# 15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (TURBINE PLANT)

## 15.1.1 MODERATE FREQUENCY INCIDENTS

### 15.1.1.1 Decrease in Feedwater Temperature

#### 15.1.1.1.1 Identification of Causes and Frequency Classification

→(DRN 02-1479)

The estimated frequency of a decrease in feedwater temperature classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. A decrease in feedwater temperature is caused by the loss of one or more feedwater heaters. The loss could be due to interruption of extraction steam flow or to the opening of a bypass flowpath around one or more heaters. There are three parallel trains of feedwater heaters each consisting of heaters that operate at low, intermediate, and high pressures. The maximum credible reduction in feedwater enthalpy is 140 Btu/lbm and results from the loss of one of the three trains of feedwater heaters.

←(DRN 02-1479)

#### 15.1.1.1.2 Sequence of Events and Systems Operation

A decrease in feedwater temperature causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient and a decrease in the reactor coolant system (RCS) and steam generator pressures. Detection of these conditions is accomplished by the RCS and the steam generator pressure alarms and the high reactor power alarm. If the transient were to result in an approach to specified acceptable fuel design limits, trip signals generated from information provided by the core protection calculators would assure that low departure from nucleate boiling ratio (DNBR) or high local power density limits are not exceeded.

→(DRN 02-1479)

A comparison of the RCS temperatures shows that the maximum RCS temperature decrease rate for the decrease in feedwater temperature event is comparable to that for the increased main steam flow event. Therefore, the system operations described above, and the resulting sequence of events would produce consequences comparable to those following an increase in main steam flow (see Subsection 15.1.1.3). The consequences of a single malfunction of a component or system concurrent with a decrease in feedwater temperature are discussed in Subsection 15.1.2.1.

←(DRN 02-1479)

#### 15.1.1.1.3 Core and System Performance

→(DRN 02-1479)

The core and system performance parameters following a decrease in feedwater temperature would be comparable to those following an increased main steam flow (see Subsection 15.1.1.3).

←(DRN 02-1479)

#### 15.1.1.1.4 Barrier Performance

The barrier performance parameters following a decrease of feedwater temperature are less adverse than those following increased main steam flow (see Subsection 15.1.1.3).

15.1.1.1.5 Radiological Consequences

→(LBDCR 15-039, R309)

The radiological consequences of this event are less severe than those for the inadvertent opening of a steam generator atmospheric dump valve (see Subsection 15.1.2.4.5).

←(LBDCR 15-039, R309)

15.1.1.2 Increase in Feedwater Flow

15.1.1.2.1 Identification of Causes and Frequency Classification

The estimated frequency of an increase in feedwater flow classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. An increase in feedwater flow is caused by:

a) Further opening of a feedwater control valve or an increase in feedwater pump speed.

b) Startup of emergency feedwater with normal feedwater in the manual mode: The Emergency Feedwater System supplies relatively cold water from the condensate storage pool to the steam generators; the starting of this system would simultaneously increase feedwater flow and decrease feedwater temperature. If normal feedwater were in the automatic mode, the feedwater control valves would compensate for the increase in feedwater flow, and startup of the emergency feedwater would only result in a reduction in the feedwater enthalpy of no more than 20 Btu/lb.

15.1.1.2.2 Sequence of Events and System Operation

An increase in feedwater flow causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient, a decrease in the RCS and steam generator pressures and an increase in steam generator water level. Detection of these conditions is accomplished by the pressurizer and steam generator low-pressure alarms, high reactor power alarm, and high steam generator water level alarm. Protection against the violation of specified acceptable fuel design limits following an increase in feedwater flow is provided by the low DNBR and high local power density trips. Protection against high steam generator water level is provided by the high steam generator water level trip.

A comparison of the RCS temperature shows that the maximum RCS temperature decrease rate for the increase in feedwater flow event is no more than that for the increased main steam event. The smaller cooldown rate results, in less power increase and, consequently, in less DNBR decrease during the transient. Therefore, the system operations described above and the resulting sequence of events would produce consequences no more adverse than those following an increase in main steam flow, (see Subsection 15.1.1.3). The consequences of a single malfunction of a component or system following an increase in feedwater flow are discussed in Subsection 15.1.2.2.

15.1.1.2.3 Core and System Performance

The core and system performance parameters following an increase in feedwater flow would be no more adverse than those following an increase in main steam flow. (See Subsection 15.1.1.3).

15.1.1.2.4 Barrier Performance

The barrier performance parameters following an increase in feedwater flow are no more adverse than those following an increase in main steam flow. (See Subsection 15.1.1.3).

15.1.1.2.5 Radiological Consequences
→(LBDCR 15-039, R309)
The radiological consequences of this event are less severe than those of the inadvertent opening of a steam generator atmospheric dump valve. (See Subsection 15.1.2.4.5.)

15.1.1.3 Increased Main Steam Flow

15.1.1.3.1 Identification of Causes and Frequency Classification

→(DRN 06-1062, R15; EC-8458, R307)
The estimated frequency of an increase in steam flow classifies it as a moderate frequency incident as defined in Reference I of Section 15-0. The increase in main steam flow incident results in the most adverse consequences in that it causes the most rapid approach to the specified acceptable fuel design limits considered in Section 15.1. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.
←(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

→(DRN 05-543, R14)
The increase in heat removal by the steam generators as a result of increased main steam flow is defined as any rapid increase in steam generator steam flow, other than a steam line rupture, without a turbine trip. Protection against violation of a fuel design limit as a consequence of the excessive heat removal is provided by the VOPT, low DNBR and high local power density trips. The low steam generator water level trip, high reactor power trip, low steam generator pressure trip, and low pressurizer pressure trip will also serve to protect the plant from exceeding barrier design conditions.
←(DRN 05-543, R14)

An increase in main steam flow may be caused by any one of the following incidents of moderate frequency:

→(DRN 05-543, R14)
a) An inadvertent increase in the opening of the turbine admission valves caused by operator error or turbine load limit malfunction This can result in an additional 11 percent flow from full power conditions.

b) Failure in the Steam Bypass System which could result in an opening of one of the turbine bypass valves. The flowrate of one valve is approximately 12.3 percent of the full power turbine flowrate.

c) An inadvertent opening of an atmospheric dump valve or steam generator safety valve (for a discussion of this occurrence and presentation or results see Subsection 15.1.1.4) caused by operator error or failure within the valve itself. Each atmospheric dump valve can release approximately 5.7 percent of the full power steam flow. A safety valve will pass approximately 11.4 percent of full power steam flow.

The most severe of these incidents is case B.
←(DRN 05-543, R14)

15.1.1.3.2 Sequence of Events and System Operations

→(DRN 05-543, R14)

It is assumed that a failure in the Steam Bypass System (SBS) results in one valve opening and remaining open, even in the presence of closure signals generated by the SBS due to adverse steam generator conditions (e.g., low pressure, and low level) or adverse condenser conditions, until the operator takes action to close the valve or until the main steam isolation valves close. The increased main steam flow causes an increase in core power and heat flux, and a decrease in RCS temperature and pressure. The CPC VOPT will prevent the violation of fuel thermal limits. The Emergency Feedwater System in conjunction with the low steam generator water level trip signal will maintain adequate inventory in the generators. The closure of the main steam isolation valves following the low steam generator pressure signal will terminate the steam flow from the turbine bypass valve. The increased main steam flow incident results in the most adverse consequences of any of the moderate frequency incidents considered in Section 15.1 in that it causes the most rapid approach to the DNBR limit.

←(DRN 05-543, R14)

Table 15.1-1 gives the sequence of events from the generation of a "quick open" signal to the final stabilized condition.

→(DRN 00-1731; 05-543, R14)

←(DRN 00-1731; 05-543, R14)

15.1.1.3.3 Core and System Performance

15.1.1.3.3.1 Mathematical Model

→(DRN 05-543, R14; EC-13881, R304)

The NSSS response to an increased main steam flow was simulated using the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was calculated using the CETOP computer program with the WSSV-T and ABB-NV CHF correlations described in Chapter 4.

←(DRN 05-543, R14; EC-13881, R304)

15.1.1.3.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to an increased main steam flow are discussed in Section 15-0. In particular, those parameters, which were unique to the analysis discussed below, are listed in Table 15.1-2.

→(DRN 05-543, R14)

The initial conditions for the principal process variables monitored by the Core Operating Limit Supervisory System (COLSS) were varied within the reactor operating space given in Table 15.0-4 to determine the set of conditions that would produce the most adverse consequences following an increased main steam flow. Various combinations of initial core inlet temperature, core inlet flowrate, and pressurizer pressure were considered. Varying the core flowrate had very little effect on the transient. Increasing the core inlet temperature resulted in a more rapid approach to the fuel design limit and also maximized the steam generator pressure. Increasing initial RCS pressure delayed the possible occurrence of a Low Pressurizer Pressure Trip. This resulted in greater steam releases. Various combinations of moderator temperature coefficient, axial shapes and peaking factors, each set of which represents a COLSS limit, were also considered. That combination of the above parameters which results in

←(DRN 05-543, R14)

→(DRN 05-543, R14)
the lowest initial margin to DNB was chosen. For modeling VOPT, a cold leg RTD response time of 8 seconds was included along with a conservative CPC trip delay time of 0.50 seconds.

15.1.1.3.3.3 Results

The dynamic behavior of important NSSS parameters following an increase main steam flow is presented in Figures 15.1-1 through 15.1-11. The excess heat removal that occurs as a result of the opening of one of the turbine bypass valves results in a decrease in steam generator pressure and temperature. This decrease causes an increase in the RCS steam generator temperature difference which results in more heat being transferred to the steam generator than is being produced in the core. This in turn causes a decrease in the primary system temperature and pressure. The core power and consequently the heat flux then increases due to the negative moderator temperature coefficient of reactivity. The decreasing RCS pressure along with the increasing core power results in a reduction of the DNBR.

→(EC-13881, R304, LBDCR 15-039, R309)
A VOPT is generated at 23.9 seconds due to the increasing core power. At 24.4 seconds the trip breakers open and the CEAs begin to drop into the core at 25.0 seconds. The turbine admission valves begin to close at 24.6 seconds. The maximum core power of 109.6% occurs at 25.0 seconds and the maximum heat flux of 109.1% occurs at 25.1 seconds. The minimum DNBR occurs at 25.2 seconds and remains above the DNBR SAFDL of 1.24.
←(EC-13881, R304)

The cooldown continues as a result of more energy being released by the turbine bypass valve than is being produced in the core until at 347.1 seconds the low steam generator pressure setpoint is reached. This initiates a main steam isolation signal which closes the main steam and feedwater isolation valves terminating the uncontrolled cooldown. At 1800 seconds the operator initiates normal cooldown if the malfunction has not been corrected.

→(EC-13881, R304)
The maximum RCS and secondary pressure do not exceed 110 percent of design pressure following an increase in main steam flow. This assures that the integrity of the RCS and main steam system is maintained. The minimum DNBR is greater than 1.24. This ensures no violation of the fuel thermal limits.
←(DRN 05-543, R14; EC-13881, R304)
This event was assessed using the revised SCRAM curve times documented in Table 15.0-5 and it was determined there was a negligible impact to the results.
←(LBDCR 15-039, R309)

15.1.1.3.4 Barrier Performance

15.1.1.3.4.1 Mathematical Model

→(DRN 05-543, R14)
The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.1.1.3.3.
←(DRN 05-543, R14)

15.1.1.3.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of Barrier Performance are identical to those described in Subsection 15.1.1.3.3.

→(DRN 00-592, R11-A; 02-91, R11-A)
←(DRN 00-592, R11-A; 02-91, R11-A)

→(DRN 00-592, R11-A; 02-91, R11-A)
#### 15.1.1.3.4.3 Results
←(DRN 00-592, R11-A; 02-91, R11-A)

→(DRN 05-543, R14)
Steam releases are smaller than those of the inadvertent opening of a steam generator atmospheric dump valve with LOOP (See Subsection 15.1.2.4.5).
←(DRN 05-543, R14)

#### 15.1.1.3.5 Radiological Consequences
→(LBDCR 15-039, R309)
The radiological consequences of this event are less severe than those of the inadvertent opening of a steam generator atmospheric dump valve (see Subsection 15.1.2.4.5).
←(LBDCR 15-039, R309)

### 15.1.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

#### 15.1.1.4.1 Identification of Causes and Frequency Classification

The estimated frequency of an inadvertent opening of a steam generator atmospheric dump valve classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0.

→(DRN 05-543, R14)
An atmospheric dump valve may be inadvertently opened by the operator or may open due to failure in the control system that opens the valve.
←(DRN 05-543, R14)

→(DRN 06-1062, R15; EC-8458, R307)
The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging.
←(DRN 06-1062, R15; EC-8458, R307)

#### 15.1.1.4.2 Sequence of Events and Systems Operation

The worst case inadvertent opening of a steam generator atmospheric dump valve is analyzed at a power level of one MWt, as described in Subsection 15.1.1.4.3.2.

The inadvertent opening of the steam generator atmospheric dump valve results in excessive heat removal from the steam generator. The mass released from the valve is not made up by the feedwater, which is in the manual mode. Therefore, the steam generator water level begins to decrease. The decreasing pressure (and hence temperature) in the affected steam generator results in a greater temperature difference between RCS and steam generator, and hence, more heat is transferred from the RCS. This lowers the RCS temperature and results in an increase in reactor power due to the negative moderator temperature coefficient of reactivity. This increase in power results in the heatup of the RCS, since the heat entering the RCS is greater than that extracted by the steam generator. The pressurizer pressure and RCS temperatures begin to increase. The increase in RCS temperature results in a greater RCS-to-steam generator temperature difference, resulting in more heat being transferred to the steam generator and causing the steam generator temperature and pressure to increase. As the power increases, the fuel temperature increases. As a result, the Doppler reactivity contribution increases. This decreases the positive reactivity and decreases core power and heat flux. Eventually the affected steam generator water level reaches the low level trip setpoint and initiates a reactor trip. The RCS and steam generators then cool at a faster rate as a result of the decrease in core power. At 1800 seconds, the operator takes control of the plant and begins an orderly cooldown using the condenser.

Table 15.1-3 gives the sequence of events from the opening of a steam generator atmospheric dump valve to the time when the operator takes control of the plant.

### 15.1.1.4.3 Core and System Performance

#### 15.1.1.4.3.1 Mathematical Model

→(DRN 04-704, R14)
The NSSS response to an inadvertent opening of a steam generator atmospheric dump valve was simulated with the CENTS computer program described in Section 15.0
←(DRN 04-704, R14)

#### 15.1.1.4.3.2 Input Parameters and Initial Conditions

→(DRN 05-1551, R14)
The assumptions and initial conditions given in Table 15.1-4 and the parameter described in Section 15.0 are used for this analysis. The transient was analyzed at a power level of one MWt to maximize radiological consequences.
←(DRN 05-1551, R14)

→(DRN 04-704, R14)

←(DRN 04-704, R14)

→(DRN 00-592, R11-A; 04-704, R14)

←(DRN 00-592, R11-A; 04-704, R14)

a) The main steam isolation valves and main feedwater regulating valves are closed throughout the transient. This maximizes the secondary side pressure and mass and activity releases.

b) The initial steam generator water level is just below the high level trip setpoint. This maximizes the initial mass and activity available for release to the atmosphere and the time to reactor trip on low steam generator water level.

→(DRN 04-704, R14)
c) A primary-to-secondary leak of 0.375 gallons per minute is assumed for the duration of the transient for the faulted steam generator. A primary-to-secondary leak of 150 gallons per day is assumed for the intact steam generator.
←(DRN 04-704, R14)

15.1.1.4.3.3 Results

→(DRN 05-543, R14; 04-704, R14; EC-13881, R304, LBDCR 15-039, R309)
The dynamic behavior of important NSSS parameters is shown in Figures 15.1-12 through 15.1-19a. The inadvertent opening of the steam generator atmospheric dump valve results in excessive heat removal from the steam generator. The mass released from the valve is not made up by the feedwater, which is in the manual mode. Therefore, the steam generator water level begins to decrease. The affected steam generator pressure begins to decrease due to the excessive heat removal. The decreasing pressure (and hence temperature) in the affected steam generator results in a greater temperature difference between RCS and steam generator and hence in the transfer of more heat from the RCS. This lowers the RCS temperatures and results in an increase in reactor power due to the negative moderator coefficient of reactivity. At about 82.9 seconds, the core power reaches its maximum value of 5.73 percent of rated power. This increase in power results in the heatup of the RCS, since the heat entering the RCS is greater than that extracted by the steam generator. The increase in RCS temperatures results in a greater RCS-to-steam generator temperature difference, resulting in more heat being transferred to the steam generator and causing the steam generator temperature and pressure to stabilize. As the power increases, the fuel temperature increases. As a result, the Doppler reactivity contribution increases. This decreases the positive reactivity and decreases core power and heat flux. At 607.0 seconds, the affected steam generator water level reaches the low level trip setpoint and initiates a reactor trip. The RCS and steam generators then cool at a faster rate as a result of the decrease in core power. At 1800 seconds, the operator takes control of the plant and begins an orderly cooldown using the condenser. Pressurizer pressure and secondary pressure do not exceed 110 percent of design pressure. Since the peak average core heat flux was only 5.69 percent of full power, thermal margin degradation was not significant. The minimum DNBR occurs prior to reactor trip, thus the revised CEA drop time does not impact minimum DNBR results.
←(DRN 05-543, R14; 04-704, R14; EC-13881, R304, LBDCR 15-039, R309)

15.1.1.4.4 Barrier Performance

15.1.1.4.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.1.1.4.3.

15.1.1.4.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.1.1.4.3.

15.1.1.4.4.3 Results

→(DRN 05-543, R14; 04-704, R14)
Figure 15.1-18 shows the steam generator atmospheric dump valve flowrate versus time for this event. At 30 minutes after the atmospheric steam dump valves are opened, approximately 313,300 pounds of steam will have been discharged. The operator will then cool the plant using the condenser.
←(DRN 05-543, R14; 04-704, R14)

15.1.1.4.5 Radiological Consequences

→(DRN 04-704, R14)
The radiological consequences of this event are less severe than the consequences of the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Loss of Offsite Power event (See Subsection 15.1.2.4.5).
←(DRN 04-704, R14)

15.1.2 INFREQUENT INCIDENTS

15.1.2.1 Decrease in Feedwater Temperature with a Concurrent Single Failure of an Active Component

15.1.2.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a decrease in feedwater temperature with a concurrent single failure of an active component classifies it as an infrequent incident as defined in Reference 1 of Section 15.0. A decrease in feedwater temperature is caused by the possibilities described in Subsection 15.1.1.1.1.

