CHAPTER 15

15.0 ACCIDENT ANALYSIS

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15.0 ACCIDENT ANALYSIS

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

Plant operations are established to be in one of four categories. These are categorized in accordance with the nomenclature adopted by the American Nuclear Society. The categories are:

a) CONDITION I - Normal Operation and Operational Transient - Events which are expected to occur frequently in the course of power operation, refueling, maintenance, or plant maneuvering.

b) CONDITION II - Faults of Moderate Frequency - Events which are expected to occur on a frequency of once per year during plant operation.

c) CONDITION III - Infrequent Faults - Events which are expected to occur once during the lifetime of the plant.

d) CONDITION IV - Limiting Faults - Events which are not expected to occur but which are evaluated to demonstrate the adequacy of the design.

15.0.1.1 Acceptance Criteria

Condition I

This condition describes the normal operational modes of the reactor. As such, occurrences in this category must maintain margin between operating conditions and the plant trip setpoints. The setpoints are established to assure maintenance of margin to design limits. The set of operating conditions, together with conservative operational uncertainties for the variables, establish the set of initial conditions for the other event categories.

Condition II

a) The pressures in reactor coolant and main steam systems should be less than 110% of design values.

b) The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded by assuring that the minimum calculated departure from nucleate boiling ratio does not exceed the applicable limits of the DNBR correlation being used (see Sections 4.4 and 15.0.10).

c) The radiological consequences should be less than 10 CFR 20 guidelines.

d) The event should not generate a more serious plant condition without other faults occurring independently.

Condition III

a) The pressures in reactor coolant and main steam systems should be less than 110% of design values.

b) A small fraction of fuel failures may occur, but these failures should not hinder the core coolability.

c) The radiological consequences should meet the guidelines of Regulatory Guide 1.183.

d) The event should not generate a limiting fault or result in the consequential loss of the reactor coolant or containment barriers.

Condition IV

a) Radiological consequences should not exceed 10 CFR 50.67 guidelines.

b) The event should not cause a consequential loss of the required functions of systems needed to cope with the reactor coolant and containment systems.

c) Additional criteria to be satisfied by specific events are:

   1) LOCA - 10 CFR 50.46 and Appendix K.

   2) Rod Ejection - Radially averaged fuel enthalpy < 280 cal/gm.

15.0.1.2 Classification of Accident Events by Category

Table 15.0.1-1 presents the event classification by category used in evaluating the acceptability of results of the analysis.

TABLE 15.0.1-1

SUMMARY OF CLASSIFICATION

I. ANTICIPATED OPERATIONAL OCCURRENCES - Condition II Events

15.1 Increase in Heat Removal by the Secondary System

15.1.1 Decrease in Feedwater Temperature
15.1.2 Increase in Feedwater Flow
15.1.3 Increase in Steam Flow

15.2 Decrease in Heat Removal by the Secondary System

15.2.2 Loss of External Electrical Load
15.2.3 Turbine Trip
15.2.4 Loss of Condenser Vacuum
15.2.5 Closure of Main Steam Isolation Valve
15.2.6 Loss of Non-Emergency A-C Power to the Station Auxiliaries
15.2.7 Loss of Normal Feedwater Flow

15.3 Decrease in Reactor Coolant Flow Rate

15.3.1 Loss of Forced Coolant Flow

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Rod Assembly Bank Withdrawal from Subcritical or Low Power
15.4.2 Uncontrolled Rod Assembly Bank Withdrawal from Power
15.4.3 Control Rod Misoperation (Dropped Full Length Assembly, Dropped Full Length Assembly Bank, or Statically Misaligned Full Length Assembly)
15.4.4 Startup of an Inactive Loop
15.4.6 Boron Dilution

15.5 Increases in Reactor Coolant System Inventory

15.5.1 Inadvertent Operation of ECCS
15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

II. POSTULATED ACCIDENTS - Condition III Events

15.1 Increase in Heat Removal by the Secondary System

15.1.5 Steam System Piping Failures (Minor) Inside and Outside Containment

TABLE 15.0.1-1 (Continued)

15.4 Reactivity and Power Distribution Anomalies

- 15.4.3 Control Rod Misoperation (Single Full Length Assembly Withdrawal at Power)
- 15.4.7 Inadvertent Loading of a Fuel Assembly into the Improper Location

15.7 Radioactive Releases from a Subsystem or Component

- 15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures
- 15.7.5 Spent Fuel Cask Drop Accidents

III. POSTULATED ACCIDENTS - Condition IV Events

15.1 Increase in Heat Removal by the Secondary System

- 15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve
- 15.1.5 Steam System Piping Failures (Major) Inside and Outside Containment

15.2 Decrease in Heat Removal by the Secondary System

- 15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment

15.3 Decrease in Reactor Coolant Flow

- 15.3.2 Reactor Coolant Pump Rotor Seizure
- 15.3.3 Reactor Coolant Pump Shaft Break

15.4 Reactivity and Power Distribution Anomalies

- 15.4.8 Spectrum of Rod Ejection Accidents

15.6 Decreases in Reactor Coolant System Inventory

- 15.6.1 Inadvertent Opening of a Pressurizer PORV
- 15.6.2 Small Break Loss-of-Coolant Accidents
- 15.6.3 Steam Generator Tube Failure (Radiological Consequences Only)
- 15.6.5 Loss of Coolant Accidents

15.7 Radioactive Releases from a Subsystem or Component

- 15.7.4 Fuel Handling Accidents

15.0.2 Plant Characteristics and Initial Conditions Used in the Accident Analyses

Six operational modes have been considered in the analysis.

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (ºF)
1	Power Operation	≥ 0.99	>5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

[a] Excluding decay heat.
[b] All reactor vessel head closure bolts fully tensioned.
[c] One or more reactor vessel head closure bolts less than fully tensioned.

These operational modes have been considered in establishing the subevents associated with each event initiator. A set of initial conditions is established for the events necessary to be analyzed with the conditions for each mode of operation.

The normal plant rated operating conditions are presented in Table 15.0.2-1 and principal fuel design characteristics in Table 15.0.2-2. The uncertainties used in the accident analysis applicable to the operating conditions are:

1. Core Power ± 0.3%[b]

2. Primary Coolant Pressure ± 40 psi[a][c]

[a]The primary coolant pressure uncertainty listed here is applied to the system transient analysis pressure which is used in the hot subchannel analysis to calculate the minimum DNB ratio. It is not applied to the system transient analysis initial-condition pressure. This is in accordance with the applicable non-LOCA transient analysis methodology (see page 203 of Reference 15.0-3). This uncertainty bounds the -30 psi specified by Technical Specification 3.4.1.

[b]The ultrasonic feedwater flow calorimetric has a maximum uncertainty of ± 0.3 percent. The feedwater venturi and steam flow calorimetrics have a maximum uncertainty of ± 2 percent; however, the maximum power level is restricted to 2300 MWt when the feedwater venturi or the steam flow calorimetrics are utilized.

[c]See pressurizer pressure range for LOCA analysis in Table 15.6.5-1.

TABLE 15.0.2-1

NOMINAL PLANT RATED OPERATING CONDITIONS

Core Thermal Power	2339 MWt***
Target Vessel Average Coolant Temperature	575.9 °F
Target Vessel Coolant Flow*	97.3 * 10^6 lb/hr
Active Core Flow*	91.9 * 10^6 lb/hr
Nominal Steam Generator Pressure (Dome)	800 psia (6% SGTP) 821 psia (0% SGTP)
Nominal Feedwater Temperature	441.5 °F
Pressurizer Pressure	2250 psia
Pressurizer Level	53.3% of span**
Steam Generator Level	52% of span
Steam Generator Total Fluid Inventory	91,000 lbs. (per steam generator)****
Steam Generator Circulation Ratio	4.13

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* Coolant flow reflects 6% steam generator tube plugging for rebuilt steam generators and is a lower bound value (based on the Technical Specification minimum), rather than a nominal value.

** TS allows a 10% band on the upper operating range. Therefore, Chapter 15 events have been dispositioned for initial pressurizer level conditions as high a 63.3% of span.

*** The total core power supported by the accident analyses is the nominal core thermal power plus the measurement uncertainty, which is 2346 MWt.

**** The nominal value for SG total fluid inventory is shown. For dose consequence analysis, conservatively larger or smaller values were used, as described in the appropriate dose analysis sections.

TABLE 15.0.2-2

NOMINAL CORE AND FUEL DESIGN
PARAMETERS USED IN ACCIDENT ANALYSIS

Number of fuel assemblies of all type in core	157
Number of part length shielding fuel assemblies	12
Fuel assembly pitch	8.466 in.
Fuel assembly design type	15x15
Fuel rods per assembly	204
Guide tubes per assembly	20
Instrument tubes per assembly	1
Fuel rod pitch	.563 in.
Fuel rod O.D.	.424 in.
Guide and instrument tube O.D. (above dashpot)	.544 in.
Active fuel length	144 in.
Fuel rod length	152 in.
Number of spacers(1)	9
Maximum spacer span length	26.2 in.

(1) The departure from nucleate boiling analyses assume 9 spacers when there are in fact 10. The highest grid spacer is outside of the heated length of the fuel and is therefore not modeled. A description of the fuel assembly and grids may be found in UFSAR Section 4.2.2.1.

15.0.3 Power Distribution

The radial and axial power peaking used in the analysis is presented in Table 15.0.3-1. The limiting axial power distribution used for most DNB events is presented in Figure 15.0.3-1. This axial power distribution is used for the majority of non-LOCA transients. Analysis of the remaining events used event specific axial power distributions reflecting a power level other than 100%. As an accident that does not experience power redistribution, the limiting axial power shape used for analysis of Small Break LOCA is presented in Reference 15.0-6. For Large Break LOCA, Reference 15.0-1 explains that the axial power shape is a sampled parameter. This means that different cases use different shapes, depending on the random selection of values within established limits. Applicable variables include time in cycle and axial skew (top vs. bottom).

The Technical Specification (Reference 15.0-2) operating limits and reactor protection system setpoints assure that the power distribution is maintained within these power distribution limits. For example, the margin to trip setpoint is automatically reduced for DNB and fuel temperature limiting events, when the difference between top and bottom power flux detectors would indicate an axial flux offset which would degrade conditions to less than those established with the allowable operating power distributions. This reduction for the OTΔT and OPΔT trips was confirmed using statistical setpoint analysis (Reference 15.0-10).*

* The OTΔT trip function and statistical setpoint analysis are described in Section 15.0.7.

TABLE 15.0.3-1

REACTOR POWER DISTRIBUTION USED IN THE ANALYSIS

Fraction of power deposited in fuel	.974
Nuclear enthalpy rise hot channel factor ($F_{\Delta H}$)	1.80
Heat flux hot channel factor (F_Q)	2.46

15.0.4 Range of Plant Operating Parameters and States Used in the Analysis

Table 15.0.4-1 presents the range of key plant operating parameters considered in the analysis. A broader range of power, vessel average coolant temperature, and primary pressure is considered in establishing the trip setpoints verified by the analysis results presented in this document. The broader range is consistent with that indicated on page 2.1-4 of Reference 15.0-2. The plant parameter inputs to the Chapter 15 analysis were based upon analyses performed prior to Amendment 176 to the RNP Technical Specifications. The requirements implemented by Amendment 176 to the Technical Specifications have been evaluated and determined to be bounded by the current analysis with the exception of the initial pressurizer level and minimum pressurizer PORV opening pressure. These two assumptions have been evaluated and determined to be acceptable. Some of the Chapter 15 analyses, or portions thereof, were performed prior to the Appendix K power uprate. However, the effect of operation at the uprated conditions has been evaluated and all of the reported analyses support operation at the uprated power level of 2339 MWt plus 0.3% uncertainty.

Operating states of the reactor are also considered in the analysis. The operating states include the exposure of the fuel as impacts fuel thermal performance and neutronics parameters. State values are selected for the event analyzed to provide the greatest challenge to the acceptance criteria for an event. Several analyses may be required to bound the range of the state variable. For example, a range of neutronic parameters is used in the analysis of rod withdrawal events in order to verify the range of protection of the challenged trip setpoints.

The range of initiating events is also considered in formulating the analysis conditions for an event. The initiating conditions are examined to identify the set which most challenge the acceptance criteria. Where not obvious, sensitivity analysis or several analyses are performed. For example, analyses are performed for uncontrolled rod withdrawal events throughout the range of reactivity insertion rate possible from shim dilution to maximum withdrawal rate of the most worthy control banks. Since the most challenging initial power level is not obvious, the range of power level as permitted by the reactor protection system is analyzed.

A further example of state variation is the impact of protective systems such as the pressurizer spray and power operated relief valves. These are assumed to be in a state which most challenges the acceptance criteria under consideration.

The various operating modes of the reactor are also considered. The modes for this analysis are as described in 15.0.2. The startup mode, for example, is relevant to the uncontrolled rod withdrawal from subcritical or low power event. All modes of operation are relevant to the CVCS malfunction event which can result in dilution of primary boron concentration.

In this manner, the permitted operating modes, states and range of plant operating variables are considered in the safety analysis. Thus, the plant may be operated within these bounds and be expected to meet the acceptance criteria as cited for each event.

Sensitivity studies performed by AREVA Inc. indicated that it was not necessary to bias the initial pressure in the ANF-RELAP or S-RELAP5 system analysis of the NSSS transient response. However, to be conservative in the core subchannel calculation of DNBR, it is necessary to bias the pressure based on the plant measurement uncertainty.

The sensitivity calculations indicate that the maximum pressure calculated in the system analysis for events with rapidly increasing pressures were controlled not by the initial pressure value but by the biased PORV, and safety valve setpoints. For events with little or no change in pressure or
decreasing pressures, the initial system pressure was again found to have little impact on the calculated pressures. The calculated pressures simply varied throughout the event by approximately the applied pressure bias.

Thus, the AREVA Inc. transient methodology is to:

- initiate the calculation of the NSSS transient response (using the ANF-RELAP or S-RELAP5 computer code) at nominal pressure

  and

- reduce the core outlet pressure calculated by the ANF-RELAP or S-RELAP5 computer code for use as input data in estimating the Minimum DNB Ratio with the XCOBRA-IIIC computer code.

TABLE 15.0.4-1

INITIAL CONDITION RANGE OF KEY PLANT
OPERATING PARAMETERS CONSIDERED IN THE ANALYSIS****

Core thermal power	Subcritical to 2346 MWt
Vessel average coolant temperature	547ºF to 575.9ºF*
Pressurizer water level	22.2% to 53.3% of span (programmed)**
Steam generator level	39% to 52% of span (programmed)
Nominal Reactor Coolant System pressure	2250 psia***

* Lower temperature operation during startup is bounded by the higher temperature listed here (see Reference 15.0-7). While temperatures ranging from Hot Zero Power to Hot Full Power are shown in the table, the safety analysis also considers uncertainty consistent with the methodology. The exception is the boron dilution analysis (event 15.4.6). It considers temperatures low and high enough to bound Mode 2-6.

** Technical Specifications allows a 10% band on the upper operating range. Therefore, Chapter 15 events have been dispositioned for initial pressurizer level conditions as high as 63.3% of span.

*** See discussion in text Section 15.0.4-1.

**** See Table 15.6.5-1 for Realistic Large Break LOCA

15.0.5 Reactivity Coefficients Used in the Safety Analysis

Table 15.0.5-1 presents the reactivity coefficients used in the analysis. As discussed in 15.0.4, the set of these parameters which most challenges the event acceptance criteria is used in each analysis. Conservative values for the moderator temperature and Doppler coefficients are used in the safety analysis to bound operating conditions. The conservatism factor is applied in a sense to most challenge the event acceptance criteria. For Doppler a 20% or greater conservatism is applied, for Moderator Temperature Coefficient (MTC) the limiting Technical Specifications value is used.

The table shows that a positive moderator coefficient was assumed in the analysis of events most challenged by BOC neutronic parameters. The assumption demonstrates safety of the system under an extreme set of initial conditions, and allows a single analysis to cover both high-power operation (for which the moderator temperature coefficient is actually a negative value) and low-power operation (for which the moderator temperature coefficient is a small positive value). Although the results of the analyses support a moderator temperature coefficient of up to +5 pcm/ºF, the plant operates with a moderator temperature coefficient of ≤ 0 pcm/ºF at rated power.

TABLE 15.0.5-1

NOMINAL REACTIVITY COEFFICIENTS USED IN THE ANALYSIS

Item	BOC	EOC
Moderator Temperature Coefficient*	+5.0 pcm/ºF	-45 pcm/ºF
Doppler Coefficient	Event Specific	Event Specific
Scram Worth	Event Specific	Event Specific

* The locked Reactor Coolant Pump (RCP) rotor event (15.3.2) used an analysis value of 0.0 pcm/ºF, which reflects the Technical Specifications limit for full power operation.

15.0.6 RCCA Insertion Characteristics Used in the Analysis

Figure 15.0.6-1 presents the negative insertion used in the analysis for reactor trip. The insertion worth includes a 0.9 multiplier and assumes that the most reactive rod is stuck out of the core. This insertion rate has been established to conservatively bound the actual or expected insertion rate for the plant.

15.0.7 Trip Setpoints and Time Delays

Table 15.0.7-1 presents the trip setpoints and time delays used in the analysis. Additional trips are available, i.e., overpower ΔT and turbine trip. If credit were taken in the analysis for such trips the results of the events would be further mitigated with less challenging results. It is, therefore, conservative not to credit the additional trips.

The overtemperature ΔT trip function is designed to preclude bulk boiling in the hot legs and to protect the DNBR safety limit over the range of allowable primary coolant pressures (Reference 15.0-10). Avoidance of bulk boiling assures that proper trip compensation is made for the DNB-influencing parameters (hot leg coolant temperatures and pressures). The trip function is set to protect against bulk boiling and DNB, with allowance for appropriate uncertainties in plant operation, temperature and pressure measurements, and trip channel performance.

The overtemperature ΔT trip function was evaluated statistically using the methodology described in Reference 15.0-10. This evaluation confirmed, on a static basis, that the trip function described in Table 15.0.7-2 provides protection against bulk boiling in the hot leg and against DNB at a 95% probability with a 95% confidence level. The transient analysis confirmed that the lead/lag compensation on the measured T_{avg} in conjunction with the static analysis provides the necessary protection for slow transients.

The statistical setpoint analysis uncertainties which were combined with local peaking uncertainties for measurement and for fuel pellet geometry were an overall trip channel uncertainty and a □I uncertainty. Both of these uncertainties were treated as two-sided 95% probability limits.

TABLE 15.0.7-1

TRIP SETPOINTS AND TIME DELAYS USED IN THE SAFETY ANALYSIS

Trip	Nominal Trip Setpoint	Setpoint Uncertainty	Biased Trip Setpoint Assumed in the Analysis	Nominal Time Delay (sec)
Power range high neutron flux, high setting	108%	±7.36	118%(of 2300 MWt)	.5
Power range high neutron flux, low setting	24%	±7.36%	35%(of 2300 Mwt)	.5
Overtemperature ΔT[1]	1.1265	0.0835[5]	1.24	0.75[2]
High pressurizer pressure	2376 psig	±25.57 psig	2415 psig	1.0
Low pressurizer pressure	1844 psig	+55.68/-54.72 psig [6]	1785 psig	1.0[3]
Low reactor coolant flow (from loop flow detectors)	94.26%	+5.82/-5.76%	87%	1.0
Low-low steam generator	16% span	+11.71%	0% span	1.0
Source Range High Neutron Flux	1.0E5 cps	N/A	N/A[7]	0.5

(1) A description of the overtemperature ΔT trip function is presented in Table 15.0.7-2.

(2) 0.75 sec. for electronic time delay. In addition, the thermal transient transport through the thermowell and the RTD response time are represented by a first order lag with a time constant of 4.0 seconds (nominal) or 5.0 seconds (in the analysis).

(3) 1.0 sec. for electronic delay. Also, the pressure signal for the low pressurizer pressure trip is compensated by a lead-lag controller with time constants of τ_{lead} = 10 seconds and τ_{lag} = 1 second with a +/-10% uncertainty. The analysis values of τ_{lead} = 9.0 seconds and τ_{ag} = 1.1 seconds accommodate that uncertainty.

(4) Deleted.

(5) Trip Channel uncertainty includes +/-2% ΔI uncertainty. An additional +/-1% ΔI uncertainty is accounted for in the statistical setpoint analysis.

(6) This is the setpoint uncertainty under harsh environment containment conditions.

(7) The source range trip is credited for mitigating the rod withdrawal from subcritical accident (UFSAR 15.4.1) because the power range high flux-low reactor trip is not required to be operable in Modes 3, 4, and 5. However, AREVA has analyzed this event assuming the reactor trips on the power range high flux-low reactor trip. This analysis remains bounding provided the source range trip setpoint remains below the power range high flux-low setpoint.

TABLE 15.0.7-2
DESCRIPTION OF OVER TEMPERATURE □T TRIP FUNCTION

$$\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{(1+\tau_1 S)}{(1+\tau_2 S)} (T-T') + K_3 (P-P') - f(\Delta I)\right]$$

where:

ΔT = Indicated □T

ΔT_o = Indicated □T at rated thermal power;*

T = Average temperature, ºF;

P = Pressurizer pressure, psig;

K_1 < 1.1265, nominal; 1.24, analysis value

K_2 = 0.01228; analysis value = 0.01228

K_3 = 0.00089; analysis value = 0.00089

$(1 + t_1S)/(1 + t_2S)$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

t_1 & t_2 = Time constants utilized in the lead-lag controller for T_{avg}, t_1 = 20.08 seconds, t_2 = 3.08 seconds;
(analysis value t_1 = 20.08, t_2 = 3.08)

T' = 575.9ºF Reference T_{avg} at rated thermal power;

P' = 2235 psig (Nominal RCS Operating Pressure);

S = Laplace transform operator, sec^{-1}; and f(Δ□I) is a function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start up tests such that:

(1) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% (analysis value = + 15%) in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% (analysis value = 2.4%/%) of the value of ΔT at rated power.

(2) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17% (analysis value = - 20%), the □T trip setpoint shall be automatically reduced by 2.4% (analysis value = 2.4%/%) of the value of ΔT at rated power.

*In the instrumentation, ΔT_o is set to 57.5ºF as the indicated temperature difference at full power. In the plant transient analysis calculation model, ΔT_o is set to the temperature difference for the initial conditions of rated thermal power plus measurement uncertainty and the minimum RCS flow allowed by Technical Specifications. In the statistical setpoint calculations, ΔT_o is set to the temperature difference for the initial conditions of 100% of rated thermal power and the RCS flow is set to the minimum allowed by the Technical Specifications.

15.0.8 COMPONENT CAPACITIES AND SETPOINTS USED IN THE ANALYSIS

Table 15.0.8-1 presents the component setpoints and capacities used in the analysis.

With regard to main and auxiliary feedwater flow to the steam generators, the main steam line break analysis in Section 15.1.5 is an exception to this table. In this particular event, increased feedwater flow contributes to the severity of the accident.

TABLE 15.0.8-1

COMPONENT CAPACITIES AND SETPOINTS USED IN THE SAFETY ANALYSIS

COMPONENT	RESPONSE TIME	NOMINAL SETPOINT	ANALYSIS SETPOINT	CAPACITY
Pressurizer Safety valves	0.7 s to 1.0 s[(a)]	2485 +/-1% (2460 to 2510 psig)	(1.02)(2410 psig) to (1.04)(2560 psig)[(b)]	293,330 lb/hr per valve [(c)] @(1.03)(2560 psig)
Steam line safety valves		1085 psig 1110 psig 1125 psig 1140 psig	1.03(1085 psig) 1.03(1110 psig) 1.03(1125 psig) 1.03(1140 psig)	667,229 lb/s 682,416 lb/s 1,001,760 lb/s 1,014,960 lb/s
Turbine stop and governor valves	0.1 s	-	-	-
Main steam isolation valves	7 s	-	-	-
Feedwater isolation valves	1.0 s[(d)]	-	-	-
Auxiliary feedwater	105 s	-	-	240 gpm[(e)]
Pressurizer PORVs (non - compensated)	3.0 s	2335 psig (open) 2327 psig (close)	2331 psig (open) 2323 psig (close)	(1.06)(511,200 lb/hr)@2477 psig
Pressurizer PORVs (non-compensated)		2340 psig (open) 2332 psig (close)	event specific (open) [(f)] event specific (close) [(f)]	

(a) The loop seal purge delay to the opening of the pressurizer safety valves ranges from 0.7 seconds (used for DNB-challenge cases) to 1.0 seconds (used for pressurization-challenge cases), based on the procedure and uncertainties given in Reference 15.0-11 and a 0.490 ft^3 loop seal liquid volume.

(b) The pressurizer safety valve setpoint used for DNB-challenge cases is based on the lower-bound rated setpoint, with 3% added for liquid-loop-seal setpoint shift and 1% subtracted for setpoint uncertainty. The setpoint used for pressurization-challenge cases is based on the upper-bound rated setpoint, with 3% added for liquid-loop-seal setpoint shift and 1% added for setpoint uncertainty.

(c) The analysis assumes that the pressurizer safety valves reach their rated capacity at a pressure 6% above the upper-bound rated pressure (based on 3% liquid-loop-seal setpoint shift and 3% accumulation).

(d) Steam Line Break analysis used a conservatively large value of 30 seconds.

TABLE 15.0.8-1

(e) A single motor-driven auxiliary feedwater pump delivering 240 gpm is credited in the Chapter 15 analysis. In most analyses, this results in two steam generators receiving 120 gpm each. However, in the small break loss-of-coolant accident (SBLOCA) analysis, a minimum of 80 gpm is assumed delivered to each generator. This assumption is applied to all three generators when, in reality, only one of the three would be receiving 80 gpm of auxiliary feedwater with the limiting safety train in operation. The limiting safety train configuration corresponds to one motor-driven pump supplying 240 gpm flow to two generators and the steam-driven auxiliary feedwater pump supplying 160 gpm flow to two generators, one of which is not being supplied flow by the motor-driven pump leaving only 80 gpm available for that generator. A capacity greater than that shown above is used in Steam Line Break analysis, because the higher flow contributes to the severity of that event.

(f) Beginning in Cycle 29, valve PCV-455C was converted from a compensated valve to a non-compensated valve. Two Chapter 15 accidents were reanalyzed with the new setpoints and results only changed for one of the accidents. The changes were determined to be negligible for the remainder of the events and no reanalysis was required.

15.0.9 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

Table 15.0.9-1 presents a tabular summary of trip functions, engineered safety features, and other equipment available for mitigation of accident effects.

TABLE 15.0.9-1

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1	Increase in Heat Removal by the Secondary System				
	Decrease in feedwater temperature/increase in feedwater flow	Power range high flux, manual	High-high steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	
	Increase in secondary steam flow	Power range high flux, overtemperature ΔT, over-power ΔT, manual		Pressurizer self-actuated safety valves, steam generator safety valves	
	Inadvertent opening of a steam generator or safety valve	Low pressurizer pressure, manual SIS, overpower ΔT, power range high flux	Low pressurizer pressure, high containment pressure, manual	Feedwater isolation valves, steam line isolation valves, Auxiliary FeedwaterFeedwater System	Safety Injection System
	Steam system piping failure	SIS, low pressurizer pressure, manual, overpower ΔT, power range high flux	Low pressurizer pressure, high containment pressure, manual	Feedwater isolation valves, steam line isolation valves, Auxiliary Feedwater System	Safety Injection System
15.2	Decrease in Heat Removal by the Secondary System				
	Loss of external electrical load/turbine trip	High pressurizer pressure overtemperature □T, manual		Pressurizer safety valves, steam generator safety valves	

TABLE 15.0.9-1 (Continued)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Loss of non-emergency AC power to the station auxiliaries	Steam generator low-low level, manual	Steam generator low-low level	Steam generator safety valves, Auxiliary Feedwater System	
	Loss of normal feed-water flow	Steam generator low-low level, manual	Steam generator low-low level	Steam generator safety valves, Auxiliary Feedwater System	
	Feedwater system pipe break	Steam generator low-low level, high pressurizer pressure, SIS, over-temperature ΔT, manual	High containment pressure, steam generator low-low water level, steam generator safety valves	Steam line isolation valves, feedline isolation, pressurizer self-actuated safety valves, Auxiliary Feedwater System	Safety Injection System
15.3	Decrease in Reactor Coolant System Flow Rate				
	Loss of forced reactor coolant flow	Low flow, undervoltage, underfrequency, manual		Steam generator safety valves	
	Reactor coolant pump shaft seizure (locked rotor)	Low flow, manual		Pressurizer safety valves, steam generator safety valves	
15.4	Reactivity and Power Distribution Anomalies				
	Uncontrolled rod cluster control assembly bank withdrawal from a sub-critical or low power startup conditions	Power range high flux, manual, souce range high flux			

TABLE 15.0.9-1 (Continued)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high flux, overtemperature ΔT, high pressurizer pressure, manual, high pressurizer level, overpower ΔT	Pressurizer safety valves, steam generator safety valves		
	Control rod misoperation	Overtemperature ΔT, manual			
	Startup of an inactive reactor coolant loop at an incorrect temperature	Low flow interlocked, manual			
	Boron dilution	Source range high flux, power range high flux, overtemperature ΔT, manual	Low insertion limit annunciators for boration		
	Spectrum of rod ejection accidents	Power range high flux, manual			
15.5	Increase in Reactor Coolant Inventory				
	Inadvertent operation of ECCS	Low pressurizer pressure, manual, safety injection trip		Safety Injection System	
15.6	Decrease in Reactor Coolant Inventory				
	Inadvertent opening of a pressurizer safety or relief valve	Pressurizer low pressure, overtemperature ΔT, manual			

TABLE 15.0.9-1 (continued)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
Steam generator tube failure	Low pressurizer pressure, overtemperature ΔT	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator shell side fluid operating system, steam generator safety and/or relief valves, steam line stop valves, Auxiliary Feedwater System	Emergency Core Cooling System Emergency Power System
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety and/or relief valves, Auxiliary Feedwater System	Emergency Core Cooling System Containment Heat Removal System

15.0.10 Effects of Fuel Rod Bowing and Mixed Assembly Types

H. B. Robinson used the same fuel assembly hydraulic design from the time AREVA Inc. (formerly known as Framatome ANP, Siemens Power Corporation, Advanced Nuclear Fuel and/or Exxon Nuclear) became the fuel supplier in the mid-1970's until the 1990 refueling outage. AREVA Inc.'s High Thermal Performance (HTP™) fuel was first introduced at the 1990 refueling outage. From a thermal-hydraulic standpoint, the major difference between the old Standard Mixing Vane (SMV) and new HTP™ design is an improved spacer or grid strap design and an increased number of them to improve mixing of coolant within the core. For the new HTP fuel, the HTP™ correlation has a DNBR safety limit of 1.141 (Reference 15.0-8). A penalty of 2% is applied to the DNB safety limit when hydraulically dissimilar fuel assemblies are used in the core. This penalty (Reference 15.0-4) accounts for any hydraulic differences between fuel types loaded into the Core.

The effects of rod bow for SMV fuel in the H. B. Robinson 2 Cycle 10 and subsequent cores of similar fuel types have been evaluated (Reference 15.0-5). A rod bow evaluation of the HTP™ fuel assemblies for burnups to 52,500 MWd/MTU showed that there is no reduction in DNB or LOCA-ECCS limits to an average assembly burnup of 47,000 MWd/MTU (Reference 15.0-9). Fuel assemblies with burnups greater than approximately 30,000 MWd/MTU cannot reach sufficiently high power densities that, even with a penalty from rod bow applied, they can be limiting with regard to DNB or to LOCA-ECCS peaking limits when compared to fuel assemblies with burnups below 30,000 MWd/MTU.

15.0.11 SINGLE ACTIVE FAILURES

As outlined in FSAR Section 3.1, the Reactor Protection System and Engineered Safety Features (ESF) are designed to be sufficiently redundant to perform their intended functions while accommodating the failure of any single active component. (For example, Main Steam Line Break is analyzed with a single active failure that minimizes the mitigating function of the ESF. This is in addition to consideration of Loss of Offsite Power and the most reactive control rod "hanging up" on reactor trip. Reference FSAR Section 3.1.1.2.7, first paragraph on page 3.1.1-7; and also Section 3.1.2.44, second to the last paragraph of response on page 3.1.2-30.)

The single failures listed in Table 15.0.11-1 are the limiting failures for each event. The purpose of this table is to show that the ESF design criteria are incorporated in postulation of the scenarios that define Chapter 15 transients.

TABLE 15.0.11-1

WORST SINGLE FAILURES

	Transient	Worst Single Failure	Comment
15.1.1	Decrease in Feedwater Temperature	----	bounded*
15.1.2	Increase in Feedwater Flow	----	bounded*
15.1.3	Increase in Steam Flow	----	no ESF involved
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	----	bounded*
15.1.5	Main Steam Line Break	one SI pump	delays arrival of boron
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	----	transient is not appli-cable to PWR plants (such as HBR)
15.2.2	Loss of Electric Load	----	no ESF involved
15.2.3	Turbine Trip ----		bounded*
15.2.4	Loss of Condenser Vacuum	----	bounded*
15.2.5	Inadvertent Closure of Main Steam Isolation Valves	----	bounded*
15.2.6	Loss of Offsite Power	----	bounded*

TABLE 15.0.11-1 (Continued)

	Transient	Worst Single Failure	Comment
15.2.7	Loss of Normal Feedwater	minimum Auxiliary Feedwater flow (3)	
15.2.8	Feedwater Line Break	----	bounded*
15.3.1	Loss of Forced Reactor Coolant Flow	----	no ESF involved
15.3.2	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	----	no ESF involved
15.3.3	Reactor Coolant Pump Shaft Break	----	no ESF involved
15.4.1	Uncontrolled Rod Withdrawal from a Subcritical or Low Power Condition	----	no ESF involved
15.4.2	Uncontrolled Rod Withdrawal at Power	----	no ESF involved
15.4.3	Control Rod Misoperation	----	no ESF involved
15.4.4	Startup of Inactive Loop	----	not analyzed*
15.4.5	Flow Controller Malfunction	----	transient is not applicable to PWR plants
15.4.6	Inadvertent Boron Dilution	----	no ESF involved

TABLE 15.0.11-1 (Continued)

	Transient	Worst Single Failure	Comment
15.4.7	Inadvertent Loading of a Fuel Assembly into an Improper Position	----	no ESF involved
15.4.8	Rod Ejection Spectrum	----	no ESF involved
15.4.9	Spectrum of BWR Control Rod Drop Accidents	----	transient is not applicable to HBR
15.5.1	Inadvertent Operation of ECCS	----	not analyzed*
15.5.2	CVCS Malfunction that Increases Reactor Coolant System Inventory	----	bounded*
15.6.1	Inadvertent Opening of Pressurizer Safety or Power Operated Relief Valve	(2)	
15.6.2	Small Break Loss of Coolant Accident	one of two diesel generators	loss of offsite power is a necessary consideration
15.6.3	SG Tube Rupture	minimum Auxiliary Feedwater flow(3)	blowdown is increased by retarding cooldown
15.6.4	Spectrum of BWR Steam Piping Failures Outside Containment	----	transient is not applicable to PWR plants
15.6.5	Large Break Loss of Coolant Accident	one of two emergency electrical buses	
15.7.1	Radioactive Waste Gas System Leak or Failure	----	no ESF involved

TABLE 15.0.11-1 (Continued)

	Transient	Worst Single Failure	Comment
15.7.2	Liquid Waste System Leak or Failure	----	no ESF involved
15.7.3	Radioactivity Release Due to Liquid Tank Failure	----	no ESF involved
15.7.4	Fuel Handling Accident	----	no ESF involved
15.7.5	Spent Fuel Cask Drop	----	no ESF involved
15.7.6	Spent Fuel Pit Water Loss	----	no ESF involved

Notes:

(1) Deleted.

(2) Although the worst single failure for this event would be the loss of one of two SI pumps, the analysis presented in Section 15.6.1 does not take credit for SI.

(3) The minimum acceptable Auxiliary Feedwater System performance is one motor driven pump delivering flow to two steam generators. This criterion is more conservative than a single active failure for this system in order to be consistent with the HBR2 Technical Specifications.

* As explained in XN-NF-83-72, Revision 2, Supplement 1, "Disposition of Chapter 15 Events", either (a) the consequences of this event are bounded by a similar Chapter 15 transient, or (b) no analysis is required for the event.

15.0.12 COMMON DOSE CONSEQUENCE INPUTS FOR ALTERNATIVE SOURCE TERM (AST) ANALYSES

15.0.12.1 Source Terms

Core inventory isotopics were developed using a bounding approach. The ORIGEN-S computer code (Reference 15.0.12-8) was used to develop isotopics for a variety of burnups, enrichments, and burnup rates (power levels). A plant specific set of high burnup equilibrium fuel cycles were postulated to cover wide variations in cycle energy, enrichment, and batch sizes. The resulting ORIGEN-S calculated isotopics were increased to account for the 2% measurement uncertainty above the then-current licensed power of 2300 MWt to bound operation at power levels (including allowances for measurement uncertainty) of up to 2346 MWt. These adjusted isotopic inventories were then compared to a bounding DOE/RW 0184, R1 (July 1992) LWR Isotopics Characteristics database of other possible isotopic inventories that were developed using generic Westinghouse-style 15x15 fuel descriptions. A bounding, conservative inventory was chosen from this full, composite set of isotopics. Therefore, the core inventory used in the AST dose analyses should bound any fuel cycles up to 5 w/o enrichment, 2346 MWt core power (including adjustments for measurement uncertainty), and 18 month cycle length. The core inventory used in the AST analyses that involve fuel damage is provided in Table 15.0.12-1.

Certain analyses require the use of RCS isotopic concentrations at the Technical Specifications limits. As specified in Reference 15.0.12-3 for iodine spiking considerations, certain events have been analyzed at higher RCS radionuclide concentrations. Unless otherwise noted in the individual analysis discussions, these events start from the RCS inventory in Table 15.0.12-2. Similarly, unless otherwise noted in the individual analysis discussions, for those events which consider releases from the secondary system, the secondary system radionuclide concentrations at the Technical Specifications limits are used, as shown in Table 15.0.12-3.

Regulatory Guide (RG) 1.183 requires that certain fuel design criteria be met in order to use the RG specified release fractions (Section 3.2 of the RG, Footnotes 10 and 11). Specifically, peak fuel burnup should not exceed 62,000 MWD/MTU (Footnote 10) and the maximum linear heat generation rate should not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU (Footnote 11). Using the bounding fuel cycle specified to develop the AST core inventory, the maximum discharge batch exposure was 60,000 MWD/MTU. Also, this source term basis fuel cycle shows that fuel batches with average burnups in excess of 54,000 MWd/MTU have heat generation rates less than 6.0 kw/ft at 2300 MWt rated thermal power. Applying the 2% increase to bound the Appendix K Measurement Uncertainty Recovery (MUR) power uprate increases this parameter to 6.12 kw/ft. Therefore, both Regulatory Guide footnote restrictions are met.

15.0.12.2 Other Common Inputs

Table 15.0.12-4 presents Control Room input parameters and Table 15.0.12-5 presents breathing rates and occupancy factors used in the dose analyses.

TABLE 15.0.12-1

CORE RADIONUCLIDE INVENTORY @ T = 0, 2346 MWt

Isotope	Curies	Isotope	Curies	Isotope	Curies
Co-58	5.99E+05	Ru-103	9.87E+07	Cs-136	3.52E+06
Co-60	4.58E+05	Ru-105	6.83E+07	Cs-137	8.87E+06
Kr-85	7.30E+05	Ru-106	3.73E+07	Ba-139	1.15E+08
Kr-85m	1.51E+07	Rh-105	6.33E+07	Ba-140	1.13E+08
Kr-87	3.03E+07	Sb-127	5.38E+06	La-140	1.17E+08
Kr-88	4.20E+07	Sb-129	2.03E+07	La-141	1.02E+08
Rb-86	1.16E+05	Te-127	5.31E+06	La-142	9.83E+07
Sr-89	5.90E+07	Te-127m	8.87E+05	Ce-141	1.04E+08
Sr-90	6.16E+06	Te-129	1.90E+07	Ce-143	9.55E+07
Sr-91	7.39E+07	Te-129m	3.84E+06	Ce-144	8.18E+07
Sr-92	7.88E+07	Te-131m	1.23E+07	Pr-143	9.34E+07
Y-90	6.62E+06	Te-132	8.91E+07	Nd-147	4.17E+07
Y-91	7.69E+07	I-131	6.20E+07	Np-239	1.25E+09
Y-92	7.93E+07	I-132	9.02E+07	Pu-238	2.81E+06
Y-93	6.07E+07	I-133	1.28E+08	Pu-239	2.44E+04
Zr-95	1.05E+08	I-134	1.41E+08	Pu-240	3.55E+04
Zr-97	1.00E+08	I-135	1.21E+08	Pu-241	9.89E+06
Nb-95	1.06E+08	Xe-133	1.28E+08	Am-241	1.18E+04
Mo-99	1.16E+08	Xe-135	3.68E+07	Cm-242	3.23E+06
Tc-99m	1.03E+08	Cs-134	1.25E+07	Cm-244	3.88E+05

TABLE 15.0.12-2

RCS EQUILIBRIUM ACTIVITY LIMITED TO 0.25 µCI/GRAM DOSE EQUIVALENT I-131
PRIOR TO THE ACCIDENT, 2346 MWt

Isotope	µCI/GRAM
Kr-85	5.41E-01
Kr-85m	1.30E-01
Kr-87	8.91E-02
Kr-88	3.20E-01
Rb-86	Negligible
I-131	1.93E-01
I-132	7.12E-02
I-133	3.11E-01
I-134	4.37E-02
I-135	1.67E-01
Xe-133	2.14E+01
Xe-135	5.88E-01
Cs-134	2.06E-02
Cs-136	2.96E-03
Cs-137	1.12E-01

TABLE 15.0.12-3

SECONDARY COOLANT SYSTEM EQUILIBRIUM ACTIVITY

Isotope	SGTR µCI/GRAM	NON-SGTR EVENTS µCI/GRAM
I-131	7.72E-02	7.72E-02
I-132	2.85E-02	2.85E-02
I-133	1.25E-01	1.25E-01
I-134	1.75E-02	1.75E-02
I-135	6.69E-02	6.69E-02
Cs-134	2.06Ee-03	4.12E-03
Cs-136	2.96E-04	5.92E-04
Cs-137	1.12E-02	2.24E-02

Note: This table presents the secondary coolant system equilibrium activity that was assumed for the Steam Generator Tube Rupture event and for other events resulting in the release of secondary side activity. The iodine nuclide activity is based on the Technical Specifications limit for Dose Equivalent Iodine-131 of 0.1 µCi/gm for the secondary side. For the SGTR, the cesium nuclide activity is based on 10% of the RCS cesium activity corresponding to a Technical Specifications limit of 0.25 µCi/gm Dose Equivalent I-131. For the non-SGTR events, the cesium nuclide activity is based on 10% of the RCS cesium activity corresponding to a Technical Specifications limit of 0.50 µCi/gm Dose Equivalent I-131.

TABLE 15.0.12-4
(page 1 of 2)

CONTROL ROOM PARAMETERS FOR DOSE ANALYSES

A. Atmospheric Dispersion Coefficients (X/Q) from various release points to the Control Room – not corrected for occupancy

Release Point	Time Period	sec/m^3
Containment Nearest Point	0 – 2 hours	4.15E-03
	2 – 8 hours	2.74E-03
	8 – 24 hours	1.17E-03
	1 – 4 days	8.18E-04
	4 – 30 days	6.74E-04
Plant Stack	0 – 2 hours	1.24E-03
	2 – 8 hours	8.97E-04
	8 – 24 hours	3.62E-04
	1 – 4 days	2.58E-04
	4 – 30 days	2.14E-04
MSSV/PORV	0 – 2 hours	2.60E-03
	2 – 8 hours	1.65E-03
	8 – 24 hours	7.22E-04
	1 – 4 days	4.97E-04
	4 – 30 days	4.01E-04
Closest Main Steam Line	0 – 2 hours	2.48E-03
	2 – 8 hours	1.57E-03
	8 – 24 hours	7.05E-04
	1 – 4 days	4.74E-04
	4 – 30 days	3.93E-04
RHR Heat Exchanger Room	0 – 2 hours	7.13E-03
	2 – 8 hours	5.49E-03
	8 – 24 hours	2.29E-03
	1 – 4 days	1.71E-03
	4 – 30 days	1.37E-03
Fuel Handling Building Wall	0 – 2 hours	1.34E-03
	2 – 8 hours	1.02E-03
	8 – 24 hours	4.31E-04
	1 – 4 days	3.21E-04
	4 – 30 days	2.56E-04

TABLE 15.0.12-4
(page 2 of 2)

CONTROL ROOM PARAMETERS FOR DOSE ANALYSES

B. Control Room Volume and Ventilation Assumptions

Control Room:

Control Room Habitability Volume	20,124 ft^3
Assumed Unfiltered Inleakage*	
0 to 1 Hour	300 cfm
1 Hour until End of Event	230 cfm

Control Room Ventilation:

Normal Mode Operation – Outside Air Intake	400 cfm
Normal Mode – Roughing Filter	
Aerosol Removal	0 %
Elemental Iodine Removal	0 %
Organic Iodine Removal	0 %
Time of Switchover from Normal to Emergency Pressurization Mode of Operation	**
Emergency Pressurization Mode – Outside Air Intake	400 cfm
Emergency Pressurization Mode – HEPA Filter, Aerosol Removal	99 %
Emergency Pressurization Mode – Charcoal Filter	
Elemental Iodine Removal	95 %
Organic Iodine Removal	95 %
Emergency Pressurization Mode – Filtered Recirculated Air Flow	2600 cfm

* See Sections 15.1.5, 15.3.2, 15.4.3.1, 15.6.3, 15.6.5, 15.7.1, and 15.7.4 for actual air inleakage values used in event specific analysis.

** See individual accident evaluation sections for values used in dose consequence analysis. The switchover to Emergency Pressurization Mode will be initiated either by the Safety Injection signal, or by the “High Radiation in the Control Room” alarm, or may be assumed to not occur at all, if sufficient dose consequence margin exists. Conservatively long switchover assumptions are made in various event analyses that are confirmed to be bounding for the cited initiation signals.

TABLE 15.0.12-5

BREATHING AND OCCUPANCY RATES FOR DOSE ANALYSES

Onsite Breathing Rates	3.5E-04 m^3/sec (0 to 30 days)
Onsite Occupancy Factors	1.0 (0 to 1 day) 0.6 (1 to 4 days) 0.4 (4 to 30 days)
Offsite Breathing Rates	3.5E-04 m^3/sec (0 to 8 hours) 1.8E-04 m^3/sec (8 to 24 hours) 2.3E-04 m^3/sec (1 to 30 days)
EAB Occupancy Factor	1.0 (0 to 2 hours)
LPZ Occupancy Factor	1.0 (0 to 30 days)

REFERENCES: SECTION 15.0

15.0-1 Robinson Nuclear Plant Realistic Large Break LOCA Analysis, ANP-2973(NP) Revision 0, AREVA NP Inc., August 2011.

15.0-2 Technical Specifications and Bases for H. B. Robinson Unit 2, Appendix A to the Facility Operating License DPR-23, Docket No. 50-261, Carolina Power and Light, Hartsville, SC.

15.0-3 ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, Richland, WA, May 1992.

15.0-4 Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, XN-NF-82-21(P)(A), Revision 1, Exxon Nuclear Company, Richland, WA, September 1983.

15.0-5 Computational Procedure for Evaluating Fuel Rod Bowing, XN-NF-75-32(P)(A), Supplements 1, 2, 3, & 4, Exxon Nuclear Company, Richland, WA, October 1983.

15.0-6 H. B. Robinson Unit 2 Plant Small Break LOCA Analysis, ANP-2972 (NP), Revision 0, AREVA NP Inc., October 2011.

15.0-7 Letter from H. G. Shaw (SPC) to B. A. Morgan (CP&L), "Minimum Temperature for Criticality for H. B. Robinson Unit 2," HGS:94:472, December 22, 1994.

15.0-8 Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, EMF-92-153 and Supplement 1, Siemens Power Corporation, Richland, WA, March 1994.

15.0-9 H. B. Robinson Unit 2, Cycle 17 Safety Analysis Report, EMF-95-031, Siemens Power Corporation, Richland, WA, April 1995.

15.0-10 Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, EMF-92-081(P)(A), Revision 1, Siemens Power Corporation, Richland, WA, February 2000.

15.0-11 Pressurizer Safety Valve Set Pressure Shift, WCAP-12910, Westinghouse Electric Corporation, Pittsburgh, PA, March 1991.

15.0-12 SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, EMF-2310(P)(A), Revision 1, Framatome ANP, Inc., May 2004

REFERENCES: SECTION 15.0 (continued)

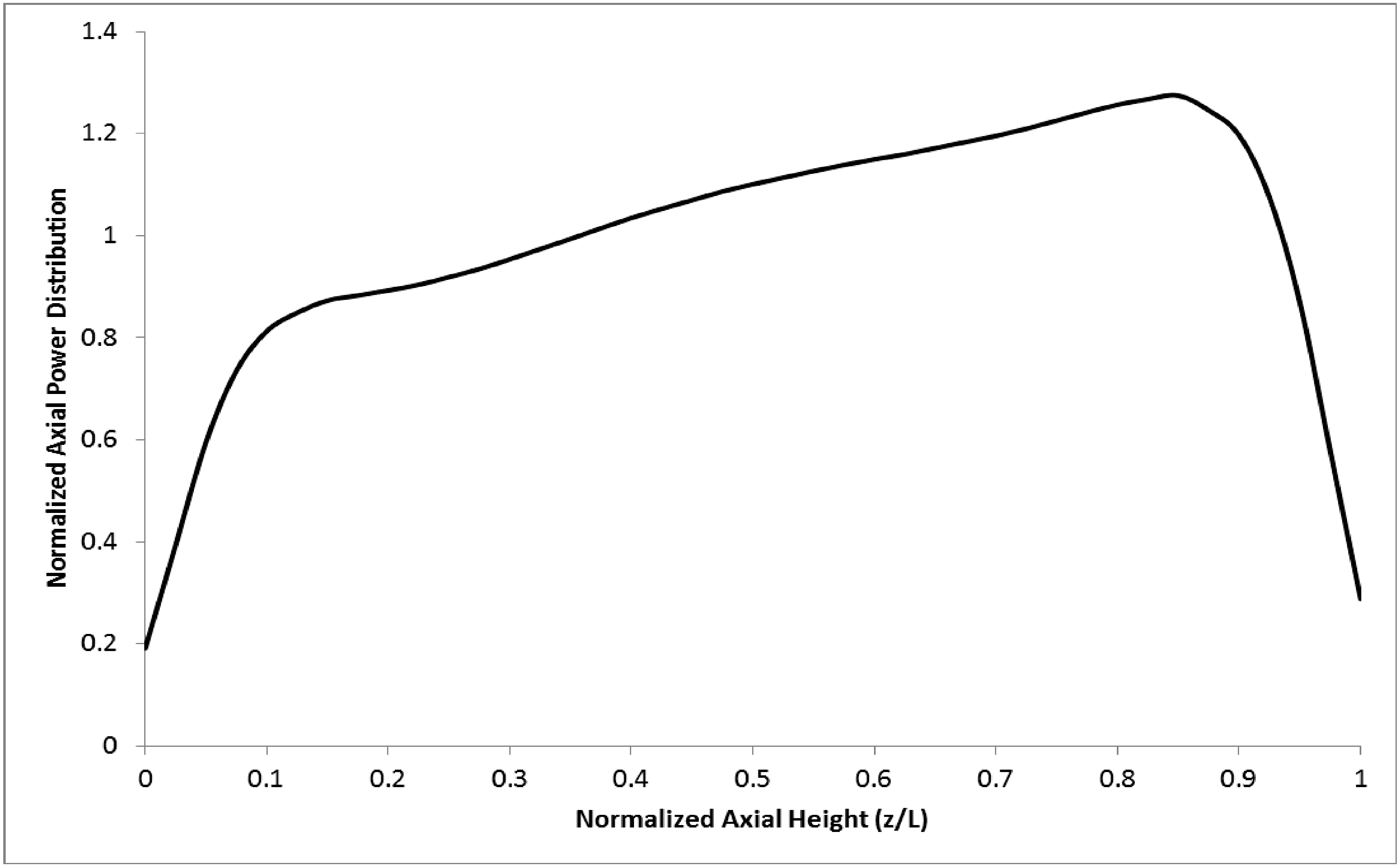
15.0.12-1 "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, May, 1997. ARCON96, RSICC Computer Code Collection No. CCC-664.

15.0.12-2 "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," NUREG/CR-6604, April 1998 and Supplement 1, June 8, 1999.

15.0.12-3 NRC Regulatory Guide No. 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents At Nuclear Power Reactors," July 2000.

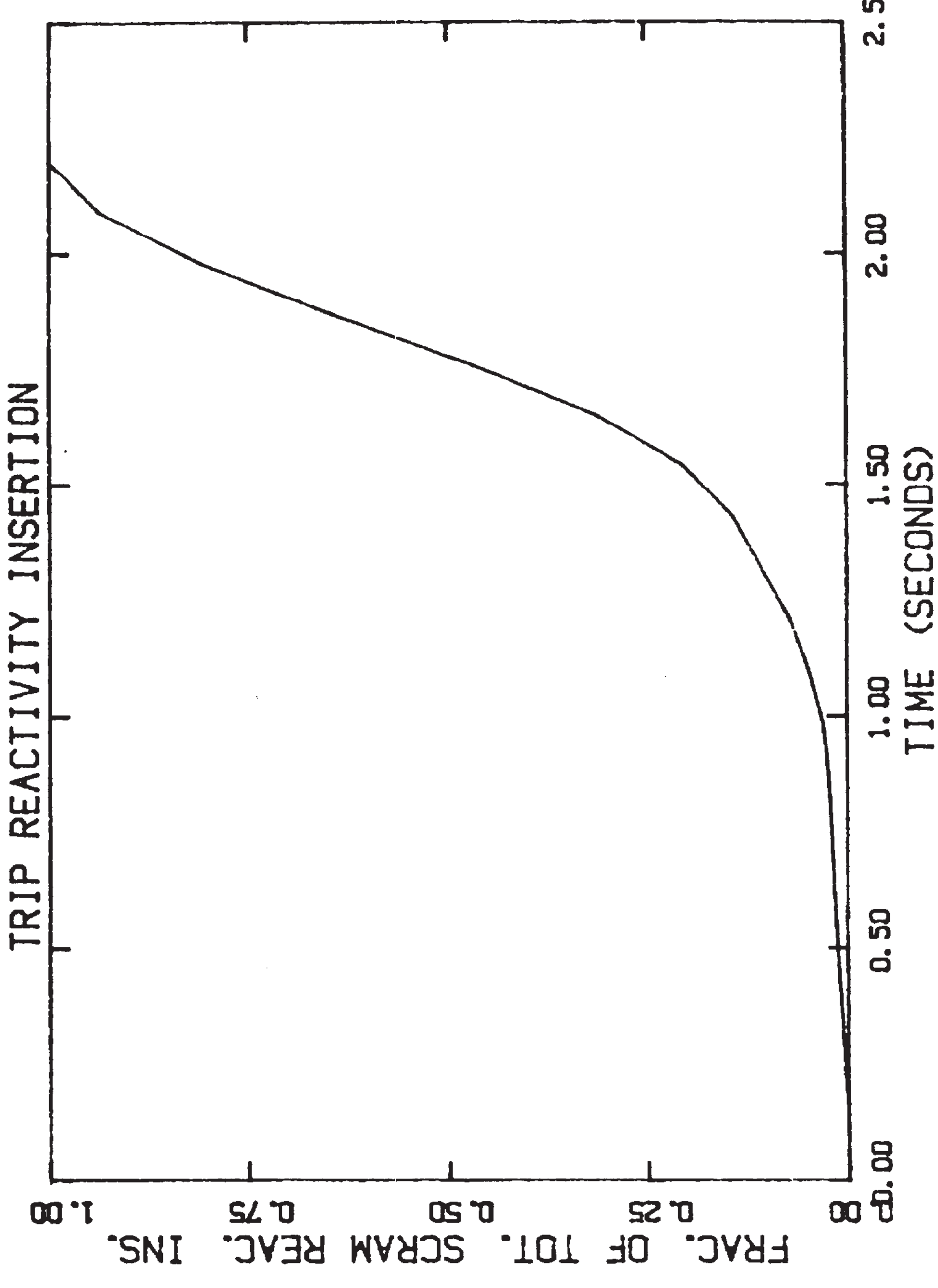
15.0.12-4 NRC Letter dated September 24, 2004, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105)."

15.0.12-5 10 CFR 50.67, "Accident Source Term"



REVISION NO. 27

H. B. Robinson Unit 2 Duke Energy UPDAED FINAL SAFETY ANALYSIS REPORT	Limiting Axial Power Distribution For Most DNB Events	FIGURE 15.0.3-1



AMENDMENT 3

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR TRIP REACTIVITY INSERTION, MOST REACTIVE ROD STUCK OUT	FIGURE 15.0.6 - 1

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the Reactor Coolant System (RCS) by the Secondary System. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented in this section:

a) Feedwater system malfunctions that result in a decrease in feedwater temperature

b) Feedwater system malfunctions that result in an increase in feedwater flow

c) Excessive increase in secondary steam flow

d) Inadvertent opening of a steam generator relief or safety valve, and

e) Steam system piping failure.

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE IN FEEDWATER TEMPERATURE

The event is caused by malfunction of a feedwater bypass valve which diverts flow around the low pressure feedwater heaters at full power. A conservative calculation of the increase in feedwater flow rate was performed, since it was assumed that all flow in that train bypassed the preheaters. The incremental heat removal capacity was calculated to be 3.8×10^8 Btu/hr. Power uprate to 2339 MWt will not impact the incremental heat removal capacity for this event. The added heat removal capacity of the 10 percent load increase event (15.1.3) is 7.8×10^8 Btu/hr. (Note: 7.8×10^8 Btu/hr corresponds to 10% of 2300 MWt. Event 15.1.3 is actually analyzed with a load increase of 10% of 2346 MWt. Both values bound the incremental increase in heat removal capacity produced by this event.) The magnitude of the initiator of this event is less than that of the 10 percent load increase event.

The thermal inertia of the steam generator will slow the reactivity insertion of the feedwater temperature decrease event in comparison to the load increase event; the cooler feedwater must mix with the generator inventory. The load increase event, however, results in a pressure decrease in the steam generator which results in a cooldown of inventory. Therefore, the event is bounded both in rate and magnitude by the 10 percent load increase event. The acceptance criteria for these events are the same.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW

This event is caused by malfunction (opening) of the main feedwater valve during startup or low power operation when feedwater is controlled by the bypass valves. A step increase in feedwater to one steam generator to rated flow at a feedwater temperature of 70ºF could occur.

Two events are relevant:

a) During full power operation, one of the steam generator feedwater regulating valves opens to full capacity; and

b) During startup, the reactor is operated on the feedwater bypass system. While operating in bypass, a steam generator feedwater flow regulating valve opens to full capacity.

To evaluate the increase in cooling capacity of the sub-event at power, the feedwater regulating valve is assumed to deliver full flow with the increased coolant delivered at 70ºF for power uprate conditions (2339 MWt) using a bounding feedwater flow value of 4.8×10^6 lb/hr. The increased heat removal capacity was calculated to be 6.8×10^8 Btu/hr. The 10 percent load increase event results in an increased cooling demand of 7.8×10^8 Btu/hr. (Note: 7.8×10^8 Btu/hr corresponds to 10% of 2300 MWt. Event 15.1.3 is actually analyzed with a load increase of 10% of 2346 MWt. Both values bound the incremental increase in heat removal capacity produced by this event.) This sub-event is, therefore, bounded by the 10 percent load increase event (15.1.3).

The calculation of the increased cooling capacity sub-event during startup conservatively assumed the feedwater regulating valve permitted full flow, even though only one feedwater train is operated in this mode and some of its capacity is used by the bypass line. Since this is a cooldown event, it is most limiting at end of cycle when the moderator temperature coefficient is most negative. Using the most negative moderator coefficient, the maximum reactivity insertion due to this malfunction was calculated to be 4.4×10^{-4} $\Delta\rho$/sec for power uprate conditions (2339 MWt) using a bounding feedwater flow value of 4.8×10^6 lb/hr. This is compared to the insertion rate used in the rod withdrawal event at subcritical or low power (15.4.1). The latter is greater than the calculated insertion rate for this event and is, therefore, bounded by the results of 15.4.1.

This sub-event may be conservatively compared to the rod withdrawal event, even though this is a cooldown in comparison to a heatup event. In 15.4.1 the coolant temperature will increase. The net effect is a reduction in thermal margin. Conversely, in 15.1.2 the coolant temperature decreases, which increases thermal margin relative to a heatup event. Therefore, the events being otherwise equal in terms of reactivity insertion rate, the rod withdrawal event produces a lower MDNBR because of the increase in coolant temperature. Similar to the rod withdrawal event, the reactor would trip on the power range (low setting) flux trip set at approximately 25 percent of rated power.

It is concluded that this event is bounded by the results of 15.1.3 and 15.4.1.

15.1.3 Increase in Steam Flow (Excess Load)

15.1.3.1 Identification of Causes and Event Description

The increase in steam flow event is initiated by an increase in steam demand. The increased steam demand may be initiated by the operator, system demand, or regulating valve malfunction. The step increase in steam flow used bounds the maximum capacity of the turbine steam regulating valves.

The event initiator is a step increase in steam flow. The feedwater regulating valves open to increase the feedwater flow to match the new steam demand and maintain steam generator water level. In response to the increased steam flow, the secondary system pressure decreases, resulting in an increase in the primary-to-secondary heat transfer rate. The primary side steam generator outlet temperature decreases due to the enhanced heat removal. As a consequence, the primary system core average temperature decreases and the primary system fluid contracts, resulting in an outsurge of fluid from the pressurizer. The pressurizer level and pressure decrease as fluid is expelled from the pressurizer.

The effect of this cooldown on the core power level will depend upon the sign of the moderator temperature coefficient and the state of the Rod Control System. If automatic control rod withdrawal is blocked[a], negative moderator feedback will increase the core power as the coolant temperature decreases, and the reactor system will reach a new steady-state condition at a power level which is consistent with the increased heat removal rate. (Positive moderator feedback, on the other hand, would decrease the core power level and not challenge the acceptance criterion.)

This event is classified as a Condition II event (Table 15.0.1-1). The relevant acceptance criteria are described in 15.0.1.1. As cited in Table 15.0.11-1, no single failure in the ESF will affect the analysis for this event.

15.1.3.2 Analysis Method

The analysis was performed using the S-RELAP5 and XCOBRA-IIIC codes. The S-RELAP5 methodology (Reference 15.0-12) was used to model the salient system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures) and produce an approximate DNBR calculation to estimate the time at which the DNBR was a minimum. The core fluid boundary conditions and average rod surface heat flux at this time were then used as input to the XCOBRA-IIIC code (Reference 15.0-4), which was used to more precisely evaluate the MDNBR value for the analysis.

15.1.3.3 Definition of Events Analyzed and Bounding Input

This event is predominantly a depressurization event, so the primary concern for this event is the challenge to the specified acceptable fuel design limits (SAFDLs). Therefore, the cases identified for analysis for this event are selected on the basis of bounding the largest challenge to the SAFDLs.

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[a] Automatic control rod withdrawal was not considered in the analysis.

This event is analyzed at full power conditions because at full power the margin to the SAFDLs is the smallest. Thus, full power conditions bound operation at lower power levels.

The core burnup (beginning of cycle or end of cycle) was selected to maximize the challenge to the SAFDLs. The time in the cycle will determine the value of the moderator reactivity temperature coefficient. If the moderator reactivity temperature coefficient is negative, there will be a positive reactivity insertion as the core average temperature decreases. The magnitude of this positive insertion is dependent on how negative the moderator reactivity temperature coefficient is. If the moderator temperature coefficient is positive, then negative reactivity will be inserted as the coolant temperature decreases, causing the power to decrease with less challenge.

The following conditions were used:

Initial power	100.3% of 2339 MWt
Moderator temperature coefficient	-45.0 pcm/°F
Doppler coefficient (biased)	-1.10 pcm/°F
Increase in steam flow	10% step increase
Core inlet temperature	Nominal
Initial RCS pressure	Nominal
Core outlet pressure used in subchannel analysis	Nominal -40 psi
Pressurizer level	Nominal
Pressurizer heater	Disable
Pressurizer level control	Disable

15.1.3.4 Analysis of Results

The event is initiated by a 10 percent step increase in turbine steam flow. The steam dome pressures drop about 53 psi, resulting in lower steam generator temperatures, increased primary to secondary heat transfer, and hence a reduction in primary side temperature. Vessel average temperature drops about 2°F. This temperature reduction results in a depressurization of the primary system about 29 psia, lowers the pressurizer level about 3%, and provides a positive reactivity insertion which peaks at about 0.024 dollars.

The positive reactivity insertion increases core power until the increased load demand is balanced.

Eventually, a new steady state operating condition is reached. This occurs at about 150 sec. The challenge to the DNBR limit results from the combination of rising power and decreasing pressure.

The transient response is shown in Figures 15.1.3-1 to 15.1.3-7. The event sequence is summarized in Table 15.1.3-1. The minimum DNBR computed for the event is 1.540, which is significantly greater than the DNBR limit of 1.141.

15.1.3.5 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted is greater than the limit. The limit assures that with 95 percent probability and confidence limits, DNB is not expected to occur; therefore, no fuel is expected to fail.

TABLE 15.1.3-1

INCREASE IN STEAM FLOW EVENT SUMMARY

TIME	EVENT	VALUE
0.0 sec.	10% Step increase in turbine flow	
13.5 sec.	Maximum reactivity	0.024 dollars
150 sec.	New quasi-steady-state operating condition	
	Core power	111% RTP
	Pressurizer pressure	2221 psia
	Minimum DNBR	1.540

15.1.4 Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve

The event is caused by malfunction of a single power operated (PORV) or safety relief valve (SRV). There are two sub-events: (1) at power; and (2) after reactor trip. The safety relief valves are independent mechanical type valves allowing no common mode failure. Similarly, the PORVs have no common mode malfunction. Therefore, the consequences of this event are limited to malfunction of a single valve. The maximum capacity of one PORV (approximately 580,000 lbs/hr) is significantly less than that for one SRV which is 1,015,000 lb/hr. The total initial steam demand for 15.1.3 at 102% of 2300 MWt for H. B. Robinson is about 10,300,000 lb/hr. Therefore, the 10 percent load increase of the 15.1.3 analysis will conservatively treat the consequences of the secondary valve malfunction event at power. This event is, therefore, bounded by the results of 15.1.3.

The second sub-event is postulated to occur after reactor trip and is most limiting at end of cycle. The acceptance criteria is that adequate shutdown margin exists to preclude penetration of the DNB SAFDL after trip with spurious relief of steam resulting from a stuck open steam generator relief or safety valve. Reference 15.1.5-1 presents results of the main steamline break analysis. That analysis shows that even with rupture of a main steamline, the DNB SAFDL is not penetrated. Since the steam relief rate of a stuck open steam generator relief or safety valve is much less than that of a main steamline rupture, and the DNB SAFDL is not penetrated for steamline break, the results of this event are acceptably bounded by those of steamline break and separate analysis of this event is not necessary.

This disposition is not impacted by power uprate to 2339 MWt.

15.1.5 Main Steamline Break Event

15.1.5.1 Introduction

The Main Steamline Break event analyzed is the most severe case of an uncontrolled steam release from a steam generator. The break would result in a large initial steam flow, decreasing during the accident as the steam pressure falls. The energy removal from the RCS would cause a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown would result in a reduction of core shutdown margin. If the most reactive control rod assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core power transient is stabilized by Doppler feedback and by the decrease in moderator density in the core. With time, the increasing boron concentration in the reactor causes the power to continuously reduce. When the operator secures auxiliary feedwater, the temperature of the RCS increases and the event is terminated.

The MSLB analysis was performed to demonstrate the acceptability of the analysis covering the range of Shutdown Margin versus Boron Concentration represented in the RNP COLR. This included the EOC case with a 1770 pcm minimum shutdown margin requirement, and an additional case representing the 640 ppm boron point with the 1000 pcm minimum shutdown margin requirement.

15.1.5.2 Analysis Basis

This event was analyzed from hot full power and hot zero power with and without offsite power available (Reference 15.1.5-1 & 15.1.5-10). For the "with off-site power" case, the "all RCP's running" condition has been determined to bound the other allowable RCP scenarios. The MSLB challenges both fuel centerline melt (FCM) and minimum departure from nucleate boiling ratio (MDNBR) criteria. The case that presents the greatest challenge to MDNBR is the hot full power (HFP) with offsite power available. The HZP with offsite power case has the least margin to the fuel acceptance criteria and is the case displayed in Figures 15.1.5-3 to 15.1.5-14.

The bases for this analysis are listed in Tables 15.1.5-1 and 15.1.5-2. This event was analyzed with the most reactive control rod assumed to be stuck out of the core. The worst single failure assumed in the analysis is the loss of one of the two operable High Head Safety Injection (HHSI) pumps. The remaining HHSI pump is assumed to take suction from the refueling water storage tank at 45°F and discharge into the BIT. The initial inventories of the BIT and of the injection lines between the RCS cold legs and the HHSI pumps were assumed to be unborated water. The initial inventory of the BIT was assumed to flow to the cold leg of the primary ahead of and unmixed with the borated water from the refueling water storage tank. Flow in the lines between the HHSI pumps and the cold leg injection locations was conservatively assumed to have no axial mixing. Initial boron concentration in these lines was assumed to be zero. Flow at the cold leg injection point was controlled by the injection delivery curve illustrated in Figure 15.1.5-1.

The HHSI pump can provide safety injection as soon as the primary system pressure at the injection point drops below the HHSI pump shut-off head(1435 psia). Safety injection by the HHSI pump was delayed for 15 seconds after the SI actuation trip setpoint was reached to account for the time required for the HHSI pump to come up to speed when offsite power is available.

The limiting break location is downstream of the steamline flow nozzle (i.e., flow meter) of the affected steam generator. The break flow area for the affected steam generator is at the integral flow restrictor of that steam generator.

No credit was taken for the steam line check valves until the Main Steam Isolation Valves (MSIVs) close: Instead of preventing initial reverse flow from the intact loops out the break, the intact loops make a significant contribution to the blowdown until MSIV closure is complete. This basis bounds the case for a (different) break location outside the Reactor Containment Building between the steam line check valves and the steam line header (for a break inside the Reactor Containment Building, check valve operability is necessary to prevent continued blowdown of an intact steam line if one MSIV fails to close as an alternative single active failure). Although in actuality a break located inside containment gives a High Steam Line Differential Pressure signal because of immediate check valve closure, only a High Steam Line Flow Coincident with Low Steam Line Pressure (see Table 15.1.5-2) was used in the analysis.

For the HFP cases, all of the initial main feedwater (MFW) flow is assumed to be delivered to the affected steam generator. The MFW is modeled to be isolated 30 seconds after receiving the isolation signal.

For the HZP cases, the MFW pumps are assumed to take suction from the condensate storage tank (CST) at a temperature of 33°F. The MFW flow is based on the operation of one of two Feedwater trains since this is the maximum number of trains that would be operable under HZP conditions. Flow as a function of steam generator pressure is given in Figure 15.1.5-2. This flow is based on the combined head/flow characteristics of one condensate pump and one feedwater pump connected in series with a conservative treatment of the hydraulic characteristics in the feedwater train including the control valve. This results in a conservatively high MFW flow rate. Like the HFP cases, the MFW will be isolated 30 seconds after receiving the isolation signal.

AFW flow is assumed to start at break initiation and is held constant over the entire transient. The operator is assumed to terminate the AFW flow at 600 seconds. The AFW pumps are assumed to take suction from the CST at 33°F. All of the AFW flow is assumed to be delivered to the affected steam generator.

Trips for the SIS, main feedwater valves, and main steam isolation valves (MSIVs) are given in Table 15.1.5-2. Uncertainty biases are included in the trip setpoints as shown. Delay times given are between the time the trip setpoint is reached and completion of the safety action.

The reactor kinetics were calculated using a point kinetics model. The minimum bounding values for the Moderator Temperature Coefficient at full power based on a 0 ppm boron condition (-45 pcm/°F) and a 640 ppm boron condition were used in the event analysis. The minimum shutdown margin that
corresponds to these conditions is 1770 pcm (0 ppm boron) and 1000 pcm (640 ppm boron). The parameters in the steamline break analysis that have a direct impact on core return to power have been biased or modeled to conservatively maximize the return to power. Specifically, biases or models were utilized which: (1) delay injection of boron into the core; (2) delay closure of the MSIVs; (3) increase the flow rate and decrease the temperature of the feedwater; (4) limit mixing between loops; and, (5) increase positive reactivity feedback and decrease negative reactivity feedback. Other parameters in the analysis such as plant geometric parameters and plant thermal hydraulic initial conditions have been taken at their nominal values.

15.1.5.3 Calculation Results

These calculations were performed in accordance with the AREVA steamline break methodology (Reference 15.1.5-2). NSSS response is computed with S-RELAP5, detailed core neutronics characteristics are computed with PRISM, and the detailed core flow distribution is computed with XCOBRA-IIIC.

15.1.5.3.1 NSSS simulation

The S-RELAP5 simulation of the NSSS response during a steamline break is illustrated in Figures 15.1.5-3 through 15.1.5-10. A tabulation of the limiting steamline break event sequence of events is presented in Table 15.1.5-3 for the limiting case.

15.1.5.3.1.1 Secondary system thermal hydraulic parameters

Steam flow out the break is the source of the NSSS cooldown. Break flow rates from both sides of the break are plotted in Figure 15.1.5-3. The break flow from the unaffected steam generators (MSIV side) begins to terminate when the MSIVs start to close approximately 22 seconds after the break was initiated. Steam generator pressures and steam generator masses for the affected and unaffected steam generators are plotted in Figure 15.1.5-4 and 15.1.5-5, respectively. The pressures in these intact steam generators recovered once the MSIVs closed.

The affected steam generator continued to blow down through the break throughout the transient. The pressure and mass flow rate dropped rapidly, after which they reached a quasi-steady state with the affected steam generator boiling off its inventory. The main feedwater flow terminated approximately 40 seconds after the break. The auxiliary feedwater was assumed to continue feeding the affected steam generator at the maximum achievable rate. Termination of auxiliary feedwater would have made the event less severe.

15.1.5.3.1.2 Primary system thermal hydraulic parameters

The primary system pressure and core coolant temperature responses resulting from the break flow are illustrated in Figures 15.1.5-6 through 15.1.5-10. The primary system pressure decays rapidly as the coolant contracts due to cooldown and the pressurizer empties. After the pressurizer empties of liquid, the pressure continues to drop as steam flows from the pressurizer and is condensed in the hot leg. Continued pressure reduction in the primary system due to expansion of steam in the pressurizer ultimately causes the relatively hot stagnant liquid in the head of the reactor pressure vessel to flash. This retards the pressure decay from that point forward in time.

