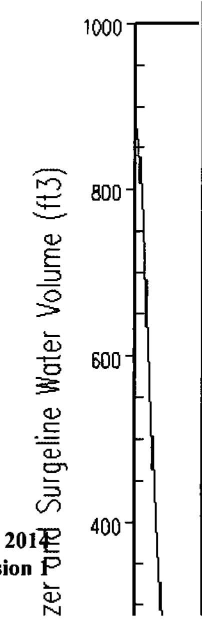

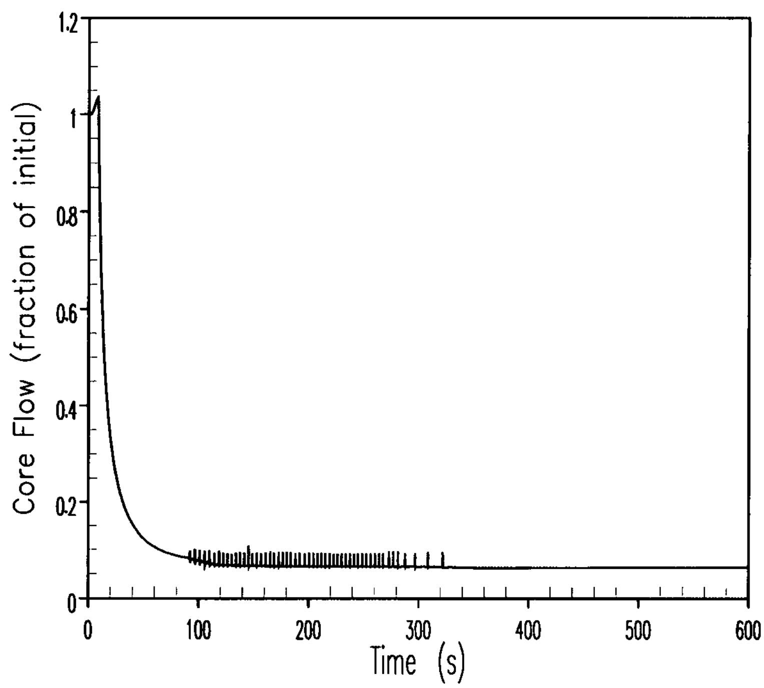

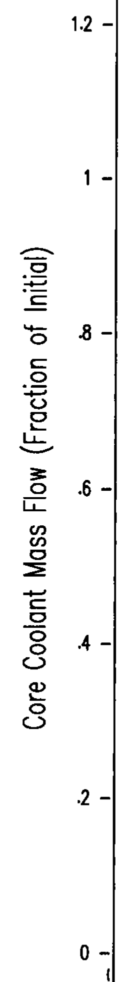

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Figure 15.1.5-7

Core Flow Transient Steam System Piping Failure

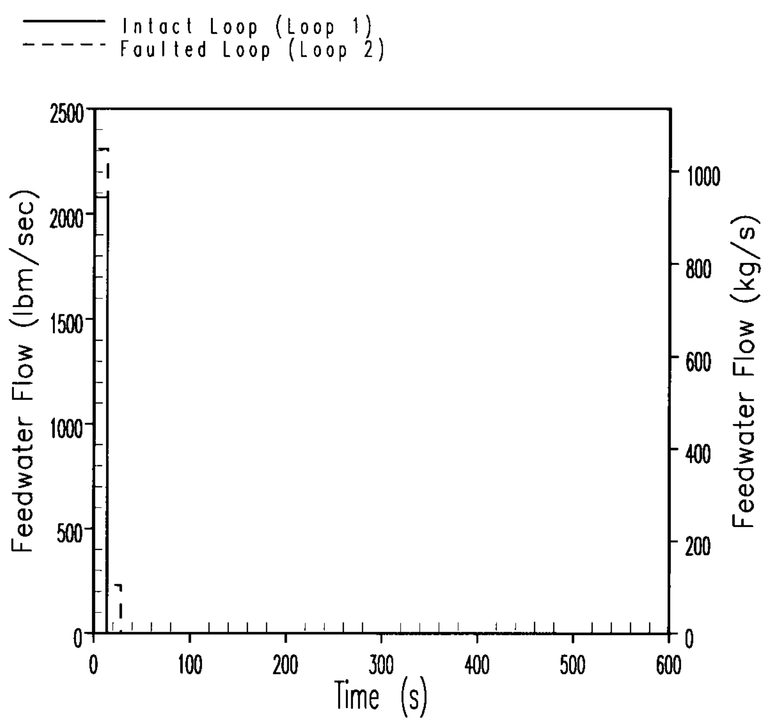

Figure 15.1.5-8

Feedwater Flow Transient Steam System Piping Failure

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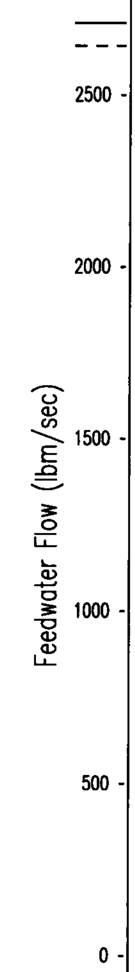

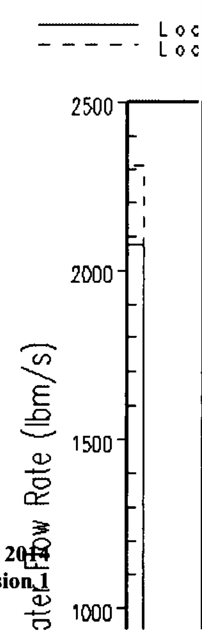

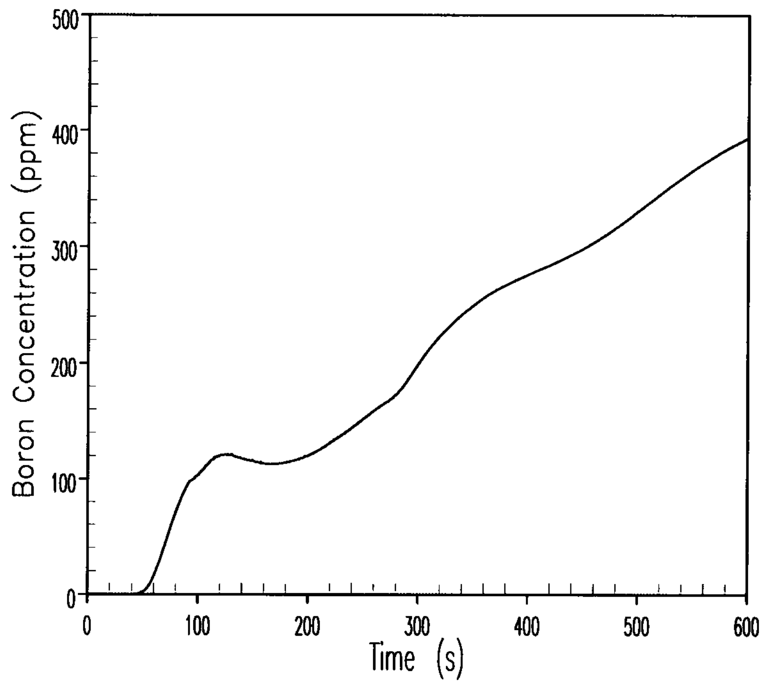

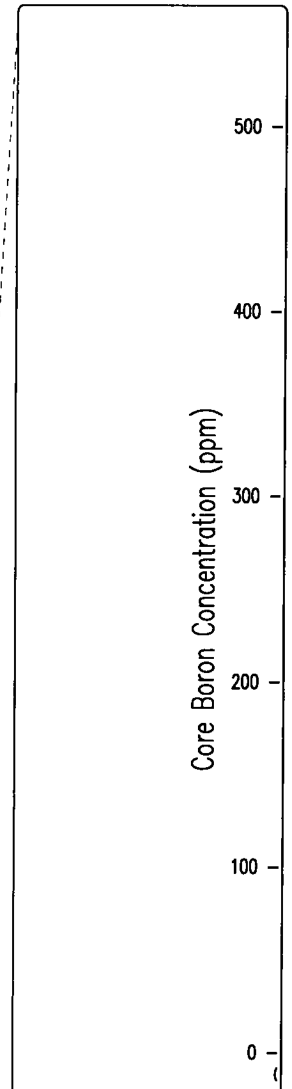

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Figure 15.1.5-9

Core Boron Concentration Transient Steam System Piping Failure

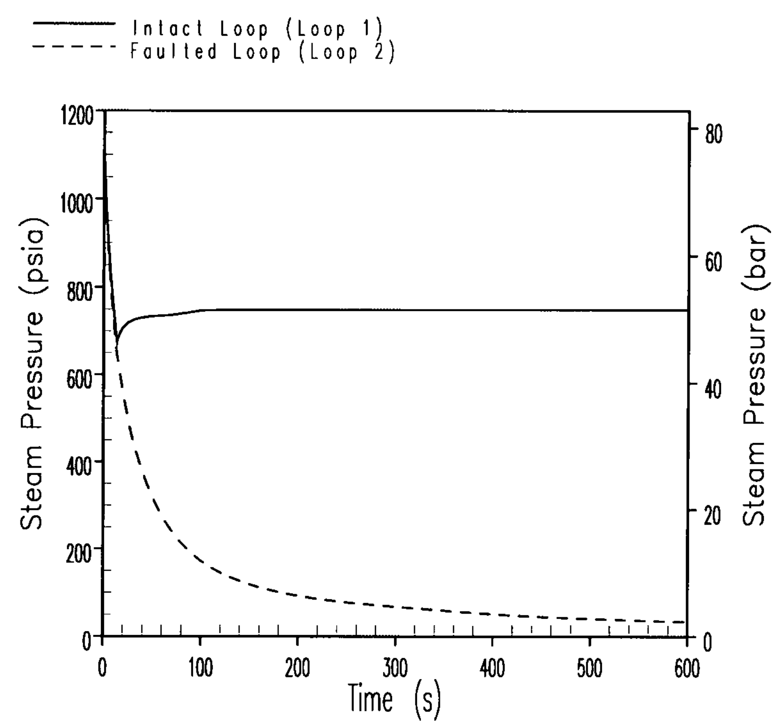

Figure 15.1.5-10

Steam Pressure Transient Steam System Piping Failure

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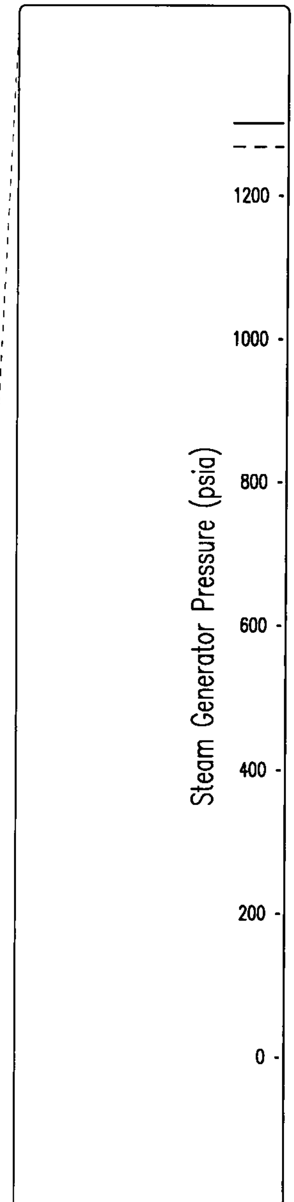
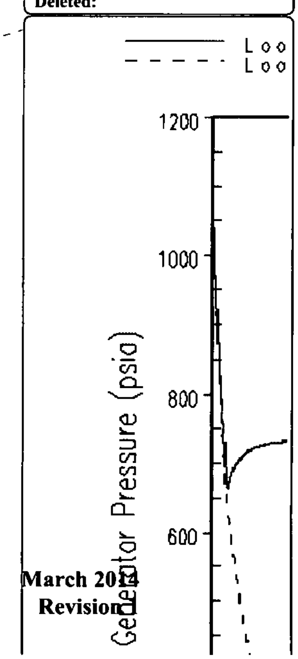




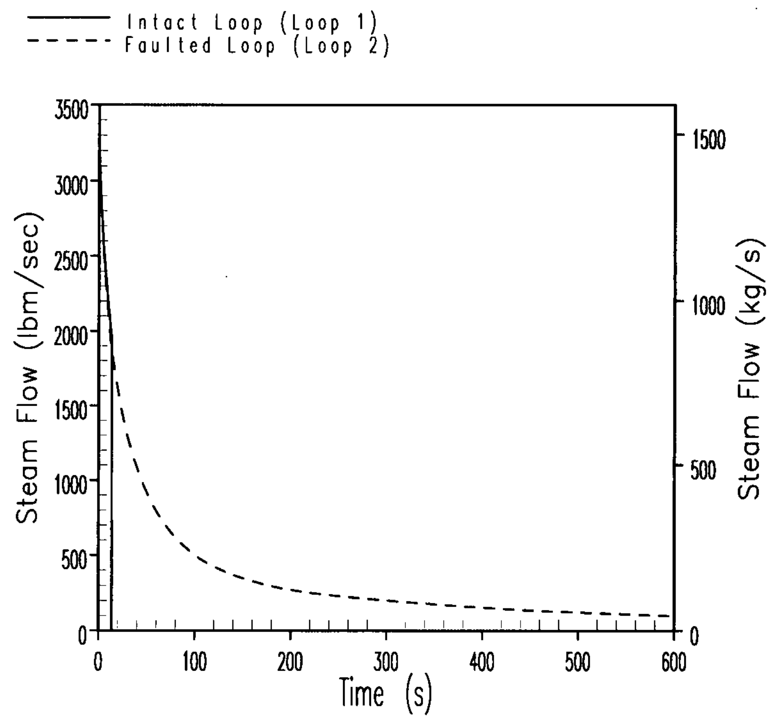
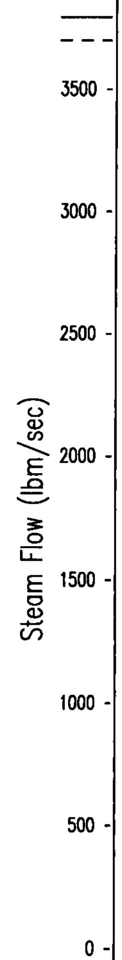


Figure 15.1.5-11

Steam Flow Transient Steam System Piping Failure



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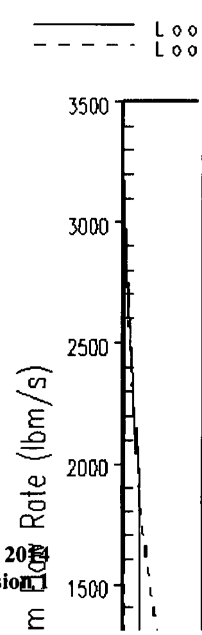


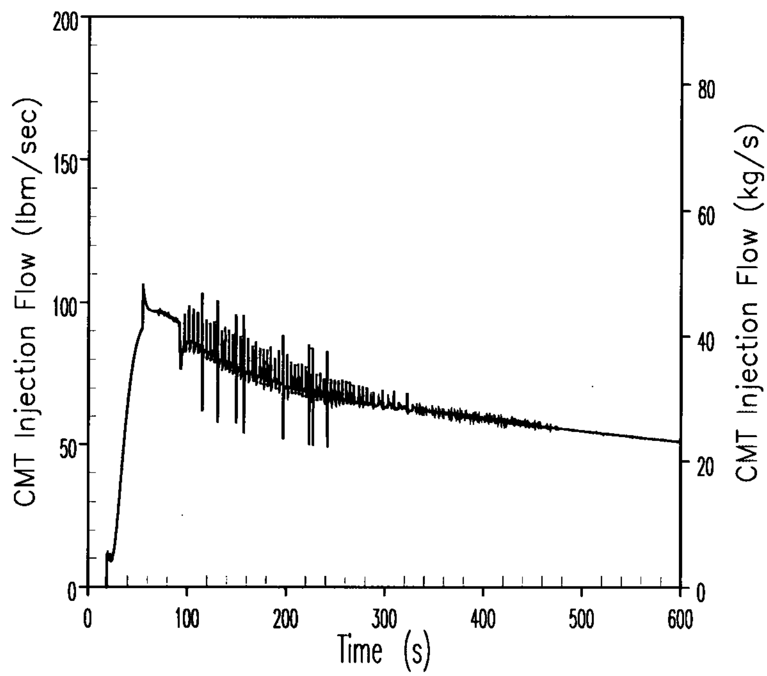
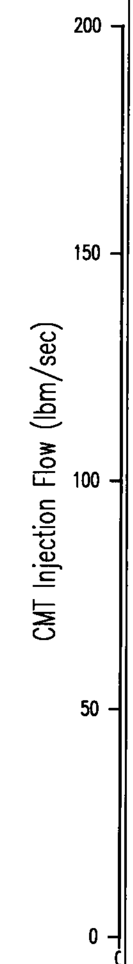


Figure 15.1.5-12

Core Makeup Tank Injection Flow
Steam System Piping Failure



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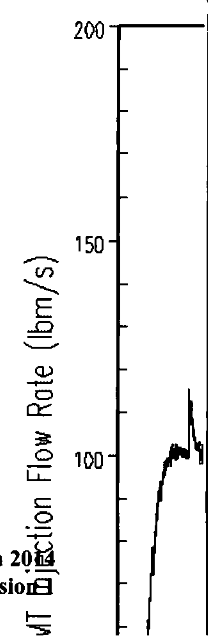


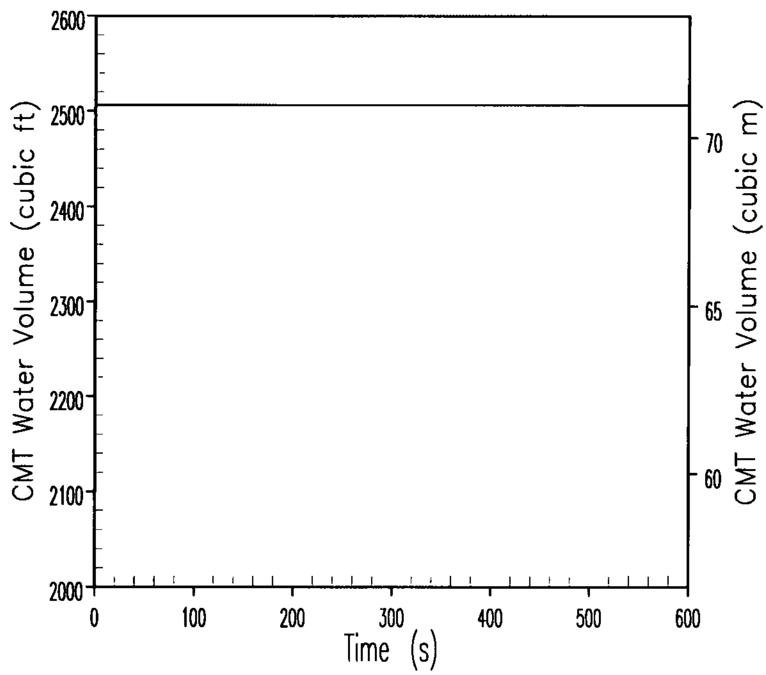
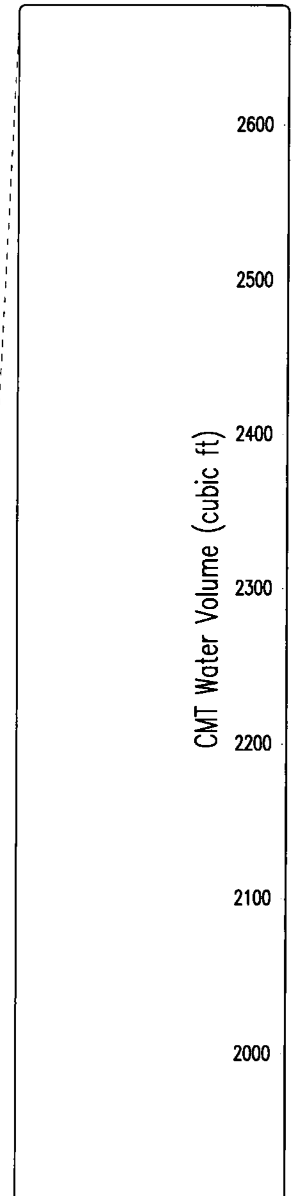


Figure 15.1.5-13

Core Makeup Tank Water Volume Steam System Piping Failure



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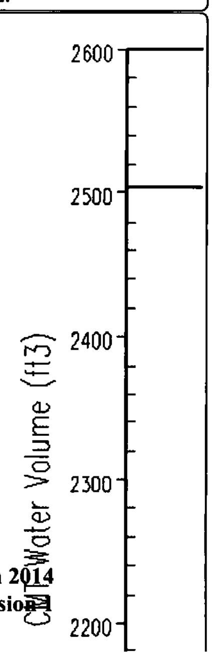

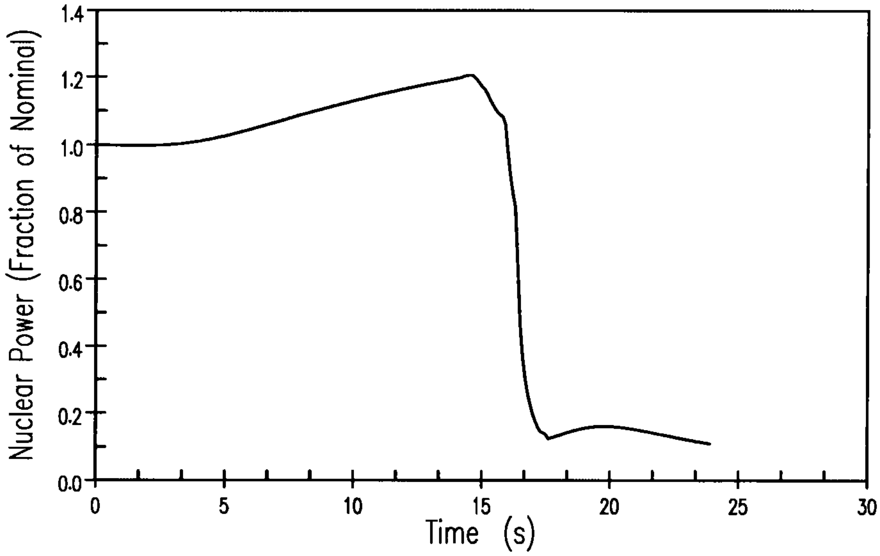

Figure 15.1.5.5-1
Nuclear Power Transient
Steam System Piping Failure at Full Power – 0.87 ft² Break Size

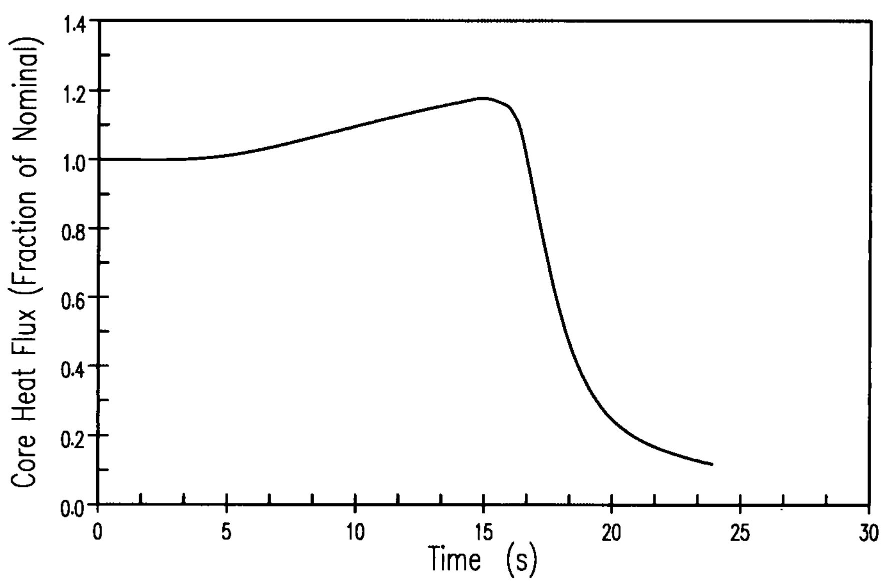

Figure 15.1.5.5-2
Core Heat Flux Transient
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

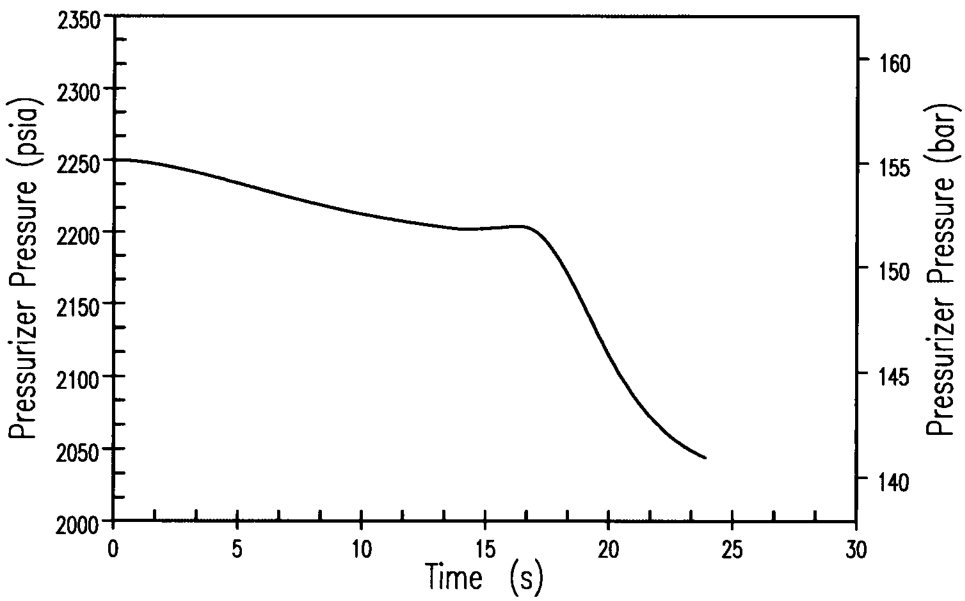

Figure 15.1.5.5-3

Pressurizer Pressure Transient

Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

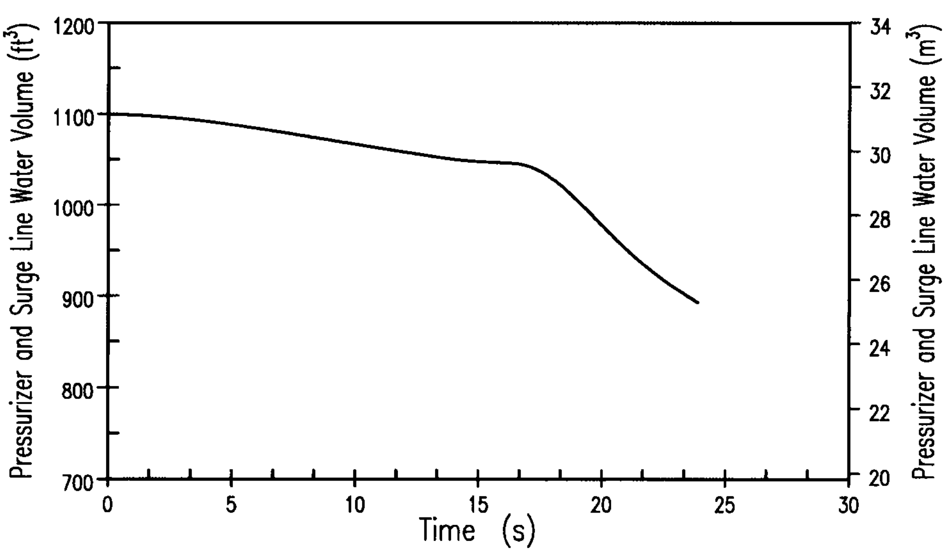

Figure 15.1.5.5-4
Pressurizer Water Volume Transient
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

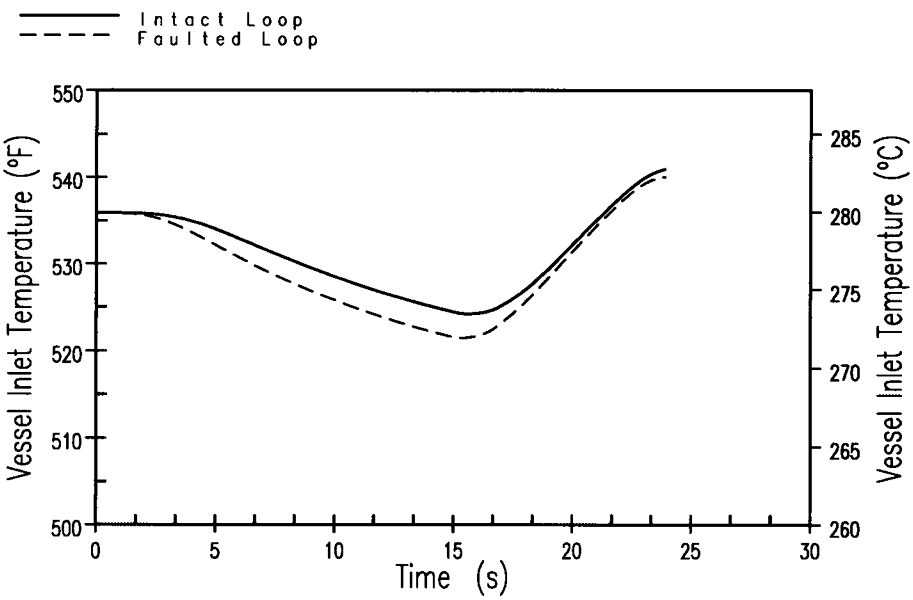

Figure 15.1.5.5-5
Vessel Inlet Temperature Transient
(Intact and Faulted Loops)
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

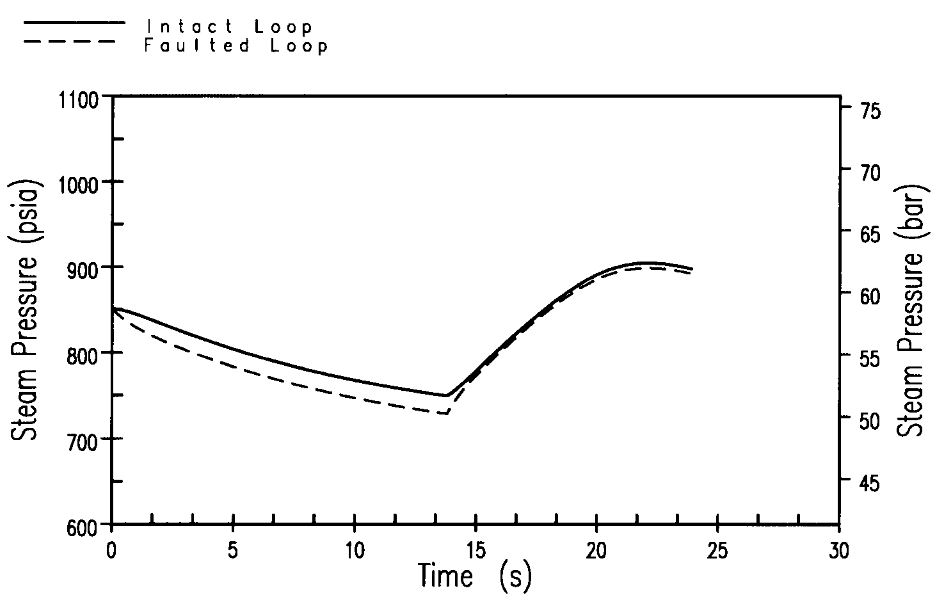

Figure 15.1.5.5-6
Steam Generator Pressure Transient
(Intact and Faulted Loops)
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

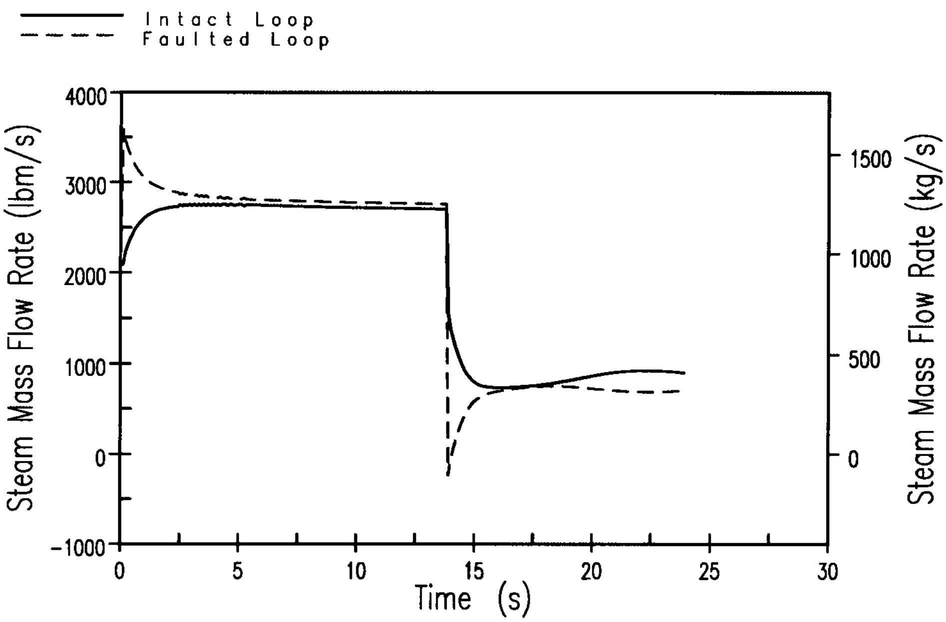

Figure 15.1.5.5-7
Steam Flow Transient(Intact and Faulted Loops)
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

Figures 15.1.6-1 through 15.1.6-8 not used.

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.2	Change Summary Description
[15.2-1]	15.2.2, Loss of External Electrical Load	Editorial changes incorporated.
[15.2-2]	15.2.3, Turbine Trip	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), use of the digital ΔT signal, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Additionally, the moderator density function was modeled as a function of density.
[15.2-3]	15.2.6, Loss of ac Power to the Plant Auxiliaries	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), containment backpressure effects on PRHR heat transfer, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients. The loss of ac power to the plant auxiliaries case presented in the DCD, where feedwater flow is lost at time zero, and power to the reactor coolant pumps is lost as a result of the turbine trip, was renamed Loss of Normal Feedwater Flow with loss of offsite power and was moved into Section 15.2.7. The case presented in Section 15.2.6 now assumes a loss of reactor coolant pumps and loss of feedwater pumps at event initiation.
[15.2-4]	15.2.7, Loss of Normal Feedwater Flow	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), containment backpressure effects on PRHR heat transfer, addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Editorial changes were made to the loss of feedwater analyses to identify an operator action to open the safety related reactor vessel head vent to prevent filling the reactor coolant system water solid. An additional case, Loss of Normal Feedwater Flow with loss of offsite power was added to this section (See the description of changes for Change Number 15.2.6-1).
[15.2-5]	15.2.8, Feedwater System Pipe Break	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), containment backpressure effects on PRHR heat transfer, addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.2-6]	15.2.10 References	Added new reference, WCAP-14565 – consistent with the change to Section 15.2.3.2.1
[15.2-7]	Table 15.2-1	Updated in Revision 1 due to revised CVS makeup flows.
[15.2-8]	Figures 15.2.7-1 through 15.2.7-13	Updated in Revision 1 due to revised CVS makeup flows.

Change No.	Chapter 15 Section 15.2	Change Summary Description
[15.2-9]	15.2.7	Updated in Revision 1 due to revised CVS makeup flows.

15.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents that could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system are postulated. Analyses are presented in this section for the following events that are identified as more limiting than the others:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of ac power to the station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

The above items are considered to be Condition II events, with the exception of a feedwater system pipe break, which is considered to be a Condition IV event.

The radiological consequences of the accidents in this section are bounded by the radiological consequences of a main steam line break (see subsection 15.1.5).

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

There are no steam pressure regulators in the AP1000 whose failure or malfunction causes a steam flow transient.

15.2.2 Loss of External Electrical Load

Comment [B1]: [15.2-1]

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from a loss of electrical load due to an electrical system disturbance. The ac power remains available to operate plant components such as the reactor coolant pumps; as a result, the standby onsite diesel generators do not function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves occurs. The automatic turbine bypass system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system function properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available. For this transient, feedwater flow is maintained by the startup feedwater system.

For a loss of electrical load without subsequent turbine trip, no direct reactor trip signal is generated. The plant trips from the protection and safety monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent exists after total loss of external electrical load because of the steam demand of plant auxiliaries.

If a safety limit is approached, protection is provided by high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external electrical load, the maximum turbine overspeed is not expected to affect the voltage and frequency sensors. Any increased frequency to the reactor coolant pump motors results in a slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, an overfrequency condition is not seen by the protection and safety monitoring system equipment or other safety-related loads. Safety-related loads and the protection and safety monitoring system equipment are supplied from the 120-Vac instrument power supply system, which in turn is supplied from the inverters. The inverters are supplied from a dc bus energized from batteries or by a regulated ac voltage.

If the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. This would cause steam generator shell side pressure and reactor coolant temperature to increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the reactor coolant system and steam generator against overpressure for load losses, without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control.

The steam generator safety valve capacity is sized to remove the steam flow at the nuclear steam supply system thermal rating from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load. The pressurizer safety valves can then relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

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A discussion of overpressure protection can be found in WCAP-7769, Revision 1 (Reference 1) and WCAP-16779 (Reference 9).

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A loss-of-external-load event is classified as a Condition II event, fault of moderate frequency.

A loss-of-external-load event results in a plant transient that is bounded by the turbine trip event analyzed in subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss-of-external-load event.

The primary side transient is caused by a decrease in heat transfer capability, from primary to secondary, due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feedwater flow not be reduced, a larger heat sink is available and the transient is less severe). Reduction of steam flow to the turbine following a loss-of-external load event occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.15 seconds. The transient in primary pressure, temperature, and water volume is less severe for the loss-of-external-load event than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss-of-external-load event is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Refer to subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis bound those expected for the loss-of-external-load event, as discussed in subsection 15.2.2.1.

Plant systems and equipment that may be required to function in order to mitigate the effects of a complete loss of load are discussed in subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to terminate core heat input and to prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may open to maintain system pressures below allowable limits. No single active failure prevents operation of any system required to function. Normal plant control systems and engineered safety systems are not required to function. The passive residual heat removal (PRHR) system may be automatically actuated following a loss of main feedwater, further mitigating the effects of the transient.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event and considerations described in subsection 15.2.2.1, the applicable Standard Review Plan, subsection 15.2.1, evaluation criteria for a loss-of-external-load event, are met (see subsection 15.2.3).

15.2.3 Turbine Trip

Comment [B2]: [15.2.2]

15.2.3.1 Identification of Causes and Accident Description

The turbine stop valves close rapidly (about 0.15 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

- Generator trip
- Low condenser vacuum
- Loss of lubricating oil
- Turbine thrust bearing failure
- Turbine overspeed
- Manual trip
- Reactor trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass. The loss of steam flow results in a rapid increase in secondary system temperature and pressure, with a resultant primary system transient, described in subsection 15.2.2.1, for the loss-of-external-load event. A slightly more severe transient occurs for the turbine trip event due to the rapid loss of steam flow caused by the abrupt valve closure.

The automatic turbine bypass system accommodates up to 40 percent of rated steam flow. Reactor coolant temperatures and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere and main feedwater flow is lost. For this situation, feedwater flow is maintained by the startup feedwater system to provide adequate residual and decay heat removal capability. Should the turbine bypass system fail to operate, the steam generator safety valves may lift to provide pressure control. See subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as a Condition II event, fault of moderate frequency.

A turbine trip is a more limiting than a loss-of-external-load event, loss of condenser vacuum, and other events which result in a turbine trip. As such, this event is analyzed and presented in subsection 15.2.3.2.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent of full power, without rapid power reduction, primarily to show the adequacy of the pressure-relieving devices, and to demonstrate core protection margins. The turbine is assumed to trip without actuating the rapid power reduction system. This assumption delays reactor trip until conditions in the reactor coolant system result in a trip due to other signals. Thus, the analysis assumes a bounding transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater or the PRHR heat exchanger (except for long-term recovery) to mitigate the consequences of the transient.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses are performed to evaluate the effects produced by a possible consequential loss of offsite power during a complete loss of steam load. As discussed in subsection 15.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

The turbine trip transients are analyzed by using a modified version of the LOFTRAN code (Reference 2), as described in Reference 6 The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures, and power level.

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In the turbine trip analyses, which include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the departure from nucleate boiling ratio (DNBR) analyses. First, the LOFTRAN code (References 2 and 6) is used to calculate the plant system transient. The FACTRAN code (Reference 7) or the VIPRE-01 fuel rod model (Reference 8), which is equivalent to FACTRAN, is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN (or VIPRE-01 fuel rod model) and flow from LOFTRAN.

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The major assumptions used in the analysis are summarized below.

Initial Operating Conditions

Two sets of initial operating conditions are used. Cases performed to evaluate the minimum DNBR obtained are analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 5). Instrument bias on the RCS temperature signal is also considered to ensure it is reflected in either the modeled initial conditions or in the safety analysis DNBR limit value.

Cases performed to evaluate the maximum calculated RCS pressure include uncertainties on the initial conditions. Initial core power, reactor coolant temperature, and pressure are assumed to be at the nominal full-power values plus or minus uncertainties. The direction of the uncertainties is chosen to maximize the RCS pressure.

Reactivity Coefficients

Two cases are analyzed:

- Minimum reactivity feedback – A least-negative moderator temperature coefficient and a least-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).
- Maximum reactivity feedback – A conservatively large negative moderator temperature coefficient and a most-negative Doppler-only power coefficient are assumed (see Figure 15.0.4-1).

Rod Control

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From the standpoint of the maximum RCS pressure and minimum DNBR attained, it is conservative to assume that the reactor is in manual rod control. If the reactor is in automatic rod control, the control rod banks move prior to trip and reduce the severity of the transient.

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Steam Release

No credit is taken for the operation of the turbine bypass system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

Pressurizer Spray

Two cases for both the minimum and maximum reactivity feedback cases are analyzed:

- Full credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are also available. These cases are analyzed primarily to address DNBR concerns.
- No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable. These cases are analyzed to address RCS overpressure concerns.

Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for startup feedwater flow or the PRHR heat exchanger, because a stabilized plant condition is reached before initiation of the startup feedwater or the PRHR heat exchanger is normally assumed to occur. The startup feedwater flow or PRHR heat exchanger removes core decay heat following plant stabilization.

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Reactor Trip

Reactor trip is actuated by the first reactor trip setpoint reached, with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, low RCP speed, high pressurizer water level, or low steam generator water level.

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Plant characteristics and initial conditions are further discussed in subsection 15.0.3. Plant systems and equipment that may be required to function in order to mitigate the effects of a turbine trip event are discussed in subsection 15.0.8 and listed in Table 15.0-6.

The protection and safety monitoring system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure prevents operation of systems required to function. Cases are analyzed, both with and without the operation of pressurizer spray, to determine the worst case for presentation.

Availability of Offsite Power

Each case is analyzed with and without offsite power available. As discussed in subsection 15.0.14, the loss of offsite power is considered to be a consequence of an event due to disruption of the electrical grid following a turbine trip during the event. The grid is assumed to remain stable for 3 seconds following the turbine trip. In the analysis for the complete loss of steam load, the event is initiated by a turbine trip. Therefore, offsite power is assumed to be lost 3 seconds after the start of the event. For the loss of steam load analysis, the primary impact of the loss of offsite power is a coastdown of the reactor coolant pumps.

Main Steam System Pressure

Additional cases are performed to evaluate the maximum Main Steam System (MSS) pressure, with initial condition uncertainties chosen to maximize MSS pressure. The additional cases include cases with and without offsite power available for minimum and maximum reactivity feedback.

15.2.3.2.2 Results

The transient responses for a turbine trip from 100 percent of full-power operation are shown for eight cases. The eight analysis cases are performed assuming minimum and maximum reactivity feedback, with and without credit for pressurizer spray, and with and without offsite power available. The results of the analyses are shown in Figures 15.2.3-1 through 15.2.3-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Minimum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-1 through 15.2.3-7 show the transient responses for two cases analyzed for DNBR concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the overtemperature ΔT trip function. The transient DNBR is shown in Figure 15.2.3-6; the minimum DNBR remains above the safety analysis DNBR limit value at all times. Based on this, the DNB design basis defined in Section 4.4 is met.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. The minimum DNBR remains above the safety analysis DNBR limit value at all times, as shown in Figure 15.2.3-6; therefore, the DNBR design basis defined in Section 4.4 is met. This case is the limiting case with respect to the DNBR margin of the turbine trip cases.

Maximum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-8 through 15.2.3-14 show the transient responses for the other two cases analyzed for DNBR concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the overtemperature ΔT trip function. The transient DNBR

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for the case, is shown in Figure 15.2.3-13; the minimum DNBR remains above the safety analysis DNBR limit value at all times. Based on this, the DNBR design basis defined in Section 4.4 is met for this case.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. The DNBR transient is similar to, and bounded by, the minimum feedback case with pressurizer spray and without offsite power discussed above. The minimum DNBR remains above the safety analysis DNBR limit value at all times, as shown in Figure 15.2.3-13; therefore the DNBR design basis defined in Section 4.4 is met.

Minimum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

The results for these cases analyzed to address RCS pressure concerns are shown in Figure 15.2.3-15 through 15.2.3-20. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. The pressurizer safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value.

If offsite power is lost, the reactor is tripped by the low reactor coolant pump speed reactor trip function. Offsite power is assumed to be lost 3 seconds after turbine trip. This causes a reduction in the reactor coolant system flow, which is illustrated in Figure 15.2.3-20.

The pressurizer safety valves actuate in both of these cases and maintain the reactor coolant system pressure below 110 percent of the design value. RCS pressure for these cases is shown in Figure 15.2.3-16. Note that the with and without power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run. With respect to maximum reactor coolant system pressure, this case with offsite power available is the most limiting for turbine trip cases.

Maximum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-21 through 15.2.3-26 show the transient responses for the two other cases analyzed to address RCS pressure concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure function.

The case without offsite power is tripped by the low reactor coolant pump speed trip function. RCS pressure for both cases is shown in Figure 15.2.3-22, the pressure within the reactor coolant system is maintained below 110 percent of the design value. Note that with and without

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power cases have different assumptions regarding initial pressure. The initial pressure assumptions were based upon sensitivities that were run.

The additional cases performed to address maximum MSS pressure concerns confirm that the steam generator safety valves provide sufficient pressure relief to prevent overpressurization of the MSS.

15.2.3.3 Conclusions

Results of the analyses show that a turbine trip presents no challenge to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The analyses show that the predicted DNBR is greater than the safety analysis DNBR limit value at any time during the transient. Thus, the departure from nucleate boiling design basis, as described in Section 4.4, is met.

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15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves results in a turbine trip with no credit taken for the turbine bypass system. Turbine trips are discussed in subsection 15.2.3.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in subsection 15.2.3. A loss of condenser vacuum prevents the use of steam dump to the condenser. Because steam dump is assumed to be unavailable in the turbine trip analysis, no additional adverse effects result if the turbine trip is caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in subsection 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, listed in subsection 15.2.3.1, are covered by subsection 15.2.3. Possible overfrequency effects, due to a turbine overspeed condition, are discussed in subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of ac Power to the Plant Auxiliaries

Comment [B3]: [15.2-3]

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in subsection 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- Upon reaching one of the trip setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal.
- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below occurs:

- Plant vital instruments are supplied from the Class 1E and uninterruptable power supply.
- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The onsite standby power system, if available, supplies ac power to the selected plant non-safety loads.
- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.
- If startup feedwater is not available, the PRHR heat exchanger is actuated.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, keeps the reactor coolant subcooled indefinitely. After the IRWST water reaches saturation, steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

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Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine trip initiated decrease in secondary heat removal without loss of ac power, which is discussed in subsection 15.2.3. A loss of offsite power to the plant auxiliaries will also result in a loss of normal feedwater.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

15.2.6.2.1 Method of Analysis

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, the PRHR heat exchanger, and the reactor coolant system natural circulation capability in removing long-term (approximately 36,000 seconds) decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.

A modified version of the LOFTRAN code (Reference 2), described in WCAP- 15644 (Reference 6), is used to simulate the system transient following a plant loss of offsite power. The simulation describes the plant neutron kinetics and reactor coolant system, including the natural circulation, pressurizer, and steam generator system responses. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in this analysis minimize the energy removal capability of the PRHR heat exchanger and maximize the coolant system expansion.

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The assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design power rating with initial reactor coolant temperature 8°F below the nominal value and the pressurizer pressure 50 psi above the nominal value.

- Core residual heat generation is based on ANSI 5.1 (Reference 3). ANSI 5.1 is a conservative representation of the decay energy release rates.

- Reactor trip occurs on RCP speed-low

- A heat transfer coefficient is assumed in the steam generator associated with reactor coolant system natural circulation flow conditions following the reactor coolant pump coastdown.

- The PRHR heat exchanger is actuated by the low steam generator water level (narrow range coincident with low start up feed water flow).

- For the loss of ac power to the station auxiliaries and following reactor trip, the main safety function required is core decay heat removal. That is accomplished by the secondary steam relief through the steam generator safety valves and the PRHR heat exchanger. One of two parallel valves in the PRHR outlet line is assumed to fail to open. This is the worst single failure.

- The pressurizer safety valves are assumed to function.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of ac power to the station auxiliaries are discussed in subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of ac power. The PRHR heat exchanger is required to function with an overall minimum capability to extract heat from the reactor coolant system.. No single active failure prevents operation of any system required to function.

Parameters used in the analysis are selected to maximize the pressurizer water volume. Input parameters are not selected to maximize the transient primary side and secondary side pressure. Transient primary side and secondary side pressures during a loss of ac power to station auxiliaries are bounded by those calculated for the turbine trip analyses presented in Section 15.2.3

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With respect to DNB concerns, the loss of ac power to station auxiliaries event is bounded by the loss of ac power case analyzed for the turbine trip event presented in Section 15.2.3.

15.2.6.2.2 Results

The transient response of the reactor coolant system following a loss of ac power to the plant auxiliaries is shown in Figures 15.2.6-1 through 15.2.6-12. The calculated sequence of events for this event is listed in Table 15.2-1.

The loss of ac power event results in a pressurizer water volume increase until the actuation of the steam generator safety valves. Actuation of the steam generator safety valves attenuates the pressurizer water volume until actuation of the PRHR which turns around the pressurizer water volume increase. PRHR heat extraction and steam generator safety valve relief results in a consequential decrease in the water volume until the safety valve relief stops. After the steam generator safety valve flow stops the pressurizer water volume begins a slight increase until the PRHR heat extraction matches and then exceeds the decay heat addition resulting in a reduction in the pressurizer water volume.

15.2.6.3 Conclusions

Results of the analysis show that for the loss of ac power to plant auxiliaries event, all safety criteria are met. The heat extraction provided by the steam relief capacity of the steam generator safety valves and the operation of the PRHR is sufficient to prevent water relief through the pressurizer safety valves.

The analysis demonstrates that sufficient long-term reactor coolant system heat removal capability exists, via the steam generator safety valves, natural circulation and the PRHR heat exchanger, following reactor coolant pump coastdown to prevent fuel or cladding damage and reactor coolant system overpressure.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related PRHR heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.

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A small secondary system break can affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break. This scenario is addressed by the assumptions made for the feedwater system pipe break (see subsection 15.2.8).

The following occurs upon loss of normal feedwater (assuming main feedwater pump fails or valve malfunctions):

- The steam generator water inventory decreases as a consequence of the continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow leads to the reactor trip on a low steam generator water level signal. The same signal also actuates the startup feedwater system (see subsection 15.2.6.1).

- As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the decay heat and to maintain the plant at the hot shutdown condition, if the startup feedwater is used to supply water to the steam generator.

- If startup feedwater is not available, the PRHR heat exchanger is actuated on either a low steam generator water level (narrow range), coincident with a low startup feedwater flow rate signal or a low steam generator water level (wide range) signal.

- The PRHR heat exchanger extracts heat from the reactor coolant system leading to an "S" signal on a Low T_{cold} signal. This actuates the core makeup tanks. Both core makeup tanks inject mass into the reactor coolant system and the pressurizer level continues to increase until the operators take action to end the pressurizer level increase transient. The operators are assumed to be alerted that a potential filling event is occurring on the high-2 pressurizer level signal. The operator action assumed in the analysis is to open the reactor vessel head vent following receipt of the high-3 pressurizer level signal; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal. When the head vent is opened, the pressurizer level increase slows and ultimately the level begins to decrease.

A loss-of-normal-feedwater event is classified as a Condition II event, a fault of moderate frequency.

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15.2.7.2 Analysis of Effects and Consequences

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, and the capability of the PRHR heat exchanger in removing long-term (approximately 36,000 seconds) decay heat following a loss of normal feedwater. Those systems in conjunction with the operator action to open the reactor head vent show that the loss of water from the reactor coolant system is prevented. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization.

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15.2.7.2.1 Method of Analysis

An analysis using a modified version of the LOFTRAN code (Reference 2), described in WCAP-15644 (Reference 6), is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the neutron kinetics, reactor coolant system (including the natural circulation), pressurizer, and steam generators. The program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Two cases are analyzed. One case assumes a consequential loss of ac power to the plant auxiliaries resulting from the turbine trip after reactor trip. The loss of ac power results in a coast down of the reactor coolant pumps. A second case does not assume the consequential loss of ac power, which maintains the reactor coolant pumps at normal speed until automatically tripped when the core makeup tanks are actuated.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design power rating.
- Reactor trip occurs on steam generator low (narrow range) level.
- The principle safety function required after reactor trip is the core decay heat removal. That function is carried out by the PRHR heat exchanger. The worst single failure is assumed to occur in the PRHR heat exchanger. The actuation of the PRHR heat exchanger requires the opening of one of the two fail-open valves arranged in parallel at the PRHR heat exchanger discharge. Because no single failure can be assumed that impairs the opening of both valves, the failure of a single valve is assumed.

  The PRHR heat exchanger is actuated by the low steam generator water level narrow range signal, coincident with low start up feedwater flow or by the low steam generator water level wide range signal.

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- Plant cool down with the PRHR heat exchanger may cause a reduction in the low cold leg temperature such that the Safeguards setpoint is reached which will actuate the core makeup tanks. The additional borated fluid added by the core makeup tanks may cause excessive pressurizer water volume. Prevention of pressurizer filling is accomplished by an operator action to open the reactor head vent.
- Secondary system steam relief is achieved through the steam generator safety valves.
- The initial reactor coolant average temperature is 8°F lower than the nominal value, and initial pressurizer pressure is 50 psi lower than nominal.

The loss of normal feedwater analyses are performed to demonstrate the adequacy of the protection and safety monitoring system and the PRHR heat exchanger in removing long-term decay heat. Such decay heat removal prevents excessive heatup of the reactor coolant system with possible resultant reactor coolant system overpressurization or loss of reactor coolant system water. The assumptions used in this analysis minimize the energy removal capability of the system, and maximize the coolant system expansion.

With respect to the overpressure evaluation, the loss of normal feedwater transient with and without ac power available events are bounded by the turbine trip event.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

Plant systems and equipment necessary to mitigate the effects of a loss of normal feedwater accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The protection and safety monitoring system is required to function following a loss of normal feedwater. The PRHR heat exchanger is required to function with an overall minimum capability to extract heat from the reactor coolant system. No single active failure prevents operation of any system to perform its required function.

15.2.7.2.2 Results

Figures 15.2.7-1 through 15.2.7-13 show the significant plant parameters following a loss of normal feedwater.

The loss of main feedwater results in an increase in the pressurizer water volume until reactor trip on low steam generator water level (narrow range). The pressurizer water volume then decreases briefly due to the reactor trip. Later in the transient, the pressurizer water level decreases again when the steam generator safety valves open. Steam relief and a consequential reduction in the pressurizer water volume continues until the steam generator pressure falls below the safety

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valve setpoints stopping the steam relief. The pressurizer water volume then increases until the PRHR actuates.

The capacity of the PRHR heat exchanger, when the reactor coolant pumps are operating, is much larger than the decay heat, and in the first part of the transient, the reactor coolant system is cooled down and the pressurizer pressure and water volume decrease. The cool down continues until the reactor coolant temperature reaches the low T_{cold} setpoint. When the low T_{cold} setpoint is reached, the reactor coolant pumps are tripped and the core makeup tanks are actuated.

The pressurizer water volume then increases due to the cold borated water injected by the core makeup tanks and the reduced PRHR efficiency due to the loss of forced flow resulting from the reactor coolant pump trip. Pressurizer water volume increases during this period. The operators are alerted to the pressurizer level increase when the level exceeds the high-2 pressurizer level setpoint. The operator action assumed in the analysis is to open the reactor vessel head vent following receipt of the high-3 pressurizer level signal; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal. After that point, the pressurizer water volume begins to decrease.

The DNBR transient for the loss of normal feedwater event is shown in Figure 15.2.7-12.

The calculated sequence of events for this accident is listed in Table 15.2-1.

In the loss of normal feedwater event, the operator action to open the reactor vessel head vent and the capacity of the PRHR heat exchanger is sufficient to avoid water relief through the pressurizer safety valves.

Figures 15.2.7-14 through 15.2.7-26 show the significant plant parameters following a loss of normal feedwater with a consequential loss of ac power to plant auxiliaries.

The first increase in pressurizer water volume is turned around by the heat extraction provided by the steam generator safety valves. Due to the steam generator safety valve relief, the pressurizer water volume decreases until the heat extraction provided by the steam generator safety valves relief stops once the steam pressure decreases below the steam generator safety valve setpoints. With no steam generator safety valve relief, the pressurizer water volume begins to increase until the PRHR heat extraction approaches the magnitude of the decay heat addition resulting in a peak pressurizer water volume at 3584 seconds.

15.2.7.3 Conclusions

Results of the analyses show that a loss of normal feedwater, or a loss of normal feedwater with a consequential loss of ac power to the plant auxiliaries do not adversely affect the core, the reactor

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coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger, the steam generator safety valves and the fluid relief capacity of the reactor vessel head vent are such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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15.2.8 Feedwater System Pipe Break

Comment [B5]: [15.2-5]

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedwater line check valve would affect the plant only as a loss of feedwater. This case is covered by the evaluation in subsections 15.2.6 and 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in subsection 15.1.5. Therefore, only the reactor coolant system heatup effects are evaluated for a feedwater line rupture in this subsection.

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- Feedwater flow to the steam generators is reduced. Because feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.
- Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main feedwater after trip.

A major feedwater line rupture is classified as a Condition IV event.

The severity of the feedwater line rupture transient depends on a number of system parameters, including the break size, initial reactor power, and the functioning of various control and safety-related systems. Sensitivity studies presented in WCAP-9230 (Reference 4) illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line.

At the beginning of the transient, the main feedwater control system is assumed to malfunction due to an adverse environment. Interactions between the break and the main feedwater control system result in no feedwater flow being injected or lost through the steam generator feedwater nozzles. This assumption causes the water levels in both steam generators to decrease equally until the low steam generator level (narrow range) reactor trip setpoint is reached. After reactor trip, a full double-ended rupture of the feedwater line is assumed such that the faulted steam generator blows down through the break and no main feedwater is delivered to the intact steam generator. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analysis is performed at full power assuming the loss of offsite power at the time of the reactor trip. This is more conservative than the case where power is lost at the initiation of the event. The case with offsite power available is not explicitly examined because, due to the fast generation of an "S" signal (generated by the low steam line pressure), the reactor coolant pumps would be tripped by the protection and safety monitoring system shortly after the reactor trip. The only difference between the cases with and without offsite power available would be a small difference in when the reactor coolant pumps are tripped.

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The following provides the protection for a main feedwater line rupture:

- A reactor trip on any of the following five conditions:
  - High pressurizer pressure
  - Overtemperature ΔT
  - High-3 pressurizer water level
  - Low steam generator water level in either steam generator
  - "S" signals from either of the following:
    - Two out of four low steam line pressure in either steam generator
    - Two out of four high containment pressure (high-2)

Refer to Sections 7.1 and 7.2 for a description of the actuation system.

The PRHR heat exchanger functions to:

- Provide a passive method for decay heat removal. The heat exchanger is a C-tube type, located inside the IRWST. The heat exchanger is above the reactor coolant system to provide natural circulation of the reactor coolant. Operation of the PRHR heat exchanger is initiated by the opening of one of the two parallel power-operated valves at the PRHR heat exchanger cold leg.

- Prevent substantial overpressurization of the reactor coolant system (less than 110 percent of design pressures).

- Maintain sufficient liquid in the reactor coolant system so that the core remains in place, and geometrically intact, with no loss of core cooling capability.

Refer to subsection 6.3.2.2.5 for a description of the PRHR heat exchanger.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

An analysis using a modified version, described in WCAP-15644 (Reference 6), of the LOFTRAN code (Reference 2) is performed to determine the plant transient following a feedwater line rupture. The code describes the reactor thermal kinetics, reactor coolant system (including natural circulation), pressurizer, steam generators, and feedwater system responses and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The case analyzed assumes a double-ended rupture of the largest feedwater pipe at full power. Major assumptions used in the analysis are as follows:

- The plant is initially operating at 101 percent of the design plant rating. The main feedwater flow measurement supports a 1-percent power uncertainty.

- Initial reactor coolant average temperature is 8.0°F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value.

- The pressurizer spray is turned on.

- Initial pressurizer level is at a conservative maximum value and a conservative initial steam generator water level is assumed in both steam generators.

- At the start of the transient, interaction between the break in the feedline and the main feedwater control system is assumed to result in a complete loss of feedwater flow to both steam generators. No feedwater flow is delivered to or lost through the steam generator nozzles.

- Reactor trip is assumed to be initiated by the low steam generator water level (narrow range) signal on the ruptured steam generator. A two-second delay is assumed following the low level setpoint being reached to allow for the system response times.

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- After reactor trip, the faulted steam generator blows down through a double-ended break area of 1,117 ft^2. A saturated liquid discharge is assumed until all the water inventory is discharged from the faulted steam generator. This minimizes the heat removal capability of the faulted steam generator and maximizes the resultant heatup of the reactor coolant. No feedwater flow is assumed to be delivered to the intact steam generator.

- The PRHR heat exchanger is assumed to be actuated by the low steam generator water level (wide range) signal. A 17-second delay is assumed following the low level setpoint being reached to allow for the system response times and the valve stroke time.

- Credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

- No credit is taken for charging or letdown.

- Pressurizer safety valve setpoint is assumed to be at its minimum value.

- Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases. The heat transfer remains approximately 100 percent in the faulted steam generator until the liquid mass reaches about 11 percent. The heat transfer is then reduced to 0 percent with the liquid inventory.

- Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip (Reference 3).

- No credit is taken for the following four protection and safety monitoring system reactor trip signals to mitigate the consequences of the accident:

  - High pressurizer pressure
  - Overtemperature ΔT
  - High pressurizer water level
  - High containment pressure

The PRHR heat exchanger is initiated once the steam generator water level drops to the low steam generator level (wide range). Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives an "S" signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

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No credit is taken for the plant control system, to mitigate the consequences of the event. The protection and safety monitoring system is required to function following a feedwater line rupture as analyzed here. No single active failure prevents operation of this system.

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The engineered safety features assumed to function are the PRHR heat exchanger, core makeup tank, and steam line isolation valves. The single failure assumed is the failure of one of the two parallel discharge valves in the PRHR outlet line (see Table 15.0-7).

A description and analysis of the core makeup tank is provided in subsection 6.3.2.2.1. The PRHR heat exchanger is described in subsection 6.3.2.2.5.

Deleted: For the case without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the reactor coolant system is shown (see subsection 15.2.6) to be sufficient to remove core decay heat following reactor trip for the loss of ac power transient. Pump coastdown characteristics are demonstrated in subsections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.¶

15.2.8.2.2 Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2.8-1 through 15.2.8-10. The calculated sequence of events for the case analyzed is listed in Table 15.2-1.

The results presented in Figures 15.2.8-5 and 15.2.8-7 show that pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure decreases after reactor trip on the low steam generator water level (narrow range) due to the loss of heat input.

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In the first part of the transient, due to the conservative analysis assumptions, the system response following the feedwater line rupture is similar to the loss of ac power to the station auxiliaries (subsection 15.2.6). Accordingly, like the loss of ac power event documented in subsection 15.2.6, the feedwater line rupture event is bounded by the turbine trip event presented in Section 15.2.3 with respect to DNB concerns.

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After

After the trip, the core makeup tanks are actuated, on low steam line pressure in the ruptured loop while the PRHR heat exchanger is actuated on a low steam generator water level (wide range).

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The addition of the PRHR heat exchanger and the core makeup tanks flow rates helps to cool down the primary system and to provide sufficient fluid to keep the core covered with water.

Pressurizer safety valves open due to the mismatch between decay heat and the heat transfer capability of the PRHR heat exchanger. In the first part of the transient, there is a cooling effect due to the core makeup tanks that inject cold water into the reactor coolant system and receive hot water from the cold leg. This effect decreases due to the heatup of the core makeup tanks from recirculation flow. Also, the injection driving head is lowered as the core makeup tanks heat up.

Reactor coolant system temperatures are low (approximately 510°F at about 2,500 seconds) and, in this condition, the PRHR heat exchanger cannot remove the entire decay heat load. Reactor coolant system temperatures increase until an equilibrium between decay heat power and heat absorbed by the PRHR heat exchanger is reached. After about 26,400 seconds, the heat transfer capability of the PRHR heat exchanger exceeds the decay heat power and the reactor coolant system temperatures, and pressure start to steadily decrease. Since subcooling is maintained throughout the transient and the reactor coolant system inventory increases (i.e., net core makeup tank injection exceeds net pressurizer safety valve relief), core cooling capability is maintained.

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15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the capacity of the PRHR heat exchanger is adequate to remove decay heat, to prevent overpressurization the reactor coolant system, and to maintain the core cooling capability. Radioactivity doses from postulated ruptures of the feedwater lines are less than those presented for the postulated main steam line break. The Standard Review Plan, subsection 15.2.8, evaluation criteria are therefore met.

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15.2.9 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.2.10 References

1. Cooper, L., Miselis, V., and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972. (Also letter NS-CE-622, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1, April 16, 1975).

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

3. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

4. Lang, G. E., and Cunningham, J. P., "Report on the Consequences of a Postulated Main Feedline Rupture," WCAP-9230 (Proprietary) and WCAP-9231 (Nonproprietary), January 1978.

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

6. "AP1000 Code Applicability Report," WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

7. Hargrove, H. G., "FACTRAN – A FORTRAN-TV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

8. Sung, Y .X. ,et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary) and WCAP -15306-NP-A (Nonproprietary), October 1999.

