15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (TURBINE PLANT)

15.2.1 MODERATE FREQUENCY INCIDENTS

15.2.1.1 Loss of External Load

15.2.1.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of external load classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. A loss of external load may be caused by abnormal events in the electrical distribution network.

15.2.1.1.2 Sequence of Events and Systems Operation

A loss of external load results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves. The Steam Bypass System is normally in automatic mode and would be available upon turbine trip. In the event of a turbine trip with the Steam Bypass System in the manual mode, reactor trip would occur as a result of high pressurizer pressure. With the Steam Bypass System in the manual mode and with no credit for immediate operator action, the steam generator safety valves open to relieve steam. Following a loss of external load, offsite power is normally available. The case of loss of all normal ac power is described in Subsection 15.2.1.4. If necessary, the operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip.

The systems operations described above and the resulting sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum, which is described in Subsection 15.2.1.3, since the condenser is available for plant cooldown, assuming operator action after 30 minutes.

15.2.1.1.3 Core and System Performance

The core and system performance parameters following a loss of external load would be no more adverse than those following a loss of condenser vacuum, which is described in Subsection 15.2.1.3.

15.2.1.1.4 Barrier Performance

The barrier performance parameters following a loss of external load would be less adverse than those following a loss of condenser vacuum (See Subsection 15.2.1.3), because the Steam Bypass System would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.1.1.5 Radiological Consequences

→(DRN 04-704, R14)
The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.2.4.5.
←(DRN 04-704, R14)

15.2.1.2 Turbine Trip

15.2.1.2.1 Identification of Causes and Frequency Classification

→(DRN 05-543, R14)
The estimated frequency of a turbine trip classifies it as a moderate frequency incident as defined in Reference 2 of Section 15.0. A turbine trip can be produced by any of the following signals:
←(DRN 05-543, R14)

a) Manual emergency trip

b) Turbine bearing oil pressure low

c) Condenser vacuum loss (See Subsection 15.2.1.3)

d) Rotor Vibration Excessive

e) Stator rectifier cooling water flow low

f) Seal oil differential pressure low

g) Differential Expansion Excessive

h) Exhaust Hood Temperature High

i) Feedwater Heaters 5 & 6 level high
→(EC-13881, R304)
j) Reactor trip
←(EC-13881, R304)
k) Excessive thrust bearing wear
→(EC-13881, R304)
l) Turbine overspeed
←(EC-13881, R304)
m) Moisture separator/reheater drain tank level high

n) Turbine Oil Tank level low

o) DEH DC Power Bus under voltage

p) Generator Lockout relays tripped

15.2.1.2.2 Sequence of Events and Systems Operation

A turbine trip results in a reduction of steam flow due to closure of the turbine stop valves. The Steam Bypass System is normally in the automatic mode and will function following turbine trip. It is assumed that the reactor trip on turbine trip is unavailable. In the event that the turbine stop valves were to close with the Steam Bypass System in the manual mode, reactor trip would occur as a result of high pressurizer pressure.

With the Steam Bypass System in the manual mode and with no credit for immediate operator action, the steam generator safety valves will open to relieve steam. Following a turbine trip, offsite power is available to provide power to the auxiliaries. The operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip. The case of loss of all normal ac power is described in Subsection 15.2.1.4.

The systems operations described above, and the resulting sequence of events, would produce consequences no more adverse than those following a loss of condenser vacuum, described in Subsection 15.2.1.3, since the condenser is available to cool the plant, assuming operator action after 30 minutes.

15.2.1.2.3 Core and System Performance

The core and system performance parameters following a turbine trip would be no more adverse than those following a loss of condenser vacuum, as described in Subsection 15.2.1.3.

15.2.1.2.4 Barrier Performance

The barrier performance parameters following a turbine trip would be less adverse than those following a loss of condenser vacuum (See Subsection 15.2.1.3), because the Steam Bypass System would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.1.2.5 Radiological Consequences

→(DRN 04-704, R14)
The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.2.4.5.
←(DRN 04-704, R14)

15.2.1.3 Loss of Condenser Vacuum

15.2.1.3.1 Identification of Causes and Frequency Classification
→(DRN 06-1062, R15; EC-8458, R307. LBDCR 15-039, R309)
The estimated frequency of a loss of condenser vacuum classifies it as a moderate frequency incident, as defined in Reference 1 of Section 15.0. A loss of condenser vacuum may occur due to failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove noncondensible gases, or excessive leakage of air through a turbine gland packing. 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 evaluation using the revised SCRAM curve time presented in Table 15.0-5.
←(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

15.2.1.3.2 Sequence of Events and Systems Operation
→(EC-13881, R304)
The turbine generator trip that occurs due to a loss of condenser vacuum would lead to a reactor trip on high-pressurizer pressure. The turbine bypass system and the turbine driven main feedwater pumps are unavailable following a loss of condenser vacuum, and it is assumed that the reactor trip on turbine trip is unavailable. The pressure increases in the primary and secondary systems following reactor trip are limited by the pressurizer and steam generator safety valves. The loss of condenser vacuum causes a turbine trip. Following turbine trip, offsite power is available to provide power to the auxiliaries. The operator may cool the NSSS using emergency feedwater and the atmospheric dump valves any time after reactor trip. The case of loss of all normal power is described in Subsection 15.2.1.4.
←(EC-13881, R304)

In this analysis, it is conservatively assumed that operator action is delayed until 30 minutes after first indication of the event.

➔(DRN 05-543, R14)
Because this event bounds the results of the event with any single malfunction of an active component or system, the consequences of a single malfunction of an active component or system following a loss of condenser vacuum are discussed within this section.
🡐(DRN 05-543, R14)

Table 15.2-1 gives the sequence of events following a loss of condenser vacuum.

15.2.1.3.3 Core and System Performance

15.2.1.3.3.1 Mathematical Model
➔(DRN 05-543, R14; EC-13881, R304)
The NSSS response to a loss of condenser vacuum 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, with the WSSV-T and ABB-NV correlations described in Chapter 4.
🡐(EC-13881, R304)

15.2.1.3.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a loss of condenser vacuum are discussed in Section 15.0. In particular, those parameters that were unique to the analysis discussed below are listed in Table 15.2-2. Selection of the mode of operation for the pressurizer control systems has a negligible effect on the limiting parameters and merely influences the timing of the sequence of events.

➔(DRN 00-592; EC-8458, R307)
The initial conditions for the principal process variables monitored by the COLSS were varied within the reactor operating region given in Table 15.0-4 to determine the set of conditions that would produce the most adverse consequences following a loss of condenser vacuum. Various combinations of initial core inlet temperature, core inlet flow rate, and pressurizer pressure were considered and their effects on peak pressurizer and steam generator pressures were assessed. Decreasing the initial core inlet temperature delays the opening of the steam generator safety valves, because of a lower initial secondary pressure. The Loss of Condenser Vacuum analyses has assumed a 533°F initial RCS inlet temperature and an initial Steam Generator Pressure of 732.0 psia. This accounts for the full range of values allowed by Technical Specifications for indicated RCS inlet temperature of 536°F to 549°F. Decreasing the initial RCS pressure delays the high-pressurizer pressure reactor trip and the opening of the pressurizer safety valves. At an initial RCS pressure below 2090 psia, the steam generator safety valves open sufficiently to offset the delay in the energy removing capability of the pressurizer safety valves.
🡐(DRN 00-592)

Increasing the coolant flow rate produces faster transport through the RCS of the primary energy increase (RCS heatup) which results from the loss of secondary heat removal. Above 115 percent of design flow, the high pressurizer pressure trip signal is generated soon enough to negate the effect of faster heat transport.
🡐(EC-8458, R307)

For the peak primary pressure analysis, input parameters include:

- The least negative (BOC) Doppler coefficient was assumed.

- A maximum (BOC) delayed neutron fraction and minimum neutron lifetime.
🡐(DRN 05-543, R14)

➔(DRN 05-543, R14)

- The CEA insertion curve assumed accounts for a 0.6 second holding coil delay. A CEA worth of -6.0% $\Delta\rho$ was conservatively assumed.

- An HPPT setpoint of 2422 psia and a response time of 0.9 seconds were assumed.

- An MSSV tolerance of +3% was applied.

- An initial core power of 3735 MWt, based on a rated power of 3716 MWt and a 0.5% uncertainty, was assumed.

- A most positive/least negative MTC of $-0.2 * 10^{-4}$ $\Delta\rho$/°F at hot full power (HFP) was used.

- The minimum HFP core inlet temperature of 533°F was assumed.

- A PSV tolerance of +3% was applied.

- Parametric analyses were performed on coolant flow, temperature, pressurizer level, and SG level. The most limiting conditions were maximum RCS flow, minimum inlet temperature, maximum pressurizer level, and minimum SG level.

A similar analysis was formed to determine a conservative peak secondary pressure. This secondary system analysis is effectively the same as the peak primary analysis with the following differences:

➔(EC-8458, R307)

- The pressurizer level and pressure control system are in automatic operation.