This acts to limit the delivery of boron into the core due to the pressure versus flow characteristics of the SIS.

A comparison of cold leg temperatures indicates significant differences between loops.

15.1.5.3.1.3 Reactivity and core power

The system response for the core is shown in Figure 15.1.5-11.

The reactivity transient calculated by S-RELAP5 is illustrated in Figure 15.1.5-13. Initially, the core is assumed to be at HZP. All control rods, except for the most reactive one, are assumed to be in the core at the initiation of the steamline break and the reactor is initially subcritical by the shutdown margin appropriate for the boron concentration as determined by the COLR.

Cooldown of both the coolant and fuel brings the core critical due to moderator and doppler reactivity feedback. Shortly thereafter power and moderator temperature begin to level out and core reactivity is essentially zero. The HHSI pump reaches full speed at about 25 seconds. The first wave of water with low concentration boric acid passes through the core at about 334 seconds. Insertion of negative reactivity from sustained boron injection from the HHSI system and dryout of the affected steam generator initiates a power descent. Ultimately, the transient was assumed to be terminated by operator intervention at 10 minutes.

The transient experienced by the core power is illustrated in Figure 15.1.5-14. Peak power is calculated to occur at about 48 seconds. The maximum power level is approximately 622 MWt for the limiting case. The S-RELAP5 core power calculation is conservative as demonstrated by a comparison to PRISM.

During a normal cooldown, the safety injection actuation on Low Pressurizer Pressure, High Steam Line Differential Pressure, and High Steam Line Flow with Low Steam Line Pressure or Low Tavg is manually blocked by the Operator upon the receipt of the appropriate permissives to allow the plant to be intentionally cooled down without the initiation of safety injection. This analysis bounds steam line breaks in Mode 3 below the point where safety injection has been manually blocked due to the fact that the RCS is borated sufficiently to ensure that the core remains subcritical during the steam line break (Reference 15.1.5-9).

15.1.5.3.2 Core analysis

The reference reactivity and both the axial and radial power distributions for subcritical HZP conditions were calculated with PRISM. The axial distribution is skewed to the top of the core and is essentially independent of the radial location within the core. As the reactor comes to power without offsite power, the axial power shape becomes more bottom-peaked as a result of the increased reactivity supplied by the moderator cool-down and the heating of the coolant by the core. Because the stuck rod region has the bulk of the power and the coolant heating, the axial power shape for this region is more depressed in the upper portion of the core. The radial peaking in the stuck rod region is reduced with the return to power. When offsite power is present, the axial power shape shifts upwards and the radial peaking in the upper part of the core increases. The input data taken from S-RELAP5 for input to PRISM is listed in Table 15.1.5-4. The radial and axial power distributions computed by PRISM at MDNBR form the basis for the subsequent XCOBRA-IIIC core flow distribution calculation and also for the DNBR calculation.

PRISM calculates a reactivity less than that calculated by S-RELAP5 at the time of MDNBR. Thus, indicating that the S-RELAP5 power calculation is conservative.

15.1.5.3.3 DNBR analysis

The limiting MDNBR conditions are from the HFP with offsite power available case. The inputs are as described below and in the referenced tables.

An XCOBRA-IIIC core analysis was conducted to define the axial and radial flow distribution within the high power assembly. The limiting calculation was based on the core power and boundary conditions from S-RELAP5 and the power distributions from PRISM at the MDNBR point in time for the HFP with offsite power available, 1770 pcm SDM case. Specifically the calculation was based on the data listed in Table 15.1.5-4. The resultant mass flux and enthalpy axial distributions in the hot assembly are shown in Figures 15.1.5-16 and 15.1.5-17.

The Biasi DNB correlation used for the MSLB analysis has a DNB acceptance limit of 1.122 excluding a 2% mixed core penalty. The calculated minimum DNB ratio is 1.751, occurring at 336 seconds. No fuel rods would be expected to fail.

15.1.5.3.4 Fuel centerline melt analysis

The fuel centerline melt criterion was also used to determine the extent of fuel failure. Maximum post-scram LHGR values were determined. The maximum LHGR was calculated as follows:

$$LHGR_{max} = LHGR_{avg} \times F_Q^N \times f_e$$

Where $LHGR_{avg}$ is the LHGR at the time of maximum post-scram core average LHGR (based on the core power) from S-RELAP5, F_Q^N is the nuclear heat flux hot channel factor from PRISM and f_e is the engineering factor uncertainty.

The EOC case remains the limiting case for fuel failure due to the limited margin available to the EOC fuel centerline melt limit.

The margin to the fuel centerline melt LHGR limit, along with the core average LHGR and F_Q^N values, are tabulated in Table 15.1.5-5. The peak LHGR value was below fuel centerline melt LHGR for the limiting EOC case, therefore, no fuel failures were predicted to occur.

15.1.5.4 Radiological Consequences

The NRC has approved implementation of the Alternative Source Term methodology (Reference 15.0.12-3) for analysis of the radiological consequences of this event (Reference 15.1.5-8).
In the event of a main steam line break (MSLB) outside containment, it is assumed that the affected steam generator (SG) will rapidly depressurize and release radionuclides initially contained in the secondary coolant and primary coolant activity transferred via SG tube leaks, directly to the outside atmosphere. The steam line break outside containment will bound any break inside containment, since the outside break provides a means for direct release to the environment.
In fuel cycles where no fuel melt or fuel clad breach is predicted for the MSLB, the released activity is dependent on the maximum primary and secondary coolant activity allowed by the Technical Specifications. Two cases of iodine spiking are evaluated (Reference 15.0.12-3, Appendix E).

(1) For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB. The primary coolant iodine concentration has been raised to the maximum value of 60 µCi/gm DE I-131 permitted by Technical Specifications. Primary coolant is released into the faulted steam generator via a fraction of the total Technical Specifications primary-to-secondary allowable tube leakage limit. Activity is released to the environment from the faulted steam generator via the postulated main steam line break and also via steaming of the postulated tube leak from the faulted SG loop until the leakage into the faulted steam generator is terminated at 98.8 hours. The unaffected steam generators are used to cool down the plant during the MSLB event. Primary-to-secondary tube leakage is also postulated into the unaffected SGs. Activity is released via steaming from the unaffected SG relief valves until the decay heat generated in the reactor core can be removed by the RHR system, 53.2 hours into the MSLB event.

(2) For the case of the accident induced iodine spike, the postulated MSLB event induces an iodine spike. RCS activity is conservatively assumed to be initially at twice the activity presented in Table 15.0.12-2. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. For fuel cycles where fuel breach may occur, a conservative, bounding analysis was performed to assess the impact of postulating the breach of 2 fuel assemblies during a MSLB accident.

Other assumptions used in the radiological consequence analysis:

- Coincident loss of offsite power.
- 157 fuel assemblies in the core.

- Fuel gap inventory fractions:

Krypton 85:	10%
Other Noble Gases:	5%
Iodine 131:	8%
Other Halogens:	5%
Alkali Metals (Cs, Rb):	12%

- Volume of the fluid of the RCS is 8254 ft^3 (minimum volume, used to determine RCS concentration) and 9623 ft^3 (maximum volume, used to determine iodine equilibrium appearance rate) at 575.9°F and 2235 psig.
- RCS activity conservatively remains constant throughout the Pre-Accident Iodine Spike MSLB and the 2 failed fuel assembly events (no dilution of the RCS activity from the safety injection system is considered).
- RCS mass remains constant throughout the MSLB event (no change in the RCS mass as a result of the MSLB or from the safety injection system).
- Data used to calculate the iodine equilibrium appearance rate:

Maximum Nominal Letdown Flow:	120 gpm @ 130 °F, 2235 psig
Uncertainty Applied to Letdown Flow:	10%
Maximum Identified RCS Leakage:	10 gpm
Maximum Unidentified RCS Leakage:	1 gpm

- Maximum radial peaking factor is 1.8.
- The primary-to-secondary leak rate in the steam generators is based on the leak-rate-limiting condition for operation specified in the Technical Specifications of 75 gpd increased by a factor of 2 (150 gpd, which is 0.104 gpm). The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized. The operational primary-to-secondary leakage is conservatively assumed to be 0.11 gpm through any one SG and 0.3 gpm total to all three SGs. Since the tube leak into the faulted SG and subsequently to the environment continues until the RCS temperature drops below 212 °F at 98.8 hours, it is conservative to assign the maximum allowed 0.11 gpm to the faulted SG with the remainder of 0.19 gpm assigned to the unaffected SGs.

- The mass of the fluid of the SGs secondary side is 88,461 lbm/SG (minimum mass).
- SG volume in the unaffected SGs is assumed to remain constant throughout and dilution by incoming Auxiliary Feedwater is not considered.
- Noble gas radionuclides released from the primary to the secondary system are immediately released to the environment without holdup, reduction, or mitigation.
- For the faulted SG, primary leakage immediately flashes to vapor and is immediately released to the environment with no mitigation (no scrubbing).
- For the unaffected SGs, all leakage that does not immediately flash mixes with the bulk water. The radioactivity within the bulk water is assumed to become a vapor at a rate that is the function of the steaming rate and the partition coefficient. The partition coefficient of 100 is utilized for iodine and the alkali metals. For the HBR-2 MSLB event, steam generator dryout is not postulated for the unaffected SGs and thus none of the primary-to-secondary leakage is postulated to immediately flash to steam.
- The integrated mass of the steam released during the MSLB event, based on 3.7°F/hour cooldown rate (within time periods, flow rate is assumed to be constant, mass and associated activity release is assumed to be linear):

Time	Integrated Steam Release from Ruptured SG (lbm)(2)	Integrated Steam Release from Unaffected SGs (lbm)
0 hours	161,194(1)	0
0 – 2.0 hours	161,304.2	300,116.1
0 – 8 hours	161,634.7	861,350.9
0 – 24 hours	162,516.0	1,971,677.3
0 – 53.2 hours	164,124.3	3,582,768.8
0 – 98.8 hours	166,636.1	3,582,768.8

(1) Includes 23,900 lbm of feedwater flow prior to isolation plus the 137,294 lbm of SG steam release. Feedwater activity is conservatively considered to be the same as main steam activity. Auxiliary Feedwater mass is not included, since little activity is expected in the Auxiliary Feedwater system.

(2) Includes SG tube leakage.

- Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3 % organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

15.1.5.5 Conclusions

In a conservative estimation of the consequences, the shutdown margin is lost and the core returns to power. When the auxiliary feedwater flow is terminated, heatup of the primary with resulting negative moderator and doppler feedback effects will augment the negative reactivity inserted from the boron to terminate the power excursion. Evaluation of the peak fuel Linear Heat Generation Rate shows that fuel centerline melting does not occur and core subchannel calculations show that DNB does not occur. The limiting cases are the Hot Zero Power case for Fuel Centerline Melting and Hot Full Power for MDNBR.

For the MSLB with a pre-accident iodine spike, the 2-hour dose at the EAB is 0.26 rem TEDE. The dose at the LPZ is 0.03 rem TEDE. The Control Room doses at inleakages of 300 and 500 cfm are 0.14 and 0.21 rem TEDE, respectively.

For the MSLB with an accident induced iodine spike, the 2-hour dose at the EAB is 0.75 rem TEDE. The dose at the LPZ is 0.10 rem TEDE. The Control Room doses at inleakages of 300 and 500 cfm are 0.45 and 0.73 rem TEDE, respectively.

For the MSLB with a breach of two fuel assemblies, the 2-hour dose at the EAB is 2.93 rem TEDE. The dose at the LPZ is 0.42 rem TEDE. The Control Room doses at inleakages of 300 and 500 cfm are 1.61 and 2.71 rem TEDE, respectively.

The offsite dose acceptance criterion established by Reference 15.0.12-3 for the pre-accident iodine spike or fuel damage is that doses should be less than the 10 CFR 50.67 guideline of 25 rem TEDE. The offsite dose acceptance criterion established by Reference 15.0.12-3 for the accident induced iodine spike is that doses should be less than 10% of the 10 CFR 50.67 guideline, or less than 2.5 rem TEDE. The Control Room dose acceptance criterion established by 10 CFR 50.67 for the MSLB is 5 rem TEDE.

Therefore, the offsite and Control Room TEDE doses due to a MSLB event meet the dose acceptance criteria.

TABLE 15.1.5-1

S-RELAP5 THERMAL-HYDRAULIC INPUT

Initial Condition Thermal-Hydraulic Input	
Total Core Power	HFP - 2,339 MWt HZP - 1 Watt
Primary Pressure	2250 psia
Primary Temperature	HFP – 575.9°F HZP – 546.7°F
Primary Flow Rate	97.3×10^6 lbm/hr
Pressurizer Level	HFP - 53.3% of Span HZP - 22.2% of Span
Secondary Pressure	HFP - 821 psia HZP - 1025 psia
Steam Flow Rate	HFP - 10.3×10^6 lbm/hr HZP – 41,000 lbm/hr
Feedwater Flow Rate	HFP - 10.3×10^6 lbm/hr HZP – 41,000 lbm/hr
Secondary Mass	HFP – 265,923 lbm HZP – 403,500 lbm
Break Characteristics	
Minimum Flow Area	
Affected Steam Generator	1.388 ft^2
Combined Intact Steam Generators	2 x 1.388 ft^2
Location	In steamline downstream of flow meter
SIS System	
Total Pumps	2*
Single Failure	1 of 2 HHSI pumps
BIT Boron Concentration	0 ppm
Refueling Water Storage Tank Concentration	1,950 ppm
Refueling Water Storage Tank Temperature	45ºF
BIT Volume	120 ft^3
BIT to Cold Leg Injection Total Volume	20 ft^3
HHSI Pumps to BIT	30 ft^3

* The "B" safety injection pump serves as a maintenance replacement for the "A" and "C" safety injection pumps.

TABLE 15.1.5-1 (Continued)

Feedwater	
Auxiliary	
Flow	1325 gpm
Temperature	33°F
Main	
Flow	See Figure 15.1.5-2
Temperature	33°F

TABLE 15.1.5-2

ACTUATION SIGNALS AND DELAYS FOR MSIV,
SIS AND FEEDWATER SAFETY ACTIONS

PARAMETER SETPOINTS		Setpoint	Uncertainty	Analysis Value
1.	High Steam Line Flow - HFP	(2)	(3)	4,013,280 lbm/hr
	- HZP	(2)	(3)	2,263,680 lbm/hr
2.	Low Steam Line Pressure	614 psig	+24.22/-23.80 psi	572.7 psia
3.	Low Pressurizer Pressure	1715 psig	+40.00 psi	1684.7 psia

MSIV CLOSURE

Required Actuation Signal

A. (1) in two of three lines coincident with
(2) in two of three lines.

Delay - 10 seconds

SIS ACTUATION

Required Actuation Signal

A. (1) in two of three lines coincident with
(2) in two of three lines

B. (3)

Delay - 15 seconds with offsite power available, 35.5 seconds for a loss of offsite power

MAIN FEEDWATER VALVE CLOSURE

Required Actuation Signal

A. Any SIS actuation signal

Delay - 30 seconds[1]

---

(1) From Reference 15.1.5-6, containment analysis shows acceptable results with this response time extended to 82 seconds. This will allow credit for feedwater block valve closure, in case of feedwater regulating valve's failure to close.
(2) 37.25% of Full Power Steam Flow at ≤ 20% RTP - ramping linearly to 109% of Full Power Steam Flow at 100% RTP.
(3) 6.15% of Full Power Steam Flow at < 20% RTP; 7.05% of Full Power Steam Flow at 20% RTP; 6.7% of Full Power Steam Flow at 100% RTP.

TABLE 15.1.5-3
LIMITING MAIN STEAMLINE BREAK EVENT SUMMARY
(HZP, Offsite Power Available, 1770 pcm SDM)

Time (s)	Event
0	Control rods inserted and break initiates
9.69	Low pressurizer pressure ESFAS signal initiates SI and MFW isolation
12.06	High 2 of 3 steam line flow with low 2 of 3 steam line pressure initiates MSIV closure
16	Reactor returns to critical
22.07	MSIVs begin to close
24.68	HHSI available and flow begins
39.68	MFW isolation valves closed
48	Maximum post-scram power reached
334	Boron begins to affect core reactivity
600	Calculation ends (AFW manually isolated)

TABLE 15.1.5-3b

Table Deleted

TABLE 15.1.5-4

PRISM INPUT AND S-RELAP5 OUTPUT AT LIMITING POINT
(HFP with offsite power available, 1770 pcm SDM)

Time of MDNBR	336 seconds
Core Parameters	
Power	538.88 MW_t
Exit Pressure	1129.1 psia
Total Inlet Flow	28483 lbm/sec
Unaffected Core Sector	
Inlet Temperature	459.13°F
Total Inlet Flow	18645 lbm/sec
Affected Core Sector	
Inlet Temperature	397.91°F
Inlet Flow	6058.9 lbm/sec
Stuck Rod Sector	
Inlet Temperature	397.91°F
Inlet Flow	3779.3 lbm/sec

Note: Table 15.1.5-4 contains the data used to calculate MDNBR of 1.751.

TABLE 15.1.5-5

LIMITING PEAK LHGR AND FUEL FAILURE RESULTS
(HZP, Pumps On, 1770 pcm SDM)

Time of Peak LHGR	48 seconds
Core Maximum LHGR	18.49 kW/ft
F_Q^N	11.10
Margin to fuel centerline melt LHGR limit	16.7%

TABLE 15.1.5-6

EQUILIBRIUM IODINE CONCENTRATIONS PRIOR TO ACCIDENT

Equilibrium iodine concentrations prior to the accident with a leak rate of 0.1 gpm to steam generators and primary activity associated with 1 percent fuel defects.

Nuclide	Cp	Cw	Cs	Cb
I-131	1.49×10^{0}	1.28×10^{-2}	4.70×10^{-5}	2.32×10^{-2}
I-132	5.53×10^{-1}	2.35×10^{-4}	8.60×10^{-7}	2.72×10^{-4}
I-133	2.41×10^{0}	6.95×10^{-3}	2.55×10^{-5}	1.20×10^{-2}
I-134	3.49×10^{-1}	6.15×10^{-5}	2.25×19^{-7}	4.50×10^{-5}
I-135	1.29×10^{0}	1.50×10^{-3}	5.50×10^{-6}	2.25×10^{-3}

Cp = equilibrium iodine concentrations in the primary coolant

Cw = equilibrium iodine concentrations in the steam generator water

Cs = equilibrium iodine concentrations in the main steam

Cb = equilibrium iodine concentrations in the blowdown tank water

Equilibrium iodine concentrations prior to the accident with a leak rate of 3 gpm in steam generators and primary activity associated with 1 percent fuel defects.

Nuclide	Cp	Cw	Cs	Cb
I-131	1.43×10^{0}	3.66×10^{-1}	1.36×10^{-3}	6.72×10^{-1}
I-132	5.50×10^{-1}	6.96×10^{-3}	2.58×10^{-5}	8.08×10^{-3}
I-133	2.34×10^{0}	2.04×10^{-1}	7.55×10^{-4}	3.54×10^{-1}
I-134	3.47×10^{-1}	1.80×10^{-3}	6.67×10^{-6}	1.34×10^{-3}
I-135	1.28×10^{0}	4.36×10^{-2}	1.62×10^{-4}	6.70×10^{-2}

REFERENCES: SECTION 15.1

15.1.3-1 Deleted by Revision No. 14.

15.1.3-2 Deleted by Revision No. 14.

15.1.3-3 Deleted by Revision No. 14.

15.1.3-4 Deleted by Revision No. 14.

15.1.3-5 Deleted by Revision No. 14.

15.1.5-1 ANP-2560, "Robinson Nuclear Plant Chapter 15 Safety Analysis Supporting MTC Change," AREVA NP Inc., December 2006.

15.1.5-2 EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., May 2004.

15.1.5-3 Deleted by Revision No. 17.

15.1.5-4 Deleted by Revision No. 13.

15.1.5-5 Deleted by Revision No. 13.

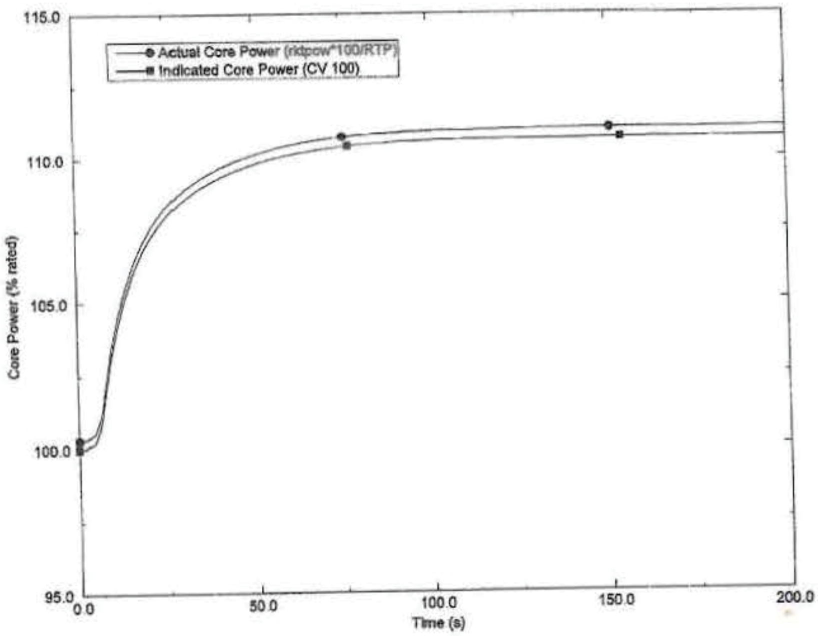
15.1.5-6 Deleted by Revision No. 20.

15.1.5-7 Deleted by Revision No. 14.

15.1.5-8 NRC Letter dated September 24, 2004, “H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105).”

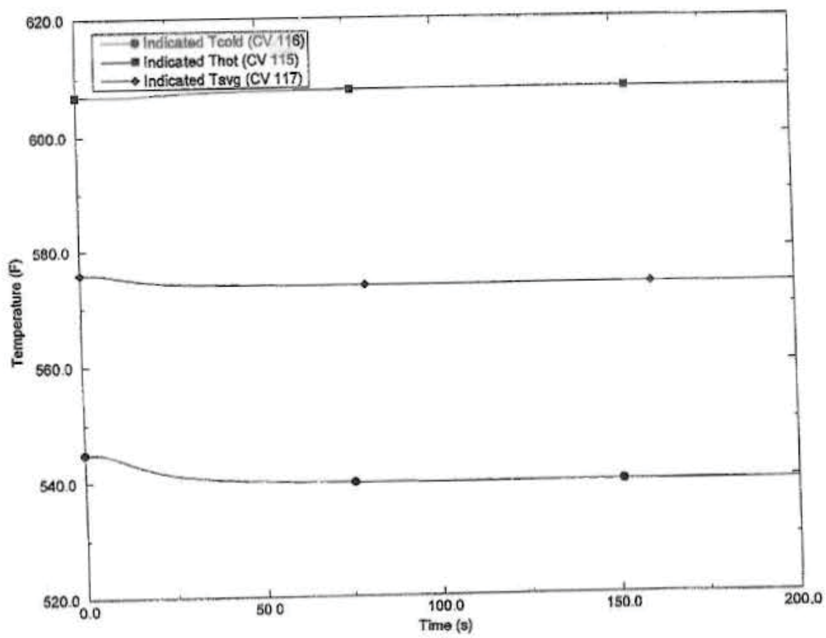
15.1.5-9 NSAL-02-14, “Steam Line Break During Mode 3,” Westinghouse, October 2002.

15.1.5-10 ANP-3526, "Robinson Nuclear Plant Cycle 31 Safety Analysis Report," AREVA, Inc., December 2016.



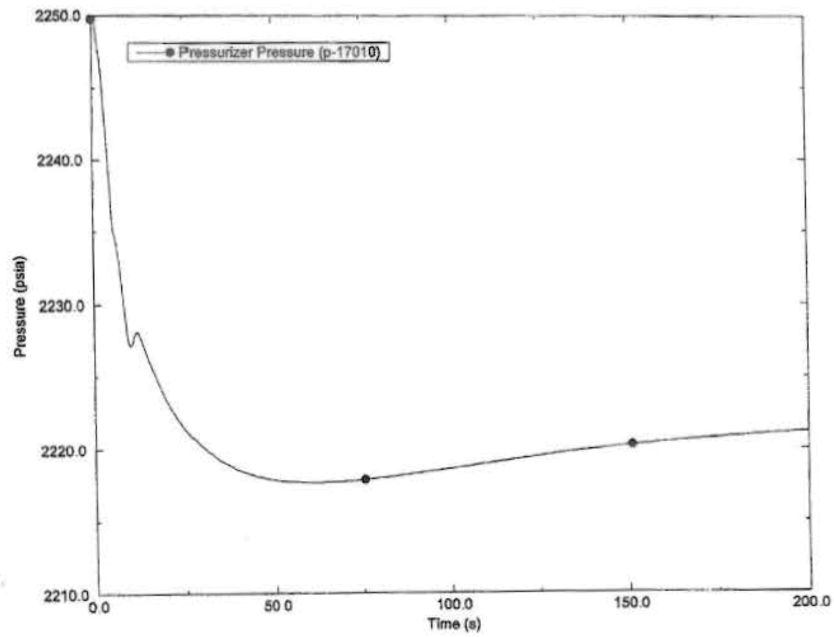
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase In Steam Flow: Core Power	FIGURE 15.1.3-1



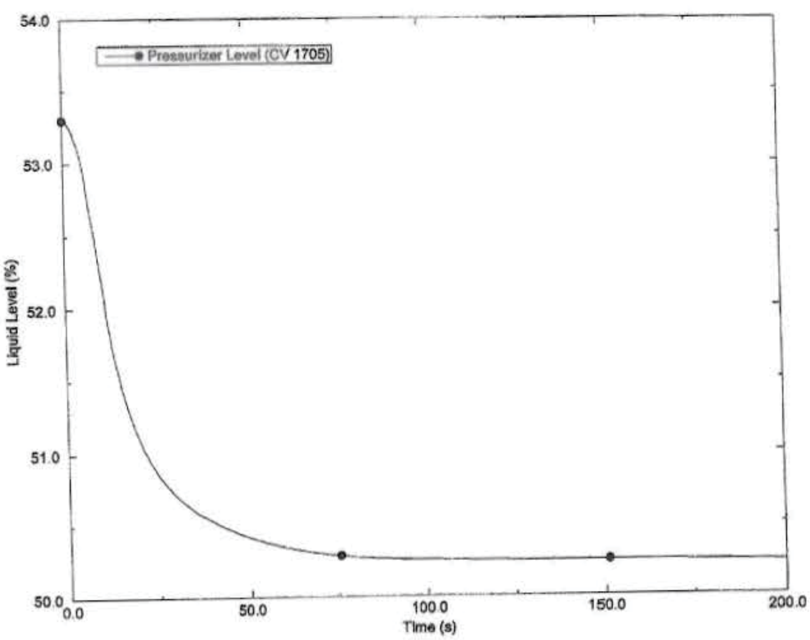
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase in Steam Flow: Reactor Coolant Temperatures	FIGURE 15.1.3-2



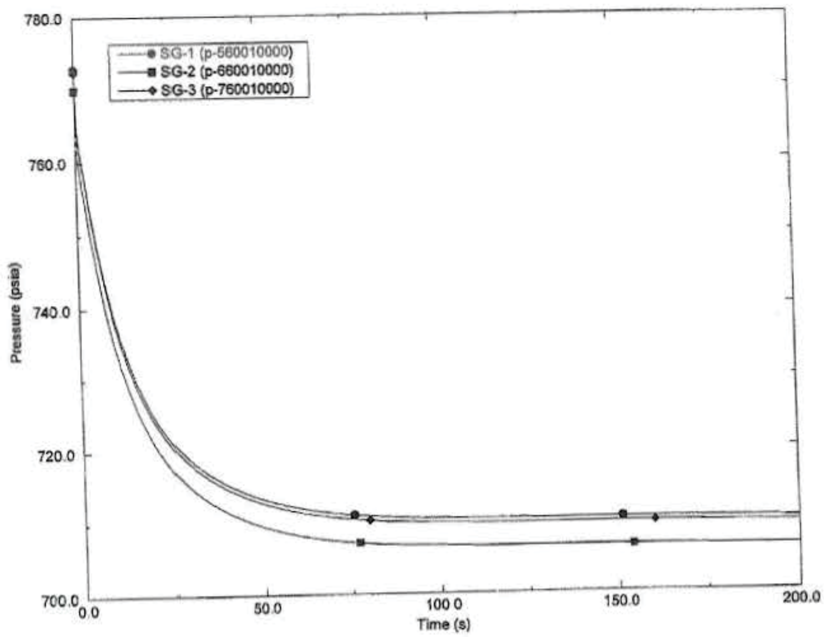
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase in Steam Flow: Pressurizer Pressure	FIGURE 15.1.3–3



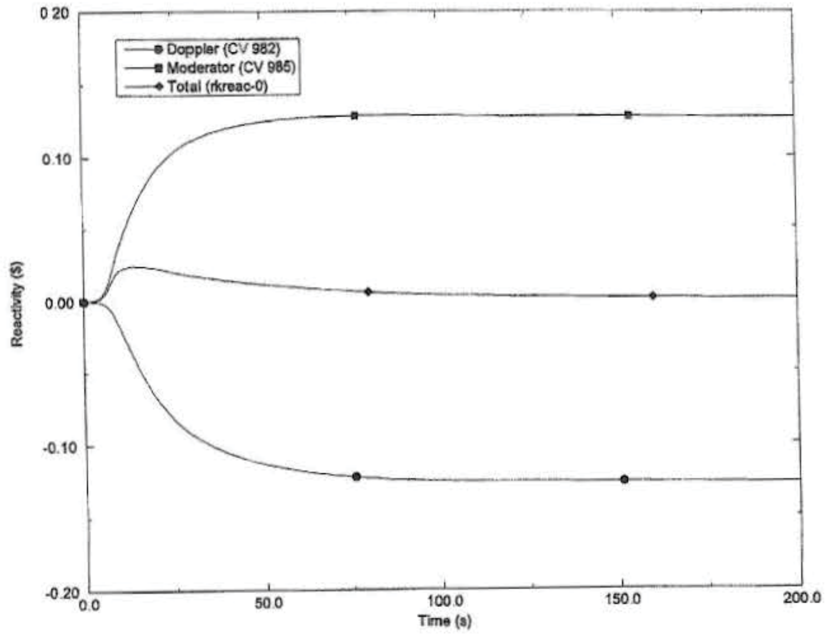
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase in Steam Flow: Pressurizer Liquid Level	FIGURE 15.1.3-4



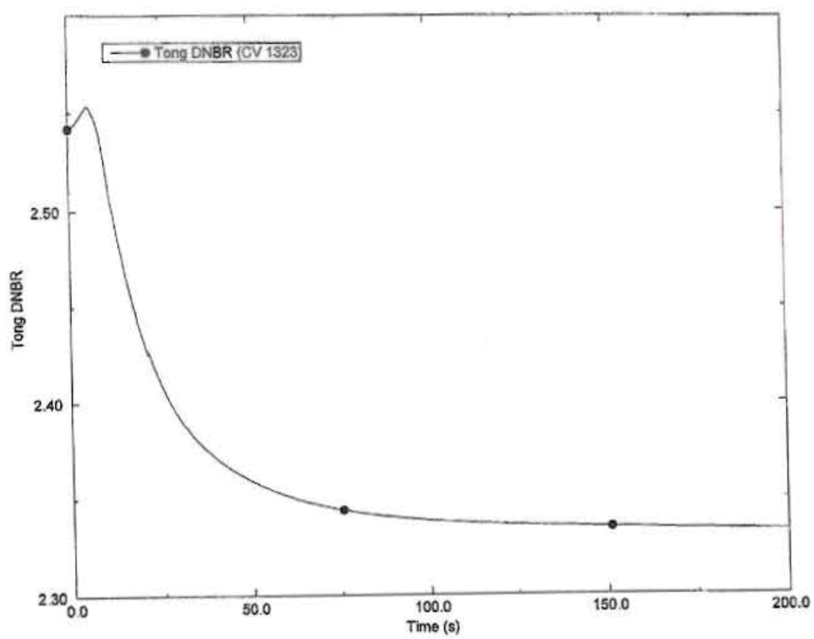
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase in Steam Flow: Steam Line Pressures	FIGURE 15.1.3–5



REVISION NO. 22

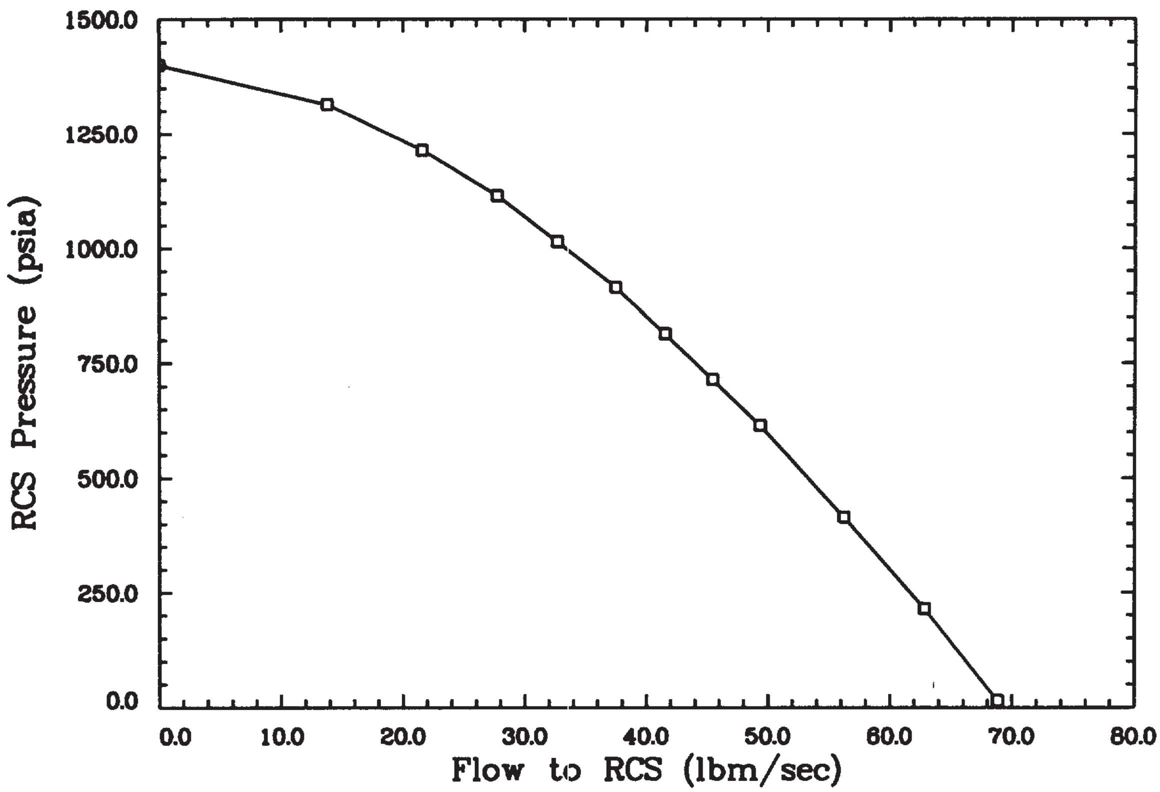
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase in Steam Flow: Reactivity Feedback	FIGURE 15.1.3–6



Note: Tong DNBR value is a RELAP calculated core-wide DNBR trend used to estimate approximate time of minimum DNBR to establish core parameters for input into the XCOBRA-IIIC code to evaluate MDNBR.

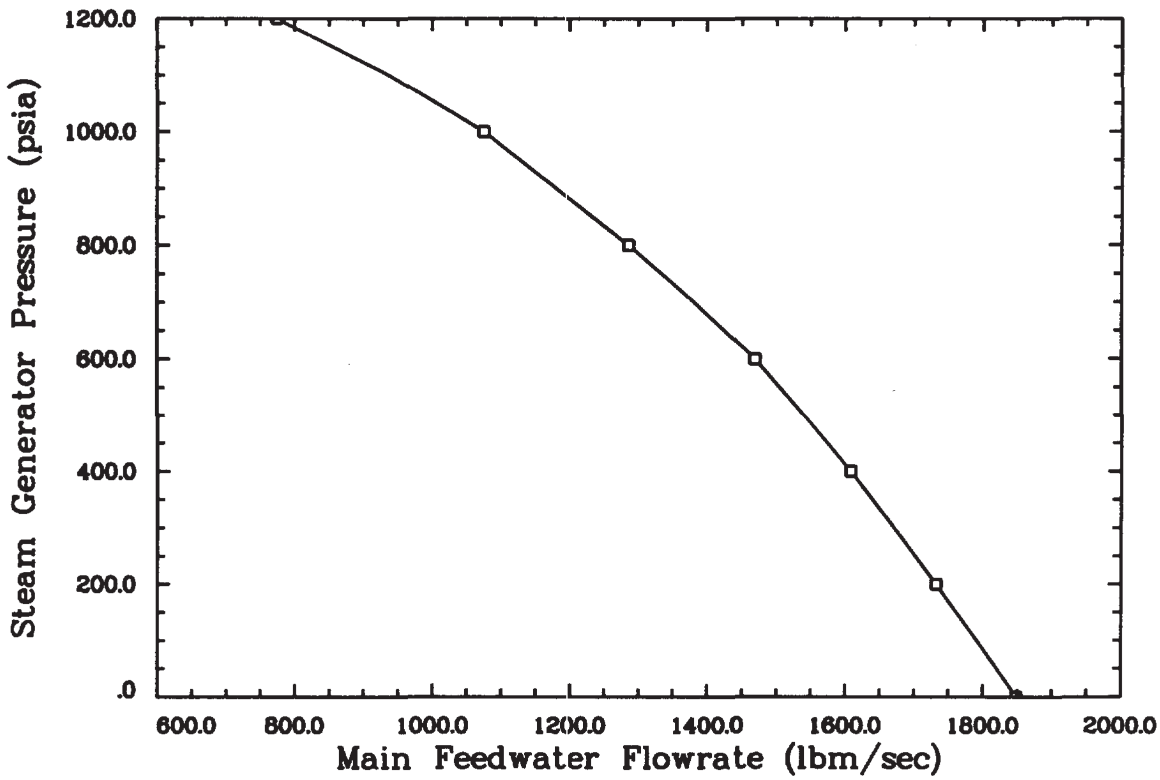
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Increase In Steam Flow: DNBR Trend	FIGURE 15.1.3-7



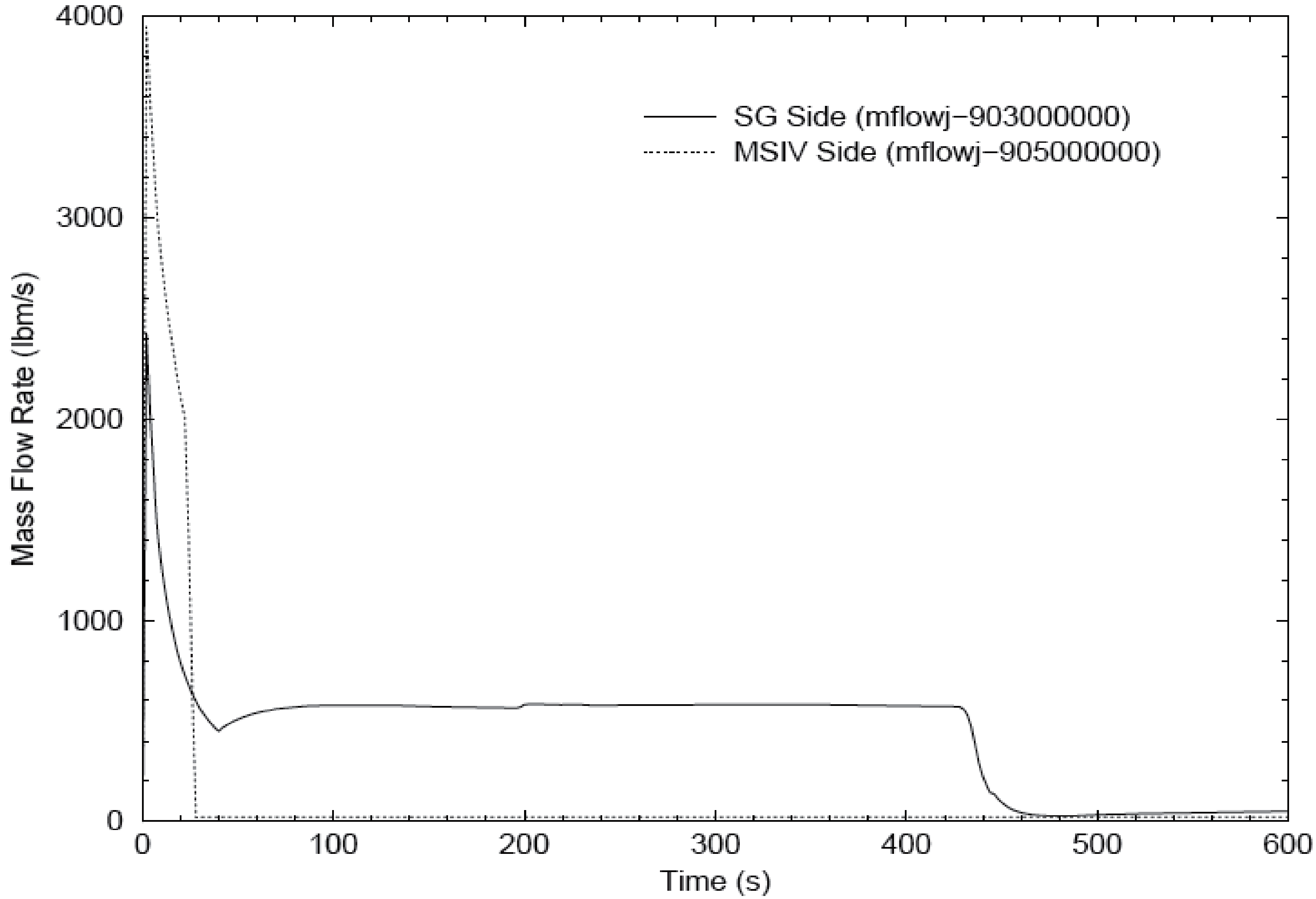
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	High Head Safety Injection Delivery Curve Based on One Pump with No Spills	FIGURE 15.1.5-1



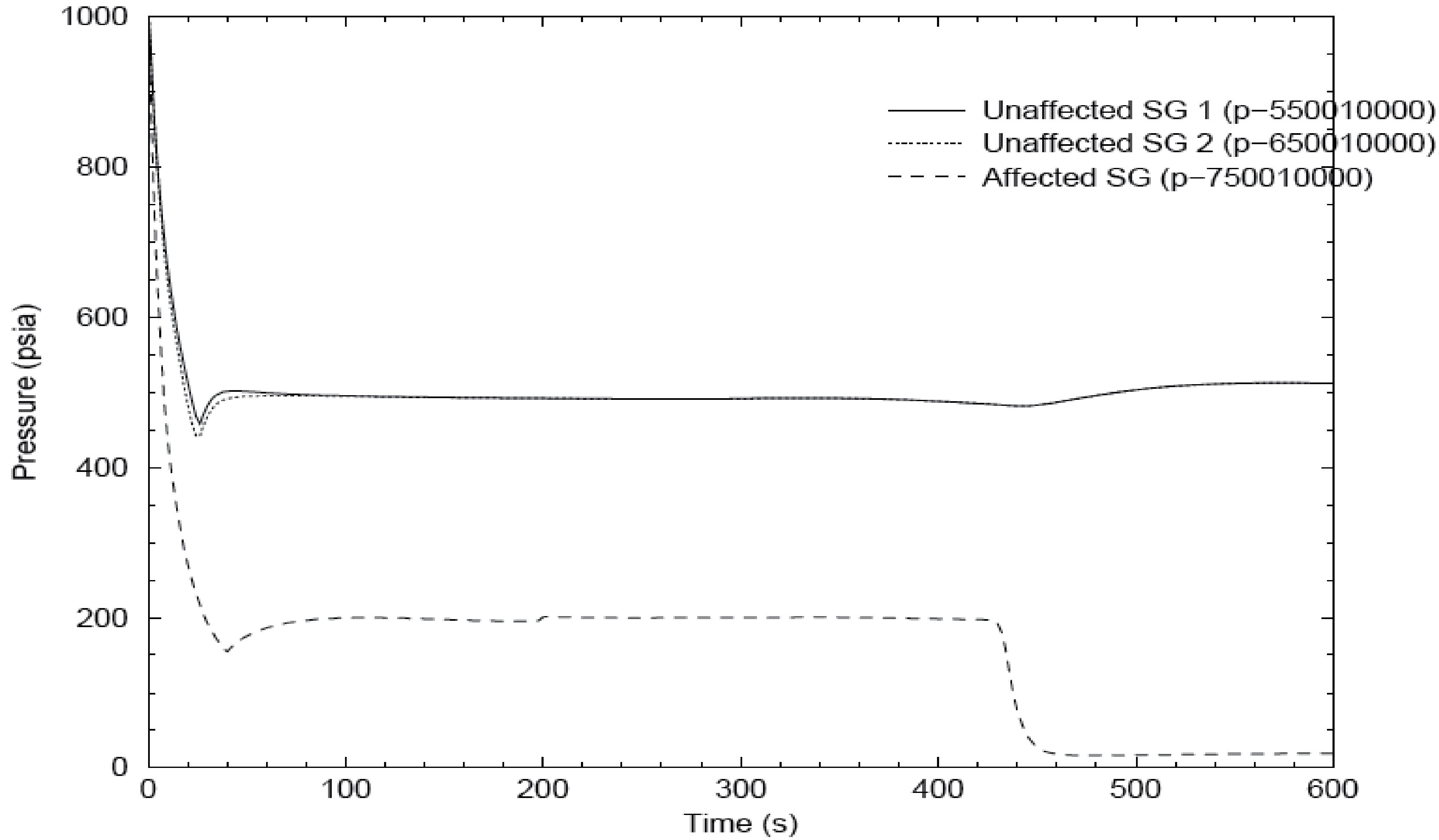
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Main Feedwater Delivery to Affected Steam Generator from Single Feedwater Train	FIGURE 15.1.5-2



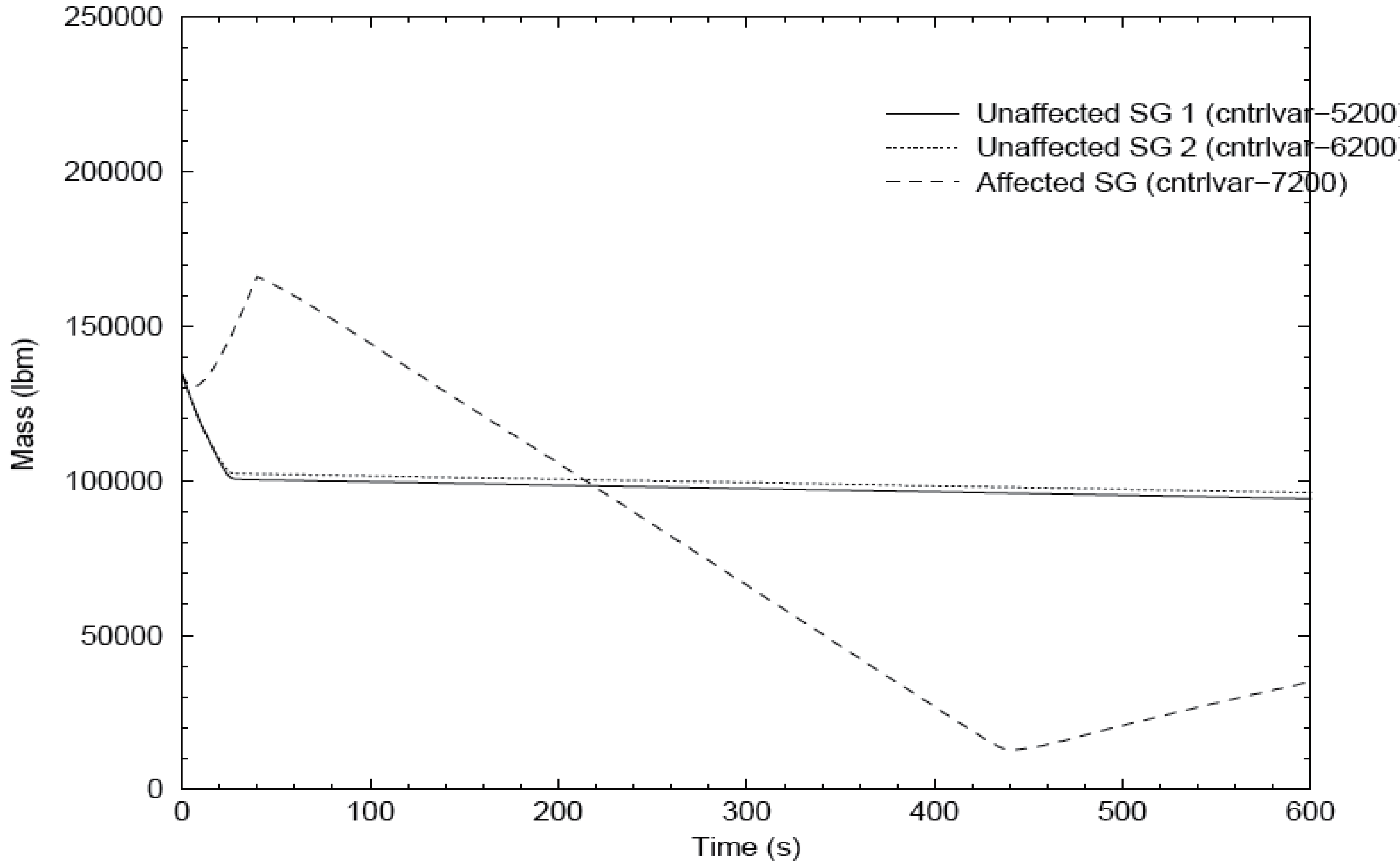
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Steamline Break Flow Rates for Limiting MSLB	FIGURE 15.1.5-3



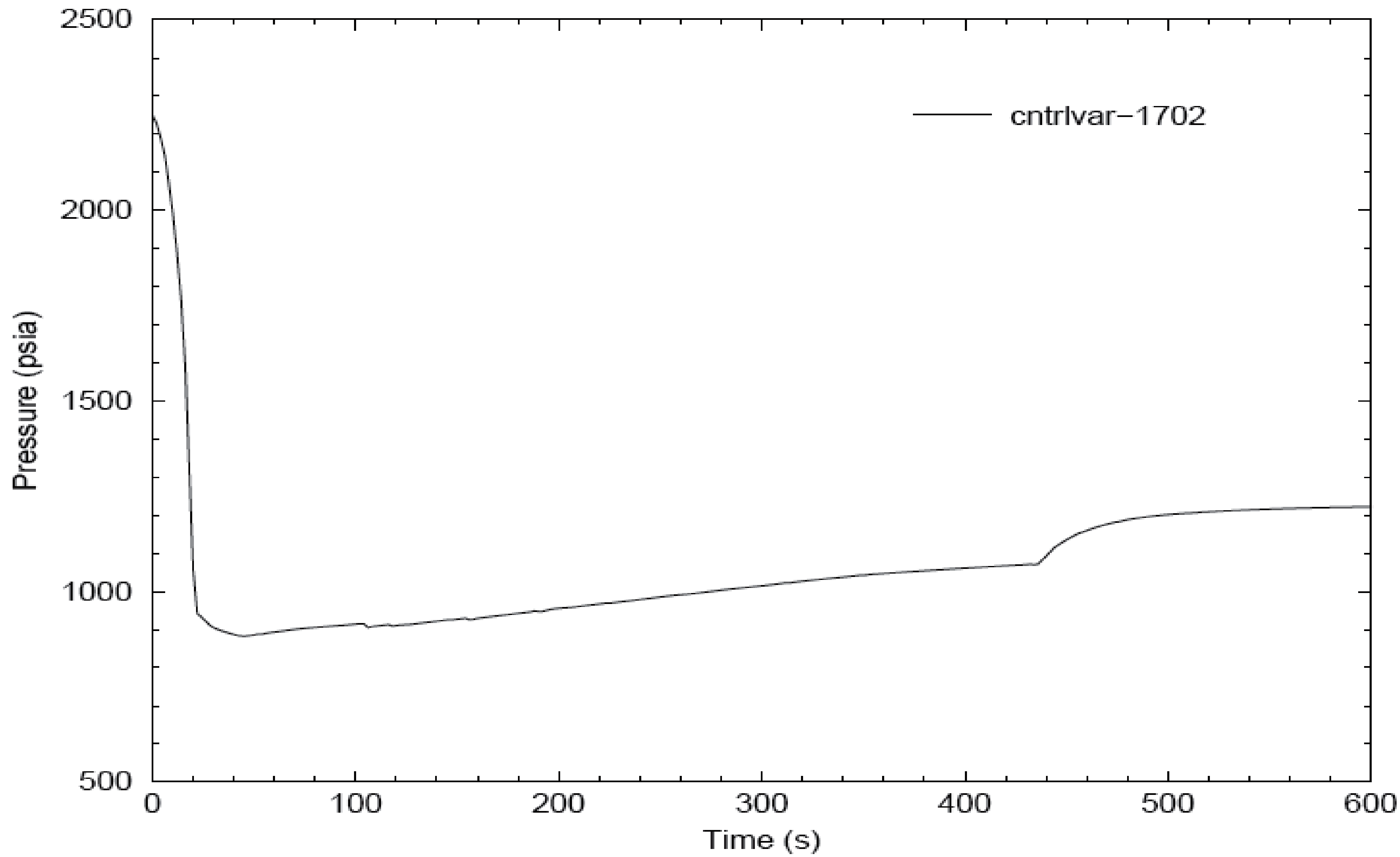
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Pressure for Limiting MSLB	FIGURE 15.1.5-4



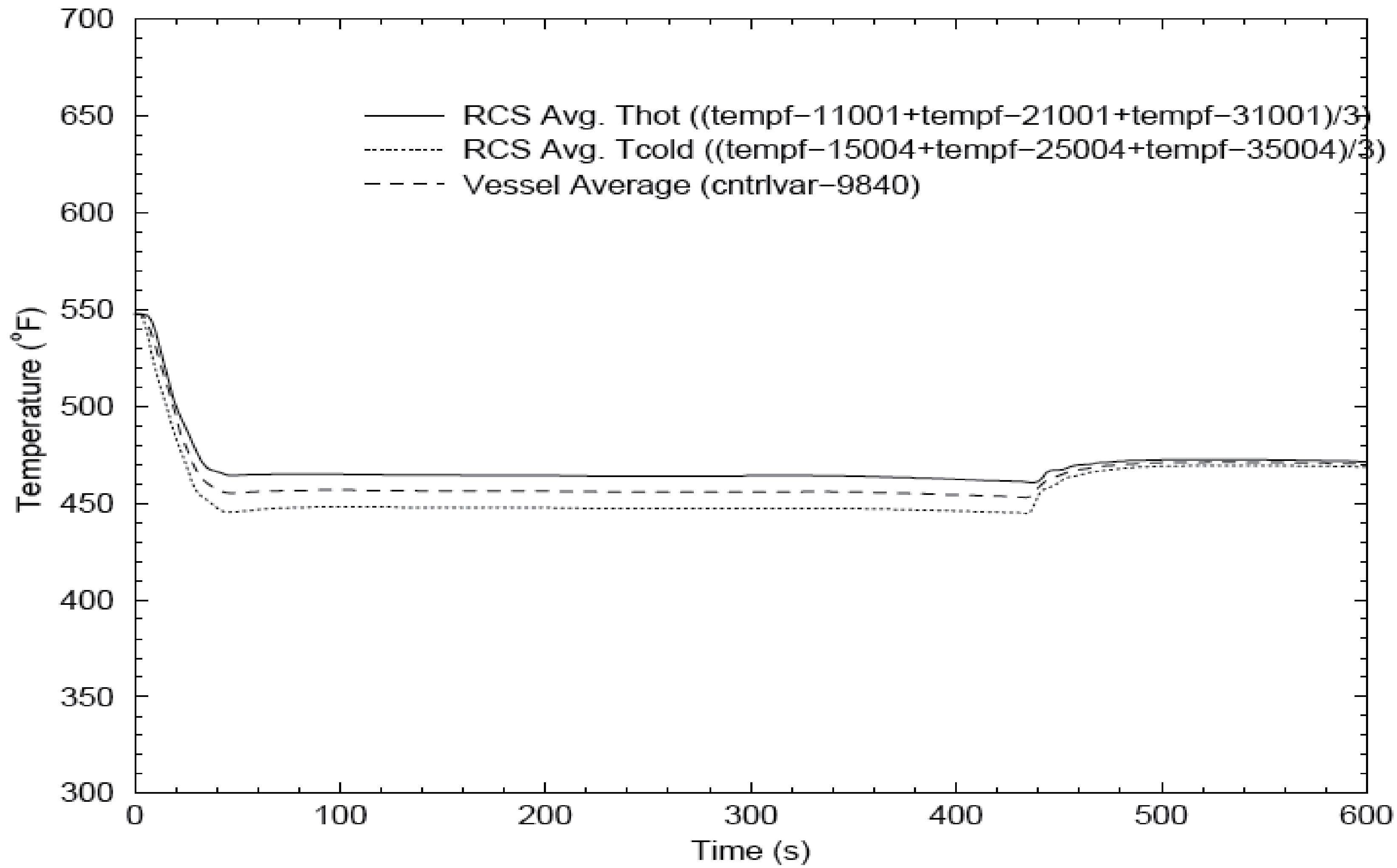
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Mass for the Limiting MSLB	FIGURE 15.1.5-5



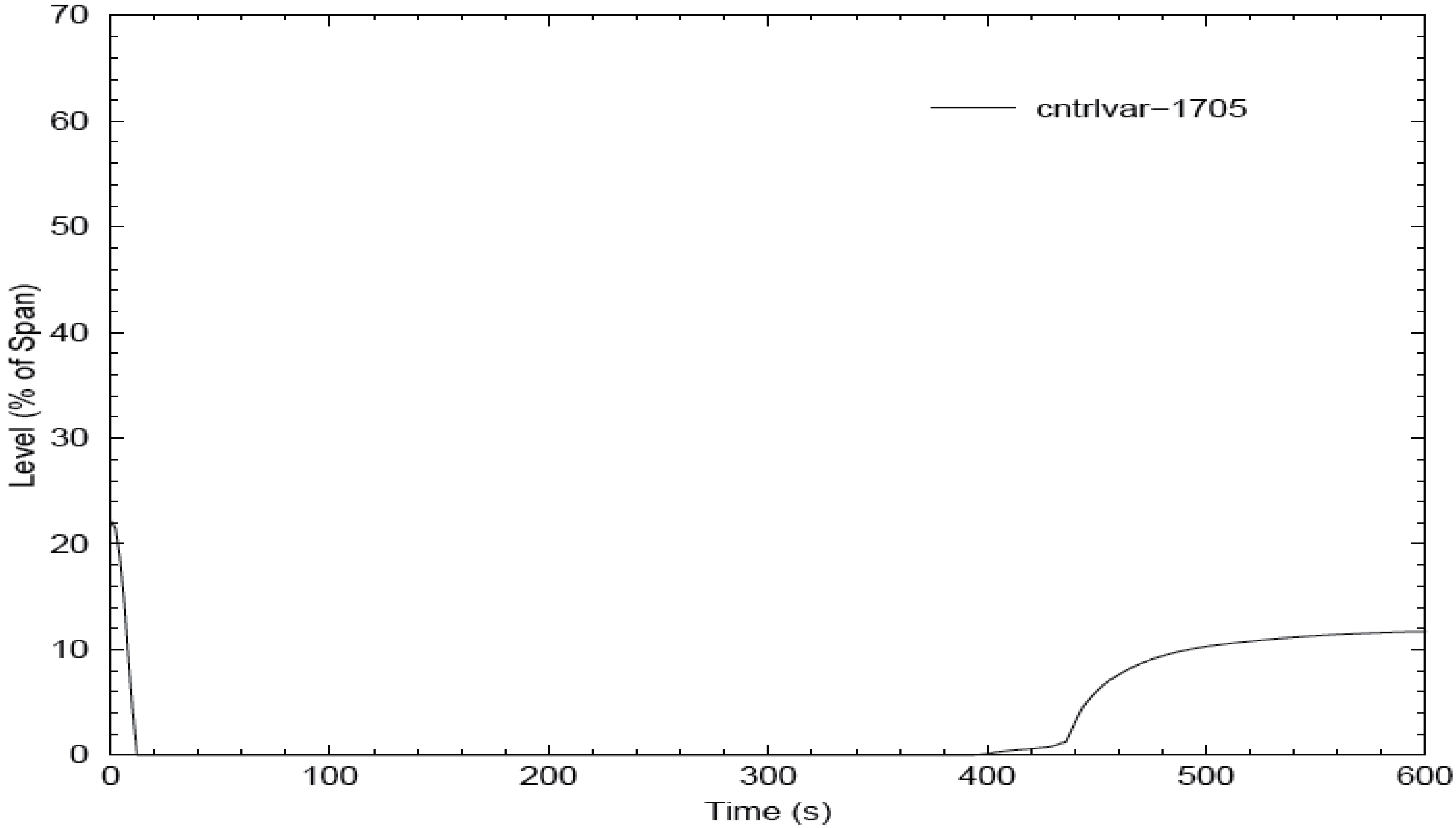
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure for Limiting MSLB	FIGURE 15.1.5-6



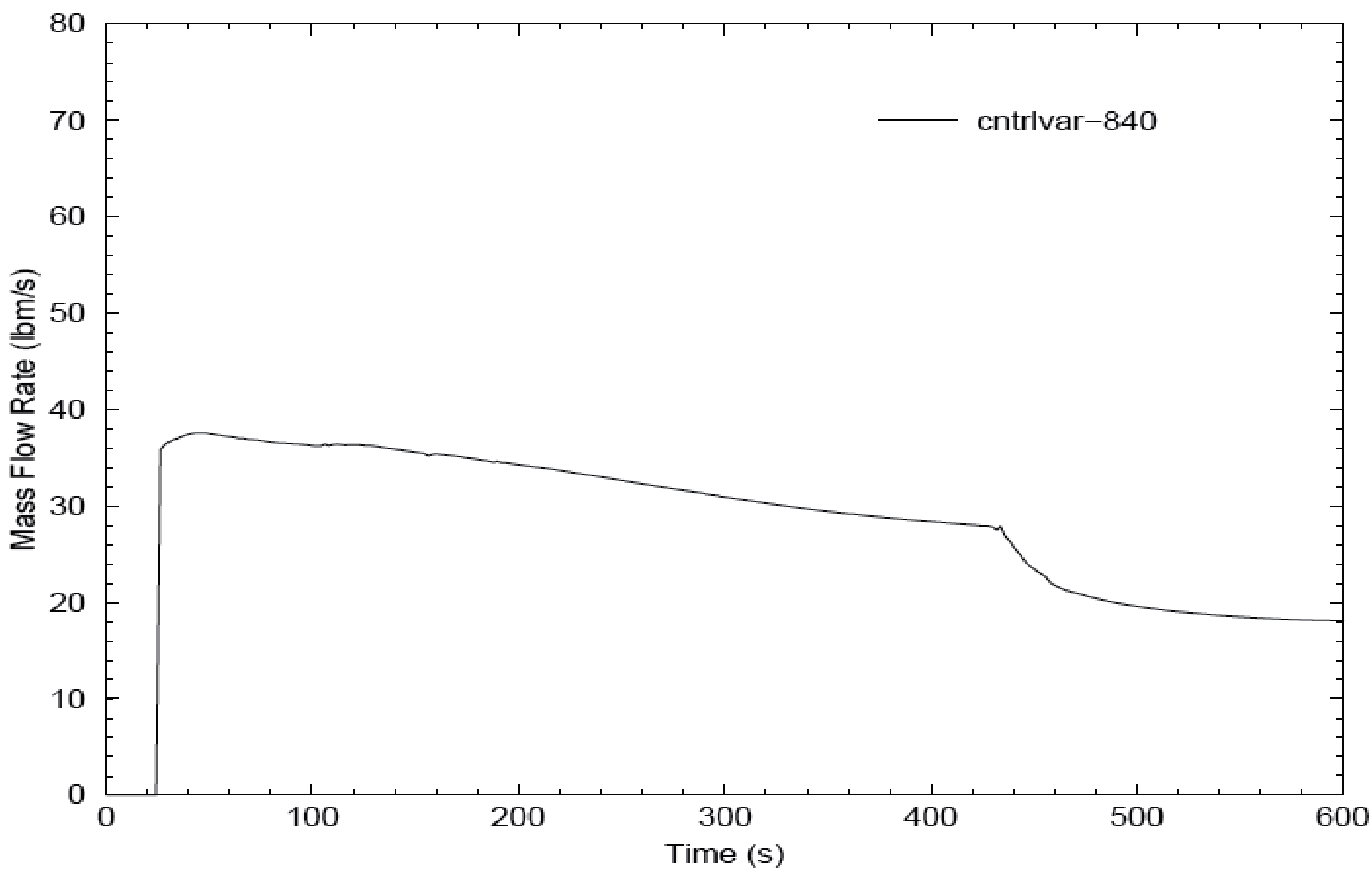
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Average Primary System Temperature for Limiting MSLB	FIGURE 15.1.5-7



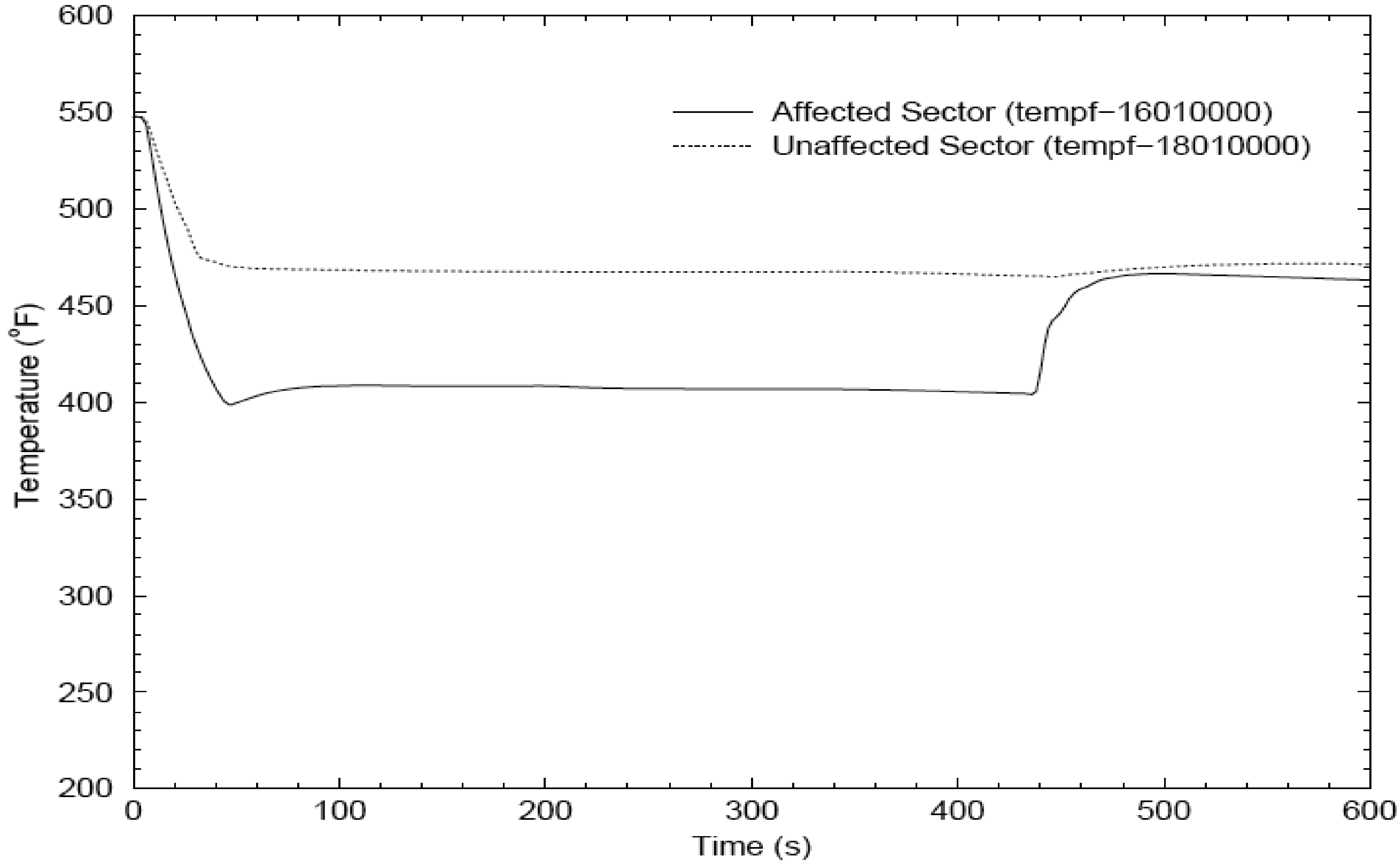
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Level for Limiting MSLB	FIGURE 15.1.5-8



REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Total HHSI Flow Rate for Limiting MSLB	FIGURE 15.1.5-9



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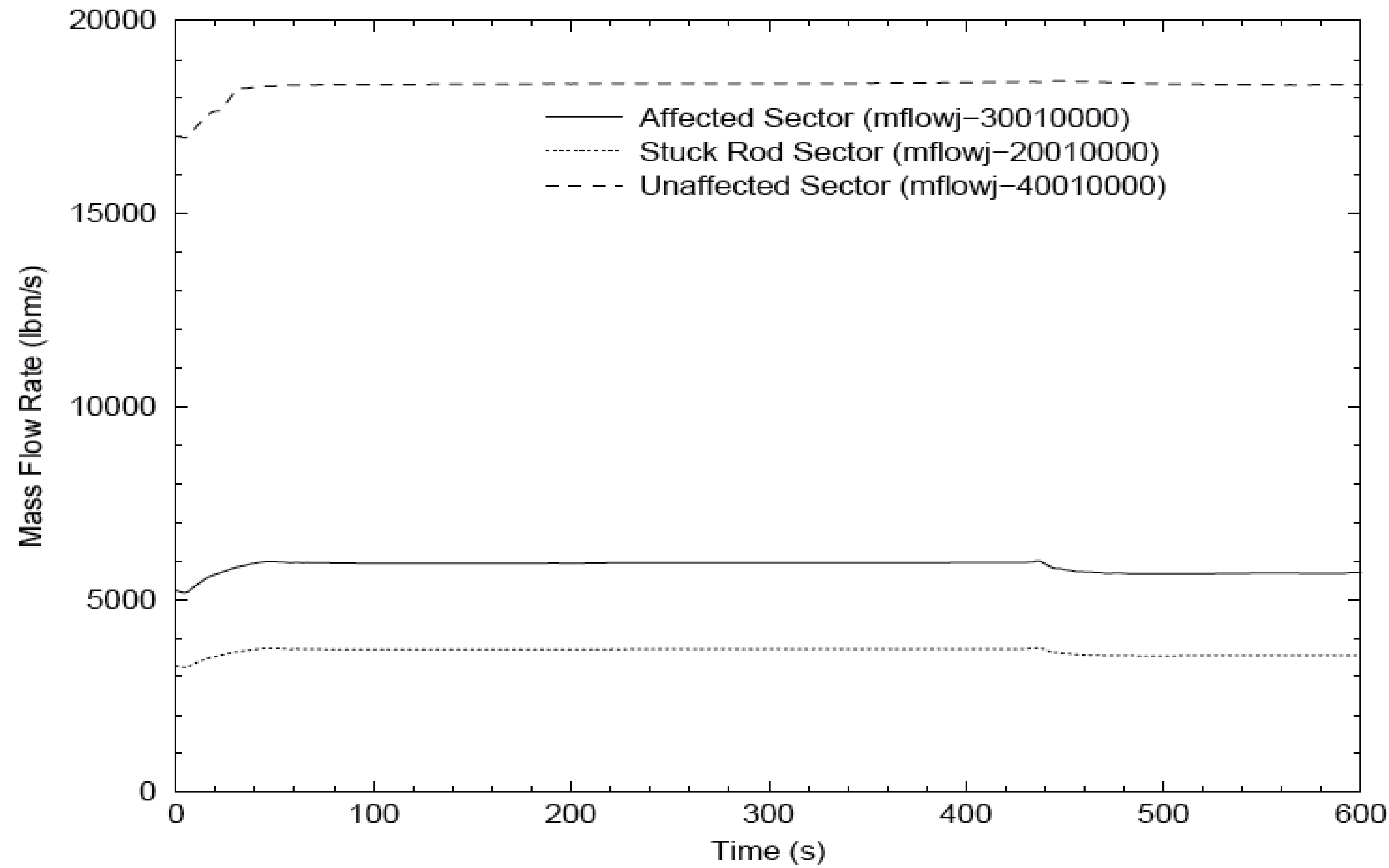
H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Core Inlet Temperature for Limiting MSLB	FIGURE 15.1.5-10



REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Core Inlet Flow Rates for Limiting MSLB	FIGURE 15.1.5-11

