15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 MODERATE FREQUENCY INCIDENTS

15.4.1.1 Uncontrolled CEA Withdrawal from Subcritical Conditions

15.4.1.1.1 Identification of Causes and Frequency Classification

→(DRN 02-1479, R12; 05-543, R14; 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)

The estimated frequency of a control element assembly (CEA) withdrawal from subcritical conditions classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. An uncontrolled withdrawal of CEAs is assumed to occur as a result of a single failure in the control element drive mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), Reactor Regulating System, or operator error. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.

←(DRN 02-1479, R12; 05-543, R14; 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)

15.4.1.1.2 Sequence of Events and Systems Operation

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase with corresponding increases in reactor coolant temperatures and Reactor Coolant System (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the approach to specified fuel design limits and to RCS and secondary system pressure limits, thereby requiring the protective action of the Reactor Protection System (RPS).

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at subcritical conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. Depending on the initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high logarithmic power level trip, high pressurizer pressure trip, or the removal of the CPC bypass while in a tripped condition. The secondary system pressure increases following reactor trip and is limited by the steam generator safety valves.

→ (DRN 02-1479, R12)

Operating procedures provide assurance that criticality cannot occur during withdrawal of shutdown CEAs.

← (DRN 02-1479, R12)

Table 15.4-1 gives the sequence of events for the limiting CEA withdrawal transient from subcritical conditions discussed in Subsection 15.4.1.1.3.

15.4.1.1.3 Core and System Performance

a) Mathematical Model

→ (DRN 05-543, R14)

The nuclear steam supply system (NSSS) response to a CEA withdrawal from subcritical conditions was simulated using the CENTS computer program described in Section 15.0.

← (DRN 05-543, R14)

→ (EC-13881, R304)

The thermal margin on DNBR in the reactor core was simulated using the CETOP computer program with the WSSV-T and ABB-NV correlations described in Chapter 4.

← (EC-13881, R304)

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a CEA withdrawal from subcritical conditions are discussed in Section 15.0. In particular, those parameters which were unique to the analysis discussed below are listed in Table 15.4-2.

The initial conditions and NSSS characteristics assumed in this analysis have been identified as a limiting set of conditions allowed by the limiting conditions for operation (LCOS) in terms of providing the nearest and most rapid approach to the fuel design limits.

→(DRN 02-1479, R12)

An uncontrolled CEA withdrawal from subcritical conditions may result from the inadvertant withdrawal of either shutdown or regulating CEA banks. Additionally, the subcritical withdrawal events could potentially occur with plant configurations not normally designed for power operations (i.e., less than four reactor coolant pumps operating). Protection for the subcritical CEA withdrawal event is provided by the High Logarithmic Power Level trip (analytical setpoint of 4.4% power), the automatic removal of the Core Protection Calculators (CPC) bypasses (nominally at 10^{-4} % power in the safety analysis), or the High Pressurizer Pressure trip (at 2422 psia in the safety analysis).

→(DRN 05-543, R14; 05-1551, R14)

Input parameters employed in the evaluation of the Uncontrolled CEA Withdrawal from Subcritical Conditions include:

←(DRN 05-1551, R14)

- The least negative Doppler was assumed.
- A minimum delayed neutron fraction and neutron lifetime consistent were assumed.
- A CEA coil decay time of 0.6 seconds was assumed followed by negative reactivity insertion proportional to the CEA position post trip. A stuck rod was assumed.
- A high logarithmic power level trip setpoint of 4.4% of rated power and a response time of 0.4 seconds were assumed in the analysis. This timing is a conservative assumption with respect to the CPC delay times and nuclear instrumentation response times.
- An initial power fraction of $1x10^{-8}$% of rated thermal power was assumed.
- A source strength of $1x10^{-11}$ of rated thermal power was assumed.
- An initial K_{eff} of 0.91 was assumed.

Initial RCS pressure is chosen to avoid a trip on high pressurizer pressure and to provide the minimum thermal margin.

←(DRN 05-543, R14)

→(DRN 02-1479, R12)

←(DRN 02-1479, R12)

→(DRN 05-543, R14)

c) Results

Typical dynamic behavior of important NSSS parameters following a CEA withdrawal from subcritical conditions is presented on Figures 15.4-1 through 15.4-7a. The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core causing a sudden increase in core power and in the core heat flux around the time of a trip. This results in increasing temperature and pressure which, together with a top peaked axial power distribution, produce the nearest and most rapid approach to the specified acceptable fuel design limit on DNBR. Since the transient is initiated at subcritical conditions, moderator feedback does not contribute to any appreciable extent to the transient. At approximately 593 seconds into the transient, a reactor trip is actuated on high logarithmic power, and the CEAs begin dropping into the core at 594 seconds. (These

←(DRN 02-1479, R12; 05-543, R14)

→ (DRN 02-1479, R12; 04-1098, R14; EC-13881, R304; LBDCR 15-039, R309)

times and values are based on a 4.4% analytical setpoint.) This terminates the transient with a minimum DNBR of ≥ 1.24 in the hot channel. The peak linear heat generation rate during the transient exceeds the steady state value of 21 Kw/ft corresponding to the fuel melt limit, however, a detailed deposited energy calculation has demonstrated that the fuel centerline temperatures do not exceed 4663°F.

← (DRN 04-1098, R14; EC-13881, R304; LBDCR 15-039, R309)

The highest RCS pressure produced by this event is well below 110 percent of the design pressure, or 2750 psia.

← (DRN 02-1479, R12)

15.4.1.1.4 Barrier Performance

a) Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.4.1.1.3.

b) Input Parameters, Initial Conditions and Results

In Subsection 15.4.1.2.4, it was determined that the most adverse CEA withdrawal event, in terms of degradation in barrier performance, is one initiated from low power conditions such that the reactivity addition rate combined with the natural plant feedback mechanisms result in a new steady state, not tripped condition.

The fuel performance conditions of the CEA withdrawal from subcritical conditions is combined with barrier performance and its associated steam releases to determine the radiological consequences.

15.4.1.1.5 Radiological Consequences

→ (DRN 04-704, R14)

The radiological consequences due to steam releases from the secondary system are less severe than those from the inadvertent opening of the atmospheric dump valve with a loss of offsite power, Subsection 15.1.2.4.5.

← (DRN 04-704, R14)

15.4.1.2 Uncontrolled CEA Withdrawal from Low Power Conditions

15.4.1.2.1 Identification of Causes and Frequency Classification

→ (DRN 02-1479, R12; 05-543, R14; 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)

The estimated frequency of a control element assembly (CEA) withdrawal from low power conditions classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. An uncontrolled withdrawal of CEAs is assumed to occur as a result of a single failure in the control element drive mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), Reactor Regulating System, or operator error. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.

← (DRN 02-1479, R12; 05-543, R14; 06-1062, R15; EC-8458, R307; LBDCR 15-039, R309)

15.4.1.2.2 Sequence of Events and Systems Operation

The withdrawal of CEAs from low power conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase with corresponding increases in reactor coolant temperatures and Reactor Coolant System (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the approach to specified fuel design limits and to RCS and secondary system pressure limits, thereby requiring the protective action of the Reactor Protection System (RPS).

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at lower power conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. Depending on the initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a variable overpower trip, high pressurizer pressure trip, a low departure from nucleate boiling ratio (DNBR) trip or a high local power density trip. The secondary system pressure increases following reactor trip and is limited by the steam generator safety valves.

→(DRN 05-543, R14)

Table 15.4-3 gives the sequence of events for the limiting CEA withdrawal transient at low power as discussed in Subsection 15.4.1.2.3.

←(DRN 05-543, R14)

15.4.1.2.3 Core and System Performance

→(DRN 05-543, R14)

a) Mathematical Model

→(EC-13881, R304)

The nuclear steam supply system (NSSS) response to a CEA withdrawal from low power conditions was simulated the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the CETOP computer program with the WSSV-T and ABB-NV correlations described in Chapter 4.

←(DRN 05-543, R14; EC-13881, R304)

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a CEA withdrawal from low power conditions are discussed in Section 15.0. In particular, those parameters which were unique to the analysis of fuel performance discussed below are listed in Table 15.4-4.

→(DRN 05-543, R14)

The initial conditions and NSSS characteristics assumed in this analysis have been identified as a limiting set of conditions allowed by the limiting conditions for operation (LCOS) in terms of providing the nearest and most rapid approach to the fuel design limits. These initial conditions are as follows:

←(DRN 05-543, R14)

Power	10^{-6} percent

→(DRN 02-1479, R12; 05-543, R14)

Temp	552°F
Flow	148 x 10^6 lbm/hr gpm

←(DRN 02-1479, R12)

Pressure	2090 psia

The initial RCS pressure is chosen to be the lowest allowable pressure, since this delays actuation of the high pressurizer trip (maximum trip point of 2422 psia including uncertainties). The initial core average axial power distribution assumed in the analysis corresponds to an axial shape index (ASI) which was varied between 0.0 to -0.9. This is outside the envelope of Table 15.0-4. The worst ASI was found to be at -0.6 with a one pin radial peaking factor of 3.17 including uncertainties. This radial peaking factor is the highest radial peak expected for any CEA configuration at any time in core life.

←(DRN 05-543, R14)

Parametric analyses have shown that an initial power level of 10^{-6} percent results in the nearest and most rapid approach to fuel design limits during the CEA withdrawal transient. Transients initiated from a power level just above the nominal high logarithmic power level trip bypass setting of 10^{-4} percent power are terminated by a high pressurizer pressure trip, a variable overpower trip, a low DNBR trip, or a high local power density trip at a much later time. At power levels above 10^{-4} percent, reactivity feedback mechanisms prevail and dampen the severity of the transient.

→(DRN 02-1479, R12)

The most positive moderator temperature coefficient of +0.5 x 10^{-4} $\Delta\rho$/°F is assumed for this analysis. The regulating CEAs are initially in the fully inserted position when the withdrawal is initiated. Based on calculated CEA worths and the maximum CEA withdrawal rate of the CEA drive system, the rate assumed is conservative. For this analysis, the reactivity insertion is 1.6 x 10^{-4} $\Delta\rho$/sec., i.e., 3.20 x 10^{-4} percent $\Delta\rho$ per inch of rod. This rate corresponds to approximately twice the largest insertion rate expected from the sequential withdrawal of the CEA groups with forty percent overlap at the maximum speed of thirty inches per minute.

←(DRN 02-1479, R12)

→(DRN 05-543, R14)

A conservative CPCS VOPT setpoint of 35% of rated power and a response time of 0.429 seconds were assumed. This timing is a conservative assumption with respect to the CPC delay times and nuclear instrumentation response times.

c) Results

The dynamic behavior of important NSSS parameters following a CEA withdrawal from low power conditions is presented on Figures 15.4-8 through 15.4-11d.

