FINAL SAFETY ANALYSIS REPORT

CHAPTER 15

TRANSIENT AND ACCIDENT ANALYSIS

15.0 TRANSIENT AND ACCIDENT ANALYSIS

This chapter of the U.S. EPR Final Safety Analysis Report (FSAR) is incorporated by reference with departures and supplements as identified in the following sections.

The U.S. EPR FSAR includes the following COL Item in Section 15.0:

A COL applicant that references the U.S. EPR design certification will provide, prior to the first cycle of operation, a report that demonstrates compliance with the following items:

- Examine fuel assembly characteristics to verify that they are hydraulically compatible based on the criterion that a single package of assembly specific critical heat flux (CHF) correlations can be used to evaluate the assembly performance.
- Verify that uncertainties used in the setpoint analyses are appropriate for the plant and cycle being analyzed.
- Verify that the DNBR and LPD satisfy SAFDL with a 95/95 assurance.
- Review the U.S. EPR FSAR Tier 2 analysis results for the first cycle to confirm that the static setpoint value provides adequate protection for at least three limiting AOO.

The COL Item is addressed as follows:

A report that demonstrates compliance with evaluation of fuel assembly performance, analysis of setpoint uncertainties, verification that DNBR and LPD satisfy SAFDL, and a review of U.S. EPR Tier 2 analysis for the first cycle confirming static setpoint values provide adequate protection for at least three limiting AOO shall be submitted to the NRC staff for review prior to the first cycle of operation.

15.0.1 Radiological Consequence Analysis

No departures or supplements.

15.0.2 Computer Codes Used in Analysis

No departures or supplements.

15.0.3 Radiological Consequences of Design Basis Accidents

15.0.3.1 {Introduction

BBNPP will depart from the U.S. EPR FSAR by utilizing the site-specific short-term atmospheric dispersion factors for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). U.S. EPR FSAR Tier 1 Table 5.0-1 and U.S. EPR FSAR Table 2.1-1 provide the Accident Atmospheric Dispersion Factors (χ/Q) for the EAB of 1.00E-03 sec/m^3 and provide for χ/Qs for the Low Population Zone (LPZ) during the following periods: 0-2 hours, 2-8 hours and 8-24 hours of 1.75E-04 sec/m^3, 1.35E-04 sec/m^3, and 1.00E-04 sec/m^3, respectively. The corresponding BBNPP site-specific EAB and LPZ χ/Q values are provided in Table 15.0-1.

The site-specific Accident Atmospheric Dispersion Factors, including the 0-2 hour (EAB at 0.33 mile) χ/Q of 1.495E-03 sec/m^3, the 0-2 hour (LPZ at 1.5 miles) χ/Q of 2.766E-04 sec/m^3, the 2-8 hour (LPZ at 1.5 miles) χ/Q of 1.648E-04 sec/m^3, the 8-24 hour (LPZ at 1.5 miles) χ/Q of

1.038E-04 sec/m^3, the 24-96 hour (LPZ at 1.5 miles) χ/Q of 5.106E-05 sec/m^3 and the 96-720 hour (LPZ at 1.5 miles) χ/Q of 1.845E-05 sec/m^3 were used in calculation of doses resulting from the accident scenarios specified in this Section. In each case, the resulting EAB and LPZ doses were determined to be below regulatory limits as shown in Table 15.0-2.

15.0.3.2 Event Categorization

No departures or supplements.

15.0.3.3 Analytical Assumptions

No departures or supplements.

15.0.3.3.1 Non-Safety-Related Systems Credited in the Analyses and Operator Action

No departures or supplements.

15.0.3.3.2 Loss of Offsite Power Assumptions

No departures or supplements.

15.0.3.3.3 Atmospheric Dispersion Factors

Table 15.0-1 provides the short-term atmospheric dispersion factors for the EAB and LPZ that are utilized to calculate the radioactive doses associated with the various design basis events for BBNPP.