9. Matthys, C.,"Overpressure Protection Report for AP1000 Nuclear Power Plant, "WCAP-16779-NP, April 2007.

Comment [B6]: [15.2-6]

Table 15.2-1 (Sheet 1 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
I. Turbine trip		
A.1. With pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater	0.0
	Minimum DNBR (2.336) occurs	10.7
	Initiation of steam release from steam generator safety valves	11.5
	OTDT reactor trip setpoint reached	19.1
	Rods begin to drop	21.1
A.2. With pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Minimum DNBR (1.575/1.554, typical/thimble) occurs	6.2
	Initiation of steam release from steam generator safety valves	16.6

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Table 15.2-1 (Sheet 2 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
B.1. With pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR (2.393) occurs	0.0[1]
	Initiation of steam release from steam generator safety valves	11.7
	OTDT reactor trip setpoint reached	21.0
	Rod motion begins	23.0
B.2. With pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Minimum DNBR (2.168/2.117 typical/thimble) occurs	5.2
	Initiation of steam release from steam generator safety valves	18.8

(1) Minimum DNB never drops below initial value.

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Table 15.2-1 (Sheet 3 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
C.1. Without pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	5.1
	Rods begin to drop	7.1
	Initiation of steam release from steam generator safety valves	8.9
	Peak RCS pressure (2728psia) occurs	8.9
C.2. Without pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Peak RCS pressure (2708 psia)occurs	6.4
	Initiation of steam release from steam generator safety valves	10.7

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Table 15.2-1 (Sheet 4 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
D.1. Without pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip	5.1
	Rods begin to drop	7.1
	Peak RCS pressure (2710 psia) occurs	8.2
	Initiation of steam release from steam generator safety valves	8.8
D.2. Without pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Low reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Peak RCS pressure (2668 psia) occurs	6.1
	Initiation of steam release from steam generator safety valves	10.9

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Table 15.2-1 (Sheet 5 of 8)		
TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM		
Accident	**Event**	**Time (seconds)**
II.A. Loss of ac power to the plant auxiliaries	Offsite ac power is lost, feedwater is lost, RCPs begin to coast down, turbine trip	0.0
	RCP speed- low reactor trip set point is reached	0.5
	Rods begin to drop	1.3
	Pressurizer safety valves open	~3.0
	Maximum pressurizer pressure reached	3.0
	Pressurizer safety valves close	~7.5
	Pressurizer safety valves open	47.0[1]
	Steam generator 1 safety valves open	89.0[1]
	Steam generator 2 safety valves open	91.0[1]
	Maximum pressurizer water volume reached	401.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up flow rate)	401.0
	PRHR heat exchanger extracted heat matches decay heat	~18,500

1. The pressurizer safety valves open and close from 47.0 seconds until the time the maximum pressurizer water volume is reached. The steam generator safety valves in Loops 1 and 2 also cycled open and closed from 89.0 and 91.0 seconds, respectively, until the time the maximum pressurizer water volume was reached.

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Table 15.2-1 (Sheet 6 of 8) TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM		
Accident	**Event**	**Time (seconds)**
IIIA. Loss of normal feedwater flow	Feedwater is lost	0.0
	Low steam generator water level (narrow range) reactor trip reached	48.2
	Rods begin to drop	50.2
	Minimum DNBR is reached	51.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up feeedwater flow rate)	110.2
	Cold leg temperature reaches low T_{cold} setpoint	1,915.7
	Reactor coolant pump trip on low T_{cold} "S" signal	1,922.4
	Steam line isolation on low T_{cold} "S" signal	1,927.7
	Core makeup tank actuation on low T_{cold} "S" signal	1,932.7
	The chemical volume and control system is isolated on "S" signal and Pressurizer Water Level –High1	1,953.2
	Pressurizer safety valves open	~2,452.0
	High-2 pressurizer level setpoint reached	2,602.0
	High-3 pressurizer level setpoint reached	3,958.0
	Operator opens reactor vessel head vent (at least 30 minutes after high-2 pressurizer level setpoint is reached)	4,402.0
	Pressurizer safety valves reclose	~4,394.0
	Maximum pressurizer water volume reached	5,894.0

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Table 15.2-1 (Sheet 7 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
III.B Loss of normal feedwater flow with a consequential loss of ac power	Feedwater is lost	10.0
	Low steam generator water level setpoint is reached	58.2
	Rods begin to drop	60.2
	Minimum DNBR is reached	61.0
	RCP trip due to loss of ac power	67.6
	Steam generator safety valves open	98.6
	Pressurizer safety valves open	~104.5
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up flow rate)	120.2
	Pressurizer safety valves close	~137.0
	Pressurizer safety valves open	~1744
	Steam generator safety valves close	~2018[1]
	Pressurizer safety valves close	~2822[2]
	PRHR heat extraction matches decay heat addition	~3165
	Maximum pressurizer water volume reached	3584

1. Between 98.6 seconds and 2018 seconds the steam generator safety valves cycle open and closed. After 2018 seconds the steam generator safety valves intermittently relieve steam, but with a relief rate less than 1 lbm/second, which has a negligible effect on the transient.

2. Between 1744 seconds and 2822 seconds the pressurizer safety valves cycle open and closed.

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Table 15.2-1 (Sheet 8 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
IV. Feedwater system pipe break	Main feedwater flow to both steam generators stops due to interaction between the break and the main feedwater control system	10.0
	Low steam generator water level (narrow range) setpoint reached	60.3
	Rods begin to drop	62.3
	Reverse flow from the faulted steam generator through a full double-ended rupture starts	62.3
	Loss of offsite power	70.3
	Low steam line pressure setpoint is reached	76.7
	Core makeup tank valves fully opened	76.7
	Low steam generator water level (wide range) setpoint reached	81.7
	All steam isolation valves close	88.7
	PRHR heat exchanger actuation on low steam generator water level (wide range)	98.7
	Faulted steam generator empties	122.0
	Intact steam generator safety valves open for the first time	251.9
	Pressurizer safety valves open for the first time	1,792
	PRHR heat exchanger extracted heat matches decay heat	~26,400

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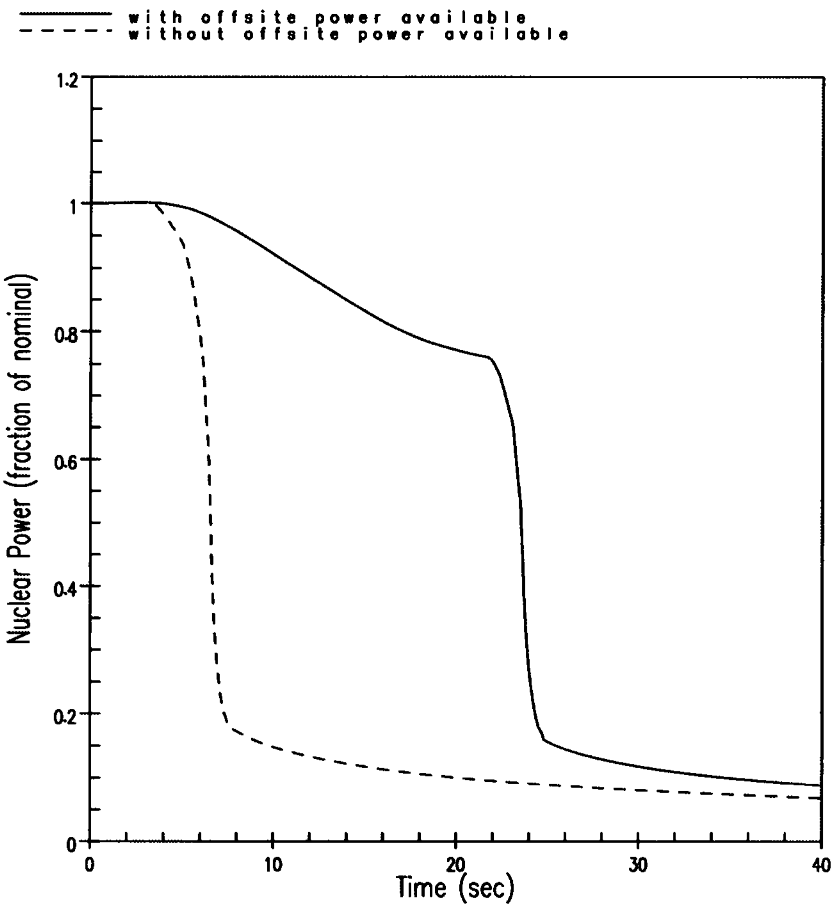

Figure 15.2.3-1

Nuclear Power versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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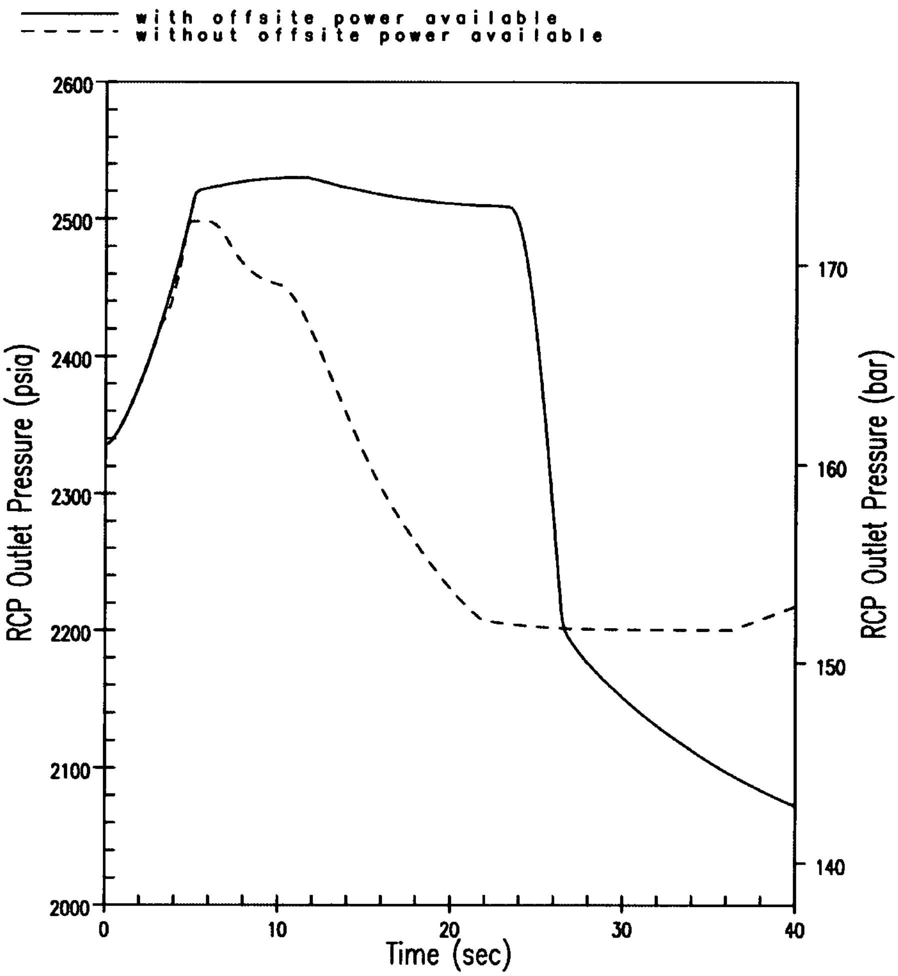

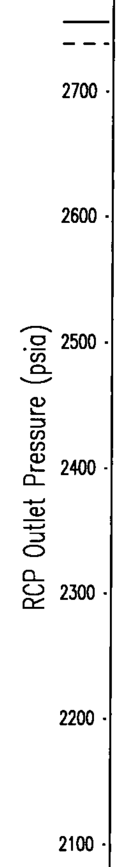

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Figure 15.2.3-2

RCP Outlet Pressure versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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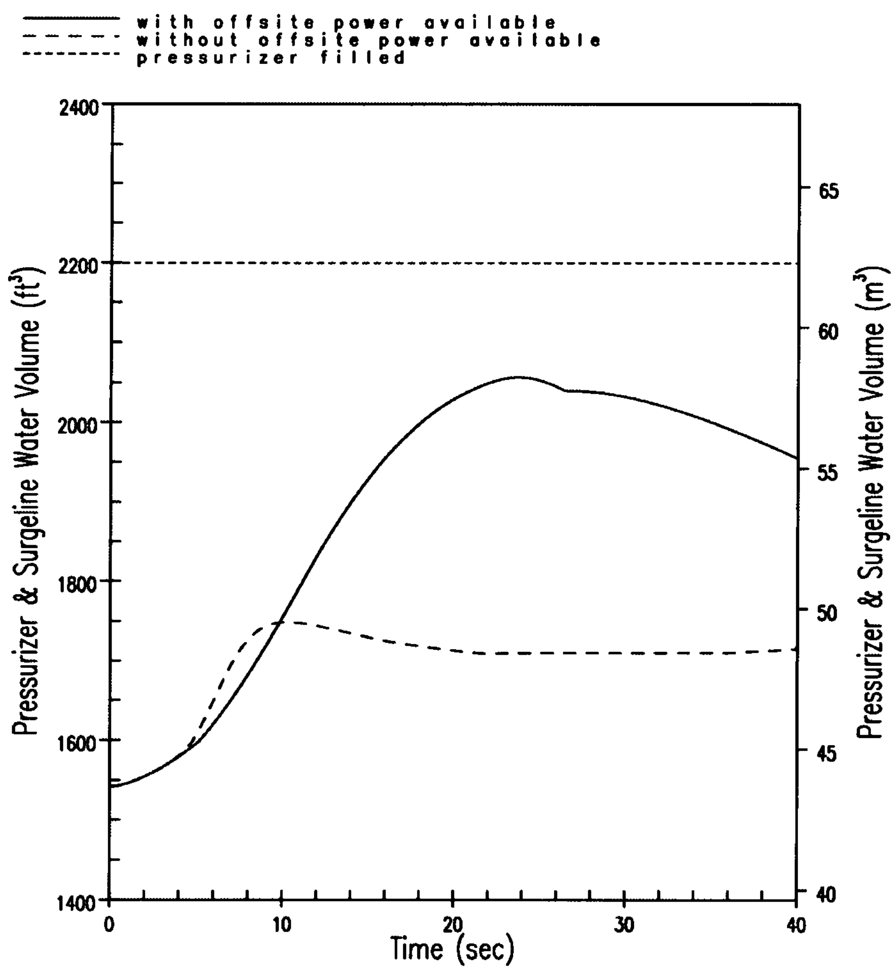

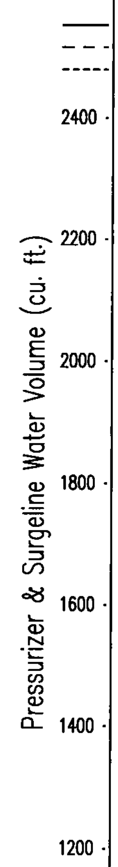

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Figure 15.2.3-3

Pressurizer & Surgeline Water Volume versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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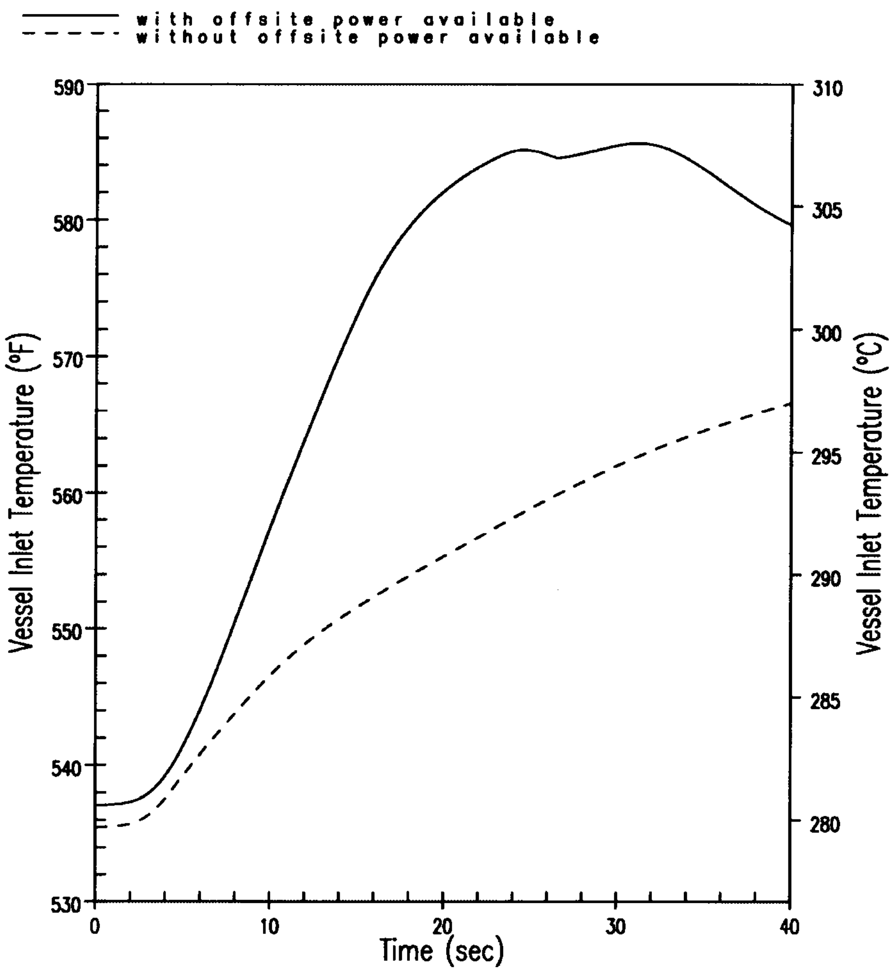

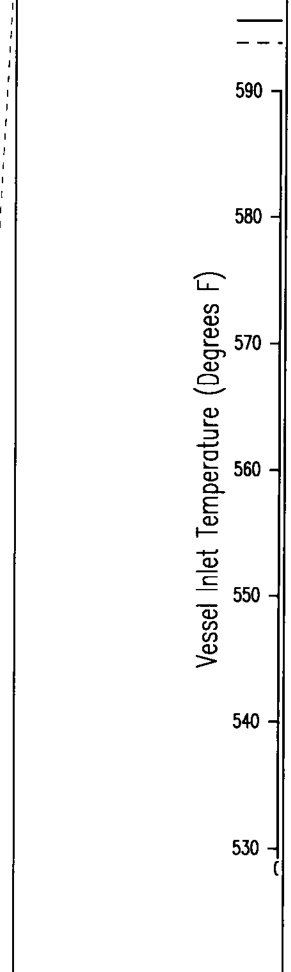

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Figure 15.2.3-4

Vessel Inlet Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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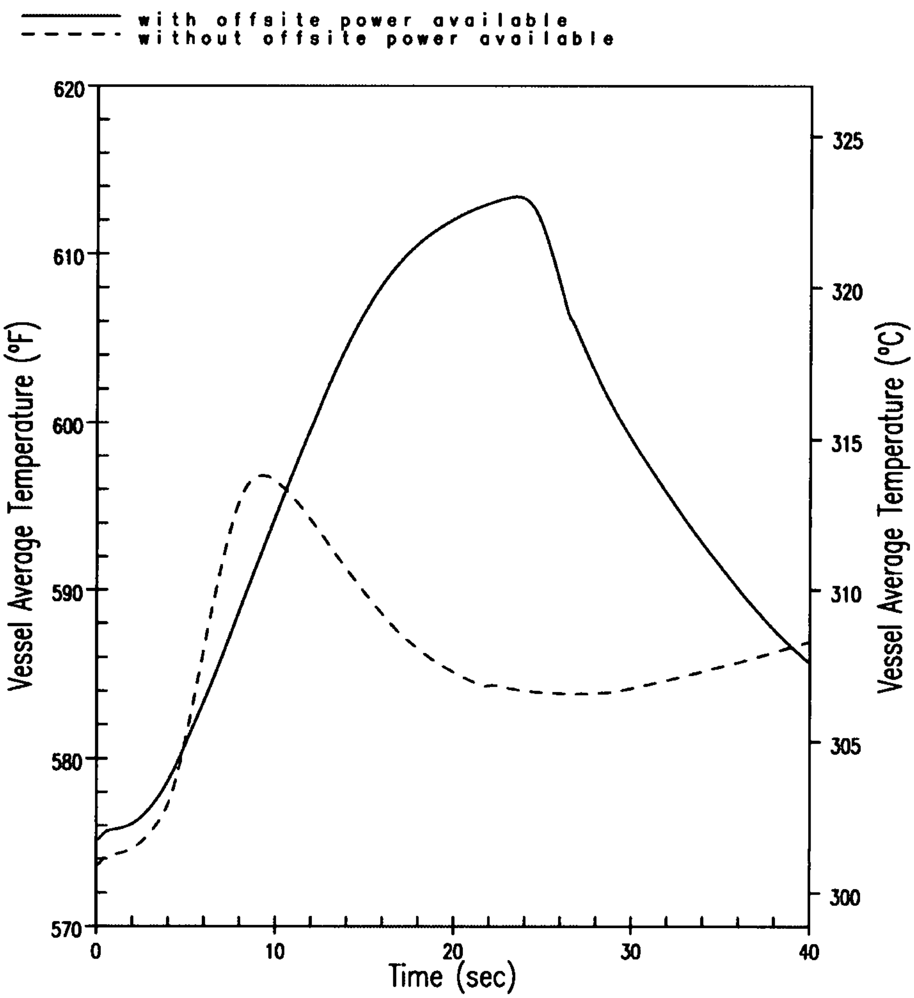

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Figure 15.2.3-5

Vessel Average Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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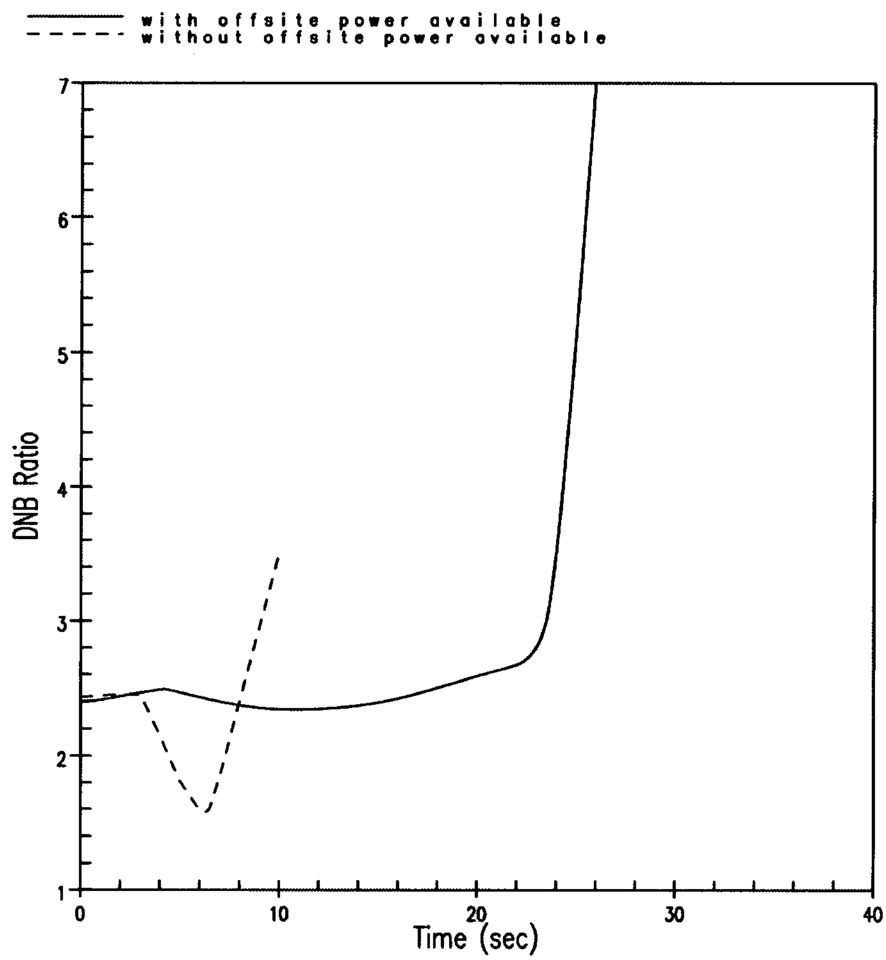

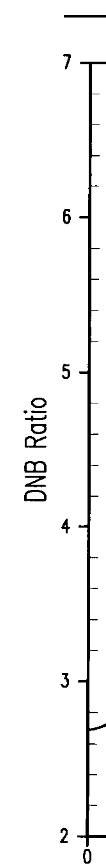

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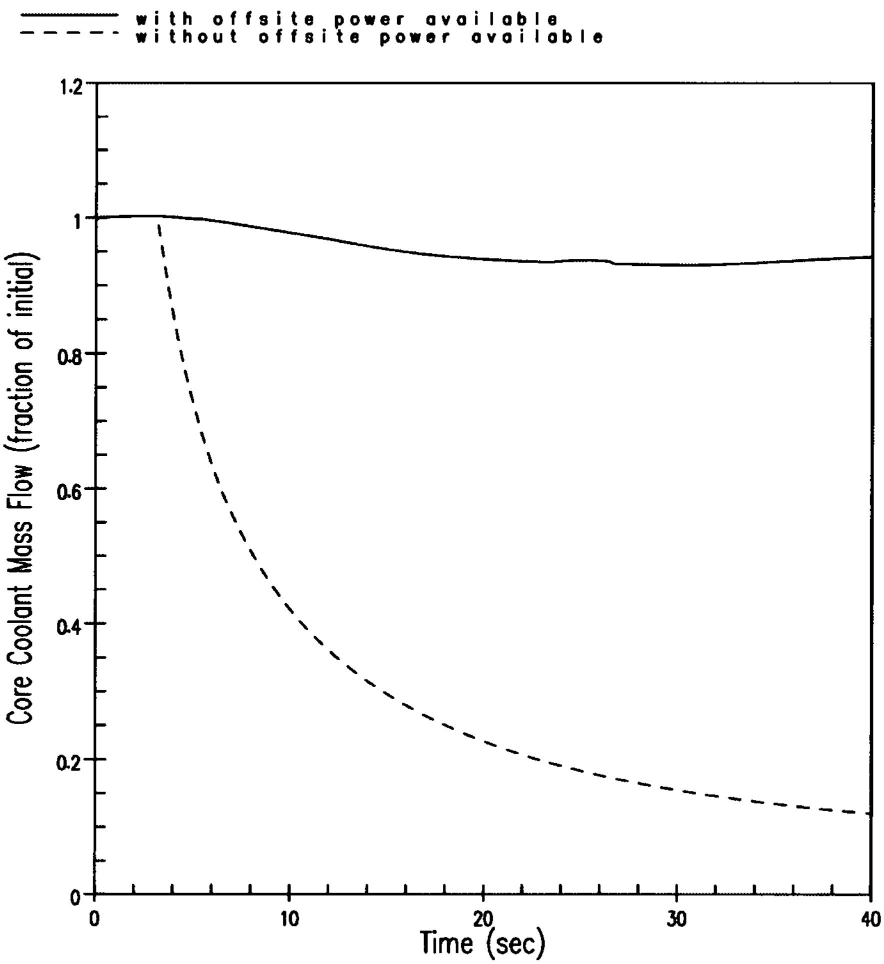
Figure 15.2.3-6

DNBR versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback



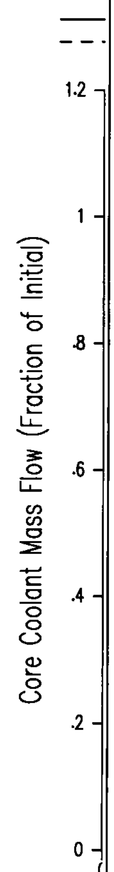


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Figure 15.2.3-7

Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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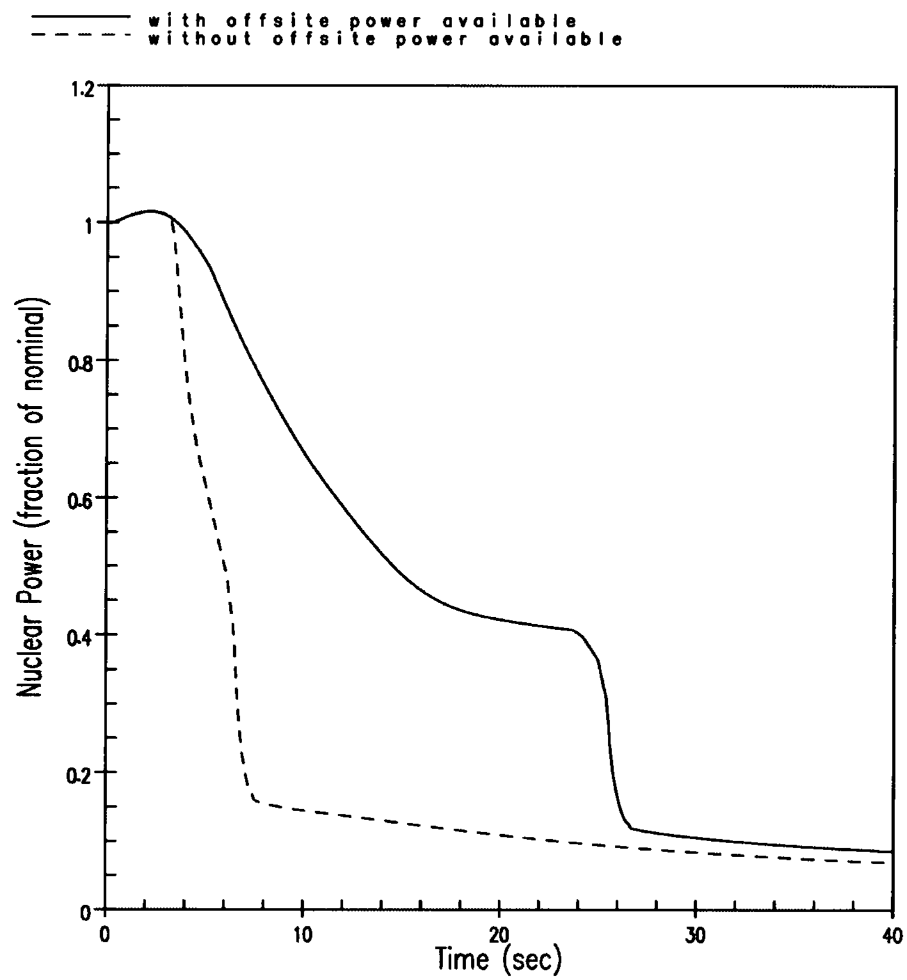

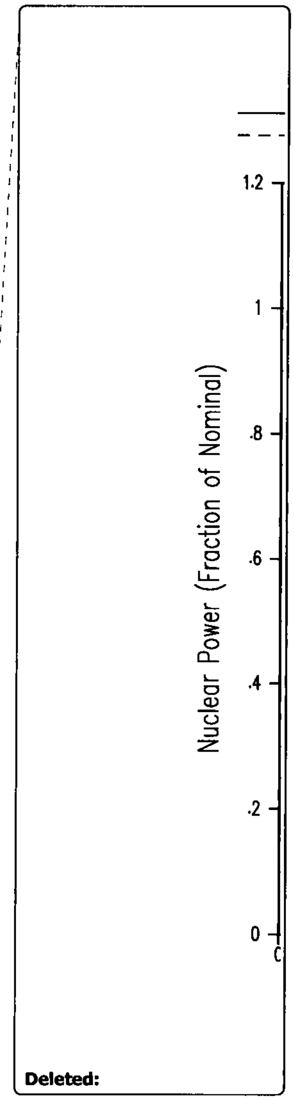

Figure 15.2.3-8

Nuclear Power versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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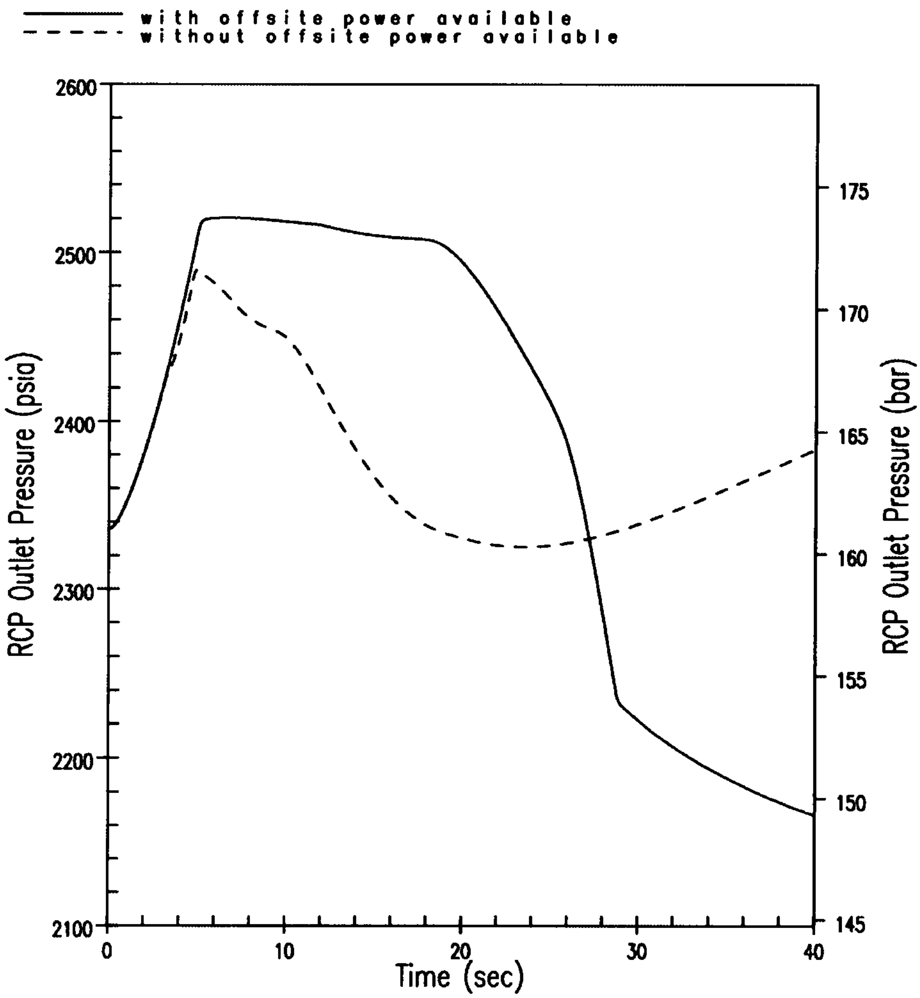

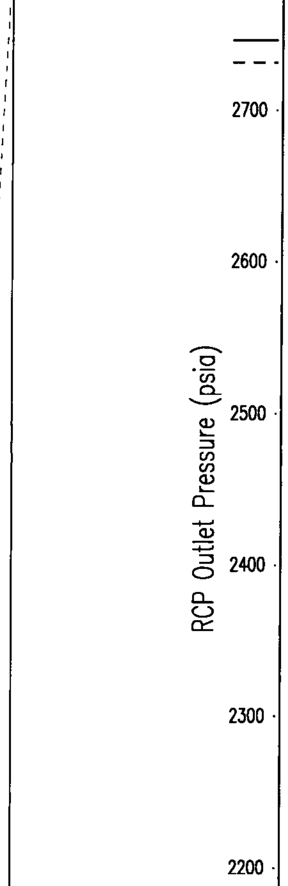

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Figure 15.2.3-9

RCP Outlet Pressure versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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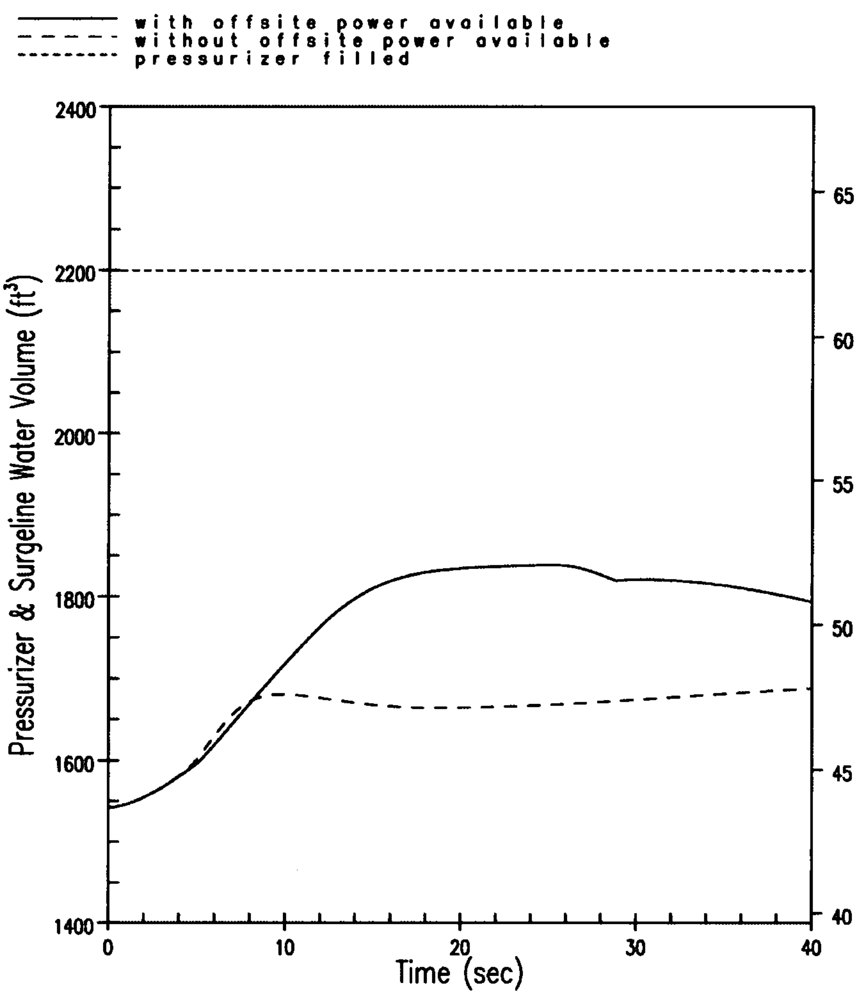

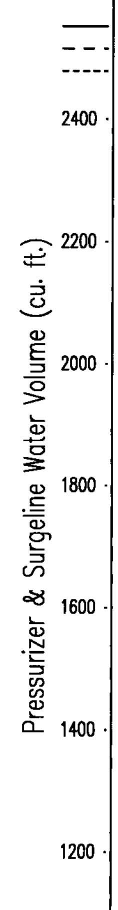

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Figure 15.2.3-10

Pressurizer & Surgeline Water Volume versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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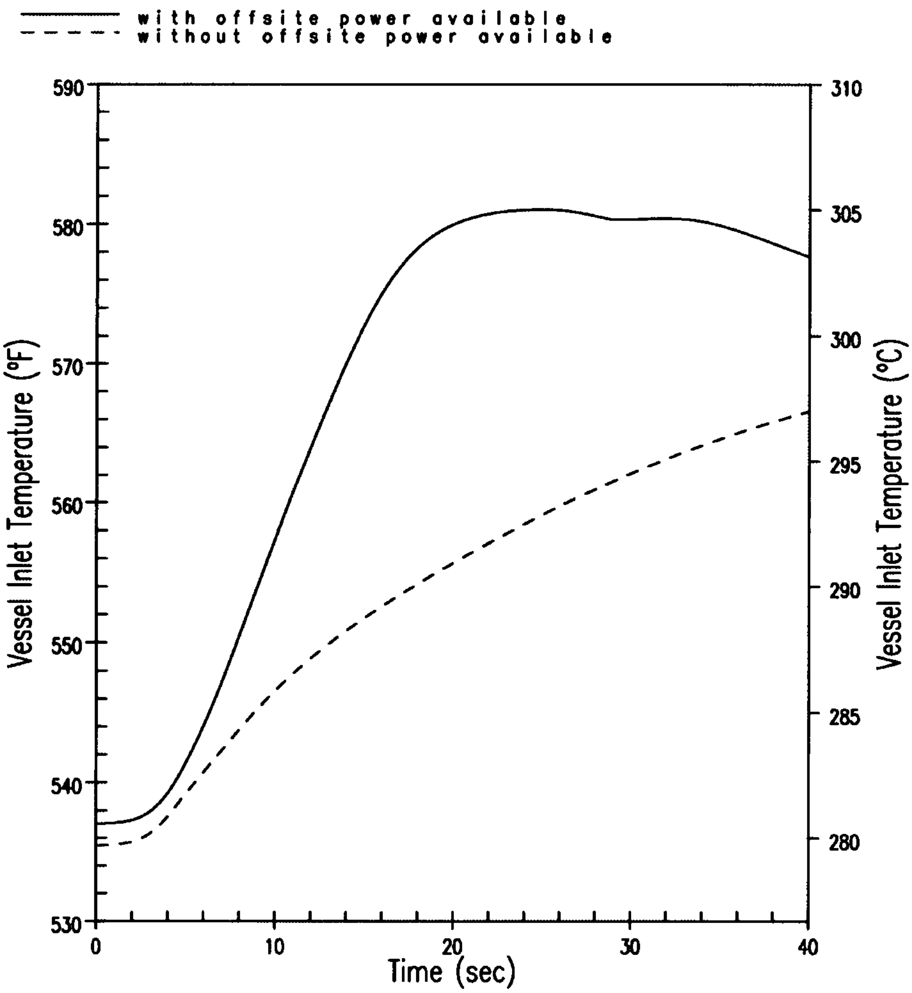

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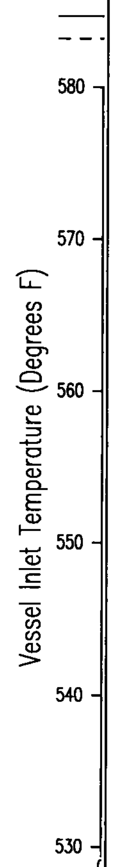

Figure 15.2.3-11

Vessel Inlet Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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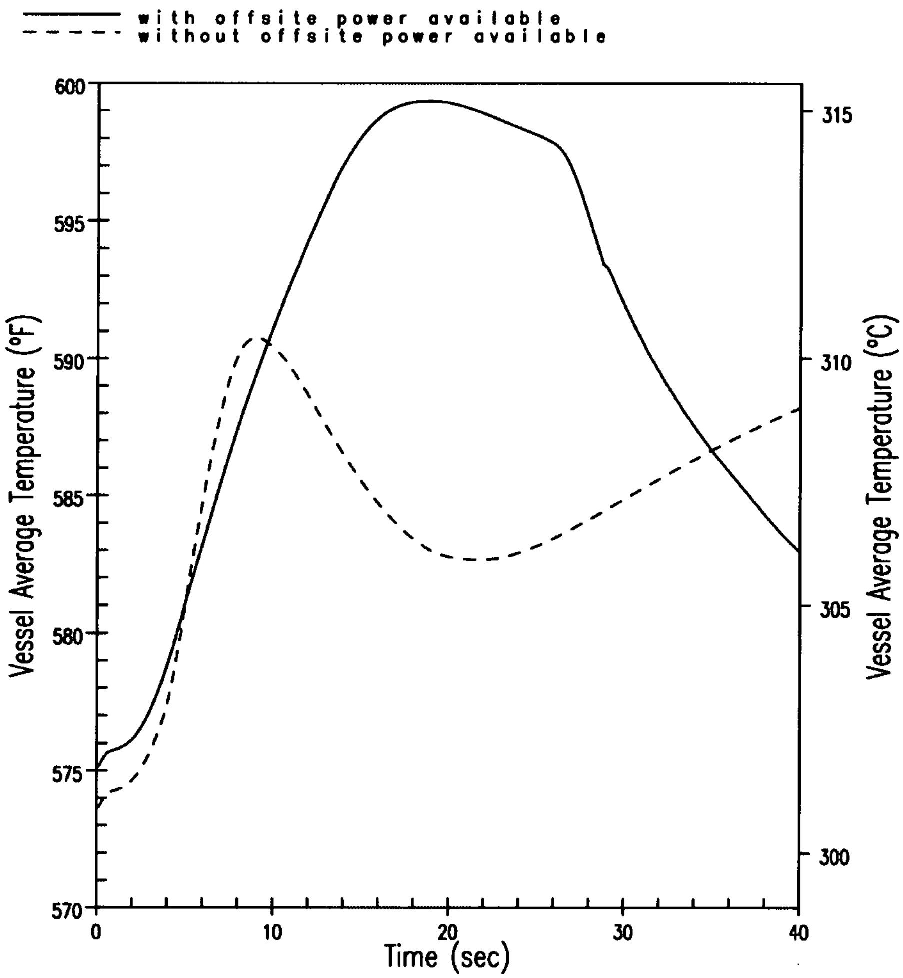

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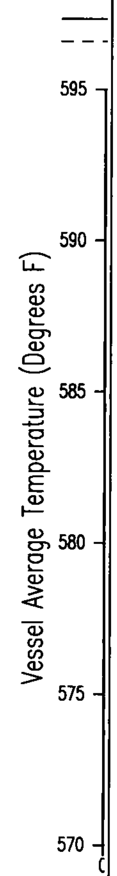

Figure 15.2.3-12

Vessel Average Temperature versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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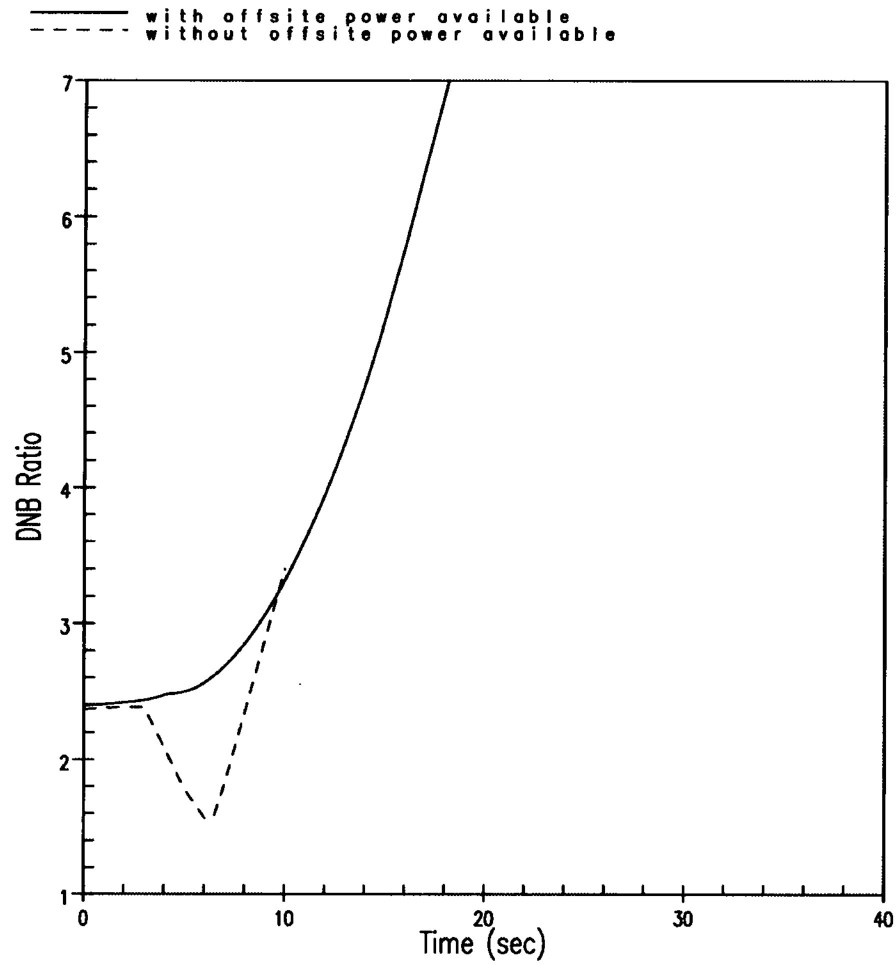

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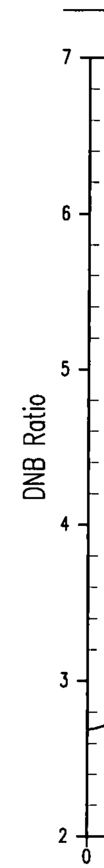

Figure 15.2.3-13

DNBR versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

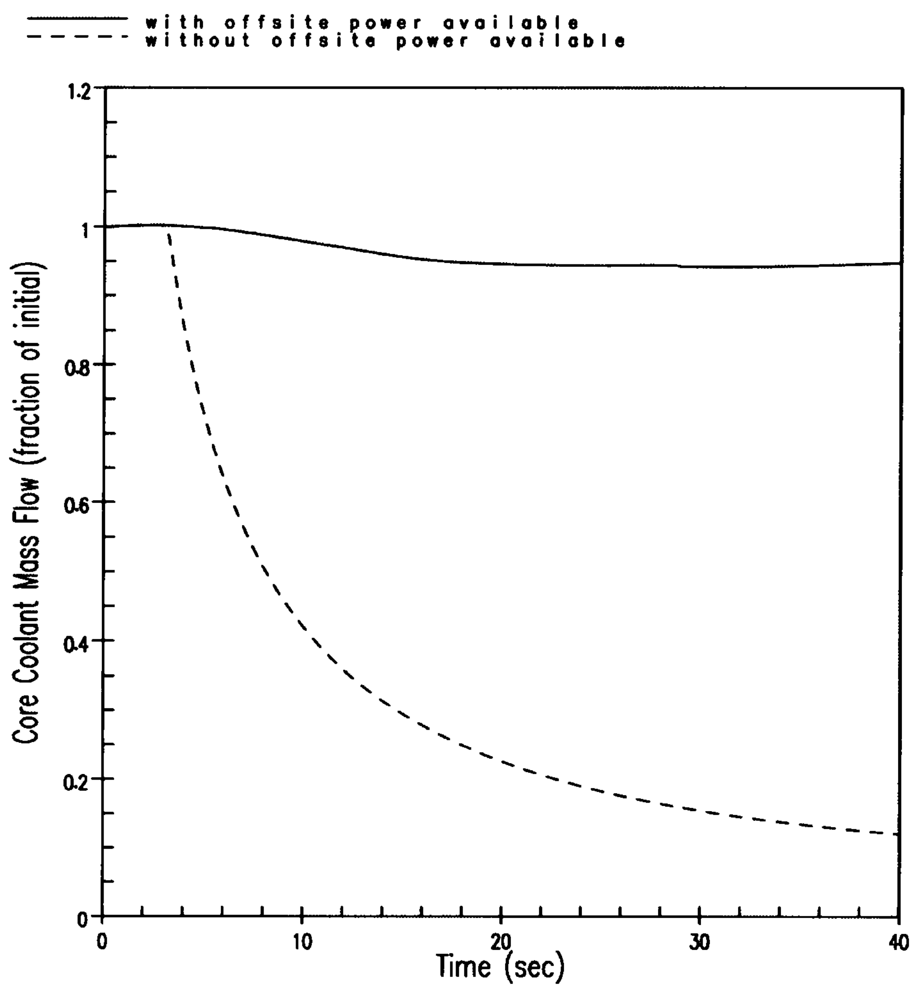

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Figure 15.2.3-14

Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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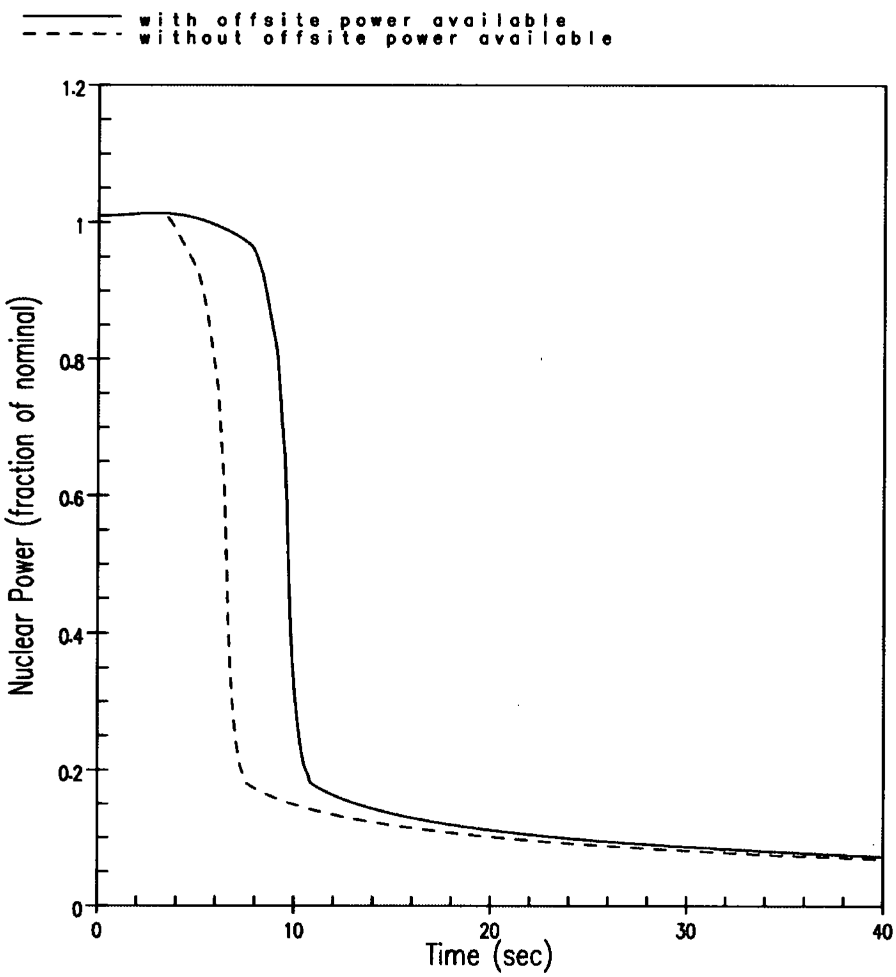

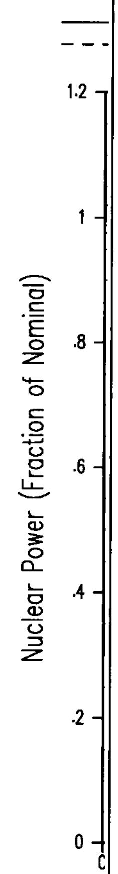

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Figure 15.2.3-15

Nuclear Power versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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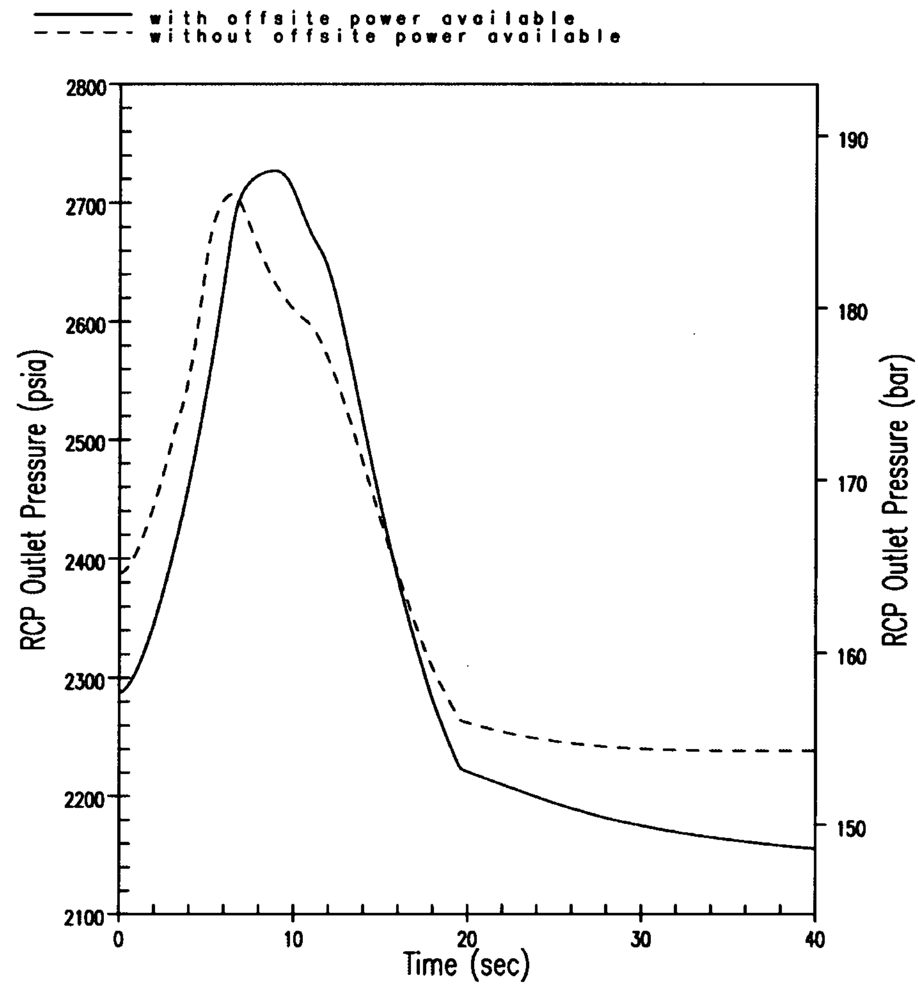

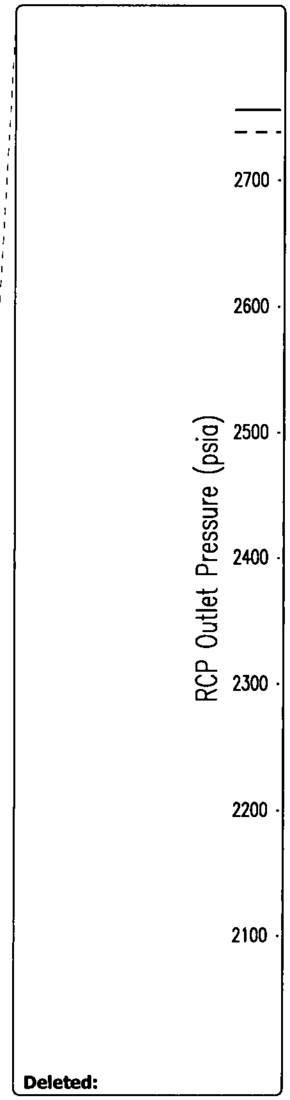

Figure 15.2.3-16

RCP Outlet Pressure versus Time for Turbine Trip
Accident Without Pressurizer Spray and Minimum Moderator Feedback

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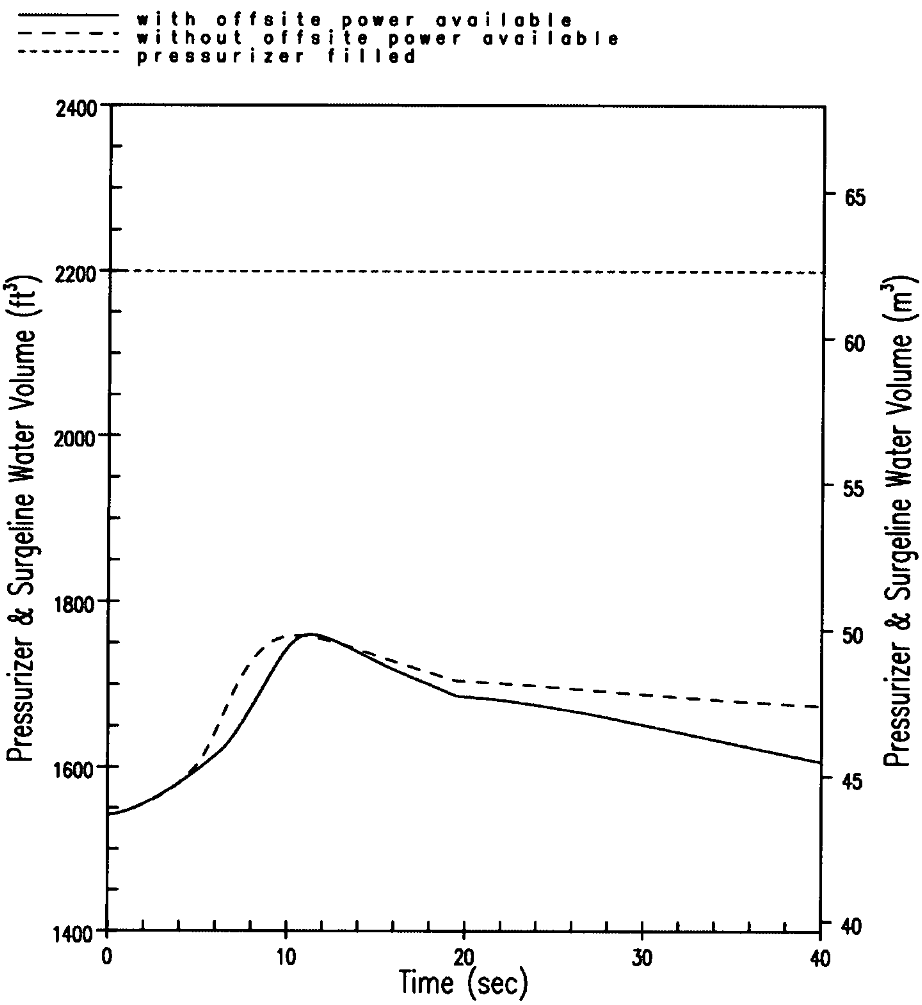

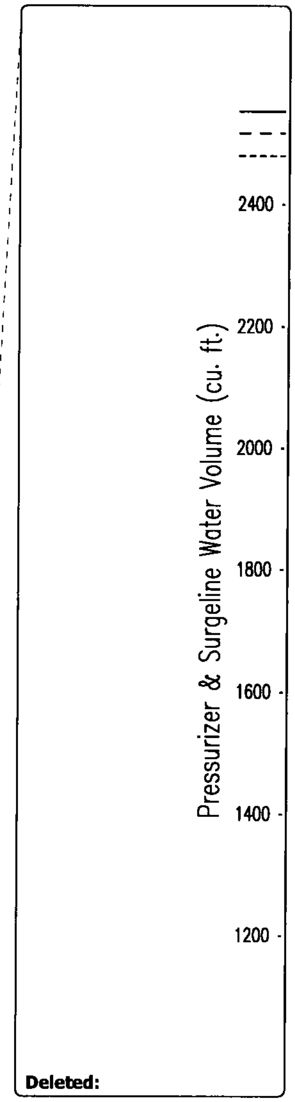

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Pressurizer & Surgeline Water Volume versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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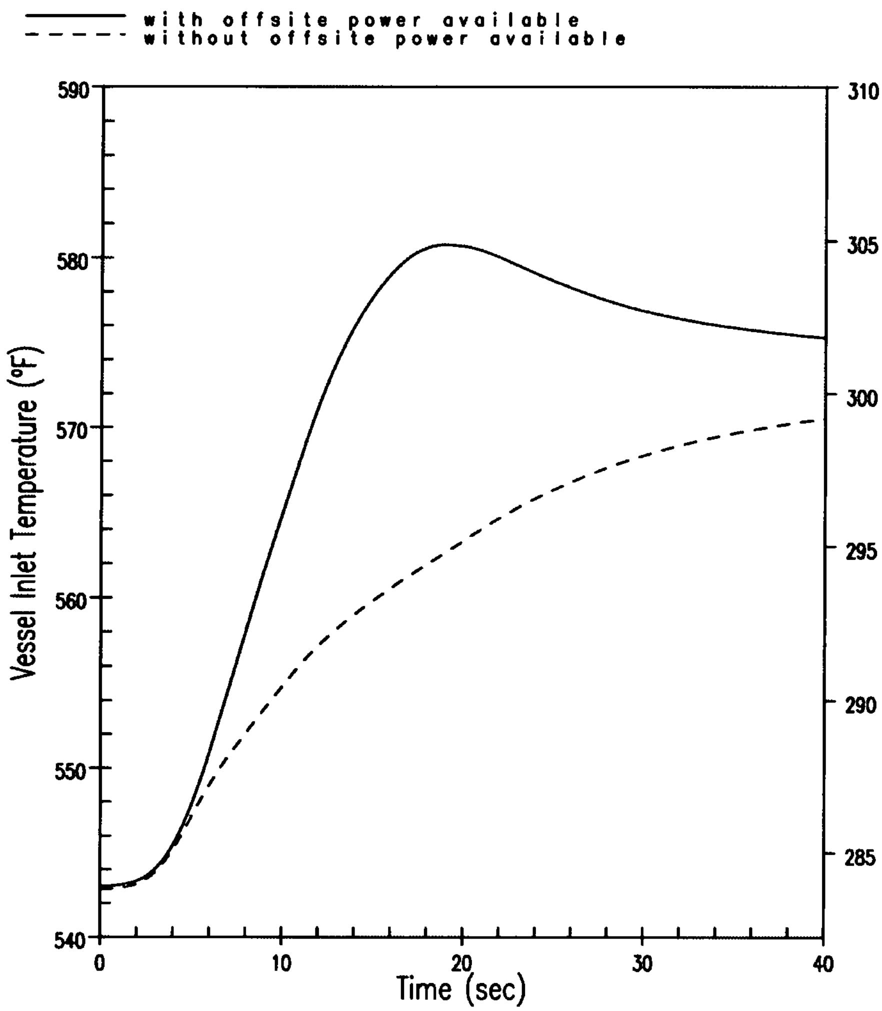

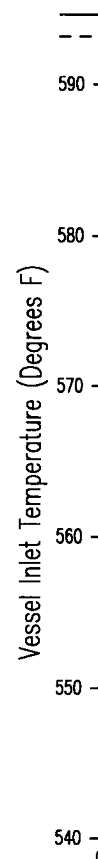

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Figure 15.2.3-18

Vessel Inlet Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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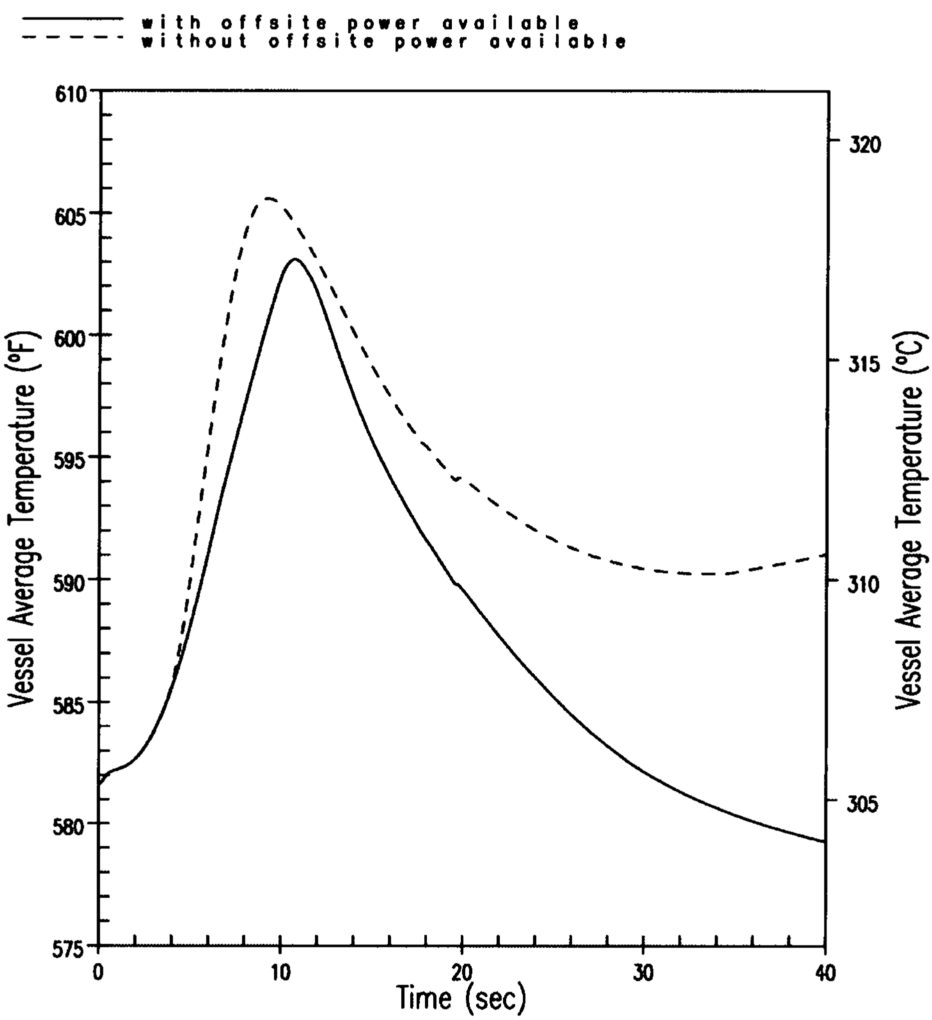