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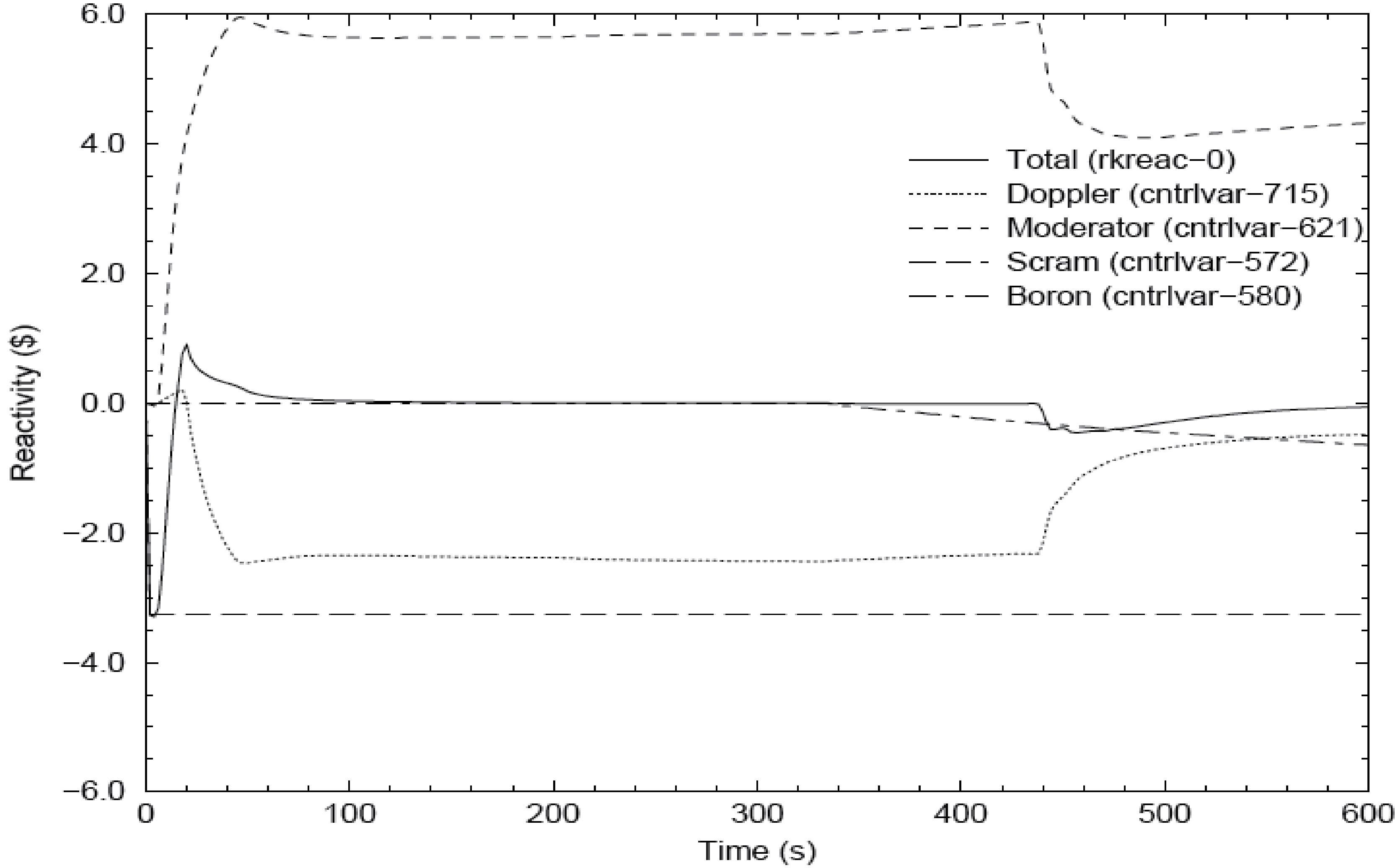
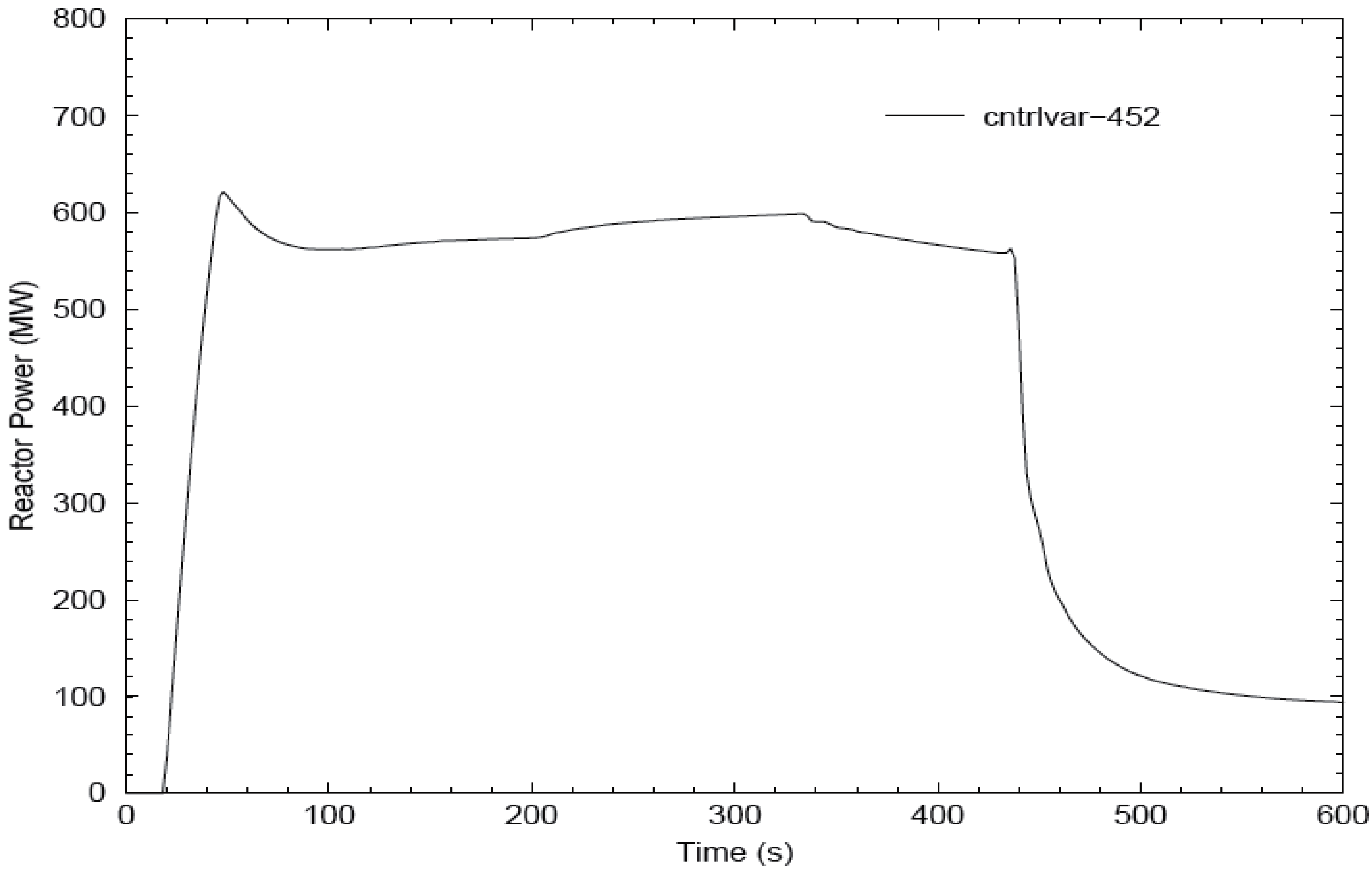
FIGURE 15.1.5-12



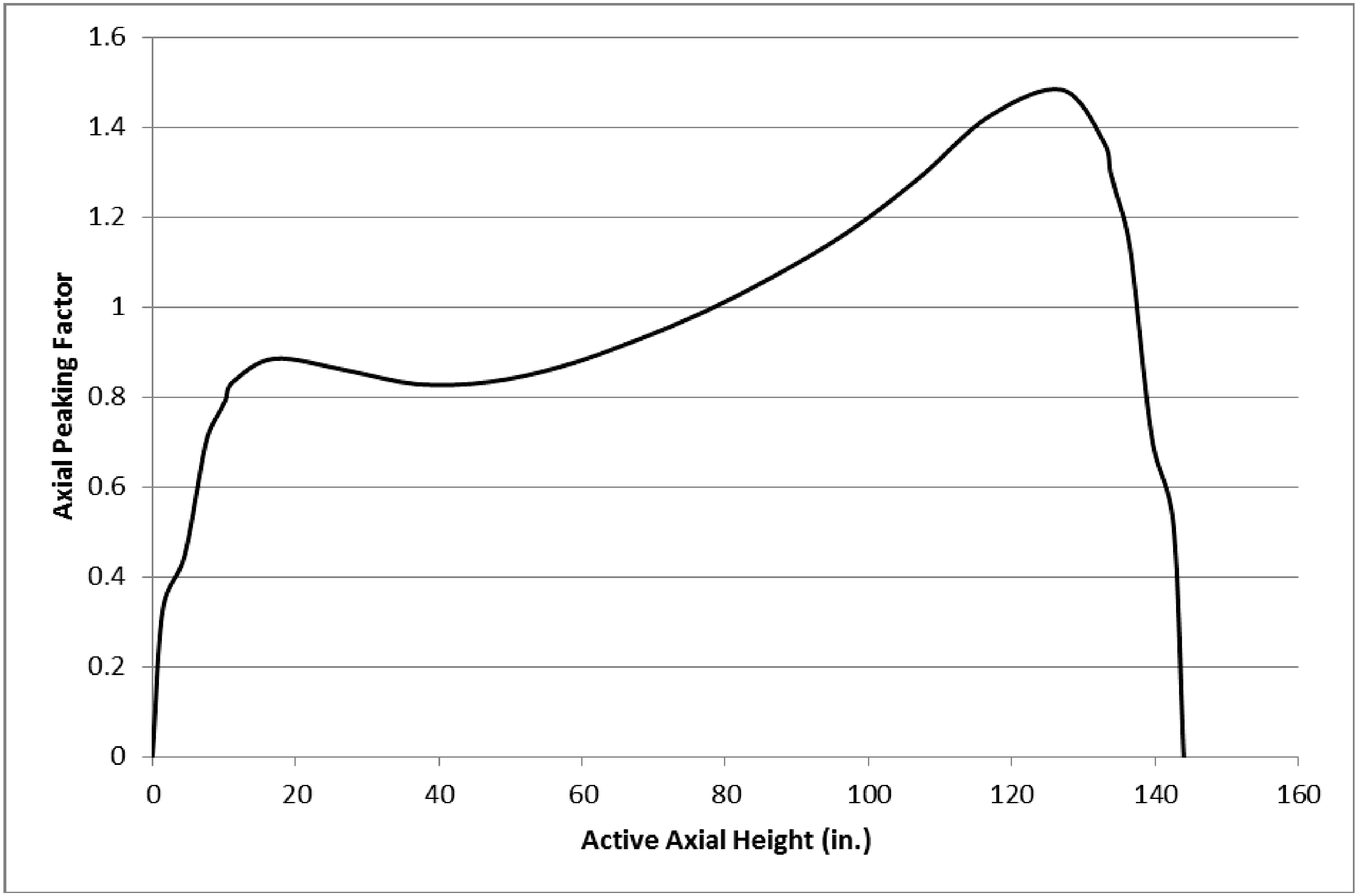
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Reactivity Feedback for Limiting MSLB	FIGURE 15.1.5-13



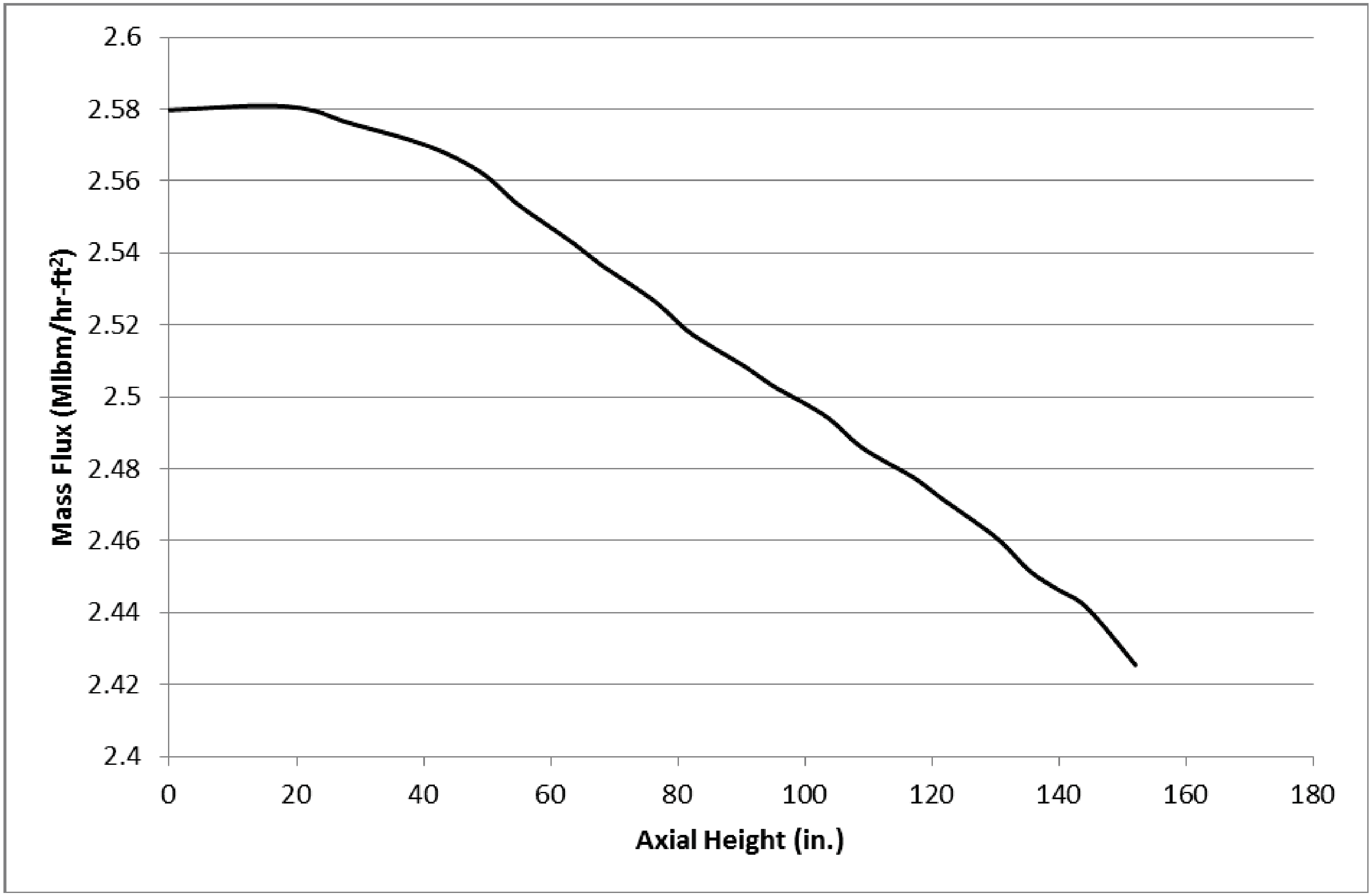
REVISION NO. 26

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Power for Limiting MSLB	FIGURE 15.1.5-14



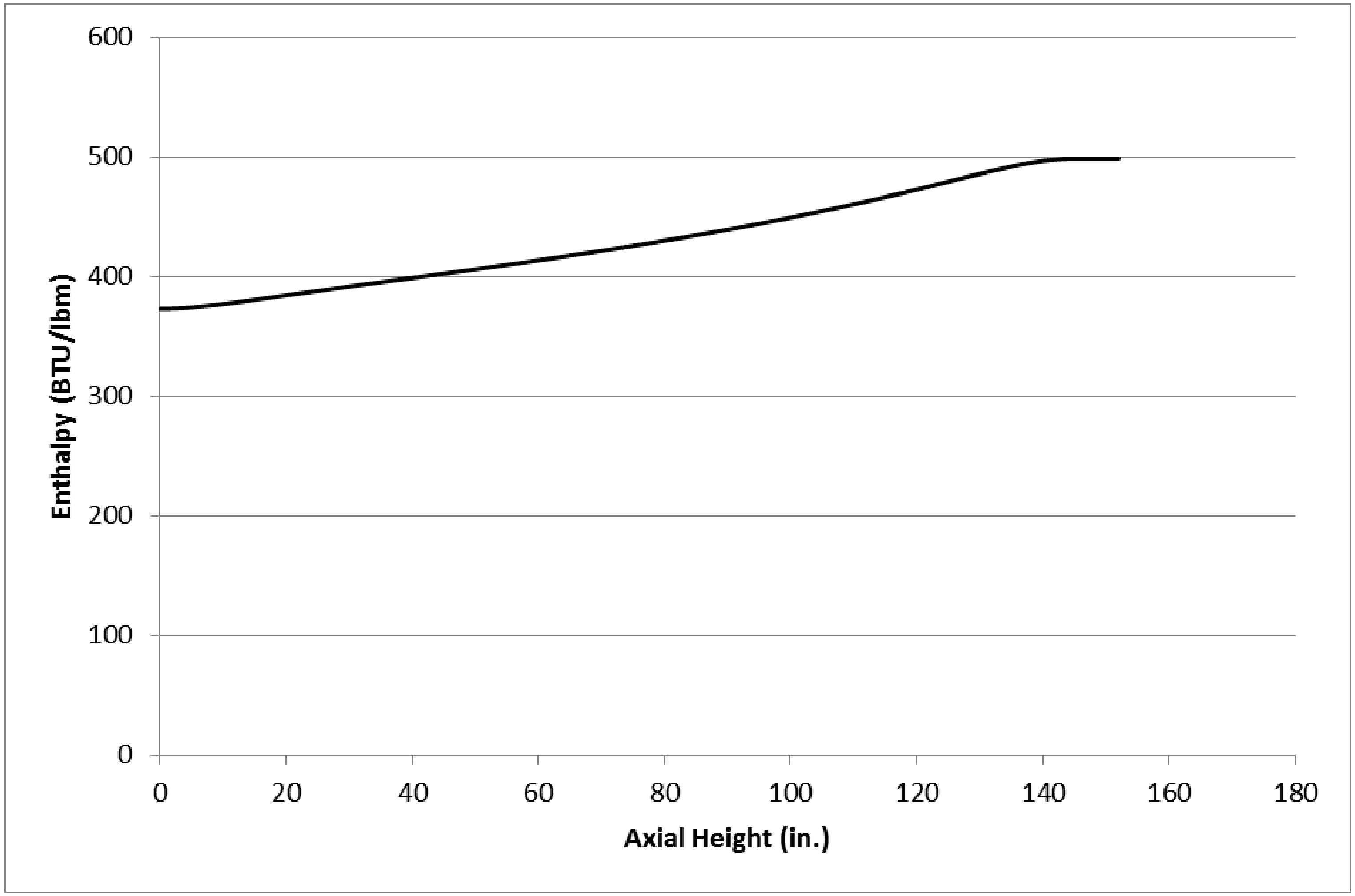
REVISION NO. 27

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Hot Assembly Axial Power Distribution for Limiting MSLB	FIGURE 15.1.5-15



REVISION NO. 27

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Hot Assembly Average Mass Flux for Limiting MSLB	FIGURE 15.1.5-16



REVISION NO. 27

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Hot Assembly Average Enthalpy for Limiting MSLB	FIGURE 15.1.5-17

15.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents have been postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the Reactor Coolant System (RCS). Detailed analyses are presented in this section for several such events which have been identified as more limiting than the others.

Discussion of the following RCS coolant heatup events are presented in Section 15.2:

a) Steam pressure regulator malfunction or failure that results in decreasing steam flow

b) Loss of external electrical load

c) Turbine trip

d) Loss of condenser vacuum and other events resulting in turbine trip

e) Inadvertent closure of main steam isolation valves

f) Loss of non-emergency AC power to the station auxiliaries

g) Loss of normal feedwater flow, and

h) Feedwater system pipe break.

15.2.1 STEAM PRESSURE REGULATOR MALFUNCTION OR FAILURE THAT RESULTS IN DECREASING STEAM FLOW

There are no steam pressure regulators in the H. B. Robinson (HBR) Nuclear Power Plant whose failure or malfunction could cause a steam flow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Event Description

A major load loss on the generator can result from the loss of external electrical load due to an electrical system disturbance. Offsite electrical power is available to operate the reactor coolant system pumps and other station auxiliaries. Following the loss of generator load, the turbine control/stop valves close, terminating the steam flow and causing the secondary system temperature and pressure to increase. The primary-to-secondary heat transfer decreases as the secondary system temperature increases.

If the reactor is not tripped when the turbine is tripped, the primary system temperature continues to rise. The primary liquid will expand and the pressurizer steam space is compressed, causing the pressurizer pressure to rise. If this continues, the reactor will trip on high pressurizer pressure, reducing the primary heat source. As the heat load into the primary system decreases, the primary system pressurization will begin to diminish. If the setpoint for opening the primary system code safety valves is exceeded during the initial system overpressurization, these valves will open to relieve pressure and to mitigate the pressure transient. The mitigative features of the pressurizer spray, pressurizer relief valves, and the steam bypass system are assumed not to function, so as to exacerbate the overpressurization of the primary system. For the minimum DNBR case, the mitigative features of the pressurizer spray and pressurizer relief valves are assumed to function. This minimizes the pressurization of the primary system, resulting in a conservative evaluation of the MDNBR for this event. Energy is removed during the early phase of the transient through the steam generator safety valves when the steam generator pressure exceeds the safety valve opening setpoint.

The primary challenge of this transient is to the primary and secondary system overpressurization acceptance criterion (peak pressure less than 110 percent of the design value). This case also provides the limiting analysis for maximum pressurizer level for Condition II events. The challenge to the specified acceptable fuel design limit is also evaluated because of the increasing core inlet temperature and the potential for the reactor core power to increase prior to reactor trip. Reactor control is assumed to be in the manual mode, so the reactor power will not be reduced when the primary system average temperature begins to increase.

This event is a moderate frequency (Condition II) event (Table 15.0.1-1). The acceptance criteria for this event are listed in Section 15.0.1.1. As cited in Table 15.0.11-1, no single failure in the ESF will affect the analysis for this event.

15.2.2.2 Analysis Method

The analysis was performed using the ANF-RELAP and XCOBRA-IIIC codes. The ANF-RELAP code (Reference 15.0-3) was used to model the salient system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures). A DNBR calculation was performed to estimate the approximate time at which the DNBR was a minimum. The core fluid boundary conditions and average rod surface heat flux at this time were then used as input to the XCOBRA-IIIC code (Reference 15.0-4), which was used to evaluate the MDNBR.

15.2.2.3 Definition of Events Analyzed and Bounding Input

The purpose of analyzing this event is to demonstrate that the primary and secondary pressure relief capability is sufficient to limit the pressures to less than 110 percent of their respective design pressure limits. This event is also analyzed to ensure that reactor protection systems are properly set to prevent penetration of the SAFDLs under the limiting assumptions of no credit for a direct reactor trip on turbine trip and the unavailability of the secondary system relief capacity of the turbine bypass system.

Three cases are analyzed for this event: one challenging the RCS pressure criterion, one challenging the Secondary Side pressure criterion, and one challenging to the fuel design limits. In these, the input parameters are biased to maximize the increase in reactor power during the transient. However, in each case, the parameters and the equipment operational states are selected to maximize the challenge to the criterion of specific interest.

The bounding operating mode for this event is full power initial conditions with the reactor control system in the manual mode.

Pressurizer and steam safety valve setpoints and other parameters are biased per Table 15.0.8-1 for pressurization challenge and DNB challenge, except as provided below:

	Maximum Secondary Side Pressure	Maximum RCS Pressurization	Minimum DNBR.
Rod Control	Manual	Manual	Manual
Initial Power	2300 MWt +2%	2300 MWt +2%	2300 MWt +2%
Moderator temperature coefficient	0.0pcm/°F	5.0 pcm/°F	5.0 pcm/°F
Doppler coefficient	-0.968 pcm/°F	-0.8 pcm/°F	-0.8 pcm/°F
Core inlet temperature	Nominal	Nominal	Nominal
Initial RCS pressure	Nominal	Nominal	Nominal
Core outlet pressure used in subchannel analysis	______	______	Nom. -40 psi
Pressurizer level	Nominal	Maximum	Nominal
Pressurizer spray	Available	Disable	Available
Pressurizer PORVs	Available	Disable	Available
Steam bypass	Disable	Disable	Disable
Steam PORVs	Disable	Disable	Disable

	Maximum Secondary Side Pressure	Maximum RCS Pressurization	Minimum DNBR.
Reactor trip on turbine trip	Disable	Disable	Disable
High pressurizer pressure trip	Available	Available	Disable
Overtemperature ΔT trip	Available	Disable	Available

15.2.2.4 Analysis of Results

The maximum RCS pressurization case initiates with a turbine control/stop valve closure. Steam line pressure increases until the safety valves open at 13.2 sec. The pressurization of the secondary side results in decreased primary to secondary heat transfer, and a substantial rise in cold leg temperature. The average primary temperature increases about 12 F peaking at 9.2 sec. This results in a large insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The reactor trips on high pressure with rods beginning to insert at 6.1 sec. The pressurizer pressure reaches the safety valve opening setpoint at about 9.2 seconds, however, the pressure does not stay above the setpoint long enough to clear the safety valve loop seals (1 second clearing time assumed). As a result, the pressurizer safety valves do not open in the simulation of this transient. The increase in coolant temperature also causes the core power to rise to about 104 percent (of 2300 MWt) due to positive moderator feedback. The transient is terminated when the reactor scrams, decreasing temperature and hence pressure.

The maximum secondary side pressurization case is initiated in the same manner as the RCS pressurization case. The secondary pressure was demonstrated to be more limiting at zero % Steam Generator plugging. The steamline pressure increase caused the first MSSV to open at 11.3 seconds and the last MSSV to open at 19.3 seconds. The reactor trip signal on OT delta T occurred at 20.1 seconds and the peak secondary pressure of 1207.4 psia occurred at 27 seconds. The maximum secondary side pressurization case produced the most severe challenge to the pressurizer overfill criterion. However, the liquid level remains below the Pressurizer PORV and safety valve inlets, thereby preventing overfill and satisfying the acceptance criteria.

The minimum DNBR case is initiated in the same manner. Steam line pressure increases until the secondary side safety valves open at 12.9 sec. The pressurization of the secondary side results in decreased primary-to-secondary heat transfer and a substantial rise in cold leg temperature. The average primary temperature increased about 28ºF, peaking at 16.0 seconds. The rapid increase in primary side temperatures result in a large insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The pressurizer compensated and uncompensated PORV's opened at 3.8 and 4.9 seconds, respectively. Limiting the pressure rise prevents the reactor scram on high pressure. Therefore, in this case the reactor power reaches about 118 percent (of 2300 MWt), scramming on overtemperature ΔT with rod insertion commencing at 13.9 sec. The DNBR challenge results from the core power and primary coolant temperature increase. The challenge is further exacerbated by the limitation on primary pressure rise.

The transient response to the maximum RCS pressurization case is shown in Figures 15.2.2-1 to 15.2.2-5. The transient response to the maximum Secondary Side pressurization case is shown in Figures 15.2.2-1a to 15.2.2-5a. An event summary is shown in Table 15.2.2-1. The maximum reactor coolant system boundary pressure computed for the maximum RCS pressurization case is 2739.2 psia in the vessel lower head. This is below the 110 percent design allowable of 2750 psia. The response to the minimum DNBR case is given in Figures 15.2.2-6 to 15.2.2-12, with an event summary shown in Table 15.2.2-1. A minimum DNB ratio of 1.293 was calculated.

This is greater than the DNB limit of 1.141. The maximum secondary side pressure predicted is 1207.4 psia, at the bottom of the steam generators. This is below the 110% design allowable pressure of 1208.2 psia.

Per NSAL-03-1 (Reference 15.2.2-1), lower RCS temperatures may delay the opening of the MSSVs, which could increase the maximum RCS pressure during a Loss of External Load event; however, the use of nominal temperatures in the Maximum RCS Pressurization case bounds lower temperatures due to the fact that the MSSVs are actuated after the time of maximum RCS pressure.

15.2.2.5 Conclusion

The maximum pressure is less than the acceptance limit of 110 percent of design pressure and the minimum DNBR is greater than the approved safety limit. Therefore, acceptance criteria are met.

TABLE 15.2.2-1

LOSS OF EXTERNAL LOAD EVENT SUMMARY

RCS Pressurization Case

Time	Event	Value
0.0 s	Turbine tripped	-
5.1 s	Pressurizer pressure reached high-pressure trip setpoint (see Figure 15.2.2-3)	2430 psia
6.0 s	Core power peaked (see Figure 15.2.2-1)	104% of 2300 MWt
6.1 s	Scram rod insertion began	
9.2 s	Vessel average temperature peaked (see Figure 15.2.2-2)	587ºF
9.4 s	RCS pressure peaked (vessel lower head)	2740 psia
13.2 s	Steam line safety valves opened	1132 psia

Secondary Side Pressurization Case
(0% SG Tube Plugging)

Time (sec)	Event	Value
0.0	Turbine trip	-
0.0	Pressurizer Sprays – On	-
4.3	Pressurizer Uncompensated PORVs – Open	2359 psia
11.25	SG Main Steam Safety Valve (MSSV)-1 – Open	1132 psia
14.07	SG MSSV-2 – Open	1158 psia
15.8	SG MSSV-3 – Open	1173 psia
17.8	Peak pressurizer pressure	2425.6 psia
19.33	SG MSSV-4 – Open	1189 psia
20.09	Reactor trip – OTDT signal	-
20.84	Scram occurs	-
27.0	Peak secondary pressure	1207.4 psia

TABLE 15.2.2-1

LOSS OF EXTERNAL LOAD EVENT SUMMARY (continued)

Minimum DNBR Case

Time	Event	Value
0.0 s	Turbine tripped	-
3.8 s	Compensated pressurizer PORV opened	+96 psi error
4.9 s	Uncompensated pressurizer PORV opened (see Figure 15.2.2-9)	2346 psia
12.9 s	Steam line safety valves opened	1132 psia
13.2 s	Indicated vessel temperature rise reached OTΔT trip setpoint (see Figure 15.2.2-8)	54°F
13.9 s	Scram rod insertion began	-
13.9 s	Core power peaked (see Figure 15.2.2-6)	118% of 2300 MWt
14.4 s	Minimum DNBR occurred (see Figure 15.2.2-12)	1.293
16.0 s	Vessel average temperature peaked (see Figure 15.2.2-7)	603°F

15.2.3 TURBINE TRIP

The event is caused by a turbine trip signal. Turbine trip causes a direct reactor trip which results in earlier trip than analyzed in Section 15.2.2. Since the turbine stop valve closure time is used in the loss of load analysis, the consequences of this event are bounded by the conditions assumed for analysis of the loss of load event.

15.2.4 LOSS OF CONDENSER VACUUM AND OTHER EVENTS RESULTING IN TURBINE TRIP

This event may be caused by loss of the condenser cooling pumps. Loss of condenser vacuum disables steam bypass. Since steam bypass and direct trip on turbine trip is defeated in the loss of load event, the results of this event will be less severe than loss of load and are, therefore, bounded by the loss of load event. This disposition is not impacted by power uprate to 2339 MWt.

15.2.5 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES (MSIVs)

The event is assumed to be caused by malfunction of the valve controllers. The MSIVs close more slowly than the turbine stop valves, as is imposed in the loss of load event. Therefore, the consequences of this event will be bounded by the results of the loss of load analysis. This disposition is not impacted by power uprate to 2339 MWt.

15.2.6 LOSS OF NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

The event may be caused by a break in connection to the main grid. The H. B. Robinson plant is designed to accept a substantial load loss without trip, maintaining all auxiliaries as turbine load. The plant is designed to bring the station to safe shutdown even with turbine trip and consequent loss of auxiliaries.

The most likely result of break in connection to the grid is a sudden loss in load but maintaining station auxiliaries such as primary coolant pumps and main feedwater pumps. This branch of the event is similar to the loss of load event with respect to safety except the steam bypass and dump are expected to operate. The station sheds extra load in an orderly fashion and DNB margin increases. The challenging aspects are bounded by the loss of load results.

The second branch of this event is the possibility that the turbine trips with consequent loss of primary coolant and main feedwater pumps. The early stages of this subevent will be bounded by the results of the loss of forced reactor coolant flow event because direct trip will occur prior to low flow trip (for 3-pump coastdown), and MDNBR occurs prior to significant reduction in steam generator inventory. In the longer term, the results are bounded by the loss of normal feedwater event with concurrent loss of primary coolant pumps as the reactor trips directly on turbine trip and steam generator level at or above the low low trip. Therefore, this event is bounded by the loss of normal feedwater and 3-pump coastdown events. This disposition is not impacted by power uprate to 2339 MWt.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Event Description

The loss of normal feedwater event could result from loss of feedwater pumps, isolation of the feedwater regulating valves, or loss of offsite AC power. This results in reduction of heat removal capacity from the reactor system. An alternative supply of feedwater from the condensate storage tank is available with the steam-driven auxiliary or diesel engine generator motor-driven auxiliary feedwater pumps. This supply assures long-term and orderly recovery of the unit. The initial inventory in the steam generators assures a short-term controllable response.

The reactor trips early, due to either high pressurizer pressure or the steam generator low-low level reactor trip. Sufficient heat rejection capacity remains at this steam generator water level to avoid approach to DNB. The DNB aspects of the event are bounded by those of the loss of flow event, since the trip is delayed in the loss of flow event until loop flow coasts down to the low flow trip setpoint with consequent lower primary flow.

The objective of this analysis is to demonstrate the adequacy of relief capacity and setpoint of the steam generator safety valves, auxiliary feedwater capacity, and steam generator inventory to maintain primary system pressure below the 110 percent pressure vessel design rating and to avoid the expelling of liquid from the primary pressurizer safety valves. The latter assures long-term cooling capability to a safe shutdown condition and precludes pressure surge related to packing.

This event is classified as a Condition II event (Table 15.0.1-1). The acceptance criteria are as described in Section 15.0.1.1. As noted in Tables 15.0.8-1 and 15.0.11-1, minimum Auxiliary Feedwater flow is used as a conservative basis for the analysis; one motor driven pump delivering flow to two steam generators. The event is analyzed with and without primary coolant pump coastdown in order to bound results of all causative events.

15.2.7.2 Analysis Method

The analysis was performed using the ANF-RELAP thermal-hydraulic code (Reference 15.0-3) to simulate the system response. The ANF-RELAP code includes relevant aspects of the primary and secondary systems. Minimum auxiliary feedwater flow was used as a conservative basis for the analysis: one motor-driven pump delivering flow to two steam generators. The event was analyzed with and without offsite power available. The following assumptions were made:

1. The main feedwater valves are ramped closed at the initiation of the event.

2. The reactor trips on high pressurizer pressure or steam generator low-low level.

3. Depending on the case to be analyzed, all reactor coolant pumps may be tripped at the time of reactor trip and coast down.

4. The starting sequence for the auxiliary feedwater pump diesel generators (which includes a time delay) is initiated when the ESF steam generator low-low level signal is issued.

15.2.7.3 Definition of Events Analyzed and Bounding Input

Two cases were analyzed:

1. Reactor coolant pumps trip at reactor scram (loss of offsite power).

2. Reactor coolant pumps operate throughout transient (offsite power available).

These cases bound all operational modes for this event. Conservative conditions were used:

Initial power	102% of 2300 MWt
Moderator temperature coefficient	+5.0 pcm/ºF
Doppler coefficient	-0.8 pcm/ºF
Condensate storage tank temperature	Maximum [115ºF]
Steam-driven auxiliary feedwater pump	Disabled
Diesel generator-driven auxiliary feedwater pump	One available[(a)]

15.2.7.4 Analysis of Results

The event was initiated by shutting off the main feedwater flow to all steam generators, using a conservatively short 1.0 second rampdown. Steam generator pressures rose slowly due to the cessation of feedwater flow. This resulted in a reduction of reactor system heat removal, which caused a primary temperature rise and an increase in reactor power, as well as pressurizer level and pressure. The pressurizer pressure reached the high-pressure trip setpoint of 2430 psia at 40.0 seconds. The reactor scrammed at 41.0 seconds. The turbine tripped at reactor scram, which caused a rapid increase in the secondary pressure, primary pressure, pressurizer level, and primary coolant temperature. The steam generator level reached the low-low setpoint at 41.9 seconds. Auxiliary feedwater flow began 105 seconds after the steam generator low-low level setpoint was reached, delivering a total of 240 gpm to two steam generators.

For the PUMPS OFF case (see Figures 15.2.7-1 through 15.2.7-7 and Table 15.2.7-1), the reactor coolant pumps were tripped and began to coast down at reactor scram (41.0 seconds). The maximum vessel average coolant temperature was calculated to be 584□F (at 43.5 seconds), and the maximum primary coolant system pressure (at the bottom of the reactor vessel) was calculated to be 2543 psia (at 44.0 seconds). The maximum pressurizer level was calculated to be 59.0% of span (at 44.5 seconds). The maximum secondary

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[(a)] Delivering a total of 240 gpm to two steam generators.

coolant system pressure (at the bottom of one of the steam generators) was calculated to be 1180 psia (at 61.0 seconds). The liquid inventories of the fed steam generators (C and A) reached an individual minimum of 32,348 lb (at 672.0 seconds). This minimum represented 35.51% of the initial inventory. Dryout occurred in the unfed steam generator (B) at 1690.0 seconds. Following dryout of the unfed steam generator, the primary system experienced a relatively rapid heatup until the auxiliary feedwater to the remaining steam generators restored sufficient liquid level to remove the power from the primary.

For the PUMPS ON case (see Figures 15.2.7-8 through 15.2.7-14 and Table 15.2.7-2), the maximum primary coolant system pressure (at the bottom of the reactor vessel) was calculated to be 2535 psia (at 42.5 seconds), and the maximum vessel average coolant temperature was calculated to be 583ºF (at 42.5 seconds). The maximum pressurizer level was calculated to be 58.6% of span (at 43.5 seconds). The maximum secondary coolant system pressure (at the bottom of one of the steam generators) was calculated to be 1185 psia (at 61.5 seconds). The liquid inventories of the fed steam generators reached an individual minimum of 9,079 lb (at 4325.0 seconds). This minimum represented 14.18% of the initial inventory. Dryout occurred in the unfed steam generator at 1825.0 seconds. Following dryout of the unfed steam generator, the primary system experienced a relatively rapid heatup until the auxiliary feedwater to the remaining steam generators restored sufficient liquid level to remove the power from the primary.

Of the two cases, the PUMPS OFF case provided the greater challenge to the acceptance criteria of pressurizer pressure and level swell. The PUMPS ON case provided the greater challenge to the minimum steam generator inventory criterion. The minimum steam generator liquid inventory was less for the PUMPS ON case because of the higher heat load (from the operating pumps). The maximum pressurizer pressure for both cases did not exceed the primary safety valves opening setpoint of 2677 psia.

15.2.7.5 Conclusion

The SAFDLs are bounded by the loss of flow event (15.3.1). The primary pressure is less than the 110 percent vessel design rating of 2750 psia. The pressurizer retains steam to avoid expelling of liquid. The steam generators retain adequate inventory to maintain heat removal capability and the auxiliary feedwater system provides adequate coolant to assure orderly recovery and maintain the unit in safe condition.

TABLE 15.2.7-1

LOSS OF NORMAL FEEDWATER
EVENT SUMMARY FOR PUMPS OFF CASE

Time	Event	Value
0.0 s	Main feedwater flow was shut off	-
40.0 s	Pressurizer pressure reached high-pressure trip setpoint (see Figure 15.2.7-4)	2430 psia
41.0 s	Scram rod insertion began	-
41.0 s	Reactor coolant pumps were tripped	-
41.9 s	Steam generator level reached low-low level ESF setpoint	0.0% of span
44.0 s	Primary pressure peaked (vessel lower head)	2543 psia*
44.5 s	Pressurizer liquid level peaked	59.0% of span*
61.0 s	Secondary pressure peaked (bottom of steam generators)	1180 psia
108.9 s	Auxiliary feedwater began feeding steam generators C and A	-
672.0 s	Liquid inventory of steam generators C and A reached minimum (see Figure 15.2.7-7)	32,348 lb (SG C)
1690.0 s	Steam generator B dried out	-

*This value is based on an initial level of 53.3%.

Pressurizer level and pressure will increase slightly when the Technical Specifications maximum initial value of 63.3% is incorporated into the analysis. Current evaluations demonstrate the results will remain well within the acceptance criteria.

TABLE 15.2.7-2

LOSS OF NORMAL FEEDWATER
EVENT SUMMARY FOR PUMPS ON CASE

Time	Event	Value
0.0 s	Main feedwater flow was shut off	-
40.0 s	Pressurizer pressure reached high-pressure trip setpoint (see Figure 15.2.7-11)	2430 psia
41.0 s	Scram rod insertion began	-
41.9 s	Steam generator level reached low-low level ESF setpoint	0.0% of span
42.5 s	Primary pressure peaked (vessel lower head)	2535 psia*
43.5 s	Pressurizer liquid level peaked	58.6% of span*
61.5 s	Secondary pressure peaked (bottom of steam generators)	1185 psia
108.9 s	Auxiliary feedwater began feeding steam generators C and A	-
1825.0 s	Steam generator B dried out	-
4325.0 s	Liquid inventory of steam generators C and A reached minimum	9,079 lb (SG C and SG A)

*This value is based on an initial level of 53.3%.

Pressurizer level and pressure will increase slightly when the Technical Specifications maximum initial value of 63.3% is incorporated into the analysis. Current evaluations demonstrate the results will remain well within the acceptance criteria.

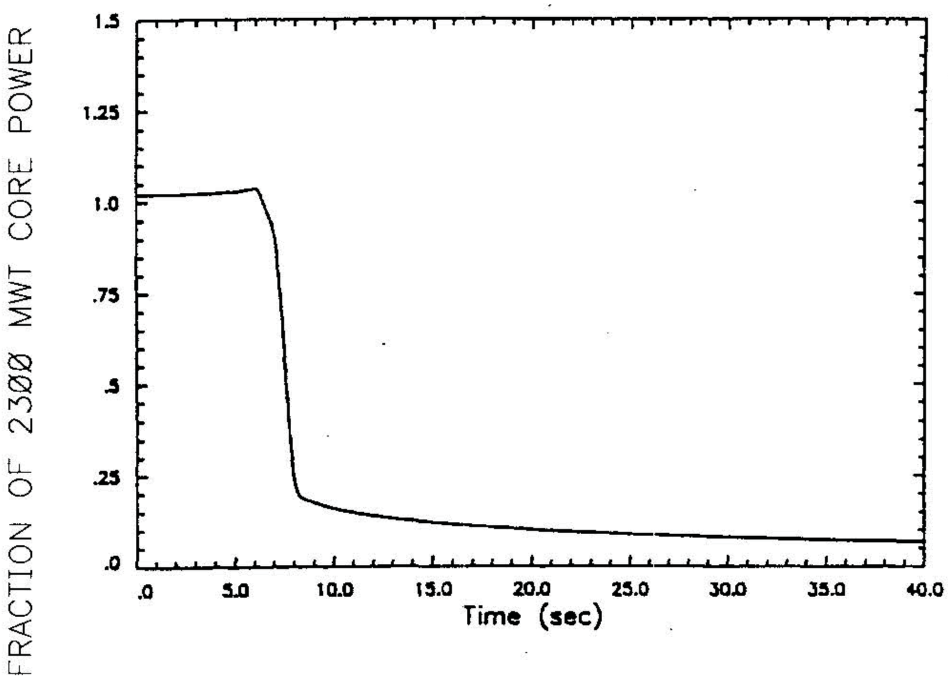
15.2.8 FEEDWATER SYSTEM PIPE BREAK

This event is postulated to be caused by the instantaneous severence of a feedwater line.

The H. B. Robinson steam generators are fed by a single 16-inch line. Auxiliary feedwater enters the same nozzle. The sparger is approximately 3 feet above the top of the tube bends and approximately 7 feet below the top of the downcomer. The sparger is approximately 2 feet above the low range liquid level tap. The feedwater sparger is of the J tube type. Upon rupture, some liquid may initially blow down; however, substantial liquid will remain. In many PWRs, feedwater is introduced at the bottom of the steam generator and a feedwater pipe break potentially results in a major or total loss of steam generator liquid inventory and subsequent primary system heatup. In the case of H. B. Robinson, however, this event will be a cooldown event and will be bounded by the steam line break results as the feedwater pipe is much smaller in area than the minimum area for flow in the new steam generator integral flow restrictors. This disposition is not impacted by power uprate to 2339 MWt.

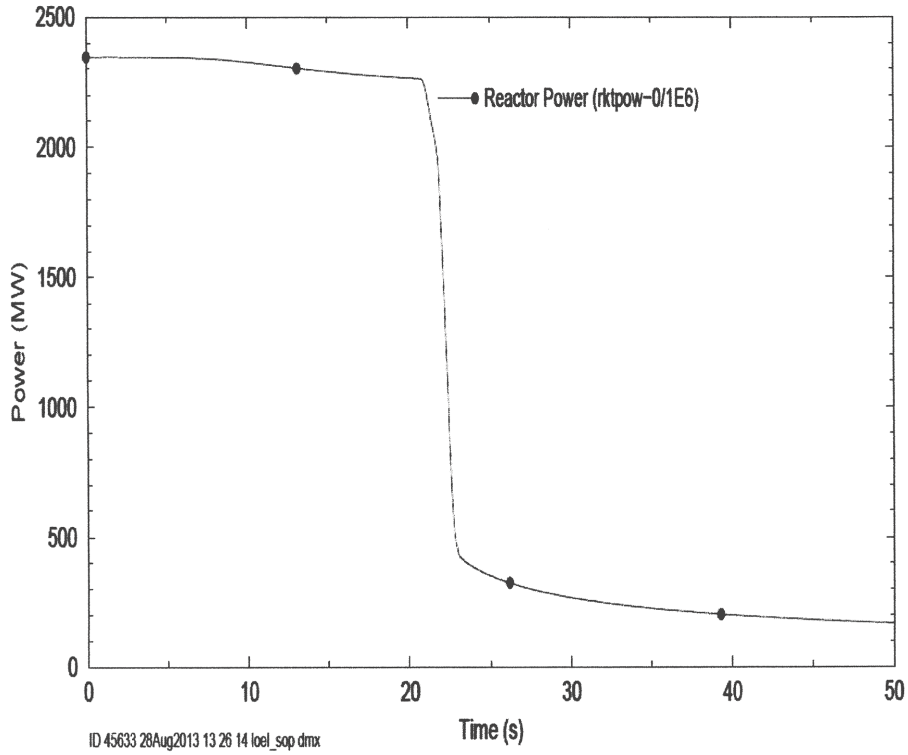
REFERENCES: SECTION 15.2

15.2.2-1 NSAL-03-1, “Safety Analysis Modeling Loss of Load/Turbine Trip,” Westinghouse, January 2003.



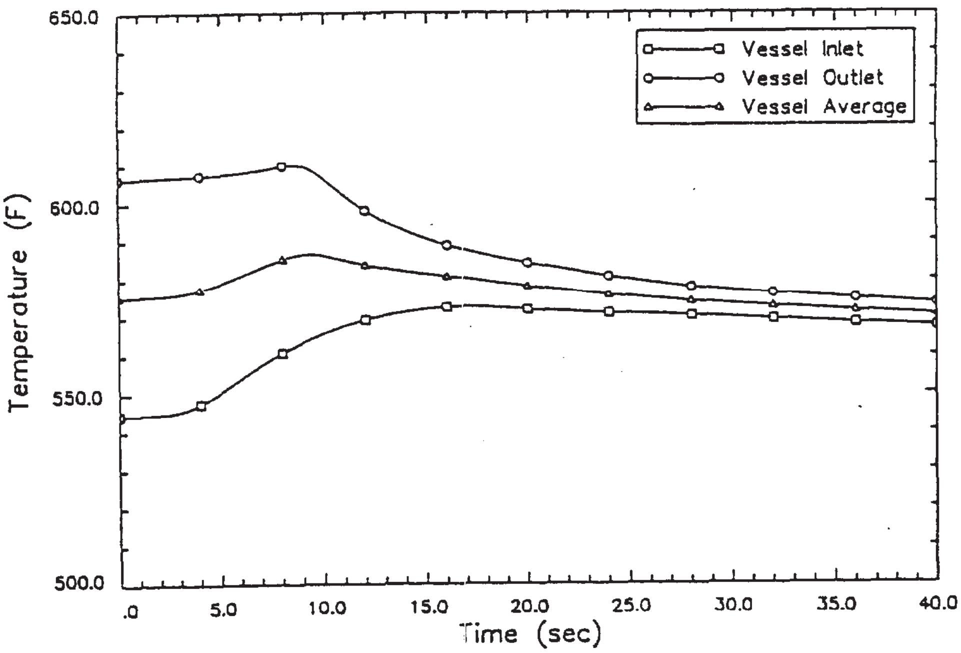
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, RCS Pressurization Case: Core Power	FIGURE 15.2.2-1



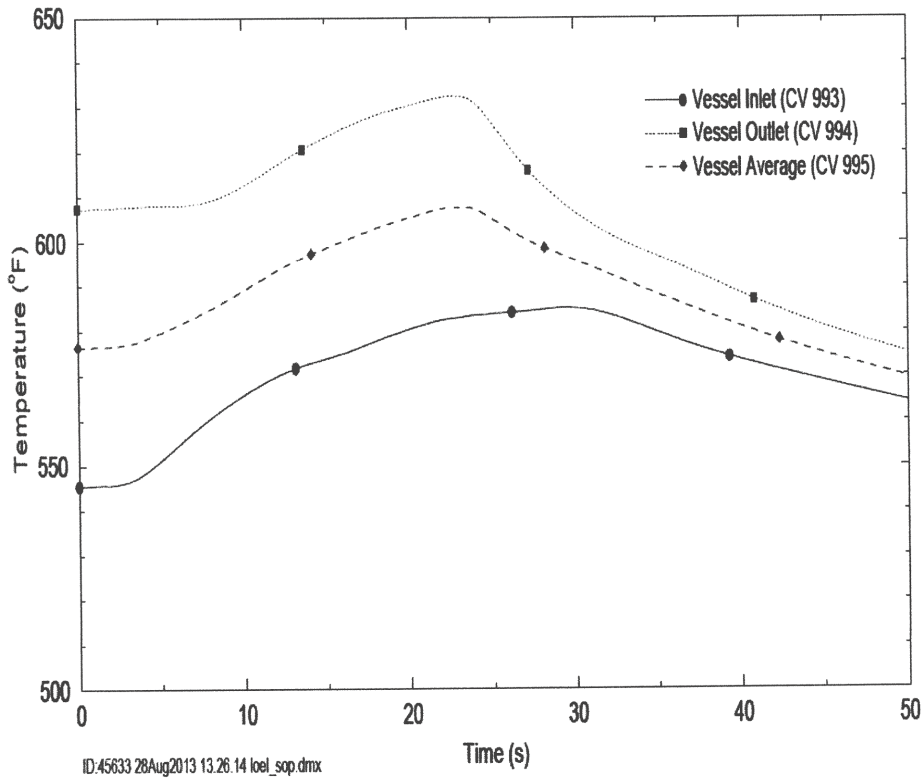
REVISION NO. 25

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF EXTERNAL LOAD, SECONDARY SIDE PRESSURIZATION CASE: CORE POWER	FIGURE 15.2.2-1a



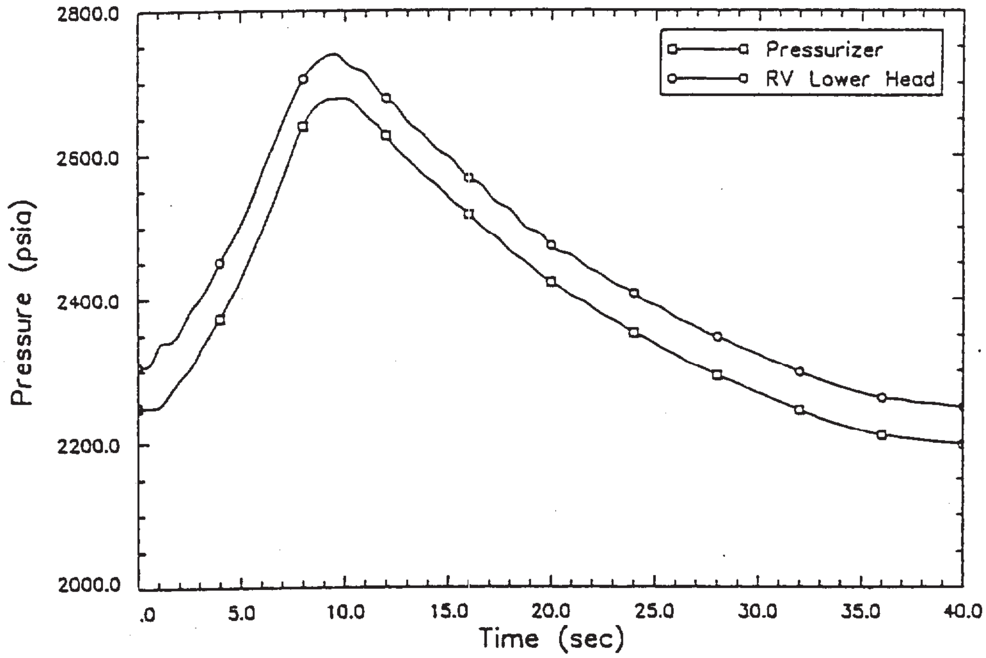
REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, RCS Pressurization Case: Reactor Coolant Temperatures	FIGURE 15.2.2-2



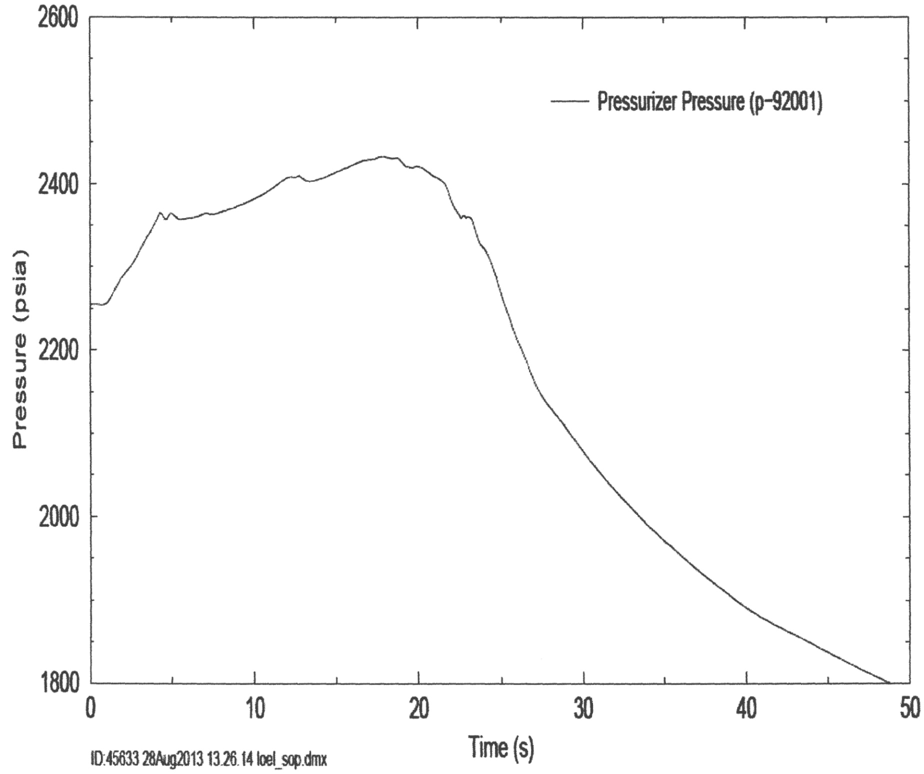
REVISION NO. 25

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF EXTERNAL LOAD, SECONDARY SIDE PRESSURIZATION CASE: REACTOR COOLANT TEMPERATURES	FIGURE 15.2.2-2a



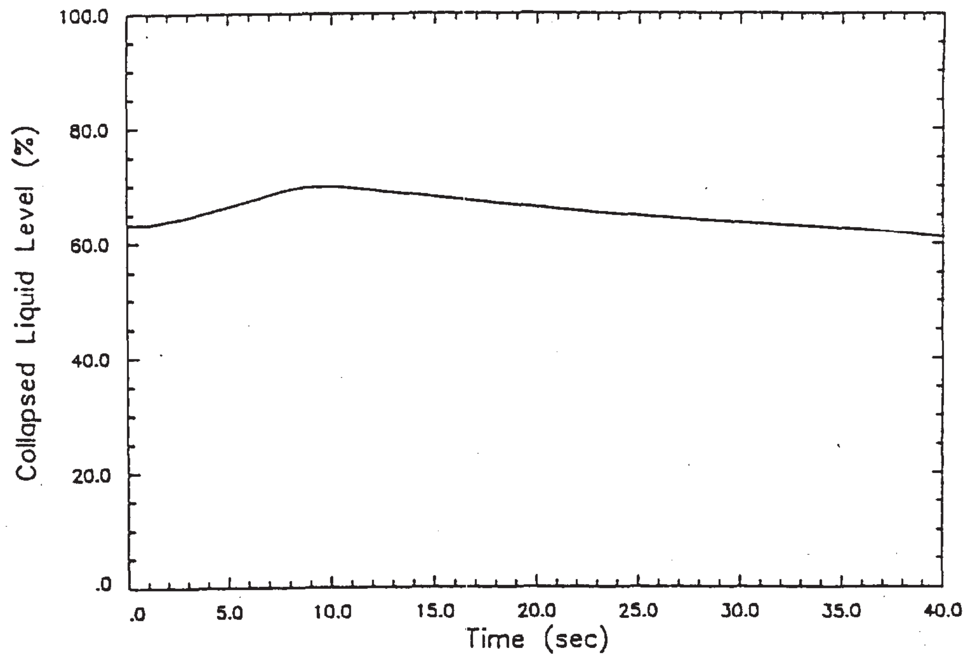
REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, RCS Pressurization Case: Pressurizer Pressure and Reactor Vessel Lower Head Pressure	FIGURE 15.2.2-3



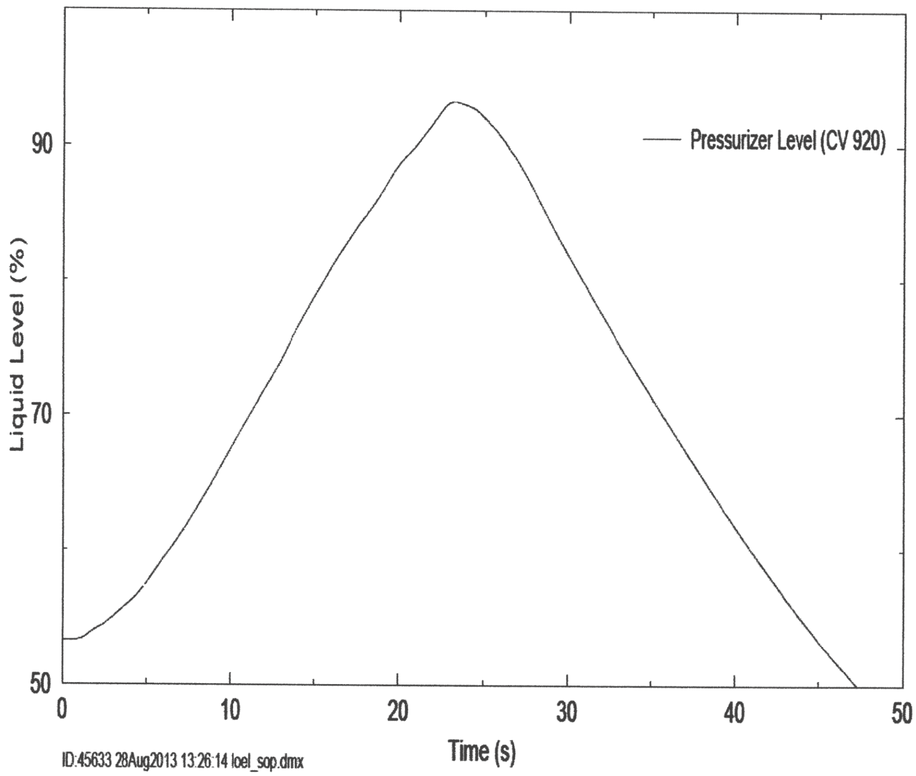
REVISION NO. 25

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF EXTERNAL LOAD, SECONDARY SIDE PRESSURIZATION CASE: PRESSURIZER PRESSURE	FIGURE 15.2.2-3a



REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, RCS Pressurization Case: Pressurizer Liquid Level	FIGURE 15.2.2–4



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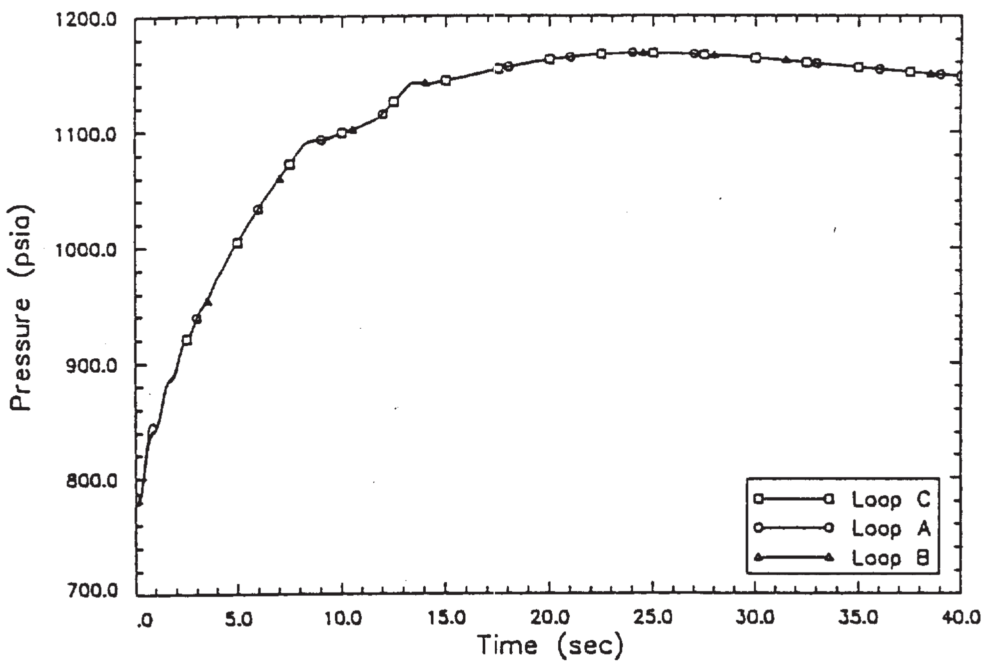
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF EXTERNAL LOAD, SECONDARY SIDE PRESSURIZATION CASE: PRESSURIZER LIQUID LEVEL	FIGURE 15.2.2-4a



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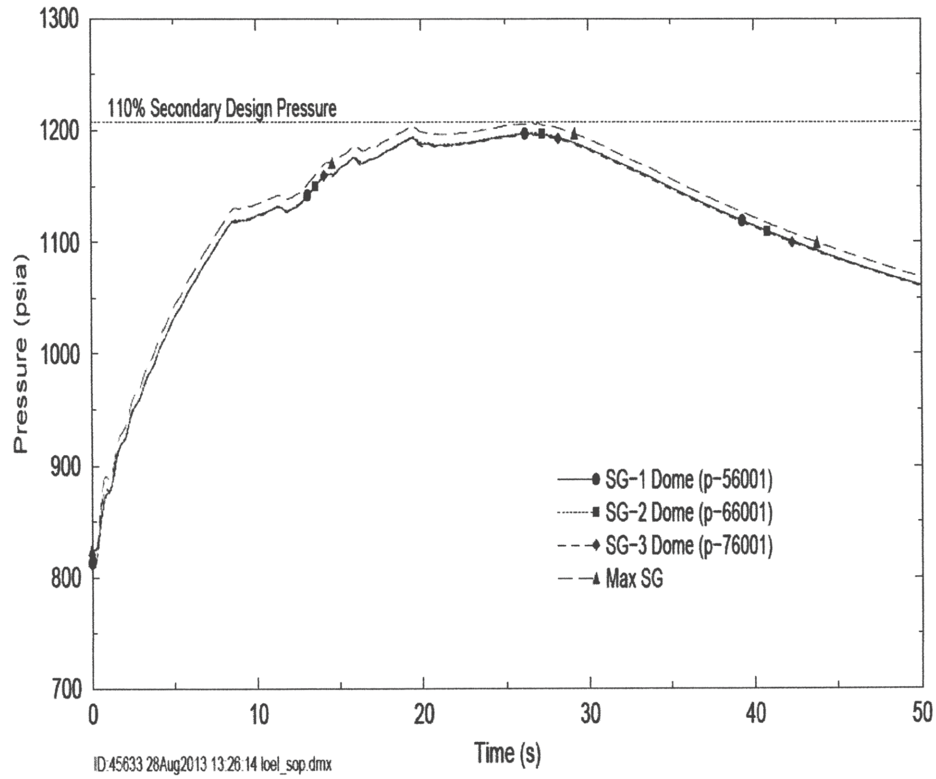
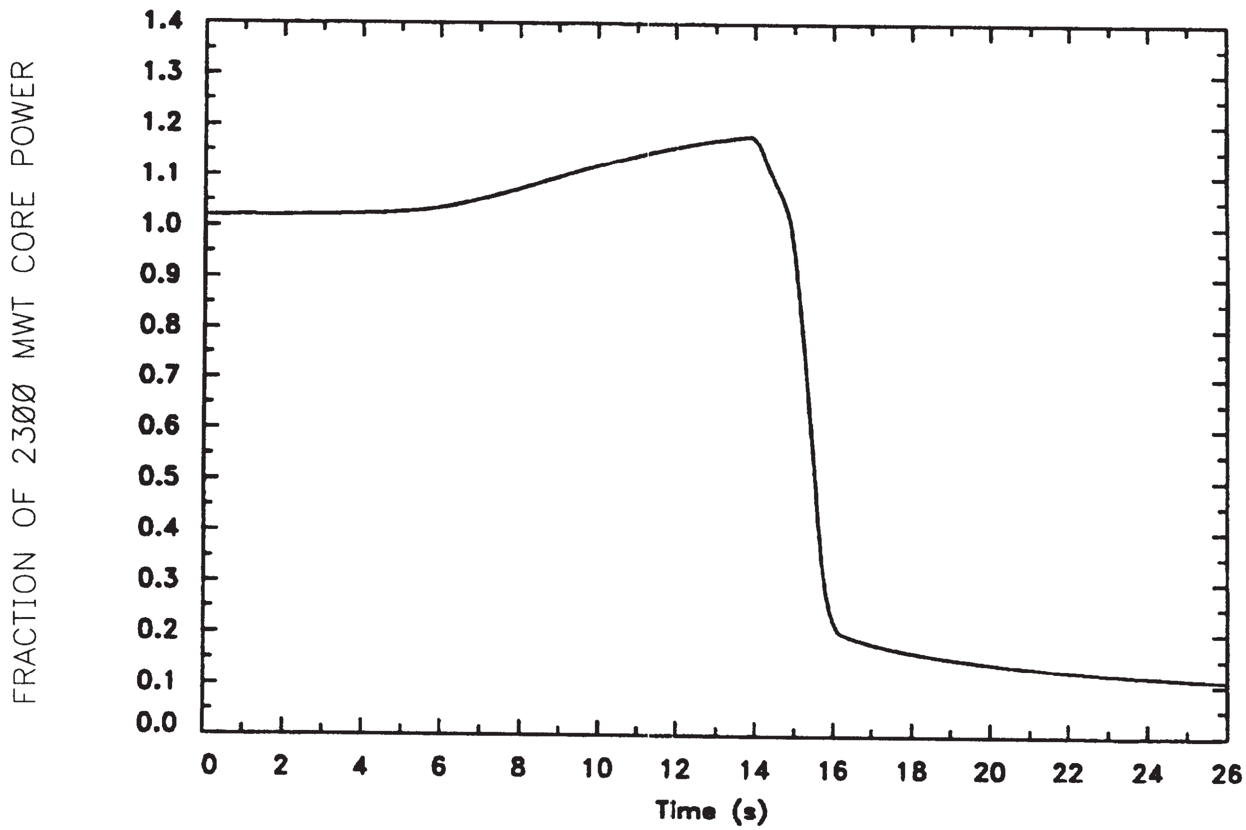
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, RCS Pressurization Case: Steam Generator Pressures	FIGURE 15.2.2-5



REVISION NO. 25

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	LOSS OF EXTERNAL LOAD, SECONDARY SIDE PRESSURIZATION CASE: STEAM GENERATOR PRESSURES	FIGURE 15.2.2-5a



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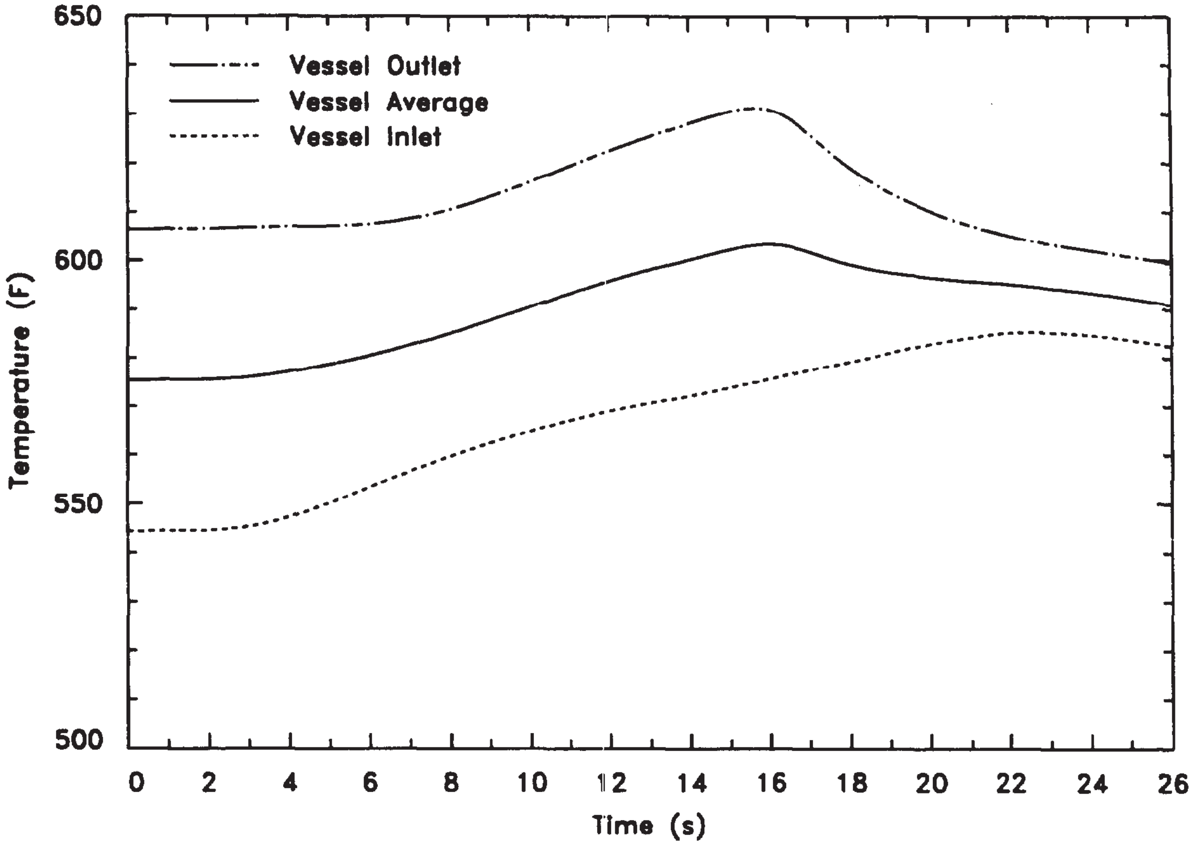
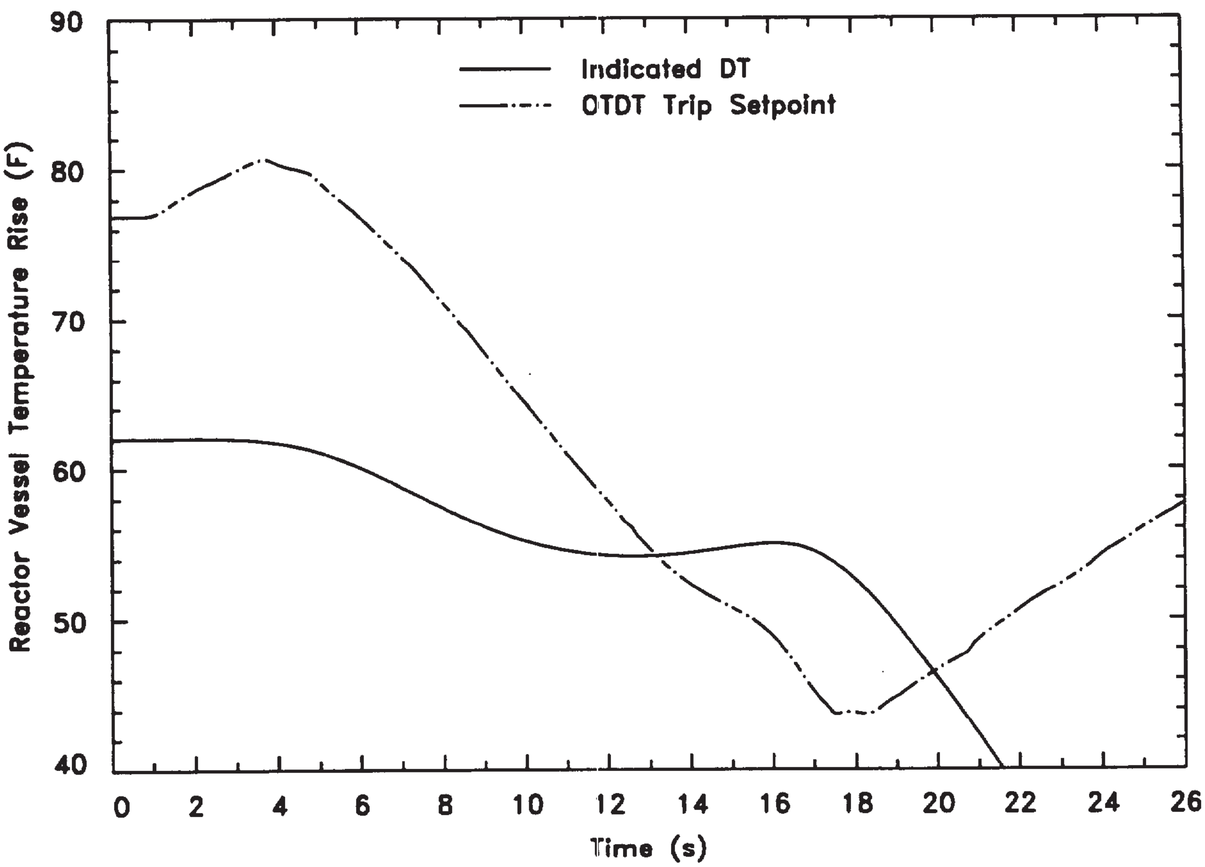
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, MDNBR Case: Core Power	FIGURE 15.2.2-6



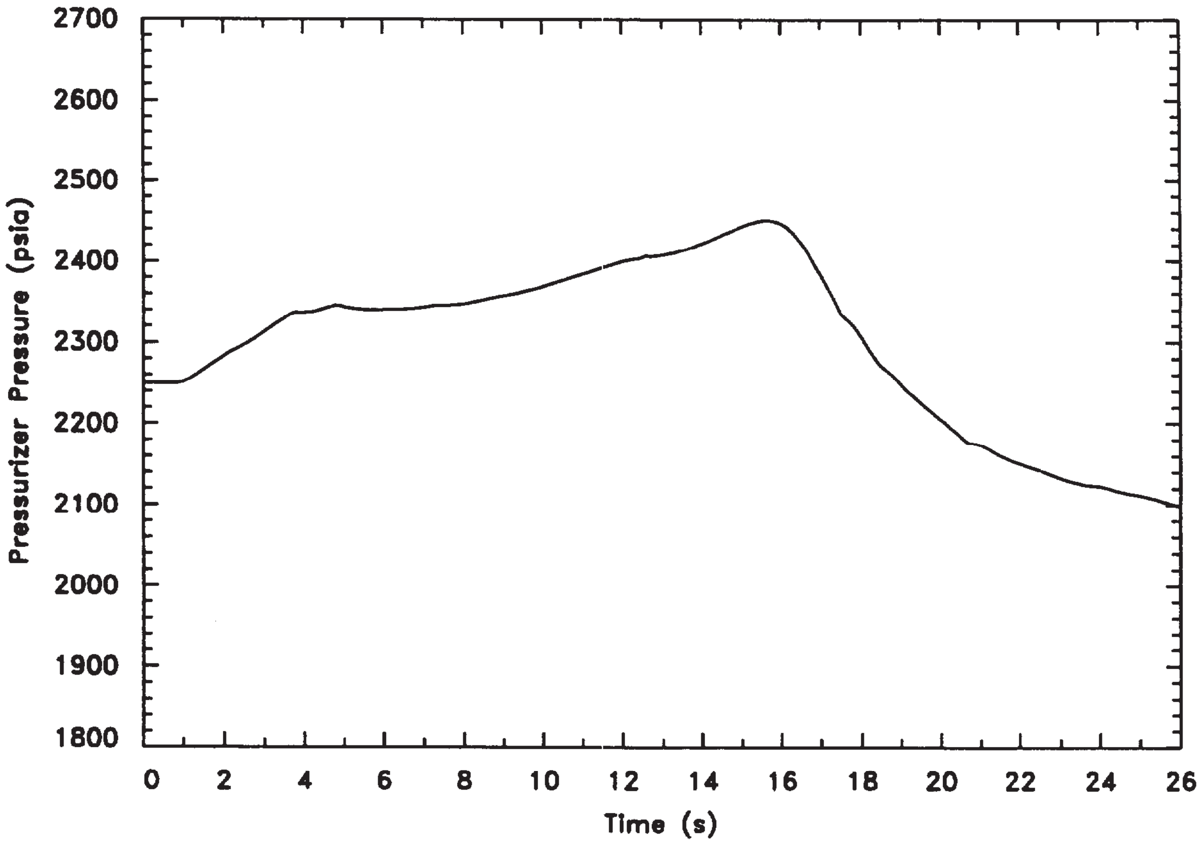
REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	**Loss of External Load, MDNBR Case: Reactor Coolant Temperatures**	FIGURE 15.2.2-7



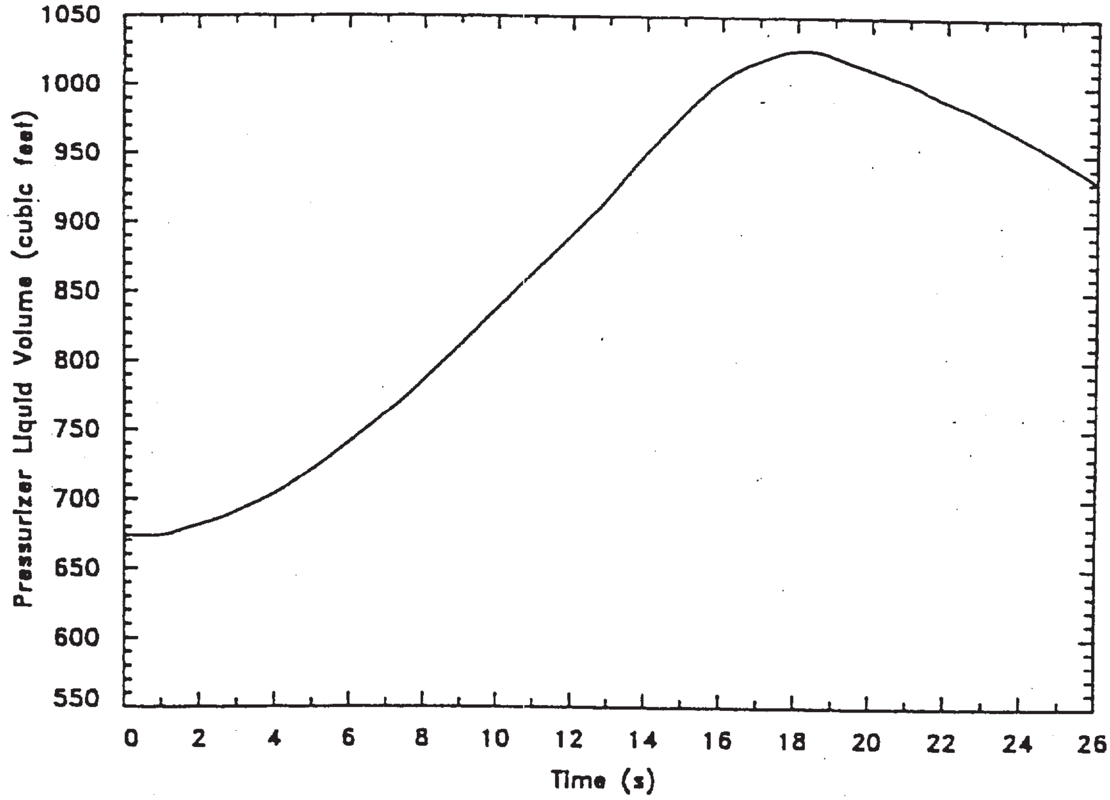
REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	**Loss of External Load, MDNBR Case: Reactor Vessel Temperature Rise and OTΔT Setpoint**	FIGURE 15.2.2-8



REVISION NO. 15

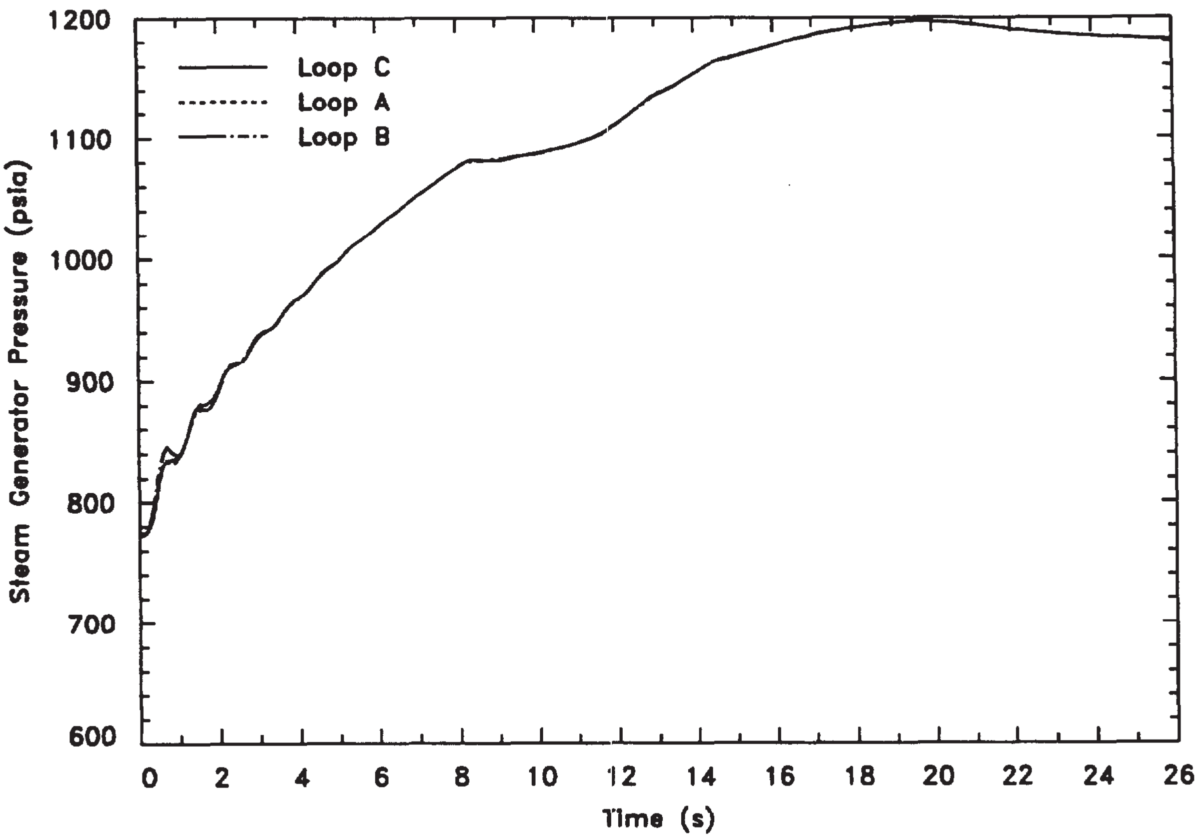
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	**Loss of External Load, MDNBR Case: Pressurizer Pressure**	FIGURE 15.2.2-9



NOTE: Reviewed analysis of this event combined with consideration of a higher initial pressurizer water level identified in Technical Specifications will result in slightly higher liquid levels than shown here. However, overfill is still avoided.