15.1.2.1.2 Sequence of Events and Systems Operation
→(DRN 02-1479, R12)
The systems operations following a decrease in feedwater temperature with a concurrent single failure of an active component are the same as those described in Subsection 15.1.1.1.2. The single malfunction of a component or system is discussed in Subsection 15.1.2.3.1 for the increased main steam flow with a concurrent single failure of an active component. The cooldown rate and the resultant sequence of events would produce consequences comparable to those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).

15.1.2.1.3 Core and System Performance

The core and system performance following a decrease in feedwater temperature with a concurrent single failure of an active component would be comparable to those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).

15.1.2.1.4 Barrier Performance

The barrier performance following a decrease in feedwater temperature with a concurrent single failure of an active component would be comparable to those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).
←(DRN 02-1479, R12)
15.1.2.1.5 Radiological Consequences

The radiological consequences of this event are less severe than the results of the inadvertent opening of a steam generator atmospheric dump valve with a concurrent loss of offsite power. (See Subsection 15.1.2.4.5).

15.1.2.2 Increase in Feedwater Flow with a Concurrent Single Failure of an Active Component

15.1.2.2.1 Identification of Causes and Frequency Classification

The estimated frequency of an increase in feedwater flow with a concurrent single failure of an active component classifies it as an infrequent incident as defined in Reference 1 of Section 15.0. Possible causes of increase in feedwater flow are described in Subsection 15.1.1.2.1.

15.1.2.2.2 Sequence of Events and Systems Operation

The systems operations following an increase in feedwater flow with a concurrent single failure of an active component are the same as those described in Subsection 15.1.1.2.2. The single malfunction of a component or system is discussed in Subsection 15.1.2.3.1 for the increased main steam flow with a concurrent single failure of an active component. Because of the lower cooldown rate, the resultant sequence of events would produce consequences no more adverse than those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).

15.1.2.2.3 Core and System Performance

The core and system performance following an increase in feedwater flow with a concurrent single failure of an active component would be no more adverse than those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).

15.1.2.2.4 Barrier Performance

The barrier performance following an increase in feedwater flow with a concurrent single failure of an active component would be less adverse than those following an increased main steam flow with a concurrent single failure of an active component. (See Subsection 15.1.2.3).

15.1.2.2.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve with a concurrent loss of offsite power. (See Subsection 15.1.2.4.5).

15.1.2.3 Increased Main Steam Flow with a Concurrent Loss of Offsite Power

→(DRN 05-543, R14)

←(DRN 05-543, R14)

15.1.2.3.1 Identification of Causes and Frequency Classification

→(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

The estimated frequency of an increased main steam flow with a concurrent single failure of an active component classifies this incident as an infrequent incident as defined in Reference 1 of Section 15.0. The cause of the increased main steam flow is discussed in Subsection 15.1.1.3.1. The results presented in this section are based on the original steam generators, which bound the replacement steam generators with up to 10% steam generator tube plugging, and the revised SCRAM curve times presented in Table 15.0-5.

←(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

A review of potential active component single failures to determine which failure would have the most adverse effect following an increased main steam flow indicates that the following single failures are most limiting: (1) loss of offsite (all normal ac) power at any time during the transient and (2) failure or unavailability of all the condenser circulating water pumps. A parametric analysis has shown that the loss of all normal ac power when a reactor trip condition exists produces the most adverse consequences (minimum DNBR) following an increased main steam flow. This failure is an independent loss of all offsite power.

15.1.2.3.2 Sequence of Events and System Operation

→(DRN 05-543, R14)

The systems that operate following an increased main steam flow with loss of offsite power, when a reactor trip condition exists, are the same as those described in Subsection 15.1.1.3.2 following an increased main steam flow with the following exceptions. The loss of offsite power when a reactor trip condition exists will result in the closure of the turbine bypass valves since power is removed from the solenoids that act to keep the turbine bypass valves open.

Table 15.1-8 gives a typical sequence of events that occur following an increased main steam flow with concurrent loss of offsite power, when a reactor trip condition exists.

←(DRN 05-543, R14)

15.1.2.3.3 Core and System Performance

15.1.2.3.3.1 Mathematical Model

Two different analysis models are included in the discussion of this event:

→(DRN 05-543, R14)

a) Typical: The expected plant response to an increase in main steam flow with Loss of Offsite Power (LOOP) is presented. This simulation includes a modeling of the excess load portion of the event prior to the LOOP occurring simultaneous with reactor trip. For simulation of this scenario, the mathematical modeling is identical to that described in Subsection 15.1.1.3.3.

b) Worst DNB Performance Case: A second scenario is examined to minimize the transient DNBR which could occur during an anticipated operational occurrence with LOOP. For this scenario, it is assumed that the underlying AOO (in this case an excess load) degrades all of the initially preserved thermal margin and the core is at a condition such that the CPCS are just on the verge of generating a low DNBR trip. The event is then modeled as a four pump loss of flow, due to the LOOP, from a steady state condition with the peak pin in the core just above SAFDL conditions.

←(DRN 05-543, R14)

→(DRN 05-543, R14)

For this case, the increased heat removal portion of the transient has not been explicitly modeled. It is conservatively assumed that the increased heat removal due to excess Main Steam flow uses up all thermal margin initially preserved by CPC's and/or COLSS. This assumption is conservative as explicit modeling would demonstrate that thermal margin to the SAFDL would exist at the time of Loss of Offsite Power.

←(DRN 05-543, R14)

After steady-state conditions are reached, it is postulated that the perturbation in the secondary system due to the decreased secondary system pressure results in a turbine trip. A Loss of Offsite Power (LOOP) is assumed to immediately accompany the turbine trip. The LOOP interrupts power to the Reactor Coolant Pumps. The analysis performed is thus equivalent to the analysis of a 4 pump loss of forced reactor flow initiated from conditions at or just above the conditions which would result in a CPC Low DNBR Trip.

→(DRN 05-543, R14; 05-1551, R14)

Having determined the flow coastdown curve representative of the four pump coastdown, it is input into a 1-D HERMITE model in which the initial conditions have been adjusted to be at SAFDL conditions. The 1-D HERMITE model then predicts the core response to the flow reduction and subsequent reactor trip. The thermal hydraulic response of the fuel rods is examined with the CETOP code.

←(DRN 05-543, R14; 05-1551, R14)

15.1.2.3.3.2 Input Parameters and Initial Conditions

→(DRN 05-543, R14)

Table 15.1-08B presents the initial conditions assumed in the case which results in the lowest value of transient DNBR for this event.

a) Typical Case: The input parameters and initial conditions used for this evaluation of core and systems performance are identical to those described in Subsection 15.1.1.3.3. The most negative value of MTC allowed by the COLR ($-4.2 \times 10^{-4}$ $\Delta\rho/°F$) was selected to obtain the highest power rise prior to reactor trip.

b) Worst DNB Performance Case: A set of initial conditions corresponding to the DNBR SAFDL power operating limit was calculated with the CETOP-D computer code. A Moderator Temperature Coefficient (MTC) of $-1.05 \times 10^{-4}$ $\Delta\rho/°F$ is assumed. To use up the initial margin, the increased heat removal due to increased main steam flow must occur with a negative MTC present. However, reactivity feedbacks due to a negative MTC mitigate the effects for a 4 pump loss of flow. For that case, the increased temperature rise across the core prior to reactor trip provides negative reactivity feedback. The limiting MTC is thus a balance between these two competing effects. A parametric study on MTC determined that a value of $-1.05 \times 10^{-4}$ $\Delta\rho/°F$ was the most limiting value.

←(DRN 05-543, R14)

15.1.2.3.3.3 Results

→(DRN 05-543, R14)

a) Typical Case: The dynamic behavior of important NSSS parameters following an increased main steam flow with concurrent loss of offsite power when a reactor trip condition exists is shown on Figures 15.1-20 through 15.1-32. Table 15.1-8 presents the sequence of events for the event.

→(EC-13881, R304, LBDCR 15-039, R309)

The dynamic behavior of the NSSS following an increased main steam flow with loss of offsite power is similar to that of the increased main steam flow shown in Subsection 15.1.1.3 up until the time of trip. At 11.7 seconds, in addition to the generation of a reactor trip, a loss of offsite power is assumed to occur resulting in a four pump coastdown. At 11.8 seconds the core reaches its maximum power of 109 percent and maximum heat flux of 105 percent. The decreasing forced reactor coolant flow results in a minimum DNBR of 1.051 at 14.0 seconds. The steam generator pressure then begins to increase resulting in the opening of the steam generator safety valves at approximately 132 seconds. It is conservatively assumed that operator action to take control of the plant and begin an orderly cooldown is delayed until 30 minutes after first indication of the event.

The peak RCS and main steam system pressures are within 110 percent of design, assuring integrity of the RCS and main steam system is maintained following an increased main steam flow with loss of offsite power concurrent with reactor trip. The minimum DNBR of 1.051 indicates that some fuel pins will have experienced DNB.

←(DRN 05-543, R14; EC-13881, R304)

→(DRN 00-592, R11-A)

b) Worst DNB Performance Case: Table 15.1-8A contains the Sequence of Events for the Increased Main Steam Flow with LOOP for this case. The increased heat removal portion of this event (i.e., increased steam flow) begins at t = 0. The initially preserved thermal margin is reduced until some time $\Delta t$, when the core has reached a condition at or just above the condition which would result in a CPC Low DNBR Trip. The specifics of $\Delta t$ depend upon the rate and severity of the increase in the main steam flow.

←(DRN 00-592, R11-A)

→(DRN 05-543, R14)

At time $\Delta t$ the Loss of Offsite Power is assumed to occur. The CPC's sense the decreased RCS flow and generate a Low DNBR Trip. Power to the holding coils is removed at $\Delta t$ + 0.34 seconds. At $\Delta t$ + 0.94 seconds the CEA's begin to drop into the core.

→(EC-13881, R304)

For an MTC of $-1.05 \times 10^{-4}$ $\Delta\rho/°F$, the minimum DNBR value reached during the transient is 1.051.

←(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

→(DRN 05-543, R14, LBDCR 15-039, R309)

At approximately ΔT + 120 seconds the steam generator safety valves lift and a maximum steam generator pressure of 1103 psia occurs. The peak Main Steam System (MSS) pressure is within 110% of design, assuring the integrity of the MSS. It is assumed that operator action to begin an orderly cooldown of the plant is delayed until 30 minutes into the event.

Less than 8% of the fuel pins are calculated to briefly experience DNB during this event. This value is based upon the method of statistical convolution.

←(DRN 05-543, R14, LBDCR, R309)

15.1.2.3.4 Barrier Performance

15.1.2.3.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.1.1.3.3.

15.1.2.3.4.2 Input Parameters and Initial Conditions

→(DRN 04-704, R14)

For determining barrier performance for the Increased Main Steam Flow event with a Concurrent Loss of Offsite Power, the secondary steaming model of FSAR Section 15.B applies. A primary-to-secondary leak rate of 150 gallons per day per steam generator is assumed.

←(DRN 04-704, R14)

15.1.2.3.5 Radiological Consequences

→(DRN 04-704, R14)

15.1.2.3.5.1 Design Basis – Method of Analysis

→(EC-40444, R307)

For the Increased Main Steam Flow with LOOP dose analysis, the only fission product release path that needs to be considered is fission product releases via SG steaming (releases via ADVs or MSSVs). Per RG 1.183 Table 6, this pathway is analyzed until cold shutdown is established. The secondary steaming pathway assumes that an Increased Main Steam Flow event resulted in a reactor scram and that releases are occurring due to primary-to-secondary leakage from the RCS to the SG. Activity is then released to the environment through the use of the ADVs to remove decay heat and to cool the plant to shutdown cooling entry conditions. Since the control room $\chi$/Q values are worst for ADV releases, any releases which would occur through the MSSVs are instead assumed released from the ADV locations. Once shutdown cooling is initiated, the release to the environment is terminated for this pathway. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.

←(EC-40444, R307)

→(EC-5000081470, R301)

For the Increased Main Steam Flow event, the reactor is assumed to be at a power operating limit due to the excess steam demand transient for some period of time prior to the postulated LOOP. It is assumed that a transient initially occurs that degrades all the thermal margin preserved by the Core Operating Limits Supervisory System (COLSS) and brings the hot channel to DNB Ratio (DNBR) Specified Acceptable Fuel Design Limit (SAFDL) conditions. Because of this pre-trip power excursion, it is possible that the reactor could be operating above 100% rated thermal power for some period prior to the reactor trip on LOOP. The RPS includes a trip on Linear Power Level – High of 108% of rated thermal power, per Technical Specification Table 2.2-1. Thus, to account for the impact of this relatively short duration potential power excursion at the start of the event, a 10% transient overpower factor is applied to decay heat in generating steam releases.

←(DRN 04-704, R14; EC-5000081470, R301)

➔(DRN 04-704, R14)

15.1.2.3.5.2 Input Parameters and Assumptions

The input parameters and assumptions for the Increased Main Steam Flow analysis are listed in Table 15.1-27. Certain inputs and assumptions are discussed in additional detail below.

a) Source Term

As a result of the Increased Main Steam Flow analysis, an 8% fuel failure was reported in the EPU Licensing Amendment Request. Therefore, the maximum assumed fuel failure mechanism is up to 8% of the fuel rods in the core experiencing DNB. The non-LOCA gap fractions specified in Table 3 (Footnote 11) of RG 1.183 are conservatively selected for use in the Increased Main Steam Flow analysis to provide the most conservative set of results. These gap fractions are 10% for iodines and noble gases and 12% for alkali metals. Thus, this analysis conservatively applies the larger gap fractions required for CEA Ejection rather than the smaller gap fractions in the table itself which would apply for an Increased Main Steam Flow event.

b) Iodine Chemical Form

For the secondary steaming release pathway, the iodine releases fro the SG to the environment are assumed to be 97% elemental iodine and 3% organic iodine. This is consistent wit the guidelines provided in RG 1.183.

c) Release Pathways

➔(EC-40444, R307)

Conservatively, all the iodine, alkali metal and noble gas activity due to the postulated Increased Main Steam Flow transient is assumed to be in the primary coolant when determining the dose consequences due to primary-to-secondary SG leakage and subsequent secondary steaming. Releases, for cooldown purpose from ADVs, are assumed to be terminated once shutdown cooling is initiated and the SGs are no longer providing decay heat removal (no further releases would occur due to cooldown to cold shutdown conditions). However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions. This is consistent with the guidelines provided in Table 6 of RG 1.183.

←(EC-40444, R307)

➔(EC-5000081470, R301)

For the purposes of the Increased Main Steam Flow analysis, a primary-to-secondary SG leakage of 150 gpd per SG is assumed. This value is consistent with the Technical Specification limit of 75 gpd.

Operator actions to select the more favorable of the control room air intakes in terms of $\chi$/Q are credited.

←(EC-5000081470, R301)

d) Removal Coefficients

➔(DRN 05-1551, R14)

For the secondary steaming path, iodine and alkali metal releases to the secondary side via primary-to-secondary side SG leakage are assumed to be subject to a Partition Factor (PF). Consistent with RG 1.183, Section 5.5, A PF of 100 is assumed for iodines and alkali metals. For the sake of conservatism, a PF of 10 is assumed for the first 30 minutes of the event to account for potential elevated releases due to the initial transient.

←(DRN 05-1551, R14)

Per RG 1.183, all noble gases released to the secondary side via primary-to-secondary side SG leakage are assumed to be immediately released to the environment.

←(DRN 04-704, R14)

→(DRN 04-704, R14)

e) Main Control Room Model

→(EC-5000081470, R301)

The MCR ventilation model is described in FSAR Section 15.B. The Increased Main Steam Flow dose model for secondary steaming release pathway conservatively assumes a constant unfiltered in-leakage of 100 CFM for the entire duration of the secondary steaming release (7.5 hours).

←(EC-5000081470, R301)

15.1.2.3.5.2 Results

The potential radiological consequences resulting from this event have been analyzed, using assumptions and models described in previous sections. These results are listed in Table 15.1-28.

←(DRN 04-704, R14)

15.1.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component

15.1.2.4.1 Identification of Causes and Frequency Classification

→(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

The estimated frequency of this event with a concurrent single failure of an active component classifies it as an infrequent frequency incident, as defined in Reference 1 of Section 15.0. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging, as well as an assessment for peak pressure and evaluation for DNBR using the revised SCRAM curve times presented in Table 15.0-5.

←(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

15.1.2.4.2 Sequence of Events and Systems Operation

→(DRN 04-524, R13; 05-543, R14)

The systems operations following this event with a concurrent single failure of an active component are the same as those described in Subsection 15.1.1.4.2. The sequence of events for the inadvertent opening of the ADV with a concurrent LOOP is given in Table 15.1-8C. The resultant sequence of events would not result in a violation of the DNBR SAFDL.

←(DRN 04-524, R13)

15.1.2.4.3 Core and System Performance

15.1.2.4.3.1 Mathematical Model

→(EC-13881, R304; EC-8458, R307)

The CENTS code is used to model the NSSS response to the inadvertent opening of the atmospheric dump valve. The combined steam flow through the turbine and the ADV degrades core thermal margin but is insufficient to bring plant conditions to the point of reactor trip. CETOP is used to quantify the thermal margin degradation from event initiation to the steady state condition resulting from the increased steam flow due to the open ADV.

←(EC-13881, R304)

A 1-D HERMITE model is adjusted to the core conditions existing in the steady state, increased steam flow condition. Upon Loss of Offsite Power, a four pump flow coastdown is initiated in the 1-D HERMITE model and the core response to the flow coastdown and reactor trip is determined. The further thermal margin degradation during the flow coastdown and trip is quantified with the CETOP code.

←(EC-8458, R307)

15.1.2.4.3.2 Input Parameters and Initial Conditions

The initial conditions used to analyze the inadvertent opening of a steam generator atmospheric dump valve with loss of offsite power are presented in Table 15.1-8D. Two different MTCs are used in the analysis. Quantification of the new steady state resulting from the increased steam flow was performed with a $-4.2 \times 10^{-4}$ $\Delta\rho/°F$. This results in the most rapid increase in core power while minimizing the offsetting reductions in core inlet temperature. A more negative MTC is used to exacerbate the NSSS response. (Use of MTC=$-1.5 \times 10^{-4}$ $\Delta\rho/°F$ would have greatly increased the time to reach quasi-steady

←(DRN 05-543, R14)

→(DRN 05-543, R14)
state after the IOADV. Additionally, the magnitude of several NSSS responses to the IOADV portion of the event would have been smaller and therefore less significant.) However, reactivity feedbacks associated with the most negative MTC would mitigate the effects of the four pump loss of flow. Thus, the 1-D HERMITE simulation of that portion of the transient is analyzed with a $-1.05 \times 10^{-4}$ $\Delta\rho/°F$.