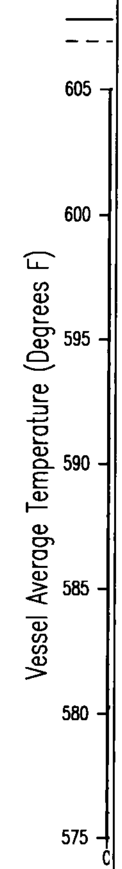

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Figure 15.2.3-19

Vessel Average Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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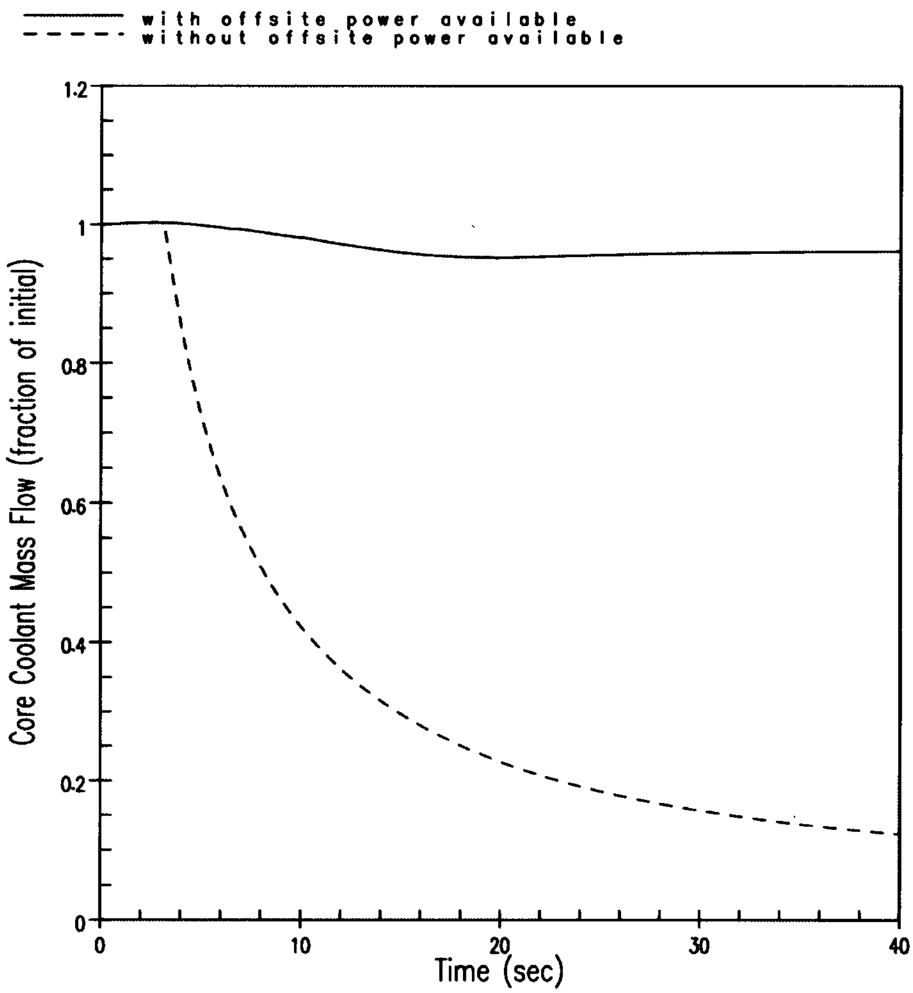

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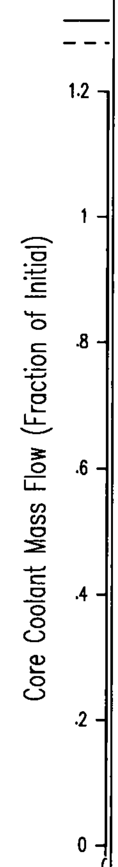

Figure 15.2.3-20

Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

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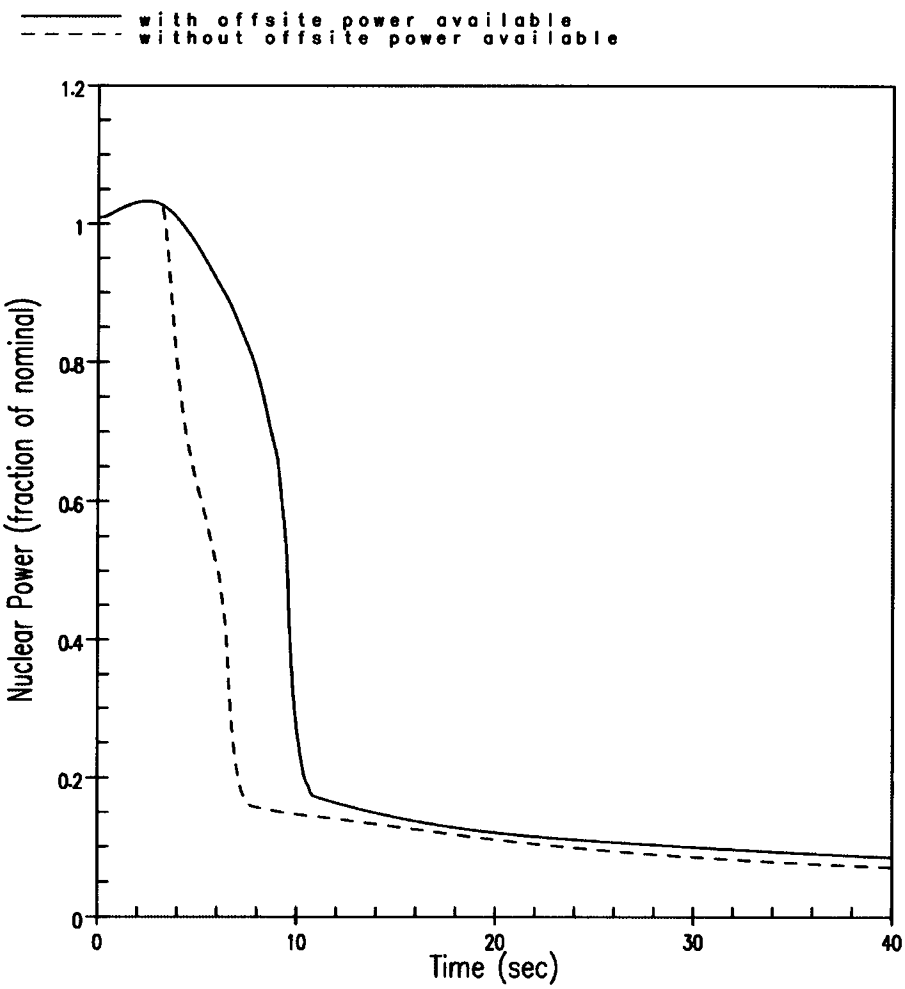

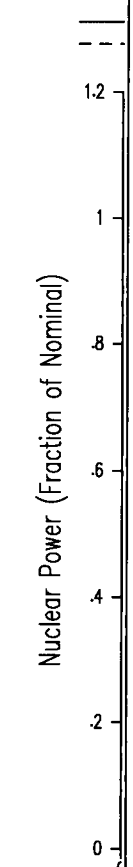

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Figure 15.2.3-21

Nuclear Power versus Time for Turbine Trip
Accident Without Pressurizer Spray and Maximum Moderator Feedback

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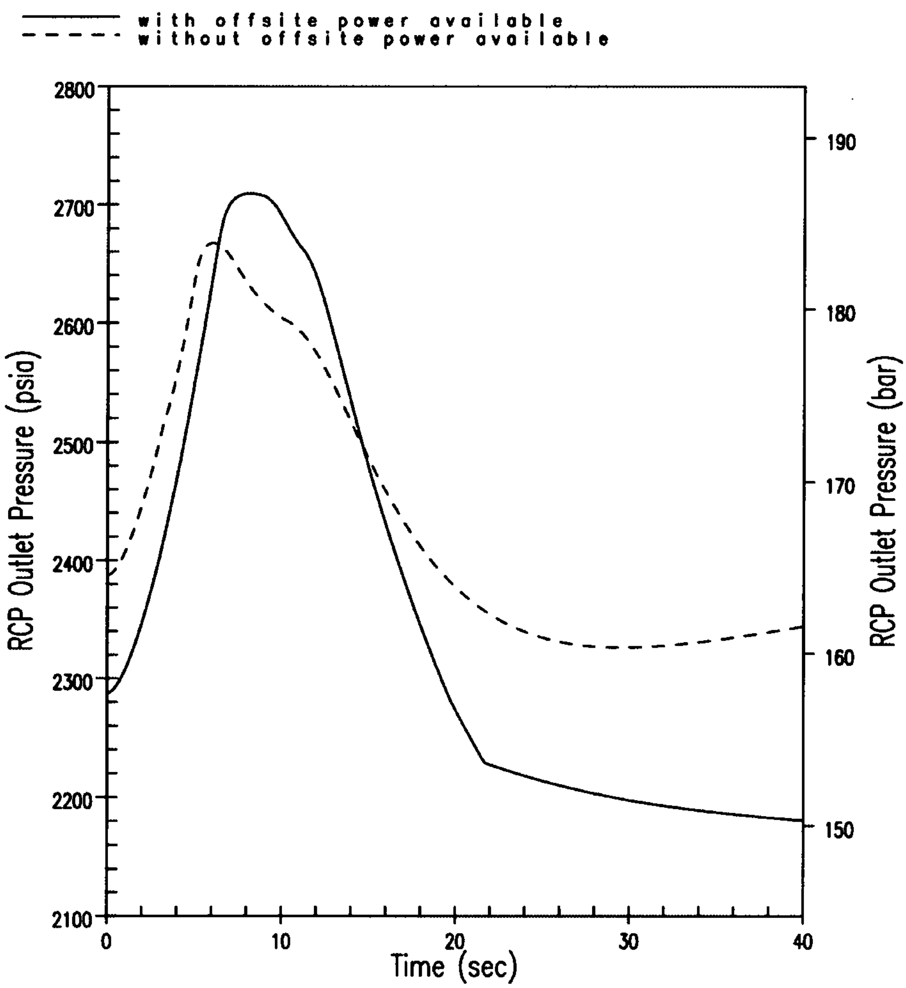

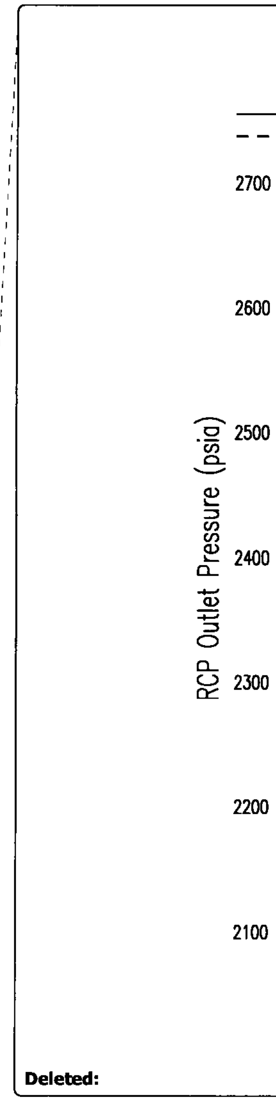

Figure 15.2.3-22

RCP Outlet Pressure versus Time for Turbine Trip
Accident Without Pressurizer Spray and Maximum Moderator Feedback

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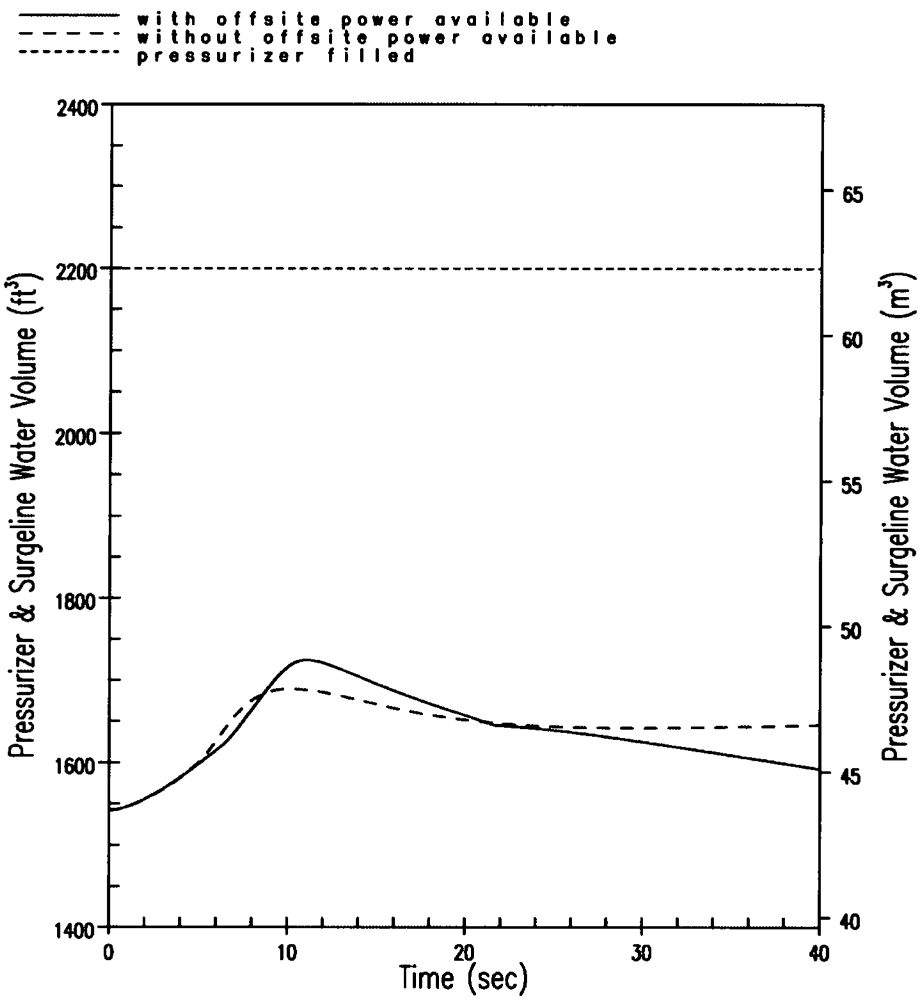

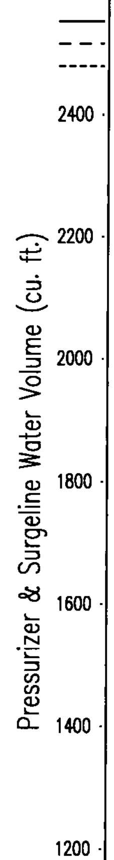

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Figure 15.2.3-23

Pressurizer & Surgeline Water Volume versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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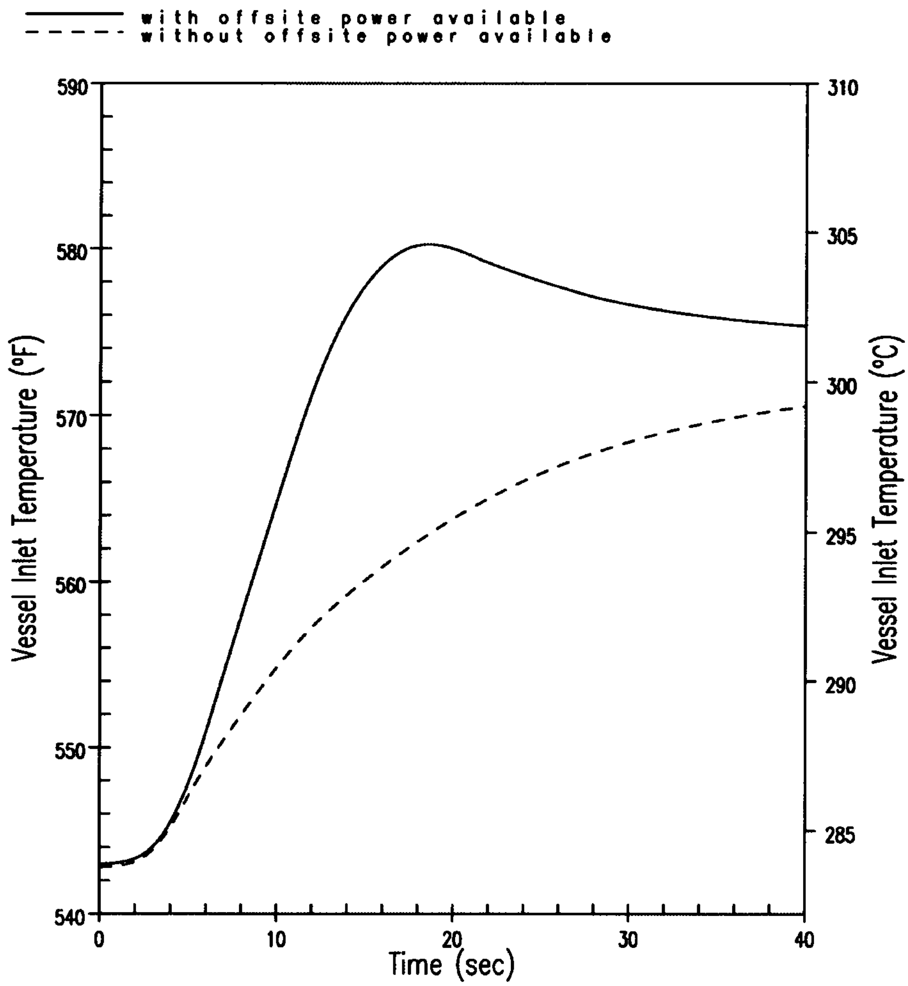

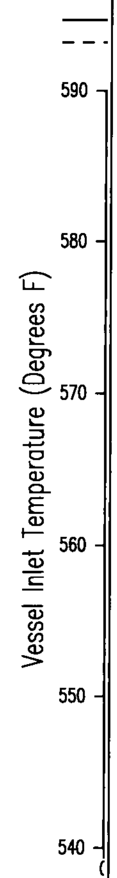

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Figure 15.2.3-24

Vessel Inlet Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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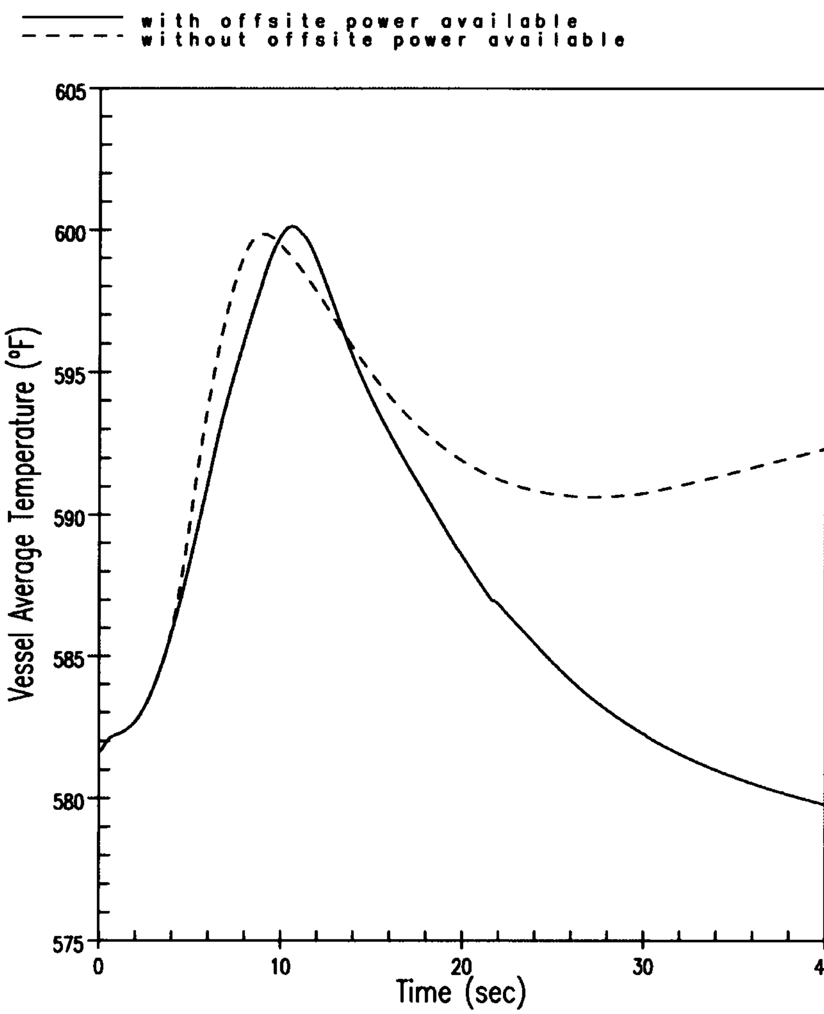

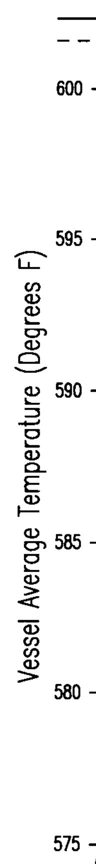

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Figure 15.2.3-25

Vessel Average Temperature versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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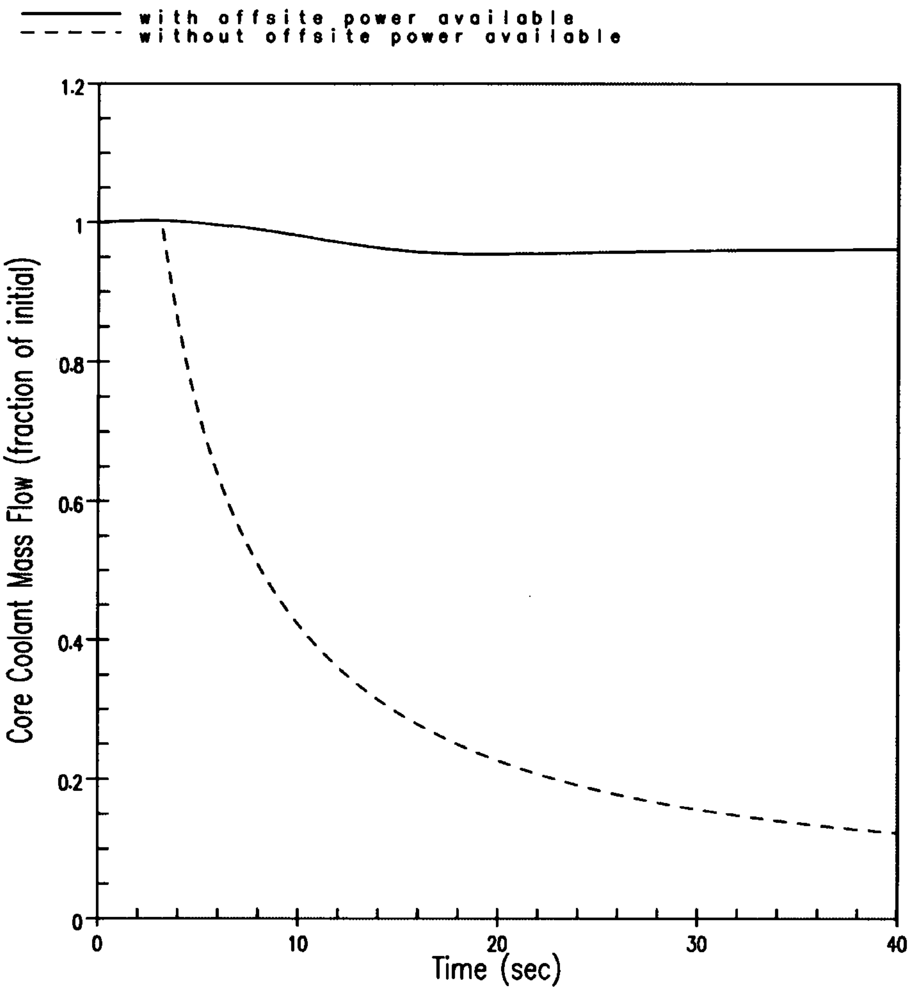

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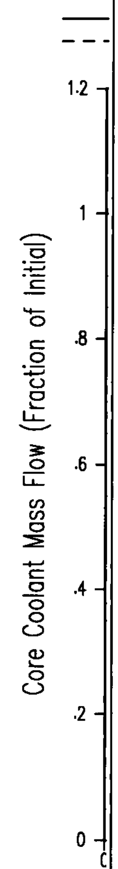

Figure 15.2.3-26

Core Coolant Mass Flow Rate versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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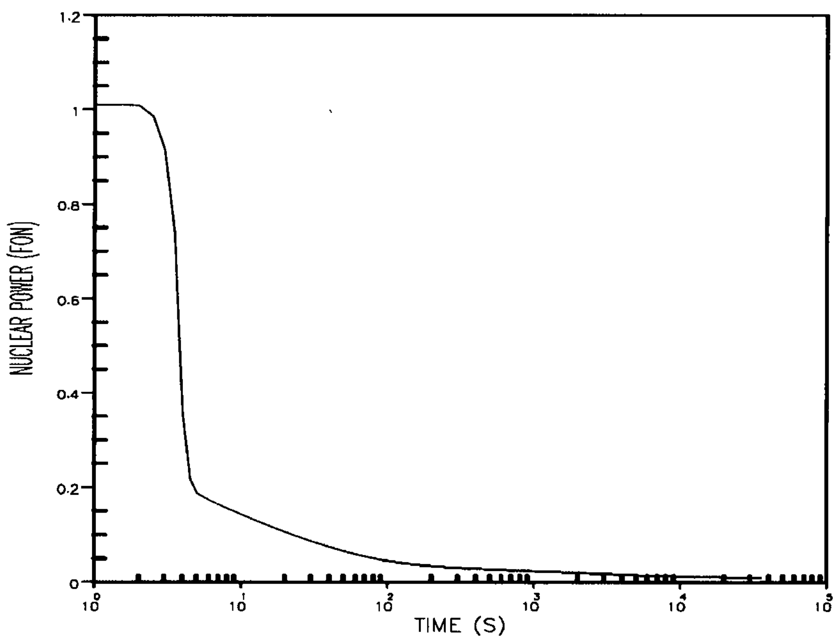

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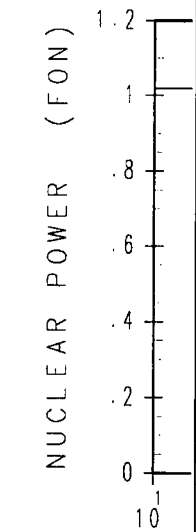
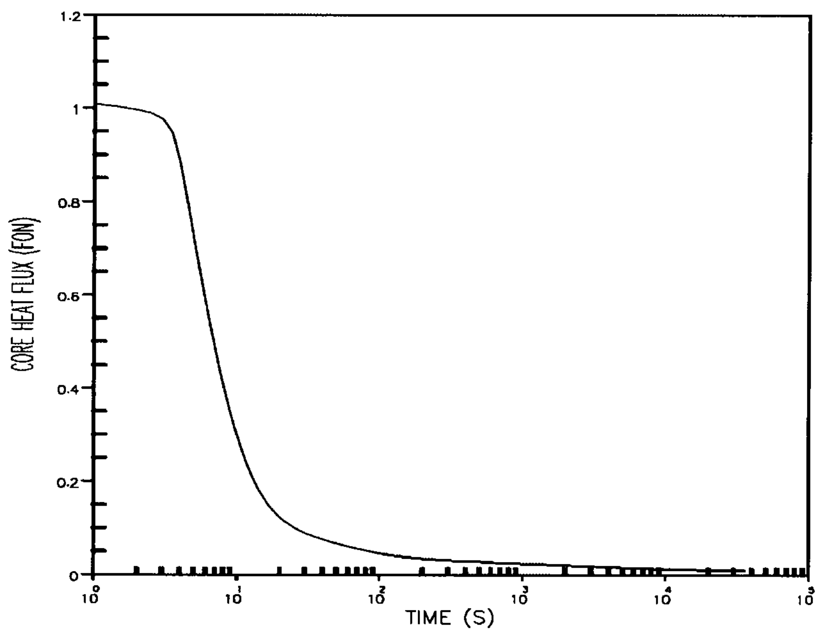


Figure 15.2.6-1

Nuclear Power Transient for Loss of ac Power to the Plant Auxiliaries



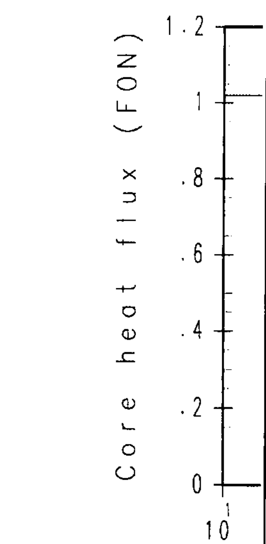
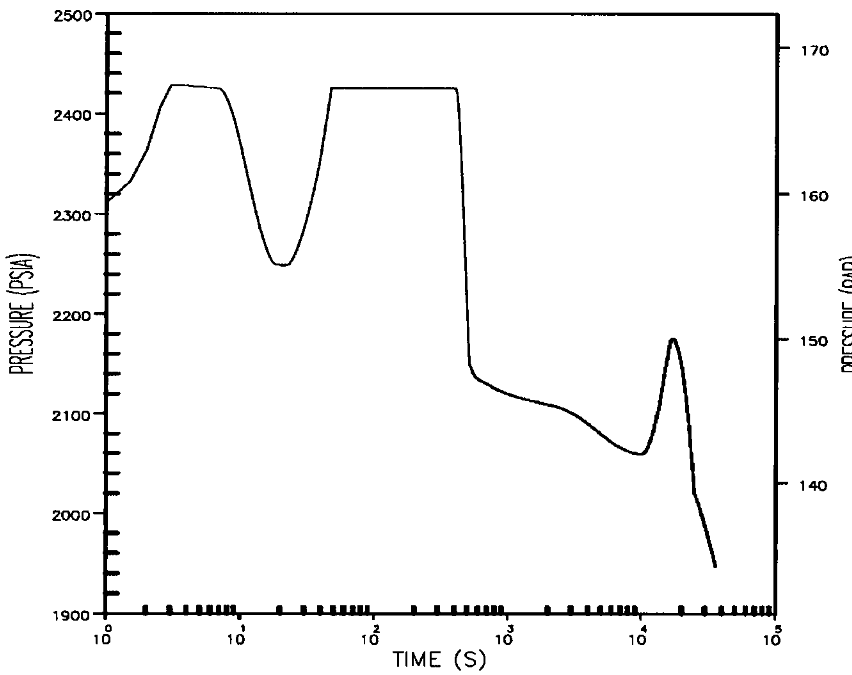


Figure 15.2.6-2

Core Heat Flux Transient for Loss of ac Power to the Plant Auxiliaries



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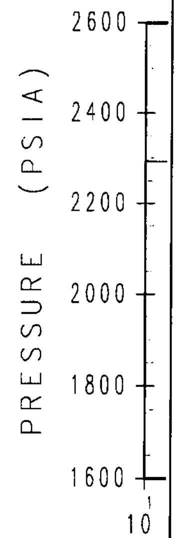
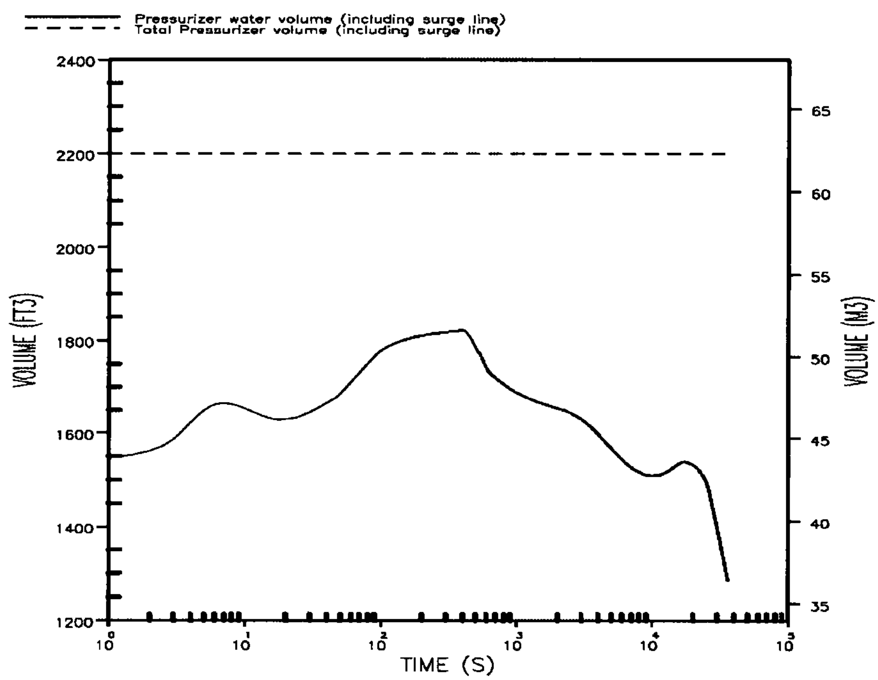


Figure 15.2.6-3

Pressurizer Pressure Transient for Loss of ac Power to the Plant Auxiliaries



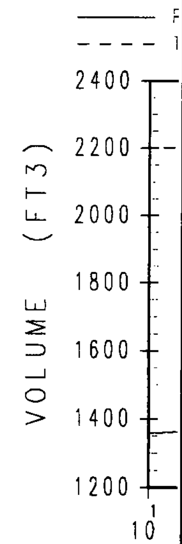


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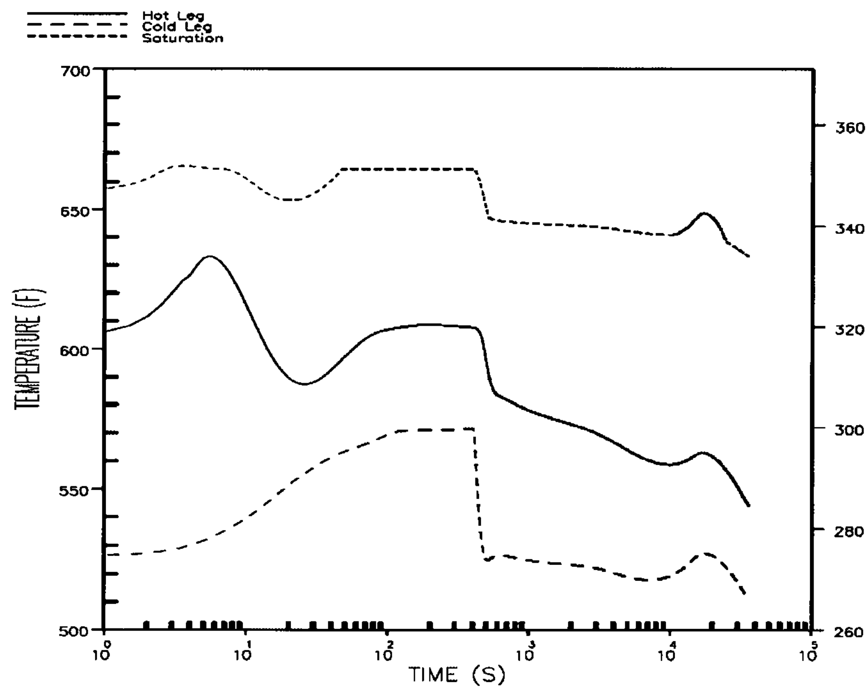
Figure 15.2.6-4

Pressurizer Water Volume Transient for Loss of ac Power to the Plant Auxiliaries



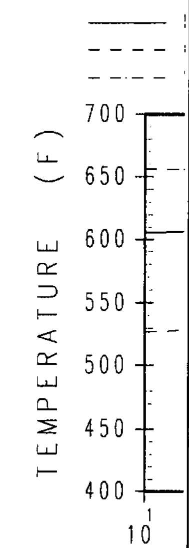


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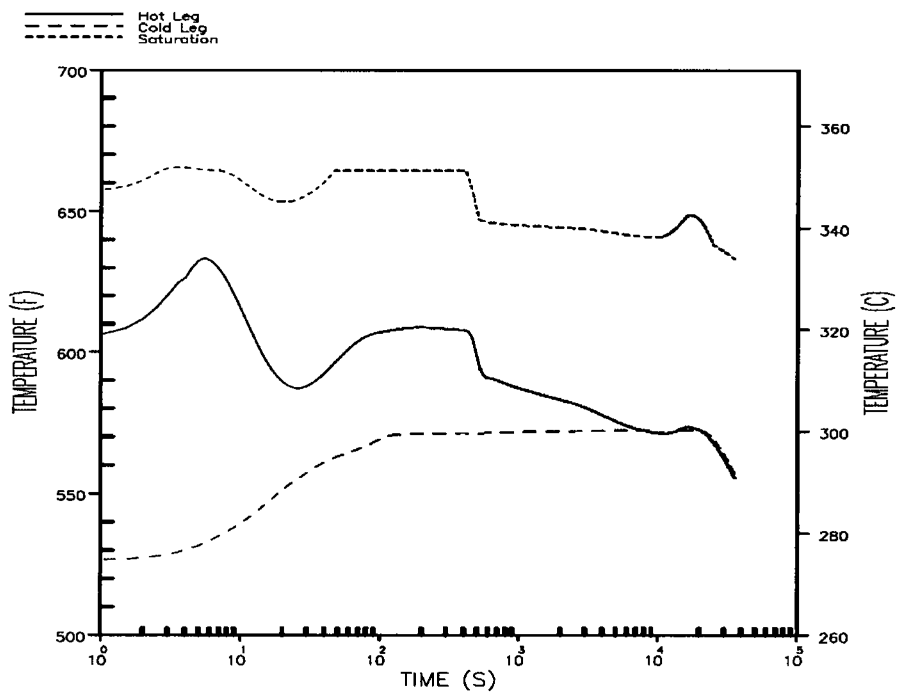
Figure 15.2.6-5

Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of ac Power to the Plant Auxiliaries



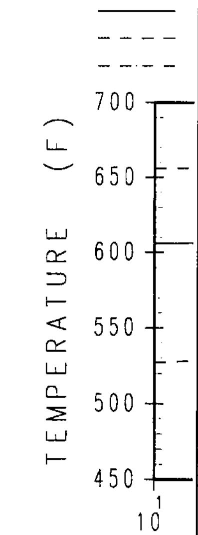


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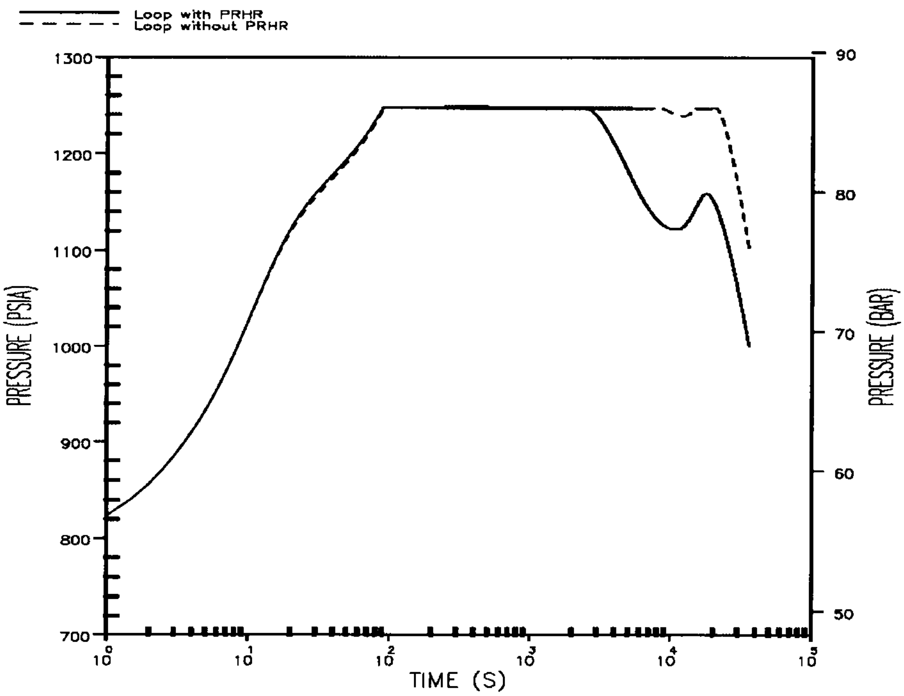
Figure 15.2.6-6

Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of ac Power to the Plant Auxiliaries



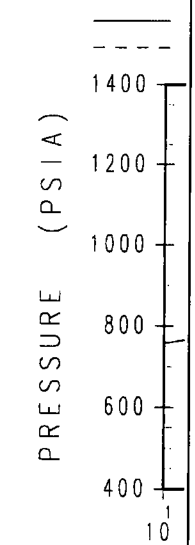


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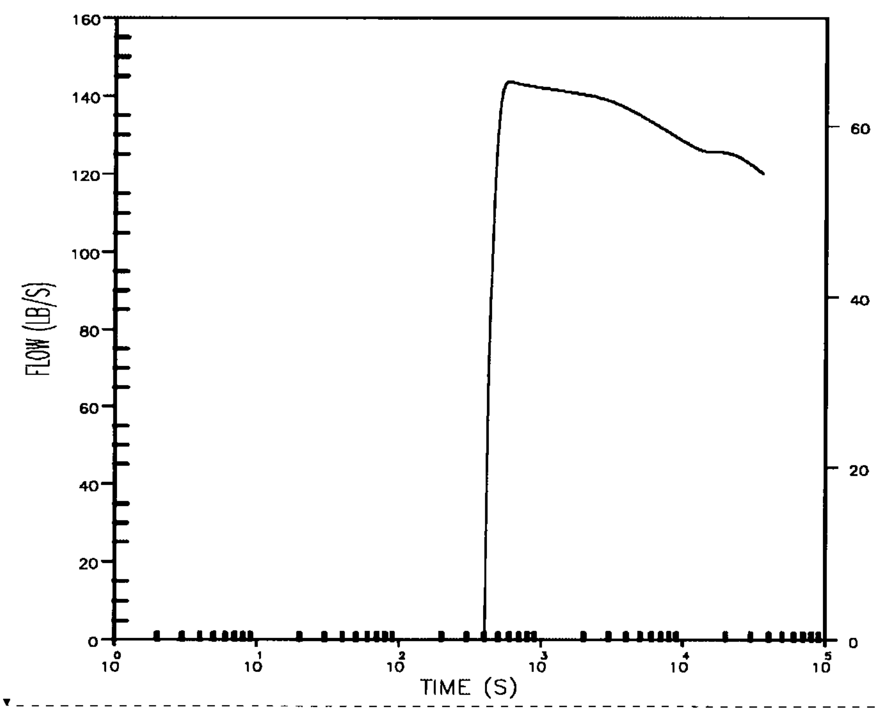
Figure 15.2.6-7

Steam Generator Pressure Transient for Loss of ac Power to the Plant Auxiliaries



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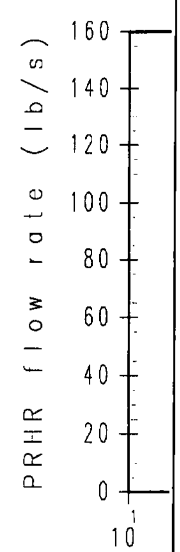
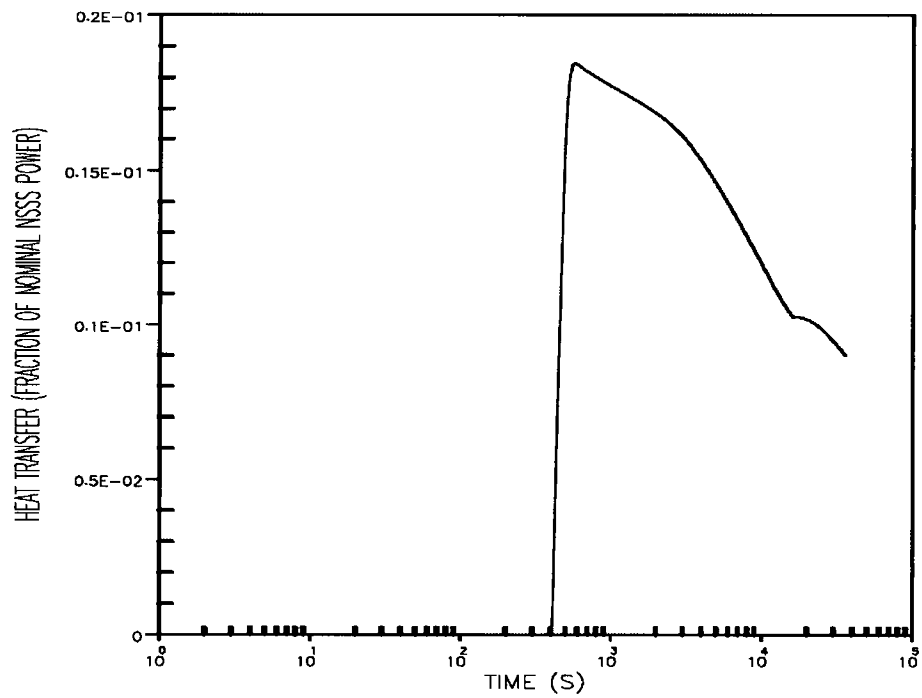


Figure 15.2.6-8

PRHR Flow Rate Transient
for Loss of ac Power to the Plant Auxiliaries



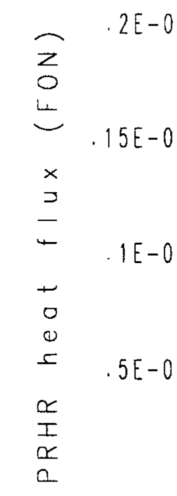


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Figure 15.2.6-9

PRHR Heat Transfer Transient
for Loss of ac Power to the Plant Auxiliaries

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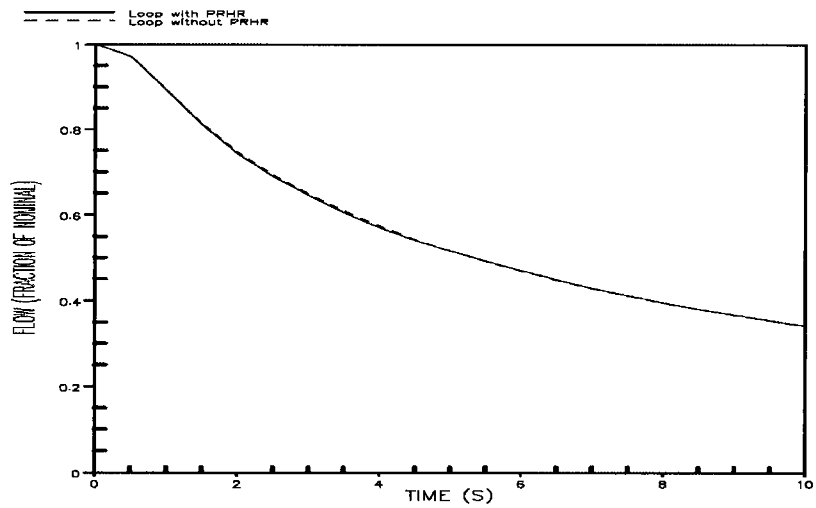

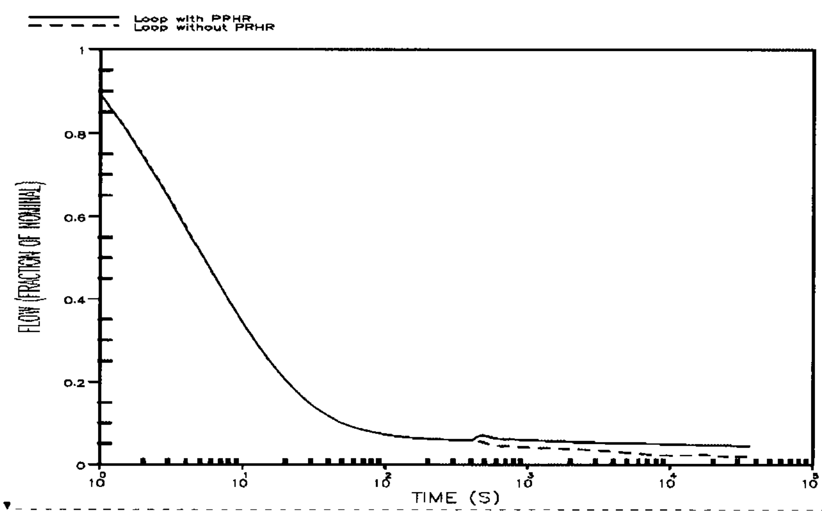

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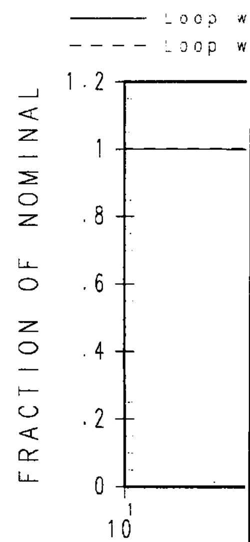
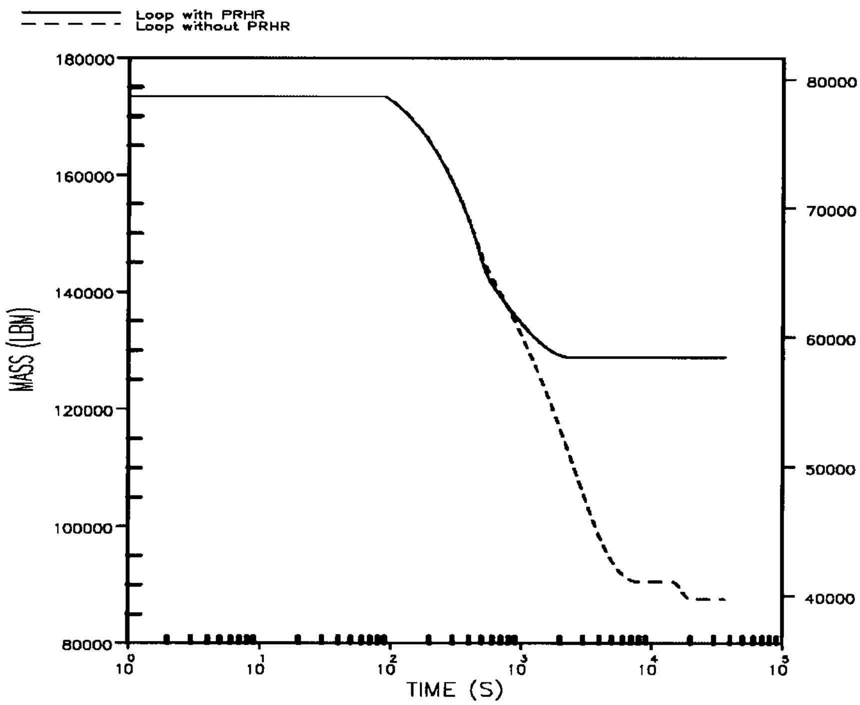


Figure 15.2.6-10

Reactor Coolant Volumetric Flow Rate
Transient for Loss of ac Power to the Plant Auxiliaries



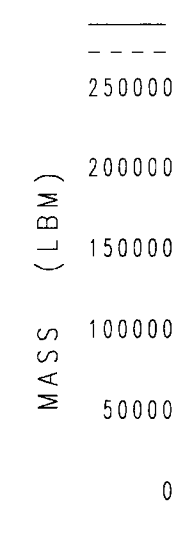


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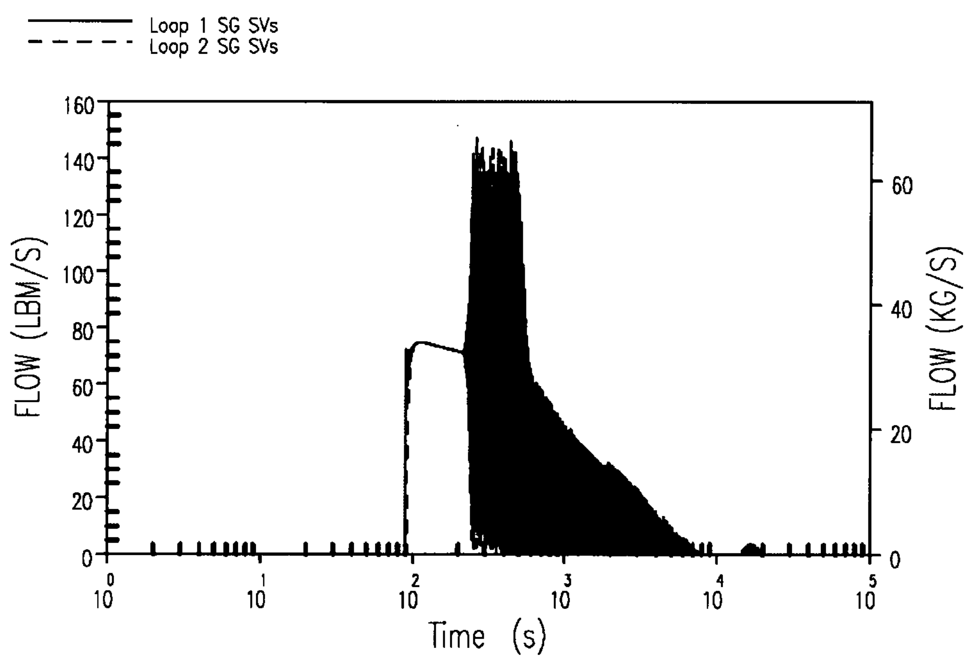
Figure 15.2.6-11

Steam Generator Inventory Transient for Loss of ac Power to the Plant Auxiliaries



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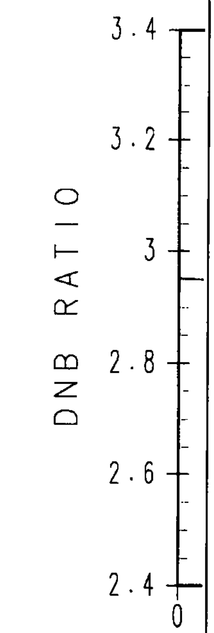

Figure 15.2.6-12

Steam Generator Safety Valve Relief
for Loss of ac Power to the Plant Auxiliaries

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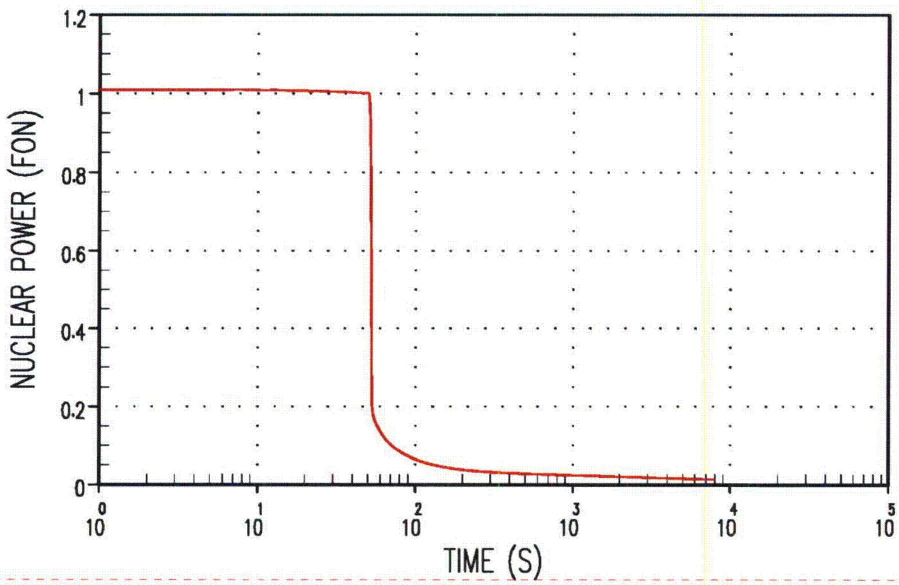
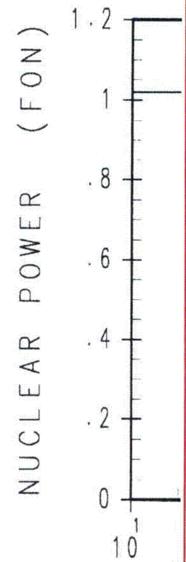


Figure 15.2.7-1

Nuclear Power Transient for Loss of Normal Feedwater



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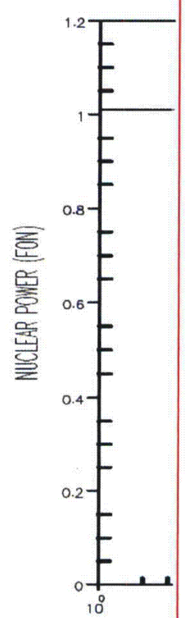

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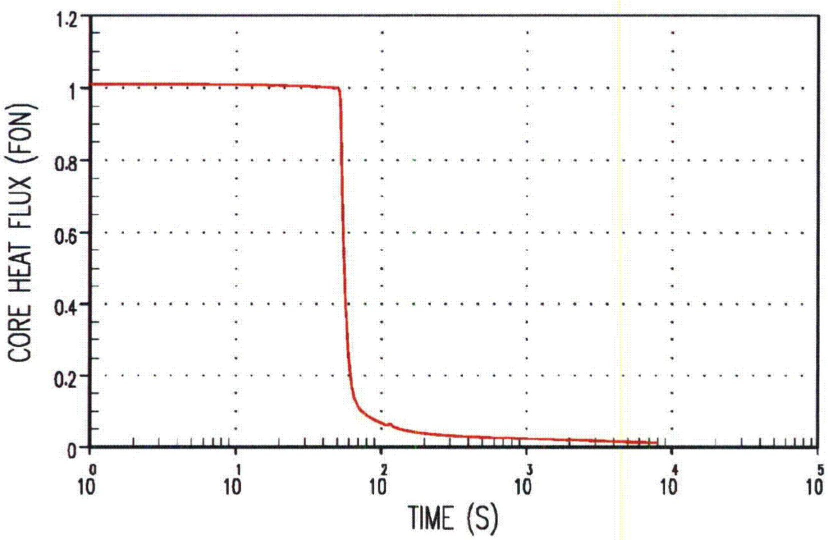

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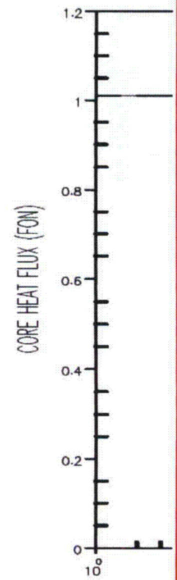

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Figure 15.2.7-2

Core Heat Flux Transient for

for Loss of Normal Feedwater

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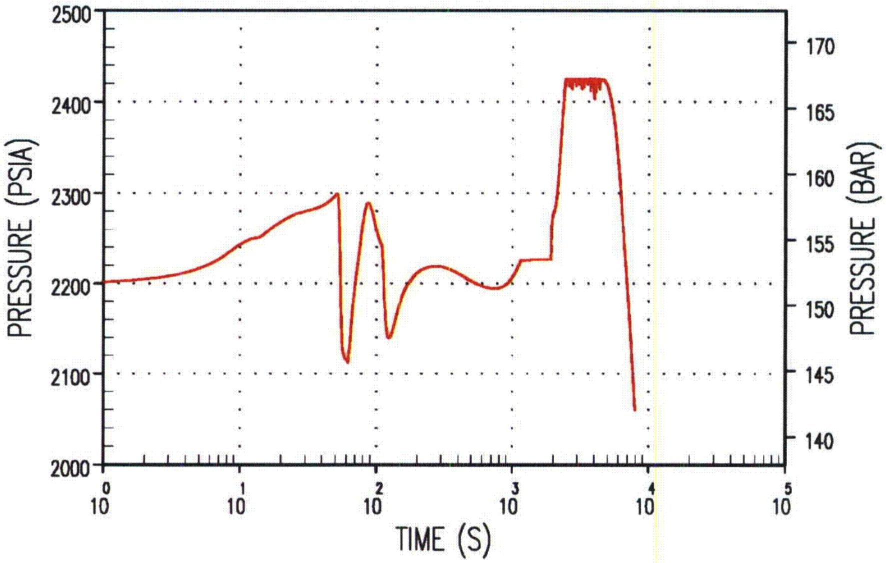
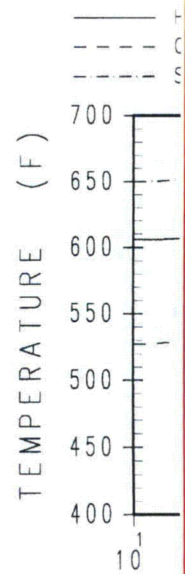


Figure 15.2.7-3

Pressurizer Pressure Transient for Loss of Normal Feedwater



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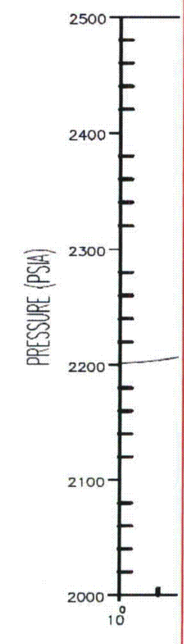

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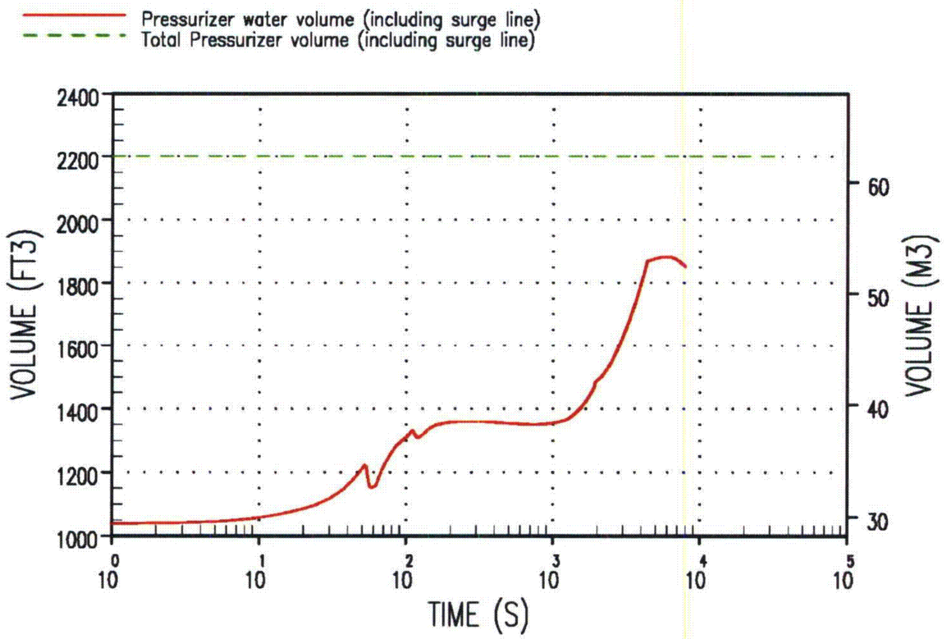

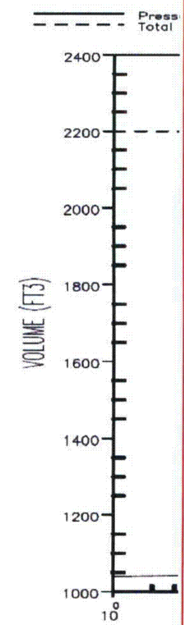

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Figure 15.2.7-4

Pressurizer Water Volume Transient for Loss of Normal Feedwater

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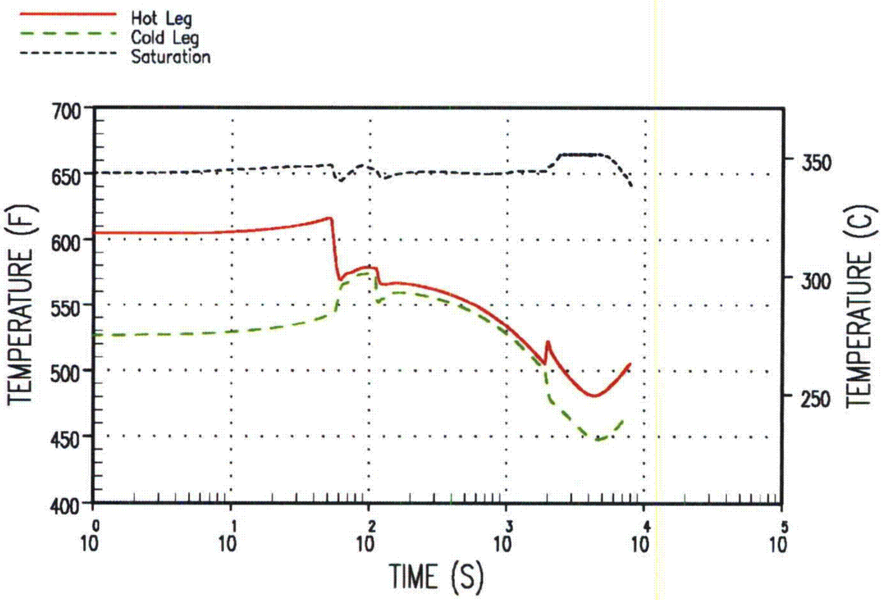

