CHAPTER 15

ACCIDENT ANALYSES

TABLE OF CONTENTS

Section	Title	Page
15.0	ACCIDENT ANALYSES	15.0-1
15.0.3.2	Initial Conditions	15.0-1
15.0.11.1	FACTRAN Computer Code	15.0-1
15.0.11.6	ANC Computer Code	15.0-1
15.0.13	OPERATOR ACTIONS	15.0-2
15.0.15	COMBINED LICENSE INFORMATION	15.0-2
15.0.16	REFERENCES	15.0-2
15.1	INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM	15.1-1
15.1.5.4.1	Source Term	15.1-1
15.1.5.4.6	Doses	15.1-1
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	15.2-1
15.2.6.1	Identification of Causes and Accident Description	15.2-1
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE	15.3-1
15.3.3.3.1	Source Term	15.3-1
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES	15.4-1
15.4.8.1.1.3	Reactor Protection	15.4-1
15.4.8.1.2	Limiting Criteria	15.4-1
15.4.8.2	Analysis of Effects and Consequences	15.4-2
15.4.8.2.1	Calculation of Basic Parameters	15.4-3
15.4.8.2.1.1	Ejected Rod Worths and Hot Channel Factors	15.4-3
15.4.8.2.1.2	Reactivity Feedback Weighting Factors	15.4-3
15.4.8.2.1.3	Moderator and Doppler Coefficients	15.4-4
15.4.8.2.1.4	Delayed Neutron Fraction βeff	15.4-4
15.4.8.2.1.5	Trip Reactivity Insertion	15.4-4
15.4.8.2.1.7	Results	15.4-4
15.4.8.2.1.8	Fission Product Release	15.4-6
15.4.8.2.1.9	Peak Reactor Coolant System Pressure	15.4-6
15.4.8.3	Radiological Consequences	15.4-6
15.4.8.3.1	Source Term	15.4-6

TABLE OF CONTENTS (Continued)

Section	Title	Page
15.4.8.3.5	Identification of Conservatisms	15.4-7
15.4.8.3.6	Doses	15.4-7
15.4.10	REFERENCES	15.4-7
15.5	INCREASE IN REACTOR COOLANT INVENTORY	15.5-1
15.6	DECREASE IN REACTOR COOLANT INVENTORY	15.6-1
15.6.2.6	Doses	15.6-1
15.6.3.3.1	Source Term	15.6-1
15.6.3.3.6	Doses	15.6-1
15.6.5.3.2	In-containment Activity Removal Processes	15.6-2
15.6.5.3.5	Main Control Room Dose Model	15.6-2
15.6.5.3.7.3	Atmospheric Dispersion Factors	15.6-2
15.6.5.3.8.1	Offsite Doses	15.6-3
15.6.5.3.8.2	Doses to Operators in the Main Control Room	15.6-3
15.6.6	REFERENCES	15.6-3
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT	15.7-1
15.7.4.5	Offsite Doses	15.7-1
15.7.6	COMBINED LICENSE INFORMATION	15.7-1
15.8	ANTICIPATED TRANSIENTS WITHOUT SCRAM	15.8-1
APP. 15A	EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS	15A-1
15A.3.1.2	Secondary Coolant Source Term	15A-1
15A.3.3	ATMOSPHERIC DISPERSION FACTORS	15A-1
APP. 15B	REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA	15B-1
15B.1	ELEMENTAL IODINE REMOVAL	15B-1

LIST OF TABLES

Number	Title
15.0-201	Summary of Initial Conditions and Computer Codes Used
15.1-201	Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break
15.3-201	Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident
15.4-201	Time Sequence of Events for Incidents which Result in Reactivity and Power Distribution Anomalies
15.4-202	Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident
15.4-203	Deleted
15.6-201	Parameters Used in Evaluating the Radiological Consequences of a Small Line Break Outside Containment
15.6-202	Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture
15.6-203	Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-Of-Coolant Accident
15.6-204	Radiological Consequences of a Loss-of-Coolant Accident with Core Melt
15.7-201	Assumptions Used to Determine Fuel Handling Accident Radiological Consequences