←(DRN 05-543, R14)

→(DRN 02-1479, R12; 05-543, R14; 04-1098, R14; EC-13881, R304; LBDCR 15-039, R309)

The withdrawal of CEAs from low power conditions (10^{-6} percent of full power) adds reactivity to the reactor core and causes both the core power level and the core heat flux to increase. This results in increasing temperature and pressure which, together with a top peaked axial power distribution, produce the nearest and most rapid approach to the specified acceptable fuel design limit on DNBR. Since the transient is initiated at low power levels, moderator feedback does not contribute to any appreciable extent to the transient. At 24.2 seconds into the transient, a reactor trip is actuated on CPC Variable Overpower and the CEAs begin dropping into the core at 25.3 seconds. This terminates the transient with a minimum DNBR of greater than 1.24 in the hot channel. Fuel centerline temperatures do not exceed 4663°F at any point in the transient.

← (DRN 02-1479, R12; 04-1098, R14; EC-13881, R304; LBDCR 15-039, R309)

The highest RCS pressure produced by an event in the category of reactivity and power distribution anomalies is less than 2750 psia.

← (DRN 05-543, R14)

15.4.1.2.4 Barrier Performance

→(DRN 05-543, R14)

The mathematical model used for evaluation of barrier performance is described in Subsection 15.4.1.2.3. The analysis of Subsection 15.4.1.2.3 demonstrates that peak RCS pressure remains below the acceptance limit of 2750 psia.

← (DRN 05-543, R14)

15.4.1.2.5 Radiological Consequences

→ (DRN 04-704, R14)

The radiological consequences due to steam releases from the secondary system are less severe than those from the inadvertent opening of the atmospheric dump valve with a loss of offsite power, Subsection 15.1.2.4.5.

← (DRN 04-704, R14)

15.4.1.3 Uncontrolled CEA Withdrawal at Power

15.4.1.3.1 Identification of Causes and Frequency Classification

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

The estimated frequency of a CEA withdrawal at power classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. A CEA withdrawal is assumed to occur as a result of a single failure in the control element drive mechanism, Control Element Drive Mechanism Control System, or Reactor Regulating System. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging. The results presented in this section are based on an assessment using the revised SCRAM curve times presented in Table 15.0-5.

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

15.4.1.3.2 Sequence of Events and Systems Operation

An uncontrolled CEA withdrawal results in an increase in core power and corresponding increase in reactor coolant temperature and pressure; further, the withdrawal of CEAs produces a time dependent redistribution of core power. These transient variations in core thermal parameters may result in a rapid approach to the fuel design limits on DNBR and fuel centerline temperature, thereby requiring the protective action of the RPS.

The net reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent upon the CEA withdrawal rate and reactivity feedback mechanisms present at the time. The net reactivity insertion rate determines the rate of approach to the fuel design limits. Depending on the initial conditions and reactivity insertion rate the uncontrolled CEA withdrawal transient is terminated by one of the following:

a) achieving a stable, steady-state condition

b) high power level trip

c) high pressurizer pressure trip

→ (DRN 05-543, R14)

d) low DNBR trip

← (DRN 05-543, R14)

e) high local power density trip

→ (DRN 05-543, R14)

The CPC system has dynamic compensation lead-lag filters that project increases in core heat flux and more power for the VOPT. These dynamic compensation filters in conjunction with static power correction factors ensure that the CEA Withdrawal transient is terminated prior to any SAFDL violation.

The secondary system pressure increase following reactor trip is limited by the turbine bypass valves and the steam generator safety valves.

Table 15.4-7 gives the sequence of events for the limiting CEA withdrawal at power transient described in Subsection 15.4.1.3.3.

15.4.1.3.3 Core and System Performance

a) Mathematical Model

→ (EC-13881, R304; LBDCR 15-039, R309)

The NSSS response to a CEA withdrawal at power was simulated using the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program with the WSSV-T and ABB-NV correlations described in Chapter 4.

← (EC-13881, R304; LBDCR 15-039, R309)

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a CEA withdrawal at power are discussed in Section 15.0. Those parameters which were unique to the analysis discussed below are listed in Table 15.4-8.

← (DRN 05-543, R14)

The combination of the initial conditions for the CEA withdrawal event are selected from the range of parameters in Table 15.0-4 such that the minimum margin to the DNBR SAFDL exists. The event is run at a spectrum of initial power levels to ensure that the reactor protection system provides adequate function from all conditions.

→ (DRN 05-1551, R14)

The physics parameters assumed for the CEA withdrawal are selected to ensure the most adverse rate of power increase. The minimum delayed neutron fraction is used to make the core power more responsive to the insertion of positive reactivity. The least negative fuel temperature coefficient is used to minimize the negative reactivity feedback as core power begins to increase. A spectrum of differential road worths up to the maximum at HFP or $0.55 \times 10^{-4}\% \Delta\rho$/sec is

← (DRN 05-543, R14; 05-1551, R14)

→(DRN 05-543, R14)

considered. Analyses have shown that the most adverse CEAW results occur with the maximum reactivity addition rates. This analysis therefore uses the maximum reactivity addition rate with a CEA withdrawal speed of 30 in/minute. The CEA insertion curve assumed during reactor trip is based upon +0.3 ASIU to delay the negative reactivity being inserted by the falling CEAs.

Additional initial conditions and assumptions for this evaluation are:

- The BOC Doppler curve was assumed.
- The CEA insertion curve based on the limiting +0.3 ASI shape was assumed. This curve accounts for a 0.6-second CEA holding coil delay.

→(DRN 05-1551, R14)

- Maximum reactivity insertion rate (RIR) of 0.55×10^{-4} $\Delta\rho$/sec was assumed.

←(DRN 05-1551, R14)

- The CPCS VOPT ceiling was not credited. The CPCS VOPT follow trip of 10% was assumed resulting in a power trip at 110.5% of full power.
- The analysis of the high-power CEAW event covers a spectrum of CEA bank worths from the maximum worth, presented herein, to lesser worths which result in slower trip response times\different primary trips. The entire spectrum of CEA bank worths are used to ensure adequate initial thermal margin and acceptable trip response. Initial thermal-hydraulic conditions for the analysis presented herein are selected to be the most adverse for tuning of the CPCS.

←(DRN 05-543, R14)

c) Results

The dynamic behavior of important NSSS parameters following a CEA withdrawal at power are shown on Figures 15.4-25 through 15.4-37. The sequence of events for this transient as given in Table 15.4-7.

→(DRN 05-543, R14)

The withdrawal of CEAs produces a reactivity addition and a corresponding increase in core power. The power transient causes an increasing temperature and pressure transient, which together with the power distribution shift to the top of the core, causes a rapid approach to the fuel design limits. In the presence of a positive moderator temperature coefficient and with a constant secondary load, the moderator reactivity feedback accelerates the approach to the fuel design limits. At 11.6 seconds, a reactor trip is actuated on CPC VOPT. At 12.1 seconds, the trip breakers are opened and the CEAs begin dropping into the core at 12.7 seconds. This terminates the transient, with a minimum hot channel DNBR of greater than or equal to the DNBR limit.

The peak linear heat generation rate for this transient remains well below those which would cause centerline melt. The peak fuel centerline temperature for any CEA withdrawal at power is no worse than those attained for the CEA misoperation transient discussed in Subsection 15.4.1.4. That event is characterized by a transient increase in the radial power peaking factors and is accompanied by automatic CEA motion, producing a more rapid approach to the fuel centerline melt limit than does the CEA withdrawal at power.

←(DRN 05-543, R14)

15.4.1.3.4 Barrier Performance

a) Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.4.1.3.3.

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.4.1.3.3.

c) Results

→(DRN 05-543, R14)

Figure 15.4-37 shows the steam generator safety valve flows vs time for the uncontrolled CEA withdrawal at power. For secondary release considerations, the limiting CEA withdrawal transient is one initiated from a lower power level, since a greater primary power to secondary system load mismatch results. For peak RCS pressure considerations, the limiting CEA withdrawal is one initiated from a low power level, since these cases are characterized by a greater overshoot in pressure above the high pressurizer pressure trip setpoint.

←(DRN 05-543, R14)

15.4.1.3.5 Radiological Consequences

→(DRN 04-704, R14)

As discussed in Subsection 15.4.1.2.4, the CEA withdrawal resulting in the largest secondary releases is one initiated from a lower power. The radiological consequences of this event are less severe than the consequences of the inadvertent opening of the atmospheric dump valve with a loss of offsite power discussed in Subsection 15.1.2.4.5.

←(DRN 04-704, R14)

15.4.1.4 Control Element Assembly Misoperation

15.4.1.4.1 Identification of Causes and Frequency Classification

→(DRN 06-1062, R15; EC-8458, R307)

The estimated frequency of a CEA misoperation incident classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging.

←(DRN 06-1062, R15; EC-8458, R307)

A CEA misoperation is defined as any event which could result from a single malfunction in the reactivity control system, with the exception of sequential group withdrawals which are considered in Subsections 15.4.1.1 through 15.4.1.3. A list of the events which could be caused by a single malfunction in the reactivity control system is included in Subsection 7.2.2.1.1. CEA misalignment may be caused by a malfunction of the CEDM or the CEDMCS, or by an operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. Inadvertent withdrawal of a single CEA may be caused by a single malfunction of the Reactor Regulating System or of the CEDMCS or by an operator error. A dropped CEA may be caused by the opening of the electrical circuit of the CEDM holding coil. A dropped CEA subgroup could be caused by an electrical failure in the CEA coil power programmers. A CEA subgroup is defined as any one set of four symmetrical CEAs which is moved by the same CEDMCS.

The dropping of multiple CEA subgroups can be caused either by the operator erroneously depressing the "DROP" pushbutton on the Reactor Power Cutback control panel or by a fault in the CEA drop circuitry in the reactor power cutback module.

→(DRN 02-1479, R12; 05-543, R14)

←(DRN 02-1479, R12; 05-543, R14)

→(DRN 05-543, R14)

15.4.1.4.2 CEA DROP

15.4.1.4.2.1 Sequence of Events and Systems Operation

The effect of any misoperated CEA on core power distribution assessed by the CEA calculators (CEACS) and an appropriately augmented power distribution penalty factor is supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing moderator and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip if specified acceptable fuel design limits (SAFDLS) are approached.

The single CEA drop is the most adverse CEA misoperation event in terms of approach to the SAFDLs since dropped CEA subgroup is symmetric and produces less power distribution distortion.

→(EC-13881, R304)

The transient is initiated by the release and subsequent drop of a full-length CEA. A rapid decrease in power follows, accompanied by a decrease in reactor coolant temperatures and pressure. In the presence of a negative moderator temperature coefficient (MTC), positive reactivity is added. After the CEA drop, the mismatch between the secondary side (initial turbine demand) and primary side (post-drop power less than initial demand) results in feedback side that act to restore the primary power to the initial power level. The resultant increase in the hot pin radial peaking factor coupled with a return to 100% of full power (following the temporary power depression) can result in a DNBR that violates the DNBR SAFDL at some time during the transient.

←(EC-13881, R304)

→(DRN 02-1479, R12)

Sufficient initial thermal margin is preserved by the Limiting Conditions for Operation (LCOS) so that the plant may experience a CEA drop without requiring a reactor trip to ensure that the SAFDLs are not violated.

