15. ACCIDENT ANALYSES

15.0 TRANSIENT ANALYSES

This chapter describes the analytical evaluation of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. These events are considered here despite the many precautions taken in the design, construction and operation of the plant to prevent such occurrences. The potential consequences of such events are then examined to determine the effect on the plant and to verify that the design is adequate to assure that the health and safety of the public and of plant personnel are protected in all cases.
→(DRN 05-543, R14)
The structure of this section is based on the eight by three matrix specified in Reference 1. Initiating events are placed in one of the eight categories of process variable perturbation. The frequency of each incident was estimated, and each incident [a] was placed in one of three frequency categories specified in Reference 1.
→(DRN 02-526, R12; 05-543, R14; EC-13881, R304)
The original licensed power for Waterford 3 was 3390 MWt. During Cycle 12, the licensed power level was increased to 3441 MWt via an Appendix K uprate. Waterford 3 implemented an extended power uprate to a licensed thermal power level of 3716 MWt for Cycle 14. All of the Chapter 15 design basis events were re-evaluated for the extended power uprate. The Westinghouse reload analysis methodology was applied to determine the impact of the extended power uprate on Waterford 3. The results presented in Chapter 15 are either the result of the new analysis which was performed for extended power uprate or for which the pre-uprate analyses remain valid and bounding.
←(DRN 02-526, R12; 05-543, R14)
→(DRN 06-1062, R15)
All of the extended power uprate events were evaluated for up to 20% of the steam generator tubes plugged.
←(DRN 06-1062, R15)
→(EC-9533, R302; EC-8458, R307)
All of the events were evaluated for the implementation of Next Generation Fuel (NGF), Reference 26, with Optimized ZIRLO™.
←(EC-9533, R302; EC-8458, R307)
→(LBDCR 15-039, R309)
Waterford 3 replaced its steam generators the outage prior to Cycle 19. All of the Chapter 15 design basis events were re-evaluated for the replacement steam generators. The Westinghouse reload analysis methodology was applied to determine the impact of the replacement steam generators on Waterford 3. The results presented in Chapter 15 are either the result of the new analysis which was performed for replacement steam generators or for which the previous analyses remain valid and bounding.
←(LBDCR 15-039, R309)
All of the replacement steam generator events are also evaluated for up to 10% of the steam generator tubes plugged.
←(EC-13881, R304; EC-8458, R307)

15.0.1 IDENTIFICATION OF CAUSES AND FREQUENCY CLASSIFICATION

The analyses of events considered in this chapter are presented according to the format explained by Table 15.0-1 and illustrated in the Table of Contents for this chapter. The initiating events are each placed in one of the categories of process variable perturbation listed in Table 15.0-1. The initiating events for which analyses are presented are listed in Table 15.0-2 along with their respective section designations.

Certain initiating events which are suggested for consideration in Reference 1 have not been explicitly analyzed. Justification for omission of these analyses is provided in the appropriate subsections.

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a. Incidents are defined in this section as either the initiating event or initiating event in combination with one or more coincident component system malfunctions and the resulting transient.

The frequency of each incident has been estimated, and each incident has been placed in one of the frequency categories listed in Table 15.0-1. These frequency categories are defined as follows:

a) Moderate Frequency Incidents

These are incidents any one of which may occur during a calendar year for a particular plant.

b) Infrequent Incidents

These are incidents any one of which may occur during the lifetime of a particular plant.

c) Limiting Faults

These are incidents that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

→(DRN 05-543, R14)
←(DRN 05-543, R14)

15.0.2 SYSTEMS OPERATION

During the course of an incident, various systems may be called upon to function. These systems are described in Chapter 7 and include those systems designed to perform a safety function (see Sections 7.2 through 7.6), i.e., the operation of which is necessary to mitigate the consequences of the incident.

The Reactor Protective System (RPS) is described in Section 7.2. Table 15.0-3 lists the RPS trips for which credit is taken in the analyses discussed in this chapter, the setpoints with uncertainties, and the trip delay times associated with each trip utilized in the analyses. The response times of actuated devices after the trip setting is reached is also factored into the analyses.

The interval between the time at which the setpoint condition is reached at the sensor and the time at which the trip breakers are open is defined as the trip delay time. The trip delay times shown in Table 15.0-3 are divided into sensor delay time and RPS delay times. Sensor delay time is defined as the interval between the time at which the setpoint condition is reached at the sensor until the sensor output signal reaches the trip setpoint. This time is determined by manufacturer's test on typical sensor models. The RPS delay time is defined as the interval between the time at which the sensor output signal reaches the trip setpoint and the time at which the trip circuit breakers are fully open. This time is determined during the preoperational test of the RPS.