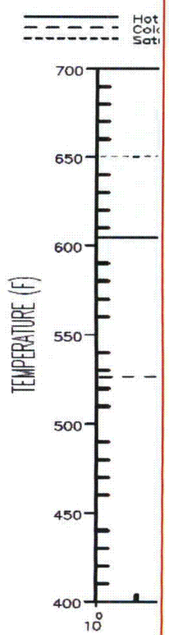

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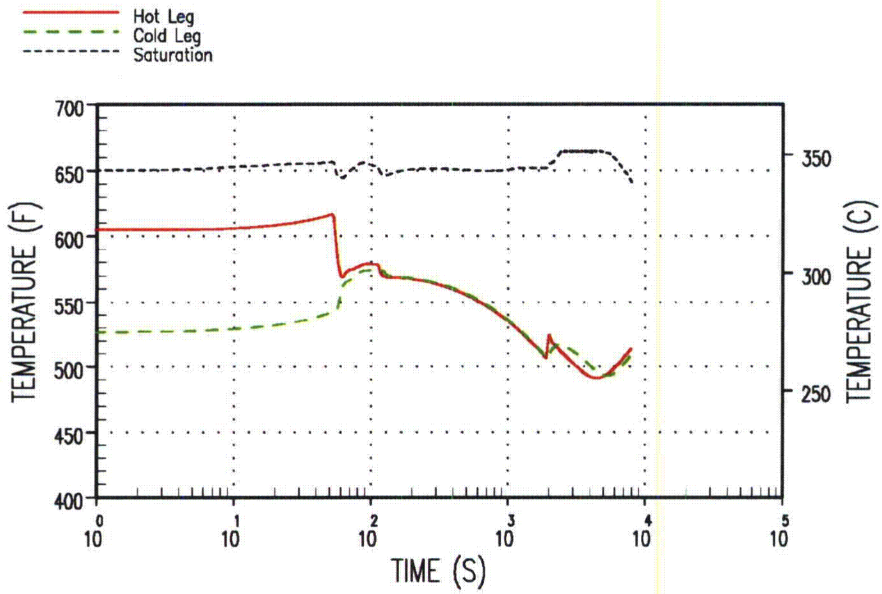
Figure 15.2.7-5

Comment [B12]: [15.2-8]

Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of Normal Feedwater Flow



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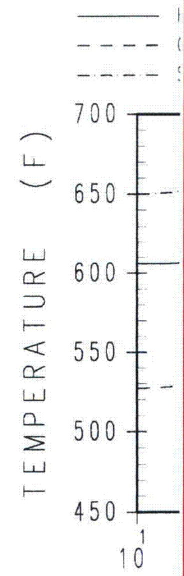

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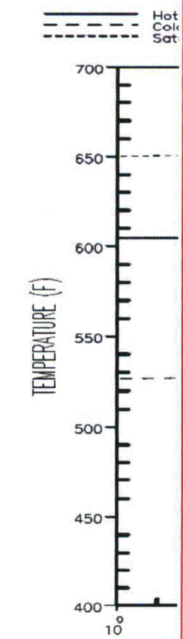

Figure 15.2.7-6

Reactor Coolant System Temperature Transient in Loop Not Containing the PRHR for Loss of Normal Feedwater Flow

Comment [B13]: [15.2-8]

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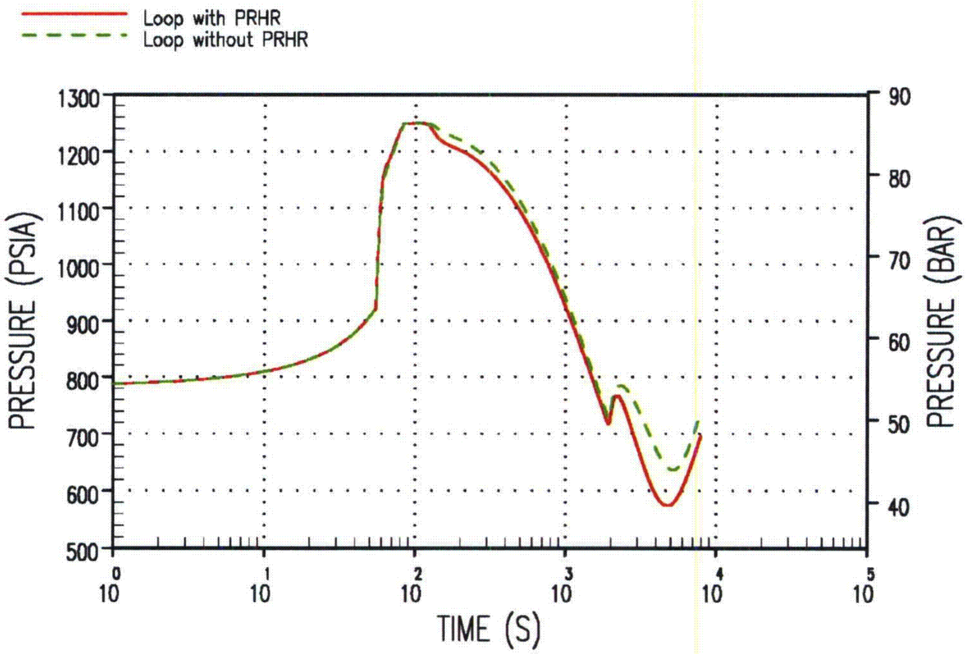
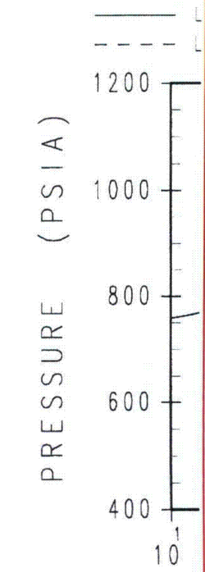


Figure 15.2.7-7

Steam Generator Pressure Transient for Loss of Normal Feedwater



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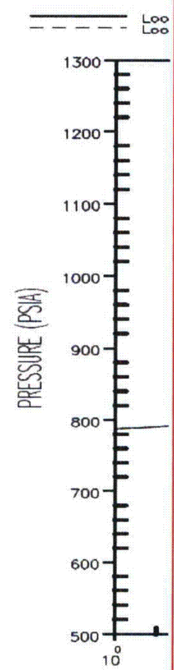

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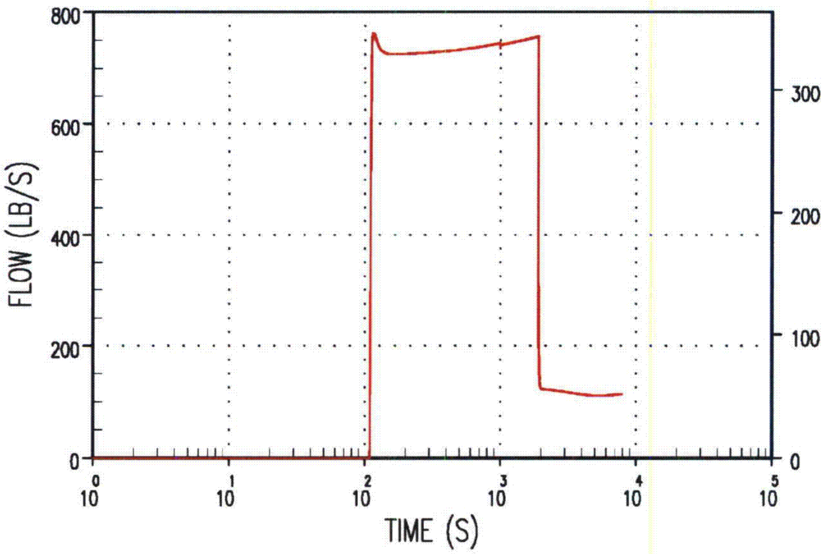
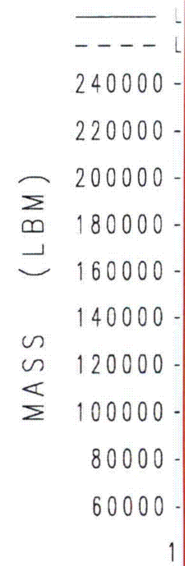


Figure 15.2.7-8

PRHR Flow Rate Transient for Loss of Normal Feedwater



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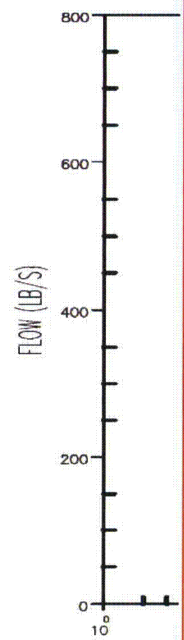

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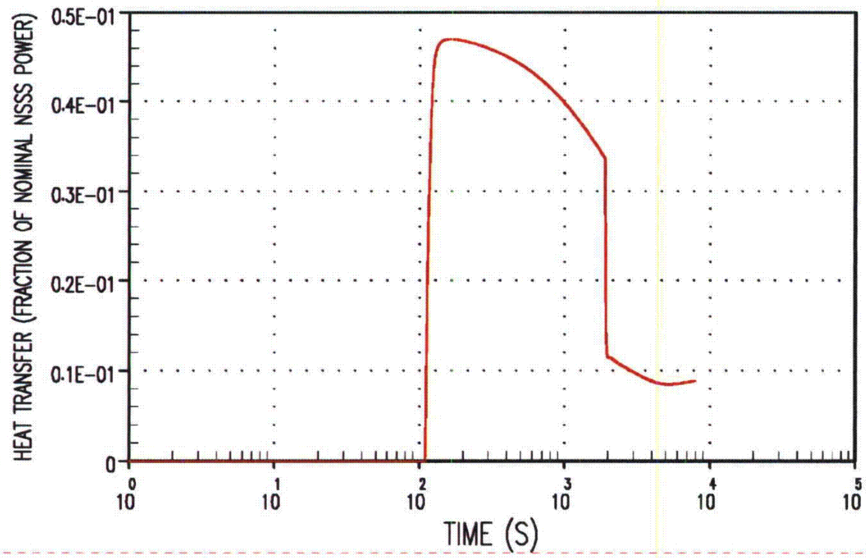

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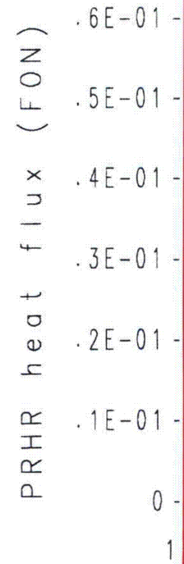

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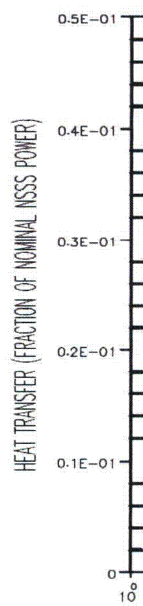

Figure 15.2.7-9

PRHR Heat Transfer Transient for Loss of Normal Feedwater

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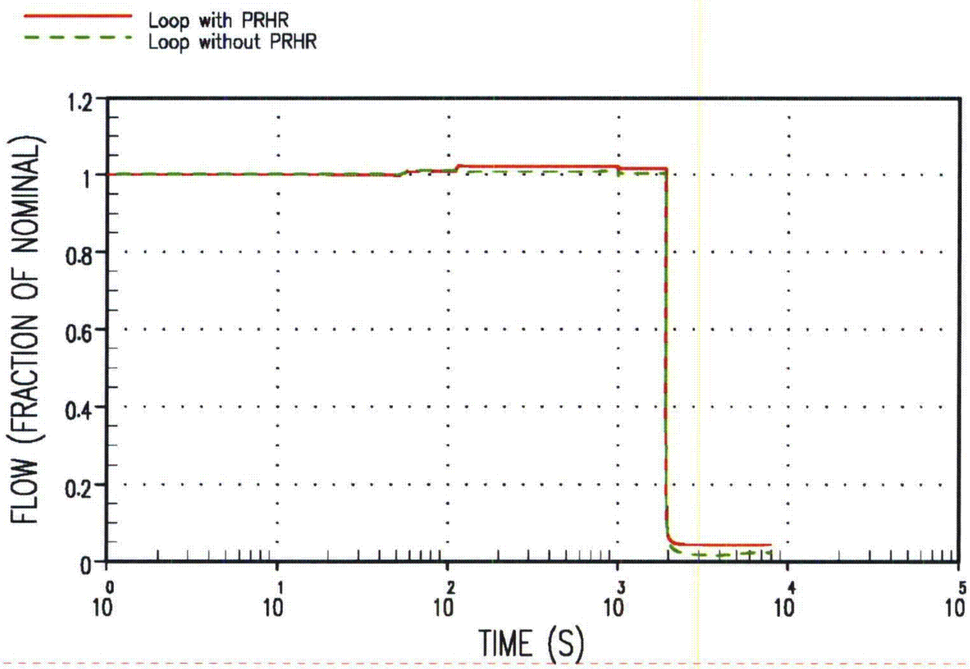

Figure 15.2.7-10

Reactor Coolant Volumetric Flow
Transient for Loss of Normal Feedwater

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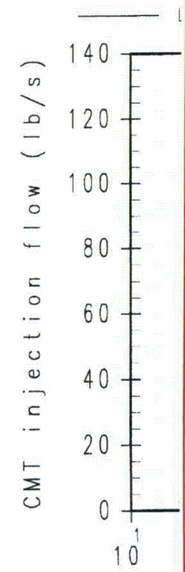

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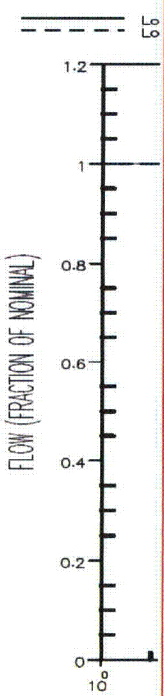

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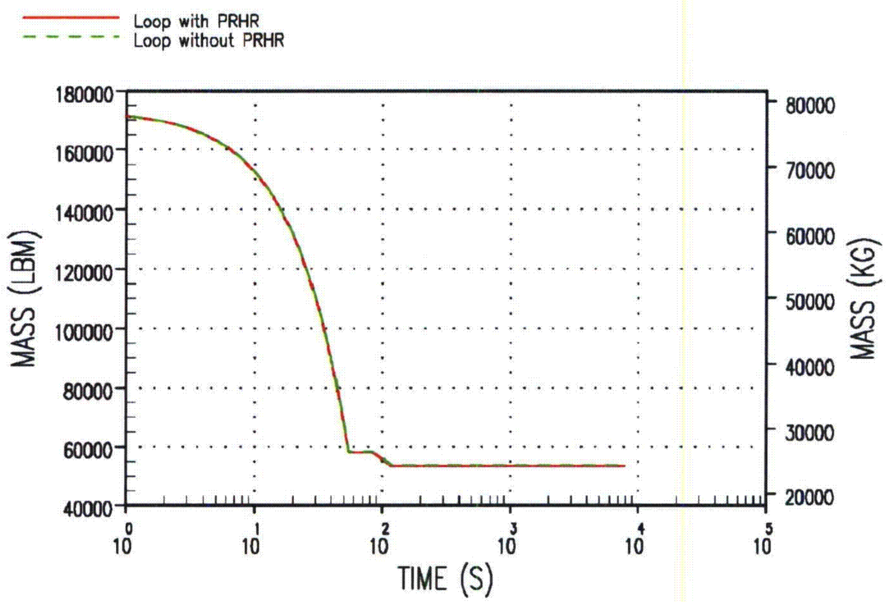
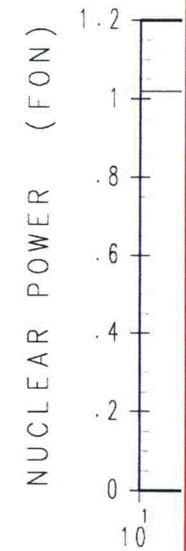


Figure 15.2.7-11

Steam Generator Inventory Transient for Loss of Normal Feedwater



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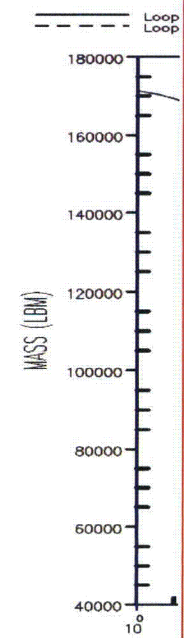

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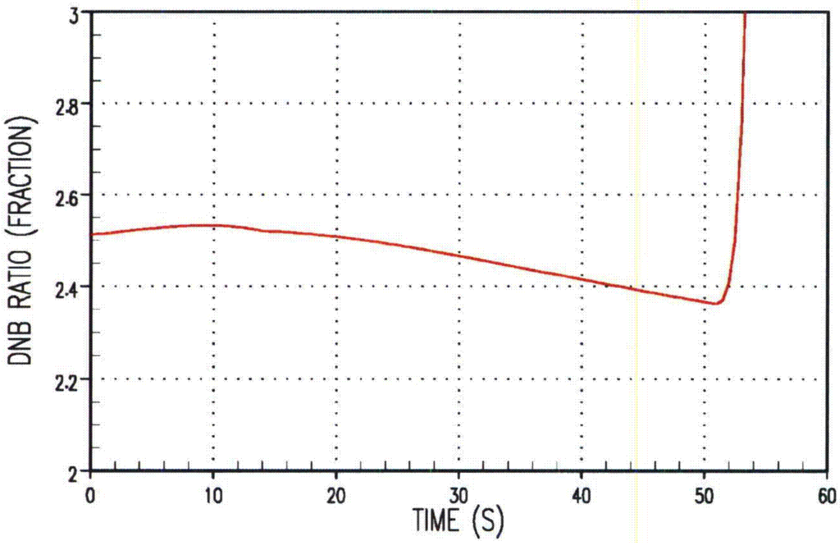

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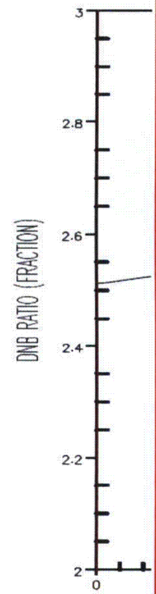

Figure 15.2.7-12

DNB Ratio Transient
for Loss of Normal Feedwater

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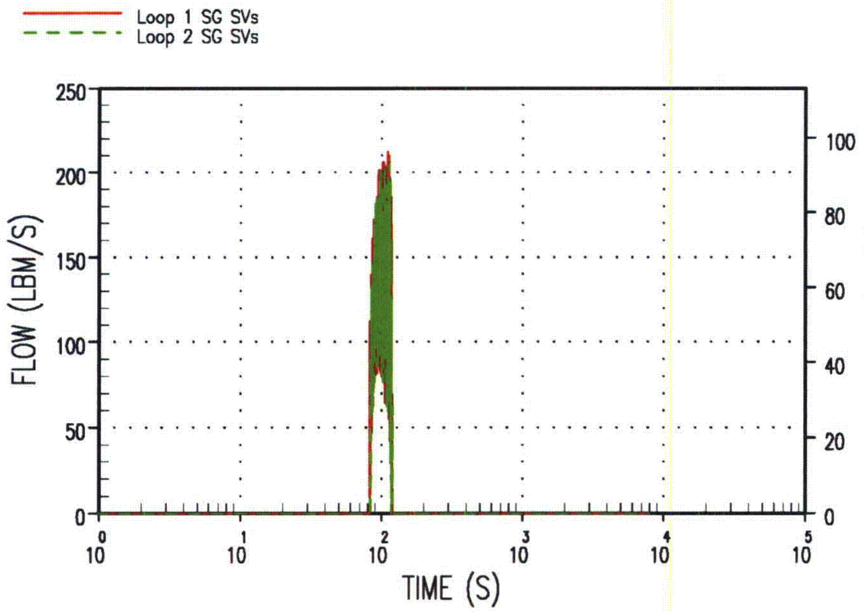

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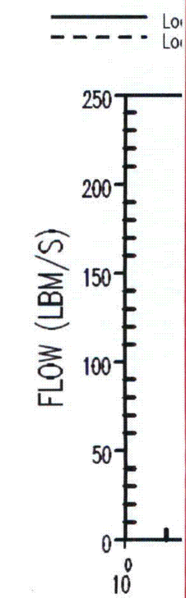

Figure 15.2.7-13

Steam Generator Safety Valve Relief Transient for Loss of Normal Feedwater

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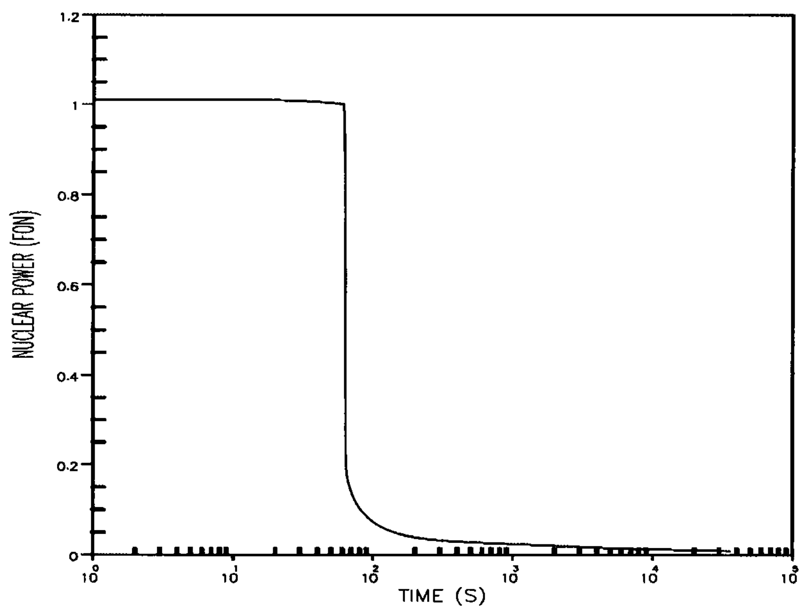


Figure 15.2.7-14
Nuclear Power Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

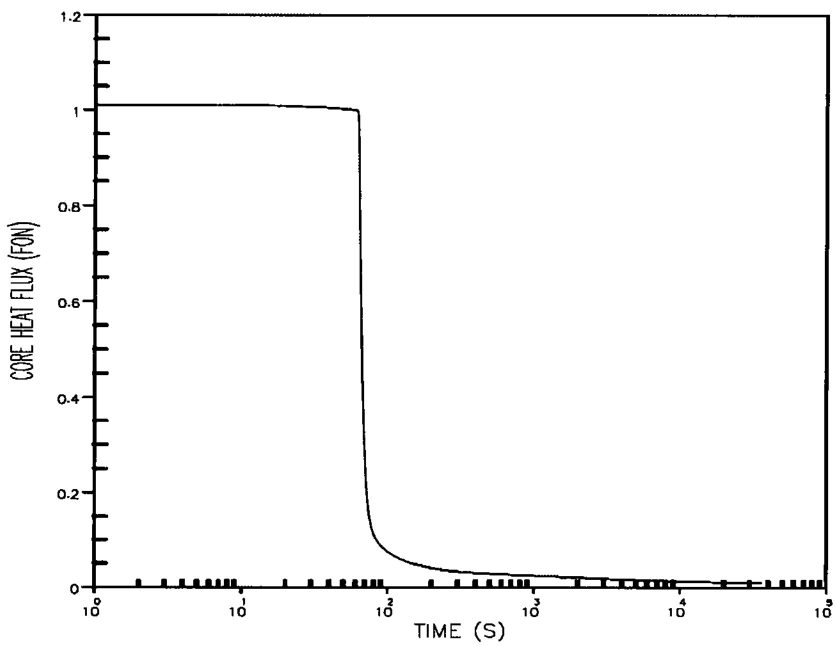


Figure 15.2.7-15

Core Heat Flux Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

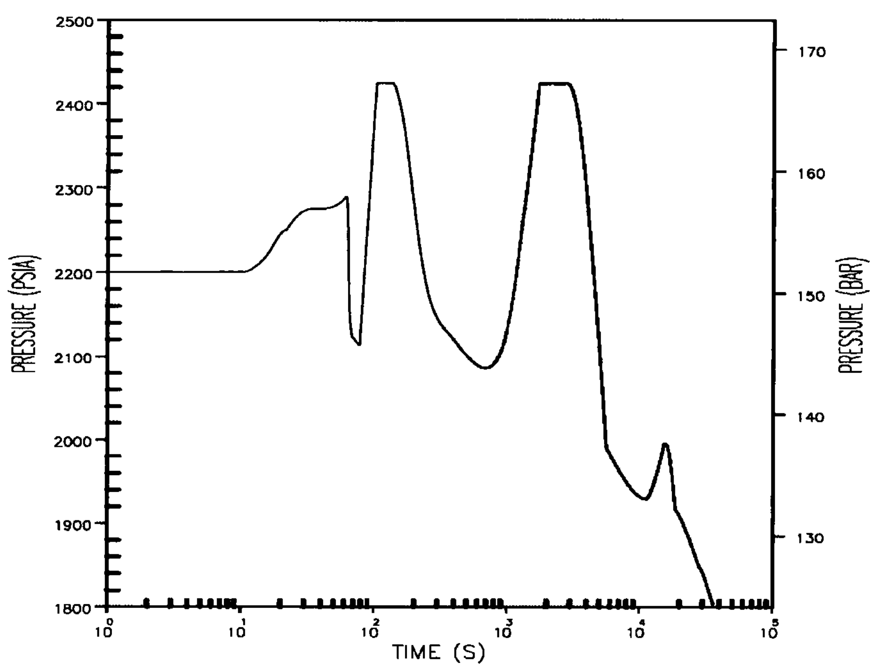


Figure 15.2.7-16

Pressurizer Pressure Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

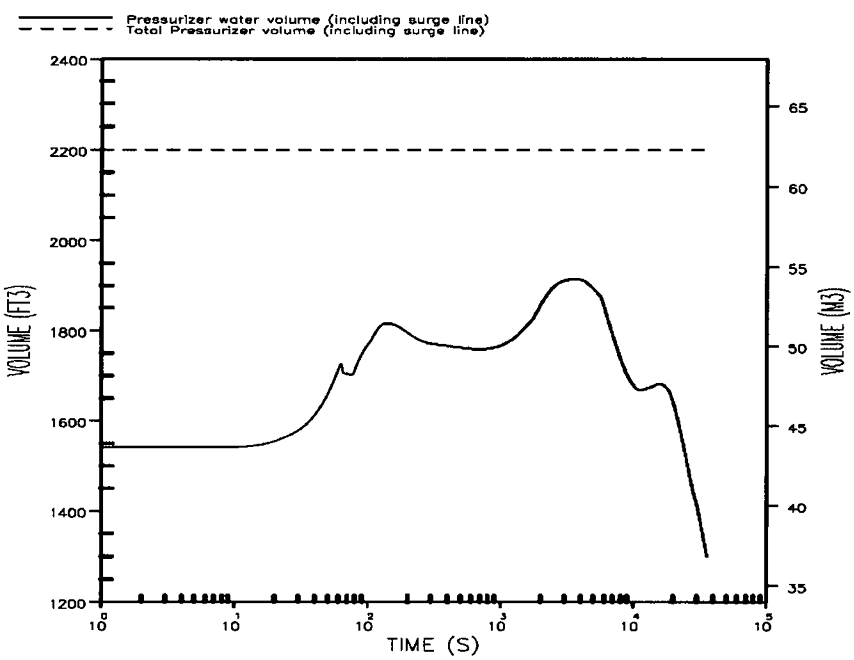


Figure 15.2.7-17

Pressurizer Water Volume Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

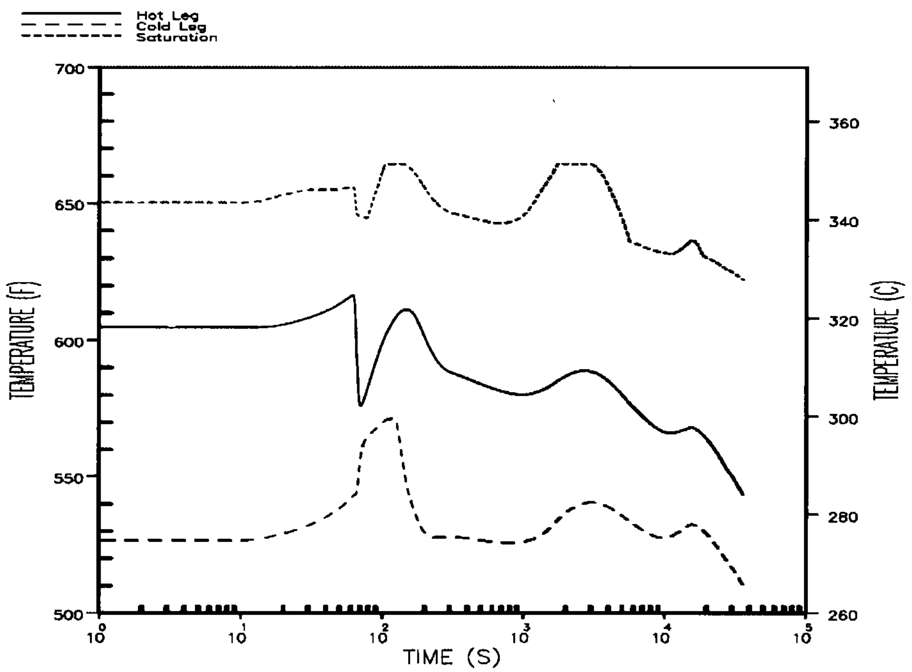
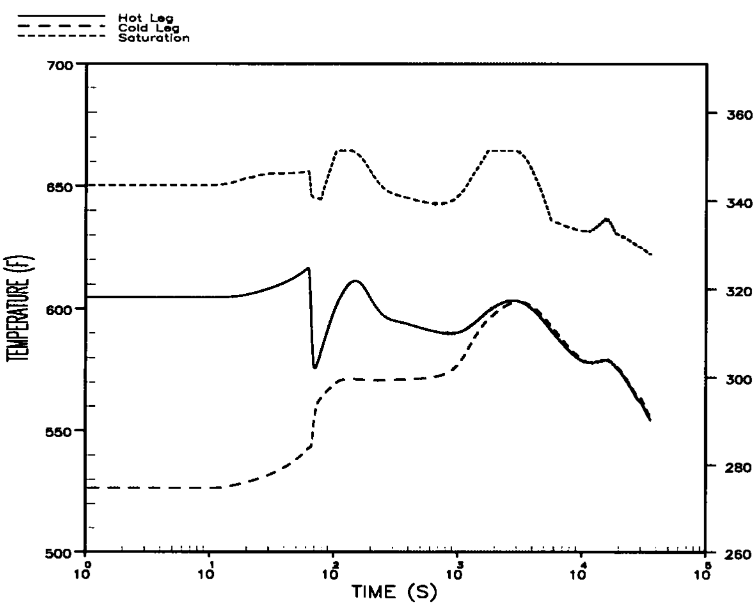


Figure 15.2.7-18

Reactor Coolant System Temperature Transients in Loop Containing the PRHR for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries



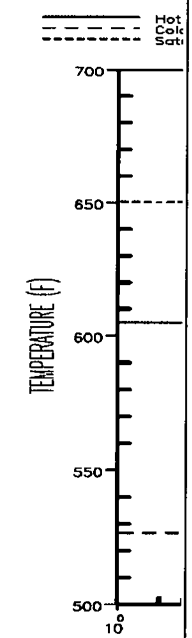


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Figure 15.2.7-19

Reactor Coolant System Temperature Transients in Loop Not Containing the PRHR for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plan Auxiliaries

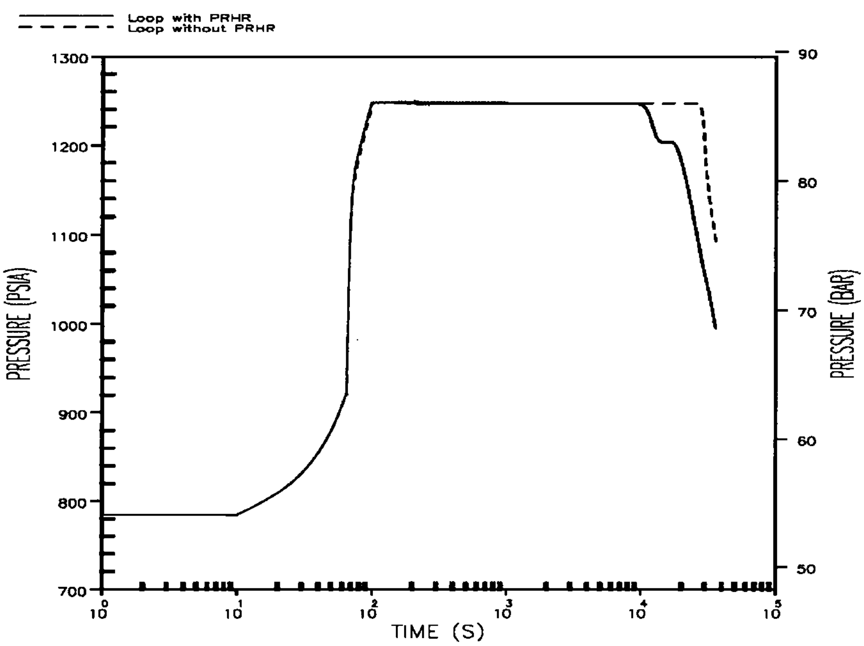


Figure 15.2.7-20

Steam Generator Pressure Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

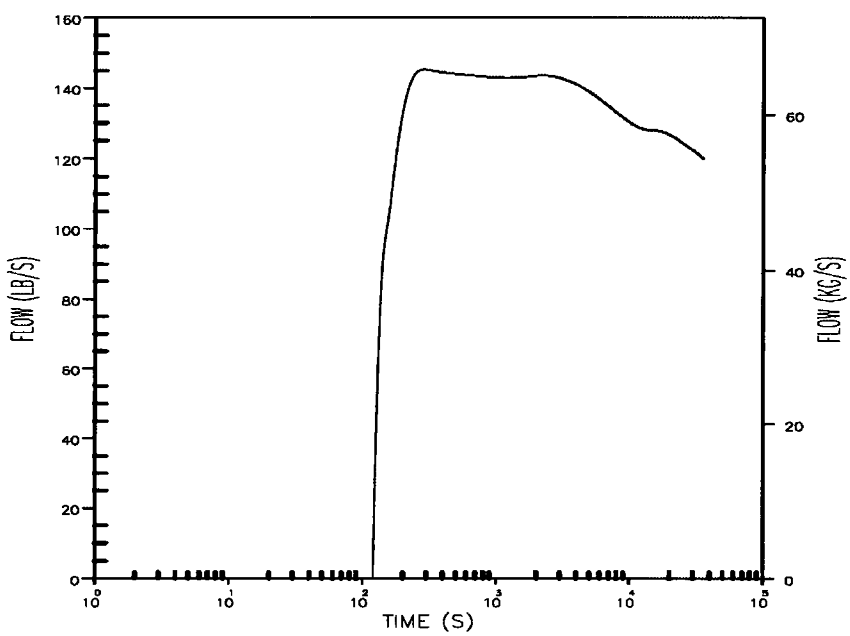


Figure 15.2.7-21

PRHR Flow Rate Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

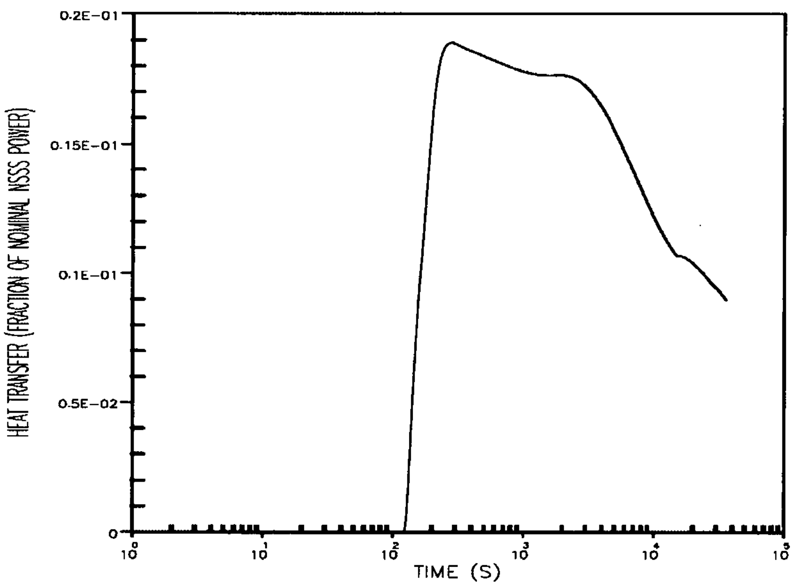


Figure 15.2.7-22

PRHR Heat Transfer Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

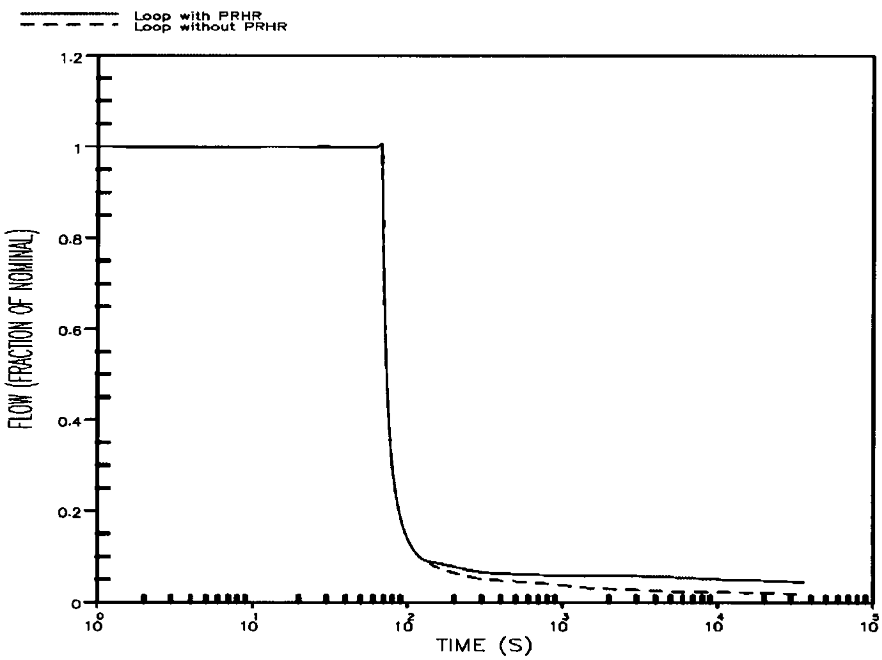


Figure 15.2.7-23

Reactor Coolant Volumetric Flow Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

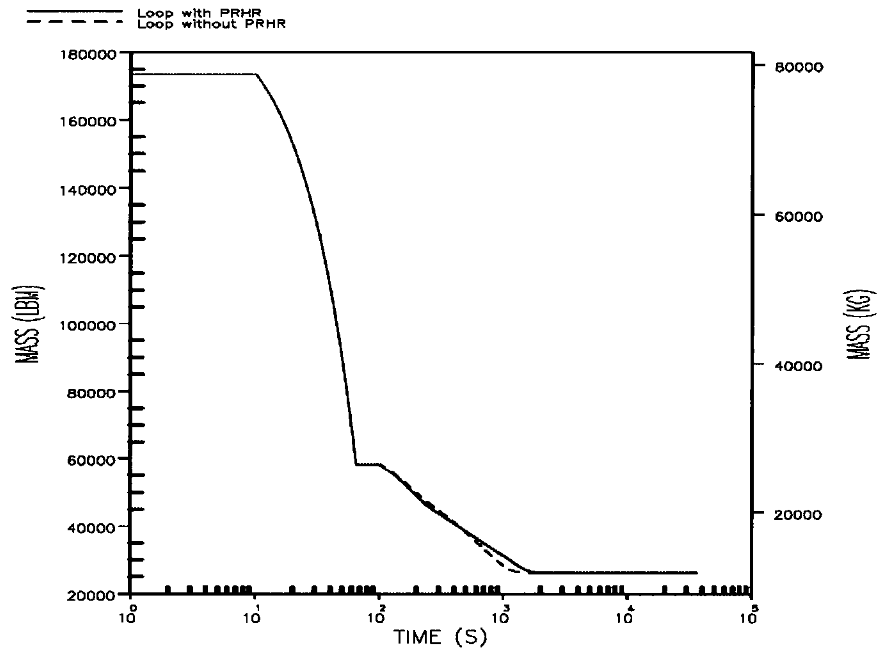


Figure 15.2.7-24

Steam Generator Inventory Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

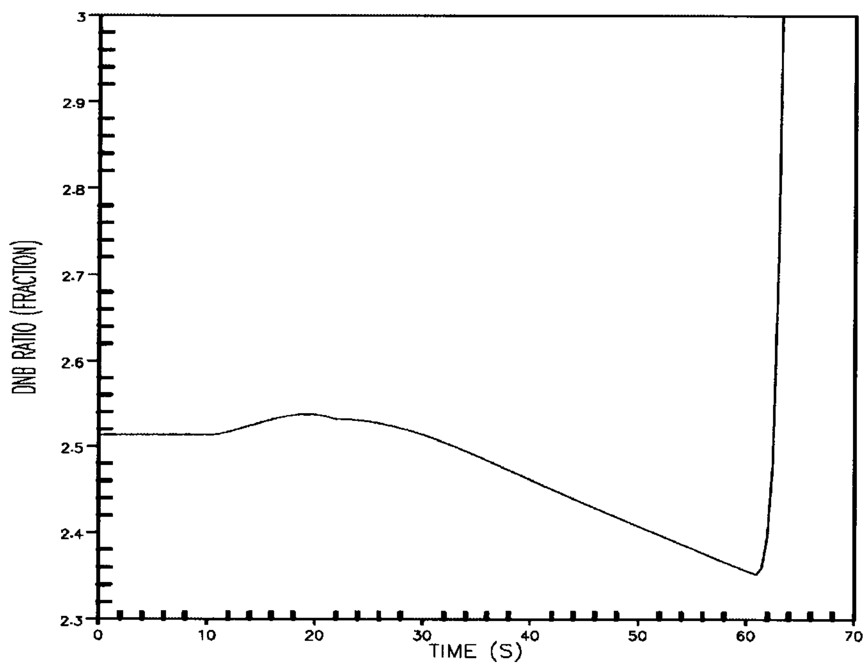


Figure 15.2.7-25

DNB Ratio Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

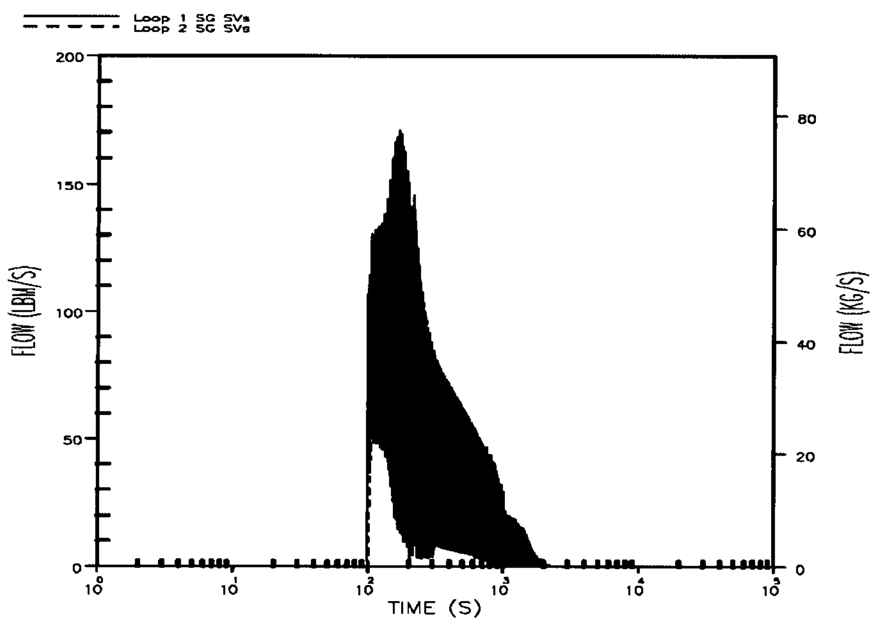


Figure 15.2.7-26

Steam Generator Safety Valve Relief Transient for Loss of Normal Feedwater with a Consequential Loss of ac Power to the Plant Auxiliaries

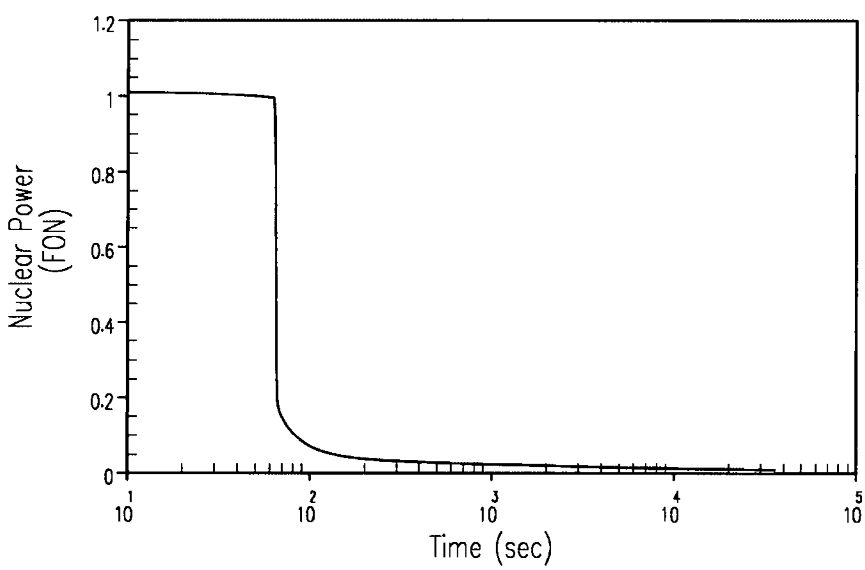
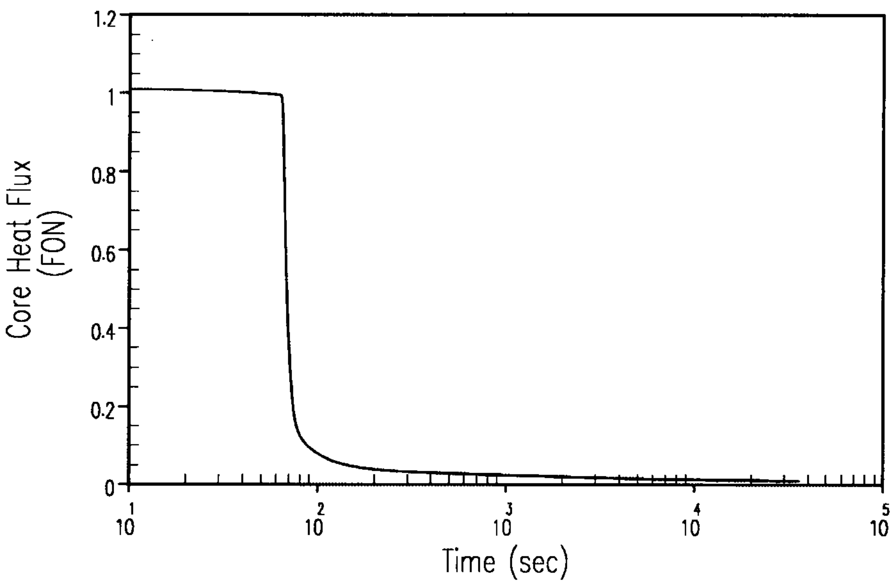


Figure 15.2.8-1

Nuclear Power Transient for Main Feedwater Line Rupture



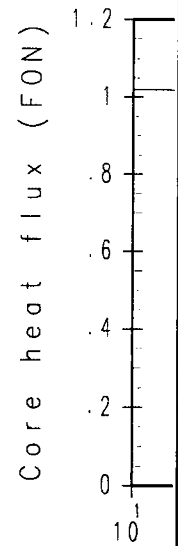


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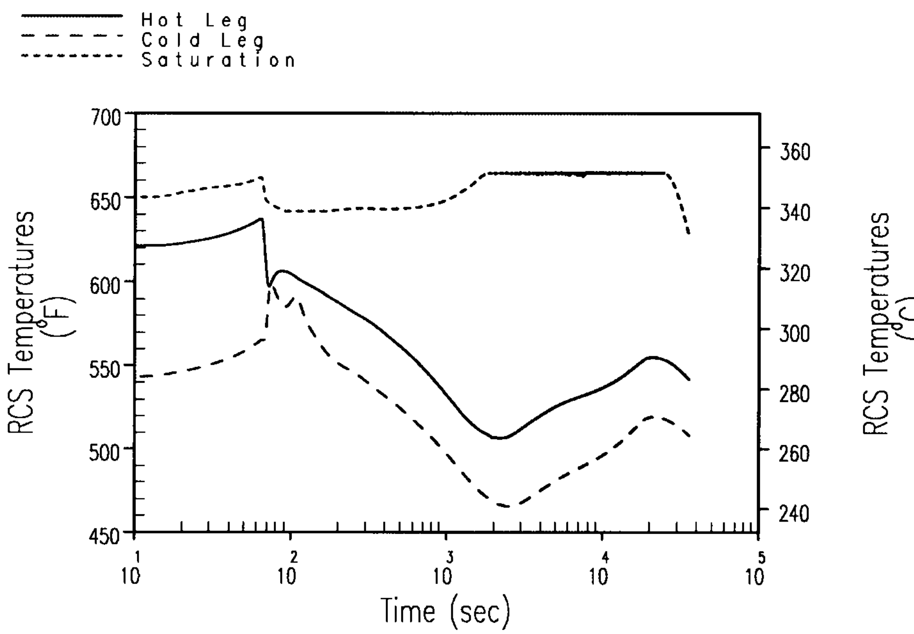
Figure 15.2.8-2

Core Heat Flux Transient for
Main Feedwater Line Rupture



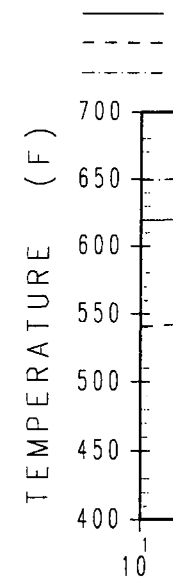


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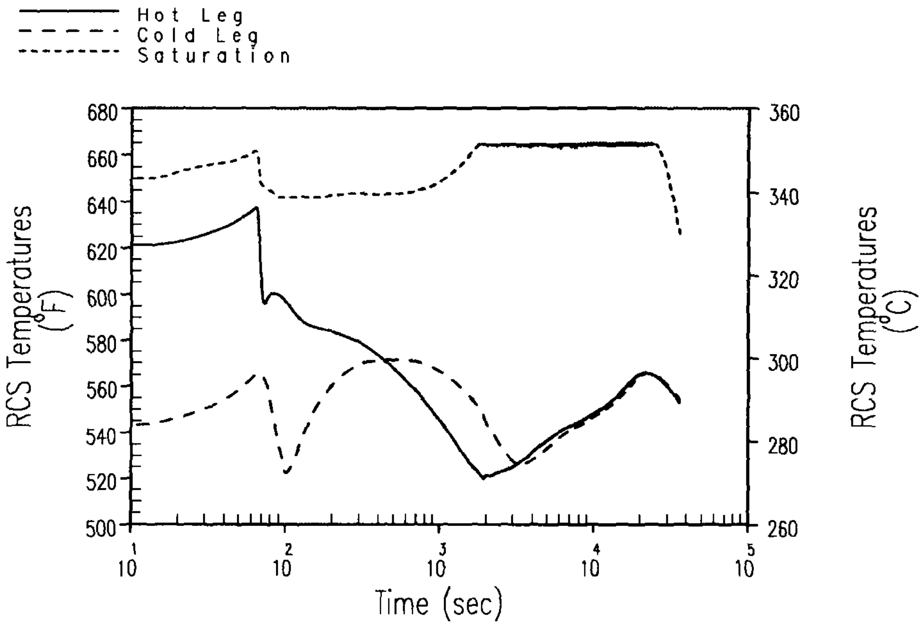
Figure 15.2.8-3

Faulted Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture



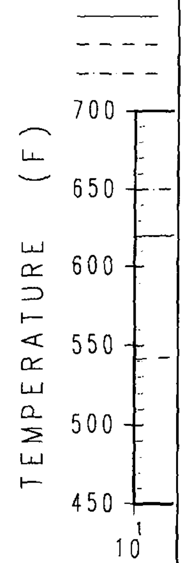


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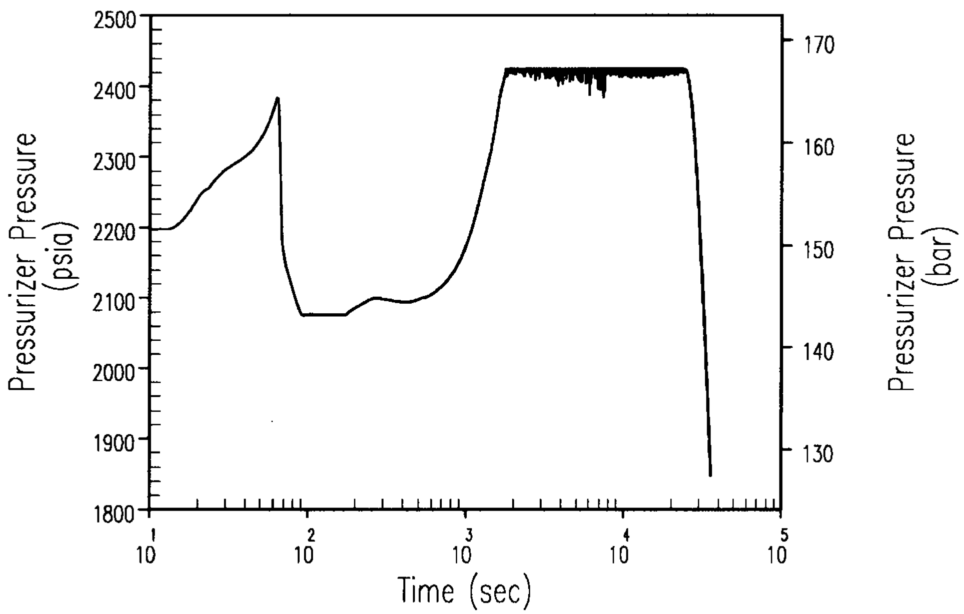
Figure 15.2.8-4

Intact Loop Reactor Coolant System
Temperature Transients for Main Feedwater Line Rupture



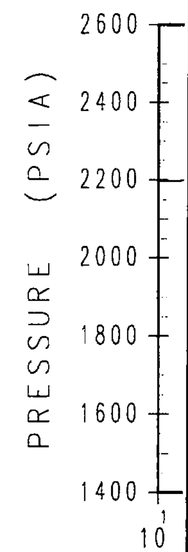


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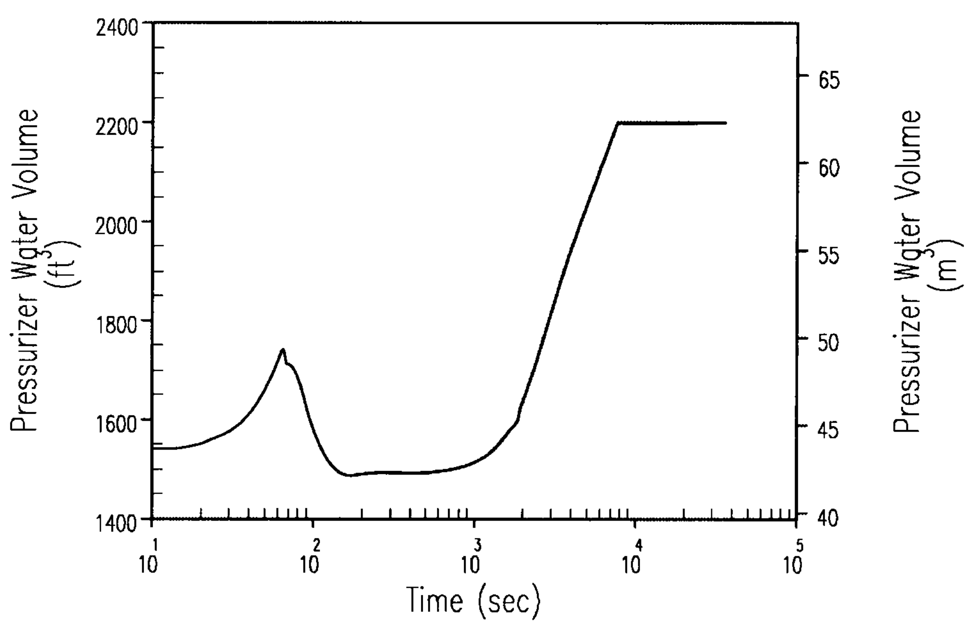
Figure 15.2.8-5

Pressurizer Pressure Transient for Main Feedwater Line Rupture



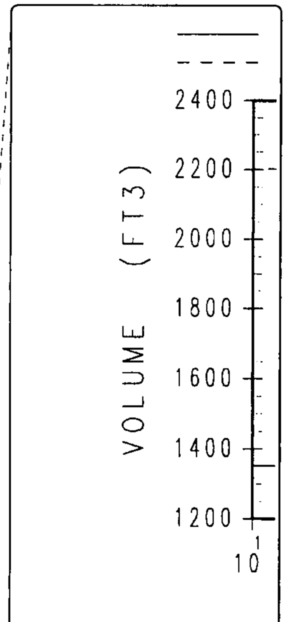
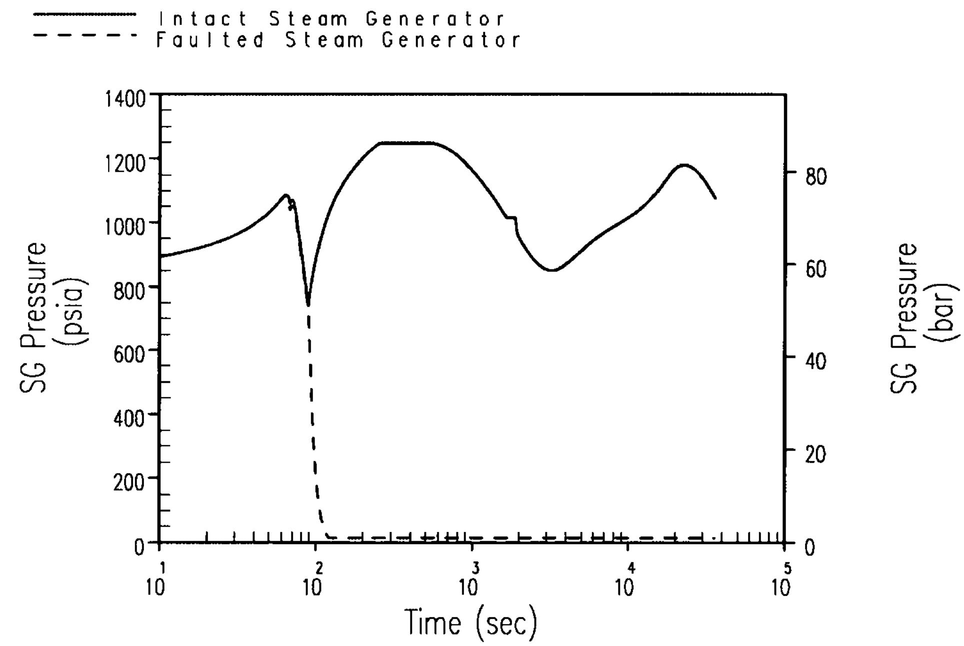


Figure 15.2.8-6

Pressurizer Water Volume Transient for Main Feedwater Line Rupture



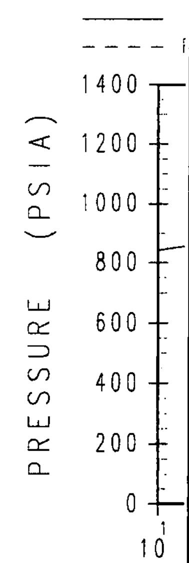


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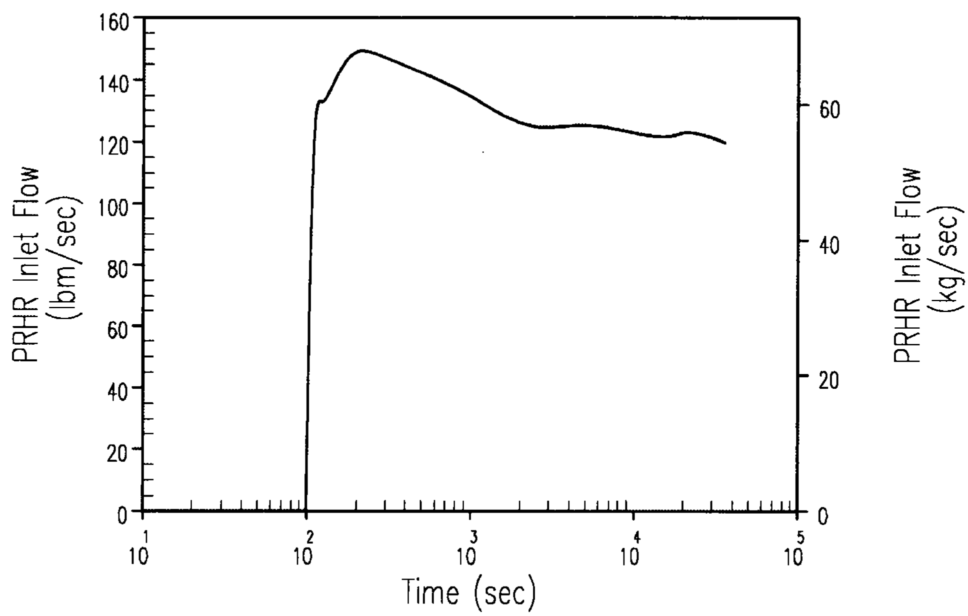
Figure 15.2.8-7

Steam Generator Pressure Transient for Main Feedwater Line Rupture



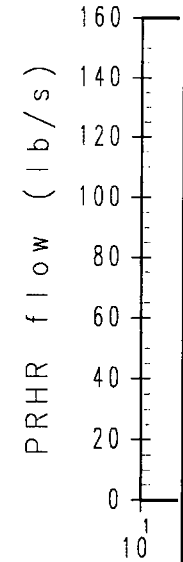


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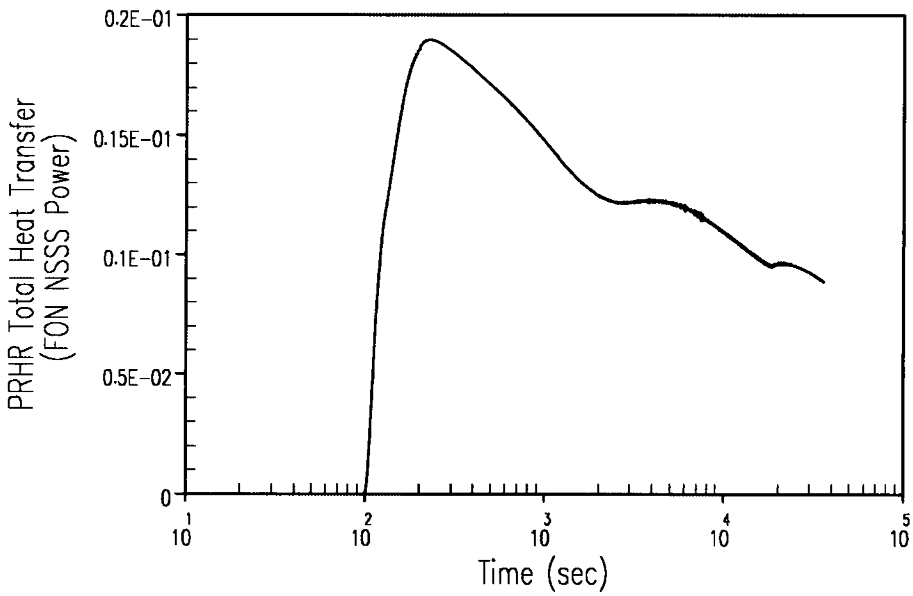
Figure 15.2.8-8

PRHR Flow Rate Transient for
Main Feedwater Line Rupture



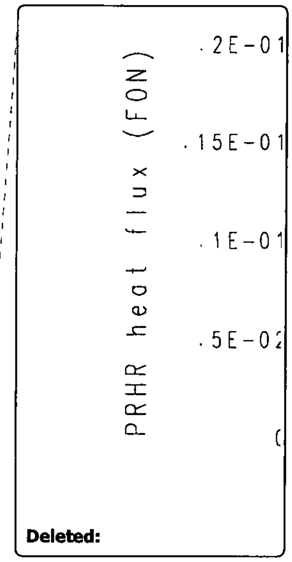
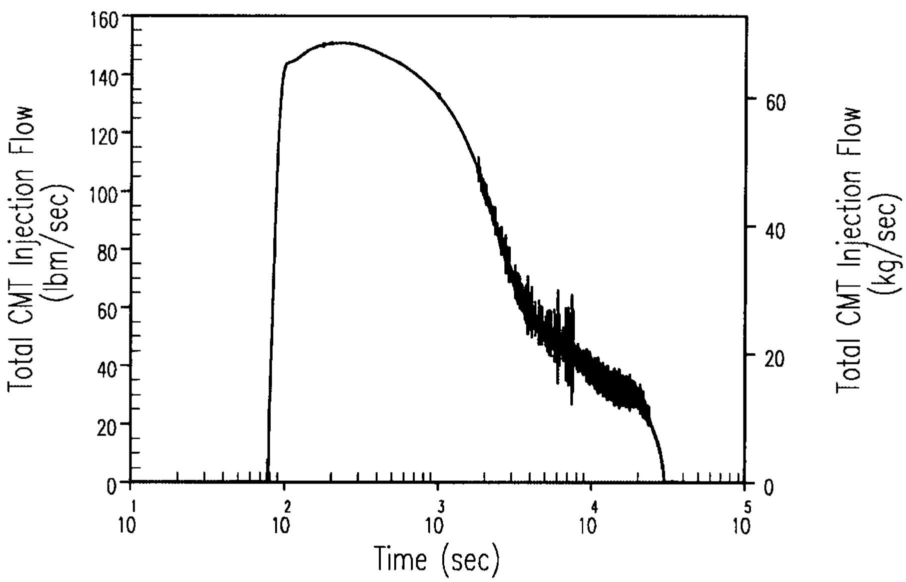


Figure 15.2.8-9

PRHR Heat Flux Transient for Main Feedwater Line Rupture



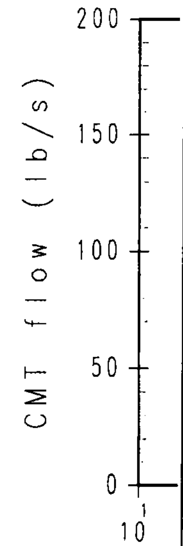


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Figure 15.2.8-10

CMT Injection Flow Rate Transient for Main Feedwater Line Rupture

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.3	Change Summary Description
[15.3-1]	15.3.1, Partial Loss of Forced Reactor Coolant Flow	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Additionally, the moderator density function was modeled as a function of density.
[15.3-2]	15.3.2, Complete Loss of Forced Reactor Coolant Flow	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.3-3]	15.3.3, Reactor Coolant Pump Shaft Seizure (Locked Rotor)	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Additionally, the moderator density function was modeled as a function of density.
[15.3-4]	15.3.3.3, Reactor Coolant Pump Shaft Seizure (Locked Rotor) Radiological Consequences.	Editorial Changes. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses.
[15.3-5]	15.3.3.3, Reactor Coolant Pump Shaft Seizure (Locked Rotor) Radiological Consequences.	See Change No. [15.3-4]
[15.3-6]	15.3.3.3, Reactor Coolant Pump Shaft Seizure (Locked Rotor) Radiological Consequences.	See Change No. [15.3-4]
[15.3-7]	15.3.4, Reactor Coolant Pump Shaft Break	Editorial changes incorporated.
[15.3-8]	15.3.6 References	Added new reference, WCAP-14565 – consistent with the change to Sections 15.3.1.2.1 and 15.3.3.2.1
[15.3-9]	15.3.6 References	Added new reference, WCAP-15644 – consistent with the change to Section 15.3.1.2.1

Change No.	Chapter 15 Section 15.3	Change Summary Description
[15.3-10]	Table 15.3-3	The radial peaking factor was increased to 1.75 from 1.65. Secondary mass updated based on revised NSSS models. Alkali metal partition factor updated to be consistent with moisture carryover.

