15.3 DECREASE IN REACTOR COOLANT FLOW RATE

15.3.1 MODERATE FREQUENCY INCIDENTS

15.3.1.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a partial loss of forced reactor coolant flow classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. A partial loss of forced reactor coolant flow is caused by loss of power to one pump or the loss of power from one pump bus.

The loss of power from one pump bus is more severe since it causes a more rapid reduction of the reactor coolant flow.

15.3.1.1.2 Sequence of Events and Systems Operation

→(DRN 05-543, R14)

A loss of electric power from one pump bus produces a decrease in flow in one cold leg of each coolant loop. This, in turn produces a reduction of coolant flow through the reactor core. Since the remaining pumps continue to produce a positive flow head, the core flow decreases more slowly than during a total loss of forced reactor coolant flow. The reduced coolant flow through the core causes an increase in the average coolant temperature in the core and a decrease in the margin to DNB. The coastdown of the involved pumps results in the generation of a CPCS low pump speed trip. This trip, in combination with the initial thermal margin prevents the minimum DNBR from violating the SAFDL at any time during the transient.

←(DRN 05-543, R14)

Since the power operating limit calculated by the COLSS and reactor trip signal generated by the CPCs are based on the more rapid loss of thermal margin resulting from a total loss of forced reactor coolant flow, the consequences of a partial loss of forced reactor coolant flow are no more adverse than those following a total loss of forced reactor coolant flow, which is described in Subsection 15.3.2.1. The consequences of a single malfunction of an active component or system following a partial loss of forced reactor coolant flow are discussed in Subsection 15.3.2.2.

15.3.1.1.3 Core and System Performance

The core and system performance parameters following a partial loss of forced reactor coolant flow are no more adverse than those following a total loss of forced reactor coolant flow, which is described in Subsection 15.3.2.1.

15.3.1.1.4 Barrier Performance

The barrier performance parameters following a partial loss of forced reactor coolant flow would be less adverse than those following a total loss of forced reactor coolant flow, which is described in Subsection 15.3.2.1.

15.3.1.1.5 Radiological Consequences

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

15.3.2 INFREQUENT INCIDENTS

15.3.2.1 Total Loss of Forced Reactor Coolant Flow

15.3.2.1.1 Identification of Causes and Frequency Classification

→ (DRN 05-543, R14; 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)
The estimated frequency of a total loss of forced reactor coolant flow classifies it as an infrequent incident as defined in ANSI N18.2 . A total loss of forced reactor coolant flow is caused by simultaneous loss of electric power to all four reactor coolant pumps. During normal power operation, electrical power is supplied by the main generator. Power is provided through two buses each bus supplies the power to one pump in each loop. In the event of a loss of the main generator, automatic fast transfer to preferred offsite ac power is actuated, and there is no significant change in reactor coolant flow. However, no credit is taken for the automatic transfer to offsite power, and the transient is postulated to occur as a result of a loss of the main generator. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an evaluation using the revised SCRAM curve times presented in Table 15.0-5.
← (DRN 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)

15.3.2.1.2 Sequence of Events and Systems Operation
→ (EC-13881, R304)
A loss of power to all reactor coolant pumps produces a reduction of coolant flow through the reactor core. The reduction in coolant flow rate causes an increase in the average coolant temperature in the core and a decrease in the margin to DNB. A reactor trip is generated by the core protection calculators due to low RCP shaft speed as described in Section 7.2 to prevent the minimum DNBR calculated with the WSSV-T and ABB-NV correlations from decreasing below the DNBR limit at any time during the transient. The reactor trip produces an automatic turbine trip. Following a turbine trip, offsite power is available to provide ac power to the auxiliaries. The case of loss of all normal ac power is discussed in Subsection 15.2.1.4. Due to the increase in the secondary system pressure following turbine trip, the main steam safety valves will lift to control secondary pressure. The combination of the action of the main steam safety valves and the emergency feedwater system will maintain plant conditions until the operators take action to perform an orderly cooldown. Operator action is assumed at 30 minutes into the event to begin a controlled cooldown.
← (EC-13881, R304)

