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 AP600 whose failure or malfunction causes a steam flow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from 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, along with operation of the steam generator safety valves. 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.

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

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

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 the computer program LOFTRAN (Reference 2). 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. The LOFTRAN code is modified to incorporate the specific passive safeguards system features for the AP600. A description of these modifications is presented in WCAP-14234, Revision 1 (Reference 6).

In the turbine trip analyses, that 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) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the WESTAR code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

The major assumptions used in the analysis are summarized below.

Initial Operating Conditions

The accident is 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).

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).

Reactor Control

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

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 with maximum capacity.

- No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable with minimum capacity.

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 remove core decay heat following plant stabilization.

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, high pressurizer water level, and low steam generator water level.

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. Normal reactor coolant system and engineered safety systems are not 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 Section 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.

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 15.2.3-28. The calculated sequence of events for the accident is shown in Table 15.2-1.

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

The results for these cases are shown in Figures 15.2.3-15 through 15.2.3-21. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. The pressure safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value. The transient DNBR for the case with offsite power available is shown in Figure 15.2.3-20. The DNB design basis defined in Section 4.4 is met for this case.

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 reactor coolant system flow, which is illustrated in Figure 15.2.3-21. The DNB design basis defined in Section 4.4 is met, and the minimum calculated DNBR for the case where offsite power is lost is 1.51. This case is the most limiting with respect to DNB margin of the loss of steam load cases. The pressurizer safety valves actuate in this case and maintain the reactor coolant system pressure below 110 percent of the design value. Pressurizer pressure for this case is shown in Figure 15.2.3-16. With respect to maximum reactor coolant system pressure, this case is also the most limiting for complete loss of steam load cases.

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 with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. Pressurizer pressure is shown in Figure 15.2.3-2, and the pressure within the reactor coolant system is maintained below 110 percent of the design value. The DNBR for the case with offsite power is shown in Figure 15.2.3-6, and 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 DNB transient is similar to, and bounded by, the minimum reactivity feedback case without pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met. The pressurizer pressure is shown in Figure 15.2.3-2, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

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 with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure trip function. The pressure safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value. Pressurizer pressure is shown in Figure 15.2.3-9. The transient DNBR for the case with offsite power available is shown in Figure 15.2.3-20. The DNB 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 DNB transient is similar to, and bounded by, the minimum feedback case without pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met. The pressurizer pressure is shown in Figure 15.2.3-9, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

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

Figures 15.2.3-22 through 15.2.3-28 show the transient responses with and without offsite power available. In the case with offsite power available, the reactor is tripped by the high pressurizer pressure function.

Pressurizer pressure is shown in Figure 15.2.3-23, and the pressure within the reactor coolant system is maintained below 110 percent of the design value. The DNBR for the case with offsite power is shown in Figure 15.2.3-27, and 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 DNB transient is similar to, and bounded by, the minimum feedback case without pressurizer spray and without offsite power. The DNB design basis defined in Section 4.4 is met. The pressurizer pressure is shown in Figure 15.2.3-23, and the pressure within the reactor coolant system is maintained below 110 percent of the design value.

WCAP-7769, Revision 1 (Reference 1), presents additional results of analysis for a complete loss of heat sink, including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Conclusions

Results of the analyses, including those in Reference 1, 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 design limit at any time during the transient. Thus, the departure from nucleate boiling design basis, as described in Section 4.4, is met.

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

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.

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.

- 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.

- The onsite standby power system, if available, supplies ac power to the selected plant nonsafety loads.

- If startup feedwater is not available, the PRHR heat exchanger is actuated. The PRHR heat exchanger transfers the core decay heat and sensible heat to the in-containment refueling water storage tank (IRWST) and provides an uninterrupted core heat removal capability following loss of normal and startup feedwater.

The startup feedwater system, if available, is started automatically when low levels occur in either steam generator.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system. 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 (in about 5 hours), 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.

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 can also result in a loss of normal feedwater.

Following the reactor coolant pump coastdown caused by the loss of ac power, the natural circulation capability of the reactor coolant system removes residual and decay heat from the core, using the PRHR heat exchanger. 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.

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

Two analyses are performed for the loss of ac power to plant auxiliaries. The first analyses examine 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 30,000 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. This analysis will be referred to as the "decay heat removal analysis" in the following methodology discussions.

The second analysis demonstrates the ability of the protection and safety monitoring system to detect the event and trip the reactor such that the predicted DNBR is always greater than the design limit defined in Section 4.4. This analysis will be referred to as the "DNB analysis" in the following methodology discussions.

Decay Heat Removal Analysis

An analysis using a modified version of the LOFTRAN code (Reference 2), described in WCAP-14234 (Reference 6), is performed 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.

The transient response of the plant following a loss of ac power to plant auxiliaries is similar to the loss of normal feedwater flow accident (see subsection 15.2.7), except that power is assumed to be lost to the reactor coolant pumps at the time of the reactor trip.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design power rating with initial reactor coolant temperature 7°F below the nominal value and the pressurizer pressure 50 psi below 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 steam generator low level (narrow range). Offsite power is assumed to be lost at the time of reactor trip. This is more conservative than the case in which offsite power is lost at time zero because of the lower steam generator water mass at the time of the reactor trip.
- 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 (wide range).
- Conservative PRHR heat exchanger heat transfer coefficients (low) associated with the low flow rate caused by the reactor coolant pump trip are assumed.
- For the loss of ac power to the station auxiliaries, the only safety function required is core decay heat removal. That is accomplished by 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.
- Secondary system steam relief is achieved through the steam generator safety valves.
- 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 a minimum heat transfer capability. No single active failure prevents operation of any system required to function.

DNB Analysis

A combination of three computer codes is used to perform the DNB analysis. First, the LOFTRAN code (References 2 and 6) is used to calculate the plant system response.

LOFTRAN is used to calculate the core flow, the time of reactor trip, the nuclear power, and reactor coolant system fluid temperatures. The FACTRAN code (Reference 7) is then used to calculate the core heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (see Section 4.4) is used to calculate the DNBR during the transient, based on heat flux from FACTRAN, and flow and fluid temperature from LOFTRAN.

The assumptions used in the DNB analysis are the same as those used in the decay heat removal analysis except for the following:

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

- As was done in the decay heat removal analysis, the event is initiated with a complete loss of feedwater. The ac power to the reactor coolant pumps is assumed to be lost following the reactor trip and subsequent turbine trip. This is consistent with the loss of offsite power assumption used in the decay heat removal analysis and is consistent with the consequential loss of ac power assumptions discussed in subsection 15.0.14.

  If the event is initiated as a loss of offsite power at the start of the event, such that a complete loss of feedwater flow and a coastdown of the reactor coolant pumps occurs at the start of the event, then the calculated DNBR transient will be the same as is predicted for the complete loss reactor coolant system flow event as presented in subsection 15.3.2.

15.2.6.2.2 Results

Decay Heat Analysis

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 15.2.6-11. The calculated sequence of events for this event is listed in Table 15.2-1.

The LOFTRAN code results show that the natural circulation flow and the PRHR system are sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

Immediately following the reactor trip, the heat transfer capability of the PRHR heat exchanger and the steam generator heat extraction rate are sufficient to slowly cool down the plant. The cooldown continues until a low T_{cold} "S" signal is reached at approximately 500 seconds. The "S" signal actuates the core makeup tanks. During this transient, the core makeup tanks operate in water recirculation mode. The cold borated water injected by the core makeup tanks accelerates the cooldown of the plant. The core makeup tank flow slowly decreases as the core makeup tank fluid temperature increases due to water recirculation.

As the plant cools down, the heat removal capacity of the PRHR heat exchanger is lowered. At approximately 3,500 seconds, the heat removal rate from the reactor coolant system, due to the core makeup tank injection and the PRHR heat exchanger, decreases below the core decay heat produced. The reactor coolant system then begins heating up again. As the reactor coolant system temperature is elevated, the heat removal capacity of the PRHR heat exchanger increases. The reactor coolant system temperature slowly increases until the heat removal rate of the PRHR heat exchanger matches the core decay heat produced. This occurs at approximately 19,800 seconds.

Pressurizer safety valves open to discharge steam to containment and reclose later in the transient when the heat removal rate of the PRHR heat exchanger exceeds the decay heat production rate.

The capacity of the PRHR heat exchanger is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.6-5 and 15.2.6-6, in the long-term the plant starts a slow cooldown driven by the PRHR heat exchanger. Plant procedures may be followed to further cool down the plant.

DNB Analysis

The DNBR, following a loss of ac power to plant auxiliaries, is shown in Figure 15.2.6-12. The calculated sequence of events for the analysis is listed in Table 15.2-1. The minimum DNBR occurs shortly after the rods begin to insert in the core at 87.2 seconds. When the reactor coolant flow begins decreasing at 89.7 seconds, the DNBR continues to increase because reactor trip has already been initiated and core heat flux has started decreasing. The DNB design basis defined in Section 4.4 is met.

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. Because DNBR remains above the design limit values, the core is not adversely affected. PRHR heat exchanger capacity 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 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.

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.

- 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-low steam generator water level (wide range) signal. The PRHR heat exchanger transfers the core decay heat and sensible heat to the IRWST so that core heat removal is uninterrupted following a loss of normal and startup feedwater.

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

The reactor trip on low narrow range water level in either steam generator provides the necessary protection against a loss of normal feedwater.

The startup feedwater system is started automatically, as discussed in subsection 15.2.6.1. If startup feed is unavailable, the PRHR heat exchanger is actuated as discussed in subsection 15.2.6.

An analysis of the system transient is presented below to show that, following a loss of normal feedwater, the PRHR heat exchanger is capable of removing the stored and decay heat to prevent either overpressurization of the reactor coolant system or loss of water from the reactor coolant system.

15.2.7.2 Analysis of Effects and Consequences

15.2.7.2.1 Method of Analysis

An analysis using a modified version of the LOFTRAN code (Reference 2), described in WCAP-14234 (Reference 6), is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant 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.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design power rating.
- Reactor trip occurs on steam generator low (narrow range) level.
- The only safety function required is the core decay heat removal that is carried by the PRHR heat exchanger; therefore, 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-low steam generator water level wide range signal.
- Secondary system steam relief is achieved through the steam generator safety valves.
- The initial reactor coolant average temperature is 6.5°F higher than the nominal value, and initial pressurizer pressure is 50 psi lower than nominal.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the protection and safety monitoring system and the PRHR heat exchanger in removing long-term decay heat and preventing 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.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value and the reactor trips via the low steam generator narrow range level trip. The reactor coolant pumps continue to run until automatically tripped when the core makeup tanks are actuated.

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 a minimum heat transfer capability. No single active failure prevents operation of any system to perform its required function. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.2.7.2.2 Results

Figures 15.2.7-1 through 15.2.7-10 show the significant plant parameters following a loss of normal feedwater.

Prior to reactor trip and the insertion of the rods into the core, the loss of normal feedwater transient is the same as the transient response presented in subsection 15.2.6 for the loss of ac power to plan auxiliaries. The DNB results, presented in Figure 15.2.6-12 for the loss of ac power to plant auxiliaries, are also applicable for a loss of normal feedwater and demonstrate that the DNB design basis is met.

Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction. Steam flow through the safety valves continues to dissipate the stored and core decay heat.

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 pressure decreases.

The cooldown continues until a low T_{cold} "S" signal is eventually reached at 1,069.7 seconds. The "S" signal actuates the core makeup tanks. During this transient, the core makeup tanks operate in water recirculation mode. The cold borated water injected by the core makeup tanks accelerates the cooldown of the plant. The core makeup tank flow slowly decreases as the core makeup tank fluid temperature increases due to water recirculation.

As the plant cools down, the heat removal capacity of the passive residual heat exchanger is lowered. At approximately 3,000 seconds, the heat removal rate from the reactor coolant system, due to the core makeup tank injection and the PRHR heat exchanger, decreases below the core decay heat produced. The reactor coolant system then begins heating up again. As the reactor coolant system temperature is elevated, the heat removal capacity of the PRHR heat exchanger increases again. The reactor coolant system temperature slowly increases until the heat removal rate of the PRHR heat exchanger matches the core decay heat produced. This occurs at approximately 22,000 seconds.

The capacity of the PRHR heat exchanger is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.7-3 and 15.2.7-4, the plant starts a slow cooldown driven by the PRHR heat exchanger. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger is 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.

15.2.8 Feedwater System Pipe Break

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.

The PRHR heat exchanger functions to:

- 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.

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-9320 (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 presented because, due to the fast core makeup tanks actuation (on an "S" signal generated by the low steam line pressure), the reactor coolant pumps are tripped by the protection and safety monitoring system a few seconds after the reactor trip. The only difference between the cases with and without offsite power available is the operating status of the reactor coolant pumps.

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

- A reactor trip on any of the following four conditions:
  - High pressurizer pressure
  - Overtemperature ΔT
  - 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 subsections 7.1 and 7.2 for a description of the actuation system.

- The PRHR heat exchanger provides 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.

  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-14234 (Reference 6), of the LOFTRAN code (Reference 2) is performed to determine the plant transient following a feedwater line rupture. The code describes the plant 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 102 percent of the design plant rating.

- Initial reactor coolant average temperature is 6.5°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.

- No credit is taken for the high pressurizer pressure reactor trip.

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

- After reactor trip, the faulted steam generator blows down through a double-ended break area of 1.12 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.

- Reactor trip is assumed to be initiated when the low steam generator narrow range level setpoint is reached on the ruptured steam generator.

- The PRHR heat exchanger is actuated by the low steam generator water level (wide range) signal. A 20-second delay is assumed following the low level signal to allow time for the alignment of PRHR heat exchanger valves.

- No 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 level
  - High containment pressure

The PRHR heat exchanger is initiated if 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.

The plant control system is not assumed to function in order 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.

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).

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.

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.

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 pressure. Pressurizer pressure decreases after reactor trip on the low steam generator water level (83.1 seconds) due to the loss of heat input.

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). The DNB results, presented in Figure 15.2.6-12 for the loss of ac power to plant auxiliaries, are also applicable to a feedwater system pipe break and demonstrate that the DNB design basis is met.

After the trip, the core makeup tanks are actuated (111.3 seconds) 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 (107.1 seconds).

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 again 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.

Reactor coolant system temperatures are low (approximately 475°F at 3,000 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 5 hours, the heat transfer capability of the PRHR heat exchanger exceeds the decay heat power and the reactor coolant system temperatures, pressure, and pressurizer water volumes start to steadily decrease. Core cooling capability is maintained throughout the transient because reactor coolant system inventory is increasing due to core makeup tank injection.

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 overpressurizing the reactor coolant system, and to maintain the core cooling capability. Radioactivity doses from ruptures of the postulated 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.

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-11398-A (Nonproprietary), April 1989.

6. Bachrach, U., Carlin, E. L., "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary), August 1997.

7. Hargrove, H. G., "FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1975.

Table 15.2-1 (Sheet 1 of 7)

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
	High pressurizer pressure reactor trip point reached	7.2
	Rods begin to drop	9.2
	Minimum DNBR occurs	10.5
	Peak pressurizer pressure occurs	11.0
	Initiation of steam release from steam generator safety valves	13.0
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	3.3
	Rods begin to drop	4.1
	Peak pressurizer pressure occurs	7.3
	Initiation of steam release from steam generator safety valves	21.5

Table 15.2-1 (Sheet 2 of 7)

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 occurs	3.5
	High pressurizer pressure reactor trip setpoint reached	7.3
	Rods begin to drop	9.3
	Peak pressurizer pressure occurs	10.0
	Initiation of steam release from steam generator safety valves	13.0
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	3.3
	Rods begin to drop	4.1
	Peak pressurizer pressure occurs	8.4
	Initiation of steam release from steam generator safety valves	26.0

Table 15.2-1 (Sheet 3 of 7)

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	5.9
		Rods begin to drop	7.9
		Minimum DNBR occurs	9.0
		Peak pressurizer pressure occurs	10.0
		Initiation of steam release from steam generator safety valves	13.0
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	3.3
		Rods begin to drop	4.1
		Minimum DNBR occurs	6.1
		Peak pressurizer pressure occurs	7.4
		Initiation of steam release from steam generator safety valves	22.0

Table 15.2-1 (Sheet 4 of 7)

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
	Minimum DNBR occurs	4.0
	High pressurizer pressure reactor trip	5.8
	Rods begin to drop	7.8
	Peak pressurizer pressure occurs	9.5
	Initiation of steam release from steam generator safety valves	13.0
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	3.3
	Rods begin to drop	4.1
	Peak pressurizer pressure occurs	7.3
	Initiation of steam release from steam generator safety valves	26.0

Table 15.2-1 (Sheet 5 of 7)

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 (decay heat removal analysis)	Feedwater is lost	10.0
	Low steam generator water level reactor trip setpoint is reached	84.1
	Rods begin to drop, ac power is lost, reactor coolant pumps start to coastdown	86.1
	Pressurizer safety valves open	89.1
	Maximum pressurizer pressure reached	90.5
	Steam generator safety valves open	94.0
	Pressurizer safety valves reclose	95.9
	Maximum pressurizer water volume reached	95.7
	PRHR heat exchanger actuation on low steam generator water level (wide range)	151.1
	Core makeup tank actuation on low T_{cold} "S" signal	483.1
	Steam line isolation on low T_{cold} "S" signal	495.1
	Steam generator 1 safety valves close	665.1
	Steam generator 2 safety valves close	735.1
	Pressurizer safety valves open	2,632.0
	Pressurizer safety valves reclose	19,360.0
	Second pressurizer water volume peak is reached	19,448.0
	PRHR heat exchanger extracted heat matches decay heat	~ 19,800.0

Table 15.2-1 (Sheet 6 of 7)

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

Accident	Event	Time (seconds)
II.B. Loss of ac power to the plant auxiliaries (DNB analysis)	Feedwater is lost	10.0
	Low steam generator water level reactor trip setpoint is reached	85.1
	Turbine trip signal occurs	86.7
	Rods begin to drop	87.1
	Minimum DNBR occurs	87.2
	ac power is lost, reactor coolant pumps start to coast down	89.7
III. Loss of normal feedwater flow	Feedwater is lost	10.0
	Low steam generator water level (narrow range) reactor trip reached	83.9
	Rods begin to drop	85.9
	Steam generator safety valves open	87.6
	Pressurizer safety valves open	88.1
	Maximum pressurizer pressure reached	88.4
	Pressurizer safety valves reclose	88.9
	PRHR heat exchanger actuation on low steam generator water level (wide range)	150.9
	Steam generator safety valves reclose	182.1
	Steam line isolation on low T_{cold} "S" signal	1,081.7
	Reactor coolant pump trip on low T_{cold} "S" signal	1,084.7
	Core makeup tanks actuation on low T_{cold} "S" signal	1,091.7
	Pressurizer safety valves open	3,488
	Pressurizer safety valves reclose	22,540
	Passive residual heat removal heat exchanger extracted heat matches decay heat	~22,000
	Maximum pressurizer water volume reached	22,409

Table 15.2-1 (Sheet 7 of 7)

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 an the main feedwater control system	10.0
	Low steam generator water level (narrow range) setpoint reached	83.1
	Reverse flow from the faulted steam generator through a full double-ended rupture starts	83.1
	Rods begin to drop	85.1
	Loss of offsite power occurs	85.1
	Low steam generator water level (wide range) setpoint reached	85.1
	Pressurizer safety valves open	86.0
	Low steam line pressure setpoint reached	89.3
	Pressurizer safety valves close	96.0
	All steam and feedline isolation valves close	101.3
	PRHR heat exchanger actuation on low steam generator water level (wide range)	107.1
	Faulted steam generator empties	111.0
	Core makeup tank valves fully opened	111.3
	Intact steam generator safety valves open	140.0
	Intact steam generator safety valves reclose	768.0
	Pressurizer safety valves open	6,304.0
	PRHR heat exchanger extracted heat matches decay heat	18,500