←(DRN 05-543, R14)

→(DRN 02-1479, R12)

←(DRN 02-1479, R12)

←(DRN 02-1479, R12)

→(DRN 05-543, R14)

15.4.1.4.2.2 Core and System Performance

←(DRN 05-543, R14)

a) Mathematical Model

→(DRN 05-543, R14; EC-13881, R304)

The NSSS response to a CEA Drop was simulated using the CENTS computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was calculated using the CETOP computer program, with the WSSV-T and ABB-NV correlations described in Chapter 4.

←(DRN 05-543, R14; EC-13881, R304)

b) Input Parameters and Initial Conditions

→(DRN 05-543, R14)

The input parameters and initial conditions used to analyze the NSSS and hot channel response to a CEA misoperation are discussed in Section 15.0. Those parameters which are unique to the analyses discussed below are listed in Table 15.4-9.

←(DRN 05-543, R14)

The sets of initial conditions (power, pressure temperature, coolant flow rate, radial peaking factor, and axial power distribution) that produce the most adverse consequences following CEA misoperation events were determined by performing parametric analyses of each incident over the range of reactor operating conditions given in Table 15.0-4.

Each of the combinations of these initial conditions which results in the minimum distance, in terms of thermal margin, from the SAFDL constitutes a Power Operating Limit (POL). Operation with less than the required initial thermal margin (closer to the SAFDL than a POL) is prohibited by plant Technical Specifications.

Parametric analyses are performed on CEA misoperation to determine the most adverse CEA misoperation in terms of power distortion and reactivity consequences. This information is combined with the most adverse sensitivity of thermal margin to changes in power distribution to determine the amount of initial thermal margin which must be preserved by the LCOs to prevent violation of the SAFDLs. This is done whether the event requires a reactor trip for protection or if a reactor trip can be avoided due to the initial thermal margin which is preserved by the LCOS.

→ (DRN 02-1479, R12; 05-543, R14)

The worst CEA drop incident is that case which produces the maximum increase in the radial peak-to-average power density factor with the minimum negative reactivity insertion. Parametric studies have shown that the results of the analysis are more sensitive to the radial peaking factor than to the reactivity insertion. The calculated increase in radial peak-to-average power density for any dropped CEA is less than or equal to 18 percent for cases initiated at 100 percent power. This value increases as power decreases and at 70 percent power the distortion is less than or equal to 45 percent. These distortion values include both the prompt power distortion accompanying the dropped CEA and an allowance for a redistribution of xenon in the core.

← (DRN 02-1479, R12; 05-543, R14)

Overall reactor power is conservatively assumed to return to the pre-drop power levels. This power return is driven by a constant turbine demand and Doppler and moderator reactivity feedback mechanisms. No analytical credit is taken for the decrease in core inlet temperature accompanying the cooldown driven return to initial power.

⟶ DRN 02-1479, R12; 05-543, R14)

The worst CEA drop incident is that case which produces the maximum increase in the radial peak-to-average power density factor with the minimum negative reactivity insertion. Parametric studies have shown that the results of the analysis are more sensitive to the radial peaking factor than to the reactivity insertion.

← (DRN 02-1479, R12; 05-543, R14)

→ (DRN 02-1479, R12)
← (DRN 02-1479, R12)

c) Results

→ (DRN 02-1479, R12; 05-543, R14; LBDCR 15-039, R309)

The sequence of events for a typical single CEA drop are shown in Table 15.4-13. The transient behaviors of important plant parameters during this event are shown in Figures 15.4-38 through 15.4-44b.

It is seen that the dropping of the single CEA causes a rapid reduction in core power to 88.6 percent of rated. Reactivity feedback and a slight RCS cooldown due to the power-to-load mismatch causes an increase in core power until at 41 seconds the core power has returned to its initial value. A new steady state condition is gradually established. Changes in the distribution of xenon within the core continue to gradually degrade thermal margin until the SAFDL is approached by 900 seconds. At this point the operators are required by plant Technical Specifications to reduce power if the dropped CEA has not been realigned, thereby maintaining sufficient margin to the SAFDL. Since the single CEA drop event does not generate a reactor trip, the revised CEA drop times, as documented in Table 15.0-5, would have no impact on the results of this event.

← (DRN 02-1479, R12; 05-543, R14; LBDCR 15-039, R309)

→(DRN 02-1479, R12)
←(DRN 02-1479, R12)

→(DRN 05-543, R14)
15.4.1.4.2.3 Barrier Performance
←(DRN 05-543, R14)

The barrier performance parameters following a CEA misoperation would be less adverse than those following the CEA withdrawal at low power incident (See Subsection 15.4.1.2.4) since the secondary safety valves are open for a much shorter period of time.

→(DRN 05-543, R14)
15.4.1.4.2.4 Radiological Consequences
←(DRN 05-543, R14)

→(DRN 04-704, R14)
The radiological consequences of this event are less severe than the consequences of the inadvertent opening of the atmospheric dump valve with a loss of offsite power in Subsection 15.1.2.4.5.
←(DRN 04-704, R14)

15.4.1.5 CVCS Malfunction (Inadvertent Boron Dilution)

15.4.1.5.1 Identification of Causes and Frequency Classification
→(DRN 00-592; 05-543, R14; 06-1062, R15; EC-8458, R307)
The estimated frequency of a Chemical and Volume Control System (CVCS) malfunction resulting in inadvertent boron dilution classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. The CVCS malfunction which results in unborated water being pumped at the maximum possible rate into the RCS by the demineralized water supply system is assumed to occur. For this to occur, one or more charging pumps must be on, the primary water makeup water pumps must be on, and the demineralized water supply system must be aligned to supply water to the charging pump suction via the volume control tank. Since at least three simultaneous equipment malfunctions would be required to produce the above conditions, the incident could only be the result of improper operator action accompanied by a single equipment malfunction. Since boron dilution is conducted under strict equipment procedural controls which specify limits on the rate and magnitude of any required change in boron concentration, the probability of a sustained or erroneous dilution is very low. Analysis of the Inadvertent Boron Dilution event initiated during each operational mode (defined in the Technical Specifications) was performed prior to Cycle 3 due to the potential for smaller primary system active mixing volume for modes 4 and 5 (with the RCS filled). Under some conditions the primary flow through a steam generator might stagnate. This could occur during an RCS cooldown using the Shutdown Cooling System if the Steam generator secondary side temperature becomes greater than the RCS temperature. In this scenario, flow through a steam generator would make a transition from forward flow to reverse flow and might include a short period of stagnant flow. This flow stagnation could also occur near the end of a long outage where core decay heat levels may be too small to provide sufficient driving head to move water through the steam generators. The boron dilution analysis showed that for the same K_{eff} and number of charging pumps, Mode 5 with the RCS partially drained resulted in the shortest available time for detection of the boron dilution event. Other Modes of Operation could have a shorter time to loss of shutdown margin depending on the initial conditions (K_{eff}, etc.) and number of charging pumps. COLR Section 3.1.2.9 provides the allowed number of charging pumps that can be operated in the various MODES and reactivities, as well as the required monitoring frequencies for backup boron dilution detection. 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 05-543, R14; 06-1062, R15; EC-8458, R307)

In Mode 4, operation of two charging pumps is allowed by Technical Specifications. The remaining charging pump is required to be isolated with power removed from it. In Mode 5 with K_{eff} less than or equal to 0.97, operation with two charging pumps is allowed. In Mode 5 with K_{eff} greater than 0.97, operation with one charging pump is allowed. In Mode 6, operation with one charging pump is allowed.

15.4.1.5.2 Sequence of Events and Systems Operation

The core is assumed to be initially subcritical with the shutdown margin at the minimum value consistent with the technical specification limit for the RCS condition. A CVCS malfunction occurs which causes unborated water to be pumped into the RCS. The resulting decrease in RCS boron concentration adds reactivity to the core. Prior to 15 minutes (30 minutes for Mode 6) before shutdown margin is lost, a high neutron flux alarm on the startup flux channels provides an indication that a boron dilution event is in progress.

←(DRN 00-592)

This boron dilution alarm alerts the operators so that the dilution can be terminated. Operational procedures, in addition to this alarm, assure detection and termination of the boron dilution event before the shutdown margin is lost in accordance with the requirement of SRP 15.4.6.

SRP 15.4.6 specifies minimum intervals between the time when a boron dilution alarm announces an unplanned moderator dilution and the time of loss of shutdown margin:

a) During refueling - 30 minutes

b) During startup, cold shutdown, hot standby and power operation - 15 minutes

In satisfaction of these requirements LP&L provided to the NRC, in letter W3P82-3783, the Boron Dilution Alarm System setpoint analysis. The current setpoints meet these time requirements.

15.4.1.5.3 Core and System Performance

a) Mathematical Model

Assuming complete mixing of boron in the RCS, the rate of change of boron concentration during dilution is described by the following equation:

$$M\frac{dC}{dt} = -W \, , \qquad (1)$$

where

M = RCS mass

C = RCS boron concentration

W = charging mass flowrate of unborated water

dC/dt is maximized by maximizing W and minimizing M.

Assuming

W = constant, equal to the maximum possible value,

and choosing

M = constant equal to the minimum value occurring during the boron dilution incident,

the solution of equation (1) can be written

→(DRN 00-592)

$$C(t) = C(o)e^{-t/\Upsilon} \qquad (2)$$

←(DRN 00-592)

where

Υ = M/W = boron dilution time constant

C(o) = initial boron concentration

The time T required to dilute to criticality is given by

→(DRN 00-592)

$$T = \Upsilon \, \ell n \frac{C(o)}{C_{crit}} \qquad (3)$$

←(DRN 00-592)

where

C_{crit} = critical boron concentration

b) Input Parameters and Initial Conditions

→(DRN 05-543, R14)

Since coolant is circulated through the RCS by the Shutdown Cooling System, complete mixing of boron within the active RCS volume is assumed. The initial conditions and analysis parameters are chosen to minimize the interval from initiation of dilution to the time at which criticality is reached. The parameters for a representative Mode 5 partially drained case are presented below as an example.

←(DRN 05-543, R14)

1) The technical specification lower limit on shutdown margin for the assumed RCS condition is used. This results in a subcriticality of at least 2.0%$\Delta\rho$ for either technical specification condition of any CEA withdrawn or all CEAs fully inserted.

2) The technical specification upper limit on K_{eff} for Modes 3 - 5 is 0.99. However, the most adverse initial condition would be for an initial K_{eff}, corresponding to 2.0%$\Delta\rho$ subcritical, since this would result in the core reaching criticality with no shutdown CEAs available at time of trip. Therefore, a maximum K_{eff} of 0.98 is assumed.

3) The RCS is assumed to be drained so that the water level is at the centerline of the hot leg nozzle. Assuming the coolant temperature is 200°F, the technical specification upper limit for cold shutdown, the resulting mass is 274,820 lbm.

→(DRN 05-543, R14)

4) The flow from the charging pumps is assumed to be at 44 gpm per pump. The corresponding mass flowrate, assuming cold liquid flow, is 6.12 lbm/sec per pump.