🡐(EC-8458, R307)

- The maximum HFP core inlet temperature of 552°F was assumed.

➔(EC-8458, R307)

- Parametric analyses were performed on coolant flow, temperature, pressurizer level, and SG level. The most limiting conditions were minimum RCS flow, maximum inlet temperature, and maximum SG level.

🡐(EC-8458, R307)

- A minimum RCS flow of 148×10^6 lbm/hr was assumed.

For the inoperable MSSV analysis (one or two valves), the following assumptions either complement or replace the peak secondary system pressure case.

- An ESFAS low SG level actuation of 5% and a response time of 0.9 secondary were credited.

- Initial core powers were based on a HFP value of 3716 MWt. For one inoperable MSSV bank, an initial core power of 87.3% RTP was used. For two inoperable MSSV banks, an initial core power of 69.9% RTP was used.

- For the one or two inoperable MSSVs per steam line, the lowest opening valve(s) were assumed to be inoperable.

➔(EC-8458, R307)

The single failure considered for the peak primary pressure cases was the failure of the pressurizer level measurement channel. The effect of that failure is a false low pressurizer level signal, which would result in the activation of the charging pumps and the closing of the letdown valves. No single failure of the peak secondary side was assumed since there was no single failure, including a loss of offsite power, that resulted in a peak secondary pressure larger than the case without a single failure.

🡐(DRN 05-543, R14; EC-8458, R307)

15.2.1.3.3.3 Results

The dynamic behavior of important NSSS parameters following a loss of condenser vacuum is shown in Figures 15.2-1 through 15.2-13.

→(DRN 05-543, R14; EC-8458, R307)
The loss of steam flow due to closure of the turbine stop valves produces a rapid increase in the secondary pressure. This produces a rapid decrease in the primary-to-secondary heat transfer and causes a rapid heatup of the primary coolant. The insurge to the pressurizer increases the pressurizer pressure producing high-pressurizer pressure reactor trip condition at 5.5 seconds. The CEAs begin dropping into the core at 7.0 seconds, terminating the transient.
→(LBDCR 15-039, R309)
The opening of the steam generator safety valves (at 16.8 seconds) and the pressurizer safety valves (at 7.1 seconds) combines with the decreasing core power after reactor trip to reduce the primary and secondary pressures after reaching a maximum RCS pressure of 2712 psia at 7.9 seconds. The pressurizer safety valves close at 11.7 seconds. The steam generator safety valves continue to cycle and relieve steam to the atmosphere until the atmospheric steam dump valves are opened by operator action at 30 minutes. The plant is then cooled to 350°F and shutdown cooling is initiated.

The maximum RCS and secondary pressures remained below their respective acceptance criteria of 110% of design pressures. The peak RCS pressure case yielded a peak RCS pressure of 2712 psia The peak steam generator pressure case showed a peak SG pressure of 1181 psia.
←(EC-8458, R307, LBDCR 15-039, R309)

Additional simulations of the loss of condenser vacuum are run with 1 and 2 inoperable main steam safety valves. To offset the reduction in steam relief capability while maintaining peak system pressures within their acceptance criteria, the core power for these cases is restricted to less than full power operation. The combinations of allowed core power and inoperable main steam safety valves are found in Table 3.7-2 of plant technical specifications.

→(EC-8458, R307)
←(EC-8458, R307)

→(EC-13881, R304)
The minimum DNBR remains above the SAFDL limit of 1.24 indicating no violation of the fuel thermal limits.
←(EC-13881, R304)

The results of this event would be less adverse than those following an increased main steam flow with concurrent loss of offsite power, as described in Subsection 15.1.2.3. No fuel failure is predicted for the loss of condenser vacuum event.

Analyses were also conducted to demonstrate that adequate operator action times exist to prevent the pressurizer from becoming liquid filled for this event. A failure in the Pressurizer Level Control System (PLCS) was assumed that resulted in terminating letdown flow and starting all three charging pumps. Operator action at 15 minutes into the event to trip charging pumps and, if applicable, restore letdown demonstrated that margin exists to a solid pressurizer condition. This time for operator action is greater than the 10 minutes allowed per NUREG-0800. For this analysis, minimum core inlet temperature and most positive MTC are assumed to maximize the resulting pressurizer level.
←(DRN 05-543, R14)

15.2.1.3.4 Barrier Performance

15.2.1.3.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.2.1.3.3.

15.2.1.3.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.2.1.3.3.

15.2.1.3.4.3 Results

➔(DRN 05-543, R14)
Figures 15.2-12 and 15.2-13 show the pressurizer and steam generator safety valve flow rate vs. time for the loss of condenser vacuum transient. The steam discharged from the pressurizer is completely condensed in the quench tank and hence not released to the atmosphere. The steam released to the atmosphere through the steam generator safety valves is not greater than that following the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.1.4.
🡐(DRN 05-543, R14)

15.2.1.3.5 Radiological Consequences
➔(DRN 04-704, R14)
The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.2.4.5.
🡐(DRN 04-704, R14)

15.2.1.4 <u>Loss of Normal AC Power</u>

15.2.1.4.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of normal ac power classifies it as a moderate frequency incident, as defined in Reference 1 of Section 15.0. The loss of normal ac power is assumed to result in the loss of all power to the station auxiliaries and a concurrent turbine generator trip. This situation could result either from a complete loss of external grid (offsite) or a loss of the onsite ac distribution system. As a result, electrical power would be unavailable for the reactor coolant pumps, condensate pumps, circulating water pumps, and pressurizer pressure and level control systems. Under such circumstances, the plant would experience simultaneous losses of load, feedwater flow, and forced reactor coolant flow.

15.2.1.4.2 Sequence of Events and System Operation

➔(DRN 05-543, R14; EC-8458, R307)
At time zero, when all normal ac power is assumed to be lost to the plant, the turbine stop valves close. It is assumed that the flow area of the turbine control valves is instantaneously reduced to zero and that the feedwater flow to both steam generators goes instantaneously to zero. The reactor coolant pumps coast down and reactor coolant flow begins to decrease. A reactor trip will occur as a result of a low DNBR condition as soon as the flow coastdown begins. The low DNBR trip ensures that the minimum DNBR will not be less then the SAFDL of 1.24. The pressure increases in the RCS and steam generator are limited by the primary (pressurizer) and steam generator safety valves.
🡐(DRN 05-543, R14; EC-8458, R307)

→(DRN 05-543, R14)
On loss of normal ac power, the emergency diesel generators are automatically started. After reactor trip, stored heat and fission product decay heat must be dissipated by the Main Steam System. In the absence of forced reactor coolant flow, convective heat transfer is supported by natural circulation of reactor coolant flow. Initially, the residual water inventory in the steam generators provides a heat sink, and the resultant steam is released to atmosphere by the steam generator safety valves. Emergency feedwater is automatically initiated on a low steam generator water level signal. Additional equipment which must operate to maintain safe shutdown conditions is listed in Table 8.3-1. Plant cooldown is controlled by the atmospheric steam dump valves if normal ac power cannot be restored within 30 minutes.
←(DRN 05-543, R14)

The consequences of a single malfunction of a component or system following a loss of normal ac power are discussed in Subsection 15.2.2.4.

→(DRN 05-543, R14)
←(DRN 05-543, R14)

15.2.1.4.3 Core and System Performance

→(DRN 05-543, R14)

→(DRN 05-543, R14)
The DNBR/fuel performance aspects of this event is bounded by the loss of flow, Section 15.3.2.1. The peak pressure aspects of this event is bounded by the LOCV, Section 15.2.1.3.
←(DRN 05-543, R14)

15.2.1.4.4 Barrier Performance

→(DRN 05-543, R14)
The barrier performance parameters following a loss of normal ac power would be less adverse than those following an inadvertent opening of the atmospheric dump valve, as described in Section 15.1.1.4.
←(DRN 05-543, R14)

15.2.1.4.5 Radiological Consequences
→(DRN 04-704, R14)
The radiological consequences due to steam releases for the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.2.4.5.
←(DRN 04-704, R14)

15.2.1.5 Steam Pressure Regulator Failure

The failure of the steam pressure regulator would result in closing the turbine throttle valves. This transient is less severe than a loss of condenser vacuum described in Subsection 15.2.1.3, where the maximum rate of throttle valve closure is assumed.

15.2.2 INFREQUENT INCIDENTS

15.2.2.1 Loss of External Load with a Concurrent Single Failure of an Active Component

15.2.2.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of external load 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 loss of external load is caused by abnormal events in the electrical distribution network.

15.2.2.1.2 Sequence of Events and Systems Operation

→(DRN 05-543, R14)
The systems operations following a loss of external load with a concurrent single failure of an active component are the same as those described in Subsection 15.2.1.1.2. The single malfunction of a component or system is discussed Subsection 15.2.1.3 for the loss of condenser vacuum with a concurrent single failure of an active component. The resultant sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum which is described in Subsection 15.2.1.3.