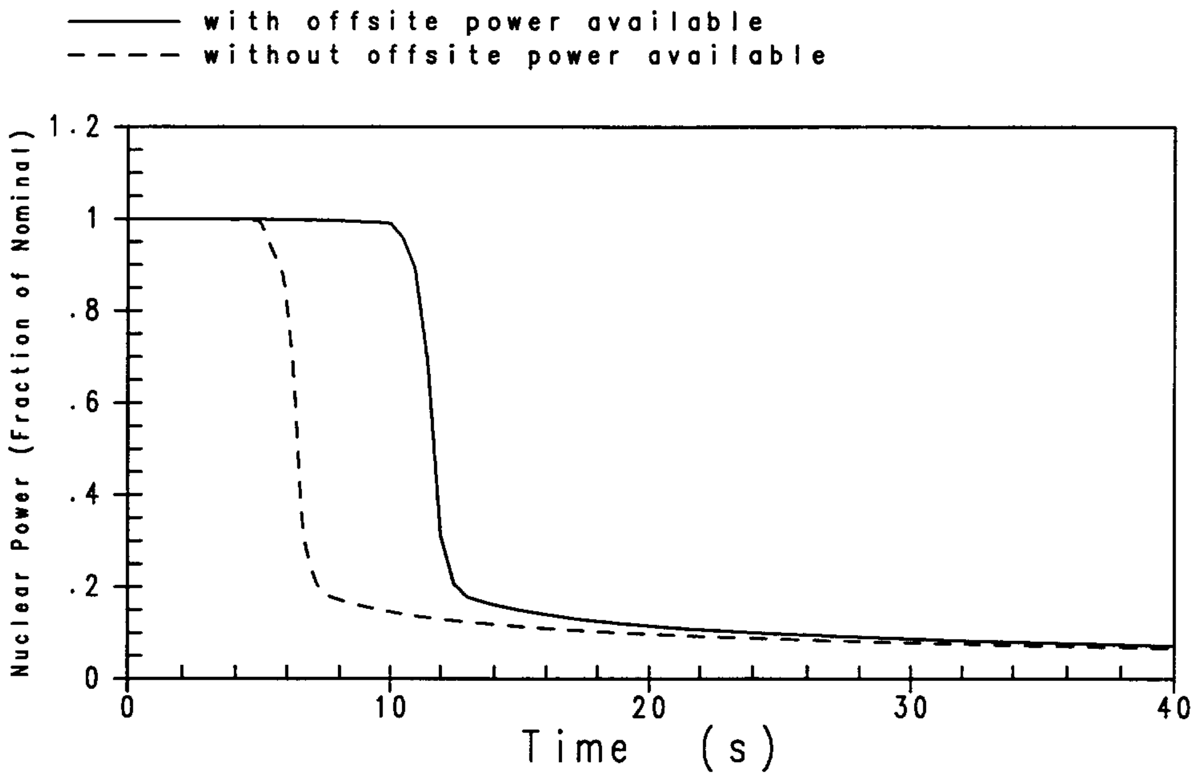

Figure 15.2.3-1

Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

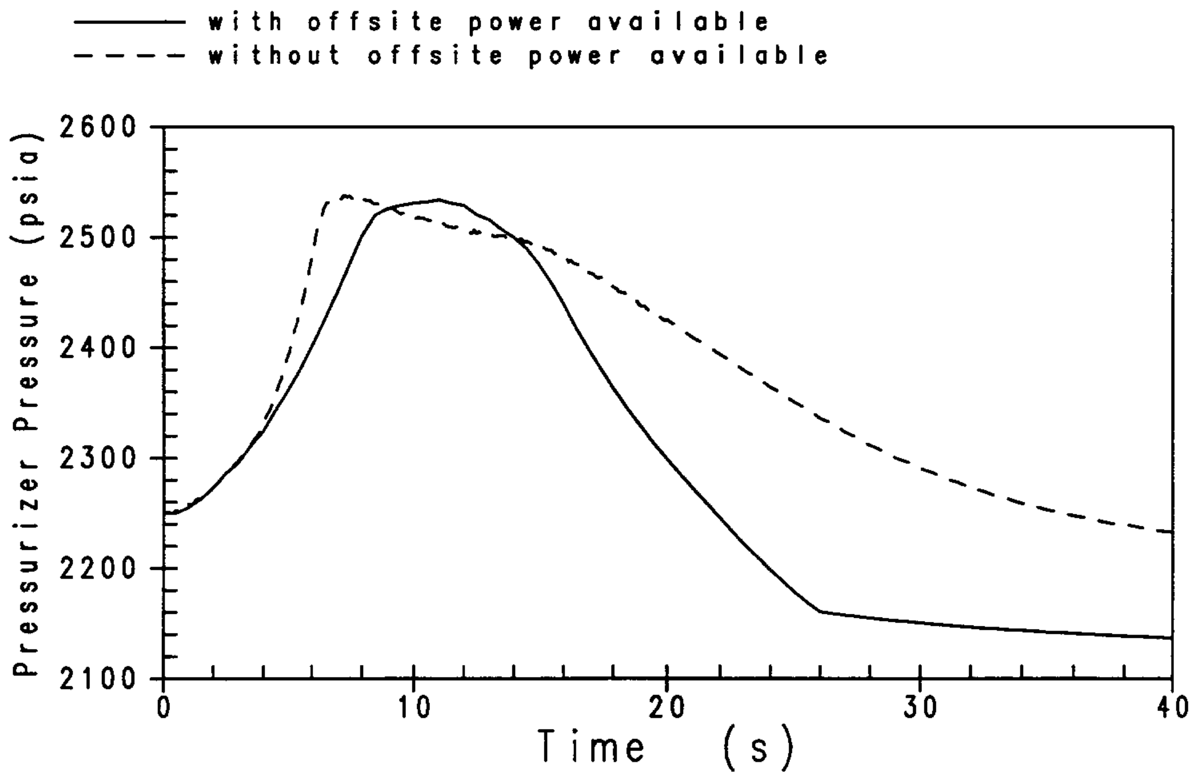

Figure 15.2.3-2

Pressurizer Pressure (psia) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

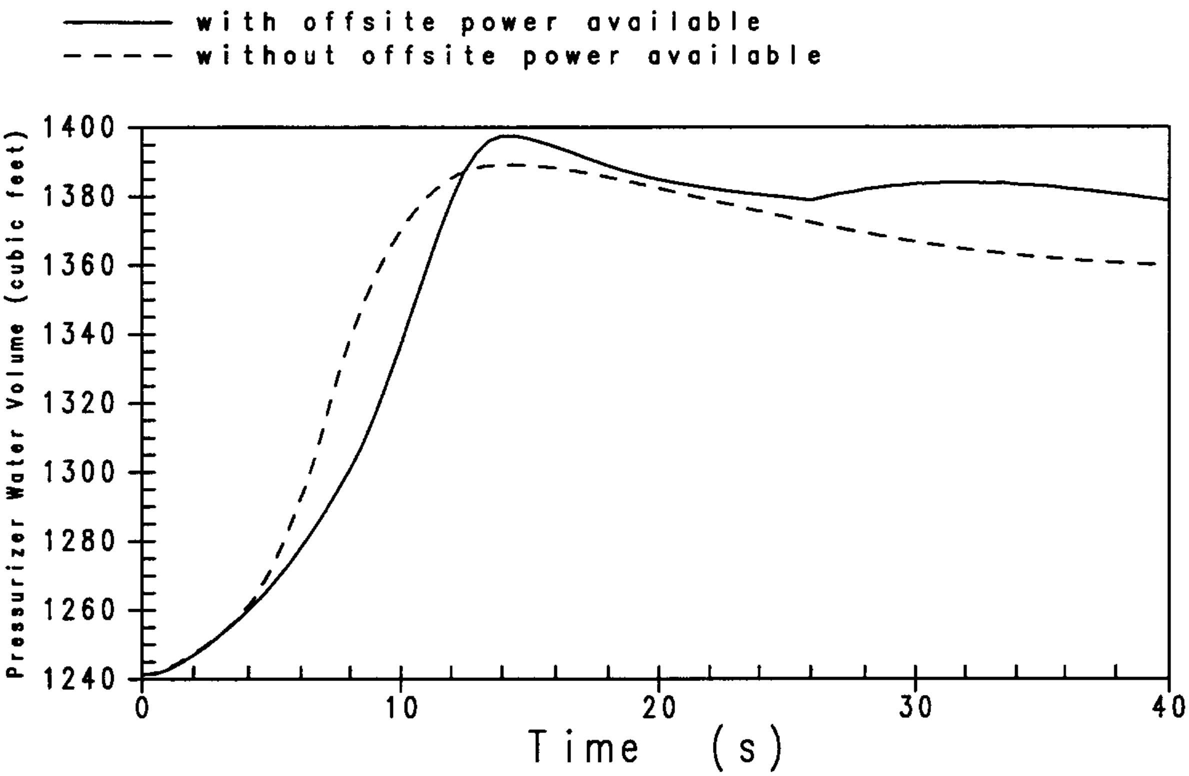

Figure 15.2.3-3

Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

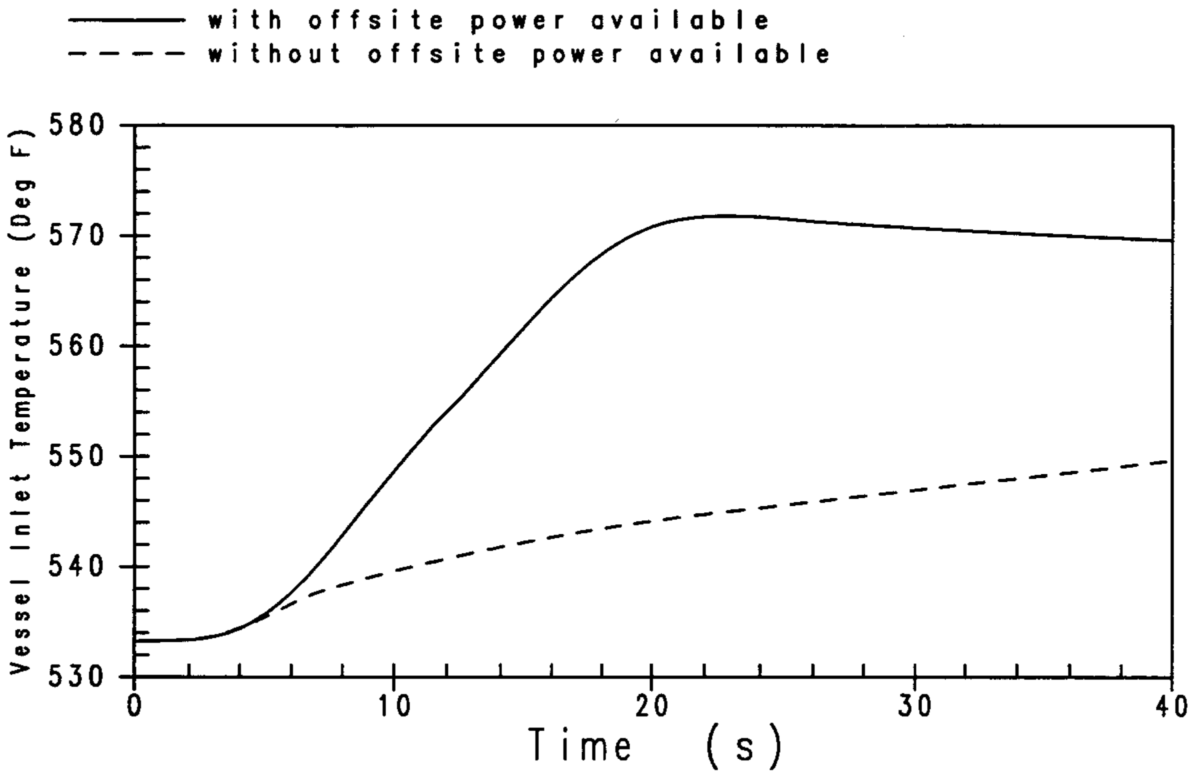

Figure 15.2.3-4

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

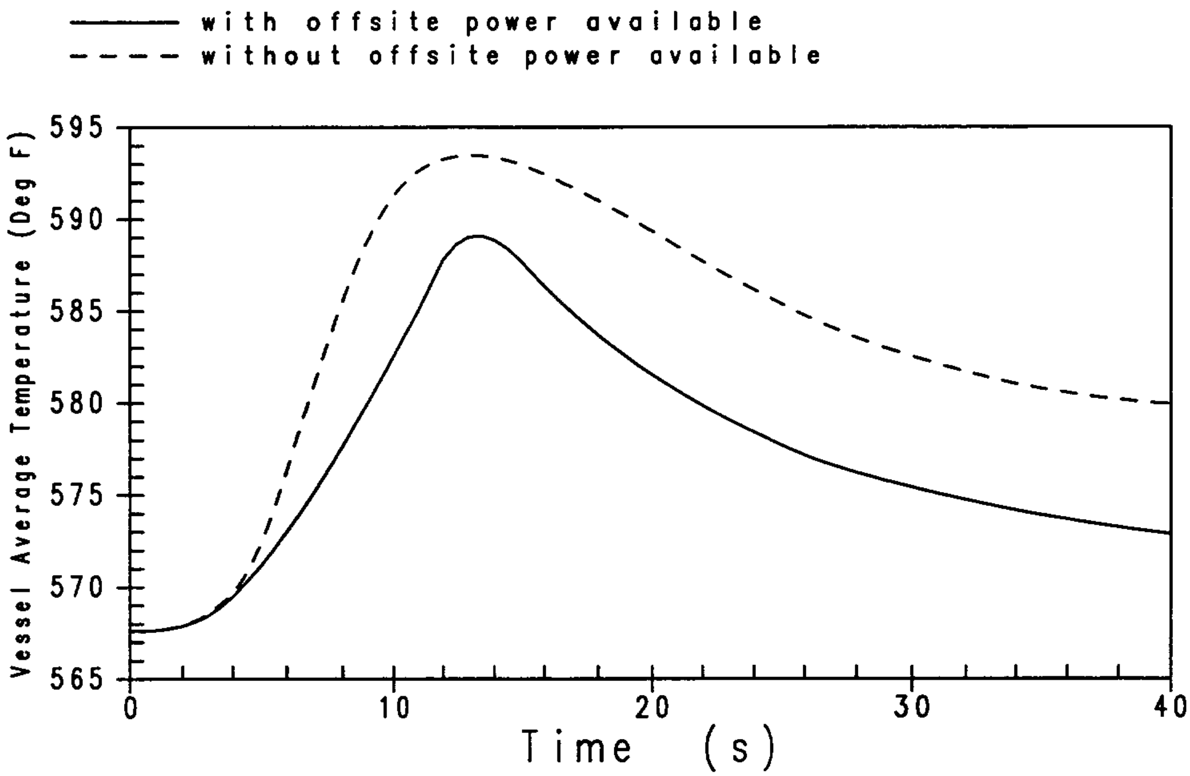

Figure 15.2.3-5

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

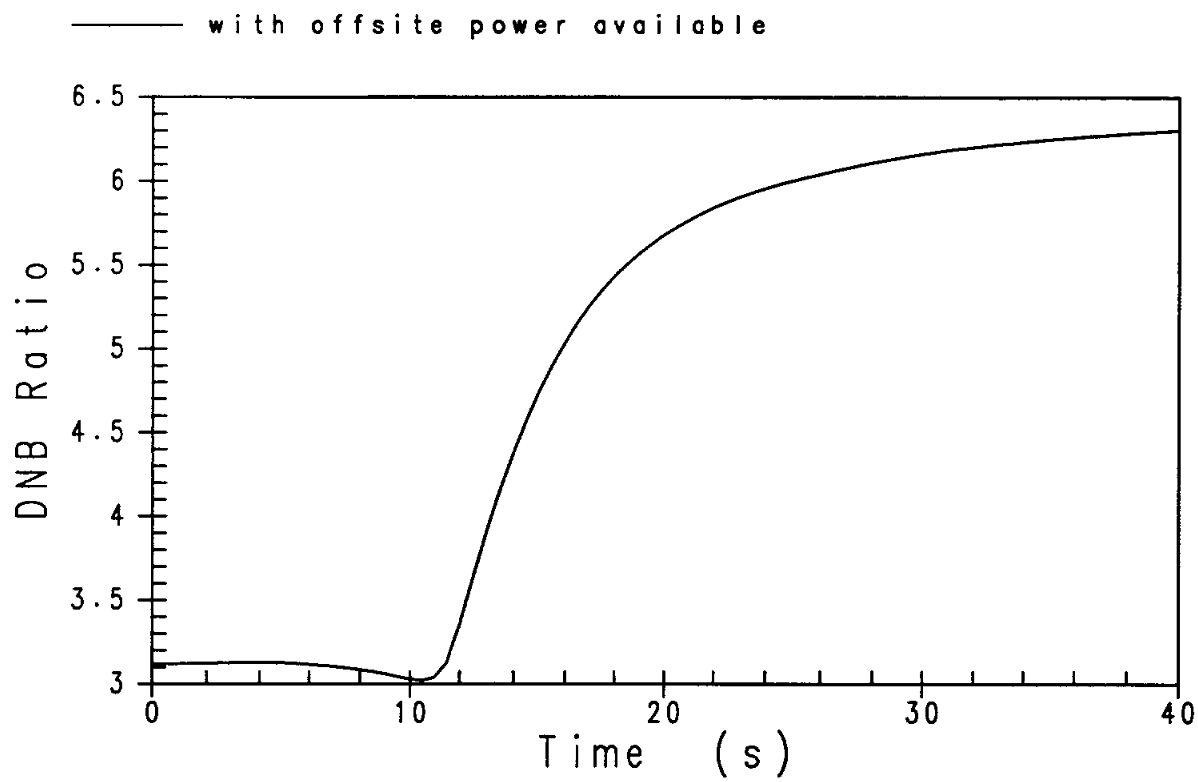

Figure 15.2.3-6

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

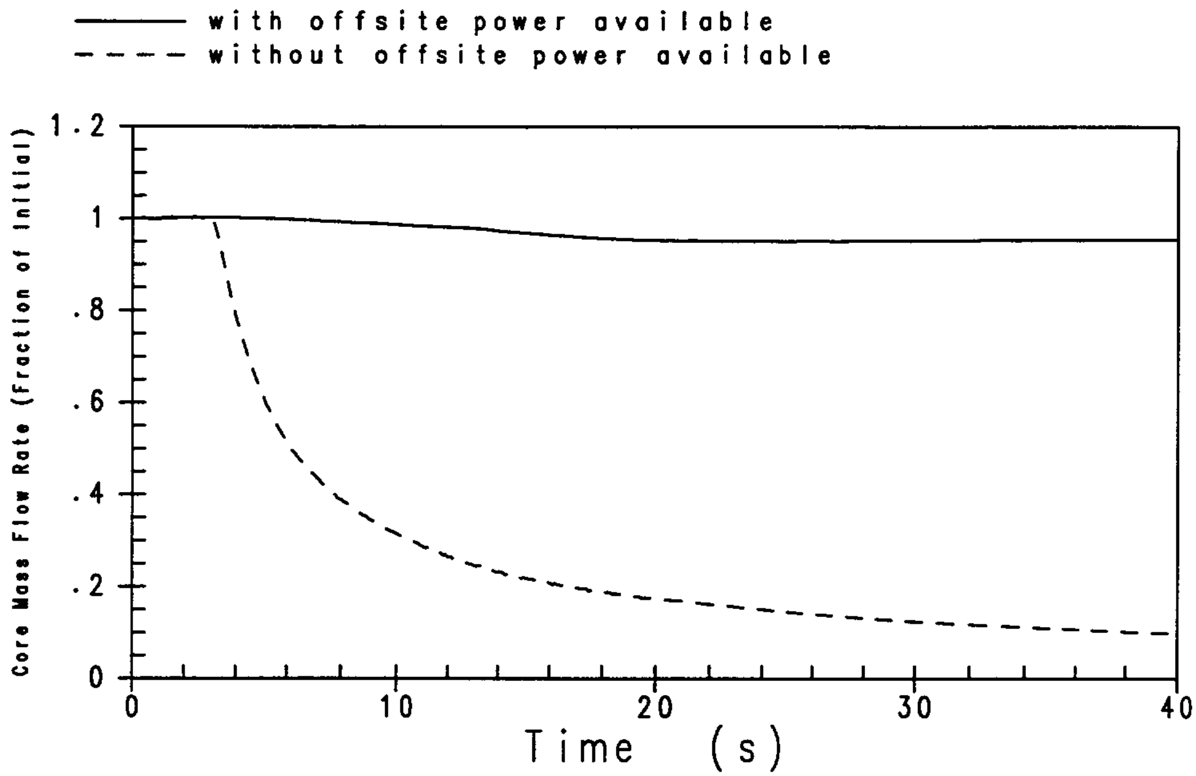

Figure 15.2.3-7

Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

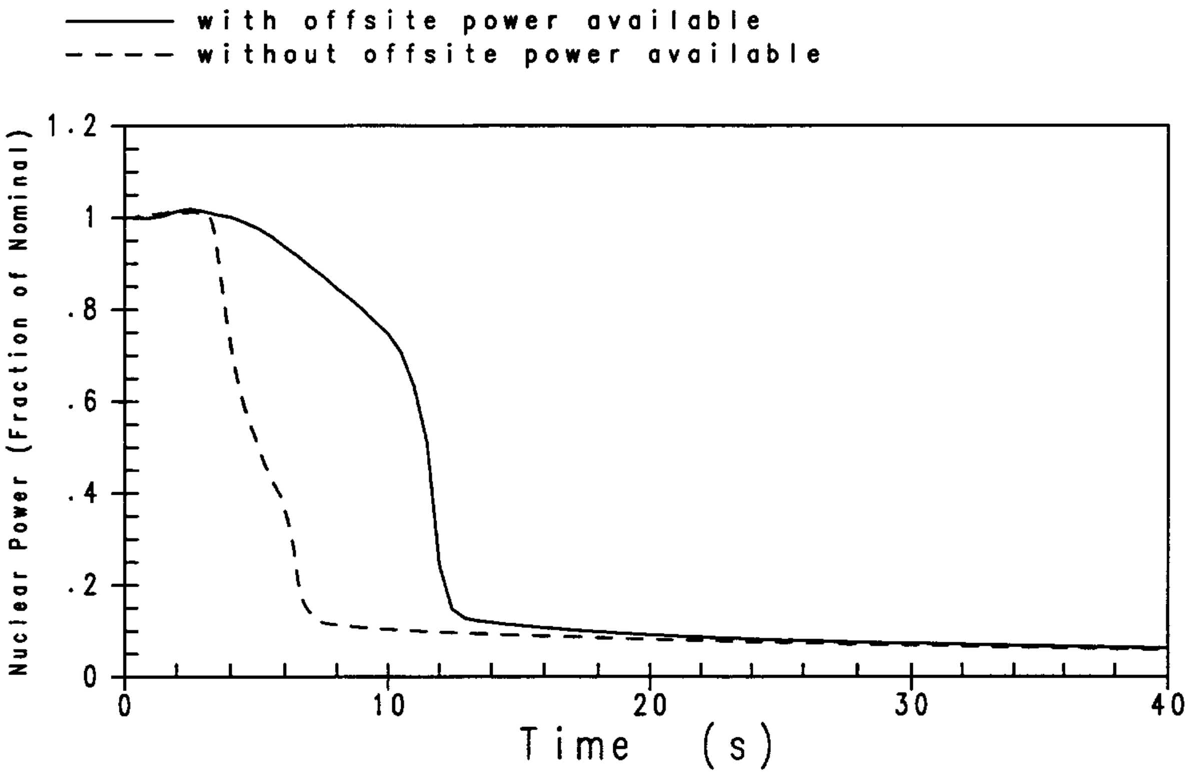

Figure 15.2.3-8

Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

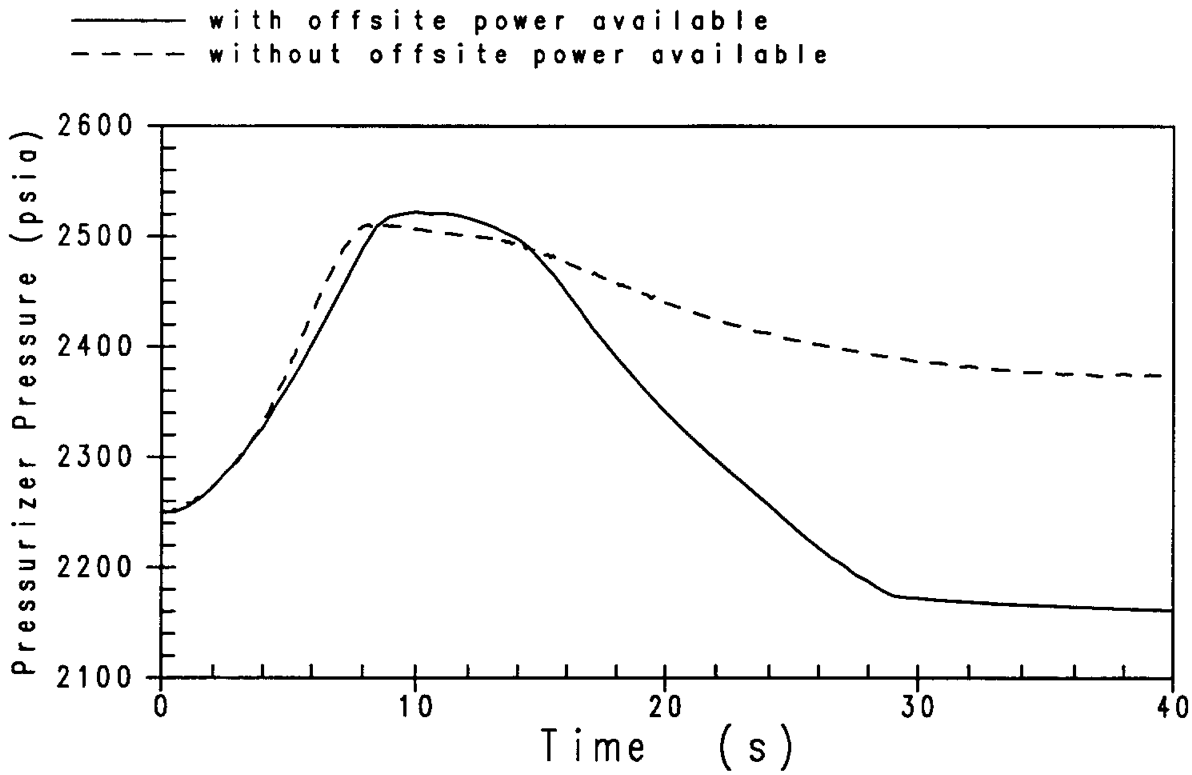