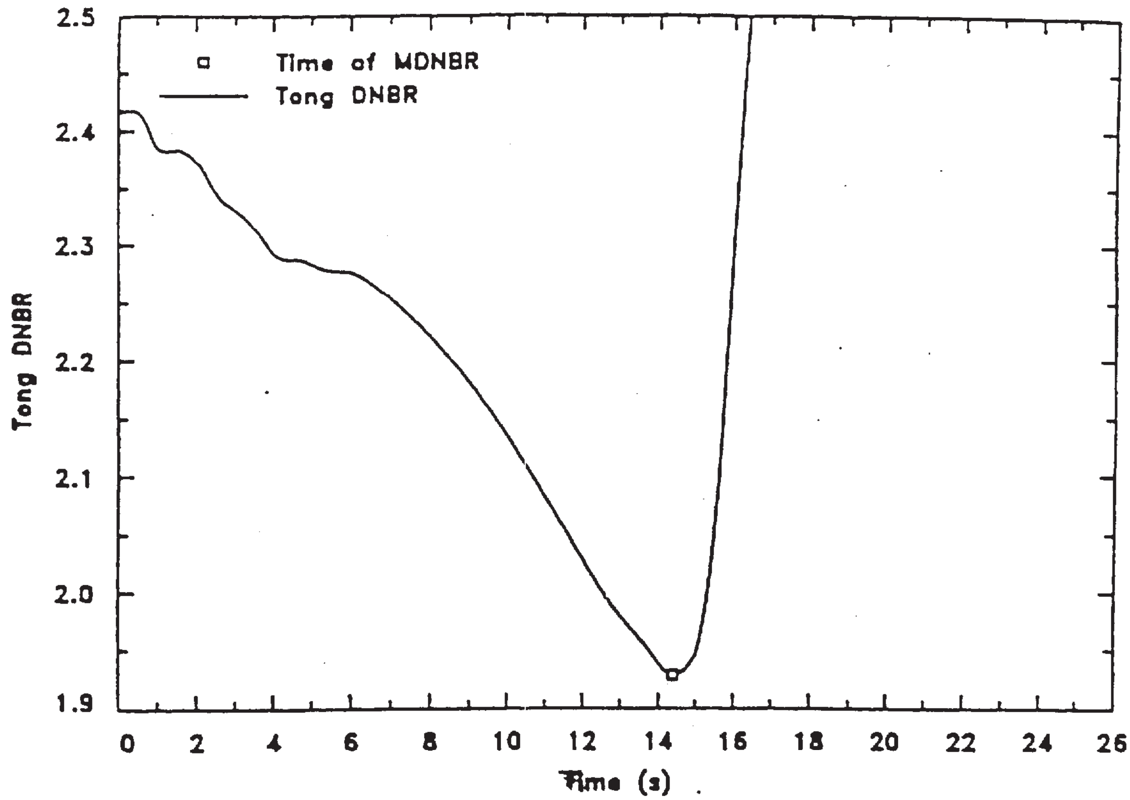
REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, MDNBR Case: Pressurizer Liquid Volume	FIGURE 15.2.2-10



REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	**Loss of External Load, MDNBR Case: Steam Generator Pressures**	FIGURE 15.2.2-11



Note: Tong DNBR value is a RELAP calculated core-wide DNBR trend used to estimate approximate time of minimum DNBR to establish core parameters for input into the XCOBRA IIIC code to evaluate MDNBR.

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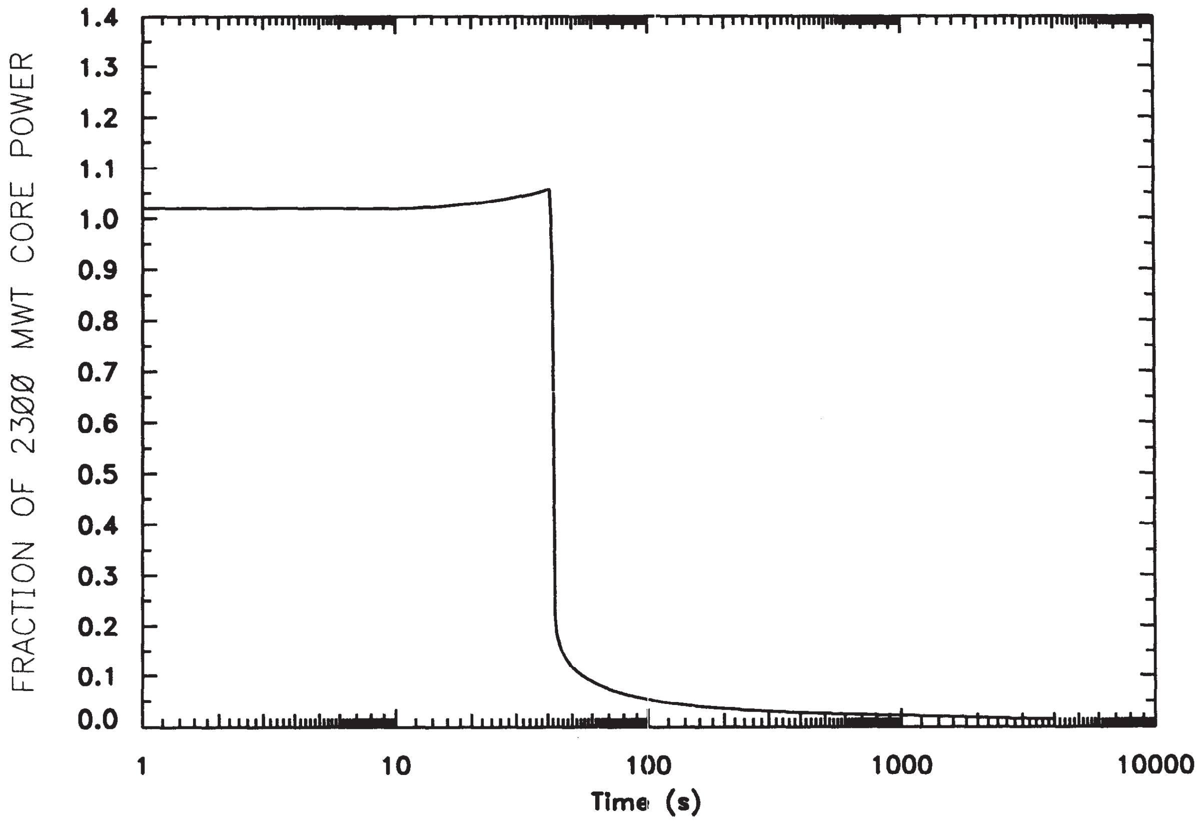
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of External Load, MDNBR Case: DNBR Trend	FIGURE 15.2.2-12



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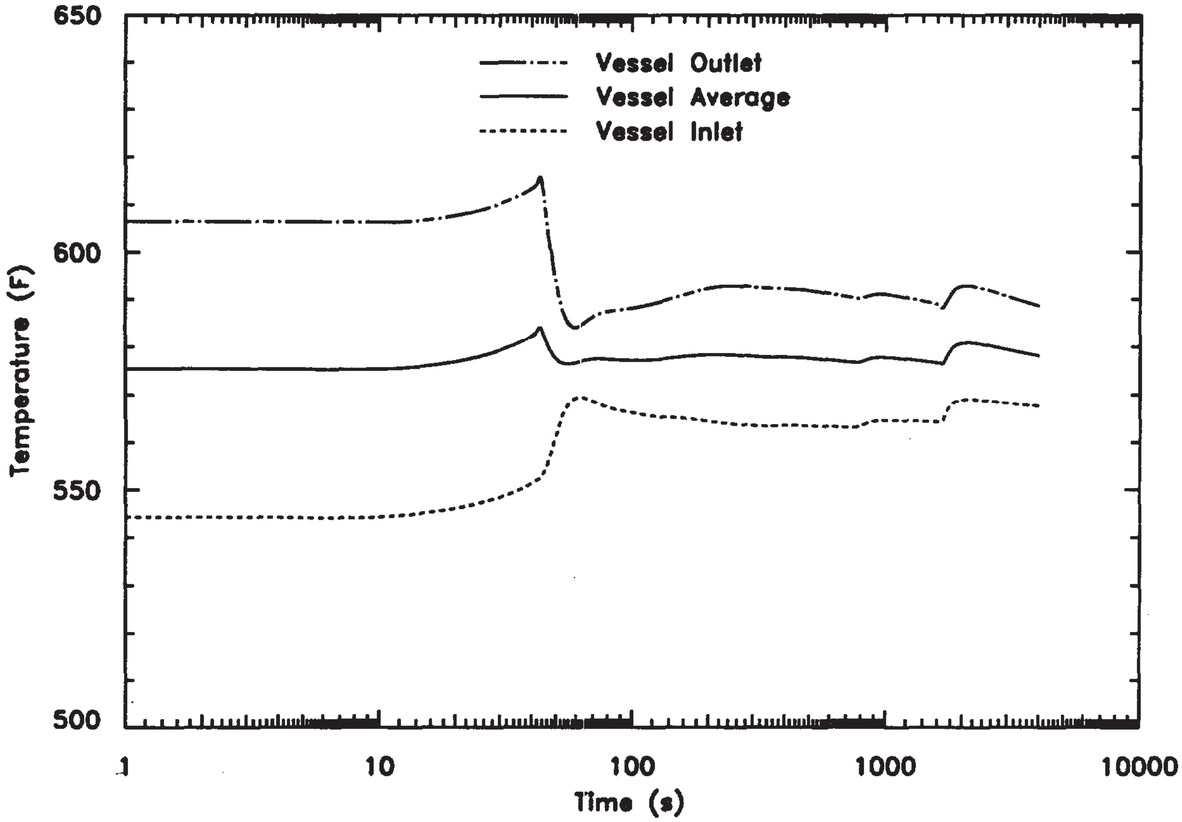
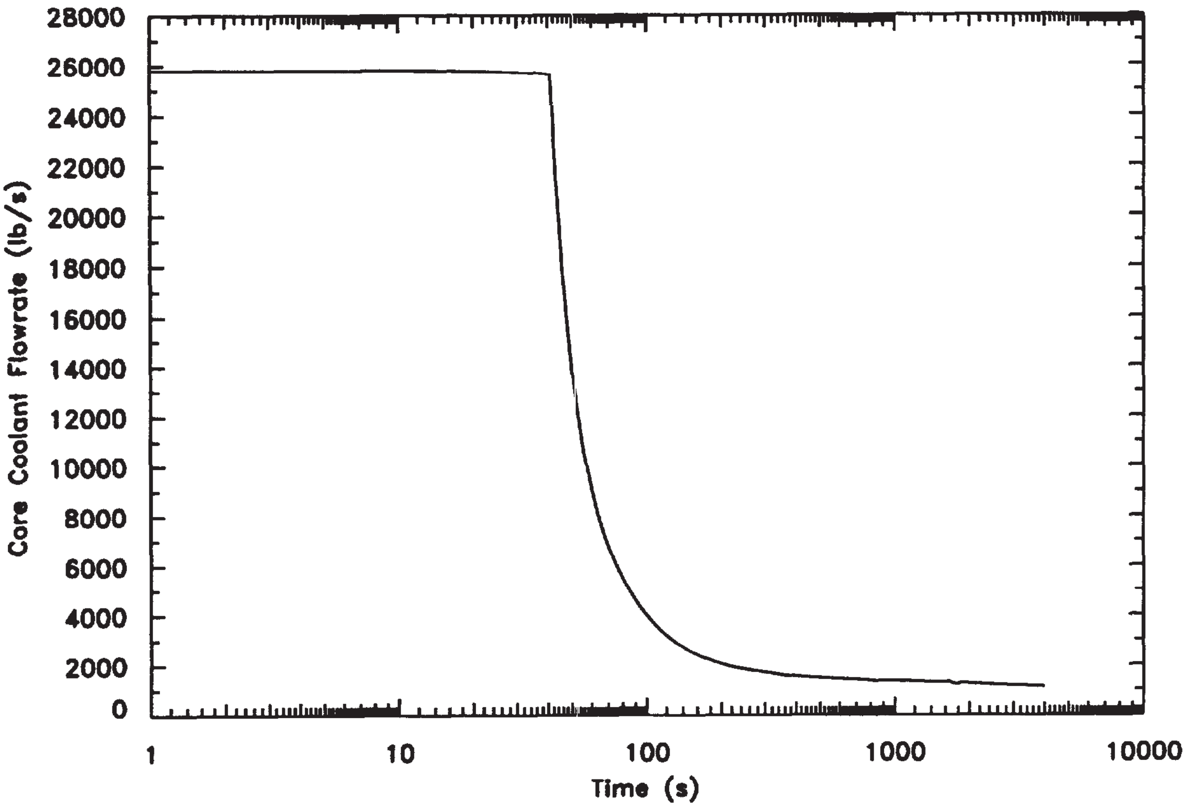
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Core Power	FIGURE 15.2.7-1



REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Reactor Coolant Temperatures	FIGURE 15.2.7-2



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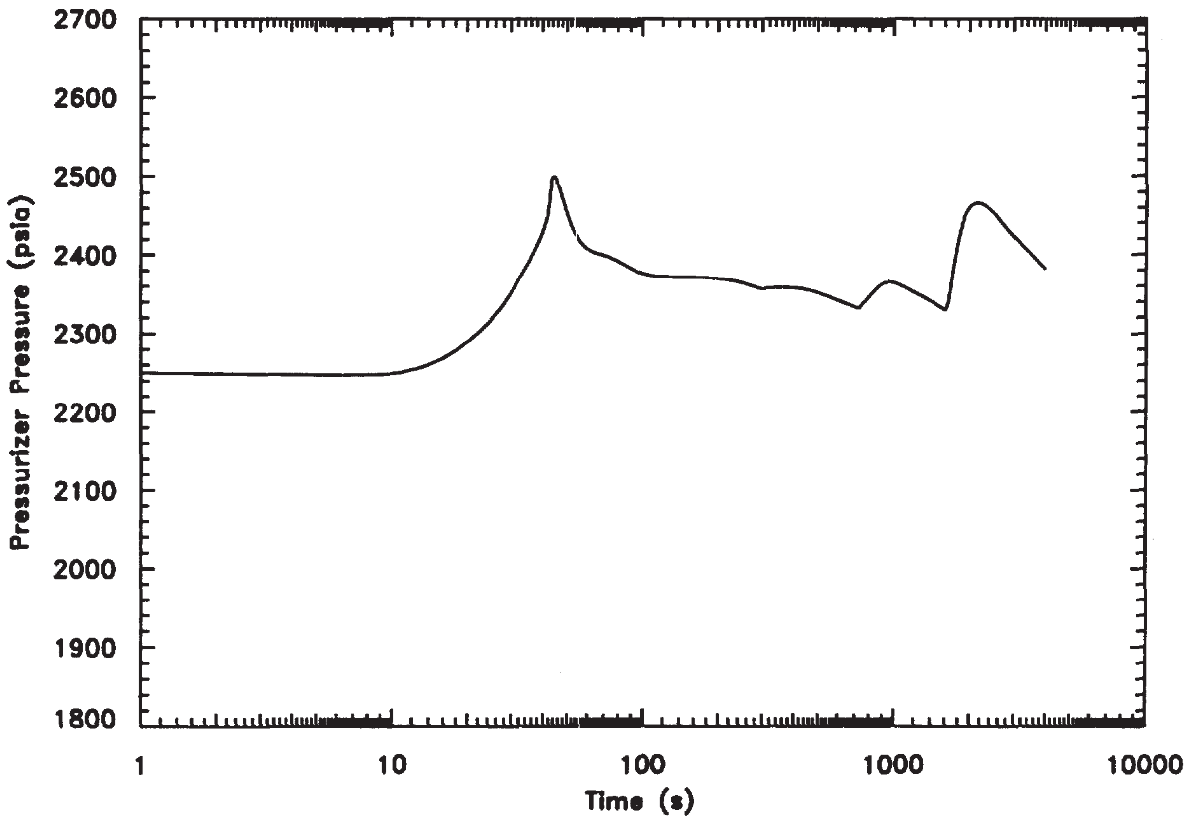
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Core Coolant Flowrate	FIGURE 15.2.7-3



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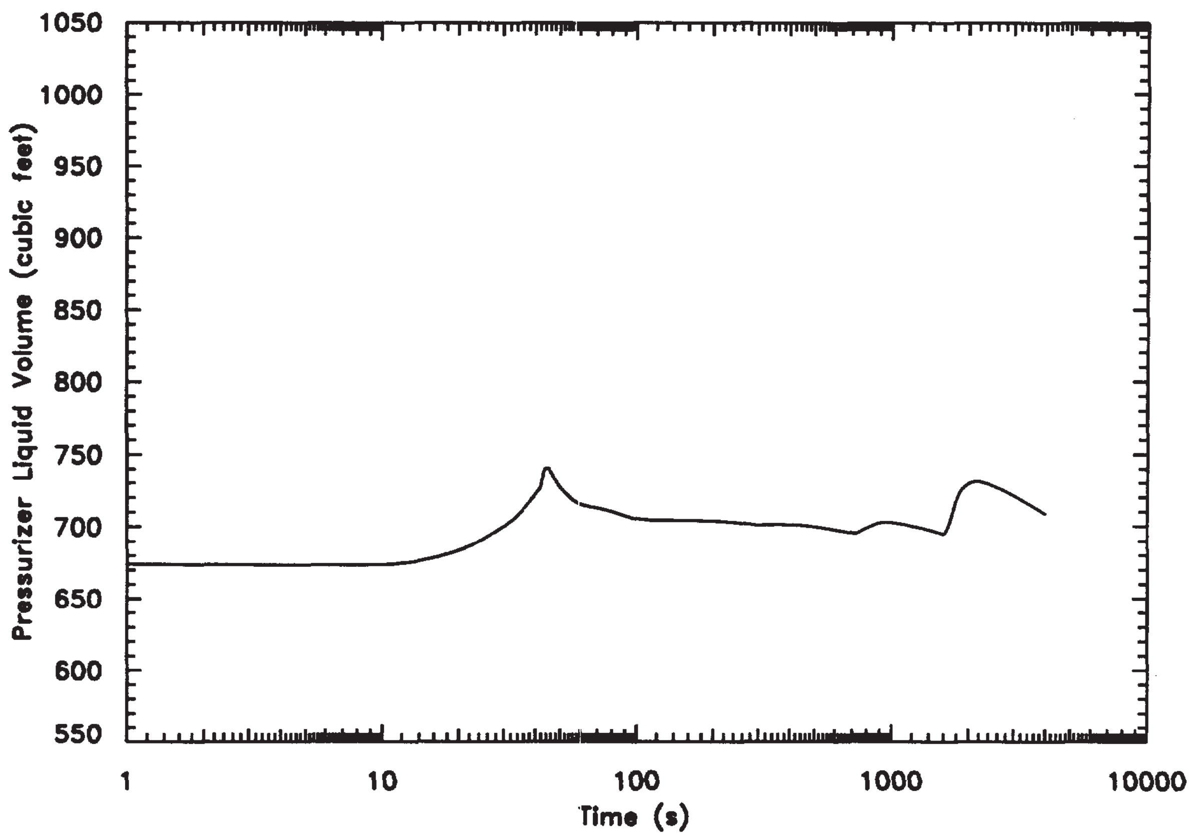
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Pressurizer Pressure	FIGURE 15.2.7-4



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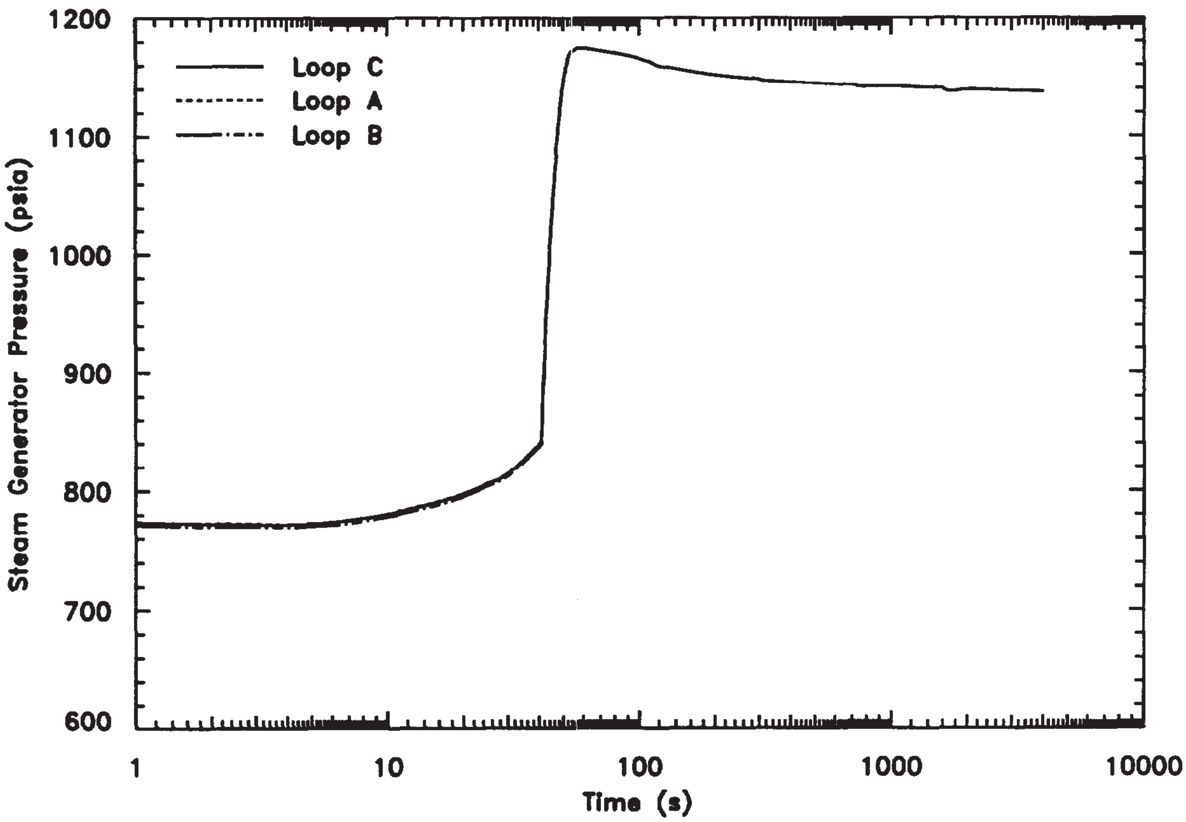
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Pressurizer Liquid Volume	FIGURE 15.2.7-5



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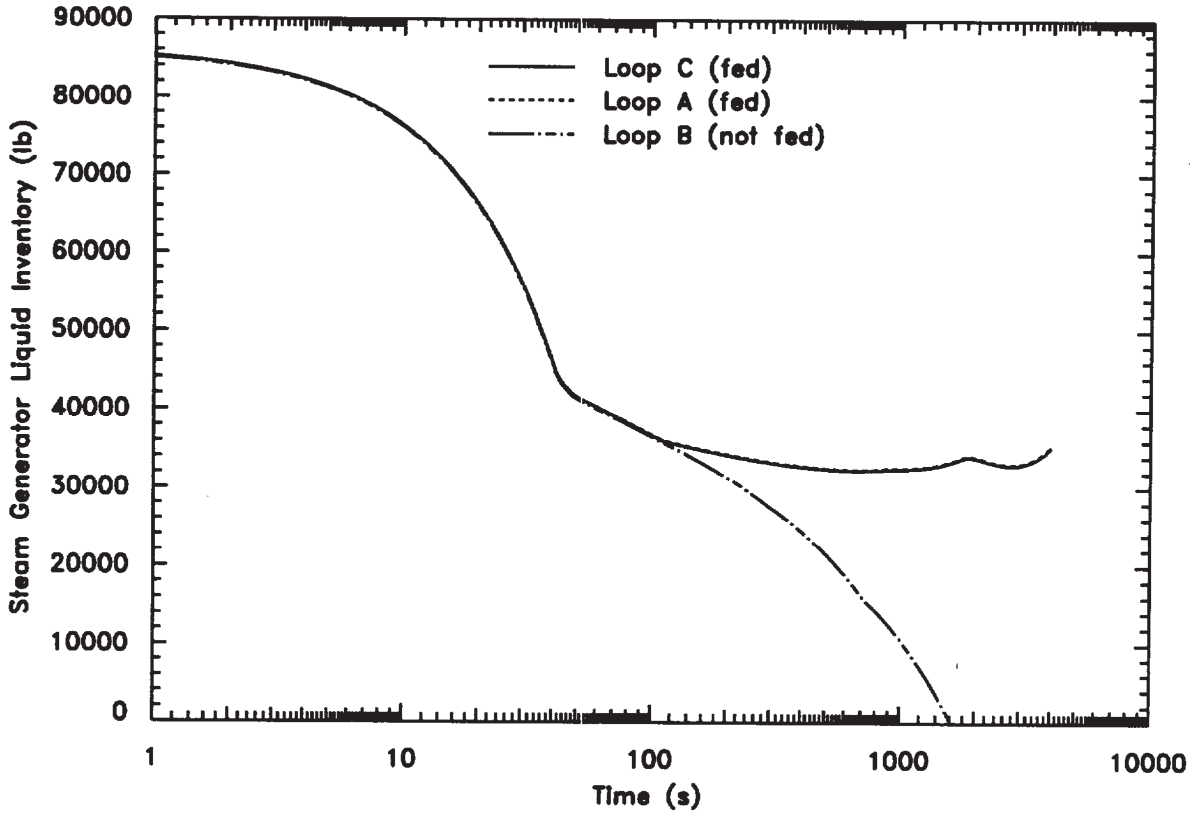
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Steam Generator Pressures	FIGURE 15.2.7-6



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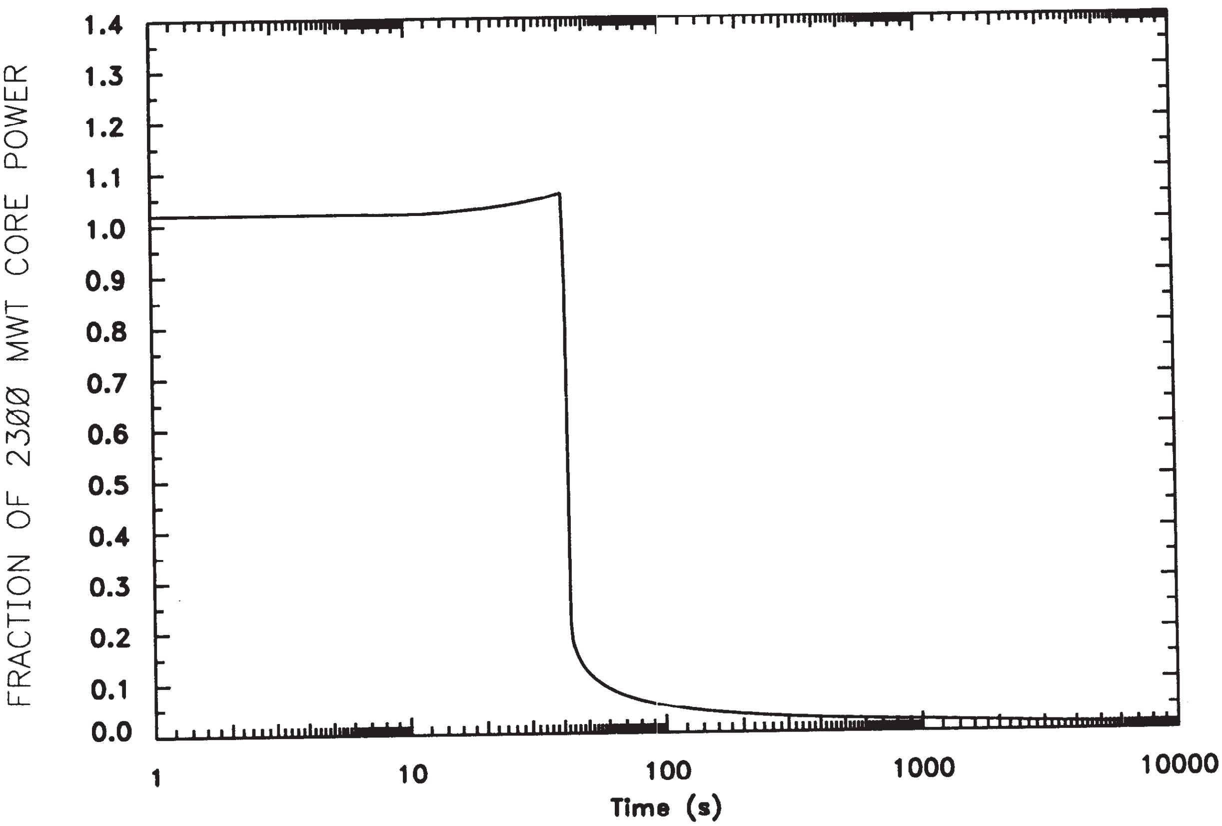
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS OFF Case: Steam Generator Liquid Inventories	FIGURE 15.2.7-7



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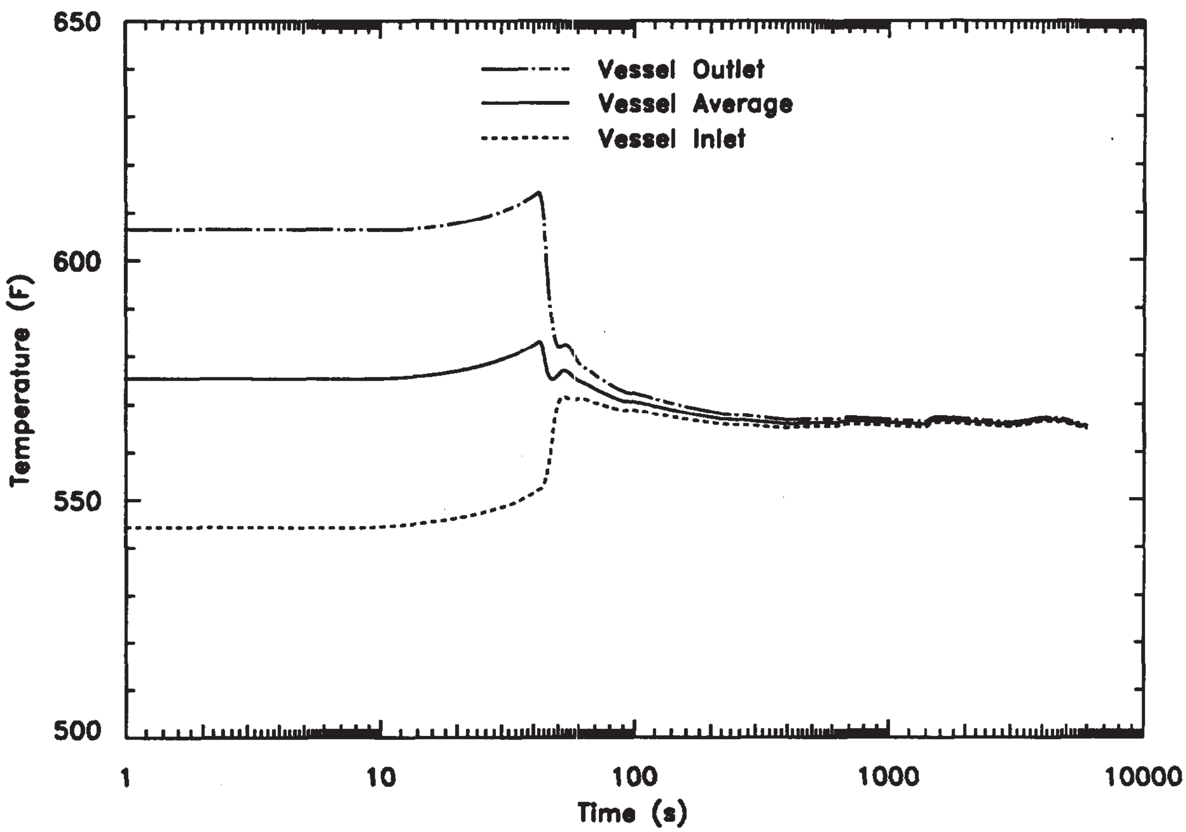
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Core Power	FIGURE 15.2.7-8



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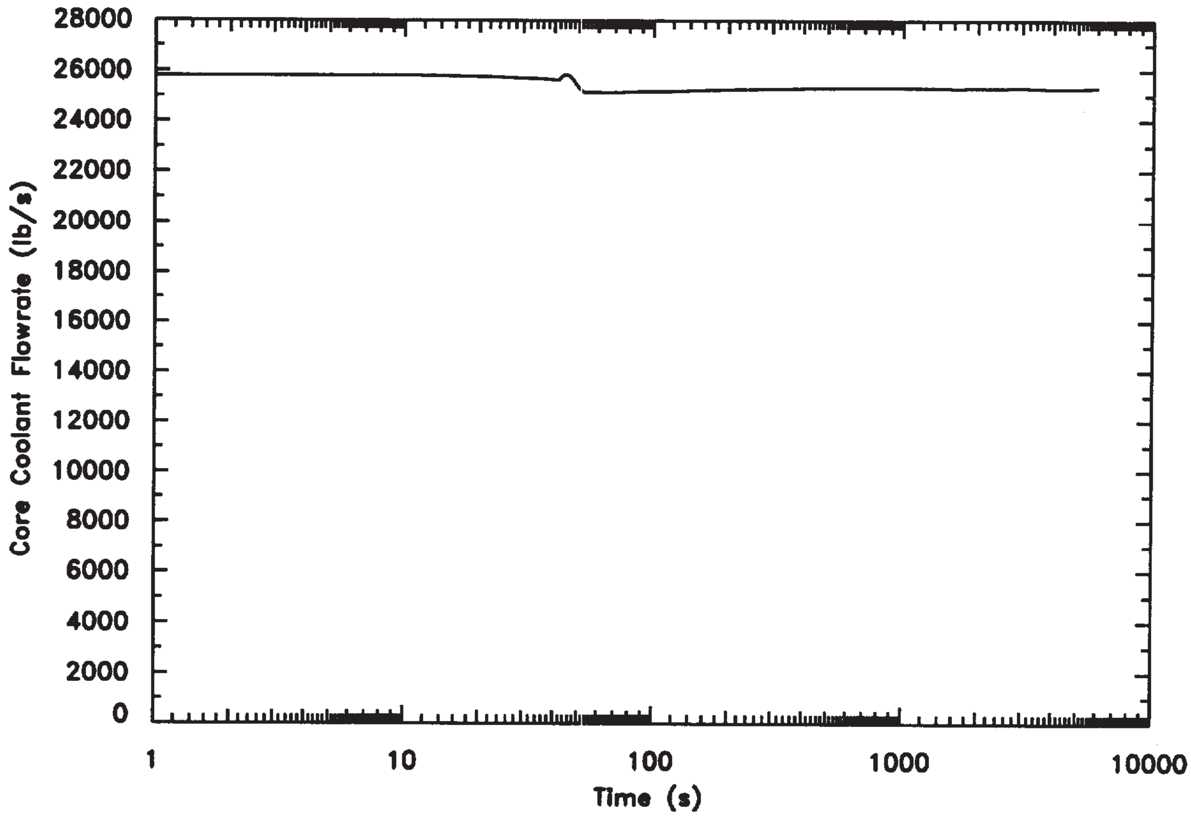
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Reactor Coolant Temperatures	FIGURE 15.2.7-9



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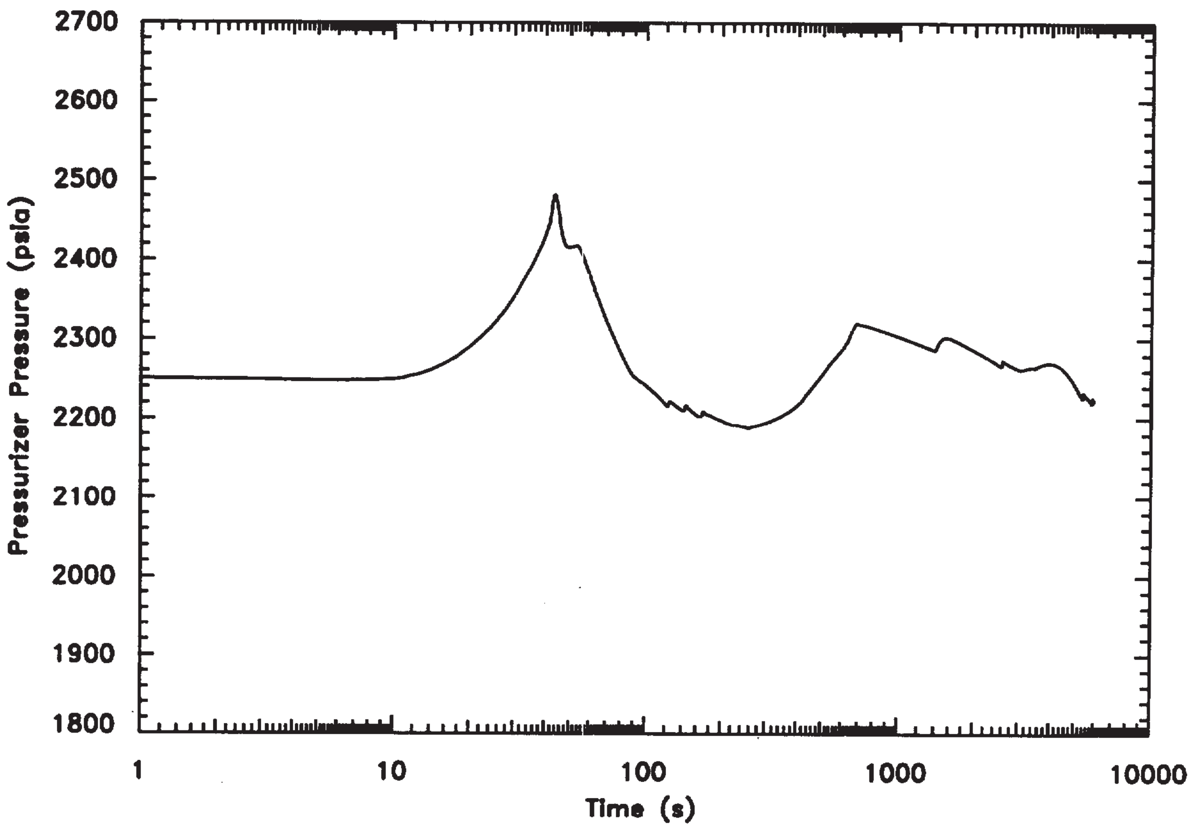
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Core Coolant Flowrate	FIGURE 15.2.7-10



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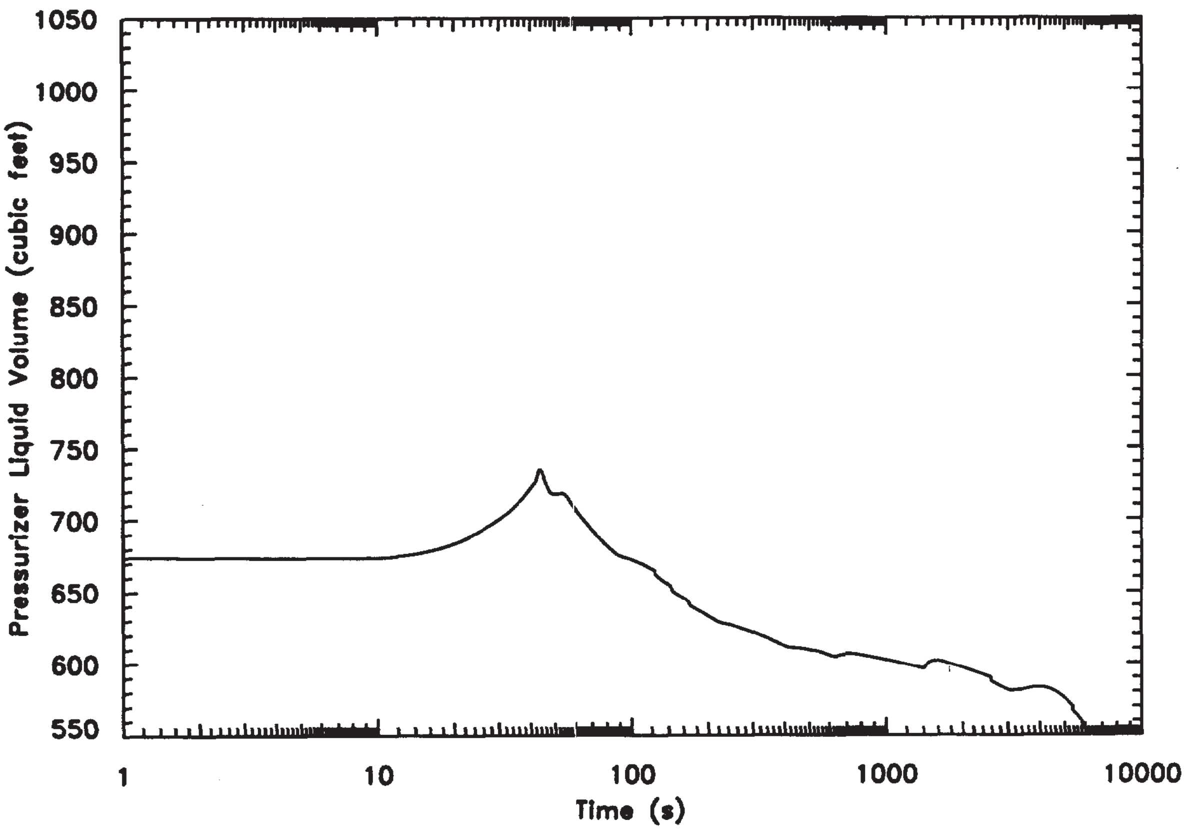
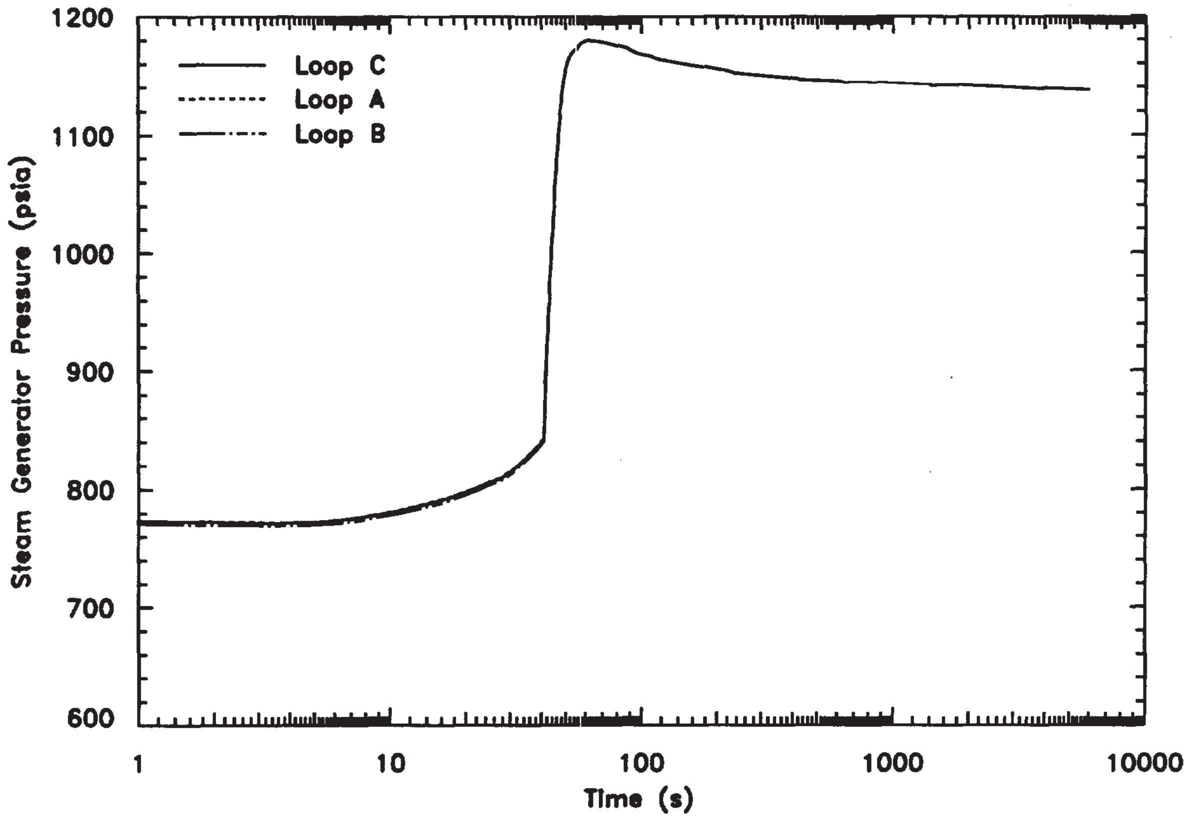
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Pressurizer Pressure	FIGURE 15.2.7-11



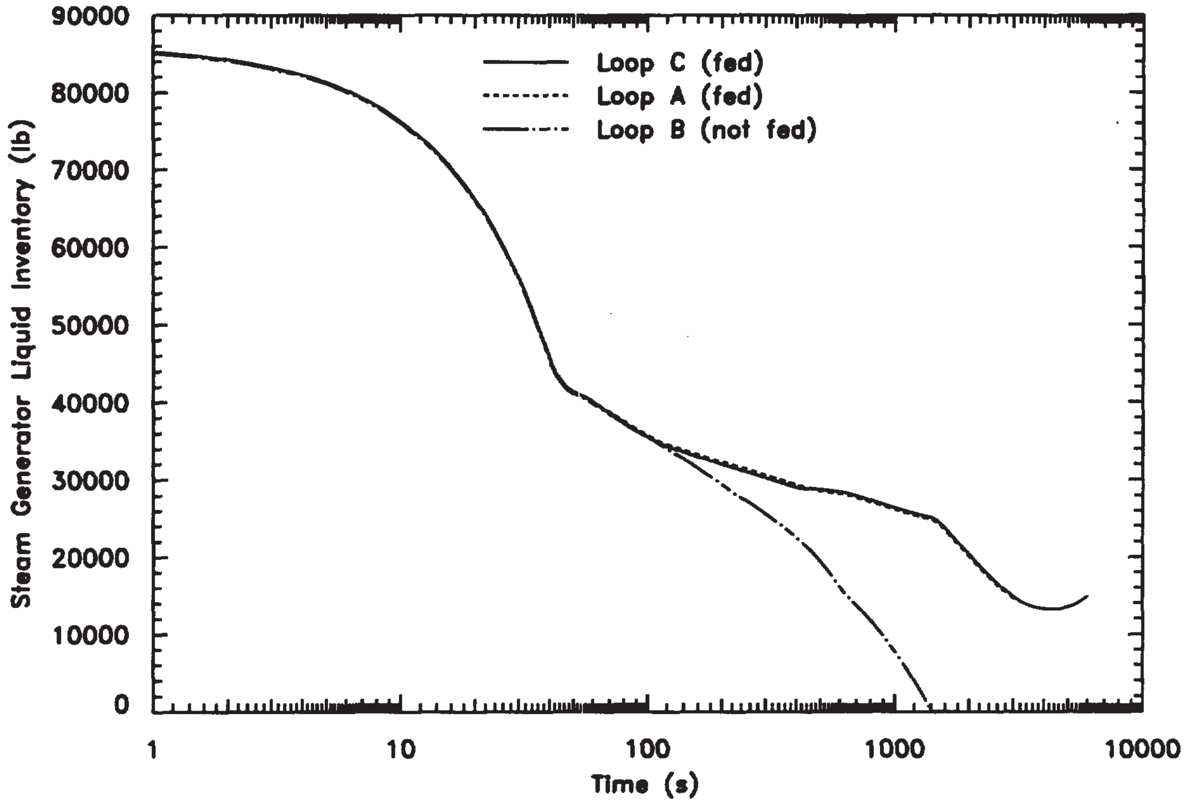
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Pressurizer Liquid Volume	FIGURE 15.2.7-12



REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Steam Generator Pressures	FIGURE 15.2.7-13



REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater, PUMPS ON Case: Steam Generator Liquid Inventories	FIGURE 15.2.7-14

15.3 Decrease in Reactor Coolant System (RCS) Flow Rate

A number of faults are postulated which could result in a decrease in RCS flow rate. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented in Section 15.3:

1. Loss of forced reactor coolant flow,

2. Reactor coolant pump shaft seizure (locked rotor), and

3. Reactor coolant pump shaft break.

15.3.1 Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Event Description

This event is characterized by a total loss of forced reactor coolant flow which is caused by the simultaneous loss of electric power to all of the reactor primary coolant pumps. Following the loss of electrical power, the reactor coolant pumps begin to coast down. The pump coastdown is governed by the pump flywheel inertia and rated torque.

If the reactor is at power when the event occurs, the loss of forced coolant flow causes the reactor coolant temperatures to rise rapidly. This results in a rapid reduction in DNB margin, and could result in DNB if the reactor is not tripped promptly. Also as the reactor coolant temperatures rise, the primary coolant expands, which cause an insurge into the pressurizer, a compression of the pressurizer steam space, and a rapid increase in reactor coolant system pressure. The primary system overpressurization will be mitigated by the action of the primary system safety valves and the reduction in core power following reactor trip.

Reactor trip signals are provided based on signals from reactor coolant pump power supply undervoltage or underfrequency and low reactor coolant loop flow. However, in the analysis, credit is taken only for the low reactor coolant loop flow trip.

The minimum DNBR is controlled by the interaction of the primary coolant flow decay and the core power decrease following reactor trip. The power to flow ratio initially increases, peaks, and then declines as the challenge to the SAFDLs is mitigated by the decline in core power due to the reactor trip. If a reactor trip can be obtained promptly, the power to flow ratio will first peak and then decrease during the transient such that the SAFDLs will no longer be challenged.

The overpressurization challenge was not evaluated in this analysis. Experience has shown that the Loss of Forced Reactor Coolant Flow event does not present the most severe challenge to the maximum pressure criterion (Reference 15.3.1-1).

After the pumps coast down, natural circulation flow is developed in the primary system and the steam generators are available to remove the decay power. Therefore, long-term cooling of the core can be achieved.

The primary concern with this event is the challenge to the SAFDLs. The purpose for analyzing this event is to verify that the reactor protection system can respond fast enough to prevent penetration of the SAFDLs.

This event is classified as a Condition II event (Table 15.0.1-1). The acceptance criteria is as described in Section 15.0.1.1. This event is the bounding decrease in reactor coolant flow rate event for the Condition II events. As cited in Table 15.0.11-1, no single failure of the ESF will affect the analysis.

15.3.1.2 Analysis Method

The overall response of the primary and secondary systems for this event was calculated by the ANF-RELAP code (Reference 15.0-3). The MDNBR for the event was calculated using the thermal hydraulic conditions from the ANF-RELAP calculation as input to the XCOBRA-IIIC code (Reference 15.0-4).

15.3.1.3 Definition of Events Analyzed and Bounding Input

This event was analyzed for a full power initial condition. The core thermal margin is at a minimum at full power. This is the bounding mode of operation for this event.

Conservative conditions were used:

Rod Control	Manual
Initial Power	102% of 2300 MWt
Pump flywheel inertia	90% of rated
Moderator temperature coefficient	+5.0 pcm/ºF
Doppler coefficient	-1.0 pcm/ºF
Core inlet temperature	Nominal
Initial Pressurizer pressure	Nominal
Core outlet pressure used in subchannel analysis	Nominal -40 psi
Pressurizer level	Nominal
Pressurizer PORVs	Available
Pressurizer spray	Not available (No pump head)
Reactor trip setpoint	low flow -3%

15.3.1.4 Analysis of Results

The transient is initiated by tripping all three primary coolant pumps. As the pumps coast down, the core flow is reduced, causing a reactor scram with rod insertion beginning at 2.7 sec.

As the flow coasts down, primary temperatures increase. The average core temperature increases about 7°F before being turned around due to the power decrease following reactor scram. This increase in temperature causes subsequent power rise due to moderator reactivity feedback as a result of the coefficient. The power peaks at about 107 percent (of 2300 MWt).

The temperature increase also causes an insurge into the pressurizer and resultant pressurization of the reactor coolant system. The pressurizer PORVs are allowed to open to minimize pressure and maximize the DNB challenge. This occurs at 4.6 sec and prevented the pressure from exceeding 2338 psia. This peak pressure occurred at 4.7 seconds. The pressure then decreased as the core power level continued to drop. The principal DNB challenge was caused by the decrease in flowrate and resultant increase in coolant temperatures.

The transient response is shown in Figure 15.3.1-1 through 15.3.1-6. An event summary is given in Table 15.3.1-1. The minimum DNBR was calculated to be 1.223 which meets the acceptance criteria of 1.141.

15.3.1.5 Conclusion

Experience has shown that this event is not the most severe challenge to the maximum pressure criterion. Substantial margin to Departure from Nucleate Boiling is calculated. Therefore, event acceptance criteria are met.

TABLE 15.3.1-1

LOSS OF FORCED COOLANT FLOW EVENT SUMMARY

Time	Event	Value
0.0 s	Three-pump coastdown was initiated	-
1.7 s	Primary loop flowrate reached low-flow trip setpoint	87% of TS min.
2.7 s	Scram rod insertion began	-
2.7 s	Core power peaked (see Figure 15.3.1-1)	107% of 2300 MWt
3.7 s	Minimum DNBR occurred (see Figure 15.3.1-6)	1.223
4.6 s	Compensated pressurizer PORV opened	+96 psi error
4.7 s	Pressurizer pressure peaked (see Figure 15.3.1-4)	2338 psia
5.0 s	Average vessel coolant temperature peaked (see Figure 15.3.1-2)	582°F

15.3.2 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.2.1 Identification of Causes and Event Description

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal.

Following reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will begin to rise. This will cause the coolant to expand and force fluid in to the pressurizer. The insurging fluid will compress the pressurizer steam space causing the pressurizer pressure to rise rapidly. Simultaneously, the reduced primary coolant flow rate will cause a reduction in the primary to secondary heat transfer which will cause the primary side steam generator outlet temperature to increase. The reduction in primary to secondary heat transfer will be further degraded as the steam generator pressures and temperatures increase following termination of steam flow on turbine trip. This overall reduction in primary to secondary heat transfer will further increase the rise in primary system temperatures and pressures. As the primary system pressure increases, the pressurizer spray and power operated relief valves would actuate to mitigate the overpressurization transient. For conservatism in evaluating the overpressurization challenge of the event, no credit is taken for the pressure relief capacity of these components. The pressurizer code safety valves will lift to relieve the primary system pressure. Eventually the primary system pressure will decrease due to the reduced core power following reactor trip.

On the secondary side, the rise in shell side steam generator pressure would normally be controlled by the operation of the steam bypass system or steam generator PORVs following closure of the turbine stop valves. However, no credit is taken for these two systems; the mechanism for removing energy from the steam generator following closure of the turbine stop valves is through the steam generator code safety valves. Energy removal through these valves will help to mitigate the primary pressure transient.

The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin. This event causes a challenge to both the specified acceptable fuel design limits and system overpressurization. The system pressurization is less severe than the Loss of External Load event presented in Section 15.2.2, so only the DNB case is presented.

This event is a Condition IV (Limiting Fault) event (Table 15.0.1-1). The acceptance criteria for this event is presented in Section 15.0.1.1. This event represents the bounding decrease in reactor coolant flow rate event for the Condition IV events. As cited in Table 15.0.11-1, no single failure in the ESF affects the analysis for this event. However, it was conservatively assumed that a loss of offsite power coincident with the turbine trip, trips the pumps in the two unaffected loops.

15.3.2.2 Analysis Method

The transient response of the primary and secondary systems is calculated using the ANF-RELAP computer code (Reference 15.0-3). The MDNBR is calculated using the ANF-RELAP conditions at time of MDNBR as input to the XCOBRA-IIIC methodology (Reference 15.0-4).

15.3.2.3 Definition of Events Analyzed and Bounding Input

The bounding operating mode for this event is full power. Conservative conditions were used:

Rod Control	Manual
Initial Power	102% of 2300 MWt
Pump flywheel inertia	90% of rated
Moderator temperature coefficient	0.0 pcm/°F
Doppler coefficient	-1.0 pcm/°F
Core inlet temperature	Nominal
Initial RCS pressure	Nominal
Core outlet pressure used in subchannel analysis	Nominal -40 psi
Pressurizer level	Nominal -10%
Pressurizer heater	Disable
Pressurizer PORVs	Available
Reactor trip setpoint	Low flow -3%

15.3.2.4 Analysis of Results

The pump in one loop seizes at initiation and the flow in the loop is abruptly reduced. This generates a low flow scram signal which results in rod insertion beginning at 1.075 sec. Turbine trip occurred at 1.10 sec., resulting in the assumed loss of power to the unaffected RCP's. The flows in the other two loops decreased slowly, as the unaffected pumps began coasting down (following the loss of offsite power at initiation). The relatively high flows in the unaffected loops maintained a substantial reactor vessel inlet-to-outlet pressure drop, which caused the flow in the affected loop to reverse at about 1.5 seconds. By 2.00 seconds, the core flowrate was about 60% of nominal. The flow reduction caused the core average coolant temperature to rise. The increasing temperatures and decreasing flowrate resulted in MDNBR occurring shortly after scram, at 2.25 seconds. The reactor pressures and temperatures were eventually reduced by heat transfer to the steam generators (mostly to the two unaffected-loop steam generators).

The transient response is shown in Figures 15.3.2-1 to 15.3.2-6. An event summary is given in Table 15.3.2-1.

The minimum DNBR for this transient is calculated to be 1.071, which is below the safety limit. Eight fuel assemblies are conservatively predicted to experience fuel clad failure, and no assemblies are predicted to experience fuel melting.

15.3.2.5 Radiological Consequences

The NRC has approved implementation of the Alternative Source Term methodology (Reference 15.0.12-3) for analysis of the radiological consequences of this event (Reference 15.3.2-1).

An instantaneous seizure of a reactor coolant pump is assumed to occur, rapidly reducing flow through the affected reactor coolant loop. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through either the steam generator atmospheric relief valves or safety valves. In addition, radioactivity is contained in the primary and secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

This event can result in fuel damage. In order to bound the maximum number of fuel assemblies expected to experience fuel clad damage, this analysis conservatively assumes that 17 assemblies experience clad damage, but no fuel melting is assumed to occur.

Other assumptions used in the radiological consequence analysis:

- Loss of offsite power at the time of reactor trip. This drives the release from the secondary coolant system through the SG relief valves, since condenser cooling is lost.
- Maximum radial peaking factor is 1.8.
- 157 fuel assemblies in the core.
- For the 17 damaged fuel assemblies, the activity released from the fuel clad failure is based on the following gap inventory fractions:

Krypton 85:	10%
Other Noble Gases:	5%
Iodine 131:	8%
Other Halogens:	5%
Alkali Metals (Cs, Rb):	12%

  All gap activity in the damaged fuel rods is instantaneously released.
- The chemical form of the radioiodine released from the damaged fuel is 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.
- The minimum volume (hot) of the reactor coolant system is 8,254 ft^3, based on a temperature of 575.9°F and a pressure of 2235 psig.
- The primary-to-secondary leakage to the steam generators mixes instantaneously and homogeneously with the secondary water without flashing.

- RCS equilibrium activity concentration is conservatively assumed to be twice the values in Table 15.0.12-2. Since fuel damage is assumed, no iodine spiking is assumed for the equilibrium RCS activity.
- The primary-to-secondary leak rate is limited to 0.3 gpm total through the 3 steam generators which bounds TS limit of 75 gpd per S/G.
- The minimum volume of the secondary side coolant is 88,641 lbs per steam generator.
- The integrated mass of steam released from the steam generators as a function of time is 301,967.3 lbm (0 – 2 hours), 868,735.6 lbm (0 – 8 hours), 1,993,731.4 lbm (0 – 24 hours), and 3,631,641.5 lbm (0 – 53.2 hours).
- The halogen and alkali metal partition coefficient for the steam generators is 100.
- Iodine releases from the steam generators to the environment are 97% elemental and 3% organic.
- All noble gas radionuclides released from the primary system are released to the environment without holdup, reduction, or mitigation.
- The time required for one train of the RHR System to establish adequate shutdown cooling to terminate releases from the steam generators is 53.2 hours.

15.3.2.6 Conclusion

For the locked rotor accident, the 2-hour dose at the EAB is 2.24 rem TEDE. The dose at the LPZ is 0.21 rem TEDE. The Control Room doses at inleakages of 300 and 500 cfm are 0.86 and 1.43 rem TEDE, respectively.

The offsite dose acceptance criterion established by Reference 15.0.12-3 for this accident is that doses should be less than 10% of the 10 CFR 50.67 guidelines, or less than 2.5 rem TEDE. The Control Room dose acceptance criterion established by 10 CFR 50.67 for this accident is 5 rem TEDE.

Therefore, the offsite and Control Room TEDE doses due to a locked rotor event meet the dose acceptance criteria.

TABLE 15.3.2-1

LOCKED ROTOR EVENT SUMMARY - DNBR

Time	Event	Value
0.00 s	Single primary coolant pump seized	-
0.075 s	Affected-loop flow reached low-flow trip setpoint	87% of TS min.
1.075 s	Scram rod insertion began	-
1.10 s	Turbine trip	
1.10 s	Assumed LOOP and unaffected RCP's trip	
1.50 s	Affected-loop flow reversed	-
2.25 s	Minimum DNBR occurred (see Figure 15.3.2-6)	1.071

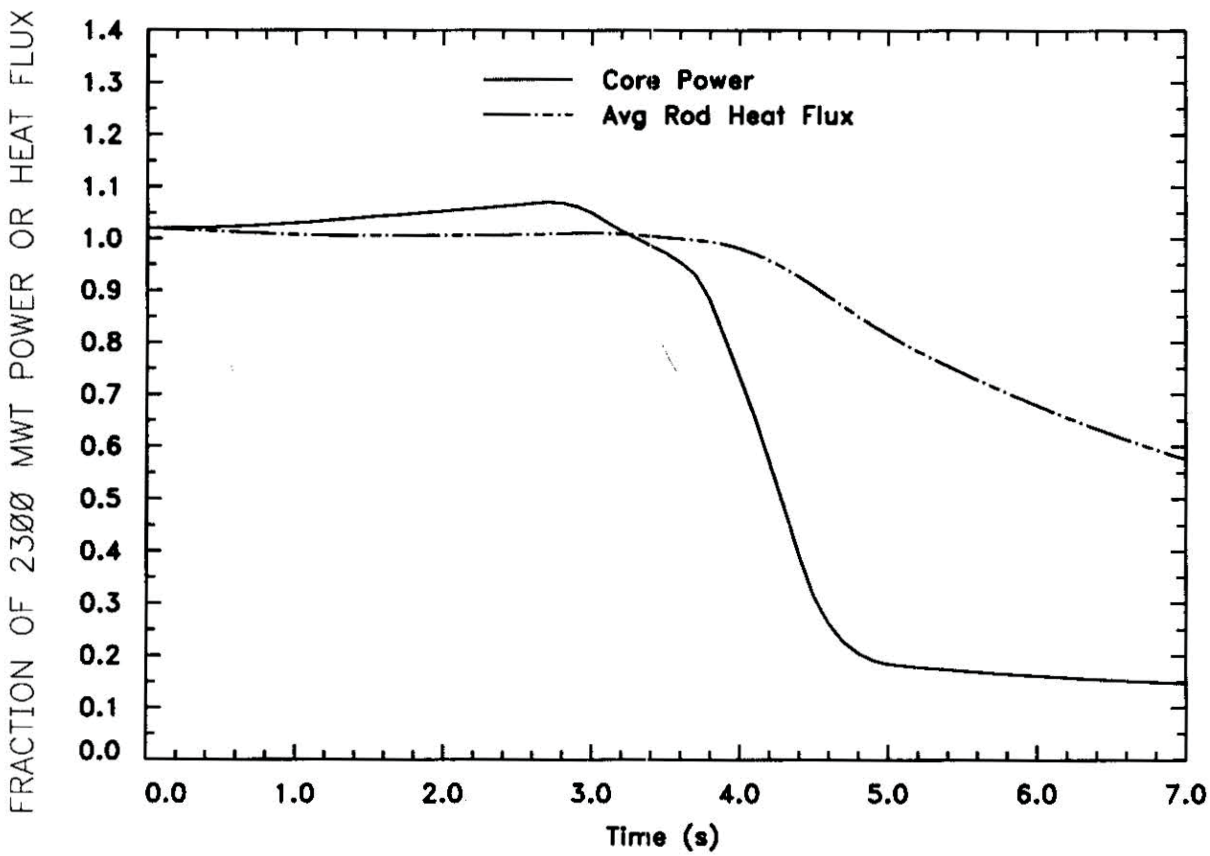
15.3.3 REACTOR COOLANT PUMP SHAFT BREAK

This event is postulated to be caused by the instant severance of the pump impeller shaft. The reactor trips on low flow slightly later than for the pump seizure event because of the higher flow associated with a free-wheeling impeller in comparison to a locked impeller. The reverse flow associated with a free-wheeling impeller in reverse direction is larger, but MDNBR occurs prior to significant flow reversal due to momentum effects. The results of this event are bounded by those of the pump seizure event. This disposition is not impacted by power uprate to 2339 MWt.