The consequences of a single malfunction of an active component or system following a partial loss of forced reactor coolant flow are discussed in Subsection 15.3.2.2.
← (DRN 05-543, R14)

→(DRN 05-543, R14)
Table 15.3-1 gives the sequence of events that would occur during a total loss of reactor coolant flow, assuming Main Steam Safety Valve operation for a representative set of initial conditions that cause the most adverse consequences.
←(DRN 05-543, R14)

15.3.2.1.3 Core and System Performance

15.3.2.1.3.1 Mathematical Models

→(DRN 05-543, R14)
The method used to analyze the total loss of forced reactor coolant flow is the space-time method described in Reference 4. The flow coastdown associated with the loss of forced circulation is determined with the CENTS code. This flow behavior is an input into the HERMITE code (Reference 3) providing spatial detail of the core response. The CETOP code is then used to determine the approach to DNBR of the limiting channel in the core.
←(DRN 05-543, R14)

15.3.2.1.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a total loss of forced reactor coolant flow are discussed in Section 15.0. These parameters, which are unique to the analysis discussed below, are listed in Table 15.3-2.

→(DRN 05-543, R14)

The principal process variables that determine thermal margin to DNB in the core are monitored by the COLSS. The COLSS computes a power-operating limit which ensures that the thermal margin available in the core is equal to or greater than that needed to cause the minimum DNBR to remain greater than the DNBR limit. The COLSS is described in Section 7.7. The initial conditions chosen for the analysis presented in this section are one of a very large number of combinations within the reactor operating envelope given in Table 15.0-4 which would provide the minimum thermal margin required by the COLSS power operating limit. The consequences following a total loss of forced reactor coolant flow initiated from any of these combinations of conditions would be no more adverse than those presented herein.
←(DRN 05-543, R14)

15.3.2.1.3.3 Results

→(DRN 05-543, R14)
The dynamic behavior of the relevant NSSS parameters following a total loss of forced reactor coolant flow is presented on Figures 15.3-1 through 15.3-10.

→(DRN 05-543, R14)

←(DRN 05-543, R14)

The reduced primary coolant flow causes:

a) A reduction in margin to DNB

b) An increase in average reactor coolant temperature in the core

c) An increase in primary system pressure

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

The CPC low RCP shaft speed trip setpoint is reached at 0.49 seconds and the CEAs begin dropping into the core at 1.39 seconds. The minimum DNBR in the core occurs at 2.80 seconds and is greater than the DNBR limit.

← (LBDCR 15-039, R309)

Due to the turbine trip that occurs on reactor trip, secondary pressure increases, opening the steam generator safety valves.

After the steam generator safety valves close, the turbine bypass valves continue to relieve steam to the condenser until the operator takes manual control. Even if initiation of operator control is delayed until 30 minutes after the event, the only steam released to the atmosphere is that which passes through the steam generator safety valves.

← (DRN 05-543, R14)

The maximum RCS and secondary pressures do not exceed 110 percent of the design pressures following a total loss of forced reactor coolant flow, thus ensuring the integrity of the RCS and Main Steam System. The minimum DNBR is greater than the DNBR limit, which ensures that the specified acceptable fuel design limit is not violated.

15.3.2.1.4 Barrier Performance

15.3.2.1.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.3.2.1.3.

15.3.2.1.4.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.3.2.1.3.

15.3.2.1.4.3 Results

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

The maximum RCS pressure reached during the transient is 2315 psia. The steam released to the atmosphere through the steam generator safety valves is no greater than that following a loss of all normal ac power, discussed in Subsection 15.2.1.4.

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

15.3.2.1.5 Radiological Consequences

The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in Subsection 15.1.2.4.

15.3.2.2 Partial Loss of Forced Reactor Coolant Flow With a Concurrent Single Failure of an Active Component

It is not possible to identify any credible single failure of an active component which would affect the DNBR during the first two to four seconds of the transient (the most limiting time period) and cause the partial loss of forced reactor coolant flow to produce consequences more severe than those following a single reactor coolant pump shaft seizure, which is presented in Subsection 15.3.3.1.