LIST OF FIGURES

Number	Title
15.4-201	Nuclear Power Transient Versus Time for the PCMI Rod Ejection Accident
15.4-202	Nuclear Power Transient Versus Time for the High Cladding Temperature Rod Ejection Accident
15.4-203	Nuclear Power Transient Versus Time for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident
15.4-204	Not Used

CHAPTER 15

ACCIDENT ANALYSES

15.0 ACCIDENT ANALYSES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.0.3.2 Initial Conditions

Add the following paragraph at the end of DCD Subsection 15.0.3.2.

STD COL 15.0-1

The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System (Reference 201), which will be calibrated (in a certified laboratory using a piping configuration representative of the plant piping design) prior to installation and will be tested after installation in the plant in accordance with the LEFM CheckPlus commissioning procedure. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. The calculated calorimetric is done in accordance with a previously accepted Westinghouse methodology (Reference 202). Administrative controls implement maintenance and contingency activities related to the power calorimetric instrumentation.

15.0.11.1 FACTRAN Computer Code

LNP DEP 6.4-1

Revise the first paragraph and first bullet of DCD Subsection 15.0.11.1 to read as follows:

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal-clad UO2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients.

15.0.11.6 ANC Computer Code

LNP DEP 6.4-1

Add new DCD Subsection 15.0.11.6 to read as follows:

The ANC computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. ANC can also solve the three-dimensional kinetics equations for six delayed neutron groups.

15.0.13 OPERATOR ACTIONS

LNP DEP 3.2-1

Revise the first sentence of the first paragraph of DCD Subsection 15.0.13 to read as follows:

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable shutdown condition.

Revise the first sentence of the second paragraph of DCD Subsection 15.0.13 to read as follows:

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition for at least 72 hours.

15.0.15 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 15.0.15.1.

STD COL 15.0-1

This COL item is addressed in FSAR Subsection 15.0.3.2.

15.0.16 REFERENCES

Add the following text to the end of DCD Subsection 15.0.16.

201. Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, "Caldon Ultrasonics Engineering Report ER-157P, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus™ System'," (TAC No. ME1321). August 16, 2010. ADAMS Accession No. ML102160694.

202. Final Safety Evaluation for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) - Issuance of Amendment re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves. September 24, 2001, ADAMS Accession No. ML012490569.

LNP DEP 6.4-1

Table 15.0-201
Summary of Initial Conditions and Computer Codes Used

			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.4	Spectrum of RCCA ejection accidents	ANC, VIPRE	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8	Refer to Subsection 15.4.8

15.1 INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.1.5.4.1 Source Term

Revise the fourth paragraph of DCD Subsection 15.1.5.4.1 to read as follows:

LNP DEP 6.4-1

The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level.

Revise the last paragraph of DCD Subsection 15.1.5.4.1 to read as follows:

The secondary coolant is assumed to have an iodine source term of 0.01 μCi/g dose equivalent I-131. This is 1 percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be 1 percent of the primary concentration.

15.1.5.4.6 Doses

Revise the text of DCD Subsection 15.1.5.4.6 to read as follows:

LNP DEP 6.4-1

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.6 rem at the site boundary for the limiting 2-hour interval (4.8 to 6.8 hours) and 1.1 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. The 30-day contribution to the dose at the site boundary and the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

Table 15.1-201
Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break

LNP DEP 6.4-1

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is 5 hours.
– Preaccident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.32 E+05
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb)	
0 – 2 hr	3.321E+05
2 – 72 hr	3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb)	
0 – 2 hr	3.321E+05
2 – 72 hr	3.66 E+03
Nuclide data	See Table 15A-4

Note:
a) Equivalent to 150 gpd cooled liquid at 62.4 lb/ft^3.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add a new third paragraph to Section 15.2 to read as follows:

LNP DEP 3.2-1

For events in this section where PRHR HX actuation occurs, transients are presented until the PRHR HX heat removal matches decay heat generation. After that point in time, PRHR HX performance is driven by the performance of the passive containment cooling systems to control containment pressure and the ability of the condensate collection features to return condensate to the in-containment refueling water storage tank. The performance of these systems, for extended decay heat removal, is described in subsection 6.3.1.1.1.