15.2.2.1.3 Core and System Performance

The core and system performance parameters following a loss of external load, with a concurrent single failure of an active component, would be no more adverse than those following a loss of condenser vacuum which is described in Subsection 15.2.1.3.

15.2.2.1.4 Barrier Performance

The barrier performance parameters following a loss of external load with a concurrent single failure of an active component would be less adverse than those following a loss of condenser vacuum (see Subsection 15.2.1.3), since the Steam Bypass System would be available to remove steam to the condenser.
←(DRN 05-543, R14)

15.2.2.1.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in Subsection 15.1.2.4.

15.2.2.2 Turbine Trip with A Concurrent Single Failure of an Active Component

15.2.2.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a turbine trip with a concurrent single failure of an active component classifies it as an infrequent incident defined in Reference 1 of Section 15.0. The conditions that can produce a turbine trip are listed in Subsection 15.2.1.2.1.

15.2.2.2.2 Sequence of Events and Systems Operation

→(DRN 05-543, R14)
The systems operations following a turbine trip with a concurrent single failure of an active component are the same as those described in Subsection 15.2.1.2.2. The single malfunction of a component or system is discussed in Subsection 15.2.1.3 for the loss of condenser vacuum with a concurrent single failure of an active component. The resultant sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum as described in Subsection 15.2.1.3.

15.2.2.2.3 Core and System Performance

The core and system performance parameters following a turbine trip with a concurrent single failure of an active component would be no more adverse than those following a loss of condenser vacuum as described in Subsection 15.2.1.3.

15.2.2.2.4 Barrier Performance

The barrier performance parameters following a turbine trip with a concurrent single failure of an active component would be less adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component (see Subsection 15.2.1.3), since the Steam Bypass System would be available to remove steam to the condenser.
←(DRN 05-543, R14)

15.2.2.2.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in Subsection 15.1.2.4.

15.2.2.3 Loss of Condenser Vacuum with a Concurrent Single Failure

→(DRN 05-543, R14)

This event is bounded by the LOCV. See Section 15.2.1.3.

←(DRN 05-543, R14)

15.2.2.4 Loss of all Normal AC Power with a Concurrent Single Failure of an Active Component

→(DRN 05-1201, R14)
The DNBR/fuel performance aspects of this event are bounded by the loss of flow, Section 15.3.2.1. The peak pressure aspects of this event are bounded by the LOCV, Section 15.2.1.3.
←(DRN 05-1201, R14)

15.2.2.5 Loss of Normal Feedwater Flow

15.2.2.5.1 Identification of Causes and Frequency Classification
→(DRN 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)
The estimated frequency of a loss of normal feedwater flow classifies it as an infrequent incident as defined in Reference 1 of Section 15.0. 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)

A loss of normal feedwater flow is defined as a reduction in feedwater flow to the steam generators when operating at power, without a corresponding reduction in steam flow from the steam generators. This flow imbalance results in a reduction in the steam generator water inventory and a consequent heatup of the reactor coolant. The complete loss of normal feedwater case is analyzed since this condition requires the most rapid response from the plant protection system (PPS). A complete loss of normal feedwater flow can result from the loss of both main feedwater pumps or the loss of three condensate pumps or by a control system malfunction causing the feedwater control valves to close. Manually closing the feedwater control or isolation valves will also result in a complete loss of normal feedwater flow.

15.2.2.5.2 Sequence of Events and Systems Operation

The complete loss of normal feedwater flow is analyzed by assuming an instantaneous stoppage of feedwater flow to both steam generators. The PPS provides protection against the loss of the secondary heat sink by the steam generator low water level trip and by automatic initiation of the emergency feedwater system. The emergency feedwater system consists of one turbine-driven and two motor-driven emergency feedwater pumps. The high pressurizer pressure trip provides protection in the event the RCS pressure limit is approached. The steam bypass control system is assumed to be in the automatic mode, which maximizes the decrease in steam generator water inventory. Table 15.2-6 gives the sequence of events for the complete loss of normal feedwater.

The consequences of a single malfunction of a component or system following a loss of normal feedwater flow are discussed in Subsection 15.2.3.2.

15.2.2.5.3 Core and System Performance

15.2.2.5.3.1 Mathematical Model

→(DRN 05-1201, R14; EC-13881, R304)
The NSSS response to a loss of normal feedwater flow 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, with the WSSV-T and ABB-NV correlations described in Chapter 4.
←(DRN 05-1201, R14; EC-13881, R304)

15.2.2.5.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a complete loss of normal feedwater are discussed in Section 15.0. In particular, those parameters which were unique to the analysis discussed below are listed in Table 15.2-7.

→(DRN 05-543, R14)
The initial conditions for the principal process variables monitored by the COLSS system were varied within the reactor operating space given in Table 15.0-4 to determine the set of conditions that would produce the maximum decrease in steam generator water inventory following a complete loss of normal feedwater flow. No set of initial conditions could be found such that for a complete loss of normal feedwater flow the RCS pressure would approach 110 percent of the design pressure. Various combinations of initial core inlet temperature, initial pressurizer pressure, and initial core flowrate were considered. Increasing the initial core inlet temperature increases the secondary side pressure; this causes the Steam Bypass System to open sooner after the cessation of the feedwater flow. Therefore, an inlet temperature of 552°F was used in this analysis. Lowering the initial pressurizer pressure to 2090 psia ensures that the reactor trip signal will not be generated from a high pressurizer pressure signal. A reactor trip on low steam generator water level will minimize the steam generator water inventory during this transient. The initial core flowrate has little effect on the transient minimum steam generator water inventory.
←(DRN 05-543, R14)

The initial steam generator water level was set at the high-level alarm setting. At this setting, the reactor trip on low steam generator water level is delayed, so that the reactor coolant temperature will be increased to the maximum possible value. Increasing the reactor coolant temperatures will increase the secondary pressure and minimize the steam generator water inventory.

→(DRN 05-543, R14)
Assumptions for this analysis include:

- The EOC Doppler curve was assumed with a 1.3 multiplier.
- A BOC delayed neutron fraction and neutron lifetime were assumed.
- The CEA insertion curve based on the limiting +0.3 ASI shape was assumed. This curve accounts for a 0.6 second holding coil delay. A CEA work of -6.0% $\Delta\rho$ was conservatively assumed.
- An MSSV tolerance of -3% was applied.
- An initial core power of 3735 MWt, based on a rated power of 3716 MWt and a 0.5% uncertainty, was assumed.
- A most negative MTC of $-4.2 * 10^{-4}$ $\Delta\rho$/°F was used for the no single failure case.

←(DRN 05-543, R14)

Finally a large bottom peaked axial shape was utilized to ensure conservative power reduction as the CEAs are inserted on reactor trip.

15.2.2.5.3.3 Results

→(EC-13881, R304)
The dynamic behavior of important parameters following a loss of normal feedwater is shown in Figures 15.2-26 through 15.2-36a.
←(EC-13881, R304)

→(DRN 05-543, R14; EC-8458, R307, LBDCR 15-039, R309)
The complete loss of normal feedwater results in an increase in the secondary pressure and temperature which in turn cause the RCS temperatures to increase. The turbine continues to operate, depleting secondary steam generator inventory. The RCS pressure increases as the temperature and power increases. Feedwater is halted at 0.0 seconds and at 8.1 seconds the reactor power level increases to 101.0%. The reactor reaches reactor trip conditions on low steam generator water level at 43.2 seconds. The CEAs begin to drop at 44.7 seconds. The negative reactivity added by the CEAs rapidly reduces the reactor core power. The steam removed through the Steam Bypass System and the steam generator safety valves rapidly cools the RCS following the reactor trip. The maximum pressures in the RCS and Main Steam Systems are 2268 and 1072 psia, respectively. Emergency feedwater from the three EFW pumps reaches the steam generators 50 seconds after actuation of low steam generator level trip. The total steam generator inventory in the affected generator reaches its minimum value of 42,830 lbm at 190.1 seconds. The Steam Bypass System operates to remove decay heat until operator action is taken. In this analysis, it is conservatively assumed that operator action is delayed until 30 minutes after initiation of the event. The primary system is then cooled to 350°F at which point shutdown cooling is initiated. Should the main condenser become unavailable, cooldown will be accomplished using the atmospheric dump valves.
←(DRN 05-543, R14, LBDCR 15-039, R309)

For complete loss of normal feedwater flow, the DNB ratio remains above the SAFDL limit and the PPS assures both that the steam generator heat removal capability is maintained and that the RCS pressure does not exceed 110 percent of design.
←(EC-8458, R307)

15.2.2.5.4 Barrier Performance

15.2.2.5.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.2.2.5.3.

15.2.2.5.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.2.2.5.3.

15.2.2.5.4.3 Results

→(DRN 05-543, R14)

The results of this event would be less adverse then those following an increased main steam flow with concurrent loss of offsite power, as described in Subsection 15.1.2.3. No fuel failure is predicted for the loss of normal feedwater flow event.