15.1.2.4.3.3 Results

The dynamic behavior of the NSSS to the inadvertent opening of a steam generator atmospheric dump valve with loss of offsite power is seen in Figures 15.1-32a through 15.1.32m. Following the opening of the atmospheric dump valve, the RCS initially undergoes a cooldown and associated pressure reduction. The minimum pressure is reached at 21.2 seconds at which time the increasing core power and pressurizer pressure control system act to restore RCS pressure towards its initial value. At 91.6 seconds, the core power has achieved a level consistent with the increased steam demand, 106 percent of rated. By 130.3 seconds the core heat flux is also in equilibrium with this new power level.
→(EC-13881, R304, LBDCR 15-039, R309)
At 150 seconds, with the initial thermal margin having been partially degraded by the increased steam flow, a four pump loss of flow is assumed to occur. The low RCP pump speed trip setpoint is reached at 150.5 seconds and a reactor trip is generated on low RCP speed at 150.8 seconds. The shutdown CEAs begin to enter the core at 151.4 seconds and the minimum DNBR occurs at 152.75 seconds and remains greater than the SAFDL limit.

Following reactor and RCP trip, the RCS begins to heat up and increase in pressure until the pressurizer safety valves open at approximately 247 seconds. This limits peak RCS pressure to 2584 PSIA. The steam generator safety valves first open at 279 seconds; opening as necessary until the operators take control of the plant at 1800 seconds to perform an orderly cooldown to shutdown cooling entry conditions.
←(EC-13881, R304, LBDCR 15-039, R309)
→(DRN 05-1551, R14)
Although not modeled or shown in the Sequence of Events, Table 15.1-8C, an EFAS signal on low SG level might be reached prior to 1800 seconds. Should EFAS occur, EFW would not adversely affect DNB performance or releases to the environment. Consequently, EFW was not modeled prior to 1800 seconds.
←(DRN 05-1551, R14)
The peak pressures of both the primary and secondary systems remain below 100% of design pressure. Due to the limited increase in steam flow associated with the atmospheric dump valve, the event, even in combination with a single failure which results in the most rapid degradation of thermal margin, the loss of offsite power, will not violate the DNBR SAFDL.

15.1.2.4.4 Barrier Performance

The barrier performance of the inadvertent opening of a steam generator atmospheric dump valve with loss of offsite power would be bounded by that of the feedwater line break event, Subsection 15.2.3.1.3.1.4, in that

a) No SAFDL violation is predicted to occur, hence no increase from the assumed technical specification limits on coolant activity are predicted.

b) The contents of one steam generator are assumed to be released to the environment.
→(DRN 04-704, R14)
c) The assumed steam generator leakage of 0.375 gpm governs the post dryout direct release from the affected steam generator and the indirect release via steaming from the unaffected steam generator. A steam generator primary-to-secondary leakage of 150 gallons per day is assumed for the intact steam generator.
←(DRN 04-704, R14)

d) The operators are assumed to have only the atmospheric dump valve in the unaffected steam generator available to perform the controlled cooldown.
←(DRN 05-543, R14)

15.1.2.4.5 Radiological Consequences

15.1.2.4.5.1 Physical Model

→(DRN 04-704, R14)

For the IOADV event, the ADV associated with one steam generator (SG) is assumed to open and remain open throughout the event. The event scenario bounds all initial power levels and bounds cases both with and without a loss of offsite power. The ADV of the unaffected SG is used to cooldown the plant after reactor trip until shutdown cooling is initiated. The IOADV event is documented/quantified in terms of the dose assessment for the Feedwater Line Break (FWLB) event (See Subsection 15.2.3.1.3.1.5). That is, the releases are assumed to be bounded by the releases associated with that event. The IOADV sequences are quantified in terms of the FWLB event since the plant response and accident progression characteristics for these two events are similar, as modeled for dose analyses. The faulted or affected SG share sufficient plant response characteristics that their releases can be quantified as a single release applicable to both scenarios. In both cases, the unaffected SG is used to cooldown and depressurize the primary system to cold shutdown conditions using the ADV to remove decay heat. However, for the FWLB, the unaffected SG undergoes significant depletion of its inventory prior to reactor trip. In the FWLB event, the SG time constant is calculated using the mass versus time information. Additionally, the iodine decontamination factor is held at unity for the four hours it takes for the unaffected SG to have its tubes covered. The operators are assumed to not commence the cooldown of the RCS until this 4 hour time period. The model maintains for reactor coolant pumps in operation until the plant reaches 500°F in this delayed cooldown, maximizing steam releases due to the increased heat load. Therefore, it is conservative to use the FWLB releases fro the unaffected SG for this analysis. No fuel failure is predicted for the IOADV or the FWLB.

The analysis also considers the secondary steaming pathway from the intact SG. This is a minor contributor compared to the releases from the affected SG, contributing less than 1% to the overall doses. The operators are assumed to not commence cooldown for this event until 4 hours into the event, when the level has recovered to above the top of the U-tubes; this is consistent with the assumptions on PF. At that point, a cooldown of 50°F/hr is assumed such that shutdown cooling is entered and releases from the intact SG are secured at 8 hours into the event. Note that the cooldown analysis neglects any cooldown that occurs due to the stuck open ADV. Also note that for IOADV, level would be recovered in significantly less time that the 4 hours assumed herein. That level recovery time is based on FWLB analyses. The doses due to releases from the affected and unaffected (intact) SGs are added to obtain the total dose.

←(DRN 04-704, R14)

15.1.2.4.5.2 Assumptions, Parameters, and Calculational Methods

→(DRN 04-704, R14)

The major assumptions, parameters, and calculational methods used to evaluate the radiological consequences are given Table 15.1-10. Additional clarification is provided as follows:

The results of this analysis are developed to be applicable for both Hot Full Power (HFP) and Hot Zero Power (HZP) conditions. A HZP steam generator inventory is assumed for radiological releases calculations to maximize the release; however, assumptions for partition factor (PF) are based on assuming the smaller HFP mass inventory.

a) Source Term

→(DRN 05-1551, R14)

No fuel damage is predicted for the IOADV event (as is also the case for FWLB). Two cases of iodine spiking are considered:

←(DRN 04-704, R14; 05-1551, R14)

→(DRN 04-704, R14)

1. A PIS case where a reactor transient is postulated to have occurred prior to the event and has raised primary coolant iodine concentration to the maximum value (60 μCi/gm DEI-131) allowed per Technical Specifications.

2. A GIS case, where the primary system transient associated with the event causes an iodine spike in the primary system. A spiking factor of 500 is assumed.

A maximum RCS DEI-131 activity of 1.0 μCi/gm is assumed. Initial reactor coolant isotopic activity distribution is per FSAR Table 11.1-2. This distribution is based on predicted activity distributions for power uprate conditions and is applicable for events for which no fuel failure will occur. The activity of FSAR Table 11.1-2 is adjusted to correspond to the Technical Specification activity limit of 100/E μCi/gm.

A maximum Technical Specification secondary activity of 0.1 μCi/gm DEI-131 is assumed. Since no fuel failure is postulated, the small contribution to dose consequences from alkali metals has been ignored.

b) Iodine Chemical Form

Per RG 1.183, iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. Since the same filter efficiency is specified for all iodine forms, there is no impact on the results of this assumption.

c) Release Pathways

→(EC-40444, R307, LBDCR 15-039, R309)

Consistent with RG 1.183, Table 6, guidance for the Main Steam Line Break (MSLB) and the modeling for the FWLB, the IOADV event is analyzed for a release duration until cold shutdown is established. This corresponds to terminating releases at the point of shutdown cooling initiation for the unaffected SG, and a period of 36 hours for the affected SG. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions. As discussed in FSAR Section 9.3.6.3.3, Waterford 3 can achieve cold shutdown under natural circulation conditions within the requirements of BTP RSB 5-4, which includes achieving cold shutdown within 36 hours.

←(EC-40444, R307, LBDCR 15-039, R309)

The release pathway for the intact SG is due to secondary steaming with radiological releases resulting from primary-to-secondary leakage from the RCS into the SGs. Activity is assumed to be released to the environment through the use of the ADVs to remove decay heat and to cool the plant to cold shutdown. Once cold shutdown is reached, the release to the environment is terminated. The doses due to releases from the affected and unaffected (intact) SGs are added to obtain the total dose.

Steam releases from the affected SG may be greater than the inventory of that SG alone, due to releases from the intact SG backflowing through steam lines prior to MSIV closure. The initial steam generator inventory is based on assuming a conservative Hot Zero Power (HZP) SG liquid inventory of 262, 586 lb and the Technical Specification limit of 0.1 μCi/gm dose equivalent I-131. It is conservatively assumed that one-third of the intact steam generator blows down out the break before it is isolated.

The release pathway for the affected SG is due to primary-to-secondary leakage from the RCS to the secondary side of the SGs. Due to the postulated open ADV, activity transferred from the secondary side is assumed to be released immediately to the environment. Activity is assumed released until cold shutdown is achieved.

←(DRN 04-704, R14)

→(DRN 04-704, R14; EC-8458, R307)

The primary-to-secondary leak rate for the intact SG is assumed to be 150 gpd. The primary-to-secondary leak rate for the faulted SG is assumed to be 540 gpd.

←(EC-8458, R307)

d) Removal Coefficients

A PF of 1.0 was assumed for the duration of the event for the affected SG. For the intact SG, a PF of 1.0 was assumed for the first 4 hours to account for the inventory loss due to the initial transient and the time to recover the SG U-tubes. A PF of 100 for the non-faulted SG is subsequently assumed.

→(EC-5000081470, R301)

e) The MCR ventilation model is described in Section 1.3 of Reference 6. The FWLB/IADVO dose model for secondary steaming assumes an unfiltered in-leakage of 100 CFM for the event duration. It is conservatively assumed that the pressurized mode is initiated at the start of the event. The filtered intake in the pressurized mode is assumed to be 225 CFM. The out-leakage from the control room envelope was assumed to be equal to the unfiltered in-leakage rate (100 CFM).

←(EC-5000081470, R301)

15.1.2.4.5.3 Conclusions

The potential radiological consequences resulting from this event have been analyzed, using assumptions and models described in previous sections. These results are listed in Table 15.1-11.

←(DRN 04-704, R14)

15.1.3 LIMITING FAULTS

15.1.3.1 Steam System Piping Failures Post-Trip Return-to-Power

15.1.3.1.1 Identification of Causes and Frequency Classification

→(DRN 05-543, R14; 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

The estimated frequency of a steam line break classifies it as a limiting fault as defined in Reference 1 of Section 15.0. A steam line break is defined as a pipe break in the Main Steam System. Section 15.1.3.1 evaluates performance and fuel integrity with respect to reactivity control after reactor trip (“Return-to-Power”) whereas Section 15.1.3.3 evaluates the thermal performance and fuel integrity due to the initial MSLB transient (“Pre-Trip Power Excursion”). The radiological consequences are evaluated in a combined manner in Section 15.1.3.3. Results of the evaluation presented in this section are valid for up to 10% of the steam generator tubes plugged for the Replacement Steam Generators. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.

←(DRN 05-543, R14; 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

15.1.3.1.2 Sequence of Events and Systems Operation

The increase in steam flow resulting from a pipe break in the Main Steam System causes an increase in energy removal by the affected steam generator from the Reactor Coolant System. This results in a reduction of the reactor coolant temperature and pressure. With a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The reactor trips which may occur due to a steam line break (assuming no loss of offsite power) are:

Low steam generator pressure

CPC - High variable overpower

Low steam generator water level

High linear power level

Low primary system pressure

High containment pressure

➔(DRN 05-1551, R14)
Where a concurrent loss of offsite power is assumed, a reactor trip may also be caused by an RCS steam generator ΔP low flow trip or a low DNBR trip. For any reactor trip, the CEA of maximum worth is assumed to remain fully withdrawn. The low steam generator pressure signal also initiates a main steam isolation signal (MSIS) which closes the main steam isolation valves (MSIV) and main feedwater isolation valves. The cooldown and contraction of the primary coolant empties the pressurizer and initiates a safety injection actuation signal (SIAS). The emptying of the steam generator associated with the ruptured steam line and the initiation of safety injection boron cause a decrease in core reactivity. Potential single failures that have a significant impact on the SLB transient are: (1) failure of a main steam isolation valve to close after MSIS which results in continued blowdown of the intact steam generator and (2) failure of a high pressure safety injection (HPSI) pump to start after SIAS. An analysis has shown that the failure of a HPSI pump to start on SIAS has the most adverse effect (higher post-trip fission power due to less boron injection) with respect to core damage and radiological consequences. Therefore, one HPSI pump is conservatively assumed to fail. The operator may initiate plant cooldown by manual control of the steam generator atmospheric dump valves associated with the intact steam generator anytime after reactor trip. In this analysis, it is conservatively assumed that operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350°F at which point shutdown cooling is initiated.
🡐(DRN 05-1551, R14)

➔(EC-8458, R307)
Due to the presence of the flow limiting venturis in the main steam system piping, upstream of the containment penetrations, the cooldown rate associated with outside containment break locations is significantly smaller than that for inside containment break locations. The limiting cases for the potential for core damage are therefore the inside the containment breaks.
🡐(EC-8458, R307)

Four scenarios are examined for the return to power steam line break.

- Hot Full Power, Loss of Offsite Power
- Hot Full Power, no Loss of Offsite Power
- Hot Zero Power, Loss of Offsite Power
- Hot Zero Power, no Loss of Offsite Power

Sequences of events for the four steam line break scenarios are given in Table 15.1-12 through 15.1-14B.
🡐(DRN 05-543, R14)

15.1.3.1.3 Core and System Performance

15.1.3.1.3.1 Mathematical Model

➔(DRN 05-543, R14; EC-9533, R302)
The NSSS response to a steam line break was simulated using the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the HRISE computer program and the MacBeth CHF correlation described in References 3 and 4. ANC and HERMITE were used to assess reactivity feedback and core power distribution.
🡐(DRN 05-543, R14; EC-9533, R302)

➔(DRN 05-543, R14)

15.1.3.1.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response of the four inside containment steam line scenarios having the most potential for core damage due to the post trip return-to-power are listed in Tables 15.1-15 through 15.1-16A. The initial conditions over which the initial process variables were varied within the reactor operating space were given in Table 15.0-4. The combinations which provided the most adverse results are listed in Table 15.1-15 through 15.1-16A.

Selection of the maximum RCS pressure has the effect of delaying SIAS, and after SIAS does occur, it tends to minimize the negative reactivity added to the system due to boron from the safety injection system. The positive reactivity added by the moderator during the transient is a function of the change in density of the moderator. Decreasing the initial RCS flow and selecting a maximum initial core inlet temperature initiates the transient at a higher average moderator temperature. This results in the cooldown of the RCS in a range of greater changes in moderator density. This increases the potential for return to power.

Input parameters for HFP and HZP and the bounding physics data include:

➔(DRN 05-1551, R14; EC-8458, R307)

- A double-ended guillotine break causes the greatest cooldown of the RCS and the most severe degradation of shutdown margin. The RSGs have inline nozzle flow restrictors that reduce the effective cross sectional area of a steam line break from 7.88 $ft^2$ to 2.78 $ft^2$. Steam flow from the intact SG is also limited to the 2.78 $ft^2$ effective area of the inline flow restrictors.

🡐(DRN 05-1551, R14; EC-8458, R307)

- A break inside or outside the containment building, upstream of the MSIVs causes a non-isolatable condition in the affected SG.
- A SIAS is actuated when the pressurizer pressure drops below 1560 psia. Time delays associated with the safety injection pump acceleration and valve opening are taken into account. An 18.5-second HPSI response time was assumed for the offsite power available case while a 30-second delay (conservatively greater than that specified in the Technical Requirements Manual [TRM]) was assumed for the LOOP case. Additionally, the event was initiated from the highest pressure allowed by the Technical Specifications to delay the effect of the safety injection boron.
- The cooldown of the RCS is terminated when the affected steam generator blows dry. As the coolant temperatures begin increasing, positive reactivity insertion from moderator reactivity feedback decreases. The decrease in moderator reactivity combined with the negative reactivity inserted via boron injection cause the total reactivity to become more negative.
- CENTS is used to model the reactor coolant pump (RCP) coast down on a LOOP.
- Low SG pressure trip setpoint of 576 psia was assumed with a 0.9-second response time.
- MSIS is actuated on a LSGP setpoint of 576 psia. The MSIVs and Main Feedwater Isolation Valves (MFIVs) all receive an MSIS signal to close. A response time of 8.0 seconds was assumed for the MSIVs.

➔(EC-9533, R302)

- The HERMITE code was used to calculate the reactivity for the post-trip return to power portion of the analysis. This was done since the HERMITE code, which is a three-dimensional, coupled neutronics, open channel thermal hydraulics code, can more accurately model the effects of moderator temperature feedback on the power distribution and reactivity for the critical configuration existing during the return to power. The HERMITE results used in the Waterford 3 analysis were actually obtained from a parametric study performed for Calvert Cliffs Unit 1 Cycle 7. Waterford 3 specific ANC calculations were used to confirm the applicability of these parametric results to Waterford 3.

🡐(DRN 05-543, R14; EC-9533, R302)

➔(DRN 05-543, R14; EC-9533, R302)

- Three-dimensional power distribution peaks (Fq) were determined with the ANC evaluations mentioned above. Axial profiles consistent with these conservative power distribution peaks were utilized in the analysis.

🡐(EC-9533, R302)

- Reactor core thermal margin (DNBR) was simulated using the HRISE computer program, which employed the MacBeth critical heat flux (CHF) correlation and a 1.3 DNBR limit. RCS conditions from CENTS (RCS temperature, pressure, flow and power) are used in the HRISE thermal margin calculations.

➔(DRN 05-1551, R14)

- An EOC Doppler coefficient was assumed. This was based on the most negative fuel temperature coefficient (FTC). This FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the steam line break event.

🡐(DRN 05-1551, R14)

- The delayed neutron fraction assumed is the maximum value including uncertainties for EOC conditions (total delayed neutron fraction, $\beta$, 0.005662).
- A minimum initial RCS flow of 148,000,000 lbm/hr was assumed.
- A maximum initial RCS temperature results in the greatest increase in density of the coolant during the event. This maximizes the positive reactivity added by the moderator. The analytical value of 552°F was used in this analysis.

🡐(DRN 05-543, R14)

Conservative assumptions regarding initial plant conditions and postulated system failures include:

a) End-of-cycle core conditions to yield the most negative moderator temperature coefficient, void coefficient, and Doppler coefficient,

b) Loss of offsite ac power to the plant at the most adverse time. The most adverse time for the loss of offsite ac power to occur was found to be coincident with the steam line break,

c) The CEA of maximum worth stuck in the fully withdrawn position after reactor trip,

d) A failure of one HPSI pump as the worst single active component failure,

e) Feedwater flow at the start of the transient corresponding to initial steady-state operation.