15.3 Decrease in Reactor Coolant System Flow Rate

A number of faults that could result in a decrease in the reactor coolant system flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following reactor coolant system flow decrease events:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

The first event is a Condition II event, the second is a Condition III event, and the last two are Condition IV events.

The four limiting flow rate decrease events described above are analyzed in this section. The most severe radiological consequences result from the reactor coolant pump shaft seizure accident discussed in subsection 15.3.3. Doses are reported only for that case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

Comment [B1]: [15.3-1]

15.3.1.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or an electrical failure of a reactor coolant pump or from a fault in the power supply to the pump or pumps. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature. For the AP1000 plant design, there are two potential partial loss of flow scenarios. These scenarios include the coast down of one reactor coolant pump and the coast down of two reactor coolant pumps in diametrically opposite loops. Although both scenarios are analyzed, the loss of two reactor coolant pumps bounds the loss of one pump since it results in a more severe flow coast down. Thus, the two pump partial loss of flow is used as the basis for the discussion within this section.

Normal power for the pumps is supplied through four buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power and the pumps continue to operate.

A partial loss of coolant flow is classified as a Condition II incident (a fault of moderate frequency), as defined in subsection 15.0.1.

Protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated by two-out-of-four low-flow signals. Above permissive P10, low flow in either hot leg actuates a reactor trip (see Section 7.2).

As specified in GDC 17 of 10 CFR Part 50, Appendix A, the effects of a loss of offsite power are considered in evaluating partial loss of forced reactor coolant flow transients. As discussed in subsection 15.0.14, the loss of offsite power is considered to be a potential consequence of the event due to disruption of the electrical grid following a turbine trip during the event. A delay of 3 seconds is assumed between the turbine trip and the loss of offsite power. In addition, turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down. However, since the loss of offsite power would occur no earlier than 8 seconds into the event, it is well beyond the critical time frame of interest for the partial loss of flow events (i.e., time of rod insertion). Thus, it is not explicitly modeled in the case runs.

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15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

This transient is analyzed using three computer codes. First, the LOFTRAN code (References 1 and 8) is used to calculate the core flow during the transient based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 2) or the VIPRE-01 fuel rod model (Reference 7), which is equivalent to FACTRAN, is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The calculated DNBR transient represents the minimum of the typical cell or the thimble cell.

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15.3.1.2.2 Initial Conditions

Initial reactor power, pressurizer pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are statistically accounted for in the DNBR limit, as described in WCAP-11397-P-A (Reference 5).

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Plant characteristics and initial conditions assumed in this analysis are further discussed in subsection 15.0.3.

15.3.1.2.3 Reactivity Coefficients

The reactivity feedback parameters are chosen so as to maximize the energy transferred to the primary coolant during the flow coastdown. A most-negative Doppler-only power coefficient (see Figure 15.0.4-1) is applied to maximize the positive reactivity addition during the reactor trip and rod motion, which acts to slow the rate of power reduction; the equivalent total integrated Doppler reactivity from 0 to 100 percent power of 0.016 Δk. As there is an initial heatup due to the reduction in RCS flow, a least-negative (minimum feedback) moderator temperature coefficient is most conservative. Therefore, a constant moderator density coefficient of 0.0 Δk/g/cc is modeled. Finally, a curve of trip reactivity versus time based on a 2.3-second rod cluster control assembly insertion time to the dashpot is applied (see subsection 15.0.5).

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15.3.1.2.4 Flow Coastdowns

Conservative flow coastdowns are used to simulate the transient. The flow coastdowns are calculated externally to the LOFTRAN code using the COAST computer code which is described in Section 15.0.11.

15.3.1.2.5 Protection Systems

Plant systems and equipment necessary to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.1.2.6 Results

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Figures 15.3.1-1 through 15.3.1-6 show the transient response for the loss of two reactor coolant pumps with offsite power available. Figure 15.3.1-6 demonstrates that the DNBR is always greater than the safety analysis limit value, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4.

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The affected reactor coolant pumps coast down and the core flow reaches a new equilibrium value. The plant is tripped by the low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values. With the reactor tripped, a stable plant condition is attained and plant shutdown may then proceed.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1.

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In the event that a loss of offsite power occurs as a consequence of a turbine trip during a partial loss of reactor coolant flow, the DNB design basis continues to be met as discussed in subsection 15.3.1.1.

15.3.1.3 Conclusions

The analysis shows that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the safety analysis limit value at any time during the transient, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4. The applicable Standard Review Plan, subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature. Electric power for the reactor coolant pumps is normally supplied through buses, connected to the generator through the unit auxiliary transformers. When a generator trip occurs, the buses receive power from external power lines and the pumps continue to supply coolant flow to the core.

A complete loss of flow accident is a Condition III event (an infrequent fault), as defined in subsection 15.0.1. The following signals provide protection against this event:

1. Reactor coolant pump underspeed
2. Low primary coolant loop flow

The reactor trip on reactor coolant pump underspeed protects against conditions that can cause a loss of voltage to two-out-of-four reactor coolant pumps. This function is blocked below approximately 10-percent power (permissive P10). The reactor trip on reactor coolant pump underspeed also protects against an underfrequency condition resulting from frequency disturbances on the power grid, so long as the maximum grid frequency decay rate is less than approximately 5 hertz per second. WCAP-8424, Revision 1 (Reference 3), provides analyses of grid frequency disturbances and the resulting protection requirements that are applicable to the AP1000.

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15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps.

For the scenario of a complete loss of voltage, which results in all the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in subsection 15.3.1, with two exceptions. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the reactor coolant pump underspeed trip instead of the low primary coolant flow trip. Also, rather than the bounding value of 0.0 Δk/g/cc, a less limiting, yet still conservative, moderator density coefficient (MDC) curve (MDC as a function of coolant density) was modeled.

A complete loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. However, the results of the complete loss of voltage scenario (i.e., free spinning pump coastdown) bound the results of the complete loss of flow initiated by a frequency decay of up to 5 hertz per second. This is due to the reactor coolant pump design, which initially (during the critical time frame of the transient) has a more rapid coastdown as a free spinning pump than for an electrical frequency decay. Therefore, only the results of the complete loss of voltage case scenario presented in subsection 15.3.2.2.2.

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is tripped on the reactor coolant pump underspeed signal. Figure 15.3.2-6 demonstrates that the DNBR is always greater than the safety analysis limit value, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With respect to DNB concerns, the event is essentially over shortly after reactor trip. However, if the event was extended beyond the time frame analyzed for DNB, the reactor coolant pumps continue to coast down, and natural circulation flow would be established, as demonstrated in subsection 15.2.6. With the reactor tripped, a stable plant condition is attained and plant shutdown may then proceed.

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15.3.2.3 Conclusions

The analysis demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit value at any time during the transient, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4. The applicable Standard Review Plan, subsection 15.3.1 (Reference 4), evaluation criteria are met.

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15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Comment [B3]: [15.3-3]

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low-flow signal.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant temperature to increase and expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Consistent with the AP1000 design, the peak pressure and fuel rod thermal analyses assume a 5 second delay in turbine trip following reactor trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

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This event is classified as a Condition IV incident (a limiting fault), as defined in subsection 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code (Reference 1) calculates the resulting core flow transient following the pump seizure and the nuclear power following reactor trip. This code is also used to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code (Reference 2) or the VIPRE-01 fuel rod model (Reference 7) which is equivalent to

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FACTRAN. This fuel thermal calculation uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be in operation under the most adverse steady-state operating conditions, that is, maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in subsection 15.0.3. The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at 3.0 seconds following turbine/generator trip. Turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia), which allows for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

15.3.3.2.2 Evaluation of the Pressure Transient and Fuel Rod Thermal Design Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1.45 seconds after the flow in the affected loop reaches the reactor trip setpoint. No credit is taken for the pressure-reducing effect of the pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to result in a lower peak reactor coolant system pressure, an additional conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.

For this accident, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to cladding temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be 3 times the average rod power (that is, $F_Q = 3.0$) at the initial core power level.

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15.3.3.2.3 Evaluation of Departure from Nucleate Boiling in the Core During the Accident

An analysis is performed to determine the percentage of fuel rods that experience DNB. The percentage is determined to be less than the limit value used for the fraction of fuel rods that are predicted to experience a DNB in the radiological consequences calculations reported in Section 15.3.3.3.

15.3.3.2.4 Film-Boiling Coefficient

The film-boiling coefficient is calculated in the FACTRAN code (Reference 2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient because they are the most conservative with respect to cladding temperature response. For conservatism, DNB is assumed to start at the beginning of the accident.

15.3.3.2.5 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and the cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient is assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/h-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value of the gap coefficient is released to the cladding at the initiation of the transient.

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above a cladding temperature of 1800°F. The Baker-Just parabolic rate equation is used to define the rate of the zirconium-steam reaction:

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45{,}500}{1.986T}\right)$$

where:

w = amount reacted (mg/cm^2)

t = time (s)

T = temperature (Kelvin)

The reaction heat is 1510 cal/g.

The effect of the zirconium-steam reaction is included in the calculation of the hot spot cladding temperature transient.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3.3-1 through 15.3.3-7 show the transient results for one locked rotor with four reactor coolant pumps in operation. The without-offsite-power case bounds the results for the case with offsite power. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure reached during the transient is less than that which causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

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The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition is eventually attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant. Refer to Section 15.3.3.3.1 and Table 15.3-3.

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Comment [B4]: [15.3-4]

As a result of the accident, it is determined that no fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. However, a conservative analysis has been performed assuming 10 percent of the rods are damaged. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

15.3.3.3.1 Source Term

The significant radionuclide releases due to the locked rotor accident are the iodines, alkali metals (cesiums, rubidiums) and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident.

Comment [B5]: [15.3-5]

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Based on NUREG-1465 (Reference 6), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 8 percent of the inventory for I-131, 10 percent for Kr-85, 5 percent for other iodines and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.

The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

15.3.3.3.2 Release Pathways

There are two components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant. As steam is released, a portion of the iodine and alkali metal activity in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

Credit is taken for the decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.3.3.3.3 Dose Calculation Models

The models used to calculate offsite doses are provided in Appendix 15A.

15.3.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.3-3.

Two separate accident scenarios are addressed. In the first scenario, it is assumed that the non-safety grade startup feedwater system is not available to provide feedwater to the steam generators. In this event, the water level in the steam generators drops, resulting in tube uncovery and there is flashing of a portion of the primary coolant assumed to be leaking into the secondary side of the steam generators. Also, the period of steaming is terminated at 1.5 hours when the capacity of the passive residual heat removal system exceeds the decay heat generation rate.

In the second scenario, it is assumed that the startup feedwater system is available to maintain water level in the steam generators such that the tubes remain covered. In this scenario, direct release of flashed primary coolant is not considered. Also, the passive residual heat removal system does not actuate, resulting in a longer period of steaming releases.

15.3.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on conservative assumptions (Refer to Table 15.3-3); whereas, the expected activities based on the fuel defect level are far less, (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

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15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be less than 0.5 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 0.2 rem at the low population zone outer boundary for the scenario in which there is no feedwater available to maintain water level in the steam generators. The doses for the scenario in which it is assumed that water level in the steam generators is maintained are 0.4 rem at the exclusion area boundary for the limiting 2-hour interval of 6 to 8 hours and 0.4 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the locked rotor event, the resulting total dose remains less than the value reported above.

15.3.4 Reactor Coolant Pump Shaft Break

Comment [B7]: [15.3-7]

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a low-flow signal in the affected loop.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator in the faulted loop is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer

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safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (limiting fault), as defined in subsection 15.0.1.

15.3.4.2 Conclusion

With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as is the case when a locked rotor occurs. This results in a decrease in the end point (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculated results presented for the locked rotor analysis bound the reactor coolant pump shaft break event.

15.3.5 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.3.6 References

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2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Baldwin, M. S., et al., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May 1975.

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

6. Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

7. Sung.Y.X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary) and WCAP -15306-NP-A9Nonproprietary),October 1999.

8. "AP1000 Code Applicability Report," WCAP-15644_P (Proprietary) and WCAP-15644-NP-A (Nonproprietary), Revision 2, March 2004.

Comment [B8]: [15.3-8]

Comment [B9]: [15.3-9]

Table 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Accident	Event	Time (seconds)
Partial loss of forced reactor coolant flow		
– Loss of two pumps with four pumps running	Two pumps lose power and begin coasting down	0.00
	Low-flow reactor trip setpoint reached	1.45
	Rods begin to drop	3.42
	Minimum DNBR occurs	5.50
Complete loss of forced reactor coolant		
– Loss of four pumps with four pumps running	All pumps lose power and begin coasting down	0.00
	Reactor coolant pump underspeed trip setpoint reached	0.55
	Rods begin to drop	1.35
	Minimum DNBR occurs	3.20
Reactor coolant pump shaft seizure (locked rotor)		
– One locked rotor with four pumps running without offsite power available	Rotor on one pump locks	0.00
	Low-flow trip point reached	0.10
	Rods begin to drop	1.55
	Maximum reactor coolant system pressure occurs	3.40
	Maximum cladding temperature occurs	4.10

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4.900

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Low-flow trip point reached
Rods begin to drop
Maximum reactor coolant system pressure occurs
Maximum cladding temperature occurs

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0.10
1.55
2.30
3.90

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Table 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS (FOUR REACTOR COOLANT PUMPS OPERATING INITIALLY)

Maximum reactor coolant system pressure (psia)	2716.30
Maximum cladding average temperature, core hot spot (°F)	2013
Zr-H_2O reaction, core hot spot (percentage by weight)	0.57

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Table 15.3-3 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Comment [B10]: [15.3-10]

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	1.75
Fission product gap fractions I-131 Kr-85 Other iodines and noble gases Alkali metals	 0.08 0.10 0.05 0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.04 E+05
Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.5[b]
Partition coefficient in steam generators iodine alkali metals	 0.01 0.003
Accident scenario in which startup feedwater is not available Duration of accident (hr) Steam released (lb) 0-1.5 hours[c] Leak flashing fraction[d] 0-60 minutes > 60 minutes	 1.5 hr 6.48 E+05 0.04 0

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Table 15.3-3 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Accident scenario in which startup feedwater is available	
Duration of accident (hr)	8.0 hr
Steam release rate (lb/sec)	60
Leak flashing fraction	Not applicable

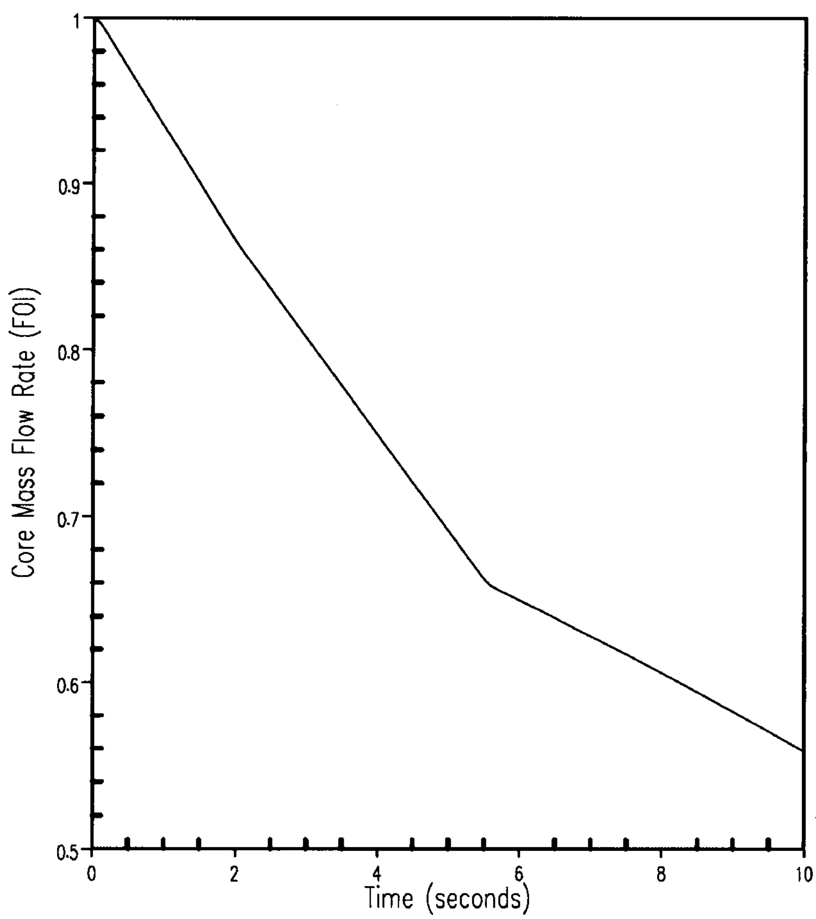
Notes:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity released to the coolant from the assumed fuel failures, it is not significant.

b. Equivalent to 300 gpd cooled liquid at 62.4 lb/ft^3.

c. Heat removal is achieved by steaming and by passive core cooling system operation in the limiting case where the startup feedwater system is not available. When heat removal by the passive core cooling system exceeds the decay heat load, steam releases are terminated.

d. No credit for iodine partitioning is taken for flashed leakage. Credit is taken for a partition coefficient of 0.10 for alkali metals. Flashing is terminated by the passive core cooling system operation reducing the RCS below the saturation temperature of the secondary.



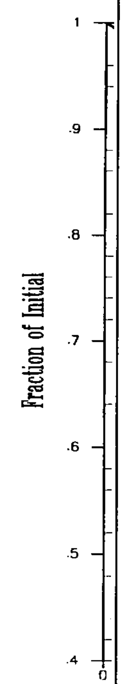


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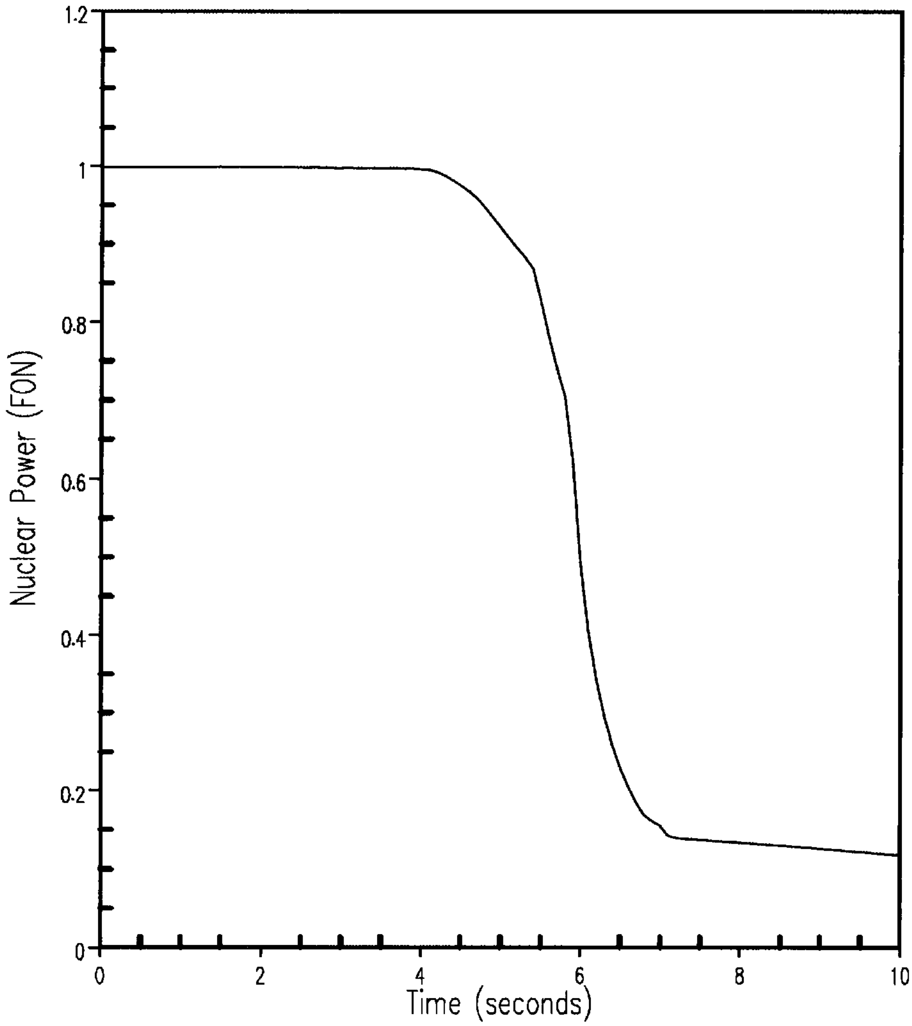
Figure 15.3.1-1

Core Mass Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down



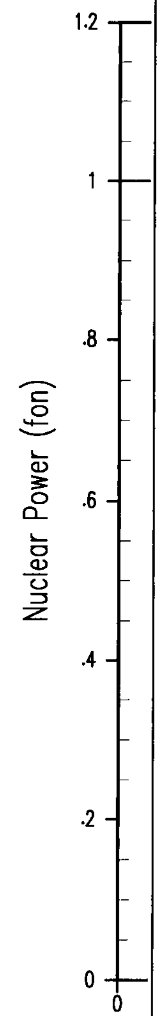


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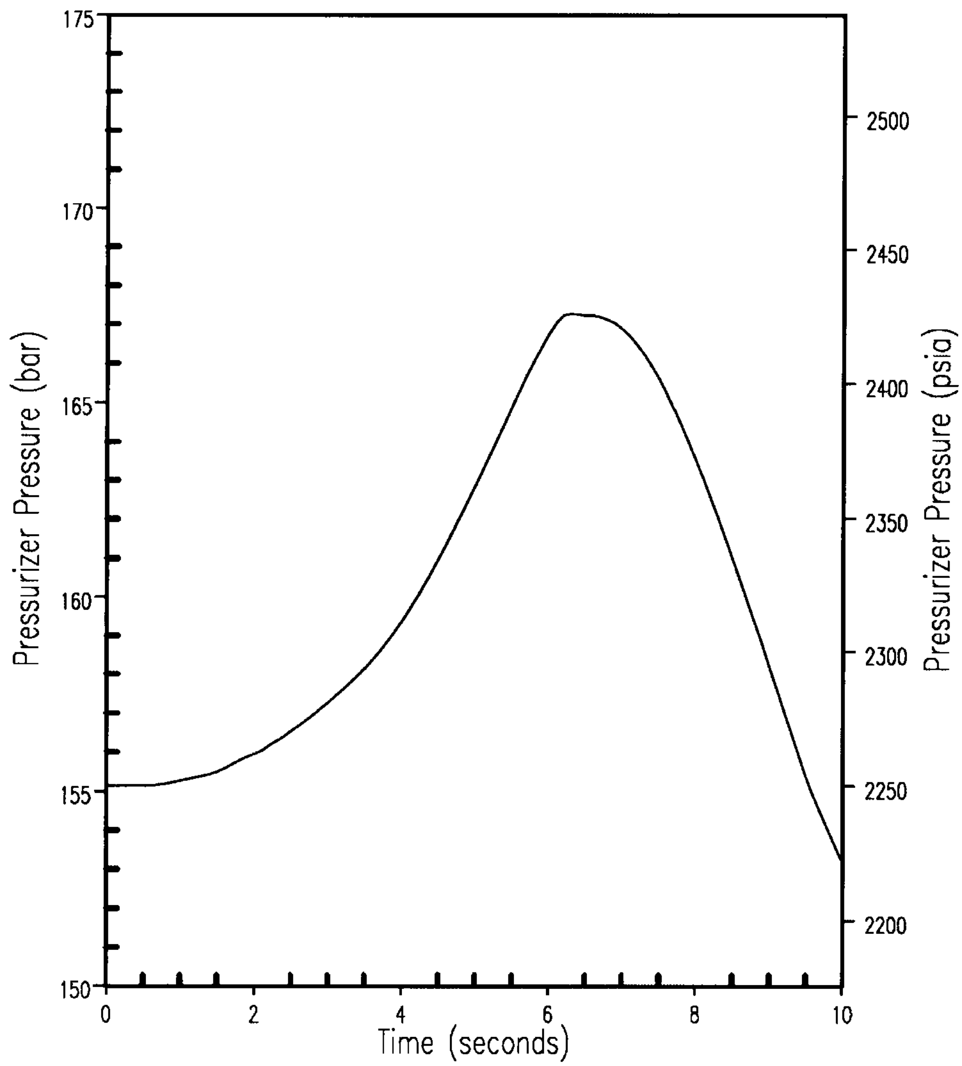
Figure 15.3.1-2

Nuclear Power Transient for Four Cold Legs in Operation, Two Pumps Coasting Down



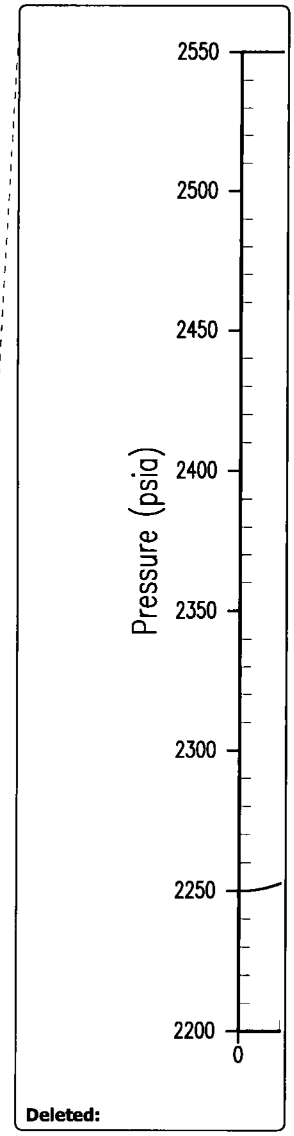
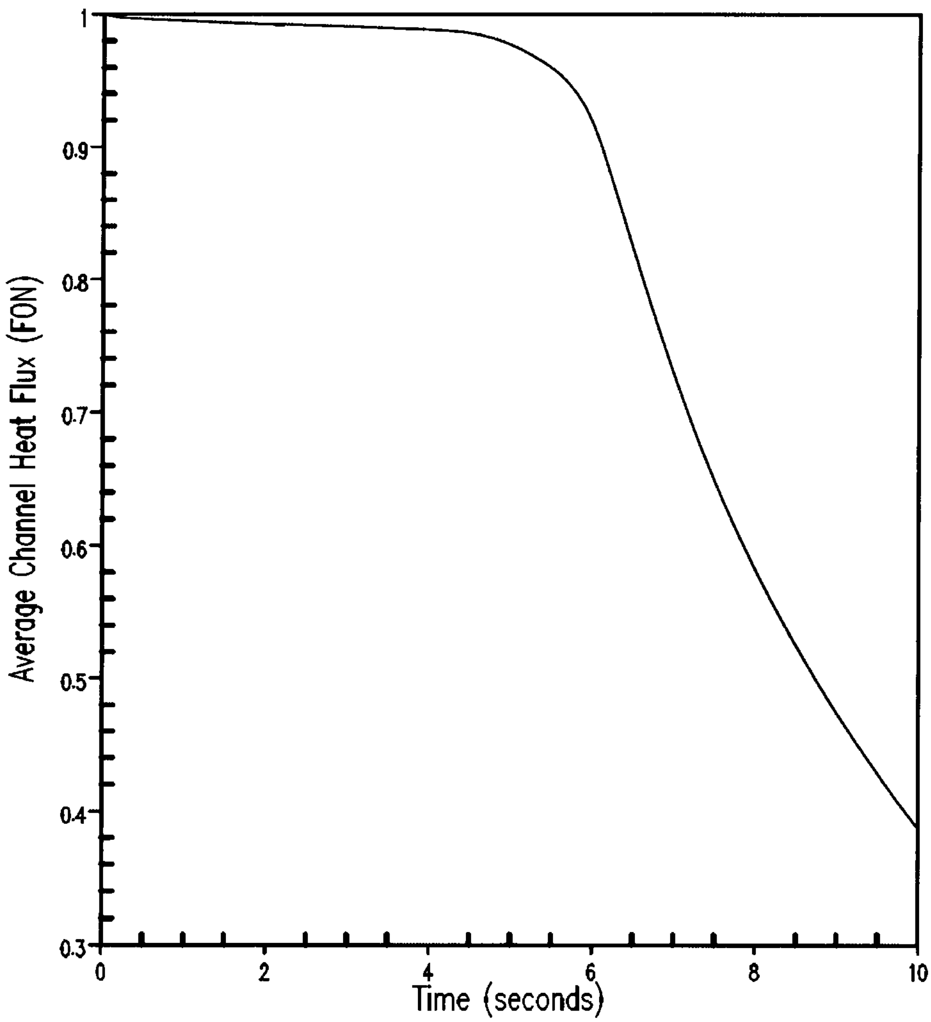


Figure 15.3.1-3

Pressurizer Pressure Transient for Four Cold Legs in Operation, Two Pumps Coasting Down



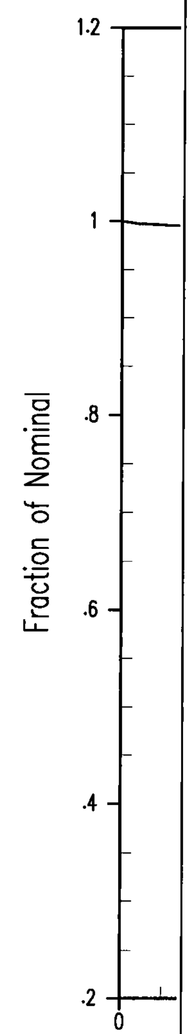


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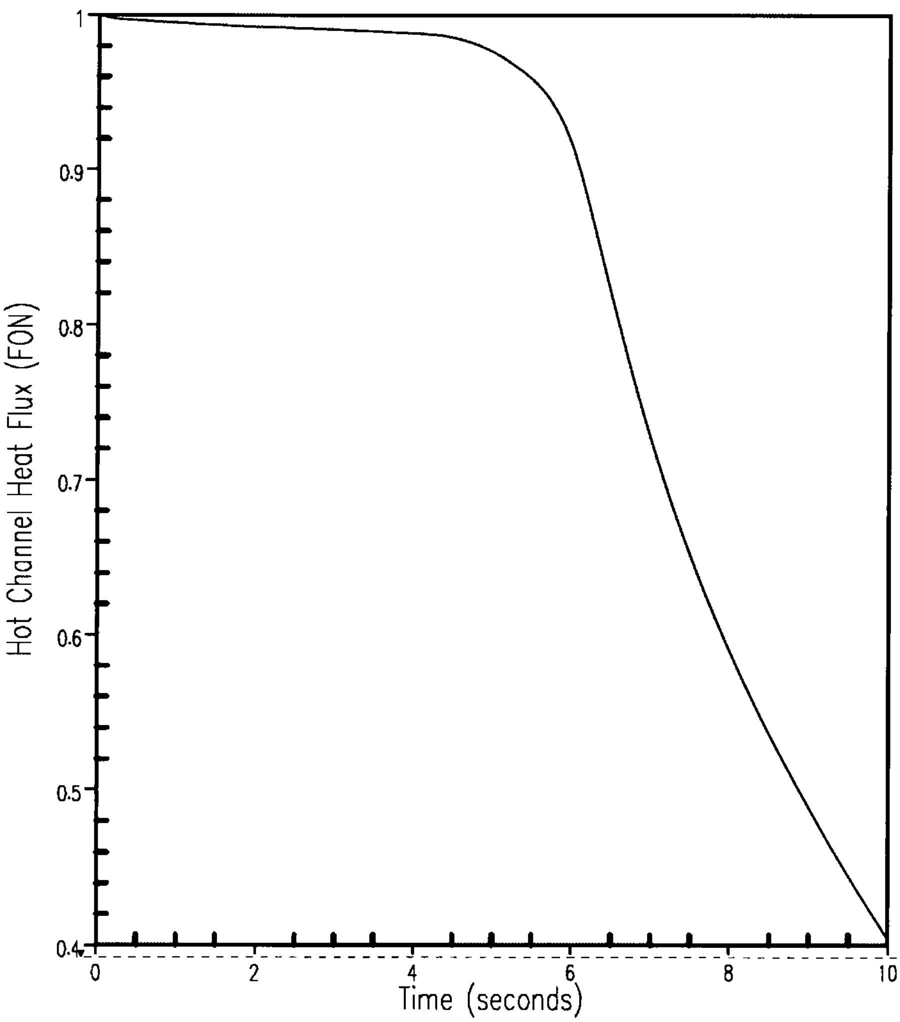
Figure 15.3.1-4

Average Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down



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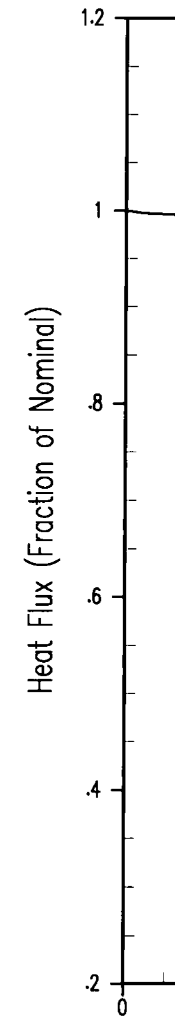
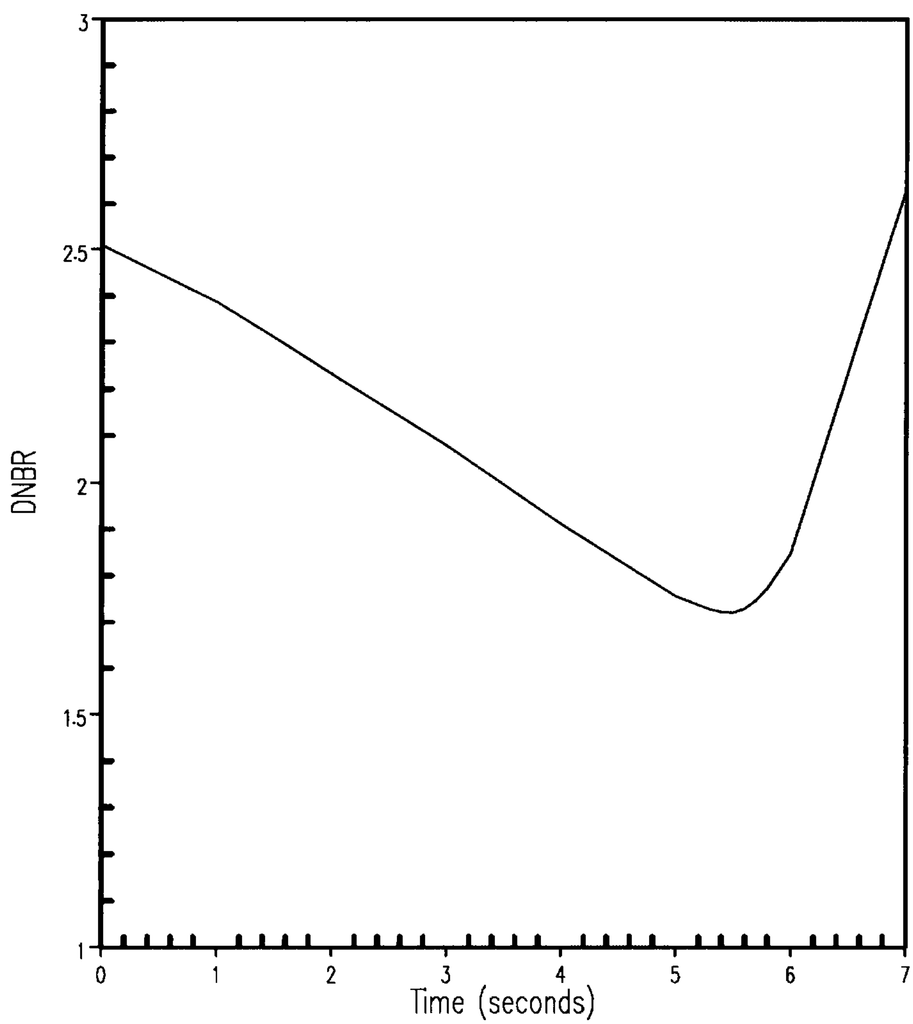


Figure 15.3.1-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down



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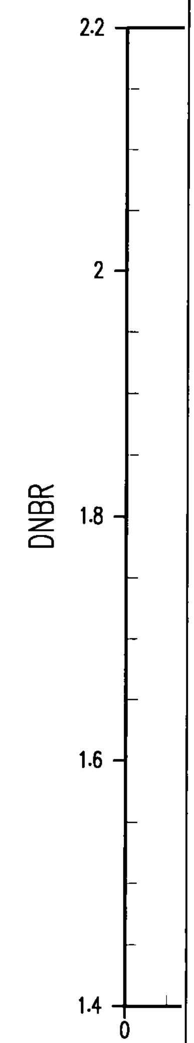

Figure 15.3.1-6

DNBR Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

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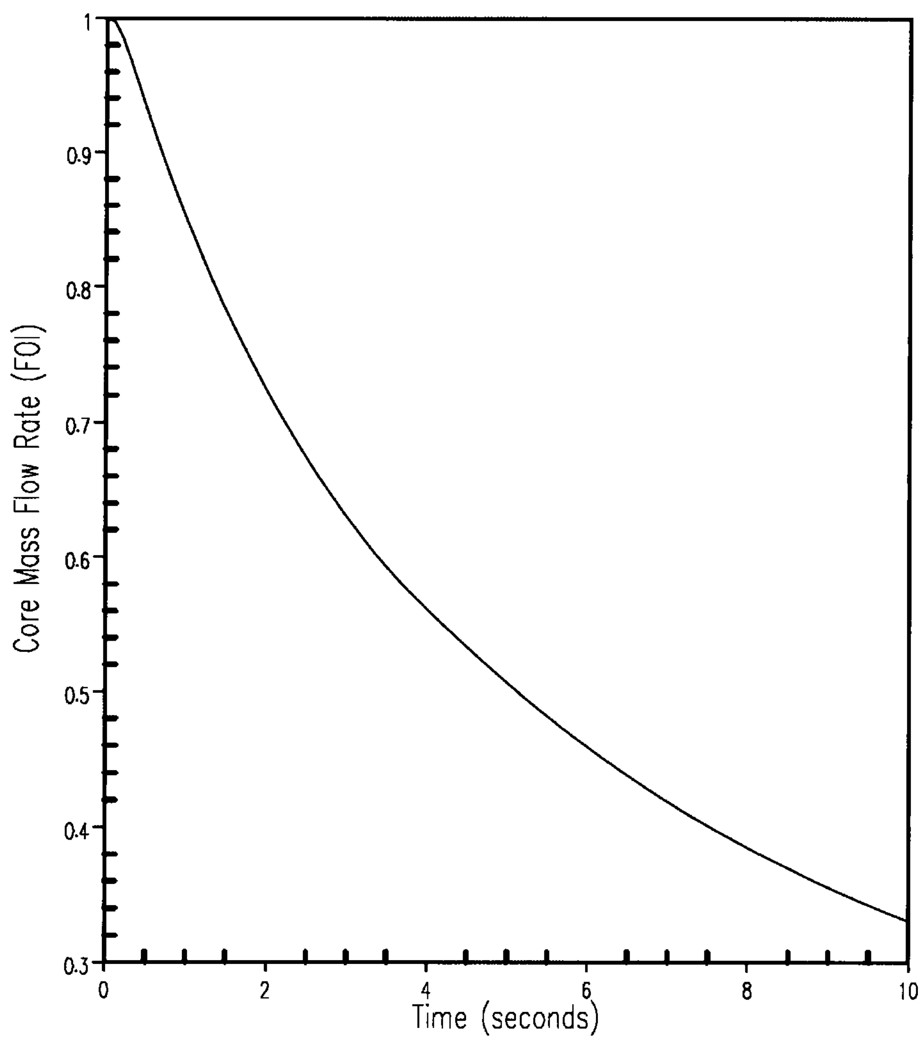

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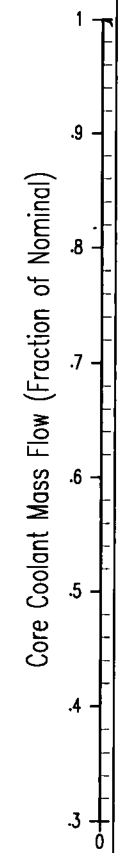
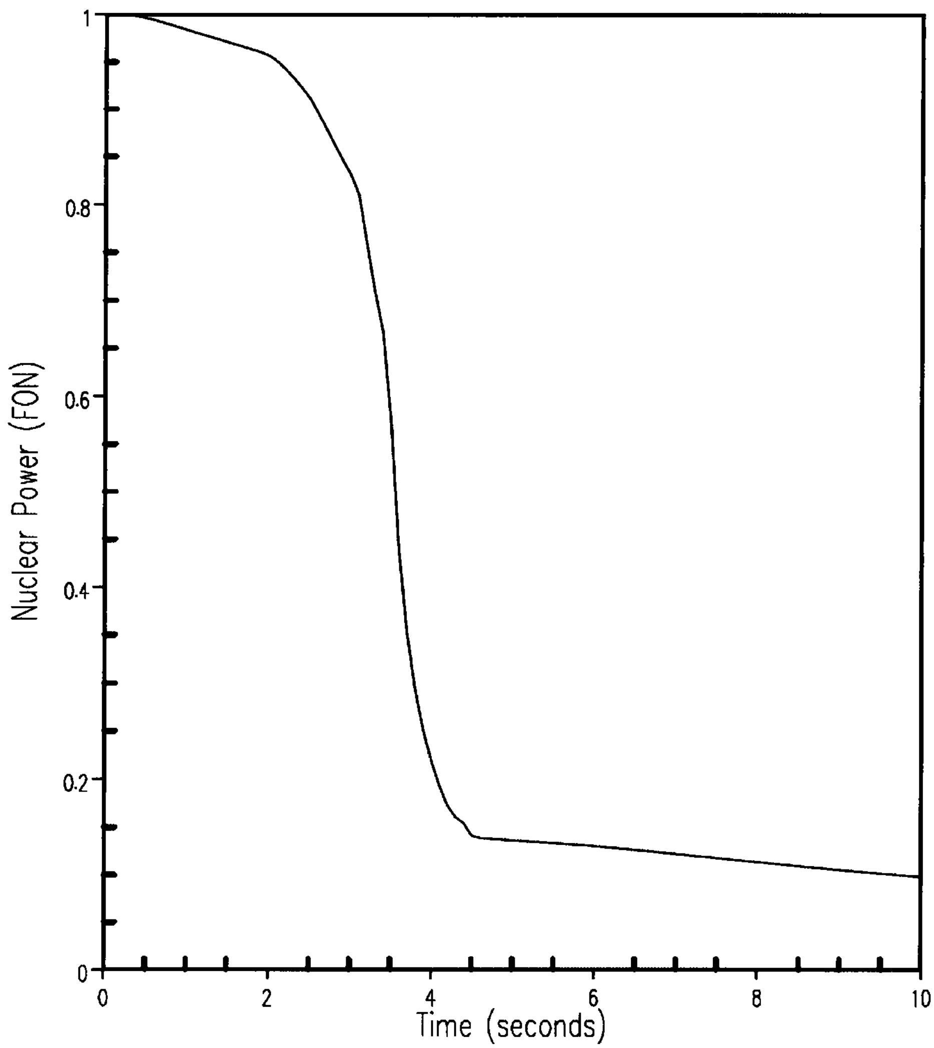


Figure 15.3.2-1

Core Mass Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



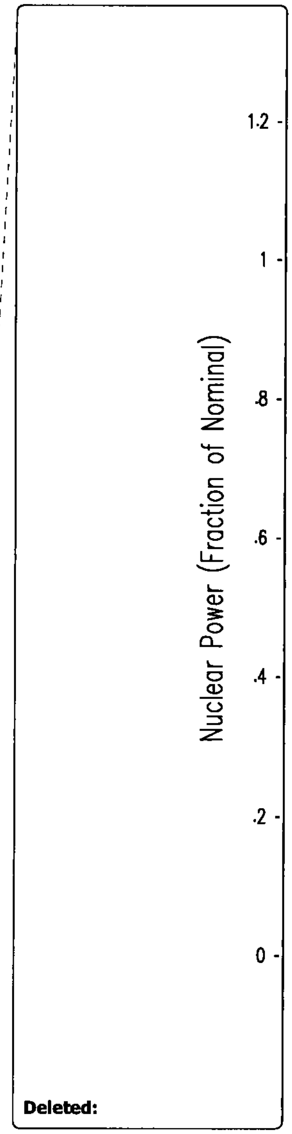
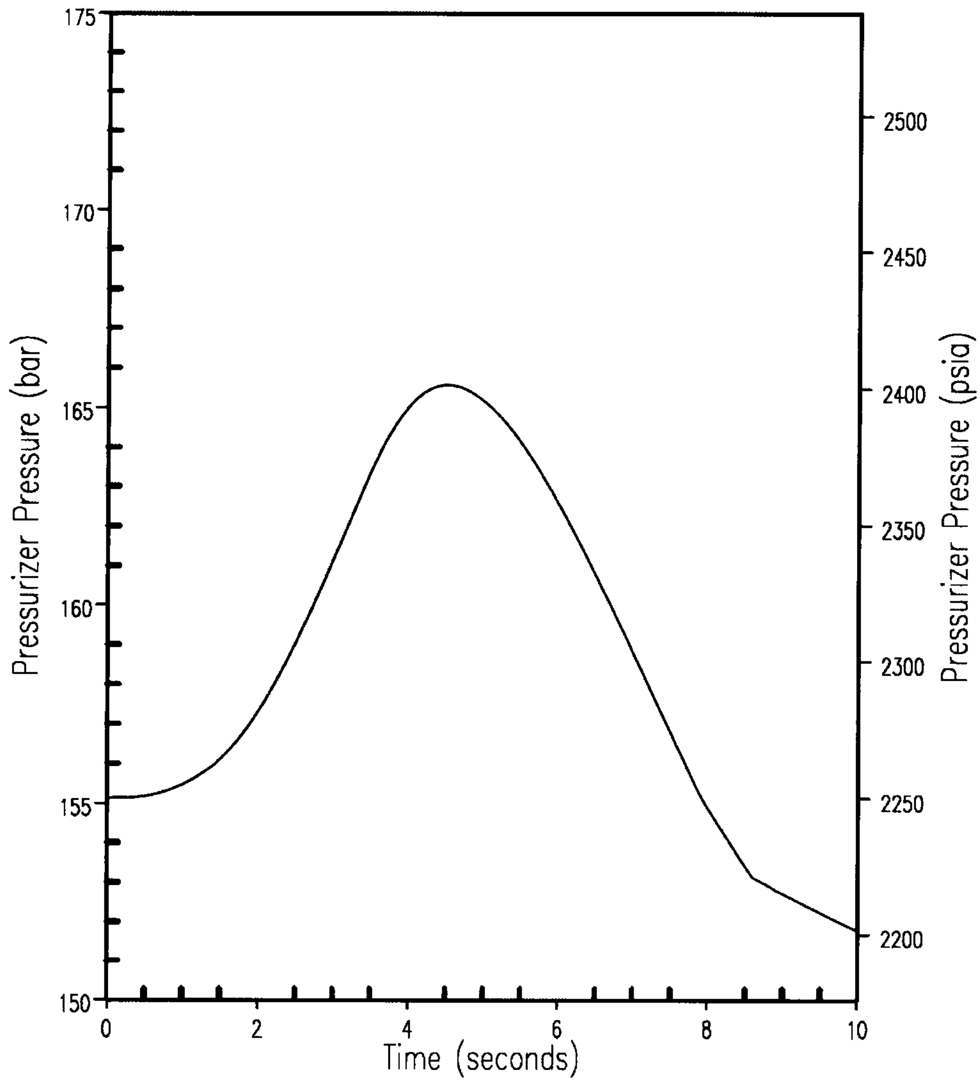


Figure 15.3.2-2

Nuclear Power Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



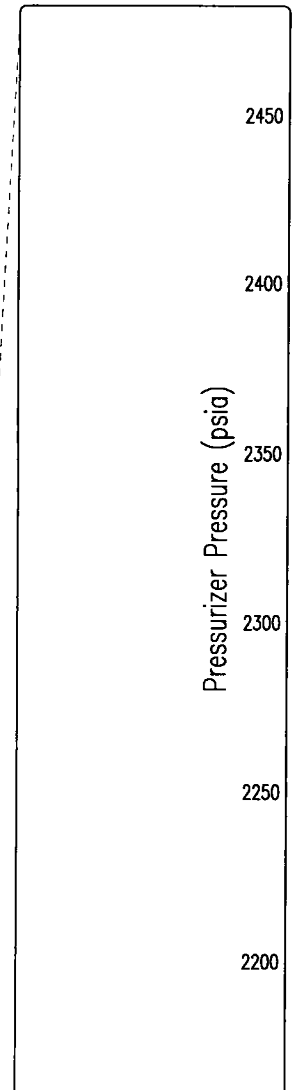


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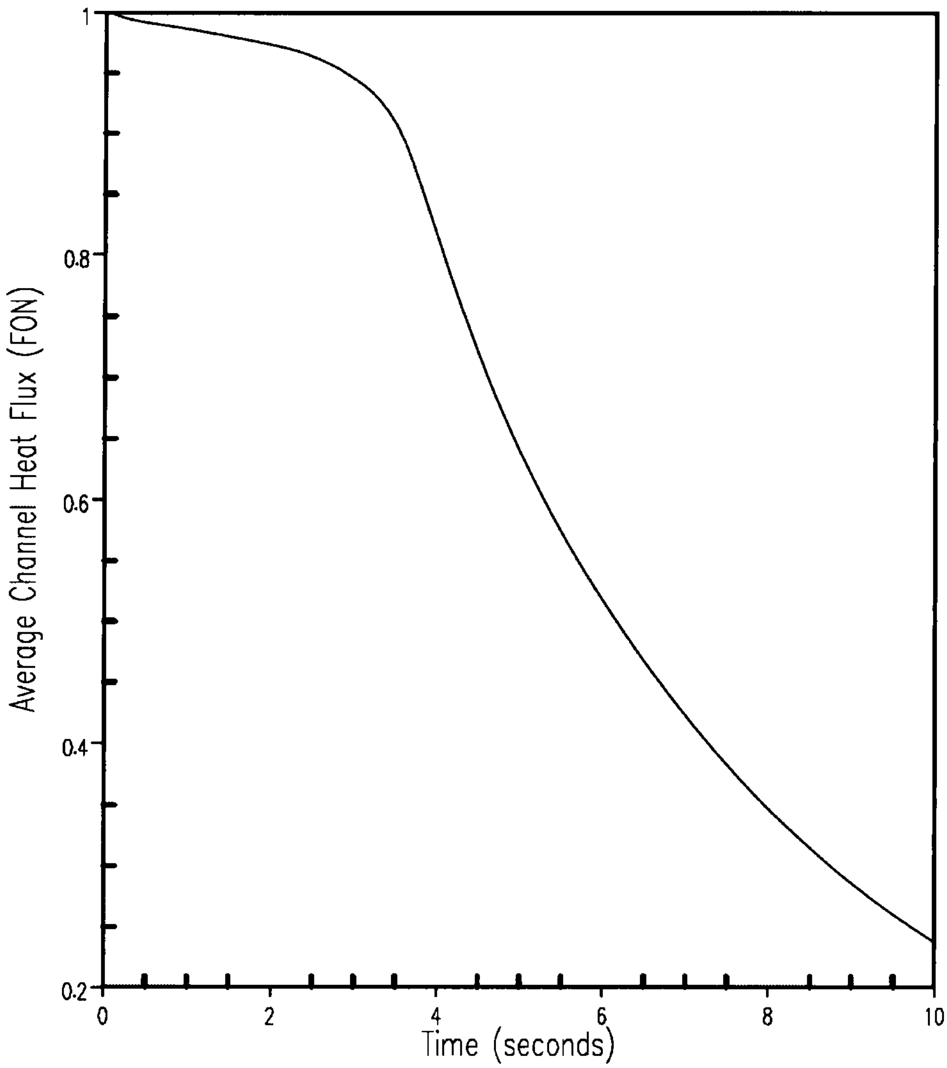
Figure 15.3.2-3

Pressurizer Pressure Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



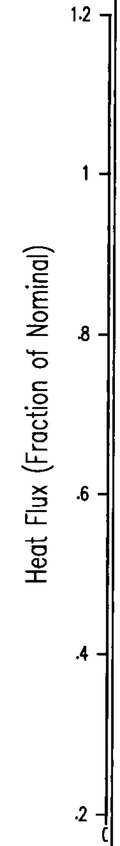


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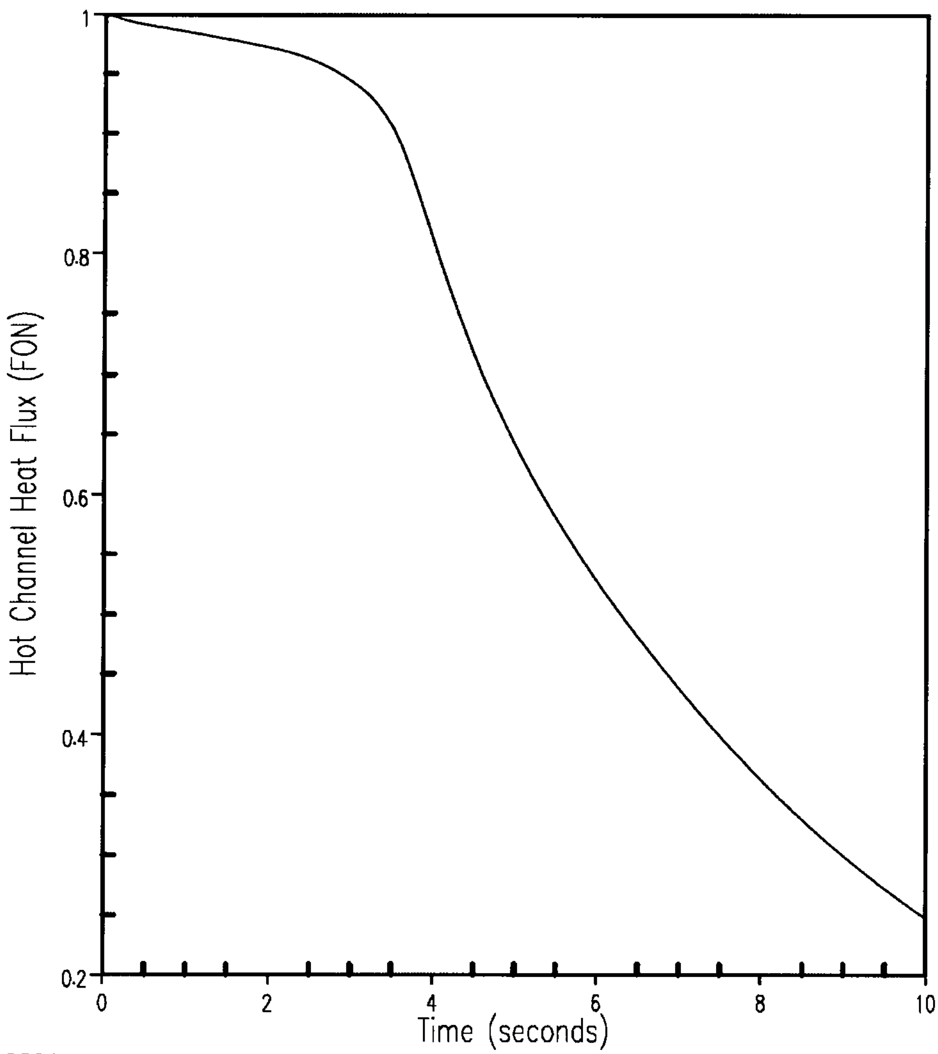
Figure 15.3.2-4

Average Channel Heat Flux Transient for
Four Cold Legs in Operation, Four Pumps Coasting Down



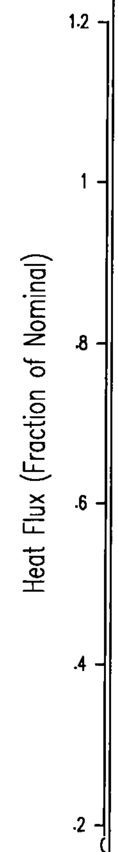


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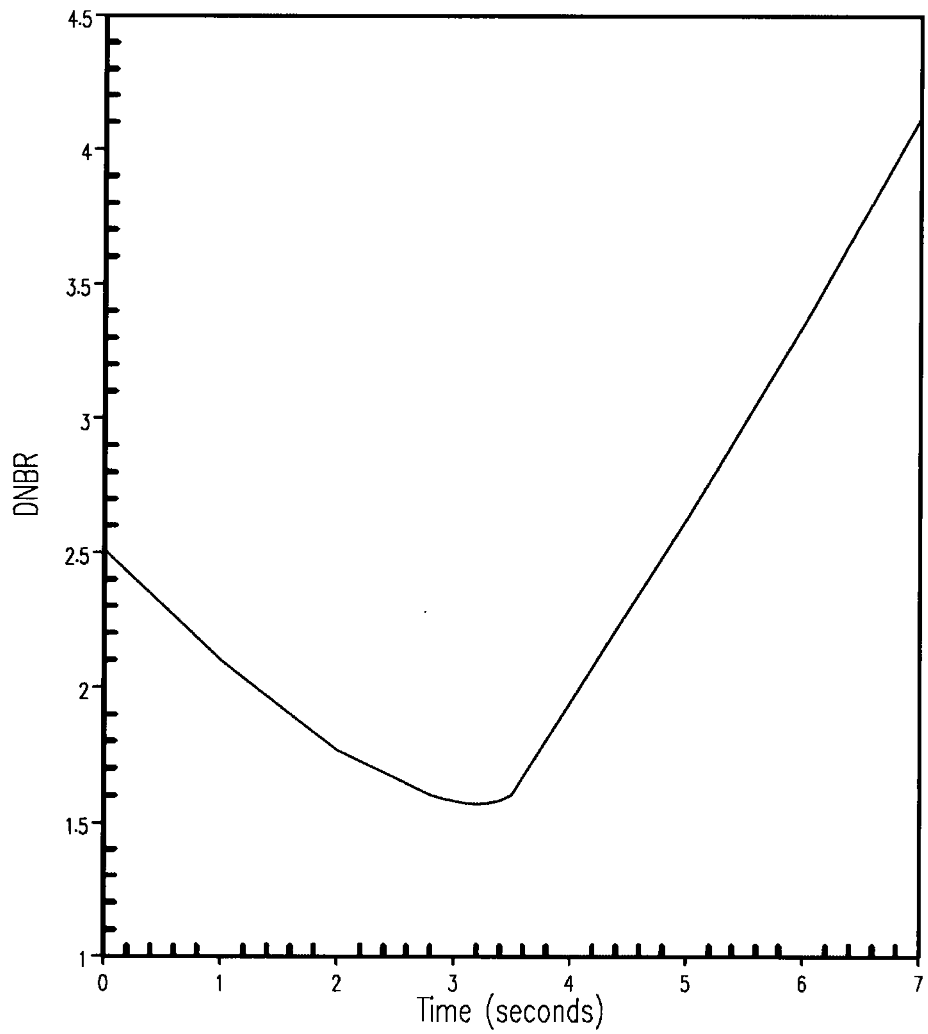
Figure 15.3.2-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



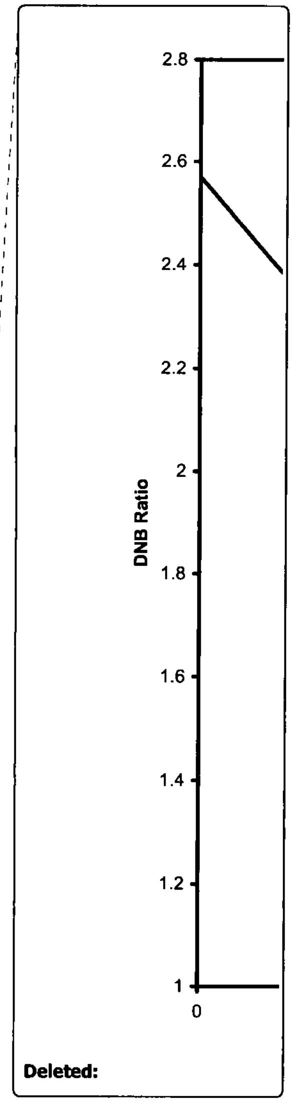
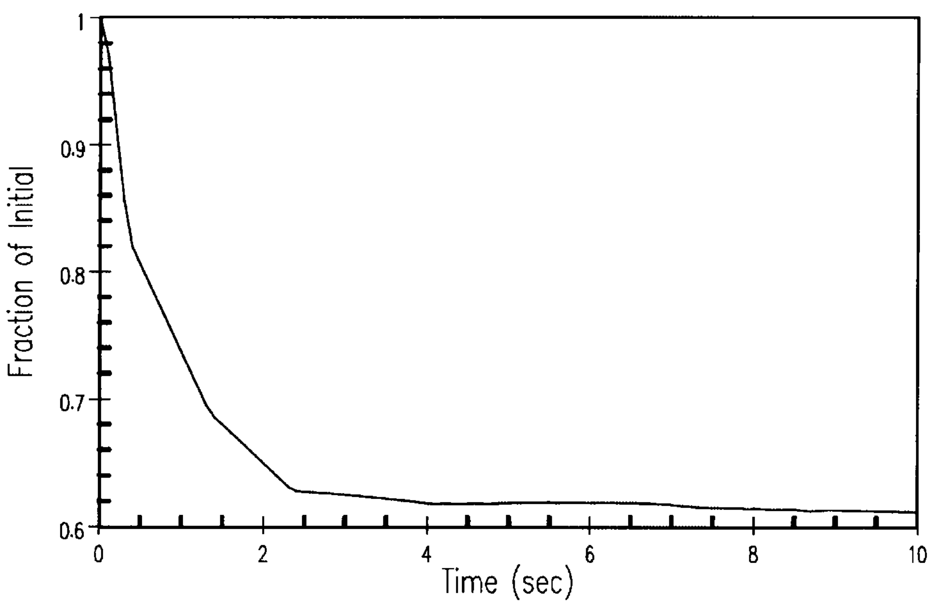


Figure 15.3.2-6

DNBR Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



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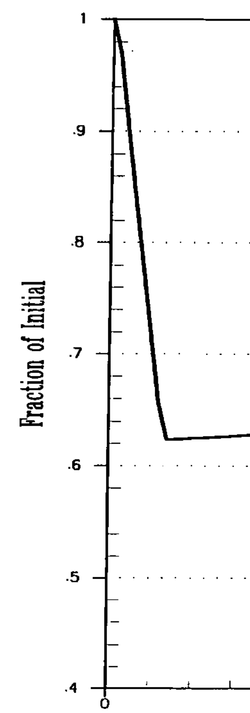
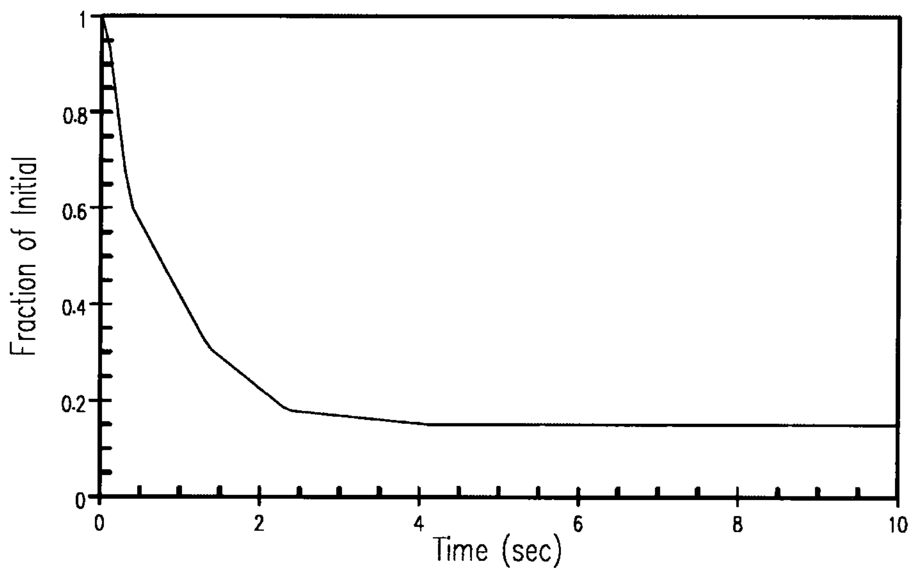


Figure 15.3.3-1

Core Mass Flow Transient for
Four Cold Legs in Operation, One Locked Rotor



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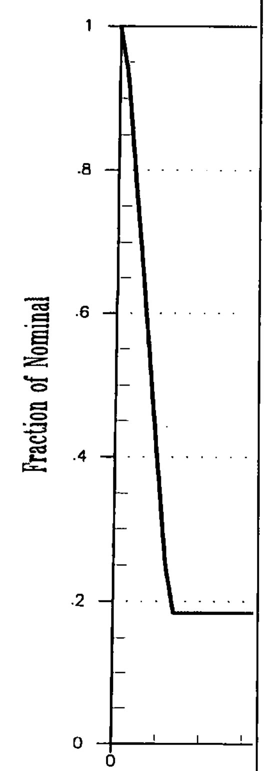
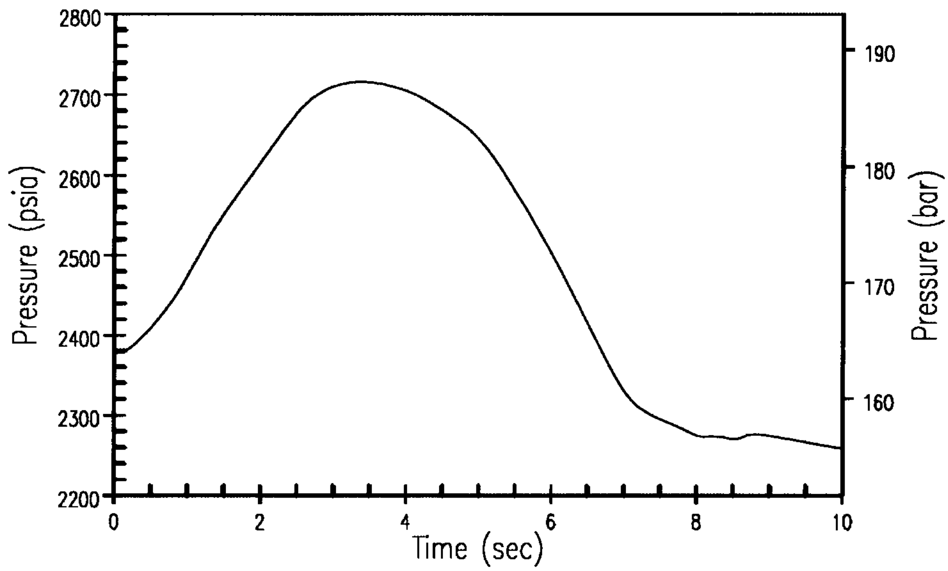