→(DRN 05-543, R14, LBDCR 15-039, R309)
The safety analyses conservatively assume 0.6 seconds between a trip breaker opening and when the magnetic flux of the CEA holding coils decay enough to allow CEAs to fall. Finally, the analyses conservatively assume 2.6 seconds for actual CEA insertion, i.e., from beginning of movement to 90 percent insertion, corresponding to a 3.2 second time between when the trip breakers opens until the CEA's are 90% inserted. The safety analyses assume all holding coils release their CEAs at the same time and all CEAs move at a uniform rate. The reactivity insertion due to CEA insertion is characterized by the average CEA insertion.

For example, 3.6 seconds covers the total elapsed time from a high linear power level condition at the sensor until the average CEA position is 90 percent inserted. The 3.6 seconds comprises 0.4 seconds for trip delay, 0.6 seconds for CEA holding coil magnetic flux decay, and 2.6 seconds to reach 90 percent inserted.
←(DRN 05-543, R14, LBDCR 15-039, R309)

The manner in which these systems function during each incident is discussed in the appropriate subsection.

The instrumentation which must be available to the operator in order to assist him in evaluating the incident and determining required action is described in Section 7.5. The use of this instrumentation by the operator during each incident is also discussed in the appropriate subsection.

➔(DRN 00-592, R11-A)
Systems which are not required to perform safety functions are described in Section 7.7. In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incidents more adverse. In this case, the particular control system is assumed to be inoperative. No credit is taken in the analyses for any operator action prior to initation of the event which would mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation (LCO) at the initiation of all events.
🡐(DRN 00-592, R11-A)

➔(DRN 05-543, R14)
All of the events included the impact of a single failure in the reactor protection system. The effects of malfunctions single active components which are called upon to respond to the transient are also considered. The complicating presence of these single active failures in addition to the event initiators would move an event to a less frequent event category. In some instances, a loss of offsite ac power, which would require the failure to fast transfer of two independent buses, is the most severe failure. TRM Bases Sections 3/4.3.1 and 3/4.3.2 document the CPC delay times for the different CPC functions that are acceptable to assume in accident and transient analyses.

➔(DRN 00-126, R11)
🡐 DRN 00-126, R11; 05-543, R14)

15.0.3 CORE AND SYSTEM PERFORMANCE

15.0.3.1 Mathematical Model

The Nuclear Steam Supply System (NSSS) response to various incidents is simulated using digital computer programs and analytical methods most of which are documented in Reference 2 and have been approved for use by the NRC by Reference 3. Those programs and methods not documented in Reference 2 are documented either in topical reports which have been submitted to the NRC for review or in quality assured record calculations.

15.0.3.1.1 Loss of Flow Analysis Method

➔(DRN 05-543, R14; EC-13881, R304; EC-8458, R307)
The method used to analyze incidents which are initiated by a decrease in reactor coolant flowrate (Section 15.3) is the method documented in topical report CENPD-183, Reference 4, except that the HERMITE Space-Time Kinetics Code, Reference 17, is used to determine the spacial variation in core power during the CEA insertion following a reactor trip was used. This method is detailed in Appendix 15-A of Reference 2. The loss of flow methodology uses the TORC computer code and the WSSV-T and ABB-NV CHF correlation Reference 27, (See Section 4.4) to calculate both the time and value of the minimum DNBR during the transient. UFSAR Figures (DNBR vs. Time) and Sequence of Events Tables have been revised using the WSSV-T and ABB-NV CHF correlation as part of the Cycle 17 full core NGF implementation.
🡐(DRN 05-543, R14; EC-13881, R304; EC-8458, R307)

15.0.3.1.2 CEA Ejection Analysis Method

➔(DRN 05-543, R14)
The method used for analyzing the reactivity and power distribution anomalies initiated by CEA ejection (Subsection 15.4.3.2) is documented in topical report CENPD-190.
🡐(DRN 05-543, R14)

15.0.3.1.3 Anticipated Transients Without Scram (ATWS) Analysis Method

The method used to analyze the consequences of ATWS (See Section 15.8) is described in topical report CENPD-158, Revision 1.

15.0.3.1.4 CESEC Computer Program

The CESEC computer program is used to simulate the NSSS. This program is described in Reference 7 and was used in the analysis for Reference 2.

→(DRN 00-592, R11-A)
CESEC computes key system parameters during a transient including core heat flux, pressures, temperatures, and valve actions. Symmetric and asymmetric plant responses over a wide range of operating conditions can be determined by CESEC. The following is a partial list of the dynamic functions included in this NSSS simulation:
←(DRN 00-592, R11-A)

point kinetics
doppler and moderator reactivity feedback
boron and CEA reactivity effects
multi-node average and hot channel reactor core thermal hydraulics
reactor coolant pressurization and mass transport
Reactor Coolant System safety valve behavior
steam generation
steam generator water level
main steam bypass
secondary safety and turbine valve behavior (and alarms)
control system functions
protective system functions
→(EC-8458, R307)
engineered safety feature system functions
←(EC-8458, R307)

The turbine and its associated controls are not included in the simulation. Steam generator feedwater enthalpy and flowrate are provided as input to CESEC.