15.0.3.3.4 Core Radionuclide Inventory Assumptions

BBNPP will depart from the U.S. EPR FSAR by utilizing a different core radionuclide inventory for the Fuel Handling and Rod Ejection Accidents. The radionuclide inventory for the Fuel Handling and Rod Ejection Accidents is derived using a fuel enrichment of 5 wt% in U-235 and maximized burnup (ranging between approximately 5 and 41 GWD/MTU). The resulting core inventory is shown in Table 15.0-4. BBNPP incorporates by reference the core radionuclide inventory provided in U.S. EPR FSAR Table 15.0-14 for all other DBAs.

15.0.3.3.5 Iodine Appearance Rates

No departures or supplements.

15.0.3.3.6 Analytical Methods

No departures or supplements.

15.0.3.4 Receptor Variables

BBNPP will depart from the U.S. EPR FSAR EAB definition. The BBNPP EAB is shown in Figure 1.1-1.

15.0.3.4.1 Main Control Room/Technical Support Center Modelling

No departures or supplements.

15.0.3.4.2 Offsite Receptors

Table 15.0-3 provides BBNPP offsite receptor variables.

15.0.3.5 Small Line Carrying Primary Coolant Break Outside of the Reactor Building Accident

No departures or supplements.

15.0.3.5.1 Sequence of Events and Systems Operations

BBNPP will depart from the U.S EPR FSAR by using the RELAP-5 computer code (AREVA, 2007) to determine the flashing fractions in the sampling line break analysis.

15.0.3.5.2 Input Parameters and Initial Conditions

BBNPP will depart from the U.S. EPR FSAR scenario representation through refinement of the bounding assumption that the NSS sampling-line break locations are at the connecting points to the RCS, which maximized the break flow by ignoring the pressure drop along the lines. The BBNPP analysis accounts for the pressure drop in the sampling line through use of the RELAP-5 computer code (AREVA, 2007). The scenario involves the postulated break of one of the three NSS sampling lines and is based on simplified representations of the sampling lines (namely, straight piping without bends or couplings).

Table 15.0-5 and U.S. EPR FSAR Table 15.0-22 summarize the key design inputs for the small line break accident scenario.

15.0.3.5.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-23 for the small line break outside of the Reactor Building. The BBNPP TEDE doses at the EAB and LPZ for the small line break outside of the Reactor Building are provided in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.6 Steam Generator Tube Rupture Accident

No departures or supplements.

15.0.3.6.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.6.2 Input Parameters and Initial Conditions

No departures or supplements.

15.0.3.6.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-29 for the steam generator tube rupture. The BBNPP TEDE doses at the EAB and LPZ for the steam generator tube rupture for both of the source terms are presented in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.7 Main Steam Line Break Outside of Reactor Building Accident

No departures or supplements.

15.0.3.7.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.7.2 Input Parameters and Initial Conditions

No departures or supplements.

15.0.3.7.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-34 for the main steam line break outside of reactor building. The BBNPP TEDE doses at the EAB and LPZ for the main steam line break outside of reactor building for each of the four source terms are presented in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.8 Locked Rotor Accident

No departures or supplements

15.0.3.8.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.8.2 Input Parameters and Initial Conditions

BBNPP will depart from the U.S. EPR FSAR values of the primary-to-secondary steam generator (SG) tube leakage, which is reduced from 0.125 gpm to the Technical Specification limit of 0.104 gpm (150 gpd). In addition, the U.S. EPR FSAR analysis used a conservative clad failure of 9.5%, which was a maximized value determined to yield 90% of the regulatory dose limit at the critical receptor. The BBNPP analysis uses a more realistic, but still conservative, maximum calculated clad failure of 8%, based on fuel performance. Table 15.0-6 and the U.S. EPR FSAR Table 15.0-37 summarize the key design inputs for the LRA scenario.

15.0.3.8.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-38 for the locked rotor accident. The BBNPP TEDE doses at the EAB and LPZ for the locked rotor accident are provided in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.9 Rod Ejection Accident

No departures or supplements.