15.3.3 LIMITING FAULTS

15.3.3.1 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

15.3.3.1.1 Identification of Causes and Frequency Classification

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

The estimated frequency of a single reactor coolant pump shaft seizure or reactor coolant pump sheared shaft classifies it as a limiting fault incident as defined in ANSI N18.2.[1] A single reactor coolant pump shaft seizure can only be caused by mechanical failure of the pump shaft. This might be produced by a manufacturing defect. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an evaluation using the revised SCRAM curve times presented in Table 15.0-5.

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

15.3.3.1.2 Sequence of Events and Systems Operation

Following seizure of a Reactor Coolant Pump (RCP) shaft or a sheared reactor coolant pump shaft, the core flow rapidly decreases to the value that would occur with only three reactor coolant pumps operating. The reduction in coolant flow causes an increase in the average coolant temperature in the core and may produce a departure from nucleate boiling (DNB) in some portions of the core. For an RCP shaft seizure, a low DNBR reactor trip is generated by the core protection calculators (CPCs) due to low RCP shaft speed (see Section 7.2). For an RCP sheared shaft, no DNBR trip would be generated since the RCP shaft continues to spin and the CPCs perceive full flow. A reactor trip is generated on reactor coolant low flow which occurs as the coastdown decreases the steam generator primary pressure drop to the low flow differential pressure setpoint. The reactor trip produces an automatic turbine/generator trip. Offsite power is assumed to be lost, resulting in coastdown of the remaining three RCPS. The steam generator safety valves open to relieve steam and remove decay heat. After 30 minutes the operator initiates a cooldown to shutdown cooling conditions. Table 15.3-3 gives the sequence of events following a single reactor coolant pump shaft seizure/sheared shaft.

15.3.3.1.3 Core and System Performance

15.3.3.1.3.1 Mathematical Models

→(DRN 05-543, R14)

The method used to analyze the single reactor coolant pump shaft seizure/sheared shaft is the space-time method described in Reference 4. The flow coastdown associated with the loss of force circulation is determined with the CENTS code. This flow behavior is an input into the HERMITE code (Reference 3), which provides spatial detail of the core response. The CETOP or TORC code in then used to determine the approach to DNBR of the limiting channel in the core.

←(DRN 05-543, R14)

15.3.3.1.3.2 Input Parameters and Initial Conditions

→(DRN 00-119)

The input parameters and initial conditions used to analyze the NSSS response to a single reactor coolant pump shaft seizure/sheared shaft are discussed in Section 15.0. Those parameters that are unique to the analysis discussed below are listed in Table 15.3-7.

←(DRN 00-119)

Of the two events, the single RCP shaft seizure and sheared shaft, the single RCP sheared shaft is more limiting with respect to time to generate a reactor trip signal. This is due to the longer time to reach the differential pressure low flow trip setpoint credited for the sheared shaft compared to the rapid CPC pump speed trip generated during the RCP seized shaft. The longer times to generate a reactor trip results in a larger power-to-flow mismatch at the time of minimum DNBR for the sheared shaft event. This analysis compared the RCP sheared shaft event with the RCP seized rotor event to determine which event would produce the most adverse fuel performance.

→(DRN 05-543, R14)

←(DRN 05-543, R14)

The principal process variables that determine thermal margin to DNB in the core are monitored by the COLSS. The COLSS computes a power operating limit that ensures that the thermal margin available in the core is equal to or greater than that needed to cause the minimum DNBR to remain greater than the DNBR limit for the limiting anticipated operational occurrence. The single reactor coolant pump seized/sheared shaft may result in a minimum DNBR below the DNBR limit due to the rapid decrease in flowrate. This could result in some fuel damage and corresponding radiological releases.

The initial conditions chosen for the analysis presented in this section are one of an infinite number of combinations within the reactor operating envelope (see Table 15.0-4) which would provide the minimum thermal margin required by the COLSS power operating limit. The case presented was chosen to maximize calculated fuel damage. The consequences following a single reactor coolant pump seized/sheared shaft initiated from any one of these combinations of conditions would be no more adverse than those presented herein.

15.3.3.1.3.3 Results

→(DRN 05-543, R14)

The dynamic behavior of the relevant NSSS parameters following a reactor coolant pump seized/sheared shaft are shown on Figures 15.3-11 through 15.3-18a.

