15.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY

15.6.1 MODERATE FREQUENCY INCIDENTS

There are no moderate frequency incidents resulting from a decrease in reactor coolant inventory.

15.6.2 INFREQUENT INCIDENTS

There are no infrequent incidents resulting from a decrease in reactor coolant inventory.

15.6.3 LIMITING FAULTS

15.6.3.1 Primary Sample or Instrument Line Break

15.6.3.1.1 Identification of Causes and Frequency Classification

→(DRN 02-1742, R12-B; 03-220, R12-B; 06-1062, R15; EC-8458, R307)
The estimated frequency of a primary sample or instrument line rupture classifies it as a limiting fault incident as defined in Reference 1 of Section 15.0. A primary sample or instrument line break provides a release path for reactor coolant outside containment. The line break selected for analysis is the letdown line (two inch Schedule 160 Pipe) which penetrates the containment. This is the largest penetration whose failure could result in an event in this category. The break size was investigated to determine the maximum RCS mass release outside containment. The charging lines which penetrate the containment are provided with check valves to prevent blowdown of reactor coolant resulting from a break outside the containment. 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 02-1742, R12-B; 03-220, R12-B; 06-1062, R15; EC-8458, R307)

15.6.3.1.2 Sequence of Events and Systems Operation

→(DRN 03-220, R12-B; 05-543, R14)
The integrity of lines containing primary coolant external to the containment is significant radiologically since a rupture of this barrier results in the release of reactor coolant outside containment. Following such a break, the RCS pressure decreases due to the loss of reactor coolant. When the RCS pressure has decreased sufficiently, a CPCS trip on Low DNBR will occur. To conservatively maximize the amount of primary mass released outside of containment, this trip is not explicitly credited in the analysis. When pressurizer pressure decreases such that it is outside the allowed range of the CPCS, a CPCS trip on Out-of-Range Low Pressurizer Pressure is initiated. The safety injection actuation signal (SIAS) on low pressurizer pressure terminates the break flow by isolating the letdown line inside containment, and the reactor coolant inventory is replenished by the safety injection pumps and by the charging pumps.
←(DRN 03-220, R12-B; 05-543, R14)

Operation of the HPSI pumps as well as the charging pumps ensures that the core will not uncover and prevents any significant increase in clad temperatures.

Table 15.6-1 shows the sequence of events following a letdown line break.

15.6.3.1.3 Core and System Performance

15.6.3.1.3.1 Mathematical Model

→(DRN 03-220, R12-B; 05-543, R14)
The NSSS response to a letdown line break was simulated using the CENTS III computer program described in Section 15.0. The CETOP code is used to calculate the minimum DNBR.
←(DRN 03-220, R12-B; 05-543, R14)

15.6.3.1.3.2 Input Parameters and Initial Conditions

The initial conditions and input parameters of the NSSS assumed in the analysis are listed in Table 15.6-2.

➜(DRN 05-543, R14)
Parametric analyses were performed to determine whether minimum or maximum values of important input parameters where limiting. A decrease in core inlet temperature results in a lower core outlet temperature and steam generator (SG) pressure values. For a given break area, a lower core inlet temperature results in a slightly larger primary mass release. This has a very small effect as the event is performed parametric in break size.

An increase in RCS flow rates results in lower core outlet temperature and steam generator pressure values. For a given break area, a higher initial RCS flow rate results in a slight increase in primary mass release for a given break size. This has a very small effect as the event is performed parametric in break size.

Initial conditions and input assumptions for this analysis are given in Table 15.6.-2. Additional assumptions are summarized below:

- The most negative MTC of $-4.2 * 10^{-4}$ $\Delta\rho$/°F was assumed.
- A parametric analysis on break size was performed to generate a reactor trip at the time of operator action that is conservatively assumed to occur at 1800 seconds.
- A CPCS hot leg saturation trip and out-of-range (low pressurizer pressure) trip were conservatively assumed.
- An initial core power of 3735 MWt was assumed, based on a rated power of 3716 MWt and a 0.5% uncertainty.
- An initial RCS flow of $178.9 * 10^6$ lbm/hr was assumed.
- An initial minimum core inlet temperature of 533°F was assumed.
- An initial pressurizer pressure of 2312 psia was assumed.
- It was determined from parametric studies that the limiting break size for mass releases is the largest size that allows the event to go to the assumed 30 minute operator action time without having generated a reactor trip. A double-ended break is not limiting due to the earlier trip and break isolation time.

15.6.3.1.3.3 Results

➜(DRN 03-220, R12-B; EC-13881, R304)
The response of the NSSS following a letdown line break begins with a decrease in pressurizer level and pressure. The transient response of important system parameters is shown in Figures 15.6-1a through 15.6-1i. At approximately 1800 seconds after the break, the RCS pressure has decreased such that the CPCS Out-of-Range Trip on low pressurizer pressure occurs. The turbine trip on reactor trip results in an increase in secondary side pressure to the steam generator safety valve set pressure. A SIAS is subsequently generated on low pressurizer pressure. Operator action to isolate the RCS and initiate plant cooldown is assumed at 1800 seconds.
🡐(DRN 03-220, R12-B; 05-543, R14; EC-13881, R304)

The reactor coolant inventory is replenished by the HPSI pumps and by the charging pumps. Operation of these pumps ensures that the core will not uncover and prevents any significant increase in clad temperature.

After 30 minutes, the operator is assumed to start a plant cooldown.

15.6.3.1.4 Barrier Performance

15.6.3.1.4.1 Mathematical Model

The mathematical model used for evaluation of Barrier Performance is described in Subsection 15.6.3.1.3.

15.6.3.1.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of Barrier Performance are the same as those described in Subsection 15.6.3.1.3.

15.6.3.1.4.3 Results

→(DRN 03-220, R12-B; 05-543, R14)
At about 1800 seconds into the transient, the ruptured line is isolated, terminating the leak flow. Prior to isolation of the line, less than 78,000 pounds of primary coolant have been released from the RCS.
←(DRN 03-220, R12-B; 05-543, R14)

15.6.3.1.5 Radiological Consequences

→(EC-5000081470, R301)
15.6.3.1.5.1 Design Basis, Pre-Existing Iodine Spike
←(EC-5000081470, R301)

15.6.3.1.5.1.1 Physical Model

A break in fluid-bearing lines which penetrate the containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS which penetrate the containment. There are, however, other piping lines from the RCS to the Chemical and Volume Control System (CVCS) and the Process Sampling System which penetrate the containment. The most severe rupture with respect to radioactivity release during normal plant operation is the rupture of the letdown line outside containment. For such a break, the reactor coolant letdown flow would have passed from the cold leg and through the regenerative heat exchanger.

→(DRN 03-220, R12-B; 04-704, R14)
It was assumed that about 1800 seconds would elapse before an SIAS is initiated on low pressurizer pressure and the letdown line isolation valves are shut. The reactor coolant mass released to the Reactor Auxiliary Building (RAB) is less than 78,000 lbm.
←(DRN 03-220, R12-B)

15.6.3.1.5.1.2 Assumptions and Parameters
←(DRN 04-704, R14)
The major assumptions and parameters used in the analysis are listed in Table 15.6-2 and 15.6-3 are discussed below:

→(DRN 03-220, R12-B; EC-5000081470, R301)
a) 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.
←(EC-5000081470, R301)
→(DRN 04-704, R14)
b) Less than 78,000 pounds of reactor coolant are spilled.
←(DRN 03-220, R12-B; 04-704, R14)

c) All the noble gases in spilled reactor coolant are released to the atmosphere.

→(DRN 03-220, R12-B; 04-704, R14)

d) The fraction of water flashing to steam was calculated and was defined as the fraction of iodines in the water that volatilize. The reactor coolant temperature was assumed to be 533°F. The fraction of iodines calculated to volatilize was less than 36%.

←(DRN 03-220, R12-B; 04-704, R14)

e) No credit is taken for mixing or holdup of the activity released to the RAB atmosphere.

→(DRN 00-592, R11-A)

f) The activity released from the ruptured letdown line is assumed to be released directly to the environment during the two-hour period immediately following the accident.

←(DRN 00-592, R11-A)

g) No credit is taken for ground deposition or decay in transit to the exclusion area boundary or outer boundary of the low population zone (LPZ).

→(DRN 04-704, R14)

h) The small contribution to dose consequences from alkali metal is ignored.

→(EC-5000081470, R301)

i) Noble Gas RCS isotopic concentrations are per FSAR Table 11.1-2.

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

15.6.3.1.5.1.3 Mathematical Model

Models used in the analysis are described in the following sections:

a) The meteorological conditions assumed present during the course of the accident are based on X/Q values which are expected to be conservative 95 percent of the time.

Calculational methods for X/Qs are presented in Subsection 2.3.4. For the design basis accident, five percent level X/Qs were used (Table 2.3-136).

→(DRN 04-704, R14)

b) The potential dose to an individual exposed at the exclusion area boundary or LPZ outer boundary are obtained using the models given in Appendix 15B.

←(DRN 04-704, R14)

15.6.3.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the Reactor Auxiliary Building (RAB) is collected in the floor drain sumps. From there, it is pumped to the radwaste treatment system. Thereafter, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

15.6.3.1.5.1.5 Uncertainties and Conservatisms in the Evaluation of the Results

The principal uncertainties and conservatisms in the calculation of the resultant doses following a letdown line rupture arise from the unknown extent or reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of radionuclides that volatilize, the fraction of the spilled activity that escapes the RAB, and the meteorological conditions at the time of the accident. Each of these uncertainties is treated by taking worst-case or conservative assumptions.

→(DRN 03-220, R12-B)

a) Reactor coolant equilibrium activities are based on the Technical Specification limit, which is greater than that normally observed in past PWR operation.

b) The quantity of coolant spilled is maximized by determining the break size that produces the largest mass release.

→(DRN 04-704, R14)

c) The fraction of iodines calculated to volatilize is based on a reactor coolant temperature of 533°F. This temperature does not take credit for cooling provided by the regenerative heat exchanger. The resulting fraction of iodines released (36 percent) would decrease as the coolant temperature decreased.

←(DRN 03-220, R12-B; 04-704, R14)

→(EC-5000081470, R301)

d) No credit is taken for the effects of retention of radioactivity (plate out) which could occur within the RAB, reducing the amount of activity released to the environment.

←(EC-5000081470, R301)

e) The meteorological conditions assumed during the course of the accident are based on X/Q values which are expected to be conservative 95 percent of the time. This condition results in the poorest values or atmospheric dispersion calculated for the exclusion area boundary or LPZ outer boundary. Further, no credit is taken for the transit time of activity from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.3.1.5.1.6 Results

Offsite Doses

→(DRN 04-704, R14)

The radiological consequences resulting from a letdown line rupture have been conservatively calculated using assumptions and models described above. The Total Effective Dose Equivalent (TEDE) was calculated for the worst 2 hour period at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-4. These results meet the criteria set forth in 10CFR50.67.

Control Room Doses

TEDE doses were determined for control room personnel for the duration of the accident. The dose consequences meet the criteria set forth in 10CFR50.67 and GDC 19.

←(DRN 04-704, R14)

15.6.3.1.5.2 Design Basis, Iodine Spike Caused by the Accident

→(EC-5000081470, R301)

In this evaluation, the radiological consequences of the letdown line rupture were evaluated assuming that the accident causes an iodine spike. The mathematical models, assumptions and parameters used in this analysis are identical with the design basis evaluation without an iodine spike discussed in Subsection 15.6.3.1.5.1 with the following exception.

←(EC-5000081470, R301)

→(DRN 03-220, R12-B)

At the initiation of the letdown line break, the I-131 equivalent source term is assumed to increase as shown in Figure 15.1.-75a. This figure is based on a factor of 500 increase in the iodine release rate.

←(DRN 03-220, R12-B)

➔(DRN 04-704, R14; EC-5000081470, R301)
The activity released is presented in Table 15.6-3. Radiological consequences are presented in Table 15.6-4 and are within a small fraction of the guidelines of 10CFR50.67 for EAB and LPZ dose, and meet GDC19 for Control Room dose.
🡐(DRN 04-704, R14; EC-5000081470, R301)

➔(DRN 03-220, R12-B)
🡐(DRN 03-220, R12-B)

15.6.3.2 Steam Generator Tube Rupture

➔(DRN 05-1201, R14)
15.6.3.2.1 Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

15.6.3.2.1.1 Identification of Causes and Frequency of Classification

➔(DRN 05-543, R14)
The estimated frequency of a steam generator tube rupture (SGTR) with concurrent loss of offsite power (LOOP) classifies it as a limiting fault incident as defined in Reference 1 of Section 15.0. Integrity of the barrier between the RCS and Main Steam System is radiologically significant, since a leaking steam generator tube allows transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with shell side water in the affected steam generator. A steam generator tube rupture is a penetration of this barrier. As a result of the LOOP, electrical power is unavailable for the reactor coolant pumps and the main circulating water pumps. In such circumstances, the plant experiences simultaneous losses of load, normal feedwater flow, forced reactor coolant flow, condenser vacuum, and steam generator blowdown.
🡐(DRN 05-543, R14)
The analysis of the radiological consequences of the SGTR to account for the impact of potential uncovery of the rupture during the event. Since greater radiological releases result if early operator actions are assumed, the first operator action was postulated to occur at 7 minutes after reactor trip. This is consistent with ANSI/ANS-58.8-1984, 'American National Standard Response Design Criteria for Nuclear Safety Related Operator Actions.'

The order of operator actions assumed for this analysis are based upon Waterford 3 procedures.
🡐(DRN 05-1201, R14)
Note that the Waterford 3 procedures and ANSI/ANS-58.8-1984 were used to derive a conservative sequence and timing for the assumed operator actions. It is not necessary or intended to maintain an exact correspondence between procedures and the assumptions of this SGTR Analysis. Indeed, to do so would risk increasing the radiological releases during a real event due to the conservative assumptions in this analysis.

➔(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)

➔(DRN 05-1201, R14)
15.6.3.2.1.2 Sequence of Events and Systems Operation

🡐(DRN 05-1201, R14)
Table 15.6-24 lists the sequence of events which occur from the time of the double-ended rupture of a Steam Generator U-tube until reaching Shutdown Cooling entry conditions.