Figure 15.3.3-2

Faulted Loop Volumetric Flow Transient for Four Cold Legs in Operation, One Locked Rotor



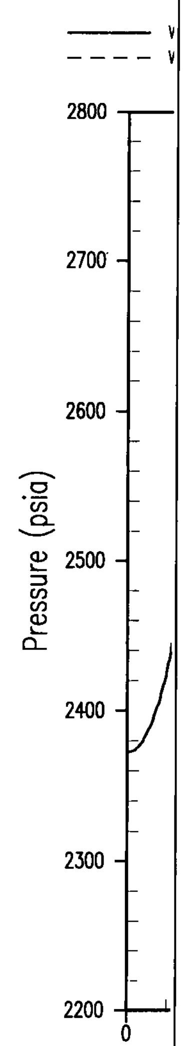


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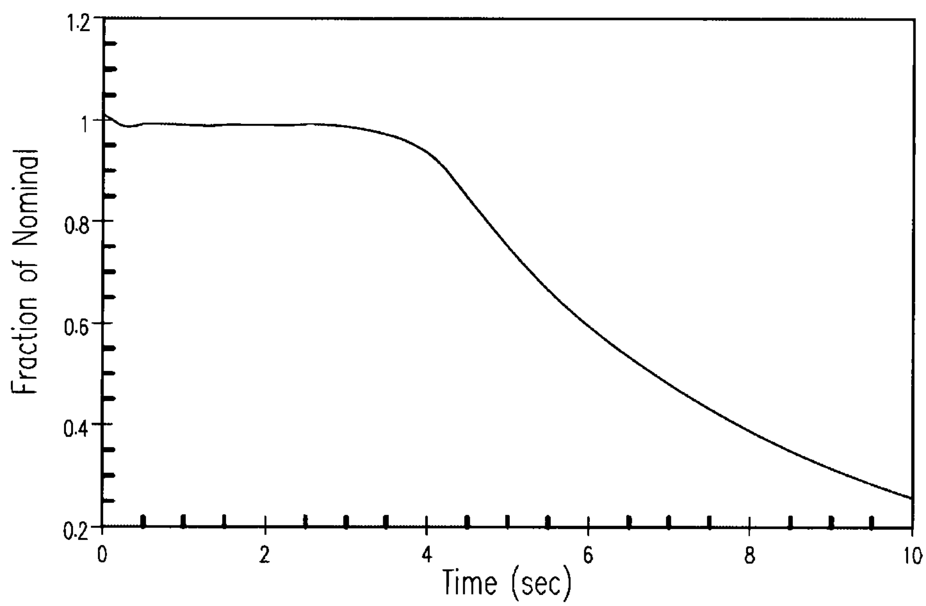
Figure 15.3.3-3

Peak Reactor Coolant Pressure for Four Cold Legs in Operation, One Locked Rotor



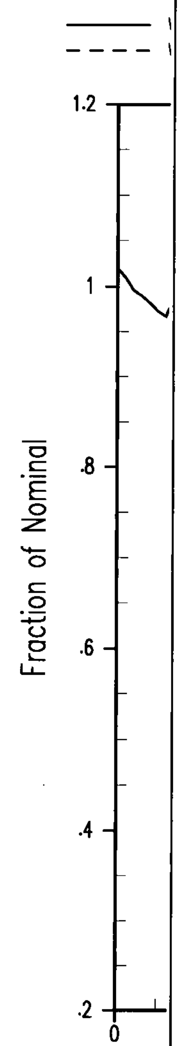


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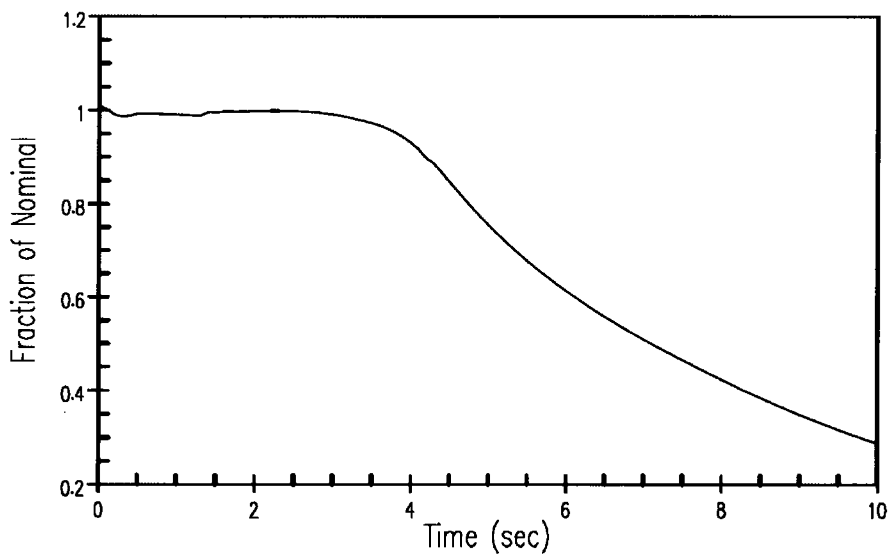
Figure 15.3.3-4

Average Channel Heat Flux Transient for
Four Cold Legs in Operation, One Locked Rotor



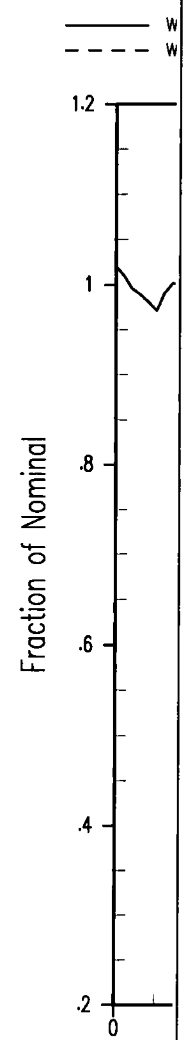
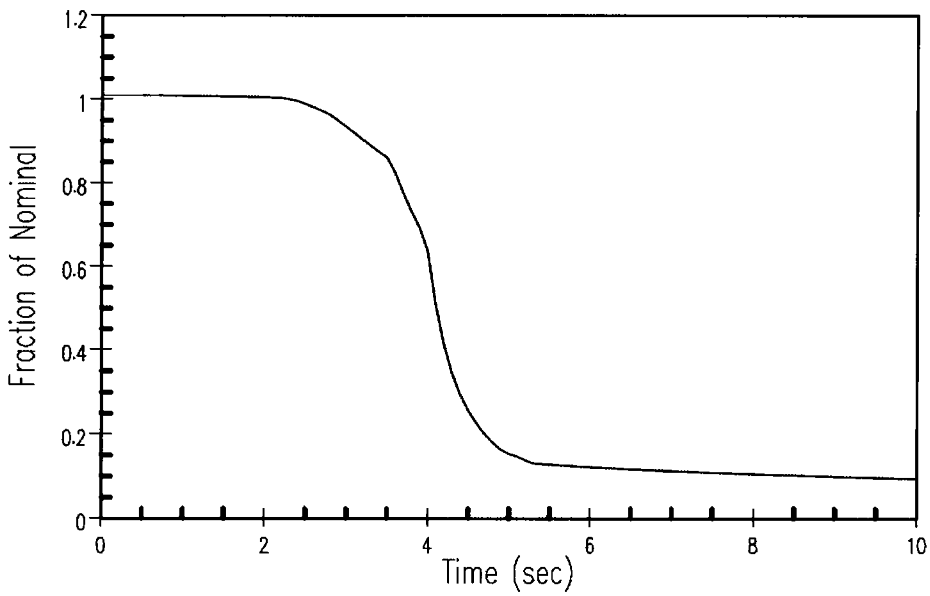


Figure 15.3.3-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, One Locked Rotor



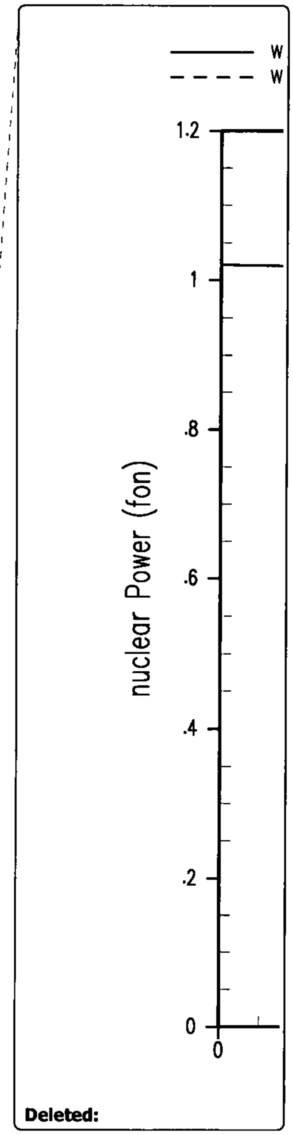
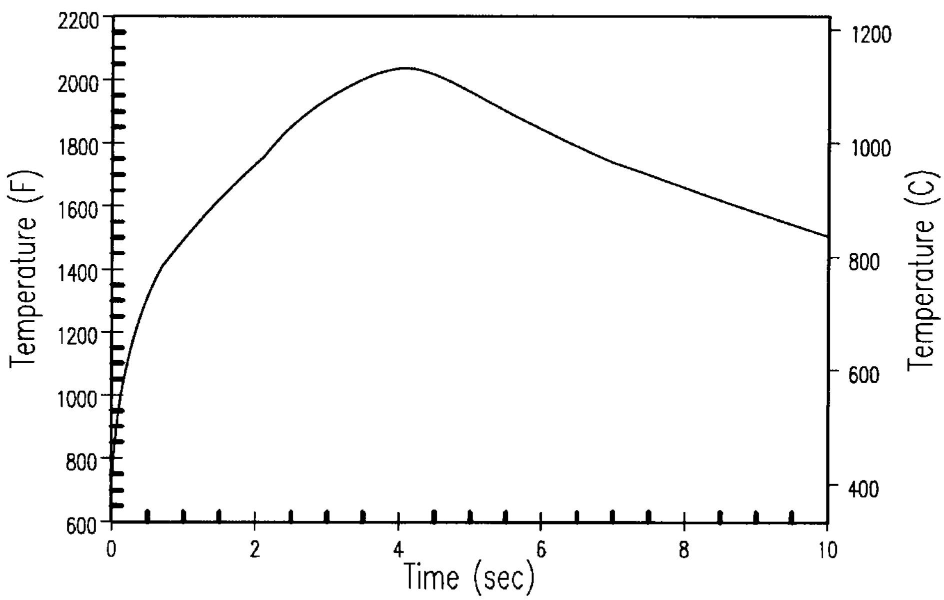


Figure 15.3.3-6

Nuclear Power Transient for
Four Cold Legs in Operation, One Locked Rotor



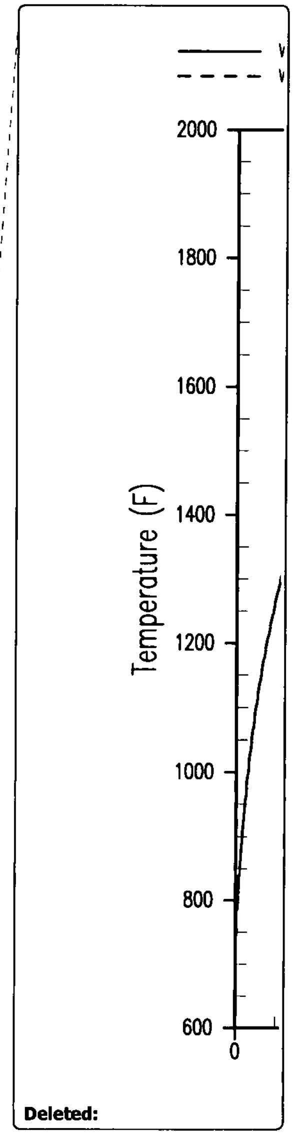


Figure 15.3.3-7

Cladding Inside Temperature Transient for
Four Cold Legs in Operation, One Locked Rotor

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.4	Change Summary Description
[15.4-1]	15.4.1, Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-power Startup Condition	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.4-2]	15.4.2, Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.4-3]	15.4.3, Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.4-4]	15.4.6, Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.4-5]	15.4.8, Spectrum of Rod Cluster Control Assembly Ejection Accidents	The AFC was analyzed in accordance with WCAP-15806-P-A to determine acceptability with respect to the criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3. WCAP-15806-P-A is generally applicable to all Westinghouse reactors, and describes the 3D methods to analyze the rod ejection transient. The complete analysis and summary of conclusions are presented in Section 15.4.8.
[15.4-6]	15.4.8.3, Radiological Consequences	Editorial Changes. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The rod ejection dose analysis was revised based on SRP Section 4.2, Revision 3, Appendix B, which requires the enthalpy increase following a rod ejection be considered in the source term generated for the dose analysis, and presents an equation to use. More recent NRC guidance i.e. Draft Guide 1199 (DG-1199) and the subsequent clarification to DG-1199 expand upon the SRP 4.2 Rev 3 requirements, changing the pre-accident gap fractions and the increased gap activity due to a reactivity insertion event. The changes to the gap fraction were incorporated into the rod ejection dose analysis. The doses were revised based on updated analysis.

Change No.	Chapter 15 Section 15.4	Change Summary Description
[15.4-7]	15.4.10 Reference	References updated consistent with updated Section 15.4. Additionally, the edition date of Reference 10 was corrected to "1973".
[15.4-8]	Table 15.4-4 (Sheets 1 and 2)	The radial peaking factor was increased to 1.75 from 1.65. Gap fractions were updated and fuel enthalpy was added as part of the inclusion of the updated DG-1199 guidance. Leak rate updated based on the value modeled in the analysis. Alkali metal partition factor updated to be consistent with moisture carryover.
[15.4-9]	15.4.6.2.6 Dilution During Full Power Operation (Mode 1)	The existing boron dilution analysis was calculated using an initial boron concentration consistent with the control rods at the all rods out (ARO) position; this analysis was updated to model a concentration consistent with the rods at the rod insertion limit (RIL).

15.4 Reactivity and Power Distribution Anomalies

A number of faults are postulated that result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Analyses are presented for the most limiting of these events.

The following incidents are discussed in this section:

A. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

B. Uncontrolled RCCA bank withdrawal at power

C. RCCA misalignment

D. Startup of an inactive reactor coolant pump at an incorrect temperature

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that results in an increased reactor coolant flow rate (not applicable to AP1000)

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

G. Inadvertent loading and operation of a fuel assembly in an improper position

H. Spectrum of RCCA ejection accidents

Items A, B, D, and F above are Condition II events, item G is a Condition III event, and item H is a Condition IV event. Item C includes both Conditions II and III events.

The applicable transients in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing as discussed in subsection 15.4.8.

Radiological consequences are reported only for the limiting case.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-power Startup Condition

Comment [B1]: [15.4-1]

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power. The at-power case is discussed in subsection 15.4.2.

The reactor may be brought to a critical condition by either RCCA withdrawal or boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see subsection 15.4.6).

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The RCCA drive mechanisms are grouped into preselected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks are withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event (a fault of moderate frequency) as defined in subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion limits the power during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Source range high neutron flux reactor trip

  This trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

- Intermediate range high neutron flux reactor trip

  This trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (low setting)

  This trip function is actuated when two out of four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (high setting)

  This trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.

- High nuclear flux rate reactor trip

  This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above a preset setpoint.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

15.4.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the departure from nucleate boiling ratio (DNBR) calculation. In the first stage, the average core nuclear calculation is performed using spatial neutron kinetics methods, using the code TWINKLE (Reference 1), to determine the average power generation with time, including the various total core feedback effects (doppler reactivity and moderator reactivity).

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in VIPRE-01 (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. The following assumptions are made to give conservative results for a startup accident:

- Because the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values, as a function of power, are used (see Table 15.0-2).
- Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. After the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux (see Table 15.0-2).
- The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect and thereby increase the neutron flux peak. The initial effective multiplication factor (k_{eff}) is assumed to be 1.0 because this results in the worst nuclear power transient.
- Reactor trip is assumed to be initiated by the power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent uncertainty increase is assumed for the power range flux trip setpoint, raising it to 35 percent from the nominal value of 25 percent.

  Because the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See subsection 15.0.5 for RCCA insertion characteristics.
- The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential RCCA banks having the greatest combined worth at maximum speed (45 inches per minute). Control rod drive mechanism design is discussed in Section 4.6.

- The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

- The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

- Four reactor coolant pumps are assumed to be in operation.

- Pressurizer pressure is assumed to be 50 psi below nominal for steady-state fluctuations and measurement uncertainties.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or components adversely affects the consequences of the accident. A loss of offsite power as a consequence of a turbine trip disrupting the grid is not considered because the accident is initiated from a subcritical condition where the plant is not providing power to the grid.

15.4.1.2.2 Results

Figures 15.4.1-1 through 15.4.1-4 show the transient behavior for the uncontrolled RCCA bank withdrawal from subcritical incident. The accident is terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest-worth sequential rod cluster control banks, both assumed to be in their highest incremental worth region.

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Figure 15.4.1-1 shows the average neutron flux transient. The energy release and the fuel temperature increases are relatively small. The heat flux response (of interest for DNB considerations) is also shown in Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is margin to DNB during the transient because the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. Figures 15.4.1-3 and 15.4.1-4 shows the response of the average fuel temperature and the inner clad temperature, respectively. The minimum DNBR at all times remains above the design limit value (see Section 4.4).

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The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. Subsequently, the plant may be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected because the combination of thermal power and the coolant temperature results in a DNBR greater than the safety analysis limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

Comment [B2]: [15.4-2]

15.4.2.1 Identification of Causes and Accident Description

An uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system (PMS) is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4).

This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

The automatic features of the PMS that prevent core damage following the postulated accident include the following:

- Power range neutron flux instrumentation actuates a reactor trip if two out of four divisions exceed an overpower setpoint. In particular, the power range neutron flux instrumentation provides the following reactor trip functions:

  1. Reactor trip on high power range neutron flux (high setpoint)
  2. Reactor trip on high power range positive neutron flux rate

  The latter trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against reactivity insertion rate accidents at mid and low power, and it is always active.

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- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against violating the DNB design basis. The overtemperature ΔT reactor trip function initiates a reactor trip to prevent the plant from exceeding the core thermal limits. With the overtemperature ΔT reactor trip function,

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setpoints are selected to match the non-linear characteristics of the core thermal limits. Dynamic compensation is included to account for transport times from the hot and cold legs to the core and to provide protection in a timely fashion such that the core thermal limits are not exceeded.

- Reactor trip is actuated if any two out of four ΔT power divisions exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable linear heat generation rate (kW/ft) from being exceeded.

- A high pressurizer pressure reactor trip is actuated from any two out of four pressure divisions when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A high pressurizer water level reactor trip is actuated from any two out of four level divisions that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).

In addition to the preceding reactor trips, there are the following RCCA withdrawal blocks:

- High neutron flux (two out of four power range)
- Overpower ΔT (two out of four)
- Overtemperature ΔT (two out of four)

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- High neutron flux (fixed setpoint)
- High pressurizer pressure (fixed setpoint)
- Low pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of ac power during an uncontrolled RCCA bank withdrawal at power event have been evaluated and did not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of ac power in comparison to the reactor shutdown time for an uncontrolled RCCA bank withdrawal at power event. The primary effect of the loss of ac power is to cause the reactor coolant pumps (RCPs) to coast down. The PMS includes a five second minimum delay between the reactor trip and the turbine trip. In addition, a three second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coast down due to the loss of ac power, it is clear that the plant

Deleted: The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7 and Reference 13.¶
Figure 15.0.3-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include adverse instrumentation and setpoint uncertainties so that under nominal conditions, a trip occurs well within the area bounded by these lines.¶

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shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this uncontrolled RCCA bank withdrawal at power analysis because the plant will be shut down well before the RCPs begin to coast down.

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15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed using the LOFTRAN (References 3 and 11) code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0.3-1 are used to define the inputs to LOFTRAN that determine the minimum DNBR during the transient.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. In performing a conservative analysis for an uncontrolled RCCA bank withdrawal at-power accident, the following assumptions are made:

- The nominal initial conditions are assumed in accordance with the revised thermal design procedure. Uncertainties in the initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 9).

- Two sets of reactivity coefficients are considered:

  Minimum reactivity feedback — A least-negative moderator temperature coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed (see Figure 15.0.4-1).

  Maximum reactivity feedback — A conservatively large positive moderator density coefficient corresponding to the end of core life and a large (in absolute magnitude) negative Doppler power coefficient are assumed (see Figure 15.0.4-1).

- The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The high positive flux rate trip is assumed to be actuated when the power range neutron flux changes at a rate higher than 9% per second with a two second rate-lag time constant. The overtemperature ΔT trip includes adverse instrumentation and setpoint uncertainties. The delays for trip actuation assumed are given in Table 15.0-4a.

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- The RCCA trip insertion characteristic is based on the assumption that the highest-worth assembly is stuck in its fully withdrawn position.

- A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks, having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to the DNBR limit.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident.

15.4.2.2.2 Results

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative rapid (80 pcm/s) RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the transient. Because this is rapid with respect to the thermal time constants of the fuel, small changes in temperature and pressure result, and the DNB design basis described in Section 4.4 is met.

The transient response for a representative slow (5 pcm/s) RCCA withdrawal from full power is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The DNB design basis described in Section 4.4 is met.

Figure 15.4.2-13 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback. Minimum DNBR, occurs immediately after rod motion. Three reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux, high positive flux rate and overtemperature ΔT channels. The minimum DNBR is greater than the design limit value described in Section 4.4. Note that the high positive flux rate trip was needed for only one case (100% power, minimum reactivity feedback, 110 pcm/s) to prevent the peak heat flux from exceeding 118%.

Figures 15.4.2-14 and 15.4.2-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively. Minimum DNBR, occurs immediately after rod motion. The results are similar to the 100-percent power case, except as the initial power is

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The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to PMS action in initiating a reactor trip.

Referring to Figure 15.4.2-14, for example, it is noted that:

A. For high reactivity insertion rates (between 38 pcm/s and 110 pcm/s), reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases.

B. For minimum reactivity feedback cases that assume reactivity insertion rates of less than 38 pcm/s, protection is provided by the overtemperature ΔT trip.

C. Reactor trip is initiated by overtemperature ΔT for the entire range of reactivity insertion rates for the maximum reactivity feedback cases.

D. For most of the minimum feedback cases and all of the maximum feedback cases, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which removes additional heat from the reactor coolant system, sharply decreases the rate of increase of reactor coolant system average temperature. This decrease in the rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation. This causes the overtemperature ΔT setpoint to be reached later, with resulting lower minimum DNBRs.

For transients initiated from full power (see Figure 15.4.2-13), both minimum and maximum reactivity feedback, the minimum DNBR occurs for the lower reactivity insertion rates that trip on overtemperature ΔT (higher reactivity insertion rates trip on high neutron flux).

At lower reactivity insertion rates the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) because for these lower reactivity insertion rates, the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

Steam generator safety valves never open before the reactor trip, for transients initiated at full power.

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Because the RCCA bank withdrawal at-power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For fast reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature still remains below the fuel melting temperature.

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For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before the DNB design basis is violated. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak centerline temperature remains below the fuel melting temperature.

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The reactor is tripped during the RCCA bank withdrawal at-power transient that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

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The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may be cooled down further by following normal plant shutdown procedures.

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15.4.2.2.3 Overpressure Evaluation Results

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In addition to the DNB cases discussed above, several cases are analyzed to ensure that the maximum reactor coolant system pressure does not exceed 110% of the design pressure. The cases cover a range of reactivity insertion rates from less than 1 pcm/s to 110 pcm/s and power levels from 10% to 100% power. Initial condition uncertainties on power, pressure and average temperature are conservatively included and the thermal design flow rate is assumed. The most limiting case was for a reactivity insertion rate of 36 pcm/s and an initial power level of 65% power. The peak pressure calculated is 2698.4 psia which is well below the limit of 2748.5 psia.

15.4.2.3 Conclusions

The power range neutron flux instrumentation, overtemperature ΔT and high positive flux rate trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The DNB design basis, as defined in Section 4.4, is met for all cases. The maximum reactor coolant system pressure remains below 110% of design.

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15.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

Comment [B3]: [15.4-3]

15.4.3.1 Identification of Causes and Accident Description

RCCA misoperation accidents include:

- One or more dropped RCCAs within the same group
- Statically misaligned RCCA
- Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod-at-bottom signal, which actuates a local alarm and a main control room annunciator. Group demand position is also indicated.

RCCAs are moved in preselected banks, and the banks are moved in a preselected sequence. Each bank of RCCAs is divided into one or two groups of four or five RCCAs each. The rods comprising a group operate in parallel. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Because the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which causes rod withdrawal affects the entire group. A single electrical or mechanical failure in the plant control system could, at most, result in dropping one or more RCCAs within the same group. Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are Condition II incidents (incidents of moderate frequency) as defined in subsection 15.0.1. The single RCCA withdrawal event is a Condition III incident, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could withdraw a single RCCA in the control bank because this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage.

The event is classified as a Condition III incident consistent with the philosophy and format of American National Standards Institute, ANSI N18.2. By definition, "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . ." (Reference 10).

This selection of criterion is in accordance with General Design Criterion 25, which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>." (Emphases have been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with General Design Criterion 25.

A dropped RCCA or RCCA bank may be detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication

Misaligned RCCAs are detected by one or more of the following:

- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod deviation alarm
- Rod position indicators

The resolution of the rod position indicator channel is ± 5 percent span (± 7.5 inches). A deviation of any RCCA from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span.

If one or more of the rod position indicator channels is out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs. The operator also takes action as required by the Technical Specifications.

In the extremely unlikely event of multiple electrical failures that result in single RCCA withdrawal, rod deviation and rod control urgent failure are both displayed to the operator, and the rod position indicators indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indication. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III Standard Review Plan Section 15.4.3 evaluation criteria are met; however, due to the increase in local power density, the limits in Figure 15.0.3-1 may be exceeded.

Plant systems and equipment available to mitigate the effects of the various control rod misoperations are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

15.4.3.2.1.1 Method of Analysis

- One or more dropped RCCAs from the same group

  A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. Following the drop, the reduced core power and continued steam demand to the turbine causes the reactor coolant temperature to decrease. In the manual control mode, the plant will establish a new equilibrium condition. The new equilibrium condition is reached through reactivity feedback. In the presence of a negative moderator temperature coefficient, the reactor power rises monotonically back to the initial power level at a reduced inlet temperature with no power overshoot. The absence of any power overshoot establishes the automatic operating mode as a limiting case. If the reactor coolant system temperature reduction is very large, the turbine power may not be able to be maintained due to the reduction in the secondary-side steam pressure and the volumetric flow limit of the turbine system. In this case, the equilibrium power level is less than the initial power. In the automatic control mode, the plant control system detects the drop in core power and initiates withdrawal of a control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to the initial power level. The magnitude of the power overshoot is a function of the plant control system characteristics, core reactivity coefficients, the dropped rod worth, and the available control bank worth.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (References 3 and 11). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures and power level.

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system transient conditions and reactor power. By combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling design basis is shown to be met using the VIPRE-01 code.

- Statically misaligned RCCA

Steady-state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the VIPRE-01 code to calculate the DNBR.

15.4.3.2.1.2 Results

- One or more dropped RCCAs

Figures 15.4.3-1 through 15.4.3-4 show the transient response of the reactor to a dropped rod (or rods) in automatic control. The nuclear power and heat flux drop to a minimum value and recover under the influence of both rod withdrawal and thermal feedback. The prompt decrease in power is governed by the dropped rod worth because the plant control system does not respond during the short rod drop time period. The plant control system detects the reduction in core power and initiates control bank withdrawal to restore the primary side power. Power overshoot occurs after which the core power is restored to the initial power level.

The primary system conditions are combined with the hot channel factors from the nuclear analysis for the DNB evaluation. Uncertainties in the initial conditions are included in the DNB evaluation as discussed in subsection 15.0.3.2. The calculated minimum DNBR, for

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any single or multiple rod drop from the same group is greater than the design limit value described in Section 4.4. The sequence of events for a representative case is shown in Table 15.4-1.

The analysis described previously includes consideration of drops of the RCCA groups which can be selected for insertion as part of the rapid power reduction system. This system is provided to allow the reactor to ride out a complete loss of load from full power without a reactor trip and is described in subsection 7.7.1.10. If these RCCAs are inadvertently dropped (in the absence of a loss-of-load signal), the transient behavior is the same as for the RCCA drop described. The evaluation showed that the DNBR remains above the design limit value as a result of the inadvertent actuation of the rapid power reduction system.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the dropped RCCA event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR occurs before the reactor coolant pumps begin to coast down.

- Statically misaligned RCCA

  The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim or axial offset rod banks are inserted up to their insertion limit with one RCCA fully withdrawn while the reactor is at full power. Multiple independent alarms, including a bank insertion limit or rod deviation alarm, alert the operator well before the postulated conditions are approached.

  For RCCA misalignments in which the mechanical shim or axial offset banks are inserted to their respective insertion limits, with any one RCCA fully withdrawn, the DNBR remains above the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and reactor coolant system temperature are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the DNB evaluation as described in subsection 15.0.3.2.

  DNB does not occur for the RCCA misalignment incident, and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature is that corresponding to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which causes fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator takes action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single Rod Cluster Control Assembly Withdrawal

15.4.3.2.2.1 Method of Analysis

Power distributions within the core are calculated using the computer codes described in Table 4.1-2. The peaking factors are then used by VIPRE-01 to calculate the DNBR for the event. The case of the worst rod withdrawn from the mechanical shim or axial offset bank inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at beginning of life because this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.4.3.2.2.2 Results

For the single rod withdrawal event, two cases are considered as follows:

A. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. In the overall system response, this case is similar to those presented in subsection 15.4.2. The increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum DNBR from falling below the safety analysis limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip is expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5 percent.

B. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA result in the immobility of the other RCCAs in the controlling bank. The transient then proceeds in the same manner as case A.

   For such cases, a reactor trip ultimately occurs although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety analysis limit value. Following reactor trip, normal shutdown procedures are followed.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the single RCCA withdrawal event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit value (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs in those groups selected to be inserted as part of the rapid power reduction system, it is shown that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For cases of any one RCCA fully inserted, or the mechanical shim or axial offset banks inserted to their rod insertion limits with any single RCCA in one of those banks fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value (see Section 4.4).

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the mechanical shim or axial offset banks at their insertion limits, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The Technical Specifications (3.4.4) require all RCPs to be operating while in Modes 1 and 2. The maximum initial core power level for the startup of an inactive loop transient is approximately zero MWt. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. There will be no increase in core power, and no automatic or manual protective action is required.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This subsection is not applicable to the AP1000.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

Comment [B4]: [15.4-4]

15.4.6.1 Identification of Causes and Accident Description

Other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Normal boron dilution with these systems is manually initiated under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup water boron concentration to that of the reactor coolant system during normal charging.

An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller, operator or mechanical failure. The chemical and volume control system and demineralized water transfer and storage system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, allowing sufficient time for automatic or operator response to terminate the dilution.

An inadvertent dilution from the demineralized water transfer and storage system through the chemical and volume control system may be terminated by isolating the makeup flow to the reactor coolant system, by isolating the makeup pump suction line to the demineralized water transfer and storage system storage tank, or by tripping the makeup pumps. Lost shutdown margin may be regained by adding borated water to the reactor coolant system from the boric acid tank.

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Generally, to dilute, the operator would need to perform two actions:

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- Switch control of the makeup from the automatic makeup mode to the dilute mode.
- Start the chemical and volume control system makeup pumps.

Failure to carry out either of those actions prevents initiation of dilution. Because the AP1000 chemical and volume control system makeup pumps do not run continuously (they are expected to be operated once per day to make up for reactor coolant system leakage), a makeup pump is started when the volume control system is placed into dilute mode.

The status of the reactor coolant system makeup is available to the operator by the following:

- Indication of the boric acid and blended flow rates

- Chemical and volume control system makeup pumps status
- Deviation alarms, if the boric acid or blended flow rates deviate by more than the specified tolerance from the preset values
- When reactor is subcritical
  - High flux at shutdown alarm
  - Indicated source range neutron flux count rate
  - Audible source range neutron flux count rate
  - Source range neutron flux-multiplication alarm
- When the reactor is critical
  - Axial flux difference alarm (reactor power ≥ 50 percent rated thermal power)
  - Control rod insertion limit low and low-low alarms
  - Overtemperature ΔT alarm (at power)
  - Overtemperature ΔT reactor trip
  - Power range neutron flux-high, both high and low setpoint reactor trips.

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This event is a Condition II incident (a fault of moderate frequency), as defined in subsection 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for critical/key parameters are used (high reactor coolant system critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual reactor coolant system volumes). These assumptions (see Table 15.4-2) result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

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In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is considered for the boron dilution case initiated from the power mode of operation (Mode 1) with the reactor in manual control. This is the analyzed Mode 1 boron dilution case that produces a reactor and turbine trip (Section 15.4.6.2.6). The loss of offsite power is assumed to occur as a direct result of a turbine trip that would disrupt the grid and produce a consequential loss of offsite ac power. As discussed in subsection 15.0.14, that scenario can occur only with the plant at power and connected to the grid. Therefore, only a boron dilution case initiated from full power will be addressed with respect to the consequential loss of offsite power.

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15.4.6.2.1 Dilution During Refueling (Mode 6)

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls, which isolate the reactor coolant system from the potential source of unborated water by locking closed specified valves in the chemical and volume control system during refueling operations. These valves block the flow paths that allow unborated makeup water to reach the reactor coolant system. Makeup which is required during refueling uses water supplied from the boric acid tank (which contains borated water).

15.4.6.2.2 Dilution During Cold Shutdown (Mode 5)

The following conditions are assumed for inadvertent boron dilution while in this operating mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 200°F and 14.7 psia.
- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region. No calculations are performed assuming that the active reactor coolant system volume is reduced to the mid-plane of the hot leg. Technical Specification 3.4.8 requires that at least one RCP be operating any time that unborated water sources are not isolated.
- Control rods are fully inserted, which is the normal condition in cold shutdown and a critical boron concentration is 1483 ppm. This is a conservative boron concentration with control rods inserted and accounts for the most reactive rod stuck in the fully withdrawn position.
- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value identified by the core operating limits report (COLR) for the cold shutdown mode. Combined with the critical boron concentration identified above, this gives an initial boron concentration of 1675 ppm.
- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that, when in Mode 5, at least one RCP shall be operating with a flow of at least 3000 gpm. This provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1

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- A Boron Dilution Protection System (BDPS) safety analysis limit (SAL) flux multiplier setpoint of 3.0 is assumed.

In the event of an inadvertent boron dilution transient during cold shutdown, the source range nuclear instrumentation detects an increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier. Upon detecting a sufficiently large flux increase, an alarm is sounded for the operator, and valves are actuated to terminate the dilution automatically.

Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. In addition, the makeup pumps are tripped for equipment protection purposes.

No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 30.75 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.3 Dilution During Safe Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 420°F and 401 psia.
- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region.
- All control rods are fully inserted, except the most reactive rod which is assumed stuck in the fully withdrawn position. The critical boron concentration is 1449 ppm.
- The shutdown margin is equal to 1.6-percent Δ k/k, the minimum value required by the core operating limits report (COLR) for the hot shutdown mode. Combined with the critical boron concentration given above, this gives an initial boron concentration of 1649 ppm.
- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that at least one reactor coolant pump shall be operating with a flow of at least 3000 gpm when in Mode 4. This provides sufficient flow through the system to

Deleted: <#>At least one reactor coolant pump will be normally operating during plant operation in Mode 5. It may be possible under some conditions, however, to operate the plant in Mode 5 with no reactor coolant pumps operating. For this reason, the mixing volume assumed for the analysis in Mode 5 will include the reactor coolant loop and normal residual heat removal syste... [8]

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maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1.

- A Boron Dilution Protection System (BDPS) Safety Analysis Limit (SAL) setpoint 3.0 is assumed.

In the event of an inadvertent boron dilution transient during safe shutdown, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier, automatically initiates valve movement to terminate the dilution, and sounds an alarm.

Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. Also, the makeup pumps are tripped for equipment protection purposes.

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No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 28.83 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

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15.4.6.2.4 Dilution During Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 557°F and 2250 psia.

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- The reactor coolant system volume is 7605.9 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running. The assumed active volume does not include the volume of the reactor vessel upper head region.

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- Critical boron concentration is 1281 ppm. This is a conservative boron concentration assuming control rods are fully inserted minus the most reactive rod, which is assumed stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value required by the core operating limits report (COLR) for the hot standby mode. Combined with the critical boron concentration given above, this gives an initial boron concentration of 1509 ppm.

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- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that, at least one reactor coolant pump shall be operating with a flow of at least 3000 gpm when in Mode 3. This provides sufficient flow through the system to maintain the system well mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1.

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In the event of an inadvertent boron dilution transient in hot standby, the source range nuclear instrumentation detects a sufficiently large increase in the neutron flux by comparing the current source range flux to that of about 50 minutes earlier, automatically initiates valve movement to terminate the dilution, and sounds an alarm. Upon the actuation of a source range flux multiplier signal, the makeup flow to the reactor coolant system and the makeup pump suction line to the demineralized water transfer and storage system storage tank are isolated. This thereby terminates the dilution. Also, the makeup pumps are tripped for equipment protection purposes.

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No operator action is required to terminate this transient. The analysis demonstrates that the flux multiplier SAL will be reached 32.07 minutes after the dilution transient begins and that there is sufficient time at this point for the automatic protective features to terminate the dilution prior to losing all shutdown margin. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.5 Dilution During Startup (Mode 2)

The plant is in the startup mode only for startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. Normal actions taken to change power level, either up or down, require operator actuation. The Technical Specifications require an available shutdown margin of 1.6-percent Δk/k and four reactor coolant pumps operating. Other conditions assumed are the following:

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 565.83°F (5% power) and 2250 psia.

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- Minimum reactor coolant system water volume is 8425.5 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.

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- The initial maximum boron concentration, corresponding to the rods inserted to the insertion limits, is 2031 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 1097 ppm, which gives a critical boron concentration of 934 ppm.

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This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control. For a normal approach to criticality, the operator manually withdraws control rods and dilutes the reactor coolant with unborated water at controlled rates until criticality is achieved. Once critical, the power escalation is slow enough to allow the operator to manually block the source range reactor trip after receiving the P-6 permissive signal from the intermediate range detectors (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

Upon any reactor trip signal, or low input voltage to the Class 1E dc and uninterruptable power supply system battery chargers, a safety-related function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

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After reactor trip, the dilution would have to continue for approximately 205 minutes to overcome the available shutdown margin.

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15.4.6.2.6 Dilution During Full Power Operation (Mode 1)

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The COLR and Technical Specifications require an available shutdown margin of 1.6-percent $\Delta k/k$ and four reactor coolant pumps operating. With the plant at power and the reactor coolant system at pressure, the dilution rate is limited by the capacity of the chemical and volume control system makeup pumps. The analysis is performed assuming two chemical and volume control system pumps are in operation, even though normal operation is with one pump. Conditions assumed for a dilution in this mode are the following:

Comment [B5]: [15.4-9]

- A dilution flow of 175 gpm of unborated water exists. The dilution flow is assumed to be at 40°F and 14.7 psia. The fluid conditions of the RCS are assumed to be 581.6°F (full power) and 2250 psia.

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- Minimum reactor coolant system water volume is 8425.5 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.

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- An initial maximum boron concentration, corresponding to the rods inserted to the insertion limits, is 1811 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 877 ppm, which gives a critical boron concentration of 934 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.

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With the reactor in automatic rod control, the pressurizer level controller limits the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller throttles charging flow down to match the letdown rate. For the safety analysis, a conservative dilution flow rate of 175 gpm is assumed. With the reactor in automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. This action by the controller results in at least three alarms to the operator:

Deleted: <#>With the reactor in manual rod control, an initial maximum boron concentration, corresponding to the rods out, is 2150 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 1216 ppm, which gives a critical boron concentration of 934 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.¶

A. Rod insertion limit- low level alarm

B. Rod insertion limit- low-low level alarm if insertion continues

C. Axial flux difference alarm (ΔI outside of the target band)

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. The operator has at least 170.6 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at the beginning of the cycle. The time is significantly longer at the end of the cycle because of the lower initial and critical boron concentrations.

Because the analysis for the boron dilution event with the reactor in automatic rod control does not predict a reactor and turbine trip, considering the consequential loss of offsite power for this case is not needed.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature would rise and cause the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. Upon any reactor trip signal, a safety-related function automatically isolates the unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

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The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdrawal at power (see Section 15.4.2). The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be approximately 0.6 pcm/s and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Before reaching the overtemperature ΔT reactor trip, the operator receives an alarm overtemperature ΔT and an overtemperature ΔT turbine runback.

Should a consequential loss of offsite power occur after reactor and turbine trip, it does not alter the fact that the dilution event has been terminated by automatic protection features. As indicated previously, the reactor trip signal that occurs in parallel with the turbine trip will actuate a safety-related function that automatically isolates the unborated water from the demineralized water system and thereby terminates the dilution. A subsequent loss of offsite power will cause the chemical and volume control system pumps to shut down.

After reactor trip, the automatic termination of the dilution flow from the dimineralized water transfer and storage system precludes a post-trip return to criticality.

15.4.6.3 Conclusions

Inadvertent boron dilution events are administratively prevented by the Technical Specifications (3.9.2) during refueling (Mode 6) and automatically terminated during cold shutdown (Mode 5), safe shutdown (Mode 4), and hot standby (Mode 3) modes. Inadvertent boron dilution events during startup (Mode 2) or power operation (Mode 1), if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality is prevented.

The preceding results demonstrate that in all modes of operation, an inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors can inadvertently occur, such as those arising from the inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly with pellets of the wrong enrichment. This leads to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core-loading errors is the

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Deleted: Because the realignment of the suction for the chemical and volume control system pumps to the boric acid tank is a non-safetynonsafety-related operation, the only consideration given to the reboration phase of the event in the safety analysis is the unborated purge volume.¶
After reactor trip, the dilution would have to continue for at least 325 minutes to overcome the available shutdown margin. The ... [14]

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inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

An error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichments. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing fuel assembly enrichment errors or loading errors that cause power shapes to be peaked in excess of the design value. Power-distribution-related measurements are incorporated into the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distributions measurements in Westinghouse pressurized water reactors.

Each fuel assembly is marked with an identification number and loaded in accordance with a core-loading diagram to reduce the probability of core loading errors. During core loading, the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed.

The power distortion due to a combination of misplaced fuel assemblies could significantly increase peaking factors and is readily observable with the online core monitoring system. The fixed incore instrumentation within the instrumented fuel assembly locations is augmented with core exit thermocouples. There is a high probability that these thermocouples would also indicate any abnormally high coolant temperature rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is a Condition III incident (an infrequent fault) as defined in subsection 15.0.1.

15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady-state power distributions in the x-y plane of the core are calculated at 30-percent rated thermal power using the three-dimensional nodal code ANC (Reference 7). Representative power distributions in the x-y plane for a correctly loaded core are described in Chapter 4.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown in the incore detector locations. (See Figures 15.4.7-1 through 15.4.7-4.)

15.4.7.2.2 Results

The following core loading error cases are analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered is the interchange of two assemblies near the periphery of the core (see Figure 15.4.7-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. For the particular case considered, the interchange is assumed to take place close to the core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position (see Figure 15.4.7-2).

Case C:

Enrichment error – Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4.7-3).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4.7-4).

15.4.7.3 Conclusions

Fuel assembly enrichment errors are prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

Comment [B5]: [15.4-5]

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

15.4.8.1.1.1 Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to prevent the possibility of an RCCA drive mechanism housing failure are listed below:

- Each control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.
- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head. The housings are checked during the hydrotest of the completed reactor coolant system.
- Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the safe shutdown earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.
- The latch mechanism housing and rod travel housing are each a single length of forged stainless steel. This material exhibits excellent notch toughness at temperatures that are encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional confidence that gross failure of the housing does not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds, which are subject to periodic inspections.

15.4.8.1.1.2 Nuclear Design

If a rupture of an RCCA drive mechanism housing is postulated, the operation using chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the power control (or mechanical shim) RCCAs inserted only far enough to permit load follow. The axial offset RCCAs are positioned so that the targeted axial offset can be met throughout core life. Reactivity changes caused by core depletion and xenon transients are normally compensated for by boron changes and the mechanical shim banks, respectively. Further, the location and grouping of the power control and axial offset RCCAs are selected with consideration for an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full-power operation, a less severe reactivity excursion than analyzed is expected.

It may occasionally be desirable to operate with larger than normal insertions. For this reason, a power control and axial offset rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit provides adequate shutdown capability and an acceptable power distribution. The position of the RCCAs is continuously indicated in the main control room. An alarm occurs if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low level alarm and emergency boration at the low-low level alarm.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in WCAP-15806-P-A (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

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15.4.8.1.1.4 Effects on Adjacent Housings

Failures of an RCCA mechanism housing, due to either longitudinal or circumferential cracking, does not cause damage to adjacent housings. The control rod drive mechanism is described in subsection 3.9.4.1.1.

15.4.8.1.1.5 Not Used

15.4.8.1.1.6 Not Used

15.4.8.1.1.7 Consequences

The probability of damage to an adjacent housing is considered remote. If damage is postulated, it is not expected to lead to a more severe transient because RCCAs are inserted in the core in symmetric patterns and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal. This is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.1.8 Summary

Failure of a control rod housing does not cause damage to adjacent housings that increase the severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

NUREG-0800 Standard Review Plan (SRP) 4.2 Revision 3 (Reference 24) interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2 Revision 3 Appendix B.
- The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.
- For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g. DNBR).
- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 Btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.

Deleted: Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project (Reference 5). Extensive tests of uranium dioxide (UO_2) zirconium-clad fuel rods representative of those in pressurized water reactor cores such as AP1000 have demonstrated failure thresholds in the range of 240 to 257 cal/g. Other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT (Reference 6) results, which indicated a failure threshold of 280 cal/g. Limited results indicate that this threshold decreases by about 10 percent with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods.¶ Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/g for unirradiated rods and 200 cal/g for irradiated rods. Catastrophic failure (large fuel dispersal, large pressure ri... [18]

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- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core calculation and then, a hot rod calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, fuel temperature and DNB transients are then determined by performing a conservative fuel rod transient heat transfer calculation.

A discussion of the method of analysis appears in WCAP-15806-P-A (Reference 4).

Average Core Analysis

The three-dimensional nodal code ANC (References 14, 15, 16, 17, 21, 22 and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in 3 spatial dimensions (rectangular coordinates) for 6 delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE-01 code and methods (References 18 and 19).

Hot Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature and DNBR using as input the time-dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18.

Deleted: <#>Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of the first criterion.¶

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Deleted: multiplying the average core energy generation by the hot channel factor and

Deleted: The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

Deleted: 7588, Revision 1A

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System Overpressure Analysis

If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure. This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected as described in Reference 4.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three-dimensional methods. Standard nuclear design codes are used in the analysis. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

15.4.8.2.1.2 Not Used

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentration, is adjusted in the nuclear code to obtain a moderator temperature coefficient that is conservative compared to actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements.

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.50 percent at the end of cycle. The accident is sensitive to β_{eff} if the ejected rod worth is equal

Deleted: In the hot spot analysis, the initial heat flux is equal to the nominal value multiplied by the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. The assumption is made that the hot spots before and after ejection are coincident. This is conservative because the peak after eje... [20]

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to or greater than β_{eff}. To allow for future cycles, a pessimistic estimate of β_{eff} of 0.44 percent is used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion accounts for the effect of the ejected rod and one adjacent stuck rod. The trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.583 second for the instrument channel to produce a signal, 0.167 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.7 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

The minimum design shutdown margin available at hot zero power may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Calculations show that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1-percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor is subcritical when the core returns to hot zero power.

15.4.8.2.1.6 Reactor Protection

As discussed in subsection 15.4.8.1.1.3, reactor protection for a rod ejection is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are part of the protection and safety monitoring system. No single failure of the protection and safety monitoring system negates the protection functions required for the rod ejection accident or adversely affects the consequences of the accident.

15.4.8.2.1.7 Results

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases for a typical cycle are summarized following the criteria outlined in Section 15.4.8.1.2.

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Deleted: 0.49 percent at beginning of cycle and

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Deleted: Because the control rod insertion limits for the AP1000 are multidimensional, a significant number of rodded configurations are evaluated to determine the most limiting cases, (that is, those cases that produced the least amount of margin to the Standard Review Plan Section 15.4.8 evaluation acceptance criteria). The hot zero power

Deleted: and hot full power cases

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- Pellet-Clad Mechanical Interaction (PCMI) and High Clad Temperature (Hot Zero Power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in Section 15.4.8.1.2.

- High Clad Temperature (≥ 5% Rated Thermal Power)

  The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in Section 15.4.8.3.

- Core Coolability

  The resulting maximum fuel average enthalpy is less than the criterion given in Section 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

The nuclear power transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-3.

The calculated sequence of events for the limiting cases are presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed

Deleted: <#>Beginning of cycle, full power¶
The limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.37-percent Δk and 4.9, respectively. The peak hot spot cladding average temperature is 22652290°F. (1254°C). The peak hot spot fuel center temperature reaches melting at 4900°F. (2704.44°C). However, melting is restricted to less than 10 percent of the pellet at the hot spot.¶
<#>Beginning of cycle, zero power¶
For this condition, the limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.65-percent Δk and 12.0, respectively. The peak hot spot cladding average temperature is 19071587°F, (864°C), and the peak hot spot fuel center temperature is 3018 3104°F. (1707°C).¶
<#>End of cycle, full power¶
The ejected rod worth and hot channel factor are conservatively assumed to be 0.30-percent Δk and 6.0, respectively. The peak hot spot cladding average tem... [33]

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Deleted: rod ejection accidents, as shown in Figures 15.4.8-1 through 15.4.8-4, is three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.

Deleted: THINC analysis (Reference 4). Although limited (less than 10 percent) fuel melting at the hot spot is allowed for the full-power cases, in practice, melting is not expected because the analysis conservatively assumes that the hot spots before and after ejection are coincident.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the calculation of the number of rods assumed to enter DNB for the RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.9 Peak RCS Pressure

Deleted: Surge

Calculations of the peak reactor coolant system pressure demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the pressure surge transient resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak system pressure occurs before the reactor coolant pumps begin to coast down.

Deleted: A calculation of the pressure surge for an ejection worth of about one dollar at beginning of cycle, hot full power, demonstrates that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Because the severity of the analysis does not exceed the worst-case analysis, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.¶

15.4.8.2.1.10 Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Because the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion, tending to bow the midpoint of the rods toward the hotter side of the rod.

Calculations indicate that this bowing results in a negative reactivity effect at the hot spot because the core is undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, no significant bowing is anticipated because the structural rigidity of the core is sufficient to withstand the forces produced.

Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow is sufficient to produce lattice deformation. Even if massive and rapid boiling, sufficient to distort the lattices, is hypothetically postulated, the large void fraction in the hot spot region produces a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect is therefore a negative feedback.

In conclusion, no credible mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. Refer to section 15.4.8.3.1 and Table 15.4-4.

As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of Draft Guide 1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). Draft Guide 1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was issued as a clarification to the gap fraction guidance in Draft Guide 1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see subsection 15.4.8.2.1.8). Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the

Comment [B6]: [15.4-6]

Deleted: The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with the design basis fuel defect level (0.25 percent of power produced bya limited number of fuel rods containing cladding defects) and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. Refer to section 15.4.8.3.1 and Table 15.4-4.¶
As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. In addition, a small fraction of fuel is assumed to melt and release core inventory to the reactor coolant.¶
Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.¶

lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

Deleted: The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design fuel defect level. of 0.25%. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.¶
Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 10 percent of the inventory for iodine and noble gases and 12 percent for alkali metals. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor.¶
Even though no fuel centerline melting is expected, a conservative upper limit for fuel melting was determined to be 0.25 per... [34]

15.4.8.3.2 Release Pathways

There are three components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.
- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant and, as steam is released, a portion of the iodine and alkali metal in the coolant is released. The fraction of activity released is defined by the assumed flashing fraction and the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.
- The activity from the reactor coolant system and the core is released to the containment atmosphere and is available for leakage to the environment through the assumed design basis containment leakage.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.4.8.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.4.8.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.4-4.

15.4.8.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.
- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4); whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).
- The leakage of reactor coolant into the secondary system, at 300 gallons per day, is conservative. The leakage is normally a small fraction of this.
- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.
- The leakage from containment is assumed to continue for a full 30 days. It is expected that containment pressure is reduced to the point that leakage is negligible before this time.

Deleted: <#>The reactor coolant activities are based on an assumed fuel defect level of 0.25 percent;conservative assumptions (refer to Table 15.4-4); whereas, the activities based on the expected fuel defect level isare far less than this (see Section 11.1).¶

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than the value reported above.

Deleted: Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be less than 1.8 rem at the site boundary for the limiting 2--hour interval (0 to 2 hours) and less than 2.5 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.¶

15.4.9 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.4.10 References

Comment [B7]: [15.4-7]

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Table 15.4-1 (Sheet 1 of 3) **TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES**		
Accident	**Event**	**Time (seconds)**
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux (low setting) setpoint reached	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	11.3
	Peak heat flux occurs	12.9
	Minimum DNBR occurs	12.9
	Peak average clad temperature occurs	13.5
	Peak average fuel temperature occurs	13.7
One or more dropped RCCAs	Rods drop	0.0
	Control system initiates control bank withdrawal	0.4
	Peak nuclear power occurs	21.7
	Peak core heat flux occurs	24.2
Uncontrolled RCCA bank withdrawal at power		
1. Case A - Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a fast reactivity insertion rate (80 pcm/s)	0.0
	Power range high neutron flux high trip point reached	6.2
	Rods begin to fall into core	7.1
	Minimum DNBR occurs	7.4
2. Case B - Full power with maximum reactivity feedback	Initiation of uncontrolled RCCA withdrawal at a slow reactivity insertion rate (5 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	568.3
	Rods begin to fall into core	570.3
	Minimum DNBR occurs	570.4

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Table 15.4-1 (Sheet 2 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (minutes)
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant		
1. Dilution during power operation (Mode 1)		
a. Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost	170.6
b. Manual reactor control	Dilution initiated	0.0
	Reactor trip on overtemperature ΔT due to dilution	3.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	3.5
2. Dilution during startup (Mode 2)	Power range high neutron flux-low setpoint reactor trip due to dilution	0.0
	Shutdown margin lost	205.3
3. Dilution during hot standby (Mode 3)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	32.1
	Shutdown margin lost	39.6
4. Dilution during safe shutdown (Mode 4)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	28.8
	Shutdown margin lost	35.6
5. Dilution during cold shutdown (Mode 5)	Dilution initiated	0.0
	Boron dilution protection system setpoint reached, which initiates isolation of the dilution source	30.8
	Shutdown margin lost	38.1

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Table 15.4-1 (Sheet 3 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
RCCA ejection accident		
1. PCMI Limiting Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.14
	Reactor trip setpoint reached	< 0.30
	Peak cladding temperature occurs	0.36
	Peak enthalpy deposition occurs	0.44
	Rods begin to fall into core	1.20
2. Peak Clad Temperature Limiting Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.08
	Minimum DNBR occurs	0.11
	Peak cladding temperature occurs	0.11
	Reactor trip setpoint reached	< 0.30
	Rods begin to fall into core	1.20
3. Peak enthalpy / Peak Fuel Centerline Temperature Event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.06
	Reactor trip setpoint reached	< 0.30
	Rods begin to fall into core	1.20
	Peak fuel center temperature occurs	2.50
	Peak cladding temperature occurs	2.80

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Table 15.4-2 KEY INPUT PARAMETERS FOR BORON DILUTION		
Dilution Flow Rates		
Mode	**Flow Rate (gal/min)**	**Flow Rate (m³/hr)**
1 through 5	175	39.75
Active RCS Volume		
Mode	**Volume (ft³)**	**Volume (m³)**
1 and 2	8425.5	(238.584)
3,4 and 5	7605.98	(215.375)
Boron Concentration		
Mode	**Initial concentration (ppm)**	**Critical Concentration (ppm)**
1	1811	934
2	2031	934
3	1509	1281
4	1649	1449
5	1675	1483

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Table 15.4-3 Not Used.

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PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT ... [49]

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1. HZP – Hot zero power¶
2. HFP – Hot full power¶
3. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.¶

Table 15.4-4 (Sheet 1 of 2) PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT	
Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in damaged fuel)	1.75
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fuel Enthalpy Increase (cal/gm)	60
– Fission product gap fractions	
Iodine 131	0.1238
Iodine 132	0.1338
Krypton 85	0.5120
Other Nobles Gases	0.1238
Other Halogens	0.0938
Alkali Metals	0.6860
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A
Reactor coolant mass (lb)	3.7 E+05 (1.68E+05 kg)

Note:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Comment [B8]: [15.4-8]

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Table 15.4-4 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.5[a] (47.4 kg/hr)
– Leak flashing fraction	0.04[b]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.003
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.7[c]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	3.1

Notes:

a. Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).

b. No credit for iodine partitioning is taken for flashed leakage.

c. From Appendix 15B.