Figure 15.2.3-9

Pressurizer Pressure (psia) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

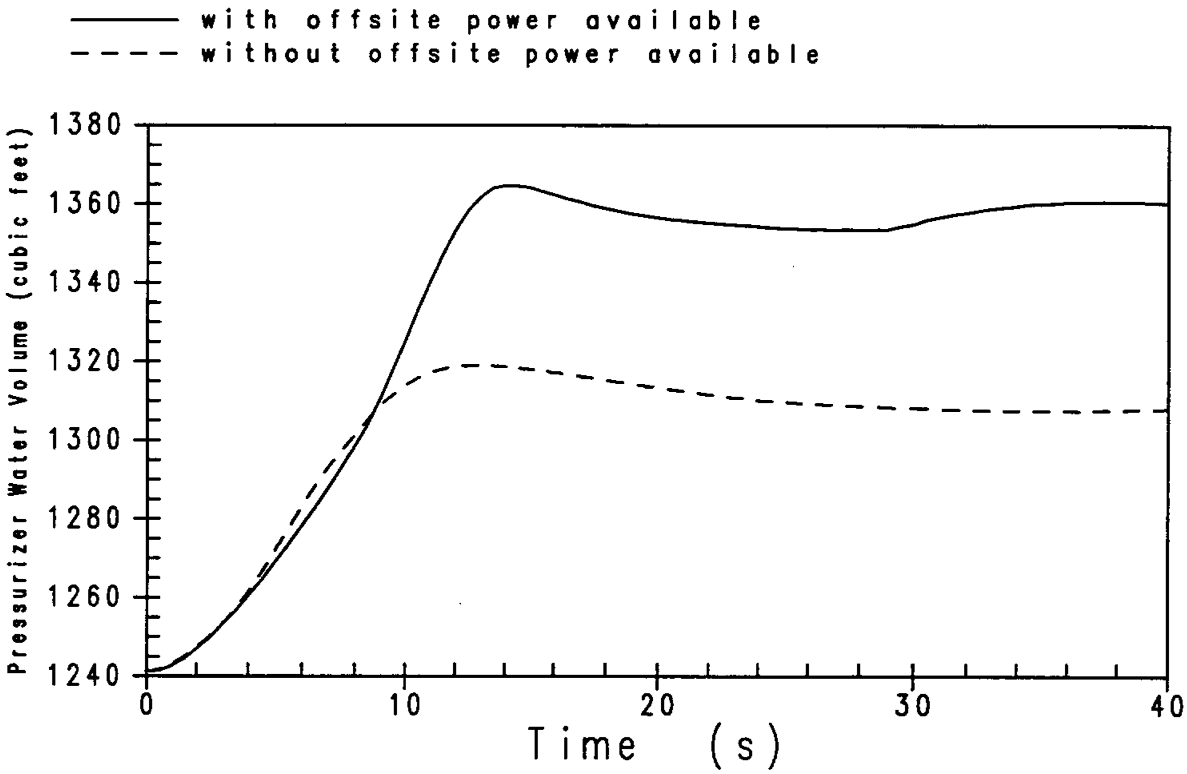

Figure 15.2.3-10

Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

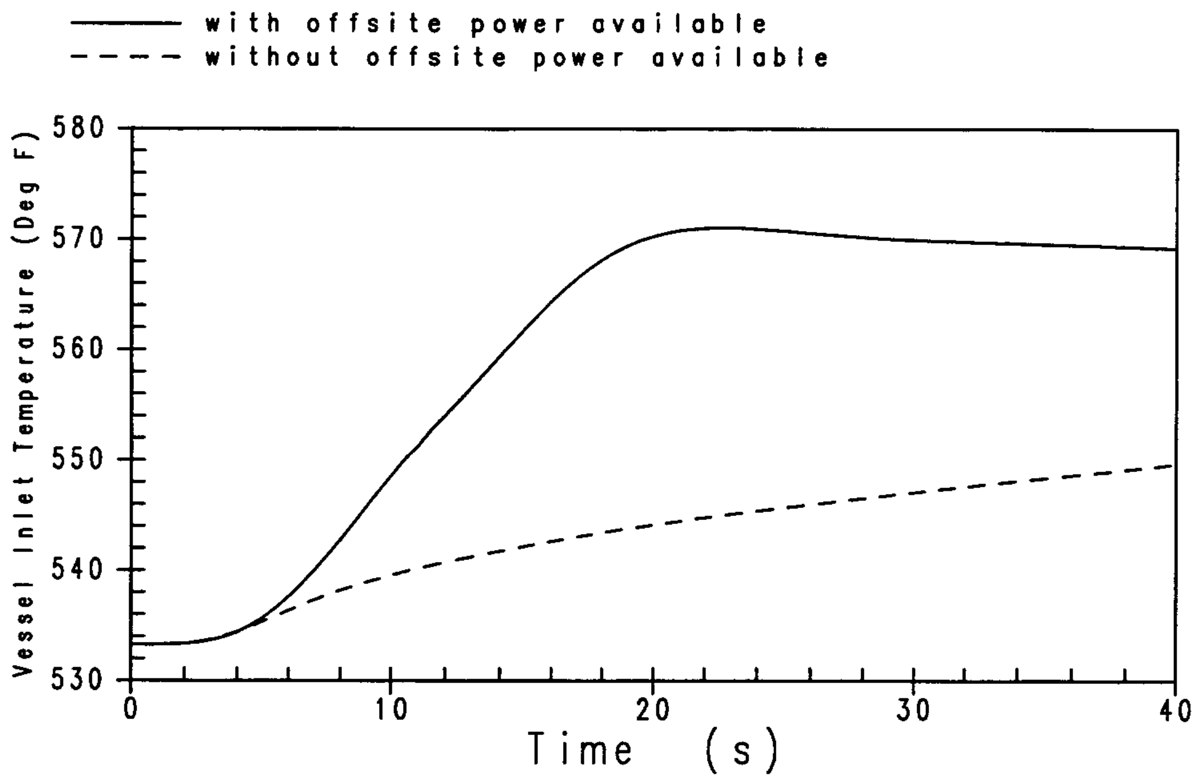

Figure 15.2.3-11

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

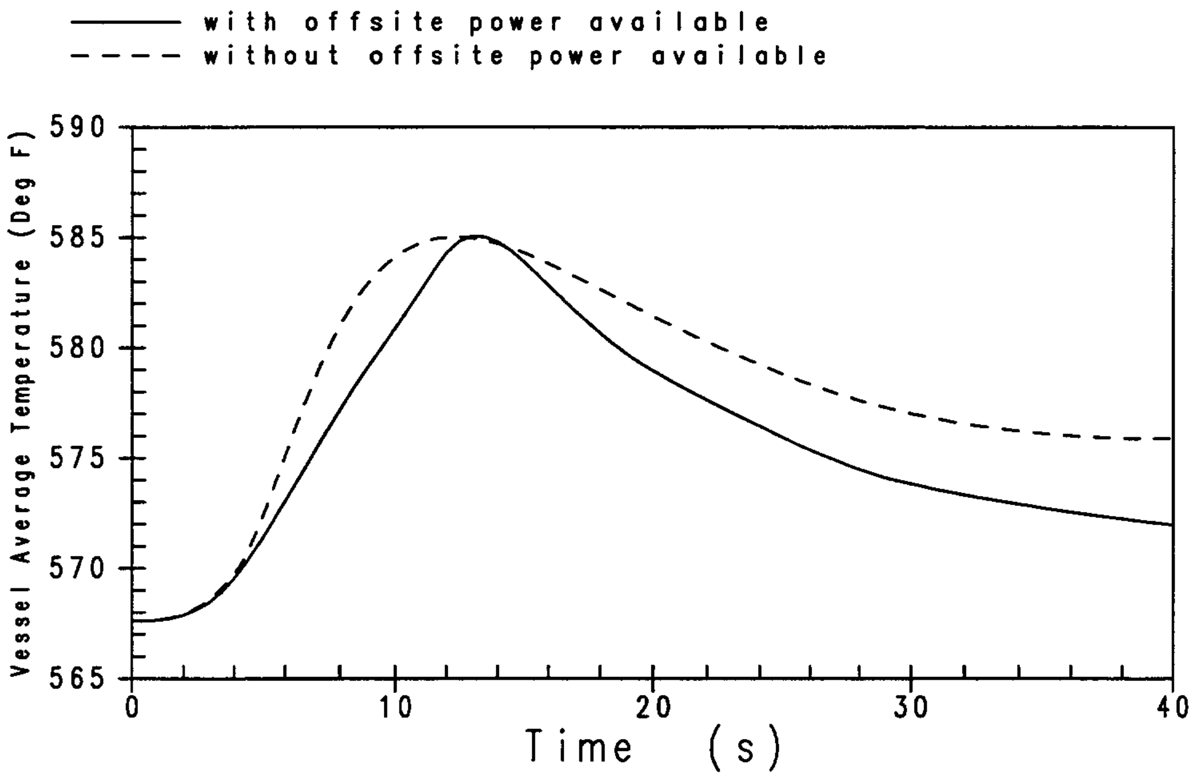

Figure 15.2.3-12

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

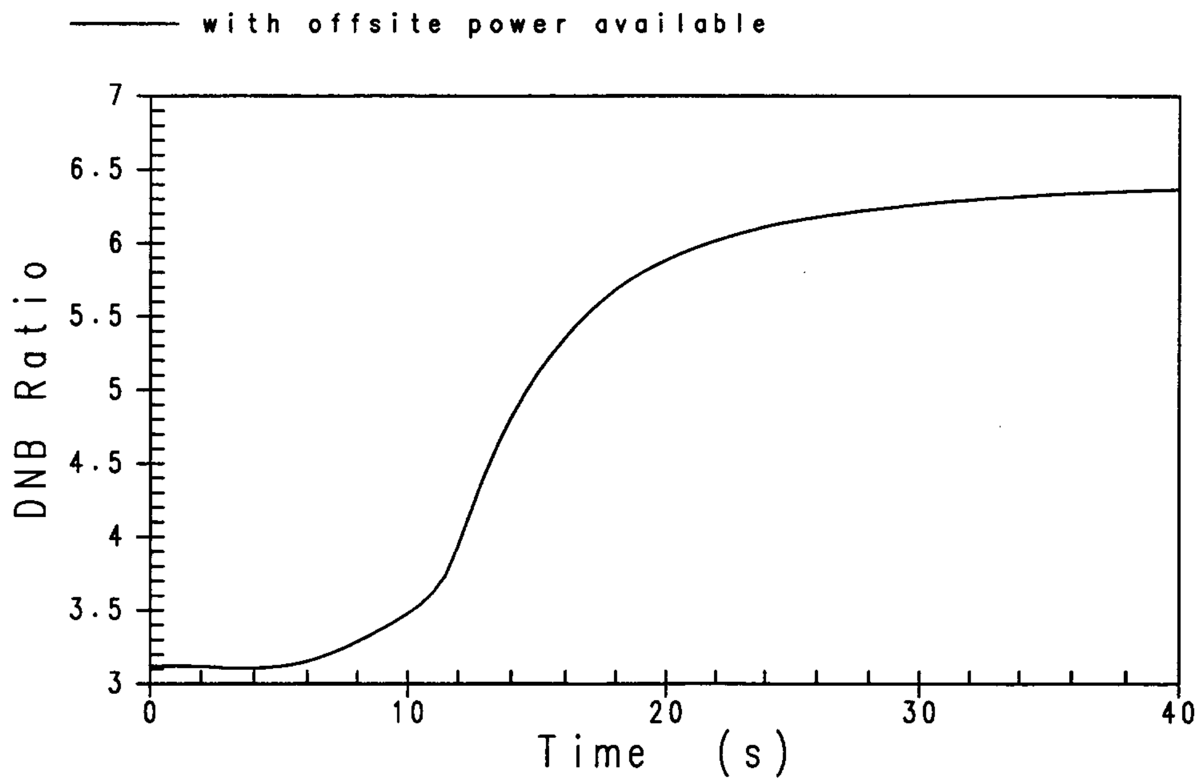

Figure 15.2.3-13

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

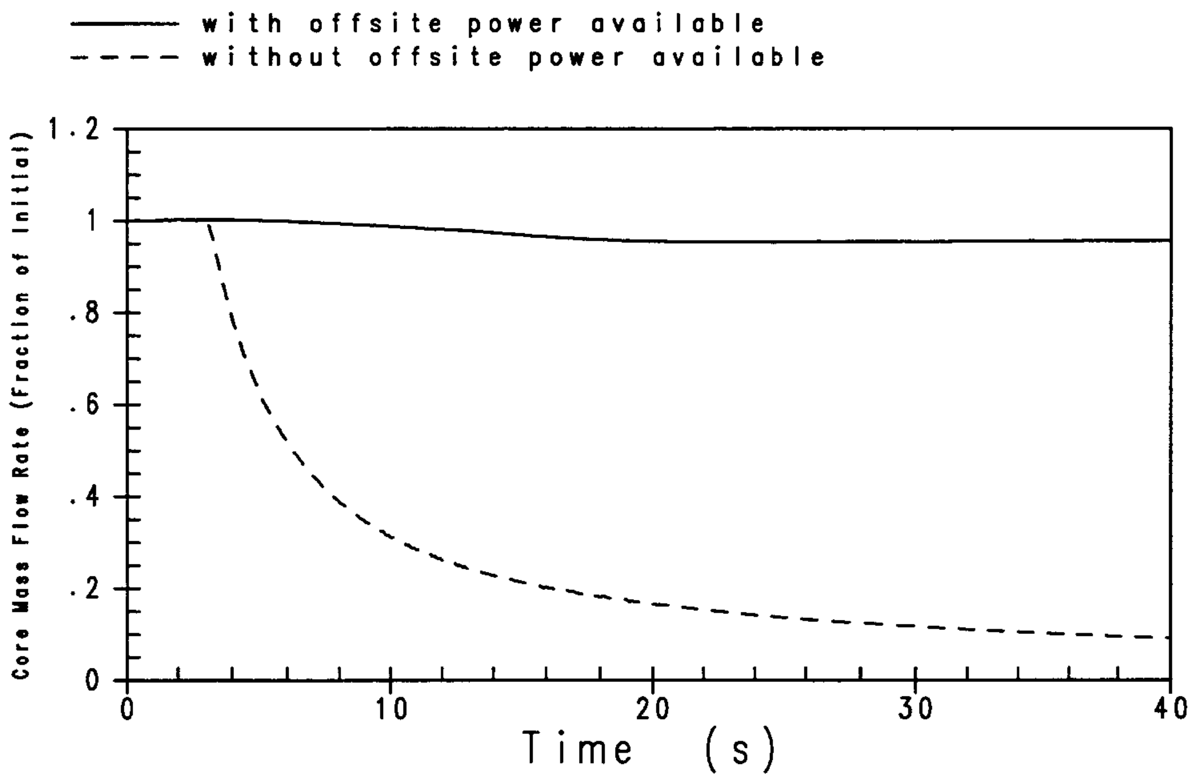

Figure 15.2.3-14

Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

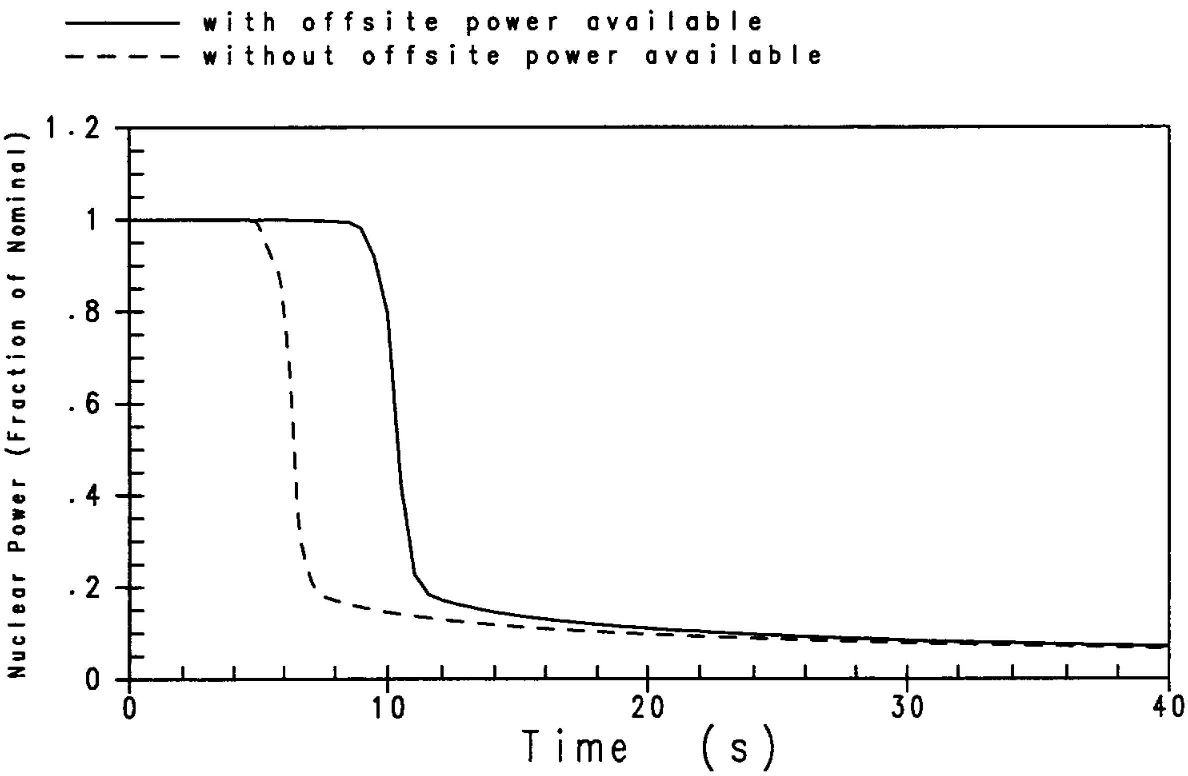

Figure 15.2.3-15

Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

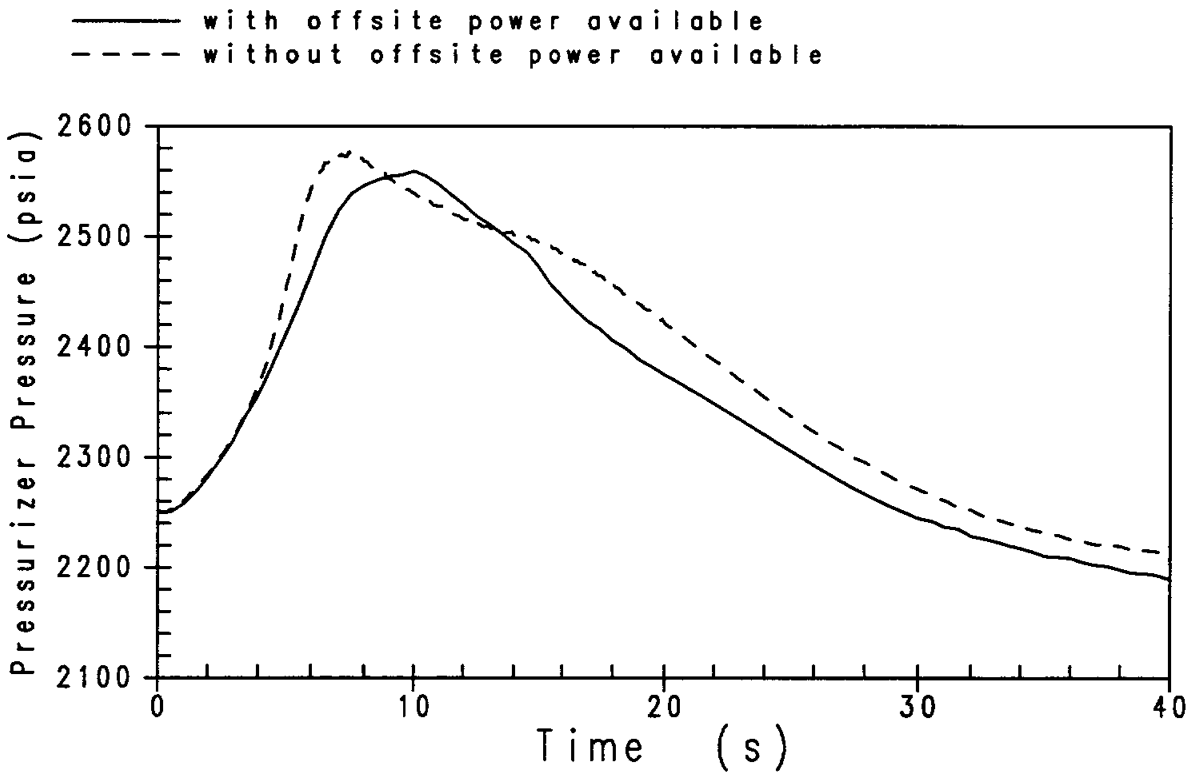

Figure 15.2.3-16

Pressurizer Pressure (psia) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

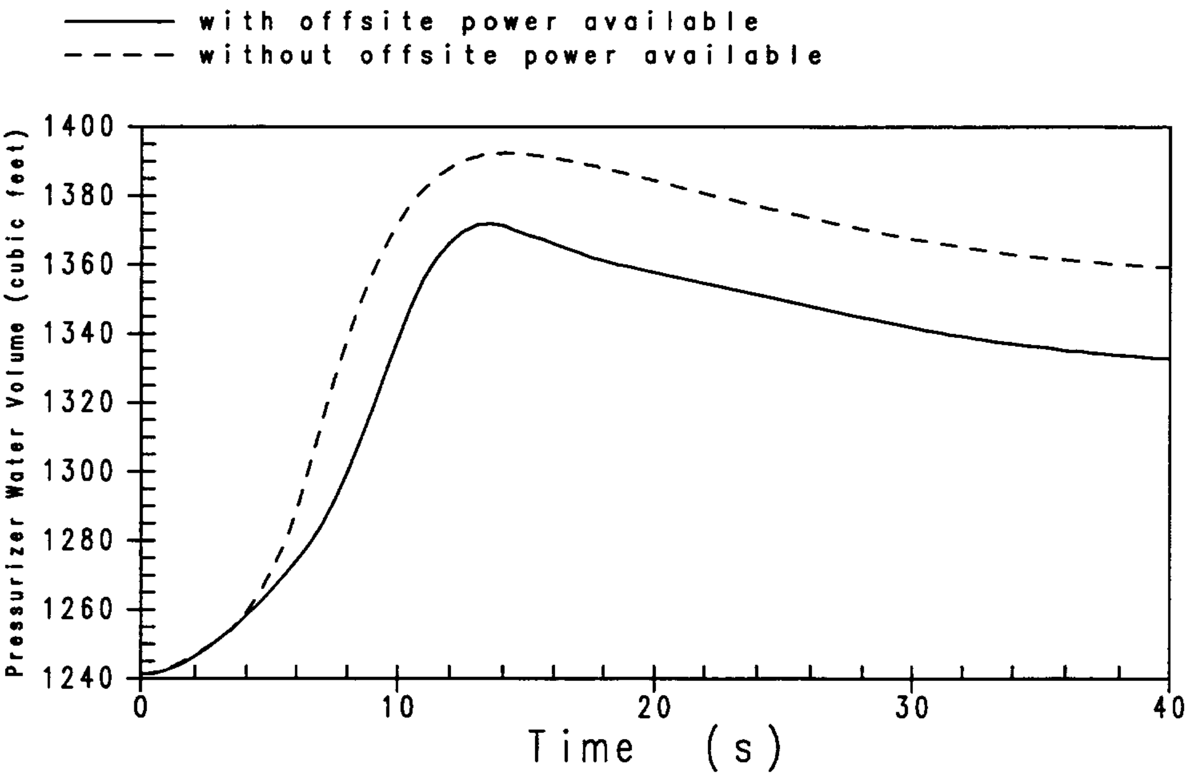