➔(DRN 00-1265; 02-1479, R12; 05-543, R14; EC-8458, R307)

Between the occurrence of the break and the termination of main feedwater flow upon closure of the MFIVs following reactor trip, increases in the feedwater delivery to the steam generators are accounted for due to the reductions in back pressure presented to the main feedwater system by the depressurizing steam generators.

🡐(DRN 00-1265; 02-1479, R12; 05-543, R14; EC-8458, R307)

➔(DRN 00-1265; 02-1479, R12)
🡐(DRN 00-1265; 02-1479, R12)

Conservative assumptions regarding parameters used in the analysis include:

a) 100 percent steam with no moisture carryover during the steam generator blowdown to yield the maximum energy removal,

➔(DRN 02-1479, R12)

b) An increase for the slope of the Doppler reactivity versus fuel temperature function accounting for physics uncertainties to assure that the calculation of the reactivity increase due to cooldown of the fuel from its nominal temperature is conservative,

c) An increase for the slope of the moderator reactivity versus coolant temperature function to assure that the calculation of the reactivity increase due to cooldown of the moderator is conservative,

🡐(DRN 02-1479, R12)

d) Moderator reactivity as a function of the lowest cold leg temperature to account conservatively for the effect of uneven temperature distribution on the moderator reactivity,

➔(DRN 02-1479, R12; 05-543, R14; EC-9533, R302)

e) Credit is taken for the local reactivity effects of moderator density changes in the hot channel. This credit is based upon three-dimensional coupled neutronic-thermal hydraulic calculations performed with the HERMITE-TORC code for Calvert Cliffs unit 1 Cycle 7 (Reference 2) which have been confirmed as valid for Waterford 3.

🡐(DRN 02-1479, R12; 05-543, R14; EC-9533, R302)

Assumptions considered as to the worst single active component failure included:

a) Failure of one HPSI pump to start after SIAS,

b) Failure of one main feedwater isolation valve to close after MSIS,

c) Failure of one main steam isolation valve to close after MSIS,

➔(DRN 05-543, R14)

d) Failure of one diesel generator to start after loss of offsite power.

🡐(DRN 05-543, R14)

The worst single active component failure was the failure of one HPSI pump to start, delaying the time for safety injection boron to reach the reactor core.

15.1.3.1.3.3 Results

➔(DRN 05-543, R14; EC-8458, R307)

The dynamic behavior of the important NSSS parameters following a double-ended steam line break inside containment from Hot Full Power with a Loss of Offsite Power is shown in Figures 15.1-33 through 15.1-38g. Table 15.1-12 provides the sequence of events.

Concurrent with a double ended steam line break inside containment, a loss of offsite power is assumed causing the reactor coolant pumps to coast down. At 2.0 seconds, a reactor trip signal on CPCS RCP Shaft Speed is initiated. Reactor trip breakers open at 2.9 seconds and the CEAs begin to drop into the core at 3.5 seconds. The main steam isolation valves are completely closed by 20.9 seconds. The closure of the MSIVs ensures that the blowdown from the unaffected steam generator ceases. The shrinkage results in the decrease in RCS pressure and a Safety Injection Actuation Signal (SIAS) is generated on low pressurizer pressure at 28.9 seconds. Low level in the affected SG will result in the generation of a Emergency Feedwater Actuation Signal. As a result of the closure of the MSIV's, by the time the EFW pumps are ready to deliver emergency

🡐(DRN 05-543, R14; EC-8458, R307)

➔(DRN 05-543, R14; 05-1551, R14; EC-8458, R307)
feedwater, the two SG's will have developed a differential pressure greater than the setpoint used to identify the ruptured SG. Due to this differential pressure, EFW will not be delivered to the ruptured SG. Delivery of emergency feedwater to the intact SG has little effect on the transient as the return to criticality is being driven by the uncontrolled blowdown of the ruptured SG. The minimum DNBR of 1.47 occurs at 170.8 seconds. The affected steam generator empties at 468.3 seconds. Minimum Pressurizer pressure occurs at 541.3 seconds. The Pressurizer does not fully empty at any point during the transient. At 1800 seconds, the operators take control of the plant and begin an orderly cooldown.

➔(EC-9533, R302)
The dynamic behavior of the important NSSS parameters following a double-ended steam line break inside containment from Hot Full Power with no Loss of Offsite Power is shown in Figures 15.1-47 through 15.1-58c. Table 15.1-13 provides the sequence of events.

A double ended steam line break inside containment is assumed. At 5.0 seconds, a reactor trip signal on CPC VOPT is initiated. Reactor trip breakers open at 5.1 seconds and the CEAs begin to drop into the core at 5.7 seconds The main steam isolation valves are completely closed by 25.0 seconds. The closure of the MSIVs ensures that the blowdown from the unaffected steam generator ceases. The cooldown results in shrinkage of the RCS causing the pressurizer to empty at 11.8 seconds. The shrinkage also results in the decrease in RCS pressure and a Safety Injection Actuation Signal (SIAS) is generated on low pressurizer pressure at 45.0 seconds. Low level in the affected SG will result in the generation of an Emergency Feedwater Actuation Signal. As a result of the closure of the MSIV's, by the time the EFW pumps are ready to deliver emergency feedwater, the two SG's will have developed a differential pressure greater than the setpoint used to identify the ruptured SG. Due to this differential pressure, EFW will not be delivered to the ruptured SG. Delivery of emergency feedwater to the intact SG has little effect on the transient as the return to criticality is being driven by the uncontrolled blowdown of the ruptured SG. The affected steam generator empties at 213.7 seconds. At 1800 seconds, the operators take control of the plant and begin an orderly cooldown. There is no return-to-power for this event with the implementation of the Replacement Steam Generators, thus the Peak Linear Heat Rate remains below the SAFDL limit.
🡐(DRN 05-1551, R14; EC-9533, R302)

The dynamic behavior of the important NSSS parameters following a double-ended steam line break inside containment from Hot Zero Power with a Loss of Offsite Power is shown in Figures 15.1-61 through 15.1-72c. Table 15.1-14A provides the sequence of events.

Concurrent with a double ended steam line break inside containment, a loss of offsite power is assumed causing the reactor coolant pumps the coast down. At 2.0 seconds, a reactor trip signal on CPCS RCP Shaft Speed is initiated. Reactor trip breakers open at 2.9 seconds and the CEAs begin to drop into the core at 3.5 seconds. The main steam isolation valves are completely closed by 20.3 seconds. The closure of the MSIVs ensures that the blowdown from the unaffected steam generator ceases. The cooldown results in shrinkage of the RCS causing a decrease in RCS pressure leading to a Safety Injection Signal (SIAS) to be generated on low pressurizer pressure at 22.9 seconds. Low level in the affected SG will result in the generation of a Emergency Feedwater Actuation Signal. As a result of the closure of the MSIV's, the two SG's will have developed a differential pressure greater than the setpoint used to identify the ruptured SG. Due to this differential pressure, EFW will not be delivered to the ruptured SG. As there is little change in the intact SG level for the HZP case, EFW will not be delivered to the intact SG prior to the point of reactivity turn around. The minimum DNBR of 1.61 occurs at 190.4 seconds. The pressurizer empties at 314.3 seconds. At 1800 seconds, the operators take control of the plant and begin an orderly cooldown.
🡐(DRN 05-543, R14; EC-8458, R207)

→(DRN 05-543, R14; EC-8458, R307)
The dynamic behavior of the important NSSS parameters following a double-ended steam line break inside containment from Hot Zero Power with no Loss of Offsite Power is shown in Figures 15.1-72d through 15.1-72o. Table 15.1-14B provides the sequence of events.

A double ended steam line break inside containment is assumed. At 5.0 seconds, a reactor trip signal on CPC VOPT is initiated. Reactor trip breakers open at 5.1 seconds and the CEAs begin to drop into the core at 5.7 seconds. The main steam isolation valves are completely closed by 21.2 seconds. The closure of the MSIVs ensures that the blowdown form the unaffected steam generator ceases. The cooldown results in shrinkage of the RCS causing the pressurizer to empty at 11.2 seconds. The shrinkage also results in the decrease in RCS pressure and a Safety Injection Actuation Signal (SIAS) is generated on low pressurizer pressure at 40.2 seconds. Low level in the affected SG will result in the generation of a Emergency Feedwater Actuation Signal. As a result of the closure of the MSIV's, the two SG's will have developed a differential pressure greater than the setpoint used to identify the ruptured SG. Due to this differential pressure, EFW will not be delivered to the ruptured SG. As there is little change in the intact SG level for the HZP case, EFW will not be delivered to the intact SG prior to the point of reactivity turn around. The peak return to power linear heat rate occurs at 206.7 seconds. This linear heat rate does not result in fuel damage. The affected steam generator empties at a time greater than 450 seconds. At 1800 seconds, the operators take control of the plant and begin an orderly cooldown.
←(EC-8458, R307)

The maximum RCS pressure does not exceed 110 percent of design pressure following a steam line break.

→(EC-8458, R307)
Due to the combination of natural circulation flow and high local power production of the region of the stuck CEA, the LOOP cases have the potential for violation of DNBR SAFDL. The peak local power is sufficiently low that fuel failure will not occur due to violation of the melting SAFDL (21 kW/ft). Evaluations of fuel rods operating at up to 21 kW\ft at thermal hydraulic conditions consistent with the time of peak power during a SLB with LOOP conditions have shown that fuel temperatures are below those which result in fuel melting.
←(EC-8458, R307)

Each reload, the overall reactivity state and power distribution of the core is assessed to ensure that the LOOP cases will not result in more than 2% of the pins in violation of the DNBR SAFDL. For example, cycle specific Scram worth and moderator cooldown reactivity curves are considered to ensure acceptance criteria for fuel failures are maintained.

The no-LOOP cases, by virtue of the maintenance of full forced circulation do not approach the DNBR SAFDL. The peak local power is sufficiently low that fuel failure will not occur due to violation of the fuel melting SAFDL.

Each reload the overall reactivity state and power distribution of the core is assessed to ensure that the no-LOOP cases will not result in violation of the fuel melting SAFDL, i.e., that there is not fuel damage for the cases with no LOOP.

No fuel failure is permitted for the cases of outside containment MSLBs, either with or without LOOP. The only fuel failure permitted to be assumed for the Return-to-Power MSLB is 2% fuel failure for the inside containment case with LOOP.
←(DRN 05-543, R14)
→(LBDCR 15-039, R309)
This event was assessed using the revised SCRAM curve times documented in Table 15.0-5 and it was determined there was an insignificant impact to the results.
←(LBDCR 15-039, R309)

➔(DRN 05-543, R14)

15.1.3.1.4 Barrier Performance and Radiological Consequences

The return-to-power steam line break is combined with the pre-trip portion of the steam line break to develop a combined fuel failure/barrier performance assessment which captures the total radiological consequences of the steam line break. This combined behavior is discussed in Subsection 15.1.3.3.4 and 15.1.3.3.5.
🡐(DRN 05-543, R14)

15.1.3.1.5 Radiological Consequences

➔(DRN 04-704, R14)
The radiological consequences of the outside containment Post-Trip Main Steam Line Break are less severe than the results of a Feedwater System Pipe Break (See Subsection 15.2.3.1.3.1.5).

The radiological consequences of the inside containment Post-Trip Main Steam Line Break are analyzed in conjunction with the Pre-Trip Main Steam Line Break (See Subsection 15.1.3.3.5).

➔(DRN 05-543, R14)
🡐(DRN 05-543, R14)

🡐(DRN 04-704, R14)

➔(DRN 02-1479, R12)

15.1.3.2 Steam System Piping Failures Inside and Outside Containment Modes 3 and 4 With All CEAs on the Bottom

🡐(DRN 02-1479, R12)

15.1.3.2.1 Identification of Causes and Frequency Classification

➔(DRN 02-1479, R12; 05-543, R14
🡐(DRN 02-1479, R12)

➔(DRN 06-1062, R15; EC-8458, R307)
The limiting steam line break is a large steam line break inside containment during Mode 3 operation with loss of offsite power in combination with a single failure and Technical Specification shutdown margin. The analysis of this limiting event is presented in detail. Results of the evaluation presented in this section are valid for up to 10% of the steam generator tubes plugged for the Replacement Steam Generators.
🡐(DRN 06-1062, R15; EC-8458, R307)

Steam line break events during Mode 3 and 4 operation are analyzed to demonstrate the adequacy of the shutdown margin as specified by Technical Specifications 3.1.1.1 (any CEA withdrawn) and 3.1.1.2 (all CEAs inserted) to prevent degradation in fuel performance as a result of post trip return to power. The numerical values of these two shutdown margin requirements are contained in COLR Sections 3.1.1.1 and 3.1.1.2, respectively.

➔(DRN 05-1551, R14)
For current cycles, the any CEA withdrawn shutdown margin requirement is 5.15% delta rho for all values of RCS temperatures above 200°F. For all CEAs inserted, the shutdown margin requirement is, for Tcold greater than 500°F, a constant at 4.6% delta rho. At or below 500°F, the all CEAs inserted shutdown margin decreases linearly with decreasing temperature until 400°F. Below 400°F the all CEAs inserted shutdown margin is constant at 1.5%.
🡐(DRN 05-543, R14; 05-1551, R14)

➔(DRN 05-543, R14)
The moderator cooldown curve for the any CEA withdrawn scenario is identical to the curve used for the HZP analysis, and the shutdown margin requirements are the same as the HZP case. The change in moderator density which adds the positive reactivity to the core is greater at higher RCS temperatures. Thus, a HZP, any CEA withdrawn, case is more adverse than a any CEA withdrawn case initiated from the lower RCS temperatures allowed by Mode 3 and 4.

Therefore, the limiting Mode 3 and 4 cases are the all CEAs inserted cases. Cold leg temperatures above and below 500°F are considered in the following discussions.
🡐(DRN 05-543, R14)

➔(EC-8458, R307)
The largest possible steam line break size is the double ended rupture of a steam line upstream of the main steam isolation valve (MSIV). In the Waterford 3 design, the blowdown area for the affected steam generator is limited by the nozzle flow restrictor to 2.78 $ft^2$. The blowdown area of the intact steam generator is also limited to 2.78 $ft^2$.
🡐(EC-8458, R307)

➔(DRN 05-543, R14)
This case is verified each cycle. Minimum safety injection flow rates, feedwater isolation valve closure time, and steam generator differential pressure (delta p) isolation (lockout) setpoint are employed.
🡐(DRN 05-543, R14)

For steam line breaks initiated at or below Tcold equal to 500°F, the positive reactivity insertion due to cooldown is less than the subcriticality due to the shutdown margin from Technical Specification 3.1.1.2 plus the minimum worth of the most reactive rod without taking credit for safety injection boron. Since the total positive reactivity insertion without taking credit for safety injection boron is less than the initial subcriticality, there is no significant post trip return to power. Thus the adequacy of the Technical Specification 3.1.1.2 shutdown margin is demonstrated in this region.

The results of the analyses show that the shutdown margin is sufficiently large to prevent a post trip return to power for any zero power steam line break.

15.1.3.2.2 Sequence of Events and Systems Operation

Steam line breaks (SLBS) are characterized as cooldown events due to the increased steam flow rate which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). This results in a decrease in reactor coolant temperatures and in RCS and steam generator pressures. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Modes 3 and 4 steam line breaks are initiated from a subcritical reactivity condition. In Modes 3 and 4, detection of the cooldown may be accomplished by the pressurizer low pressure alarm, the steam generator low pressure alarm, the low pressurizer level alarm, or the low steam generator water level alarm. Reactor trip, although not meaningful for the "all rods in" configuration, is provided by loss of power to the CEA holding coils as a result of the loss of offsite power.

For a steam line break that occurs with a concurrent loss of offsite power, the coastdown of the reactor coolant pumps is assumed to be initiated simultaneously. In Mode 3 the depressurization of the affected steam generator results in actuation of a main steam isolation signal (MSIS). This signal closes the MSIVS, isolating the unaffected steam generator from blowdown, and closes the main feedwater isolation valves (MFIVs) terminating main feedwater flow to both steam generators. After the reduction of steam flow that occurs following MSIV closure for Mode 3 steam line breaks (or eventually for Mode 4 steam line breaks), the level in the affected steam generator falls below the low steam generator water level setpoint. This causes emergency feedwater flow to be initiated to both steam generators if in Mode 3. If

→(DRN 05-543, R14)
in Mode 4 the EFAS system may not be available. If the differential pressure between the two steam generators exceeds its setpoint, the flow is isolated to the affected steam generator and diverted to the intact steam generator. For Mode 3 the pressurizer pressure may decrease to the point where a safety injection actuation signal (SIAS) is initiated. The isolation of the unaffected steam generator and subsequent emptying of the affected steam generator terminate the cooldown. The introduction of safety injection boron upon SIAS causes core reactivity to eventually decrease. For Mode 3 or Mode 4 steam line breaks the operator, via the appropriate emergency operating procedures, may initiate plant cooldown by manual control of the atmospheric steam dump valves. Cooldown may be initiated any time after the affected steam generator empties. The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after event initiation. The plant is then cooled to 350°F at which time the shutdown cooling system is manually actuated.
←(DRN 05-543, R14)

A study of single failures that would have an adverse impact on the SLB event has determined that the failure of one of the high pressure safety injection (HPSI) pumps to start following SIAS has the most adverse effect for those cases that result in generation of SIAS.
→(DRN 00-1731)
Note: The maximum allowed MSIV closure time was changed from 3.0 to 7.0 seconds. This is greater than the closure time assumed in this analysis. The impact of the change on this analysis was insignificant because there is sufficient shutdown margin to prevent a significant post trip return to power.
←(DRN 00-1731)

### 15.1.3.2.3 Core and System Performance

#### 15.1.3.2.3.1 Mathematical Models

→(LBDCR 15-039, R309)
The NSSS response to a steam line break was simulated using the CENTS computer program described in Section 15.0.
←(LBDCR 15-039, R309)

#### 15.1.3.2.3.2 Input Parameters and Initial Conditions

→(DRN 02-1479, R12; 05-543, R14; EC-8458, R307)
The initial conditions assumed in the analysis of the NSSS response to a large inside containment steam line break during Mode 3 operation are presented in Table 15.1-21. Above core inlet temperatures of 500°F the shutdown margin is 4.6% delta rho. Below 500°F the required shutdown margin decreases linearly with temperature to 1.5% at 400°F, then remains at 1.5% below 400°F. The initial core inlet temperature of 551°F was selected to demonstrate the adequacy of the shutdown margin above 500°F. Analysis at other initial cold leg temperatures in this region but below 551°F will produce results less severe and parameter trends similar to those presented here. For steam line breaks initiated at and below Tcold equal to 500°F there is no significant post trip return to power since the total reactivity insertion without taking credit for safety injection boron is less than the initial subcriticality. The credited initial subcriticality is the sum of the SHUTDOWN MARGIN Technical Specification, TS 3.1.1.2 and the reactivity worth of the most reactive CEA. This is acceptable as the definition of SHUTDOWN MARGIN has reduced the actual reactivity state of the core by an amount equal to the most reactive CEA due to the possibility of it's failure to insert upon reactor trip, however, the plant may only be in the LCO defined by Technical Specification 3.1.1.2 after having confirmed that all CEAs are in fact fully inserted. Initially four reactor coolant pumps are assumed to be operating, as allowed in Mode 3, to maximize the cooldown. A spectrum of initial RCS pressures were examined. While the higher RCS pressures resulted in a slightly earlier SIAS, the lower initial RCS pressures experienced slightly larger changes in moderator density. It was found that an initial RCS pressure of 1800 PSIA resulted in a slightly smaller margin of subcriticality at the point of peak reactivity. As these Mode 3 steam line break cases do not
←(DRN 02-1479, R12; 05-543, R14; EC-8458, R307)

→(DRN 05-543, R14, LBDCR 15-039, R309)
experience a return to power, the initial core coolant conditions and conditions existing at the time of minimum subcriticality are cycle independent. These conditions are valid to use to perform a reactivity balance with cycle specific reactivity feedback coefficients provided that the results of such an evaluation demonstrates continued subcriticality. The event was analyzed assuming that the coolant temperature decreased from its initial temperature to 212°F, thus conservatively enveloping the final temperature that might be observed in future operating cycles.
←(DRN 05-543, R14, LBDCR 15-039, R309)

→(DRN 02-1479, R12)
#### 15.1.3.2.3.3 Results

→(EC-8458, R307)
The RSG analysis of the Mode 3 Steam Line Break event is bounded by the previous EPU and OSG condition analysis with respect to subcriticality. The results presented below are for the original steam generators which bounds and are applicable to the replacement steam generators.
←(EC-8458, R307)

The dynamic behavior of the salient NSSS parameters following the steam line break event with simultaneous loss of power during Mode 3 operation is presented in Figures 15.1-76 through 15.1-86. The analysis indicates that the subcriticality due to the shutdown margin of 4.1% delta rho plus the minimum worth of the most reactive rod of 1.2% delta rho is sufficient to prevent a return to criticality. Table 15.1-22 summarizes the major events, times, and results for this transient.