During execution, CESEC obtains steady-state and transient solutions to the set of equations that mathematically models the NSSS dynamic functions mentioned above. Simultaneous numerical integration of a set of nonlinear, first-order ordinary differential equations with time-varying coefficients is carried out by means of a predictor corrector Runge-Kutta scheme. As the time variable evolves, edits of the principle system parameters are printed at specified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water and zircaloy is incorporated into this program.

→(DRN 05-543, R14)
←(DRN 05-543, R14)

15.0.3.1.5 CESEC-ATWS Computer Program

The CESEC-ATWS computer program is used to simulate the NSSS. The program is described in References 8 through 12 and was used in the analyses for Reference 2.

→(DRN 05-543, R14)
15.0.3.1.6 CENTS Computer Program

CENTS in an interactive, high fidelity computer code for the simulation of the Nuclear Steam Supply System (NSSS) components. It calculates the transient behavior of a PWR for normal and abnormal conditions including accidents. CENTS determines the core power and heat transfer throughout the NSSS. It also computes the thermal and hydraulic behavior of the reactor coolant in the primary and secondary systems. Primary and secondary thermal-hydraulic behavior is calculated with detailed multi-mode and flowpath models. CENTS includes the primary and secondary control systems and the balance-of-plant fluid systems. CENTS also calculates the time dependent flow for RCP coastdown, replacing the COAST code for this function.
←(DRN 05-543, R14)

→(DRN 05-543, R14)
CENTS incorporates a number of features that enhance is usefulness. First principle models provided a high degree of fidelity and flexibility. Use of nonequilibrium, nonhomogeneous models allows a full range of fluid conditions to be represented, including forced circulation, natural circulation and coolant voiding. The code simulates a wide range of variations in plant state from steady state conditions to severe accidents. It provides a full range of interactions between the analyst, the reactor control systems and the NSSS. Further information on CENTS can be found in Reference 13.

15.0.3.1.7 STRIKIN-II Computer Program

The STRIKIN-II computer program is used to simulate the heat conduction within a reactor fuel rod and its associated surface heat transfer. The STRIKIN-II program is described in Reference 14.

The STRIKIN-II computer program provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot cylindrical fuel rod, and the extent of the zirconium water reaction for that fuel rod. STRIKIN-II includes:

a) Incorporation of all major reactivity feedback mechanisms

b) A maximum of six delayed neutron groups

c) Both axial (maximum of 20) and radial (maximum of 20) segmentation of the fuel element

d) CEA trip initiation on high neutron power

15.0.3.1.8 TORC and CETOP Computer Program

The TORC and CETOP computer programs are used to simulate the fluid conditions within the reactor core and to predict the existence of DNB on the fuel rods. The TORC program is described in Section 4.4 and was used in the analysis for Reference 2. The CETOP computer program is described in Reference 19.

15.0.3.1.9 HERMITE Computer Program
→(DRN 00-1822, R10)
The HERMITE space-time kinetics computer code is used for the analysis of transients by means of numerical solution to the multi-dimensional, few-group, time-dependent neutron diffusion equation including feedback effects of fuel temperature, coolant temperature, coolant density and control rod motion. The time-dependent neutron diffusion equation is solved by a finite element method. The heat conduction equation in the pellet, gap and clad is solved by a finite difference method. Continuity and energy conservation equations are solved for the coolant enthalpy and density. Further information on HERMITE is found in Reference 17.
←(DRN 00-1822, R10)

15.0.3.1.10 Reactor Physics and Fuel Performance Computer Programs

Numerous computer programs are used to produce the reactor physics and fuel rod performance input parameters required by the NSSS simulation and reactor core programs performance described above. These computer programs are described in Sections 4.3 and 4.4.

15.0.3.1.11 Loss of Coolant Accident Analysis Method
→(EC-13881, R304)
The method used to analyze the consequences of the loss of coolant accident (Section 15.6) is described in topical reports CENPD-132, Reference 15, and CENPD-137P, Reference 16.
←(EC-13881, R304)

→(DRN 00-1822, R10)

15.0.3.1.12 DNBR Propagation Analysis Method

→(DRN 04-524, R13)

The Waterford 3 fuel performance analysis has relied on use of No Clad Lift Off (NCLO) methodology. It is a requirement of this methodology that "no DNBR propagation" must be demonstrated in accordance with the approved topical report, Reference 21.

←(DRN 04-524, R13)

Since the length of time in DNB for all of the events evaluated for DNB propagation is substantially less than the limiting values of Reference 21, it is concluded that DNB propagation is not a concern.