Analyses were also conducted to demonstrate that adequate operator action times exist to prevent the pressurizer from becoming liquid filled for this event. A failure in the Pressurizer Level Control System (PLCS) was assumed that resulted in terminating letdown flow and starting all three charging pumps. Operation action at 15 minutes into the event to trip charging pumps and, if applicable, restored letdown demonstrated margin exists to a solid pressurizer condition. This time for operator action is greater than the 10 minutes allowed per NUREG-0800. For this analysis, minimum core inlet temperature and most positive MTC are assumed to maximize the resulting pressurizer level.

←(DRN 05-543, R14)

15.2.2.5.5 Radiological Consequences

→(DRN 04-704, R14)

The radiological consequences of this event are less severe than the consequences of the Increased Main Steam Flow with LOOP, discussed in Subsection 15.1.2.3.5.

←(DRN 04-704, R14)

15.2.3 LIMITING FAULTS

15.2.3.1 Feedwater System Pipe Breaks

→(DRN 02-1713, R12-B; 05-543, R14; 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

This section describes the large and small Feedwater System Pipe Break events. The peak RCS pressure criteria for the large and small breaks with concurrent loss of normal AC power is 120% of the design pressure; the criteria for small break without loss of AC power and worst single active failure is 110% of the design pressure. 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 02-1713, R12-B; 05-543, R14; 06-1062, R15; EC-8458, R307, LBDCR 15-039, R309)

15.2.3.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a Feedwater System pipe break classifies it as a limiting fault incident as defined in Reference 1 of Section 15.0. A feedwater system pipe break may occur due to a pipe failure in the Feedwater System.

15.2.3.1.2 Sequence of Events and Systems Operation

A Feedwater System pipe break may result in a total loss of normal feedwater flow and a rapid blowdown of one steam generator. With a concurrent loss of normal AC power, the plant will undergo a simultaneous loss of forced reactor coolant flow and a turbine trip. In addition, the normal pressurizer

level and pressure control functions will be lost, and the Steam Bypass System will not be available. This situation results in a rapid decrease in the heat removal capability of both steam generators and finally in the loss of one steam generator, causing heatup and pressurization of the RCS.

→(DRN 02-1713, R12-B)
The NSSS is protected during this transient by the primary (pressurizer) safety valves and by the following reactor trips:
←(DRN 02-1713, R12-B)

a) Steam generator low water level

b) Steam generator low pressure
→(DRN 02-1713, R12-B)
c) High pressurizer pressure (HPPT)
←(DRN 02-1713, R12-B)
d) Low DNBR
→(DRN 02-1713, R12-B)
e) High Containment Pressure

→(DRN 05-543, R14)
Depending on the particular set of initial conditions, any of these trips may terminate the transient. The NSSS is also protected by the steam generator safety valves and the Emergency Feedwater System which together provide the ability for heat removal following reactor trip. In this analysis, the most adverse single active failure for the small FWLB is a failure to Fast Bus Transfer, resulting in coast down of two RCPs. For the large FWLB the most adverse single active failure is the LOOP. The operator can initiate a controlled plant cooldown using the atmospheric steam dump valves any time after reactor trip occurs. In this analysis, it is conservatively assumed that operator action is delayed until 30 minutes after the event.
←(DRN 02-1713, R12-B)
→(DRN 05-1551, R14; EC-8458, R307)
The Large Feedwater Pipe Break and Small Feedwater Pipe Break analyses employ differing analytical assumptions as detailed in FSAR Section 15.2.3.1.3 and 15.2.3.1.4, respectively. An evaluation parametric in break size was performed using both large and small break assumptions, and is reported in Figures 15.2-53a.1 and 15.2-53a.2, respectively. Using the large break model, peak RCS pressure is seen to increase as break size is reduced until a break size of about 0.12 ft^2. Using the small break model, peak RCS pressure increases as break size is increased until a break size of approximately 0.19 ft^2 is reached. Thus both large and small break models show that an intermediate break size is most limiting. The limiting break size for the small break model is slightly larger than that of the large break model due to differing modeling assumptions.
←(DRN 05-1551, R14; EC-8458, R307)

→(DRN 00-1731, R11; 02-1713, R12-B)
←(DRN 00-1731, R11; 02-1713, R12-B)

→(DRN 02-1713, R12-B, LBDCR 15-039, R309)
15.2.3.1.3 Large FW Pipe Break

The limiting break size for the large FWLB event is 0.12 ft^2. The limiting large break size is due to reverse steam flow from the intact SG through the main steam lines to the faulted SG and out of the break. This steam flow contributes to the heat removal capability of the intact SG and therefore helps to mitigate the RCS heat up and pressurization. Reverse steam flow can be established only after the faulted SG dries out and increases with increasing break size. The initial SG liquid mass in the faulted SG and the initial RCS pressure are adjusted to assure that the faulted SG dryout occurs before the HPPT condition to maximize the RCS pressure.
←(DRN 05-543, R14, LBDCR 15-039, R309)
Table 15.2-8 gives the sequence of events for a large FWLB with loss of offsite power.
←(DRN 02-1713, R12-B)

→(DRN 02-1713, R12-B)

15.2.3.1.3.1 Core and System Performance

15.2.3.1.3.1.1 Mathematical Model

→(DRN 05-543, R14; EC-13881, R304)

The NSSS response to a Feedwater System pipe break was simulated using the CENTS computer program described in Section 15.0 along with the blowdown model described below. Using the core heat flux and core inlet conditions calculated by CENTS, the thermal margin on DNBR in the reactor core was simulated using the CETOP computer program with the WSSV-T and ABB-NV correlations described in Chapter 4.

←(DRN 02-1713, R12-B; 05-543, R14; EC-13881, R304)

Blowdown of the steam generator nearest the feedwater line break was modeled assuming frictionless critical flow calculated by the Henry-Fauske correlation (Reference 1). The enthalpy of the blowdown is initially assumed to be that of saturated liquid. As the steam generator liquid mass decreases, the quality of the blowdown is allowed to increase to that value which is calculated by assuming that all of the liquid mass is contained in the downcomer region, and that it forms a homogeneous two-phase mixture with a two-phase level which remains at the height of the break (bottom of the feedwater ring). This model underestimates the blowdown quality and energy and overestimates the discharge rate, thereby leading to a more rapid blowdown and subsequent loss of steam generator heat removal capability.

Assuming that the two-phase mixture level remains at the feedwater ring as the quality increases also provides a very conservative prediction of the minimum steam generator liquid mass existing in the steam generator connected to the ruptured feedwater line at the time of the low water level trip. Since this model underestimates the quality in the downcomer, the two-phase density and static head between the level sensors are overestimated. This method, therefore, predicts a higher level for a given liquid mass than can actually exist, thereby delaying the low level trip.

→(DRN 02-1713, R12-B)

15.2.3.1.3.1.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a large feedwater pipe break are discussed in Section 15.0. In particular, those parameters that are unique to the analysis are discussed below and are listed in Table 15.2-9.

←(DRN 02-1713, R12-B)

The initial conditions for the principal process variables monitored by the 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 a Feedwater System pipe break. The full spectrum of break areas was considered up to a break size of the combined area of the flow distributing nozzles in the bottom of the feedwater ring. The initial intact steam generator inventory and the initial RCS pressure were adjusted within the plant operating space in order to produce as nearly as possible simultaneous trip conditions for:

a) Low water level in the intact steam generator

b) Low water level in the ruptured steam generator

c) High pressurizer pressure

→(DRN 05-543, R14; 05-1201, R14)
The principal conservative assumptions and analytical methods utilized in the analysis of this event include:

a) Conservative estimation of the break flow and enthalpy, i.e., discharge of liquid from the downcomer until the SG is dry. Consistent with the pre-uprate licensing basis, the steam generator is considered dry when 9,000 lbm or less of liquid inventory is left. To establish this, the break is assumed to occur at the bottom of the SG, that is, at the tube sheet elevation.
b) Delay of heat transfer degradation in the affected steam generator until the liquid inventory is depleted. The heat transfer area is ramped from unity at 9,000 lbm liquid mass in the SG to zero at 2,000 lbm liquid mass in the SG.
c) Initializing key parameters such that a reactor trip occurs on high pressurizer pressure coincident with low SG level trip, further, the low SG level trip is delayed until liquid mass inventory in the affected SG is depleted. Parametric study shows that the coincident trips result in the most adverse results. Although depletion of the liquid mass inventory results in a low SG level trip much earlier, no credit is taken for the low SG level trip until the affected SG liquid mass inventory is depleted. The SG low level trip is activated when the SG liquid mass is less than 9,000 lbm.
d) Delaying the emergency feedwater actuation signal until liquid mass inventory in the affected SG is depleted (the SG liquid mass is less than 9,000 lbm).