←(DRN 05-543, R14)

5) The critical boron concentration in Mode 5 cold shutdown with all CEAs inserted except for the single highest worth rod stuck out (N-1) and zero xenon is 1412 ppm. This concentration is assumed to minimize the time to loss of shutdown margin. The initial subcritical boron concentration is found by adding the product of the inverse boron worth (84 ppm/%$\Delta\rho$) and the minimum shutdown margin required to the critical boron concentration for Mode 5 with K_{eff} of 0.98 is 1580 ppm.

The parameters discussed above are summarized in Table 15.4-17.

c) Results

→(DRN 05-543, R14)

Using the above conservative parameters in equation 3, the minimum possible time interval to dilute from the K_{eff}, of 0.98 to criticality for Mode 5, partially drained with one charging pump operational is greater than 80 minutes. For all other combinations of RCS conditions, K_{eff}, and number of charging pumps, the time to loss of shutdown margin is greater than 46 minutes. A high neutron flux alarm on the startup flux channel will assure detection of a boron dilution event with at least 15 minutes (30 minutes for Mode 6) prior to loss of shutdown margin as per requirements of SRP 15.4.

Each reload cycle, the cycle specific values of inverse boron worth and critical boron concentration for each operational mode are examined to ensure that Technical Specification 3.1.2.9 and its associated COLR values preserve the necessary times for operator action prior to loss of shutdown margins.

←(DRN 05-543, R14)

→ (LBDCR 15-039, R309)

Since all CEAs have been already inserted at event initiation, the revised CEA drop times, as documented in Table 15.0-5, would have no impact on the CVCS malfunction event.

← (LBDCR 15-039, R309)

15.4.1.5.4 Barrier Performance

The barrier performance parameters during an inadvertent boron dilution would be less adverse than those during uncontrolled CEA withdrawal at power (Subsection 15.4.1.3) because there is no associated power transient.

15.4.1.5.5 Radiological Consequences

There are no radiological consequences associated with this event.

→(DRN 02-1479, R12)

15.4.1.6 <u>Start of an Inactive Reactor Coolant System Pump - Modes 3, 4 or 5 With All CEAs on the Bottom</u>

←(DRN 02-1479, R12)

No specific analysis of this event is provided for Modes 1 and 2 since the Technical specifications require that the reactor be subcritical when less than four reactor coolant pumps are operating.

15.4.1.6.1 Identification of Causes and Frequency Classification

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

The estimated frequency of an inadvertent startup of an inactive reactor coolant system pump classifies it as a moderate frequency incident as defined in Reference 1 of section 15.0.

The idle reactor coolant pump startup event is defined as the start of a reactor coolant pump without observance of prescribed operating procedures, assuming an initial condition in which one or more reactor coolants pumps are idle. 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 02-1479, R12; 05-543, R14; 06-1062, R15; EC-8458, R307)

15.4.1.6.2 Sequence of Events and Systems Operation

→(DRN 02-1479, R12)

Below Mode 3, the Technical Specifications do not require that a reactor coolant pump be operable. If the temperature of the secondary system is significantly different from the core coolant temperature, then upon reactor coolant pump startup a rapid change in core coolant temperature occurs. Depending on the isothermal temperature coefficient, a corresponding change in core reactivity may occur.

→ (DRN 05-543, R14)

Two cases have been evaluated to determine the reactivity addition due to the inadvertent startup of a reactor coolant pump in Modes 4 and 5. This reanalysis assumed bounding values of the temperature difference between the RCS and the steam generators. The first case is defined as the set of conditions where the steam generator temperature is less than the reactor coolant system temperature. In this case the startup of an inactive pump could cause a cooldown of the reactor coolant system.

←(DRN 05-543, R14)

The bounding value of RCS cooldown is 100°F. In the presence of a negative isothermal temperature coefficient this would insert positive reactivity. The second case is defined as the set of conditions where the steam generator temperature is greater than the reactor coolant system temperature. In this case the startup of an inactive pump could cause a heatup of the reactor coolant system. The bounding value of RCS heatup is 190°F. In the presence of a positive isothermal temperature coefficient this would insert positive reactivity.

15.4.1.6.3 Core and System Performance

→ (DRN 05-543, R14; EC-8458, R307)

The more limiting case of the two is when the steam generator temperature is greater than the reactor coolant system temperature with a negative isothermal temperature coefficient. This event was evaluated parametrically on cold leg temperature for a range of temperatures consistent with Mode 4, and 5 operation. From Technical Specification 3.1.1.2 it is seen that the shutdown margin must always be greater than the value specified in the COLR. This shutdown margin determined based on the most reactive rod stuck out of the core results in an initial subcriticality greater than the positive reactivity addition due to the inadvertent startup of a reactor coolant pump. The available shutdown margin is thus sufficient to preclude criticality due to an inadvertent reactor coolant system pump startup. The results indicate that the reactor remains subcritical and the specified acceptable fuel design limits are not exceeded, thus maintaining clad integrity.

← (DRN 05-543, R14; EC-8458, R307)

→ (LBDCR 15-039, R309)

Since all CEAs are fully inserted at event initiation, the revised CEA drop times, as documented in Table 15.0-5, would have no impact on this event.

← (LBDCR 15-039, R309)

15.4.1.7 <u>Uncontrolled CEA Withdrawal from a Subcritical Condition - Modes 3, 4, or 5 With All CEAs on the Bottom</u>

← (DRN 02-1479, R12)

15.4.1.7.1 Identification of Causes and Frequency Classification

→ (DRN 06-1062, R15; EC-8458, R307)

The estimated frequency of a control element assembly (CEA) withdrawal from subcritical conditions classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. The results presented in this section are based on the original steam generators and bound the replacement steam generators with up to 10% steam generator tube plugging.

← (DRN 06-1062, R15; EC-8458, R307)

An uncontrolled withdrawal of a shutdown bank is assumed to occur as a result of a single failure in the Control Element Drive Mechanism Control System or as a result of operator error. An uncontrolled sequential withdrawal of regulating CEAs is assumed to occur as a result of a single failure in the Control Element Drive Mechanism Control System, Reactor Regulating System, or as a result of operator error. These events are analyzed in support of the reduced Technical Specification shutdown margin requirements in subcritical modes for Cycle 2.

15.4.1.7.2 Sequence of Events and Systems Operation

→ DRN 02-1479, R12)

The withdrawal of CEAs from subcritical conditions with all CEAs on the bottom adds reactivity to the reactor core causing the core power level to increase. The withdrawal of CEAs also produces a time dependent redistribution of core power. As the power level continues to increase a trip is generated by the Core Protection Calculators (CPCs) when the CPC bypass is automatically removed at 2.4 x 10^{-4}% of rated thermal power.

← (DRN 02-1479, R12)

This is due to a standing trip condition existing prior to the CPC bypass caused by the withdrawal of CEAs which result in an abnormal CEA configuration. The trip will occur on integrated radial peaking factor out of range, low DNBR, and/or high local power density. This trip causes the shutdown of the reactor prior to the point of adding sensible heat.

15.4.1.7.3 Core and System Performance

Due to the prompt CPC trip at 2.4 x 10^{-4}% of rated thermal power the consequences of these events do not result in DNBR or fuel centerline melt specified acceptable fuel design limits being violated.

→ (LBDCR 15-039, R309)

Since all CEAs are fully inserted at event initiation, the revised CEA drop times, as documented in Table 15.0-5, would have no impact on this event.

← (LBDCR 15-039, R309)

15.4.2 INFREQUENT EVENTS

There are no infrequent events resulting from reactivity and power distribution anomalies.

15.4.3 LIMITING FAULTS

15.4.3.1 Inadvertent Loading of a Fuel Assembly into the Improper Position

15.4 3.1.1 Identification of Causes

Two accidents are considered in this subsection, first, the erroneous loading of fuel pellets or fuel rods of different enrichment in a fuel assembly, and second, the erroneous placement or orientation of fuel assemblies.

The likelihood of an error in assembly, fabrication, or core loading is considered to be extremely remote because of the extensive quality control and quality surveillance programs employed during the fabrication process as well as the strict procedural control used during core loading.

15.4.3.1.1.1 Erroneous Loading of Fuel Pellets or Fuel Pins of Different Enrichment in a Fuel Assembly

The probability of manufacturing a fuel assembly with an incorrect enrichment is considered an unlikely event. The extensive quality control and quality surveillance programs in effect during the manufacture of the fuel pellets, in the loading of the fuel rods and in the assembly of the fuel bundles preclude the possibility of manufacturing a fuel bundle with an incorrect enrichment(s).

→ (DRN 03-1610, R13)

During the manufacture and assembly of the fuel rods, numerous check points and assay tests ensure that the enrichment is as specified and the fuel rods and fuel assemblies are properly loaded and assembled. An assay is made of each lot of UO_2 powder to ensure that the enrichment is as required by the fuel specification. During the manufacture of the pellets, each powder lot is isolated during processing. After sintering, additional enrichment check of the fabricated pellets is made by random sampling. Assembly of the fuel rods is performed by loading the fuel pellets into cladding onto which one end cap has been welded. The end cap is marked prior to welding, thereby identifying the enrichment to be loaded into the fuel rod.← (DRN 03-1610, R13)

During assembly of the fuel bundle the quality control procedures require verification of each fuel rod in the assembly. Each fuel assembly is identified by a serial number which is engraved on the upper assembly plate in a prominent location.

→(DRN 03-1610, R13)

The fuel vendor uses a record-keeping system that provides a mechanism by which an accurate record of all fuel rods within a fuel assembly can be defined as well as the enrichment, weight of U-235 and UO_2, and lot number of the fuel within each fuel rod.

15.4.3.1.1.2 Erroneous Placement or Orientation of Fuel Assemblies

A coded serial number is marked on the upper end fitting of each fuel assembly. This serial number is used as a means of positive identification for each assembly in the plant. During refueling outages, a visual display is provided in the main control room showing a schematic representation of the reactor core, spent fuel pool, and new fuel storage area. During core loading operations, the location of each control element assembly, fuel assembly, and source is depicted on the display by a marker carrying its identification number. The display is kept updated during core loading/unloading operations by a designated member of the plant staff. This individual is in constant communication with personnel in each area where fuel movement is occurring. Fuel movement is not allowed unless these lines of communication exist. At the completion of core loading, all fuel assemblies in the core are confirmed to be in the proper location and to have the correct rotational orientation as specified by design documentation. Independent verification of this activity is performed.

If, however, in spite of these precautions it is assumed that an assembly is placed in the wrong core position, then many possibilities exist. The worst situation would be the interchange of two assemblies of different enrichments or reactivities.

←(DRN 03-1610, R13)

15.4.3.1.2 Analysis of Effects and Consequences

15.4.3.1.2.1 Erroneous Placement or Orientation of Fuel Assemblies

If, in spite of the extreme precautions described above, it is postulated that a fuel assembly is misloaded, several situations may be postulated. The misloading of a fuel assembly may effect the core power distribution only slightly, for example, if assemblies of similar enrichments and reactivities are misloaded. Alternatively, the core power distribution may be affected enough so that core performance would be affected if assemblies having different enrichments or reactivities are misloaded.