←(DRN 00-1822, R10)

→(DRN 05-543, R14)

15.0.3.1.13 ORIGEN2 Computer Program

ORIGEN2 is a computer code system for calculating buildup, decay, and processing of radioactive materials. ORIGEN2 is a revised version of ORIGEN and incorporates updates of the reactor models, cross sections, fission product yields, decay data, and decay photon data as well as the source code. The ORIGEN2 program is described in Reference 25.

←(DRN 05-543, R14)

15.0.3.2 Initial Conditions

The incidents discussed in this section have been analyzed over a range of values for the principal process variables that affect the margin to fuel thermal design limits. These variables are core power level, core power distribution, core inlet coolant flowrate, core inlet coolant temperature, and system pressure.

Analyses over a range of initial conditions are compatible with the monitoring function performed by both COLSS, described in Section 7.7, and the other limiting conditions for operation. The required margin to initiating event produces the most rapid loss of margin to DNB before reactor trip and the maximum loss of margin to DNB after reactor trip. Most often, postulated initiating events do not require as much initial margin, as evidenced by the fact that the reactor trip may be delayed somewhat (i.e., the time of trip may be greater than 0.6 seconds) without causing a violation of the specified acceptable fuel design limit on DNB. The required margin to fuel centerline melting incorporated in COLSS is established by the loss of coolant accident (LOCA) as described in Subsection 15.6.3.3.

→(LBDCR 15-039, R309)

The range of values of each of the principal process variables that were considered in analyses of all incidents discussed in this section are listed in Table 15.0-4. Values beyond this range listed in Table 15.0-4 are used in certain analyses to provide bounding or conservative results. It is emphasized that no plant operational or safety problems have been identified for operating conditions outside of the range shown in Table 15.0-4. This range merely represents a range of expected normal reactor operation.

←(LBDCR 15-039. R309)

15.0.3.3 Input Parameters

The parameters used in the analyses were based upon those generated in the design of a specific fuel cycle. Where possible the input parameters are modified in the conservative direction to encompass minor cycle-by-cycle variations. For each fuel cycle, the parameters for the proposed core are compared to the input parameters used in the current analysis of record. If the input parameters for a particular transient have not become worse, then the transient is "bounded" by the analysis of record. If the input parameters have become more adverse, then it is necessary to determine if the results (e.g., % of fuel failure, minimum DNBR, etc.) of the transient are worse than the results reported for the analysis of record.

15.0.3.3.1 Doppler Coefficient

→(DRN 05-543, R14)
The Doppler reactivity values used in the safety analysis vary according to the burnup dependent isotopics in the core, varying from minimum reactivity feedbacks at BOC to maximum reactivity feedbacks at EOC. Included in the values used are multipliers to account for uncertainty of 0.85 for those cases in which it is conservative to minimize Doppler and a multiplier of 1.30 for those cases in which it is conservative to maximize Doppler.
←(DRN 05-543, R14)

15.0.3.3.2 Moderator Temperature Coefficient

→(DRN 05-543, R14)
Allowances are included to account for:

a) Changes in values from cycle to cycle
←(DRN 05-543, R14)

b) Changes in coefficient that might occur due to design changes

→(EC-13881, R304)
c) Changes in coefficient that might occur due to differences between design parameters and as built parameters (such as shim loadings, enrichments, etc.)
←(EC-13881, R304)

d) Any changes in parameters that might occur during a cycle

→(EC-13881, R304)
e) Calculational uncertainties or biases
←(EC-13881, R304)

→(DRN 05-1201, R14)
The analyses considered the entire range of moderator temperature coefficient values allowed by plant COLR. The most positive MTC values vary from $+0.5 \times 10^{-4}$ $\Delta\rho$/°F at zero power to -0.2×10^{-4} $\Delta\rho$/°F. The most negative MTC specified in the COLR will vary between cycles; most but not all (e.g., Return-to-Power Main Steam Line Break) analyses have assumed a value of -4.2×10^{-4} $\Delta\rho$/°F at HFP. The most restrictive value for each Cycle is specified in the COLR.

→(EC-13881, R304)
In some circumstances, due to variations of other core parameters with burnup, it is desirable to model the MTC at BOC and later in the cycle separately. When this is done, the BOC behavior is taken as the most positive MTC quoted above.
←(DRN 05-1201, R14; EC-13881, R304)

→(DRN 00-1822; 05-543, R14)
←(DRN 00-1822; 05-543, R14)

15.0.3.3.3 Shutdown CEA Reactivity

→(DRN 05-543, R14; EC-13881, R304)
The shutdown reactivity and the rapidity with which it acts during the transients is dependent on the CEA worth available on reactor trip, the axial power distribution, the position of the regulating CEAs, and the time in cycle life. Most of the Chapter 15 analyses assume a minimum total negative reactivity worth of the CEAs available of -6.0% $\Delta\rho$ at full power and -5.0% $\Delta\rho$ at hot zero power. These represent values reduced from explicit calculations to encompass possible variations in scram worth in future cycles. These values are based on the assumption that the most reactive CEA is stuck in the fully withdrawn position and includes effects of cooldown to hot zero power conditions.
←(EC-13881, R304)