REFERENCES: SECTION 15.3

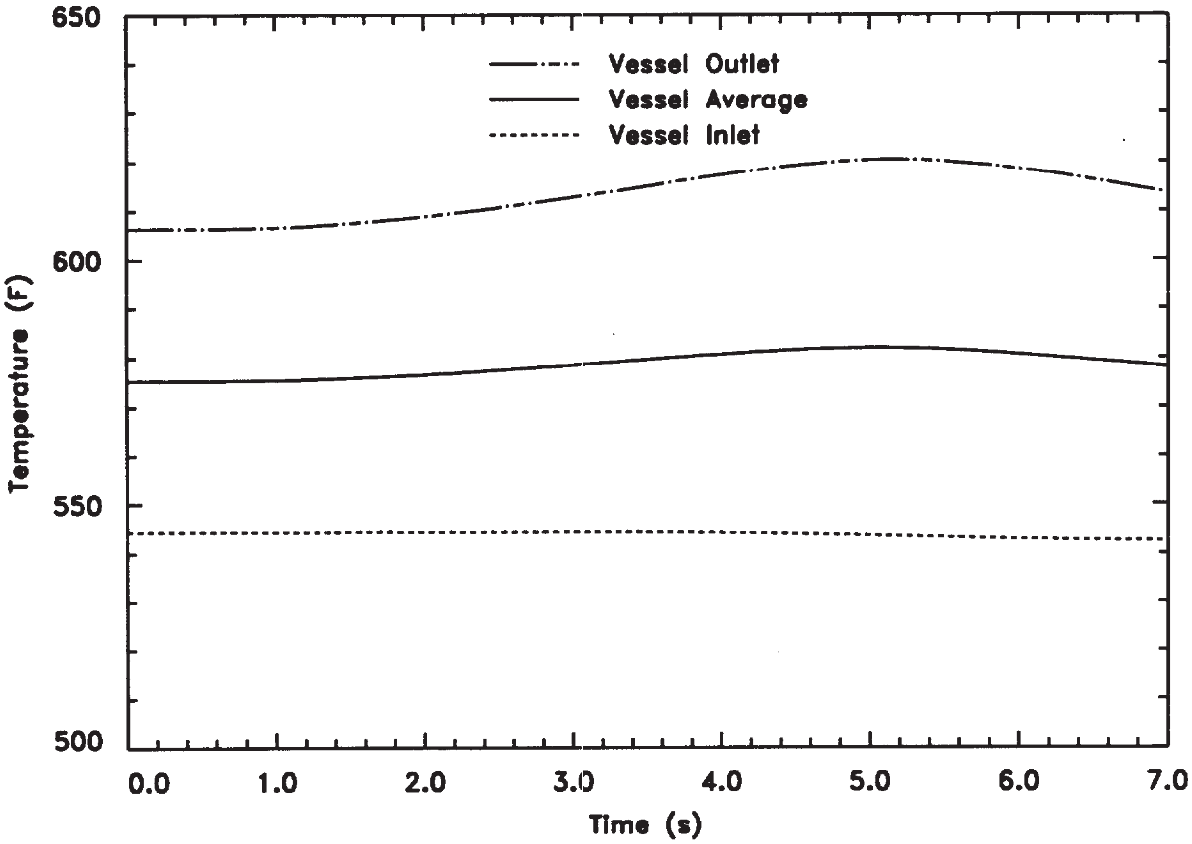
15.3.1-1 "Plant Transient Analysis for H. B. Robinson Unit 2 at 2300 MWt with Increased FΔH," XN-NF-84-74(P), Revision 1, Exxon Nuclear Company, April 1986.

15.3.2-1 NRC Letter dated September 24, 2004, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105)."



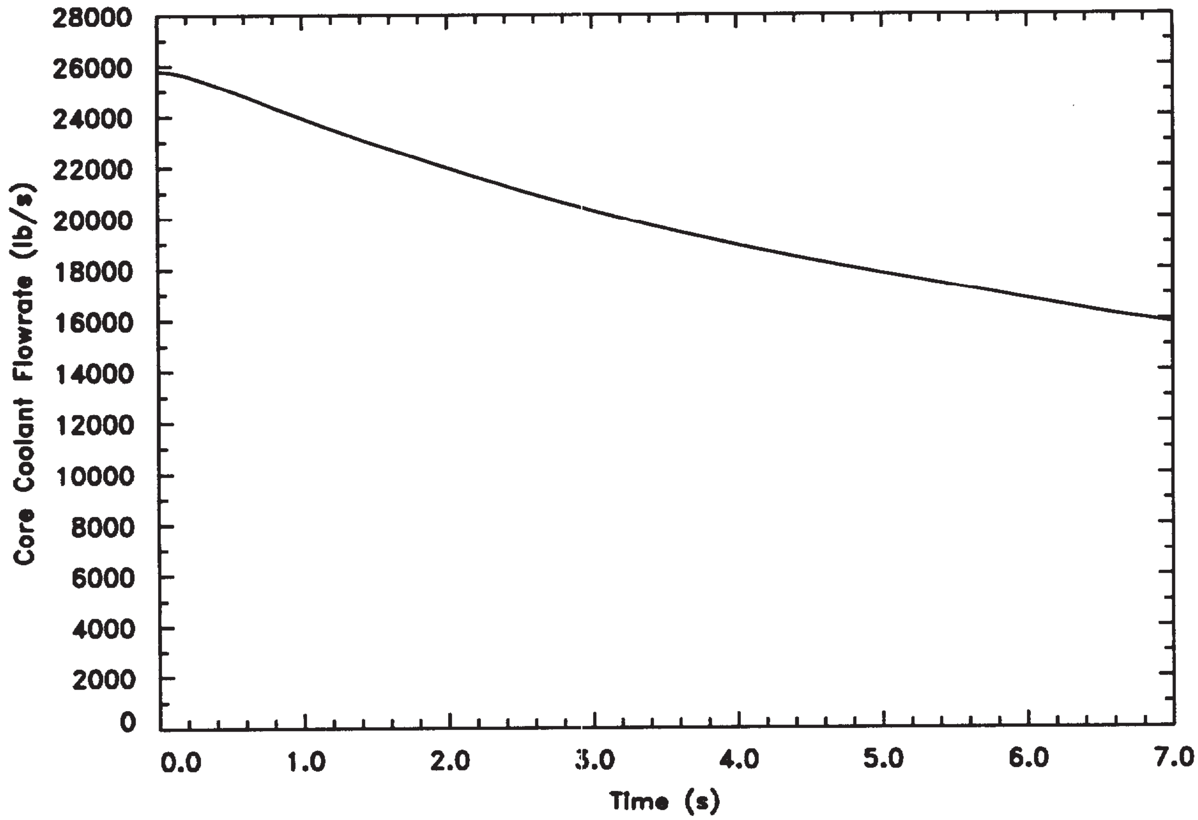
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: Core Power and Average Rod Heat Flux	FIGURE 15.3.1-1



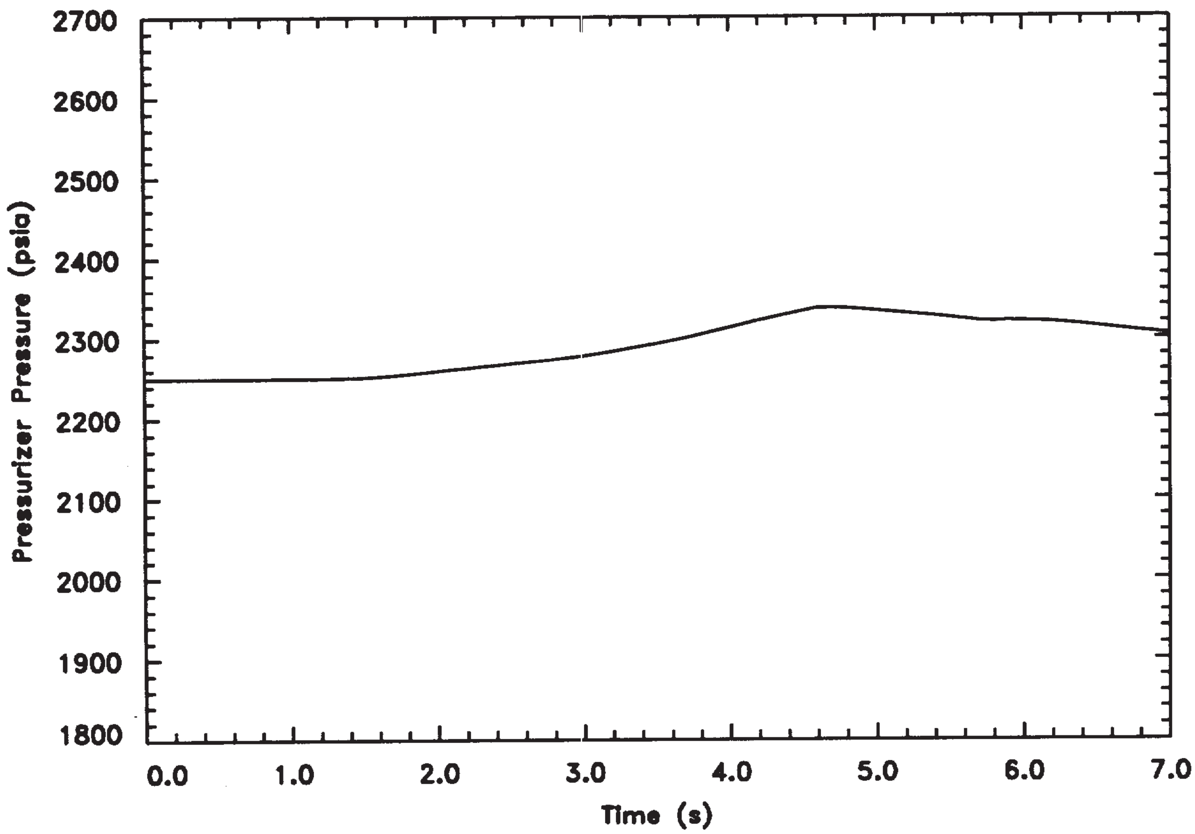
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: Reactor Coolant Temperatures	FIGURE 15.3.1-2



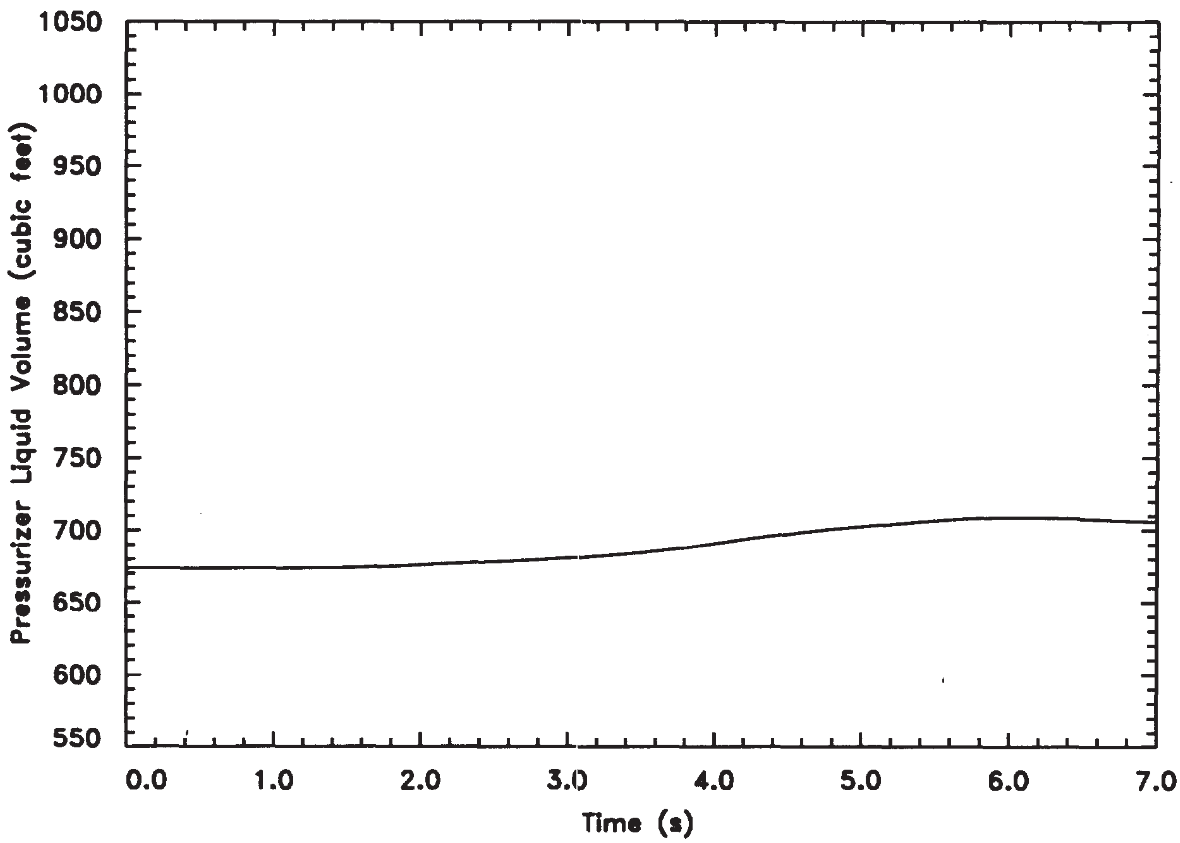
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: Core Coolant Flowrate	FIGURE 15.3.1-3



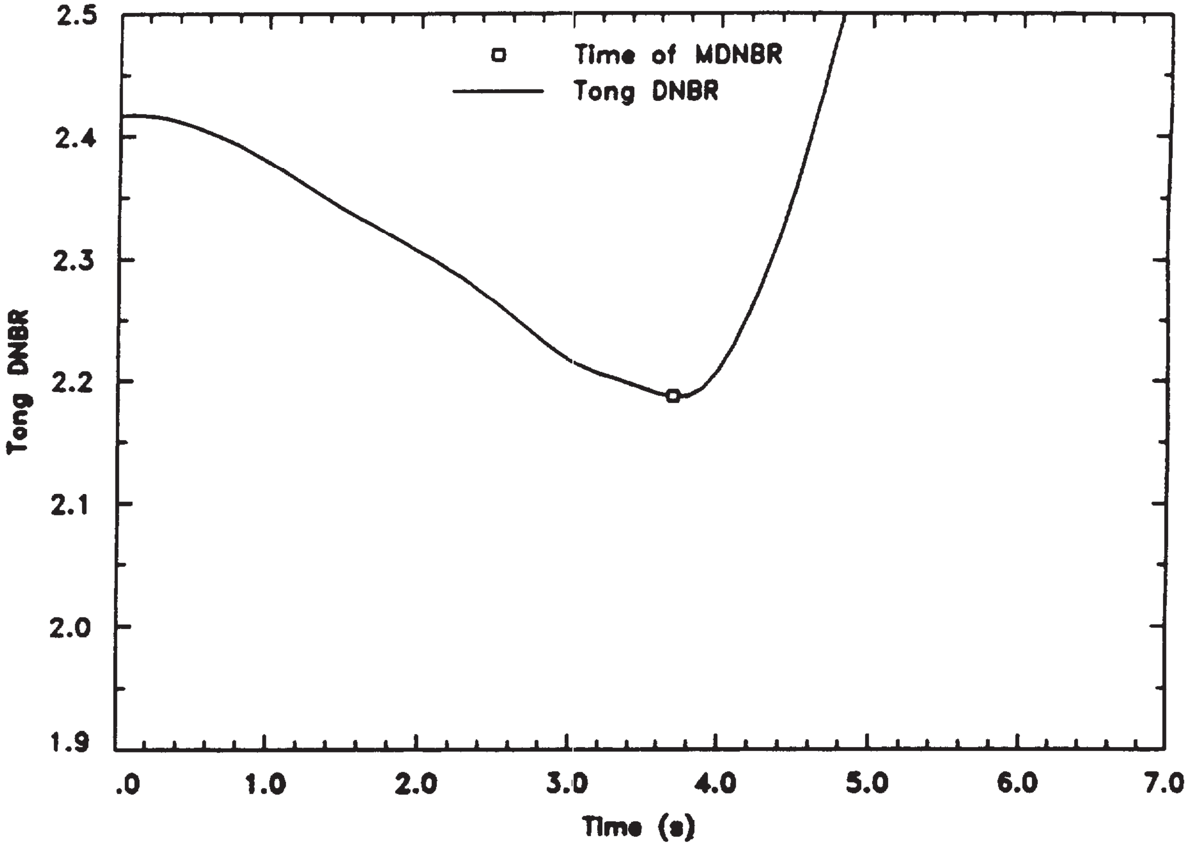
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: Pressurizer Pressure	FIGURE 15.3.1-4



REVISION NO. 13

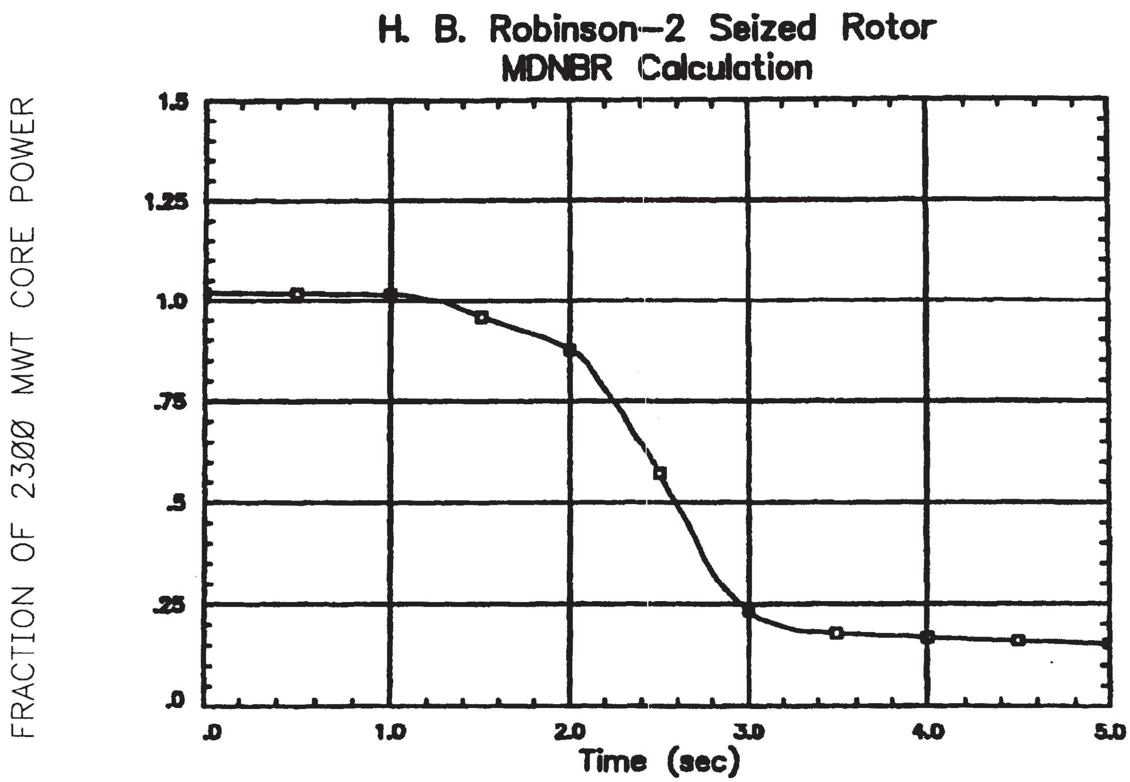
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: Pressurizer Liquid Volume	FIGURE 15.3.1-5



Note: Tong DNBR value is a RELAP calculated core-wide DNBR trend used to estimate approximate time of minimum DNBR to establish core parameters for input into the XCOBRA IIIC code to evaluate MDNBR.

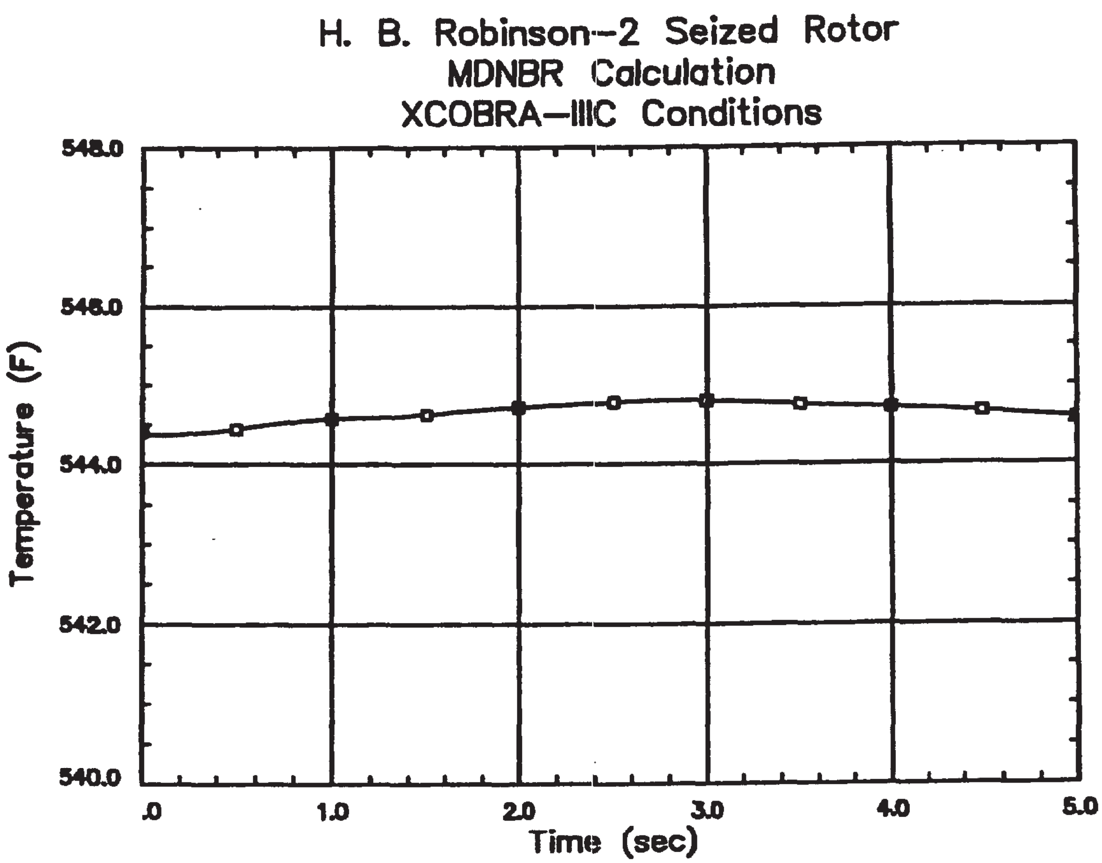
REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Forced Coolant Flow: DNBR Trend	FIGURE 15.3.1-6



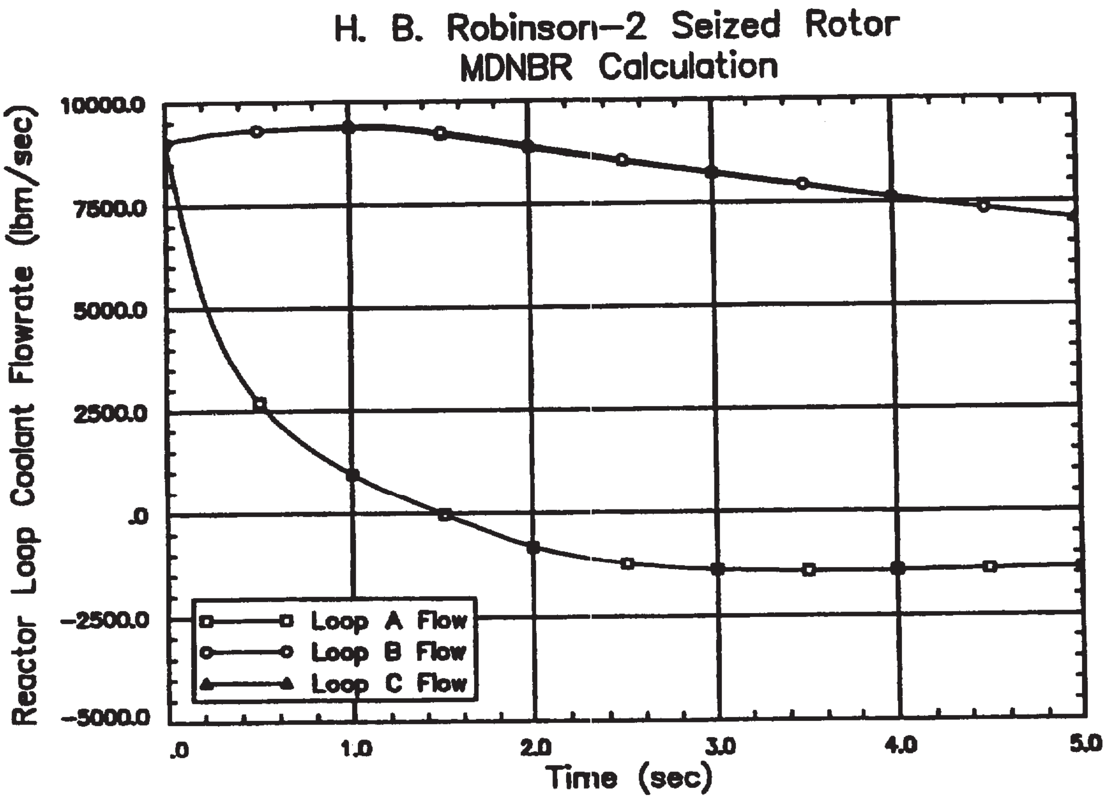
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: Core Power	FIGURE 15.3.2-1



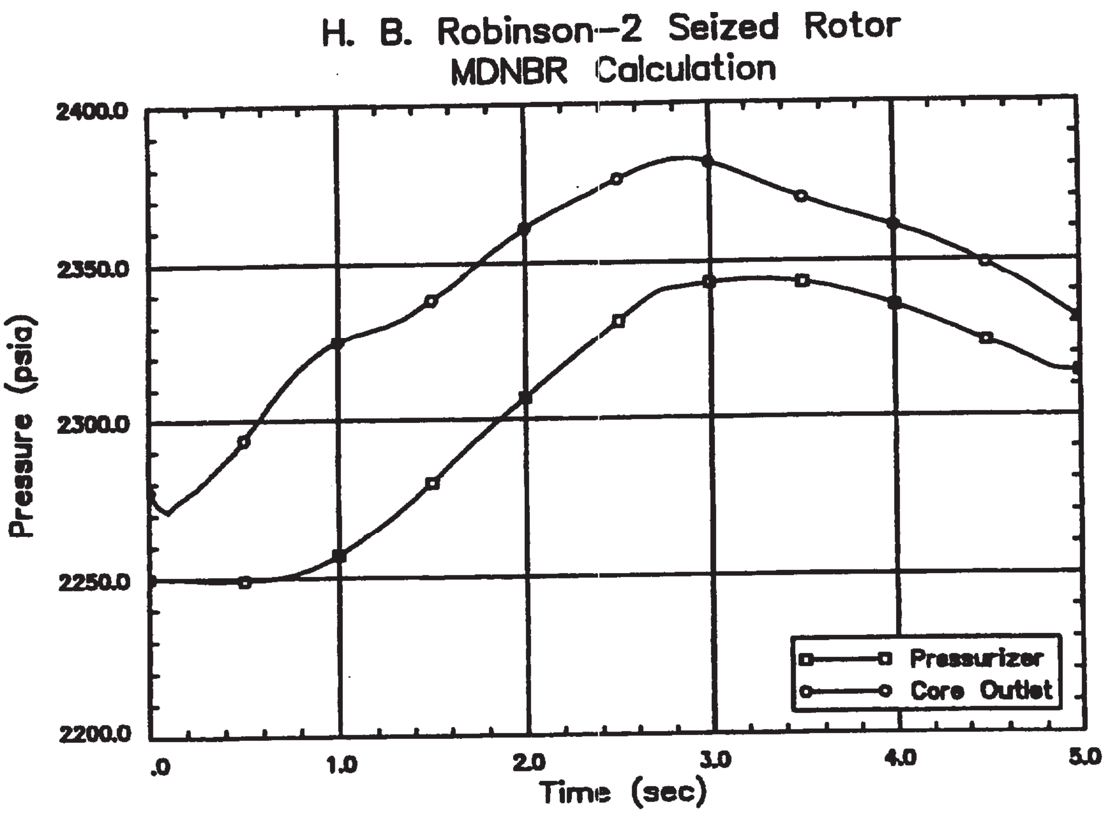
REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: Reactor Coolant Temperatures	FIGURE 15.3.2-2



REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: Primary Loop Coolant Flowrates	FIGURE 15.3.2-3



REVISION NO. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: Pressurizer Pressure	FIGURE 15.3.2-4

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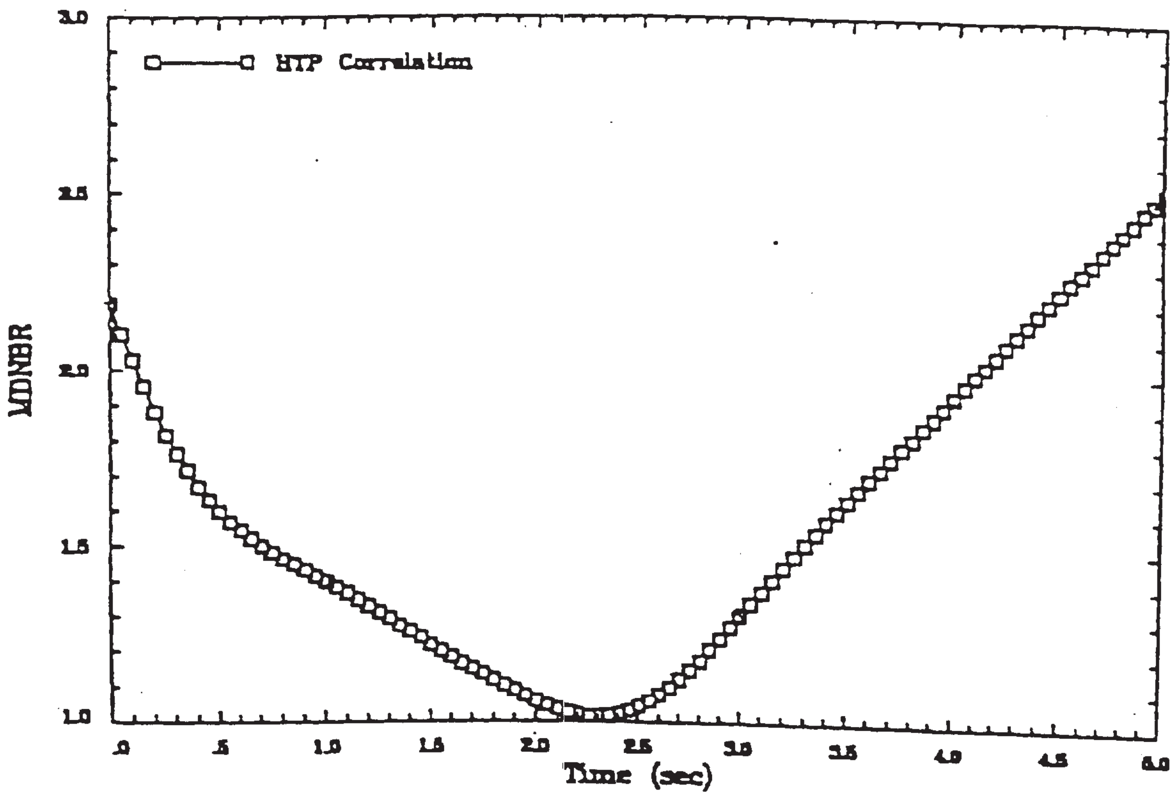
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: Pressurizer Liquid Volume	FIGURE 15.3.2-5



Note: MDNBR value is a RELAP calculated core-wide DNBR trend used to estimate approximate time of DNBR to establish core parameters for input into the XCOBRA-IIIC code to evaluate MDNBR.

REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Coolant Pump Shaft Seizure: DNBR Trend	FIGURE 15.3.2-6

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From Subcritical or Low Power

15.4.1.1 Identification of Causes and Event Description

This event is defined to result from an uncontrolled control rod bank withdrawal from subcritical or low power. The event could be caused by a control system malfunction. The malfunction could result in a rapid and large reactivity insertion, which is terminated by the low range setting of the power range flux trip. The maximum insertion rate is determined from the bounding worth of rod banks which are wired in common together with a bounding control rod withdrawal rate.

The reactivity insertion rate is rapid enough that very high neutron powers are calculated, but of short enough duration that excessive energy deposition does not occur. Rod surface heat flux approaches a significant fraction of full power. As the event can be very rapid, primary coolant temperature lags behind power. The reactivity insertion rate is initially countered by the fuel Doppler followed by trip and rod insertion.

There are four safety mechanisms which limit this event. These are:

1. Source range flux trip
2. Intermediate range flux trip
3. Intermediate range rod stop
4. Power range trip (low setting)

The source and intermediate range trips are bypassed when permissives are reached before reaching the respective trip setpoints. For events that are initiated below the point where the Power Range trips are required to be operable (i.e., Modes 3, 4, and 5), the Source Range flux trip is credited to mitigate the event in accordance with the licensing basis (Reference 15.4.1-2). If the event is initiated below the point where the source range trip is bypassed, the event is bounded by the Hot Zero Power condition. The power range (low setting) trip is set at 24% of 2339 MWt.

Initial power levels ranging up to 2% of 2300 MWt were considered for this event. Higher initial powers ranging to 2346 MWt are analyzed in Section 15.4.2.

The objective of this analysis is to bound plant operational modes below approximately 2 percent of 2300 MWt to where the operational state (shutdown margin greater than or equal to 1.77% at end of cycle, 1% at beginning of cycle) precludes return to power in an anticipated operational occurrence. The analysis examined the possible operational modes and state conditions between these two limits to develop a bounding case.

The event is classified as a Condition II event (Table 15.0.1-1). The acceptance criteria is as described in Table 15.0.1-1 with the addition of fuel centerline melt criterion. For this analysis, the systems challenged in this event are redundant; no single active failure will adversely affect the consequences of the event.

15.4.1.2 Analysis Method

The analysis is performed using the ANF-RELAP code and XCOBRA-IIIC. The ANF-RELAP code models the salient system components and calculates neutron power, fuel thermal response, surface heat transport and fluid conditions, including coolant flow rate, temperature and primary pressure. An approximate DNB calculation is performed to identify the time and parametrically the fluid conditions for which DNBR is minimum. The fluid boundary conditions and rod surface heat flux at the time of MDNBR are then transposed to the XCOBRA-IIIC methodology (Reference 15.0-4).

15.4.1.3 Definition of Events Analyzed and Bounding Input

One case was analyzed for DNB and fuel centerline melt. The case input and initial conditions bound hot shutdown and startup modes. The lowest initial power yields the maximum margin to trip, and hence maximum time for withdrawal to trip. This yields the largest prompt multiplication which maximizes overshoot past trip. The initial power conservatively bounds the initial power possible in hot shutdown and startup operation. Maximum coolant temperature for the mode of operation minimizes DNBR and is, therefore, appropriate. The bias selection for the pellet to cladding heat transfer coefficient minimizes Doppler feedback.

Maximization of power peaking and minimization of core flow rate reduce DNBR. The use of two primary coolant pumps appropriately represents the operational mode and results will bound those with 3 pump operation. If less than two reactor coolant pumps are operating, this event is mitigated by maintaining 6% Shutdown Margin or by making the rod control withdrawal system not capable of withdrawal. A boron concentration that provides 6% Shutdown Margin will be sufficient to keep the reactor subcritical assuming an initial ARI-MRR condition with Control Banks A, B, C, and D being fully withdrawn in overlap. A boron concentration that provides shutdown margin of 6% will also be sufficient to keep the reactor subcritical assuming an initial ARI MRR condition with any one shutdown bank SA or SB being fully withdrawn.

Consistent beginning of cycle parameters are used as this minimizes Doppler and provides maximum positive moderator coefficient which provides positive feedback for an increasing coolant temperature.

Conservative conditions are established for the analysis:

Initial Power	10^{-9} x 2300 MWt
Primary Coolant Pumps Operating	2
Reactivity Insertion Rate	Maximum differential bank worth for banks wired in common
Radial Power Distribution	Hot Zero Power
Axial Power Distribution	See Figure 15.4.1-1
Moderator Temp. Coefficient	+5.0 pcm/°F
Doppler	0.8 x [Bounding Temperature Dependent BOC Value] See Reference 15.4.1-1
Pellet to Clad HTC	Maximum core-average BOC value

15.4.1.4 Analysis of Results

The event is initiated with control bank withdrawal. At approximately 15.7 sec reactor power reached 1% of 2300 MWt. The peak nuclear power of 184% of 2300 MWt is reached at 16.3 seconds. The rapid power increase results in a fuel temperature increase which produces negative Doppler reactivity which first reduces power. The trip signal occurs at 16.0 sec on the high flux (low setting) trip with rod insertion beginning at 16.5 seconds. A peak core-average surface heat flux equivalent to 51% of 2300 MWt occurs at 17.8 seconds. This results in a maximum LHGR less than that for fuel centerline melt. The minimum DNB ratio calculated for this event was 2.284. A summary of sequence of events is presented in Table 15.4.1-1. Neutron power, rod surface heat flux and fuel rod temperature as a function of time are presented in Figures 15.4.1-2 and 15.4.1-3.

15.4.1.5 Conclusions

The analysis demonstrated that the SAFDLs are not penetrated and, therefore, event acceptance criteria are met.

TABLE 15.4.1-1

BANK WITHDRAWAL FROM SUBCRITICAL EVENT SUMMARY

TIME	EVENT	VALUE
0.0 s	Bank withdrawal began	-
16.0 s	Core power reached high-flux trip setpoint (see Figure 15.4.1-2)	35% of 2300 MWt
16.3 s	Core power peaked (see Figure 15.4.1-2)	184% of 2300 MWt
16.5 s	Scram rod insertion began	-
17.8 s	Core-average rod surface heat flux peaked (see Figure 15.4.1-2)	51% of heat flux corresponding to 2300 MWt
17.8 s	Minimum DNBR occurred (see Figure 15.4.1-4)	2.284
18.0 s	Core-average fuel temperature peaked (see Figure 15.4.1-3)	867°F
18.2 s	Hot rod centerline temperature peaked	2598°F
20.0 s	Average vessel coolant temperature peaked (see Figure 15.4.1-3)	564°F

15.4.2 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Event Description

This event is defined to result from an uncontrolled control bank withdrawal at power. The power range to be considered is from 2 percent of 2300 MWt to rated power. The event could be caused by misoperation of the most reactive control rod banks wired in common withdrawing at up to the maximum rate.

The reactor protection trip system is designed and set to preclude penetration of the SAFDLs. Because of the design of this analysis, the overtemperature ΔT and power range (high setting) high flux trips are principally challenged. Both trip setpoints include allowance for process variable measurement, processing channel drift, and operating variances from that indicated.

The overtemperature ΔT function is designed and set to protect against DNB. Principal DNB parameters such as power (measured as core coolant temperature rise), core coolant temperature, primary pressure and core power distribution are measured, and the function decreases margin to trip setpoint when process variables indicate a decrease in operating margin. This function is established based on the core protection boundaries, operation within which assures protection of the SAFDLs.

For the maximum possible reactivity insertion rates, the core temperature rise lags behind nuclear power. The power range reactor trip protects the system from these events.

A broad range of reactivity insertion rates and initial operating conditions are possible. The range of reactivity insertion is from very slow, as would be associated with a gradual boron dilution, and bounded on the fast end of the range by bank withdrawal.

The objective of the analysis is to demonstrate the adequacy of the trip setpoints to assure meeting the acceptance criteria. To assure this objective, the analysis is performed for a spectrum of reactivity insertion rates and initial powers. Since neutronic feedback is a function of cycle exposure and design, these effects are also included in the analysis.

Each transient in the spectrum of cases analyzed is characterized by the following sequence of events:

1. Reactivity is inserted.

2. Core power ascends.

3. Clad heat flux increases, lagging behind the core power ascent.

4. Primary coolant temperatures increase.

5. The reactor trips on core temperature rise or high neutron flux.

This event is classified as a Condition II event (Table 15.0.1-1).

The acceptance criteria are as described in Section 15.0.1.1 with the added condition of fuel centerline melt criteria. The systems challenged in this event are redundant and no single active failure will adversely affect the consequences of the event.

Maximum RCS pressure is bounded by Loss of External Load in Section 15.2.2 because the secondary system is isolated.

15.4.2.2 Analysis Method

The analysis is performed using the S-RELAP5 code and XCOBRA-IIIC. The S-RELAP5 code models the salient system components and calculates neutron power, fuel thermal response, and fluid conditions. The fluid conditions and rod surface heat transport at the time of MDNBR are transposed to be XCOBRA-IIIC methodology (Reference 15.0-4) for calculation of the MDNBR.

Systems which minimize DNBR are enabled in the analysis. These include (e.g.) pressurizer spray and PORVs.

15.4.2.3 Definition of Events Analyzed and Bounding Input

The analysis bounds power operation. Two case series are analyzed: one for negative and the other for positive neutronic feedback.

Case Series	Initial Power	Reactivity Rate	Neutronics
1	100.3% of RTP	Low to high	Neg. Feedback
	60.3% of RTP	Low to high	Neg. Feedback
2	100.3% of RTP	Low to high	Pos. Feedback
	60.3% of RTP	Low to high	Pos. Feedback
	10.3% of RTP	Low to High	Pos. Feedback

Conservative conditions are established for analysis of each subevent.

Control	Manual
Core power	100.3% of 2339 MWt
Core coolant inlet temperature	Nominal
Initial RCS pressure	Nominal
Core outlet pressure used in subchannel analysis	Nom. -40 psi
Pressurizer spray	Available
Reactor coolant system flow rate	Minimum allowed by Tech Specs
Pressurizer PORVs	Available
Pressurizer level	Nominal
Steam bypass	Disable

Steamline PORVs	Disable	
Reactor Trips	OT-ΔT Power Range high flux (high)	
Reactivity insertion rate	Maximum to very low	
	<u>Max. Pos.</u>	<u>Max. Neg.</u>
Moderator temperature coefficient	+5 pcm/°F	-45 pcm/°F
Doppler coefficient	-0.9 pcm/°F	-1.75 pcm/°F

The maximum reactivity insertion rate used bounds the most reactive banks wired in common withdrawing at maximum rate. The minimum reactivity insertion rate used is typical of boron dilution.

15.4.2.4 Analysis of Results

For full power and 60% power, a spectrum of reactivity insertion rates ranging from 0.1 pcm/sec to 50 pcm/sec were analyzed for both positive reactivity feedback and negative reactivity feedback. For 10% power, reactivity ranged from 5 pcm/sec to 50 pcm/sec with positive (BOC) feedback.

The limiting rod bank withdrawal event is from 10% power initial conditions with an insertion ramp of 6.8 pcm/sec and positive reactivity feedback.

Figures 15.4.2-1, 15.4.2-2, and 15.4.2-2a present MDNBR vs. Reactivity Insertion Rate results for full power, mid-power, and low power, respectively.

Figures 15.4.2-3 through 15.4.2-9 show the characteristic plant response for a limiting case: 10% power with positive reactivity feedback. Table 15.4.2-1 presents the sequence of events. Power increased steadily in response to the reactivity insertion, until the reactor scrammed. Coolant temperatures also increased steadily, due to the primary-to-secondary system power mismatch. The pressure increase due to coolant expansion and pressurizer insurge was limited by the primary PORVs, with the pressurizer pressure peaking at 2414.3 psia. The overtemperature ΔT trip setpoint was reached at about 63.7 seconds, and rod insertion began at about 64.4 seconds. The calculated DNBR reached a minimum value at 64.8 seconds.

15.4.2.5 Conclusions

Reactivity insertion transient calculations demonstrate that the DNB safety limit of 1.141 will not be breached during any credible reactivity insertion transient at full power, mid power, or low power. The MDNBR of 1.213, reached during the most limiting transient, occurs at 10% power and retains margin to the MDNBR limit.

The fuel melt SAFDL is also met because the OPΔT trip function provides protection during slow transients which are not characterized by localized radial power redistribution.

TABLE 15.4.2-1

LIMITING BANK WITHDRAWAL AT POWER EVENT SUMMARY

TIME	EVENT	VALUE
0.0 s	Bank withdrawal began	-
63.7 s	Vessel temp. rise reached OTΔT trip setpoint (see Figure 15.4.2-5)	45.03°F
64.4 s	Scram rod insertion began	-
64.8 s	Minimum DNBR occurred	1.213
65.6 s	Pressurizer pressure peaked (see Figure 15.4.2-6)	2414.3 psia

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

Rod Cluster Control Assembly (RCCA) misoperation events include:

1. Withdrawal of a single full length RCCA

2. Static misalignment of a single full length RCCA

3. Dropped full length RCCA

4. Dropped full length RCCA bank

Each RCCA has a position indicator which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod bottom light. Group demand position is also indicated. The full-length RCCAs are always moved in preselected banks and the banks are always moved in the same preselected sequence.

The statically misaligned RCCA, dropped RCCA, and dropped RCCA bank events are classified as Condition II events. The withdrawal of a single RCCA event is classified as a Condition III event. Acceptance criteria are presented in Section 15.0.1.1.

The analyses are performed using S-RELAP5 to model system response and XCOBRA-IIIC to calculate minimum DNB ratios. Bounding values were obtained by coupling conservative local power peaking to the MDNBR calculations. The power peak associated with each event is characterized through an augmentation factor which relates the maximum power peak to the steady state power peak. The steady state power distributions and augmentation factors are calculated with the PRISM reactor simulator (Reference 15.4.3-1).

For control rod misoperation events, the maximization of power peaking results in a reduction in the DNBR. To assure that bounding values are determined for the radial power peaking, the following approach is used for each event. The increase in power peaking above that associated with equilibrium steady state conditions is determined for a spectrum of cycle exposures and applicable control rod configurations. Based on these results, a conservative augmentation factor is derived. This augmentation factor is then applied to the allowable $F_{\Delta H}$ to ensure a bounding value for the peak pin power input to the DNB analysis.

15.4.3.1 Withdrawal of a Single Full-Length RCCA

15.4.3.1.1 Identification of causes and event description

The event is initiated by the inadvertent withdrawal of a single control rod at power. The ensuing reactivity insertion causes core power to increase. In the event that the secondary steam dump control system does not respond to the increased power production, secondary system temperature and pressure will increase, causing a corresponding increase in primary coolant temperature. This increase in primary coolant temperature occurs slowly enough that the pressurizer pressure control system, if available, is capable of suppressing the primary pressure increase. The degradation of coolant conditions coupled with the power increase is essentially the same as expected for RCCA bank withdrawals at power, and may approach DNB conditions in the hot channel.

The single RCCA withdrawal is distinguished from the withdrawal of an RCCA bank by a severe radial power redistribution. High radial power peaking is quite localized in the region of the single withdrawn RCCA and may, in severe cases, surpass the design limits. Thus, assemblies in the immediate vicinity of the withdrawn RCCA may experience boiling transition. Such exposure would be limited to short time periods. Some fuel damage might occur.

Primary protection for this event is afforded by the high nuclear flux trip and the overtemperature ΔT trip.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted RCCA
bank during full power operation. Procedures are available to permit the operator to withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an RCCA should one be accidentally dropped. The event can occur only as the result of multiple wiring failures or multiple operator action. The probability of such a combination of conditions is low. This event is, therefore, classified as a Condition III event during which some fuel damage is permitted.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, the rod position indicators would indicate the relative positions of the assemblies in the bank. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would similarly result in the same visual indications. Withdrawal of a single RCCA results both in a positive reactivity insertion tending to increase core power, and in an increase in local power density in the core area associated with the RCCA.

15.4.3.1.2 Analysis method

The transient response of the reactor system exclusive of radial power redistribution effects is as calculated with the S-RELAP5 methodology (Reference 15.0-12) for the most limiting case of uncontrolled RCCA withdrawal at power. The coolant flow rate, primary pressure, and core inlet coolant temperature boundary conditions at the time of MDNBR (determined by S-RELAP5) are transferred to the XCOBRA-IIIC computer code (Reference 15.0-4) for calculation of MDNBR. The core average heat flux at the time of MDNBR is adjusted to include design power peaking and a radial peaking augmentation factor calculated to describe the radial power peaking redistribution due to the single withdrawn RCCA.

The fraction of the fuel to experience boiling transition for the event is conservatively taken to be the number of fuel assemblies with calculated minimum DNB ratios below the safety limit, divided by the total number of assemblies.

15.4.3.1.3 Definition of events analyzed and bounding input

The initial input is selected such that the analysis bounds power operation.

The initial input for the case analyzed is the same as that previously identified to provide the limiting transient response for the uncontrolled RCCA bank withdrawal at power. For the withdrawal of a single full-length RCCA, a maximum radial power peaking augmentation factor of 1.134 was used.

15.4.3.1.4 Radiological Consequences

The NRC has approved implementation of the Alternative Source Term methodology (Reference 15.0.12-3) for analysis of the radiological consequences of this event (Reference 15.4.3-2).

The single RCCA rod withdrawal event causes an insertion of positive reactivity which results in a power excursion transient and may cause fuel damage. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the steam generator atmospheric relief valves or safety valves. In addition, radioactivity is contained in the primary and secondary coolant before the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

This event can result in fuel damage. In order to bound the maximum number of fuel assemblies expected to experience fuel clad damage, this analysis conservatively assumes that four assemblies experience clad damage, and that three of those four assemblies also experience melting.

Other assumptions used in the radiological consequence analysis:

- Loss of offsite power at the time of reactor trip. This drives the release from the secondary coolant system through the SG relief valves, since condenser cooling is lost.
- Maximum radial peaking factor is 1.8.
- 157 fuel assemblies in the core.
- For the 4 damaged fuel assemblies, the activity released from the fuel clad breach and fuel melting is based on the following fractions:

Krypton 85:	10% breach and 100% melt
Other Noble Gases:	5% breach and 100% melt
Iodine 131:	8% breach and 40% melt
Other Halogens:	5% breach and 40% melt
Alkali Metals (Cs, Rb):	12% breach and 30% melt

  All gap activity in the damaged fuel rods is instantaneously released.
- The chemical form of the radioiodine released from the damaged fuel is 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.
- The minimum volume (hot) of the reactor coolant system is 8,254 ft^3, based on a temperature of 575.9°F and a pressure of 2235 psig.

- The primary-to-secondary leakage to the steam generators mixes instantaneously and homogeneously with the secondary water without flashing.
- RCS equilibrium activity concentration is conservatively assumed to be twice the values in Table 15.0.12-2. Since fuel damage is assumed, no iodine spiking is assumed for the equilibrium RCS activity.
- The primary-to-secondary leak rate is limited to 0.3 gpm total through the 3 steam generators which bounds TS limit of 75 gpd per S/G.
- The minimum volume of the secondary side coolant is 88,641 lbs per steam generator.
- The integrated mass of steam released from the steam generators as a function of time is 301,967.3 lbm (0 – 2 hours), 868,735.6 lbm (0 – 8 hours), 1,993,731.4 lbm (0 – 24 hours), and 3,631,641.5 lbm (0 – 53.2 hours).
- The halogen and alkali metal partition coefficient for the steam generators is 100.
- Iodine releases from the steam generators to the environment are 97% elemental and 3% organic.
- All noble gas radionuclides released from the primary system are released to the environment without holdup, reduction, or mitigation.
- The time required for one train of the RHR System to establish adequate shutdown cooling to terminate releases from the steam generators is 53.2 hours.

15.4.3.1.5 Conclusions

The minimum DNB ratio calculated for the event is 0.991, which is less than the safety limit. The extreme radial power peaking calculated for the single RCCA withdrawal is localized in the neighborhood of the withdrawn RCCA. Only one of the 157 fuel assemblies in the core is calculated to experience boiling transition. The peak pellet LHGR was calculated to be under the threshold limit.

The single RCCA withdrawal event is classified as a Condition III event. Less than 10 percent of the core experiences boiling transition. Reactor vessel pressurization is well below 110 percent of the design limit. It is not anticipated that core cooling would be significantly hindered by less than 10 percent fuel failures. No more limiting fault is engendered by the occurrence of the event. The result of the analysis is thus in conformance with the acceptance criteria for a Condition III event and is, therefore, acceptable.

For the single RCCA withdrawal accident, the 2-hour dose at the EAB is 1.76 rem TEDE. The dose at the LPZ is 0.24 rem TEDE. The Control Room doses at inleakages of 300 and 500 cfm are 0.75 and 1.22 rem TEDE, respectively.

The offsite dose acceptance criterion established by Reference 15.0.12-3 for this accident is that doses should be less than 10% of the 10 CFR 50.67 guidelines, or less than 2.5 rem TEDE. The Control Room dose acceptance criterion established by 10 CFR 50.67 for this accident is 5 rem TEDE.

Therefore, the offsite and Control Room TEDE doses due to a single RCCA withdrawal event meet the dose acceptance criteria.

15.4.3.2 Static Misalignment of a Single RCCA

15.4.3.2.1 Identification of Causes and Event Description

The static misalignment of an RCCA is defined as a malfunction of the Control Rod Drive (CRD) mechanism, or of the rod control power supply, which causes an RCCA to be out of alignment with its bank; i.e., either higher or lower than any of the other RCCAs in the same bank. The reactor is in the steady state at rated full power conditions, and no excursion of core temperature, pressure, flow, or power occurs. For extreme RCCA misalignments, the core radial power distribution may be characterized by peaking factors in excess of design limits. Highly localized increases in clad surface heat flux, coolant temperature, and flow diversion may occur. In severe cases, the SAFDL on DNB may be approached.

The full-length RCCAs are always moved in preselected sequence. A quadrant tilt monitor alarm (upper and lower ex-core neutron detectors) is provided to indicate significant power tilts. If this alarm is temporarily out of service, periodic checks of individual rod positions and ex-core detector currents, and even core symmetry checks using in-core thermocouples and movable detectors can be made.

The operator is provided with rod position indication for each RCCA. An alarm is actuated when any RCCA bottom defeat switch is actuated so that an RCCA can be inserted into the core. This defeat switch must be actuated to prevent a load cutback.

15.4.3.2.2 Analysis Method

Primary system pressure, core inlet temperature, and coolant flow rate at the rated full power operating point are input to the XCOBRA-IIIC code to calculate MDNBR. The rated full power core average clad surface heat flux is input to the MDNBR calculation after having been adjusted to include the design radial and axial power peaking distribution factors and a radial peaking augmentation factor calculated to bound the radial power redistribution characteristics of a misaligned RCCA.

15.4.3.2.3 Definition of events analyzed and bounding input

The event is analyzed at the rated full power operating point to bound power operation. Analysis inputs reflect the following allowance from nominal full power operating conditions:

Power	102% of 2300 MWt
Core Inlet Temperature	544.4°F
Pressurizer Pressure	Nominal -40 psi
Coolant Flow	Minimum allowed by Technical Specifications

The radial peaking factor augmentation used in the analysis is 1.134.

Two cases are analyzed to bound the misalignment conditions where the single RCCA is stuck fully out of the core or stuck fully in the core. For the condition with the single RCCA misaligned above the RCCA's in the same bank, it is conservatively assumed that Bank D is fully inserted to the full extent allowed by the Rod Insertion Limits (RIL) controlled by the Technical Specifications as specified on a cycle-specific basis by the Core Operating Limits Report. However, it is assumed that the most reactive "D" bank RCCA is fully withdrawn from the core (Case 1). For the condition with a single RCCA misaligned below its corresponding bank, the analysis is performed with all Banks fully withdrawn and the most reactive RCCA fully inserted to the bottom of the core (Case 2).

15.4.3.2.4 Analysis of Results

Case 1 represents the most limiting case in the current analysis. The calculated MDNBR for the Static Misalignment of a Full-Length RCCA is 1.432, which is greater than the 1.141 DNB limit. The peak pellet linear heat generation is 19.558 kw/ft, which is below the threshold limit, so that fuel centerline melt does not occur. Since no fuel failure is calculated to occur, there is no radiological release consequent to this event. The result of the analysis is, thus, in conformance with the acceptance criteria for Condition II events and is, therefore, acceptable.

15.4.3.2.5 Conclusions

An RCCA out of position can result only from a malfunction in the mechanism or its associated power supply and, in such a case, it is clearly indicated to the operator by independent monitoring systems. The cases discussed above have indicated that the DNB ratio remains greater than the safety limit in the event of a rod misalignment. The DNB SAFDL is, therefore, satisfied for this event.

15.4.3.3 Dropped RCCA and RCCA Bank

15.4.3.3.1 Identification of Causes and Event Description

The event is defined to be initiated by a dropped RCCA or RCCA bank. The dropped RCCA/bank promptly inserts negative reactivity which reduces reactor power and disturbs the power distribution, resulting in increased local power peaking. The moderator temperature decreases as a result of the reduction in reactor power. Consequently, a negative moderator temperature coefficient can return the reactor to a full power condition with an elevated radial power peaking factor corresponding to the new radial power distribution caused by the dropped RCCA/bank.

If a RCCA/bank drops into the core during power operation, it would be detected by either a rod bottom signal device or by the use of the excore chambers. The rod bottom signal device provides an individual position indication signal for each RCCA. The other independent indication of an RCCA/bank drop is obtained through the excore power range channel signals. This rod drop detection circuit is actuated upon sensing a rapid decrease in local flux such as could occur from depression of flux in one region by a dropped RCCA/bank.

15.4.3.3.2 Analysis Method

The analyses are performed by coupling a conservative power peak to transient characterized through an augmentation factor which relates the maximum power peak to the steady state power peak. The steady state power distributions and augmentation factors are calculated with the PRISM reactor simulator. Standard neutronic methodology is used to calculate neutronics parameters such as control rod worth and power peaking.

The system response to a dropped RCCA/bank is analyzed with the S-RELAP5 code. The DNB analysis is performed using the XCOBRA methodology, using the operating conditions from the S-RELAP5 calculation. Local power redistribution effects due to the dropped rod/bank are input to the XCOBRA methodology by a local power augmentation factor. The Technical Specification value of the allowed $F_{\Delta H}$ is multiplied by this augmentation factor.

15.4.3.3.3 Definition of Events Analyzed and Bounding Input

The initial input is selected such that the analysis bounds power operation. No single failure assumption is required since manual rod control is assumed. A spectrum of dropped rod/bank worth cases is analyzed.

Key analysis conditions include:

Initial power	100.3% of 2339 MWt
Moderator temperature coefficient	-45 pcm/°F
Doppler coefficient	-1.75 pcm/°F
Dropped rod worth	25 pcm (as a bounding minimum value)
Dropped bank worth	1500 pcm (as a bounding maximum value)
Radial peaking augmentation factor (dropped rod)	1.124
Radial peaking augmentation factor (dropped bank)	1.270

15.4.3.3.4 Analysis of Results

The limiting MDNBR case of this event was initiated by a step negative reactivity insertion representing a dropped RCCA bank. The reactor power dropped quickly in response, which in turn caused a decrease in moderator temperature. Due to the strongly negative moderator temperature coefficient, the reactor power recovered to near the initial power level. A similar system response was predicted in the limiting LHGR case of this event, which was initiated by a step negative reactivity insertion representing a dropped RCCA bank.

The limiting case MDNBR was calculated to be 1.339, which is greater than the DNB safety limit of 1.141. The peak pellet LHGR for each case is below the threshold limit, so that fuel centerline melt does not occur.

The event summary for the limiting MDNBR case is provided in Table 15.4.3-1. The event summary for the limiting LHGR case is provided in Table 15.4.3-2. Figures 15.4.3-1 through 15.4.3-7 depict the system response to a dropped RCCA of low worth. Figures 15.4.3-8 through 15.4.3-13 depict the system response to a dropped bank.

15.4.3.3.5 Conclusions

For the case of a dropped full-length RCCA or RCCA bank, the minimum calculated DNBR is greater than the safety limit. The peak LHGR is less than the fuel centerline melt limit. Therefore, the event acceptance criteria on DNBR and fuel centerline melt are met.

TABLE 15.4.3-1

DROPPED ROD/BANK EVENT SUMMARY FOR MDNBR CASE

TIME	EVENT	VALUE
0.0 s	Rod/Bank began to drop	-
2.2 s	Rod/Bank reached bottom of core	-
178 s	Core power level returned	2237.2 MWt
178 s	Minimum DNBR occurred	1.339

TABLE 15.4.3-2

DROPPED ROD/BANK EVENT SUMMARY FOR PEAK LHGR CASE

TIME	EVENT	VALUE
0.0 s	Rod/Bank began to drop	-
2.2 s	Rod/Bank reached bottom of core	-
~134 s	Core power level returned	2131 MWt
~134 s	Minimum LHGR margin reached	1.09%

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP AT AN INCORRECT TEMPERATURE

The H. B. Robinson plant technical specifications do not permit operation with less than three primary coolant pumps during power operation. Therefore, analysis of this event is unnecessary.

15.4.5 RECIRCULATION LOOP AT INCORRECT TEMPERATURE OR FLOW CONTROLLER MALFUNCTION

The H. B. Robinson Unit 2 plant has no primary loop isolation valves nor means to control primary flow. Therefore, this event is not applicable to H. B. Robinson Unit 2.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Identification of Causes and Event Description

Reactivity can be added to the core with the CVCS by feeding reactor makeup water into the RCS via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of reactor makeup water to the RCS from the reactor makeup water system, and inadvertent dilution can be readily terminated by isolating this single source. The operation of the reactor makeup water pumps which take suction from this tank provides the only supply of makeup water to the RCS. In order for makeup water to be added to the RCS, the makeup pumps must be running in addition to the reactor charging pumps.

The rate of addition of unborated water makeup to the RCS is limited to the capacity of the charging pumps. This limiting addition rate is conservatively assumed to be 242.55 gpm. For totally unborated water to be delivered at this rate to the RCS at pressure, three charging pumps must be operated. Normally only one charging pump and one reactor make-up pump are operating.

A minimum of two separate operations are required for dilution. First, the operator must position the makeup mode switch from the "automatic makeup" mode to the "dilute" or "alternate dilute" mode. Second, the control switch must be positioned to "start." Omitting either step would prevent dilution. A dilution could also be initiated by manual operator action at the control board by repositioning individual component control switches. More than two separate actions would be required to initiate a dilution manually. This makes the possibility of inadvertent dilution very small.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all modes of plant operation, boron dilution during refueling, cold shutdown, hot shutdown, startup, and power operation are considered in this analysis. Surveillance procedures for control rod exercise, control rod drop test, NARPI calibration, and CRDM operation test have also been considered. Several procedures allow withdrawal of control rods 5 steps from the bottom of the core to prevent thermal binding. This does not need to be explicitly considered in the safety analysis because it inserts a negligible amount of reactivity. In Modes 4 and 5 with cooling via RHR, the analysis assumes all rods are inserted and the RCS will be borated to account for any stuck rod, consistent with the analysis methodology.

15.4.6.2 Analysis Methods

The dilution time required to overcome the shutdown margin is calculated by solving the differential equation,

$$M \times \frac{dC(t)}{dt} = -W \times C(t)$$

so that the dilution time is given by

$$T_D = \frac{M}{W} \times \ln \frac{C_{initial}}{C_{critical}}$$

where:

M = mass of water in the primary system
C = boron concentration in the primary system
W = mass flow of unborated water

The model described above is commonly referred to as the "instantaneous" or "perfect" mixing model and is applicable when RCS flow rates are sufficient to ensure continuous and uniform mixing in the reactor vessel. Sufficient RCS flow for use of this model is assumed if at least one RCP is in operation.

The dilution front model is used when the system is being cooled via the RHR system and the RCS flow is slower than would occur with at least one RCP running. The time for the first dilution front to reach the core is calculated by dividing the RCS mass from the mixing location to the bottom of the core by the shutdown cooling + dilution flow. The time for subsequent fronts is calculated by dividing the mass of the RCS by the RHR + dilution flow. The time to criticality is determined by iteratively tracking the number of dilution fronts. The analysis considers a range of RCS flow rates between 2800 and 7500 gpm. While operating on RHR, a low flow alarm, set at 3000 gpm, will alert the Operators to low flow conditions. If the low flow condition is not due to intentional Operator action, then the Operator is instructed to restore RHR flow to greater than or equal to 2800 gpm. If flow cannot be restored then the Operator is referred to the Abnormal Operating Procedures. This provides a high degree of assurance that this minimum flow rate will be maintained.

The critical boron concentration and a conservative boron worth are determined utilizing the PRISM reactor simulator code.

15.4.6.3 <u>Definition of Events and Bounding Input</u>

15.4.6.3.1 Dilution During Refueling (MODE 6)

During refueling the following conditions exist:

a) One residual heat removal pump is running to ensure continuous mixing in the reactor vessel.

b) The valve in the seal water header to the reactor coolant pumps is closed.

c) The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution.

d) The boron concentration of the refueling water is 1950 ppm.

e) Monitoring of the core is provided by BF_3 detectors and fission chambers which are installed in instrument wells in the primary shield wall outside of the reactor vessel and are connected to instrumentation giving audible and/or visual count rates. Irradiated fuel assemblies or neutron sources generate an adequate neutron flux level in the core to provide indication on the Source Range instrumentation channels during Refueling.

A minimum water volume in the RCS of 3200 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel to the centerline of the nozzles to ensure mixing via the residual heat removal (RHR) loop. The conservative maximum dilution flow rate of 242.55 gpm and uniform mixing are also considered.

15.4.6.3.2 Dilution during Cold Shutdown (MODE 5)

For the cold shutdown case a minimum water volume (3200 ft^3) in the RCS is used, which corresponds to the volume necessary to fill the reactor vessel to the centerline of the nozzles to ensure mixing via the RHR loop. With the reactor in this configuration, a minimum shutdown margin of 2.6% $\Delta\rho$ is maintained.

15.4.6.3.3 Dilution during hot shutdown (MODE 4)

Conditions at hot shutdown require the reactor to have available at least 2.6%$\Delta\rho$ shutdown margin. Dilution flow is conservatively assumed to be 242.55 gpm. The volume of the reactor coolant is assumed to be 4042 ft^3 which is a conservative approximation of the volume of the RCS excluding the pressurizer, upper head, and steam generators.

15.4.6.3.4 Dilution during Hot Standby (MODE 3)

Conditions at hot standby require the reactor to have available at least 1.1%$\Delta\rho$ shutdown margin. Dilution flow is conservatively assumed to be 242.55 gpm. The volume of the reactor coolant is assumed to be 7472 ft^3 which is a conservative approximation of the volume of the RCS excluding the pressurizer and upper head and accounts for a maximum steam generator tube plugging level of 6%.

15.4.6.3.5 Dilution during Startup (MODE 2)

Conditions at startup require the reactor to have available at least 1.0%$\Delta\rho$ shutdown margin. Dilution flow is conservatively assumed to be 242.55 gpm. The volume of the reactor coolant is assumed to be 7472 ft^3, which is a conservative approximation of the volume of the RCS excluding the pressurizer and upper head and accounts for a maximum steam generator tube plugging of 6%. Mixing of the reactor coolant is maintained by operation of the reactor coolant pumps. High source level and all reactor trip alarms are effective.

15.4.6.3.6 Dilution during Power Operation (MODE 1)

Dilution rate during power operation is dependent on charging pump capacity and coolant boron concentration. The conservative maximum reactivity addition rate for a boron dilution flow of 242.55 gpm during power operation is approximately $1.1 \times 10^{-5} \Delta\rho$/sec. The reactivity insertion rates considered in Sections 15.4.1 and 15.4.2 cover any rate achievable by boron dilution and demonstrate that the core is protected from DNB.

15.4.6.4 Analysis of Results

The results of the analysis for this event are summarized in Table 15.4.6-1 with the exception of boron dilution during power operations. The results show that there is adequate time for the operator to manually terminate the source of dilution flow. The reactor will be in a stable condition. The operator can then initiate reboration to recover the shutdown margin. Boron dilution during power operation is bounded by the analyses presented in Sections 15.4.1 and 15.4.2.

15.4.6.5 Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and to take corrective action before shutdown margin is lost.

TABLE 15.4.6-1

RESULTS OF THE ANALYSES OF CVCS MALFUNCTION

Mode	Reactor Conditions	Critical Boron Concentration (ppm)	Initial Boron Concentration (ppm)	Time to Criticality* (minutes) Calculated **	Time to Criticality* (minutes) Minimum Acceptable
6	Refueling	1252.5	1950.0	30.69	30
5	Cold Shutdown	1124.1	1511.8	15.21	15
4	Hot Shutdown	1199.5	1470.5	16.49	15
3	Hot Standby	1121.4	1245.5	16.03	15
2	Startup	819.6	952.7	16.28	15
1	Power Operation	------- Bounded by analysis in Section 15.4.1 and 15.4.2 -------			

* The time it takes to go from an initial condition to critical (i.e., K_{eff}=1.0).
** This event was bounded by previous AOR and therefore the times to criticality are based on the bounding calculations.

15.4.7 Inadvertent Loading of a Fuel Assembly into the Improper Location

15.4.7.1 Identification of Causes and Event Description

Core loading errors arise from the loading of one or more fuel assemblies into improper core locations. This can result in changes in the power distribution and increases in local power density that may go undetected by incore instrumentation.

Reactor protection for the misloaded fuel assembly event depends on administrative plant procedures. To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a fuel loading or shuffle procedure to achieve the cycle specific core loading plan. The location of each assembly is verified prior to replacing the upper internals.

Incore instrumentation is used to determine the core power distribution and can also be used to monitor for possible misloaded assemblies. The instrumentation includes 46 incore thimble tubes to accommodate incore neutron flux probes. A minimum of 36 operable thimbles is required for power distribution flux maps. For excore instrument calibrations, 15 thimbles are required with at least 2 thimbles per core quadrant. Incore flux maps are taken at cycle startup and during initial power ascension at power levels of 30%, 70%, and 100% of rated thermal power, and at monthly surveillance intervals thereafter.

In the unlikely event that a loading error occurs, the power distribution will be changed by an amount proportional to the change in reactivity of the misloaded assembly. Large changes in the measured power distribution relative to the projected power distribution will be readily detectable by the incore instrumentation system at startup and during initial power ascension. However, small changes in the measured power distribution may go undetected by startup power ascension flux maps and continued operation at rated power can result in an increase in the radial peaking factor primarily for the case where the misloaded assemblies are the fresh gadolinia-bearing assemblies. If power operation persists with radial peaking factors in excess of Technical Specification limits due to an undetected misloading event, the DNBR SAFDL may be penetrated.

15.4.7.2 Analysis Method

A spectrum of misloading events has been analyzed with the PRISM (Reference 15.4.7-1) code using a full core 3-dimensional sixteen (16) axial node model. Full core power distributions were calculated for the correctly loaded core and for a spectrum of misloading configurations. A misloading that resulted in an assembly power deviation greater than or equal to 10% in detector locations, or a ratio greater than 1.10 between assembly powers in symmetric detector locations were considered to be detectable. The initial low power map (e.g., 30% of rated thermal power) can be used as an early detection of a misloaded assembly since the power distribution changes only slightly during power escalation.

For undetectable misloading cases, the analysis focuses on core power peaking limits. If power peaking values for the misloaded core are calculated not to exceed Technical Specification limits (including uncertainties), no further evaluation is necessary, as DNB will not be exceeded. If calculations indicate that Technical Specification peaking limits could be exceeded, additional analysis is necessary. The additional analysis includes a DNBR determination. If penetration of the critical heat flux correlation safety limit has occurred, then a determination of the fraction of the fuel to experience boiling transition is made and the radiological consequences of such failures is assessed.

15.4.7.3 Definition of Events Analyzed and Bounding Input

A spectrum of misloading cases was analyzed. These cases represent the misloading of assemblies into core locations which are designated to be occupied by exposed or fresh fuel with different assembly reactivity characteristics.

For those cases which are found to be undetectable at beginning-of-cycle, a cycle depletion calculation was performed to determine the power history as a function of cycle exposure. From the results of the depletion calculation, the peak $F_{\Delta H}$ can be assessed relative to the Technical Specification limit. Since plant procedures require that measured power distributions be taken at monthly intervals, some of the undetectable events at BOL will be prevented from exceeding the Technical Specification limit by this periodic assessment. For those misloading events that remain undetectable, a DNB analysis is performed to determine the potential impact on the core.

Several thimble locations were assumed unavailable and not credited for the misload analysis. These locations are H-01, R-08, A-09, J-15, L-05, N-12, N-5, and D-12. These locations may not be counted in determining the detection criteria below:

Cycle 31 Thimble Requirements for Initial Low-Power Flux Map

Number of Operable Thimbles	Criteria for Difference Between Measured and Predicted	Criteria for Difference Between Symmetric Thimble Reaction Rates
37	5%	5%
40	7%	7%
41	7%	10%
38	10%	5%
42	10%	10%

These detection criteria should be compared against the maximum difference between measured and predicted reaction rates and the maximum difference between symmetric thimbles measured in the initial low power flux map (e.g., 30% of rated thermal power). If the measured values exceed the criteria, then further evaluation of the flux map for a potential misload is required.

15.4.7.4 Analysis of Results

The fuel misloading analysis determined the maximum value of $F_{\Delta H}$ and F_Q which can be expected to go undetected. The events analyzed can be categorized as the replacement of:

Category 1. Exposed fuel with exposed fuel,

Category 2. Exposed fuel with fresh fuel, and

Category 3. Fresh fuel assemblies with different reactivity characteristics; either different burnable absorber or enrichment designs.

The maximum allowed value of $F_{\Delta H}$ to meet the MDNBR safety limit is 2.243. This maximum allowed $F_{\Delta H}$ ensures that $F_{\Delta H}$ values at or below this value will meet DNBR safety limits.

The maximum allowed value for F_Q to meet the fuel centerline melt limit is 3.636. This maximum allowed F_Q ensures that F_Q values at or below this value will meet the fuel centerline melt limit.

15.4.7.5 Conclusion

It has been determined that the peaking factor threshold values will not be exceeded during a misloaded assembly event. For the initial flux map and periodic Technical Specification surveillance requirements provided that the detector operability constraints established above are satisfied.

15.4.8 Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

15.4.8.1.1 Design precautions and protection

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

1. The mechanism housings were hydrotested to 3105 psig when they were installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed RCS.

2. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III for Class A components, and

3. The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range, together with the energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and rod travel housing are threaded joints reinforced by canopy-type rod welds.

The use of a chemical shim (soluble boron) in the reactor coolant is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only a few rods in the core are at full power. There are low level insertion monitors, each with both visual and audio signals. Operating instructions require boration at the low level alarm. The control rod position monitoring and alarm systems are described in detail in Section 7.3.

15.4.8.1.2 Event classification and acceptance criteria

The probability of a rod being rapidly ejected from the core is so low that Rod Ejection is classified as a Condition IV event. The acceptance criteria require that doses in the exclusion area and low population zone be less than the 10 CFR 50.67 and Regulatory Guide 1.183 guidelines.

15.4.8.2 Analysis Method

The analysis was performed using the ANF-RELAP and XCOBRA-IIIC Codes. The ANF-RELAP code (Reference 15.0-3) was used to model the salient system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures). A DNBR calculation was performed to estimate the approximate time at which the DNBR was a minimum. The core fluid boundary conditions[a] and average rod surface heat flux at this time were then used as input to the XCOBRA-IIIC code (Reference 15.0-4), which was used to evaluate the MDNBR.

The Rod Ejection event was also evaluated with the procedures developed in the SPC Generic Rod Ejection Analysis (Reference 15.4.8-1) to determine the fuel pellet energy deposition resulting from an ejected rod.