Figure 15.2.3-17

Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

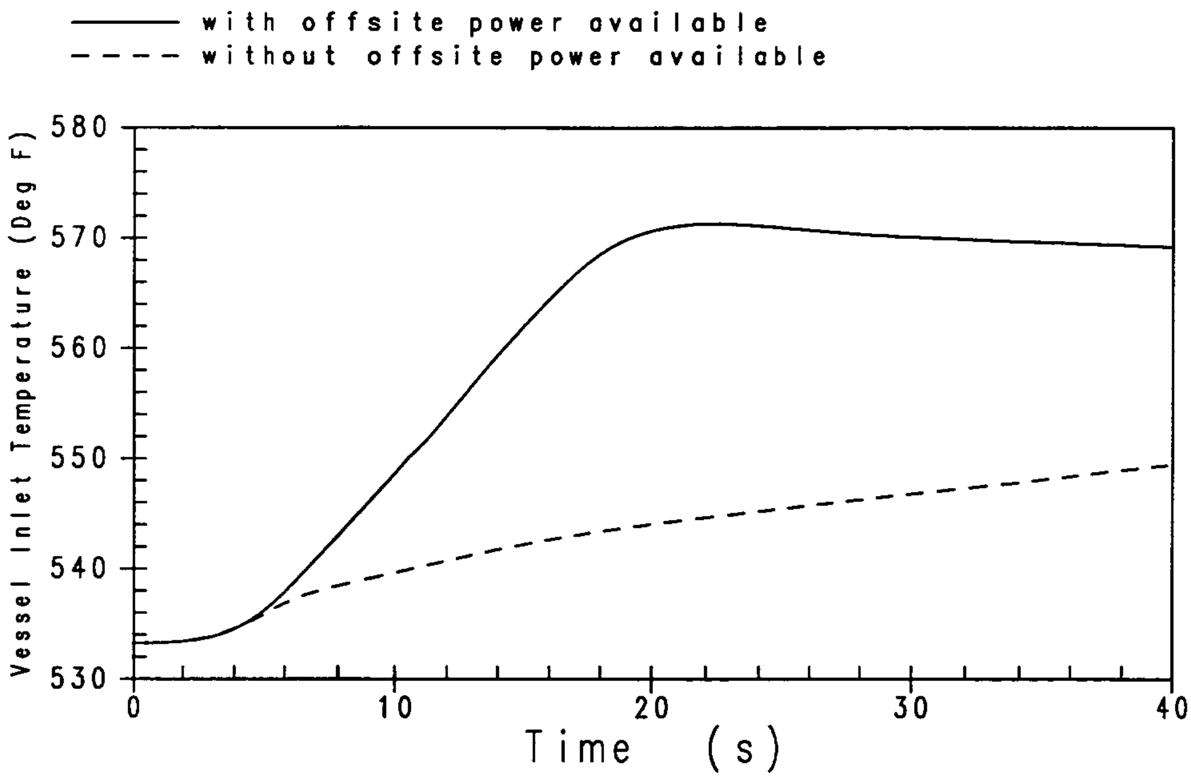

Figure 15.2.3-18

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

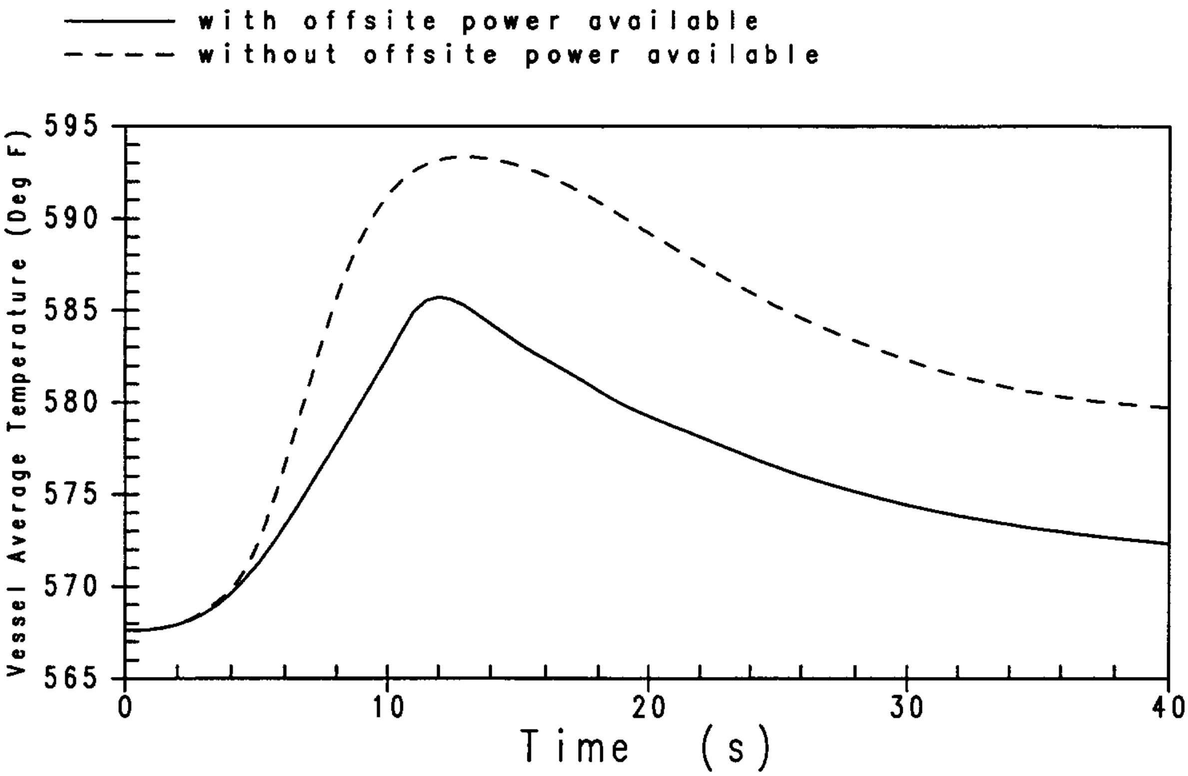

Figure 15.2.3-19

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

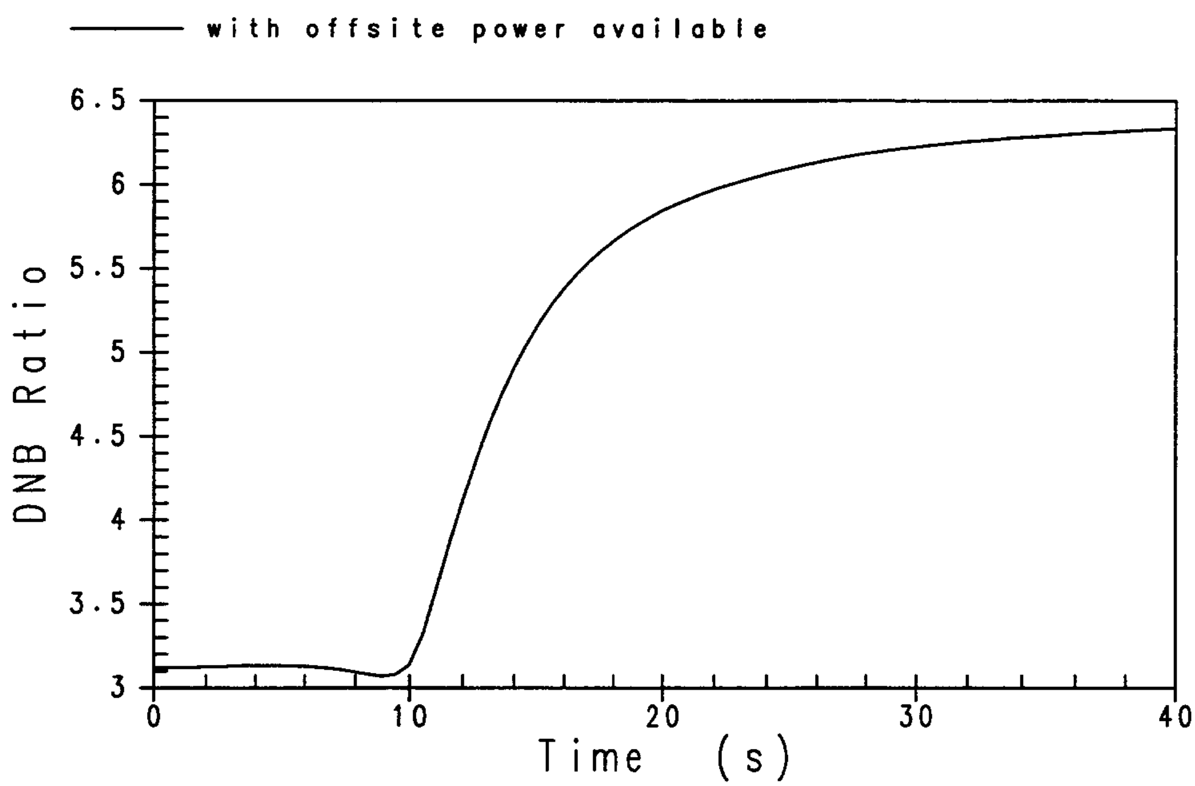

Figure 15.2.3-20

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

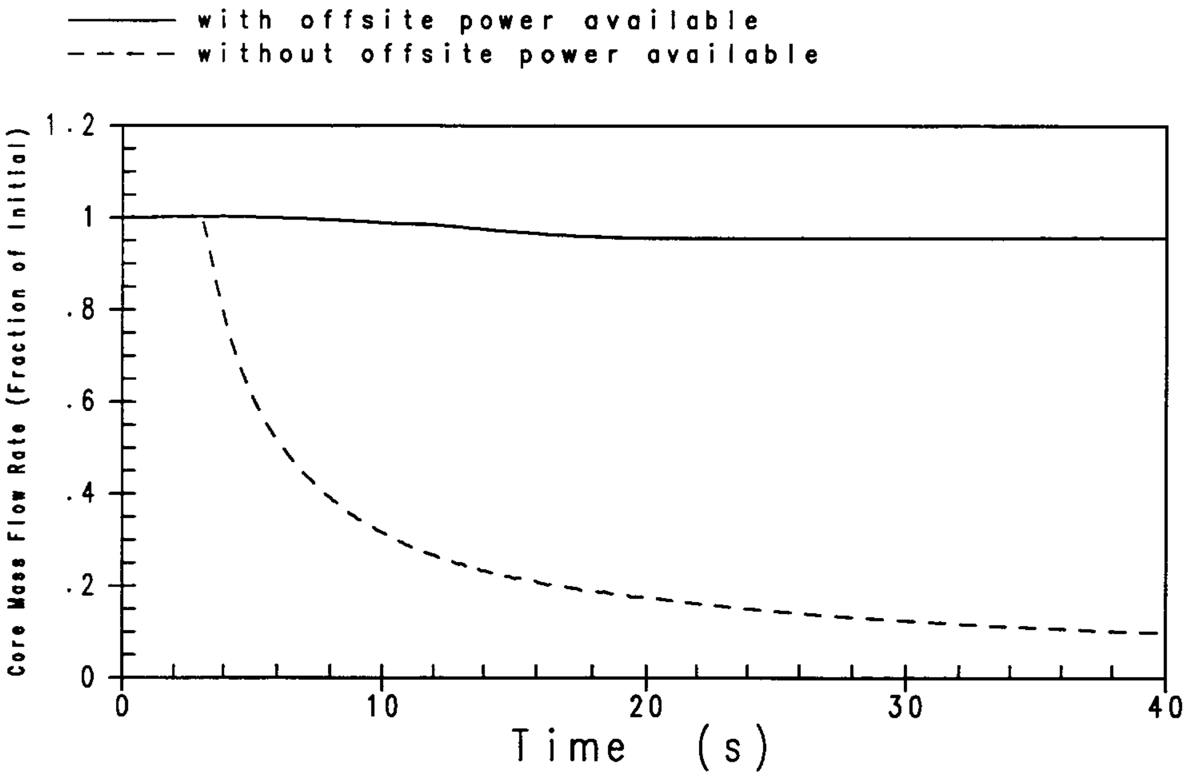

Figure 15.2.3-21

Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

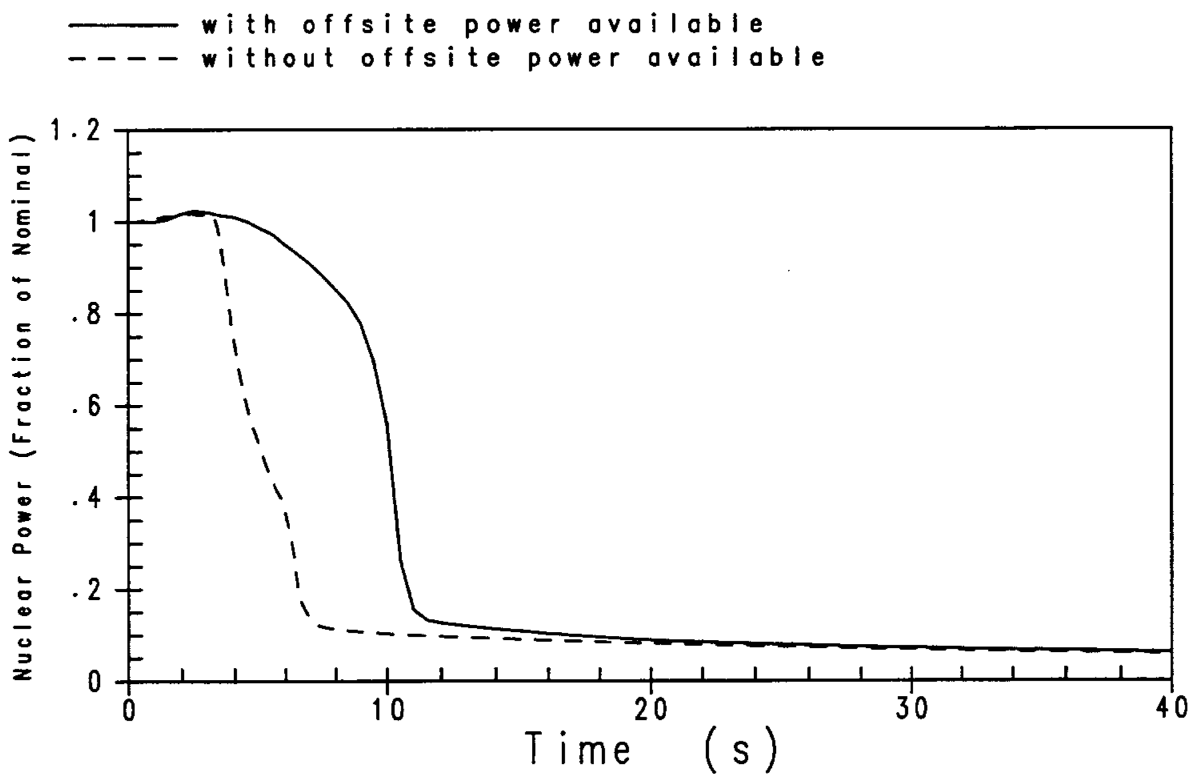

Figure 15.2.3-22

Nuclear Power (Fraction of Nominal) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

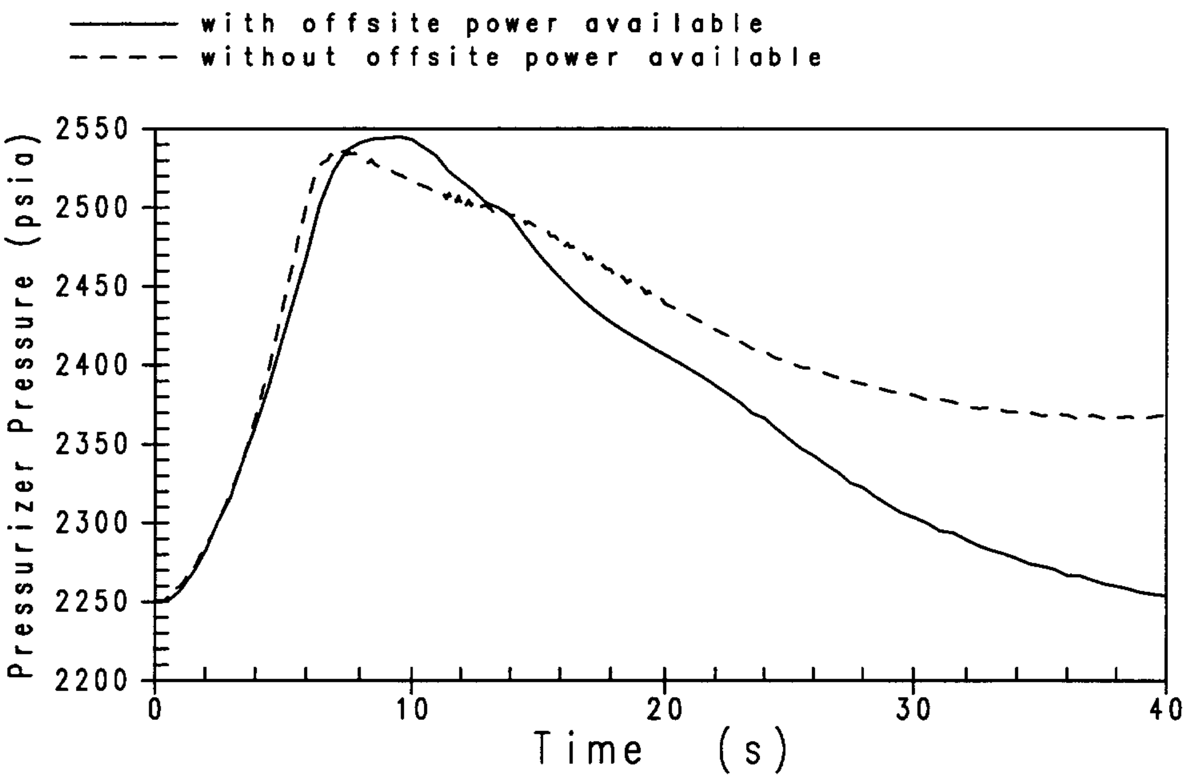

Figure 15.2.3-23

Pressurizer Pressure (psia) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

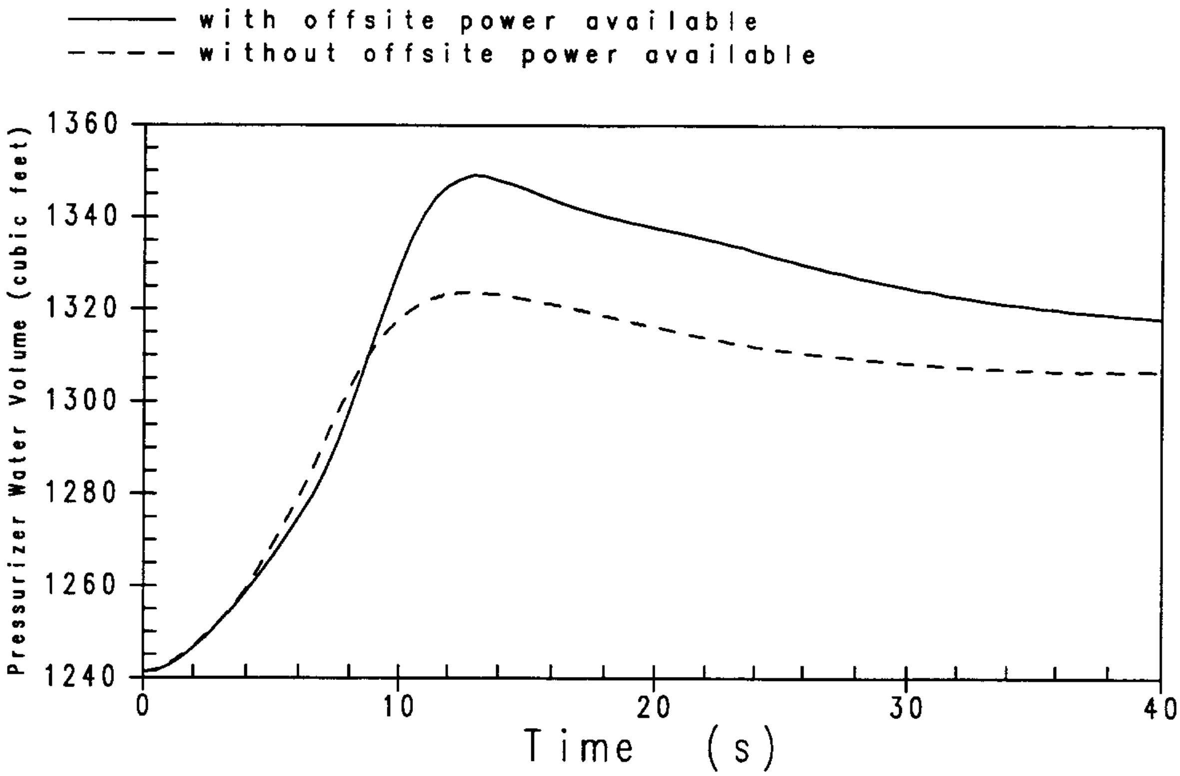

Figure 15.2.3-24

Pressurizer Water Volume (ft^3) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