←(DRN 05-543, R14)

→(DRN 00-592)

The rapid reduction in primary coolant flow causes an increase in the average coolant temperature in the core, a corresponding reduction in the margin to DNB, and an increase in the primary system pressure. For the sheared shaft, reactor trip occurs when the differential pressure low flow setpoint is reached.

←(DRN 00-592)

→ (DRN 00-592; 05-543, R14; 05-1551, R14)
At 1.50 seconds the turbine/generator is tripped, resulting in a loss of offsite power and subsequent coastdown of the remaining three RCPs. This causes a further decrease in the reactor coolant flow. The CEAs begin to drop into the core in response to the trip at 2.10 seconds.
← (DRN 00-592; 05-1551, R14)

→ (EC-13881, R304; LBDCR 15-039, R309))
After the turbine/generator trip, RCS heat is removed by the release of steam through the steam generator safety valves. At 30 minutes it is assumed that the operator initiates a controlled system cooldown using the atmospheric steam dump valves and emergency feedwater. This event results in a transient minimum DNBR of 1.131 at 3.50 seconds. The percentage of fuel pins calculated to experience DNB is less than 15 percent. The calculation of the number of failed fuel pins used the statistical convolution method described in Reference 4 of Section 15.0. All fuel pins that experience DNB are conservatively assumed to fail for radiological release calculations.
← (EC-13881, R304; LBDCR 15-039, R309)

The maximum RCS and secondary pressures do not exceed 110 percent of the design pressures following a single reactor coolant pump seized/sheared shaft, thus assuring the integrity of the RCS and Main Steam System. The minimum DNBR is less than the DNBR limit, but no more than 15.0 percent of the fuel rods will experience DNB.
←(DRN 05-543, R14)

15.3.3.1.4 Barrier Performance

15.3.3.1.4.1 Mathematical Model
→(DRN 04-704, R14)
For purposes of radiological analyses, the analysis model for the RCP Seized Rotor/Sheared Shaft is very similar to that for the Control Element Assembly (CEA) Ejection event described in Subsection 15.4.3.2.5.
←(DRN 04-704, R14)

15.3.3.1.4.2 Input Parameters and Initial Conditions
→(DRN 04-704, R14)
For the purpose of the RCP Seized Rotor/Sheared Shaft analysis, the maximum assumed fuel failure is 15% of the fuel rods in the core experiencing Departure from Nucleate Boiling (DNB). The non-LOCA gap fractions specified in Table 3 of RG 1.183 (Footnote 11) are conservatively selected for use in the RCP Seized Rotor/Sheared Shaft analysis to provide the most conservative set of results. These gap fractions are 10% for iodines and noble gases and 12% for alkali metals. Thus, this analysis conservatively applies the larger gap fractions required for CEA Ejection rather than the smaller gap fractions in the table itself which would apply for an RCP Seized Rotor/Sheared Shaft event.
→(EC-40444, R307)
Conservatively, all the iodine, alkali metal and noble gas activity due to the postulated RCP Seized Rotor/Sheared Shaft accident is assumed to be in the primary coolant when determining the dose consequences due to primary-to-secondary SG leakage and subsequent secondary steaming. As the loss of offsite power will cause loss of condenser availability, the operators will use the atmospheric dump valves to achieve plant cooldown. Releases, for cooldown purpose from ADVs, are assumed to be terminated once shutdown cooling is initiated (no further steam releases would occur due to cooldown to cold shutdown conditions). However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions. This is consistent with the guidelines provided in RG 1.183, Table 6. For this event, a primary-to-secondary SG leakage of 150 gpd is assumed.
←(EC-40444, R307)

For the secondary steaming release pathway, the iodine releases from the SG to the environment are assumed to be 97% elemental iodine and 3% organic iodine. This is consistent with the guidelines provided in Appendix G of RG 1.183.
← (DRN 04-704, R14)

→(DRN 04-704, R14)

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

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

→(EC-5000081470, R301)

The RCP Seized Rotor/Sheared Shaft dose model for secondary steaming release pathway conservatively assumes a constant unfiltered in-leakage of 100 CFM to the Main Control Room for the entire duration of the secondary steaming release (7.5 hours).