15.2.6.1 Identification of Causes and Accident Description

LNP DEP 6.3-1

Revise the seventh sentence of the fourth paragraph of DCD Subsection 15.2.6.1 to read as follows:

The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.3.3.3.1 Source Term

LNP DEP 6.4-1

Revise the last paragraph of DCD Subsection 15.3.3.3.1 to read as follows:

The initial secondary coolant activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

Table 15.3-201
Parameters Used in Evaluating the Radiological Consequences of a Locked Rotor Accident

LNP DEP 6.4-1

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/gm of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	1.75
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.04 E+05
Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.5[(b)]
Partition coefficient in steam generators	
iodine	0.01
alkali metals	0.0035
Accident scenario in which startup feedwater is not available	
Duration of accident (hr)	1.5 hr
Steam released (lb)	
0-1.5 hours(c)	6.48 E+05
Leak flashing fraction(d)	
0-60 minutes	0.04
> 60 minutes	0

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.4.8.1.1.3 Reactor Protection

Revise the first paragraph of DCD Subsection 15.4.8.1.1.3 to read as follows:

LNP DEP 6.4-1

The reactor protection in the event of a rod ejection accident is described in WCAP-15806 P -A (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.2 Limiting Criteria

Revise DCD Subsection 15.4.8.1.2 to read as follows:

LNP DEP 6.4-1

This event is a Condition IV incident (ANSI N18.2). See Subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

NUREG-0800 Standard Review Plan (SRP) 4.2, Revision 3 (Reference 24), interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2, Revision 3, Appendix B.
- The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.
- For intermediate (greater than 5 percent rated thermal power) and full power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g., DNBR).
- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 Btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.

- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst that must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the “Service Limit C” as defined in the ASME code.

15.4.8.2 Analysis of Effects and Consequences

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2 to read as follows:

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core calculation and then, a hot rod calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, fuel temperature, and DNB transients are then determined by performing a conservative fuel rod transient heat transfer calculation.

A discussion of the method of analysis appears in WCAP-15806-P-A (Reference 4).

Average Core Analysis

The three-dimensional nodal code ANC (References 14, 15, 16, 17, 21, 22, and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in three spatial dimensions (rectangular coordinates) for six delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE 01 code and methods (References 18 and 19).

Hot Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature, and DNBR using as input the time dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18.

System Overpressure Analysis

If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure.

This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

LNP DEP 6.4-1 Revise DCD Subsection 15.4.8.2.1 to read as follows:

Input parameters for the analysis are conservatively selected as described in Reference 4.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

Revise DCD Subsection 15.4.8.2.1.1 to read as follows:

LNP DEP 6.4-1

The values for ejected rod worths and hot channel factors are calculated using three dimensional static methods. Standard nuclear design codes are used in the analysis. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

15.4.8.2.1.2 Reactivity Feedback Weighting Factors

LNP DEP 6.4-1 Revise DCD Subsection 15.4.8.2.1.2 to read as follows:

15.4.8.2.1.2 Not Used

15.4.8.2.1.3 Moderator and Doppler Coefficients

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.3 to read as follows:

The critical boron concentration is adjusted in the nuclear code to obtain a moderator temperature coefficient that is conservative compared to actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements.

15.4.8.2.1.4 Delayed Neutron Fraction βeff

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.4 to read as follows:

Calculations of the effective delayed neutron fraction (βeff) typically yield values no less than 0.50 percent at end of cycle. The accident is sensitive to βeff if the ejected rod worth is equal to or greater than βeff. To allow for future cycles, a pessimistic estimate of βeff 0.44 percent is used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.5 to read as follows:

The trip reactivity insertion accounts for the effect of the ejected rod and one adjacent stuck rod. The trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.583 second for the instrument channel to produce a signal, 0.167 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.47 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

15.4.8.2.1.7 Results

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.7 to read as follows:

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases, a

typical cycle are summarized following the criteria outlined in Subsection 15.4.8.1.2.