→(DRN 03-1610, R13; 05-543, R14)

In the unlikely event that two assemblies of different initial reactivities would be interchanged, most misloadings would be detected using either the incore or excore neutron detectors (See Figure 15.4-98 for incore detector locations) during the power ascension physics testing. Misloadings that result in significant azimuthal tilt would be detected by the differences in signals between the four excore detectors during routine monitoring of their signals during the initial power ascension. In addition, analysis has shown that if all the incore detectors in the vicinity of the misload are operational, then most of the misloads that may result in a significant increase in power peaking will be detected during the power ascension testing. If several of the incore detectors in the vicinity of the misloaded assembly are inoperable, then some of the misloads that result in substantial increase in core power peaking may not be detected during the initial power ascension physics testing. Of this small class certainly the worst case that can be envisioned is the interchange of a fresh shimmed assembly with a once-burned assembly near the center of the core. Although this type of misload may result in high local power peaking, it may not be detectable if most of the incore detectors in the vicinity of the misload are inoperable since it produces essentially no core wide global power tilt during the power ascension physics testing. Figure 15.4-99 shows the mid-cycle power distribution for a worst case misload of this type.

For Waterford 3, operation with an undetectable mislead may result in a potential nonconservatism of the COLSS and CPC online margin assessment due to the potential non-conservatism of the measurement of local power peaking for the misloaded core. If the CECOR measured values for the radial peaking factors are conservative then safety limits will not be exceeded even during operation with an undetectable mislead, since these peaking factors are used by COLSS and CPC to calculate margin to the SAFDLs.

If all the ICIs in the vicinity of the misload are operable then most of the impact of the misload on power peaking will be accounted for. However, if some of the ICIs in the vicinity of the misload are inoperable, then the CECOR measurement radial peaking factor may be significantly non-conservative for a core with an undetectable misload. However, even in this case the SAFDL will not be exceeded provided that the overpower margin associated with the error in measured peaking factor does not exceed the minimum (required) overpower margin (ROPM) reserved by COLSS. Thus, this analysis determines the ROPM that must be reserved by COLSS to assure that the DNB SAFDLs is not violated due to a fuel assembly misloading.

→(DRN 05-1551, R14)

The methodology used herein for the assembly misloading analysis is summarized as follows:

←(DRN 05-1551, R14)

- Several candidate worst case misloads are selected for analysis. The selection of the representative worst undetectable misload was based on the reactivities and fissile content of the assemblies as well as reactivities of other assemblies in the vicinity of the misloaded assembly.

- A determination is made as to whether the misload is detectable during the startup tests. This is done by using the ROCS code to calculate the signal at each of the ICI locations for the fuel symmetry verification test for each postulated assembly misloading. These signals were then used by CECOR to infer the “measured” power distribution. These CECOR cases consider several different configurations of failed instruments in the vicinity of the high-power misloaded assembly consistent with the requirements for ICI operability.

- If the misload is judged detectable then it is eliminated from further consideration.

- Of the misloads judged undetectable several of the worst candidates are selected for analysis under full power operating conditions. The worst candidates are defined as those having the highest CECOR decalibration factor (DF).

←(DRN 05-543, R14)

→(DRN 05-543, R14)

- For each of the assembly misloadings selected for full power analysis the power distribution was calculated by both ROCS and CECOR at several different burnup points. The CECOR cases consider several different configurations of failed instruments consistent with the ICI operability criteria. The ROCS and CECOR core power distributions are then used to establish the maximum CECOR Decalibration Factor during full power operation.

- The power distributions calculated by ROCS and CECOR for the burnup point corresponding to the maximum CECOR DF is used to calculate the minimum DNBR and ROPM using the CETOP code for the worst case misload.

- The largest value of ROPM calculated for the worst undetectable misload is compared to the ROPM installed in COLSS to assume that the DNB SAFDL will not be violated under any allowed operating condition.

Table 15.4-37 shows that if all the incore detectors in the vicinity of the misloaded assembly are operational, then the misload will be detected during power ascension physics testing since it would exceed the physics startup test acceptance criteria of 10% for the maximum power difference between a detector and the average of its symmetric detector group. If three out of the four incore instruments in the vicinity of the misload are inoperable, then the misload may not fail the power ascension test acceptance criteria and operation under a misloaded condition at power may occur. Table 15.4-38 shows the cycle-maximum power peaking factors associated with this misload. If all of the incore detectors in the vicinity of the misload are operable then the error in measured power peaking (using the CECOR code, Reference 35 of Section 4.3) due to the misload will be 6%. Since the measured power peaking is used by COLSS to determine the available overpower margin and since the minimum COLSS available overpower margin under any condition is at least 20%, including allowances for uncertainties, no fuel damage is expected. If three out of the four incore instruments in the vicinity of the misload are inoperable, then the error in measured power peaking due to the misload will be 15%. This corresponds to an overestimate in the available COLSS overpower margin of 18% which is still less than minimum COLSS overpower margin, which is typically ≥20%. Furthermore, since it is very improbable that three out of four of the incore detectors in the vicinity of the misload would be inoperable, it is highly likely that the misload will be detected during power ascension or early in the cycle before a significant decrease in operating margin occurs.

←(DRN 03-1610, R13)

The Misloading analysis assumes the following limitations and initial conditions:

→(LBDCR 16-002, R309)

• Nominal Cycle Length	330 to 590 EFPD
• Number of fresh fuel assemblies	60 to 104
• Maximum Pellet Enrichment	5.0 wt % U-235
• Maximum # Erbia Roads/Assembly	100
• Fuel Assembly Loading Pattern Symmetry	Octant except for twice burnt
• ROPM	At least 118%
• Low Leakage Loading	Assemblies with two faces adjacent to the shroud must be at least once-burned
• ICI Operability	At least 75% of all detector locations (per TRM 3/4 3.3.2)
• Vector Tilt	No greater than 3%

←(LBDCR 16-002, R309)

Conformance to the above assumptions and conditions is verified as part of the reload design and startup testing processes.

←(DRN 05-543, R14)

→ (DRN 05-543, R14)

Based on these results:

- If all the ICIs in the vicinity of the misload are operational, then the ROPM associated with the representative worst case undetectable misload is expected to be <10%.

- With the current TRM requirement of 1 operable ICI in every 4-by-4 array of fuel assemblies, the worst undetectable misload will not result in fuel failure under any allowed operating condition even with up to 25% failed ICIs.

- The MDNBR at nominal core conditions for the representative worst case undetectable misload including allowance for physics calculational uncertainties is significantly greater than 1.26. In addition, the peak linear heat generation rate (PLHGR) for this worst undetectable misload is also well below the LHR SAFDL for fuel centerline melt. These results indicate that there is sufficient OPM for the representative worst case undetectable misload to accommodate a reasonable combination of off-nominal core operating conditions and/or expected operational occurrences without resulting in doses that exceed a small fraction of 10CFR50.67.

← (DRN 05-543, R14)

→ (LBDCR 15-039, R309)

Since a reactor trip is not assumed to occur, the revised CEA drop times, as documented in Table 15.0-5, would have no impact on this event.

← (LBDCR 15-039, R309)

15.4.3.2 <u>Control Element Assembly (CEA) Ejection</u>

15.4.3.2.1 Identification of Causes and Frequency Classification

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

The estimated frequency of a CEA ejection classifies it as a limiting fault as defined in Reference 1 of Section 15.0. For a CEA ejection to occur, a mechanical failure of the control element drive mechanism (CEDM) housing or of the CEDM nozzle must be postulated such that the Reactor Coolant System pressure ejects the CEA and drive shaft to the fully withdrawn position. For this analysis, it is assumed that a complete and instantaneous circumferential rupture of the CEDM housing or of the CEDM nozzle results in the ejection of a CEA. 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)

15.4.3.2.2 Sequence of Events and Systems Operation

The transient following a CEA ejection accident is as follows: The reactor core power rises rapidly for a short period of time. This increase is terminated by Doppler feedback or neutronic effects. Following this, a reactor trip is initiated on high neutron power, and the power transient is terminated. The potential for fuel damage from a CEA ejection is minimized by the Reactor Protective System and by restrictions on CEA patterns and/or power dependent insertion limits (PDIL) during operation. These combine to limit the transient fuel enthalpy, fuel and clad temperatures, and RCS pressure to acceptable values.

→ DRN 05-543, R14)

The consequences of the loss of coolant resulting from the RCS rupture are less severe than those for small RCS breaks as discussed in Subsection 6.3.3.

← (DRN 05-543, R14)

15.4.3.2.3 Core and System Performance

a) Mathematical Model

The NSSS response to a CEA ejection was simulated using the method of analysis described in Reference 1. The procedure outlined on Figure 2-1 of Reference 1 was used to determine the energy deposition in the fuel rod. The number of fuel pins predicted to experience DNB was calculated using the STRIKIN-II computer program described in Section 15.0, with the CE-1 CHF correlation described in Chapter 4. The procedure used to calculate the number of fuel pins which experience DNB is as follows:

→ (DRN 05-543, R14)

1) The minimum transient DNBR is determined from STEP 7 of the CE Synthesis Method(1) for various combinations of initial and ejected radial fuel pin peaking factors.

← (DRN 05-543, R14)

2) A matrix relating the initial and ejected peaking factors to a pin census edit is obtained from STEP 6 of the CE Synthesis Method (that is, each pin is traced from its initial to its ejected value).

3) The number of fuel pins with ejected peaking factors greater than the threshold peaking factors, determined in (1) above for each initial peaking factor, is found from the matrix obtained in (2) above.

→ (DRN 05-543, R14)

4) This number is then conservatively equated to the number of fuel pins experiencing clad failure due to DNB using the method of statistical convolution, Reference 6.

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

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a CEA ejection are discussed in Section 15.0. Those parameters which were unique to the analysis discussed below are listed in Tables 15.4-23 and 15.4-24.

→ (DRN 05-543, R14; 05-1551, R14)

The CEA Ejection event is analyzed across a range of initial power levels to examine the competing effects of available thermal margin and CEA insertion worth. The case presented is analyzed from hot full power initial conditions. Analyses were performed using conservative core conditions that were a mix of BOC and EOC conditions. A least negative/most positive BOC MTC was used to maximize the reactivity inspection that accompanied the coolant temperature increase. The BOC (minimum) Doppler coefficients were used to minimize the reactivity decrease due to increased temperature. EOC (minimum) delayed neutron fractions and lifetimes were used to maximize the reactivity increase during the transient. Using the mix of conditions ensures that a conservative analysis was performed. Maximum radial peaking factors and maximum ejected CEA worths were assumed to maximize the potential for fuel enthalpy increase. The initial axial power distributions was chosen to maximize the energy content of the hottest fuel pellet.

← (DRN 05-1551, R14)

Additional initial conditions and assumptions for this evaluation are:

- The CEA insertion curve assumed a 0.6-second CEA holding coil delay.