15.4.8.3 Definition of Events Analyzed and Bounding Input

The control rod ejection event was analyzed at both BOC and EOC conditions and at HFP and HZP conditions, for a total of four cases (HFP EOC, HFP BOC, HZP EOC, and HZP BOC). Consistent with Regulatory Guide 1.77, a loss of offsite power was not assumed.

An over-pressure case was not analyzed, because pressures are more limiting for the Loss of External Load event (15.2.2). The initial increase in power due to an ejected rod causes an over-power condition of about 30%, resulting in a 30% power overload on the secondary. The Loss of External Load event, on the other hand, results in a 100% power overload on the secondary. Both of these events are very rapid and have approximately the same time to peak pressure (less than 10 seconds). Therefore, the Loss of External Load event bounds the Rod Ejection event with respect to over-pressure.

The systems challenged in this event are redundant and no single active failure will adversely affect the consequences of this event.

The least negative Doppler coefficient at each cycle exposure (i.e., BOC or EOC) was used, which minimizes negative Doppler feedback. A minimum delayed neutron fraction at each cycle exposure was conservatively used to convert reactivity to dollars in ANF-RELAP because it maximizes the worth of the ejected rod. A maximum pellet-to-clad heat transfer coefficient at each cycle exposure was conservatively used because it maximizes the heat flux at the rod surface and minimizes negative Doppler feedback.

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[a] The core outlet pressure at the time of MDNBR was reduced to account for the pressure loss due to the opening created by the ejected control rod.

The key analysis conditions for the limiting case are summarized below:

Initial power	102% of 2300 MWt
Ejected RCCA worth	Bounding (maximum) value [136.8 pcm]
Moderator temp. coefficient	+5.0 pcm/°F
Doppler coefficient	-0.976 pcm/°F
Delayed neutron fraction, β	Bounding (minimum) value [0.006252]
Pellet-to-clad HTC	Bounding (maximum) core-average value 1387 $\frac{\text{BTU}}{\text{hr-ft}^2\text{-°F}}$

15.4.8.4 Analysis of Results

The sequence of events for the analysis is given in Table 15.4.8-1. The transient tripped the reactor on the high-flux reactor trip. The key system response parameters are shown in Figures 15.4.8-1 through 15.4.8-4.

The pellet energy deposition was conservatively evaluated for BOC and EOC, at HFP and HZP, using the SPC Generic Rod Ejection methodology. The results of this analysis show that the peak deposited energy is 175.1 cal/g, which is less than the 280 cal/g limit as stated in Regulatory Guide 1.77.

15.4.8.5 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met. The predicted MDNBR is 1.199. This is greater than the 1.141 DNB limit. The predicted peak energy deposition is less than the 280 cal/g limit. Therefore, no fuel failures are predicted to occur, and there is no significant radiological release due to this event.

TABLE 15.4.8-1

ROD EJECTION EVENT SUMMARY

TIME	EVENT	VALUE
0.50 s	RCCA was ejected	-
0.58 s	Core power reached high-flux trip setpoint (see Figure 15.4.8-1)	118% of 2300 MWt
1.05 s	Core power peaked (see Figure 15.4.8-1)	128% of 2300 MWt
1.08 s	Scram rod insertion began	-
1.90 s	Core-average rod surface heat flux peaked (see Figure 15.4.8-1)	109% of heat flux corresponding to 2300 MWt
2.00 s	Minimum DNBR occurred (Figure 15.4.8-4)	1.199

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS

This event is not applicable to pressurized water reactors.

REFERENCES: SECTION 15.4

15.4.1-1 ANF-89-151 (A), "ANF-Relap Methodololgy for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland, WA, April 1992.

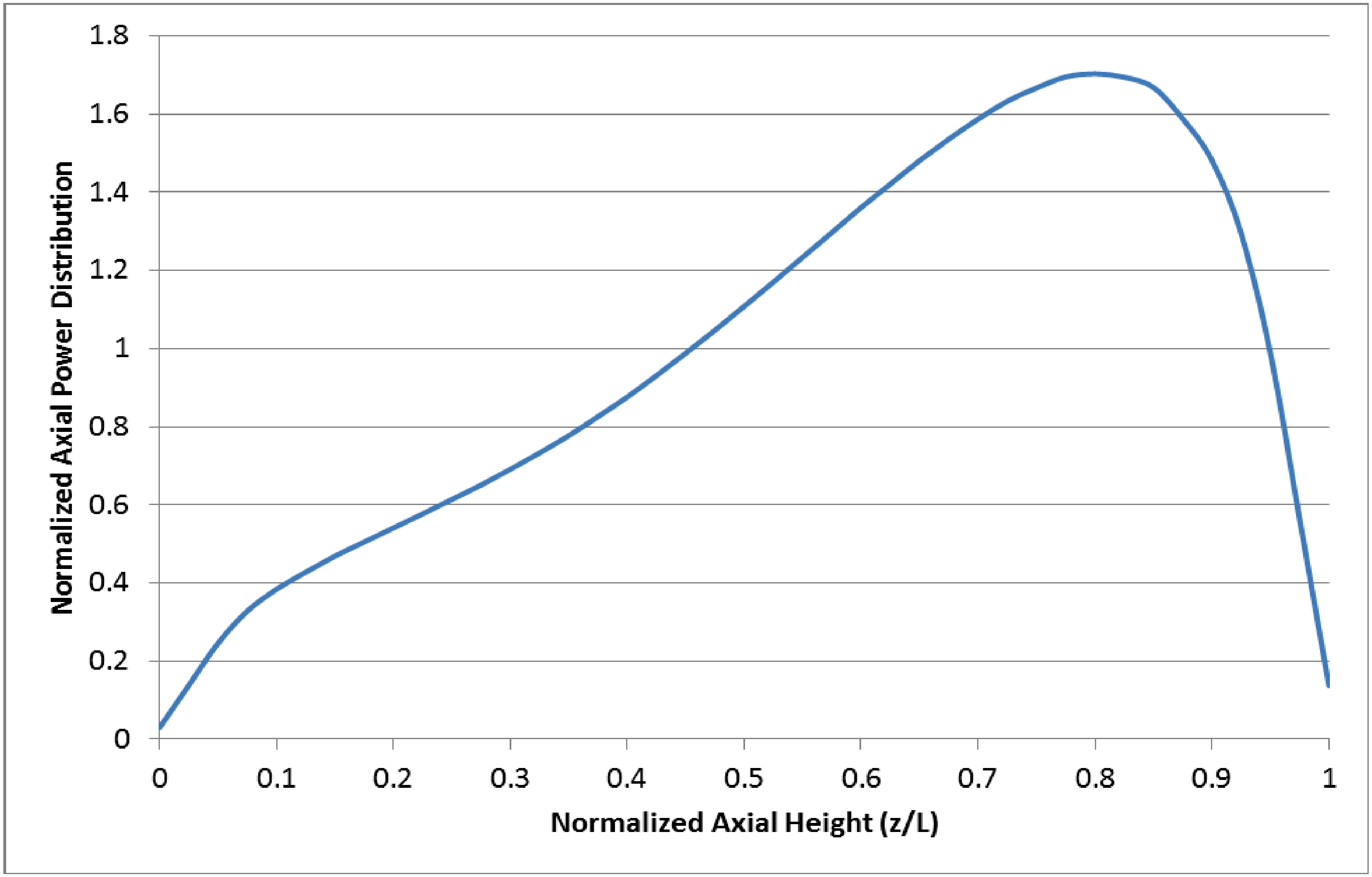
15.4.1-2 NSAL-00-016, “Rod Withdrawal from Subcritical Protection in Lower Modes,” Westinghouse, 2000.

15.4.3-1 EMF-96-029 (P)(A), Volumes 1 and 2, and Attachment, "Reactor Analysis System for PWRs Volume 1 - Methodology Discription, Volume 2 - Benchmarking Results" January 1997.

15.4.3-2 NRC Letter dated September 24, 2004, “H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105).”

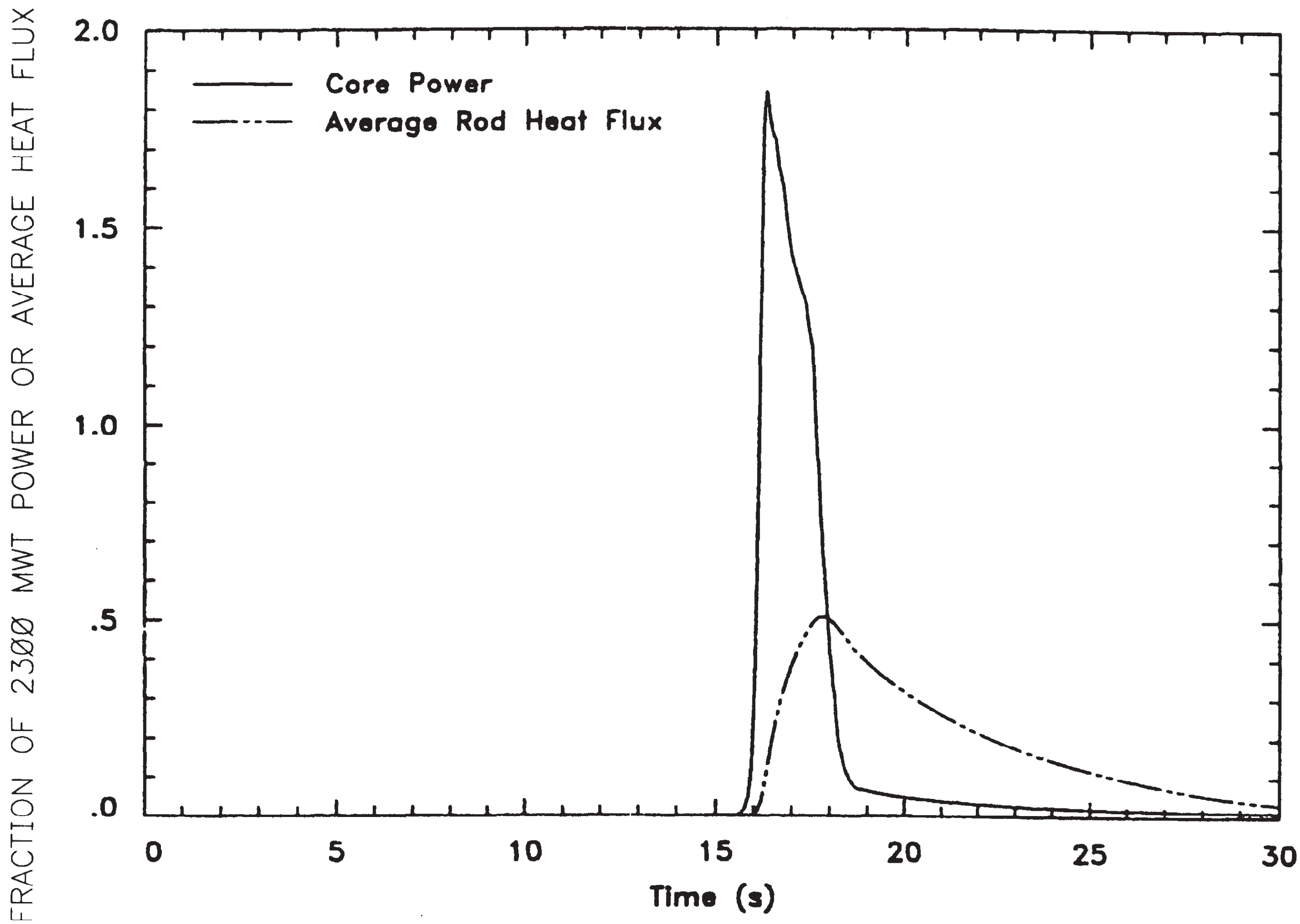
15.4.7-1 EMF-96-029 (P)(A), Volumes 1 and 2, and Attachment, "Reactor Analysis System for PWRs Volume 1 - Methodology Discription, Volume 2 - Benchmarking Results" January 1997.

15.4.8-1 XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for PWR's," Exxon Nuclear Company, Richland, WA, October 1983.



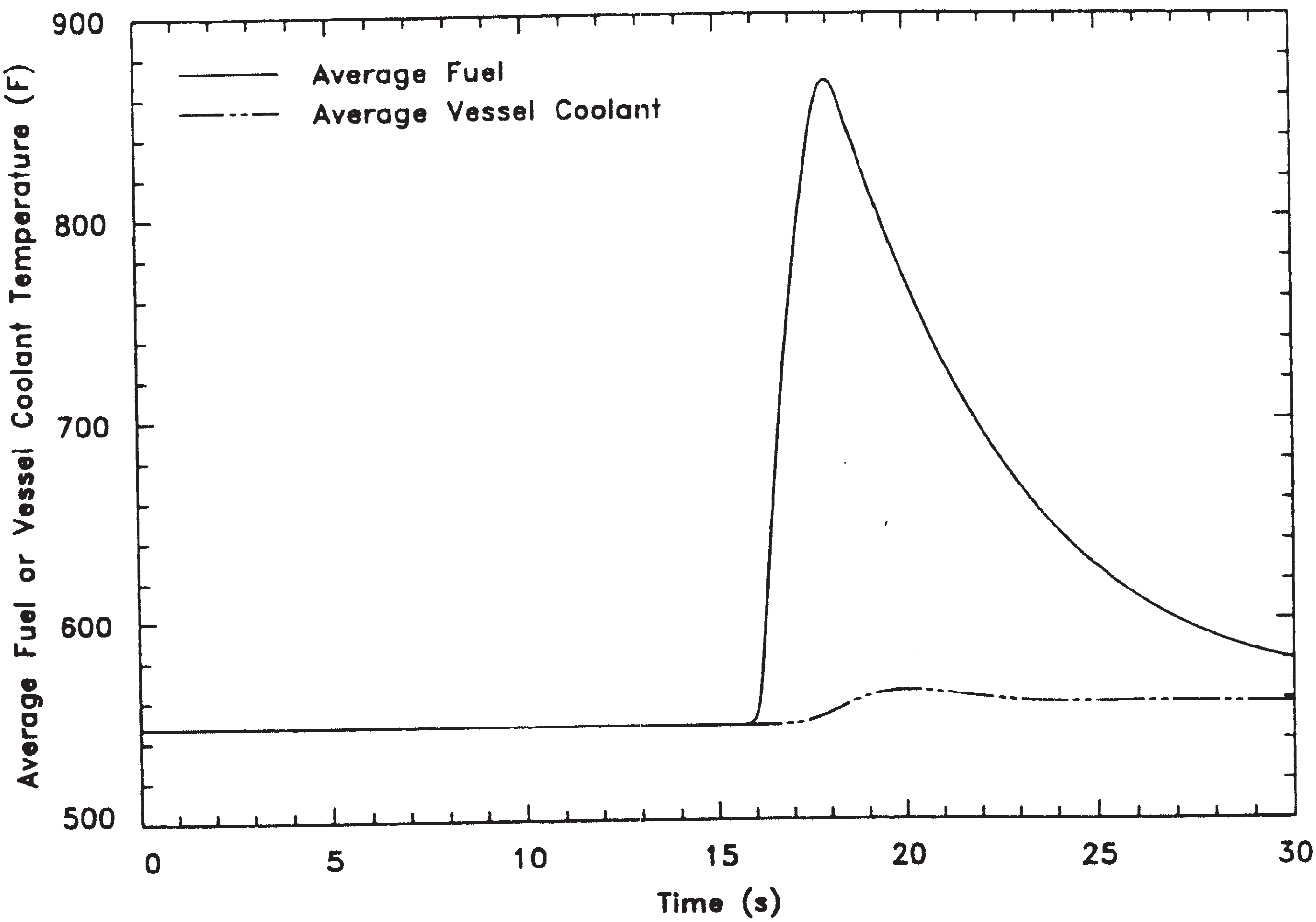
REVISION NO. 27

H. B. Robinson Unit 2 Duke Energy UPDATED FINAL SAFETY ANALYSIS REPORT	Bank Withdrawal from Subcritical: Axial Power Distribution	FIGURE 15.4.1-1



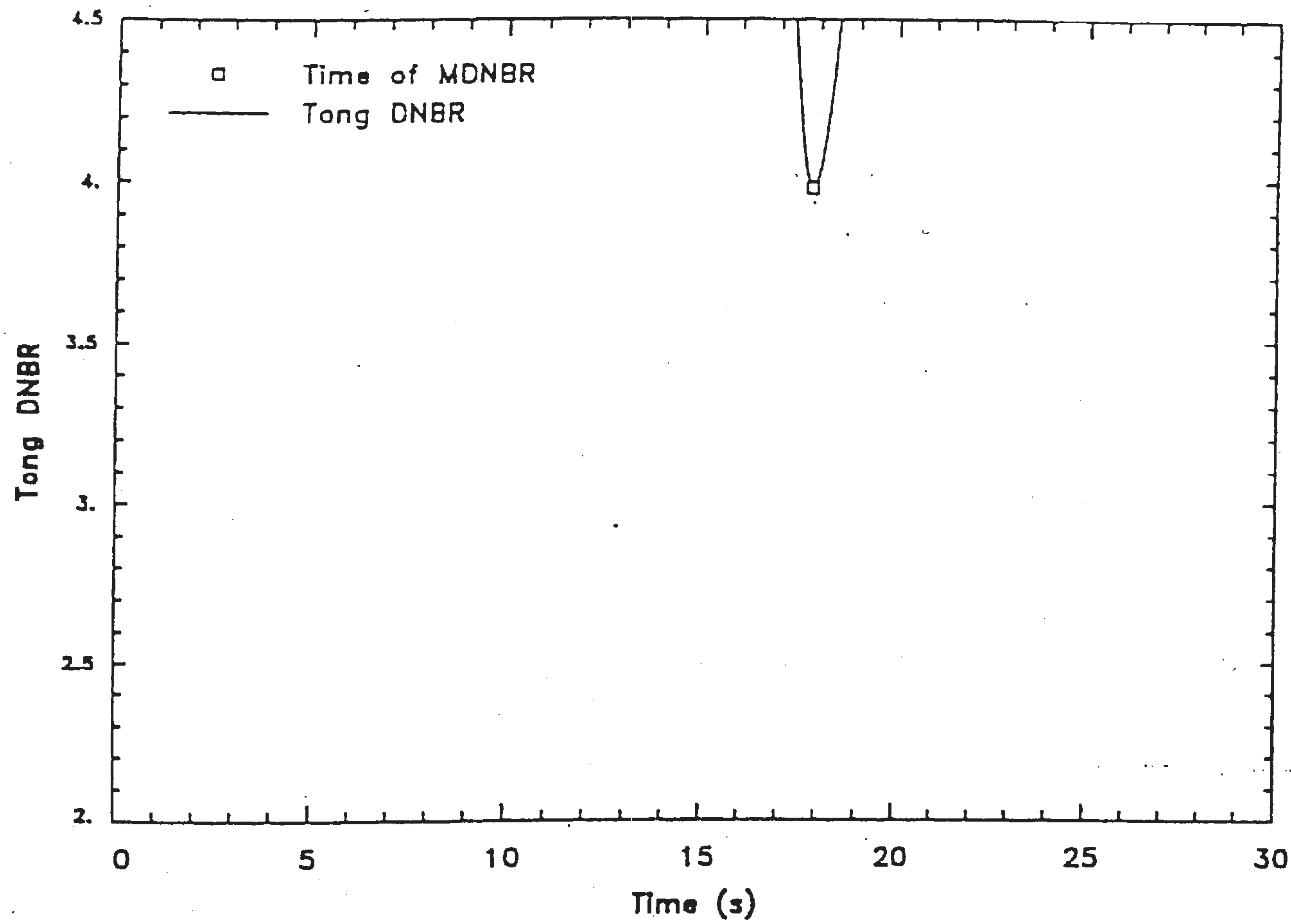
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Bank Withdrawal from Subcritical: Core Power and Average Rod Heat Flux	FIGURE 15.4.1-2



REVISION NO. 16

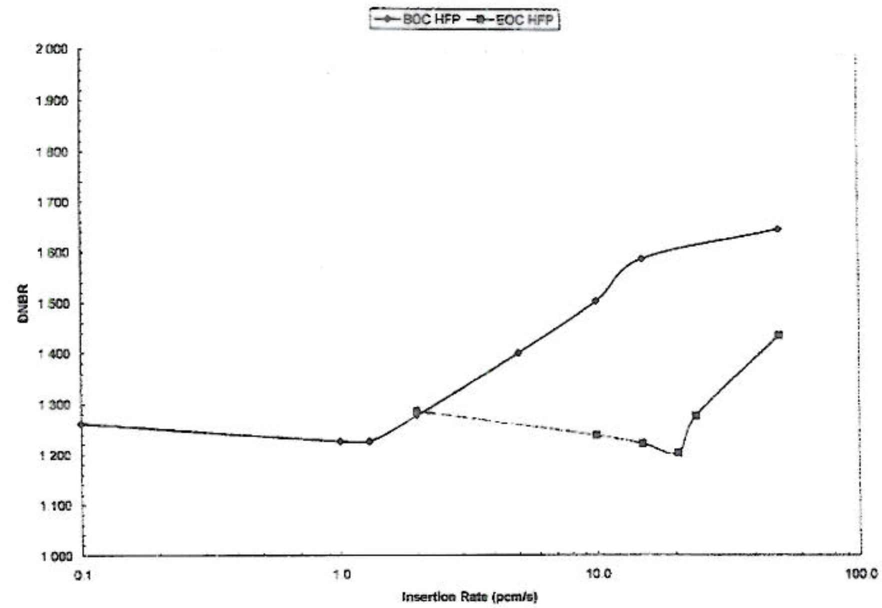
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Bank Withdrawal from Subcritical: Average Fuel and Vessel Coolant Temperatures	FIGURE 15.4.1-3



Note: Tong DNBR value is a RELAP calculated core-wide DNBR trend used to estimate approximate time of minimum DNBR to establish core parameters for input into the XCOBRA IIIC code to evaluate MDNBR.

REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Bank Withdrawal from Subcritical: DNBR Trend	FIGURE 15.4.1-4

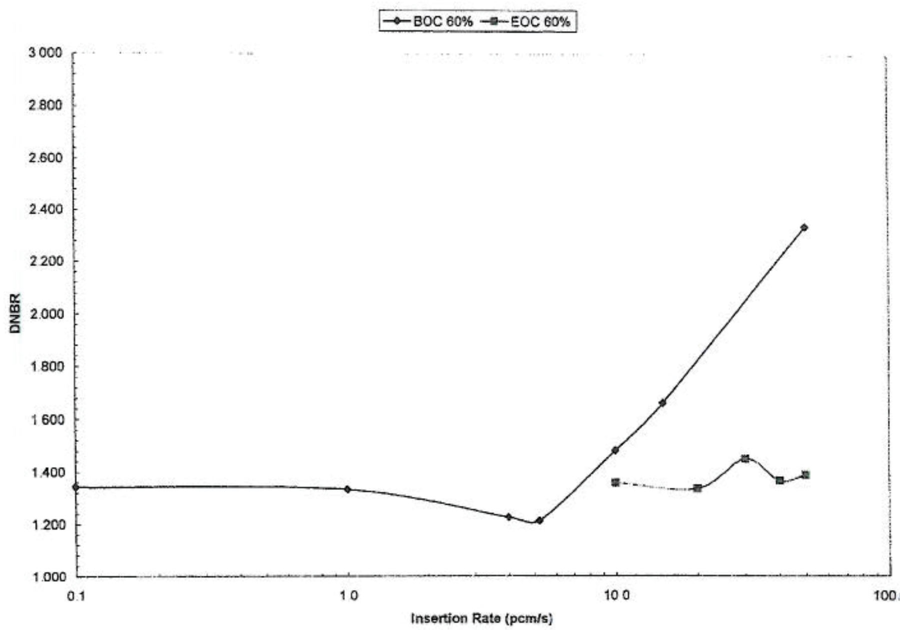


NOTE:

The purpose of this plot is to show the relative trend in DNBR as a function of reactivity insertion rate and cycle exposure.

REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Spectrum of Bank Withdrawals at 100.3% of 2339 MWt: MDNBRs	FIGURE 15.4.2-1

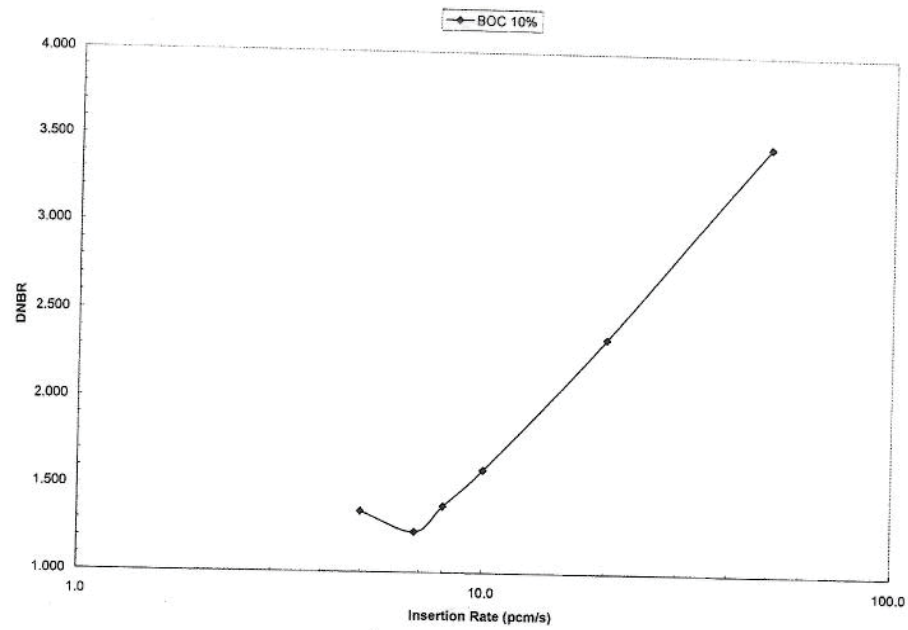


NOTE:

The purpose of this plot is to show the relative trend in DNBR as a function of reactivity insertion rate and cycle exposure.

REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Spectrum of Bank Withdrawals at 60% of 2339 MWt: DNBRs	FIGURE 15.4.2-2

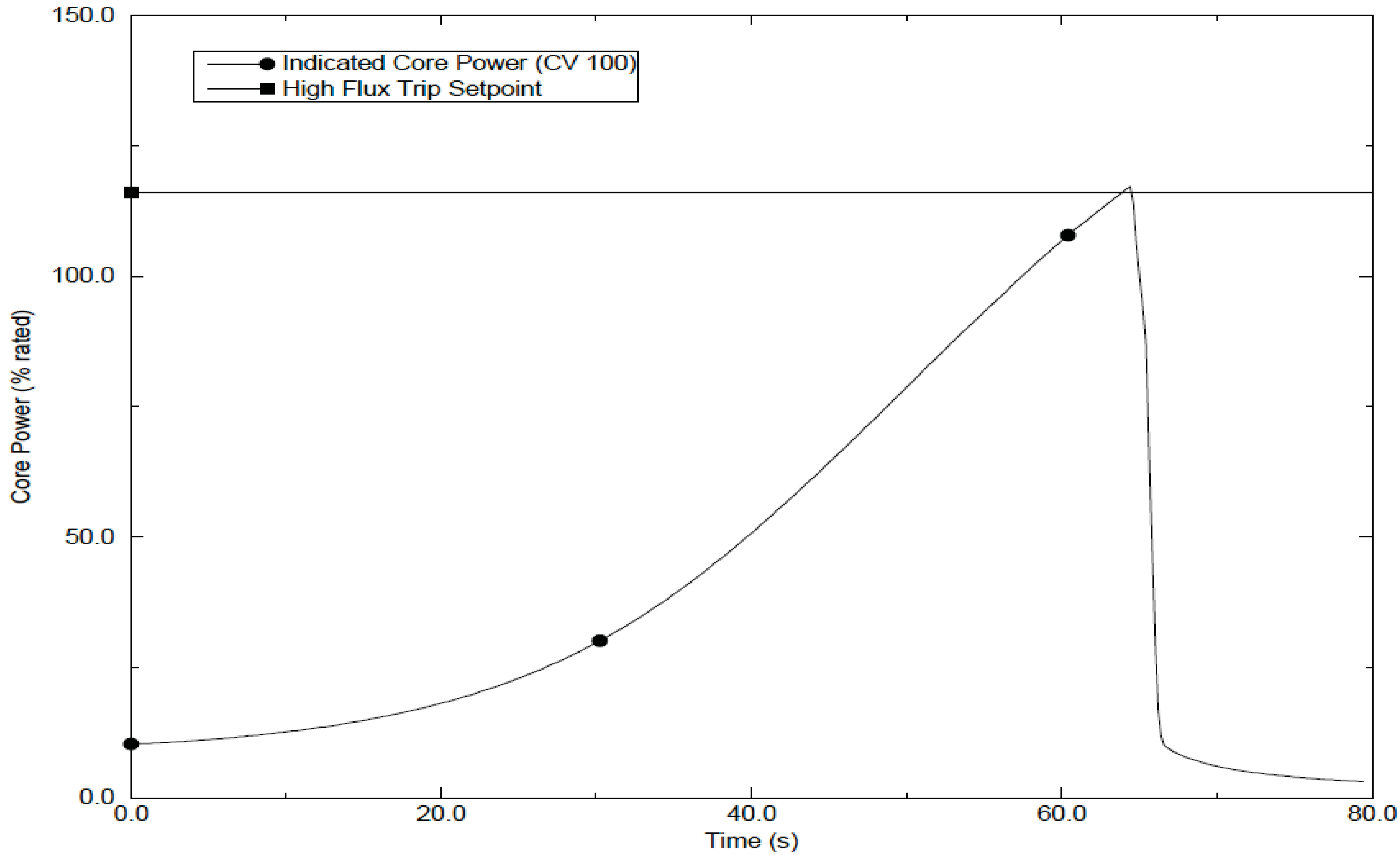


NOTE:

The purpose of this plot is to show the relative trend in DNBR as a function of reactivity insertion rate and cycle exposure.

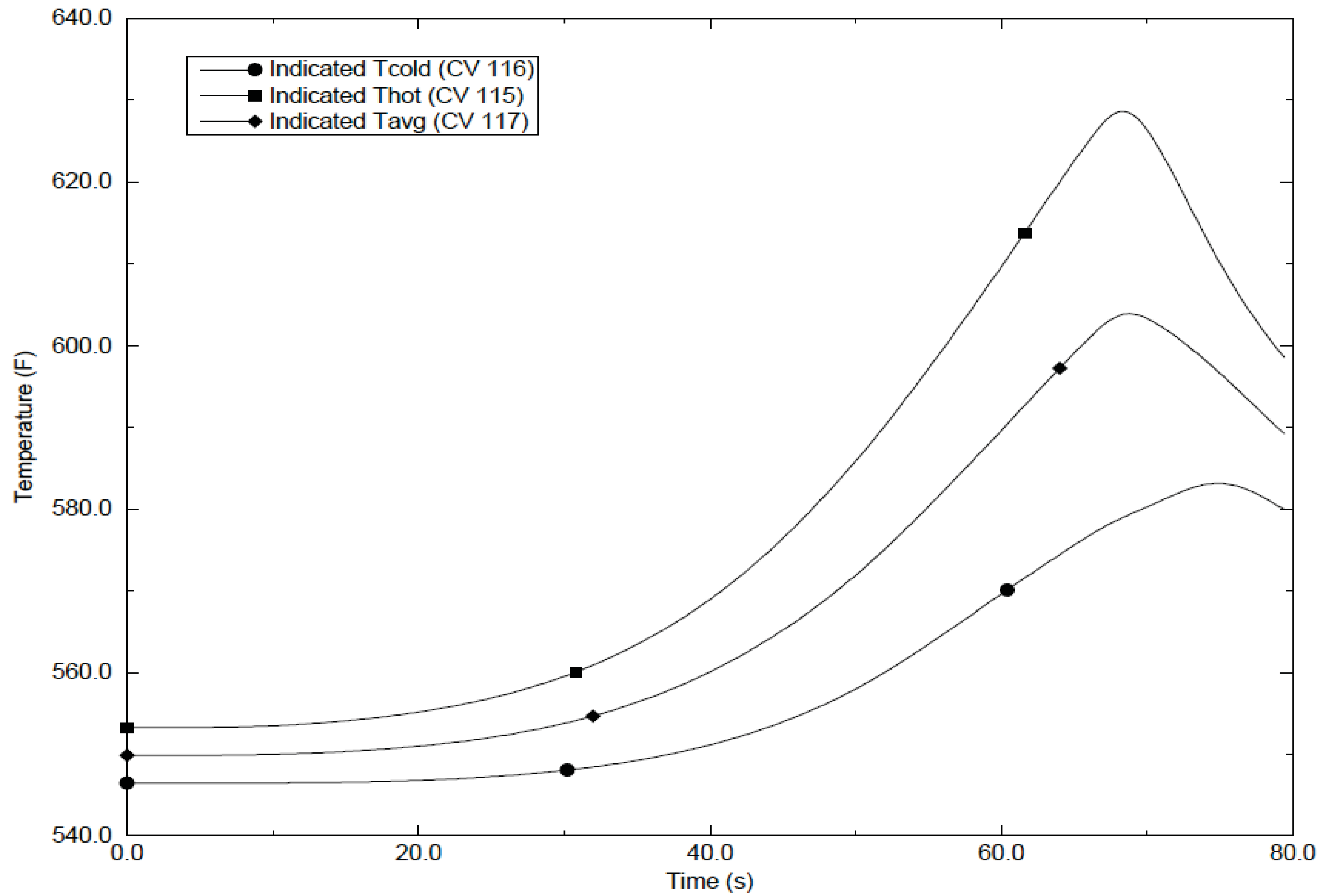
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Spectrum of Bank Withdrawals at 10% of 2339 MWt: DNBRs	FIGURE 15.4.2–2a



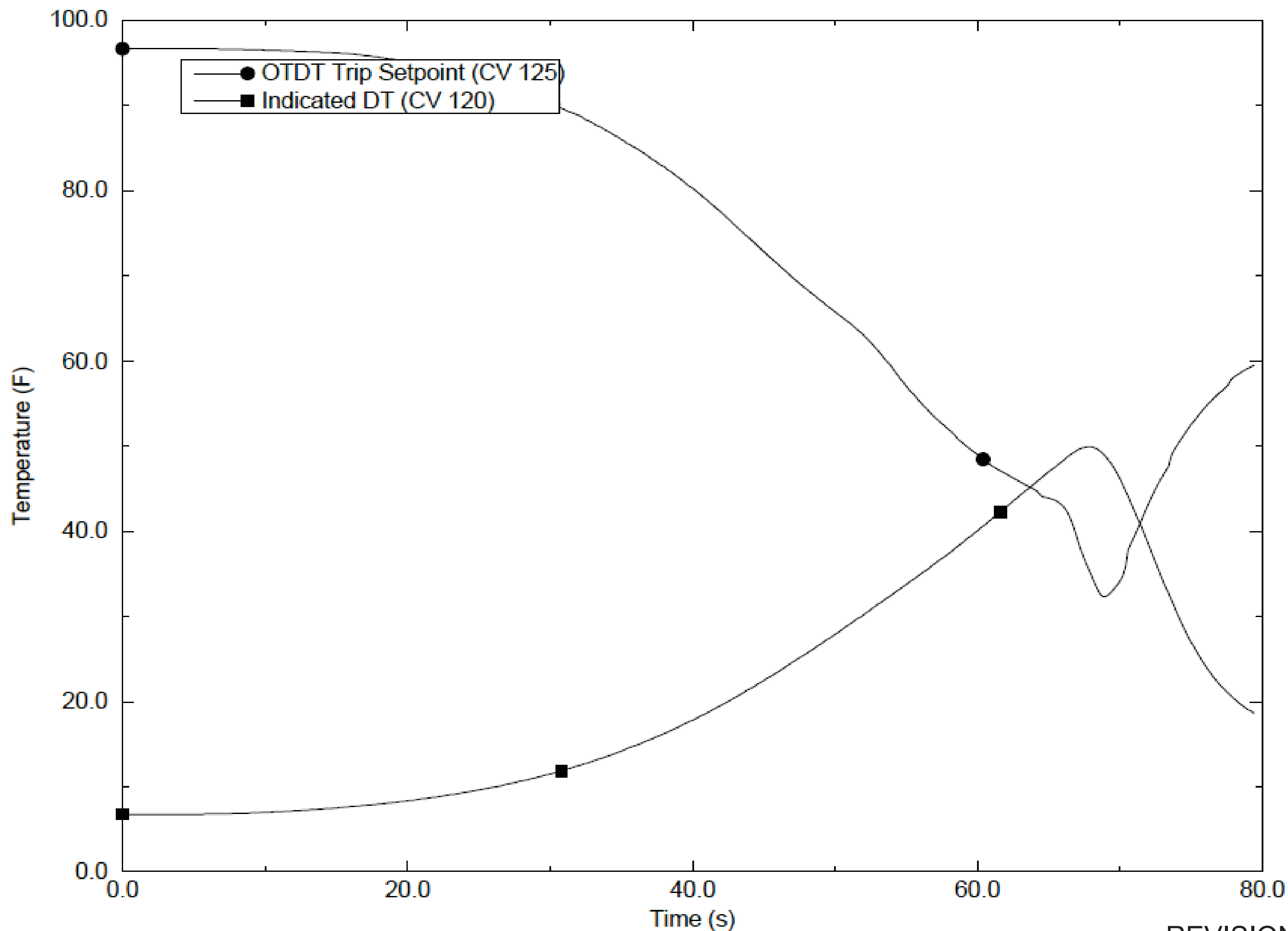
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Indicated Core Power	FIGURE 15.4.2-3



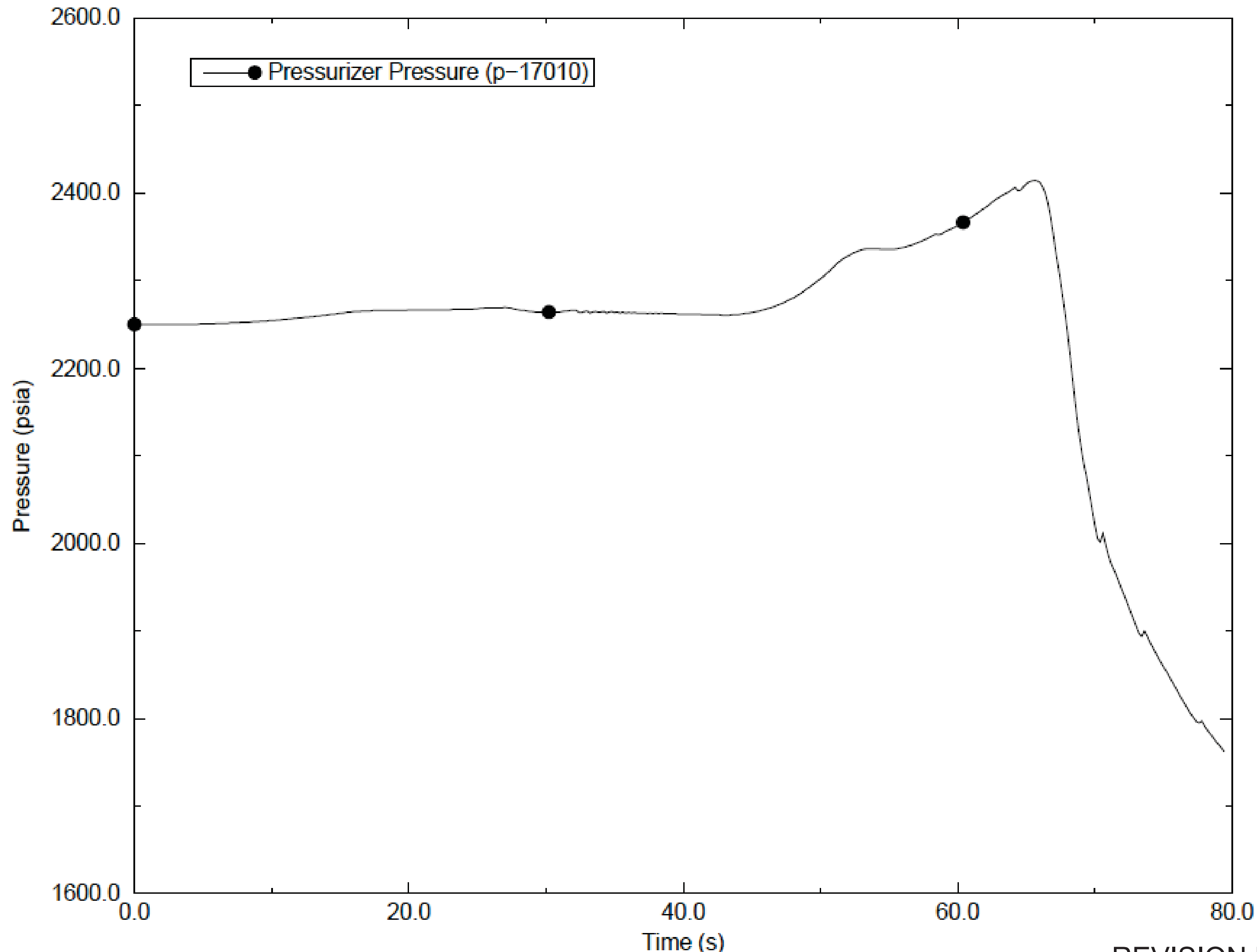
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Reactor Coolant Temperatures	FIGURE 15.4.2-4



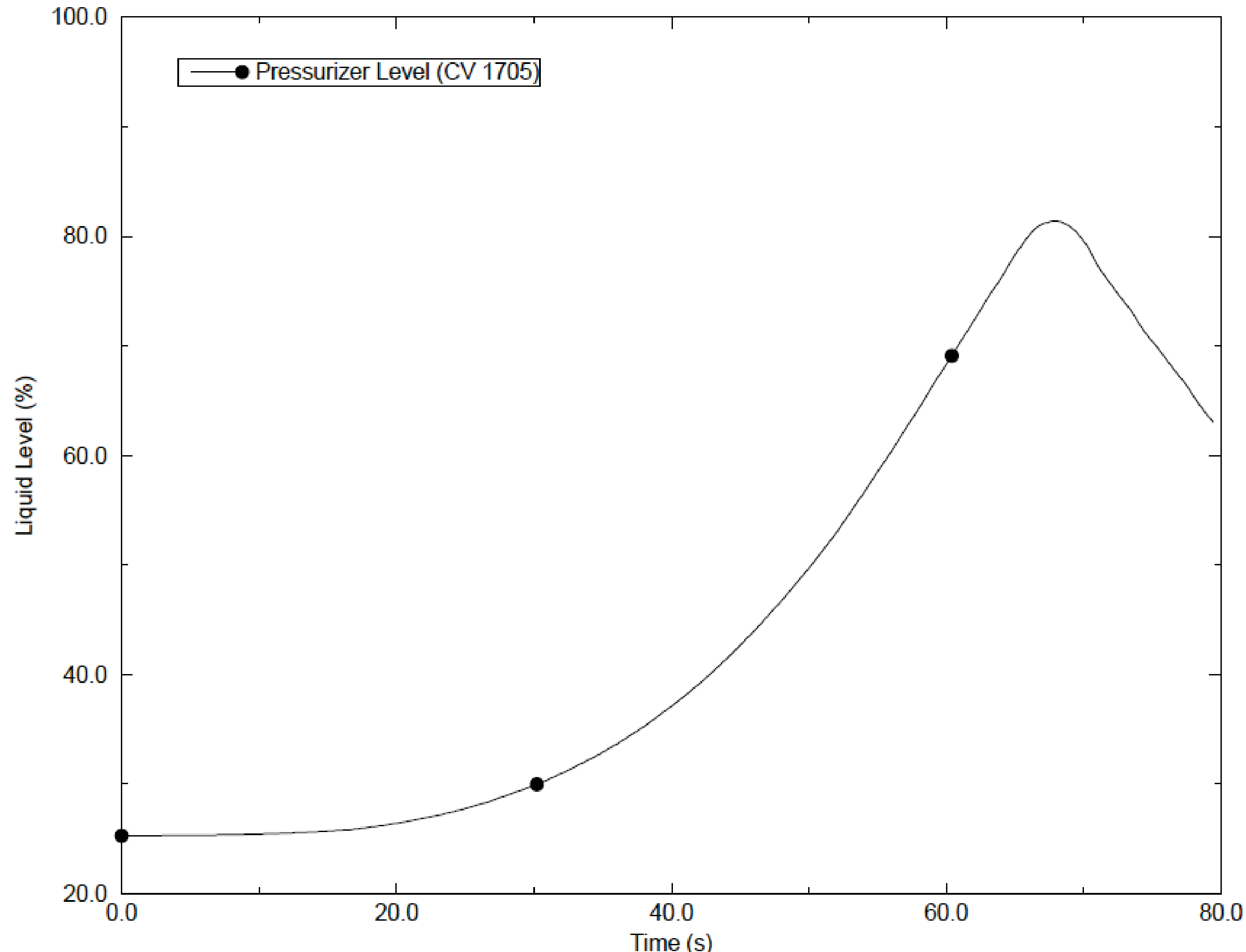
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Reactor Vessel Temperature Rise and OTΔT Setpoint	FIGURE 15.4.2-5



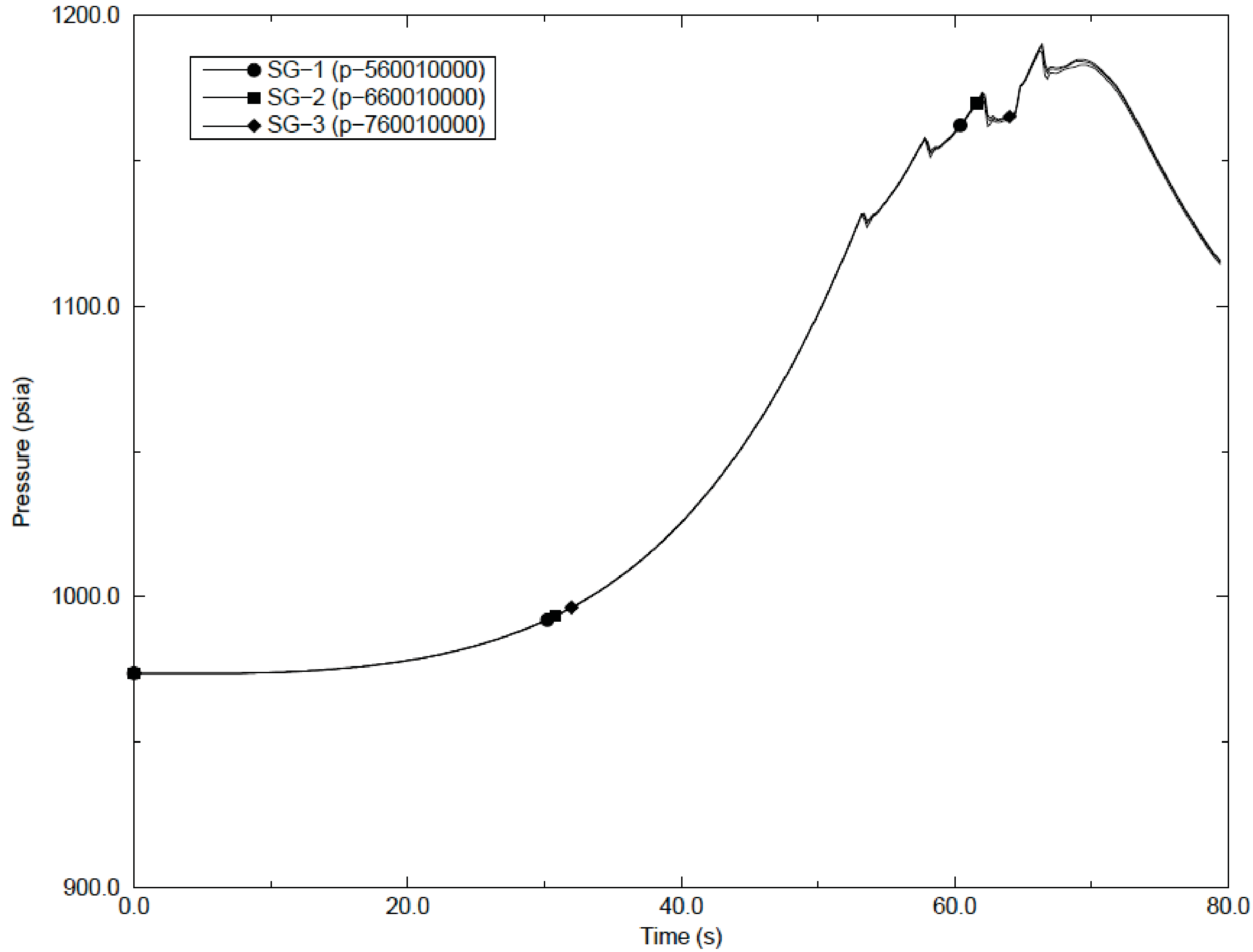
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Pressurizer Pressure	FIGURE 15.4.2-6



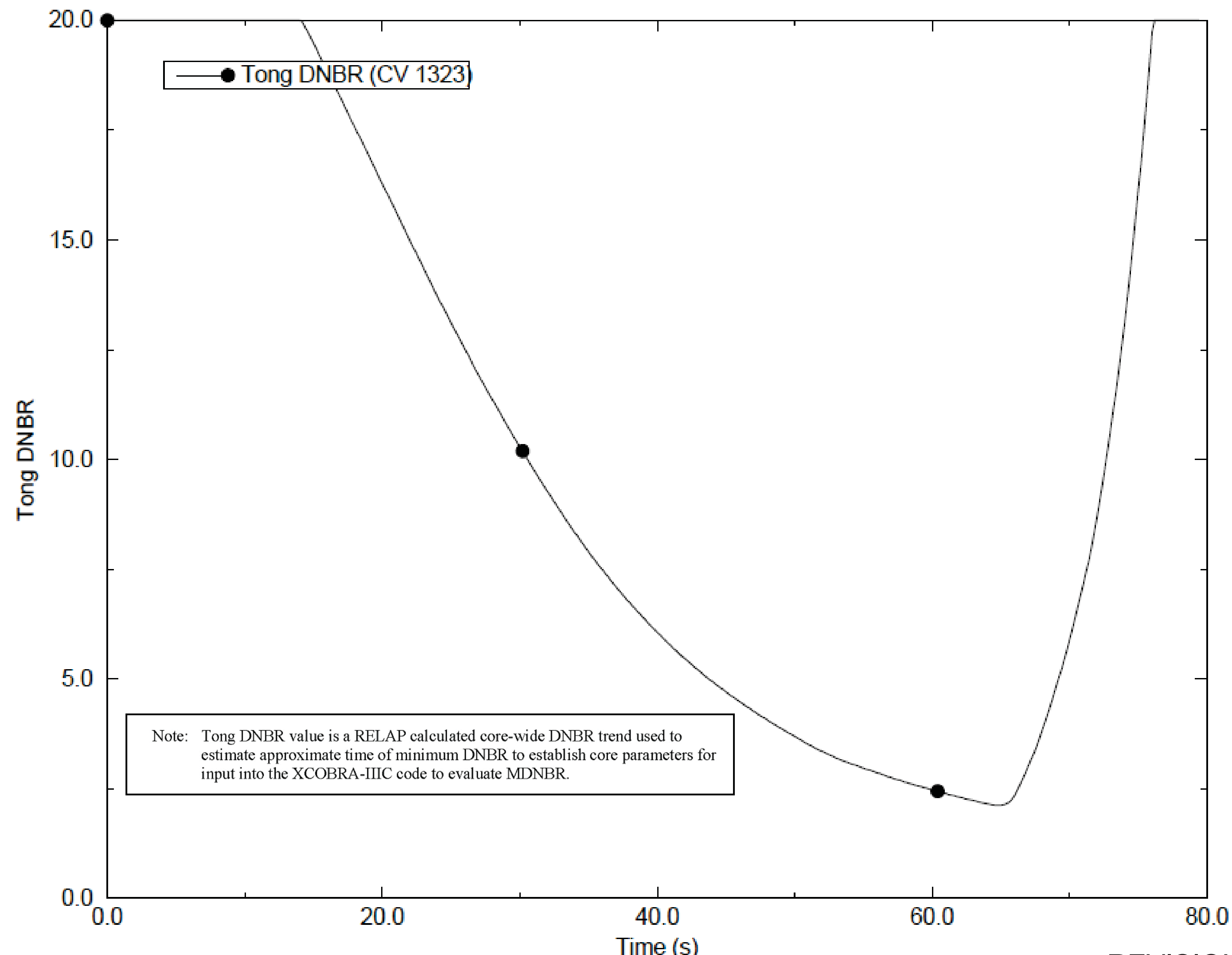
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Pressurizer Liquid Level	FIGURE 15.4.2-7



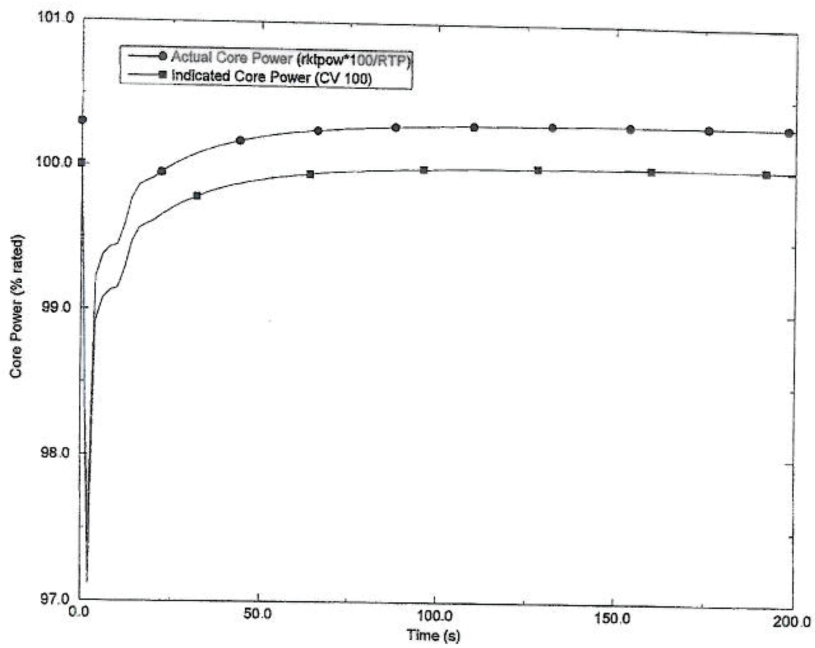
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: Steam Line Pressures	FIGURE 15.4.2-8



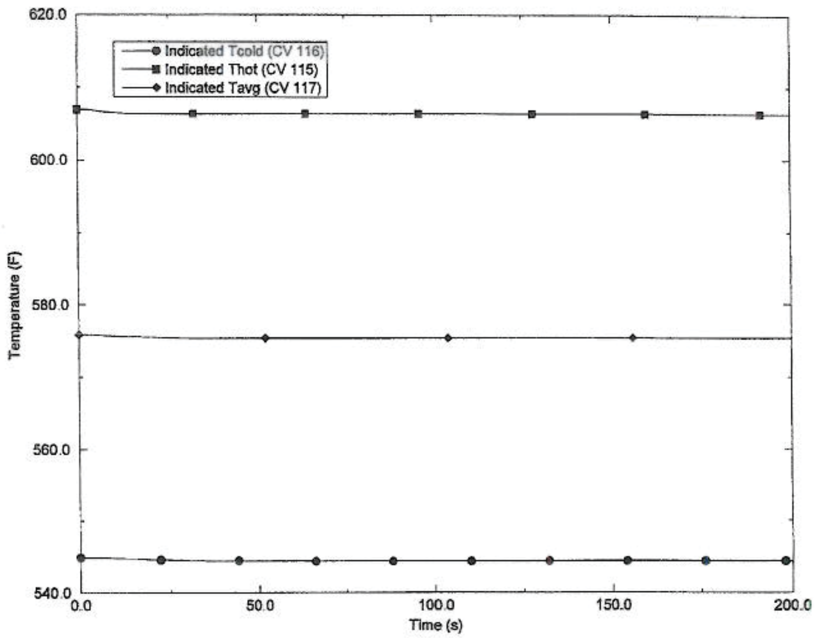
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Limiting Bank Withdrawal at Power: DNBR Trend	FIGURE 15.4.2-9



REVISION NO. 22

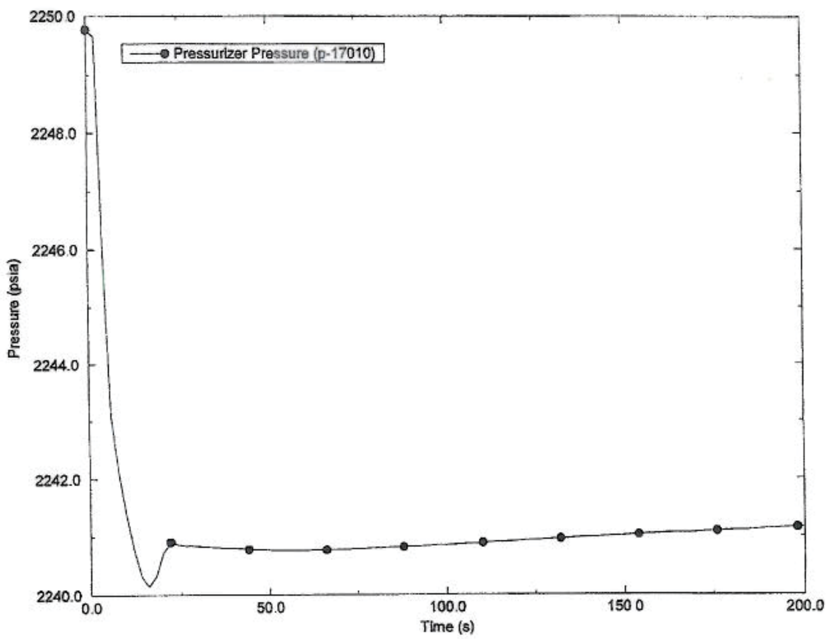
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: Core Power	FIGURE 15.4.3-1



REVISION NO. 22

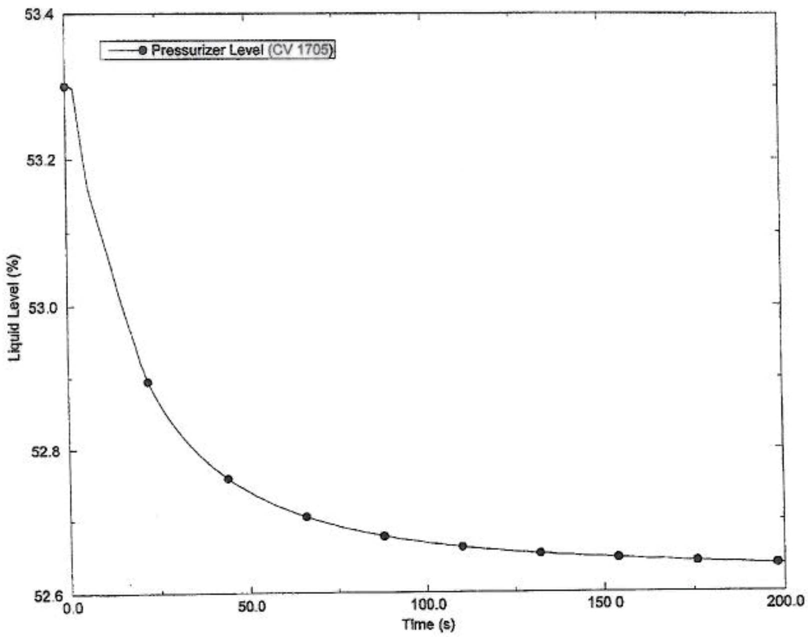
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: Reactor Coolant Temperatures	FIGURE 15.4.3-2

Figure 15.4.3-3 Deleted By Revision No. 17



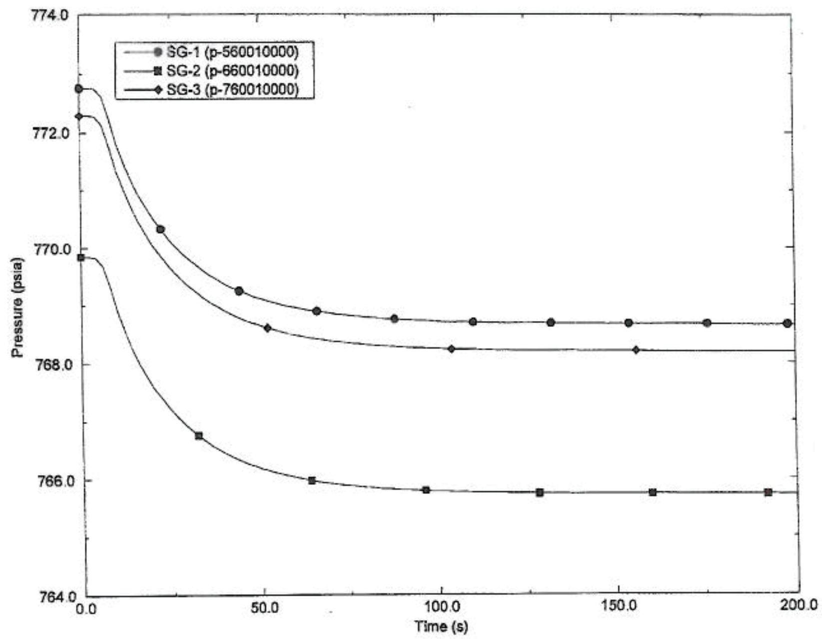
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: Pressurizer Pressure	FIGURE 15.4.3-4



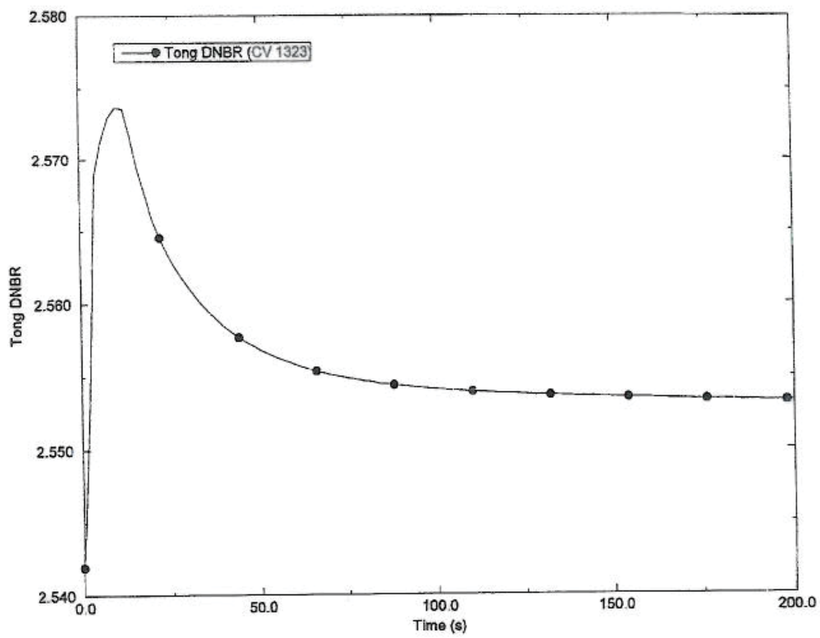
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: Pressurizer Liquid Level	FIGURE 15.4.3-5



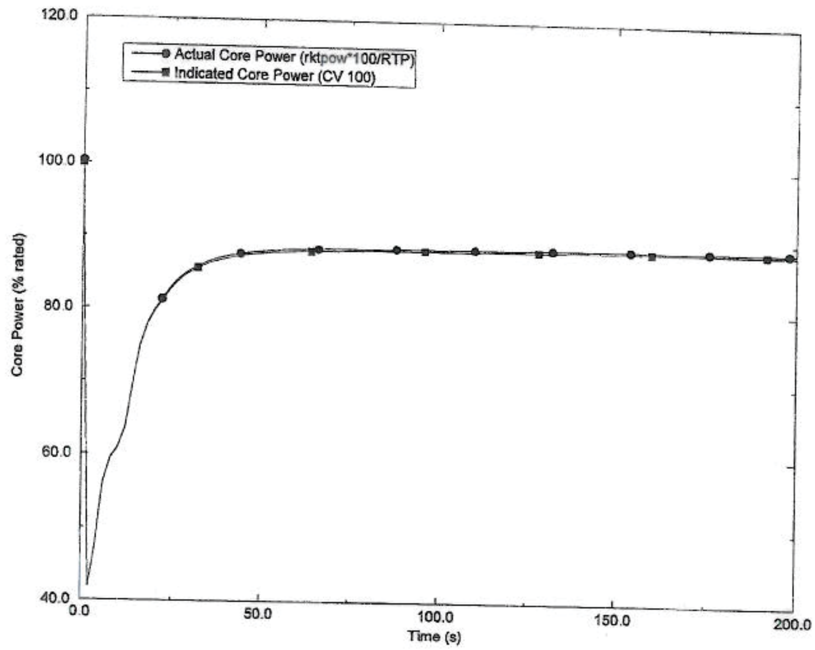
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: Steam Line Pressures	FIGURE 15.4.3–6



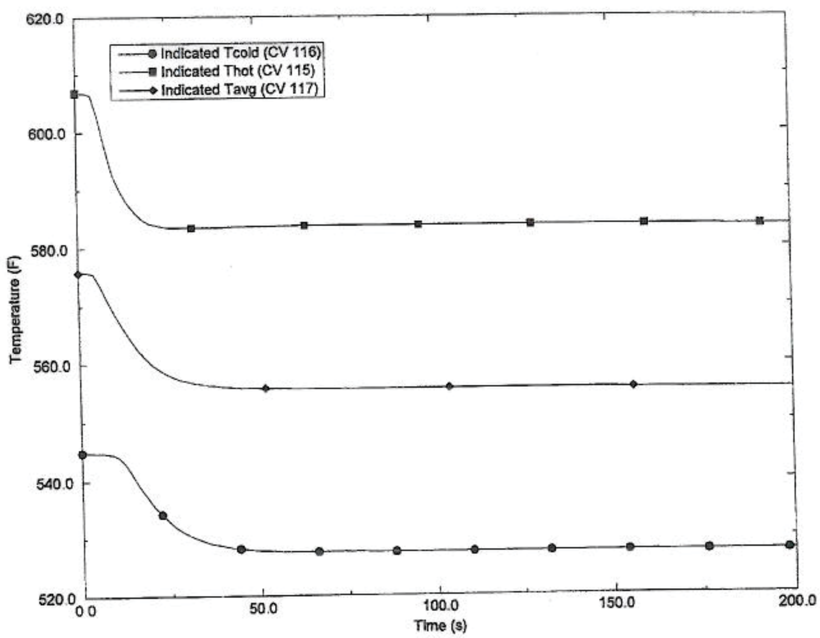
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod: DNBR Trend	FIGURE 15.4.3-7



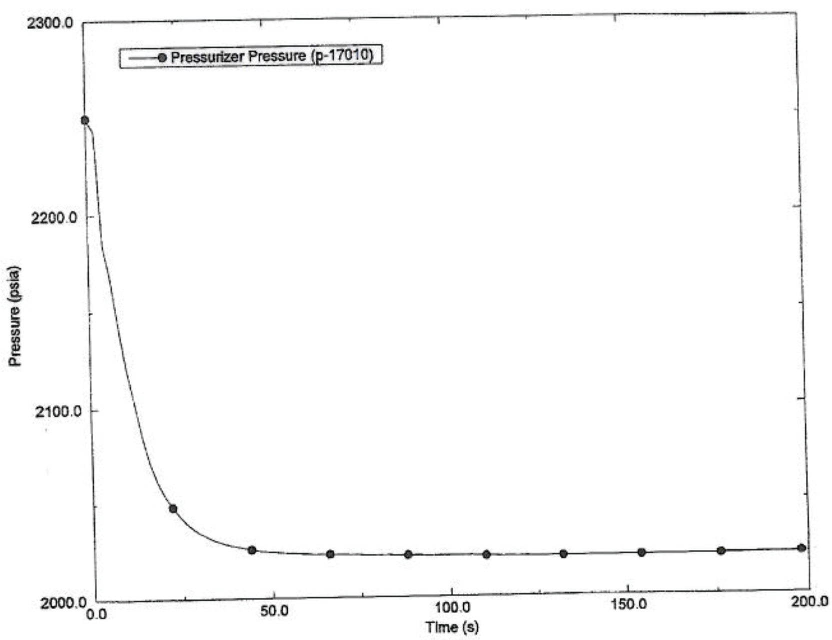
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Bank: Core Power	FIGURE 15.4.3-8



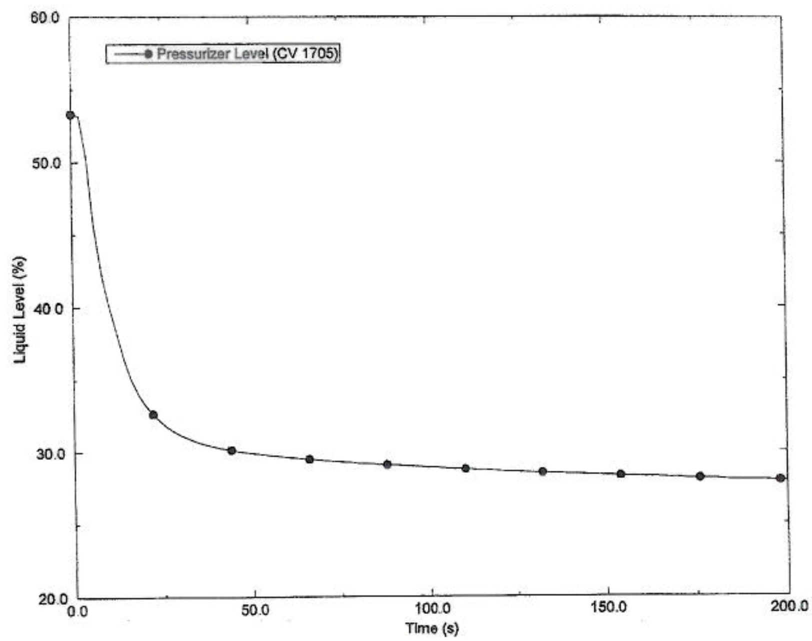
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Bank: Reactor Coolant Temperatures	FIGURE 15.4.3-9



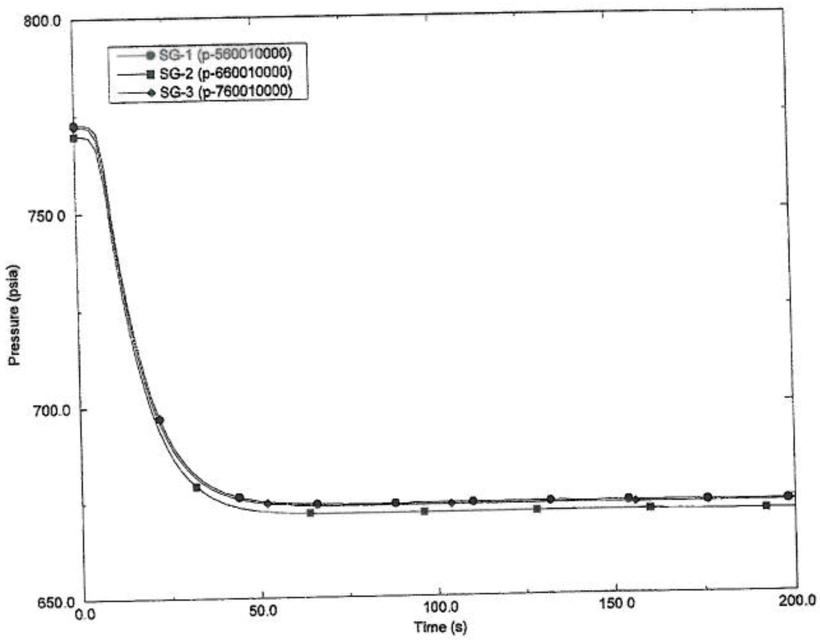
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Bank: Pressurizer Pressure	FIGURE 15.4.3–10



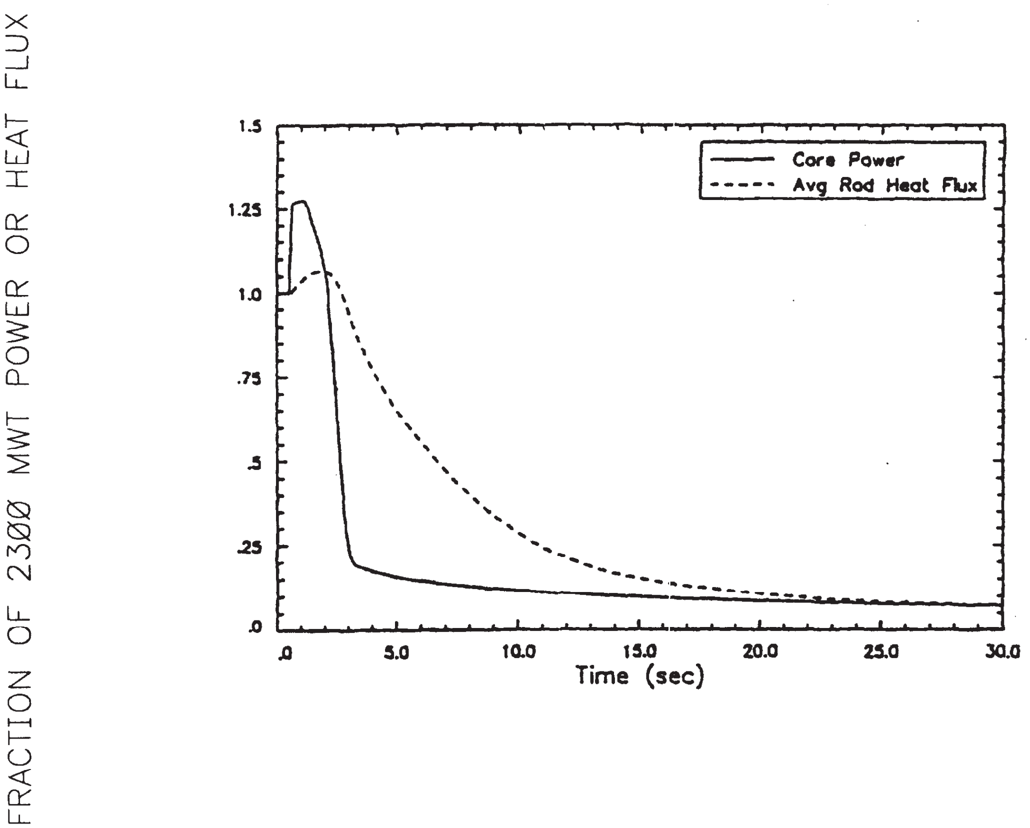
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Bank: Pressurizer Liquid Level	FIGURE 15.4.3-11



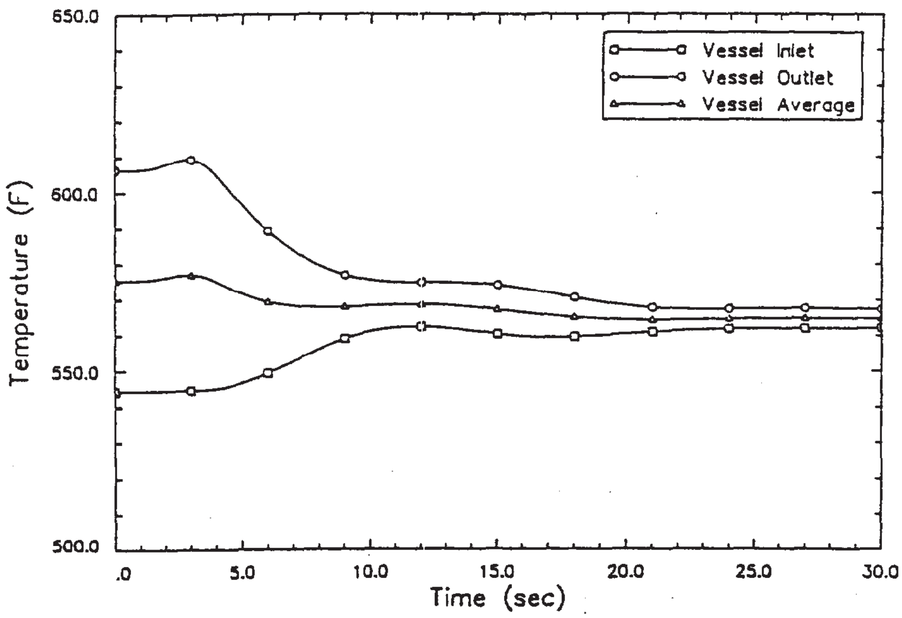
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Bank: Steam Line Pressures	FIGURE 15.4.3-12



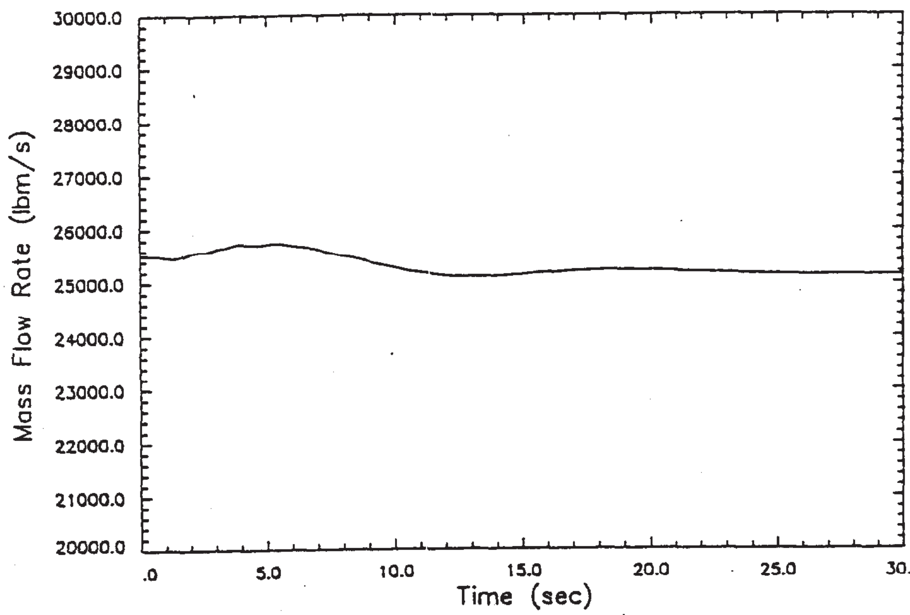
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Roc Ejection: Indicated Core Power Average Rod Heat Flux	FIGURE 15.4.8-1



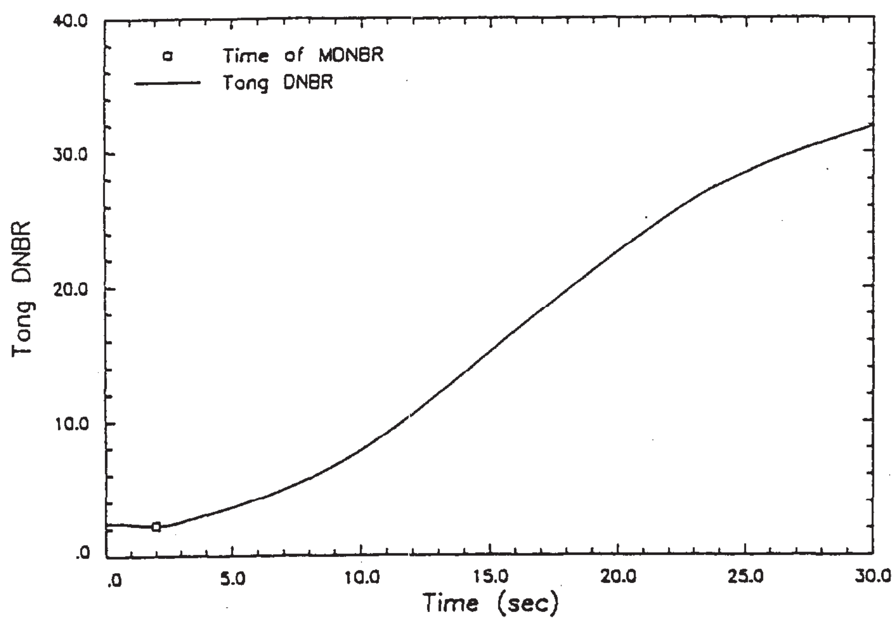
REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Ejection: Reactor Coolant Temperatures	FIGURE 15.4.8-2



REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Ejection: Core Coolant Flowrate	FIGURE 15.4.8-3



REVISION NO. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Rod Ejection: DNBR Trend	FIGURE 15.4.8-4

15.5 INCREASES IN REACTOR COOLANT SYSTEM INVENTORY

Increase in reactor coolant system inventory can be caused by inadvertent operation of the ECCS or primary coolant system charging pumps.