- PCMI and high cladding temperature (hot zero power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in Subsection 15.4.8.1.2.

- High cladding temperature (≥ 5% rated thermal power)

  The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in Subsection 15.4.8.3.

- Core coolability

  The resulting maximum fuel average enthalpy is less than the criterion given in Subsection 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

The nuclear power and fuel transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-3.

The calculated sequence of events for the limiting cases is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

15.4.8.2.1.8 Fission Product Release

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.8 to read as follows:

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.

15.4.8.2.1.9 Peak Reactor Coolant System Pressure

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.2.1.9 to read as follows:

Calculations of the peak reactor coolant system demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

15.4.8.3 Radiological Consequences

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.3 to read as follows:

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. See Subsection 15.4.8.3.1 and Table 15.4-4.

As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see Subsection 15.4.8.2.1.8) such that the activity contained in the fuel cladding gap is released to the reactor coolant. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

15.4.8.3.1 Source Term

LNP DEP 6.4-1

Revise DCD Subsection 15.4.8.3.1 to read as follows:

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a preexisting iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance

compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG 1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of Draft Guide 1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). Draft Guide 1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was issued as a clarification to the gap fraction guidance in Draft Guide 1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see Subsection 15.4.8.2.1.8). Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see Subsection 15.4.8.2.1.8).

15.4.8.3.5 Identification of Conservatisms

Revise second bullet of DCD Subsection 15.4.8.3.5 to read as follows:

LNP DEP 6.4-1

- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4); whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).

15.4.8.3.6 Doses

Revise DCD Subsection 15.4.8.3.6 to read as follows

LNP DEP 6.4-1

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “well within” is taken as being 25 percent or less.

15.4.10 REFERENCES

LNP DEP 6.4-1

Revise DCD Subsection 15.4.10 as follows:

4. Beard, C. L. et al., “Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi Dimensional Kinetics,” WCAP-15806-P-A (Proprietary) and WCAP-15807-NP-A (Nonproprietary), November 2003.
7. Liu, Y. S., et al., “ANC – A Westinghouse Advanced Nodal Computer Code,” WCAP-10965-P-A (Proprietary) and WCAP-10966-A (Nonproprietary), September 1986.

8. Not Used.
10. American National Standards Institute N18.2, “Nuclear Safety Criteria for the Design of Stationary PWR Plants,” 1973.
13. Not Used.
14. Nguyen, T. Q., et al., “Qualifications of the PHOENIX P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Nonproprietary), June 1988.
15. Ouisloumen, M., et al., “Qualification of the Two-Dimensional Transport Code PARAGON,” WCAP-16045-P-A (Proprietary) and WCAP-16045-NP-A (Nonproprietary), August 2004.
16. Liu, Y. S., “ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery,” WCAP-10965-P-A, Addendum 1 (Proprietary) and WCAP-10966-A Addendum 1 (Nonproprietary), April 1989.
17. Letter from Liparulo, N.J. (Westinghouse) to Jones, R. C., (NRC), “Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC),” NTD-NRC-95-4533, August 22, 1995.
18. Sung, Y. X., Schueren, P., and Meliksetian, A., “VIPRE 01 Modeling and Qualification for Pressurized Water Reactor Non LOCA Thermal Hydraulic Safety Analysis,” WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.
19. Stewart, C. W., et al., “VIPRE-01: A Thermal/Hydraulic Code for Reactor Cores,” Volumes 1, 2, 3 (Revision 3, August 1989), and Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute, Palo Alto, California.
20. Foster, J. P. and Sidener, S., “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),” WCAP-15063-P-A, Revision 1 with Errata (Proprietary) and WCAP-15064-NP-A (Nonproprietary), July 2000.
21. Zhang, B., et al., “Qualification of the NEXUS Nuclear Data Methodology,” WCAP-16045-P-A, Addendum 1 A (Proprietary) and WCAP-16045-NP-A, Addendum 1-A (Nonproprietary), August 2007.
22. Zhang, B., et al., “Qualification of the New Pin Power Recovery Methodology,” WCAP-10965-P-A, Addendum 2-A (Proprietary), September 2010.
23. Smith, L. D , et al., “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” WCAP-15025-P-A (Proprietary) and WCAP-15026-NP-A (Nonproprietary), April 1999.
24. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, “Fuel System Design,” Appendix B, “Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents,” March 2007.
25. Draft Regulatory Guide DG-1199, “Proposed Revision 1 of Regulatory Guide 1.183; Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” October 2009. NRC ADAMS Accession Number: ML090960464.
26. NRC Memorandum from Anthony Mendiola to Travis Tate, “Technical Basis for Revised Regulatory Guide 1.183 (DG-1199) Fission Product