➔(DRN 05-1201, R14)
Following a tube rupture, the RCS pressure decreases. The drop in RCS pressure results in startup of all charging pumps and in a reactor trip due to the CPC hot leg saturation trip. Steam generator safety valves open to control secondary pressure.

Behavior of the systems varies depending upon the size of the rupture. For leak rates up to the capacity of the charging pumps, reactor coolant inventory is maintained and an automatic reactor trip does not occur.
🡐(DRN 05-1201, R14)

Since the original Waterford 3 SGTR analyses were performed, it has been recognized that releases due to an SGTR are significantly influenced by operator actions. Thus, in recent years the SGTR dose methodology has undergone significant revision so that the analyses follow the plant specific procedures for responding to an SGTR.

The operator actions assumed in this analysis are consistent with Waterford 3 emergency procedures which apply to SGTR's. Major operator actions assumed in the analysis are summarized below. The timing of operator actions is based upon ANSI/ANS-58.8-1984, 'American National Standard Response Design Criteria for Nuclear Safety Related Operator Actions.' This document specifies time response criteria for safety related operator actions at nuclear plants. Based on these guidelines, the first intervention by the operator is assumed at seven minutes after reactor trip. Subsequently, a time delay of two minutes between discrete operator actions is assumed.

→(DRN 05-1201, R14; EC-13881, R304)

(1) An automatic Emergency Feedwater Actuation Signal (EFAS) is generated if the level in the SG falls below 27.4% NR (71% WR). However, Emergency Feedwater (EFW) flow to the SG will not commence until the level falls below the Critical Level (see Figure 7.3-12). Since the level had not fallen below the Critical Level at seven minutes after reactor trip, EFW flow had not yet been initiated. Thus, it was assumed that the operator would take manual control of the EFW system and initiate EFW flow subsequent to that time.

←(EC-13881, R304)

(2) At seven minutes after the trip, the operator opens the ADV's of both steam generators to cooldown the RCS at a rate of approximately 100°F/hr. Procedures call for a rapid cooldown prior to isolating the affected SG. It is assumed that the affected SG may be isolated only after the hot leg is cooled to 500°F or less. This conservatively allows releases from the affected Steam Generator to continue for a longer time.

(3) SG level is maintained above the tubes in the unaffected generator. EFW flow to the affected SG is terminated at the time of isolation.

←(DRN 05-1201, R14)

(4) The operator isolates the affected steam generator when the hot leg temperature is 500°F or less. The initial cooldown of the RCS is aimed at preventing re-opening of the MSSV's on the affected steam generator.

→(EC-34230, R306)

(5) The operator initiates auxiliary spray flow in order to depressurize the RCS to 1000 psia two minutes after the isolation of the affected steam generator. The operator uses the HPSI system, pressurizer backup heaters, and auxiliary sprays as necessary to control RCS inventory and subcooling. Note that the pressurizer backup heaters are not used / activated in the current SGTR analysis of record.

←(EC-34230, R306)

→(DRN 05-1201, R14)

(6) After isolating the affected steam generator, the operator cools the RCS at 50°F/hr for the remainder of the first two hours using the unaffected steam generator. This cooldown rate is the maximum allowed per procedure under natural circulation conditions and helps to maximize the releases which contribute to the 2 hour dose.

The RCS is brought to shutdown cooling entry conditions at 8 hours. By delaying entry into Shutdown Cooling, this reduction in cooldown rate increases the radiological release during the long term cooldown. This maximizes the primary heat to be removed through the ADV's within the 0-8 hour time period.

←(DRN 05-1201, R14)

(7) The operator maintains a subcooling margin of greater than 28°F in the affected loop during the cooldown. Maintaining a subcooling of 28°F in the isolated loop means that the RCS pressure will be approximately 200 psia higher than the isolated SG. This leads to continued primary to secondary leakage, which fills the affected SG. This increases the calculated dose since the ADV of the affected SG will be opened to maintain water level in the affected SG at or below 94% WR. This is consistent with procedures.

→(DRN 05-1201, R14)

15.6.3.2.1.3 Core and System Performance

←(DRN 05-1201, R14)

A. Mathematical Model

→(DRN 05-1201, R14)

The thermal hydraulic response of the Nuclear Steam Supply System (NSSS) to the SGTR with a loss of offsite power was simulated using the CENTS computer program. Operator actions to mitigate the effects of the SGTR event and bring the plant to shutdown cooling entry conditions were simulated using CENTS. The CENTS code is described in Section 15.0.

←(DRN 05-1201, R14)

B. Input Parameters and Initial Conditions

The initial conditions and input parameters used in the analyses of the system response to a SGTR with concurrent LOOP are listed in Table 15.6-25. Additional discussion on the input parameters and the initial conditions are provided in Section 15.0. Conditions were chosen to maximize the radiological releases.

→(DRN 05-1201, R14)

Additional input assumptions include:

- The BOC Doppler curve was assumed.

→(EC-13881, R304)

- A BOC delayed neutron fraction and neutron lifetime.

←(EC-13881, R304)

- 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 HFP was used.
- The maximum HFP core inlet temperature of 552°F was assumed.
- A minimum RCS flow of 1.48×10^{8} lbm/hr was assumed.

←(DRN 05-1201, R14)

→(DRN 00-592; 05-543, R14)

The input parameters and initial conditions were chosen to obtain an early trip of the reactor. Prior to reactor trip, radioactivity released through the tube rupture is transported through the turbine to the condenser where radiation is released via the condenser air ejector system. The amount of radiation released is very small because of the high decontamination factors for the condenser. After reactor trip, assuming that the condenser becomes unavailable, the steam generators release steam directly to the atmosphere.

←(DRN 00-592; 05-543, R14)

→(EC-13881, R304)

The reactor trip signal is generated on margin to hot leg saturation. A combination of initial conditions is chosen to force an early reactor trip signal due to exceeding the CPC hot leg saturation temperature limit. These conditions include the minimum allowed RCS pressure, maximum core power, minimum core coolant flow, and maximum core coolant inlet temperature. Also, the total amount of energy to be removed from the system before reaching shutdown cooling system entry conditions is higher. These factors lead to increased steam releases to the atmosphere, hence to higher offsite doses.

←(EC-13881, R304)

→(EC-13881, R304)
The automatic mode of operation for the ADV's is conservatively assumed. This results in a slightly earlier steam release to the atmosphere than if the ADV's are initially in the manual mode and only the main steam safety valves release steam. Overall, the difference in radiation doses is expected to be small. Additionally, the maximum initial steam generator pressure is chosen, which leads to an earlier opening of ADV's after the reactor trip.
←(EC-13881, R304)
→(DRN 05-543, R14)
The initial SG water level is conservatively assumed to be at the minimum value. If the level in that SG exceeds 94% WR, the analysis assumes steam is released (per procedures) through the ADV to prevent overfilling of the isolated generator. This release from the affected SG increases offsite doses. The affected SG water level is expected to continue to rise after isolation due to tube leak. Minimum initial SG level results in a higher primary-to-secondary leak rate. This is more limiting due to the additional primary to secondary flow available for release to the atmosphere. Also, releases are assumed to flash to vapor whenever the top of the steam generator U-tubes are uncovered, but a DF of 100 is applied when the SG liquid level is over the top of the U-tubes. Thus, the lower SG level also will result in more flashing since the break would remain uncovered earlier and longer.
←(DRN 05-543, R14)

C. Results

→(DRN 05-543, R14)
The dynamic behavior of important NSSS parameters following a steam generator tube rupture is presented in Figures 15.6-35A through 15.6-35S.

For a double ended rupture, the primary to secondary leak rate (initially 39.5 lbm/sec) exceeds the capacity of the charging pumps. As a result, the pressurizer pressure gradually decreases from an assumed initial value of 2090 psia. The primary to secondary leak rate and drop in pressurizer water level causes the second and third charging pumps to turn on. Even with all three charging pumps turned on, the pressurizer pressure and level continue to drop.
→(DRN 05-560, R14)
At about 445 seconds, a reactor trip signal is generated due to exceeding the CPC hot leg saturation temperature range limit. Reactor trip is followed by turbine/generator trip and a subsequent loss of offsite power. Following the loss of offsite power, the reactor coolant pumps coast down and natural circulation flow is established in the RCS. The loss of offsite power is assumed to occur 3 seconds after the turbine/generator trip. This 3 second delay is based on an evaluation of grid stability and is conservative for the Waterford grid. The delay in loss of offsite power was presented to the NRC in Reference 22. Credit for this 3 second delay for the SGTR analysis for Waterford-3 was explicitly reviewed and approved by the NRC in Reference 24.
←(DRN 05-560, R14)
Following turbine trip and with turbine bypass unavailable, the main steam system pressure increases until the steam generator ADV's open at 445 seconds and MSSV's open at 450 seconds.

A maximum main steam system pressure of 1130 psia occurs at 450 seconds (Figure 15.6-35K). Subsequently, the main steam system pressure decreases, resulting in closure of the MSSV's at 455 seconds. From then on, the steam is released through ADV's only. ADV capacity is assumed equally shared between both SG's.

Following reactor trip, the main feedwater flow is terminated due to the loss of offsite power. As the level in the steam generators decrease below 27.4% NR (71% WR), an EFAS signal is generated. However, the EFW flow to steam generators will not occur unless the SG level reaches the Critical Level (see Figure 7.3-12). Since EFW flow had not yet begun at seven minutes after trip, the operator was assumed to initiate EFW flow in the Manual mode.

At 485 seconds a Safety Injection Actuation Signal (SIAS) is generated. By 515 seconds safety injection flow begins to enter the RCS when the RCS pressure has decreased to below the shutoff head of the HPSI pumps. Maximum safety injection flow is assumed to increase the primary-to-secondary releases.
←(DRN 05-543, R14)

➔(DRN 05-543, R14)
The pressurizer empties at 595 seconds resulting in rapid decrease in the RCS pressure. Reactor coolant begins to flash due to the depressurization. Consequently the RCS pressure decreases at a lower rate (Figure 15.6-35D).

At 875 seconds, the operator takes control of the plant. The operator begins to cool down the plant at a rate of 100°F/hr by closing the ADV of the affected SG and adjusting the ADV of the intact SG. Subsequently, operators are assumed to manually control EFW flow to the unaffected SG. Since MSIVs are still open, the affected steam generator continues to steam through the ADV of the intact SG. The pressures of the two steam generators remain approximately equal.

➔(DRN 05-359, R14)
The operator controls SG level to above the tubes using SG NR and WR instrumentation via a combination of control of EFW, backflow via primary-to-secondary system pressure difference and, if necessary, steaming.
🡐(DRN 05-359, R14)

➔(DRN 05-560, R14)
By 1980 seconds the hot leg temperature is below 500°F (Figure 15.6-35F). The operator isolates the affected SG, stopping steam and radiation release from the affected SG. The RCS cooldown proceeds using the intact steam generator only.
🡐(DRN 05-543, R14; 05-560, R14)
After isolation of the affected SG, the two steam generator pressures diverge. The isolated steam generator pressure increases due to the continued addition of RCS fluid through the tube break. The intact steam generator pressure continues to decrease due to steaming via the ADV.

➔(DRN 05-543, R14)
The operator initiates auxiliary pressurizer spray in order to depressurize the RCS and thus reduces the leak flow. This also increases the safety injection flow to the RCS, and thus regains water level in the pressurizer (Figure 15.6-35H).
🡐(DRN 05-543, R14)
➔(EC-34230, R306)
Per procedures, the operator controls the safety injection flow, auxiliary spray flow and the pressurizer backup heaters as necessary to maintain a minimum subcooling of 28°F in the isolated loop and the pressurizer level between 33% and 60%. Note that the pressurizer backup heaters are not used / activated in the current SGTR analysis of record.
🡐(EC-34230, R306)
➔(DRN 05-543, R14)
The RCS is brought to shutdown cooling entry conditions (392 psia and 350°F) at 8 hours. This is assumed to increase the releases for the event duration and thus obtain a conservative estimate of the radiation doses for the duration of the event.

At 23630 seconds, the affected steam generator level increases to 94% WR, and the operator opens the ADV to reduce the level. After this time, the operator periodically steams the affected steam generator to prevent it from overfilling.
🡐(DRN 05-543, R14)
➔(EC-40444, R307)
After reaching shutdown cooling entry conditions and engaging the shutdown cooling system, it is assumed that no further steam release, for cooldown purpose from ADVs, occurs from the steam generators. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.
🡐(EC-40444, R307)

The maximum RCS and secondary pressures do not exceed 110% of design pressure following a steam generator tube rupture event with a loss of offsite power, thus assuring the integrity of the RCS and the main steam system.

→(DRN 05-543, R14; 05-1551, R14)
Figure 15.6-35Q gives the integrated ADV releases from the steam generators. Figure 15.6-35R gives the integrated primary to secondary leak flow. During the SGTR, a total of 325,702 lbm of primary coolant passes through the rupture into the affected steam generator. A total of 245,600 lbm of steam is released to the atmosphere through the ADV (and MSSV) of the affected steam generator. Of this 138,969 lbm are released during the initial steaming prior to isolation. A total of 910,107 lbm of steam is released through the ADV (and MSSV) of the unaffected steam generator during cooldown of the plant.
←(DRN 05-543, R14; 05-1551, R14)

The RCS and secondary system pressures remain well below the 110% of the design pressure limits, thus assuring the integrity of these systems.

→(DRN 05-1201, R14)
15.6.3.2.1.4 Radiological Consequences

The analysis of the radiological consequences considers the most severe release of secondary activity as well as primary system activity leaked from the tube break. The iodine fission product activity available for release to the environment is a function of the primary to secondary coolant leakage rate, the assumed increase in fission product concentration, and the mass of steam discharged to the environment.
←(DRN 05-1201, R14)
In evaluating the radiological consequences of a SGTR with a Loss of Offsite Power (LOOP), a double-ended rupture of a single steam generator U-tube is assumed to occur with the reactor operating at full power. As a result of the reactor trip the turbine-generator trips and offsite power is assumed to be lost.