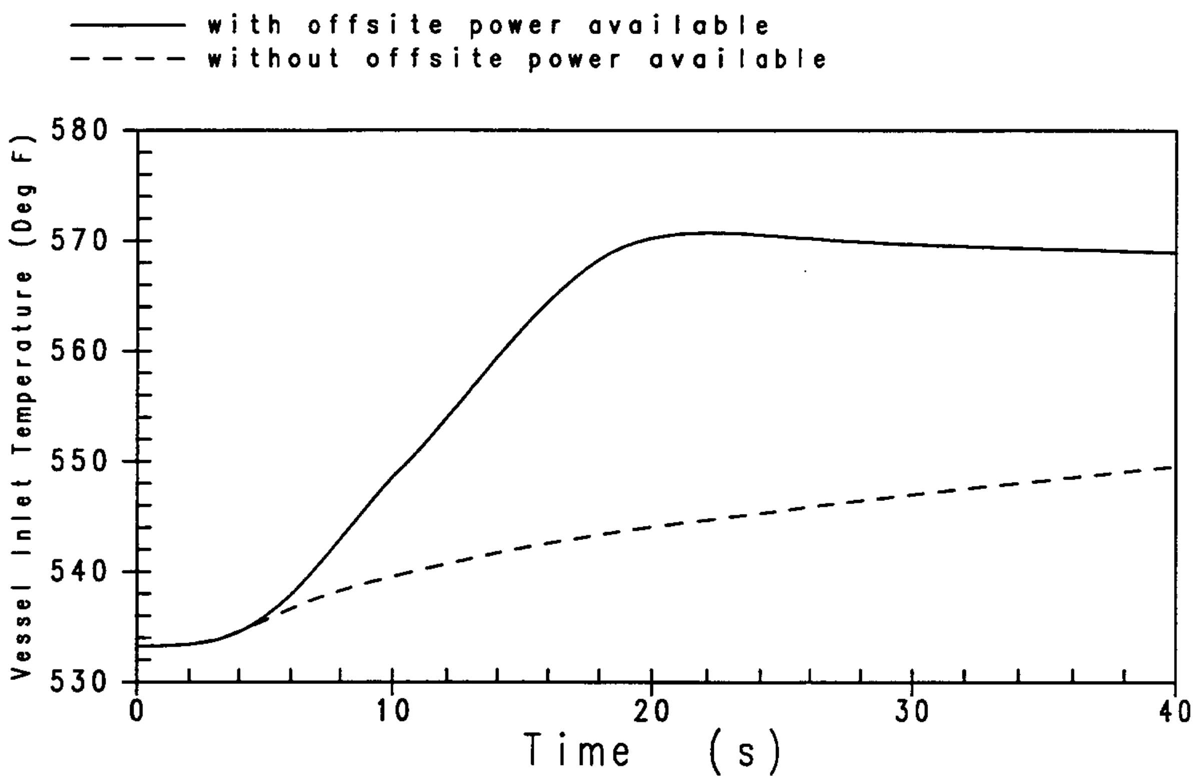

Figure 15.2.3-25

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

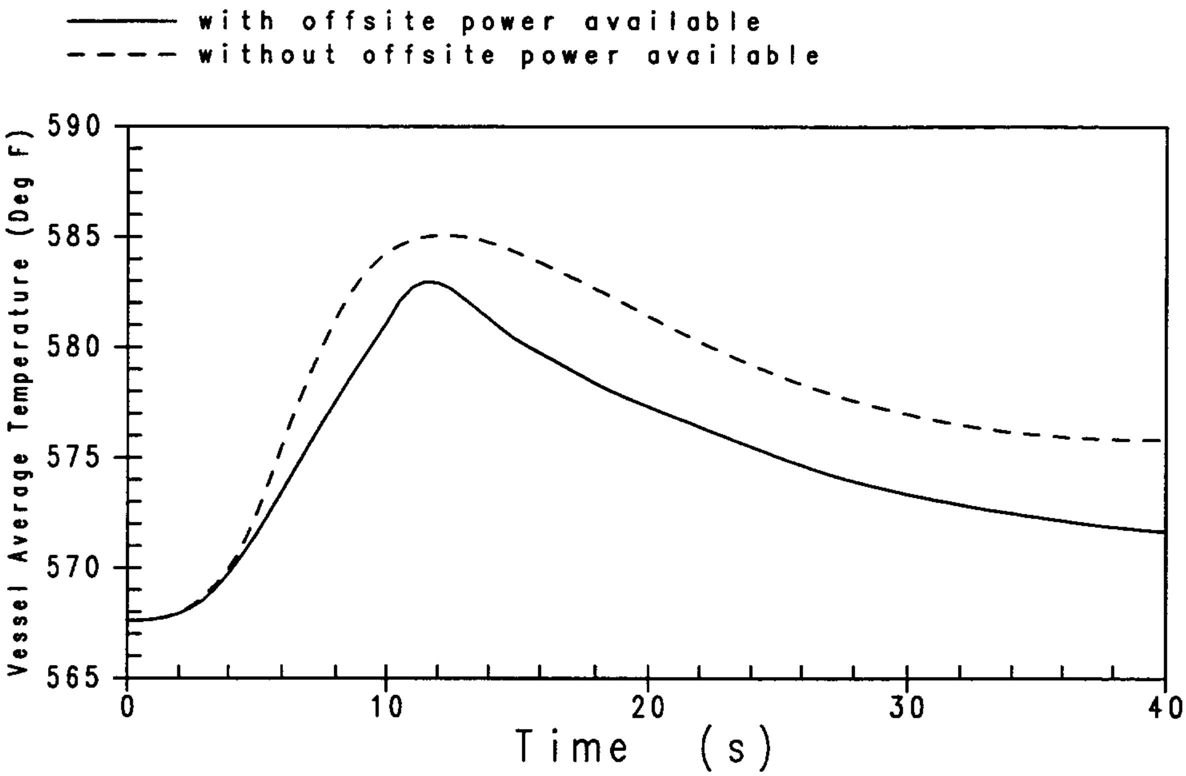

Figure 15.2.3-26

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

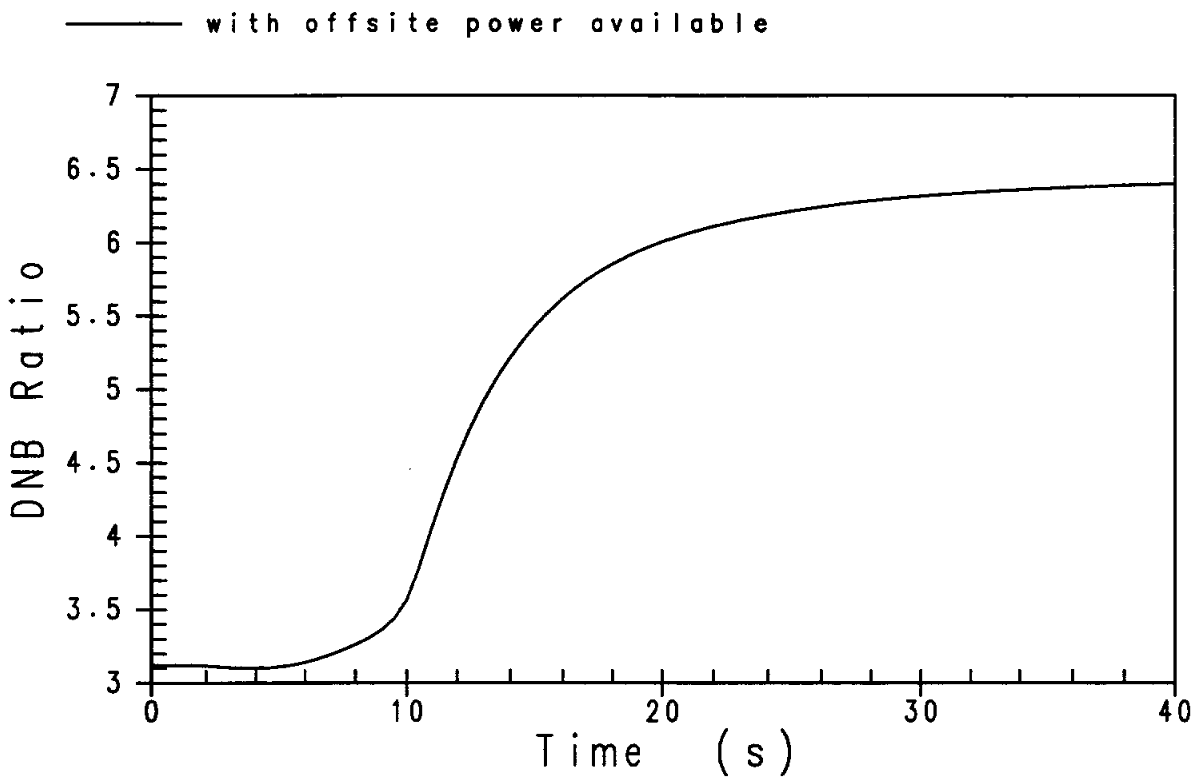

Figure 15.2.3-27

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

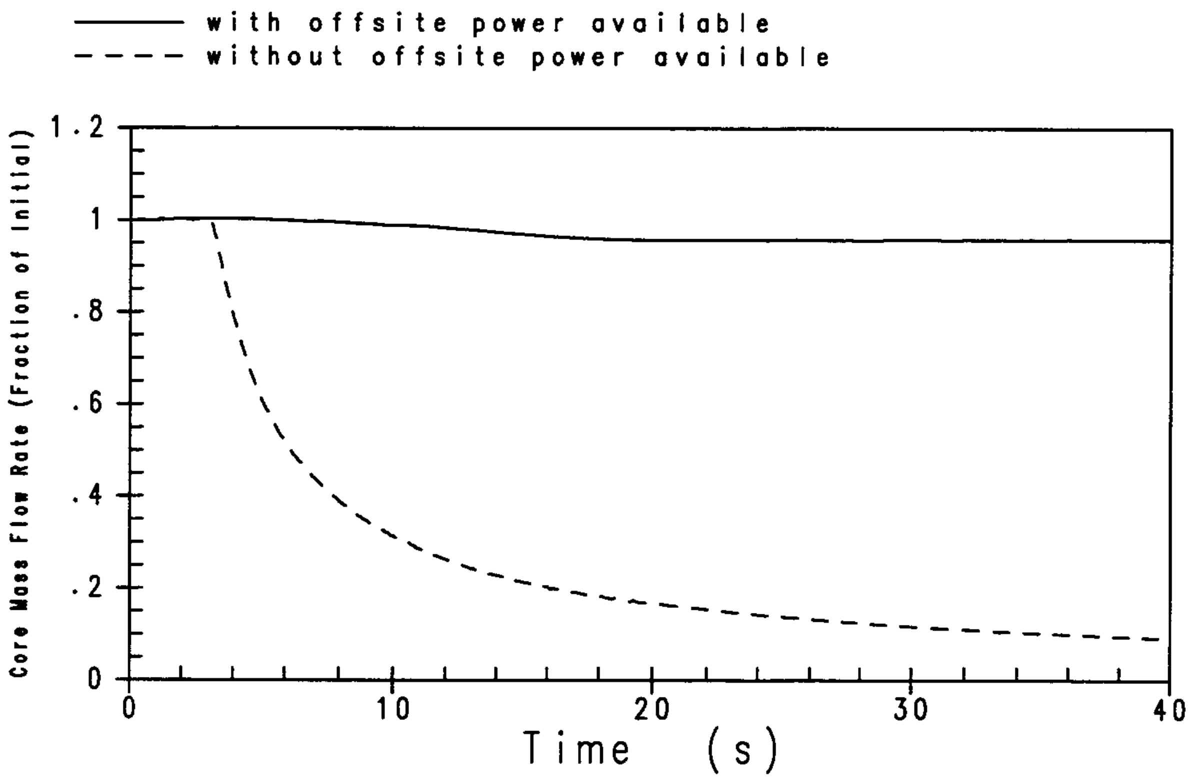

Figure 15.2.3-28

Core Mass Flow Rate (Fraction of Initial) versus Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