←(DRN 05-543, R14; 05-1201, R14)

→(DRN 02-1713, R12-B; 05-1201, R14)
Loss of normal electrical power is assumed at the time of turbine trip. Selection of these conditions maximizes both the RCS pressure and the mismatch between core power and steam generator heat removal at the time of trip. Core inlet temperature and flow had negligible effects on the peak RCS pressure for a given blowdown rate. However, maximizing the core inlet temperature also maximizes the steam generator pressure thereby increasing the blowdown rate. The maximum inlet temperature of 552°F also maximizes the RCS energy content, thereby increasing the releases associated with safety valve and dump valve discharges. Other conservative assumptions include failure of the pressurizer pressure and level control system.
←(DRN 02-1713, R12-B; 05-1201, R14)

→(DRN 02-1713, R12-B)
←(DRN 02-1713, R12-B

→(DRN 02-1713, R12-B)
15.2.3.1.3.1.3 Results

The dynamic behavior of important parameters following a large FWLB is illustrated on Figures 15.2-37 through 15.2-53.

→(DRN 00-592, R11-A; 05-1201, R14; EC-8458, R307 LBDCR 15-039, R309)
It is assumed that feedwater flow to the steam generator connected to the intact feedwater line drops instantaneously to zero at the time of the break, and that critical flow of saturated liquid from the other steam generator exits through the break. The absence of sub-cooled feedwater produces a gradual heatup of the primary and secondary systems and the steam generator associated with the ruptured line finally empties. This loss of heat removal capability results in a rapid increase in RCS temperature and pressure. A low steam generator inventory trip condition is reached at 24.0 seconds; the trip breakers open at 24.9 seconds and the CEAs begin to drop into the core at 25.5 seconds. The loss of ac power is assumed to occur at 24.9 seconds (time of turbine trip). During this sequence, the pressurizer safety valve setpoint is reached at 26.8 seconds. System temperature and pressure continue to increase briefly until the energy production from the core is matched by the heat removal from the intact steam generator, after which pressure drops rapidly.
←(DRN 00-592, R11-A; 02-1713, R12-B; 05-1201, R14; EC-8458, R307 LBDCR 15-039, R309)

→(DRN 00-592, R11-A; 02-1713, R12-B; 05-543, R14; EC-8458, R307 LBDCR 15-039, R309)
At 27.9 seconds, the peak RCS pressure 2746 psia is reached and, at 36.6 seconds, the peak steam generator pressure of 1123.1 psia is reached.
←(DRN 00-592, R11-A; 02-1713, R12-B; 05-543, R14; EC-8458, R307 LBDCR 15-039, R309)

→(DRN 02-1713, R12-B)
←(DRN 02-1713, R12-B)

The steam generator safety valves continue to cycle until the atmospheric dump valves are opened at 30 minutes. The plant is then cooled to 350°F and shutdown cooling is initiated.

→(DRN 02-1713, R12-B; 05-543, R14; EC-13881, R304)
The minimum DNBR remains above the SAFDL limit of 1.24 indicating no violation of the fuel thermal limits.
←(EC-13881, R304)

The EFW flow of 575 gpm is sufficient to prevent dryout of the unaffected steam generator. System parameters are reported in Tables 15.2-8 and 15.2-9b for the Large and Small Feedwater System Pipe Breaks, respectively.
←(DRN 02-1713, R12-B; 05-543, R14)

→(DRN 02-1713, R12-B)
←(DRN 02-1713, R12-B)

→(DRN 02-1713, R12-B)
15.2.3.1.3.1.4 Barrier Performance

15.2.3.1.3.1.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.2.3.1.1.

15.2.3.1.3.1.4.2 Input Parameters and Initial Conditions
←(DRN 02-1713, R12-B)

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.2.3.1.3.
→(DRN 02-1713, R12-B)
15.2.3.1.3.1.4.3 Results

Figures 15.2-49 and 15.2-50 give the pressurizer and steam generator safety valves flow rates versus time for the Large FWLB. The total steam release through SG safety valves and atmospheric dump valve to the environment for this event is bounded by the steam release due to a main steam line break event discussed in 15.1.3.1.

15.2.3.1.3.1.5 Radiological Consequences
←(DRN 02-1713, R12-B)

➔(DRN 04-704, R14)

15.2.3.1.3.1.5.1 Design Basis – Method of Analysis

15.2.3.1.3.1.5.1.1 Mathematical Model

To evaluate the radiological consequences due to a postulated feedwater line break, it is assumed that there is a complete severance of a feedwater line outside of the containment building. The outside containment line break is evaluated since this event bounds a break inside of containment due to the retention and delay of releases to the environment. No SAFDL violation occurs as a result of this event, therefore no fuel damage is assumed to occur as a result of this event.

➔(EC-40444, R307)

Two release pathways are considered: secondary steaming from an intact steam generator, and direct releases to the environment via primary-to-secondary leakage for a affected steam generator. Activity is released to the environment through the use of the Atmospheric Dump Valves (ADVs) to remove decay heat and to cool the plant to cold shutdown. Once cold shutdown is achieved the release, for cooldown purpose from ADV, to the environment is assumed to be terminated. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.

🡐(EC-40444, R307)

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.

Single failure of a DC bus resulting in the failure of controls to one ADV is not considered. If a steam generator is assumed to have an ADV that is inoperable due to the assumed loss of a DC bus, the activity would be contained in that steam generator before manual action is assumed to operate the ADV. Under these conditions delays in taking manual control of the ADV would postpone the releases due to cooldown to later in the event, when atmospheric dispersion factor values are more favorable and allowing more time for the operator actions to select to preferred control room air intake.

15.2.3.1.3.1.5.1.2 Input Parameters and Initial Conditions

The major assumptions, parameters, and calculational methods used in the design basis analysis are presented in Table 15.2-11. Additional clarification is provided as follows:

a) Reactor coolant activity

No fuel damage is predicted for the feedwater line break event. Two cases of iodine spiking are considered:

1. A Pre-existing Iodine Spike (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. An Accident Generated Iodine Spike (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.

b) Secondary system activity

The activity in the steam generators is conservatively assumed to be equal to 0.1 μCi/gm DEI-131. This activity is the maximum limit allowed as presented in the Technical Specifications.

🡐(DRN 04-704, R14)

➔(DRN 04-704, R14)

c) Iodine chemical form

Iodine releases from the steam generators to the environment are assumed to be 97% elemental iodine, and 3% organic iodine.

d) Primary-to-secondary leakage

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 36 hours, since Waterford 3 can achieve cold shutdown conditions under natural circulation within 36 hours of shutdown (FSAR sub-section 9.3.6.3.3).

e) Removal coefficients

A partitioning factor (PF) of 1.0 was assumed for the duration for the affected steam generator. For the intact steam generator 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.

15.2.3.1.3.1.5.2 Results

➔(EC-5000081470, R301)

The results of the feedwater line break dose consequence analyses are presented in Table 15.2-12. The results demonstrate that the dose consequences meet the acceptance criteria set forth by 10CFR50.67 for the pre-existing iodine spike case and meet the criteria of a small fraction of 10CFR50.67 for the accident generated iodine spike case.

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

➔(DRN 02-1713, R12-B)

15.2.3.1.4 Small FW Pipe Break Event with the Limiting Single Failure and Offsite Power Available

➔(DRN 05-543, R14)

This sub-section is provided to document the analysis of small feedwater system pipe breaks. This licensing basis analysis was originally requested in Supplement No. 1 of the “Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3,” (NUREG-0787, October, 1981); this analysis was in response to confirmatory item 15.3.2.

🡐(DRN 05-543, R14)

15.2.3.1.4.1 Purpose

➔(EC-8458, R307)

The purpose of this analysis is to show that the results of the small feedwater system pipe break event with the limiting single failure and offsite power available produce maximum pressures less than 110 percent of design.

🡐(EC-8458, R307)

15.2.3.1.4.2 Background

→(DRN 05-543, R14)
The large feedwater system pipe break event presented in Subsection 15.2.3.1.3 demonstrates that breaks of all sizes, when combined with the loss of offsite power, produce maximum pressures well below 120 percent of design. The NRC has concluded that the 120 percent of design maximum pressure criterion is appropriate for large feedwater system pipe break events and small feedwater system pipe break events combined with the loss of offsite power, provided that a probability argument is presented to demonstrate that the recurrence frequency of the events are sufficiently low. In addition to providing the probability arguments to close out the confirmatory item in Reference 3 on feedwater system pipe breaks, Waterford 3 demonstrates that the small feedwater system pipe break events with the limiting single failure and offsite power available meet the maximum pressure criterion of 110 percent of design.

In order to demonstrate compliance with this criterion, analysis of small breaks with a modified methodology is required. The methodology used in Subsection 15.2.3.1.3 is applicable to the full spectrum of break sizes. However, it is extremely conservative when applied to the smaller break sizes. As a result, a new method of analysis which is still conservative was developed, and is discussed in the following section.
←(DRN 05-543, R14)

The recurrence frequency for a feedwater system pipe break is sufficiently low to allow the large feedwater system pipe break event and the small feedwater system pipe break event combined with the loss of offsite power to meet the 120 percent of design maximum system pressures. This recurrence frequency is estimated as follows.