→(DRN 04-704, R14)
The increase in secondary system activity is detected by radiation monitors (Refer to Section 11.5). The LOOP precludes operation of the turbine bypass valves. Steam Generator pressure increases rapidly, resulting in releases to the atmosphere through the ADV's and MSSV'S. The ADVs from both SG are used to cool the RCS until the affected SG is isolated. The affected SG is assumed to be isolated when hot leg temperature is 500°F or less. Until it is isolated, both SG's contribute to releases through the ADV of the intact SG.

Once the affected SG is isolated, the intact SG is used to cool the RCS until Shutdown Cooling can be initiated. The ADV of the affected SG is periodically opened to prevent overfilling of that SG.
←(DRN 04-704, R14)

A. Assumptions and Conditions

Major assumptions and parameters assumed in the dose calculations are listed in Table 15.6-26. The assumptions and parameters used in evaluating of radiological releases include:

(1) Technical Specification limits for the primary system and secondary system activity concentrations are assumed for the design basis dose calculation.

For the pre-existing Iodine Spike (PIS) case, RCS equilibrium activity is assumed to be 60 μCi/gm Dose Equivalent I-131. This is the Technical Specification limit for full power operation following an iodine spike for up to 48 hours.
→(DRN 04-704, R14)
For the event Generated Iodine Spike (GIS) case, it was assumed that an iodine spike was caused by the reactor trip subsequent to the SGTR. The RCS activity prior to the SGTR was assumed to be at the longterm Technical Specification limit of 1.0 μCi/gm. Upon reactor trip, the I-131 equivalent source term (released from fuel) is assumed to increase with an assumed Iodine spiking factor of 335 for the GIS case.
←(DRN 04-704, R14)

(2) Technical Specification limits (0.1 μCi/gm) for secondary system activity were assumed for both SG's prior to the event.

→(DRN 05-645, R14; 06-1062, R15)

(3) For the period before the top of the U-tubes are submerged, the portion of the leaking primary fluid that flashes into steam upon entering the SG is assumed to be released to the atmosphere if a path for its release exists. The activity concentration of the steam is assumed to be the same as that of the primary fluid (i.e., a Decontamination Factor (DF) of 1). A DF of 100 is assumed after the top of the U-tubes are submerged. The tube leak rate for the rupture and the flashing fraction of the leaking fluid are calculated conservatively.

←(DRN 06-1062, R15)

(4) Tube leakage of 150 gal/day to the unaffected steam generator is assumed for the duration of the transient. This is conservative since the Technical Specification limits of 75 gal/day per steam generator.

←(DRN 05-645, R14)

→(DRN 04-704, R14)

(5) An iodine Partition Factor (PF) of 100 is assumed for activity transported to the secondary side prior to reactor trip. The un-flashed portion of the tube rupture flow mixes with the SG inventory and is released with a PF of 100. RG 1.183, Appendix F, for PWR SGTR, endorses the Appendix E, Position 5.5.4 which calls for assuming an iodine PF of 100.

A PF of 100 is assumed for the 0.375 gpm primary-to-secondary leak rate assumed for the unaffected SG.

Prior to reactor trip, any releases from the condenser could be assumed to have an iodine PF of 100 applied to those releases.

All noble gas release to the secondary side is assumed to be immediately released to the environment.

←(DRN 04-704, R14)

(6) The activity released from the faulted and intact steam generators is immediately vented to the atmosphere. Since the activity is released directly to the environment with no credit for plateout, retention, or decay, the results of the analysis are based on the most direct leakage pathway available.

(7) Conservative atmospheric dispersion factors when used for dose calculations. For the design basis calculation, 5% level dispersion factors are used.

→(DRN 04-704, R14)

(8) Following the accident, no additional steam and radioactivity are released to the environment when the shutdown cooling system is placed in operation.

(9) The SGTR analysis assumes that operators select the preferred control room air intake when pressurizing the control room, rather than assuming the pressurization flow is initially directed from the worst case air intake. Operators would diagnose which SG is subject to the tube rupture and use the Main Control Room air intake least impacted by releases from the ADV of the affected SG.

←(DRN 04-704, R14)

→(DRN 04-704, R14)

(10) Operator action is considered to maintain level in the affected SG to prevent SG overfill. These late releases, between 6.5 and 8.0 hours in the event scenario, are included in the analysis. Thus, all releases from both generators until shutdown cooling is entered are considered.

(11) Since no fuel failure is postulated, the small contribution to dose consequences from alkali metals is ignored for this event.

(12) For the SGTR event, a low initial SG level corresponding to 106,300 lbm per SG is assumed since this results in earlier uncovery of the top of the SG tubes and a lower SG decontamination factor. This is a very conservative value that corresponds approximately to the reactor trip setpoint on Steam Generator Level Low.

(13) Due to geometry considerations, the pre-trip releases from the condenser are assumed to contribute only to the off-site dose consequences. Because the condenser release point and the worst case ADV release locations are in the opposite directions from the worst case control room (MCR) air intake, and since the MCR envelope is isolated on any high radiation signal prior to the radiation entering the envelope, releases from the condenser are not assumed to contribute to the control room dose. Were wind speed and direction conditions such that releases from the condenser were to be directed to the MCR air intakes, the atmospheric dispersion factors for the ADVs would be greatly reduced. Also, the control room would be isolated on a high radiation signal prior to any of the release entering the control room envelope. Thus, any scenario involving releases from the condenser to the MCR would be less limiting than scenarios involving worst case atmospheric dispersion factors for releases from the ADVs to the control room.

(14) Since each SG contributes to the source of the pressurization flow, a scaled χ/Q can be developed to account for the relative contribution from each of the sources. Thus, a scaled effective χ/Q can be defined as:

$$\chi/Q_{eff} = ((R_1 \times \chi/Q_1) + (R_2 \times \chi/Q_2))/(R_1 + R_2)$$

where R_i is the release fraction for each source/volume (i.e., SG_1 or SG_2) and χ/Q_i is the corresponding atmospheric dispersion factor.

←(DRN 04-704, R14)

B. Mathematical Model

(1) The atmospheric dispersion factors used in the analysis, which are based on meteorological conditions assumed present during the event, are calculated according to the model described in Section 2.3. For the design basis analysis, the 5% level X/Q's presented in Table 2.3-136 are assumed.

→(DRN 04-704, R14)

(2) The mathematical model employed in the evaluation of the Total Effective Dose Equivalent is described in Appendix 15B.

←(DRN 04-704, R14)

C. Identification of Uncertainties and Conservatisms

(1) SG equilibrium activity for both SG's is assumed equal to the Technical Specification limit. This limit has been conservatively derived based on accidents such as SGTR.

(2) The break is assumed to be a double-ended rupture of a single U-tube. This is a conservative assumption since the steam generator tubes are constructed of ductile materials. The more probable failure mode is a minor leak of undetermined origin. Activity in the secondary system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceed the limits established in the Technical Specifications would lead to reactor shutdown. It is unlikely that the total amount of activity calculated to be available for release in this analysis would ever be reached.

→(DRN 05-645, R14)

(3) The coincident (i.e., 3 second delay) Loss of Offsite Power due to reactor trip following SGTR is a conservative assumption. With offsite power available, the turbine bypass valves will open, relieving steam to the main condenser. This will reduce the amount of steam and activity discharged directly to the environment.

←(DRN 05-645, R14)

→(DRN 00-592, R11-A)

(4) The meteorological conditions assumed during the course of the accident are based on the worst 5.0 percentile X/Q values. This condition results in poor values of atmospheric dispersion for the exclusion area boundary (EAB) or the low population zone (LPZ). Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the EAB or LPZ boundaries.

←(DRN 00-592, R11-A)

D. Results

→(DRN 05-1551, R14)

The TEDE dose was analyzed for the worst-case two hour period, at the exclusion area boundary (EAB), and for the duration of the accident for the low population zone (LPZ). The results are listed in Table 15.6-28. The radiological releases calculated for the SGTR event with a loss of offsite power are well within the acceptance limits. The GIS results are less than 10% of the 10CFR50.67 guidelines.

←(DRN 05-1551, R14)

15.6.3.3 Loss of Coolant Accident (LOCA)

→(DRN 05-543, R14)

←(DRN 05-543, R14)

15.6.3.3.1 Identification of Causes and Frequency Classification

The estimated frequency of a LOCA classifies it as a limiting fault as defined in Reference 1 of Section 15.0. A LOCA is defined as a hypothetical break in a pipe in the reactor coolant pressure boundary resulting in the loss of reactor coolant at a rate in excess of the capability of the coolant makeup system. [2] For this analysis, the particular breaks assumed are described in Subsection 6.3.3.2.3 and 6.3.3.3.3.

15.6.3.3.2 Sequence of Events and Systems Operations

The transient behavior during a LOCA is as follows. During the blowdown phase, the primary system depressurizes as primary coolant is ejected through the break into the containment, and the reactor is shutdown either by moderator voiding, or by CEA insertion. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core. When the core has been completely recovered, the long-term cooling mechanisms described in Subsection 6.3.3.4 will maintain acceptable core temperatures until the plant is secured.

The sequence of important events which occur in the short-term is listed in Table 15.6-12 for large-break LOCAs and in Table 15.6-12a for small break LOCAs. The sequence of events for long-term cooling is discussed in Subsection 6.3.3.4.

15.6.3.3.3 Core and System Performance

15.6.3.3.3.1 Large Break LOCA

➔(DRN 05-543, R14; EC-9533, R302; EC-8458, R307)
The large break LOCA calculations described in the following subsections pertain to a spectrum of large breaks. The spectrum analysis contains consistent base data, input, and mathematical models and is based upon the extended power uprate to 3716 MWt with replacement steam generators and up to 10% SG tubes plugged for the full core implementation of CE 16x16 NGF assemblies.
🡐(EC-8458, R307)

The break spectrum analysis utilized an actual SIT discharge line flow resistance K-factor input, an initial containment temperature of 95°F, the Westinghouse 1999 Evaluation Model for Combustion Engineering designed PWRs described in References 3 and 4 and approved by the NRC in Reference 16, and an axial power shape which is in conformance with Reference 3.
🡐(DRN 05-543, R14; EC-9533, R302)

15.6.3.3.3.1.1 Mathematical Model

➔(DRN 05-543, R14)
The large break calculations reported in this section are performed using the Westinghouse 1999 large break evaluation model for Combustion Engineering designed PWRs described in Reference 3 and 4. In the Westinghouse model, the CEFLASH-4A[1] computer program is used to determine the primary system flow parameters during the blowdown phase and the COMPERC-II[5] computer program is used to determine the system behavior during the refill and reflood phases. The core flow and thermodynamic parameters from these two codes are used as input in the STRIKIN-II[6] program which is used to calculate the hot rod clad temperature transient and peak local clad oxidation percentage. Steam cooling heat transfer coefficients calculated by the PARCH/REM module of STRIKIN-II code[7] were used for the time interval during which the reflood rate was less than 1.0 inch/second. The core-wide clad oxidation percentage is obtained from the results of both the STRIKIN-II and COMZIRC[5, Suppl. 1] computer programs.

➔(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307)
The break spectrum ECCS analysis reported in this section is based upon the extended power uprate to 3716 MWt with replacement steam generators and up to 10% SG tubes plugged for the full core implementation of CE 16x16 NGF assemblies and is performed using the Westinghouse ECCS Evaluation Model Flow Blockage Analysis described in Reference 13. In this Westinghouse model, new rupture temperature, rupture strain, and flow blockage models, adopted from NUREG-0630 (Reference 14), are used in the STRIKIN-II code and its PARCH/REM module. Also the steam cooling heat transfer coefficients calculated by the PARCH/REM module of STRIKIN-II, for use during the less than 1.0 inch/second reflood rate time interval, are calculated using an explicit method for redistribution of steam flow around the blockage region, described in Reference 13. Also, the steam cooling heat transfer coefficients were calculated using an improvement to the 1999 EM including the beneficial aspects of spacer grid heat transfer effects as documented in Reference 3, Addendum 1-P-A. In addition, the analysis utilized an axial power shape of 1.510 peak at a core height of 65%. This shape is conservative relative to the sensitivity study performed in Reference 3 to determine the limiting axial power distribution for ECCS analysis. The core-wide clad oxidation percentage is obtained from the COMZIRC results of the spectrum analysis.
🡐(DRN 05-543, R14; 06-1062, R15; EC-9533, R302; EC-8458, R307)

➔(DRN 05-543, R14)

15.6.3.3.3.1.2 Input Parameters and Initial Conditions

➔(EC-9533, R302)

Important input parameters and initial conditions used in the large break LOCA analysis are presented in Table 15.6-13 and, for containment related parameters, in Section 6.2.1.5. As discussed in Section 15.0, the initial conditions for the principal process variables monitored by COLSS are varied within the operating envelope given in Table 15.0-4 to determine the most adverse conditions for many of the accidents analyzed in this chapter. For the LOCA described in Section 15.6.3.3.1, the predicted consequences have been shown by experience with other Combustion Engineering designed plants and by generic model sensitivity studies[3] to be insensitive to many of these parameters over the ranges specified. For other parameters, such as linear heat rate and coolant flow, the results vary monotonically with input. Thus, only the most conservative end point must be analyzed.

🡐(DRN 05-543, R14)

The specific analysis input assumptions used for the Waterford 3 analysis were chosen to yield predicted ECCS performance results which conservatively bound the results based on the expected range of plant operating conditions. The Large Break LOCA described in Subsection 15.6.3.3.3.1 is valid over the range of initial conditions for the principal process variables given in Table 15.0-4 for the full core implementation of CE 16x16 NGF assemblies.