←(EC-5000081470, R301)

→(DRN 00-119)

The input parameters and assumptions for the RCP Seized Rotor/Sheared Shaft analysis are listed in Table 15.3-4.

←(DRN 00-119; 04-704, R14)

15.3.3.1.4.3 Results

→(DRN 00-119; 04-704, R14; EC-5000081470, R301)

The radiological consequences in terms of Rem TEDE are listed below based on an assumed 15% fuel failure for an assumed primary-to-secondary SG leakage of 150 gpd and a control room unfiltered in-leakage of 100 CFM.

←(EC-5000081470, R301)

	Dose Results	**Acceptance Criteria**
EAB (worst two hour dose)	≤ 2.5	2.5 Rem TEDE
LPZ (duration)	≤ 2.5	2.5 Rem TEDE
MCR	≤ 5.0	5 Rem TEDE

→(EC-5000081470, R301)

Thus, the radiological consequences for the RCP Seized Rotor/Sheared Shaft are < 2.5 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR, based on 15% fuel failure, a 150 gpd primary-to-secondary leak rate per SG and maximum control room unfiltered in-leakage of 100 CFM.

←(DRN 00-119; 04-704, R14; EC-5000081470, R301)

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

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

SECTION 15.3 REFERENCES

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

2. "CE Methods for Loss of Flow Analysis", CENPD-183 Amendment 1, June 1980.

3. "HERMITE Space-Time Kinetics", CENPD-188-A, July 1975.

4. Combustion Engineering Standard Safety Analysis Report (CESSAR), Appendix 15-A, Docket No. STN 50-470, December 1975.

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

TABLE 15.3-1 Revision 309 (06/16)

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

SEQUENCE OF EVENTS FOR THE TOTAL LOSS OF FORCED REACTOR COOLANT FLOW

Time (sec)	Event	Setpoint or Value
0.0	Loss of ac power to all reactor coolant pumps	---
0.49	Low RCP shaft speed trip condition	96.5% of initial shaft speed
0.78	Reactor trip breakers open	---
1.38	CEAs begin to drop	---
2.60	Minimum WSSV-T / ABB-NV DNBR	≥1.24
7.9	Maximum RCS pressure, psia	2415
183.5*	Steam generator safety valves open, psia	1117
183.5*	Maximum steam generator pressure, psia	1117
212.8*	Steam generator safety valves close, psia	1062

* These are typical values for the loss of forced RCS flow event.

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

TABLE 15.3-2 Revision 309 (06/16)

ASSUMED INITIAL CONDITIONS FOR THE TOTAL LOSS OF FORCED REACTOR COOLANT FLOW

→ (DRN 02-526, R12; 05-543, R14; LBDCR 15-039, R309)

Parameter	Assumption
Initial Core Power Level, MWt	3735
← (DRN 02-526, R12) Core Inlet Temperature, °F	533
Pressurizer Pressure, psia	2310
RCS Flowrate, 10^6 lb_m /hr	183.1
Pressurizer Level, **	44
SG Pressure, psia **	741
SG Level, % NR **	67.2
MTC, $x10^{-4} \Delta\rho/°F$	0.0
Doppler Coefficient Multiplier	0.85
Kinetics	Maximum β
CEA worth for trip, $\%\Delta\rho$	-5.0

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

← (DRN 02-526, R12; 05-543, R14; LBDCR 15-039, R309)

TABLE 15.3-3 Revision 309 (06/16)

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

Sequence of Events for the RCP Sheared Shaft Event
With a Loss of AC Power at Time of Trip

Time (sec)	Event	Setpoint or Value
0.0	Single reactor coolant pump Sheared shaft	---
0.80	Low RCS flow trip condition reached	0.6 of initial
1.50	Low RCS flow trip signal generated	---
1.50	Turbine generator trip; Loss of offsite power occurs	---
2.10	CEAs begin to drop into core	---
3.50	Minimum WSSV-T / ABB-NV DNBR	1.131
8.45 *	RCS maximum pressure (psia)	2442
113.6 *	Unaffected loop steam generator Safety valves open (psia)	1117
113.6 *	Affected loop steam generator Safety valves open (psia)	1117
113.6 *	Maximum steam generator pressure (psia)	1118
1800	Atmospheric dump valves opened by Operator to initiate plant cooldown	---

* These are typical values that would occur for the RCP Sheared Shaft with LOAC event at current power conditions. The event was only analyzed for the time of interest (i.e., Minimum DNBR within 5 seconds after event initiation). Beyond this time, typical values are given.