The feedwater system pipe break analyzed in Waterford 3 Subsection 15.2.3.1.3 is postulated to occur in the specific length of piping between either steam generator nozzle and the containment wall. Using the methods and data contained in WASH-1400, (Appendix III, Table III 6-9) a summary of pipe rupture rates per plant year for various sized “LOCA sensitive” piping is provided. The median LOCA recurrence frequency for large piping (>6” diameter) is given as 1E-3 per plant year. For Waterford 3 design, the total length of “LOCA sensitive” piping greater than 6” in diameter is approximately 723 feet. The total length of main feedwater system piping “sensitive” with respect to the event analyzed in Subsection 15.2.3.1A is 232 feet. Therefore, using the methodology discussed in WASH-1400, Appendix IIII, Section 6.4, the estimated recurrence frequency for the above postulated main feedwater line break is 5.6E-3 per plant year.

Thus it is shown that the initiating event which is analyzed in Subsection 15.2.3.1.3 is in fact a very low probability event that is highly unlikely to occur in a plant’s lifetime.

Using the WASH-1400 value of 3.16E-2 for the conditional loss of offsite power, the probability of a main feedwater system pipe break with concurrent loss of normal a/c power is less than (3.16E-2 * 5.6E-3) = 1.77E-4 per plant year.

Since the recurrence frequencies presented in the preceding paragraphs apply to pipes greater than 6 inches in diameter, the reanalysis need only consider breaks less than approximately 0.20 ft^2. Therefore, in the following, subsections “small” breaks refer to those which are less than 0.20 ft^2.

15.2.3.1.4.3 Method of Analysis

15.2.3.1.4.3.1 Mathematical Models

→(DRN 05-543, R14)
The NSSS response to the small feedwater system pipe break event with the limiting single failure and offsite power available was modeled using the CENTS computer program described in Chapter 15.0.
←(DRN 02-1713, R12-B; 05-543, R14)

➔(DRN 02-1713, R12-B; EC-8458, R307)
The methods used for Waterford (Appendix 15B of the CESSAR-FSAR) produce a more realistic, but still conservative treatment of heat transfer and downcomer water level behavior in the affected steam generator (i.e., the generator nearest the pipe break) when compared to the original methods. The original methods conservatively assumed that the affected steam generator heat transfer degradation (due to high fluid quality) and low water level trip were delayed until the generator's liquid inventory was completely depleted. However, use of the steam generator model described in Reference 4 indicates that at least 21,000 lbm of liquid remain in the steam generator prior to heat transfer degradation. The steam generator model also indicates that the steam generator low level reactor trip (a downcomer liquid level of approximately 27 feet above the tubesheet) corresponds to a liquid inventory of over 80,000 lbm. However, the reanalysis of the small feedwater system pipe break event conservatively delays the low level trip until heat transfer degradation begins with 21,000 lbm of inventory remaining in the affected steam generator.
🡐(EC-8458, R307)
The method aligns the SG low level trips from both faulted and intact SGs, the high pressurizer pressure trip (HPPT) and the beginning of the heat transfer ramp down (21000 lbm) in the faulted SG by adjusting both the initial pressure and initial SG water masses. This method is less conservative than that of the large FWLB because the heat transfer area ramp down (from 21000 to 2000 (dryout condition) lbm) occurs right after HPPT conditions.

15.2.3.1.4.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response are discussed in Section 15.0. The initial conditions for the principal process variables were varied within the range given in Table 15.0-4 to determine the limiting set of initial conditions.

In addition to conservatively delaying steam generator low level trip coincident with the assumed heat transfer degradation, the initial primary system pressure was adjusted within the range specified in Table 15.0-4 to achieve, where possible, a coincident reactor trip signal on high pressurizer pressure. This maximizes the primary pressurization potential of the small feedwater system pipe break event, by maximizing the primary system pressure at the time of the reactor trip.

➔(DRN 05-543, R14; 05-1551, R14)
The following provides the method for selecting the worst single failure for the small FWLB event. There are no single failures which can adversely impact the consequences (i.e., pressurization) associated with the loss of feedwater inventory event. As a result of the evaluation method applied to the feedwater system pipe break analysis, the only mechanisms for mitigation of the reactor coolant system (RCS) pressurization are the pressurizer safety valves, the reactor coolant flow and the main steam safety valves. The last two influence the RCS-to-steam generator heat transfer rate. There are no credible failures which can degrade pressurizer safety valve or main steam safety valve capacity. Nor are there any credible failures which can reduce steam flow to the affected steam generator. A decrease in RCS-to-steam generator heat transfer due to reactor coolant flow coastdown can only be caused by a failure to fast transfer to offsite power or a loss of offsite power following turbine trip (i.e., two or four pump coastdown, respectively). Because offsite power is assumed to be available for this analysis, the failure to fast transfer is assumed following the turbine trip. This results in the coastdown of two reactor coolant pumps in diagonally opposite loops. If one of the fast transfer buses is blocked, fast bus transfer may result in LOOP which causes 4 RCP pumps to trip and loss of PLCS and PPCS but two-pump coastdown is more limiting for the peak pressure case. Meanwhile, the steam generator heat transfer capability degrades when the SG liquid inventory decreases below 21,000 lbm, and is totally lost when the SG dries out (2,000 lbm SG liquid inventory). The HPPT is followed by the lifting of primary (PSV) and secondary (MSSV) safety valves, which are set at their most adverse lift and blowdown pressures. Primary and secondary cooldown is provided by the heat removal through the PSVs and MSSVs.
🡐(DRN 05-1551, R14)
The transient is continued until the primary and secondary pressures, temperatures, and pressurizer level have turned around and are decreasing.
🡐(DRN 02-1713, R12-B; 05-543, R14)

15.2.3.1.4.4 Results

→(DRN 05-543, R14; EC-8458, R307)
A spectrum of small breaks, of size less than or equal to 0.20 ft^2, were analyzed to determine the limiting break size. The results of this analysis are provided on Figure 15.2-53a.2, which plots maximum primary pressure vs. break size. As can be seen, the limiting break size is the 0.19 ft^2 break. Table 15.2-9a contains the initial conditions and assumptions used in the limiting small break analysis. Table 15.2-9b provides the sequence of events for the limiting small break event. The dynamic behavior of important parameters following a small FWLB is illustrated on Figures 15.2-53a.2 through 15.2-53s.
←(EC-8458, R307; LBDCR 15-039, R309)
An evaluation of the potential for pressurizer fill demonstrates that the operators must act to turn off the charging pumps by 10 minutes following SIAS to avoid filling of the pressurizer, which is equivalent to the 10 minutes allowed prior to the first operator action per NUREG-0800. Since this case is run to determine operator response times, less excess conservatism is used for some of the input assumptions to avoid an unrealistic conclusion.
→(LBDCR 15-039, R309)
Important system parameter are reported in the Sequence of Events, Table 15.2-9b. The maximum RCS and steam generator pressures remains below their acceptance limits of 2750 psia and 1210 psia, respectively.

The analysis demonstrates that the 575 gpm EFW flow is sufficient to prevent dryout of the unaffected steam generator. DNBR remains above 1.70 throughout the transient, and consequently there is no violation of the DNBR SAFDL.
←(DRN 02-1713, R12-B; 05-543, R14)

15.2.3.2 Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System

15.2.3.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of normal feedwater flow with a concurrent single failure of an active component classifies this incident as a limiting fault incident as defined in Reference 1 of Section 15.0. The causes of a loss of normal feedwater flow are discussed in Subsection 15.2.2.5.1. Various active component single failures were considered to determine which failure had the most adverse effect following a loss of normal feedwater flow. The single active failures considered were:

a) A loss of all normal ac power on reactor trip.
→(EC-8458, R307)
b) Failure of the Steam Bypass System (open).
←(EC-8458, R307)
c) Loss of 50 percent of emergency feedwater.
→(DRN 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)
The failure of the Steam Bypass System produces the minimum steam generator inventory in the shortest period of time following a loss of normal feedwater flow. This failure could be caused by an electrical malfunction providing a quick opening signal to all the turbine bypass valves. It is assumed that the failure in the Steam Bypass System causes these valves to remain open, even in the presence of closure signals generated by the system due to adverse steam generator or condenser conditions (e.g.. low pressure and low level) until a main steam isolation signal (MSIS) is generated. 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 evaluation using the revised SCRAM curve times presented in Table15.0-5.
←(DRN 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)
15.2.3.2.2 Sequence of Events and Systems Operation

The systems and reactor trip which operate following a loss of normal feedwater flow with failure of the Steam Bypass System open are the same as those described in Subsection 15.2.2.5.2, except for the operation of the bypass system and the generation of an MSIS. The MSIS is generated due to low steam generator pressure and provides protection against emptying the steam generators.

Table 15.2-10 gives a sequence of events that occur following a loss of normal feedwater flow with the turbine bypass valves open.

→(DRN 00-1731, R11; 05-543, R14)

←(DRN 00-1731, R11; 05-543, R14)

15.2.3.2.3 Core and System Performance

15.2.3.2.3.1 Mathematical Model

The mathematical model used for evaluation of core and system performance is identical to that described in Subsection 15.2.2.5.3.