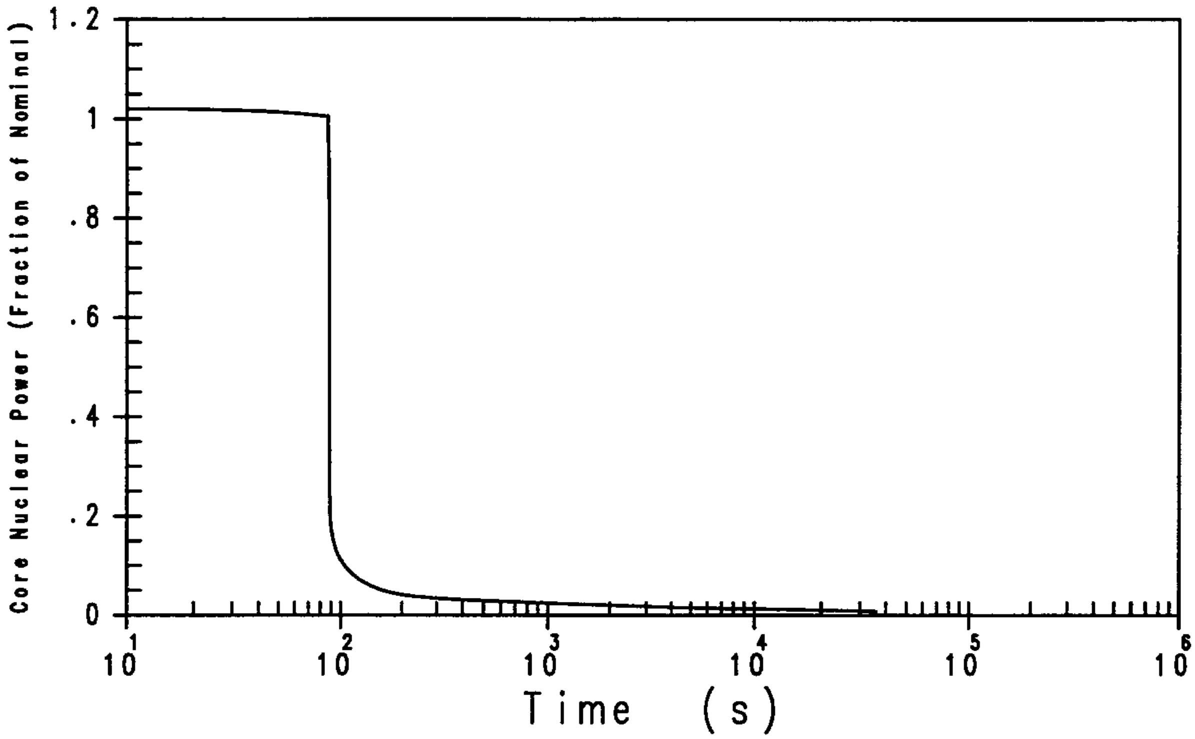

Figure 15.2.6-1

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

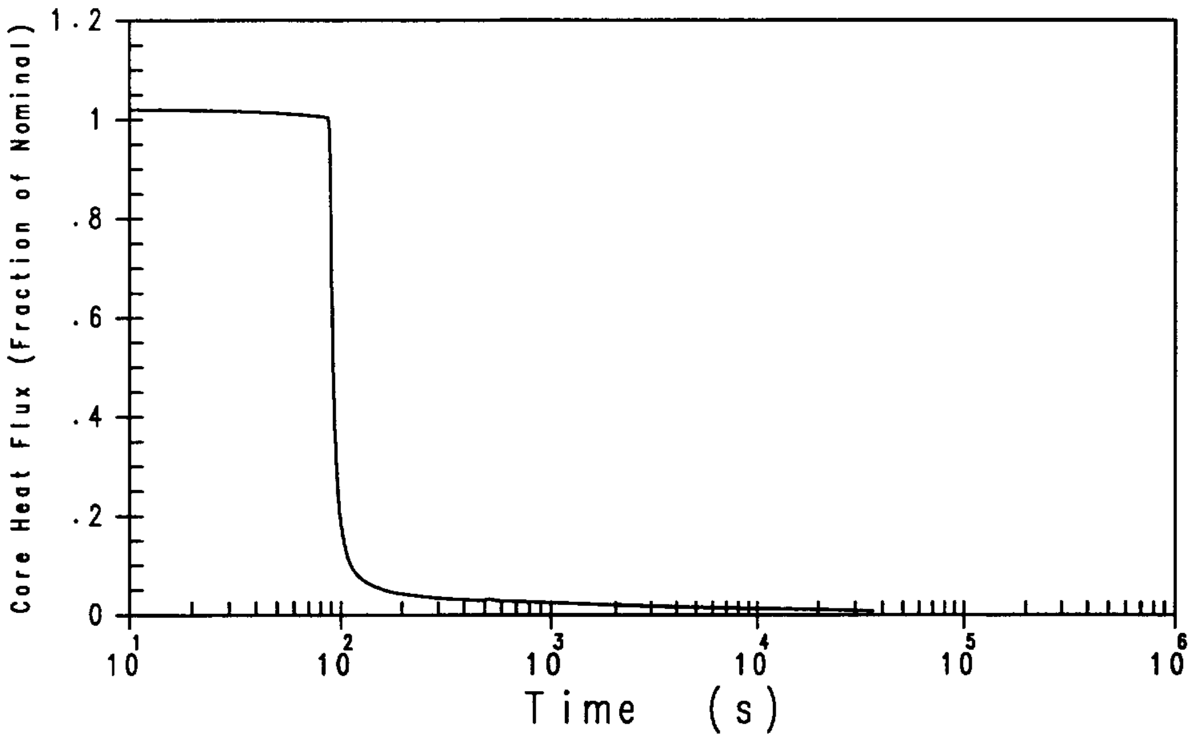

Figure 15.2.6-2

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

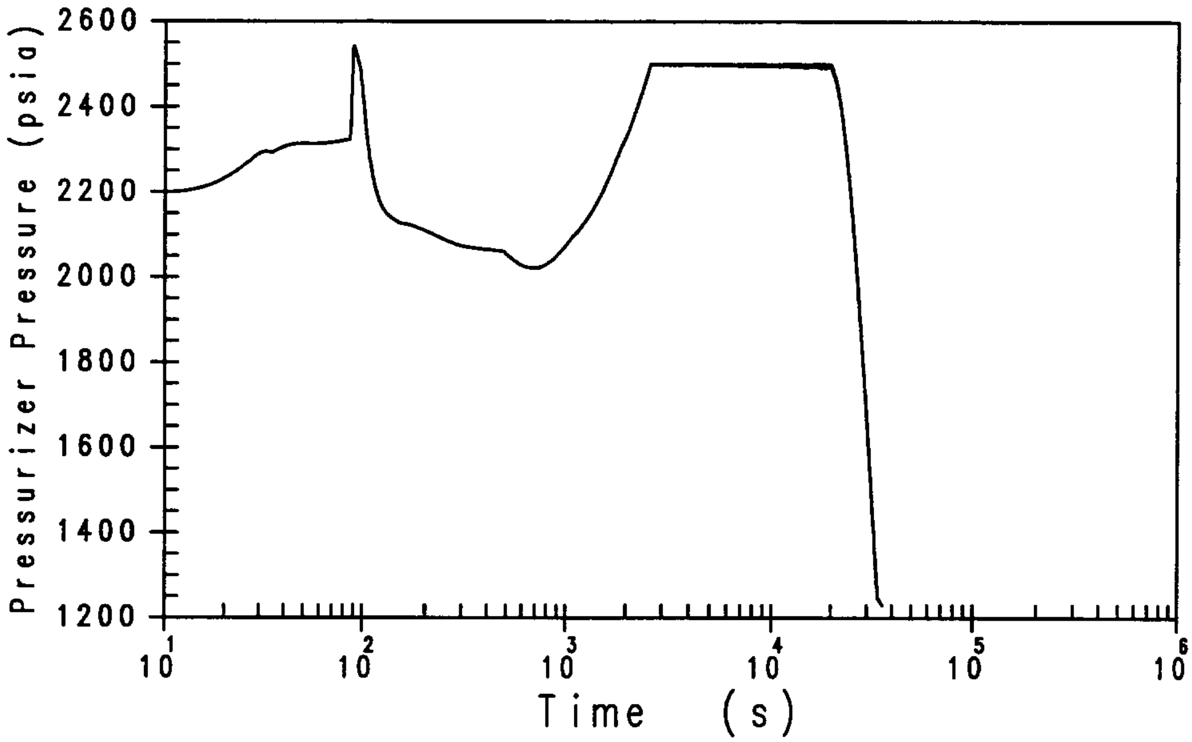

Figure 15.2.6-3

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

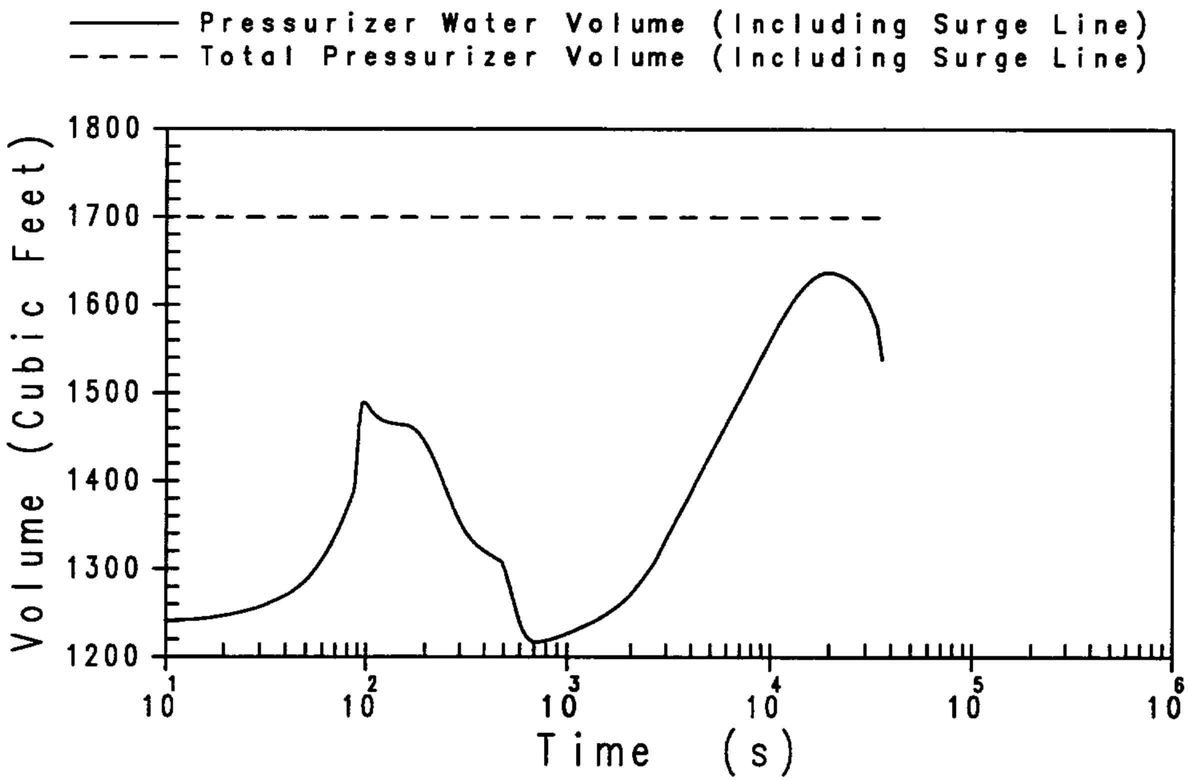

Figure 15.2.6-4

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

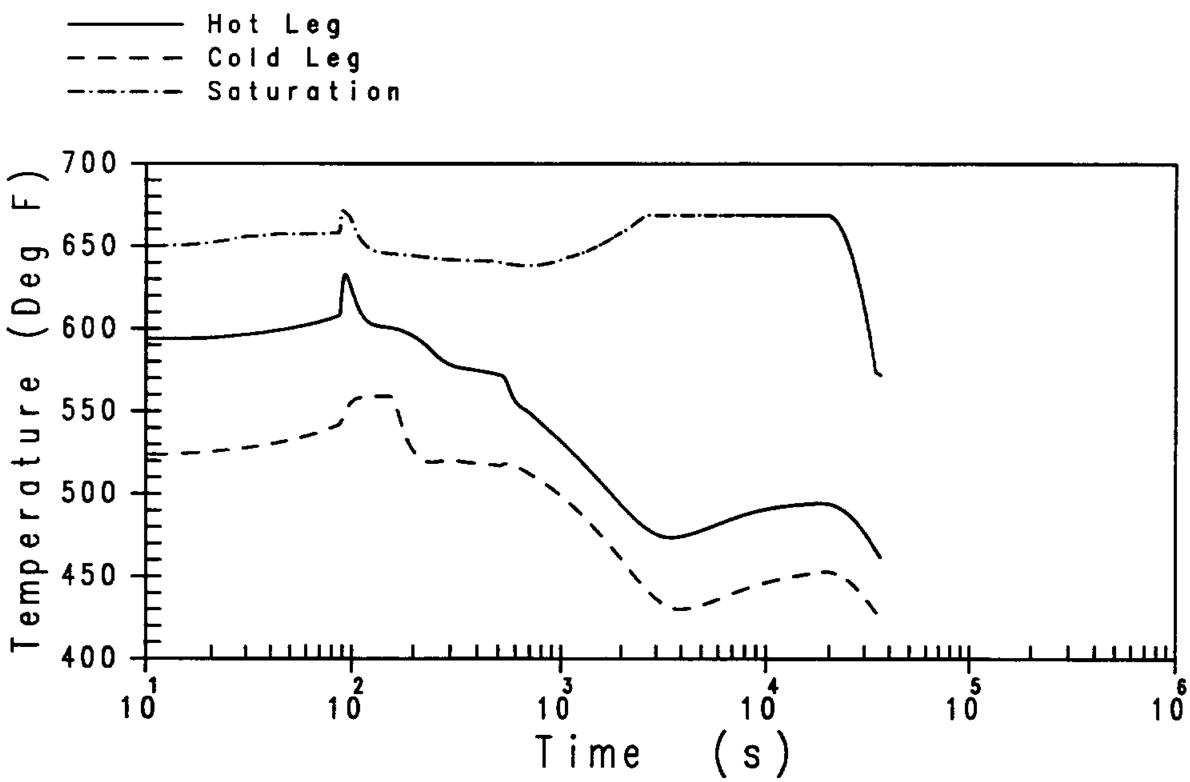

Figure 15.2.6-5

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

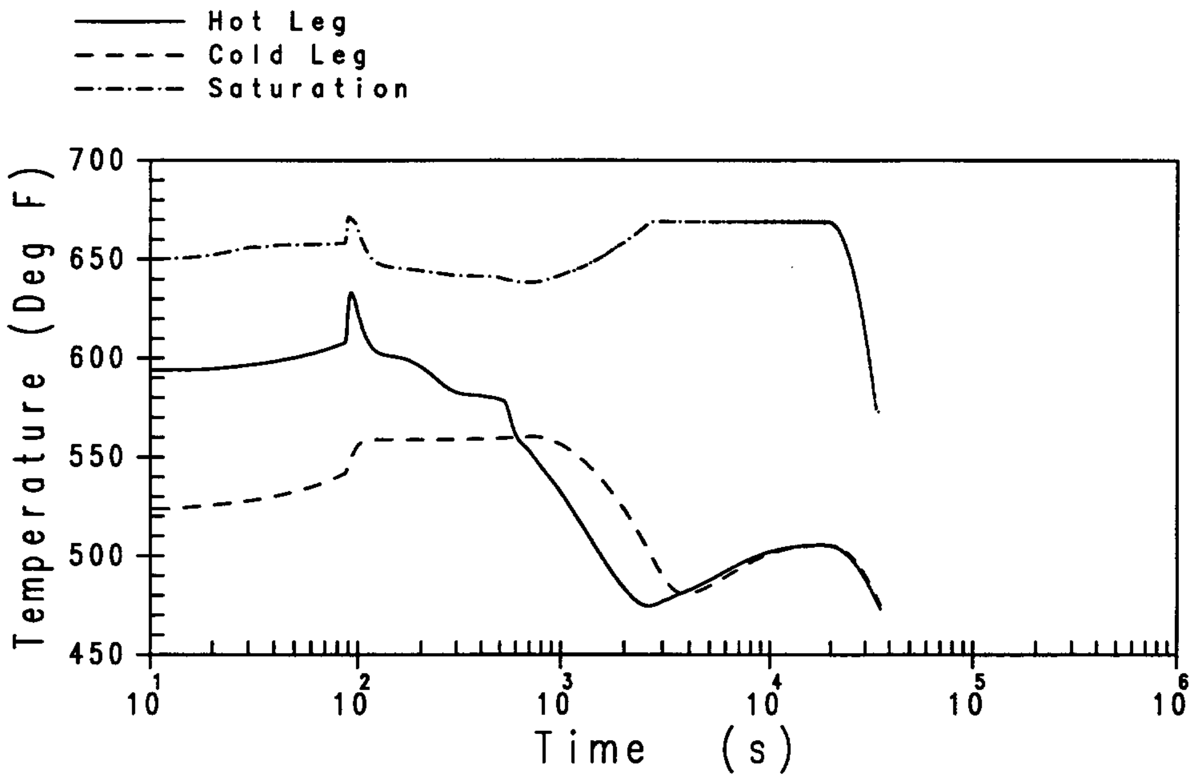

Figure 15.2.6-6

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

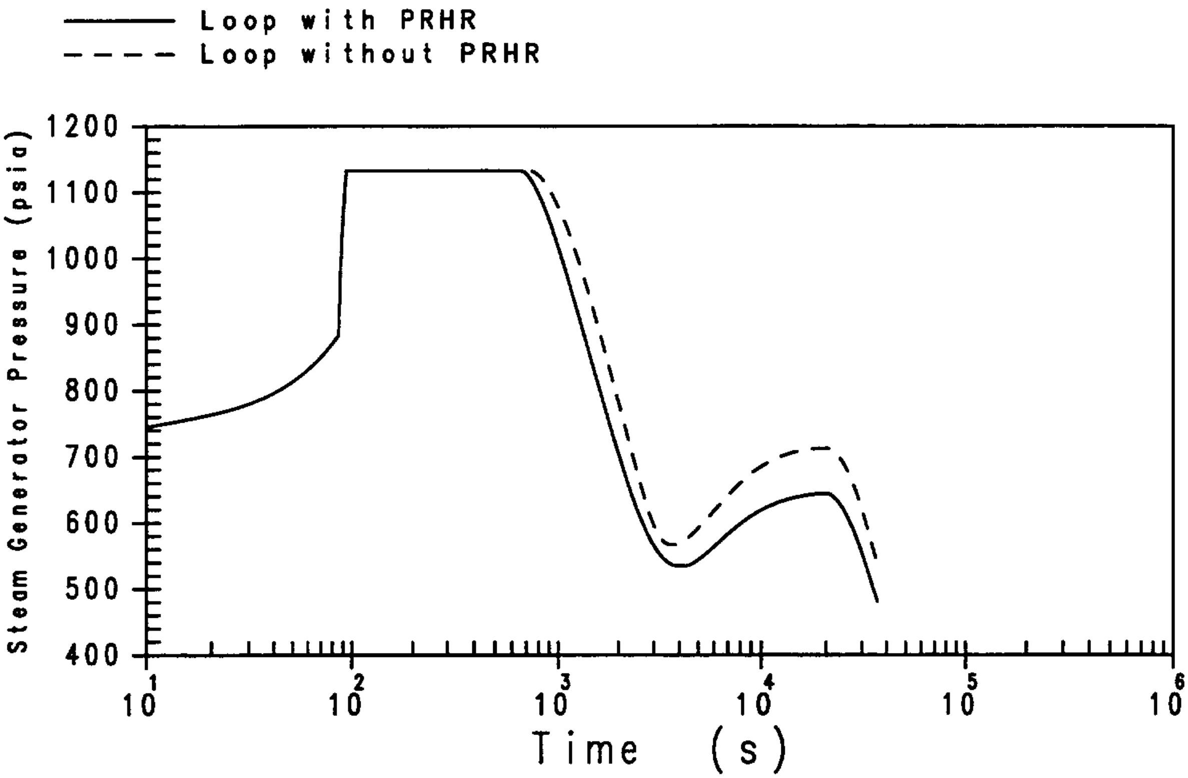

Figure 15.2.6-7

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

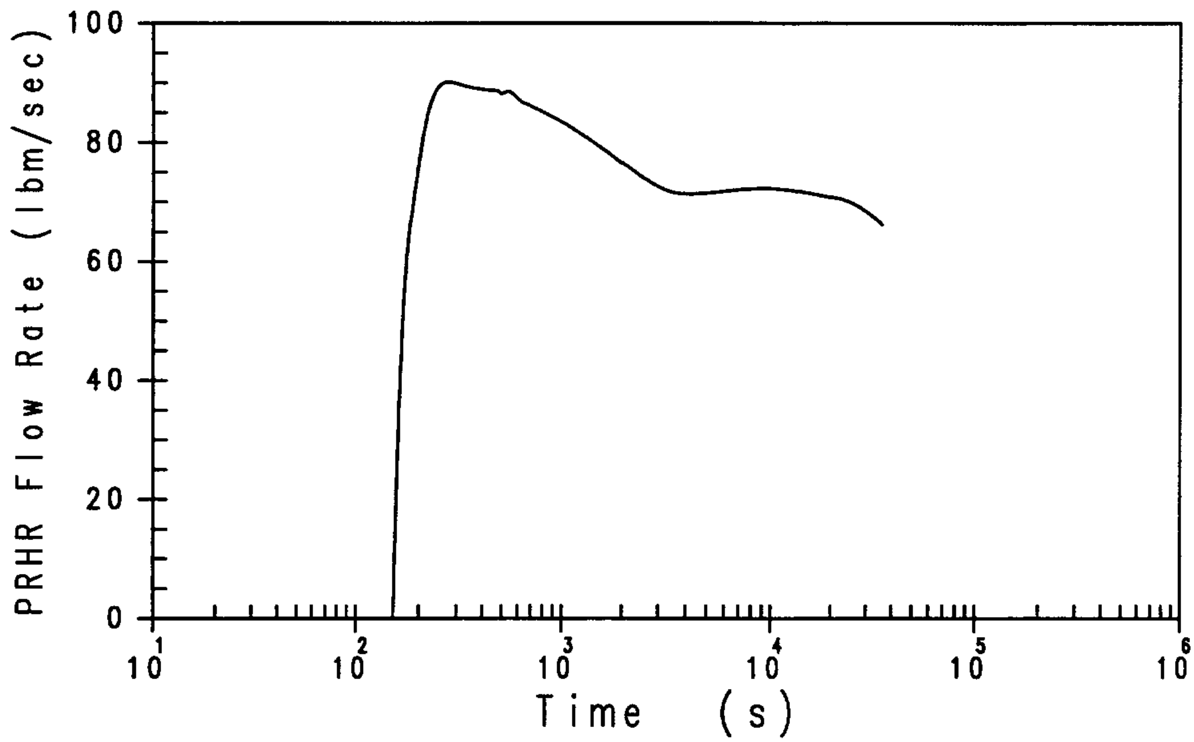

Figure 15.2.6-8

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

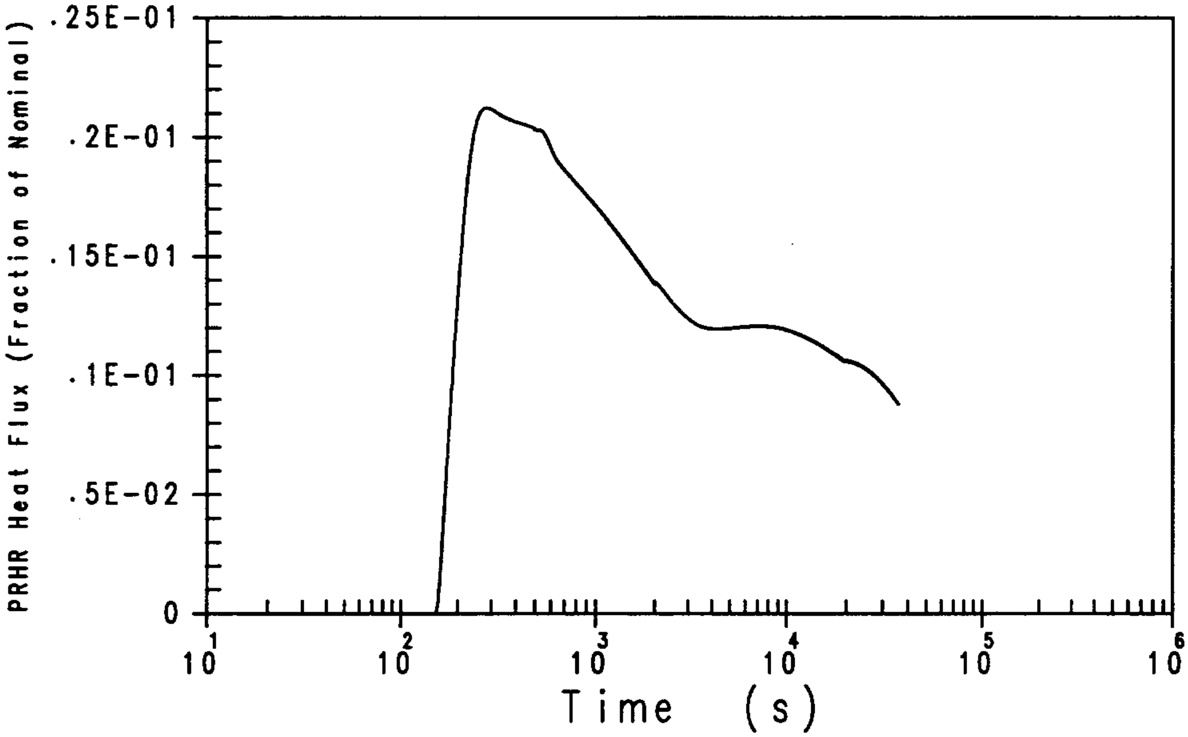

Figure 15.2.6-9

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

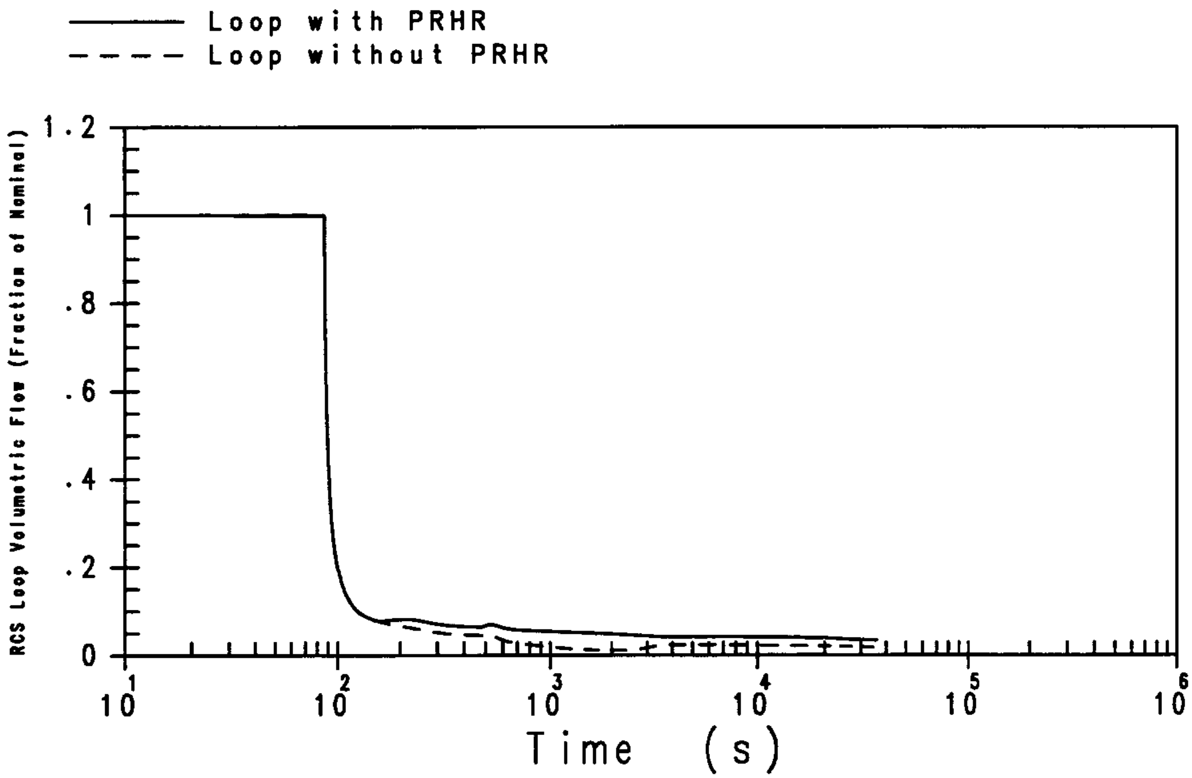

Figure 15.2.6-10

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

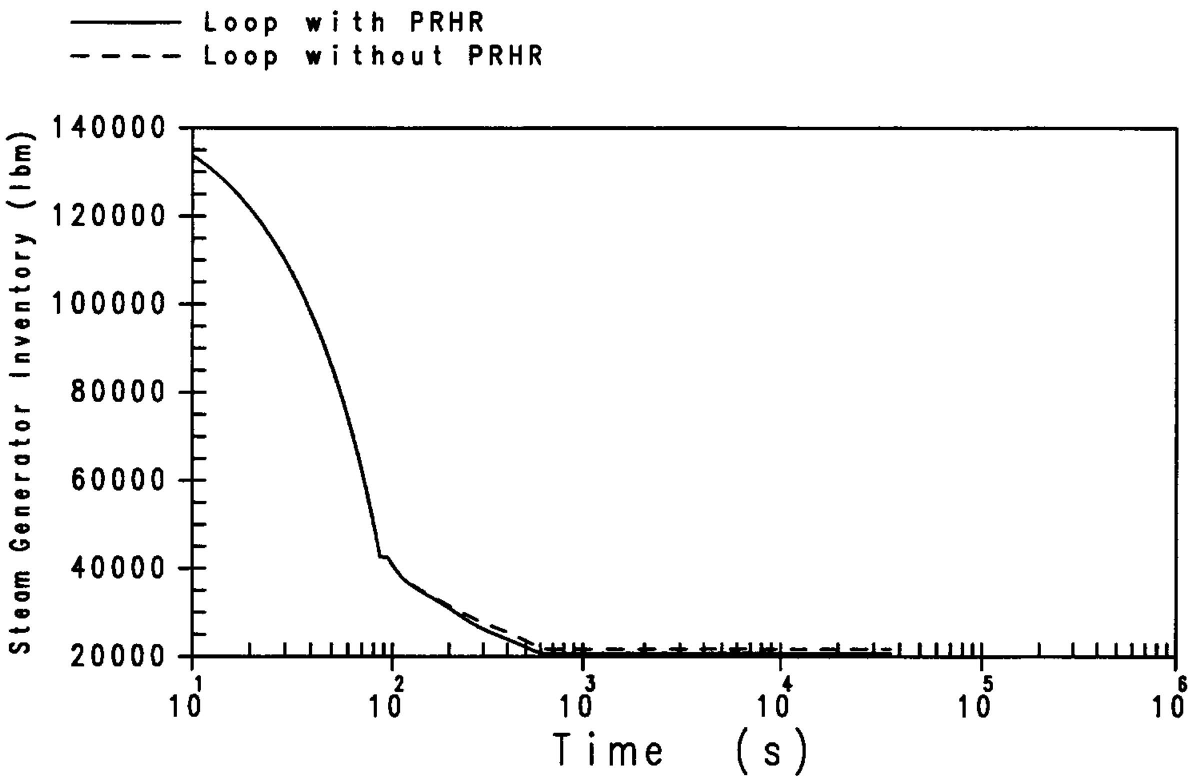

Figure 15.2.6-11

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

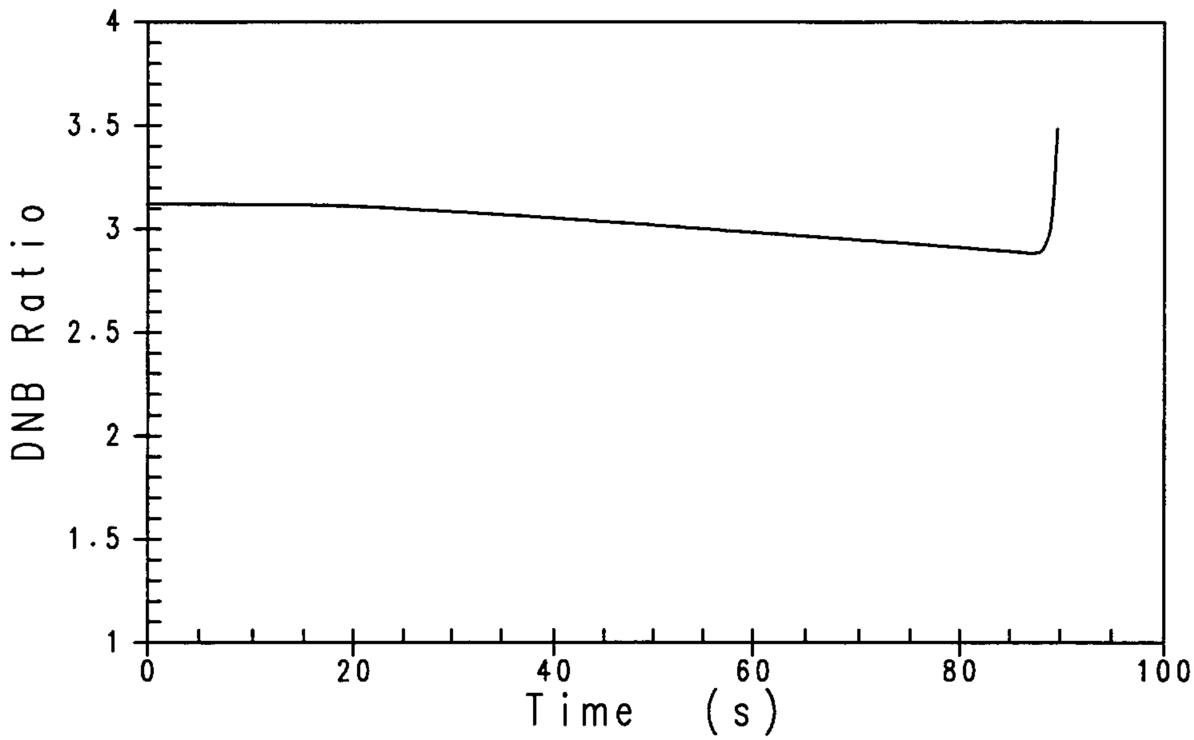

Figure 15.2.6-12

DNBR versus Time for Loss of ac Power to the Plant Auxiliaries

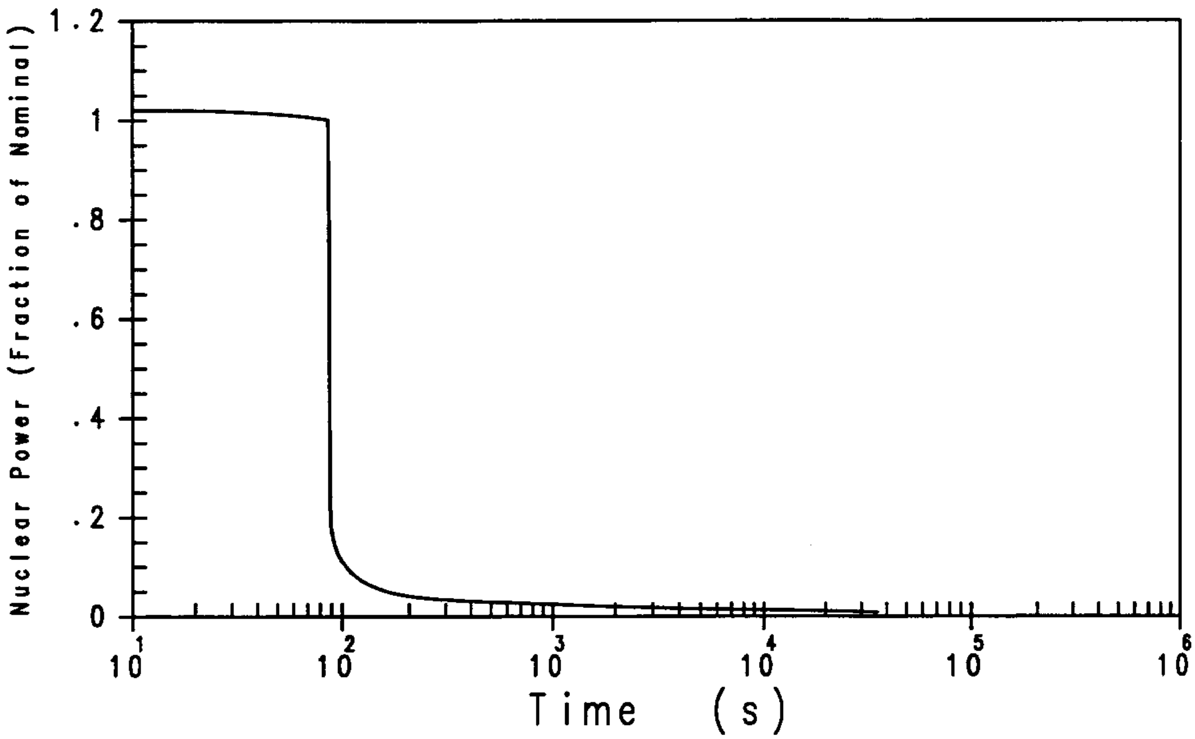

Figure 15.2.7-1

Nuclear Power Transient for Loss of Normal Feedwater Flow

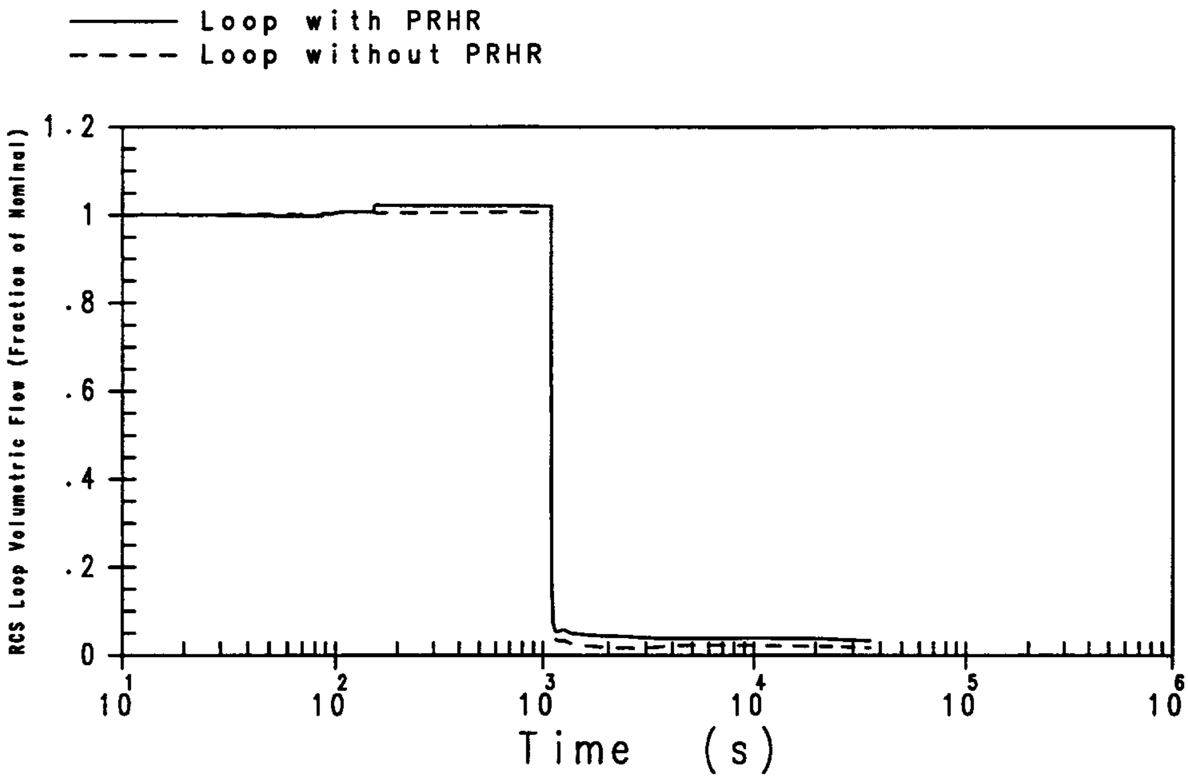

Figure 15.2.7-2

Reactor Coolant System Volumetric Flow Transient for Loss of Normal Feedwater Flow