Concurrent with the steam line break, a loss of offsite power occurs. At this time an actuation signal for the emergency diesel generators is initiated. Also at this time the CEDM coils are assumed to lose power. At 2.3 seconds the steam generator pressure falls below the main steam isolation signal (MSIS) setpoint of 675 psia resulting in the generation of MSIS, which initiates closure of the MSIVs and MFIVS. The MSIVs close by 13.2 seconds.

The pressurizer empties at 10.6 seconds in the transient. After the pressurizer empties, the RCS pressure continues falling rapidly resulting in SIAS and subsequent inflow of boron into the RCS. Alternately SIAS may be manually actuated by the operator. After the inflow of boron into the RCS, the margin to recriticality continues decreasing, then starts to increase at 247.8 seconds.

→(DRN 05-543, R14)
When the steam generator water level falls below the low steam generator level setpoint an Emergency Feedwater Actuation Signal occurs. Emergency feedwater enters the intact steam generator at 54.0 seconds. The pressure difference between the two steam generators increases above the analysis setpoint of 230 psid before the emergency feedwater can enter the affected steam generator. Hence there is automatic isolation of the emergency feedwater to the affected steam generator prior to delivery.
←(DRN 05-543, R14)

At 247.8 seconds the transient positive reactivity insertion is just under 4.8% delta rho which is less than the subcriticality available for mitigation of the event. This point represents the largest value of positive transient reactivity insertion during the event. Since there is no return to criticality, there is no DNBR concern. Eventually the affected steam generator is expected to blow down to atmospheric pressure. This would terminate further RCS cooldown.
←(DRN 02-1479, R12)

→(DRN 02-1479, R12)
←(DRN 02-1479, R12)
At a maximum of 30 minutes, the operator, via the appropriate emergency operating procedure, initiates a controlled plant cooldown utilizing the atmospheric dump valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.
→(LBDCR 15-039, R309)
Since all CEAs have been inserted at event initiation, the revised CEA drop times, as documented in Table 15.0-5, have no impact on the MSLB event inside and outside containment for Modes 3 and 4.
←(LBDCR 15-039, R309)

15.1.3.2.4 Conclusion

For the large steam line break during Mode 3 or 4 operation with or without a loss of offsite power and in combination with a single failure the shutdown margin is sufficient to prevent a significant post trip return to power, thus there will be no fuel failures.

15.1.3.3 Steam System Piping Failure: Pre-Trip Power Excursion Analysis

15.1.3.3.1 Identification of Causes and Frequency Classification

→(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)
The estimated frequency of a steam line break classifies it as a limiting fault as defined in Reference 1 of Section 15.0. A steam line break is defined as a pipe break in the Main Steam System. Results of the evaluation presented in this section are valid for up to 10% of the steam generator tubes plugged for the Replacement Steam Generators. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.
←(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

15.1.3.3.2 Sequence of Events and Systems Operation
→(DRN 05-543, R14; EC-8458, R307)
The increase in steam flow resulting from a pipe break in the Main Steam System causes an increase in energy removal by the affected steam generator from the Reactor Coolant System (RCS). This results in a reduction of the reactor coolant temperature and pressure. With a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. Prior to a reactor trip, this increase in core reactivity causes the core power to increase. The increase in core power combined with decreases in RCS pressure cause an approach to the DNBR limit. The assumptions for this analysis are chosen to maximize the approach to DNB as a result of the pre-trip power excursion during a steam line break (SLB).
←(DRN 05-543, R14)

Steam line breaks inside containment may be postulated to have break areas up to the cross section of the integral flow restrictors of the RSGs (2.78 $ft^2$). Those SLBs occurring outside the containment building have break areas also limited to an effective break area of 2.78 $ft^2$.
←(EC-8458, R307)

Several Reactor Protection System (RPS) trips are available to mitigate the pre-trip power excursion. These trips are: Low Steam Generator Pressure (LSGP), High Containment Pressure (HCP), Low Steam Generator Water Level (LSGWL), High Linear Power Level (HLPL), Low Pressurizer Pressure (LPP) and the array of Core Protection Calculator (CPC) trips. In crediting the above trip functions for mitigation of the power excursion, some consideration must be given to the location of the break. SLBs inside containment may result in a degradation of the performance of the RPS trips due to the accident related adverse environment.

In addition to the environmental degradation, transient related degradation of certain of the RPS trip functions must be considered. The RPS trips credited for both inside and outside containment pre-trip power excursion SLBs and major considerations in their use are described in Table 15.1-23.

The severity of the power excursion and the response of the RPS trips in mitigating it are dependent upon several competing effects. Smaller break flow areas tend to delay the action of the LSGP function leading to larger power excursions before LSGP trip action. Temperature shadowing of the excore detectors is, however, slower at smaller break sizes leading to a smaller amount of transient decalibration on the HLPL Trip function. A more negative Moderator Temperature Coefficient (MTC) will cause a higher power excursion for a

given amount of RCS cooldown than will a less negative MTC. This will result in a higher power level prior to LSGP trip actuation for the more negative MTC values. A competing effect is that with a more negative MTC, a given break area may result in a faster HLPL trip as the fixed amount of temperature decalibration combined with the faster power excursion will lead to the High Linear Power channels reading a higher power, and cause an earlier reactor trip. The limiting pre-trip power excursion is found by performance of a parametric analysis in both break area and time-in-life (MTC effects) to determine the highest power level reached prior to the response of the various trip functions.

The addition of the CPC Variable Overpower Trip (VOPT) function with the COLSS/CPC improvement program has provided the plant with a temperature shadowing compensated High Power trip function. This trip was seen in the above described parametric analysis to act prior to most of the other trips for early termination of the pre-trip power excursion. With the VOPT trip function, the effect of the increase in power between the generation of the trip and the actual turnaround in core power during insertion of the CEAs became relatively more important and the most severe results are obtained towards the more negative MTC and larger break portions of the parametric analysis.

→(DRN 00-592, R11-A)
As the pre-trip power excursion SLB event degrades margin to DNB prior to reactor trip, no single failure of a system required to mitigate the event is identified that adversely impacts the fuel performance. However, a survey was performed to determine the most adverse timing of a loss of offsite power (and resultant reactor coolant pump coastdown) in combination with the power excursion. Long term plant recovery from the pre-trip power excursion SLB is similar to that for the post-trip return-to-power SLB of Subsection 15.1.3.1.
←(DRN 00-592, R11-A)
→(DRN 00-1822; 05-543, R14, LBDCR 15-039, R309)
The sequence of events for the limiting inside and outside containment pretrip power excursions are given in Tables 15.1-24A and 15.1-24, respectively. The presence of the inline flow restrictors in the main steam lines, upstream of the containment penetrations limits the rate of steam flow out of the break so that the SAFDLs are not violated for breaks downstream of the flow restrictors (outside containment breaks). Therefore, the limiting break locations are inside containment where the full 7.88 $ft^2$ flow area results in power excursions rapid enough to result in SAFDL violation.
←(DRN 00-1822; 05-543, R14, LBDCR 15-039, R309)

### 15.1.3.3.3 Core and System Performance

#### 15.1.3.3.3.1 Mathematical Model
→(DRN 05-543, R14; EC-13881, R304)
The NSSS response to a steam line break was simulated using the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the CETOP computer program described in Section 15.0 and the WSSV-T and ABB-NV correlations described in Chapter 4.
←(DRN 05-543, R14; EC-13881, R304)

15.1.3.3.3.2 Input Parameters and Initial Conditions

→(DRN 00-1822; 05-543, R14)

The initial conditions that resulted in the greatest extent of fuel failure are seen in Table 15.1-25 and 15.1-26 for the inside and outside containment events, respectively. The following clarifications are applicable:

- The BOC Doppler was assumed. This maximizes the power increase prior to reactor trip.
- An EOC delayed neutron fraction and neutron lifetime were assumed.
- The CEA insertion curve accounts for a 0.6-second holding coil delay. A CEA worth of -6.0% $\Delta\rho$ was assumed.

→(DRN 05-1551, R14; EC-8458, R307)

- The analysis is conservatively based on a guillotine SLB with an effective area of 2.78 $ft^2$ for both the outside-containment break, and the Inside Containment (IC) break.

←(DRN 05-1551, R14)

- The VOPT of the CPCS was employed in the analysis. A cold leg RTD response time of 8 seconds was accounted for along with a CPCS trip delay time of 0.630 seconds, which is conservative and exceeds the expected delay time of 0.370 seconds plus the full power nuclear instrument delay time of 5 milliseconds.

←(EC-8458, R307)

- An initial core power of 3735 MWt, based on a rated power of 3716 MWt and a 0.5% uncertainty, was assumed.
- The most negative MTC of $-4.2*10x^{-4}$ $\Delta\rho/°F$ was assumed.

The listed values are the result of an extensive parametric analysis which varied time-in-life (for Moderator Temperature Coefficient variations) timing of loss of offsite power and break size to obtain the worst power excursion. Additional constraints on the values come from the plant limiting conditions for operation. The thermal-hydraulic evaluation of the fuel performance accompanying the event was made with the assumption that the plant was initially operating at a Power Operating Limit (POL). This POL initial condition represents the minimum initial margin to DNB. Thus a degradation in the thermal-hydraulic parameters from this POL initial condition will result in the greatest amount of fuel pins exceeding the DNBR Limit.

←(DRN 00-1822; 05-543, R14)

15.1.3.3.3.3 Results

→(DRN 05-543, R14; EC-13881, R304; EC-8458, R307, LBDCR 15-039, R309)

The dynamic behavior of important plant parameters is seen in Figures 15.1-87 through 15.1-94. The sequence of events for the inside containment break is provided in Table 15.1-24A. Following the break, a CPCS VOPT is generated at 4.53 seconds. At 5.16 seconds the trip breakers open. The plant is assumed to lose offsite power at 5.3 seconds and the reactor coolant pumps begin to coast down. The maximum core power of 116.5% is reached at 5.40 seconds. The maximum heat flux of 108.7% is reached at 5.63 seconds. The minimum DNBR of 1.365 is reached at 6.4 seconds.

The sequence of events for the outside containment break is provided in Table 15.1-24. Following the break, a CPCS VOPT is generated at 4.23 seconds. At 4.86 seconds the trip breakers open. The plant is assumed to lose offsite power at 5.15 seconds and the reactor coolant pumps begin to coast down. The maximum core power of 115.6% is reached at 5.25 seconds. The maximum heat flux of 108.1% is reached at 5.47 seconds. The minimum DNBR of 1.381 is reached at 6.1 seconds.

←(EC-13881, R304, DRN 05-543, R14; EC-8458, R307, LBDCR 15-039, R309)

→(DRN 05-543, R14; EC-8458, R307, LBDCR 15-039, R309)
The Pre-Trip Steam Line Break event no longer violates the DNB SAFDL limit. Inline nozzle flow restrictors on the Replacement Steam Generators limit the effective break size from 7.88 $ft^2$ to 2.78 $ft^2$ allowing the DNBR values for both inside and outside containment to reach 1.365 and 1.381 respectively. Calculated fuel failure is no longer a concern; however, the 8.0% fuel pin allowance applied to this event remains conservative.
←(DRN 05-543, R14; EC-8458, R307, LBDCR 15-039, R309)

### 15.1.3.3.4 Barrier Performance

→(DRN 04-704, R14; EC-40444, R307)
The Inside Containment Main Steam Line Break analysis assumes a 0.375 GPM (540 gal/day) limit on primary-to-secondary leakage for the affected steam generator for the duration of the event. The affected Steam Generator blows down into the containment. A Main Steam Isolation Signal (MSIS) is actuated to shut the main steam isolation valves for both steam generators. Steam is vented to atmosphere through the Main Steam Safety Valves and Atmospheric Dump Valves (ADV's) of the intact Steam Generator. A 150 gal/day primary-to-secondary leakage is assumed for the intact Steam Generator. Releases from the intact SG ADV are terminated upon achieving Shutdown Cooling entry conditions of 350F RCS temperature. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions. FSAR Section 15.B describes the generalized scenario assumed for secondary steaming in Waterford-3 radiological dose calculations.
←(DRN 04-704, R14; EC-40444, R307)

### 15.1.3.3.5 Radiological Consequences

→(DRN 04-704, R14)
#### 15.1.3.3.5.1 Design Basis – Method of Analysis

The bounding scenario for this series of events is the inside containment main steam line break, therefore, this section discusses the assumptions and methodology for that scenario in detail. The radiological consequences of the outside containment Pre-Trip Main Steam Line Break are bounded by the results reported for a Feedwater System Pipe Break (See Subsection 15.2.3.1.3.1.5). The reported FWLB results bound the Outside Containment MSLB results due to the modeling of releases for the FWLB event and due to the fact that no fuel failures (0%) are permitted for these two events.
←(DRN 04-704, R14)

→(DRN 04-704, R14)

15.1.3.3.5.1.1 Mathematical Model

The fuel failure limits for the inside containment main steam line break are

1. 2% fuel failure via Departure from Nucleate Boiling Ratio (DNBR) for inside containment post-trip MSLB with a LOOP.
2. 10% fuel failure (via DNBR) for inside containment MSLB with a LOOP due to combined post-trip return to power and pre-trip power excursion results.

Even though no fuel failure is allowed for cases with no LOOP, a slightly higher secondary steaming release based on the availability of offsite power (and hence of RCP heat contribution to steaming), is assumed. Two different pathways for releases exist for the inside containment MSLB:

- containment building leakage release pathway
- secondary steaming release pathway

The results from these two pathways are added together to obtain the total dose consequences. The containment building leakage pathway is a minor contributor compared to the secondary steaming release pathway, which accounts for releases through the Atmospheric Dump Valve (ADV) and Main Steam Safety Valves (MSSV's) of the intact SG. Because most of the releases will be from the ADV, which has worse atmospheric dispersion factors ($\chi$/Qs) for the control room, that is assumed to be the source of all the steaming release.

The containment building release pathway model is similar to and derived from the Large Break LOCA release model (See Subsection 15.6.3.3). Containment spray is not credited for fission product removal. It is conservatively assumed that all available activity is released to the containment within a very short time for this case. It is very conservatively assumed that there is a 30 day release duration for this pathway.

The secondary steaming pathway assumes that an inside containment MSLB has occurred and that releases are occurring due to primary-to-secondary leakage from the RCS to the Steam Generator. Activity is then released to the environment through the use of the Atmospheric Dump Valves (ADV's) to remove decay heat and to cool the plant to Shutdown Cooling entry conditions. Since the control room $\chi$/Q values are worst for ADV releases, any releases which would occur through the Main Steam Safety Valves (MSSV's) are instead assumed released from the ADV locations. Once Shutdown Cooling is initiated, the release to the environment is terminated for this pathway.

15.1.3.3.5.1.2 Input Parameters and Initial Conditions

The major assumptions, parameters, and calculational methods used in the design basis analysis are presented in Table 15.1-17. Additional clarification is provided as follows:

a) Reactor Coolant Activity and Core Release Fractions

Pre-accident RCS activity is assumed to be at the Technical Specification limit of 1.0 $\mu$Ci/gm for Dose Equivalent Iodine-131 DEI-131), and 100 / $\bar{E}$ for Noble Gases.

→(EC-5000081470, R301)

←(DRN 04-704, R14; EC-5000081470, R301)

➔(DRN 04-704, R14; EC-5000081470, R301)

The source term is based on the assumption that 10% of the fuel experience DNB, i.e., 10% fuel cladding failure. The following gap fractions are used for various isotopes.

| | |
|---|---|
| I-131 | 8% |
| Kr-85 | 10% |
| Other Noble Gases | 5% |
| Other Halogens | 5% |
| Alkali Metals | 12% |

🡐(EC-5000081470, R301)

b) Secondary system leakage

The activity in both steam generators is conservatively assumed to be the Technical Specification limit of 0.1 μCi/gm DEI-131.

c) Primary to secondary leakage

➔EC-40444, R307)

The primary to secondary leakage rate for the intact SG is assumed to be 150 gpd. The primary to secondary leakage rate for the faulted SG is assumed to be 540 gpd. The duration of the release from the SGs for the secondary steaming pathway is 7.5 hours, which corresponds to the assumed time until initiation of shutdown cooling. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.