→ (DRN 06-1062, R15)

- The variable overpower trip (VOPT) of the CPCS was employed in the analysis. A CPCS trip delay time of at least 0.420 seconds was assumed.

← (DRN 06-1062, R15)

- The VOPT assumed an excore uncertainty of 40%. This is a conservative allowance covering the asymmetry in core power production/excore response and assumes failures in the excore detector channels seeing the greatest power increase.
- An initial core power of 3735 MWt was assumed based on a rated power of 3716 MWt and a 0.5% uncertainty.
- Of the four excore channels, the one closest to the ejected CEA location is assumed inoperable and a failure of the next best detector channel is assumed.

→ (DRN 05-543, R14)

→ (DRN 05-543, R14)

The initial conditions for the principal process variables monitored by the COLSS were varied within the operating envelope given in Table 15.0-4 to determine the set of conditions which would produce the most adverse consequences following a CEA ejection. Various combinations of core inlet flowrate and pressurizer pressure were chosen to be consistent with the power level of the case in question. The initial pressurizer level and steam generator water level and pressure, as controlled within the operating limits, have an insignificant effect on the consequences of the CEA ejection analyses.

The axial shape was chosen to maximize the energy content in the hottest fuel pellet. The radial fuel pin peaking factor was chosen as the maximum initial pin peak. The remaining parameters were chosen based on the results shown in Chapter 4 of Reference 1. These parameters were varied in the most adverse direction until a COLSS limit was achieved.

→ (LBDCR 15-039, R309)

Initial conditions for the RCS pressure calculation are seen in Table 15.4-2. The CEA ejection was assumed to produce a break at the top of the reactor vessel head. To obtain a conservative estimate of the effect of the break size on the magnitude of overpressurization, the initial pressure response was predicted without taking credit for coolant flow through the break area.

← (LBDCR 15-039, R309)

c) Results

→ (DRN 05-1551, R14)

Figures 15.4-67 through 15.4-72 show the reactor power, heat flux, clad and fuel temperatures, and reactivity components during the significant portion of the transient. Table 15.4-19 gives sequence of events that occur during the full power CEA ejection accident. This table reflects the results of the STRIKIN-II calculation for fuel performance.

← (DRN 05-1551, R14)

The core power increases rapidly as a result of the approximate step change in reactivity due to the ejected CEA. However, the rapid increase in core power is terminated by the combination of neutronic and Doppler feedback effects. The increased core power results in a high power trip.

← (DRN 05-543, R14)

In the hot channel, the increase in heat flux is such that DNB is calculated to occur, resulting in:

1) A rapid decrease in the surface heat transfer coefficient.

2) A rapid decrease in heat flux.

3) A rapid increase in clad temperature.

The rapid increase in clad temperature is sufficient to override the decreased surface heat transfer coefficient, resulting in a second peak in the hot channel heat flux. At this time the shutdown CEAs are nearly fully inserted, resulting in a rapid reduction in the core power level, and the heat flux decreases for the remainder of the transient.

→DRN 05-543, R14; LBDCR 15-039, R309)

Figures 15.4-73 through 15.4-73e show the RCS pressure transient for the full power. These figures reflect the limiting CENTS RCS pressure calculation which determined the maximum hypothetical peak RCS pressure during a CEA ejection from full power of 2632 psia. It was conservatively assumed that there was no break in the reactor vessel head. Results of the RCS pressure calculations for full power initial conditions, with and without the break, are shown in Table 15.4-27.

← (LBDCR 15-039, R309)

The calculated average enthalpy and total centerline enthalpy of the hottest fuel pellet for the cases analyzed are shown in Table 15.4-28. These results show that the CEA ejection accident will not result in a radial average fuel enthalpy greater than 280 cal/gm at any axial location in any fuel rod. No more than 15% of the fuel roads will experience DNB as a result of the CEA Ejection. The initial power level which results in the maximum number of pins experiencing DNB may vary on a cycle by cycle basis depending upon the specific core design. The extent of fuel performance shown in Table 15.4-28 is verified not to be exceeded across the spectrum of power levels each cycle in the reload analysis process.

← (DRN 05-543, R14)

The results of the RCS pressure calculation given in Table 15.4-27 show that the maximum pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the faulted condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.

→ (DRN 05-543, R14)

← (DRN 05-543, R14)

→ (DRN 05-1551, R14; LBDCR 15-039, R309)

The threshold value for incipient centerline melting is given in Reference 3, 4 and 5 as 231.3 cal/gm. It was demonstrated that no fuel rods experience violation of this threshold.

← (DRN 05-1551, R14; LBDCR 15-039, R309)

15.4.3.2.4 Barrier Performance

→ (DRN 05-543, R14)

The consequences of the loss of coolant resulting from the RCS rupture are less severe than those for small breaks as discussed in Subsections 6.3.3 and 15.6-3. However, the overpower transient caused by the ejected CEA results in a secondary system response which is more adverse than small break loss of coolant events of the same break size.

a) Mathematical Model

The power and core average heat flux from Subsection 15.4.3.2.3 were used as input to the CENTS computer program to determine the primary and secondary system responses.

→ (DRN 05-543, R14)

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.4.3.2.3. The full power BOL case which results in the most adverse radiological consequences, was chosen for this analysis. The secondary releases were maximized by assuming a loss of offsite power at the time of turbine trip. It was also assumed that there was no break in the reactor vessel head. Thus, all heat must be removed through the steam generators, resulting in a longer period before initiation of shutdown cooling. In this analysis, it is assumed that operator action is delayed until 30 minutes after the event.

→(DRN 04-704, R14)

An additional source of radiological releases is via containment leakage to the environment.

←(DRN 04-704, R14)

c) Results

→(DRN 05-543, R14)

The transient behavior of the NSSS is similar to that for the loss of offsite power described in Subsection 15.2.1.3. This is due to the fact that a reactor trip occurs for both events within the first second of the transient. A low steam generator level signal occurs which initiates emergency feedwater flow to each steam generator. This feedwater flow remains constant for the first 1800 seconds of the transient.

→(DRN 05-1551, R14)

In order to provide for long-term cooling of the NSSS, emergency feedwater flow is assumed to be added to maintain steam generator level after 1800 seconds until shutdown cooling is initiated.

←(DRN 05-543, R14; 05-1551, R14)

15.4.3.2.5 Radiological Consequences

15.4.3.2.5.1 Design Basis - Method of Analysis

15.4.3.2.5.1.1 Design Basis - Physical Model

→(DRN 04-704, R14)

The evaluation of the radiological consequences of a postulated CEA ejection accident assumes a loss of offsite power at time of turbine trip. It is also assumed that the activity in the coolant systems prior to the accident is the Technical Specification limit for the primary system. Following a postulated CEA ejection accident, activity is released from the fuel into the reactor coolant. A maximum limit of 15% fuel failure due to DNBR is assumed for the radiological dose consequences analysis.

The activity released to the containment (from the reactor coolant discharged through the ruptured CEDM pressure housing) is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere. The removal processes considered in the containment are radioactive decay, natural deposition, and leakage from the containment.

Per RG 1.183, two different fission product release paths are considered:

1. Fission product releases via SG steaming (releases via ADVs or MSSVs). Per RG 1.183 Table 6, this pathway is analyzed until cold shutdown is established.

2. Fission product releases via containment leakage. Per RG 1.183 Table 6, this pathway is analyzed for a 30-day release duration.

←(DRN 04-704, R14)

→(DRN 04-704, R14)
Each of these release pathways must independently meet the RG 1.183 AST dose limits.

→(EC-40444, R307)
The secondary steaming pathway assumes that a CEA has been ejected concurrent with a LOOP and that releases are occurring due to 150 gallons/day per SG primary-to-secondary leakage from the RCS to the SG. Activity is then released to the environment through the use of the ADVs to remove decay heat and to cool the plant to shutdown cooling entry conditions. Since the control room χ/Q values are worst for ADV releases, any releases which would occur through the MSSVs are instead assumed released from the ADV locations. Once shutdown cooling is initiated, the release, for cooldown purpose from ADVs, to the environment is terminated for this pathway. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.
←(EC-40444, R307)

The fission product release via normal containment leakage assumes that the ejected CEA has caused a small break in the RCS coolant boundary. This results in the release to containment of the activity, due to the transient, in the RCS water inventory. A maximum limit of 15% fuel failure due to DNBR is assumed. Fission product removal via the operation of containment sprays is conservatively not credited for this calculation. Also, per RG 1.183, it is conservatively assumed that all available fission project activity is released to the containment within a very short time for this case. This analysis assumes a 0.5% per day by volume containment leakage rate for the first 24 hours, then 0.25% by volume per day afterward, consistent with the guidelines in RG 1.183.
←(DRN 04-704, R14)

15.4.3.2.5.1.2 Design Basis - Assumptions, Parameters, and Calculational Methods
→(DRN 04-704, R14)
The major assumptions, parameters, and calculational methods used in the design basis analysis are itemized in Table 15.4-31. In the evaluation of the CEA ejection accident, recommendations listed in Appendix H of Regulatory Guide 1.183 were utilized. Additional clarification of the assumptions and parameters listed in Table 15.4-31 is provided.
←(DRN 04-704, R14)
→(DRN 05-1551, R14)
a) Reactor Coolant Activity and Secondary System Activity before Accident
←(DRN 05-1551, R14)

The reactor coolant equilibrium activity is the Technical Specification limit of 1.0 μCi/gm Dose Equivalent (DEQ) Iodine - 131 (I-131).
→(DRN 05-1551, R14)
←(DRN 05-1551, R14)

The activity of liquid in both steam generators is conservatively assumed to be equal to the Technical Specification limit of 0.1 μCi/gm Dose Equivalent Iodine – 131 (I-131).

→(DRN 04-704, R14; 05-1551, R14)
b) Removal Coefficients

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 to 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 released to the environment.

For the Reactor Containment Leakage release pathway, the Powers 10% Aerosol Decontamination Factor is assumed for natural deposition. This model is contained in the RADTRAD analysis code documented in NUREG/CR-6604. A natural deposition factor of 0.4/hour is assumed for elemental iodine and 0.0 for organic iodine.
←(DRN 04-704, R14; 05-1551, R14)

→(DRN 04-704, R14; 05-1551, R14)

No credit is taken for the removal of fission products due to containment spray initiation.

←(DRN 05-1551, R14)

c) Activity Available for Release from Containment at Time Zero

→(EC-5000081470, R301)

The activity available for leakage from containment is based on Regulatory Guide 1.183, Appendix H assumptions. The activity in the fuel clad gap is assumed to be 10 percent of the iodines, 10 percent of the noble gases, and 12 percent for the alkali metals (Cs and Rb), assuming continuous maximum full power operation. Additionally, a core radial peaking factor of 1.65 is conservatively applied to fuel rods experiencing cladding failure. No fuel melting occurs for this event. The source terms available for release are presented in Table 15.4-31.