The CEA ejection event uses these values further reduced by an allowance for the ejected CEA. The return to power SLB event factors a cycle specific calculation of scram worth into an overall reactivity balance.
←(DRN 05-543, R14)

For all accident analyses except major RCS pipe rupture analyses, moderator void reactivity feedback effects are not taken into account. If no credit is taken for the negative reactivity which would accompany the possible generation of voids during the course of a transient, then it is conservative to ignore the positive reactivity feedback associated with the subsequent collapse of these voids. This assumption is conservative because any positive feedback associated with collapse of the voids will not exceed the negative feedback associated with void generation. Since there is substantial void formation in the core during the major RCS pipe rupture transients, the reactivity feedback effect of the growth and collapse of these voids is modeled for analyses.

The shutdown worth vs. position is calculated by assuming that the core is initially unrodded, i.e., all CEAs are fully withdrawn. This assumption is made since:

→(EC-8458, R307)
a) The unrodded core allows the highest permissible axial peak to be used for the transient calculation

b) Dropping CEAs into an initially unrodded core is more conservative in terms of the initial rate of negative reactivity insertion
←(EC-8458, R307)
→(LBDCR 15-039, R309)
The CEA SCRAM rod insertion curve has been revised to change the average CEA drop time (at 90% insertion) from 3.0 seconds to 3.2 seconds. Table 15.0-5 documents both the original insertion curve and the revised insertion curve for use in the Chapter 15 analyses.
←(LBDCR 15-039, R309)
→(DRN 05-543, R14)
←(DRN 05-543, R14)

15.0.3.3.4 Effective Delayed Neutron Fraction

→(DRN 05-543, R14)
The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup. The delayed neutron fraction is largest at BOC and steadily becomes smaller throughout core life. Allowances for uncertainties are applied to increase (BOC) or decrease (EOC) the delayed neutron fractions in the conservative direction. The events use the value of delayed neutron fraction which gives the most conservative result, regardless of the time in cycle used for other parameters.

	Neutron Lifetime (10^{-6} Sec)	Delayed Neutron Fraction
Beginning of Cycle (Max)	10	0.007879
End of Cycle (Min)	50	0.004120

←(DRN 05-543, R14)

→(DRN 05-543, R14)
The sole exception to this is the return-to-power steam line break event. This event is more adverse at EOC, but also is more adverse with a larger delayed neutron fraction. For this event an EOC maximum delayed neutron fraction is computed in which uncertainties are applied to make the EOC delayed neutron fraction larger.

	Neutron Lifetime (10^{-6} Sec)	Delayed Neutron Fraction
End of Cycle (Min)	50	0.005662

←(DRN 05-543, R14)

15.0.3.3.5 Decay Heat Generation Rate

Analyses based upon full power initial conditions utilize a decay heat generation rate based conservatively upon infinite reactor operation at full power.

15.0.4 BARRIER PERFORMANCE

15.0.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.0.3.1.

15.0.4.2 Initial Conditions

The initial conditions used for evaluation of barrier performance are identical to those described in Subsection 15.0.3.2.

15.0.4.3 Input Parameters

The input parameters used for evaluation of barrier performance are identical to those described in Subsection 15.0.3.3.

15.0.5 RADIOLOGICAL CONSEQUENCES

The assumptions, parameters, and calculational methods used to determine the doses that result from postulated accidents are summarized in this subsection and the accidents for which radiological consequences were quantitatively analyzed are listed below. The radiological consequences of other accidents are referred to these accidents as appropriate.

Accidents for which radiological consequences are quantitatively analyzed are:

a) Moderate Frequency Incidents

→(EC-13881, R304)

←(EC-13881, R304)

b) Infrequent Incidents

➔(DRN 05-543, R14)

1) Subsection 15.1.2.3 - Increased Main Steam Flow with a Concurrent Loss of Offsite Power

2) Subsection 15.1.2.4 - Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with Concurrent Single Failure of an Active Component.