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

→(DRN 00-119)

TABLE 15.3-4 (Sheet 1 of 2) Revision 307 (07/13)

←(DRN 00-119)

→(DRN 04-704, R14)

ASSUMPTIONS USED FOR RCP SEIZED ROTOR/SHEARED SHAFT RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
Core Inventory:	FSAR Table 12.2-12A
Fission Product Gap Fractions:	
Iodines	10%
Noble Gases	10%
Alkali metals (Cs & Rb-86)	12%
Fraction of Fuel Rods in Core Failing (maximum)	15%
Secondary Steaming Pathway	
Primary-to-Secondary Leak Rate:	150 gpd per SG
→(EC-40444, R307) Total MSSV/ADV Combined Leakage per Steam Line ←(EC-40444, R307)	280 lb/hr Until Cold Shutdown
Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Particulate	0%
Steaming PF (Iodine and Alkali Metals):	
0-30 minutes	10
> 30 minutes	100
Duration of Release:	7.5 hours

→(DRN 05-1551, R14)

Main Control Room χ/Q Assumed:

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

←(EC-5000081470, R301)

* factor of 4 reduction credited per SRP 6.4.

←(DRN 05-1551, R14)

←(DRN 04-704, R14)

TABLE 15.3-4 (Sheet 2 of 2) Revision 14 (12/05)

→(DRN 04-704, R14)

ASSUMPTIONS USED FOR RCP SEIZED ROTOR/SHEARED SHAFT RADIOLOGICAL ANALYSIS

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

→(DRN 05-1551, R14)

0-2 hr Steaming	2-7.5 hr Steaming
609,744	858,838

←(DRN 05-1551, R14)

0-15 min	15-30 min	½-1 hr	1-2 hr	2-4 hr	4-6 hr	6-7.5 hr
2.70	3.54	1.73	6.03	17.73	23.16	19.56

Alkali Metal Source Term Data, Ci Released:

Cs-134	18.506
Cs-136	4.866
Cs-137	9.855
Rb-86	0.035

←(DRN 04-704, R14)

→(DRN 00-119)

TABLE 15.3-4a Revision 14 (12/05)

←(DRN 00-119)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

Table 15.3-4b has been deleted.

→ (DRN 00-119)

TABLE 15.3-5 Revision 11 (05/01)

TABLE 15.3-5 HAS BEEN DELETED

← (DRN 00-119)

Table 15.3-5a has been deleted.

→ (DRN 00-119)

TABLE 15.3-6 Revision 11 (05/01)

TABLE 15.3-6 HAS BEEN DELETED

← (DRN 00-119)

→ (DRN 02-526, R12; DRN 05-543, R14; EC-13881, R304; LBDCR 15-039, R309)

Assumed Initial Conditions for the RCP Sheared Shaft Event
With a Loss of AC Power at Time of Trip

Parameter	Assumption
Initial Core Power (MWth)	3735
Initial Core Inlet Temperature (°F)	560 *
Initial Pressurizer Pressure (psia)	2090 *
Initial RCS Flow (10^6 lbm/hr)	167.6
Initial Thermal Margin to DNBR SAFDL (%)	123
Initial Pressurizer Level (ft / % cylindrical height) **	16.1 / 54.0%
Initial SG Water Level (ft / % narrow range) **	37.0 / 71.2%
Initial SG Pressure (psia) **	734.0
MTC ($x10^{-4}$ $\Delta\rho$/°F)	0.00
FTC (multiplier)	0.85 (least negative)
Kinetics	Maximum β
CEA Worth at Trip – WRSO (%$\Delta\rho$)	5.0

* This value is set such that the initial conditions meet the initial thermal margin value.

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

← (DRN 02-526, R12; DRN 05-543, R14; EC-13881, R304; LBDCR 15-039, R309)