🡐(EC-9533, R302)

➔(DRN 05-543, R14)

🡐(DRN 05-543, R14)

15.6.3.3.3.1.3 Results

➔(DRN 00-1822; 05-543, R14; 06-1062, R15; EC-9533, R302; EC-8458, R307)

The important results of the analysis are summarized in Table 15.6-14 and the transient behavior of the important NSSS parameters is shown on the figures listed in Tables 15.6-15 through 15.6-17. Cladding rupture is predicted to occur during the reflood period. The maximum clad temperature is calculated to occur during late reflood for all breaks in the large break spectrum. In no case does the maximum clad temperature, local clad oxidation, or core-wide oxidation exceed the limits established by the acceptance criteria for ECCS performance listed in Reference 2. The worst case large break LOCA values for PCT, local cladding oxidation, and core wide cladding oxidation are 2092°F, 13.0% and <1%, respectively. The allowable peak linear heat generation rate is 12.9 kw/ft as specified in the Core Operating Limits Report (COLR).

🡐(DRN 00-1822; 05-543, R14; 06-1062, R15; EC-9533, R302; EC-8458, R307)

15.6.3.3.3.2 Small Break LOCA

➔(DRN 04-632, R13-B; 05-543, R14)

🡐(DRN 04-632, R13-B; 05-543, R14)

➔(DRN 00-0551)

NOTE

Any material changes in the Waterford 3 Steam Electric Station's (Waterford 3) Small Break Loss of Coolant Accident (SBLOCA) analysis from that used in the analysis dated April 1998 must be reported to the Nuclear Regulatory Commission's (NRC) Staff prior to implementation. This includes changes in the application of uncertainties in the previously stated SBLOCA analysis. Deletion or modification of this commitment must be reported to the NRC Staff prior to any change.

🡐(DRN 00-0551)

➔(DRN 00-0551)

15.6.3.3.3.2.1 Mathematical Model

➔(DRN 04-632, R13-B; 05-543, R14)

The small break LOCA ECCS performance analysis was performed with the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse Electric Company (WEC) small break LOCA evaluation model for CE design PWRs (Reference 9, Supplement 2). NRC approval of the S2M for use in licensing applications of CE design PWRs, including reference in plant technical specifications and core operating limits reports, is contained in Reference 18. In the S2M small break LOCA evaluation model, the CEFLASH-4AS computer code (Reference 1, Supplements 1 and 3) is used to perform the hydraulic analysis of the RCS until the time the safety injection tanks begin to inject. After injection from the safety injection tanks begins, the COMPERC-II computer code (Reference 5) is used to perform the hydraulic analysis in conjunction with CEFLASH-4AS. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer code (Reference 6) during the initial period of forced convection heat transfer and by the PARCH computer code (Reference 7 and Reference 9, Supplement 2) during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively calculated as the rod average cladding oxidation of the hot rod. The initial steady state fuel rod conditions used in the analysis are determined using the FATES3B computer code (Reference 19).
🡐(DRN 04-632, R13-B)
➔(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307)
COMPERC-II was not run in the Waterford 3 SBLOCA analysis because the limiting break size did not credit injection from the SITs. As is typical of S2M analyses, the limiting break size was determined to be the largest small break for which the Peak Cladding Temperature (PCT) occurs at approximately the same time that injection from the SITs starts. In this case, the PCT for the limiting break size was calculated to occur approximately 6 seconds after SIT injection would have started had it been credited.
🡐(EC-8458, R307)

The SBLOCA analysis was performed for the fuel rod conditions that result in the maximum initial stored energy in the core. The calculations included the analysis of UO_2 fuel rods, Erbia and ZrB_2 IFBA fuel rods in both the CE 16x16 NGF and standard fuel assembly designs, and Zircaloy-4 and ZIRLO™ and Optimized ZIRLO fuel rod claddings. In addition, studies were performed using PARCH to determine the fuel rod internal pressures that cause cladding rupture to occur at the times that result in the maximum PCT and the maximum cladding oxidation for the limiting break.
🡐(EC-9533, R302)
Two modifications were made in the application of the S2M in the Waterford 3 SBLOCA analysis. The modifications were a consequence of crediting an ADV in the analysis. The following is a brief description of the two modifications.
🡐(DRN 06-1062, R15)

First, the CEFLASH-4AS model for representing steam generator secondary side steam relief valves was modified. Previously, the model was limited to representing both steam generators with steam relief valves that had the same opening pressures and relief areas. The model was modified to allow different opening pressures and relief areas for the two steam generators. This was required to represent one ADV and the MSSVs on one steam generator and only the MSSVs on the other steam generator.

Secondly, the CEFLASH-4AS nodalization of the cold legs of the intact loop was modified. Previously, the two cold legs of the intact loop were lumped together into a single set of nodes and flow paths to minimize the number of nodes and flow paths and therefore to minimize computer time. The nodalization was modified to explicitly represent the two cold legs using the same nodalization as used for the broken loop (see Figure B14 of Reference 9). This change was made to better model the asymmetry in RCS flows when the two steam generator secondary side pressures are different due to crediting an ADV on one of the steam generators.
🡐(DRN 00-0551; 05-543, R14)

➔(DRN 00-0551)

15.6.3.3.3.2.2 Input Parameters and Initial Conditions

The important input parameters and initial conditions used in the small break LOCA analysis are listed in Table 15.6-13a.
🡐(DRN 00-0551)

➔(DRN 00-0551; 04-632, R13-B)

15.6.3.3.3.2.3 Results

➔(DRN 05-543, R14)
The important results of the small break LOCA analysis are summarized in Table 15.6-14a. Table 15.6-16a lists the variables plotted versus time. The plots for the break spectrum analysis are presented in Figures 15.6-206 through 15.6-233. A plot of peak cladding temperature (PCT) versus break size is presented on Figure 15.6-177.

For none of the break sizes analyzed does the peak cladding temperature, maximum cladding oxidation or core wide cladding oxidation exceed the limits established by the acceptance criteria for ECCS performance listed in Reference 2.

➔(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307)
The highest PCT for the three breaks analyzed (see Subsection 6.3.3.3.3) is 1925°F. This is bounded by the PCT of the 1.0xDEG Pump Discharge Leg Break which is the limiting LOCA (see Subsection 6.3.3.3.1).
🡐(DRN 00-0551; 04-632, R13-B; 05-543, R14; 06-1062, R15; EC-9533, R302; EC-8458, R307)

15.6.3.3.4 Barrier Performances

This section is not applicable for the spectrum of postulated reactor coolant system pipe breaks.

➔(DRN 04-704, R14)

15.6.3.3.5 Radiological Consequences – Large Break LOCA

15.6.3.3.5.1 Method of Analysis - Radiological Design Basis

15.6.3.3.5.1.1 Containment Leakage Contribution

a) Physical Model

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of this nuclear plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products is evaluated.

➔(DRN 05-1551, R14; EC-5000081470, R301)
Two release phases are assumed for this event based on the guidelines of RG 1.183. During the gap release phase the radioactivity in the fuel plenum is released to the containment building. The release during this phase is assumed to be 5 percent of the core noble gases, halogens, and alkali metals over a 30 minute period. Following the gap release phase is the Early In-Vessel release phase. During this phase the core is assumed to melt releasing the remainder of the core noble gases, 35 percent of the halogens, 25 percent of the core alkali metals, and 5 percent of the tellurium metals. Small release fractions (2 percent or less) of other core fission products are also assumed (barium and strontium, noble metals, cerium group, and the lanthanides). The Early In-Vessel phase is assumed to last 1.3 hours for a total release duration of 1.8 hours.
🡐(DRN 04-704, R14; 05-1551, R14; EC-5000081470, R301)

→(DRN 04-704, R14)

Once the gaseous and particulate fission products activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of radioactivity in the containment. These mechanisms include radioactive decay, containment sprays, containment leakage, and natural deposition. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage.

1) Radioactive decay – Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit for radioactive decay or deposition is taken in evaluating offsite dose.

2) Containment Leakage – The containment leaks at a rate incorporated as a technical specification requirement at peak calculated internal containment pressure for the first 24 hours, and at 50 percent of this leakage rate for the remaining duration of the event.

→(EC-5000081470, R301)

3) Containment Sprays – Containment sprays are credited for removal of particulate iodine from the containment atmosphere. Containment spray removal coefficients are consistent with NUREG-0800, Section 6.5.3. Removal of elemental iodine from the containment atmosphere is not modeled for the purpose of determining off-site dose, or dose due to radiological intakes or due to radiological shine to control room personnel. Removal of organic iodine due to spray is not credited.

←(EC-5000081470, R301)

→(DRN 05-1551, R14)

4) Natural Deposition – Reduction of airborne activity by natural deposition may be credited for a LOCA. The Power's 10% aerosol deposition is specified for the natural deposition of aerosols and particulate iodine. This model is described in NUREG/CR-6604. The guidance of NUREG-0800, Section 6.5.2, is applied for natural deposition of elemental iodine.

←(DRN 05-1551, R14)

The contribution to the potential Total Effective Dose Equivalent (TEDE) doses is the result of direct leakage from the containment to the annulus, bypass leakage, and leakage processed through the Controlled Ventilation Area System. The resultant activity release to the environment is assumed to be released at ground level. The activity released to the environment is treated as a semi-infinite cloud, i.e., a cloud containing radioactive material that is infinite in all directions above the ground. The concentration of radioactive material within the could is assumed to be uniform and equal to the maximum centerline ground-level concentration that would exist in the cloud at the point of immersion of an individual located at the exclusion area boundary (EAB) or the outer boundary of the low population zone (LPZ).

b) Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6-18.

The following specific assumptions were used in the analysis.

1) The reactor core inventory is based on long-term operation at a core thermal power level of 3,735 MWt (100.5 percent of 3,716 MWt).

2) The fission products are released in two distinct phases over a 1.8 hour period.

Gap Release: 5 percent of the core noble gases, halogens, and alkali metals are released over the first 30 minutes of the event.

←(DRN 04-704, R14)

→(DRN 04-704, R14)

Early In-Vessel: The remaining 95 percent of the noble gases, 35 percent of the halogens, 25 percent of the alkali metals, and 5 percent of the tellurium metals are released over the following 1.3 hours. Small fractions of other fission products are released as well (2 percent of barium and strontium, 0.25 percent of the noble metals, 0.05 percent of the cerium group, and 0.02 percent of the lanthanides).

3) Of the iodine fission product inventory released to the containment, 95 percent is in the aerosol (particulate) form, 4.85 percent is elemental, and 0.15 percent is organic. These source term assumptions are applicable since the safety injection sump remains at a pH of greater than or equal to 7.0 as discussed in Subsection 6.1.3.

4) Containment is assumed to leak at 0.5 volume percent per day during the first 24 hours immediately following the accident and 0.25 volume percent per day thereafter.

→(EC-5000081470, R301)

5) The Shield Building Maintenance Hatch Seals (MHS) are assumed to fail 2 days after the accident. This is modeled by assuming one train of the Shield Building Ventilation System (SBVS) operates continuously in the discharge mode after 2 days. The offsite and control room dose contributions due to failure of the MHS are included in the dose contributions due to containment gas release (Table 15.6-18). SBVS performance is discussed in Subsections 6.2.3 and 6.5.3.

6) Even though the Waterford 3 control room is maintained at a positive pressure with respect to the atmosphere, 100 CFM unfiltered air inleakage to the control room is assumed. The assumed unfiltered inleakage location is the east control room air intake, since this location corresponds to the most conservative dispersion factors. The dose contribution to control room occupants due to this leakage is included in the dose contributions due to containment gas release (Table 15.6-18).

→(DRN 05-791, R14)

7) The shield building pressure may rise to a less negative pressure than -0.25” w.g. for a period of about a minute after the LOCA. A 0.1 hour (6 minute) Positive Pressure Period is assumed for the shield building annulus region at the start of the event. Per BTP CSB 6-3, the total allowed containment leakage during this period is assumed to be directly released to the environment to conservatively bound the time that the shield building annulus pressure may be greater than -0.25 in w.g. The dose contribution due to this positive pressure period is included in the total LOCA dose results (Table 15.6-18).

←(DRN 05-791, R14)

8) Shield Building Ventilation System charcoal filtration is not credited for the Reactor Building airborne release path.

←(EC-5000081470, R301)

c) Mathematical Model Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

1) The mathematical models used to analyzed the activity released during the course of the event are discussed in Appendix 15B.

2) The atmospheric dispersion factors used in the analysis are based on meteorological conditions assumed present during the course of the accident. Calculation methods of X/Qs are presented in Subsection 2.3.4. For the design basis event, five percent level X/Qs are used.

3) TEDE doses to an individual exposed at the EAB or LPZ are analyzed using the models described in Appendix 15B.

←(DRN 04-704, R14)

→(DRN 04-704, R14)

4) The integrated doses to control room personnel are analyzed based on the models described in Appendix 15B.

15.6.3.3.5.1.2 Leakage from Engineered Safety Features (ESF) Components Outside Containment

→(EC-5000081470, R301)

Subsequent to the injection phase of ESF system operation, the water in the containment recirculation sumps is recirculated by the HPSI pumps and the containment spray pumps. For the purposes of dose calculations, a total leak rate of 0.5 GPM of the sump water is assumed to leak into the RAB/CVAS area (per RG 1.183, the assumed leakage is twice that permitted by procedure). This leakage includes the leakage from all of the system that may possibly leak the sump water into the RAB/CVAS area (e.g., possible leakage from the safety injection pump seals).

The source term assumptions for the containment sump are similar to those for the airborne leakage. However, noble gases do not readily dissolve in water, therefore they are not considered for leakage from ESF systems. Also, all other radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The iodine contribution is limited by the amount of liquid which is assumed to flash to steam. The flashing fraction for iodine assumed in the offsite and Main Control Room intake dose analyses is 10 percent. The iodine released is assumed to be 97 percent elemental and 3 percent organic. Finally, the remaining fission products are assumed to be particulate in form, therefore they are retained in the liquid and are not available for release to the atmosphere.

The iodine activity released to the RAB/CVAS area due to leakage of sump water is assumed to be immediately processed through the ESF-grade charcoal adsorbers of the Controlled Ventilation Area System. No credit is taken for holdup and decay for this term. The results for off-site and control room doses due to the ESF system leakage are provided in Table 15.6-18.