🡐(EC-40444, R307)

d) Main control room model

➔(EC-5000081470, R301)

The control room model used in this analysis is identical to that used in the large break LOCA. This model assumes an unfiltered inleakage of 100 cfm for the duration of the event. The analysis allows for the control room to be either pressurized or in recirculation for the duration of the event. It is assumed that the preferred control room intake is selected at the time when the operators initiate pressurization flow.

e) Removal Coefficients

For the intact SG, an iodine partition factor (PF) of 1.0 for 0-30 minutes and of 100 beyond 30 minutes is assumed for activity transported to the secondary side. The activity mixes with the SG inventory and is released with the specified PF. RG 1.183, Appendix E (for PWR MSLBs), Position 5.5.4, calls for assuming an iodine PF of 100 when the tubes are not in dry-out (i.e., when top of U-tubes are submerged). Assuming a PF of 1.0 for the first 30 minutes accounts for any potential transient which might uncover the SG U-tubes during recovery from the plant scram. Noble gases released to the secondary side of the unaffected SG are assumed to be immediately released to the atmosphere.

🡐(EC-5000081470, R301)
➔(DRN 05-1551, R14)

For releases into containment from the affected SG, no iodine scrubbing is assumed. Removal of iodine by containment sprays and natural deposition is conservatively ignored.

🡐(DRN 05-1551, R14)

15.1.3.3.5.1.3 Results

The potential radiological consequences resulting from the occurrence of a postulated main steam line break have been conservatively analyzed, using the models and assumptions described in previous sections.

No fuel failures occur for the outside containment main steam line break scenarios, thus the radiological consequences are less severe than the results of a Feedwater System Pipe Break (See Subsection 15.2.3.1.3.1.5).

The off-site dose consequences of the inside containment main steam line break meet the acceptance criteria set forth in 10CFR50.67, and control room dose consequences meet the criteria of 10CFR50.67 and GDC19. The results are listed in Table 15.1-20.

🡐(DRN 04-704, R14)

SECTION 15.1: REFERENCES

→(DRN 04-704, R14)
1. Not Used.
←(DRN 04-704, R14)

2. Calvert Cliffs Nuclear Power Plant Unit I Docket No. 50-317, "Amendment to Operating License DPR 53 Supplement 1 to Seventh Cycle License Application," September 1, 1983.

3. R. V. MacBeth, "An Appraisal of Forced Convection Burnout Data," Proc. Inst. Mech. Engrs., Vol. 180, Pt. 3C, PP 37-50, 1965 - 1966.

4. D. H. Lee, "An Experimental Investigation of Forced Convection Burnout in High Pressure Water - Part IV, Large Diameter Tubes at About 1600 psia," A.E.E.W. Report R479, 1986.

5. Letter from R.P. Barkhurst (Waterford 3) to NRC, W3F192-0020, dated April 24, 1992.

→(DRN 04-704, R14)
6. W3F1-2004-0053, “License Amendment Request NPF-38-256, Alternate Source Term, Waterford Steam Electric Station, Unit 3, Document No. 50-382, License No. NPF-38,” July 15, 2004.
←(DRN 04-704, R14)

TABLE 15.1-1 Revision 309 (06/16)

→(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

SEQUENCE OF EVENTS FOR AN INCREASE IN MAIN STEAM FLOW

| Time (Seconds) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | A postulated spurious quick open signal generated by the steam bypass control system, one of the turbine bypass valves begins to open | --- |
| 1.0 | The turbine bypass valve is fully open | |
| 23.9 | CPCS VOPT occurs, % of rated power | 108.8 |
| 24.4 | Trip breakers open | --- |
| 24.65 | Turbine Admission Valves Begin to Close | --- |
| 25.0 | CEAs begin to drop into the core | --- |
| 25.0 | Maximum core power occurs, % of rated power | 109.6 |
| 25.1 | Maximum core average heat flux occurs, % of full power average channel heat flux(2) | 109.1 |
| 25.2* | Minimum WSSV-T / ABB-NV DNBR occurs | ≥1.24 |
| 27.4* | CEAs 90% inserted | --- |
| 42.4 | Feedwater control valves fully closed, main feedwater reaches 3.5% of full flow | --- |
| 153.5 | Pressurizer empties | |
| 160.5 | Safety injection actuation signal, psia | 1560. |
| 179.1 | SIAS flow initiated | --- |
| 257.0 | Boron enters the RCS | --- |
| 347.1 | MSIS and low steam generator pressure setpoint reached | 576. |
| 348.1 | MSIS is generated | --- |
| 350.2 | Main Feedwater Isolation Signal Generated | --- |
| 355.1 | Minimum steam generator pressure occurs, psia | 568.6 |
| 355.2 | Main steam isolation valves closed | --- |
| 356.4 | Minimum pressurizer pressure, psia | 679.8 |
| 360.3 | Feedwater isolations valves closed | --- |
| 367.5 | Pressurizer begins to refill | --- |
| 1000.0 | End of case run | |
| > 1000.0 | Steam Generator Safety Valves Open (and begin cycling), psia | 1085 |
| 1800.0 | Operator initiates plant cooldown procedures if the malfunction has not already been corrected | --- |

*Sequence of events based on the original SCRAM curve insertion times documented in Table 15.0-5. This event was assessed with the revised insertion time, and it was determined that there was an insignificant impact to the analysis.

←(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

TABLE 15.1-2 Revision 14 (12/05)

→(DRN 05-543, R14)

ASSUMPTIONS FOR INCREASED MAIN STEAM FLOW EVENT

| Parameter | Assumption |
|---|---|
| Initial core power, MWt | 3735 |
| Core inlet coolant temperature, °F | 554 |
| Initial RCS flowrate, lbm/hr | $146.0 \times 10^6$ |
| Initial Pressurizer Pressure, psia | 2310 |
| Steam generator secondary pressure, psia | 892 |
| Total nuclear heat flux factor | 2.5 |
| Moderator temperature coefficient, $10^{-4}$ $\Delta\rho$/°F | -4.2 |
| Doppler coefficient multiplier | 0.85 |
| CEA worth for trip, $10^{-2}$ $\Delta\rho$ | -6.0 |
| Turbine bypass system | Fails |
| Reactor regulating system | Manual |
| Feedwater regulating system | Automatic |
| Pressurizer level control system | Manual |
| Pressurizer pressure control system | Manual |

←(DRN 05-543, R14)

TABLE 15.1-3 Revision 14 (12/05)

SEQUENCE OF EVENTS FOR THE INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE

→(DRN 05-543, R14)

| Time (Sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | One atmospheric dump valve opens fully | --- |
| 53.0 | Minimum pressurizer pressure, psia | 2045 |
| 72.7, 70.5 | Minimum steam generator pressure, psia | 956.8 affected steam generator<br>1056.3 intact steam generator |
| 82.9 | Peak core power, % of rated power | 5.73 |
| 85.1 | Peak average core heat flux, % of full power heat flux | 5.69 |
| 161.3 | Maximum pressurizer pressure, psia | 2116 |
| 188.3 | Maximum pressure in affected steam generator, psia | 968.9 |
| 200.9 | Maximum pressure in intact steam generator, psia | 1085.9 |
| 607.0 | Low steam generator level trip signal, feet above tubesheet | 27 (5% narrow range) |
| 607.9 | Trip breakers open | --- |
| 608.5 | Shutdown CEA's begin to enter core | --- |
| 1800 | Operator takes control of plant | --- |
| 28,800 | Shutdown cooling initiated | --- |

←(DRN 05-543, R14)

TABLE 15.1-4 Revision 14 (12/05)

ASSUMPTIONS FOR THE INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE

→(DRN 05-543, R14)

| Parameter | 3716 MWt Power Uprate Assumption |
|---|---|
| Initial core power level, MWt | 1 |
| Core inlet coolant temperature, °F | 552 |
| Core mass flowrate, $10^6$ $lb_m$/hr | 148 |
| Pressurizer Pressure, psia | 2090 |
| Steam generator pressure, psia | 1057 |
| Total nuclear heat flux factor | 2.26 |
| Moderator temperature coefficient with uncertainties, $10^{-4}$ $\Delta\rho$ | -4.2 |
| Doppler coefficient multiplier | 0.85 |
| CEA worth on trip, $10^{-2}$ $\Delta\rho$ | -5.0 |
| Reactor Regulating System | Manual |
| Steam Bypass System | Fails |
| Feedwater Regulating System | Manual |
| Turbine Control System | Manual |
| Pressurizer Level, % | 21 |
| Steam Generator Level, % Narrow Range | 90 |

←(DRN 05-543, R14)

TABLE 15.1-5

MASS RELEASE - INADVERTENT OPENING OF STEAM GENERATOR ATMOSPHERIC DUMP VALVE

| Time (seconds) | Mass Flow Rate Out of Steam Generators ($lb_m$ /sec) | | Integrates Mass Flow Out ($10^5 lb_m$) | | Water Mass Remaining in Steam Generators ($10^5 lb_m$) | |
|---|---|---|---|---|---|---|
| | Affected | Unaffected | Affected | Unaffected | Affected | Unaffected |
| 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 3.23 | 3.23 |
| 50.0 | 276.4 | 0.0 | 0.143 | 0.0 | 3.09 | 3.23 |
| 100.0 | 282.2 | 0.0 | 0.280 | 0.0 | 2.94 | 3.23 |
| 200.0 | 291.9 | 0.0 | 0.571 | 0.0 | 2.64 | 3.23 |
| 600.0 | 288.3 | 0.0 | 1.727 | 0.0 | 1.43 | 3.23 |
| 1000.0 | 205.4 | 0.0 | 2.719 | 0.0 | 0.045 | 3.23 |
| 1250.0 | 164.6 | 0.0 | 3.179 | 0.0 | 0.001 | 3.23 |
| 1258.8 | 0.1 | 0.0 | 3.193 | 0.0 | 0.0 | 3.23 |
| 1800.0 | 0.1 | 0.0 | 3.194 | 0.0 | 0.0 | 3.23 |
| 7200.0 | 0.1 | 0.0 | 3.199 | 0.0 | 0.0 | 3.23 |
| 11,300 | 0.1 | 0.0 | 3.203 | 0.0 | 0.0 | 3.23 |

TABLE 15.1-6 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.1-7 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.1-8 Revision 309 (06/16)

→(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

Sequence of Events for the Increased Main Steam Flow in Combination with a Loss of Offsite Power
(Typical NSSS Response)**

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Malfunction of Control System causes increased steam flow through the Turbine or the Turbine Bypass Valves | --- |
| 1.0 | Turbine or Turbine Bypass valve fully open | --- |
| 11.7 | Low DNBR trip signal generated | --- |
| 11.7 | Loss of offsite power | --- |
| 11.8 | Maximum core power, percent of rated core power | 109 |
| 11.8 | Maximum average core heat flux, percent of full power heat flux | 105 |
| 11.9 | Turbine admission valves start to close | |
| 12.6 | Shutdown CEAs begin to drop into core | |
| 14.0 | Minimum WSSV-T / ABB-NV DNBR | 1.051 |
| 132.8* | Steam Generator Safety Valves Open | 1102.1 psia |
| 132.8* | Maximum Steam Generator Pressure | 1103 psia |
| 1,800 | Operator takes control of plant | --- |
| →(DRN 05-1551, R14) 14,000 ←(DRN 05-1551, R14) | Shutdown Cooling initiated | --- |

*These times are based on analysis performed for the original SCRAM curve insertion times documented in Table 15.0-5; however, are still considered representative for the revised SCRAM curve values.

**This event has been assessed using the revised SCRAM curve insertion times documented in Table 15.0-5. Therefore, times for the Sequence of Events were not re-calculated. The Sequence of Events has been modified to match Table 15.1-8A.

←(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

TABLE 15.1-8A Revision 309 (06/16)

→(EC-13881, R304)

Sequence of Events for the Increased Main Steam Flow
in Combination with a Loss of Offsite Power
(Worst DNB Performance Case)

→(DRN 05-543, R14, LBDCR 15-039, R309)

| Time (Sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Malfunction of Control System causes increased steam flow through the Turbine or the Turbine Bypass Valves | --- |
| ΔT | Thermal Margin Initially Preserved by COLSS Depleted, the Hottest Fuel Rod is just above the DNBR SAFDL of 1.24 as calculated by the CPCs. Loss of offsite power is assumed and the coast down of the RCPs begins | --- |
| ΔT+0.34 | Low DNBR Trip generated by the CPCs, Trip breakers open | --- |
| ΔT+0.94 | CEAs begin to Drop | --- |
| ΔT+2.3 | Minimum WSSV-T / ABB-NV DNBR Occurs | 1.051 |
| ΔT+3.54 | Average CEA Position 90% inserted | --- |
| ΔT+121.1* | Steam Generator Safety Valves Open | 1102.1 psia |
| ΔT+121.1* | Maximum Steam Generator Pressure | 1103 psia |
| 1,800 | Operator takes control of plant | --- |
| →(DRN 05-1551, R14) 14,000 ←(DRN 05-1551, R14) | Shutdown Cooling Initiated | --- |

*These times are based on analysis performed for the original SCRAM curve insertion times documented in Table 15.0-5; however, are still considered representative for the revised SCRAM curve values.
←(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

→(DRN 05-543, R14)

TABLE 15.1-8B Revision 309 (06/16)

Initial Conditions for the Increased Main Steam Flow in Combination with a Loss of Offsite Power

→(LBDCR 15-039, R309)

| Parameter | Assumption |
|---|---|
| Initial Core Power, MWt | 3735 |
| Core Inlet Temperature, °F | 533 |
| Pressurizer Pressure, psia | 2310 |
| RCS Flowrate, $10^6$ lbm/hr | 121.3 |
| Pressurizer Level, % | 44 |
| SG Pressure, psia | 740 |
| MTC, $x10^{-4}$ $\Delta\rho$/°F | -1.05 |
| Doppler Coefficient Multiplier | 0.85 |
| Kinetics | Maximum $\beta$ |
| CEA Worth for Trip, %$\Delta\rho$ | 5.0 |

←(DRN 05-543, R14, LBDCR 15-039, R309)

→(DRN 05-543, R14; EC-13881, R304)

TABLE 15.1-8C Revision 309 (06/16)

Sequence of Events for the HFP IOADV with LOOP Event

→(LBDCR 15-039, R309)

| Time (Sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Operator Error or Malfunction of Control System causes increased steam flow through one Atmospheric Dump Valve | --- |
| 1.0 | Atmospheric Dump Valve fully opens | --- |
| 21.2 | Minimum Pressurizer pressure, psia | 2308 |
| 91.6 | Maximum core power, percent of rated core power | 106.0 |
| 95.0 | Minimum SG pressure, psia | 705.9 |
| 130.3 | Maximum average core heat flux, percent of full power heat flux | 106.0 |
| 150.0 | Thermal Margin Initially Preserved by COLSS is partially depleted, a quasi-steady state is reached | --- |
| 150.0 | A Loss of AC Power occurs;<br>Coast down of the RCPs begins | --- |
| 150.5 | CPCS Low RCP Shaft Speed trip condition is reached | 0.965 |
| 150.8 | CPCS Trip due to low RCP shaft speed is generated; Trip breakers open | --- |
| 150.8 | Turbine admission valves start to close | --- |
| 151.4 | Shutdown CEAs begin to drop into core | --- |
| 152.75 | Minimum WSSV-T / ABB-NV DNBR | ≥1.24 |
| 154.0 | Average CEA Position 90% Inserted | |
| 247.40* | Maximum Pressurizer Pressure, psia | 2576 |
| 247.40* | Maximum RCS Pressure, psia | 2584 |
| 279.0* | Steam generator safety valves open, psia | 1117 |
| 279.0* | Maximum steam generator pressure, psia | 1118 |
| 1,800 | Operator takes control of plant | --- |
| 14,000 | Shutdown Cooling initiated | --- |

*These times are based on analysis performed for the original SCRAM curve insertion times documented in Table 15.0-5; however, are still considered representative for the revised SCRAM curve values.

←(DRN 05-543, R14; EC-13881, R304, LBDCR 15-039, R309)

→(DRN 05-543, R14, LBDCR 15-039, R309)

TABLE 15.1-8D Revision 309 (06/16)

Assumption Table for the HFP IOADV with LOOP event

| Parameter | Assumption |
|---|---|
| Initial Core Power (MWt) | 3735 |
| Initial Core Inlet Temperature (°F) | 533 |
| Initial Pressurizer Pressure (psia) | 2310 |
| Initial RCS Flowrate ($10^6$ lbm/hr) | 175.4 |
| Initial Pressurizer Level (ft)*** | 15.8 / 44% (Nominal) |
| Initial SG Water Level (ft / %narrow range)*** | 36.4 / 67.2% (Nominal) |
| MTC ($x10^{-4}$ $\Delta\rho$/°F) | -1.05* |
| Doppler Coefficient Multiplier | 0.85 |
| Kinetics | Maximum $\beta$ |
| CEA Worth at Trip – WRSO (%$\Delta\rho$) | -5.0 |
| Pressurizer Pressure Control System (PPCS)** | Auto |
| Pressurizer Level Control System (PPLS)** | Auto |

* This MTC is used in the HERMITE core simulation. For the NSSS response simulation (performed using CENTS), MTC = $-4.2x10^{-4}$ $\Delta\rho$/°F is used.

** The PPCS and PLCS are set to auto to maintain a steady-state prior to initiation of the event. These systems are shut off when the LOAC initiates.

*** This input is not used in the HERMITE core simulation. The specified value is taken from the CENTS NSSS response simulation.