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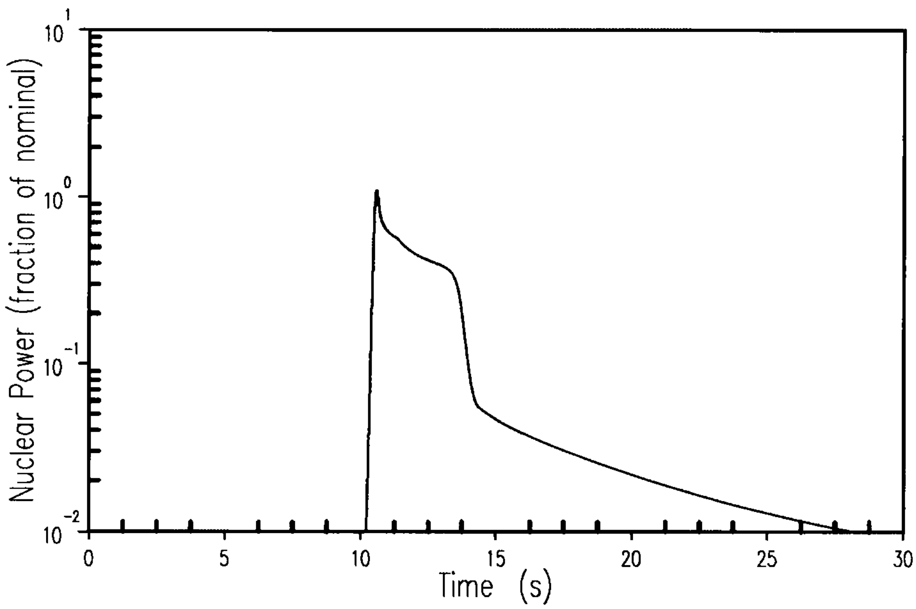

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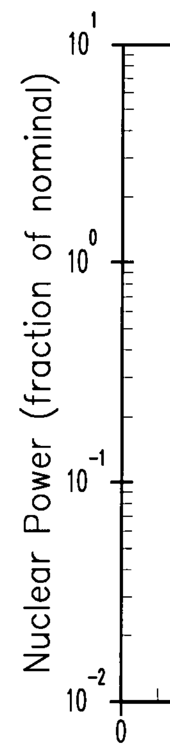
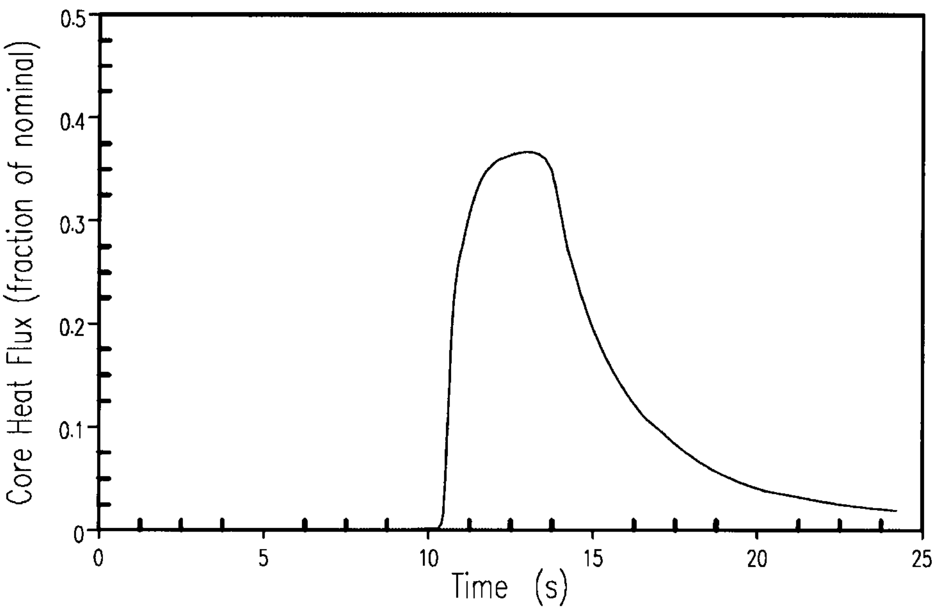


Figure 15.4.1-1

RCCA Withdrawal from Subcritical Nuclear Power



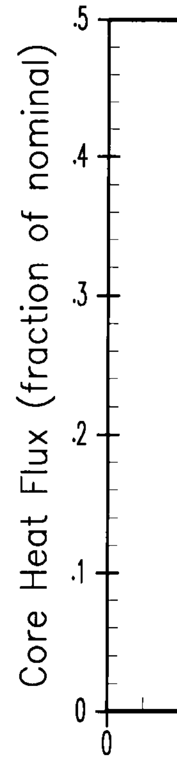


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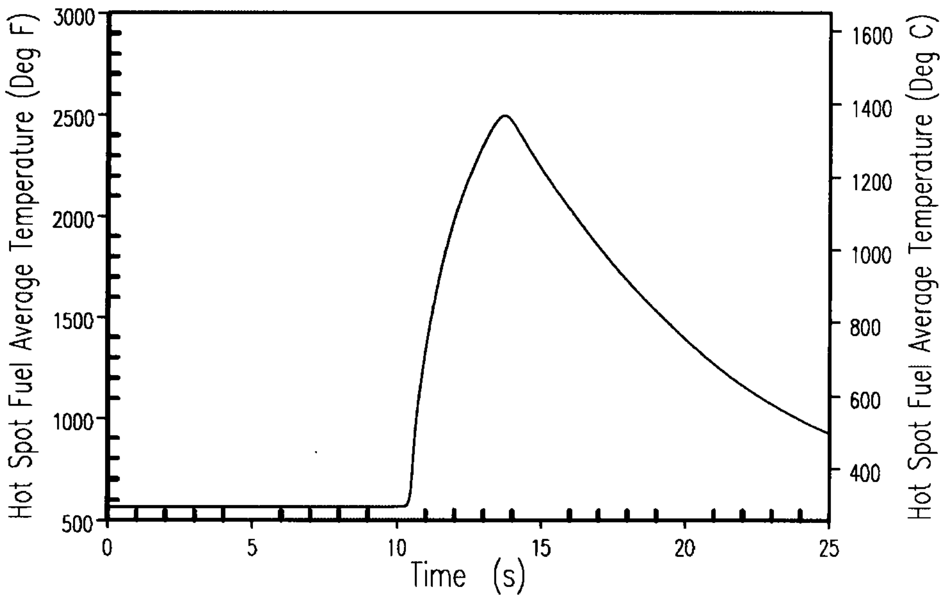
Figure 15.4.1-2

RCCA Withdrawal from Subcritical
Average Channel Core Heat Flux



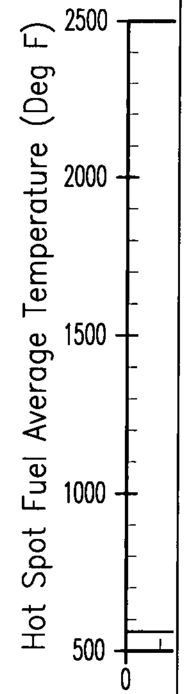


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Figure 15.4.1-3

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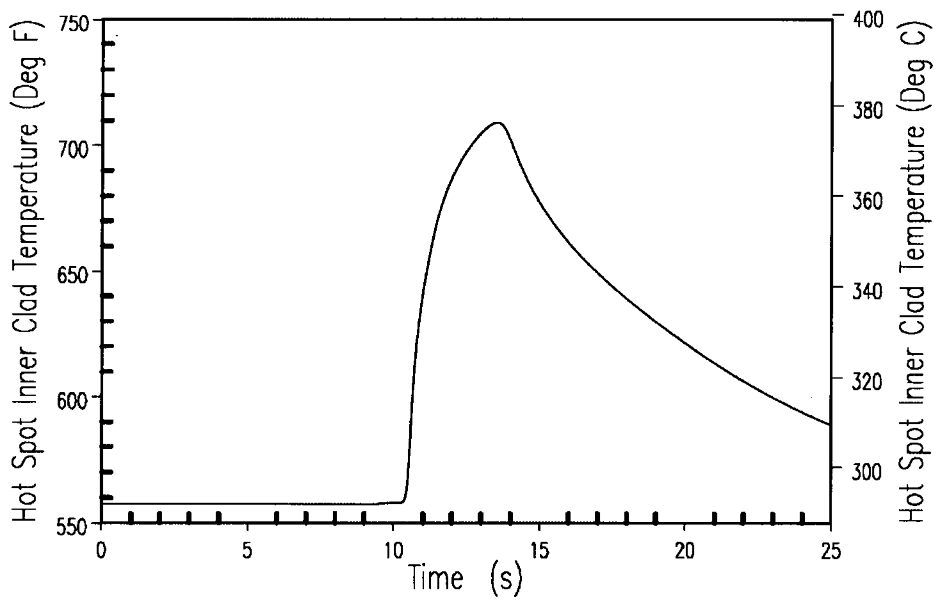
RCCA Withdrawal from Subcritical
Hot Spot Fuel Average Temperature



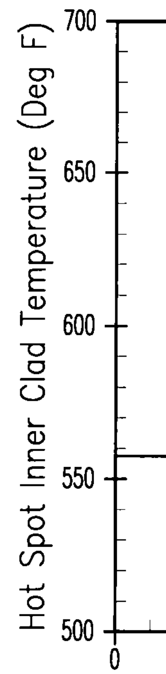


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Figure 15.4.1-4

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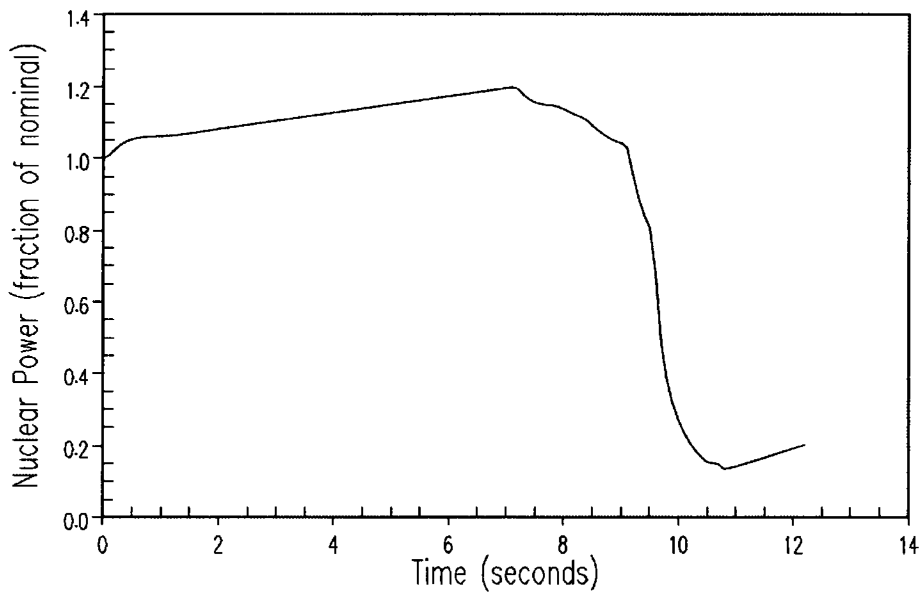
RCCA Withdrawal from Subcritical
Hot Spot Cladding Inner Temperature



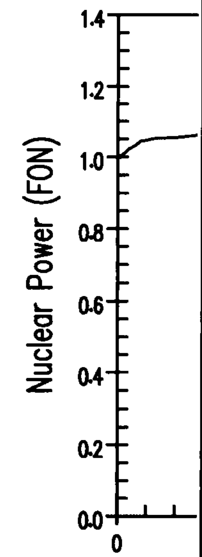


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Figure 15.4.2-1

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (80 pcm/s)

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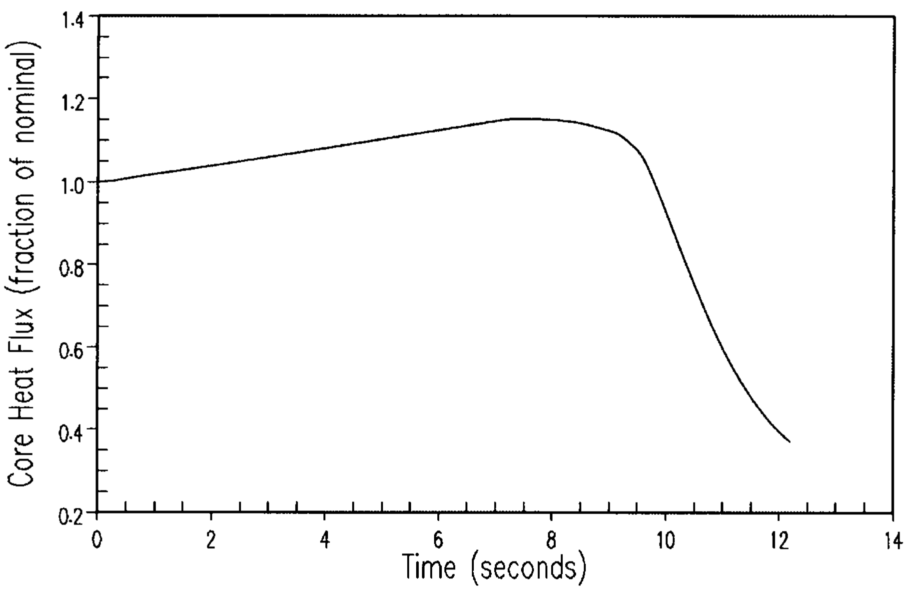

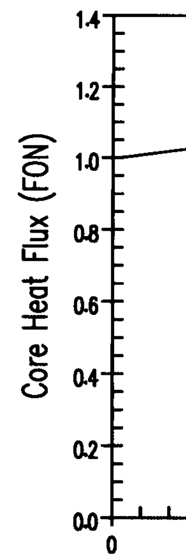

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Figure 15.4.2-2

Core Heat Flux Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
with Maximum Reactivity Feedback (80 pcm/s)

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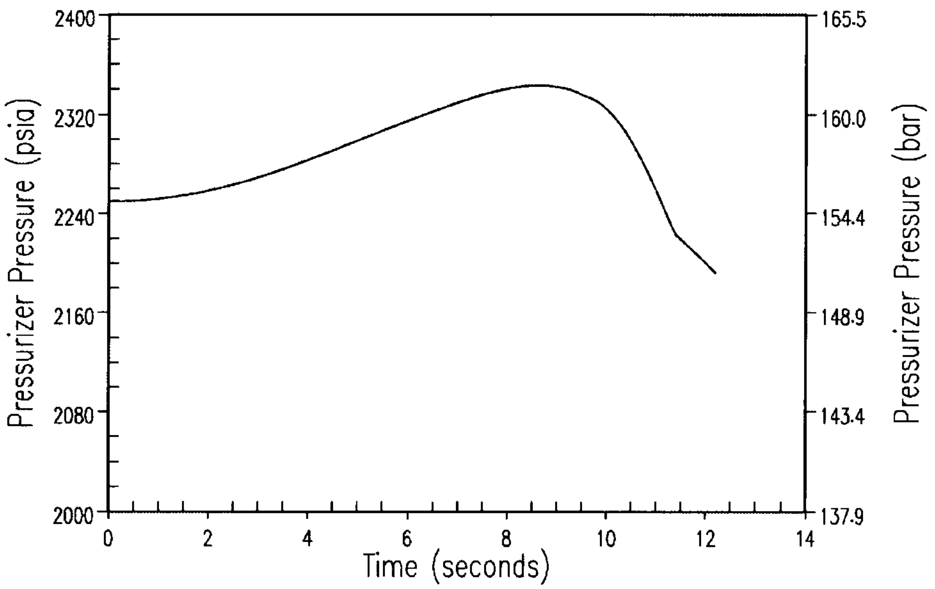

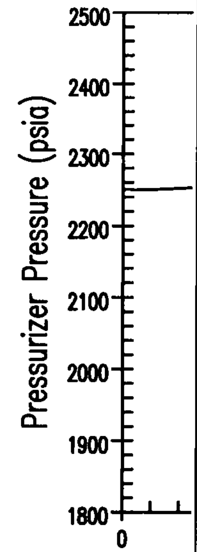

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Figure 15.4.2-3

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (80 pcm/s)

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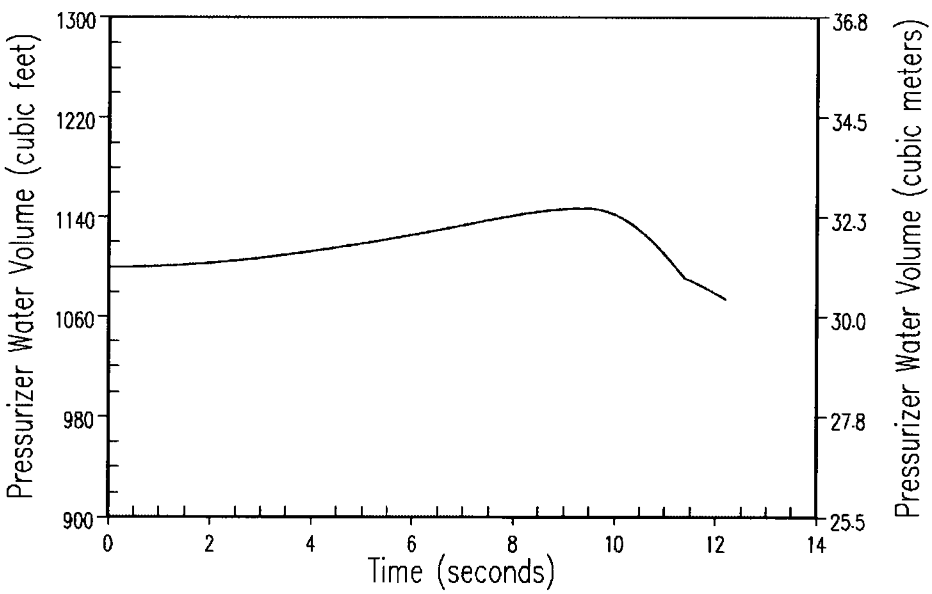

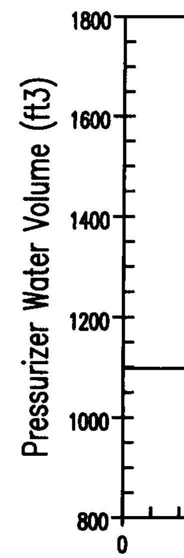

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Figure 15.4.2-4

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (80 pcm/s)

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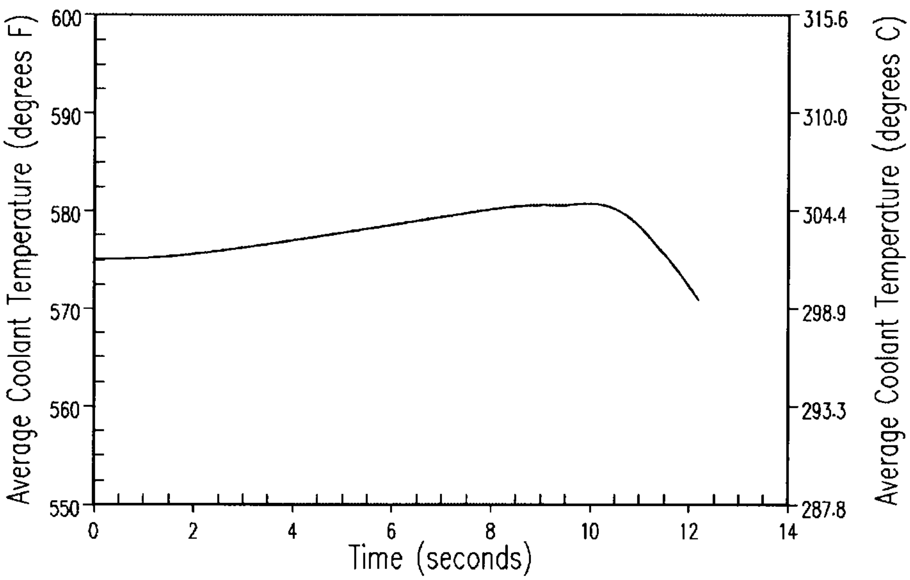

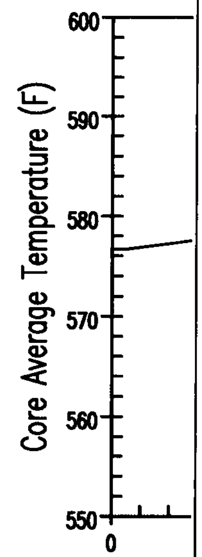

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Figure 15.4.2-5

Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (80 pcm/s)

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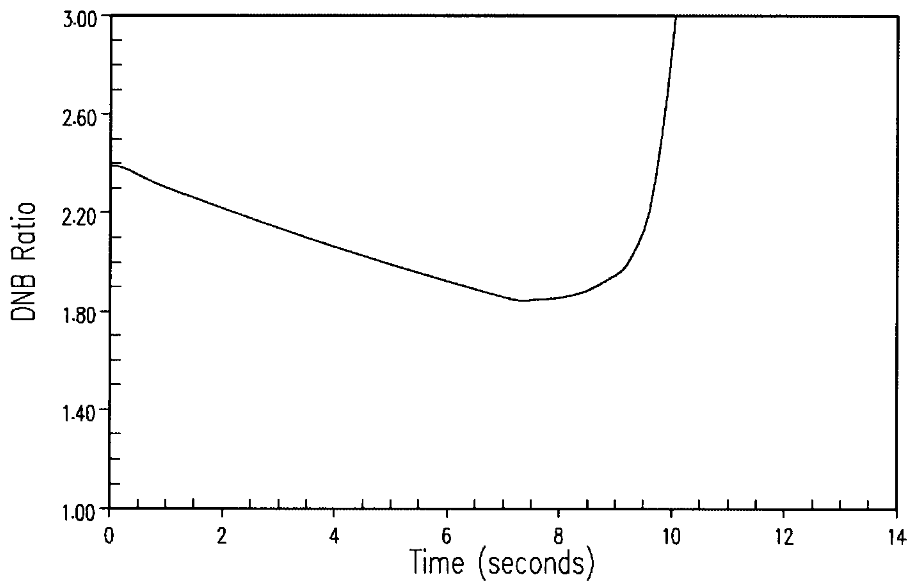

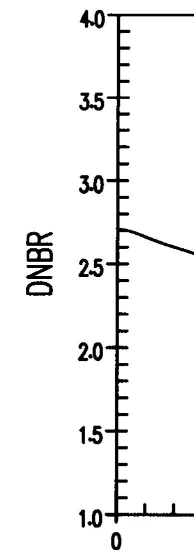

Figure 15.4.2-6

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DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
with Maximum Reactivity Feedback (80 pcm/s)

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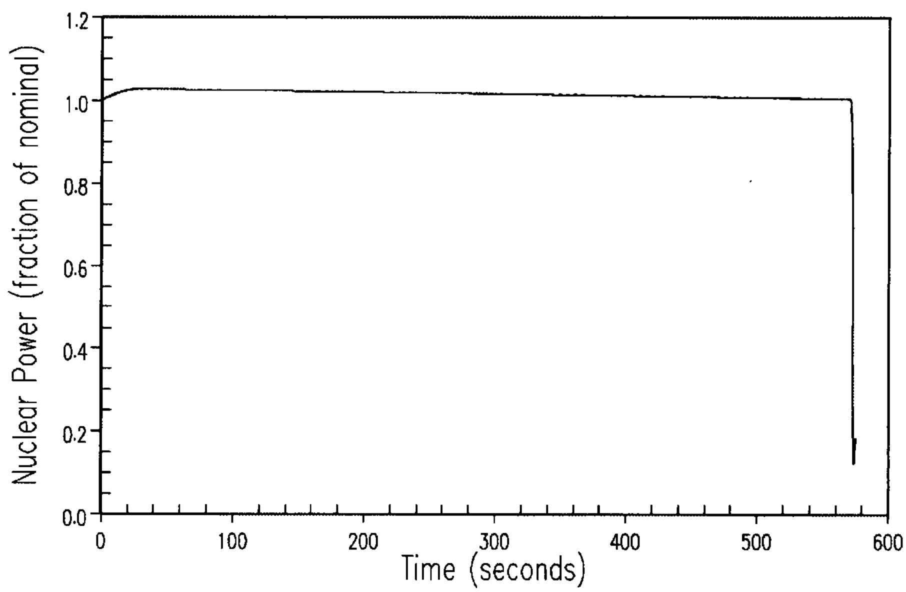

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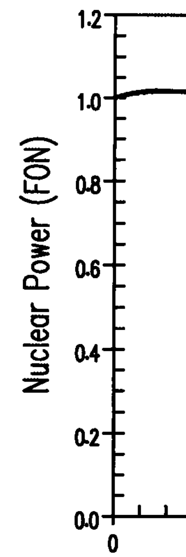

Figure 15.4.2-7

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (5 pcm/s)

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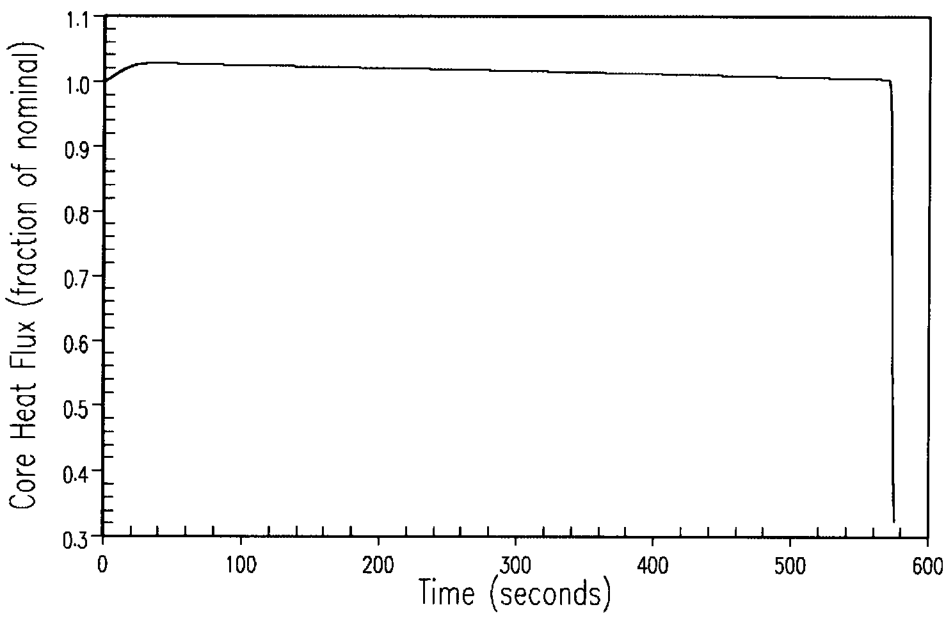

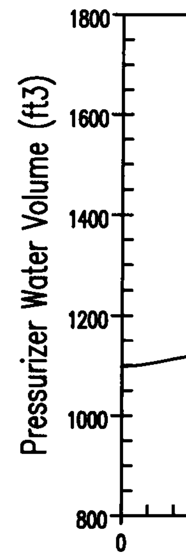

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Figure 15.4.2-8

Core Heat Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
with Maximum Reactivity Feedback (5 pcm/s)

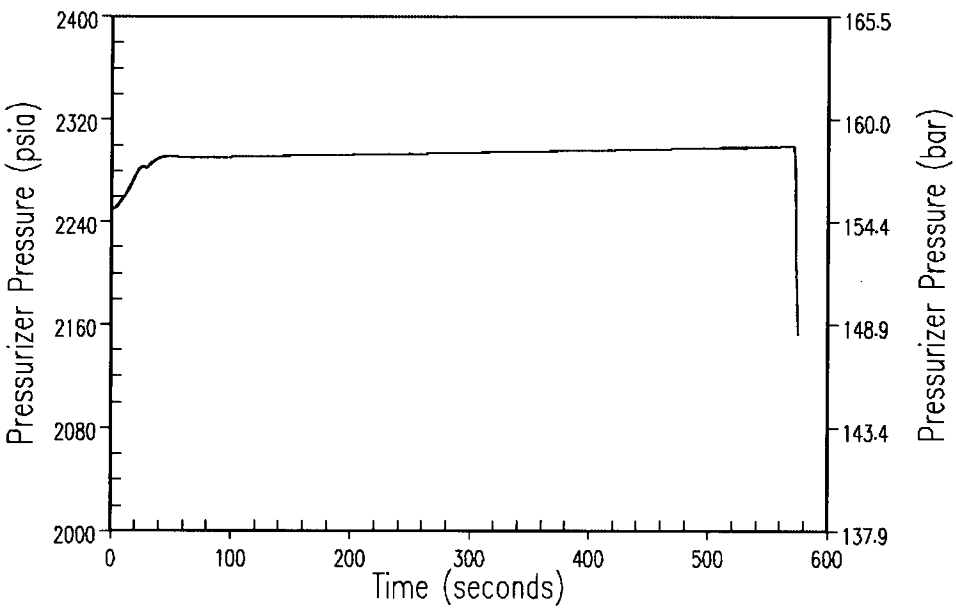

Figure 15.4.2-9

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (5 pcm/s)

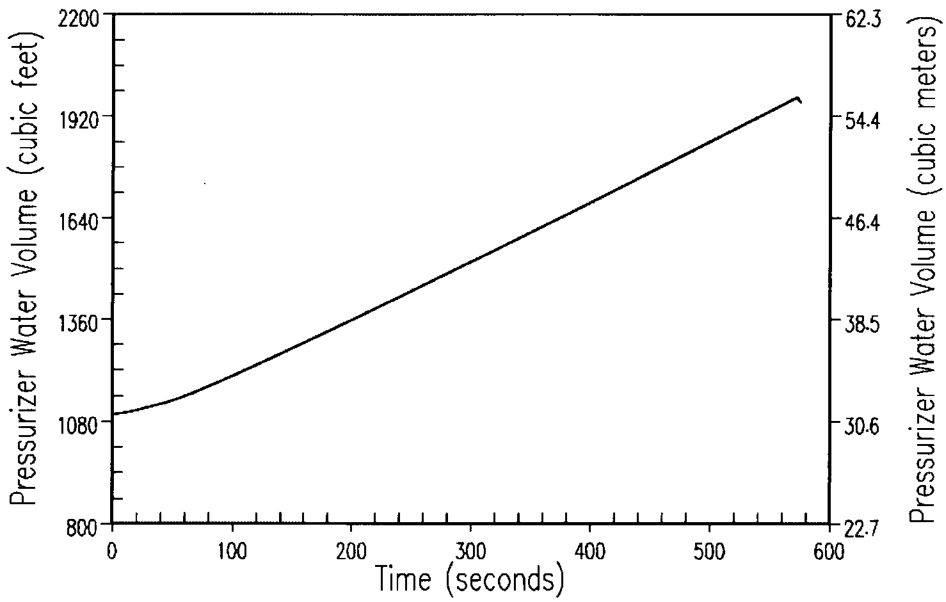

Figure 15.4.2-10

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (5 pcm/s)

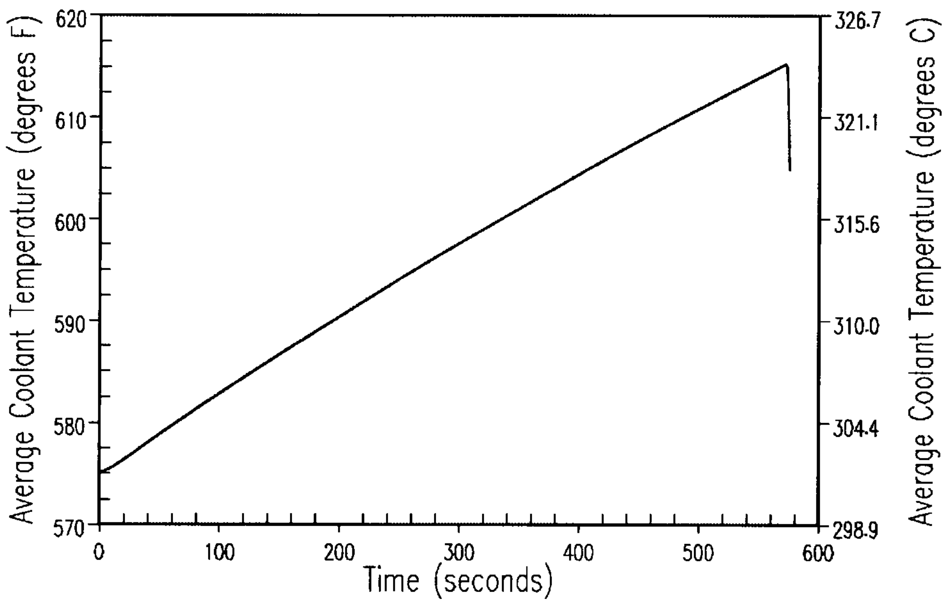

Figure 15.4.2-11

Core Coolant Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Maximum Reactivity Feedback (5 pcm/s)

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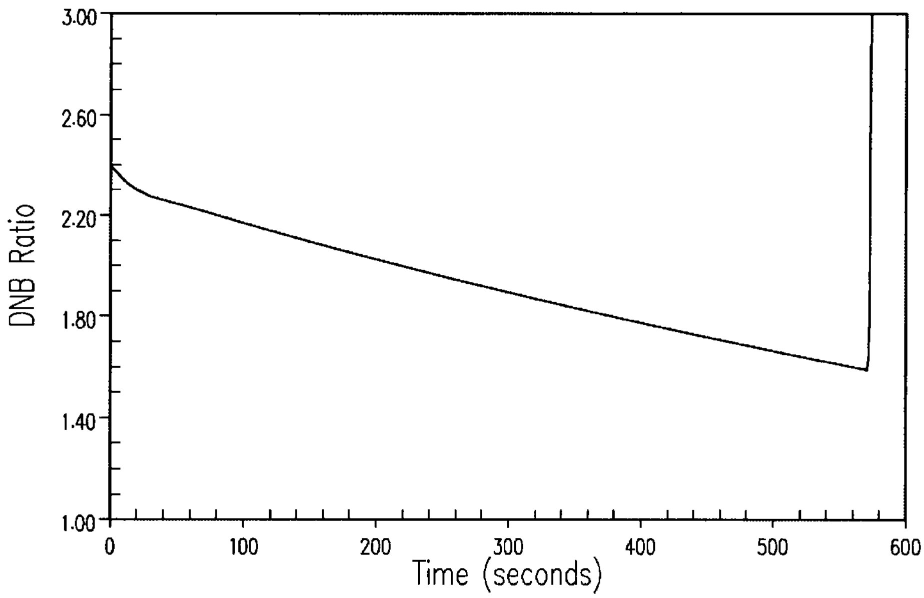

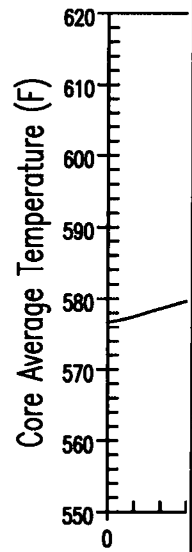

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DNBR Transient for an
Uncontrolled RCCA Bank Withdrawal from Full Power
with Maximum Reactivity Feedback (5 pcm/s)

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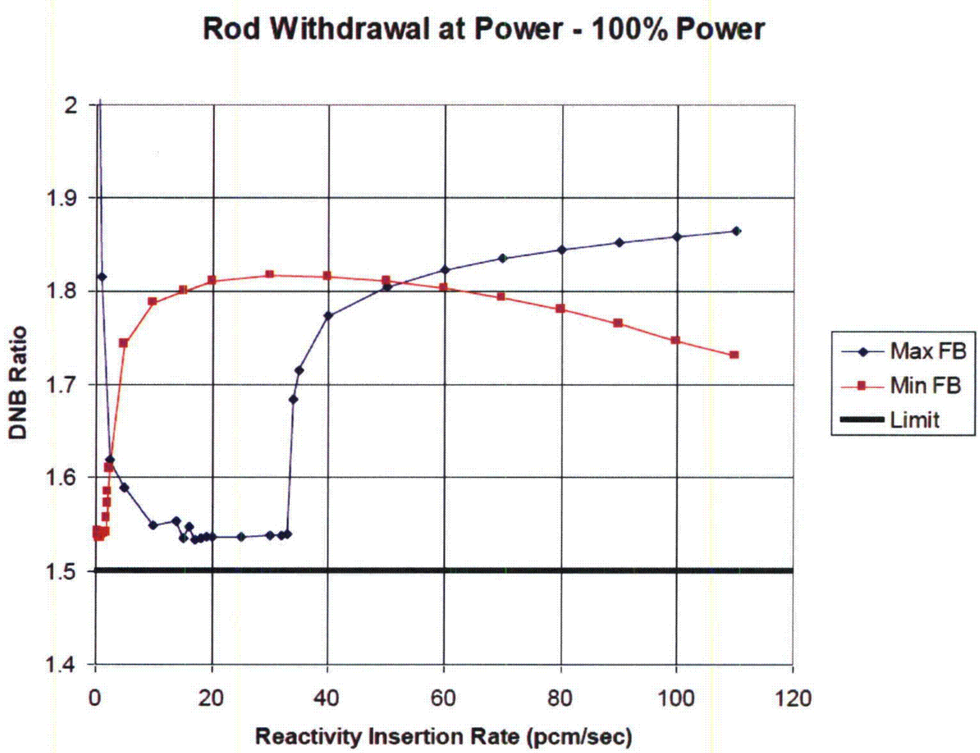

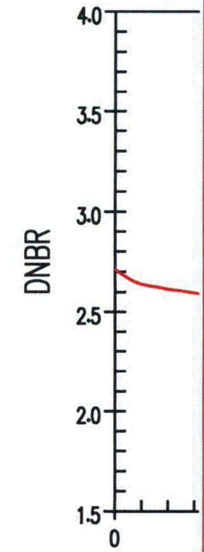

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Figure 15.4.2-13

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-percent Power

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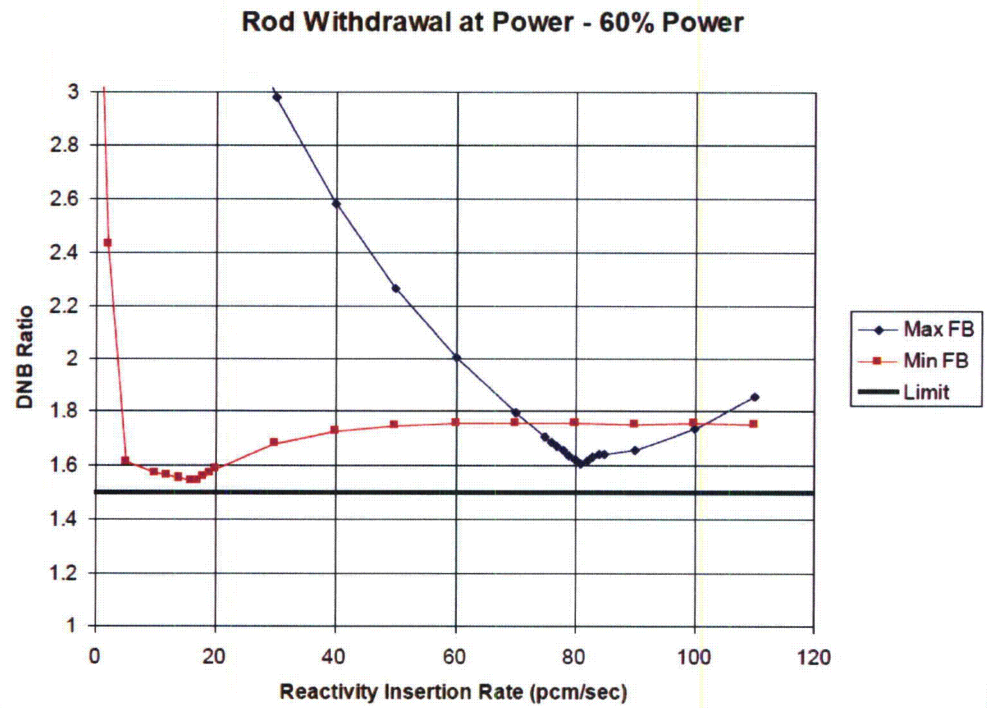
DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s)¶ ... [53]



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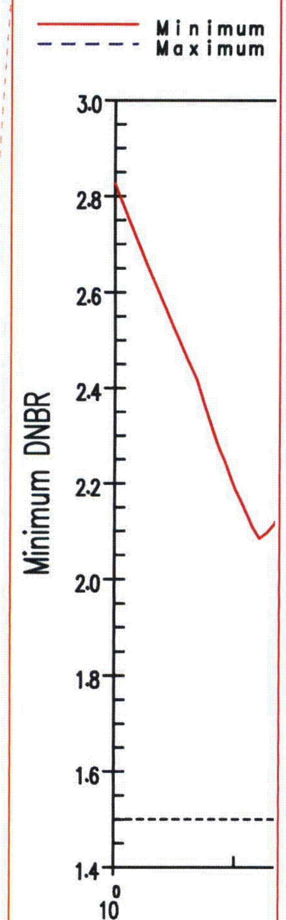
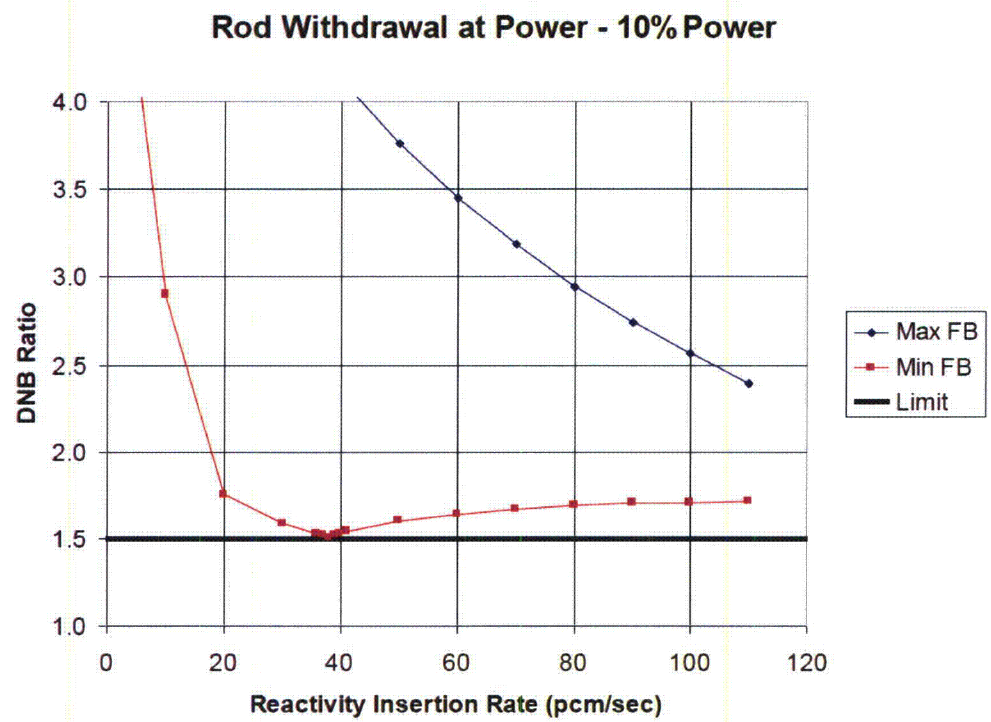


Figure 15.4.2-14

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-percent Power



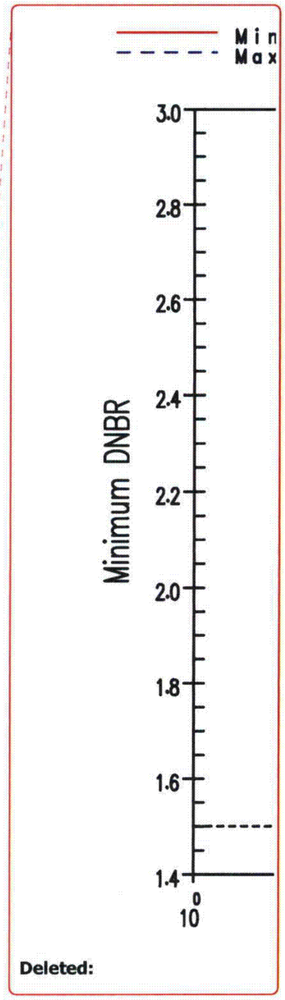


Figure 15.4.2-15

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-percent Power

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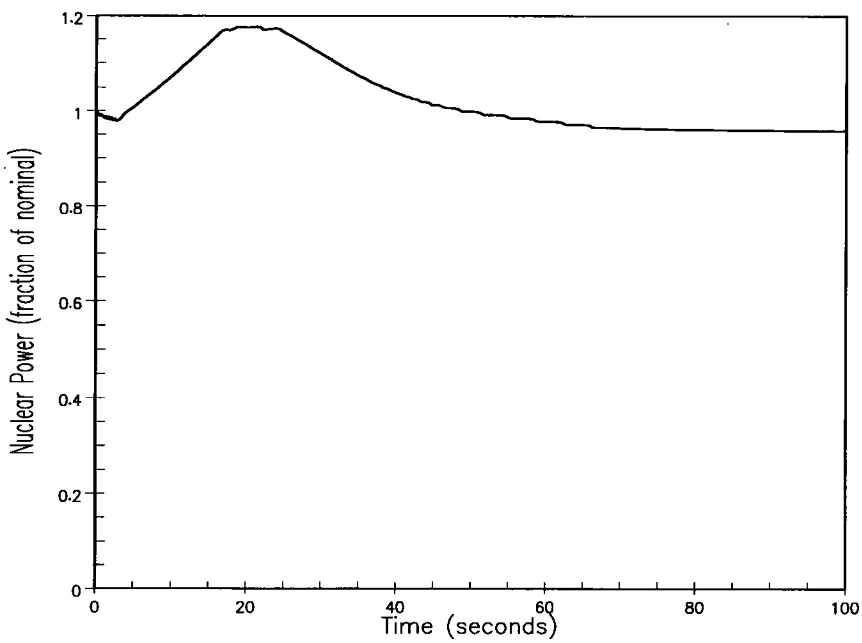

Figure 15.4.3-1

Nuclear Power Transient for Dropped RCCA

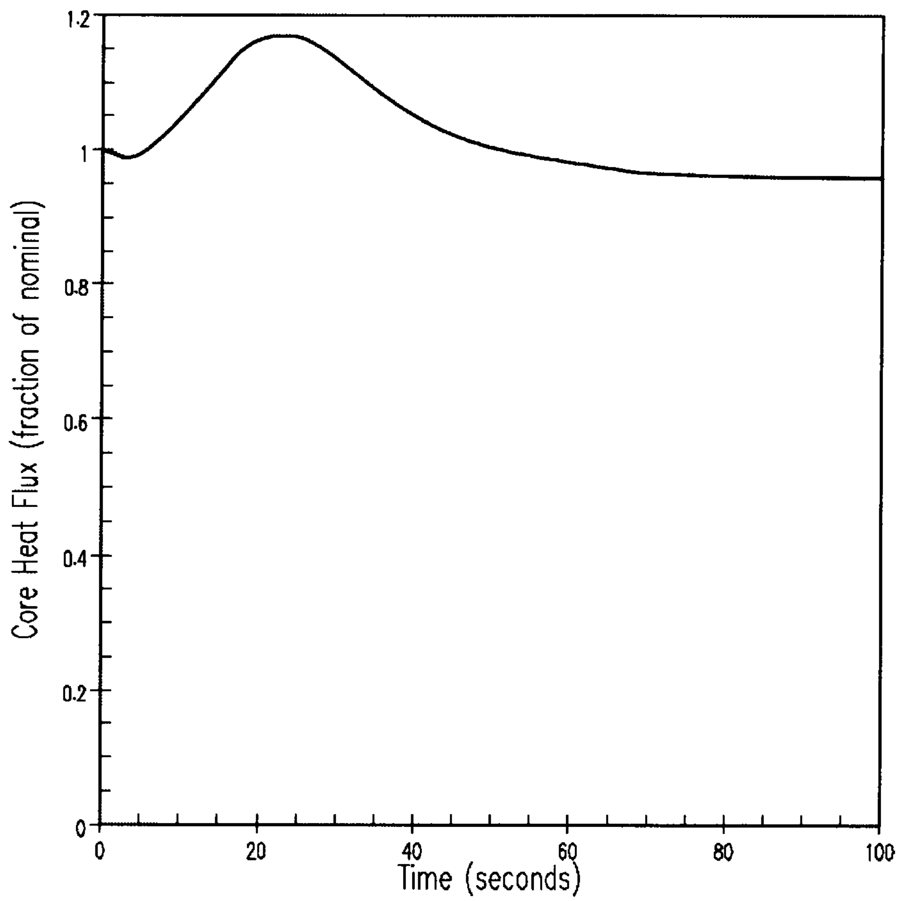

Figure 15.4.3-2

Core Heat Flux Transient for Dropped RCCA

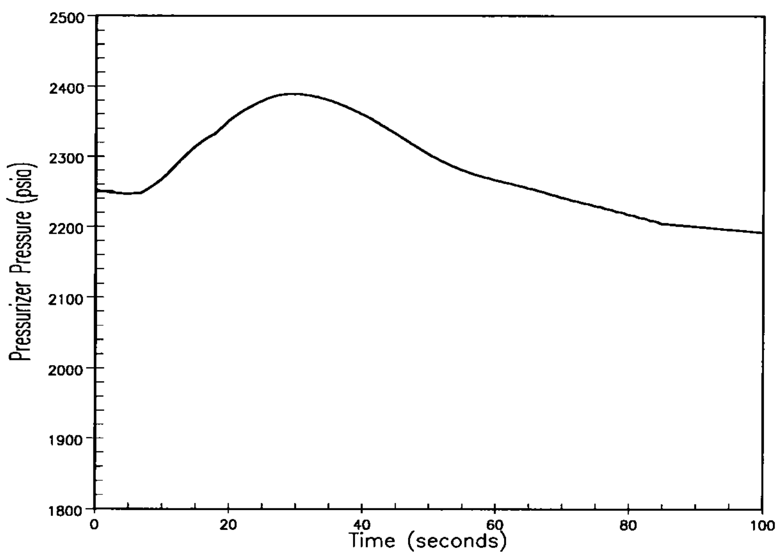

Figure 15.4.3-3

Pressurizer Pressure Transient for Dropped RCCA

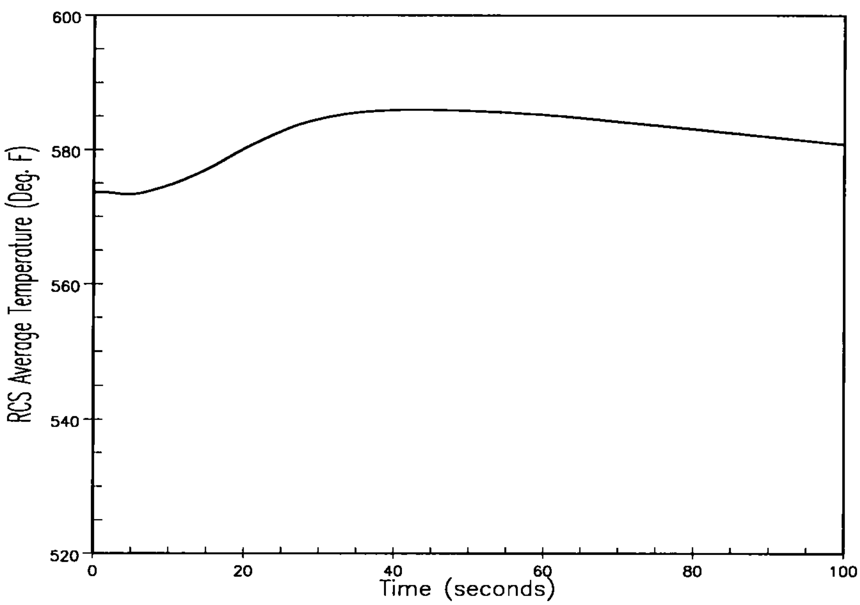

Figure 15.4.3-4

RCS Average Temperature Transient for Dropped RCCA

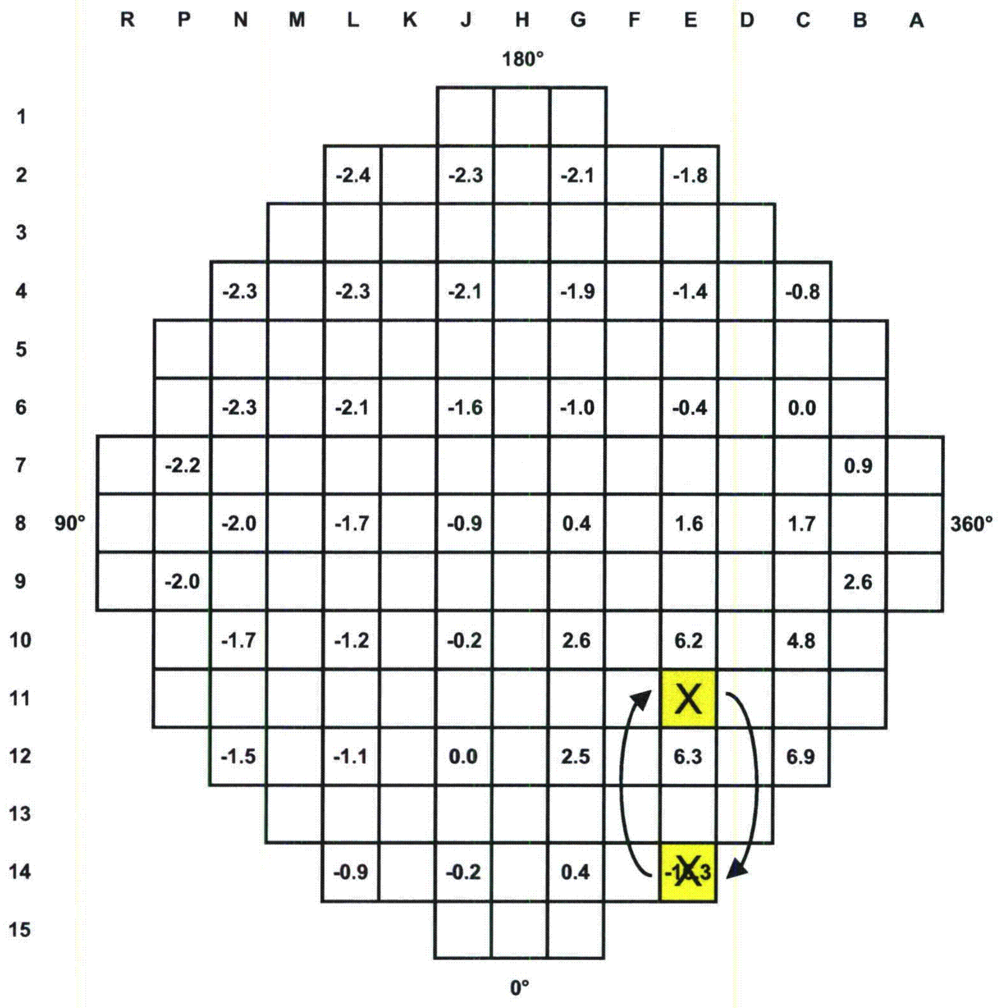

Figure 15.4.7-1

Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 3 Assembly

180°

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1																	
2						-1.6		-1.5		-1.5		-1.6					
3																	
4				-1.5		-1.2		-0.4		-0.4		-1.2		-1.5			
5																	
6				-1.5		0.2		3.5		3.5		0.2		-1.5			
7			-1.9												-1.9		
8	90°			-1.5		0.9		8.8	X	8.8		0.9		-1.5			360°
9			-2.0						X						-2.0		
10				-1.8		-0.4		1.8		1.8		-0.4		-1.8			
11																	
12				-2.0		-1.8		-1.4		-1.4		-1.8		-2.0			
13																	
14						-2.4		-2.4		-2.4		-2.4					
15																	

0°

Figure 15.4.7-2

Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 2 Assembly with the BP Rods Transferred to Region 1 Assembly

180°

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1																	
2						-2.2		-2.2		-2.2		-2.2					
3																	
4				-2.1		-1.7		-1.1		-1.1		-1.7		-2.1			
5																	
6				-2.0		-0.3		3.2		3.2		-0.3		-2.0			
7			-2.2												-2.2		
8	90°			-1.9		0.8		9.9	X	9.9		0.8		-1.9			360°
9			-2.2												-2.2		
10				-2.0		-0.3		3.2		3.2		-0.3		-2.0			
11																	
12				-2.1		-1.7		-1.1		-1.1		-1.7		-2.1			
13																	
14						-2.2		-2.2		-2.2		-2.2					
15																	

0°

Figure 15.4.7-3

Representative Percent Change in Local Assembly Average Power for Enrichment Error (Region 2 Assembly Loaded into Core Central Position)

180°

		R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1																	
2						-6.9		-6.9		-6.6		-6.0					
3																	
4				-6.7		-6.7		-6.5		-6.0		-5.0		-3.6			
5																	
6				-6.4		-6.2		-5.5		-4.2		-2.8		-1.4			
7			-6.1												1.1		
8	90°			-5.8		-5.2		-3.7		-1.3		1.6		3.2			360°
9			-5.6												6.9		
10				-5.0		-3.8		-1.4		3.3		11.1		13.8			
11																	
12				-4.1		-2.4		0.6		7.4		26.3	X	39.7			
13																	
14						-1.1		2.0		7.5		20.5					
15																	

0°

Figure 15.4.7-4

Representative Percent Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

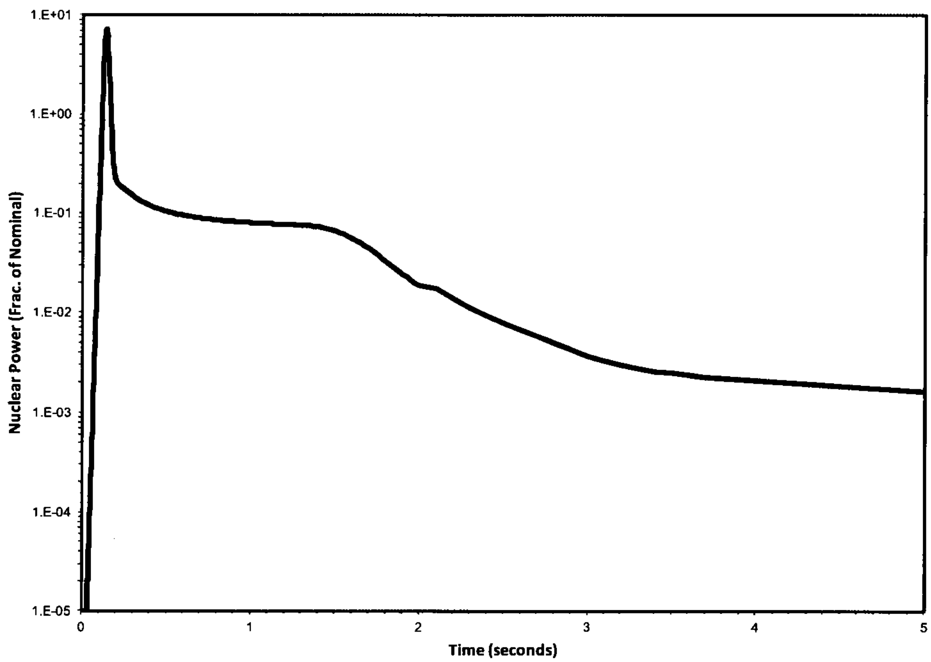

Figure 15.4.8-1

Nuclear Power Transient Versus Time for the PCMI Rod Ejection Accident

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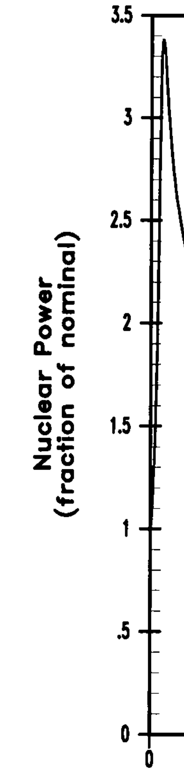

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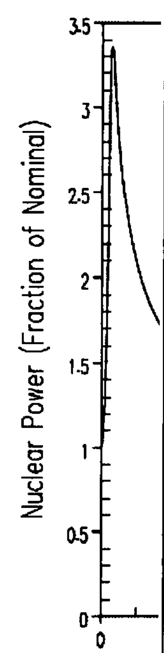

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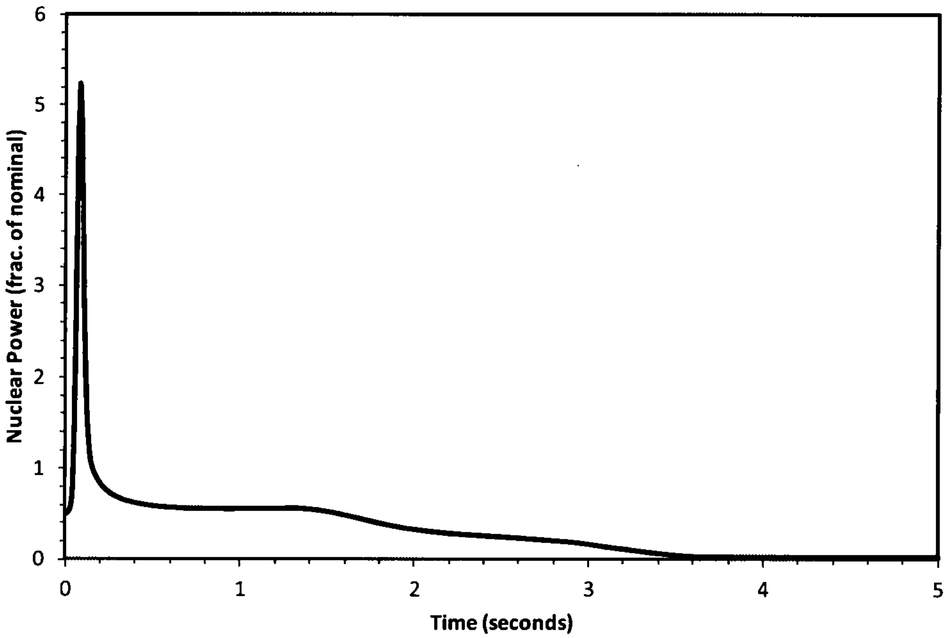
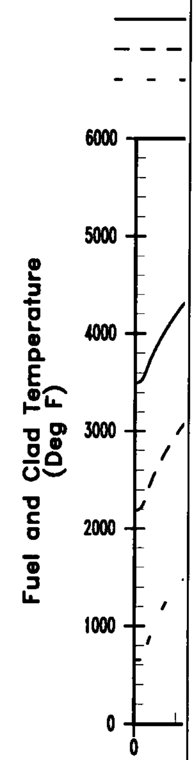


Figure 15.4.8-2

Nuclear Power Transient Versus Time for the High Clad Temperature Rod Ejection Accident



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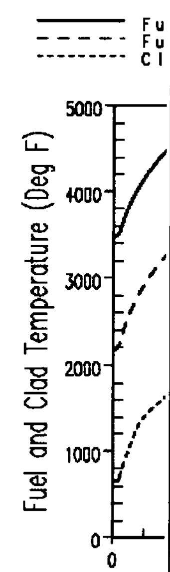

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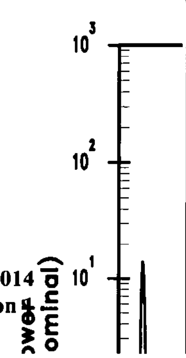


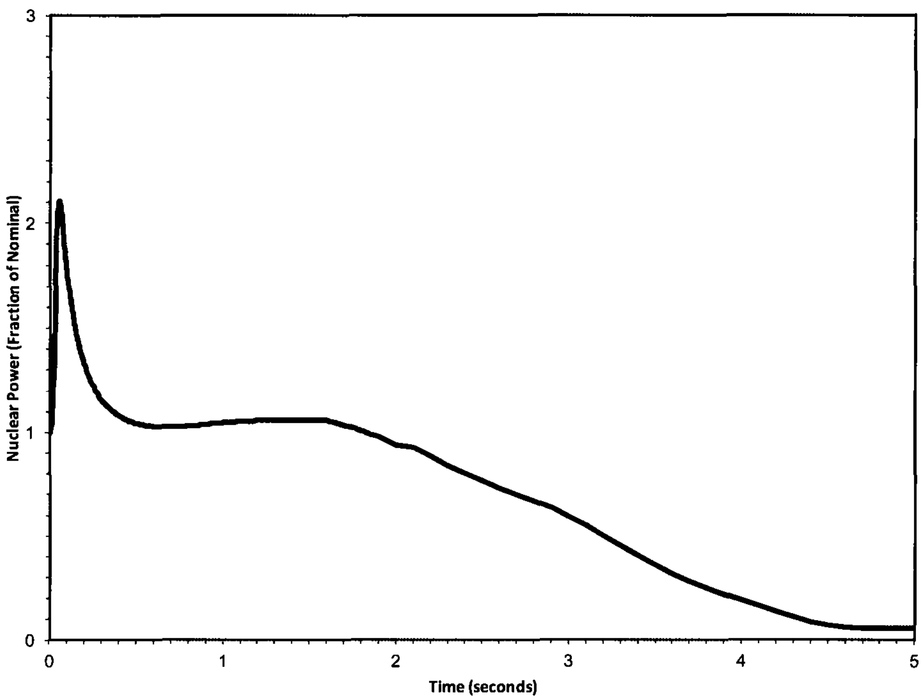
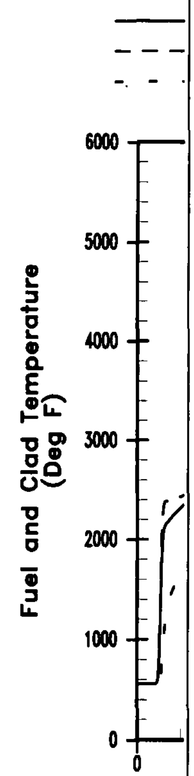


Figure 15.4.8-3

Nuclear Power Transient Versus Time
for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident



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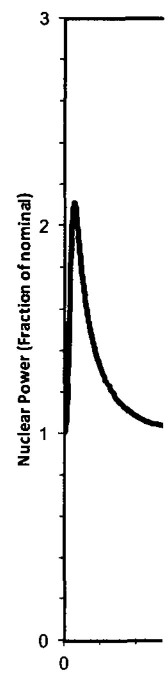

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Figure 15.4.8-4 not used

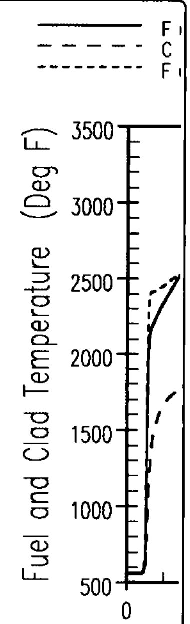

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Figure 15.4.8-4¶
Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at End of Life, Hot Zero Power

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.5	Change Summary Description
[15.5-1]	15.5.1, Inadvertent Operation of the CMT During Power Operation	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Editorial changes were made to the inadvertent CMT analyses to identify an operator action to open the safety related reactor vessel head vent to prevent filling the reactor coolant system water solid.
[15.5-2]	15.5.2, CVS Malfunction that Increases Reactor Coolant Inventory	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. Editorial changes were made to the inadvertent chemical and volume control analyses to identify an operator action to open the safety related reactor vessel head vent to prevent filling the reactor coolant system water solid.

15.5 Increase in Reactor Coolant Inventory

This section presents a discussion and analysis of the following events:

- Inadvertent operation of the core makeup tanks during power operation
- Chemical and volume control system malfunction that increases reactor coolant inventory

These Condition II events cause an increase in reactor coolant inventory.

15.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation

Comment [B1]: [15.5-1]

15.5.1.1 Identification of the Causes and Accident Description

Spurious core makeup tank operation at power could be caused by an operator error, a false electrical actuation signal, or a valve malfunction. A spurious signal may originate from any of the safeguards ("S") actuation channels as described in Section 7.3. The AP1000 protection logic is such that a single failure cannot actuate both core makeup tanks without also actuating the passive residual heat removal (PRHR) heat exchanger. A scenario such as this is the spurious "S" signal event. However, if one core makeup tank is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For the plant under automatic rod control, a reactor trip on high-3 pressurizer water level reactor trip is expected to occur followed by the PRHR actuation and eventually by an "S" signal, which would then actuate the second core makeup tank. When a consequential loss of offsite power is assumed, this event is more conservative than the spurious "S" signal event.

The inadvertent opening of the core makeup tank discharge valves, due to operator error or valve failure, results in significant core makeup tank injection flow leading to a boration similar to that resulting from a chemical and volume control system malfunction event. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result, the core makeup tank will continue injection and slowly increase the pressurizer level until the high-2 pressurizer level setpoint is reached and continues until the high-3 pressurizer level trip setpoint is reached. In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. The core makeup tank injection will increase as the steam generator outlet temperature increases resulting in a lower density in the CMT balance line. This event will then proceed similarly to a spurious "S" signal or chemical and volume control system malfunction event. However, this event is more limiting primarily due to the higher pressurizer level at the time of reactor trip and to the significant heat up of the injected fluid during the pre-trip phase of the accident. Thus, the inadvertent core makeup tank actuation event with a consequential loss of offsite power is analyzed here.

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Upon receipt of the high-3 pressurizer level reactor trip signal, the reactor is tripped; then the turbine is tripped after a 5-second delay and 3-seconds after turbine trip, a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in subsection 15.0.14. The high-3 pressurizer level signal also actuates the PRHR heat exchanger and blocks the pressurizer heaters, but a 15-second delay is built in to prevent unnecessary actuation of the PRHR heat exchanger if offsite power is maintained.

Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant cold leg temperature, in the loop without PRHR, increases and the core makeup tank starts injecting cold water into the reactor coolant system at a much higher rate. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

The PRHR heat exchanger extracts heat from the reactor coolant system leading to an "S" signal on a Low T_{cold} signal. The PRHR heat exchanger may inject asymmetrically into the steam generator outlet plenum such that a higher percentage of the PRHR flow is in one of two cold legs coming from the steam generator on the PRHR loop. To account for this, the analysis assumes that the Low T_{cold} setpoint is reached coincident with PRHR heat exchanger actuation. This actuates the second core makeup tank sooner in the transient, which is more limiting with respect to filling the pressurizer.

Both core makeup tanks inject mass into the reactor coolant system and the pressurizer level continues to increase until the operators take action to end the pressurizer level increase transient. The operators are assumed to be alerted to a potential filling event on the high-2 pressurizer level signal, which occurs well before the reactor trip on the first of two high-3 pressurizer level signals. The operator action assumed in the analysis is to open the reactor vessel head vent following receipt of the second high-3 pressurizer level signal; this action is at least 30 minutes (45 minutes as analyzed) after the operator has been alerted by the high-2 pressurizer level signal. When the head vent is opened, the pressurizer level increase slows and ultimately the level begins to decrease.

This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

The plant response to an inadvertent core makeup tank actuation is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer,

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Deleted: The cold injection flow from the second CMT initially results in a fast decrease in temperature and shrinkage of the reactor coolant. However, as the temperature decreases, the PRHR heat removal capability diminishes and a moderate heat up occurs followed by the increase of pressurizer water level. The second CMT injection rate is much lower than that experienced during the first part of the transient from the first CMT. Due to the colder cold leg temperatures, the density in balance line is much higher than during the first part of the ... [1]

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pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables, including temperatures, pressures, and power level.

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Reactor power and average temperature drop immediately following the trip, and the operating conditions never approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization occurs.

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Core makeup tank and PRHR system performance is conservatively simulated. Core makeup tank enthalpies have been maximized. This is conservative because it minimizes the cooling provided by the core makeup tanks as flow recirculates and thereby increases the peak pressurizer water volume during the transient. Core makeup tank injection and balance lines pressure drop is minimized. This maximizes the core makeup tank flow injected in the primary system. During this event, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tanks is offset by an equal volume of recirculation flow that enters the core makeup tanks via the balance lines. PRHR heat transfer capability has been minimized.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi below nominal. The initial reactor coolant system average temperature is assumed to be 8°F below nominal.

- Control systems

  The pressurizer spray system and automatic rod control system are conservatively assumed to operate. The pressurizer heaters are automatically blocked on a high-3 pressurizer level signal, so they cannot add heat to the system during the period of thermal expansion that produces the peak pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event. Other control systems are conservatively not assumed to function during the transient.

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. With these minimum feedback parameters and the operability of the pressurizer spray system and automatic rod control system assumed, the reactivity effects of the boron injection from the core makeup tanks is counteracted. As a result, the high-3 pressurizer signal is the first reactor trip signal generated during the transient.

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- Boron injection

  The transient is initiated by an inadvertent opening of the discharge valves of one of the two core makeup tanks. The core makeup tank injects 3400 ppm borated water.

- Protection and safety monitoring system actuations

  The operators are assumed to be alerted of the pressurizer level increase transient on the high-2 pressurizer level signal. Reactor trip is initiated by the first of two high-3 pressurizer level signals. The second high-3 pressurizer level signal triggers the operators to open the reactor vessel head vent; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal.

  The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

Plant systems and equipment available to mitigate the effect of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-8 show the transient response to the inadvertent operation of one of the two core makeup tanks during power operation. The inadvertent opening of the core makeup tank discharge valves occurs at 10 seconds. As the core makeup tank continues to add inventory to the primary system, the pressurizer level begins to increase until the high-2 pressurizer level setpoint is reached (556.1 seconds) and continues until the high-3 pressurizer level reactor trip setpoint is reached at about 2,589.3 seconds. After a 2-second delay, the neutron flux starts decreasing due to the reactor trip, which is followed by turbine trip after a 5-second turbine trip delay. Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. Due to the assumed loss of offsite power, the reactor coolant pumps trip at about 2,599.3 seconds. The cold leg temperature increases and the core makeup tank starts injecting cold water into the reactor coolant system at a higher rate due to the increased driving head resulting from the density decrease in the balance line and due to the reduced pressure drop between the cold leg and the injection line connection on the reactor vessel following the trip of the reactor coolant pumps. The post-trip primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached at 2,736.6 seconds, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

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Following a conservative 17-second delay, the valves are assumed to open to actuate the PRHR heat exchanger at 2,768.6 seconds.

If the PRHR heat exchanger coolant asymmetrically injects into the steam generator outlet plenum, then one cold leg could reach the Low T_{cold} "S" setpoint more quickly than if the flow were split evenly. To conservatively account for this effect, the Low T_{cold} "S" signal is modeled to actuate simultaneously with the actuation of the PRHR heat exchanger (2,768.6 seconds). The Low T_{cold} "S" signal activates the second core makeup tank, which then begins injecting additional mass into the reactor coolant system. Previous analyses have demonstrated that a more limiting pressurizer fill transient is calculated the earlier the second core makeup tank is actuated.

As the second core makeup tank begins injecting, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2 pressurizer level signal (556.1 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action at least 30 minutes later. In this analysis, since pressurizer level continues to increase, the high-3 pressurizer level reactor trip setpoint is reached within this time. The operator action assumed in this case is to open the reactor vessel head vent to preclude overfill following receipt of the second high-3 pressurizer level signal (3,256.1 seconds); this action is at least 30 minutes (45 minutes as analyzed) after the operator has been alerted by the high-2 pressurizer level signal.

The safety related reactor vessel head vent is opened by the operators and the pressurizerwater level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of an inadvertent actuation of a core makeup tank.

During the event, the departure from nucleate boiling ratio (DNBR) never drops significantly below the initial value due to the addition of highly borated water from the core makeup tanks to the reactor coolant system. At the time of reactor trip core power and heat flux drop rapidly and the DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

As noted above, the limiting case presented here models explicit operator action 45 minutes after receipt of the high-2 pressurizer level signal. For pressurizer level increase events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup (high-2), the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed

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- **Deleted:** After reactor trip, the pressure in the primary and secondary systems increases initially due to the assumed unavailability of the non-safety-related control systems. The primary and secondary system pressures eventually decrease as the PRHR system removes decay heat. The core makeup tank works in recirculation mode, meaning it is always filled with water because cold borated water injected through the injection line is ... [2]
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- **Deleted:** CMT
- **Deleted:** The pressurizer level initially shrinks due to the ... [3]
- **Deleted:** add inventory
- **Deleted:** the primary system,
- **Deleted:** begins to
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- **Deleted:** essentially stops recirculating. The PRHR ... [4]
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into service, the operator could use the safety related reactor vessel head vent valves to reduce the increase in coolant inventory (this is explicitly modeled in the case presented here). For these events, following the procedures outlined in the Emergency Response Guidelines AFR-I.1, there is sufficient time for the operator to mitigate the consequences of this event.

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15.5.1.4 Conclusions

The results of this analysis show that inadvertent operation of the core makeup tanks during power operation does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

Comment [B2]: [15.2-2]

15.5.2.1 Identification of Causes and Accident Description

An increase of reactor coolant inventory, which results from addition of cold unborated water to the reactor coolant system, is analyzed in subsection 15.4.6.