15.2.3.2.3.2 Input Parameters and Initial Conditions

→(DRN 05-543, R14)
The input parameters and initial conditions used for evaluation of core and systems performance are identical to those described in Subsection 15.2.2.5.3. and presented in FSAR Table 15.2-13. The negative moderator coefficient insures a large power increase during the cooldown caused by the turbine bypass valves failing open. The radial peak and axial shape for this case were chosen such that a DNBR trip condition would not occur before the low steam generator water level trip. This was done to allow the heat flux to increase to the maximum possible value before trip and to insure a transient which results in the minimum steam generator inventory.

In addition to the assumptions identified in Section 15.2.2.5.3.2, the following additional assumptions were employed:

- Main steam isolation signal (MSIS) is actuated on a low SG pressure setpoint of 576 psia. The main steam isolation valves (MSIVs) receive an MSIS signal to close. A response time of 8 seconds (which includes a 1-second MSIS response time) was assumed for the MSIVs.

→(EC-8458, R307)
- Safety injection actuation signal (SIAS) setpoint of 1560 psia was used. HPSI flow was initiated and maintained for a short time until the RCS pressure rose above the HPSI shutoff head.

←(EC-8458, R307)

15.2.3.2.3.3 Results

The dynamic behavior of important parameters following a loss of normal feedwater flow with failure of the Steam Bypass System (open) are presented on Figures 15.2-54 through 15.2-65.

→(DRN 00-592, R11-A; 02-91, R11-A; EC-8458, R307; LBDCR 15-039, R309))
The complete loss of normal feedwater flow results in an increase in the steam generator pressure and temperature. When the pressure exceeds 908 psia, 13.5 seconds after cessation of feedwater flow, the turbine bypass valves open. This results in an increased main steam flow incident concurrent with a loss of feedwater. As the RCS begins to cool down due to the increased steam flow, the negative moderator coefficient causes the reactor power to increase. The steam generator inventory decreases rapidly. The primary coolant temperature and pressure are decreasing rapidly when the reactor is tripped at 42.7 seconds on low steam generator water level. A safety injection actuation signal (SIAS) is initiated at 77.8 seconds on low pressurizer pressure. The pressurizer empties at 86.6 seconds with a pressurizer pressure of 1462 psia. A low steam generator pressure signal is generated at 91.8 seconds. The main steam isolation valves close at 99.0 seconds. At 119.0 seconds the total steam generator water inventory in the affected generator reaches its minimum value of 9113 lbm. The steam generator inventory increases as the Emergency Feedwater System continues to operate. The pressure in the steam
←(DRN 00-592, R11-A; 02-91, R11-A; 05-543, R14; EC-8458, R307; LBDCR 15-039, R309)

→(DRN 00-592, R11-A; 02-91, R11-A; 05-543, R14)
generators begins to increase after the isolation of the steam generators. It is conservatively assumed that no operator action is taken until 30 minutes after the event. At this time, the operator will use the atmospheric dump valves and begin cooldown of the plant. The RCS is then cooled to an average temperatures of 350°F, at which point shutdown cooling is initiated.
←(DRN 00-592, R11-A; 02-91, R11-A; 05-543, R14)

15.2.3.2.4 Barrier Performance

15.2.3.2.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.2.2.5.4.

15.2.3.2.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.2.2.5.4.

15.2.3.2.4.3 Results

→(DRN 05-543, R14)
There are no releases to atmosphere until the operator begins cooldown of the plant 30 minutes after the loss of feedwater flow.
←(DRN 05-543, R14)

15.2.3.2.5 Radiological Consequences

→(DRN 04-704, R14)
The radiological consequences of this event are less severe than the consequences of the Increased Main Steam Flow with LOOP, discussed in Subsection 15.1.2.3.5.
←(DRN 04-704, R14)

SECTION 15.2: REFERENCES

➔(EC-8458, R307)
1. "The Two-Phase Critical Flow of One Component Mixture in Nozzles, Orifices and Short Tubes", Henry, R.E. and Fauske, H.K., Journal of Heat Transfer, May 1971.
🡐(EC-8458, R307)

➔(DRN 05-543, R14)
2. Deleted.
🡐(DRN 05-543, R14)

➔(DRN 02-1713, R12-B)
3. “Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3” (Supplement 1, Subsection 15.3.2) NUREG-0787, USNRC, October 1981.

4. “Reanalysis of Small Break Loss of Feedwater Inventory Events with the Limiting Single Failure and Offsite Power Available”, CESSAR FSAR Appendix 15B (Section 15B.6).
🡐(DRN 02-1713, R12-B)

TABLE 15.2-1 Revision 309 (06/16)

SEQUENCE OF EVENTS FOR THE LOSS OF CONDENSER VACUUM

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

Time (seconds)	Event	Setpoint or Value
0.0	Closure of turbine stop valves on turbine due to loss of condenser vacuum	---
5.5	High Pressurizer Pressure Trip Condition, psia	2422
7.0	CEA's begin to drop into core	---
7.0	Pressurizer safety valves begin to open, psia Maximum pressurizer pressure, psia	2575 2576
7.9	Maximum RCS pressure, psia	2712
11.7	Pressurizer safety valves closed, psia	2525
12.0	Maximum pressurizer liquid volume	1202
16.8	Steam generator safety valves begin opening, psia	1117
22.1	Maximum steam generator pressure, psia	1135
1800.	Operator opens atmospheric steam dump valves to begin plant cooldown to shutdown cooling.	---
---	Shutdown cooling initiated	---

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

TABLE 15.2-2 Revision 307 (07/13)

➔(DRN 02-526, R12; 05-543, R14; EC-8458, R307)

ASSUMPTIONS FOR LOSS OF CONDENSER VACUUM ANALYSIS

Parameter	RSG Assumption
Initial Core Power Level, MWt *	3735
Core Inlet Coolant Temperature, °F	533
Reactor Coolant System Flow, 10^6lbm/hr	170.2
Reactor Coolant System Pressure, psia	2090
Pressurizer Level, %	67.5
Steam Generator Level, % NR	49.0
Steam Generator Pressure	732.0
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho$/°F	-0.2
Doppler Coefficient Multiplier	0.85
CEA Worth on Trip, percent $\Delta\rho$	-6.0
Steam Bypass Control System	Inoperative
Pressurizer Level Control System **	Inoperative
Pressurizer Pressure Control System **	Inoperative

🡐(EC-8458, R307)

➔(DRN 02-526, R12)

* Rated thermal power, plus power measurement uncertainty

🡐(DRN 02-526, R12)

** PLCS and PPCS are not assumed to operate for peak RCS Pressure case. PLCS and PPCS are assumed in automatic for the peak secondary pressure case.

🡐(DRN 05-543, R14)

TABLE 15.2-3 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.2-4 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.2-4a Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.2-5 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.2-6 Revision 309 (06/16)

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

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER FLOW FOR RSGS

Time (second)	Event	Setpoint or Value
0.0	Termination of all feedwater flow	---
8.12	Maximum core power (% of 3716 MWt)	101.0%
43.23	Low steam generator water level trip signal (ESFAS occurs)	5% NR
44.13	Trip breakers open	---
44.74	CEAs begin to drop into core	---
47.50	Maximum RCS pressure	2272 psia
49.0	Steam generator safety valves open	1057 psia
51.99	Maximum steam generator pressure	1072 psia
60.0	Steam generator safety valves close	1003 psia
93.2	Emergency feedwater (3 pumps) reaches steam generators	---
190.1	Minimum steam generator water inventory	42,830 lbm
226.09	Minimum RCS pressure	1594 psia
1800	Operator begins cooldown	---

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

TABLE 15.2-7 Revision 307 (07/13)

➔(DRN 02-526, R12; 05-543, R14; EC-8458, R307)

ASSUMPTIONS FOR THE LOSS OF NORMAL FEEDWATER ANALYSIS

Parameter	Assumption
Initial Core Power, MWt	3735
Core Inlet Coolant Temperature, °F	552
Reactor Coolant System Flow Rate, lbm/hr	$148 * 10^6$
Reactor Coolant System Pressure, psia	2090
Pressurizer Level, %	67.5
Steam Generator Pressure, psia	908
SG Level, % NR – 40.2 ft	79.3
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho$/°F	-4.2
Doppler Coefficient Multiplier	1.30 (EOC)
CEA Worth, % $\Delta\rho$	-6.0
Steam Bypass Control System	Automatic

🡐(DRN 02-526, R12; 05-543, R14; EC-8458, R307)

TABLE 15.2-8 Revision 309 (06/16)

➔(DRN 02-1713, R12-B; 05-543, R14; EC-8458, R307; LBDCR 15-039, R309)