←(EC-5000081470, R301)

15.6.3.3.5.1.3 Radiation Shine from ESF Charcoal Filter Trains

Radiation shine from the ESF filter trains located on the +46' elevation of the RAB is also considered for dose to control room personnel following a LB LOCA. The filter trains considered are the Controlled Ventilation Area System filter train, the Shield Building Ventilation System filter train, and the Control Room Emergency Ventilation Unit filters trains. Structural walls were credited for shielding, specifically 12 inches of concrete were credited for the control room filters, and 30 inches of concrete lie between the CVAS and SBVS filter trains and the control room. The buildup of fission products on the filter trains was modeled using conservative flow rates for each filter train.

→(DRN 05-1551, R14)

The CVAS filter trains had the most significant impact to control room doses. For the iodine loading on CVAS filter due to ESF leakage, a flashing fraction of 10 percent is assumed for the first 24 hours, and a 2 percent flashing fraction is assumed thereafter. The doses from all filter trains are included in Table 15.6-18.

←(DRN 05-1551, R14)

→(DRN 05-645, R14)

* In the Safety Evaluation Report for Amendment 198, the NRC did not find the assumption of two percent airborne iodine after 24 hours acceptable but approved the filter shine analysis results due to compensating conservatisms. The NRC recommended that the flashing fraction be increased to ten percent or the lower value of two percent be justified when the large break LOCA filter shine dose analyses are revised in the future.

←(DRN 04-704, R14; 05-645, R14)

→(DRN 04-704, R14)

15.6.3.3.6 Radiological Consequences – Small Break LOCA

For small breaks, primary pressure control and decay heat removal are accomplished through steaming from the secondary system. The release dynamics and locations for a Small Break LOCA can differ from those of the traditional Large Break LOCA (as documented in Subsection 15.6.3.3.5).

a) Physical Model

The Small Break LOCA has been analyzed for two different release pathways:

1) Reactor containment building release pathway, similar to that for Large Break LOCA.

2) Secondary steaming pathway, consisting of releases from the MSSVs or ADVs to the environment.

→(EC-5000081470, R301)

This is similar to the two different release pathways which are postulated for a PWR CEA Ejection event (Subsection 15.4.3.2). For the reactor containment building release pathway, activity released to containment is assumed to be released to the environment due to containment leaking at its design rate. For the secondary steaming pathway, secondary steaming to remove decay heat and to cooldown the plant to shutdown cooling entry conditions is assumed; primary-to-secondary leakage provides a release path for activity to the secondary system, from which it is released to the environment via secondary steaming.

←(EC-5000081470, R301)

Dynamics for a Small Break LOCA are very different than for a Large Break LOCA. The top of the core remains covered for at least ten minutes for the break sizes considered. For Waterford 3, the smallest break size for which containment spray would not actuate is small enough that the core remains covered during the transient, thus there would be no core damage. Larger break sizes would result in lower RCS pressures, resulting in discharge of the safety injection tanks. The limiting break size will be one where the hot rod cladding heat-up transient is terminated by only the high pressure safety injection pumps. Because the heat-up transient only starts after core uncovery, at least ten minutes into the event, there is no challenge to fuel melt limits. Thus, the only mechanism for fuel damage is clad damage that results in release of the gap activity.

b) Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Table 15.6-18A.

The following specific assumptions were used in the analysis.

General:

1) The reactor core inventory is based on long-term operation at a core thermal power level of 3,735 MWt (100.5 percent of 3,716 MWt).

2) The use of a NUREG-1465 AST modeling results in a gap fraction of 5.0% being assumed, consistent with the guidance for LOCA of Regulatory Guide 1.183. This is an appropriate assumption when 100% of the fuel rods are assumed to fail in a mode that releases the gas gap activity. The gas gap fraction of 5.0% is assumed for iodines, noble gases, and alkali metals (cesium and rubidium). A near instantaneous release duration of 30 seconds or less is assumed.

3) No credit will be taken for the effects of containment spray for fission product removal in the SBLOCA dose analyses.

←(DRN 04-704, R14)

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

4) A constant pressurization flow of 225 CFM and a constant unfiltered in-leakage of 100 CFM are assumed for the duration of event, for both the containment leakage pathway and the steaming pathway. It is assumed that the operators manually initiate the pressurization mode for the main control room. The assumed unfiltered inleakage location is the east control room air intake, since this location corresponds to the most conservative dispersion factors.

🡐(EC-5000081470, R301)

Containment Leakage Pathway Assumptions:

➔(DRN 05-1551, R14)

5) Of the iodine fission product inventory released, 95 percent is in the aerosol (particulate) form, 4.85 percent is elemental, and 0.15 percent is organic.

🡐(DRN 05-1551, R14)

6) Containment is assumed to leak at 0.5 volume percent per day during the first 24 hours immediately following the accident and 0.25 volume percent per day, thereafter.

7) The reactor building leakage pathway is similar to the LBLOCA release model. For the reactor containment building leakage pathway, all of the iodine, alkali metal, and noble gas activity is assumed to be released to containment. The design basis containment leak rate of 0.5% by volume per day for 0-24 hours and 0.25% by volume per day for 1-30 days is assumed.

8) For the reactor containment building leakage pathway, the Powers 10% Aerosol Decontamination Factor model is assumed for natural deposition. This model is containment in the RADTRAD analysis code of NUREG/CR-6604. A natural deposition factor of 0.40/hr is assumed for elemental iodine.

➔(EC-5000081470, R301; EC-3277, R301)

9) As in large break LOCA the SBVS charcoal filtration in not credited for iodine removal.

10) As in large break LOCA a 0.1 hour (6 minute) Positive Pressure Period is assumed for the Shield Building annulus region at the start of the event.

🡐(EC-5000081470, R301; EC-3277, R301)

Secondary Steaming Pathway Assumptions:

➔(EC-3277, R301; EC-40444, R307)

11) All the iodine, alkali metal, and noble gas activity due to the postulated SBLOCA is assumed to be in the primary coolant when determining dose consequences due to primary-to-secondary SG tube leakage and subsequent secondary steaming. Releases are assumed to be terminated once shutdown cooling is initiated and the SGs are no longer providing decay heat removal capability, thus, no further releases would occur for the 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.

🡐(EC-40444, R307)

12) A primary-to-secondary SG tube leak rate of 75 gallons per day (gpd) per SG is assumed for the analysis. This value is consistent with the Technical Specification allowable value.

13) For purposes of evaluating the secondary steaming release for MCR dose, the worst case single failure would be the failure of a DC power bus, which would result in failure of one emergency diesel generator and failure of the control logic for one ADV. It is assumed that the ADV which responds to provide decay heat removal for the event is the ADV with the worst case χ/Q, i.e., the East ADV which is assumed to contribute to unfiltered in-leakage at the East MCR Outside Air Intake. No local manual action to open the other ADV and/or to close this ADV to lower releases to the main control room is assumed.

🡐(DRN 04-704, R14; EC-3277, R301)

→(DRN 04-704, R14; EC-3277, R301)

14) For the secondary steaming path, iodine and alkali metal releases to the secondary side via SG tube leakage area assumed subject to a PF. Consistent with RG 1.183, Appendix E, Section 5.5, a PF of 100 is assumed for iodine and alkali metals. Conservatively, 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.

15) All noble gas release to the secondary side via SG tube leakage is assumed to be immediately released to the environment.

16) Of the iodine released via the secondary steaming pathway, 97 percent is elemental and 3 percent is organic.

←(EC-3277, R301)

c) Mathematical Model

Mathematical models used in the analysis are described in the following sections:

1) The mathematical models used for the releases from a Small Break LOCA are consistent with those used in for the CEA Ejection event discussed in Subsection 15.4.3.2.

2) The atmospheric dispersion factors used in the analysis are based on meteorological conditions assumed present during the course of the accident. Calculation methods of X/Qs are presented in Subsection 2.3.4. For the design basis event, five percent level X/Qs are used.

3) TEDE doses to an individual exposed at the EAB or LPZ are analyzed using the models described in Appendix 15B.

4) The integrated doses to control room personnel are analyzed based on the models described in Appendix 15B.

The radiological consequences of the event are contained in Table 15.6-18B. The results confirm that offsite doses (EAB and LPZ) meet the acceptance criteria set forth in 10CFR50.67. The calculated control room dose meets the dose limits of 10CFR50.67 and 10CFR50, Appendix A, GDC 19.

←(DRN 04-704, R14)

15.6.3.3.5.2 Method of Analysis - Realistic Analysis Assumptions

→(DRN 05-1551, R14)

Deleted.

←(DRN 05-1551, R14)

15.6.3.3.5.3 Doses From Hydrogen Purge

→(DRN 04-704, R14)

Deleted.

←(DRN 04-704, R14)

15.6.3.4 <u>Inadvertent Opening of a Pressurizer Safety Valve</u>

→(DRN 05-543, R14)

The inadvertent opening of a pressurizer safety valve was explicitly analyzed for Cycle 1 and was evaluated for the extended power uprate to 3716 MWt (Section 2.12.6 of Reference [23]). The evaluation for extended power uprate concluded that, by the nature of the event, the results for the inadvertent opening of a pressurizer safety valve are bounded by the results for the limiting small break LOCA in the reactor coolant pump discharge leg. Consequently, the conclusion of the Cycle 1 analysis, namely that the results of the event are well within the ECCS acceptance criteria of 10 CFR 50.46, are applicable to the extended power uprate.

The following sections describe the Cycle 1 analysis.

←(DRN 05-543, R14)

15.6.3.4.1 Identification of Causes and Frequency Classification

The estimated frequency of an inadvertent opening of a pressurizer safety valve classifies it as a faulted condition event as defined in Reference I of Section 15.0. The frequency of occurrence of an inadvertent opening of a pressurizer safety valve was determined from the combined operating experience of 34 pressurized water reactors. There has not been a single inadvertent opening of a pressurizer safety valve in more than 260 pressurizer safety valve years of operation. The corresponding frequency of occurrence for this event is consistent with the definition of the limiting fault category in ANSI N18.2.[12] The inadvertent opening of a pressurizer safety valve at normal RCS operating pressures could only be caused by a passive mechanical failure of the valve.

15.6.3.4.2 Sequence of Events and Systems Operation

Following an inadvertent opening of a pressurizer safety valve, the RCS pressure decreases. and a low pressurizer pressure signal initiates reactor trip. The trip quickly reduces core power to decay heat levels. The reactor coolant pumps and the main turbine are tripped at the same time due to assumed simultaneous loss of offsite power, and the emergency diesel generators are started.

The low pressurizer pressure also initiates a safety injection actuation signal (SIAS) which actuates the safety injection pumps. In this analysis, the worst single failure was assumed (failure of one diesel to start) and hence only one HPSI and one LPSI pump are available.

The HPSI pump flow prevents the core from uncovering and eventually refills the RCS, since the flow out of the pressurizer safety valve is less than the flow from the HPSI pump during the transient. As a result, the clad temperature remains very low during the transient.

Table 15.6-21 lists the sequence of events following an inadvertent opening of a pressurizer safety valve.

15.6.3.4.3 Core and System Performance

15.6.3.4.3.1 Mathematical Model

The CE small break model was employed (Reference 9). The NSSS response to an inadvertent opening of a pressurizer safety valve was simulated using the CEFLASH-4AS blowdown code (Reference 1). The temperature transient in the hottest fuel rod was calculated using the STRIKIN-II code (Reference 6 and the PARCH code (Reference 7).

15.6.3.4.3.2 Input Parameters and Initial Conditions

→(DRN 00-0551)

The initial conditions and input parameters of the NSSS assumed in the analysis are listed in Table 15.6-13c.

←(DRN 00-0551)

→(DRN 00-592)

15.6.3.4.3.3 Results

←(DRN 00-592)

The behavior of the NSSS following an inadvertent opening of pressurizer safety valve is shown on Figures 15.6-178 through 15.6-185. The decrease in reactor coolant inventory causes RCS pressure to drop as shown on Figure 15.6-179. At 413 seconds after the break, the pressurizer pressure drops to 1560 psia, initiating a reactor trip and turbine trip. A loss of offsite power is assumed to occur simultaneously with reactor trip. This assumption minimizes the heat removal by the steam generators.

→(DRN 00-592)
The low pressurizer pressure also initiates a SIAS, which actuates one safety injection train (assuming a single failure of the second diesel generator). The HPSI pump takes suction from the refueling water storage pool. At 5720 seconds, the HPSI pump injection exceeds the leak rate and the RCS begins to refill. Since the core never uncovers, there is no potential for a significant clad heatup. The peak clad temperature was calculated to be 928°F. Plant cooldown by the operator is assumed to begin at one hour after the break and follows the LOCA emergency procedure as described in Subsection 6.3.3.4.

Table 15.6-22 lists the figures that illustrate the results of this analysis. A summary of the analysis results is shown in Table 15.6-23. These results demonstrate that the ECCS response to an inadvertent opening of a pressurizer safety valve is acceptable and well within the performance criteria of 10CFR50.46.
←(DRN 00-592)

15.6.3.4.4 Barrier Performance

This section is not applicable for inadvertent opening of a pressurizer safety valve.

SECTION 15.6: REFERENCES

1. "CEFLASH-4A.A FORTRAN-FV Digital Computer Program for Reactor Blowdown Analysis." CENPD-133, April 1974 (Proprietary).

→(DRN 04-632, R13-B)

"CEFLASH4AS. A FORTRAN-IV Computer Program for Reactor Blowdown Analysis." CENPD-133, Supplement 1, August 1974 (Proprietary).

←(DRN 04-632, R13-B)

"CEFLASH-4A,A FORTRAN-[V Digital Computer Program for Reactor Blowdown Analysis (Modification)," CENPD-133, Supplement 2, December 1974 (Proprietary).

"CEFLASH-4AS,A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident", CENPD-133, Supplement 3, January, 1977 (Proprietary). CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN 77 Digital Computer Program for Reactor Blowdown Analysis", June 1985.

→ (DRN 00-0551)

2. Code of Federal Regulations, Title 10, Part 50, Section 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors”.

← (DRN 00-0551)

3. "Calculative Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132, August 1974 (Proprietary).

"Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132 Supplement 1, December 1974 (Proprietary)

"Calculational Methods for the C-E Large Break LOCA Evaluation Model, "CENPD-132, Supplement 2, July 1975 (Proprietary). CENPD-132-P, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Mode for the Analysis of C-E and W Designed NSSS", June 1985.

→(DRN 05-560, R14)

“Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model”, CENPD-132 Supplement 4-P-A, March 2001 (proprietary).

←(DRN 05-560, R14)

→(EC-9533, R302)

CENPD-132-P-A, Supplement 4-P-A, Addendum 1-P-A, “Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood,” August 2007.

←(EC-9533, R302)

4. "Reflood Heat Transfer: Application of FLECHTReflood Heat Transfer Coefficients to 16x16 Fuel Bundles," CENPD-213, January 1976 (Proprietary).

5. "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," CENPD-134, April 1974 (Proprietary).

"COMPERC-II,A Program for Emergency Refill-Reflood of the Core (Modification)," CENPD-134, Supplement 1, December 1974 (Proprietary). CENPD-134, Supplement 2-A, COMPERC-II,A Program for Emergency Refill-Reflood of the Core," June 1985.

6. "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, "CENPD-135" April 1974 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)," CENPD-135, Supplement 2, December 1974 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135, Supplement 4, August 1976 (Proprietary).

"STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," CENPD-135, Supplement 5, April 1977 (Proprietary).

7. CENPD-138, "PARCH,A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Proprietary).

CENPD- 138, Supplement 1, "PARCH,A FORTRAN IV Digital Program to Evaluate Pool Boiling Axial Rod and Coolant Heating" (Modifications), February, 1975.

CENPD-138, Supplement 2, "PARCH,A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", January, 1977 (Proprietary).

8. J.J. DiNunno, et. al., "Calculation of Distance Factors for Power and Test Reactor Sites, "TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C., 1962.

9. "Calculative Methods for the CE Small Break LOCA Evaluation Model", CENPD-137, August, 1974 (Proprietary).

"Calculative Methods for the CE Small Break LOCA Evaluation Model", CENPD-137, Supplement 1, January 1977 (Proprietary).

→(DRN 00-0551)

CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998 (Proprietary)

←(DRN 00-0551)

10. Letter, O.D. Parr (NRC) to F.M. Stern (CE), June 13, 1975.

11. Letter, K. Kniel (NRC) to A.E. Scherer (CE), September 27, 1977.

12. ANSI N18.2, "Nuclear Safery Criteria for the Design of Stationary Pressurized Water Reactor Plants", 1973.

→(DRN 06-1062, R15)

13. Enclosure I-P to LD-81-095, "CE ECCS Evaluation Model Flow Blockage Analysis," December, 1981 (Proprietary).

←(DRN 06-1062, R15)

14. D.A. Powers and R.O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April, 1980.

15. Letter, A.E. Scherer (C-E) to C.O. Thomas (NRC), LD-85-050, November 5, 1985.

→(DRN 05-560, R14)

16. Letter, S.A. Richards (NRC) to P.W. Richardson (WEC), Safety Evaluation of Topical Report CENPD-132, Supplement 4 Revision 1, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model (TAC No. MA 5660)", December 15, 2000.

←(DRN 05-560, R14)

17. Entergy Operations letter to NRC, "Cycle 6 Reload Analysis Report," W3F1-92-0053.

→(DRN 00-0551)

18. Letter, T. H. Essig (NRC) to I. C. Rickard (ABB), "Acceptance for Referencing of the Topical Report CENPD-137(P) Supplement 2, "Calculative Methods for the C-E Small Break LOCA Evaluation Model" (TAC No. M95687)," December 16, 1997.

19. CENPD-139-P-A, "C-E Fuel Evaluation Model," July 1974.

CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

←(DRN 00-0551)

→(DRN 04-632, R13-B; 04-704, R14)
20. L. Soffer, et. Al., “Accident Source Terms for Light-Water Nuclear Power Plants,” NRC Report NUREG-1465, February, 1995.
←(DRN 04-632, R13-B)

21. Regulatory Guide 1.183, Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.
←(DRN 04-704, R14)

→(DRN 05-560, R14)
22. Letter, A.E. Scherer (C-E) to D.G. Eisenhut (NRC), “Turbine Trip Time Delay”, LD-82-40, March 31, 1982.

23. W3F1-2003-0074, J.E. Venable (Entergy) to Document Control Desk (NRC), “License Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38”, November 13, 2003.

24. Letter, N. Kalyanam (NRC) to Joseph E. Venable (EOI), “Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Extended Power Uprate (TAC No. MC1355)”, April 15, 2005.
←(DRN 05-560, R14)

TABLE 15.6-1 Revision 14 (12/05)

SEQUENCE OF EVENTS FOR A LETDOWN LINE BREAK OUTSIDE CONTAINMENT

→(DRN 03-220, R12-B; 05-543, R14)

Time (sec)	Event	Setpoint or Value
0.0	Letdown line rupture occurs	---
1800	CPCS Out-of-Range Trip on Low Pressurizer Pressure occurs, psia	1736
≥ 1800	CEAs begin to drop into the core	---
≥ 1800	CEAs 90% inserted	---
≥ 1800	Isolation of ruptured letdown line (operator action)	---
≥ 1800	Safety injection actuation signal, psia	1560
≥ 1800	SIS flow initiated	---
≥ 1800	Operator initiates plant cooldown	---
28800	Shutdown cooling initiated, °F	350

←(DRN 03-220, R12-B; 05-543, R14)

TABLE 15.6-2 Revision 14 (12/05)

ASSUMPTIONS FOR LETDOWN LINE BREAK
OUTSIDE CONTAINMENT

→(DRN 03-220, R12-B; 05-543, R14)

Parameter, units	Assumptions
→(DRN 02-526, R12)	
Initial core power, MWt	3735
←(DRN 02-526, R12)	
Core inlet coolant temperature, °F	533
Core outlet coolant temperature, °F	588
Initial RCS flow rate, lbm/hr	178.9×10^6
Initial Pressurizer Pressure, psia	2312
Steam generator secondary pressure, psia	742
Secondary relief valve setpoint (lowest bank), psia	1085
Moderator temperature coefficient, 10^{-4} Δp°F	-4.2

←(DRN 03-220, R12-B; 05-543, R14)

TABLE 15.6-3 (Sheet 1 of 2) Revision 307 (07/13)

PARAMETER USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

→(DRN 04-704, R14 Core Power Level	3735 MWt
Core Inventory:	Table 12.2-12A
Secondary Streaming Pathway	
→(DRN 05-1551, R14) Primary-to-Secondary Leak Rate ←(DRN 05-1551, R14)	150 gpd for both SG's
→(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, Intact SG)	
0-30 minutes	10
> 30 minutes	100
Duration of Release	8 hours
Control Room Parameters:	
Volume	220,000 ft^3
Recirculation Flow Rate	3800 CFM
Iodine Filter Efficiency	99% (elemental/organic)
→(DRN 05-1551, R14) Pressurization Flow ←(DRN 05-1551, R14)	225 CFM (max.)
→(EC-5000081470, R301) Unfiltered Inleakage Breathing Rate ←(EC-5000081470, R301)	100 CFM 3.47E-04 m^3/sec.
Control Room Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 hours - 30 days	0.4

Main Control Room χ/Q Assumed

→(EC-5000081470, R301)

Time	Unfiltered In-leakage	Pressurization Flow
0-30 min	2.77E-03	5.15E-04
30 min-2 hr	5.368E-02	3.904E-03
2-8 hr	3.77E-02	2.914E-03

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

TABLE 15.6-3 (Sheet 2 of 2) Revision 14 (12/05)

→(DRN 04-704, R14)

PARAMETER USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE

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

0-2 hr Steaming	2-8 hr Steaming
794,217	1,357,617

	0-15 min	15-30 min	0.5-1 hr	1-2 hr	2-4 hr	4-6 hr	6-8 hr
PIS	404.81	360.15	0.70	0.12	0.23	0.21	0.24
GIS	23.00	48.33	0.06	0.11	0.19	0.20	0.27

Noble Gas Releases (Ci):

	0-15 min	15-30 min	0.5-1 hr	1-2 hr	2-4 hr	4-6 hr	6-8 hr
Kr-83m	806.68	717.98	2.1	4.57	10.1	10.28	10.28
Kr-85	16.45	14.65	0.05	0.10	0.22	0.23	0.23
Kr-85m	17.45	15.53	0.05	0.10	0.22	0.23	0.23
Kr-87	18.45	16.42	0.05	0.11	0.24	0.24	0.24
Kr-88	855.7	761.61	2.23	4.85	10.72	10.9	10.9
Xe-131m	73.86	65.74	0.20	0.42	0.93	0.95	0.95
Xe-133m	31.74	28.25	0.09	0.18	0.40	0.41	0.41
Xe-133	132.93	118.31	0.35	0.76	1.67	1.70	1.70
Xe-135m	69.04	61.45	0.18	0.40	0.87	0.88	0.88
Xe-135	62.48	55.61	0.17	0.36	0.79	0.80	0.80

←(DRN 04-704, R14)

TABLE 15.6-4 Revision 14 (12/05)

RADIOLOGICAL CONSEQUENCES OF A LETDOWN LINE RUPTURE
IN THE REACTOR AUXILIARY BUILDING

→(DRN 04-704, R14)

	TEDE Dose	**Acceptance Criteria**
PIS Case:		
EAB (worst 2 hour dose)	≤25	25 REM TEDE
LPZ (Duration)	≤25	25 REM TEDE
Main Control Room	≤5	5 REM TEDE
GIS Case:		
EAB (worst 2 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 15.6-5 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.6-6 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.6-7 Revision 5 (12/91)

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TABLE 15.6-8 Revision 5 (12/91)

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TABLE 15.6-9 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.6-10 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.6-11 Revision 5 (12/91)

THIS TABLE INTENTIONALLY DELETED

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TABLE 15.6-12 Revision 307 (07/13)

TIME SEQUENCE OF IMPORTANT EVENTS FOR LARGE LOCA
(SECONDS AFTER BREAK)

→(DRN 05-543, R14; 06-1062, R15)

Break	SI Tanks On	Time of Annulus Downflow	Start of Reflood	SI Tanks Empty	SI Pumps On	Hot Rod Rupture	Time of PCT
←(DRN 06-1062, R15)							
→(EC-9533, R302)							
Break Spectrum Analysis for Peak Cladding Temperature							
→(DRN 06-1062, R15; EC-8458, R307)							
1.0 DEG/PD [(a)]	8.9	24.3	40.3	91.2	34.2	40.2	228
0.8 DEG/PD	10.1	25.4	41.3	92.5	34.2	42.2	228
0.6 DEG/PD	11.6	27.3	43.1	94.6	34.3	47.1	230
0.4 DEG/PD	15.0	31.5	47.0	99.0	34.6	67.3	252
←(DRN 06-1062, R15; EC-8458, R307)							
Case Results for Maximum Cladding Oxidation							
→(EC-8458, R307)							
0.8 DEG/PD	10.1	25.4	41.3	92.5	34.2	47.9	228
←(EC-9533, R302; EC-8458, R307)							

[(a)] See Table 15.6-15 for an explanation of these abbreviations.

←(DRN 05-543, R14)

TABLE 15.6-12a Revision 307 (07/13)

➔(DRN 00-0551)

SEQUENCE OF EVENTS FOR SMALL BREAK LOCA
(TIME, SECONDS AFTER BREAK)

➔(DRN 04-632, R13-B; 05-543, R14)

Break Size	Reactor Trip	SIAS	HPSI Flow Delivered to RCS	LPSI Flow Delivered to RCS	SIT Flow Delivered to RCS	Time of PCT
←(DRN 04-632, R13-B) ➔(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307)						
0.04 ft^2/PD	166	165	195	(a)	>4000(b)	2313
0.05 ft^2/PD	132	131	161	(a)	1963(b)	1969
0.06 ft^2/PD	110	109	139	(a)	1554	1556

←(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307)

(a) Calculation terminated before start of LPSI flow into the RCS.

(b) Injection from the SITs was not credited. Value is the time injection would have begun had it been credited.

←(DRN 00-0551, 05-543, R14)

TABLE 15.6-13 Revision 307 (07/13)

GENERAL SYSTEM PARAMETERS AND INTIAL CONDITIONS
(LARGE BREAK LOCA SPECTRUM ANALYSIS)

Quantity	Value
→(DRN 02-526, R12; 05-543, R14) Reactor power level, Mwt (rated thermal power, plus power measurement uncertainty)	3735
←(DRN 02-526, R12) Peak linear heat rate, kw/ft	12.9[b]
→(DRN 06-1062, R15; EC-9533, R302) Peak linear heat rate of the average rod in assembly with hot rod, kw/ft	12.0
Gap conductance at peak linear heat rate [a] Btu/hr-ft^2 °F	2275
Fuel centerline temperature at peak linear heat rate[a] °F	3016
Fuel average temperature at peak linear heat rate [a] °F	1888
Hot rod gas pressure psia ←(DRN 05-543, R14; 06-1062, R15; EC-9533, R302)	1467
Moderator temperature coefficient at initial density, ΔP/°F	0.0 x 10^{-4}
System flowrate (total), lb/hr	148.0 x 10^6
Core flowrate, lb/hr	144.15 x 10^6
Initial system pressure, psia	2,250
→(DRN 05-543, R14) Cold leg temperature, °F	533
Hot leg temperature, °F	598.7
→(DRN 06-1062, R15; EC-8458, R307) Number of plugged tubes per steam generator	897[c]
←(DRN 06-1062, R15; EC-8458, R307) Low pressurizer pressure SIAS setpoint, psia	1560
Safety injection tank pressure, psia (min/max)	584.7 / 714.7
→(EC-9533, R302) Safety injection tank water volume, ft^3(min/max)	926 / 1586
←(EC-9533, R302) LPSI pump flow rate, gpm (min, 1 pump/max, 2 pump)	4084 / 11300
→(DRN 06-1062, R15) HPSI pump flow rate, gpm (min, 1 pump/max, 2 pump)	787 / 1970

→(EC-9533, R302)

(a) These quantities correspond to the burnup (32 GWD/MTU, hot rod average) yielding the highest peak clad temperature.