15.5.1 INADVERTENT OPERATION OF EMERGENCY CORE COOLING SYSTEM

The shutoff head of the H. B. Robinson high pressure safety injection system pumps is approximately 1500 psia, which is much less than the analysis trip setpoint pressure of approximately 1800 psia, and therefore, cannot increase the primary inventory during power operation.

PRESSURIZED THERMAL SHOCK (historical information)

The following information was incorporated into the UFSAR in 1985 to reflect the status the Pressurized Thermal Shock (PTS) issue and RNP's risk associated with the issue as determined at that time. This discussion has been superseded by implementation of the PTS rule and the current RNP position with regard to PTS is presented in section 5.3 of the UFSAR. Compliance with the PTS screening criteria is addressed in section 5.3 of the UFSAR and is not a criteria addressed in Chapter 15 analyses. This discussion has been retained here for historical purposes.

Pressurized thermal shock (PTS) is being addressed in the Unreviewed Safety Issue program A-49. Typical Combustion Engineering, Babcock & Wilcox and Westinghouse early design operating plants were modeled in this effort. The plants modeled were Calvert Cliffs, Oconee, and H. B. Robinson Unit 2. Approximately 200 cases have been analyzed in the thermal hydraulics portion of the H. B. Robinson program. Representative events examined were steam line break, loss of coolant accidents, and arbitrarily large step changes in coolant temperature.

Break spectrums were examined with the specific objective of achieving stagnation conditions in the primary system. In each event when primary pressure dropped below 1300 psia, the reactor coolant pumps were shut off. As required by the reactor protection logic, the safety systems were enabled injecting cold ECC water. All events were initiated at hot zero power or at power conditions in order to bound lower temperature operations. Thus, the effect of inadvertent operation of the ECCS in stagnant conditions in addition to a much broader spectrum of more limiting events has been addressed.

Probabilistic fracture mechanics analysis using these thermal hydraulic results is in progress. While not yet completed, extremely low probability of reactor vessel failure is indicated from preliminary results.

To further address the concerns of this issue, Carolina Power & Light is implementing a low radial leakage fuel management program and is installing part length shielding fuel assemblies. These actions assure that H. B. Robinson 2 will not reach the NRC screening criteria for RT_{NDT}.

The Westinghouse Owners' Group (WOG) has previously addressed this issue. This effort addressed all transients which may subject the reactor pressure vessel to overcooling thermal effects from loss of loop flow. The results of the report support the NRC screening criteria, i.e., plant operation is acceptable if the screening criteria for RT_{NDT} is not reached.

Therefore, the causes and consequences of this event and all other events which could lead to PTS have been addressed by the NRC and WOG programs and need not be further addressed in this license action.

15.5.2 CVCS MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

The consequences of unplanned additions to inventory and effect of reactivity additions due to dilution during refueling and startup are treated in Section 15.4.6. The consequences of dilutions at power are bounded by the analysis of Section 15.4.2, Uncontrolled RCCA Bank Withdrawal at Power. This disposition is not impacted by power uprate to 2339 MWt.

The consequences of volumetric addition and effect on pressure boundary are mitigated by resetting the pressurizer PORV set pressure to 400 psig prior to going below 350 psig. There are two PORVs on the pressurizer, each independently actuated. Any one valve has adequate relief capacity and response time to prevent overpressurization due to malfunction of the CVCS.

15.6 Decreases in Reactor Coolant System Inventory

15.6.1 Inadvertent Opening of Pressurizer Safety or Power Operated Relief Valve

15.6.1.1 Identification of Causes and Event Description

This event is initiated by the failure of a pressurizer PORV or safety valve in the full-open position, which causes loss of Reactor Coolant System (RCS) inventory and rapid depressurization. The primary system pressure decreases rapidly until the pressurizer liquid is depleted and the RCS is stabilized at the saturation pressure of the hot leg. However, the Reactor Protection System will scram the reactor on low pressurizer pressure or OTΔT well before the pressurizer liquid is depleted, terminating a moderator-density-feedback core power transient and further challenge to the SAFDLs.

The challenge to DNB is produced by the rapid depressurization of the primary system. Protection against this challenge is provided by the low pressurizer pressure and the OTΔT trips. In the post scram period, a challenge to fuel integrity can be produced if the core uncovers. The system response (blowdown and depressurization) for an open PORV is bounded by that for a cold leg break which corresponds to a 1.5 inch ID pipe ("small" SBLOCA). The hot rod level mixture may momentarily drop below the top of the active core for 1.5 inch and other small breaks up to ~1.7 inches. However, a single HHSI pump sufficiently quenches the core such that the hot rod cladding and fuel do not substantially heat up. Therefore, it is necessary to analyze this event only until a reactor trip occurs because the event does not result in a more limiting transient.

This event is primarily a depressurization event, but with a negative moderator density coefficient, power increases slightly, as well. Thermal margin is eroded by the significantly decreased pressures and the slightly increased power.

The objective of this analysis is to evaluate the ability of the low-pressurizer-pressure trip to protect thermal margin during a rapid depressurization. Consequently, the OTΔT trip was disabled for this analysis.

The event is classified as a Condition IV event (Table 15.0.1-1). The acceptance criterion is demonstrating that the radiological consequences meet 10 CFR 50.67 and Regulatory Guide 1.183 guidelines. The systems challenged in this event are redundant; no single active failure will adversely affect the consequences of the event.

15.6.1.2 Analysis Method

The analysis was performed using the ANF-RELAP and XCOBRA-IIIC codes. The ANF-RELAP code (Reference 15.0-3) was used to model the salient system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures). A DNBR calculation was performed to estimate the approximate time at which the DNBR was a minimum. The core fluid boundary conditions and average rod surface heat flux at this time were then used as input to the XCOBRA-IIIC code (Reference 15.0-4), which was used to evaluate the MDNBR.

15.6.1.3 Definition of Events Analyzed and Bounding Input

This event is principally of concern in the short term because of the potential challenge to the DNB SAFDL, due to depressurization before scram. The depressurization also has a small effect on core power. However, the core inlet coolant temperature and flow remain essentially constant during the transient.

A single case, at full-power conditions, was analyzed. Lower power levels present a less severe challenge to DNB.

The reactivity feedback due to the density change produced by the depressurization was derived from the maximum moderator temperature coefficient. Because a moderator temperature coefficient represents the reactivity feedback due to temperature-induced density changes (based on the thermal expansion curve for water), the reactivity change due to a given density change was set equal to the maximum moderator temperature coefficient times the temperature change which corresponded to the density change (using water property tables).

This event can be caused by the malfunction of either a pressurizer PORV or a pressurizer safety valve. Failure of a safety valve was analyzed, since the flow capacity of a safety valve (293,330 lb/hr) is larger than the flow capacity of a relief valve (255,600 lb/hr), and a malfunction of the larger-capacity valve will bound the two possible cases.

The key analysis conditions are summarized below:

Initial power	102% of 2300 MWt
OTΔT trip	Disabled
Low pressurizer pressure trip	Available
Moderator density coefficient	Calculated from Technical Specifications maximum moderator temperature coefficient

15.6.1.4 Analysis of Results

The event was initiated by fully opening a pressurizer safety valve. This caused the pressure in the primary system to decrease as fluid was lost through the open valve (see Figure 15.6.1-3). A low-pressure trip signal was issued at 44.5 seconds when the pressurizer pressure was 1863 psia. The lead filter on the compensated pressurizer pressure signal accounts for the trip occurring at a pressure higher than the 1800 psia setpoint. Reactor scram was initiated a second later (at 45.5 seconds). This ended the slow power excursion (see Figure 15.6.1-1) caused by reactivity feedback of the reduced coolant density at lower pressures.

The core-average rod surface heat flux peaked at 107% of 2300 MWt at 45.7 seconds (see Table 15.6.1-1 and Figure 15.6.1-1). The coolant temperatures remained fairly constant until reactor scram occurred (see Figure 15.6.1-2).

The minimum DNB ratio calculated for this event is 1.228 (see Table 15.6.1-1), which provides margin relative to the 1.141 DNB limit.

15.6.1.5 Conclusion

The analysis demonstrates that there is no fuel failure or significant radiological release for this event. Therefore the event acceptance criterion is met.

TABLE 15.6.1-1

OPEN PRESSURIZER SAFETY/PORV EVENT SUMMARY

TIME	EVENT	VALUE
0.0 s	Pressurizer safety valve failed fully open	-
44.5 s	Pressurizer pressure reached low-pressure trip setpoint (see Figure 15.6.1-3)	1863 psia actual 1800 psia compens.
45.5 s	Scram rod insertion began	-
45.5 s	Core power peaked	108% of 2300 MWt
45.7 s	Core-average rod surface heat flux peaked (see Figure 15.6.1-1)	107% of heat flux corresponding to 2300 MWt
45.9 s	Minimum DNBR occurred[a]	1.228

[a] For this transient event, the average rod heat flux (see Figure 15.6.1-1) serves as a better DNBR trend indicator than the Tong DNB correlation (not shown).

15.6.2 SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

15.6.2.1 Identification of Causes and Frequency Classification and Acceptance Criteria

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the Reactor Coolant System (RCS) pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 10% of the cold leg cross sectional area. This event is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 10% of the cold leg cross sectional area in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

The acceptance criteria for the loss-of-coolant accident is described in 10CFR50.46 as follows:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200ºF.

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of zircaloy in the reactor.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

Description of Small Break LOCA Transient

Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one positive displacement charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.295-inch diameter hole. This break results in a loss of approximately 10.6 lb/sec.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in the core and cause a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressures.

When the RCS depressurizes to a minimum pressure of 615 psia, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of off-site power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses. Operator Action times for tripping the reactor coolant pumps has been determined analytically to be 6 minutes (Reference 15.6.2-6).

As described in UFSAR Sections 6.2 and 6.3, the Emergency Plant Procedures have provisions for beginning realignment of the ECCS when the RWST level falls to 27%. Specifically, the ECCS can be realigned such that the RHR pump takes suction from the containment sump and discharges to the suction of the SI pumps and the containment spray pumps. This alignment is referred to as the "piggyback" mode of operation. During the period of switchover to the piggyback mode, the SI pump and RHR pump being realigned must be off. Therefore, with the single failure assumption that only one train of SI is available, this leads to there being no ECCS flow to the core for the duration of the switchover.

Once switchover is completed and ECCS flow is re-established, the Small Break LOCA event is effectively terminated, and long term core cooling can be maintained. The small break loss of coolant accident analysis is performed to a duration beyond the time of core quench.

15.6.2.2 Method of Analysis

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10CFR50 (Reference 15.6.2-1). The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10CFR50.

Small Break LOCA Evaluation Model

The analysis was performed with the approved AREVA [formerly known as Framatome ANP (FRA-ANP), Siemens Power Corporation (SPC), Advanced Nuclear Fuel (ANF) and Exxon Nuclear Corporation (ENC)] Small Break LOCA Evaluation Model (Reference 15.6.2-2 and Reference 16.6.2.2a). This methodology is based on two computer codes.

The Reactor Coolant System response is calculated with the S-RELAPS computer code, a modified version of RELAP5 which is a best estimate code to which the 10CFR50 Appendix K required Moody two-phase critical flow model has been added.

The fuel heatup response is calculated with the S-RELAPS computer code. S-RELAPS incorporates conservative fuel heatup models which meet the requirements of 10 CFR 50 Appendix K.

The RODEX2-2A computer code is used to initialize the S-RELAPS fuel rod models prior to the start of the analysis. The RODEX2-2A code conservatively predicts initial fuel rod temperatures and complies with Appendix K requirements.

Small Break Input Parameters and Initial Conditions

Table 15.6.2-1 lists important input parameters and initial conditions used in the small break analyses.

Safety injection flow into the Reactor Coolant System (RCS) as a function of the system pressure is used as part of the input. The SI delivery curve used for these analyses is depicted in Table 15.6.2-2 as a function of RCS pressure.

This table represents injection flow from one high head safety injection (HHSI) pump. The delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. The delivery data were developed based on as-built piping layout information and a composite minimum pump curve (based on system test performance) degraded by 5% of the design TDH. Other assumptions used for the development of the delivery data include no branch line header balancing, and the pump minimum flow path remains open throughout the entire injection phase.

The effect of flow from the RHR pumps is considered here but no flow occurs since their shutoff head is lower than RCS pressure during the time portion of the transient considered here.

The Safety Injection System was also assumed to initiate delivery to the RCS 40 seconds after the generation of a safety injection signal. This delay time includes the time required for diesel start up and loading of the safety injection pumps onto the emergency buses.

The worst single active failure is one Emergency Diesel Generator that does not start. With loss of offsite power, failure of one emergency electrical bus results in the loss of one HHSI pump and one of two Motor Driven Auxiliary Feedwater Pumps. The remaining Motor Driven Auxiliary Feedwater Pump in conjunction with the Steam Driven Auxiliary Feedwater Pump will supply flow to all three steam generators. With the failure of one of two HHSI pumps that automatically start, only a single HHSI pump is available to mitigate the Small Break LOCA. (The third HHSI pump is an installed spare that is not automatically supplied with electric power.)

For the low pressurizer pressure trip setpoint, the SBLOCA analysis uses a setting of 1800 psia with dynamic compensation consisting of a 1.1 second lag filter and a 9 second lead filter (Reference 15.6.2-6).

15.6.2.3 Small Break Results

A range of small break analyses is presented which establishes that the limits of 10 CFR 50.46 will not be exceeded at 100% of licensed core power operation. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4 (Reference 15.6.2-6).

As indicated in the results of clad heatup, the 2.40 inch diameter break size at End of Cycle conditions is limiting. For this limiting case, Figures 15.6.2-4 through 15.6.2-10 present the principal parameters of interest for the small break ECCS analyses for blowdown:

1. RCS and Steam Generator Pressures
2. Downcomer and Hot Assembly Collapsed Liquid Levels
3. Peak Clad Temperature
4. Combined High Head Safety Injection Flow
5. Break Flow Rates
6. Combined Accumulator Flow
7. RCS and Reactor Vessel Fluid Masses

The maximum calculated Peak Cladding Temperature for the Small Breaks analyzed is 1492°F. After error corrections, the Peak Cladding Temperature is 1552°F. These results are well below all acceptance criteria limits of 10 CFR 50.46 and demonstrate acceptability of operation with one HHSI pump at 100% of licensed core power.

The Small Break LOCA switchover sequence of events was modeled for a spectrum of break sizes as continuations of the Small Break LOCA injection phase calculations. In addition to the modeling assumptions used for the injection phase, the time at which switchover begins was chosen to be conservatively early to maximize decay heat in the core. The time at which switchover should begin was calculated assuming early containment spray activation. For example, spray was started at 3600 seconds for the 1.5 inch and smaller break sizes, consistent with containment pressure response calculated in an analysis that assumed no containment fan coolers in operation. Also, maximum spray flow was assumed (1700 gpm from each of two spray trains) even though a single failure of the emergency diesel would remove one train of spray from the RWST depletion model. The purpose of assuring a maximum spray flow is to deplete the RWST as soon as possible in the analysis to ensure that the PCT occurs at the earliest moment, thus maximizing the decay heat in the core.

This analysis was performed using NRC approved methodology (Reference 15.6.2-2) to determine the results of a 10 minute switchover duration. The results of this extended switchover analysis (Reference 15.6.2-3) are included in Tables 15.6.2-3 and 15.6.2-4. The calculated Peak Cladding Temperature during the switchover is 900ºF. This result is below the Small Break LOCA maximum Peak Cladding Temperature which occurs during the blowdown phase of the event.

15.6.2.4 Small Break Conclusions

The results of the Small Break LOCA analysis for the blowdown phase and the switchover phase, when analyzed at 102% of 2300 MWt, are well below acceptance criteria limits of 10 CFR 50.46 and demonstrate acceptability of operation with one head HHSI pump.

TABLE 15.6.2-1

INPUT PARAMETERS USED IN THE SBLOCA ANALYSIS

Parameter	Value
Reactor Power, (including 7 MW uncertainty) MWt	2346
Radial Peaking Factor ($F_{\Delta H}$) (includes uncertainty)	1.8
Total Power Peaking Factor (F_Q) (includes uncertainty)	2.5
Elevation of Peak LHGR (fraction of core height)	0.85
RCS Flow Rate (minimum) (gpm)	258100
RCS Primary Volume, ft^3	9143
Pressurizer Pressure (nominal), psia	2249.7
RCS Operating Temperature (nominal), ºF	575.9
Reactor Vessel Volume, ft^3	3635
Pressurizer Total Volume, ft^3	1318
Accumulator Volume, ft^3 (single accumulator)	1200
Accumulator Water Volume (nominal), ft^3	833
Accumulator Pressure (minimum), psia	614.7
Accumulator Fluid Temperature (maximum), ºF	130.0
Total Number of Tubes per SG	3214
SG Tube Plugging, %	6
Secondary Flow Rate/SG, lbm/hr	$3.43X10^6$
SG Secondary Pressure (nominal), psia	~780
MFW Temperature at 100% RTP (nominal), °F	440.0
AFW Temperature (maximum), ºF	115
AFW Pump Delay Time on SIAS (LOOP), sec	105
HHSI, and LHSI/RHR Fluid Temperature, ºF	100
Pressurizer Pressure – Low Reactor Trip (minimum), psia	1799.7
Reactor Scram Delay on Low Pressurizer Pressure, (maximum) sec	1.5
SIAS Activation Setpoint Pressure (minimum), psia	1674.7
HHSI Pump Delay Time on SIAS (LOOP), sec	40
MSSV lift pressures (nominal; includes 3% tolerance), psia	1132.2
	1158.0
	1173.5
	1188.9
RWST Level for Switchover Initiation, %	N/A
Maximum Containment Spray Flow Rate, gpm	3400

TABLE 15.6.2-2

HHSI DELIVERY USED IN THE SBLOCA ANALYSIS

HHSI Flow		**LHSI Flow**	
Primary Pressure (psia)	Mass Flow Rate (lbm/s)	RCS Cold Leg Pressure (psia)	Mass Flow Rate (lbm/s)
0.0	0.0	1.00	535.47
1.0	69.24	14.70	504.07
14.7	68.85	30.00	446.43
200.0	63.46	35.00	454.04
400.0	57.22	40.00	441.44
600.0	50.40	45.00	427.99
800.0	42.78	50.00	413.44
1000.0	33.93	65.00	368.45
1200.0	22.76	95.00	260.85
1300.0	15.14	120.00	115.21
1350.0	9.91	125.00	59.12
1394.7	0.0	127.85	0.119
		127.86	0.0

TABLE 15.6.2-3

SMALL BREAK LOCA TIME (SECONDS) SEQUENCE OF EVENTS (SEE NOTE 1)

Break Diameter (in.)	1	1.5	2	2.4	2.5	3	4
Break Open	0	0	0	0	0	0	0
Low Pressurizer Pressure Trip	174.5	69.8	34.9	21.5	19.1	10.7	0.7
Low Pressurizer Pressure SIAS Setpoint	192	84.3	47.7	33.25	30.6	21.3	12.1
HHSI Flow Begins	~425	~246	110	74	71	61	50
Setpoint to Start Aux. Feedwater Pump	176	71.3	36.4	23	20.6	12.2	2.2
Loop Seal 1 Clears	4568	1628	928	676	608	410	264
Loop Seal 2 Clears	-	-	-	-	-	-	262
Loop Seal 3 Clears	-	-	-	-	-	-	-
Break Uncovers	4614	1732	946	688	626	424	286
Core Uncovery Begins	~4600	~4500	~2250	~1250	~1325	~800	~170 ~230
Accumulator Injection Begins	-	-	3540	1910	1920	1096	602
PCT Occurs	1	6307	3533	1934	1940	1156	267
Beginning of Switchover Calculation							
SI Interruption time	7146	7046	6221	Note 2	Note 2	3867	Note 3
SI Reactivation time	7746	7646	6821	Note 2	Note 2	4467	Note 3
Switchover PCT Occurs	No Heatup	7842	No Heatup	Note 2	Note 2	5000	Note 3
End of Switchover Calculation	8500	8000	7500	Note 2	Note 2	5000	Note 3

Note 1: SBLOCA cases prior to switchover were performed for break sizes of 1, 1.5, 1.7, 1.8, 1.9, 2, 2.2, 2.3, 2.35, 2.4, 2.45, 2.5, 2.6, 2.75, 2.9, 3, 3.5, 4, 5, 6, 7, 8, and 9 inches. SBLOCA switchover cases were performed for 1, 1.5, 2, 3, and 4 inches. For simplicity, only the 1, 1.5, 2, 2.5, 3, 4 and the limiting break size of 2.4 inches are displayed in this table.

Note 2: The switchover analysis was not performed for this break size. For breaks larger than about 1.5”, enough accumulator injection occurs prior to the switchover time to ensure some degree of cooling in the downcomer, and to ensure sufficient reactor vessel inventory due to accumulator and SI flow, at the moment when SI flow is interrupted.

Note 3: The switchover results for 4 inches are not reported because the 2 and 3 inch cases demonstrated improving conditions as break size increased.

TABLE 15.6.2-4

SMALL BREAK LOCA FUEL CLADDING RESULTS

Break Diameter (in)	**1**	**1.5**	**2**	**2.4**	**2.5**	**3**	**4**
Time of Hot Rod Burst	-	-	-	-	-	-	-
Peak Clad Temp (ºF)	686	907	1160	1492	1297	1139	728
Time of PCT (sec)	1	6307	3533	1934	1940	1156	267
PCT Elevation (node)	27	29	31	31	31	29	27
Time of Rupture (sec)	-	-	-	-	-	-	-
Core Wide Oxidation (%)	0.0002	0.0004	0.0024	0.0203	0.004	0.0007	<0.0001
Local Maximum Oxidation (%)	0.0003	0.0099	0.075	0.44	0.123	0.032	0.0001
Switchover	Note 3		Note 2	Note 1	Note 1	Note 3	Note 1
Peak Clad Temperature (ºF) (includes calorimetric uncertainty)		900					
Time (sec)		7842					
Elevation		11					

Note 1: SBLOCA calculations prior to switchover were performed for break sizes of 1, 1.5, 1.7, 1.8, 1.9, 2, 2.2, 2.3, 2.35, 2.4, 2.45, 2.5, 2.6, 2.75, 2.9, 3, 3.5, 4, 5, 6, 7, 8, and 9 inches. SBLOCA switchover calculations were performed for 1, 1.5, 2, 3, and 4 inch break sizes. For simplicity, only the 1, 1.5, 2, 2.5, 3, 4, and the limiting break size of 2.4 inches are displayed in this table.

Note 2: TOODEE2 calculations showed that, for the 2 inch break, no heatup occurred during switchover.

Note 3: TOODEE2 switchover calculations were not performed for the 1, 3, and 4 inch break switchover calculations since the ANF-RELAP calculations did not show a heatup or large void fractions during SI interruption.

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

15.6.3.1 Event Consequences

This event is assumed to be caused by the instantaneous rupture of a steam generator tube which relieves to the lower pressure secondary. The event is similar to the primary valve malfunction event, Section 15.6.1, except the primary fluid relieves to the faulted steam generator. The primary valve malfunction event was analyzed and results reported in Section 15.6.1. That analysis demonstrated that the SAFDLs were not penetrated. Therefore, no fuel failures are expected for that event.

The results of SAFDL evaluation for this event are bounded by those of Section 15.6.1. The primary release flow for event 15.6.1 was 293,330 lb/hr, 80 lb/sec. The maximum relief flow calculated for this event was 72.8 lb/sec. Therefore, the results regarding challenge to the SAFDLs are bounded by those of the primary valve malfunction. No fuel failure is expected for this event.

15.6.3.2 Radiological Consequences

15.6.3.2.1 Introduction

The NRC has approved implementation of the Alternative Source Term methodology (Reference 15.0.12-3) for analysis of the radiological consequences of this event (Reference 15.6.3-1).

The primary consequence of this event is the release of radioactivity from the primary coolant. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. Valves in the condenser bypass lines would automatically close to protect the condenser, thereby causing steam relief directly to the atmosphere from the steam generator relief valves. This direct relief would continue until the faulty steam generator is isolated. The isolation is assumed to require 30 minutes.

The steam generator is isolated on the secondary side by closing associated inlet and outlet secondary valves. Steam dumps/secondary side power operated relief valves (PORVs) may be used for controlling secondary side pressure. The methods, implemented by site procedures, the plant has chosen to depressurize from the primary side of the affected steam generator are, in order of preference: (1) normal pressurizer spray; (2) pressurizer power operated relief valves (PORVs); (3) auxiliary pressurizer spray, and; (4) balancing charging/letdown or using unaffected steam generators for cooldown/depressurization. It should be noted that the function of depressurizing from the primary side to isolate the affected steam generator for this event is not considered to be a design basis or safety related function for any of the equipment listed above.

15.6.3.2.2 Break Flow Calculation

The RELAP5 computer code was used to model the H. B. Robinson Unit 2 steam generator secondary side so that the fluid conditions upstream of a stuck open PORV could be estimated. The stuck open PORV is the path for the primary coolant to escape from the faulted steam generator. Major calculation assumptions were that:

a) The primary pressure conservatively remained at 2280 psia instead of dropping and then recovering as would realistically be expected.

b) One-third of the core energy was removed by the faulted steam generator.

c) All pump and cooldown energy was conservatively removed by the faulted steam generator.

d) The decay heat used was 120 percent of the ANS standard with an infinite 100 percent power history.

e) The secondary side wall temperature was set low to maximize heat transfer out of the primary coolant in the faulted steam generator.

All of these assumptions maximize the discharge, and therefore will provide bounding results.

A matrix of four breaks was analyzed: normal hot leg, normal cold leg, and the hot and cold leg breaks at the cold and hot leg temperatures, respectively. This matrix was done to assure the bounding break was analyzed. The range of PORV discharge flows from this matrix was from 93,872 lbm to 95,495 lbm, with the maximum value occurring for a cold leg break at the hot leg temperature. The maximum primary to secondary transfer was 131 klbm occurring during the hot leg break at cold leg temperatures.

15.6.3.2.3 Dose Analysis Assumptions

The reactor coolant activity concentration is the maximum coolant activity allowed by the Technical Specifications (Reference 15.0.12-3, Appendix F). Two cases of iodine spiking are evaluated:

(1) A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (60 μCi/gram DEI-131) permitted by the Technical Specifications (pre-accident iodine spike case).

(2) The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated utilizing a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (0.25 μCi/gram DEI-131) specified in the Technical Specifications. The iodine spike duration is 8 hours (accident induced iodine spike case).

Other assumptions used in the dose analysis:

- No fuel melt or fuel clad breach is predicted for the SGTR event.

- Peak fuel burnup does not exceed 62,000 MWD/MTU and the maximum linear heat generation rate does not exceed 6.3 kw/foot peak rod average power for burnups exceeding 54 GWD/MTU.

- This accident is evaluated with a coincident loss of offsite power.

- The following data is used to calculate the iodine equilibrium appearance rate:

Maximum Nominal Letdown Flow:	120 gpm @ 130°F, 2235 psig
Uncertainty Applied to Letdown Flow:	10%
Maximum Identified RCS Leakage:	10 gpm
Maximum Unidentified RCS Leakage:	1 gpm

- The activity release from the breached fuel clad is based on the following gap inventory fractions.

Krypton 85:	10%
Other Noble Gases:	5%
Iodine 131:	8%
Other Halogens:	5%
Alkali Metals (Cs, Rb):	12%

- The volume of the fluid of the RCS is 8254 ft^3 (minimum volume, used to determine the RCS concentration) and 9623 ft^3 (maximum volume, used to determine iodine equilibrium appearance rate) at 575.9°F and 2235 psig.

- The activity released from the fuel is instantaneously and homogeneously mixed through the primary system.

- RCS activity conservatively remains constant throughout the Pre-Accident Iodine Spike SGTR event (no dilution of the RCS activity from the safety injection system is considered). Additionally, RCS mass remains constant throughout the SGTR event (no change in the RCS mass is assumed as a result of the rupture flow within the SGTR or from the safety injection system). For the Accident Induced Iodine Spike SGTR event, a similar assumption is made with the exception that the iodine activity increases during the first 8 hours of the transient as a result of release from the defective fuel at a rate of 335 times the iodine equilibrium appearance rate consistent with the Technical Specifications concentration (0.25 μCi/gm DEI-131).

- The primary-to-secondary leak rate in the steam generators is the leak-rate-limiting condition for operation specified in the Technical Specifications of 75 gpd increased by a factor of 2 (150 gpd, which is 0.104 gpm). The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized (a conservatively rounded 0.11 gpm assumed to pass through any one SG and 0.3 gpm total to all three SGs). Since the majority of steam release cooldown will occur in the two intact SGs, it is conservative to assign 0.11 gpm to each of the unaffected SGs with the remainder assigned to the ruptured SG.

- SG volume remains constant for both the Pre-Accident and the Accident Induced spike events and dilution by Auxiliary Feedwater is not considered.

- The integrated mass of the steam released during the SGTR event, based on 3.7°F/hour cooldown rate (within time periods, flow rate is assumed to be constant; between time periods, mass and associated activity release is assumed to be linear) is shown below:

Time	Break Flow In Ruptured SG (lbm)	Steam Release From Ruptured SG (lbm)	Integrated Steam Release from Unaffected SGs (lbm)
0 – 0.5 hour	131,000	95,500	104,640.7
0 – 2.0 hours	131,000	95,500	302,695.8
0 – 8 hours	N/A	N/A	871,641.4
0 – 24 hours	N/A	N/A	2,002,409.4
0 – 53.2 hours	N/A	N/A	3,650,872.3

- The leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG and enter the steam space and is immediately released to the environment with no mitigation. All leakage that does not immediately flash mixes with the bulk water. The radioactivity within the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. The partition coefficient of 100 is utilized for iodine and the alkali metals. Steam generator dryout is not postulated.

- Iodine releases from the SGs to the environment are 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released from normal operations, including iodine spiking.

- The percentage of the SG ruptured tube flow, which flashes and is released directly to the environment, is 30.27%.

- The time required for one RHR train to establish adequate shutdown cooling to terminate releases from the steam generators is 53.2 hours.

- Noble gas radionuclides released from the primary to the secondary system are immediately released to the environment without holdup or mitigation.

- The mass of the fluid of the SGs secondary side is 88,461 lbm/SG (minimum mass) and 137,294 lbm/SG (maximum mass).

15.6.3.2.4 Conclusions

For the SGTR with a pre-accident iodine spike, the 2-hour dose at the EAB is 23.87 rem TEDE. The dose at the LPZ is 1.21 rem TEDE. The Control Room dose at an inleakage of 300 cfm is 4.49 rem TEDE.

For the SGTR with an accident induced iodine spike, the 2-hour dose at the EAB is 1.99 rem TEDE. The dose at the LPZ is 0.10 rem TEDE. The Control Room dose at an inleakage of 300 cfm is 0.37 rem TEDE.

The offsite dose acceptance criterion established by Reference 15.0.12-3 for the pre-accident iodine spike is 25 rem TEDE. The offsite dose acceptance criterion established by Reference 15.0.12-3 for the accident induced iodine spike is that doses should be less than 10% of the 10 CFR 50.67 guideline, or less than 2.5 rem TEDE. The Control Room dose acceptance criterion established by 10 CFR 50.67 for the SGTR is 5 rem TEDE. Therefore, the offsite and Control Room TEDE doses due to a SGTR event meet the dose acceptance criteria.

15.6.4 SPECTRUM OF BOILING WATER REACTOR (BWR) STEAM PIPING FAILURES OUTSIDE CONTAINMENT

This section is not applicable to the H. B. Robinson Nuclear Power Plant.

15.6.5 Loss-of-Coolant Accidents

15.6.5.1 Identification of Causes and Event Consequences

For the purpose of LOCA analyses, a major LOCA is defined as a rupture greater than or equal to 10% of the cold leg cross-sectional area in the Reactor Primary Coolant System piping, including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system up to the first closed valve.

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A SIS signal is actuated when the appropriate setpoint (high containment or low-low pressurizer pressure) is reached. These counter measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

15.6.5.2 Method of Analysis

15.6.5.2.1 Blowdown Phase

The AREVA Inc. Realistic Large Break LOCA Methodology (Reference 15.6.5-3) was used to perform the required analysis. This methodology uses the following computer codes:

1. S-RELAP5

   This is used for calculation of the system response. The field equations are basically the same form as RELAP5/MOD2 with the addition of full two-dimensional momentum equations. This two-dimensional capability is only applied within the reactor vessel in the Realistic Large Break LOCA methodology, but can be applied anywhere in the reactor coolant system through input. The S-RELAP5 code structure was modified to be essentially the same as RELAP5/MOD3. The coding for reactor kinetics, control systems, and trip systems was also replaced from RELAP5/MOD3. Initial fuel conditions are supplied by the realistic fuel performance code, RODEX3A. To be consistent, the fuel deformation and conductivity models from RODEX3A were included in S-RELAP5. Capability for a concurrent calculation of containment backpressure based on the ICECON code was added. S-RELAP5 is documented in topical report EMF-2100 (P) (Reference 15.6.5-9).

2. RODEX3A

   RODEX3A calculates fuel rod performance for Realistic Large Break LOCA analysis. In particular, the initial operating temperature of the fuel pellets (as stored energy) and the internal fuel rod gas pressure are provided as functions of fuel exposure and power history. RODEX3A is documented in topical report ANF-90-145(P)(A) (Reference 15.6.5-10).15.6.5-1

The methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation example (Reference 15.6.5-11). This example outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis. As described in the AREVA Realistic Large Break LOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. This means that values are randomly selected within established ranges. The LBLOCA phenomenological uncertainties are provided in Reference 15.6.5-3. These phenomenological aspects include time in cycle, axial core power shape, break type (guillotine vs. split), and break size.

Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 15.6.5-1 (Reference 15.6.5-1). Note that the nominal values used for some RPS setpoints, ESFAS setpoints, and other inputs. This is consistent with the NRC approved methodology. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Two parameters, Refueling Water Storage Tank (RWST) temperature for pumped ECCS flows and diesel start time, are set at conservative bounding values for all calculations. Where applicable, the sampled parameter ranges are based on technical specification limits or supporting plant calculations that provide bounding values.

One of the parameters that is randomly selected is the availability of offsite power. The alternative possibilities are either Loss of Offsite Power (at the beginning of the accident) or offsite power continuing to be available. In either case, the worst single failure disables one of two Low Head and one of two High Head Safety Injection pumps. With Loss of Offsite Power, this may represent failure of one of two Emergency Diesel Generators. With offsite power available, this corresponds to failure of an Emergency Electrical Bus. As a conservative simplification, both trains of Containment Cooling are assumed to be operable/available because a lower Containment pressure contributes to a calculation of a higher Peak Cladding Temperature.

The scope and time period of the Realistic Large Break LOCA analysis is limited to reflooding of the core and quench to relatively low and stable temperatures. In terms of a longer term response, the switchover to sump cooling water (i.e. Recirculation) for ECCS pumped injection is considered separately.

15.6.5.2.2 Switchover To Recirculation Phase

As described in Section 6.3.2.2.5, during the transfer from the blowdown phase of a Large Break LOCA (LBLOCA) to the recirculation phase the Low Pressure Safety Injection (LPSI) pumps are shutdown and aligned to the containment sump while one High Pressure Safety Injection (HPSI) pump continues to inject RWST fluid into the primary system. If the switchover to the recirculation phase is required soon after the initiation of the LBLOCA, then a single HPSI pump may not inject enough water into the primary system to match the break flow and, during the period when only one HPSI pump is operating, partial uncovery of the core may occur. This second core uncovery can produce a second fuel rod heatup.

If operation of the Containment Spray pumps or the HPSI pumps is required during recirculation, then the LPSI pump must be shutdown again to align the discharge of the LPSI pump to the suction of the HPSI pump. During this period of time there is no ECCS flow to the core and a partial uncovery of the core may occur with a corresponding fuel rod heatup.

These switchover evolutions have been evaluated with a LBLOCA model that is specifically modified to address the phenomena important to this application. The model includes the 10 CFR 50 Appendix K requirements that are applicable to this long-term cooling analysis and consists of the following computer code:

1. S-RELAP5 with integral ICECON

   S RELAP5 is used to model the primary system and secondary side of the steam generators. The governing conservation equations for mass, energy, and momentum transfer are used along with the applicable requirements of 10 CFR 50 Appendix K. The containment analysis code ICECON has been incorporated into, and is now an integral part of, the S-RELAP5 code. Containment conditions are determined concurrently with the system conditions through the use of the integral ICECON/S-RELAP5 link.

The switchover analysis evaluated a spectrum of break sizes and break locations to identify the limiting break. Other parameters used in the LBLOCA switchover analysis are contained in Table 15.6.5-5.

The switchover analysis also evaluated two top-peaked axial power shapes corresponding to the Middle-of-Cycle (MOC) and to the End-of-Cycle (EOC). Since the calculated fuel cladding temperatures for a heatup caused by partial core uncovery are fairly sensitive to the axial power profile, the use of conservative top-peaked MOC and EOC axial power profiles assures that any fuel rod heatup calculated in this analysis is bonding.

Early containment spray activation and maximum containment spray flow was assumed in order to cause the RWST level to reach the switchover point as early as 21 minutes after the initiation of the LBLOCA. Performing the switchover analysis at 21 minutes after initiation of the LBLOCA will result in a conservatively severe transient, since the decay heat will be high and the RCS inventory low.

Although Operations personnel are instructed to align the LPSI pump suction to the containment sump and restart the LPSI flow to the RCS as quickly as possible, the time required to perform the alignment was modeled as 20 minutes to conservatively bound the actual time required to perform the alignment and to assure that any fuel rod heatup calculated in this analysis is bounding.

Likewise, the period of time when there is no ECCS flow to the core while the discharge of the LPSI pump is being realigned to the suction of the HPSI pump has been modeled as 6 minutes to conservatively bound the actual time required to perform the alignment and to assure that any fuel rod heatup calculated in this analysis is bounding.

For Hot Leg breaks, the borated water supplied by the ECCS when mixed with the RCS water and other credible dilution sources is sufficient to maintain the core subcritical following a LBLOCA. For Cold Leg breaks, the combination of boron worth and control rod worth is sufficient to maintain the core subcritical following a LBLOCA (References 15.6.5-14 and 15.6.5-15).

15.6.5.3 Results

In concurrence with GDC 35, two sets of cases were performed for the Realistic Large Break LOCA analysis. One set was run where loss of offsite power (LOOP) was assumed and a second set was run where offsite power was assumed available. The set of 59 cases that predicted the highest PCT (LOOP case) is presented.

A set of 59 transient calculations was performed for the Realistic Large Break LOCA analysis. For each transient calculation, Peak Cladding Temperature (PCT) was calculated for a UO_2 rod and for Gadolinia bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd_2O_3. The limiting PCT (2084°F)[1] occurred in Case 24 for an 8% Gd_2O_3 rod (Reference 15.6.5-1). As a result of random sampling, a few of the characteristics defining the limiting case include

- loss of offsite power,
- core average burnup of 7218 Effective Full Power Hours,
- top skewed axial power shape, and
- double ended break configuration with an area of 3.5831 ft^2 per side (as ~87% of the full intact cold leg cross-sectional flow area).

The time sequence of accident milestones for the limiting transient is characterized in Table 15.6.5-3. Table 15.6.5-2 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit.

A nominal best estimate PCT case was identified as Case 58, which corresponded to the median case out of the 59-case set. The nominal PCT was 1608°F. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 476°F.

Key parameters for the limiting PCT case are shown in Figures 15.6.5-1 through 15.6.5-33. Figure 15.6.5-27 is the plot of PCT independent of elevation; and this figure clearly indicates that the transient exhibits a sustained and stable quench.

1 – Correction of an error results in a PCT of 2088ºF.

No fuel rod heatup occurred in any of the switchover scenarios evaluated. The cladding temperatures remained below 260°F which demonstrates that the ECCS is capable of fulfilling its long-term core cooling function.

15.6.5.4 Conclusions

For break sizes up to and including the double-ended severance of a reactor primary coolant pipe, the Emergency Core Cooling System for H. B. Robinson Unit 2 will meet the acceptance criteria as specified in 10 CFR 50.46, with the 1.80 ($F_{\Delta H}$) limit and the axially dependent power peaking limit of 2.50 ($F_Q T$) (see Table 15.6.5-2).

The criteria are as follows:

1. The calculated peak fuel element clad temperature does not exceed the 2200ºF limit.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The local cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

15.6.5.5 Radiological Consequences

15.6.5.5.1 Introduction

In Reference 15.6.5-5, the NRC approved implementation of the Alternative Source Term dose consequence analysis methodology (Reference 15.0.12-3) for the HBR-2 LOCA analysis. The dose analyses were performed using the RADTRAD computer code.

15.6.5.5.2 Source Term Assumptions

1. The analysis is performed to support operation at up to 2346 MW_{th} Power, including measurement uncertainties. Initial core inventory is given in Table 15.6.5-4.

2. During the design basis LOCA, the release fractions from the damaged fuel, listed by radionuclide groups and release phase are:

Group	Isotopes	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	Xe, Kr	0.05	0.95	1.00
Halogens	I, Br	0.05	0.35	0.40
Alkali Metals	Cs, Rb	0.05	0.25	0.30
Tellurium Metals	Te, Sb, Se	0.00	0.05	0.05
Ba, Sr	Ba, Sr	0.00	0.02	0.02
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	0.00	0.0025	0.0025
Cerium Group	Ce, Pu, Np	0.00	0.0005	0.0005
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0.00	0.0002	0.0002

3. The onset of the core gap release is 0.0 seconds into the LOCA event. The duration of the gap release is 0.5 hours.

4. The early in-vessel core release begins at 0.5 hours with a release duration of 1.3-hours.

15.6.5.5.3 Fission Product Transport/Removal in Containment Assumptions

1. The chemical form of the radioiodine released to the containment is 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.

2. The radioactivity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the primary containment. The release into the containment is terminated at the end of the early in-vessel phase.

3. The primary containment free air volume is 1,958,526 ft^3 (minimum value, flooded containment condition).

4. Spray train “A” provides coverage to 82.9 % of the Containment, while spray train “B” provides coverage to 81.5% of the Containment.

The free air volume of the sprayed volume for spray train ‘A’ is:

$$1,958,526 \text{ ft}^3 * 0.829 = 1,623,618.1 \text{ ft}^3$$

The free air volume of the unsprayed volume for spray train ‘A’ is:

$$1,958,526 \text{ ft}^3 * 0.171 = 334,907.9 \text{ ft}^3$$

The free air volume of the sprayed volume for spray train ‘B’ is:

$$1,958,526 \text{ ft}^3 * 0.815 = 1,596,198.7 \text{ ft}^3$$

The free air volume of the unsprayed volume for spray train ‘B’ is:

$$1,958,526 \text{ ft}^3 * 0.185 = 362,327.3 \text{ ft}^3$$

5. Operation of the spray trains is as follows:

3 minutes	spray initiates
77 minutes	spray terminates
87 minutes	spray initiates
167 minutes	spray terminates

6. The elemental iodine spray removal coefficient is 20 hr^{-1} for both spray trains. The maximum allowable decontamination factor for the elemental iodine spray removal coefficient is 200, which is achieved at 2.01 hours for the “A” spray train and 2.03 hours for the “B” spray train.

7. The particulate iodine spray removal coefficient is 3.684 hr^{-1} for the “A” spray train and 3.627 hr^{-1} for the “B” spray train. These spray removal coefficients include a removal coefficient of 0.2 hr^{-1} for the effect of diffusiophoresis. The maximum allowable decontamination factor for the particulate iodine spray removal coefficient is 50, at which time the removal coefficient is reduced by a factor of 10, which is achieved at 2.66 hours for the “A” spray train and 2.70 hours for the “B” spray train.

8. Two safety-related Containment cooling fans, at 65,000 cfm each, begin operation at 76 seconds. The mixing rate between the sprayed and unsprayed regions was assumed to be 65,000 cfm after 76 seconds.

9. Natural deposition is credited in the containment sprayed volume (when sprays are not operational) and in the containment unsprayed region (including when sprays operational). The natural deposition removal coefficient is 0.1 hr^{-1}.

10. The maximum allowable primary containment leak rate per Technical Specifications is 0.1 % by weight of the containment air for the first 24 hours at P_a of 42 psig. Per Reference 15.0.12-3, the leak rate is reduced after the first 24 hours to 50% of the Technical Specification leak rate.

11. The sump pH is maintained > 7.0.

15.6.5.5.4 ECCS Leakage Assumptions

1. With the exception of the noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the primary containment sump water at the time of release from the core. With the exception of the non-particulate iodines, all radioactive materials in the recirculating liquid are retained in the liquid phase.

2. TRM (Technical Requirements Manual) Section 3.23 requires that the Post Accident Recirculation Heat Removal System leakage shall be ≤1 gph. Per Reference 15.0.12-3, analysis of dose consequences due leakage must consider two times the TRM allowable total leakage limit. Therefore, the ECCS leakage rate is 2 gph, initiating at 21 minutes.

3. The volume of the sump (reactor coolant system, RWST, and 3 Accumulators) is as follows:

21 – 40 minutes	35,850 ft^3
40 – 51.5 minutes	40,889 ft^3
After 51.5 minutes	43,939 ft^3

4. The leak flash fraction is 10% based on an enthalpy balance between the maximum sump fluid temperature conditions and fluid conditions at the ambient conditions expected in the area of the leak, as recommended by Reference 15.0.12-3.

5. The radioiodine that is postulated to be available for release from the sump to the environment is 97% elemental and 3% organic.

15.6.5.5.5 Control Room Ventilation System Assumptions

1. The Control Room free air volume is 20,124 ft^3

2. The Control Room outside air makeup rate during normal operation and during emergency pressurization mode operation is 400 cfm.

3. The Control Room habitability envelope unfiltered inleakage rate is 170 cfm for the first hour and 100 cfm after one hour.

4. The recirculation air flow during the emergency pressurization mode is 2600 cfm.

5. An SI signal initiates the emergency pressurization mode at 35 seconds.

6. The Control Room ventilation filter removal efficiencies are 99% for particulates, and 95% for elemental and organic iodines.

15.6.5.5.6 Other assumptions

1. The offsite and control room breathing rates are:

Offsite (EAB and LPZ)		CR	
Time	Rate (m^3/sec)	Time	Rate (m^3/sec)
0-8 hr	3.5E-04	0-30 days	3.5E-04
8-24 hr	1.8E-04		
1-30 days	2.3E-04		

2. The control room occupancy factor is 1.0 for the first 24 hours, 0.6 for 1 through 4 days, and 0.4 for 4 days through 30 days.

3. The meteorological dispersion factors (X/Q) in sec/m^3 are:

Time Period	EAB	LPZ	Cont. Nearest Point CR	RHR HX Room CR
0 – 2 hours	1.77E-03	8.92E-05	4.15E-03	7.13E-03
2 - 8 hours	1.77E-03	3.50E-05	2.74E-03	5.49E-03
8 - 24 hours	1.77E-03	2.19E-05	1.17E-03	2.29E-03
1 - 4 days	1.77E-03	7.95E-06	8.18E-04	1.71E-03
4 - 30 days	1.77E-03	1.85E-06	6.74E-04	1.37E-03

15.6.5.5.7 Results

The maximum 2-hour dose at the EAB is 24.7 rem TEDE. The 30-day dose at the LPZ is 1.62 rem TEDE. The Control Room dose is 4.51 rem TEDE. Contributing sources to the control room dose include:

Containment Leakage	2.62E+00 rem TEDE
ESF Leakage	1.83E+00 rem TEDE
Containment Shine	3.00E-02 rem TEDE
External Radioactive Cloud	3.31E-02 rem TEDE

The offsite dose acceptance criterion from 10 CFR 50.67 is 25 rem TEDE. The control room dose acceptance criterion from 10 CFR 50.67 is 5 rem TEDE. The above results meet these acceptance criteria.

TABLE 15.6.5-1

REALISTIC LARGE BREAK LOCA ANALYSIS CONDITIONS

	Event	Operating range
1.0	**Plant Physical description**	
	1.1 Fuel	
	a) Cladding Outside Diameter	0.424 in.
	b) Cladding Inside Diameter	0.364 in.
	c) Cladding Thickness	0.030 in.
	d) Pellet Outside Diameter	0.357 in.
	e) Pellet Density	96 Percent of Theoretical
	f) Active Fuel Length	144 in.
	g) Gd_2O_3 Concentrations	2, 4, 6, 8 w/o
	1.2 RCS	
	a) Flow Resistance	Analysis
	b) Pressurizer Location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot Assembly Location	Anywhere in Core
	d) Hot Assembly Type	15X15
	e) SG Tube Plugging	≤ 6 percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal Reactor Power	2346 MWt[(1)]
	b) F_Q	≤ 2.5[(2)]
	c) $F_{\Delta H}$	≤ 1.8
	d) MTC	≤ 0 at HFP
	2.2 Fluid Conditions	
	a) Loop Flow	97.3 Mlbm/hr ≤ M ≤ 113.0 Mlbm/hr
	b) RCS Average Temperature	569.9ºF ≤ T ≤ 581.9ºF[(3)]
	c) Upper Head Temperature	< Core Outlet Temperature

1 Includes 0.3% measurement uncertainties.
2 A value of 2.50 allows the COLR F_Q to be increased if necessary.
3 Sampled range of ±6°F includes both operational tolerance and measurement uncertainty.

TABLE 15.6.5-1 (continued)

REALISTIC LARGE BREAK LOCA ANALYSIS CONDITIONS

	Event	Operating range
	d) Pressurizer Pressure	2219.7 psia ≤ P ≤ 2299.7 psia[4]
	e) Pressurizer Level	43.3 percent ≤ L ≤ 63.3 percent
	f) Accumulator Pressure	614.7 psia ≤ P ≤ 674.7 psia
	g) Accumulator Liquid Volume	825 ft^3 ≤ V ≤ 841 ft^3
	h) Accumulator Temperature	80ºF ≤ T ≤ 130ºF (coupled to containment temperature)
	i) Accumulator fL/D	As-built Piping Configuration
	j) Minimum ECCS Boron	≥ 1950 ppm
3.0	**Accident Boundary Conditions**	
	a) Break Location	Any RCS Piping Location
	b) Break Type	Double-ended Guillotine or Split
	c) Break Size (each side, relative to cold leg pipe area)	0.24 ≤ A 1.0 Full Pipe Area (split) 0.24 ≤ A 1.0 Full Pipe Area (guillotine)
	d) Worst Single-failure	Loss of One Train of ECCS
	e) Offsite Power	On or Off
	f) LPSI Flow	Bounding Minimum of Current Pump Delivery
	g) HPSI Flow	Bounding Minimum of Current Pump Delivery
	h) Safety Injection Temperature	110ºF
	i) HPSI Delay	20.5 s (w/ offsite power) 40 s (w/o offsite power)
	j) LPSI Delay	29 s (w/ offsite power) 44 s (w/o offsite power)
	k) Containment Pressure	14.07 psia, nominal value
	l) Containment Temperature	80ºF ≤ T ≤ 130ºF
	m) Containment Sprays Delay	0 s
	n) Containment Spray Water Temperature	45ºF
	o) Containment Volume	1,960,000 – 2,020,000 (ft^3)

[4] The Realistic Large Break LOCA Analysis cases were re-analyzed to support a pressurizer pressure operating range of 2209.7 psia ≤ P ≤2299.7 psia.

TABLE 15.6.5-2

Fresh Fuel (8% Gd_2O_3 Rod)

Parameter	Value
PCT	
Temperature	2084ºF
Time	36.4 s
Elevation	9.235 ft
Metal-water Reaction	
Pre-transient Oxidation (%)	0.757
Transient Local Oxidation (%)	2.440
Percent Total Oxidation Maximum (%)	3.197
Percent Total Whole Core Oxidation (%)	0.0423

TABLE 15.6.5-3

Large Break LOCA/ECCS Analysis Event Times

Event	Time (sec)
Break Opened	0.0
RCP Trip	N/A
SIAS Issued	0.5
Start of Broken Loop Accumulator Injection	8.3
Start of Intact Loop Accumulator Injection (Loops 2 and 3 respectively)	10.7 and 10.7
Beginning of Core recovery (Beginning of Reflood)	33.6
Broken Loop HPSI Delivery Begin	40.5
Intact Loop HPSI Delivery Begin (Loop 2 and 3 respectively)	40.5 and 40.5
LHSI Available	44.5
Broken Loop LPSI Delivery Begin	44.5
Intact Loop LPSI Delivery Begin (Loop 2 and 3 respectively)	44.5 and 44.5
PCT Occurred (2084ºF)	36.4
Broken Loop Accumulator Emptied	49.6
Intact Loop Accumulators Emptied (Loop 2 and 3 respectively)	50.9 and 46.8
Termination of short term calculation	424.5
Start of Switchover to LPSI Recirculation (LPSI Pump Stopped)	1260.
End of Switchover to LPSI Recirculation (LPSI Pump Started)	2460.
Start of Switchover to HPSI Recirculation (LPSI Pump Stopped)	4380.
End of Switchover to HPSI Recirculation (HPSI Pump Started)	4740.

Table 15.6.5-4

Core Activity

Isotopes	Curies	Isotopes	Curies	Isotopes	Curies
Co-58	5.99E+05	Ru-103	9.87E+07	Cs-136	3.52E+06
Co-60	4.58E+05	Ru-105	6.83E+07	Cs-137	8.87E+06
Kr-85	7.30E+05	Ru-106	3.73E+07	Ba-139	1.15E+08
Kr-85m	1.51E+07	Rh-105	6.33E+07	Ba-140	1.13E+08
Kr-87	3.03E+07	Sb-127	5.38E+06	La-140	1.17E+08
Kr-88	4.20E+07	Sb-129	2.03E+07	La-141	1.02E+08
Rb-86	1.16E+05	Te-127	5.31E+06	La-142	9.83E+07
Sr-89	5.90E+07	Te-127m	8.87E+05	Ce-141	1.04E+08
Sr-90	6.16E+06	Te-129	1.90E+07	Ce-143	9.55E+07
Sr-91	7.39E+07	Te-129m	3.84E+06	Ce-144	8.18E+07
Sr-92	7.88E+07	Te-131m	1.23E+07	Pr-143	9.34E+07
Y-90	6.62E+06	Te-132	8.91E+07	Nd-147	4.17E+07
Y-91	7.69E+07	I-131	6.20E+07	Np-239	1.25E+09
Y-92	7.93E+07	I-132	9.02E+07	Pu-238	2.81E+06
Y-93	6.07E+07	I-133	1.28E+08	Pu-239	2.44E+04
Zr-95	1.05E+08	I-134	1.41E+08	Pu-240	3.55E+04
Zr-97	1.00E+08	I-135	1.21E+08	Pu-241	9.89E+06
Nb-95	1.06E+08	Xe-133	1.28E+08	Am-241	1.18E+04
Mo-99	1.16E+08	Xe-135	3.68E+07	Cm-242	3.23E+06
Tc-99m	1.03E+08	Cs-134	1.25E+07	Cm-244	3.88E+05

Table 15.6.5-5

Large Break LOCA/ECCS Condition for
Analysis of Switchover to Recirculation Phase

Calculational Basis	
Power used for analysis	2346
Heat Flux Factor at Rated Thermal Power, F_Q^{RTP}	2.50
Nuclear Enthalpy Rise Factor, $F_{\Delta H}$	1.80
Steam Generator Tube Plugging, %	6.00
Maximum peak rod average exposure, GWD/kgU	62.0

REFERENCES: SECTION 15.6

15.6.2-1 "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50, Federal Register, Volume 39, Number 3, January 4, 1974.

15.6.2-2 "Exxon Nuclear Company Evaluation Model - EXEM PWR Small Break Model", XN-NF-82-49 (P) (A), Revision 1, Supplement 1, December 1994.

15.6.2.2a "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." Including errata (January 2008), EMF-2328(P)(A), Revision 0, AREVA NP Inc.

15.6.2-3 "H. B. Robinson Unit 2 Small Break LOCA Analysis," EMF-94-203(P), Revision 3, Siemens Power Corporation, May 1998.

15.6.2-4 Deleted by Revision No. 13.

15.6.2-5 Deleted by Revision No. 19.

15.6.2-6 "H. B Robinson Unit 2 Small Break LOCA Analysis," ANP-2972(NP), Revision 0, AREVA NP Inc., October 2011.

15.6.3-1 NRC Letter dated September 24, 2004, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105)."

15.6.5-1 "Robinson Nuclear Plant Realistic Large Break LOCA Analysis," ANP-2973(NP), Revision 0, August 2011, AREVA NP Inc.

15.6.5-2 Deleted by Revision No. 22

15.6.5-3 "Realistic Large Break LOCA Methodology," EMF-2103(P)(A), Revision 0 AREVA NP Inc., April 2003.

15.6.5-4 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR50.46.

15.6.5-5 NRC Letter dated July 11, 2006, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Implementation of the Alternate Source Term for the Loss-of-Coolant Accident (TAC No. MC5709)."

15.6.5-6 Deleted by Revision No. 21.

15.6.5-7 "Radiological Consequence Analysis of the Loss of Coolant Accident with AST Implementation," RNP-M/MECH-1740

15.6.5-8 Deleted by Revision No. 21.

REFERENCES: SECTION 15.6 (continued)

15.6.5-9 "S-RELAP5 Models and Correlations Manual," EMF2100(P), Revision 11, December 2006, AREVA NP Inc.

15.6.5-10 "RODEX3 Fuel Rod Thermal-Mechanical Response Evaluation Model," ANF-90-145(P)(A), Revision 0, April 1996, AREVA NP Inc.

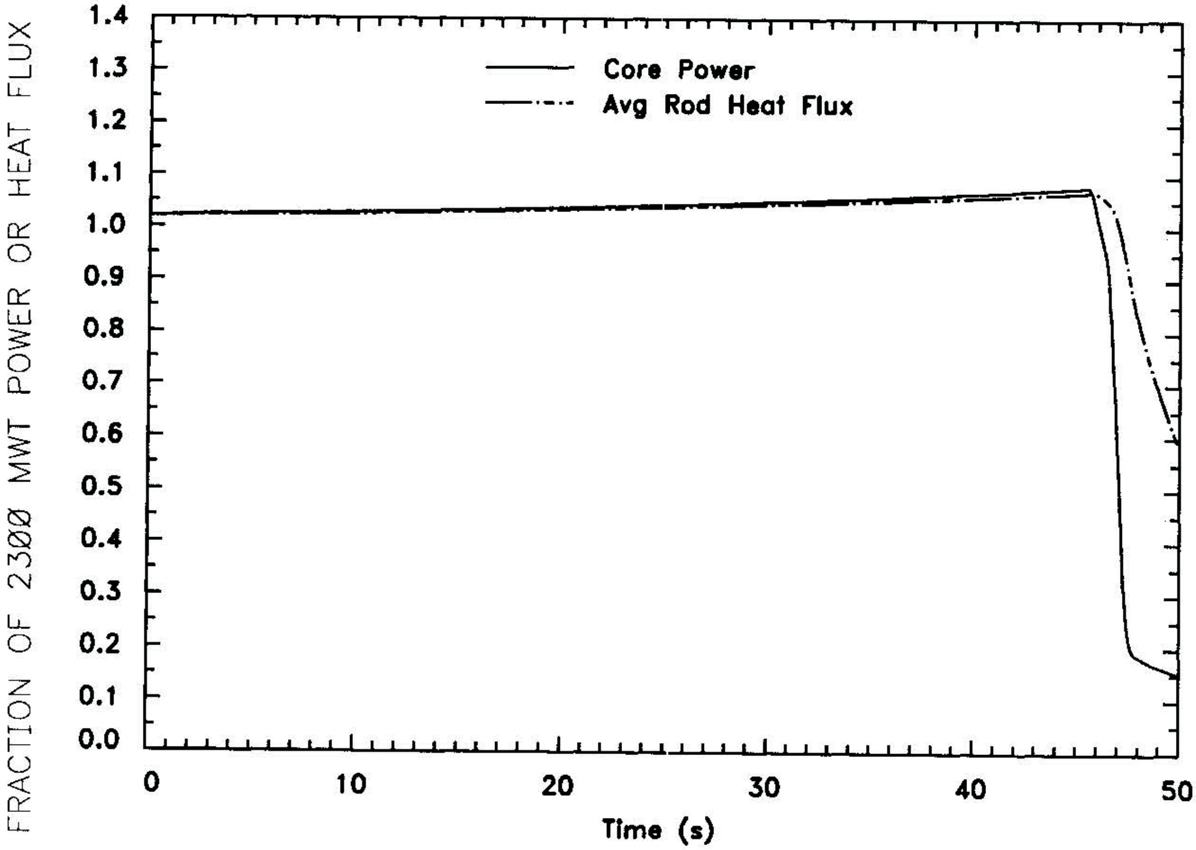
15.6.5-11 "Quantifying Reactor Safety Margins," NUREG/CR-5249, EGG-2552, October 1989 Technical Program Group.

15.6.5-12 Deleted by Revision No. 21.

15.6.5-13 H. B. Robinson Unit 2 Extended Transfer to Cold Leg Recirculation Following a LBLOCA, EMF-2286, November 2000, Siemens Power Corporation.

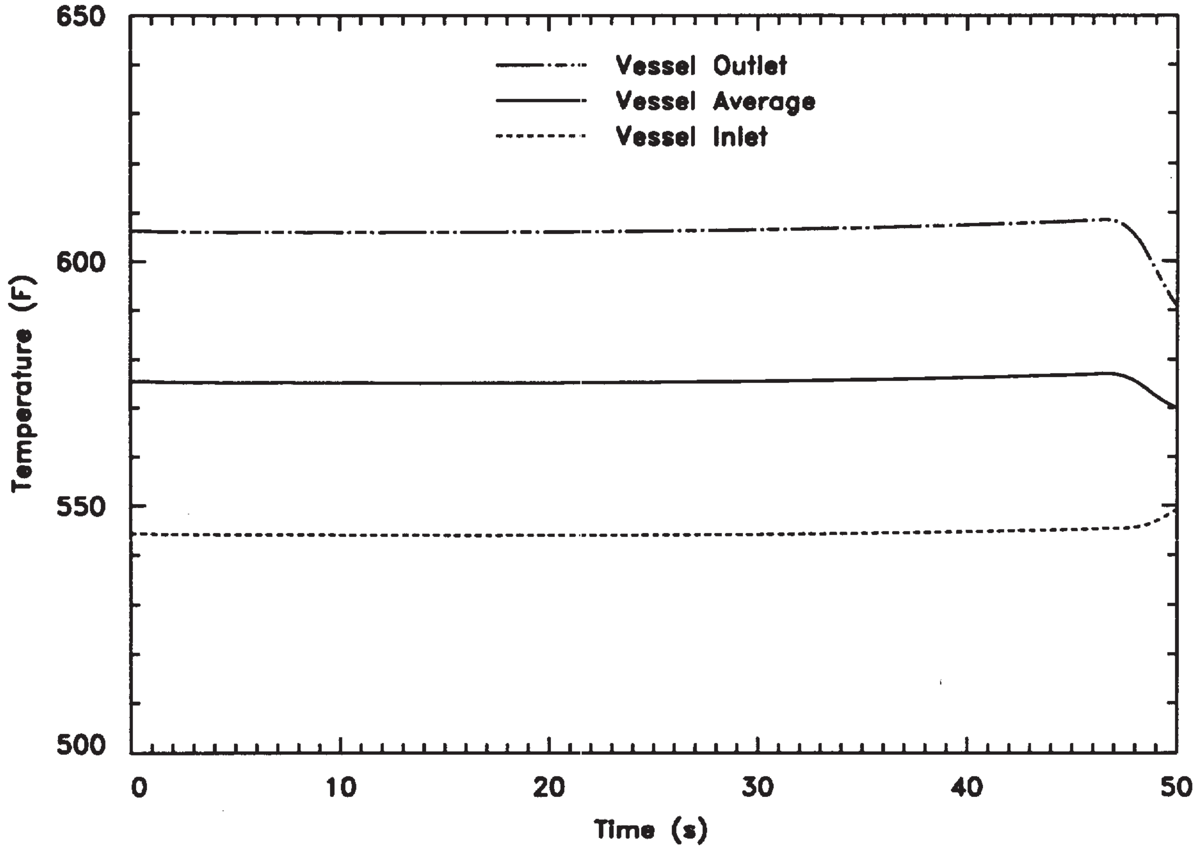
15.6.5-14 NSAL-94-016, "Core Criticality during LBLOCA Hot-Leg Switchover," Westinghouse, 1994.

15.6.5-15 ESR 00-00026, "Robinson Cycle 21 Core Reload Design," Rev. 2, 2001.