15.0.3.9.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.9.2 Input Parameters and Initial Conditions

BBNPP will depart from the U.S. EPR FSAR value for the clad failure, which is reduced from a value of 36.7%, which was a maximized value determined to yield 90% of the regulatory dose limit at the critical receptor, to the maximum calculated value of 26% (no fuel overheat) based on a fuel performance analysis, and in utilizing a core inventory based on a fuel enrichment of 5 wt% in U-235 and maximized burnup (ranging between approximately 5 and 41 GWD/MTD).

15.0.3.9.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-44 for the rod ejection accident. The BBNPP TEDE doses at the EAB and LPZ for the rod ejection accident are provided in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.10 Fuel Handling Accident

BBNPP will depart from the U.S. EPR FSAR by assuming a longer post-shutdown decay time prior to the postulated accident of 72 hours rather than 34 hours and utilizing a core inventory based on a fuel enrichment of 5 wt% in U-235 and maximized burnup (ranging between approximately 5 and 41 GWD/MTU).

15.0.3.10.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.10.2 Input Parameters and Initial Conditions

BBNPP will depart from the U.S. EPR FSAR by assuming a longer post-shutdown decay time prior to the postulated accident of 72 hours and by utilizing a core inventory based on a fuel enrichment of 5 wt% in U-235 and maximized burnup (ranging between approximately 5 and 41 GWD/MTU). The timeline associated with the radiological evaluation of a FHA is presented in Table 15.0-7. The design input associated with the FHA is presented in Table 15.0-8.

15.0.3.10.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-48 for the fuel handling accident. The BBNPP TEDE doses at the EAB and LPZ for the fuel handling accident are provided in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.11 Loss of Coolant Accident

No departures or supplements.

15.0.3.11.1 Sequence of Events and Systems Operations

No departures or supplements.

15.0.3.11.2 Input Parameters and Initial Conditions

No departures or supplements.

15.0.3.11.3 Results

BBNPP incorporates by reference the doses for the main control room presented in U.S. EPR FSAR Table 15.0-53 for the LOCA. The BBNPP TEDE doses at the EAB and LPZ for the LOCA are provided in Table 15.0-2. The BBNPP EAB and LPZ doses are below the regulatory limits.

15.0.3.12 Postaccident Reactor Building Water Chemistry Control

No departures or supplements.

15.0.3.13 Control Room Radiological Habitability

No departures or supplements.}

15.0.4 PLANT COOLDOWN

No departures or supplements.

15.0.5 Compliance with Section C.I.15, "Transient and Accident Analyses," of Regulatory Guide 1.206

No departures or supplements.

15.0.6 References

{This section is added as a supplement to the U. S. EPR FSAR.

AREVA, 2007. AREVA NP Document 43-10164PA-06, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light-Water Reactor LOCA and NON-LOCA Transient Analysis," Sep. 2007.}

Table 15.0-1— {BBNPP Atmospheric Dispersion Factors}

Receptor Variables	Atmospheric dispersion factors (sec/m^3)
EAB (Worst 2 hours) 0 to 2 hr	1.495E-03
LPZ (1.5 mi) - 0 to 2 hr	2.766E-04
LPZ (1.5 mi) - 2 to 8 hr	1.648E-04
LPZ (1.5 mi) - 8 to 24 hr	1.038E-04
LPZ (1.5 mi) - 1 to 4 days	5.106E-05
LPZ (1.5 mi) - 4 to 30 days	1.845E-05

Table 15.0-2— {BBNPP Radiological Consequences of Design Basis Accidents (rem TEDE)}

Design Basis Accident		Offsite Dose	
		EAB (0.33 mile)	LPZ (1.5 miles)
LOCA		18.7 [25] [1]	13.1 [25]
Small line break outside of Reactor Building		0.9 [2.5]	0.2 [2.5]
SGTR	Pre-incident spike	1.7 [25]	0.4 [25]
	Coincident spike	1.1 [2.5]	0.5 [2.5]
MSLB	Pre-incident spike	0.4 [25]	0.1 [25]
	Coincident spike	0.4 [2.5]	0.3 [2.5]
	Fuel rod clad failure	7.9 [25]	3.5 [25]
	Fuel overheat	8.7 [25]	3.7 [25]
RCP locked rotor/broken shaft		2.4 [2.5]	0.9 [2.5]
Rod Ejection		5.6 [6.3]	2.7[6.3]
Fuel handling accident		5.3 [6.3]	1.0 [6.3]

Notes:
[1]The values in brackets represent the dose acceptance criteria.