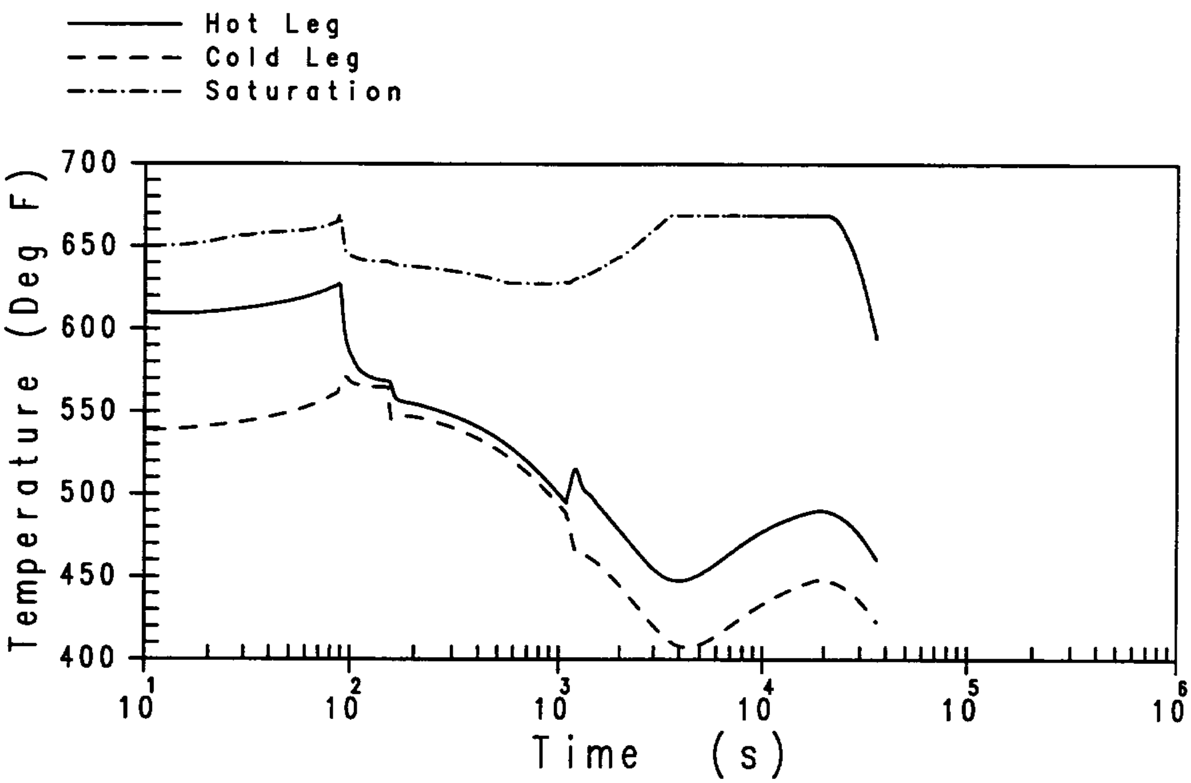

Figure 15.2.7-3

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

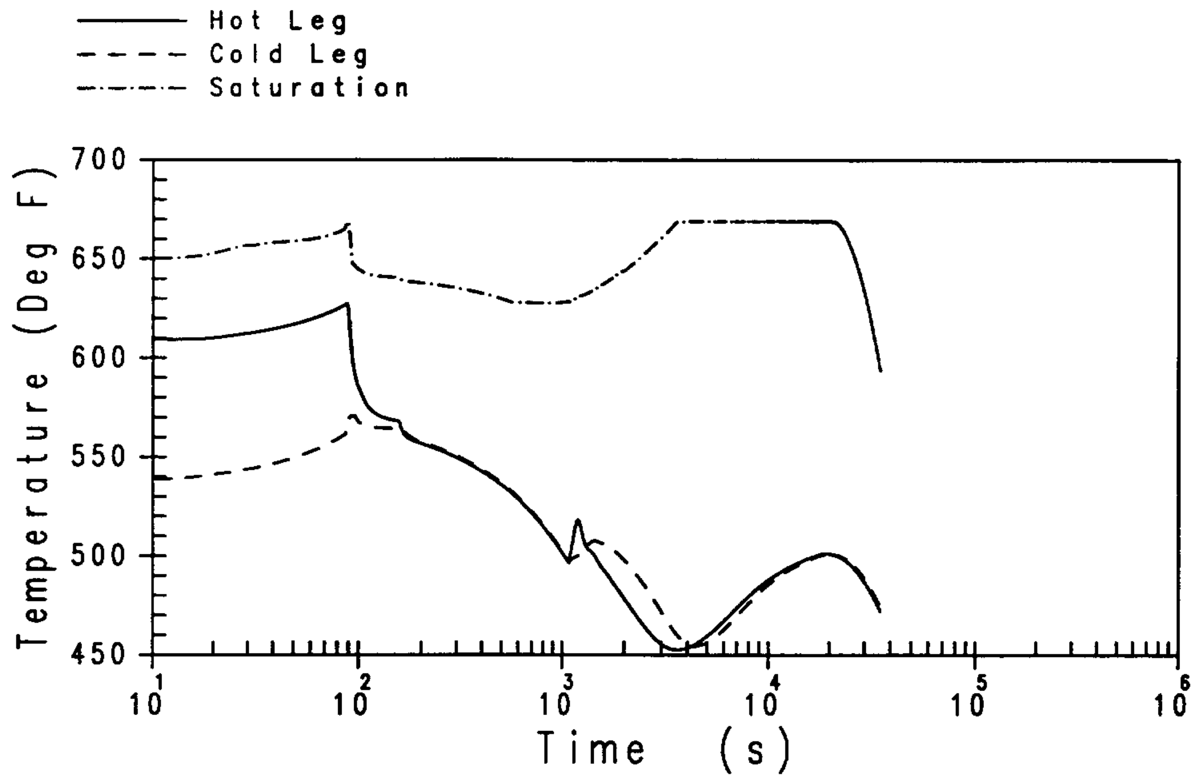

Figure 15.2.7-4

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

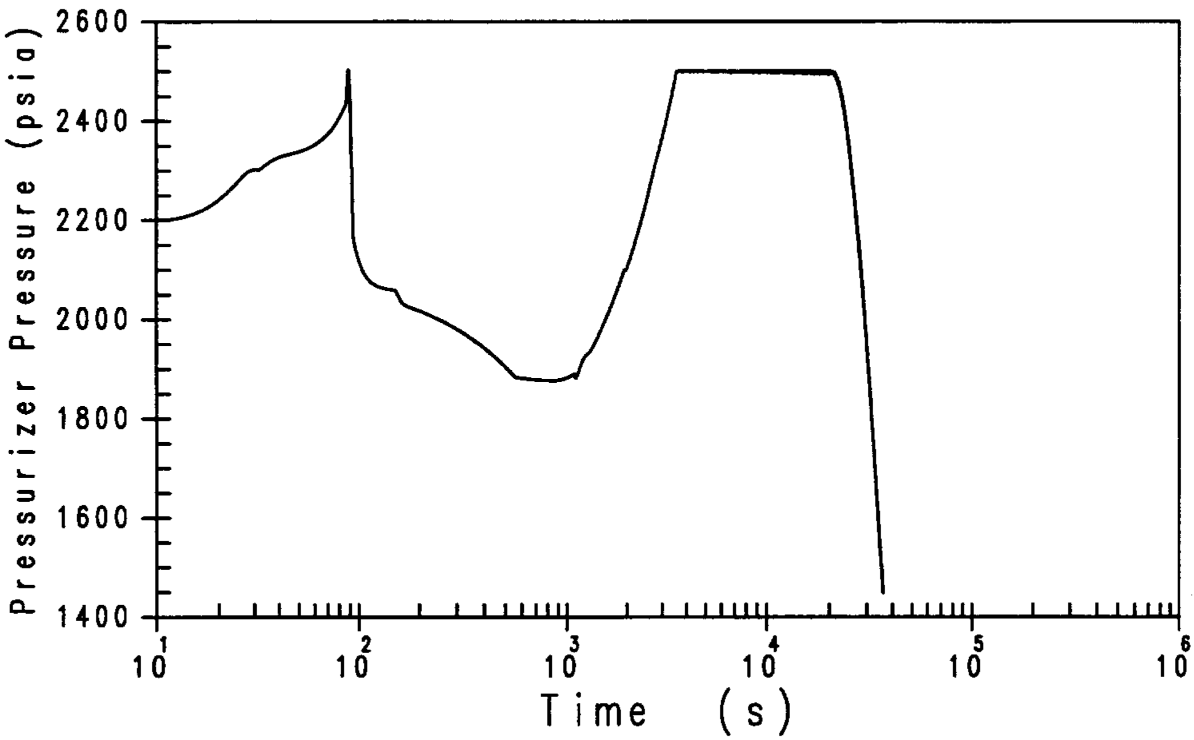

Figure 15.2.7-5

Pressure Transient for Loss of Normal Feewater Flow

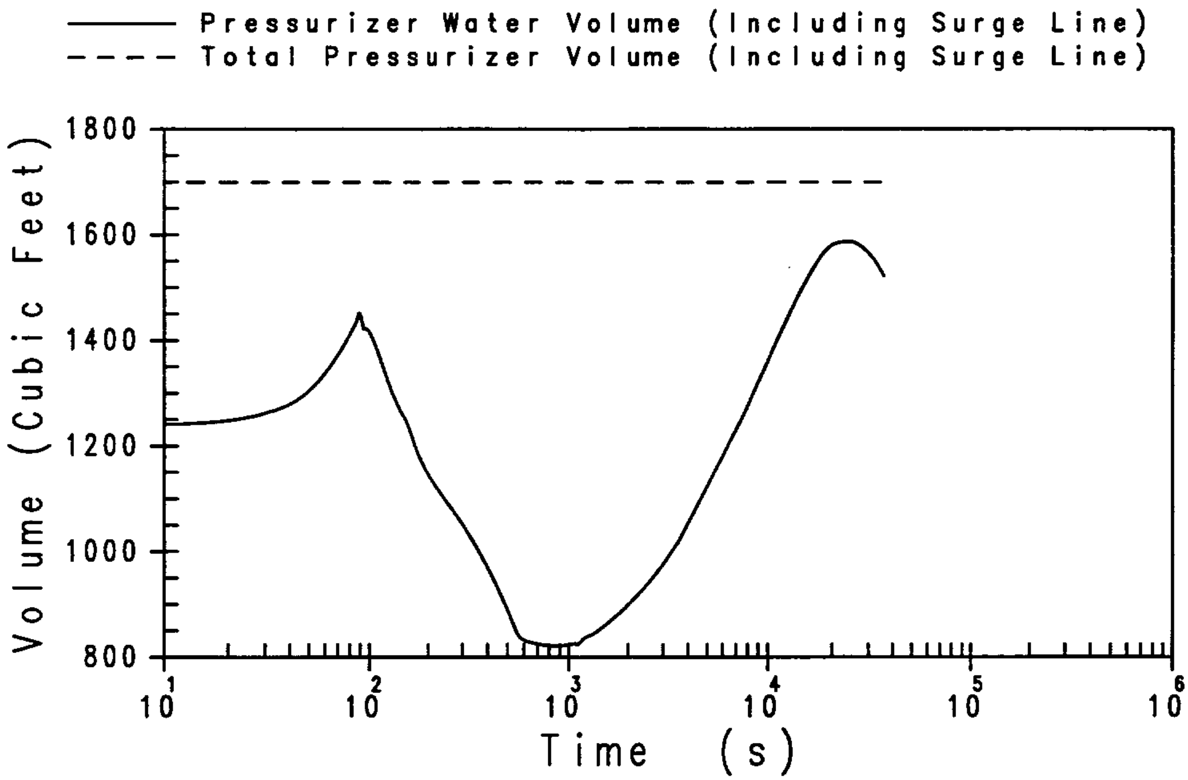

Figure 15.2.7-6

Pressurizer Water Volume Transient for Loss of Normal Feedwater Flow

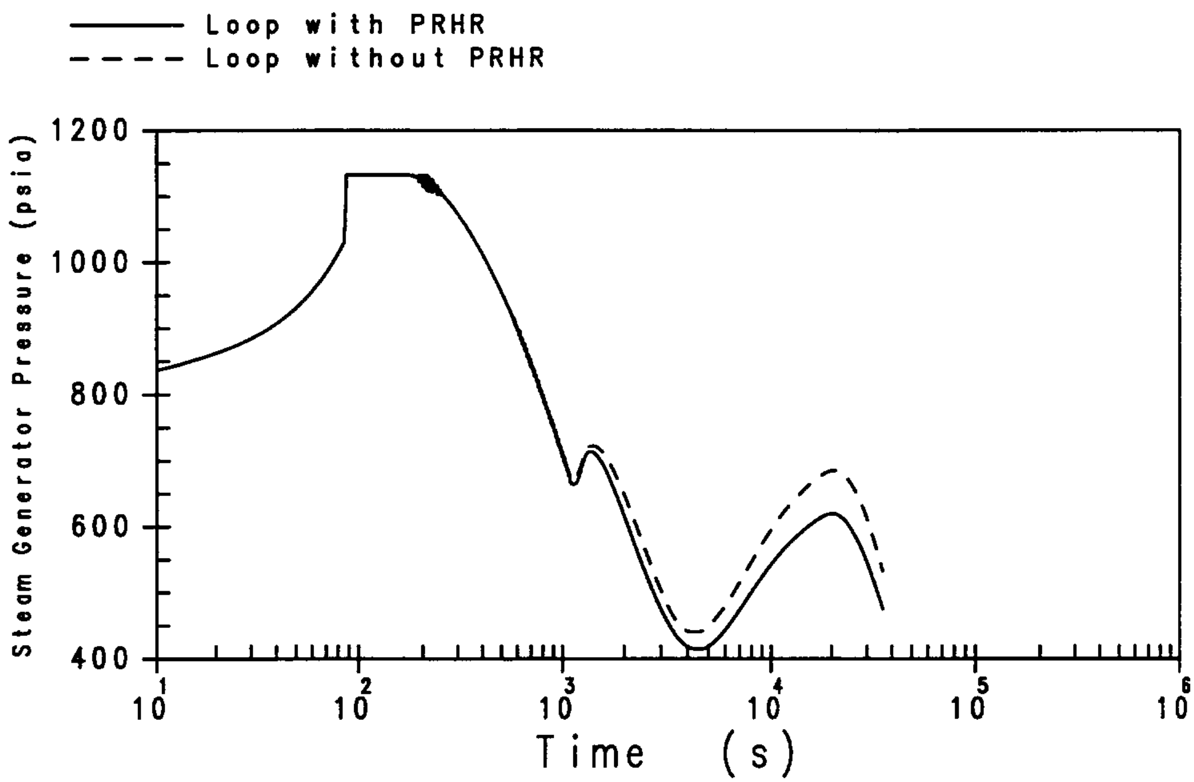

Figure 15.2.7-7

Steam Generator Pressure Transients for Loss of Normal Feedwater Flow

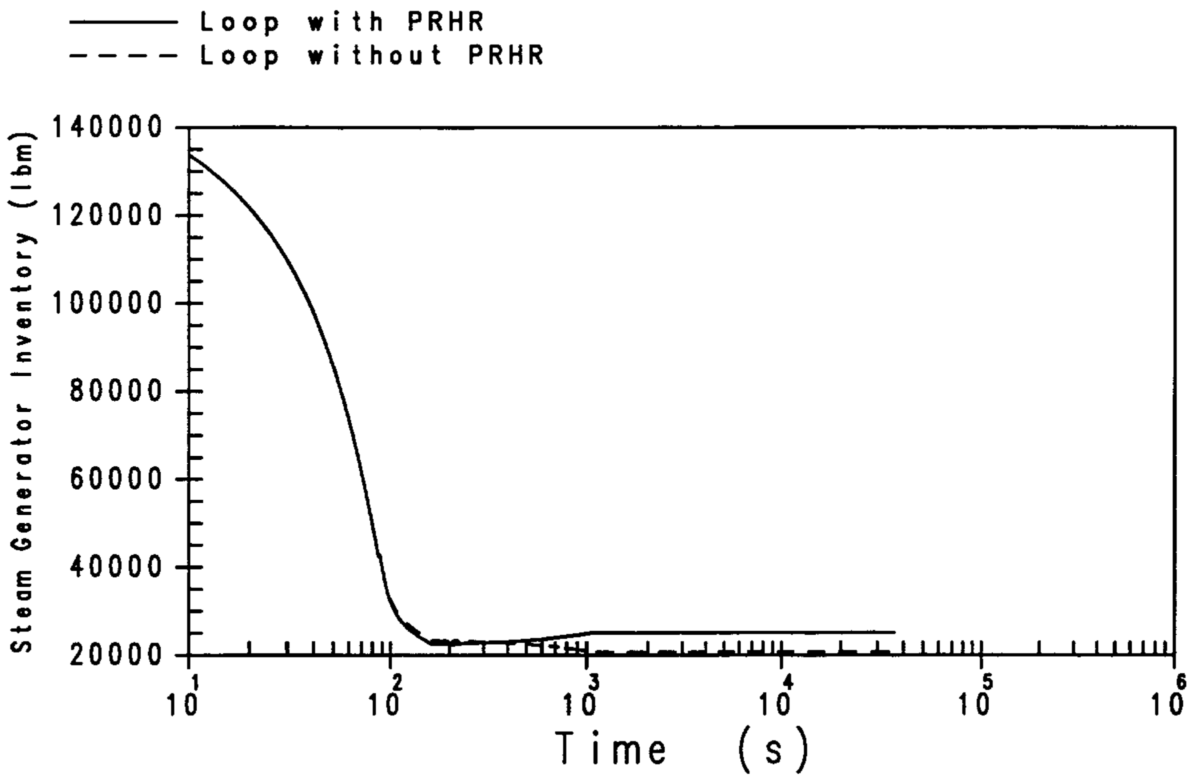

Figure 15.2.7-8

Steam Generator Inventory Transients for Loss of Normal Feedwater Flow

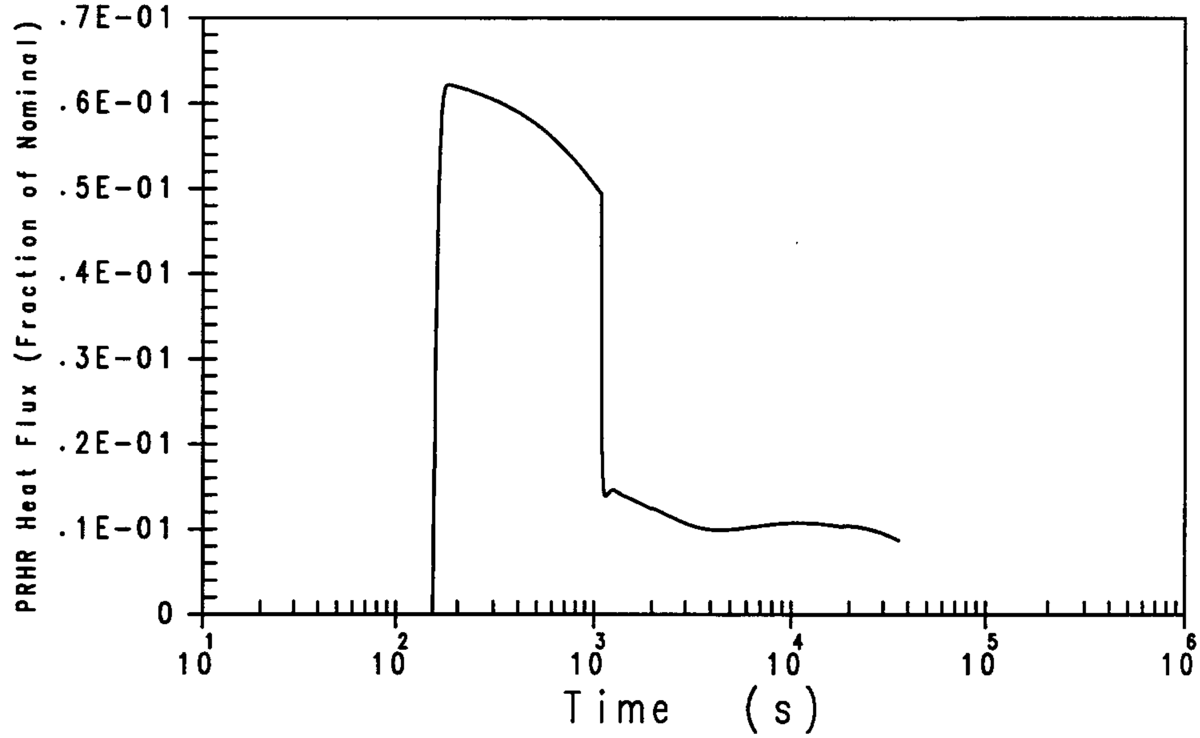

Figure 15.2.7-9

PRHR Heat Flux Transient for Loss of Normal Feedwater Flow

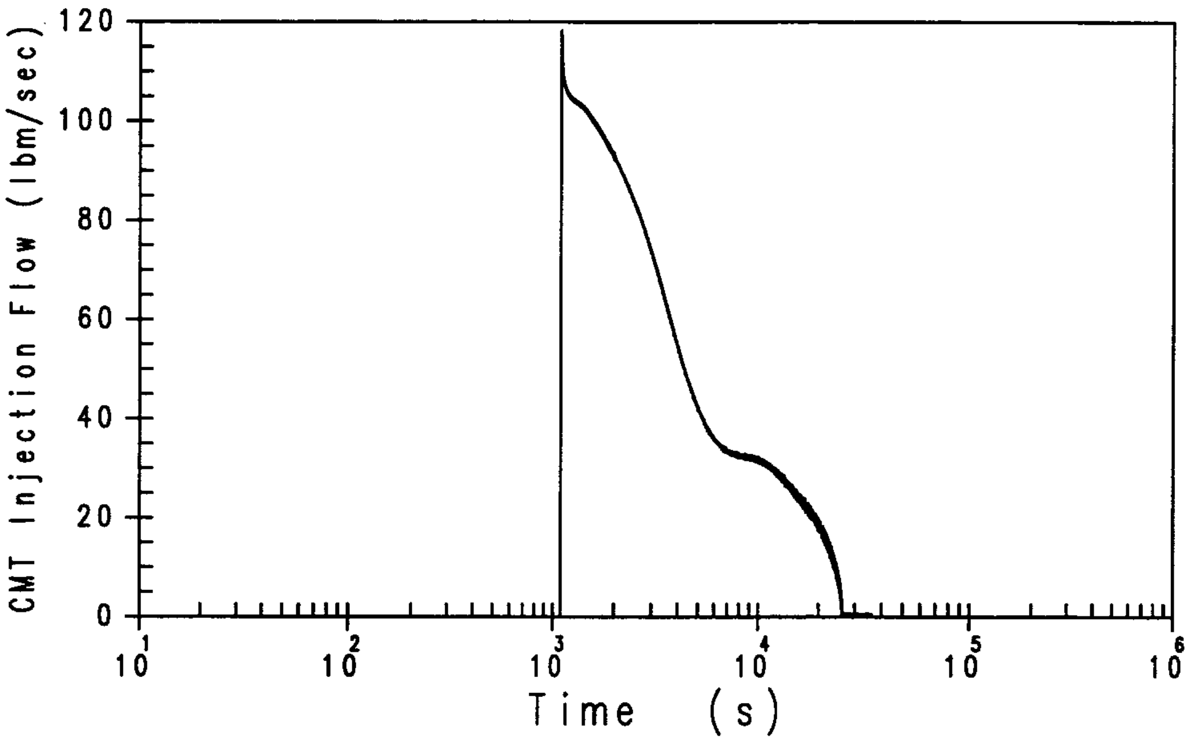

Figure 15.2.7-10

CMT Injection Flow Transient for Loss of Normal Feedwater Flow

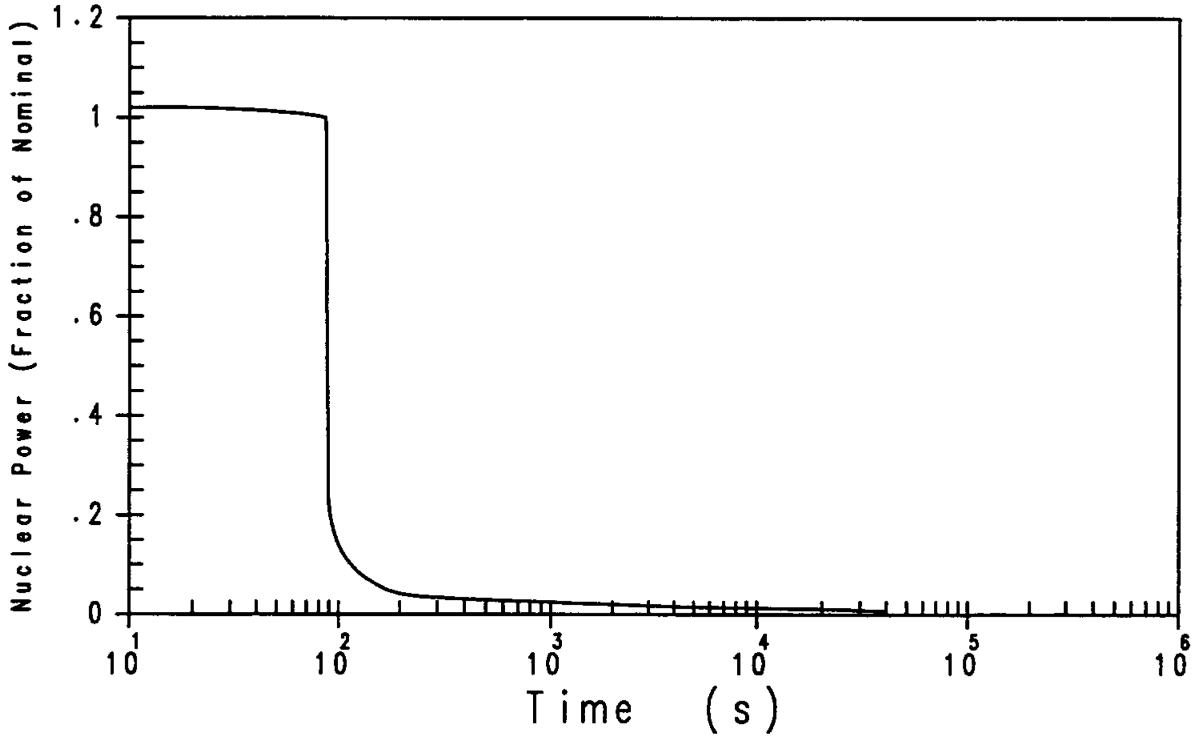

Figure 15.2.8-1

Nuclear Power Transient for Main Feedwater Line Rupture