Fuel-to-Cladding Gap Inventory," July 2011. NRC ADAMS Accession Number: ML111890397.

27. Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C. (NRC), "Process Improvement to the Westinghouse Neutronics Code System," NSD-NRC-96-4679, March 29, 1996.

Table 15.4-201
Time Sequence of Events for Incidents which Result in Reactivity and Power Distribution Anomalies

LNP DEP 6.4-1

Accident	Event	Time (seconds)
RCCA ejection accident		
1. PCMI limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.14
	Reactor trip setpoint reached	<0.30
	Peak cladding temperature occurs	0.36
	Peak enthalpy deposition occurs	0.44
	Rods begin to fall into core	1.20
2. Peak cladding temperature limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.08
	Minimum DNBR occurs	0.11
	Peak cladding temperature occurs	0.11
	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
3. Peak enthalpy/peak fuel centerline temperature event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.06
	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
	Peak fuel center temperature occurs	2.50
	Peak cladding temperature occurs	2.80

Table 15.4-202 (Sheet 1 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

LNP DEP 6.4-1

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in damaged fuel)	1.75
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fuel enthalpy increase (cal/g)	60
– Fission product gap fractions	
Iodine 131	0.1238
Iodine 132	0.1338
Krypton 85	0.5120
Other noble gases	0.1238
Other halogens	0.0938
Alkali metals	0.6860
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3
Nuclide data	See Table 15A-4
Reactor coolant mass (lb)	3.7 E+05 (1.68E+05 kg)

Note:
a) The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Table 15.4-202 (Sheet 2 of 2)
Parameters Used in Evaluating the Radiological Consequences of a Rod Ejection Accident

LNP DEP 6.4-1

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.5[a] (47.4 kg/hr)
– Leak flashing fraction	0.04[b]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.0035
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.9[c]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	2.78

Notes:
a) Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).
b) No credit for iodine partitioning is taken for flashed leakage.
c) From Appendix 15B.

Table 15.4-203
Deleted

LNP DEP 6.4-1

15.5 INCREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.6.2.6 Doses

LNP DEP 6.4-1

Revise the first paragraph of DCD Subsection 15.6.2.6 to read as follows:

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.3 rem at the exclusion area boundary and 0.6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “a small fraction” is taken as being ten percent or less.

15.6.3.3.1 Source Term

LNP DEP 6.4-1

Revise the last paragraph of DCD Subsection 15.6.3.3.1 to read as follows:

The secondary coolant iodine and alkali metal activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity.

15.6.3.3.6 Doses

LNP DEP 6.4-1

Revise the first two paragraph of DCD Subsection 15.6.3.3.6 to read as follows:

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be 0.7 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A “small fraction” is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be 1.4 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and 0.7 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

15.6.5.3.2 In-containment Activity Removal Processes

LNP DEP 6.4-1 Add the following paragraphs at the end of DCD Subsection 15.6.5.3.2.