🡐(DRN 05-543, R14)

c) Limiting Faults

➔(EC-13881, R304)

1) Subsection 15.1.3.3* - Steam System Piping Failures: Pre-Trip Power Excursion Analysis

2) Subsection 15.2.3.1 - Feedwater System Pipe Break

3) Subsection 15.3.3.1 - Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

🡐(EC-13881, R304)

4) Subsection 15.4.3.2 - CEA Ejection

5) Subsection 15.6.3.1 - Primary Sample or Instrument Line Break

6) Subsection 15.6.3.2 - Steam Generator Tube Rupture

➔(EC-13881, R304)

7) Subsection 15.6.3.3 - Loss of Coolant Accident (LOCA)

➔(DRN 05-1551, R14)

8) Subsection 15.7.3.4 - Design Basis Fuel Handling Accident

🡐(DRN 05-1551, R14)

➔(DRN 04-704, R14)
🡐(DRN 04-704, R14)

➔(DRN 05-543, R14; 05-1551, R14)

For each limiting fault an analysis using design basis assumptions was used to establish that the plant design meets 10CFR50.67 criteria. The number of fuel rods predicted to fail as a result of DNB is determined by the method of statistical convolution except for the return to power MSLB. The convolution methodology consists of integrating (or convoluting) the product of the number of fuel rods reaching various values of DNBR with the associated probability of fuel failure for that DNBR. This methodology is described in Reference 4.

🡐(DRN 05-1551, R14; EC-13881, R304)

Fuel rods also are assumed to fail as result of fuel center line melt. The fuel center line melt criterion is met if at least one of the following two conditions is met:

1) The peak linear heat rate (LHR) remains below the steady state line of 21 kW/ft, or,

2) The fuel temperature remains below the fuel centerline melt temperature during the transient as determined by calculating the deposited energy balance during the transient. Guidance for evaluating fuel centerline temperature is provided in Technical Specification 2.1.1.2 and it associated Bases.

🡐(DRN 05-543, R14)

Information used repetitively throughout the chapter is provided in Appendix 15B. This appendix contains information on dose models, containment leakage models and control room models.

➔(DRN 05-543, R14; EC-13881, R304; EC-8458, R307)
🡐(DRN 05-543, R14; EC-13881, R304; EC-8458, R307)

SECTION 15.0: REFERENCES

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3. "Combustion Engineering Standard Safety Analysis Report (CESSAR System 80 Nuclear Steam Supply System Standard Nuclear Design Preliminary Design Approval," PDA-2, Docket No. STN 50-470, NRCP December 31, 1975.

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

5. "C-E Method for Control Element Assembly Ejection Analysis," C-E, CENPD-190-A, January 1976, approved for use on June 10, 1976.

→(EC-13881, R304)

6. "Analyses of Anticipated Transients without Reactor Scram in Combustion Engineering NSSS," C-E CENPD-158, Rev. 1., May 1976, submitted to NRC on June 29, 1976.

←(EC-13881, R304)

→(DRN 00-1822)

7. "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, December 1981.

←(DRN 00-1822)

8. "ATWS Model Modifications to CESEC," CENPD-107, Supplement 1, September 1974, Proprietary Information.

9. "ATWS Model Modifications to CESEC," CENPD-107, Supplement 1, Amendment I-P, November 1975, Proprietary Information.

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12. "ATWS Model Modifications to CESEC," CENPD-107, Supplement 4-P, December 1975, Proprietary Information.

→(DRN 05-543, R14; EC-13811, R304)

13. "Technical Manual for the CENTS Code,“ CENPD-282-P-A, Rev. 2, March 2005.

←(DRN 05-543, R14; EC-13881, R304)

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→(DRN 00-1822; EC-13881, R304)
15. "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985.

→(DRN 05-543, R14)
"Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," CENPD-132, Supplement 4-P-A, March 2001.
←(EC-13881, R304)

16. "Calculative Methods for the ABB C-E Small Break LOCA Evaluation Model," CENPD-137-P, August 1974: Supplement 2-P-A, April 1998
←(DRN 00-1822; 05-53, R14

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

18. Letter from K. W. Cook (LPL) to George W. Knighton (NRC) "Waterford 3 SES, Docket No. 50-382, Reload Analysis Report (RAR)," August 29, 1986.

→(EC-13881, R304)
19. "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs 1 and 2", CEN-191(B)-P, December 1981.
←(EC-13881, R304)

20. Letter from K. W. Cook (LPL) to George W. Knighton (NRC) "Waterford 3 SES, Docket No. 50-382, Reload Cycle 2 Reports," October 1, 1986.

→(DRN 00-1822)
21. "Fuel Rod Maximum Allowable Gas Pressure," CEN-372, May 1990 (Proprietary).
←(DRN 00-1822)
→(DRN 02-526, R12)
22. Waterford 3 Technical Specification Change Request NPF-38-238 Appendix K Margin Recovery - Power Uprate Request, September 21, 2001, Letter No. W3F1-2001-0091.
←(DRN 02-526, R12)

→(DRN 05-543, R14)
23. "CE Methods for Loss of Flow Analysis," CENPD-183, July 1997.

24. "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P-A, April 1986.