←(DRN 06-1062, R15; EC-9533, R302)

(b) As specified in the Core Operating Limit Report.

←(DRN 05-543, R14)

→(DRN 06-1062, R15; EC-8458, R307)

(c) Corresponds to 10% SG tubes plugged for replacement steam generators.

←(DRN 06-1062, R15; EC-8458, R307)

TABLE 15.6-13a (Sheet 1 of 2) Revision 307 (07/13)

→(DRN 00-0551, R10)

GENERAL SYSTEM PARAMETERS AND INITIAL CONDITIONS
(FOR THE SMALL BREAK LOCA ECCS PERFORMANCE ANALYSIS)

Quantity	Value
→(DRN 02-526, R12; 04-632, R13-B; 05-543, R14) Reactor Power Level (rated thermal power, plus power measurement uncertainty), MWt ←(DRN 02-526, R12; 04-632, R13-B)	3735
Peak Linear Heat Generation Rate (PLHGR) of the Hot Rod, kW/ft	13.2
→(DRN 06-1062, R15; EC-9533, R302; EC-8458, R307) Gap Conductance at the PLHGR(a), Btu/hr/ft^2/°F ←(EC-8458, R307)	1769
Fuel Centerline Temperature at the PLHGR(a) °F →(EC-8458, R307)	3205
Fuel Average Temperature at the PLHGR(a), °F ←(EC-8458, R307)	2025
Hot Rod Gas Pressure(a), psia ←(DRN 06-1062, R15; EC-9533, R302)	705
Moderator Temperature Coefficient at Initial Density, $\Delta\rho$/°F ←(DRN 05-543, R14)	0.0×10^{-4}
Axial Shape Index, ASI units	-0.25
RCS Pressure, psia	2250
RCS Flow Rate, lbm/hr	148×10^6
Core Flow Rate, lbm/hr	144.15×10^6
→(DRN 05-543, R14) Cold Leg Temperature, °F ←(DRN 05-543, R14)	552.0
Hot Leg Temperature, °F	615.5
→(DRN 03-1964, R13; 05-543, R14; 06-1062, R15; EC-8458, R307) Number of Plugged Tubes per Steam Generator ←(DRN 03-1964, R13; 06-1062, R15)	897[(a)]
Main Steam Safety Valve First Bank Opening Pressure, psia ←(DRN 05-543, R14; EC-8458, R307)	1117.9
Low Pressurizer Pressure Reactor Trip Setpoint, psia	1560
Low Pressurizer Pressure SIAS Setpoint, psia →(DRN 04-632, R13-B; 05-543, R14)	1560
High Pressure Safety Injection Pump Flow Rate ←(DRN 04-632, R13-B; 05-543, R14)	Table 6.3-6
Time Delay for Actuation of HPSI Flow (with Loss of Offsite Power), sec ←(DRN 00-0551, R10)	30

→(DRN 06-1062, R15; EC-8458, R307)
(a) Corresponds to 10% SG tubes plugged for replacement steam generators.
←(DRN 06-1062, R15; EC-8458, R307)

TABLE 15.6-13a (Sheet 2 of 2) Revision 15 (03/07)

→(DRN 03-1964, R13)

GENERAL SYSTEM PARAMETERS AND INITIAL CONDITIONS
(FOR THE SMALL BREAK LOCA ECCS PERFORMANCE ANALYSIS)

Quantity	Value
→(DRN 00-0551, R10; 04-632, R13-B; 05-543, R14) Atmospheric Dump Valve Opening Pressure, psia ←(DRN 04-632, R13-B)	1040
Safety Injection Tank Pressure, psia ←(DRN 05-543, R14)	584.7

→(DRN 04-632, R13-B; 06-1062, R15)

(a) These quantities correspond to the rod average burnup of the hot rod (500 MWD/MTU) that yields the maximum initial fuel stored energy.

←(DRN 00-0551, R10; 04-632, R13-B; 06-1062, R15)

←(DRN 03-1964, R13)

→(DRN 05-543, R14)

←(DRN 05-543, R14)

TABLE 15.6-13b Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 00-0551)

TABLE 15.6-13c Revision 14 (12/05)

GENERAL SYSTEM PARAMETERS AND INITIAL CONDITIONS
(FOR THE INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE)

Quantity	Value
→(DRN 02-526, R12; 05-543, R14) Reactor Power Level (rated thermal power, plus power measurement uncertainty, plus reactor coolant pump heat), MWt ←(DRN 02-526, R12; 05-543, R14)	3478 [(a)]
Average Linear Heat Generation Rate (102% of Rated), kw/ft	5.6
Peak Linear Heat Generation Rate (PLHGR), kw/ft	15
Gap Conductance at the PLHGR, Btu/hr/ft^2/^{0}F	1486
Fuel Centerline Temperature at the PLHGR, ^{0}F	3683
Fuel Average Temperature at the PLHGR, ^{0}F	2322
Moderator Temperature Coefficient at Initial Density, $\Delta\rho/^0$F	$+0.15 \times 10^{-4}$
System Flow Rate (total), lbm/hr	148×10^6
Core Flow Rate, lbm/hr	142.8×10^6
Initial System Pressure, psia	2250
Cold Leg Temperature, ^{0}F	553
Hot Leg Temperature, ^{0}F	612.2
Low Pressurizer Pressure Reactor Trip Setpoint, psia	1560
Low Pressurizer Pressure SIAS Setpoint, psia	1560
Safety Injection Tank Pressure, psia	615
High Pressure Safety Injection Pump Shutoff Head, psia	1430
Low Pressure Safety Injection Pump Shutoff Head, psia ← (DRN 00-0551)	193

→(DRN 02-526, R12; 05-543, R14)

(a) The first paragraph of Section 15.6.3.4 describes the applicability of the conclusion of the inadvertent opening of the pressurizer safety valve analysis at a power level of 3478 MWt to the extended power uprate power level of 3716 MWt (3735 MWt with a 0.5% power measurement uncertainty).

←(DRN 02-526, R12; 05-543, R14)

TABLE 15.6-14 Revision 307 (07/13)

PEAK CLAD TEMPERATURE AND OXIDATION PERCENTAGES
FOR THE LARGE BREAK ANALYSIS

→(DRN 05-543, R14)

Break	Peak Clad Temperature[a] (F)	Clad Oxidation %	
		Local[b]	Core-Wide[c]
→(EC-9533, R302)			
Break Spectrum Analysis for Peak Cladding Temperature			
→(DRN 06-1062, R15; EC-8458, R307)			
1.0 DEG/PD	2092	12.8	<1
0.8 DEG/PD	2074	12.4	<1
0.6 DEG/PD	2046	11.6	<1
0.4 DEG/PD	2017	8.2	<1
←(DRN 06-1062, R15; EC-8458, R307)			
Case Results for Maximum Cladding Oxidation			
→(EC-8458, R307)			
0.8 DEG/PD	2091	13.0	<1
←(EC-9533, R302; EC-8458, R307)			

[a] Acceptance Criterion is ≤2200 °F

[b] Acceptance Criterion is ≤17%

[c] Acceptance Criterion is ≤1.0%

←(DRN 05-543, R14)

TABLE 15.6-14a Revision 307 (07/13)

→(DRN 00-0551)

PEAK CLADDING TEMPERATURE AND OXIDATION PERCENTAGES FOR SMALL BREAKS

Break, ft^2	Peak Cladding Temperature, °F	Maximum Cladding Oxidation, %	Maximum Core-Wide Cladding Oxidation, %
→(DRN 05-543, R14; 06-1062, R15; EC-9533, R302; EC-8458, R307)			
0.04	1750	6.0	<0.40
→(DRN 04-632, R13-B)	1925		
0.05		11.2	<0.65
←(DRN 04-632, R13-B)			
0.06	1865	4.4	<0.32

←(DRN 00-0551; 05-543, R14; 06-1062; R15; EC-9533, R302; EC-8458, R307)

TABLE 15.6-15 Revision 302 (12/08)

LARGE BREAK SPECTRUM

Break, Size, Type and Location	Abbreviation	Figure
Break Spectrum Analysis		
→(DRN 05-543, R14; EC-9533, R302) 1.0 x double-ended guillotine break in pump discharge leg	1.0 x DEG/PD	15.6-92 through 15.6-95 and 15.6-97 through 15.6-100k
0.8 x double-ended guillotine break in pump discharge leg ←(EC-9533, R302)	0.8 x DEG/PD	15.6-101 through 15.6-104 and 15.6-106 through 15.6-109
0.6 x double-ended guillotine break in pump discharge leg	0.6 x DEG/PD	15.6-110 through 15.6-113 and 15.6-115 through 15.6-118
→(EC-9533, R302) 0.4 x double-ended guillotine Break in pump discharge leg ←(EC-9533, R302)	04. x DEG/PD	15.6-127a through 15.6-127h
Peak clad temperature vs. break area ←(DRN 05-543, R14)		15.6-128

TABLE 15.6-16 Revision 14 (12/05)

VARIABLES PLOTTED AS A FUNCTION OF TIME FOR EACH LARGE BREAK IN THE SPECTRUM

Variables
Core Power
Pressure in center hot assembly node
Leak flow
→(DRN 05-543, R14) Hot assembly flow (below and above hot spot) ←(DRN 05-543, R14)
Hot assembly Quality
Containment Pressure
Mass added to core during reflood
Peak clad temperature

TABLE 15.6-16a Revision 14 (12/05)

VARIABLES PLOTTED AS A FUNCTION OF TIME
FOR EACH SMALL BREAK

Variable
Normalized Total Core Power
Inner Vessel Pressure
Break Flow Rate
Inner Vessel Inlet Flow Rate
→(DRN 00-0551) Inner Vessel Two-Phase Mixture Level ←(DRN 00-0551)
Heat Transfer Coefficient at Hot Spot
Coolant Temperature at Hot Spot
→(DRN 00-0551) Cladding Temperature at Hot Spot ←(DRN 00-0551)

→(DRN 05-543, R14)

ADDITIONAL VARIABLES PLOTTED AS A FUNCTION OF TIME
FOR THE WORST SMALL BREAK

Variable
Steam Generator No. 1 Pressure
Steam Generator No. 2 Pressure
Steam Generator No. 1 Secondary Flow Rate
Steam Generator No. 2 Secondary Flow Rate

←(DRN 05-543, R14)

TABLE 15.6-17 Revision 9 (12/97)

→

ADDITIONAL VARIABLES PLOTTED AS A FUNCTION OF TIME FOR THE WORST LARGE BREAK

Variables
Mid annulus flow
Qualities above and below the core
Core pressure drop
Safety injection flow into intact discharge legs
Water level in downcomer during reflood
Hot spot gap conductance
Local clad oxidation
←
Clad temperature, centerline fuel temperature, average fuel temperature and coolant temperature for hottest node
→
Hot spot heat transfer coefficient
Hot pin pressure
Core bulk channel flowrate

←

TABLE 15.6-18 (Sheet 1 of 4) Revision 14 (12/05)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Core Power Level:	3735 MWt
Containment Leak Rate:	0.50% volume/day (0-24 hrs) 0.25% volume/day (24 hrs-30 days)
Natural Deposition:	
Elemental	0.40/hr
Organic	0
Particulate	Powers 10% Aerosol Decontamination Factor (See Appendix B)
→(DRN 05-1551, R14)	
Primary Containment Volume	2.568E06 ft^3
Sprayed Volume Fraction	0.80
Unsprayed Volume Fraction	0.20
←(DRN 05-1551, R14)	
Spray Fission Product Removal (LBLOCA):	
Elemental	0
Organic	0
Particulate	3.596/hr (until PF = 50) 0.3596/hr (once PF > 50)
Containment Mixing Rate Between Sprayed and Unsprayed Regions:	17, 122 CFM
Maximum Spray Delay Time:	60 seconds
Containment Leakage Pathway:	
Controlled Ventilation Area System (CVAS)	
Filtration (RAB)	54%
Shield Building	40%
Unfiltered Direct Bypass	6%
Core Inventory:	Table 12.2-12
Iodine Chemical Form – Containment Leakage:	
Elemental	4.85%
Organic	0.15%
Particulate	95.0%
Iodine Chemical Form – ESF Liquid Leakage:	
Elemental	97%
Organic	3%
Particulate	0%

←(DRN 04-704, R14)

TABLE 15.6-18 (Sheet 2 of 4) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT

→(EC-5000081470, R301)

Shield Building Ventilation System (SBVS) and Controlled Ventilation Area System (CVAS) Filter Eff.	CVAS	SBVS
Elemental	99%	0%
Organic	99%	0%
Particulate	99%	0%

←(EC-5000081470, R301)

ESF Liquid Leakage Rate: 0.5 gpm

→(DRN 05-1551, R14)

ESF Liquid Leakage Flashing Fraction: 0.10

←(DRN 05-1551, R14)

Control Room Parameters:

→(EC-5000081470, R301)

Volume 168,500 ft^3

←(EC-5000081470, R301)

Recirculation Flow Rate 3800 CFM

Iodine Filter Efficiency 99% (elemental/organic/particulate)

→(DRN 05-1551, R14)

Pressurization Flow 225 CFM (max.)

←(DRN 05-1551, R14)

→(EC-5000081470, R301)

←(EC-5000081470, R301)

Unfiltered Inleakage 100 CFM

Breathing Rate 3.47E-04 m3/sec.

Control Room Occupancy Factors

0 – 24 hours	1.0
24 – 96 hours	0.6
96 hours – 30 days	0.4

Main Control Room X/Q Assumed:

→(EC-5000081470, R301)

Time	Unfiltered In-leakage	Pressurization Flow
0-2 hr	2.77E-03	5.15E-04 *

←(EC-5000081470, R301)

2-8 hr	1.78E-03	3.90E-04 *
8-24 hr	7.22E-04	1.79E-04 *
1-4 days	5.27E-04	1.37E-04 *
4-30 days	4.05E-04	1.08E-04 *

* factor of 4 reduction credited per SRP 6.4.