Particulates removed from the containment atmosphere to the containment shell are assumed to be washed off the shell by the flow of water resulting from condensing steam (i.e. condensate flow). The particulates may be either washed into the sump, which is controlled to a pH ≥7 post-accident or into the IRWST, which is not pH controlled post-accident. Due to the conditions in the IRWST, a portion of the particulate iodine washed into the IRWST may chemically convert to an elemental form and re-evolve, subject to partitioning, as airborne. A water-steam partition factor of 10 for elemental iodine is applied. This value bounds the time-dependent partition factors calculated using the NUREG/CR-5950 (Reference 35) models and the calculated IRWST water temperature and pH as a function of time.

The IRWST is a closed tank with weighted louvers, and without boiling, there would be no motive force for the release of re-evolved gaseous iodine from the IRWST gas space to the containment. Thus, the assumption of boiling in the IRWST liquid is imposed to force the release of the re-evolved iodine to the containment atmosphere. A portion (3%) of the re-evolved elemental iodine is assumed to convert to an organic form upon its release to containment.

15.6.5.3.5 Main Control Room Dose Model

LNP DEP 6.4-1 Revise the first sentence of the second paragraph of DCD Subsection 15.6.5.3.5 to read as follows:

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when High-2 iodine or particulate activity is detected.

Revise the second sentence of the fourth paragraph of DCD Subsection 15.6.5.3.5 to read as follows:

With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

15.6.5.3.7.3 Atmospheric Dispersion Factors

Add the following paragraph at the end of DCD Subsection 15.6.5.3.7.3.

LNP COL 2.3-4 Site-specific χ/Q (atmospheric dilution factor) values provided in Subsection 2.3.4 are bounded by the values given in DCD Table 15A-5 and Table 15A-6.

15.6.5.3.8.1 Offsite Doses

LNP DEP 6.4-1 Revise the first sentence of the second paragraph of DCD Subsection 15.6.5.3.8.1 to read as follows:

The reported exclusion area boundary doses are for the time period of 1.3 to 3.3 hours.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

LNP DEP 6.4-1 Revise the second and third sentence of the first paragraph of DCD Subsection 15.6.5.3.8.2 to read as follows:

Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings, shine from radioactivity accumulated on the VES or VBS filters, and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of these dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

15.6.6 REFERENCES

LNP DEP 6.4-1 Add the following to DCD Subsection 15.6.6.

35. Beahm, E. C. et al., NUREG/CR-5950, *"Iodine Evolution and pH Control,"* December 1992.

Table 15.6-201
Parameters Used in Evaluating the Radiological Consequences of a Small Line Break Outside Containment

LNP DEP 6.4-1

Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[a]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[b]
Fraction of reactor coolant flashing	0.47
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:
a) Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.
b) At density of 62.4 lb/ft^3.

Table 15.6-202
Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture

LNP DEP 6.4-1

Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is 8.0 hours.
– Preaccident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	1% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.7 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	15.94
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop	
– Initial secondary coolant mass (lb)	1.16 E+05
– Primary-to-secondary break flow	See Figure 15.6.3-5
– Integrated flashed break flow (lb)	See Figure 15.6.3-10
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]
Steam generator in intact loop	
– Initial secondary coolant mass (lb)	2.30 E+04
– Primary-to-secondary leak rate (lb/hr)	52.16[b]
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	3.5 E-03[a]

Notes:
a) Iodine partition coefficient does not apply to flashed break flow.
b) Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

Table 15.6-203 (Sheet 1 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-Of-Coolant Accident

LNP DEP 6.4-1

Primary coolant source data	
– Noble gas concentration	280 μCi/g dose equivalent Xe-133
– Iodine concentration	1.0 μCi/g dose equivalent I-131
– Primary coolant mass (lb)	4.39 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	16,000
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	1.0
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	1.9
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	3.89 E+04
– Volume of HVAC, including main control room and control support area (ft^3)	1.2 E+05
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	1650
• Filter efficiency	Not applicable
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6

Table 15.6-203 (Sheet 2 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-Of-Coolant Accident