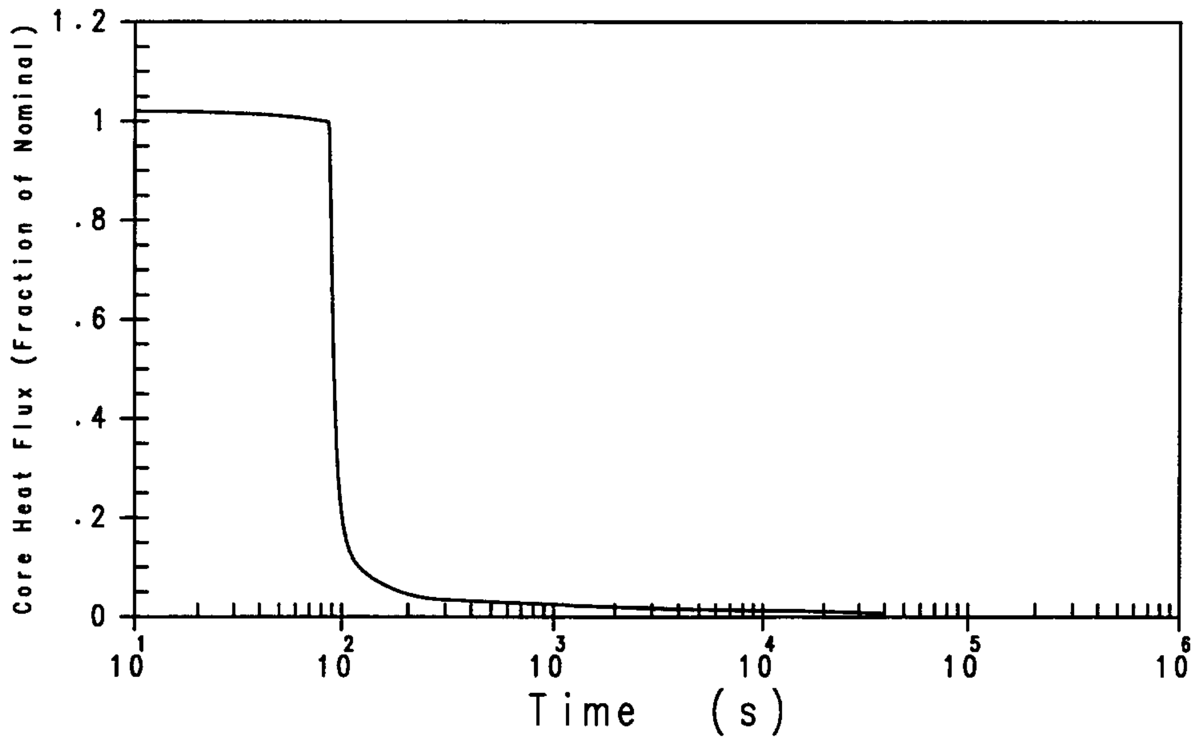

Figure 15.2.8-2

Core Heat Flux Transient for Main Feedwater Line Rupture

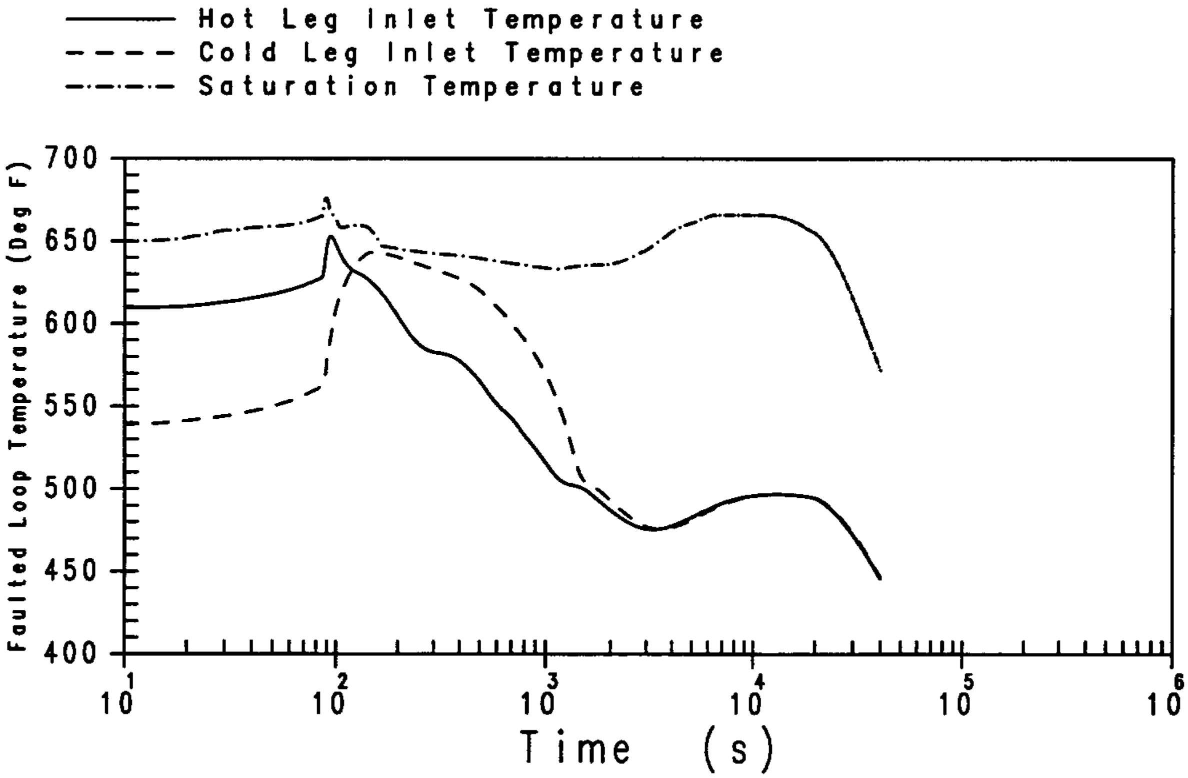

Figure 15.2.8-3

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

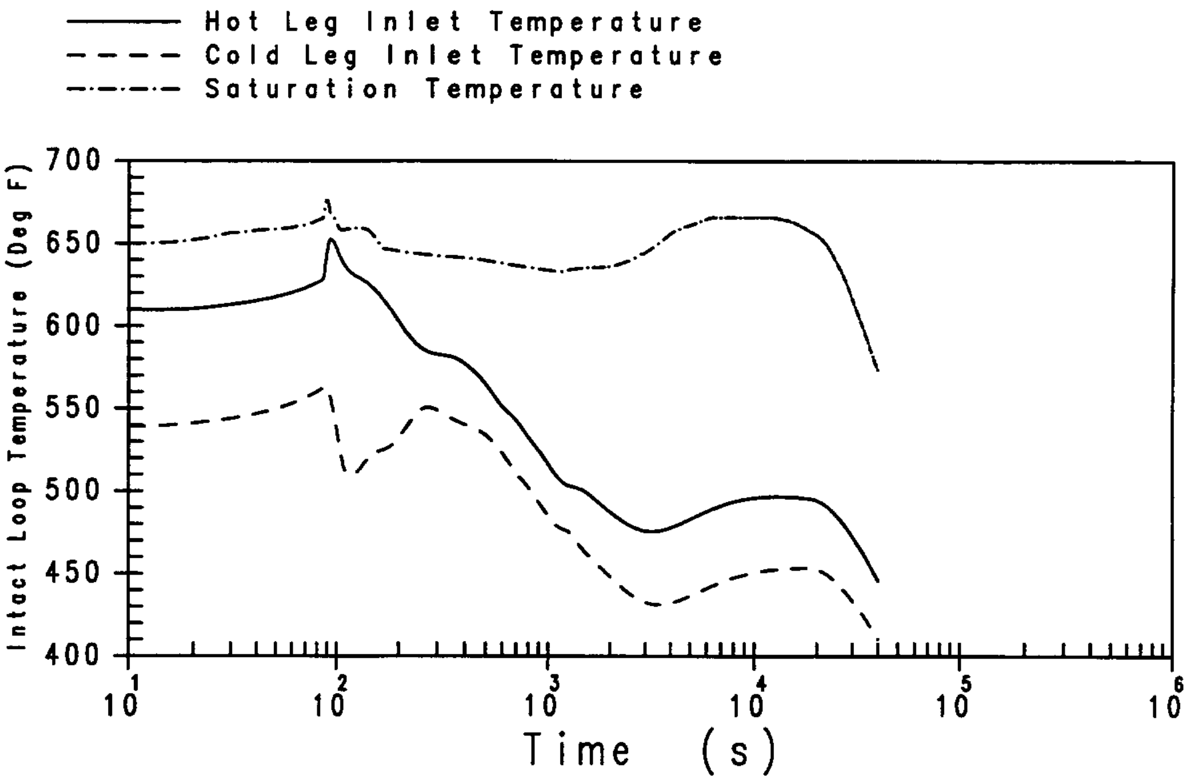

Figure 15.2.8-4

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

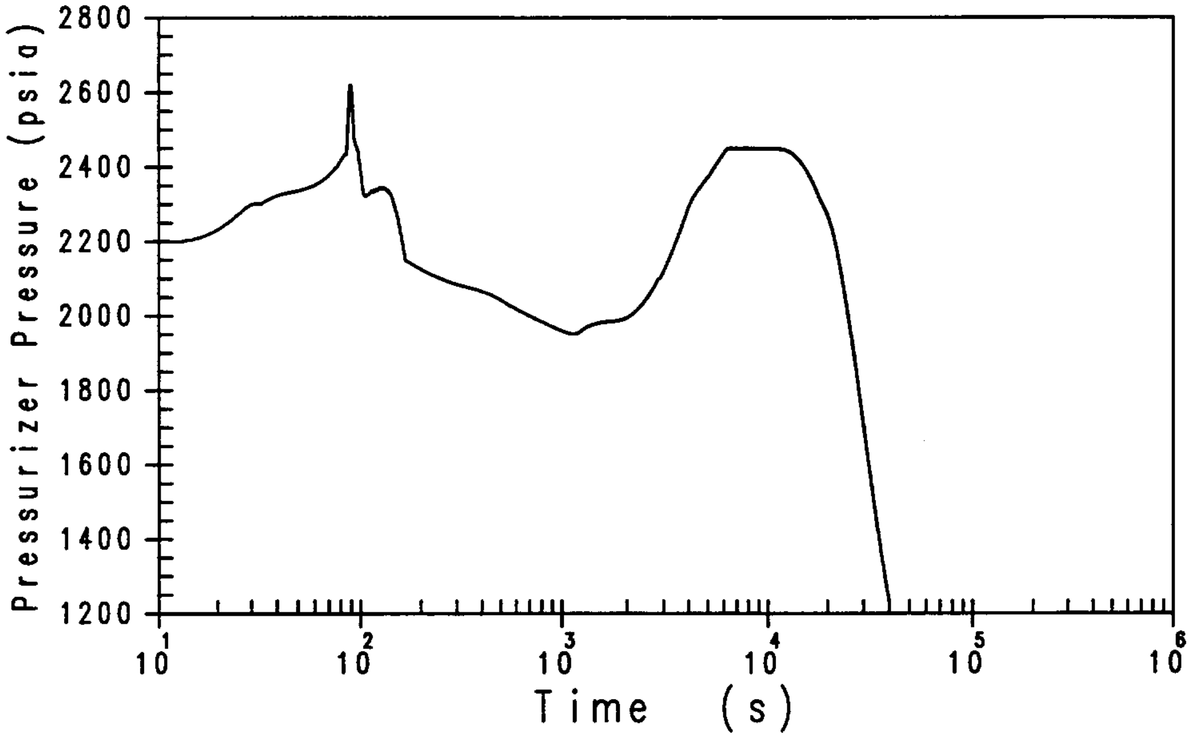

Figure 15.2.8-5

Pressurizer Pressure Transient for Main Feedwater Line Rupture

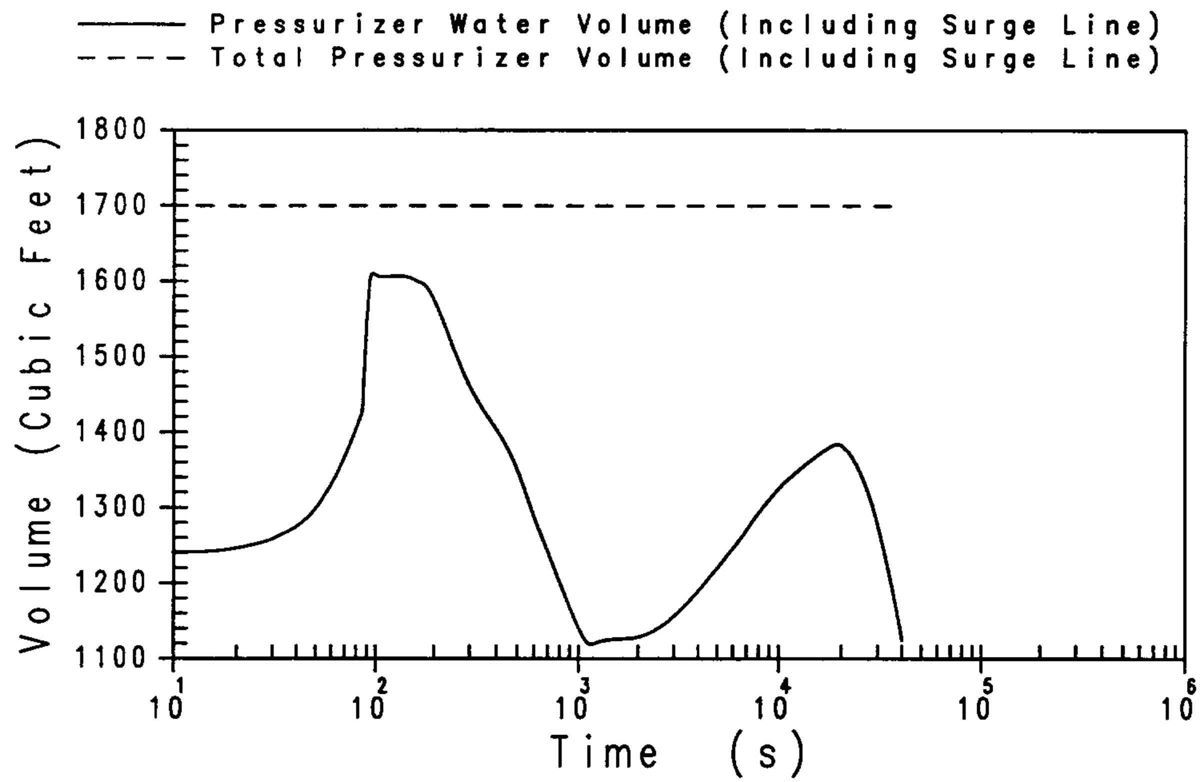

Figure 15.2.8-6

Pressurizer Water Volume Transient for Main Feedwater Line Rupture

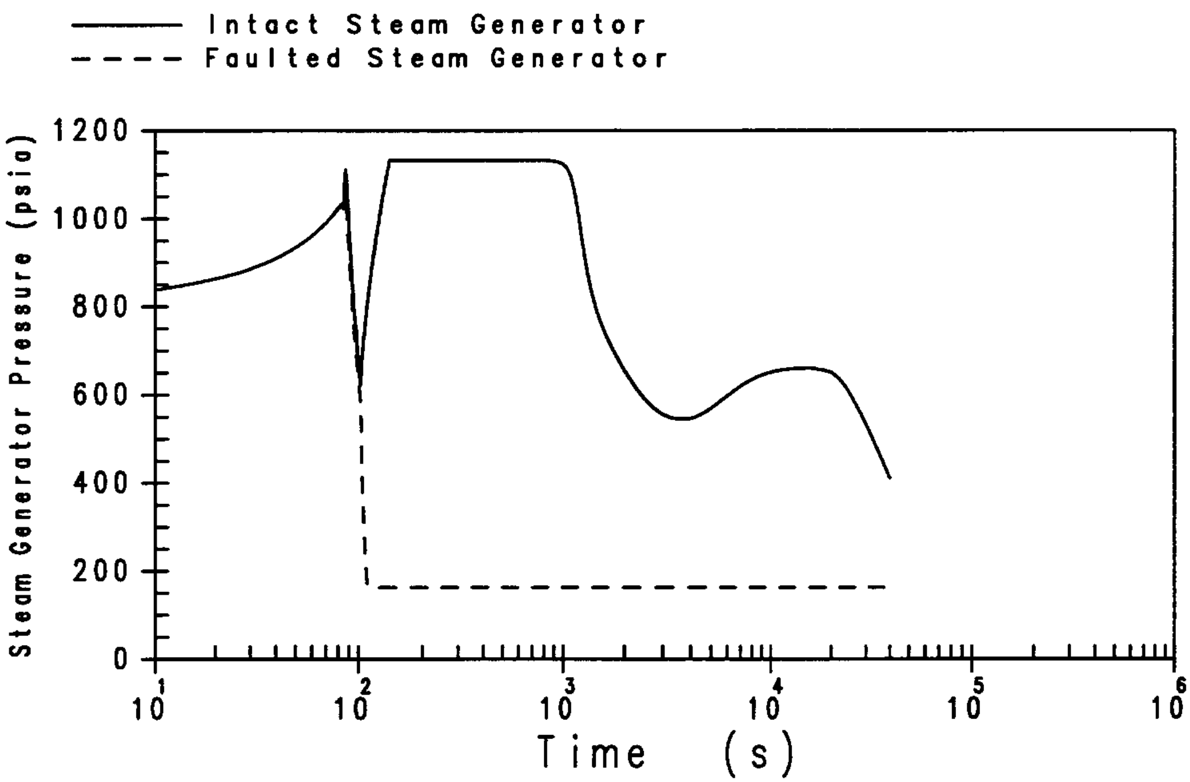

Figure 15.2.8-7

Steam Generator Pressure Transients for Main Feedwater Line Rupture

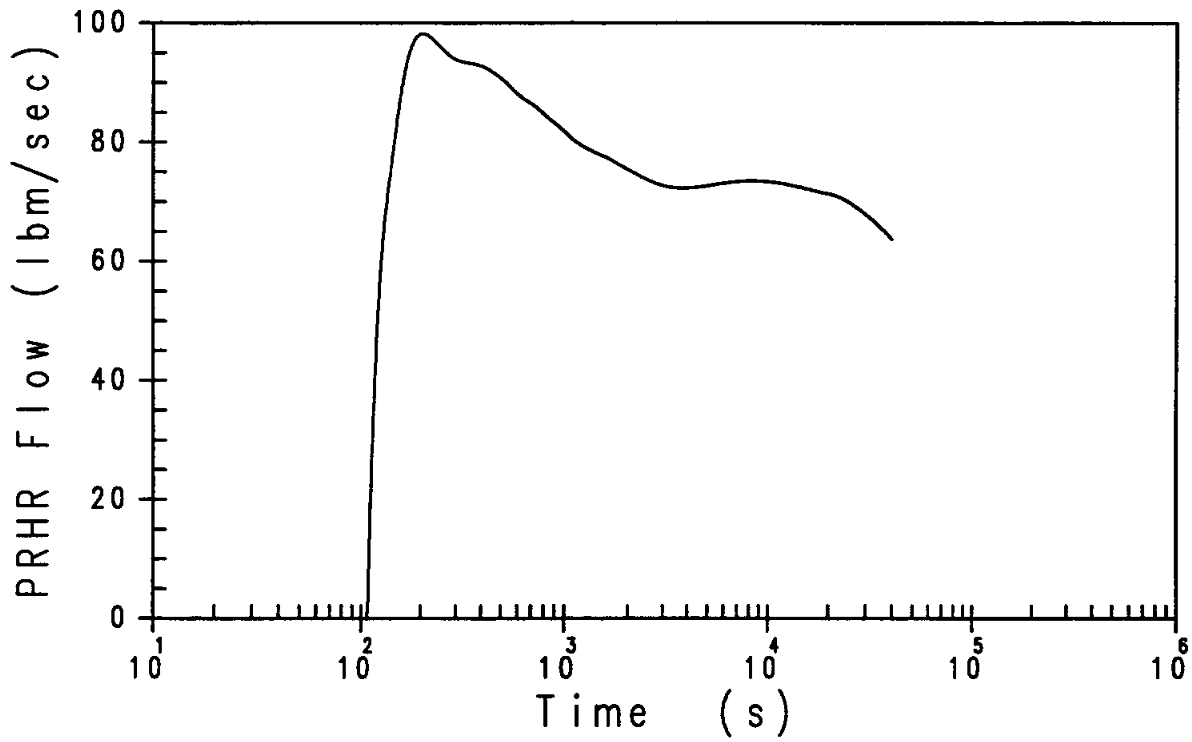

Figure 15.2.8-8

PRHR Flow Rate Transient for Main Feedwater Line Rupture

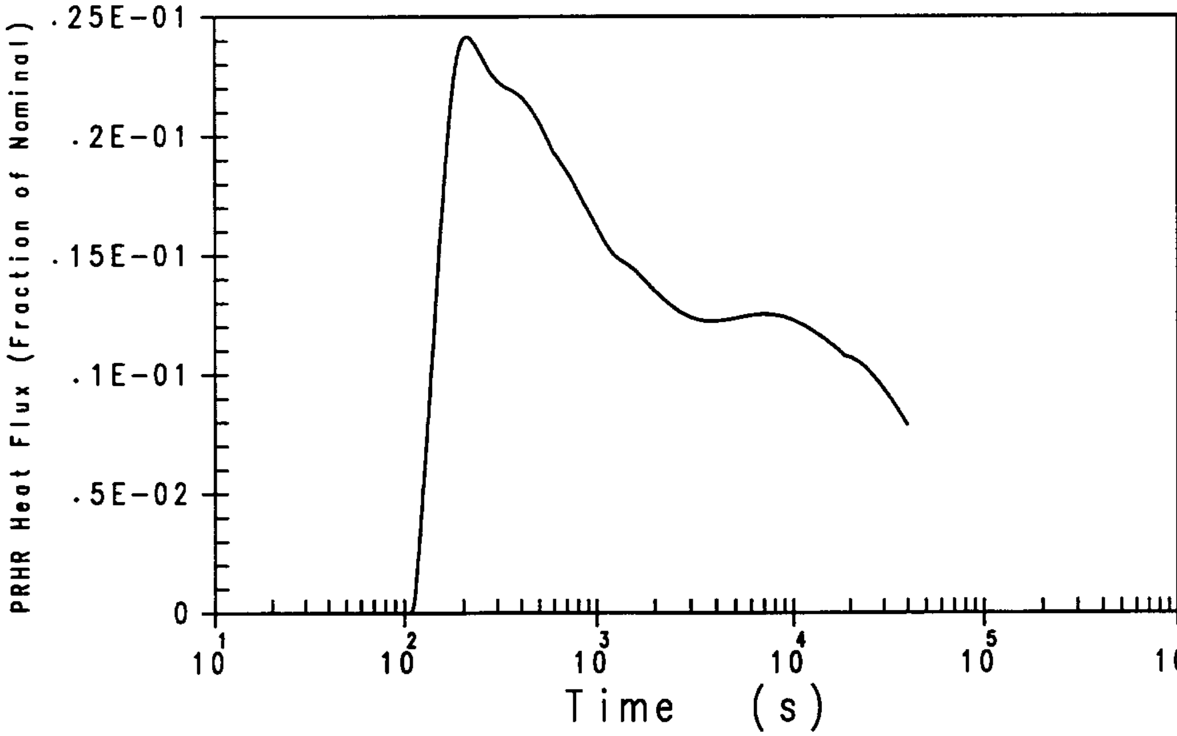

Figure 15.2.8-9

PRHR Heat Flux Transient for Main Feedwater Line Rupture

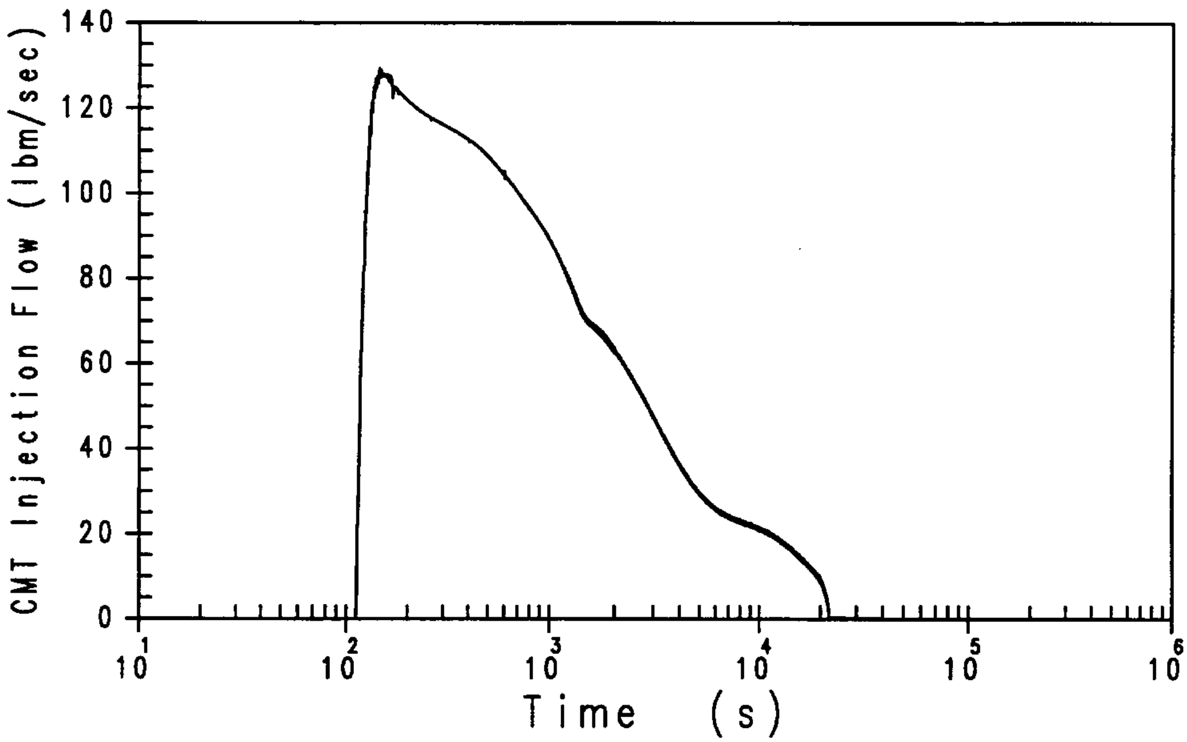

Figure 15.2.8-10

CMT Injection Flow Rate Transient for Main Feedwater Line Rupture