25. "ORIGEN2 Isotope Generation and Depletion Code – Matrix Exponential Method," CCC-371, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1983.
←(DRN 05-543, R14)
→(EC-13881, R304)
26. WCAP-16500-P-A, "CE 16 x 16 Next Generation Fuel Core Reference Report," August 2007.
→(EC-8458, R307)
27. WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," August 2007.
←(EC-13881, R304; EC-8458, R307)

TABLE 15.0-1 Revision 304 (06/10)

CHAPTER 15 SUBSECTION DESIGNATION

Each subsection is identified as 15.W.X.Y.Z with trailing zeros omitted where:

W =	1	Increase in Heat Removal by the Secondary System (Turbine Plant)
	2	Decrease in Heat Removal by the Secondary System (Turbine Plant)
	3	Decrease in Reactor Coolant System Flow Rate
	4	Reactivity and Power Distribution Anomalies
	5	Increase in Reactor Coolant System Inventory
	6	Decrease in Reactor Coolant System Inventory
	7	Radioactive Release from a Subsystem or Component
	8	Anticipated Transients Without Scram
→(EC-13881, R304)		
	9	Miscellaneous
←(EC-13881, R304)		
X =	1	Moderate Frequency Incidents
	2	Infrequent Incidents
	3	Limiting Faults
Y =		Initiating Event (see Subsection 15.0.1)
Z =	1	Identification of Causes and Frequency Classification
	2	Sequence of Events and Systems Operation
	3	Core and System Performance
	4	Barrier Performance
	5	Radiological Consequences

TABLE 15.0-2 (Sheet 1 of 2) Revision 307 (07/13)

CHAPTER 15 INITIATING EVENTS

Subsection	Event
	MODERATE FREQUENCY INCIDENTS
15.1.1.1	Decrease in feedwater temperature
15.1.1.2	Increase in feedwater flow
15.1.1.3	Increased main steam flow
15.1.1.4	Inadvertent opening of a steam generator atmospheric dump valve
15.2.1.1	Loss of external load
15.2.1.2	Turbine trip
15.2.1.3	Loss of condenser vacuum
15.2.1.4	Loss of normal ac power
➔(EC-13881, R304)	
15.2.1.5	Steam Pressure Regulator Failure
←(EC-13881, R304)	
15.3.1.1	Partial loss of forced reactor coolant flow
15.4.1.1	Uncontrolled CEA withdrawal from a subcritical
15.4.1.2	Uncontrolled CEA withdrawal at low power
15.4.1.3	Uncontrolled CEA withdrawal at power
15.4.1.4	CEA misoperation
15.4.1.5	CVCS malfunction (inadvertent boron dilution)
15.4.1.6	Startup of an inactive reactor coolant system pump
➔(EC-13881, R304)	
15.4.1.7	Uncontrolled CEA Withdrawal From A Subcritical Condition – Modes 3, 4 or 5 With All CEAs on the Bottom
←(EC-13881, R304)	
15.5.1.1	CVCS malfunction
15.5.1.2	Inadvertent operation of the ECCS during power operation
15.9.1	Assymetric steam generator transient
	INFREQUENT INCIDENTS
15.1.2.1	Decrease in feedwater temperature [(a)]
15.1.2.2	Increase in feedwater flow [(a)]
15.1.2.3	Increased main steam flow [(a)]
15.1.2.4	Inadventent opening of a steam generator atmospheric dump valve [(a)]
15.2.2.1	Loss of external load [(a)]
15.2.2.2	Turbine trip [(a)]
15.2.2.3	Loss of condenser vacuum [(a)]
15.2.2.4	Loss of normal ac power [(a)]
15.2.2.5	Loss of normal feedwater flow
15.3.2.1	Total loss of forced reactor coolant flow
15.3.2.2	Partial loss of forced reactor coolant flow [(a)]
15.5.2.1	CVCS malfunction [(a)]
	LIMITING FAULTS
Subsection	Event
➔(EC-8458, R307)	
15.1.3.1	Steam system piping failures Post-Trip Return-to-Power
←(EC-8458, R307)	
➔(EC-13881, R304)	
15.1.3.2	Steam system piping failures inside and outside containment Modes 3 and 4 with all CEAs on the bottom
15.1.3.3	Steam system piping failures: pre-trip power excursion analysis
←(EC-13881, R304)	
15.2.3.1	Feedwater system pipe breaks
15.2.3.2	Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System

(a) These incidents involve the same initiating event as the corresponding moderate frequency incidents but include either a concurrent single active component failure or a single operator error.