LNP DEP 6.4-1

Main control room model (cont.)	
– Occupancy	
• 0 - 24 hr	1.0
• 24 - 96 hr	0.6
• 96 - 720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited (VES Credited)	
– Main control room activity level at which the emergency habitability system actuation is actuated (Ci/m^3 of dose equivalent I-131)	2.0 E-07
– Response time to actuate VES based on radiation monitor response time and VBS isolation (sec)	200
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage via ingress/egress (scfm)	5
• Unfiltered inleakage from other sources (scfm)	10
• Recirculation flow through filters (scfm)	600
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1900
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168

Table 15.6-203 (Sheet 3 of 3)
Assumptions and Parameters Used in Calculating Radiological Consequences of a Loss-Of-Coolant Accident

LNP DEP 6.4-1

Control room with credit for continued operation of HVAC (VBS Supplemental Filtration Mode Credited)	
– Time to switch from normal operation to the supplemental air filtration mode (sec)	265
– Unfiltered air inleakage (cfm)	25
– Filtered air intake flow (cfm)	860
– Filtered air recirculation flow (cfm)	2740
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
Miscellaneous assumptions and parameters	
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0 - 8 hr	3.5 E-04
8 - 24 hr	1.8 E-04
24 - 720 hr	2.3 E-04

Table 15.6-204
Radiological Consequences of a Loss-of-Coolant Accident with Core Melt

LNP DEP 6.4-1

	TEDE Dose (rem)
Exclusion zone boundary dose (1.3 - 3.3 hr)[1]	23.5
Low population zone boundary dose (0 - 30 days)	22.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	3.70
– Direct radiation from adjacent structures, including sky-shine	0.30
– Filters Shine	0.32
– Spent fuel pooling boiling	0.01
– Total	4.33
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	4.50
– Direct radiation from adjacent structures, including sky-shine	0.30
– Filters Shine	0.03
– Spent fuel pooling boiling	0.01
– Total	4.84

Note:
1) This is the 2-hour period having the highest dose.

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.7.4.5 Offsite Doses

LNP DEP 6.4-1

Revise the first sentence of the first paragraph of DCD Subsection 15.7.4.5 to read as follows:

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary.

15.7.6 COMBINED LICENSE INFORMATION

LNP COL 15.7-1

This COL Item is addressed in Section 2.4.13.

Table 15.7-201
Assumptions Used to Determine Fuel Handling Accident Radiological Consequences

LNP DEP 6.4-1

Source term assumptions	
– Core power (MWt)	3434[(1)]
– Decay time (hr)	48
Core source term after 48 hours decay (Ci)	
I-130	1.28 E+05
I-131	8.18 E+07
I-132	9.10 E+07
I-133	4.06 E+07
I-135	1.17 E+06
Kr-85m	1.52 E+04
Kr-85	1.07 E+06
Kr-88	5.45 E+02
Xe-131m	1.02 E+06
Xe-133m	4.47 E+06
Xe-133	1.70 E+08
Xe-135m	1.91 E+05
Xe-135	1.04 E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.75
Percentage of fission products in gap	
I-131	8
Other iodines	5
Kr-85	10
Other noble gases	5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:
1) The main feedwater flow measurement supports a 1-percent power uncertainty.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 15A EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15A.3.1.2 Secondary Coolant Source Term

LNP DEP 6.4-1

Revise the first sentence of the first paragraph of DCD Subsection 15A.3.1.2 to read as follows:

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 1 percent of the primary coolant equilibrium source term.

15A.3.3 ATMOSPHERIC DISPERSION FACTORS

Replace the third paragraph in DCD Subsection 15A.3.3 with the following:

LNP COL 2.3-4

Site-specific χ/Q values provided in Subsection 2.3.4 are bounded by the values given in DCD Table 15A-5 and Table 15A-6.

APPENDIX 15B REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15B.1 ELEMENTAL IODINE REMOVAL

Revise the second full paragraph of DCD Subsection 15B.1 to read as follows:

The available deposition surface is 251,000 ft^2, and the containment building net free volume is 2.06×10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.9 hr^{-1}.