Table 15.0-3— {BBNPP Offsite Receptor Variables}

Description		Value	Remarks
Atmospheric dispersion (ground-level release)			See Section 2.3.4
Distance	EAB	0.33 mi	Represents the physical distance to the West sector measured from the containment building centerline and corresponds to the analytical distance of 0.31 miles used in calculating the atmospheric dispersion factors (See Section 2.3.4)
	LPZ	1.5 mi	Measured from the containment building centerline
Exposure Interval	EAB	2 hrs	RG 1.183, Section 4.1.5
	LPZ	30 days	RG 1.183, Section 4.1.6
Breathing Rate	EAB 0-2 hrs	3.5E-04 m^3/s	RG 1.183, Section 4.1.3
	LPZ 0-8hrs	3.5E-04 m^3/s	
	LPZ 8-24 hrs	1.8E-04 m^3/s	
	LPZ 1-30 days	2.3E-04 m^3/s	

Table 15.0-4— {Design Basis Core Radionuclide Inventory Used in BBNPP Fuel Handling and Rod Ejection Accidents}

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Noble Gases		Halogens		Alkalis	
Kr-83m	1.96E+07	Br-83	1.96E+07	Rb-86m	3.11E+04
Kr-85m	4.50E+07	Br-84	3.62E+07	Rb-86	3.07E+05
Kr-85	1.60E+06	Br-85	4.45E+07	Rb-88	1.29E+08
Kr-87	9.02E+07	I-129	4.75E+00	Rb-89	1.67E+08
Kr-88	1.28E+08	I-130	4.64E+06	Cs-134	2.65E+07
Kr-89	1.61E+08	I-131	1.21E+08	Cs-136	8.50E+06
Xe-131m	1.35E+06	I-132	1.75E+08	Cs-137	1.67E+07
Xe-133m	7.72E+06	I-133	2.55E+08	Cs-138	2.50E+08
Xe-133	2.55E+08	I-134	2.86E+08		
Xe-135m	4.84E+07	I-135	2.38E+08		
Xe-135	9.26E+07				
Xe-137	2.25E+08				
Xe-138	2.30E+08				

Table 15.0-5— {BBNPP Design Input for NSS and CVCS Break Locations and Flows}

Break Description	Parameter		Analytical Value	Remarks
SLB 1-Break in Nuclear Sampling System (NSS)				
Double-ended guillotine rupture of the RCS sampling line leading crossover leg 3. The break location is between the containment penetration and the sampling-line heat exchanger in the Fuel Building. No heat loss is assumed to occur within the sampling line.	Line size and Schedule		1/2 inch Sch. 40S followed by 1/4 inch Sch. 40S	
	Line transverse (flow) area at exit point		0.000723 ft^2	
	Crossover leg 3 (bounding)	Total pipe length	141 ft	RELAP-5 analysis with line losses
		RCS Pressure and Temperature	2218 psia, 563.4°F	
		Critical mass flux based on 1/2 inch Sch. 40S line	3.45E3 lbm/ft^2-sec	
		Flow rate at exit point	8.97E+03 lbm/hr	
		Flashing fraction	86.1%	
	Break isolation time		30 min	Operator manual action from MCR
SLB 2-Break in CVCS Connecting Line				
Double-ended guillotine rupture of 6-inch line between the VCT and the VCT suction valves	VCT volume		671 ft^3	
	Break Flow		176,200 lbm/hr	
	Coolant Temperature		122°F	
	Coolant density		61.7 lbm/ft^3	
	Break isolation time		30 min	Operator manual action from MCR