←(DRN 05-543, R14, LBDCR 15-039, R309)

TABLE 15.1-9 Revision 309 (06/16)

→(LBDCR 15-039, R309)

INTENTIONALLY DELETED

←(LBDCR 15-039, R309)

TABLE 15.1-10 (Sheet 1 of 2) Revision 309 (06/16)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE WITH CONCURRENT LOSS OF OFFSITE POWER

→(DRN 04-704, R14, LBDCR 15-039, R309)

| Parameter | Assumptions |
|---|---|
| Core Power Level: | 3735 MWt |
| →(EC-5000081470, R301) | |
| ←(EC-5000081470, R301, LBDCR 15-039, R309) | |
| Secondary Steaming Pathway | |
| Primary-to-Secondary Leak Rate: | 150 gpd unaffected SG, 540 gpd faulted SG |
| →(EC-40444, R307) Total MSSV/ADV Combined Leakage per Steam Line ←(EC-40444, R307) | 280 lb/hr Until Cold Shutdown |
| Iodine Chemical Form (Reactor Building Release Path): | |
| Elemental | 97% |
| Organic | 3% |
| Steaming PF (Iodine and Alkali Metals): | |
| 4 hours | 1 |
| > 4 hours | 100 |
| Duration of Release: | 36 hours |
| Control Room Parameters | |
| Volume: | 220,000 $ft^3$ |
| Recirculation Flow Rate: | 3800 CFM |
| Filter Efficiency: | 99% (elemental/organic/particulate) |
| →(EC-5000081470, R301) Pressurization Flow: ←(EC-5000081470, R301) | 225 CFM (maximum) |
| Unfiltered In-leakage: | 100 CFM |
| Breathing Rate: | 3.47E-04 $m^3$/sec |
| Control Room Occupancy Factors: | |
| 0-24 hour | 1.00 |
| 24-96 hours | 0.60 |
| 96 hours – 30 days | 0.40 |

←(DRN 04-704, R14)

TABLE 15.1-10 (Sheet 2 of 2) Revision 301 (09/07)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE WITH CONCURRENT LOSS OF OFFSITE POWER

(DRN 04-704, R14)

(DRN 05-1551, R14)

Main Control Room (MCR) $\chi$/Q Assumed
(Affected Steam Generator is Assumed to be the East SG):

(DRN 05-1551, R14)

(EC-5000081470, R301)

X/Q Values for Releases from East ADV

| Time | Unfiltered In-Leakage, East ADV to East MCR Air Intake | Pressurization Flow East ADV to West MCR Air Intake |
|---|---|---|
| 0-2 hr | 1.06E-01 | 3.40E-04 * |
| 2-8 hr | 7.45E-02 | 2.08E-04 * |
| 8-24 hrs | 3.30E-02 | 1.00E-04 * |
| 1-4 days | 2.31E-02 | 6.58E-05 * |

* factor of 4 reduction credited per SRP 6.4.

X/Q Values for Releases from West ADV

| Time | Unfiltered In-Leakage, West ADV to East MCR Air Intake | Pressurization Flow West ADV to West MCR Air Intake |
|---|---|---|
| 0-2 hr | 1.36E-03 | 7.50E-03 * |
| 2-8 hr | 8.29E-04 | 5.62E-03 * |
| 8-24 hrs | 3.55E-04 | 2.57E-03 * |
| 1-4 days | 2.48E-04 | 2.04E-03 * |

* factor of 4 reduction credited per SRP 6.4.

(EC-5000081470, R301)

Steam (lbm) and Activity (DEI-131, Ci) Releases

| 0-2 hr Steaming | 2-8 hr Steaming |
|---|---|
| 588,365 | 1,333,286 |

Activity Releases For 150 gpd Primary-to-Secondary Leakage

| | 0 - 15 min | 15 - 30 min | ½ - 1 hr | 1 - 2 hr | 2 - 4 hr | 4 - 6 hr | 6 - 8 hr |
|---|---|---|---|---|---|---|---|
| PIS (Intact SG) | 4.80 | 6.47 | 0.61 | 0.92 | 1.80 | 0.05 | 0.07 |
| GIS (Intact SG) | 4.74 | 6.14 | 0.25 | 0.44 | 1.55 | 0.07 | 0.12 |

(DRN 04-704, R14)

TABLE 15.1-11 Revision 14 (12/05)

RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE WITH CONCURRENT LOSS OF OFFSITE POWER

→(DRN 04-704, R14)

| Dose Location | EAB * | LPZ ** | Control Room |
|---|---|---|---|
| **Total for PIS** (Rem TEDE) | < 2.5 | < 2.5 | < 5.0 |
| **Total for GIS** (Rem TEDE) | < 2.5 | < 2.5 | < 5.0 |

* Worst 2 hour doses listed
** Duration

←(DRN 04-704, R14)

TABLE 15.1-12 Revision 307 (07/13)

➔(DRN 05-543, R14)

Sequence of Events for the Return to Power Steam Line Break
Hot Full Power, Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Steam line break upstream of MSIV, loss of power to RCPs<br><br>(Effective Break Area) | 7.88 $ft^2$<br><br>(2.78 $ft^2$) |
| 2.0 | CPCS RCP Shaft Speed Trip | --- |
| 2.9 | Trip breakers open | --- |
| 3.5 | Shutdown CEAs begin dropping into the core | --- |
| 12.9 | MSIS setpoint is reached | 576 psia |
| 13.9 | MSIVs begin to close | --- |
| 20.9 | MSIVs closed | --- |
| 22.9 | MFIVs closed | --- |
| 28.9 | Low RCS pressure initiates SIAS | 1560 psia |
| 58.9 | HPSI pump reaches full speed | --- |
| 168.8 | Maximum post-trip fission power occurs | 2.1% of 3716 MWt |
| 170.8 | Minimum post-trip MacBeth DNBR occurs | 1.47 |
| 489.2 | Maximum post-trip reactivity occurs | +0.0026%Δρ |
| --- | Pressurizer reaches 0.0 feet | --- |
| 468.3 | Affected SG empties | --- |
| 541.3 | Minimum pressurizer pressure | 725.7 psia |
| 1800 | Plant cooldown initiated by manual control of the ADV associated with the intact SG | --- |
| ➔(DRN 05-1551, R14)<br>----<br>🡐(DRN 05-1551, R14) | SDC initiated | --- |

🡐(DRN 05-543, R14; EC-8458, R307)

TABLE 15.1-13 Revision 309 (06/16)

→(DRN 05-543, R14; DRN 05-1551, R14; EC-9533, R302, LBDCR 15-039, R309)

Sequence of Events for the Return to Power Steam Line Break
Hot Full Power, No Loss of Offsite Power, Inside Containment

→(EC-8458, R307

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Steam line break upstream of MSIV<br><br>(Effective Break Area) | 7.88 $ft^2$<br><br>(2.78 $ft^2$) |
| 5.0 | Rx Trip, VOPT | --- |
| 5.1 | Trip breakers open | --- |
| 5.7 | Shutdown CEAs begin dropping into the core | --- |
| 11.8 | Pressurizer empties | --- |
| 17.0 | MSIS setpoint reached | 576 psia |
| 18.0 | MSIVs begin to close | --- |
| 25.0 | MSIVs Closed | --- |
| 45.0 | Low RCS pressure initiates SIAS | 1560 psia |
| 63.5 | HPSI pump reaches full speed | --- |
| 123.1 | Maximum post-trip Fission Power occurs | 3.45% of 3716 MWt |
| 53.55 | Maximum post-trip LHGR | 18.4 kW/ft |
| 213.7 | Affected steam generator empties | --- |
| 209.05 | Maximum post-trip reactivity occurs | -0.048%Δρ |
| 1800.0 | Plant cooldown initiated by manual control of the ADV associated with the intact SG | --- |

←(DRN 05-543, R14; DRN 05-1551, R14; EC-9533, R302; EC-8458, R307, LBDCR 15-039, R309)

TABLE 15.1-14 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

➔(DRN 05-543, R14; DRN 05-1551, R14)

TABLE 15.1-14A Revision 307 (07/13)

Sequence of Events for the Return to Power Steam Line Break
Hot Zero Power, Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Time (sec) | Event | Setpoint / Value |
|---|---|---|
| 0.0 | Steam line break upstream of MSIV,<br>Loss of power to RCPs<br>(Effective Break Area) | 7.88 $ft^2$<br><br>(2.78 $ft^2$) |
| 2.0 | CPCS RCP Shaft Speed Trip | --- |
| 2.9 | Trip breakers open | --- |
| 3.5 | Shutdown CEAs begin dropping into the core | --- |
| 12.3 | MSIS setpoint reached | 576 psia |
| 13.3 | MSIVs begin to close | --- |
| 20.3 | MSIVs closed | --- |
| 22.3 | MFIVs closed | --- |
| 22.9 | Low RCS pressure initiates SIAS | 1560 psia |
| 52.9 | HPSI pump reaches full speed | --- |
| 187.4 | Maximum post-trip fission power | 2.5% of 3716 MWt |
| 190.4 | Minimum post-trip DNBR | 1.61 |
| 125.4 | Maximum post-trip reactivity | 0.1324%Δρ |
| 314.3 | Pressurizer reaches 0.0 feet | --- |
| 600.0 | Minimum RCS pressure | 539 psia |
| >600.0 | Affected SG empties | ---` |
| 1800 | Plant cooldown initiated by manual control of the ADV associated with the intact SG | --- |
| --- | SDC initiated | --- |

🡐(DRN 05-543, R14; DRN 05-1551, R14; EC-8458, R307)

➔(DRN 05-543, R14; DRN 05-1551, R14)

TABLE 15.1-14B Revision 307 (07/13)

Sequence of Events for the Return to Power Steam Line Break
Hot Zero Power, no Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Time (sec) | Event | Setpoint / Value |
|---|---|---|
| 0.0 | Steam Line Break Upstream of MSIV<br><br>(Effective Break Area) | 7.88 $ft^2$<br><br>(2.78 $ft^2$) |
| 5.0 | Rx Trip, VOPT | --- |
| 5.1 | Trip breakers open | --- |
| 5.7 | Shutdown CEAs begin dropping into the core | --- |
| 11.2 | Pressurizer empties | 0.1 ft |
| 13.2 | MSIS setpoint reached | 576 psia |
| 14.2 | MSIVs begin to close | --- |
| 21.2 | MSIVs closed | --- |
| 23.2 | MFIVs closed | --- |
| 40.2 | Low RCS pressure initiates SIAS | 1560 psia |
| 58.7 | HPSI pump reaches full speed | --- |
| 74.35 | Maximum post-trip reactivity | +0.1641%Δρ |
| 206.7 | Maximum post-trip fission Power | 4.9% of 3716 MWt |
| 206.7 | Maximum post-trip LHGR | 21.25 kW/ft |
| > 450 | Affected steam generator empties | --- |
| 1800.0 | Plant cooldown initiated by manual control of the ADV associate with the intact SG | --- |

🡐(DRN 05-543, R14; DRN 05-1551, R14; EC-8458, R307)

TABLE 15.1-15 Revision 307 (07/13)

➔(DRN 05-543, R14)

Assumptions for the Return to Power Steam Line Break
Hot Full Power, Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Parameter | Assumptions |
|---|---|
| Initial Core Power, MWt | 3735 |
| Core Inlet Coolant Temperature, °F | 552 |
| RCS Flowrate, $x10^6$ lbm/hr | 148.0 |
| Pressurizer Pressure, psia | 2310 |
| Pressurizer Level, % | 21 |
| SG Pressure, psia | 867 |
| SG Level, % NR | 79.3 |
| Doppler Coefficient Multiplier | 1.15 (EOC) |
| SBCS | Inoperative |
| PPCS | Automatic |
| High Pressure Safety Injection Pumps | One Pump Inoperative |
| Blowdown Fluid | 100% steam |
| Break Area, $ft^2$ | 7.88 |

🡐(DRN 05-543, R14; EC-8458, R307)

➔(DRN 05-543, R14)

TABLE 15.1-15A Revision 307 (07/13)

Assumptions for the Return to Power Steam Line Break
Hot Full Power, no Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Parameter | Assumptions |
|---|---|
| Initial Core Power Level, MWt | 3735 |
| Core Inlet Coolant Temperature, °F | 552 |
| RCS Flowrate, $x10^6$ lbm/hr | 148.0 |
| Pressurizer Pressure, psia | 2310 |
| SG Pressure, psia | 867 |
| SG Level, % NR | 79.3 |
| Doppler Coefficient Multiplier | 1.15 |
| Steam Bypass Control System | Inoperative |
| Pressurizer Pressure Control System | Automatic |
| High Pressure Safety Injection Pumps | One Pump Inoperative |
| Blowdown Fluid | 100% steam |
| Break Area, $ft^2$ | 7.88 |
| Core Burnup | End of Cycle |

🡐(DRN 05-543, R14; EC-8458, R307)

TABLE 15.1-16 Revision 307 (07/13)

➔(DRN 05-543, R14)

Assumptions for the Return to Power Steam Line Break
Hot Zero Power, Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Parameter | Assumptions |
|---|---|
| Initial Core Power Level, MWt | 37.16 |
| Core Inlet Coolant Temperature, °F | 552 |
| RCS Flowrate, $x10^6$ lbm/hr | 148.0 |
| Pressurizer Pressure, psia | 2310 |
| Pressurizer Level, % | 21 |
| SG Pressure, psia | 1055 |
| SG Level, % NR | 79.3 |
| Doppler Coefficient Multiplier | 1.15 |
| SBCS | Inoperative |
| PPCS | Automatic |
| High Pressure Safety Injection Pumps | One Pump Inoperative |
| Blowdown Fluid | 100% steam |
| Break Area, $ft^2$ | 7.88 |

🡐(DRN 05-543, R14; EC-8458, R307)

➔(DRN 05-543, R14)

TABLE 15.1-16A Revision 307 (07/13)

Assumptions for the Return to Power Steam Line Break
Hot Zero Power, no Loss of Offsite Power, Inside Containment

➔(EC-8458, R307)

| Parameter | Assumptions |
|---|---|
| Initial Core Power Level, MWt | 37.16 |
| Core Inlet Coolant Temperature, °F | 552 |
| RCS Flowrate, $x10^6$ lbm/hr | 148.0 |
| Pressurizer Pressure, psia | 2310 |
| SG Pressure, psia | 1055 |
| SG Level, % NR | 79.3 |
| Doppler Coefficient Multiplier | 1.15 |
| Steam Bypass Control System | Inoperative |
| Pressurizer Pressure Control System | Automatic |
| High Pressure Safety Injection Pumps | One Pump Inoperative |
| Blowdown Fluid | 100% steam |
| Break Area, $ft^2$ | 7.88 |
| Core Burnup | End of Cycle |

🡐(DRN 05-543, R14; EC-8458, R307)

TABLE 15.1-17 (Sheet 1 of 3) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF AN INSIDE CONTAINMENT MAIN STEAM LINE BREAK ACCIDENT

General

| | |
|---|---|
| Core Power Level: | 3735 MWt |

→(EC-5000081470, R301)

←(EC-5000081470, R301)

| Fission Product Fractions: | Gap |
|---|---|
| Iodines | 10% |
| Noble Gases | 10% |
| Alkali metals (Cs & Rb) | 12% |

→(EC-5000081470, R301)

| | |
|---|---|
| Fuel Damage: | ≤ 10% DNBR |

Assumed fraction of the fission product activity in the clad gap:

| | |
|---|---|
| I-131 | 8% |
| KR-85 | 10% |
| Other Noble Gases | 5% |
| Other Halogens | 5% |
| Alkali Metals | 12% |

←(EC-5000081470, R301)

Containment Leakage Pathway

→(EC-5000081470, R301)

| | |
|---|---|
| Core Inventory: | Table 12.2-12 |

←(EC-5000081470, R301)

| | |
|---|---|
| Containment Leak Rate: | 0.50% volume/day (0 – 24 hrs)<br>0.25% volume/day (1 – 30 days) |
| Natural Deposition: | Not Credited |
| Spray Fission Product Removal: | Not Credited |
| Iodine Chemical Form (Reactor Building Release Path): | |
| Elemental | 4.85% |
| Organic | 0.15% |
| Particulate | 95% |
| Partition Factor (affected SG) | 1.0 |

Secondary Steaming Pathway

→(EC-5000081470, R301)

| | |
|---|---|
| Core Inventory: | Table 12.2-12a |

←(DRN 04-704, R14; EC-5000081470, R301)

TABLE 15.1-17 (Sheet 2 of 3) Revision 307(07/13)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF AN INSIDE CONTAINMENT MAIN STEAM LINE BREAK ACCIDENT

| | |
|---|---|
| Primary to Secondary Leak Rate: | |
| Unfaulted | 150 gpd |
| Faulted | 540 gpd |
| →(EC-40444, R307) | |
| Total MSSV/ADV Combined Leakage per Steam Line | 280 lb/hr Until Cold Shutdown |
| ←(EC-40444, R307) | |
| Iodine Chemical Form (Secondary Steaming Pathway) | |
| Elemental | 97% |
| Organic | 3% |
| Steaming PF (Iodine and Alkali Metals, Intact SG) | |
| 0 – 30 minutes | 1 |
| > 30 minutes | 100 |
| Duration of Release | 7.5 hours |
| →(EC-5000081470, R301) | |
| ←(EC-5000081470, R301) | |
| Control Room Parameters | |
| →(EC-5000081470, R301) | |
| Volume | 168,500 $ft^3$ |
| ←(EC-5000081470, R301) | |
| Recirculation Flow Rate | 3800 CFM |
| Iodine Filter Efficiency | 99% (elemental/particulate/organic) |
| →(EC-5000081470, R301) | |
| Pressurization Flow: | 225 CFM (maximum) |
| ←(EC-5000081470, R301) | |
| Unfiltered Inleakage: | 100 CFM |
| Breathing Rate: | 3.47E-04 $m^3/s$ |
| Control Room Occupancy Factors: | |
| 0 – 24 hours | 1.0 |
| 24 – 96 hours | 0.6 |
| 96 hours – 30 days | 0.4 |
| →(EC-5000081470, R301) | |
| Main Control Room Atmospheric Dispersion Factors ($\chi$/Q) Assumed: | |
| ←(DRN 04-704, R14;EC-5000081470, R301) | |

TABLE 15.1-17 (Sheet 3 of 3) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF AN INSIDE CONTAINMENT MAIN STEAM LINE BREAK ACCIDENT

| Time | Reactor Building Unfiltered Inleakage | Reactor Building Pressurization Flow | Secondary Steaming Unfiltered Inleakage | Secondary Steaming Pressurization Flow |
|---|---|---|---|---|
| →(EC-5000081470, R301) | | | | |
| 0 – 2 hrs | 2.77E-03 | 5.15E-04 * | 1.06E-01 | 3.075E-04 * |
| 2 – 8 hrs | 1.78E-03 | 3.90E-04 * | 7.45E-02 | 2.08E-04 * |
| ←(EC-5000081470, R301) | | | | |
| 8 – 24 hrs | 7.22E-04 | 1.79E-04 * | N/A | N/A |
| 1 – 4 days | 5.27E-04 | 1.37E-04 * | N/A | N/A |
| 4 – 30 days | 4.05E-04 | 1.08E-04 * | N/A | N/A |

* factor of 4 reduction taken for operator selection of more favorable air intake

Alkali Metal Source Term Data

| Activity Released | Unfaulted Generator (150 pgd) |
|---|---|
| Cs-134 | 19.273 Ci |
| Cs-136 | 5.067 Ci |
| Cs-137 | 10.263 Ci |
| Rb-86 | 0.036 Ci |

Steaming and Activity Releases

Steaming Releases:

| | |
|---|---|
| 0 – 2 hrs | 627,152 lbm |
| 2 – 7.5 hrs | 858,838 lbm |

Activity Releases: (Ci DEI-131)

| | |
|---|---|
| 0 – 15 min | 18.95 |
| 15 – 30 min | 18.09 |
| 0.5 – 1.0 hr | 0.95 |
| 1 – 2 hrs | 3.67 |
| 2 – 4 hrs | 11.25 |
| 4 – 6 hrs | 13.01 |
| 6 – 7.5 hrs | 12.70 |