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

d) Containment Leak Rate

The activity available for leakage from containment is assumed to be instantaneously mixed in the containment free volume. Activity is assumed to leak out at the technical specification limit for the first day (0.50 percent/day) and at half this rate for the duration of the accident (1 - 30 days).

e) Radioactive Decay

The activity in containment was assumed to decay due to holdup. After release from containment, no radioactive decay or ground deposition was assumed during transit to the dose receptor.

f) Reactor Coolant System (RCS) Activity After Accident

→(DRN 04-704, R14)

The RCS activity was calculated in order to determine the total amount of activity transported into the secondary system during the duration of the accident due to a 150 gal/day per SG primary to secondary leakage.

→(EC-5000081470, R301)

The RCS activity after the accident was based on assumption (c). The activity released as a result of cladding failure is assumed to be uniformly mixed with the reactor coolant. The cumulative source term is given in Table 15.4-31.

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

g) Secondary Mass Release to Atmosphere

→(DRN 04-704, R14; EC-40444, R307)

Release to atmosphere continues until the RCS reaches Shutdown Cooling entry conditions and the Shutdown Cooling System is placed in operation. However, a total combined MSSV/ADV leakage of 280 lb/hr per steam line is assumed until cold shutdown conditions.

←(DRN 04-704, R14; EC-40444, R307)

h) Operator Action Times

→(DRN 04-704, R14)

Operator action is assumed at 30 minutes to initiate cooldown to Shutdown Cooling entry conditions. FSAR Section 15.B discusses the general operator actions and cooldown scenarios assumed for steaming events analyzed for radiological consequences.

←(DRN 04-704, R14)

i) Meteorological Conditions

→(DRN 04-704, R14)

The meteorological conditions assumed to be present at the site 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 of atmospheric dispersion calculated for the exclusion area boundary or LPZ outer boundary or the Main Control Room. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

←(DRN 04-704, R14)

15.4.3.2.5.1.3 Design Basis - Uncertainties and Conservatisms

Reactor coolant equilibrium activities prior to the accident are based on the Technical Specification limit. This is typically greater than that normally observed in past PWR operation.

Steam generator equilibrium activity for both steam generators is assumed to be equal to the Technical Specification limit. The Technical Specification limits are conservatively derived based on acceptable offsite doses from events such as the steam generator tube rupture accident.

→(DRN 04-704, R14)

Loss of offsite power is a conservative assumption and is assumed consistent with RG 1.77.

←(DRN 04-704, R14)

The containment leakage rate is taken to be the leakage rate at maximum peak pressure for the first 24 hours and 50 percent of this value thereafter.

→(DRN 04-704, R14)

The secondary system mass release was maximized by assuming: 1) all heat must be removed through the steam generators, and 2) ignoring any potential flow into containment. Regulatory Guide 1.77 and RG 1.183 assumptions were used to determine the source available from containment leakage, and similarly conservative assumptions were used to determine sources available from secondary releases.

The meteorological conditions assumed to be present at the site 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 poor values of atmospheric dispersion calculated for the exclusion area boundary or the LPZ outer boundary or the Main Control Room. Furthermore, no credit is taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or the LPZ outer boundary or the Main Control Room. Hence, the radiological consequences evaluated under these conditions are conservative.

A 150 gpd per steam generator primary-to-secondary leakage is assumed for the duration of the accident until shutdown cooling is initiated.

15.4.3.2.5.1.4 Offsite Doses

The potential radiological consequences resulting from the occurrence of a postulated CEA ejection accident have been conservatively analyzed, using assumptions and models described in the preceding subsections.

←(DRN 04-704, R14)

→(DRN 05-1551, R14; EC-5000081470, R301)

The TEDE dose due to immersion and the thyroid dose due to inhalation have been analyzed for the worst two hour period at the exclusion area boundary and for the duration of the accident (0 to 30 days) at the LPZ outer boundary and the Main Control Room. The results are listed in Table 15.4-33. The resultant EAB and doses are well within the guidelines of 10CFR50.67 and the resultant Main Control Room dose meets GDC 19.

←(DRN 05-1551, R14; EC-5000081470, R301)

SECTION 15.4: REFERENCES

1. "CEA Ejection, C-E Method for Control Element Assembly Ejection Analysis," Combustion Engineering, Inc., CENPD-190-A.

2. "C-E Setpoint Methodology, C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," Combustion Engineering, Inc., CENPD-199-P.

3. Brassfield, H. C., et. al., "Recommended Property and Reaction Kinetics Data for Use in Evaluating a Light Water-Cooled Reactor Loss-of-Coolant Incident Involving Zircaloy-4 or 304-SS Clad UO_2, " GENP-482, April 1968.

4. Idaho Nuclear Corporation, Monthly Report, Ny-123-69, October 1969.

5. Idaho Nuclear Corporation, Monthly Report, Hai-127-70, March 1970.

→(DRN 05-543, R14)

6. “C-E Methods for Loss of Flow Analysis,” C-E CENPD-183, July 1975 submitted to NRC on August 22, 1975.

←(DRN 05-543, R14)

→(DRN 02-1479, R12; DRN 05-543, R14; EC-13881, R304)

SEQUENCE OF EVENTS FOR THE UNCONTROLLED CEA WITHDRAWAL FROM SUBCRITICAL CONDITIONS

Time (s)	Event	Setpoint or Value
0.0	Initiation of CEA bank withdrawal	---
571.5	Reactor reaches criticality	---
593.0	High logarithmic power level trip condition, % of rated thermal power	4.4%
593.4	Trip breakers open, and rod withdrawal stops	---
594.0	CEAs begin to drop	---
594.0	Peak core power, % of rated thermal power	58.9
594.15	Peak core heat flux, % of rated thermal power	17.3
594.15	Minimum WSSV-T / ABB-NV DNBR	≥ 1.24

←(DRN 02-1479, R12; DRN 05-543, R14; EC-13881, R304)

→(DRN 05-543, R14)

Assumptions for the Uncontrolled CEA Withdrawal from Subcritical Conditions

Parameter	Assumption
→(DRN 02-1479, R12) Initial Core Power, Fraction	1×10^{-10}
←(DRN 02-1479, R12) Core Inlet Temperature, F	552
→(DRN 02-1479, R12) RCS Flowrate, $x10^6$ lbm/hr	148
←(DRN 02-1479, R12) Pressurizer pressure, psia	2090
SG Pressure, psia	1056
MTC, 10^4 $\Delta\rho/°F$	+0.5
Doppler coefficient multiplier	0.85
→(DRN 02-1479, R12) CEA Reactivity Addition Rate, 10^{-4} $\Delta\rho$/sec	1.75
←(DRN 02-1479, R12) CEA worth on trip, %$\Delta\rho$	-6.25
One Pin Radial Peaking Factor, with Uncertainty	4.15

←(DRN 05-543, R14)

TABLE 15.4-3 Revision 304 (06/10)

→(DRN 00-1822; DRN 02-526, R12; DRN 05-543, R14; EC-13881, R304)

SEQUENCE OF EVENTS FOR THE UNCONTROLLED CEA WITHDRAWAL
AT HOT ZERO POWER

Time(s)		Event	Setpoint or Value
0.0		Initiation of CEA withdrawal	---
24.2		High Variable Overpower trip signal generated, % of rated thermal power	35
24.8		Trip breakers open	---
25.0		Peak core power occurs, % of rated thermal power	66.7
25.3		Shutdown CEAs begin to drop into core	---
25.8		Peak core average heat flux occurs, % of rated thermal power	32.6
25.8		Minimum WSSV-T / ABB-NV DNBR occurs	≥ 1.24
34.7		MSSV’s open, set pressure in psia	1085

←(DRN 00-1822; DRN 02-526, R12; DRN 05-543, R14; EC-13881, R304)

→ (DRN 05-543, R14)

ASSUMPTIONS FOR THE UNCONTROLLED CEA WITHDRAWAL AT HZP

Parameter	Assumption
→(DRN 02-526, R12; LBDCR 15-039, R309) Initial Core Power Fraction ←(DRN 02-526, R12; LBDCR 15-039, R309)	10^{-6}
Core Inlet Temperature, °F	552
RCS Flowrate, 10^6 lbm/hr	148
Pressurizer Pressure, psia	2090
SG Pressure, psia	1058
One pin radial peaking factor, with uncertainty	3.17
MTC, 10^{-4} $\Delta\rho$/°F	+0.5
Doppler coefficient multiplier	0.85
→(DRN 00-1822) CEA Reactivity Addition Rate 10^{-4} $\Delta\rho$/sec	1.6
(DRN 00-1822) CEA Worth on trip, %$\Delta\rho$	-5.0

←(DRN 05-543, R14)

TABLE 15.4-5 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-6 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-7 Revision 304 (06/10)

→(DRN 05-543, R14; EC-13881, R304)

SEQUENCE OF EVENTS FOR THE CEA WITHDRAWAL AT POWER

Time (sec)	Event	Setpoint or Value
0.	Withdrawal of CEA bank is initiated	---
11.6	CPCS VOPT generates a reactor trip signal	110% power
12.1	Reactor trip breakers open	---
12.3	Maximum core power, % of full power	110.5% power
12.4	Minimum WSSV-T / ABB-NV DNBR	≥ 1.24
12.7	CEAs begin to drop into core	---
14.8	Maximum RCS pressure, psia	2,287
21.9	Steam generator safety valves open, psia	1,038

←(DRN 05-543, R14; EC-13881, R304)

TABLE 15.4-8 Revision 14 (12/05)

→(DRN 05-543, R14)

ASSUMPTIONS FOR CEA WITHDRAWAL AT POWER

Parameter	Units	Assumption
Initial Core Power Level	MWt	3735
Core Inlet Coolant Temperature	°F	533
Reactor Coolant System Flow	10^6 lbm/hr	177.6
Pressurizer Pressure	psia	2090
Steam Generator Pressure	psia	733
Moderator Temperature Coefficient	10^{-4} $\Delta\rho$/°F	0.0
Doppler Coefficient Multiplier	---	0.85
CEA Worth on Trip	10^{-2} $\Delta\rho$	6.0
Hot Pin Radial Peaking Factor	---	1.81

←(DRN 05-543, R14)

TABLE 15.4-9 Revision 14 (12/05)

→(DRN 02-1479, R12; 05-543, R14)

ASSUMPTIONS FOR THE CEA DROP EVENT

←(DRN 02-1479, R12)

Parameter	Assumption
→(DRN 02-526, R12) Initial core power, MWt ←(DRN 02-526, R12)	3735
Core inlet temperature, °F	543
Pressurizer pressure, psia	2250
Pressurizer Level, %	67.5
RCS flowrate x 10^6 lb/hr	158.4
Dropped CEA reactivity worth, %$\Delta\rho$	-0.15
Time for CEA to be fully inserted, sec.	1
MTC, 10^{-4} $\Delta\rho$/°F	-4.2
Doppler coefficient multiplier	1.15
Prompt CEA radial distortion upon drop	1.147
15 minute Xenon radial distortion	1.097
PPCS	Auto
PLCS	Auto

←(DRN 05-543, R14)

→(DRN 02-1479)

Table 15.4-10 has been deleted.