TABLE 15.0-2 (Sheet 2 of 2) Revision 304 (06/10)

CHAPTER 15 INITIATING EVENTS

LIMITING FAULTS

Subsection	Event
→(EC-13881, R304)	
15.3.3.1	Single Reactor Coolant Pump (RCP) Shaft Seizure / Sheared Shaft
←(EC-13881, R304)	
→(DRN 00-592, R11-A)	
15.4.3.1	Inadvertent Loading of a Fuel Assembly into the Improper Position
←(DRN 00-592, R11-A)	
15.4.3.2	CEA Ejection
15.6.3.1	Primary Sample or Instrument Line Break
15.6.3.2	Steam Generator Tube Rupture
15.6.3.3	Loss of Coolant Accident
15.6.3.4	Inadvertent Opening of a Pressurizer Safety Valve
15.7.3.1	Radioactive Waste Gas System Leak or Failure
→(EC-13881, R304)	
←(EC-13881, R304)	
15.7.3.4	Design Basis Fuel Handling Accidents
15.7.3.5	Spent Fuel Cask Drop Accidents
15.8	Anticipated Transients Without Scram

TABLE 15.0-3 Revision 307 (07/13)

REACTOR PROTECTIVE SYSTEM TRIPS USED IN THE SAFETY ANALYSES

→(DRN 05-543, R14)

Events	Analytical Setpoint	Trip Delay Time ** (Sec)
High Logarithmic Power Level, %	4.4	0.4
→(EC-13881, R304; EC-8458, R307)		
CPC Low DNBR Safety Limit	1.24***	0.332
←(EC-13881, R304; EC-8458, R307)		
CPC Variable Overpower Trip		0.370
Offset, % RTP	8	
Floor, % RTP	30	
Ceiling, % RTP	110	
Rate, % RTP/Min	2	
CPC Low Pump Speed, %	96.5	0.232
→(EC-8458, R307)		
CPC Hot Leg Saturation, °F	13	2.952
←(EC-8458, R307)		
CPC Asymmetric Cold Leg Temp, °F	11	0.370
→(EC-8458, R307)		
CPC Pressure out of Range, PSIA	1736	0.370
←(EC-8458, R307)		
Low RCS Flow (SGΔP), % Flow	60	0.7
High Pressurizer Pressure, PSIA	2422	0.9
Low Pressurizer Pressure, PSIA	1560	0.9
Low Steam Generator Water Level, % NR*	5	0.9
Low Steam Generator Pressure, PSIA	576	0.9

* Percent of distance between the narrow range liquid level taps, referenced from the lower tap. The analysis setpoint corresponds to a water level 27.0 ft. above the tube sheet.

** Response time used in modeling the RPS trips in the Safety Analysis

←(DRN 05-543, R14)

→(EC-8458, R307)

*** Although the WSSV-T/ABB-NV CHF correlation DNBR limit is 1.24, the existing CPC algorithm will retain the CE-1 CHF correlation and DNBR limit of 1.26. The BERRi constants are calculated such that a CPC trip at a DNBR=1.26 using the CE-1 CHF correlation in CPCS assures that the bounding DNBR=1.24 for the WSSV-T/ABB-NV CHF correlations will not be violated during normal operations and anticipated operational occurrences.

←(EC-8458, R307)

TABLE 15.0-4 Revision 309 (06/16)

CHAPTER 15 INITIAL CONDITIONS

Parameter	Units	Range
→(DRN 02-526, R12)		
Core power, B *	% of rated thermal power	B ≤ rated thermal power plus power measurement uncertainty
←(DRN 02-526, R12)		
→ (DRN 05-543, R14)		
Radial 1-pin peaking factor, F_R (with uncertainty)	--	$F_R \leq 1.65$
Axial shape index, ASI[(a)]	--	-0.2 ≤ ASI ≤ +0.2**
Core inlet coolant flowrate, G	% of 148×10^6 lbm/hr	100 ≤ G ≤ 115
Core inlet coolant temperature, T	°F	533 ≤ T ≤ 552
System pressure, P	psia	2,090 ≤ P ≤ 2,310
←(DRN 05-543, R14)		
→(DRN 06-1062, R15; EC-8458, R307)		
No. of Replacement Steam Generator Tubes Plugged	# / SG	≤897
←(EC-8458, R307)		
Asymmetry in number of tubes plugged in two SGs	#	≤500
←(DRN 06-1062, R15)		

(a) $$ASI = \frac{A_L - A_u}{A_L + A_u}$$

where:

A_L = area under axial shape in lower half of core.

A_u = area under axial shape in upper half of core.

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

* 3735 Mwth

←(DRN 02-526, R12) LBDCR-039, R309

** ≥50% Rated Thermal Power

←(DRN 05-543, R14)

TABLE 15.0-5 Revision 309 (06/16)

→LBDCR 15-039, R309)

CHAPTER 15 AVERAGE CONTROL ELEMENT ASSEMBLY (CEA) INSERTION TIMES

Average CEA Insertion (%)	Original Time (Seconds)	Revised Time (Seconds)
0	0.00	0.00
0	0.60	0.60
5	0.80	0.95
10	0.95	1.15
20	1.25	1.45
30	1.55	1.75
40	1.80	2.00
50	2.05	2.25
60	2.30	2.50
70	2.535	2.75
80	2.75	2.95
90	3.0	3.20
100	3.50	3.50

←LBDCR 15-039, R309)