Table 15.0-6— {BBNPP Design Input for Locked Rotor Accident}

Description		Value	References and Remarks
Source Term			
Core inventory		See U.S. EPR FSAR Table 15.0-14	
Radial peaking factor		1.7	
Fuel rod activity gap fractions	I-131	8%	RG 1.183, Table 3
	Other halogens	5%	
	Kr-85	10%	
	Other noble gases	5%	
	Alkalis (Cs, Rb)	12%	
DNB-induced clad failure		8%	Based on fuel performance
Primary and secondary side coolant radionuclide concentrations		See U.S. EPR FSAR Tables 15.0-15 and 15.0-16	
Pre-accident halogen spike (assumed to be the same as for the MSLB)		60 µCi/gm DE-I131	RG 1.183, Appendix E, Secs. 2.1 and 2.2
Fraction of gap activity released to RCS (instantaneous release, uniform mixing)		100%	RG 1.183, Appendix E, Section 3
Reactor Coolant System Variables			
Coolant volume in RCS and pressurizer		15,009 ft^3	
Coolant mass in RCS and pressurizer		6.47E+05 lbm	
Primary to secondary leak rate used in analysis		0.104 gpm/SG	TS requirement
Secondary Side Coolant Variables			
SG water inventory	100% power	1.698E+05 lbm/SG	
	Hot shutdown	2.311E+05 lbm/SG	
	Average	2.005E+05 lbm/SG	For fractional steaming rate value
Iodine partition coefficient in secondary-side water		100	RG 1.183, Appendix E, Section 5.5.4
Alkali steam carry over fraction		1%	
Other Variables			
Duration of tube uncovered period for SG connected to MSL with stuck open MSRCV (SG 3)		15 min	
Overall steaming rate for plant cooldown		113 lbm/s	Includes analytical margin of 1.2
Time at which plant cooldown is switched from SG steaming to RHR		8 hrs	
Offsite receptor variables		See Table 15.0-3	
MCR variables		See U.S. EPR FSAR Table 15.0-18	MCR isolation actuated by PCIS
MCR composite (χ/Q) and intake filter bypass fractions for releases via MSRTs and silencers		See U.S. EPR FSAR Table 15.0-37	

Table 15.0-7— {BBNPP Fuel Handling Accident Timeline}

Action	Time
Reactor shutdown (all rods in).	0 s
Fuel movement is initiated and an FHA takes place, either in the Reactor Building (with open containment) or in the Fuel Building.	72 hrs
All activity released from the gaps of fuel rods undergoing cladding failure is released to the environment (2-hour exponential release assumption, starting at 72 hrs).	74 hrs

Table 15.0-8— {BBNPP Design Input for Fuel Handling Accident}

Description		Value	References and Remarks
Source Term			
Peak assembly radial peaking factor		1.7	
Core inventory		See Table 15.0-4	
Fuel rod activity gap fractions	I-131	8%	RG 1.183, Table 3
	Other halogens	5%	
	Kr-85	10%	
	Other noble gases	5%	
	Alkalis (Cs, Rb)	12%	
Decay time prior to PA		72 hrs	
Fuel damage resulting from PA		1 Assembly	Bounds the value in similar B&W 15x15 fuel assembly designs
Percent of damaged-fuel rod gap activity release		100%	RG 1.183, Appendix B
Atmospheric Release Resulting from Postulated FHA in Primary Containment			
Primary containment configuration during refueling operations		Open	Desired configuration
Water depth above top of fuel in refueling cavity		>23 ft	TS requirement
Overall pool decontamination factor	Noble gases	1	RG 1.183, Appendix B
	Halogens	200	
	Alkalis	Infinite	
Composition of airborne halogens above cavity	Elemental	57%	RG 1.183, Appendix B
	Organic	43%	
Release point to atmosphere		Base of vent stack	
Exhaust filtration		None credited	
Atmospheric Release Resulting from Postulated FHA in the Fuel Building			
Water depth above top of fuel in refueling cavity		>23 ft	TS requirement
Overall pool decontamination factor		See FHA in open containment	
Composition of airborne halogens above pool			
Release point to atmosphere		Base of vent stack	
Exhaust filtration		None credited	
Other Variables			
Offsite receptor variables		See Table 15.0-3	
MCR variables		See U.S. EPR FSAR Table 15.0-18	MCR isolation actuated by high rad signal in air intake duct.
MCR composite X/Qs and intake filter bypass fractions for releases via MSRTs and silencers		See U.S. EPR FSAR Table 15.0-47	