←(DRN 02-1479)

→(DRN 02-1479)

Table 15.4-11 has been deleted.

←(DRN 02-1479)

→(DRN 02-1479)

Table 15.4-12 has been deleted.

←(DRN 02-1479)

TABLE 15.4-13 Revision 304 (06/10)

→(DRN 02-1479, R12; DRN 05-543, R14; EC-13881, R304)

SEQUENCE OF EVENTS FOR THE CEA DROP EVENT

Time (sec)	Event	Setpoint or Value
0	CEA drop initiated	---
1.0	CEA reaches full insertion	---
1.2	Core power level reaches minimum value and begins to increase due to reactivity feedbacks, % of 3716 MWt (uprate)	88.6
41	Core power returns to 100% power value, % of 3716 MWt	100
305	Core inlet temperature reaches new steady state value (°F)	539.5
900	Minimum WSSV-T / ABB-NV DNBR is reached	≥ 1.24
900	Maximum PLHGR (kW/ft)	< 21
900	Maximum Fr	2.1655
900	Core power (%)	100.62
900	Operators act to reduce core power if dropped CEA has not been realigned	---

←(DRN 02-1479, R12; DRN 05-543, R14; EC-13881, R304)

→(DRN 02-526; 02-1479)

Table 15.4-14 has been deleted.

←(DRN 02-526; 02-1479)

→(DRN 02-526; 02-1479)

Table 15.4-15 has been deleted.

←(DRN 02-526; 02-1479)

→(DRN 02-1479)

Table 15.4-16 has been deleted.

←(DRN 02-1479)

TABLE 15.4-17 Revision 9 (12/97)

ASSUMPTIONS FOR THE MODE 5 PARTIALLY DRAINED INADVERTENT BORON DILUTION ANALYSIS

Parameter	Assumption
RCS mass (drained to centerline of hot leg nozzle) lb_m	274,820
Volumetric charging rate, gpm per pump	44
Mass charging rate, lb_m /sec per pump	6.12
Dilution time constant, Υ (one charging pump), sec	44,905
Initial boron concentration C(o), ppm	1580
Critical boron concentration C_{crit}, ppm	1412
Inverse boron worth, ppm/%$\Delta\rho$	84

→(DRN 03-1610, R13)

Table 15.4-18 has been deleted.

←(DRN 03-1610, R13)

→(DRN 03-1610, R13)

Table 15.4-18(a) has been deleted.

←(DRN 03-1610, R13)

→(DRN 03-1610, R13)

Table 15.4-18(b) has been deleted.

←(DRN 03-1610, R13)

TABLE 15.4-19 Revision 309 (06/16)

⟶ (DRN 05-543, R14; 05-1551, R14; 06-1062, R15)

CEA Ejection Sequence of Events (Full Power)

Time (sec)	Event	Setpoint/Value
0.00	Mechanical failure of CEDM causes CEA to eject	---
0.04	CPC VOPT, % of full power	160.0
0.05	CEA fully ejected	---
0.06	Maximum core power occurs, % of full power	190.5
0.67	Trip breakers open	---
1.27	CEAs begin to drop into core	---
→(LBDCR 15-039, R309) 2.70	Maximum fuel centerline enthalpy in hot node, cal/gm	231.3
←(LBDCR 15-039, R309) 4.77	CEAs fully inserted, core power reduced to below 10% power	---
1800	Operator takes control of plant	---
28800	SDC initiated	---

⟵ (DRN 05-543, R14; 05-1551, R14; 06-1062, R15)

TABLE 15.4-20 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-21 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-22 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-23 Revision 309 (06/16)

→(DRN 05-543, R14; 05-1551, R14; 06-1062, R15; LBDCR 15-039, R309)

Assumptions for the CEA Ejection (Full Power)

Parameter	Assumptions
Initial Core Power, MWt	3735
Delayed Neutron Fraction, β	Minimum
Pressurizer Level, %	67.5
MTC, 10^{-4} $\Delta\rho/^{o}F$	-0.2
Ejected CEA Worth, 10^{-2} $\Delta\rho$	0.200
Doppler Coefficient Multiplier	0.85
Initial Planar Fuel Pin Peaking Factor	2.23
Ejected Planar Fuel Pin Peaking Factor	4.31
CEA Worth on Reactor Trip, %Δρ	-5.00
Postulated CEA Ejection Time, sec	0.05
RCS Flowrate, 10^{6} lbm/hr	148.0
Pressurizer Pressure, psia	2090

←(DRN 05-543, R14; 05-1551, R14; 06-1062, R15; LBDCR 15-039, R309)

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

Assumptions for the CEA Ejection Peak RCS Pressure Case

Parameter	Assumptions
Initial core power level, MWt	3735
Delayed neutron fraction	Minimum
MTC, 10^{-4} $\Delta\rho$/°F	-0.2
Ejected CEA worth, 10^{-2} $\Delta\rho$	0.20
Doppler coefficient multiplier	0.85
CEA worth on trip, 10^{-2} $\Delta\rho$	-5.00
Postulated CEA ejection time, sec	0.05
Core inlet temperature, °F	533
RCS flowrate, 10^{6} lbm/hr	170.2
Pressurizer pressure, psia	2310
Pressurizer level, %	67.5
SG pressure, psia	717
SG level, %	68

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

TABLE 15.4-25 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-26 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-27 Revision 309 (06/16)

→ (DRN 05-543, R14)

CEA Ejection Peak RCS Pressure Sequence of Events

Time (sec)	Event	Setpoint/Value
0.00	Mechanical Failure of CEDM causes CEA to eject	---
0.05	CEA fully ejected	---
0.07	High power trip condition, % of Full Power	163
0.08	Maximum core power occurs, % of Full Power	187.0
0.699	Trip breakers open	---
1.299	CEAs begin to drop into core	---
→(LBDCR15-039, R309) 3.61	Maximum RCS Pressure, psia	2632
←(LBDCR15-039, R309) 4.8	CEA fully inserted, core power reduced to below 10% power	---

← (DRN 05-543, R14)

TABLE 15.4-28 Revision 309 (06/16)

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

CEA Ejection Results (Full Power)

Parameter	Results
Total Centerline Enthalpy of Hottest Fuel Pellet, cal/gm	231.3
Fraction of Rods that Experience DNB Clad Damage	≤ 0.15
Fraction of Fuel Having at Least Incipient Centerline Melting (centerline enthalpy 231.3 cal/gm)	0.0
Fraction of Fuel Having Radially Average Enthalpy ≥ 280 cal/gm	0.0

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

TABLE 15.4-29 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

TABLE 15.4-30 Revision 14 (12/05)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 00-119)

Table 15.4-31 Sheet 1 of 2 Revision 307 (07/13)

←(DRN 00-119)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A CEA EJECTION ACCIDENT

→(DRN 04-704, R14)

Core Power Level:	3735 MWt

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

Fission Product Gap Fractions:	
Iodines	10%
Noble Gases	10%
Alkali metals (Cs & Rb-86)	12%
Fraction of Fuel Rods in Core Failing (maximum):	15%

Reactor Building Release Pathway

→(EC-5000081470, R301)

Core Inventory:	Table 12.2-12

←(EC-5000081470, R301)

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 DF
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:	Table 12.2-12A

←(EC-5000081470, R301)

Primary-to-Secondary Leak Rate:	150 gpd per SG

→(EC-40444, R307)

Total MSSV/ADV Combined Leakage per Steam Line	280 lb/hr Until Cold Shutdown

←(EC-40444, R307)

Iodine Chemical Form (Reactor Building Release Path):	
Elemental	97%
Organic	3%
Particulate	0%
Steam PF (Iodine and Alkali Metals):	
0-30 minutes	10
> 30 minutes	100
Duration of Release:	7.5 hours

←(DRN 04-704, R14)

→(DRN 00-119)

Table 15.4-31 Sheet 2 of 2 Revision 301 (09/07)

←(DRN 00-119)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A CEA EJECTION ACCIDENT

→(DRN 04-704, 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/particulate/organic)

→(EC-5000081470, R301)

Pressurization Flow	225 CFM
Unfiltered Inleakage	100 CFM

←(EC-50000081470, R301)

Breathing Rate	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:

Time	Reactor Building Unfiltered Inleakage	Reactor Building Pressurization Flow	Secondary Steaming Unfiltered Inleakage	Secondary Steaming Pressurization Flow
0-2 hr	2.77E-03	5.15E-04 *	5.37E-02	3.90E-03 *
2-8 hr	1.78E-03	3.90E-04 *	3.77E-02	2.91E-03 *
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		

(→/← EC-5000081470, R301 bracket the data rows)

* 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
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 Release:

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

←(DRN 04-704, R14)

→(DRN 00-119)

TABLE 15.4-32 Revision 14 (12/05)

←(DRN 00-119)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 00-119)

Table 15.4-33 Revision 14 (12/05)

←(DRN 00-119)

RADIOLOGICAL CONSEQUENCES OF A POSTULATED CEA EJECTION ACCIDENT

→(DRN 04-704, R14)

15% Fuel Failure	**Secondary Steaming Release Pathway**	**Reactor Building Leakage Pathway**	**Acceptance Criteria**
EAB (worst two hour dose)	≤6.3	≤6.3	6.3 Rem TEDE
LPZ (duration)	≤6.3	≤6.3	6.3 Rem TEDED
MCR	≤5	≤5	5 Rem TEDE

←(DRN 04-704, R14)

→(DRN 00-119)

TABLE 15.4-34 Revision 14 (12/05)

←(DRN 00-119)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 00-119)

TABLE 15.4-35 Revision 14 (12/05)

←(DRN 00-119)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 00-119)

TABLE 15.4-36 Revision 14 (12/05)

←(DRN 00-119)

→(DRN 05-543, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 05-543, R14)

→(DRN 03-1610, R13)

TABLE 15.4-37 Revision 13 (04/04)

MAXIMUM DIFFERENCE IN MEASURED SYMMETRIC INSTRUMENT POWER FOR REPRESENTATIVE WORST CASE UNDETECTABLE MISLOAD DURING POWER ASCENSION TESTING

Case	**Maximum Difference (%)**
Nominal (no misload)	~0
Measured by CECOR for Representative Worst Case Undetectable Misload assuming all ICIs in vicinity of misload are operable	13.1
Measured by CECOR assuming three out of four of the ICIs in vicinity of misload are inoperable	8.2

←(DRN 03-1610, R13)

→(DRN 03-1610, R13)

TABLE 15.4-38 Revision 13 (04/04)

MAXIMUM POWER PEAKING FACTORS OCCURRING FOR
REPRESENTATIVE WORST CASE UNDETECTABLE MISLOAD

Case	**Cycle Max Radial Peaking Factor**
Nominal (no misload)	1.57
Representative Worst Case Undetectable Misload	1.86
Measured by CECOR for Representative Worst Case Undetectable Misload assuming all ICIs in vicinity of misload are operable	1.75
Measured by CECOR assuming three out of four of the ICIs in vicinity of misload are inoperable	1.59

←(DRN 03-1610, R13)