SEQUENCE OF EVENTS FOR THE LARGE FEEDWATER SYSTEM PIPE BREAK

Time second	Event	Setpoint or Value
0	Break of the main feedwater line	0.12 ft^2
6.8	Maximum core power (% of 3716 MWt)	100.7%
24.0	Low steam generator trip condition (Low SG liquid mass)	9000 lbm
24.0	SIAS generated (2 charging pumps on, letdown isolated) EFAS on Low SG liquid mass	9000 lbm
24.5	High Pressurizer trip condition	2422 psia
24.9	Trip breakers open	---
24.9	Turbine trip	---
24.9	LOOP	---
24.91	Turbine admission valves closed	---
25.5	CEAs begin to drop	---
26.8	PSVs open	2575 psia
26.8	SG connected to the ruptured feed line empties	2000 lbm
27.8	Maximum pressurizer surge line flow	1808 lbm/sec
27.9	Maximum RCS pressure	2746 psia
29.2	Maximum pressurizer pressure	2642 psia
35.5	SG safety valves open	1117.2 psia
36.6	Maximum steam generator pressure	1123.1 psia
84.0	EFW flow initiated	EFW Activation + 60 sec
1800	Operator takes control of plant	---

🡐(DRN 02-1713, R12-B; 05-543, R14; EC-8458, R307; LBDCR 15-039, R309)

TABLE 15.2-9 Revision 307 (07/13)

➔(DRN 02-526, R12; 02-1713, R12-B; DRN 05-543, R14; EC-8458, R307)

ASSUMPTIONS FOR THE LARGE FEEDWATER SYSTEM PIPE BREAK

Parameter	Assumption
Initial core power level, MWt	3735
Core inlet temperature, °F	552
Core Mass Flow rate, 10^6 lbm/hr	148
Reactor coolant system pressure, psia	2310
Steam generator pressure, psia	872
Moderator temperature coefficient, $10^{-4}\Delta\rho/°F$	-0.2
Doppler coefficient multiplier	0.85
CEA worth for trip, $10^{-2}\Delta\rho$	-6
Steam Bypass Control System	Inoperative
Pressurizer Pressure Control System	Inoperative
Pressurizer Level Control System	Automatic
Feedwater line break area, ft^2	0.12
Initial intact steam generator liquid inventory, lbm	97,670
Emergency feedwater flow, gpm	575

🡐(DRN 02-526, R12; 02-1713, R12-B; 05-543, R14; EC-8458, R307)

➔(DRN 02-1713, R12-B) TABLE 15.2-9a Revision 307 (07/13)

➔(DRN 05-543, R14; EC-8458, R307)

ASSUMPTIONS FOR THE LIMITING SMALL FEEDWATER SYSTEM PIPE BREAK

Parameter	Assumption
Initial core power level, MWt	3735
Core inlet temperature, °F	552
Core Mass Flow rate, 10^6 lbm/hr	148
Reactor coolant system pressure, psia	2310
Steam generator pressure, psia	866
Moderator temperature coefficient, 10^{-4} Δρ/°F	-0.2
Doppler coefficient multiplier	0.85
CEA worth for trip, 10^{-2}Δρ	-6
Steam Bypass Control System	Inoperative
Pressurizer Pressure Control System	Inoperative
Pressurizer Level Control System	Automatic
Feedwater line break area, ft^2	0.19
Initial intact steam generator liquid inventory, lbm	229,769
Emergency feedwater flow, gpm	575

🡐(DRN 02-1713, R12-B; 05-543, R14; EC-8458, R307)

TABLE 15.2-9b Revision 309 (06/16)

→(DRN 02-1713, R12-B; 05-543, R14; EC-8458, R307; LBDCR 15-039, R309)

SEQUENCE OF EVENTS FOR THE LIMITING SMALL FEEDWATER SYSTEM PIPE BREAK

Time (Sec)	Event	Setpoint or Value
0	Break of the main feedwater line	0.19 ft^2
0.1	Initial break flow	2109 lbm/sec
14.7	Maximum core power (% of 3716 MWt)	100.73%
47.1	High Pressurizer trip condition	2422 psia
47.1	SIAS generated (2 charging pumps on, letdown isolated)	---
47.2	Low steam generator trip condition (Low SG liquid mass) EFAS on Low SG liquid mass	21000 lbm
48.0	Trip breakers open	---
48.0	Turbine trip	---
48.0	FFBT (RCPs 1 and 3 coastdown)	---
48.01	Turbine admission valves closed	---
48.6	CEAs begin to drop	---
50.5	PSVs open	2575 psia
50.6	Maximum pressurizer pressure	2577.7 psia
51.1	Maximum RCS pressure	2656.5 psia
51.6	Maximum pressurizer surge line flow	1170 lbm/sec
53.4	Steam generator connected to the ruptured feed line empties	2000 lbm
60.0	SG safety valves open	1117.2 psia
60.3	Maximum steam generator pressure	1120.5 psia
97.2	EFW flow initiated	EFW Activation + 50 sec
1800	Operator takes control of plant	---

←(DRN 02-1713, R12-B; 05-543, R14; EC-8458, R307; LBDCR 15-039, R309))

TABLE 15.2-10 Revision 309 (06/16)

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

SEQUENCE OF EVENTS FOR THE LOSS OF NORMAL FEEDWATER FLOW WITH AN ACTIVE FAILURE IN THE STEAM BYPASS CONTROL SYSTEM

Time (seconds)	Event	Setpoint or Value
0.0	Termination of all feedwater flow	---
13.5	Turbine steam bypass valves fully open – ADVs	908 psia
42.69	Low steam generator water level trip signal *	5% NR
43.59	Reactor trip breakers open	---
44.19	Maximum core power	129.4%
44.2	CEAs begin to drop into core	---
77.78	Low pressurizer pressure safety injection actuation signal (SIAS flow is on for a short time until the transient RCS pressure returns above HPSI shutoff head)	1560 psia
86.6	Pressurizer empties	1462 psia
91.77	Main steam isolation signal	576 psia
98.76	Minimum steam generator pressure	522 psia
99.0	Main steam isolation valves fully closed	---
112.5	Emergency feedwater reaches steam generators	---
119.0	Minimum steam generator water inventory	9,113 lbm
1800	Operator begins cooldown	---

* NOTE CPC VOPT trip is available. However, for conservatism, Low SG water level trip is credited in this analysis.

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

TABLE 15.2-11 (Sheet 1 of 2) Revision 307 (07/13)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FEEDWATER LINE BREAK

Core Power Level:	3735 MWt
→(EC-5000081470, R301) ←(EC-5000081470, R301)	
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
Iodine Filter Efficiency	99% (elemental/particulate/organic)
→(EC-5000081470, R301) Pressurization Flow: ←(EC-5000081470, R301)	225 CFM (maximum)
Unfiltered Inleakage:	100 CFM
Breathing Rate:	3.47E-04 m^3/s
Control Room Occupancy Factors:	
0 – 24 hours	1.00
24 – 96 hours	0.60
96 hours – 30 days	0.40

←(DRN 04-704, R14)

TABLE 15.2-11 (Sheet 2 of 2) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FEEDWATER LINE BREAK

→(DRN 05-1551, R14)

Main Control Room (MCR) χ/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 *

←(EC-5000081470, R301)

* factor of 4 reduction credited per SRP 6.4.

→(EC-5000081470, R301)

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	2.04-E03 *

* factor of 4 reduction credited per SRP 6.4.

←(EC-5000081470, R301)

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

Initial Activity Release: 3.97 Ci [1 1/3 SG]

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 15.2-12 Revision 14 (12/05)

→(DRN 04-704, R14)

RADIOLOGICAL CONSEQUENCES OF A
DESIGN BASIS FEEDWATER LINE BREAK

	TEDE Dose	Acceptance Criteria
Pre-existing Iodine Spike:		
EAB (worst two hour dose)	≤25	25 Rem TEDE
LPZ (duration)	≤25	25 Rem TEDE
Main Control Room	≤5	5 Rem TEDE
Accident Generated Iodine Spike:		
EAB (worst two hour dose)	≤2.5	2.5 Rem TEDE
LPZ (duration)	≤2.5	2.5 Rem TEDE
Main Control Room	≤5	5 Rem TEDE

←(DRN 04-704, R14)

TABLE 15.2-13 Revision 307 (07/13)

➔(DRN 05-543, R14; EC-8458, R307)

ASSUMPTIONS FOR THE LOSS OF NORMAL FEEDWATER FLOW WITH AN ACTIVE FAILURE IN THE STEAM BYPASS CONTROL SYSTEM

Parameter	**Assumption**
Initial Core Power, MWt	3735
Core Inlet Coolant Temperature, °F	552
Reactor Coolant System Flow Rate, lbm/hr	$148 * 10^6$
Reactor Coolant System Pressure, psia	2090
Pressurizer Level, %	67.5
Steam Generator Pressure, psia	908
SG Level, %NR – 40.2 ft	79.3
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho$/°F	-4.2
Doppler Coefficient Multiplier	1.30
CEA Worth, % $\Delta\rho$	-6.0
Steam Bypass Control System	Automatic

🡐(DRN 05-543, R14; EC-8458, R307)