In this subsection 15.5.2, the increase of reactor coolant system inventory due to the addition of borated water is analyzed.

The increase of reactor coolant system coolant inventory may be due to the spurious operation of one or both of the chemical and volume control system pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine control valve fully opens.

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an "S" signal will be generated by either the low T_{cold} or low steam line pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similarly to, and is only slightly more limiting than, a spurious "S" signal event. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

Under typical operating conditions for the AP1000, the boron concentration of the injected chemical and volume control system water is equal to that of the reactor coolant system. If the chemical and volume control system is functioning in this manner and the pressurizer spray system functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is also terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

While these scenarios are the most probable outcomes of a chemical and volume control system malfunction, several combinations of boron concentration, feedback conditions, and plant system interactions have been identified which can result in more limiting scenarios with respect to pressurizer overfill. The key factors that make this event more limiting than a spurious "S" signal event are that the reactor coolant system is at a lower average temperature, higher pressure, and a higher pressurizer level at the time an "S" signal is generated. These factors produce a greater volume of higher density water and, thus, a larger reactor coolant system mass at the time of the "S" signal. In addition, at lower reactor coolant system average temperature, the PRHR is less effective in removing decay heat, which results in greater expansion of the cold water injected by the core makeup tanks.

The limiting analysis scenario minimizes reactor coolant system average temperature, maximizes reactor coolant system mass, and maximizes pressurizer water volume at the time of an "S" signal. This scenario is as follows:

- Both of the chemical and volume control system pumps spuriously begin delivering flow at a boron concentration slightly higher than that of the reactor coolant system. (Assuming that a chemical and volume control system malfunction results in both chemical and volume control system pumps delivering flow is a conservative assumption. One chemical and volume control system pump is automatically controlled and one is manually controlled.)

- The non-safety-related pressurizer spray is assumed to be available, so that a high pressurizer pressure reactor trip is prevented.

Due to the boron addition in the core, the plant cools down until an "S" signal is generated on low cold leg temperature. On the "S" signal, the reactor is tripped, the core makeup tank discharge valves are opened, the reactor coolant pumps are tripped, the pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated.

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Normally, the reactor coolant pumps would be tripped 15 seconds after the receipt of the "S" signal. However, to meet the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. Following reactor trip and a

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5-second timer delay the turbine is tripped 3-seconds after a turbine trip a consequential loss of offsite power is assumed. The basis for the 3-second delay is described in subsection 15.0.14. As a result, the reactor coolant pumps are conservatively assumed to trip about 10 seconds before they would otherwise trip due to the "S" signal.

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This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

15.5.2.2 Analysis of Effects and Consequences

The malfunction of the chemical and volume control system is analyzed by using a modified version of the computer program LOFTRAN (Reference 1) described in subsection 15.0.11.2. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, PRHR heat exchanger, and core makeup tanks. The program computes pertinent plant variables including temperatures, pressures, and power level.

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Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. The analysis demonstrates that no reactor coolant system overpressurization or loss of reactor coolant system water occurs.

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The assumptions are as follows:

Deleted: Using an iterative analysis process, the boron concentration is chosen such that this limiting case bounds the cases that model explicit operator action 30 minutes after the reactor trip.¶

- Initial operating conditions

  The initial reactor power is assumed to be 101 percent of nominal. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial reactor coolant system average temperature is assumed to be 8°F above nominal.

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- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. For a different set of reactivity feedback parameters, a different chemical and volume control system boron concentration can result in an identical transient.

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- Reactor control

  Rod control is not modeled.

- Pressurizer heaters

  The pressurizer heaters are automatically blocked on an "S" signal, and do not add heat to the system during the period of fluid thermal expansion that produces the peak pressurizer water volume. Thus, the pressurizer heaters are assumed to be inoperable during this event.

- Pressurizer spray

The spray system controls the pressurizer pressure so that a high pressurizer pressure reactor trip is prevented.

- Boron injection

After 10 seconds at steady state, the chemical and volume control system pumps start injecting borated water, which is slightly above the reactor coolant system boron concentration. Upon receipt of an "S" signal, the core makeup tanks begin injecting 3400 ppm borated water. The chemical and volume control system pumps are isolated on high-2 pressurizer level. In this analysis the boron concentration of the chemical and volume control system is iterated upon until the high-2 pressurizer level and the low T_{cold} "S" setpoint are reached at the same time. This begins core makeup tank injection when the chemical and volume control system pumps are isolated, which is conservative with respect to filling the pressurizer.

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- Turbine load

The turbine load is assumed constant until the turbine D-EHC drives the control valve wide open. Then the turbine load drops as steam pressure drops.

- Protection and safety monitoring system actuations

If the automatic rod control system is modeled and the pressurizer spray system functions properly, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If the automatic rod control system is not active and the pressurizer spray system is assumed to be available, reactor trip may be initiated on either low T_{cold} "S" or a low steam line pressure "S" signal.

The core decay heat is removed by the PRHR heat exchanger. The worst single failure is assumed to occur in the outlet line of the PRHR heat exchanger. One of the two parallel isolation valves is assumed to fail to open.

Plant systems and equipment available to mitigate the effect of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.5.2.3 Results

Figures 15.5.2-1 through 15.5.2-9 show the transient response to a chemical and volume control system malfunction that results in an increase of reactor coolant system inventory.

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At 2,271.3 seconds, the low T_{cold} setpoint is reached, the reactor trips on the resulting "S" signal, and the control rods start moving into the core. At the same time, the high-2 pressurizer level setpoint is reached and after a conservative delay, the chemical and volume control system injection is isolated.

The turbine is tripped as a result of the reactor trip following a 5-second turbine trip timer delay. After a 3-second delay following turbine trip, a consequential loss of offsite power is assumed and the reactor coolant pumps trip. The basis for the 3-second delay is described in subsection 15.0.14. Soon after reactor trip, the pressurizer heaters are blocked and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated and the core makeup tank discharge valves are opened. The core makeup tanks work in recirculation mode, meaning they are always filled with water because cold borated water injected through the injection lines is replaced by hot water coming from the cold leg balance lines.

The operation of the PRHR heat exchanger and the core makeup tanks cools down the plant. Due to the swelling of the core makeup tank water, the pressurizer level continues to increase. The operators are assumed to be alerted by the high-2pressurizer level signal (2,270.8 seconds) that a pressurizer level increase transient is underway, and it is assumed that the operators are ready to take corrective action at least 30 minutes later. The specific operator action assumed in this case is to open the reactor vessel head vent to preclude pressurizer overfill following the high-3 pressurizer level signal (4,070.8 seconds); this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal.

The safety related reactor vessel head vent is opened by the operators and the pressurizer water level increase slows and eventually the level begins to decrease. This demonstrates that the capacity of the reactor vessel head vent is sufficient to preclude pressurizer overfill as a result of a chemical and volume control system malfunction that causes an increase in reactor coolant inventory.

During the event, the DNBR never drops significantly below the initial value since both the chemical and volume control system and the core makeup tanks add borated water to the reactor coolant system. At the time of reactor trip, core power and heat flux drop rapidly and the DNBR is well above the design limit value defined in Section 4.4.

The calculated sequence of events is shown in Table 15.5-1.

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Deleted: matches the core decay heat, the pressurizer water volume stops increasing, and the pressurizer safety valves close. Then the core makeup tanks essentially stop injecting.¶ Figure 15.5.2-6 shows the DNBR until the time of reactor coolant pump trip and subsequent flow coastdown due to the loss of offsite power. At this time, core power and heat flux have diminish... [7]

The limiting case presented here models operator action to open the reactor vessel head vent following receipt of the high-3 pressurizer level signal; this action is at least 30 minutes after the operator has been alerted by the high-2 pressurizer level signal. For pressurizer level increase events, the operator could take other actions to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator would use the safety-related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following operations procedures, there is sufficient time for the operator to mitigate the consequences of this event.

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15.5.2.4 Conclusions

The results of this analysis show that a chemical and volume control system malfunction does not adversely affect the core, the reactor coolant system, or the steam system. Water is not relieved from the pressurizer safety valves. DNBR remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

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If the automatic rod control system and the pressurizer spray systems are assumed to function, no reactor trip signal is expected to occur. Instead, the event would be terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If manual rod control is assumed and the pressurizer spray system is assumed to be unavailable, reactor trip may be initiated on either a high pressurizer pressure, low T_{cold} "S", or a low steamline pressure "S" signal.

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15.5.3 Boiling Water Reactor Transients

This subsection is not applicable to the AP1000.

15.5.4 Combined License Information

This subsection has no requirement for additional information to be provided in support of the Combined License application.

15.5.5 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

Table 15.5-1 (Sheet 1 of 2) **TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY**		
Accident	**Event**	**Time (seconds)**
Inadvertent operation of the core makeup tanks during power operation	Core makeup tank discharge valves open	10
	High-2 pressurizer level setpoint reached	556.1
	High-3 pressurizer level setpoint reached	2,589.3
	Rod motion begins	2,591.3
	Loss of offsite power	2,599.3
	Reactor coolant pumps trip	2,599.3
	High-3 pressurizer level setpoint reached	2,735.6
	PRHR heat exchanger actuated	2,768.6
	Low T_{cold} "S" setpoint is reached	2,768.6
	Second CMT starts recirculating	2,768.6
	Main steam and feed lines are isolated	2,780.6
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached)	3,256.1
	Peak pressurizer water volume occurs	5,460.0

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Table 15.5-1 (Sheet 2 of 2) **TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY**		
Accident	**Event**	**Time (seconds)**
Chemical and volume control system malfunction that increases reactor coolant inventory	Chemical and volume control system charging pumps start	10.0
	Low T_{cold} "S" signal and high-2 pressurizer level signals are reached	2,270.8
	Core makeup tank discharge valves open	2,271.4
	Rod motion begins	2,272.8
	Loss of offsite power	2,280.8
	Reactor coolant pumps trip	2,280.8
	Main steam and feed lines are isolated	2,283.4
	PRHR heat exchanger actuated	2,288.4
	Chemical and volume control system charging pumps are isolated	2,308.9
	Operators open the reactor vessel head vent after the high-3 pressurizer level signal is reached (at least 30 minutes after high-2 pressurizer level setpoint is reached)	4,070.8
	Peak pressurizer water volume occurs	5,078.0
	Pressurizer water volume begins to decrease	5,484.0

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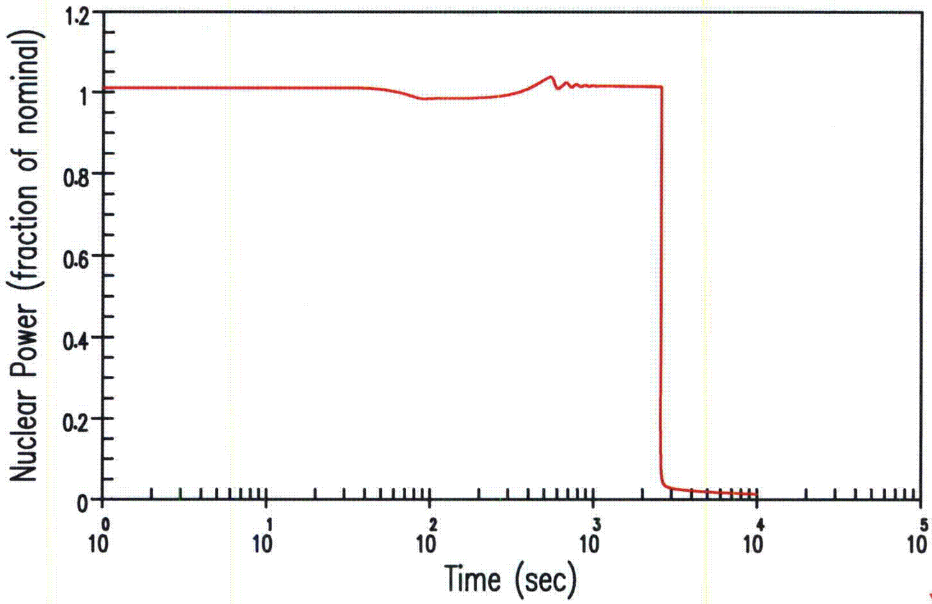

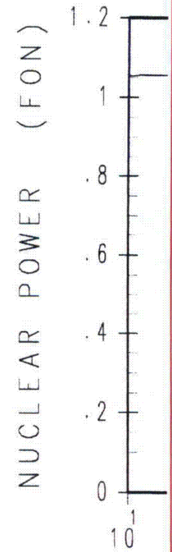

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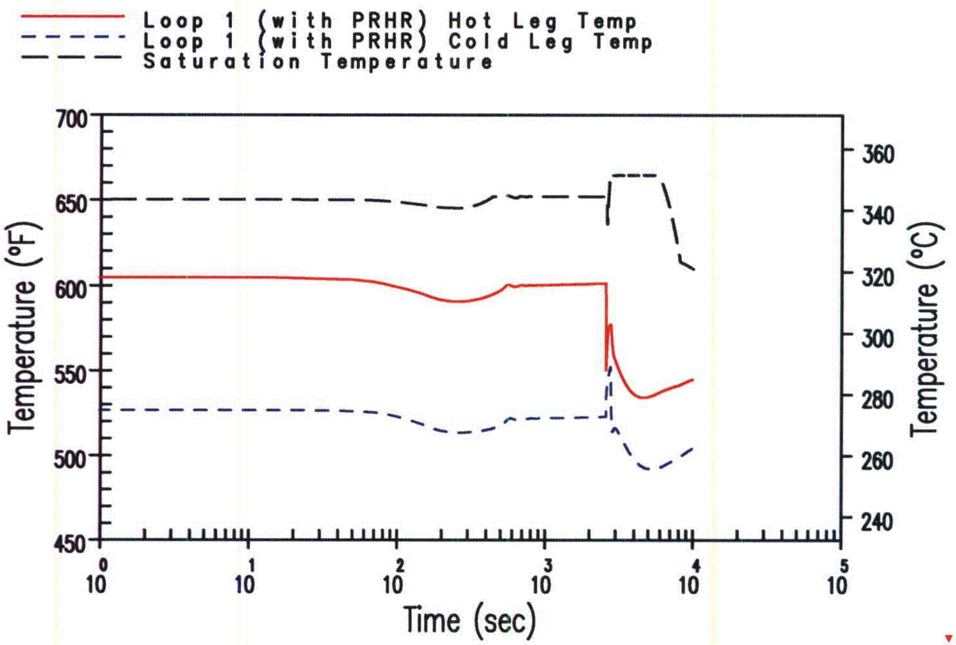
Figure 15.5.1-1

Core Nuclear Power Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



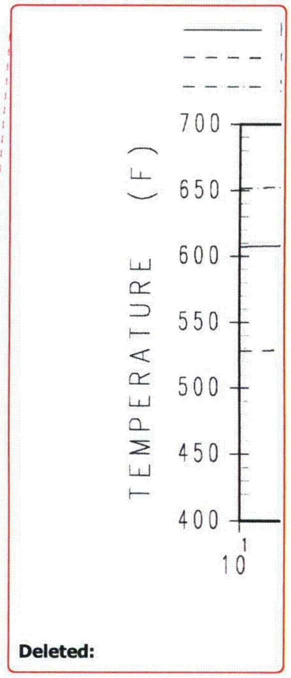
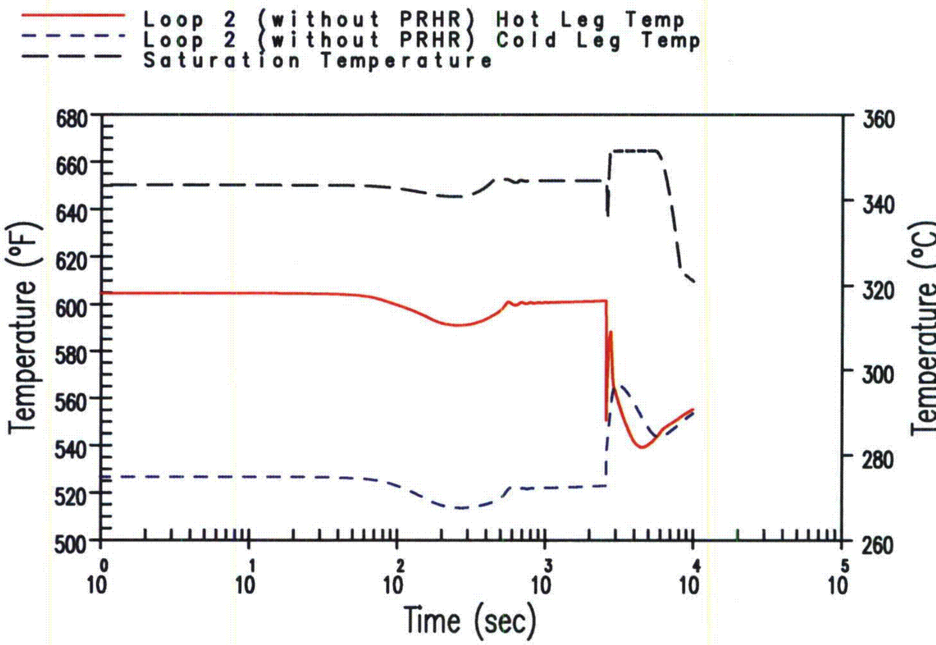


Figure 15.5.1-2

RCS Temperature Transient in Loop Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



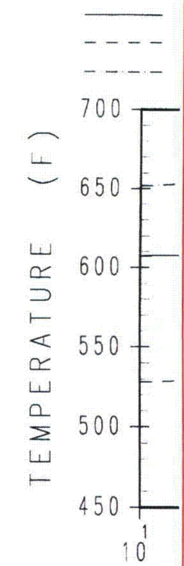


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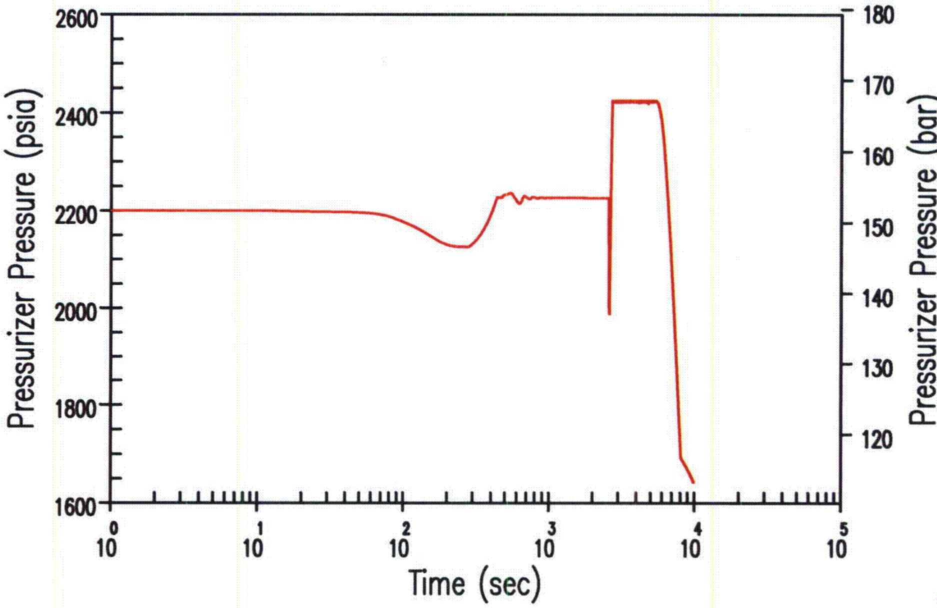
Figure 15.5.1-3

RCS Temperature Transient in Loop Not Containing the PRHR for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



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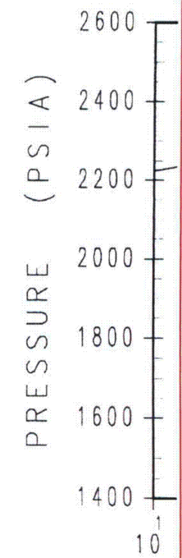
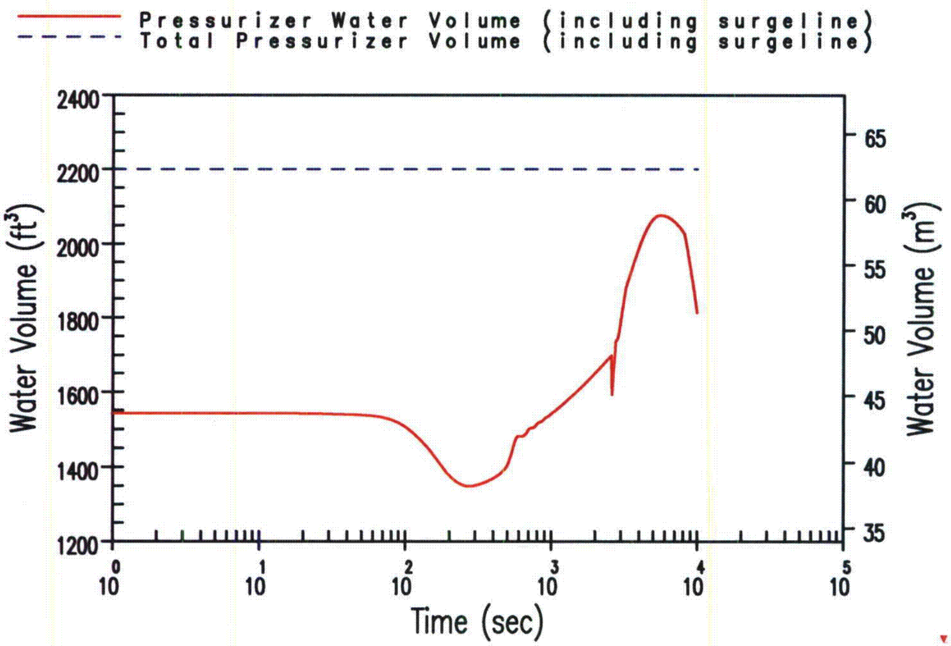


Figure 15.5.1-4

Pressurizer Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



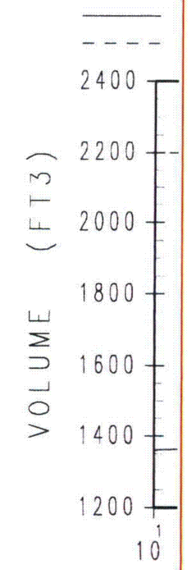


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Figure 15.5.1-5

Pressurizer Water Volume Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

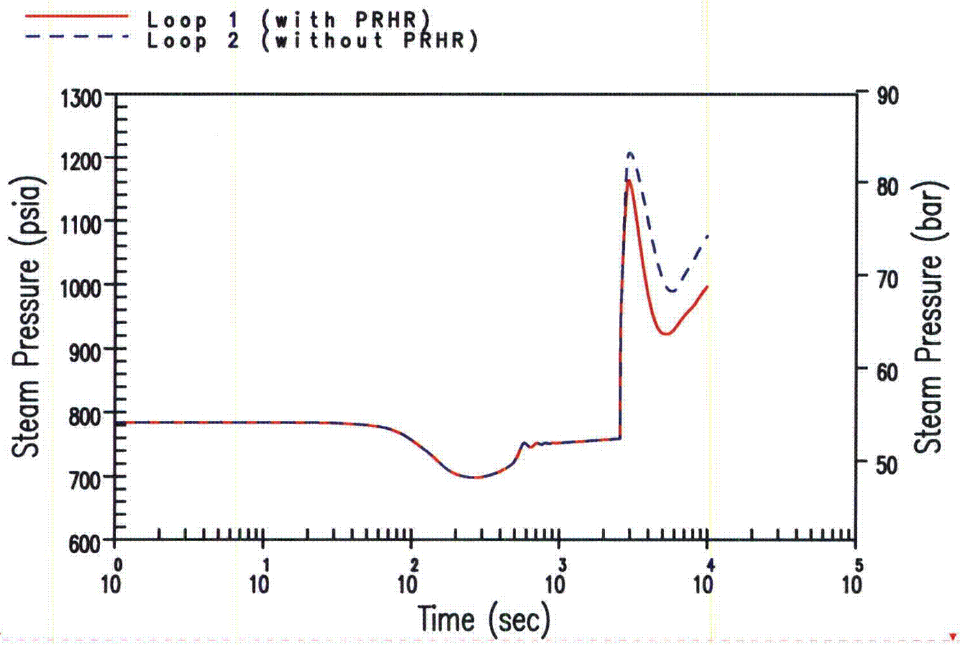

Figure 15.5.1-6

Steam Generator Pressure Transient for Inadvertent Operation of the Emergency Core Cooling System Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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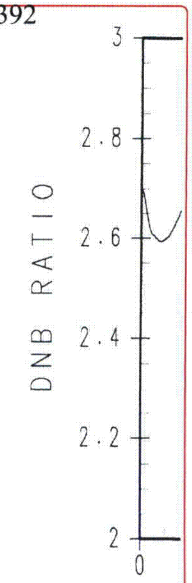

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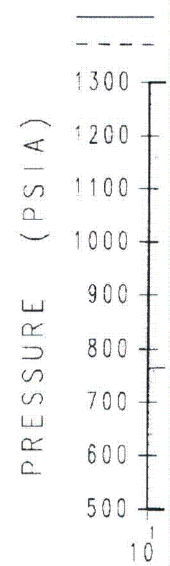

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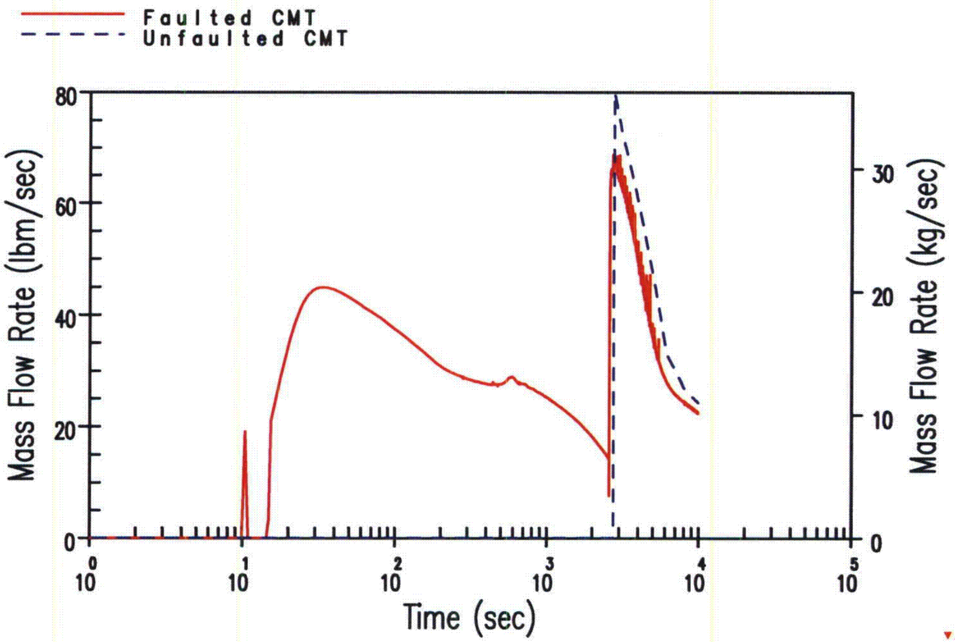

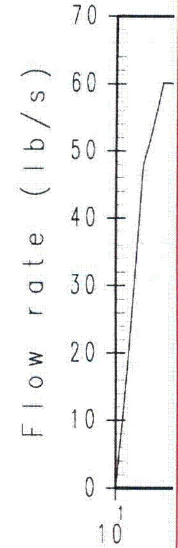

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Figure 15.5.1-7

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CMT Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves

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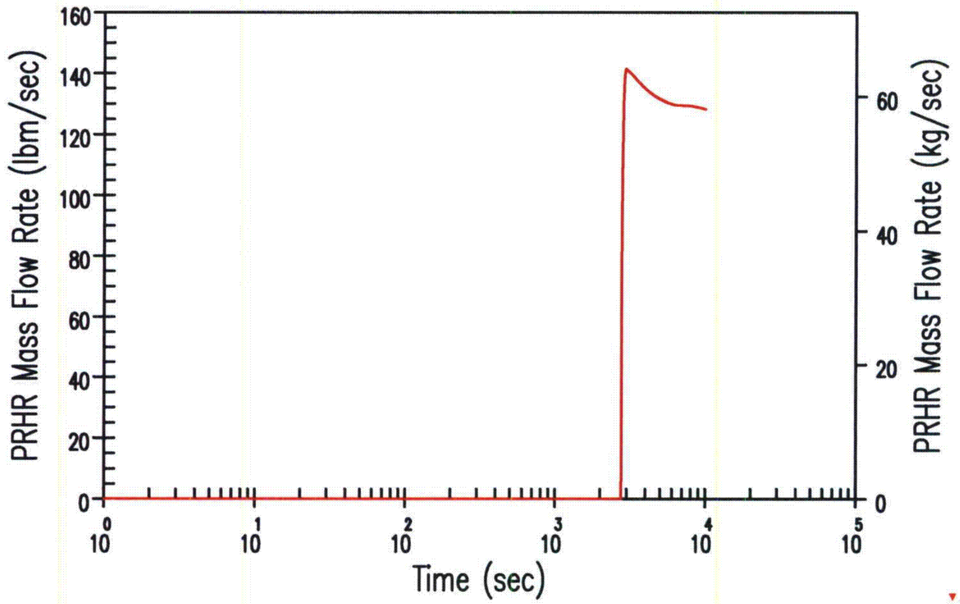
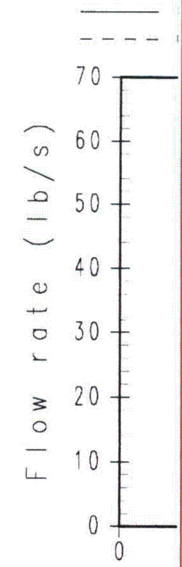


Figure 15.5.1-8

PRHR Flow Rate Transient
for Inadvertent Operation of the Emergency Core Cooling System
Due to a Spurious Opening of the Core Makeup Tank Discharge Valves



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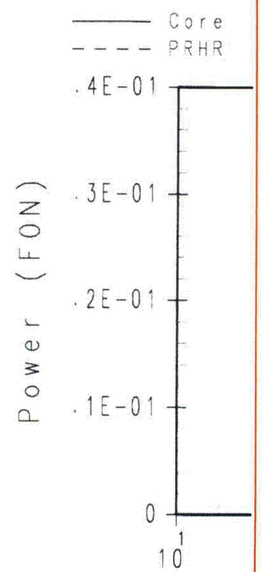

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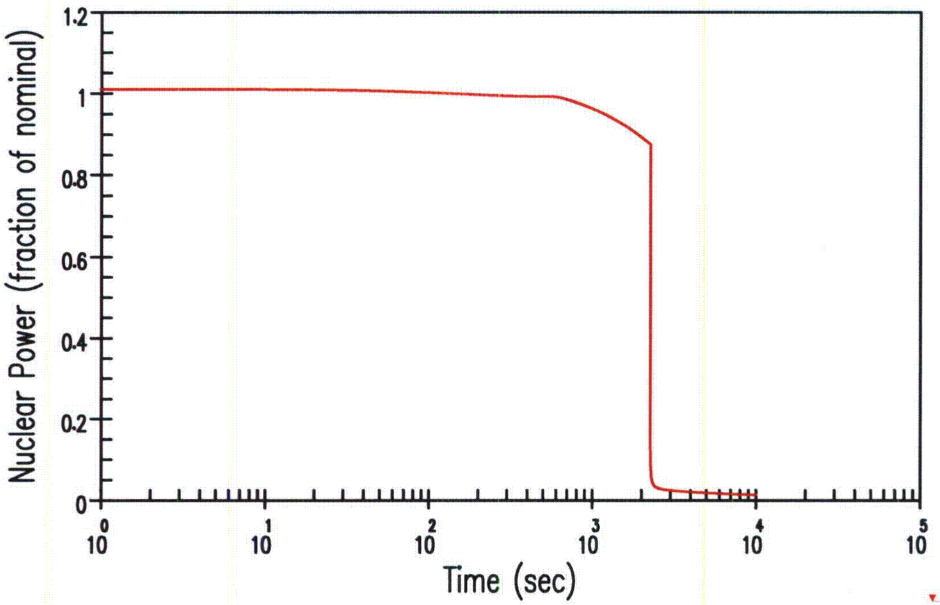

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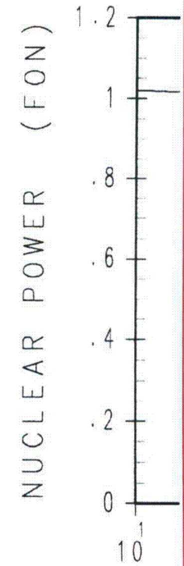
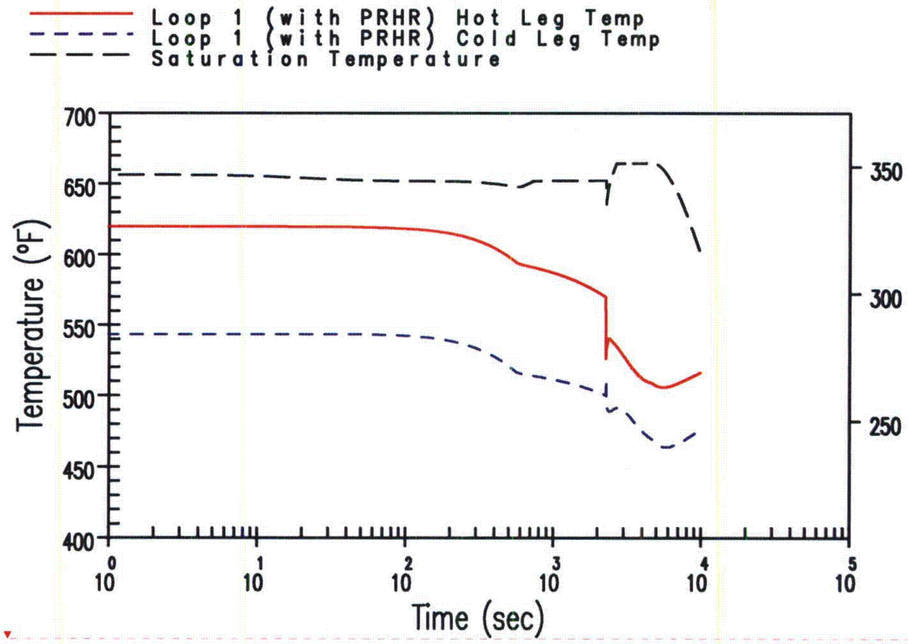


Figure 15.5.2-1

Core Nuclear Power Transient for Chemical and Volume Control System Malfunction



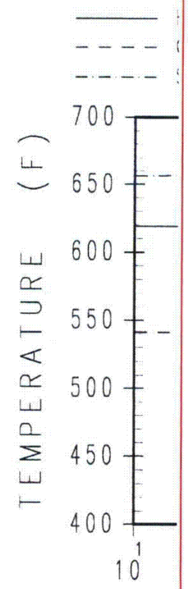


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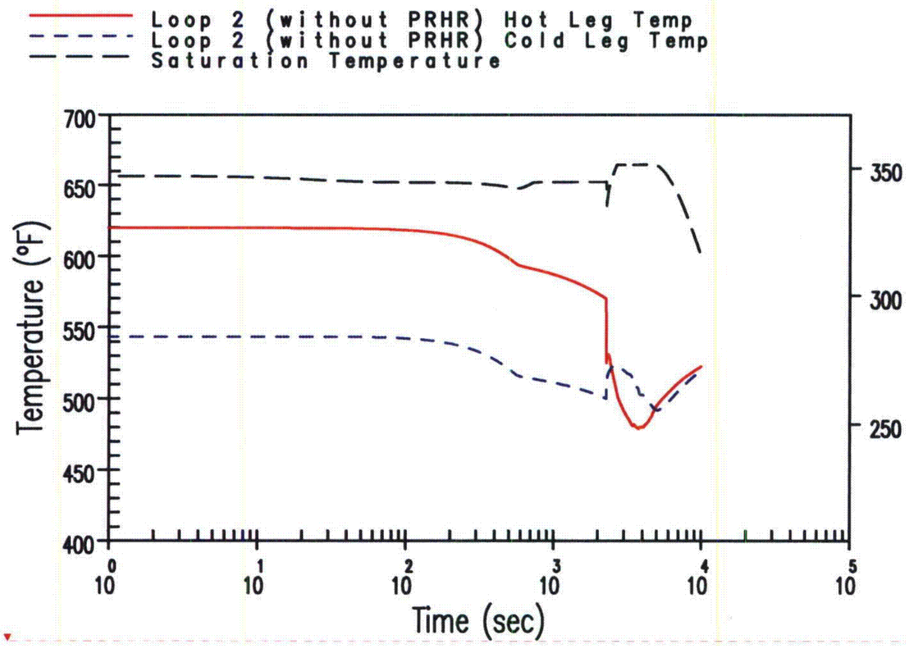
Figure 15.5.2-2

RCS Temperature Transient in Loop Containing the PRHR for Chemical and Volume Control System Malfunction



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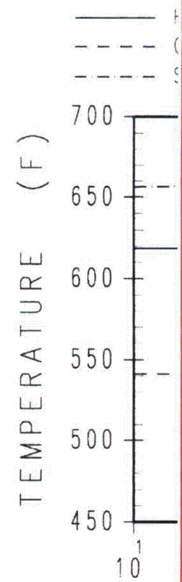
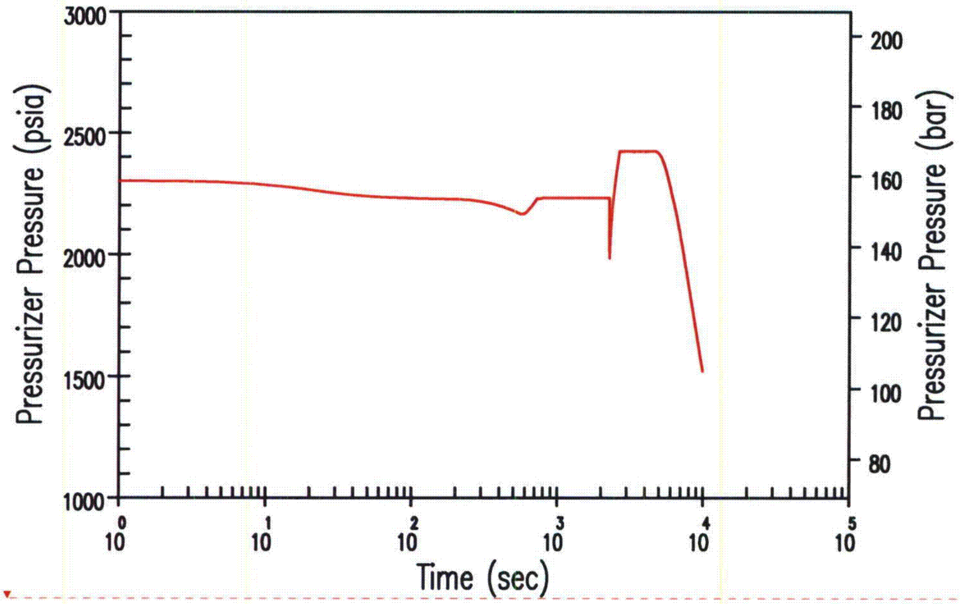


Figure 15.5.2-3

RCS Temperature Transient in Loop Not Containing the PRHR for Chemical and Volume Control System Malfunction



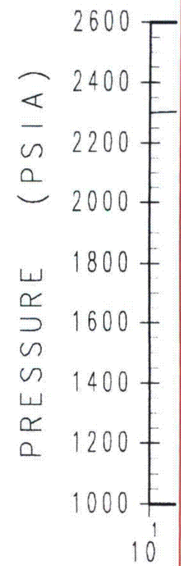


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Figure 15.5.2-4

Pressurizer Pressure Transient for Chemical and Volume Control System Malfunction

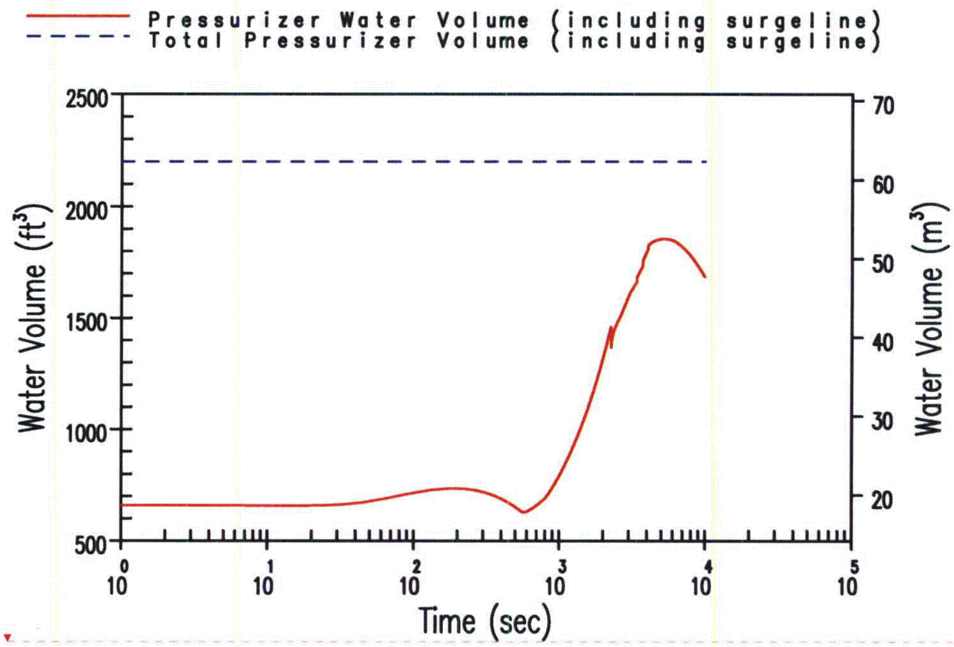

Figure 15.5.2-5

Pressurizer Water Volume Transient
for Chemical and Volume Control System Malfunction

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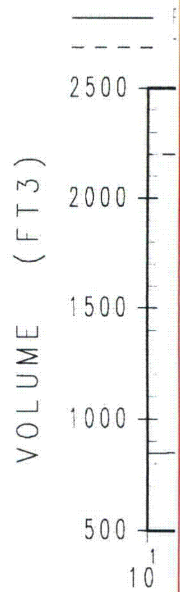

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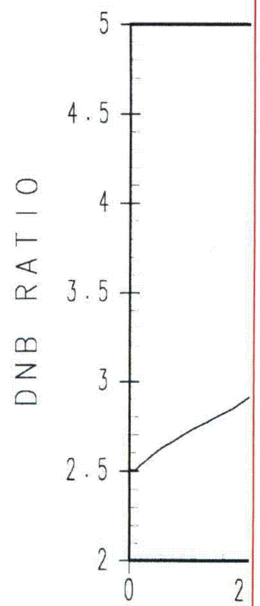

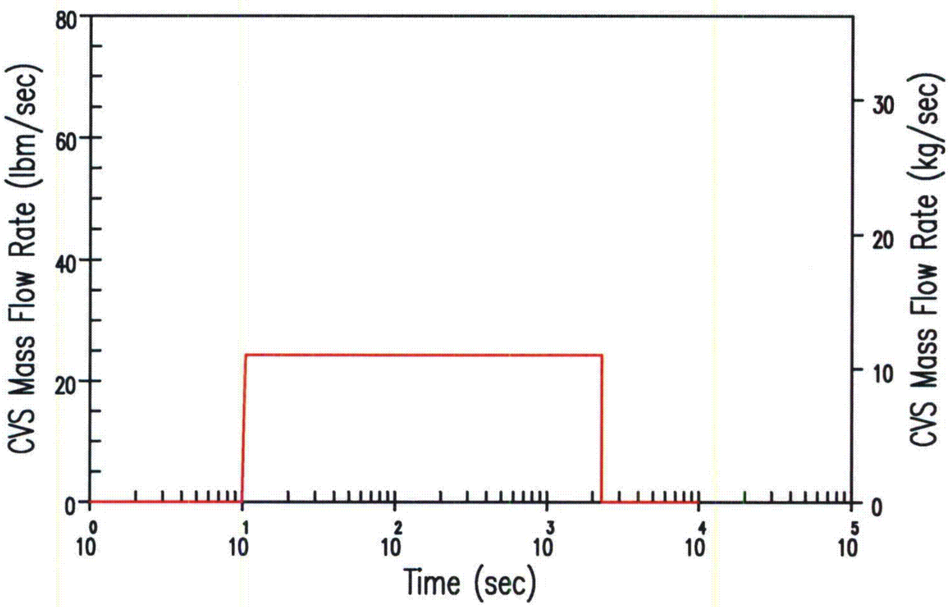

Figure 15.5.2-6

CVS Flow Rate Transient
for Chemical and Volume Control System Malfunction

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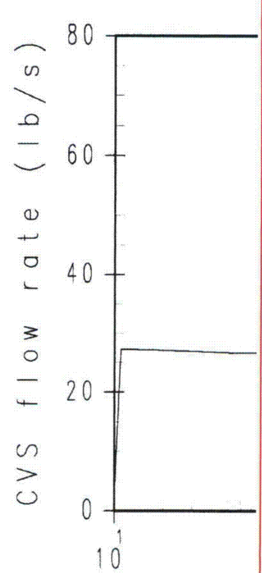
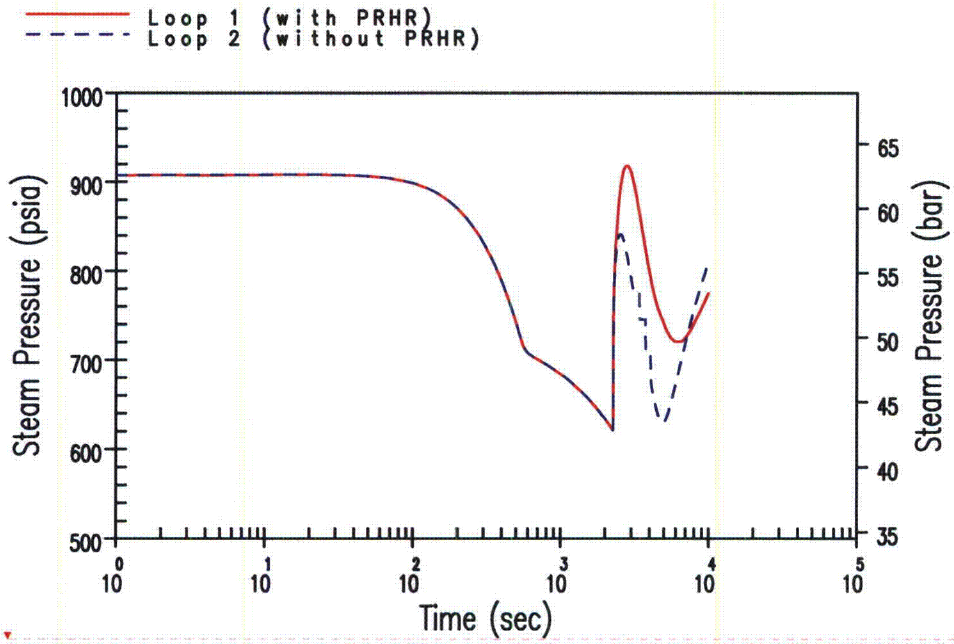


Figure 15.5.2-7¶



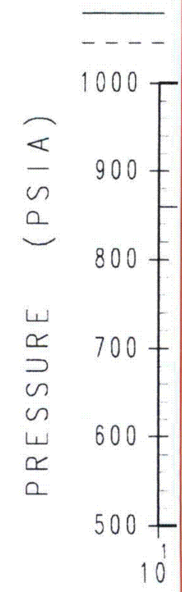


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Figure 15.5.2-7

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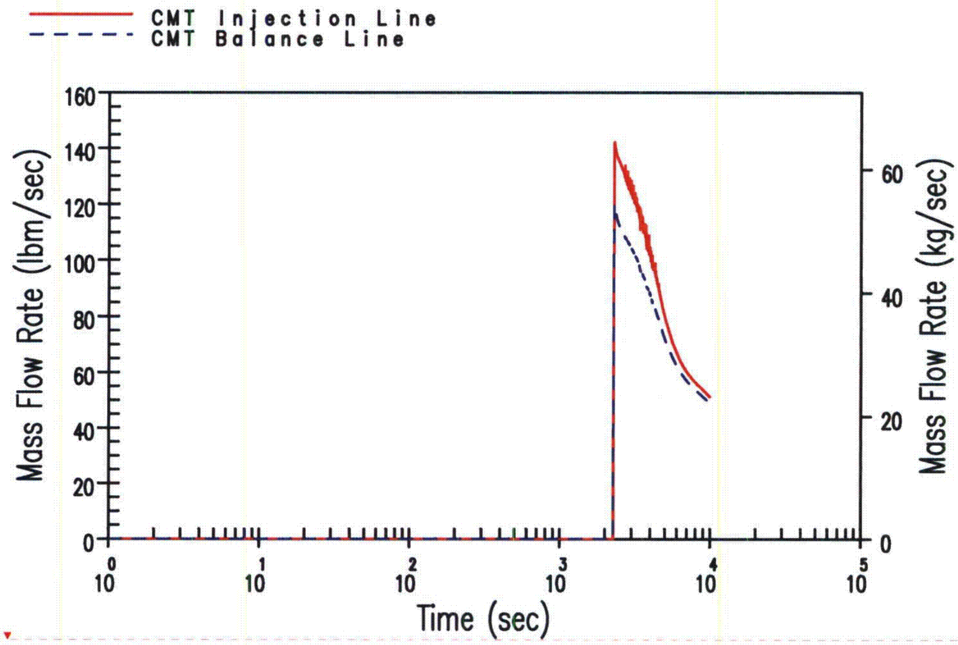
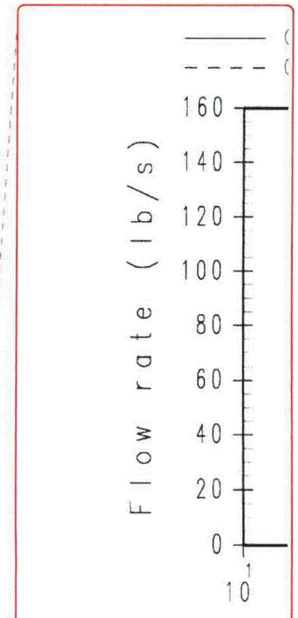
Steam Generator Pressure Transient
for Chemical and Volume Control System Malfunction





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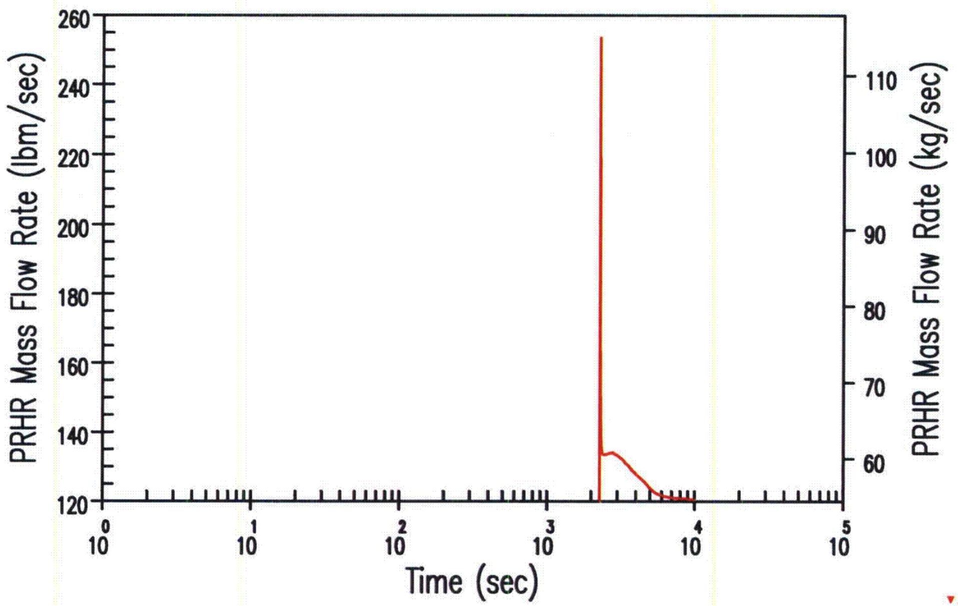
CMT Injection Line and Balance Line Flow Transient for Chemical and Volume Control System Malfunction



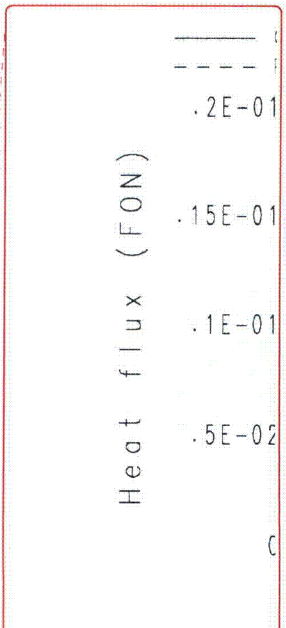


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Figure 15.5.2-9

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PRHR Flow Rate Transient
for Chemical and Volume Control System Malfunction

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AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-1]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-2]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Clarification. The ADS actuation sequence includes progression of the valves from 1-3 with associated delay timers in between such that the max valve stroke times plus delay timers ensure each valve set doesn't actuate before the other valve set.
[15.6-3]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Additional text was added to provide clarity. It was not a change in the analysis or design. The DCD description itself was updated.
[15.6-4]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Updated the description of the valve parameters. The previous value represents the max opening time. The max opening times of the ADS valves were previously revised. However, there are delay timers in place, so if the max stroke time changes the delay timers can be adjusted accordingly so the analysis is not affected. To reduce the number of possible future changes the minimum stroke time was listed, which is a hard functional requirement for the valve performance.
[15.6-5]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	By adding the comments in the preceding paragraph it was possible to omit these sections.
[15.6-6]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Additional detail on why loss of AC power need not be considered for an RCS depressurization event has been added. The previous description did not contain sufficient detail.
[15.6-7]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	Because of the RCP delay on reactor trip the inadvertent ADS valve operation does not challenge DNB. Therefore LOFTRAN is sufficient to conclude DNB margin is maintained.
[15.6-8]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	As stated in 10CFR 50 GDC 17 analysis of coincident loss of AC power for a RCS depressurization event is not required based on the Turbine and RCP response to this scenario.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-9]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	With a loss of AC power, the OTΔT is the trip signal. Now, the Low Pressurizer Pressure is the actuated protection signal.
[15.6-10]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-8]
[15.6-11]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-8]
[15.6-12]	15.6.1, Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS	See Change No. [15.6-9]
[15.6-13]	15.6.2, Failure of Small Lines Carrying Primary Coolant Outside Containment	Editorial Changes. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-14]	15.6.3, Steam Generator Tube Rupture	Editorial Changes. The analysis was revised to incorporate updates to the NSSS model and also incorporate the resolution to the containment backpressure issue. The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increase MSSV inlet piping diameter (increased 1.2 inches), increased rod drop time for the Safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-15]	15.6.3.3 Radiological Consequences (SGTR)	It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. Doses were updated based on the revised analyses.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-16]	15.6.5.3 LOCA (Radiological Consequences Only)	Editorial Changes. The analyses are based on a 1% power measurement uncertainty. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. Doses and limiting 2-hour intervals updated based on revised source terms for the Advanced First Core.
[15.6-17]	15.6.5.4A, Large-break LOCA Analysis Methodology and Results	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, the updated reactor coolant pump flywheel material and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-18]	15.6.5.4A, Large-break LOCA Analysis Methodology and Results	Reference 3 (AP600 SER) was added since Advanced Plant specific restrictions which were originally identified in the AP600 SER, and were carried to the AP1000 SER issued in 2005 and remain valid with application of ASTRUM methodology for US licensing.
[15.6-19]	15.6.5.4A.3, Signal Logic for Large-break LOCA	Editorial changes incorporated to clarify section.
[15.6-20]	15.6.5.4A.5, Large-break LOCA Analysis Results	Reference added consistent with comment 15.6.5.4A-2
[15.6-21]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-22]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Pump delay updated consistent with current timer value and timer uncertainty.
[15.6-23]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Value updated due to the additional 1.3 seconds pump delay.
[15.6-24]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Values updated due to ASTRUM methodology.
[15.6-25]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Section reworded to present the most limiting case. ASTRUM is statistical and based on probabilities. Therefore the results can change slightly each time the spectrum transient is performed. However, the most limiting transient is always chosen.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-26]	15.6.5.4A.6, Description of AP1000 Large-Break LOCA Transient	Values updated due to ASTRUM methodology.
[15.6-27]	15.6.5.4B.1 Description of Small-break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-28]	15.6.5.4B, Small-break LOCA Analyses	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, use of the digital ΔT signal, the updated reactor coolant pump flywheel material and the updated valve, nozzle and piping pressure loss coefficients.
[15.6-29]	15.6.5.4B.1 Description of Small-break LOCA Transient	Editorial changes incorporated to clarify section.
[15.6-30]	15.6.5.4B.1 Description of Small-break LOCA Transient	Minimum value replaced with the nominal value since the ASTRUM methodology uses the range of input values. Therefore the nominal value is more representative.
[15.6-31]	15.6.5.4B.2.1 NOTRUMP Computer Code	Main feedwater flow can support a 1% uncertainty. It is permissible to only model the uncertainty associated with the calorimetric measurement. In reality the main feedwater flow measurement supports a calorimetric uncertainty of 1%.
[15.6-32]	15.6.5.4B.2.1 NOTRUMP Computer Code	Value updated consistent with the 5.3 second pump delay plus a 2 second signal processing delay.
[15.6-33]	15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding	This resistance increase is due to finalized fuel design and RCS piping design. The overall change is small from 70% to 82%.
[15.6-34]	15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection	Values updated to account for the revised pressurizer diameter and height and updated line resistance calculations.
[15.6-35]	15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection	See Change No. [15.6-34]

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-36]	15.6.5.4B.3.1 Introduction	Editorial changes incorporated to clarify section.
[15.6-37]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	See Change No. [15.6-31]
[15.6-38]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	Timer delays have been updated as a result of changes to the valve stroke time. The timer delays were updated to make the valve stroke time changes transparent to the analyses.
[15.6-39]	15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System	See Change No. [15.6-32]
[15.6-40]	15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop	See Change No. [15.6-32]
[15.6-41]	15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop	Since the PXS is not the RCS, the PXS mass should not be considered in the RCS.
[15.6-42]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to clarify which CMT is being discussed.
[15.6-43]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	See Change No. [15.6-32]
[15.6-44]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to clarify what is being depicted in the cited figure.
[15.6-45]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to provide additional clarification.
[15.6-46]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Updates of detailed line resistances causes more injection flow, or less core exit flow from ADS 1-3 could cause downcomer level to remain fairly constant during this time period.
[15.6-47]	15.6.5.4B.3.5 Direct Vessel Injection Line Break	Added to provide additional clarification.
[15.6-48]	15.6.5.4B.3.6 10-inch Cold Leg Break	See Change No. [15.6-32]

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-49]	15.6.5.4B.3.6 10-inch Cold Leg Break	Due to increased ADS-4 entrainment from increased resistance calculation shown above.
[15.6-50]	15.6.5.4B.3.6 10-inch Cold Leg Break	The predictor for the onset of core boiling (x>90%) does not occur in the updated transient, therefore this paragraph is no longer applicable.
[15.6-51]	15.6.5.4B.3.6 10-inch Cold Leg Break	Updated to reflect the results of the revised analysis.
[15.6-52]	15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)	Wording updated to provide additional clarification.
[15.6-53]	15.6.5.4B.4, Conclusions	Added to clarify that this is only applicable to small break LOCAs.
[15.6-54]	15.6.5.4B.4, Conclusions	Compilation of the integrated design changes for this analysis. Namely, RCP delay times, updated line resistances, PZR geometry change. The integrated changes were not evaluated separately, therefore it is not possible to pinpoint which change contributed to the variances, only that the analysis was done in accordance with the approved licensed methodology.
[15.6-55]	15.6.5.4B.4, Conclusions	Updated based on results of DEDVI entrainment study.
[15.6-56]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	Value updated because of IRWST initial conditions and piping conditions.
[15.6-57]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	See Change No. [15.6-41] and [15.6-56].
[15.6-58]	15.6.5.4C.2, DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case	The DCD is not an appropriate place for the Sensitivity runs provided here and have therefore been removed.
[15.6-59]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	The longer equilibration time reduces uncertainty in the equilibrium conditions.

Change No.	Chapter 15 Section 15.6	Change Summary Description
[15.6-60]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	See Change No. [15.6-41] and [15.6-56].
[15.6-61]	15.6.5.4C.3, DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation	See Change No. [15.6-41] and [15.6-56].
[15.6-62]	Tables and Figures	Tables and figures have been updated to reflect the results of the revised analysis. Unless noted below, refer to the individual sections for additional details regarding changes incorporated.
[15.6-63]	Table 15.6.2-1	A more conservative method of calculating the flashing fraction was applied. Vessel outlet temperature was used in place of vessel average temperature. This is conservative.
[15.6-64]	Table 15.6.3-1	Sequence of Events updated to reflect revised SGTR analysis
[15.6-65]	Table 15.6.3-2	SGTR Mass releases updated to reflect mass releases from revised SGTR analysis.
[15.6-66]	Table 15.6.3-3	Spike duration recalculated based on revised source terms. RCS mass updated based on revised NSSS models. Steam release duration updated based on revised analysis. Ruptured and intact SG masses data updated based on updated values modeled in the analysis. Alkali metal partition factor updated to be consistent with moisture carryover.
[15.6-67]	Table 15.6.5-2 (sheets 1 through 3)	Coolant mass updated based on revised NSSS models. Containment purge rate updated to reflect the value modeled in the analysis.
[15.6-68]	Table 15.6.5-3	Doses and limiting 2-hour intervals updated based on revised source terms for the Advanced First Core.
[15.6-69]	Figure 15.6.3-1 through 15.6.3-10	Figures are updated based on the revised SGTR analysis.
[15.6-70]	15.6.5.4A	The Large-Break LOCA section was updated in Revision 1 to address the effects of thermal conduictivity degradation as described in response to CRR-008.
[15.6-71]	15.6.5.4B	The Small-Break LOCA section was updated in Revision 1 to address a change in the assumptions used in the analysis. A discussion on the changes contained in this section are described in Section 5.
[15.6-72]	Table 15.6.5-3	The results in this table have been removed. Additional information on this change is described in Section 5.

15.6 Decrease in Reactor Coolant Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

The applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences result from the major LOCA described in subsection 15.6.5. The LOCA, chemical and volume control system letdown line break outside the containment and the steam generator tube rupture (SGTR) accidents are analyzed for radiological consequences. Other accidents described in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

Comment [B1]: [15.6-1]

15.6.1.1 Identification of Causes and Accident Description

Two types of inadvertent depressurization are discussed in this section. One covers the inadvertent operation of automatic depressurization system (ADS) valves. The other covers inadvertent opening of a pressurizer safety valve.

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An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The pressure decrease causes a decrease in power via the moderator density feedback. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer low pressure

The ADS is designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault. An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

The ADS system consists of four stages of depressurization valves. The ADS stages are interlocked. For example, Stage 1 is initiated first and subsequent stages are not actuated until previous stages have completed actuation. Each stage includes two redundant parallel valve paths with two valves in series in each path such that no single failure prevents operation of the ADS stage when it is called upon to actuate and the spurious opening of a single ADS valve does not initiate ADS flow. Since each ADS path includes two valves in series, no mechanical failure could result in an inadvertent operation of an ADS stage. The ADS Stage 4 squib valves cannot be opened while the reactor coolant system is at nominal operating pressure. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

ADS Stage 1 has a minimum opening time of 20 seconds and a maximum effective flow area of $7\ in^2$ (maximum). ADS Stages 2 and 3 have a minimum opening time of 60 seconds and a maximum effective flow area of $28\ in^2$ ().

For this analysis, multiple failures and or errors are assumed which actuate both Stage 1 ADS paths. Although ADS Stages 2 and 3 have larger depressurization valves, the opening time of the Stage 1 depressurization valves is faster. This results in a more severe reactor coolant system depressurization due to ADS operation with the reactor at power.

Inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure. Although a pressurizer safety valve is smaller than the combined two Stage 1 ADS valves, the pressurizer safety valve is postulated to open in a short time.

Analyses are presented in this section for the inadvertent opening of a pressurizer safety valve and the inadvertent opening of two paths of Stage 1 of the ADS. These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values (see Section 4.4) while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of AC power during an RCS Depressurization event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of AC power in comparison to the reactor shutdown time for an RCS Depressurization event.. The primary effect of the loss of AC power is to cause the Reactor Coolant Pumps (RCPs) to coast down. The Protection & Safety Monitoring System (PMS) includes a five second minimum delay between the reactor trip and the turbine

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trip. In addition, a three second delay between the turbine trip and the loss of offsite AC power is assumed, consistent with the discussion of Section 15.0.14. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of AC power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of AC power does not adversely impact this analysis because the plant will be shut down well before the RCPs begin to coast down.

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15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by using the computer code LOFTRAN (References 14 and 15). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. The following assumptions are made to give conservative results in calculating the DNBR during the transient:

Comment [B7]: [15.6-7]

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- Initial conditions are discussed in subsection 15.0.3. Uncertainties in initial conditions are included in the DNBR limit as discussed in WCAP-11397-P-A (Reference 16).

- A least negative moderator temperature coefficient is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

- A large (absolute value) Doppler coefficient of reactivity is used such that the resulting amount of positive feedback is conservatively high to retard any power decrease.

Plant systems and equipment necessary to mitigate the effects of reactor coolant system depressurization are discussed in subsection 15.0.8 and are listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor trip protection function is reached. This is a worst case assumption. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure prevents the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6.1-1 through 15.6.1-4. The calculated sequence of events for the inadvertent opening of a pressurizer safety valve scenario is shown in Table 15.6.1-1.

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the low pressurizer pressure reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

Prior to tripping of the reactor, the core power remains relatively constant (Figure 15.6.1-1). The minimum DNBR during the event occurs shortly after the rods begin to be inserted into the core (Figure 15.6.1-2). The DNBR remains above the design limit values as discussed in Section 4.4 throughout the transient.

The system response for inadvertent operation of the ADS is shown in Figures 15.6.1-5 through 15.6.1-8. The sequence of events is provided in Table 15.6.1-1. The system response for inadvertent operation of the ADS is very similar to that obtained for inadvertent opening of a pressurizer safety valve.

15.6.1.3 Conclusion

The results of the analysis show that the low pressurizer pressure reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR remains above the design limit defined in Section 4.4. The long-term plant responses due to a stuck-open ADS valve or pressurizer safety valve, which cannot be isolated, are bounded by the small-break LOCA analysis.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample line and the discharge line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When excess primary coolant is generated because of boron dilution operations, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume control system heat exchangers and mixed bed demineralizer. The flow leaving the containment is at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out a postulated break in this line is limited to the chemical and volume control system purification flow rate of 100 gpm. Considering the low temperature of the flow and the reduced

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