←(DRN 04-704, R14)

TABLE 15.1-18

ZERO POWER STEAM LINE BREAK
SECONDARY SYSTEM MASS RELEASES

| Time (seconds) | Unaffected Steam Generator: Steam Dump Valve Release | | Unaffected Steam Generator: Release Through Break | | Affected Steam Generator: Release Through Break | |
|---|---|---|---|---|---|---|
| | Flow ($lb_m$/sec) | Integrated ($lb_m$) | Flow ($lb_m$/sec) | Integrated ($lb_m$) | Flow ($lb_m$/sec) | Integrated ($lb_m$) |
| 0 | 0.0 | 0.0 | 6,184.0 | 0.0 | 6,184.0 | 0.0 |
| 2.5 | 0.0 | 0.0 | 4,953.2 | 13,587.8 | 4,953.2 | 13,857.8 |
| 5 | 0.0 | 0.0 | 4,151.8 | 25,226.9 | 4,152.4 | 25,227.1 |
| 7.5 | 0.0 | 0.0 | 1,837.1 | 34,090.5 | 3,585.4 | 34,913.9 |
| 10 | 0.0 | 0.0 | 0.0 | 35,290.0 | 3,169.0 | 43,372.3 |
| 20 | 0.0 | 0.0 | 0.0 | 35,290.0 | 2,230.0 | 69,812.9 |
| 30 | 0.0 | 0.0 | 0.0 | 35,290.0 | 1,717.5 | 89,459.9 |
| 40 | 0.0 | 0.0 | 0.0 | 35,290.0 | 1,370.8 | 104,841.5 |
| 50 | 0.0 | 0.0 | 0.0 | 35,290.0 | 1,119.8 | 117,277.0 |
| 60 | 0.0 | 0.0 | 0.0 | 35,290.0 | 922.0 | 127,479.4 |
| 70 | 0.0 | 0.0 | 0.0 | 35,290.0 | 755.4 | 135,875.0 |
| 80 | 0.0 | 0.0 | 0.0 | 35,290.0 | 604.1 | 142,683.9 |
| 90 | 0.0 | 0.0 | 0.0 | 35,290.0 | 488.1 | 148,114.0 |
| 100 | 0.0 | 0.0 | 0.0 | 35,290.0 | 415.0 | 152,614.3 |
| 125 | 0.0 | 0.0 | 0.0 | 35,290.0 | 319.5 | 161,638.9 |
| 150 | 0.0 | 0.0 | 0.0 | 35,290.0 | 277.6 | 169,045.2 |
| 175 | 0.0 | 0.0 | 0.0 | 35,290.0 | 257.5 | 175,708.1 |
| 200 | 0.0 | 0.0 | 0.0 | 35,290.0 | 247.5 | 182,005.6 |
| 225 | 0.0 | 0.0 | 0.0 | 35,290.0 | 241.0 | 188,115.0 |
| 250 | 0.0 | 0.0 | 0.0 | 35,290.0 | 238.1 | 194,115.9 |
| 275 | 0.0 | 0.0 | 0.0 | 35,290.0 | 234.9 | 200,036.0 |
| 300 | 0.0 | 0.0 | 0.0 | 35,290.0 | 232.2 | 205,874.7 |
| 350 | 0.0 | 0.0 | 0.0 | 35,290.0 | 222.3 | 217,259.8 |
| 400 | 0.0 | 0.0 | 0.0 | 35,290.0 | 213.5 | 228,172.1 |
| 450 | 0.0 | 0.0 | 0.0 | 35,290.0 | 202.8 | 238,572.8 |
| 500 | 0.0 | 0.0 | 0.0 | 35,290.0 | 194.1 | 248,495.9 |
| 550 | 0.0 | 0.0 | 0.0 | 35,290.0 | 191.4 | 257,891.4 |
| 600 | 0.0 | 0.0 | 0.0 | 35,290.0 | 188.9 | 267,282.8 |
| 650 | 0.0 | 0.0 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 1800 | 68.3 | 0.0 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 2800 | 68.3 | 68,301.8 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 3800 | 68.3 | 136,603.5 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 4800 | 68.3 | 204,905.3 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 5800[a] | 68.3 | 273,207.0 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |
| 6940 | 68.3 | 351,033.9 | 0.0 | 35,290.0 | 0.0 | 269,841.5 |

(a) Time at which reactor coolant system temperature reaches 350°F and shutdown cooling is initiated.

TABLE 15.1-19

ZERO POWER STEAM LINE BREAK
STEAM GENERATOR LIQUID VOLUME FRACTIONS AND EMERGENCY
FEEDWATER FLOWRATE TO THE INTACT STEAM GENERATOR

| Time (seconds) | Liquid Volume Fraction | | Emergency Feedwater Flowrate ($lb_m$/sec) |
|---|---|---|---|
| | Failed Steam Generator | Intact Steam Generator | |
| 0 | 0.563 | 0.563 | 0.0 |
| 5 | 0.486 | 0.487 | 0.0 |
| 10 | 0.439 | 0.462 | 0.0 |
| 20 | 0.379 | 0.466 | 0.0 |
| 30 | 0.337 | 0.469 | 0.0 |
| 50 | 0.282 | 0.470 | 0.0 |
| 100 | 0.224 | 0.469 | 0.0 |
| 150 | 0.204 | 0.468 | 0.0 |
| 200 | 0.184 | 0.467 | 0.0 |
| 300 | 0.134 | 0.465 | 0.0 |
| 400 | 0.090 | 0.462 | 0.0 |
| 500 | 0.047 | 0.458 | 0.0 |
| 600 | 0.006 | 0.454 | 0.0 |
| 800 | 0.0 | 0.448 | 0.0 |
| 1000 | 0.0 | 0.444 | 0.0 |
| 1200 | 0.0 | 0.445 | 0.0 |
| 1400 | 0.0 | 0.446 | 0.0 |
| 1600 | 0.0 | 0.448 | 0.0 |
| 1800[a] | 0.0 | 0.450 | 68.3 |
| 3600 | 0.0 | 0.450 | 68.3 |
| 5400 | 0.0 | 0.450 | 68.3 |
| 7200 | 0.0 | 0.450 | 68.3 |

TABLE 15.1-20 Revision 14 (12/05)

→(DRN 04-704, R14)

RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED INSIDE CONTAINMENT MAIN STEAM LINE BREAK

| Dose Location | EAB * | LPZ | Control Room |
|---|---|---|---|
| Total (Rem TEDE) | < 25.0 | < 25.0 | < 5.0 |

* Worst 2 hour doses listed

←(DRN 04-704, R14)

TABLE 15.1-21 Revision 307 (07/13)

ASSUMPTIONS FOR LARGE STEAM LINE BREAKS
DURING MODE 3 OPERATION WITH AND WITHOUT CONCURRENT
LOSS OF OFFSITE POWER

| Parameter | Assumption |
|---|---|
| →(DRN 02-1479, R12) | |
| Initial Core Inlet Coolant Temperature, °F | 551 |
| Initial Reactor Coolant Mass Flow Rate, $10^6$ lbm/hr (4 RCPS) | 148.0 |
| Initial Pressurizer Pressure, psia | 1800 |
| Initial Subcriticality, %$\Delta\rho$ | -5.3 |
| ←(DRN 02-1479, R12) | |
| Doppler Coefficient Multiplier | 1.25 |
| →(DRN 00-1822; 02-1479, R12) | |
| Moderator Temperature Coefficient, x$10^4$ $\Delta\rho$/°F | -4.0 |
| ←(DRN 00-1822) | |
| Initial Steam Generator Liquid Inventory, lbm | 258,226 |
| ←(DRN 02-1479, R12) | |
| Core Burnup | End of Cycle |
| Blowdown Fluid | Saturated Steam |
| Break Area, $ft^2$ | 7.88 |
| Main Steam Flow Restrictor Minimum Throat Area, $ft^2$ | 3.14 |

→(DRN 00-1822; 02-1479, R12)
←(DRN 00-1822; 02-1479, R12)

→(EC-8458, R307)
Note: The table above represents assumptions for the EPU and OSG Mode 3 Steam Line Break Analysis. This analysis bounds the RSG analysis with respect to subcriticality.
←( EC-8458, R307)

TABLE 15.1-22 Revision 307 (07/13)

➔(DRN 05-543, R14)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK*
DURING MODE 3 OPERATION WITH CONCURRENT
LOSS OF OFFSITE POWER

🡐(DRN 05-543, R14)

| Time (Sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Steam line break and loss of offsite power occur; holding coils lose power | - |
| ➔(DRN 02-1479, R12) 2.3 | Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia; and MSIS is generated | 675 |
| 13.2 | MSIVs completely closed | - |
| 🡐(DRN 02-1479, R12) | | |
| ➔(DRN 02-1479, R12) | | |
| 🡐(DRN 02-1479, R12) | | |
| 10.6 | Pressurizer empties | - |
| ➔(DRN 02-1479, R12) 10.7 | Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia | 1560 |
| 12.9 | Voids begin to form in reactor vessel upper head | - |
| 40.7 | Safety injection flow begins | |
| 110.0 | Safety injection boron begins to reach reactor core | - |
| 247.8 | Maximum transient reactivity, $10^{-2}$ delta rho | -0.54 |
| 🡐(DRN 02-1479, R12) 1800 | Operator initiates cooldown | - |

➔(DRN 05-543, R14)

* Note: Event is not affected by EPU and hence was not reanalyzed. The setpoint for MSIS used in the analysis is retained in this table. Changing this setpoint for EFW would affect the timing of the sequence of events but not the conclusions of the analysis. Not shown in the sequence of events is EFW that would enter the intact SG at 54 sec. This EFW flow would not affect the conclusions of the analysis.

🡐(DRN 05-543, R14)

➔(EC-8458, R307)

Note: The above table represents the sequence of events for the Mode 3 Steam Line Break analysis under EPU and OSG conditions. This analysis bounds that of the RSGs with respect to subcriticality.

🡐(EC-8458, R307)

TABLE 15.1-23 Revision 14 (12/05)

REACTOR PROTECTION SYSTEM TRIPS AND TRANSIENT EFFECTS CONSIDERED FOR PRETRIP POWER EXCURSION MAIN STEAM SYSTEM PIPING FAILURES

| Trip Function | Transient Effects |
|---|---|
| Inside Containment Break Locations | |
| →(DRN 05-543, R14) High Linear Power Level | Transient decalibration due to temperature shadowing |
| Low Steam Generator Pressure | Environmentally degraded value |
| Low Steam Generator Level | Not credited due to transient and environmental effects |
| High Containment Pressure | None |
| Core Protection Calculators | Variable overpower trip function only due to environmental effects |
| ΔP Low Flow Trip | Environmentally degraded value (to determine most adverse LOAC timing) |
| Low Pressurizer Pressure | Environmentally degraded value |
| Outside Containment Break Locations | |
| High Linear Power Level ←(DRN 05-543, R14) | Transient decalibration due to temperature shadowing |
| Low Steam Generator Pressure | None |
| Low Steam Generator Level | Not credited due to transient effects |
| Low Pressurizer Pressure Trip | None |
| Core Protection Calculators | Full array of CPC trips available |

TABLE 15.1-24 Revision 309 (06/16)

→(DRN 00-1822; 05-543, R14; EC-13881, R304; EC-8458, R307, LBDCR 15-039, R309)

SEQUENCE OF EVENTS FOR THE STEAM SYSTEM PIPING FAILURE EVENT OUTSIDE CONTAINMENT PRETRIP POWER EXCURSION WITH LOSS OF OFFSITE POWER

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Failure in the MSSS Piping | 2.78 $ft^2$ |
| 4.23 | CPCS VOPT trip occurs | 108.9% of 3716 MWt |
| 4.86 | Trip breakers open | --- |
| 5.15 | LOOP occurs, RCPs begin coastdown | --- |
| 5.25 | Maximum core power | 115.6% of 3716 MWt |
| 5.47 | Maximum core heat flux | 108.1% of 3716 MWt |
| 5.46 | CEAs begin to drop | --- |
| 6.1 | Minimum DNBR | 1.381 |

←(DRN 00-1822; 05-543, R14; EC-13881, R304; EC-8458, R307, LBDCR 15-039, R309)

→(DRN 05-543, R14; EC-13881, R304)

TABLE 15.1-24A Revision 309 (06/16)

Sequence of Events for the Steam System Piping Failure Event
Inside Containment Pre-trip Power Excursion With
Loss of Offsite Power

→(EC-8458, R307, LBDCR 15-039, R309)

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Failure in the MSSS Piping | 2.78 $ft^2$ |
| 4.53 | CPCS VOPT trip occurs | 110.8% of 3716 MWt |
| 5.16 | Trip breakers open | --- |
| 5.3 | LOOP occurs, RCPs begin coastdown | --- |
| 5.40 | Maximum core power | 116.50% of 3716 MWt |
| 5.63 | Maximum core heat flux | 108.7% of 3716 MWt |
| 5.76 | CEAs begin to drop | --- |
| 6.4 | Minimum DNBR | 1.365 |

←(DRN 05-543, R14; EC-13881, R304; EC-8458, R307, LBDCR 15-039, R309)

TABLE 15.1-25 Revision 307 (07/13)

KEY PARAMETERS ASSUMED FOR THE
STEAM PIPING FAILURES EVENT
INSIDE CONTAINMENT PRETRIP POWER EXCURSIONS

➔(DRN 00-1822; 05-543, R14; EC-8458, R307)

| Parameter | Assumption |
|---|---|
| Initial Core Power, MWt | 3735 |
| Core Inlet Coolant Temperature, °F | 552 |
| Pressurizer Pressure, psia | 2310 |
| RCS Flowrate, $x10^6$ lbm/hr | 148.0 |
| Pressurizer Level, % | 55.6 |
| SG Pressure, psia | 908 |
| SG level, % NR | 79.3 (40.20 ft) |
| MTC, $10^{-4}$ $\Delta\rho$/°F | -4.2 |
| Doppler Coefficient Multiplier | BOC (Least Negative) |
| Kinetics | Minimum $\beta$ |
| CEA Worth at Trip, %$\Delta\rho$ | -6.0 |
| Break Area, $ft^2$ | 2.78 |

🡐(DRN 00-1822; 05-543, R1; EC-8458, R307)

TABLE 15.1-26 Revision 307 (07/13)

KEY PARAMETERS ASSUMED FOR THE
STEAM PIPING FAILURES EVENT
OUTSIDE CONTAINMENT PRE-TRIP POWER EXCURSIONS

➔(DRN 00-1822; 05-543, R14; EC-8458, R307)

| Parameter | RSG Assumption |
|---|---|
| Initial Core Power, MWt | 3735 |
| Core Inlet Coolant Temperature, °F | 552 |
| Pressurizer Pressure, psia | 2310 |
| RCS Flowrate, $10^6$ lbm/hr | 148.0 |
| Pressurizer Level, % | 55.6 |
| SG Pressure, psia | 908 |
| SG level, % NR | 79.3 (40.20 ft) |
| MTC, $10^{-4}$ $\Delta\rho$/°F | -4.2 |
| Doppler Coefficient Multiplier | BOC (Least Negative) |
| Kinetics | Minimum $\beta$ |
| CEA Worth at Trip, %$\Delta\rho$ | -6.0 |
| Break Area, $ft^2$ | 2.78 |

🡐(DRN 00-1822; 05-543, R14; EC-8458, R307)

TABLE 15.1-27 (Sheet 1 of 2) Revision 307 (07/13)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INCREASED MAIN STEAM FLOW EVENT WITH A CONCURRENT LOSS OF OFFSITE POWER

| | |
|---|---|
| Core Power Level: | 3735 MWt |
| Core Inventory: | See Table 12.2-12a |
| Fission Product Gap Fractions: | |
| Iodines | 10% |
| Noble Gases | 10% |
| Alkali metals (Cs & Rb-86) | 12% |
| Fraction of Fuel Rods in Core Failing (maximum): | 8% |
| Secondary Steaming Pathway | |
| Primary-to-Secondary Leak Rate: | 150 gpd per SG |
| →(EC-40444, R307) Total MSSV/ADV Combined Leakage per Steam Line ←(EC-40444, R307) | 280 lb/hr Until Cold Shutdown |
| Iodine Chemical Form (Reactor Building Release Path): | |
| Elemental | 97% |
| Organic | 3% |
| Particulate | 0% |
| Steaming PF (Iodine and Alkali Metals): | |
| 0 – 30 minutes | 10 |
| > 30 minutes | 100 |
| Duration of Release: | 7.5 hours |
| Control Room Parameters | |
| →(EC-5000081470, R301) Volume: ←(EC-5000081470, R301) | 168,500 $ft^3$ |
| Recirculation Flow Rate: | 3800 CFM |
| Filter Efficiency: | 99% (elemental/ organic/particulate) |
| →(EC-5000081470, R301) Pressurization Flow: | 225 CFM (maximum) |
| Unfiltered In-leakage: ←(EC-5000081470, R301) | 100 CFM |
| Breathing Rate: ←(DRN 04-704, R14) | 3.47E-04 $m^3$/sec |

TABLE 15.1-27 (Sheet 2 of 2) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INCREASED MAIN STEAM FLOW EVENT WITH A CONCURRENT LOSS OF OFFSITE POWER

→(EC-5000081470, R301)
←(EC-5000081470, R301)

Main Control Room $\chi$/Q Assumed:

| Time | Secondary Steaming Unfiltered Inleakage | Secondary Steaming Pressurization Flow |
|---|---|---|
| →(EC-5000081470, R301) | | |
| 0 – 2 hr | 5.37E-02 | 3.90E-03 * |
| 2 – 8 hr | 3.77E-02 | 2.91E-03 * |
| ←(EC-5000081470, R301) | | |

* factor of 4 reduction credited per SRP 6.4.

→(EC-5000081470, R301)
Steaming (lbm) Releases
←(EC-5000081470, R301)

→(DRN 05-1551, R14)

| 0 – 2 hr Steaming | 2 – 7.5 hr Steaming |
|---|---|
| ←(DRN 05-1551, R14) | |
| 609,744 | 858,838 |

→(EC-5000081470, R301)

←(DRN 04-704, R14; EC-5000081470, R301)

TABLE 15.1-28 Revision 14 (12/05)

→(DRN 04-704, R14)

RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED INCREASED MAIN STEAM FLOW EVENT WITH A CONCURRENT LOSS OF OFFSITE POWER

| Dose Location | EAB * | LPZ | Control Room |
|---|---|---|---|
| Total (Rem TEDE) | < 2.5 | < 2.5 | < 5.0 |

* Worst 2 hour doses listed

←(DRN 04-704, R14)