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the U.S. EPR FSAR is incorporated by reference.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the U.S. EPR FSAR is incorporated by reference.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

{This section of the U.S. EPR FSAR is incorporated by reference with departures and supplements as identified in the following subsections.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

No departures or supplements.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

No departures or supplements.

15.3.3 Reactor Coolant Pump Rotor Seizure

No departures or supplements.

15.3.3.1 Identification of Causes and Event Description

No departures of supplements.

15.3.3.2 Method of Analysis and Assumptions

No departures or supplements.

15.3.3.3 Results

No departures or supplements.

15.3.3.4 Radiological Consequences

Section 15.0.3.8 addresses the radiological impact associated with the Reactor Coolant Pump Rotor Seizure event (also referred to as the Locked Rotor Accident (LRA)). The U.S. EPR FSAR analysis used a conservative clad failure of 9.5%, which was a maximized value determined to yield 90% of the regulatory dose limit at the critical receptor. The BBNPP analysis in Section 15.0.3.8 departs from this approach and uses a more realistic, but still conservative, maximum calculated clad failure of 8%, based on fuel performance.

15.3.3.5 Conclusions

No departures or supplements.

15.3.3.6 SRP Acceptance Criteria

No departures or supplements.

15.3.4 Reactor Coolant Pump Shaft Break

No departures or supplements.

15.3.5 References

No departures or supplements.}

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

{This section of the U.S. EPR FSAR is incorporated by reference with departures and supplements as identified in the following subsections.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition

No departures of supplements.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

No departures or supplements.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

No departures or supplements.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

No departures or supplements.

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

No departures or supplements.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

No departures or supplements.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

No departures or supplements.

15.4.7.1 Identification of Causes and Accident Description

No departures or supplements.

15.4.7.2 Method of Anlysis and Assumptions

No departures or supplements.

15.4.7.3 Results

The radiological analysis for the offsite radiological consequences and main control room habitability includes an evaluation of the Reactor Coolant Pump Rotor Seizure event (also referred to as the Locked Rotor Accident (LRA)), with clad failure. The U.S. EPR FSAR analysis used a conservative clad failure of 9.5%, which was a maximized value determined to yield 90% of the regulatory dose limit at the critical receptor. The BBNPP analysis departs from this approach and uses a more realistic, but still conservative, maximum calculated clad failure of 8%, based on fuel performance (Section 15.0.3.8.2).

15.4.7.4 Conclusions

No departures or supplements.

15.4.7.5 SRP Acceptance Criteria

No departures or supplements.

15.4.8 Spectrum of Rod Ejection Accidents in a PWR

No departures or supplements.

15.4.9 Spectrum of Rod Drop Accidents (BWR)

No departures or supplements.

15.4.10 References

No departures or supplements

15.5 INCREASE IN REACTOR COOLANT INVENTORY

This section of the U.S. EPR FSAR is incorporated by reference.

15.6 DECREASE IN REACTOR COOLANT INVENTORY EVENTS

This section of the U.S. EPR FSAR is incorporated by reference.

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This section of the U.S. EPR FSAR is incorporated by reference.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section of the U.S. EPR FSAR is incorporated by reference.

15.9 BOILING WATER REACTOR STABILITY

This section of the U.S. EPR FSAR is incorporated by reference.

15.10 SPENT FUEL POOL CRITICALITY AND BORON DILUTON ANALYSIS

This section of the U.S. EPR FSAR is incorporated by reference.