←(DRN 04-704, R14)

TABLE 15.6-18 (Sheet 3 of 4) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Core Inventory Fraction Released into Containment:

Group	Gap Release Phase	Early In-Vessel Phase
Noble Gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metal	0.00	0.0025
Cerium group	0.00	0.0005
Lanthanides	0.00	0.0002

LOCA Release Phases:

→(DRN 05-1551, R14)

Phase	Start	Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

←(DRN 05-1551, R14)

→(EC-5000081470, R301)

Shine Dose Calculation Assumptions *:

←(EC-5000081470, R301)

Containment Leakage Pathway:	
Controlled Ventilation Area System (CVAS)	
Filtration (RAB)	60%
Shield Building	40%
Unfiltered Direct Bypass	0%
Control Room Unfiltered Inleakage	200 cfm
Liquid Leakage from ESF Systems	0.5 gpm
Flashing Fraction	10% Up to 24 hours 2% Thereafter
Filter Efficiencies	100%
Spray Fission Product Removal (LBLOCA):	
Elemental	20/hr (maximum PF = 200)
Organic	0
Particulate	3.596/hr (until PF = 50) 0.3596/hr (once PF > 50)

Note *: The assumptions used in the filter shine analyses differ slightly from the off-site/control room dose analyses.

←(DRN 04-704, R14)

TABLE 15.6-18 (Sheet 4 of 4) Revision 14 (12/05)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LARGE BREAK LOSS-OF-COOLANT ACCIDENT

Results:

Location	Results	Regulatory Limit
EAB (worst 2 hours)	≤ 25 REM TEDE	25 REM TEDE
LPZ (duration)	≤ 25 REM TEDE	25 REM TEDE
Main Control Room	≤ 5.0 REM TEDE **	5.0 REM TEDE

Note **: Includes filter shine, containment shine, and external cloud shine doses.

←(DRN 04-704, R14)

TABLE 15.6-18A (Sheet 1 of 3) Revision 307 (07/13)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL BREAK LOSS-OF-COOLANT ACCIDENT

Core Power Level:	3735 MWt
→(EC-5000081470, R301) ←(EC-5000081470, R301)	
Fission Product Gap Fractions:	
Iodines	5%
Noble Gases	5%
Alkali metals (Cs & Rb-86)	5%
Fraction of Fuel Rods in Core Failing:	100%
Reactor Building Release Pathway	
→(EC-5000081470, R301) Core Inventory: ←(EC-5000081470, R301)	Table 12.2-12
Containment Leak Rate:	0.50% volume/day (0-24 hours) 0.25% volume/day (24 hours – 30 days)
Natural Deposition:	
Elemental	0.40/hr
Organic	0
Particulate	Powers 10% Aerosol Decontamination Factor
Spray Fission Product Removal:	Not Credited
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	4.85%
Organic	0.15%
Particulate	95%
Secondary Steaming Pathway	
→(EC-5000081470, R301) Core Inventory: ←(EC-5000081470, R301)	Table 12.2-12A
Primary-to-Secondary Leak Rate:	75 gpd per SG
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Particulate	0%
(DRN 04-704, R14) →(EC-40444, R307) Total MSSV/ADV Combined Leakage per Steam Line ←(EC-40444, R307)	280 lb/hr Until Cold Shutdown

TABLE 15.6-18A (Sheet 2 of 3) Revision 301 (09/07)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL BREAK LOSS-OF-COOLANT ACCIDENT

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	168,500 ft^3
←(EC-5000081470, R301)	
Recirculation Flow Rate	3800 CFM
Iodine Filter Efficiency	99% (elemental/organic/particulate)
→(DRN 05-1551, R14)	
Pressurization Flow	225 CFM
←(DRN 05-1551, R14)	
→(EC-5000081470, R301)	
←(EC-2000081470, R301)	
→(DRN 05-1551, R14; EC-5000081470, R301)	
Unfiltered Inleakage	100 CFM
←(DRN 05-1551, R14; EC-5000081470, R301)	
Breathing Rate	3.47E-04 m3/sec.
Control Room Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 hours – 30 days	0.4

Main Control Room X/Q Assumed:

Time	Reactor Building Unfiltered In-Leakage	Reactor Building Pressurization Flow	Secondary Steaming Unfiltered In-leakage	Secondary Steaming Pressurization Flow
→(EC-5000081470, R301)				
0-2 hr	2.77E-03	5.15E-04 *	1.06E-01	3.08E-04 *
2-8 hr	1.78E-03	3.90E-04 *	7.45E-02	2.08E-04 *
8-24 hr	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
←(EC-5000081470, R301)				

* factor of 4 reduction credited per SRP 6.4.

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

→(DRN 05-1551, R14)

0-2 hr Steaming	2-7.5 hr Steaming
←(DRN 05-1551, R14)	
627,512	858,838

←(DRN 04-704, R14)

TABLE 15.6-18A (Sheet 3 of 3) Revision 14 (12/05)

→(DRN 04-704, R14)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL BREAK LOSS-OF-COOLANT ACCIDENT

0-15 min	15-30 min	½-1 hr	1-2 hr	2-4 hr	4-6 hr	6-7.5 hr
3.28	3.77	1.85	6.51	19.32	21.96	21.33

Alkali Metal Source Term Data, Ci Release:

Cs-134	16.016
Cs-136	4.211
Cs-137	8.529
Rb-86	0.029

←(DRN 04-704, R14)

TABLE 15.6-18B Revision 14 (12/05)

→(DRN 04-704, R14)

RADIOLOGICAL CONSEQUENCES OF A SMALL BREAK LOSS-OF-COOLANT ACCIDENT

Location	Secondary Steaming Release Pathway	Reactor Building Release Pathway	Regulatory Limit
EAB (worst 2 hours)	≤ 25	≤ 25	25 REM TEDE
LPZ (duration)	≤ 25	≤ 25	25 REM TEDE
Main Control Room	≤ 5	≤ 5	5.0 REM TEDE

←(DRN 04-704, R14)

➜

TABLE 15.6-19 Revision 9 (12/97)

TABLE DELETED

➜

TABLE 15.6-20 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.6-21

SEQUENCE OF EVENTS FOR AN INADVERTENT

OPENING OF A PRESSURIZER SAFETY VALVE

Time (seconds)	Event	Setpoint or Value
0.0	Valve opens	-
413	Low pressurizer pressure setpoint attained (generates reactor trip signal and SIAS), psia; Loss of Normal AC Power	1560.0
414	CEA'S begin to drop into core	-
418	CEA's fully inserted	-
443	HPSI flow initiated	-
3600	Operator initiates plant cooldown (a)	-
5720	HPSI flow exceeds leak flow (a)	-
6 hours	Shutdown cooling initiated	-

(a) It should be noted that this analysis did not include cooldown initiation at 3600 seconds. Cooldown initiation time is included here to maintain consistency with the LOCA emergency procedures. Had cooldown been credited in the analysis, the HPSI flow would have surpassed the leak flow earlier than 5720 seconds as a result of faster system depressurization.

TABLE 15.6-22

VARIABLE PLOTTED AS A FUNCTION OF TIME
FOR AN INADVERTENT OPENING OF A
PRESSURIZER SAFETY VALVE

Variable	Figure
Normalized Total Core Power	15.6-178
Inner Vessel Pressure	15.6-179
Leak Flowrate	15.6-180
Inner Vessel Inlet Flowrate	15.6-181
Inner Vessel Two-Phase Mixture Height	15.6-182
Heat Transfer Coefficient at Hot Spot	15.6-183
Coolant Temperature at Hot Spot	15.6-184
Clad Surface Temperature at Hot Spot	15.6-185

TABLE 15.6-23

RESULTS OF LOCA ANALYSIS FOR AN

INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

Parameter	Valve
Break Size, ft^2	0.0273
Peak Clad Temperature, °F	928
Maximum Local Clad Oxidation, %	0.001
Maximum Core-Wide Clad Oxidation, %	0.0002

TABLE 15.6-24 Revision 309 (06/16)

→ (DRN 05-543, R14)

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR
TUBE RUPTURE WITH LOOP

Time (sec)	Event	Setpoint or Value
0.0	Tube rupture occurs	---
45	Second charging pump turned on, on pressurizer level error, ft	-0.75
→(LBDCR15-039, R309) 70 ←(LBDCR15-039, R309)	Third charging pump turned on, on pressurizer level error, ft	-1.17
445	Steam Generator ADVs open, psia	980
445	CPC hot leg saturation trip condition reached	13°F
445.4	Trip Breakers Open	
446	CEA's begin to drop	
448	Loss of Offsite Power	---
450	Steam Generator MSSVs open, psia	1085
455	Steam Generator MSSVs close, psia	1041.6
485	SIAS actuated on pressurizer pressure, psia	1560
515	Safety Injection flow begins to enter RCS	---
595	Pressurizer empties	---
600	EFW delivered to Intact Steam Generator	---
≥ 875	Operator takes manual control of the SG ADVs, initiates plant cooldown by steaming through both SG ADV's at a rate of 100 °F/hr	---
≥ 875	Operator initiates EFW flow to unaffected SG	---
≥ 875	Operator initiates auxiliary spray in order to depressurize the RCS below 1000 psia and regain level control in the pressurizer	---
≥ 875	Operator manually controls EFW flow to the intact SG to maintain 68% to 71% WR	---
→ (EC-34230, R306)		
≥ 875	Operator manually controls safety injection, auxiliary spray flow and the pressurizer backup heater output to try to maintain as necessary subcooling (28 °F) and pressurizer level (33% - 60%). Note that the pressurizer backup heaters are not used / activated in the current SGTR analysis of record.	---
←(EC-34230, R306)		
1980	Operator isolates the affected SG	---
23630	Operator opens ADV to the affected SG as needed to maintain level below 94% WR	---
28800	Shutdown cooling entry conditions reached, RCS pressure, psia/Temperature, °F	392/350

← (DRN 05-543, R14)

TABLE 15.6-25 Revision 14 (12/05)

→(DRN 05-543, R14)

Assumptions for 3716 MWt SGTR with LOOP

Parameter	Assumption
Initial Core Power, MWt	3735
Core Inlet Temperature, °F	552
RCS Flowrate, 10^6 lbm/hr	148
Pressurizer Pressure, psia	2090
Pressurizer Level, %	33
SG Pressure, psia	872
SG Level, % NR	26.5
MTC 10^{-4} $\Delta\rho$/°F	-0.2
Doppler Coefficient Multiplier	85
CEA worth for Trip, % $\Delta\rho$	-6.0
SBCS	Inoperative
Feedwater Regulation System	Inoperative
EFS	Automatic
SG ADVs	Automatic
ADV Setpoint, psia	980
SIAS Setpoint, psia	1560

←(DRN 05-543, R14)

TABLE 15.6-26 (Sheet 1 of 2) Revision 307 (07/13)

→(DRN 05-1551, R14)

ASSUMPTIONS
STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER
RADIOLOGICAL CONSEQUENCES CASE

←(DRN 05-1551, R14)
→(DRN 04-704, R14)

Core Power Level:	3735 MWt
RCS Noble Gas Activity:	Table 11.1-2
Core Inventory:	Table 12.2-12A
RCS Initial Activity:	100/Ē μCi/gm
Pre-existing Iodine Spike (PIS):	60 μCi/gm DEI-131
Accident Generated Iodine Spike (GIS):	1.0 μCi/gm DEI-131
Iodine Spiking Factor:	335
Secondary Coolant Initial Activity:	0.1 μCi/gm DEI-131
Fraction of Fuel Rods in Core Failing:	0%
Iodine Chemical Form:	
Elemental	97%
Organic	3%
Particulate	0%
→(DRN 05-645, R14) Primary-to-Secondary Leak Rate (unaffected SG): ←(DRN 05-645, R14)	150 gpd
→(EC-40444, R307) Total MSSV/ADV Combined Leakage per Steam Line ←(EC-40444, R307)	280 lb/hr Until Cold Shutdown
Steaming PF:	100
Steam Releases:	
Affected SG, time of reactor trip to isolation (1980 sec)	139,000 lbm
Affected SG, time of reactor trip to 8 hours	245,600 lbm
Intact SG, time of reactor trip to 2 hours	351,400 lbm
Intact SG, time of reactor trip to 8 hours	910,100 lbm
Duration of Release:	8.0 hours
Control Room Parameters:	
→(DRN 05-645, R14) Volume	168,500 ft^3
←(DRN 05-645, R14) Recirculation Flow Rate	3800 CFM
Iodine Filter Efficiency	99% (elemental/organic/particulate)

←(DRN 04-704, R14)

TABLE 15.6-26 (Sheet 2 of 2) Revision 14 (12/05)

ASSUMPTIONS
STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER
RADIOLOGICAL CONSEQUENCES CASE

→(DRN 04-704, R14)

→(DRN 05-1551, R14)

Pressurization Flow	225 CFM
←(DRN 05-1551, R14)	
	0 CFM (min., 0-8 hours)
Unfiltered Inleakage	100 CFM
Breathing Rate	3.47E-04 m3/sec.
Control Room Occupancy Factors	
0-24 hours	1.0
24-96 hours	0.6
96 hours – 30 days	0.4

→(DRN 05-1551, R14)
Activity Releases for the SGTR (Ci)
←(DRN 05-1551, R14)

	DEI-131 Release		**Noble Gas Release (Table 11.2-1 distribution)**	
→(DRN 05-1551, R14)				
	0-2 hr	8 hr	0-2 hr	8 hr
←(DRN 05-1551, R14)				
→(DRN 05-645, R14)				
PIS	132.33	176.2	28,516	69,925.5
GIS	7.21	49.1	28,516	69,925.5
←(DRN 05-645, R14)				

←(DRN 04-704, R14)

TABLE 15.6-27 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.6-28 Revision 14 (12/05)

RADIOLOGICAL CONSEQUENCES
STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER
RADIOLOGICAL CONSEQUENCES CASE

→(DRN 04-704, R14)

	TEDE Dose	Acceptance Criteria
PIS case:		
EAB (worst two hour dose)	≤ 25	25 Rem TEDE
LPZ (duration)	≤ 25	25 Rem TEDE
Main Control Room	≤ 5	5 Rem TEDE
GIS case:		
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)