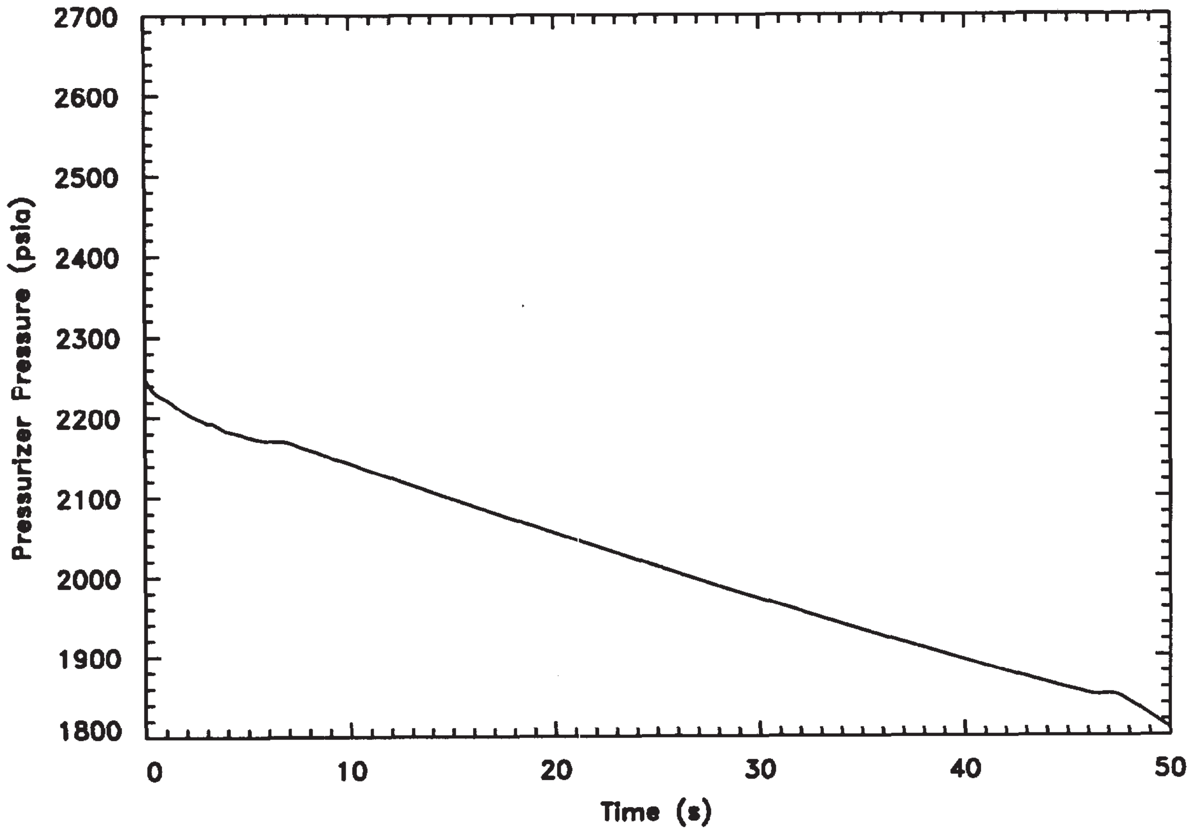
REVISION NO. 18

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Open Pressurizer Safety / PORV: Core Power and Average Rod Heat Flux	FIGURE 15.6.1–1



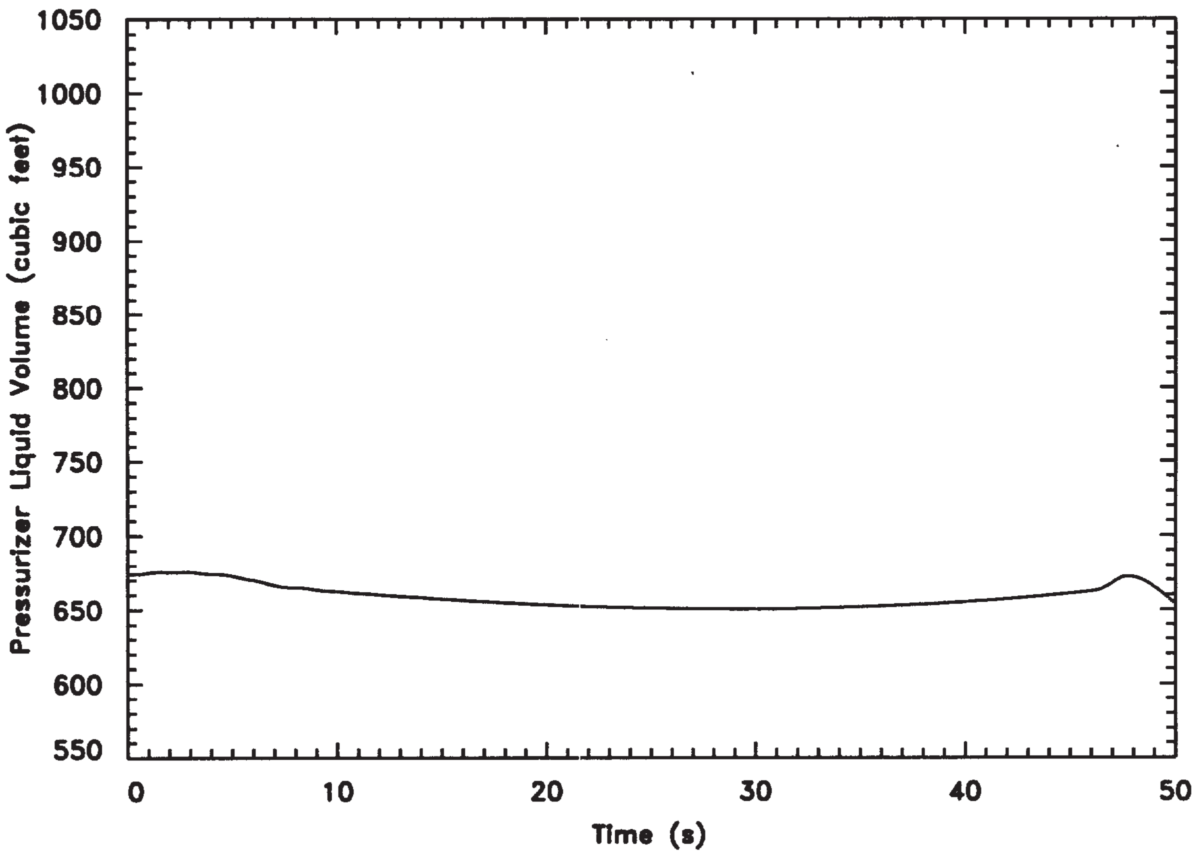
REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Open Pressurizer Safety / PORV: Reactor Coolant Temperatures	FIGURE 15.6.1-2



REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Open Pressurizer Safety / PORV: Pressurizer Pressure	FIGURE 15.6.1-3

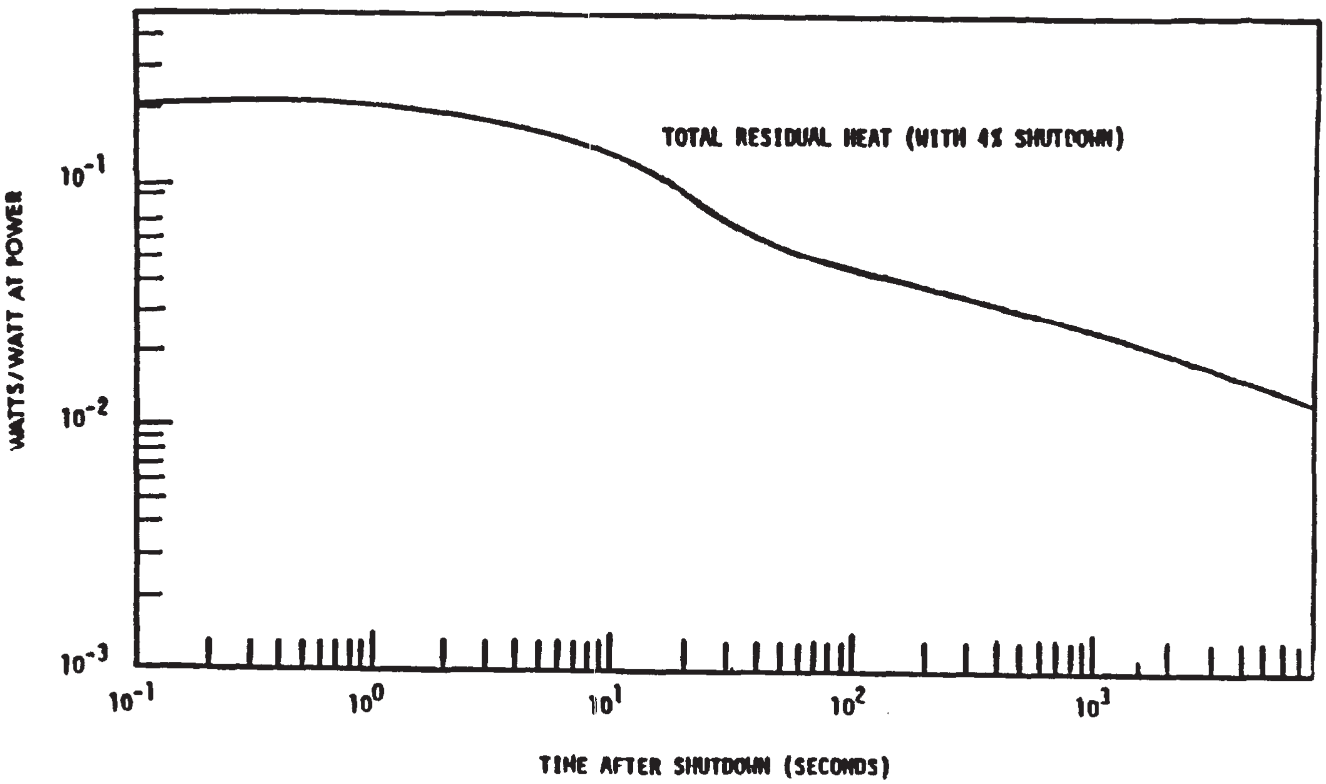


REVISION NO. 13

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Open Pressurizer Safety / PORV: Pressurizer Liquid Volume	FIGURE 15.6.1-4

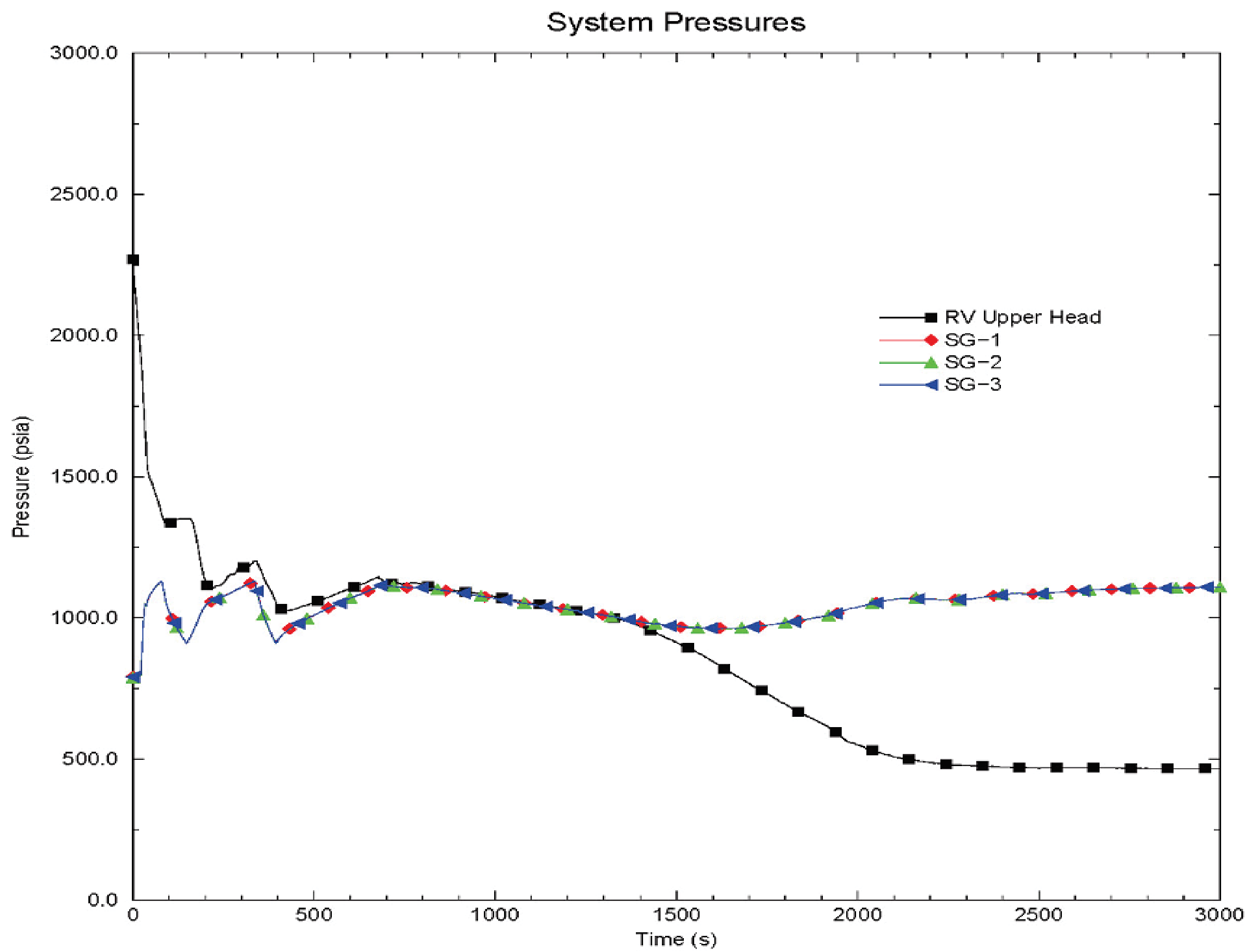
Figure 15.6.2-1 was deleted by Amendment No. 12

Figure 15.6.2-2 was deleted by Revision No. 13



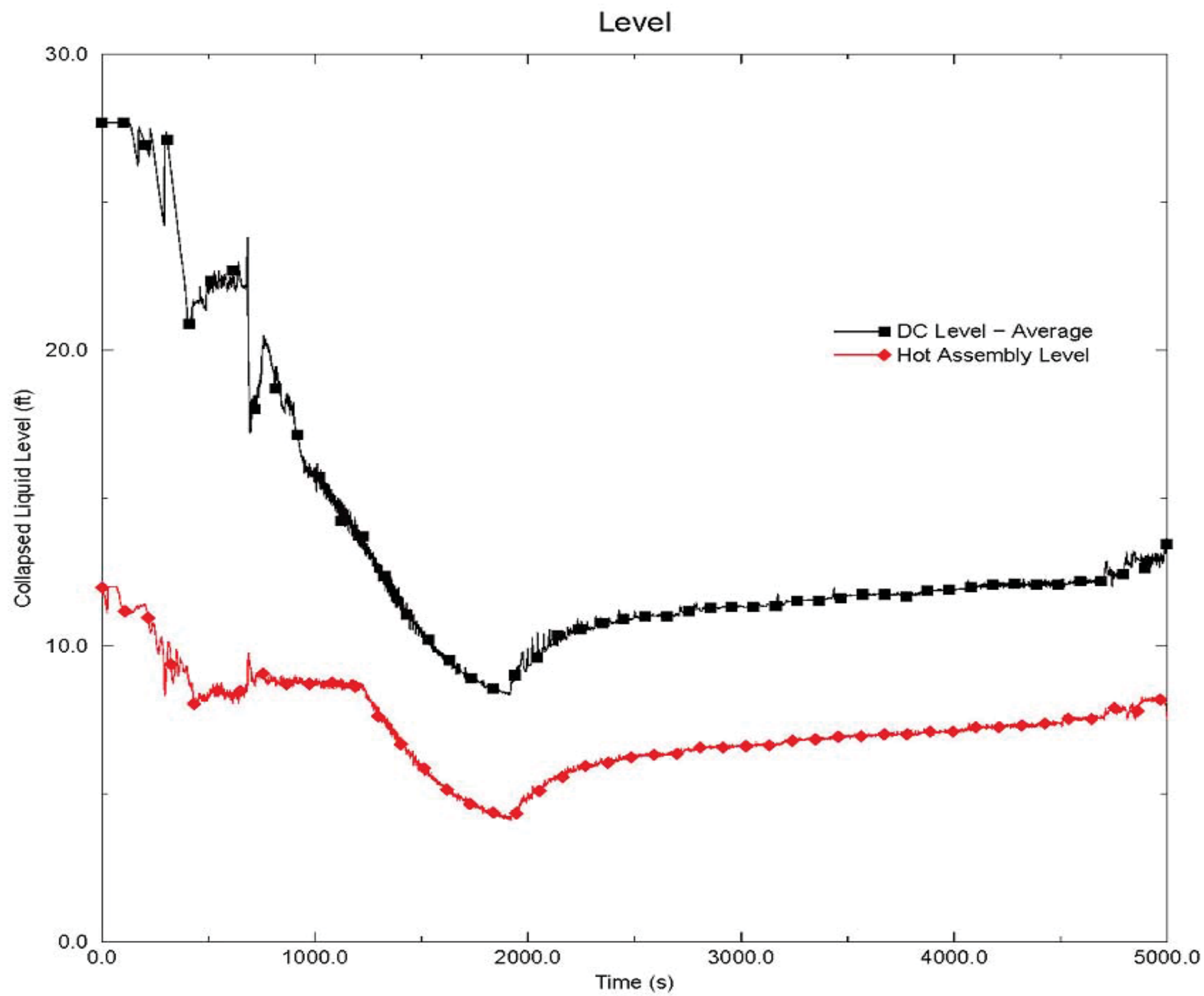
Amendment No. 7

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CORE POWER AFTER REACTOR TRIPS	FIGURE 15.6.2-3



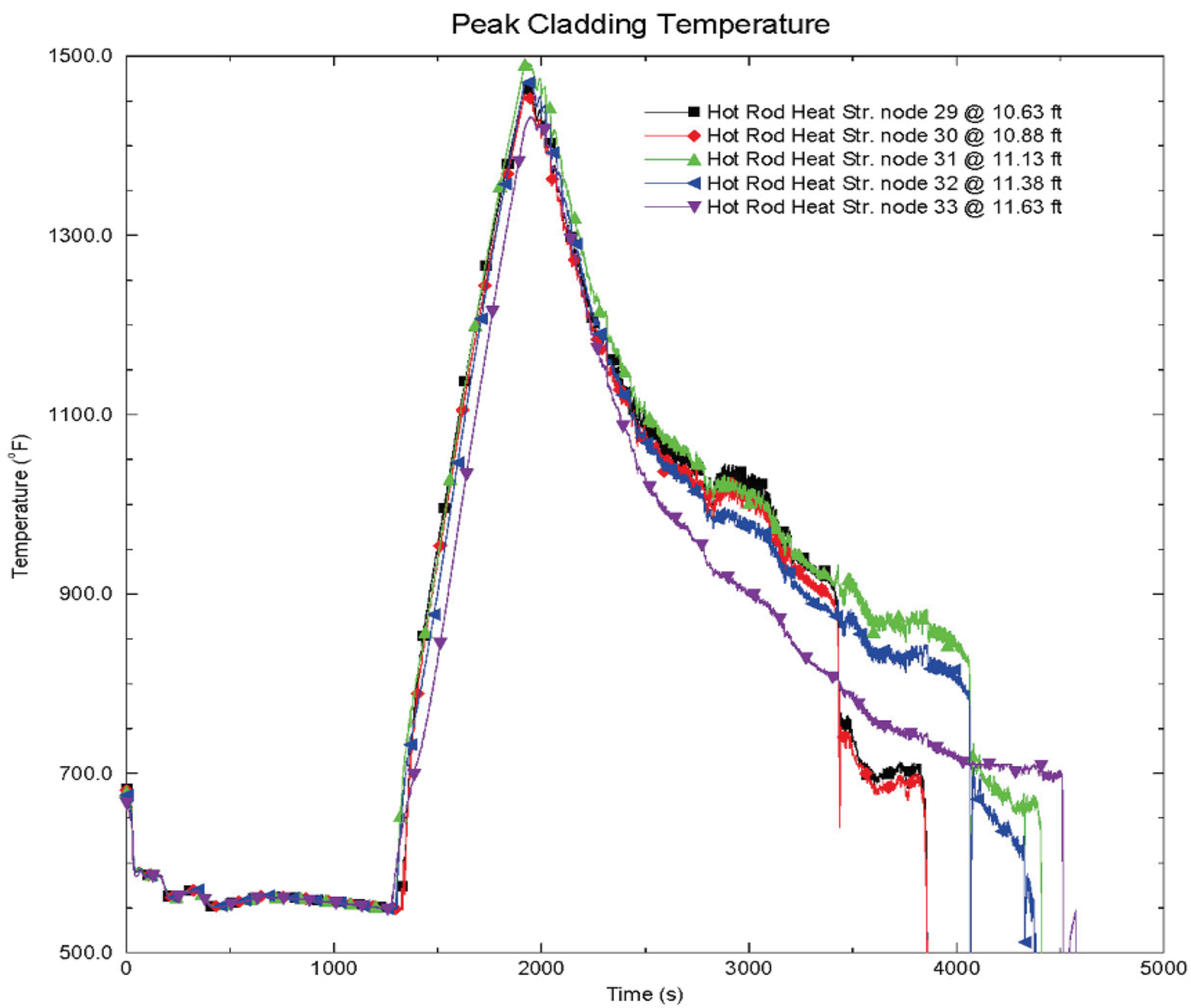
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	System Pressure for Small Break LOCA During Blowdown	FIGURE 15.6.2-4



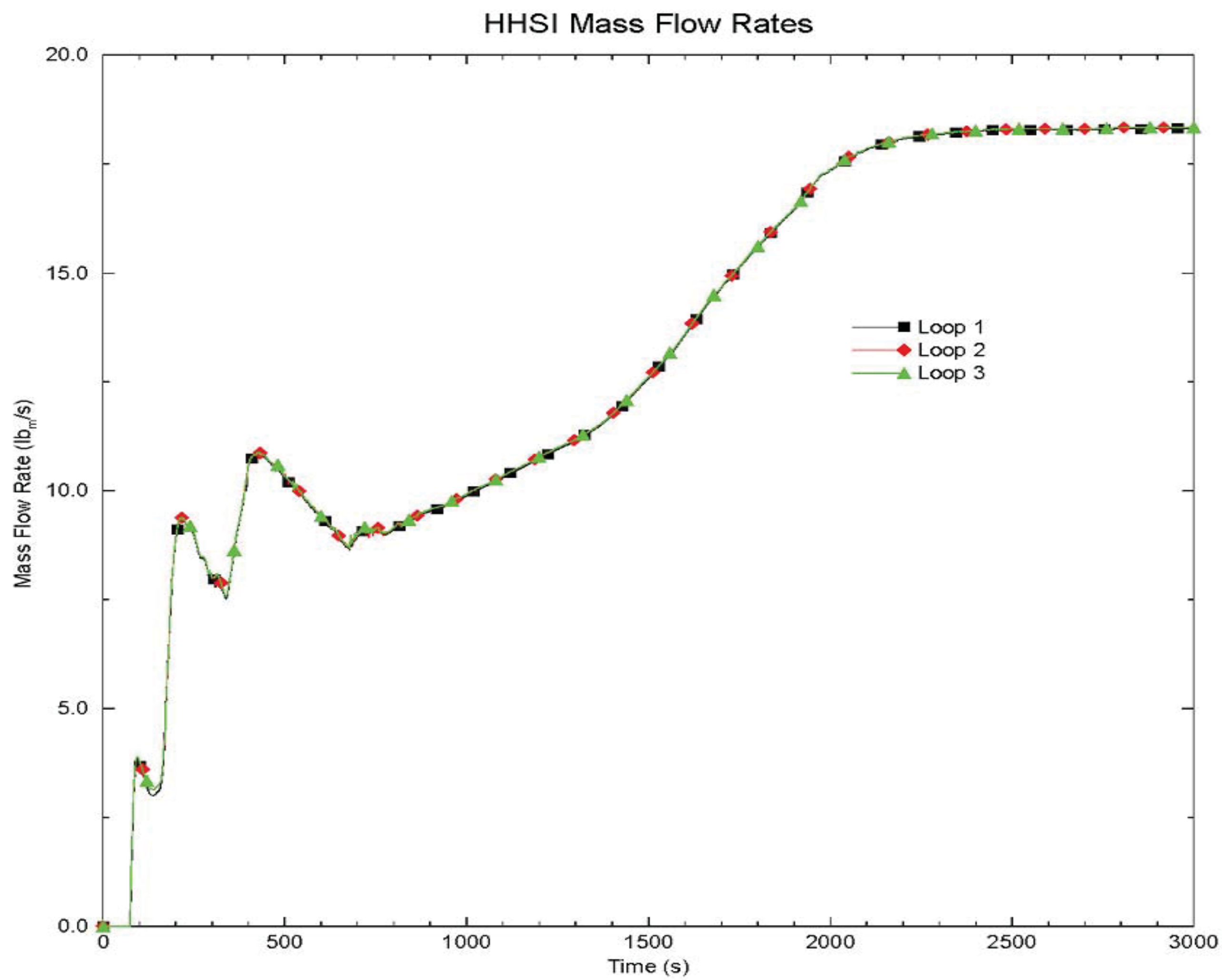
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Downcomer and Hot Assembly Collapsed Liquid Levels for Small Break LOCA During Blowdown	FIGURE 15.6.2-5



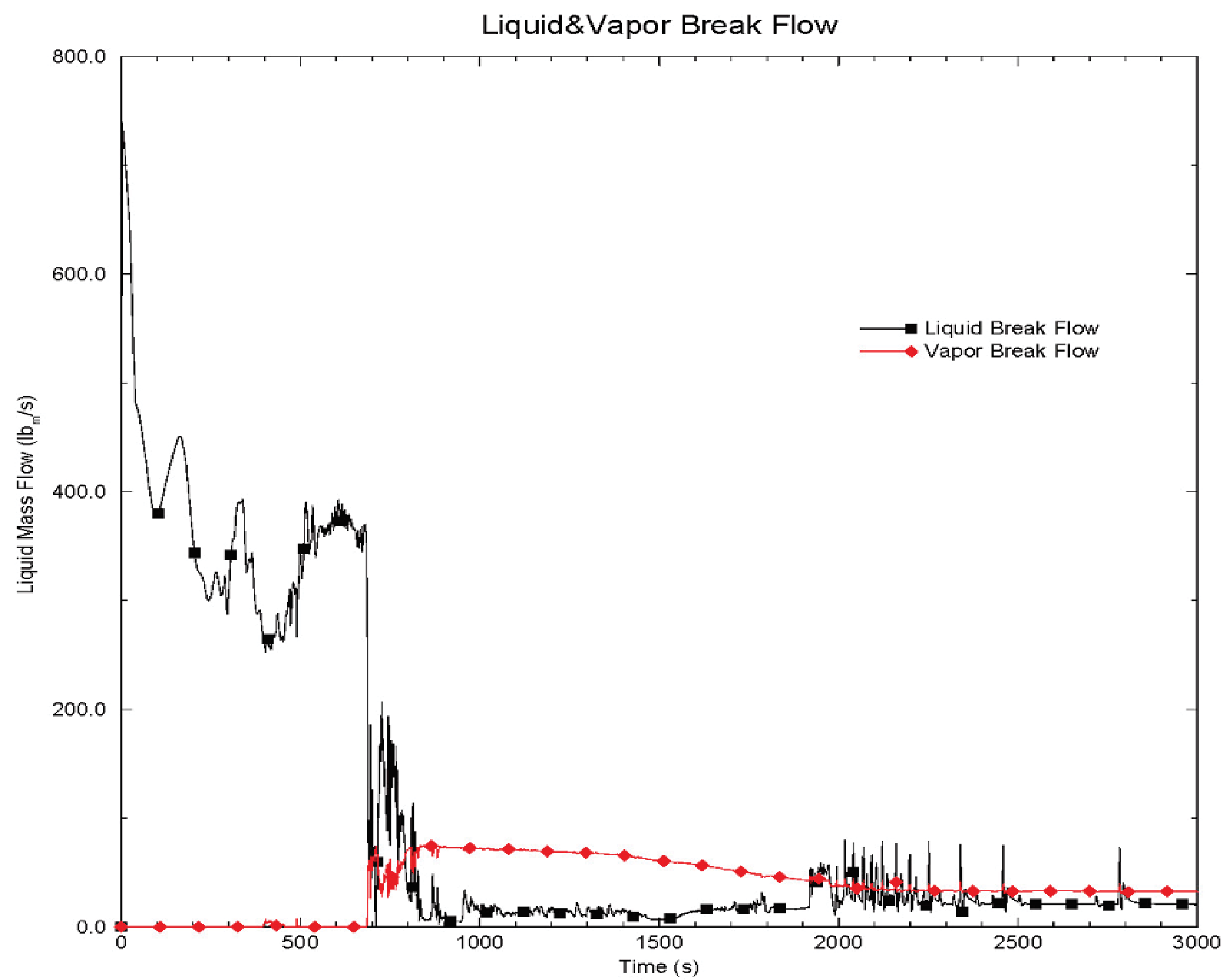
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Hot Rod Cladding Temperatures for Small Break LOCA during Blowdown	FIGURE 15.6.2-6



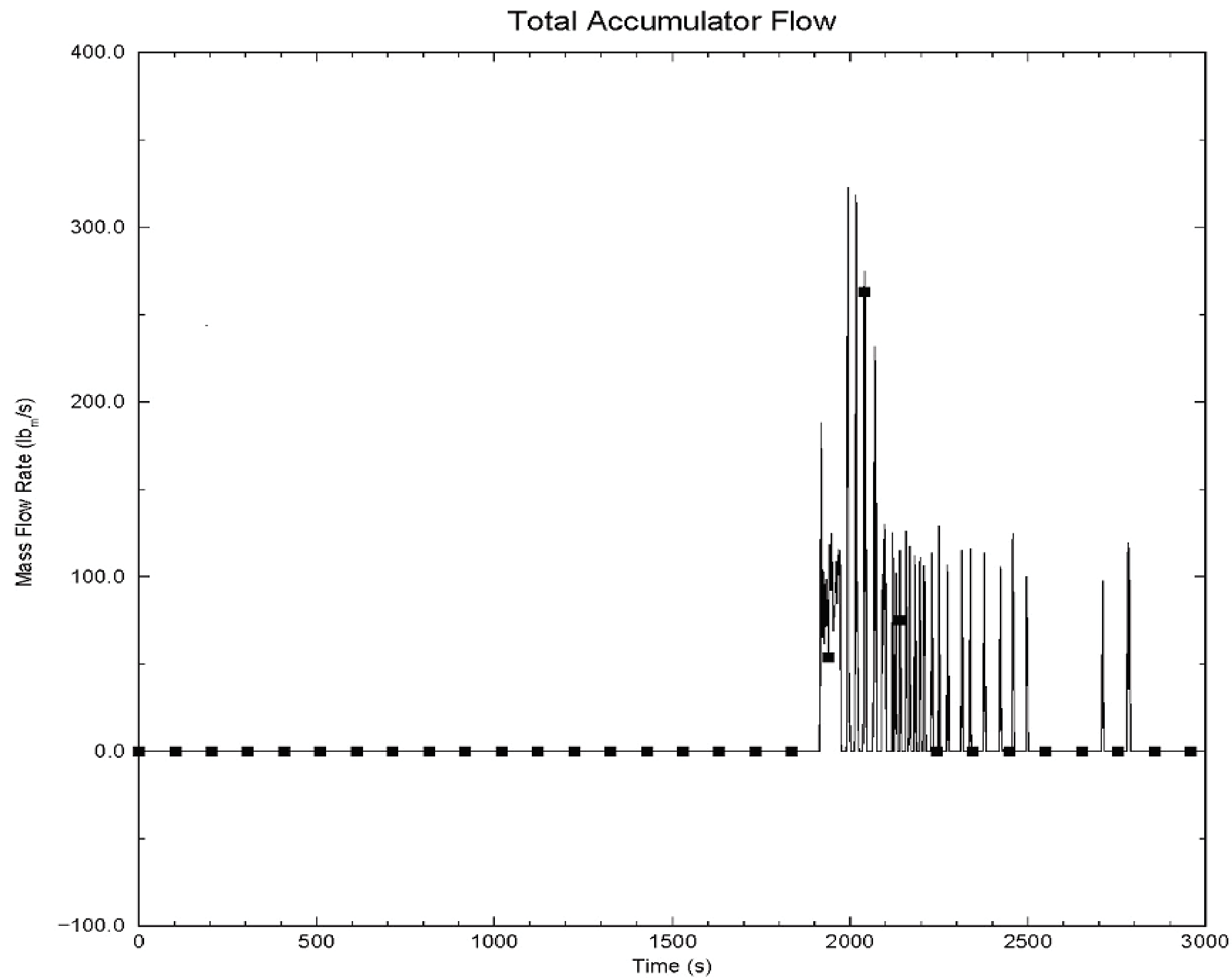
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Total SI Flow for Small Break LOCA during Blowdown	FIGURE 15.6.2-7



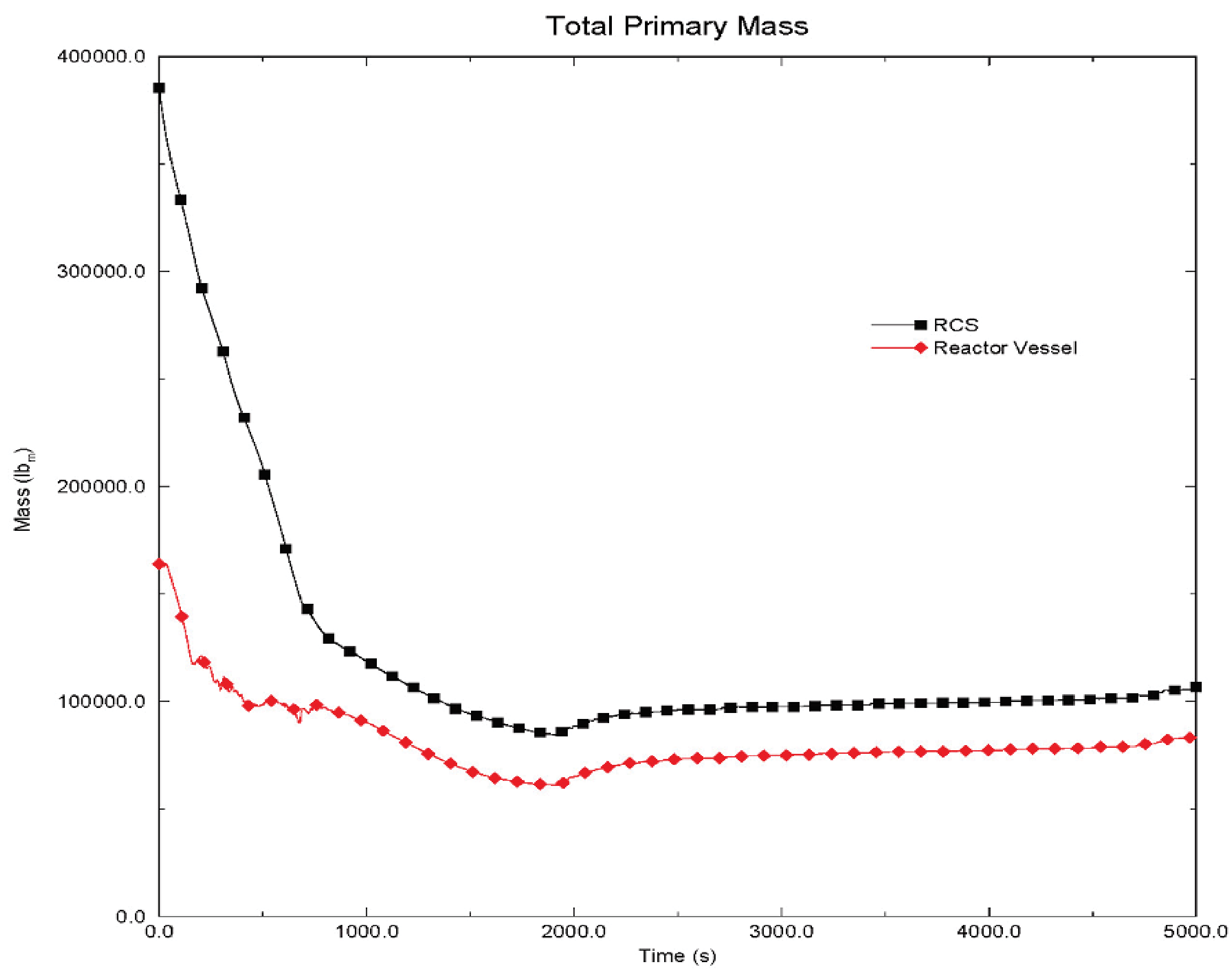
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Break Flow Rate for Small Break LOCA during Blowdown	FIGURE 15.6.2-8



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Combined Accumulator Flow for Small Break LOCA during Blowdown	FIGURE 15.6.2-9

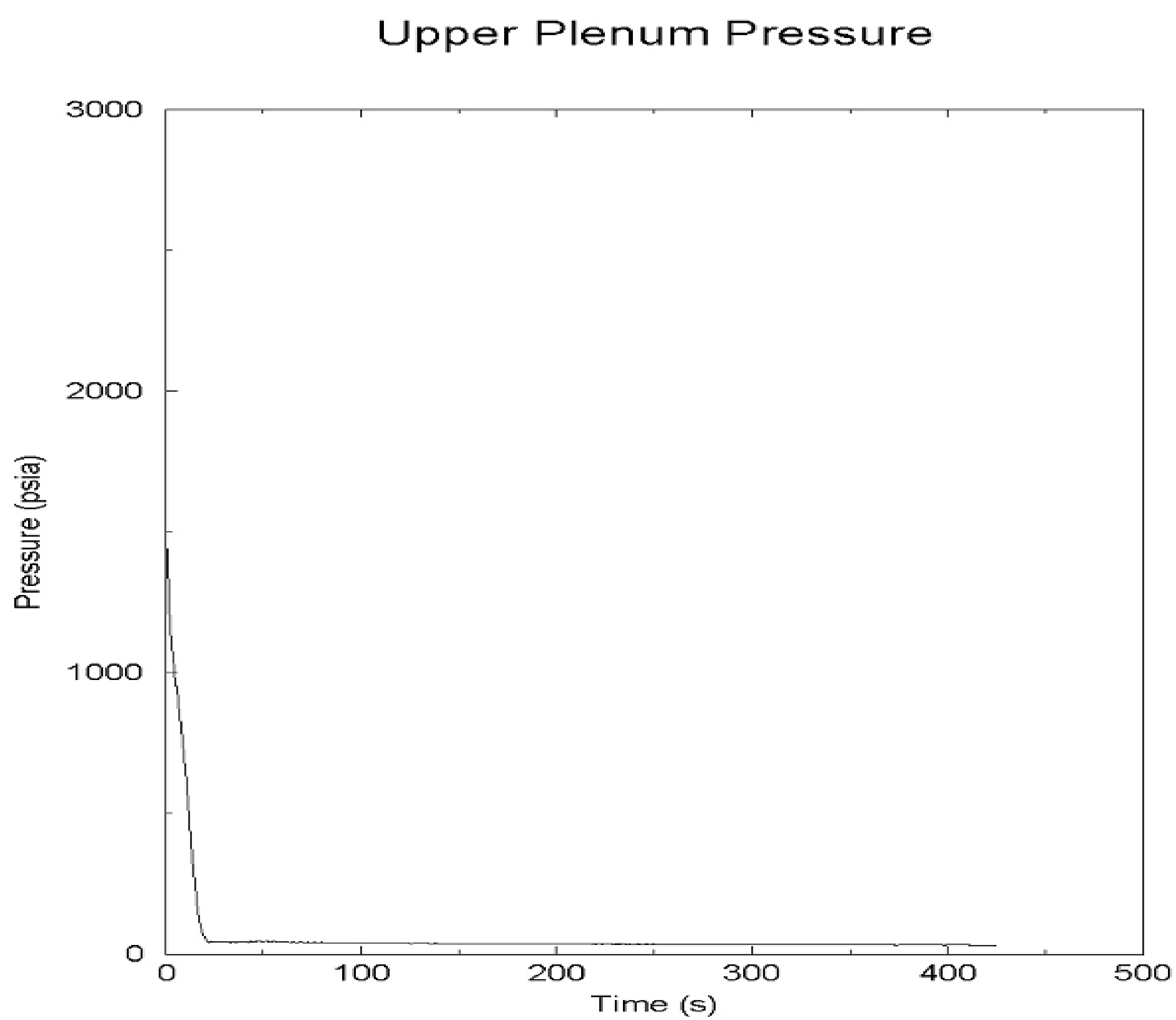


REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Primary System Fluid Mass and Reactor Vessel Mass for Small Break LOCA during Blowdown	FIGURE 15.6.2-10

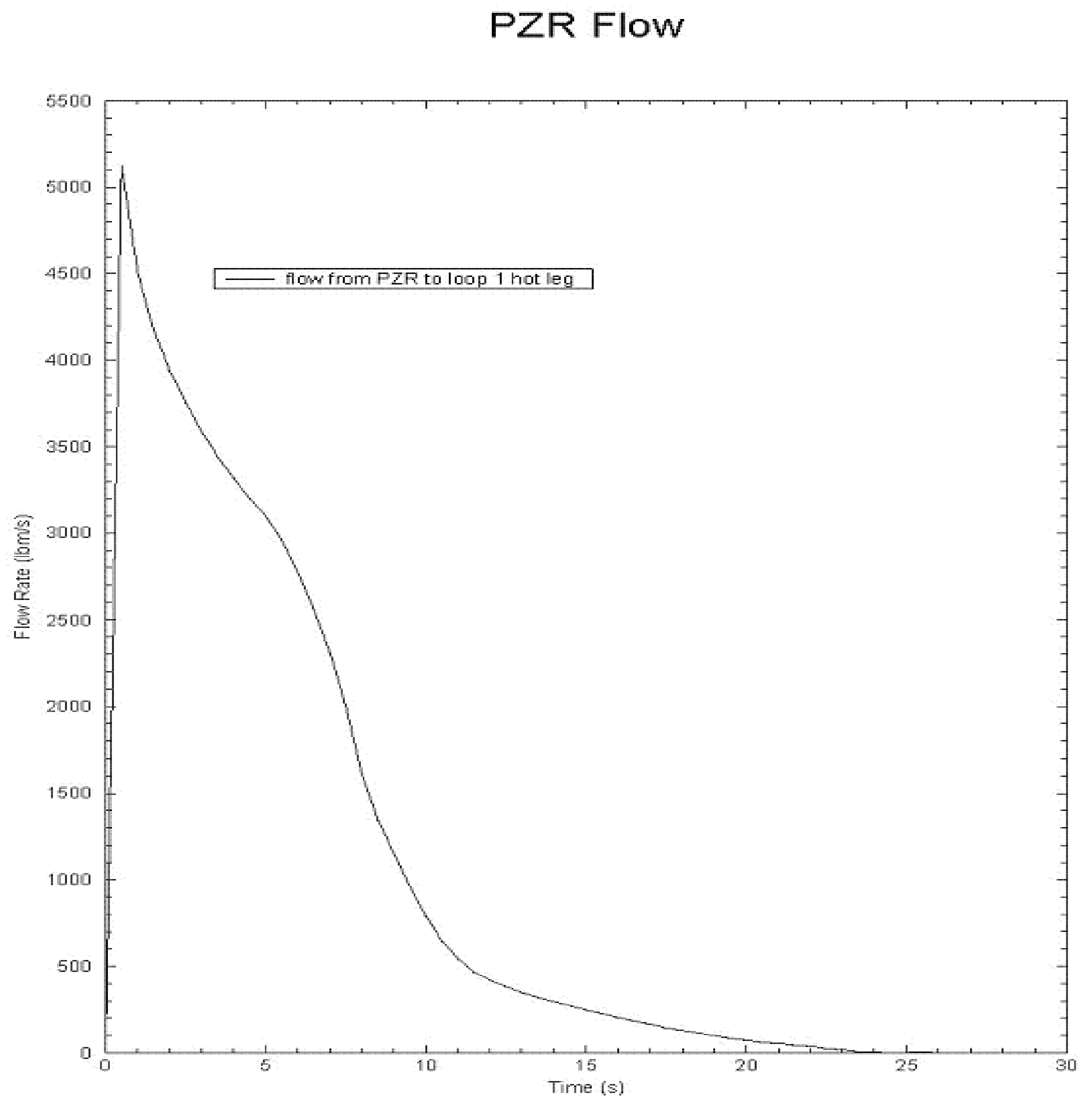
FIGURE 15.6.5-1 was deleted by Revision No. 22

FIGURE 15.6.5-2 was deleted by Revision No. 22



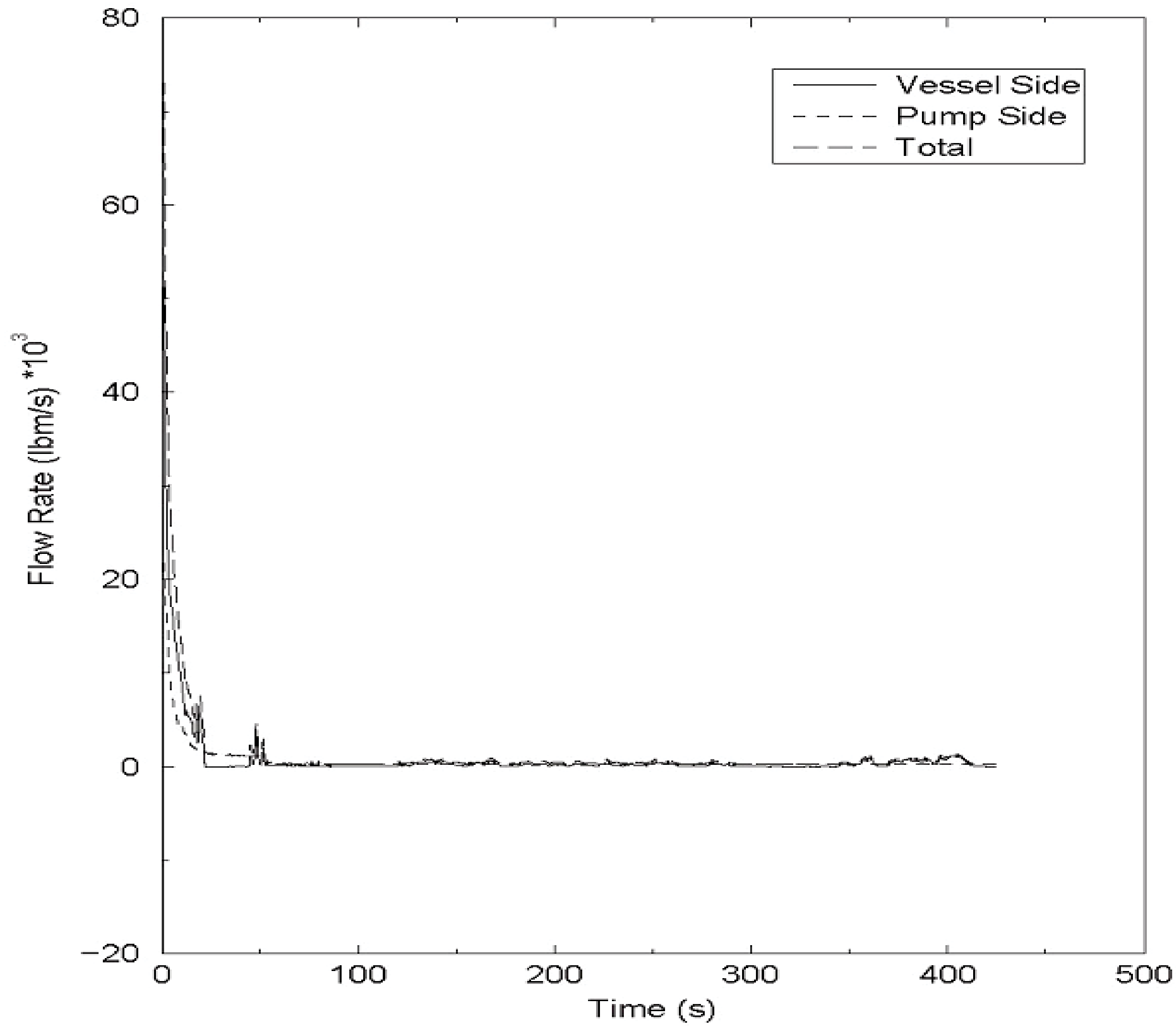
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Upper Plenum Pressure for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-3



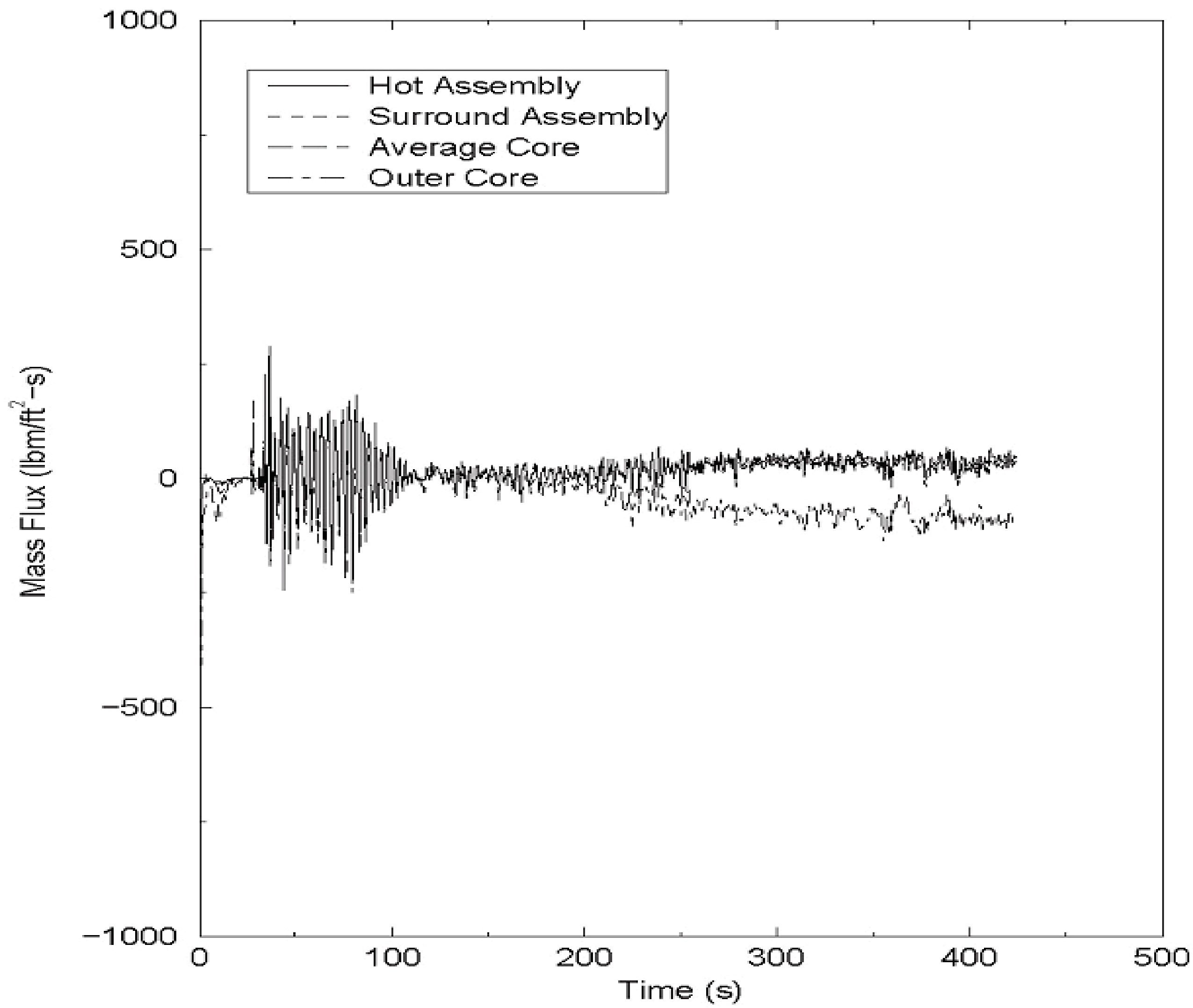
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Surge Line Flow Rate for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-4



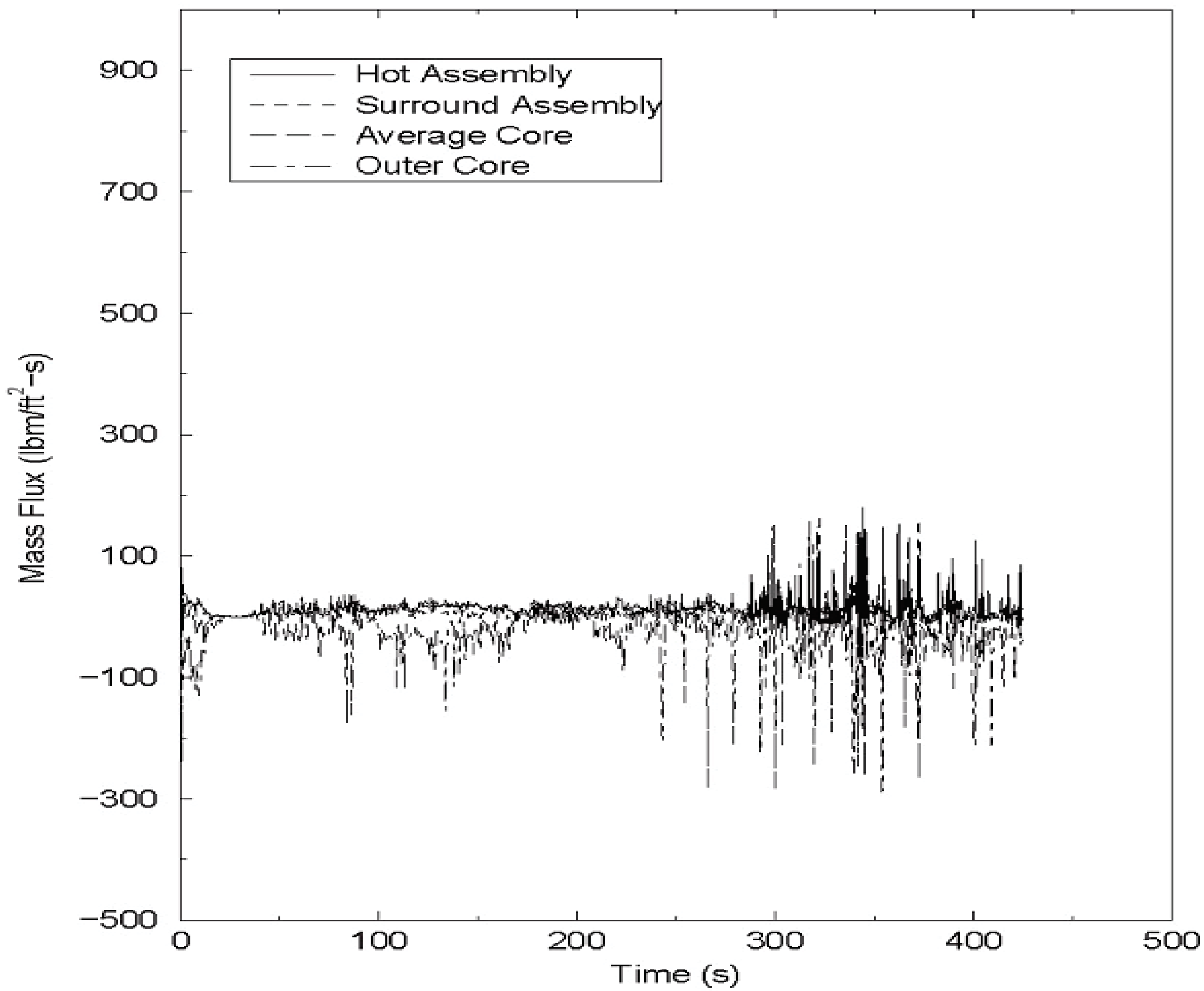
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Total Break Flow Rate for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-5



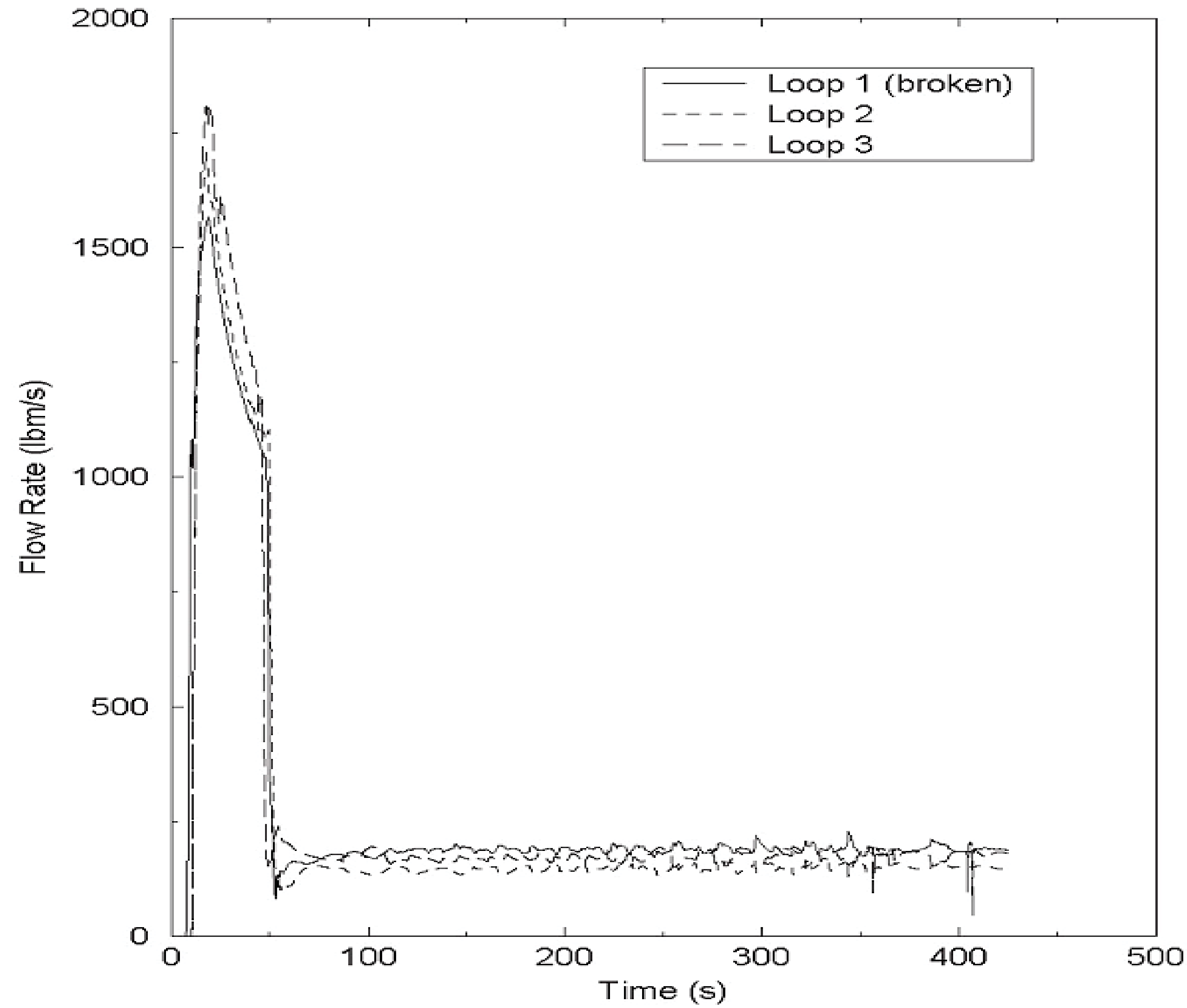
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Core Inlet Mass Flux for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-6



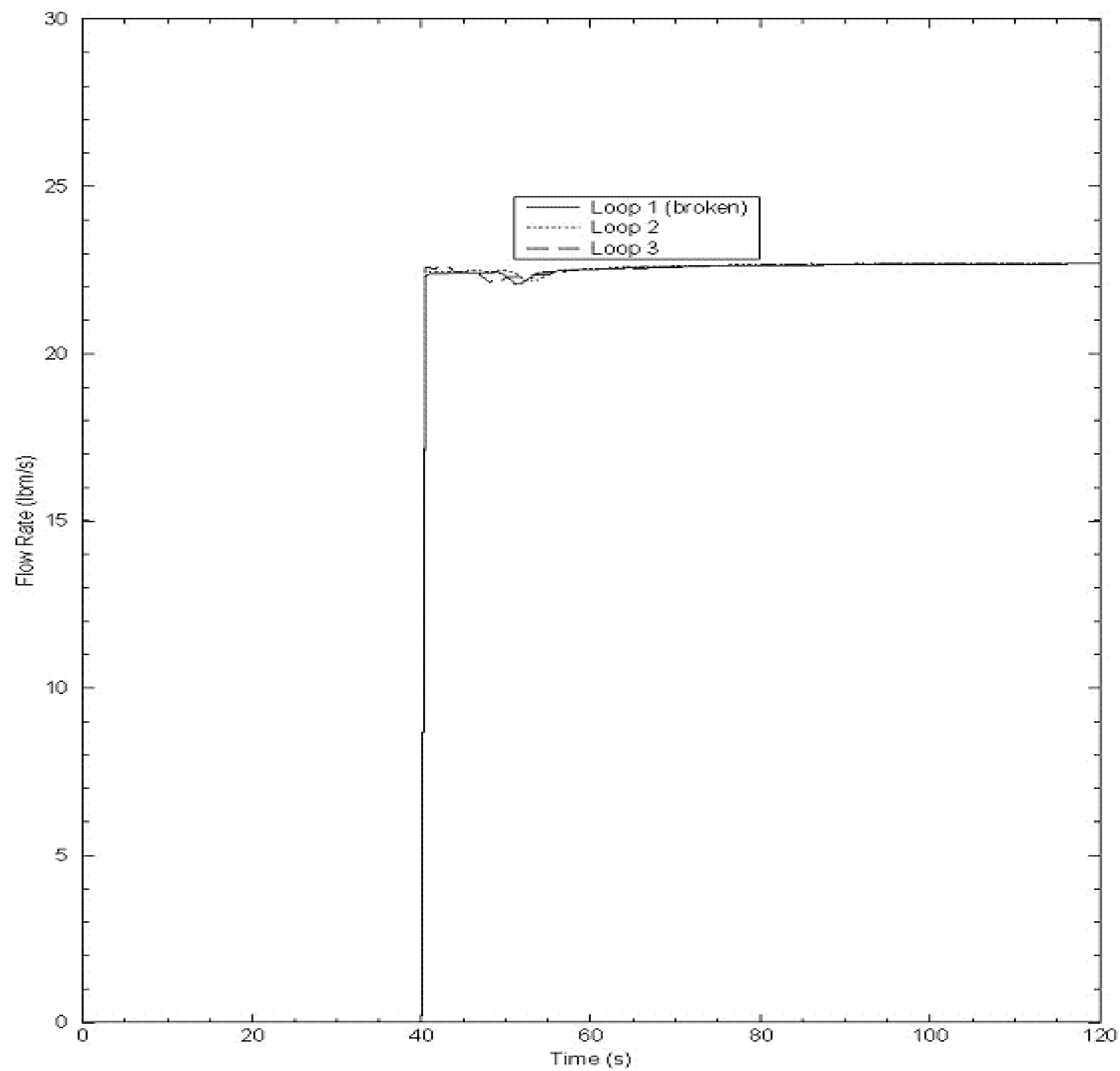
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Core Outlet Mass Flux for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-7



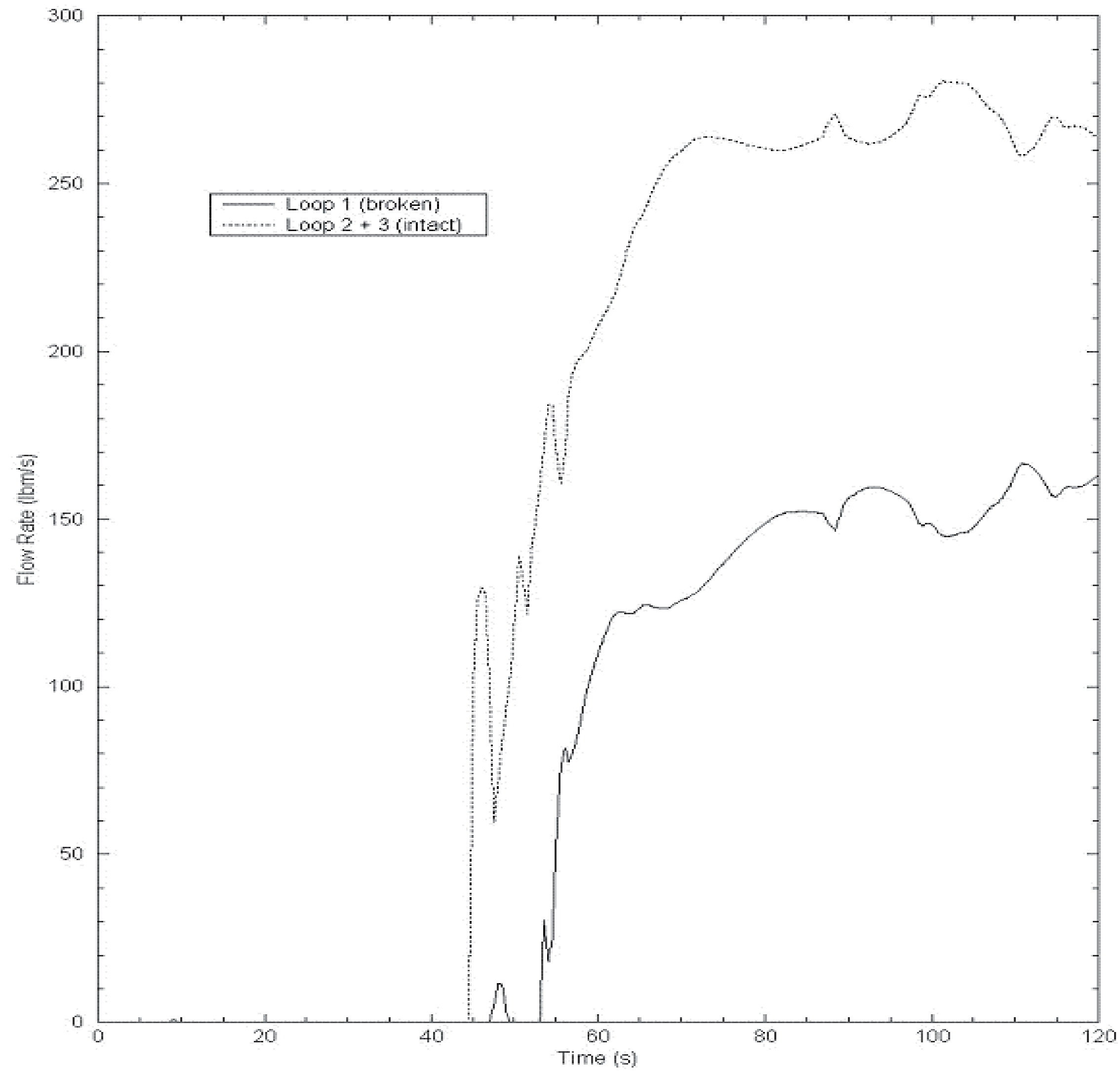
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	ECCS Flows (Accumulator, SI, and RHR) for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-8



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Intact Loop and Broken Loop HHSI Flow for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-9



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Intact Loop and Broken Loop LHIS Flow for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-10

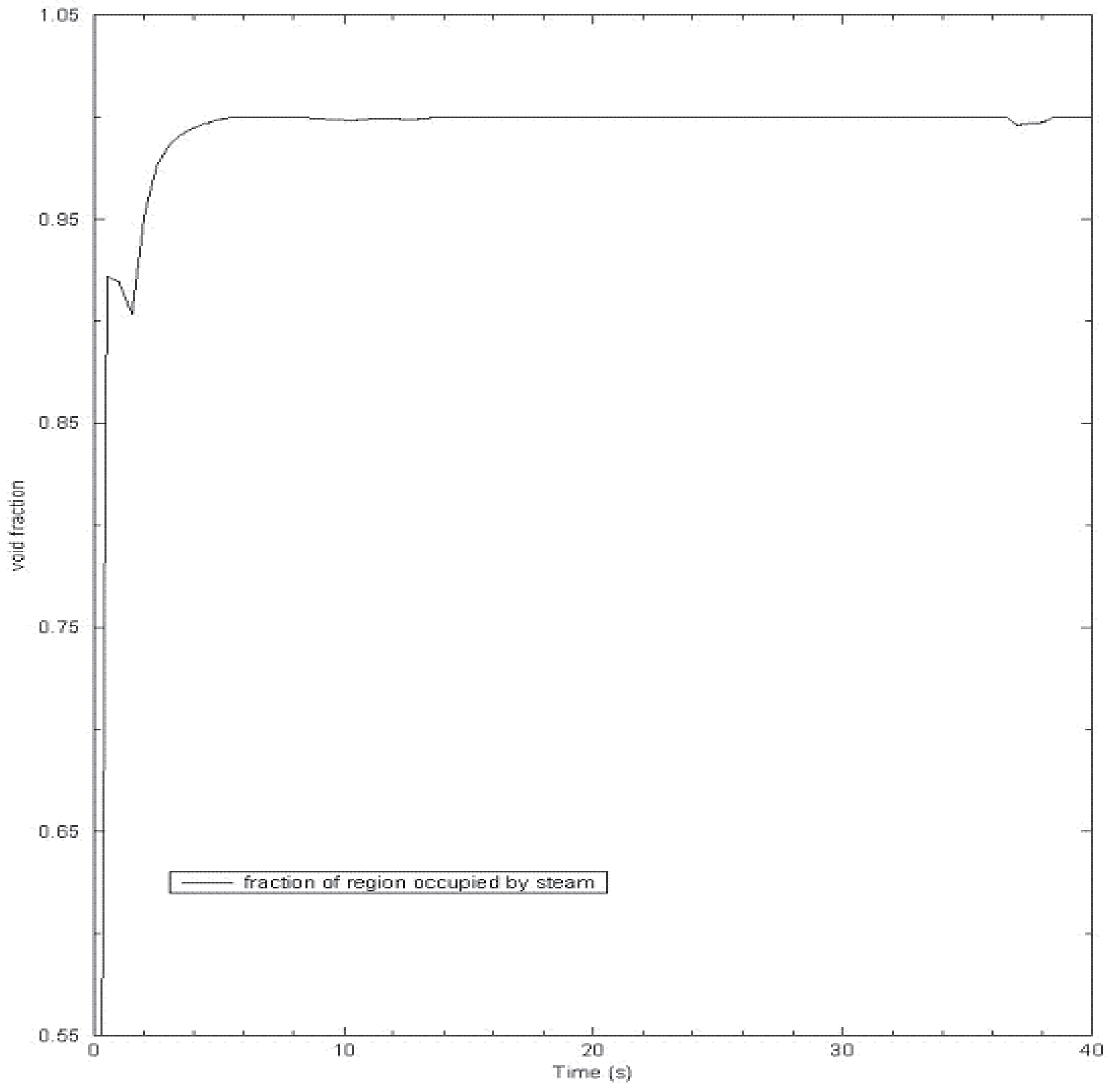
Figure 15.6.5-11 Deleted By Revision No. 17

Figure 15.6.5-12 Deleted By Revision No. 17

Figure 15.6.5-13 Deleted By Revision No. 17

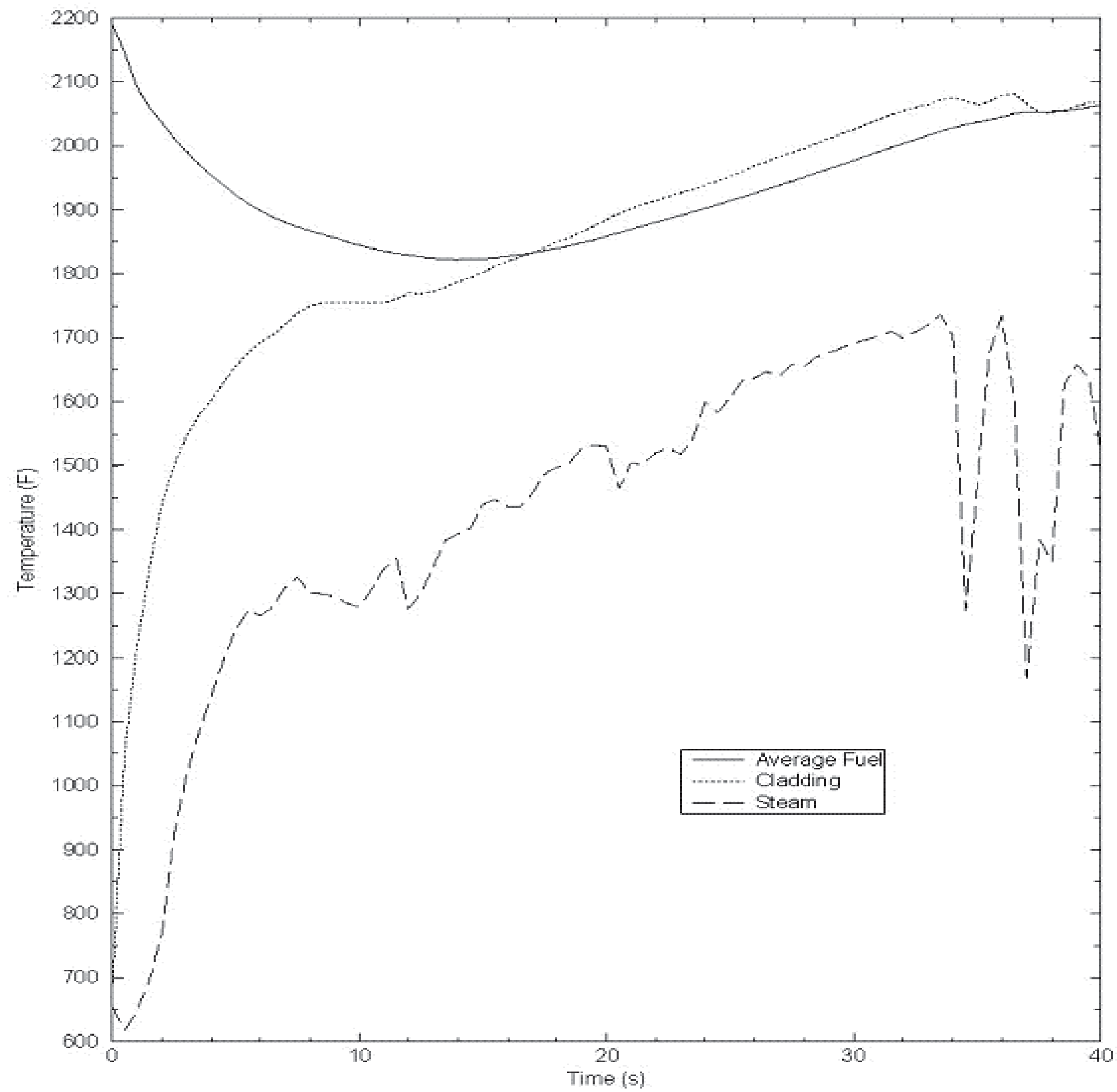
FIGURE 15.6.5-14 was deleted by Revision No. 22

FIGURE 15.6.5-15 was deleted by Revision No. 22



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PCT Node Fluid Void Friction for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-16

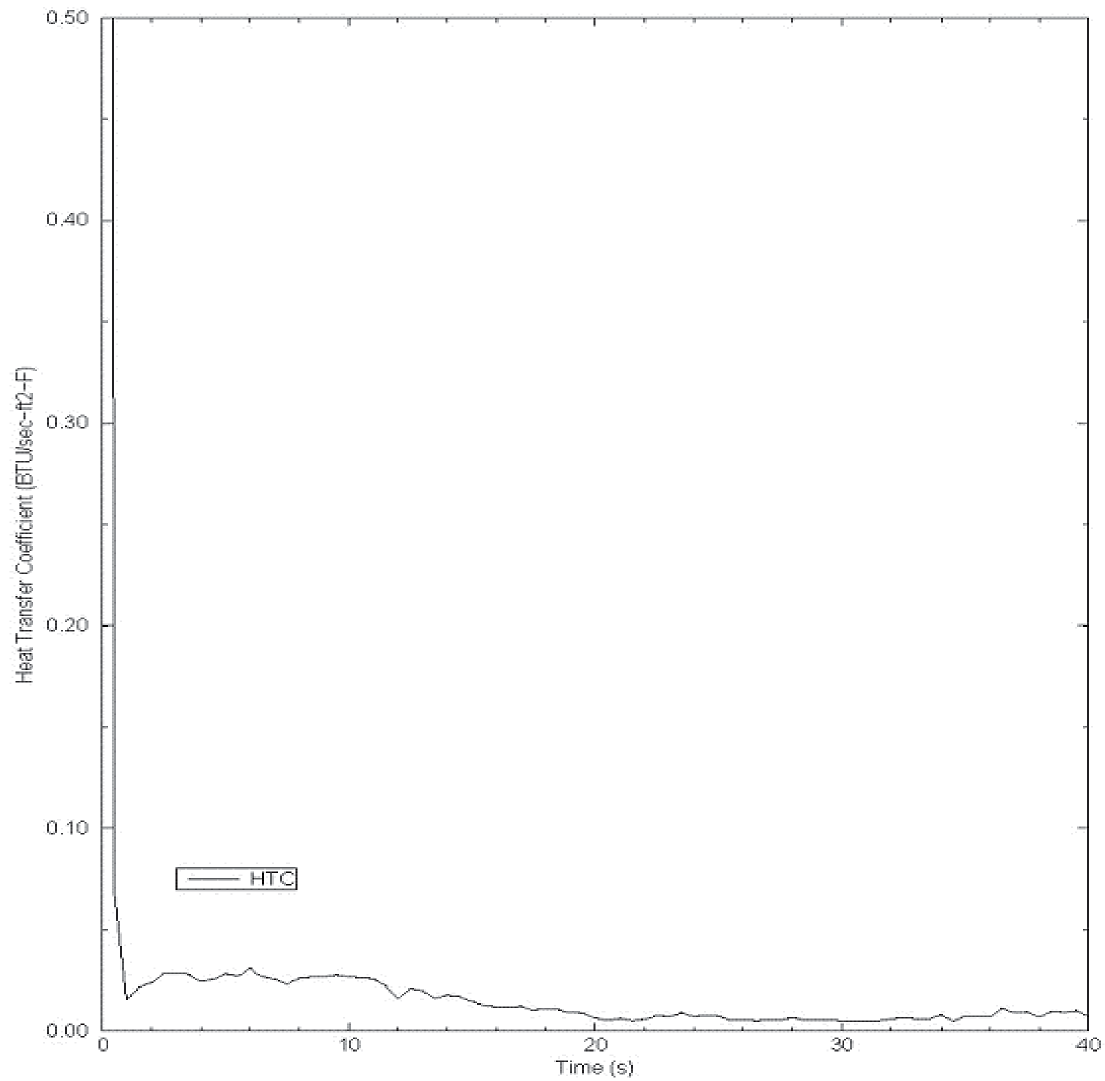


REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PCT Node Fuel (Average), Cladding and Fluid Temperatures for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-17

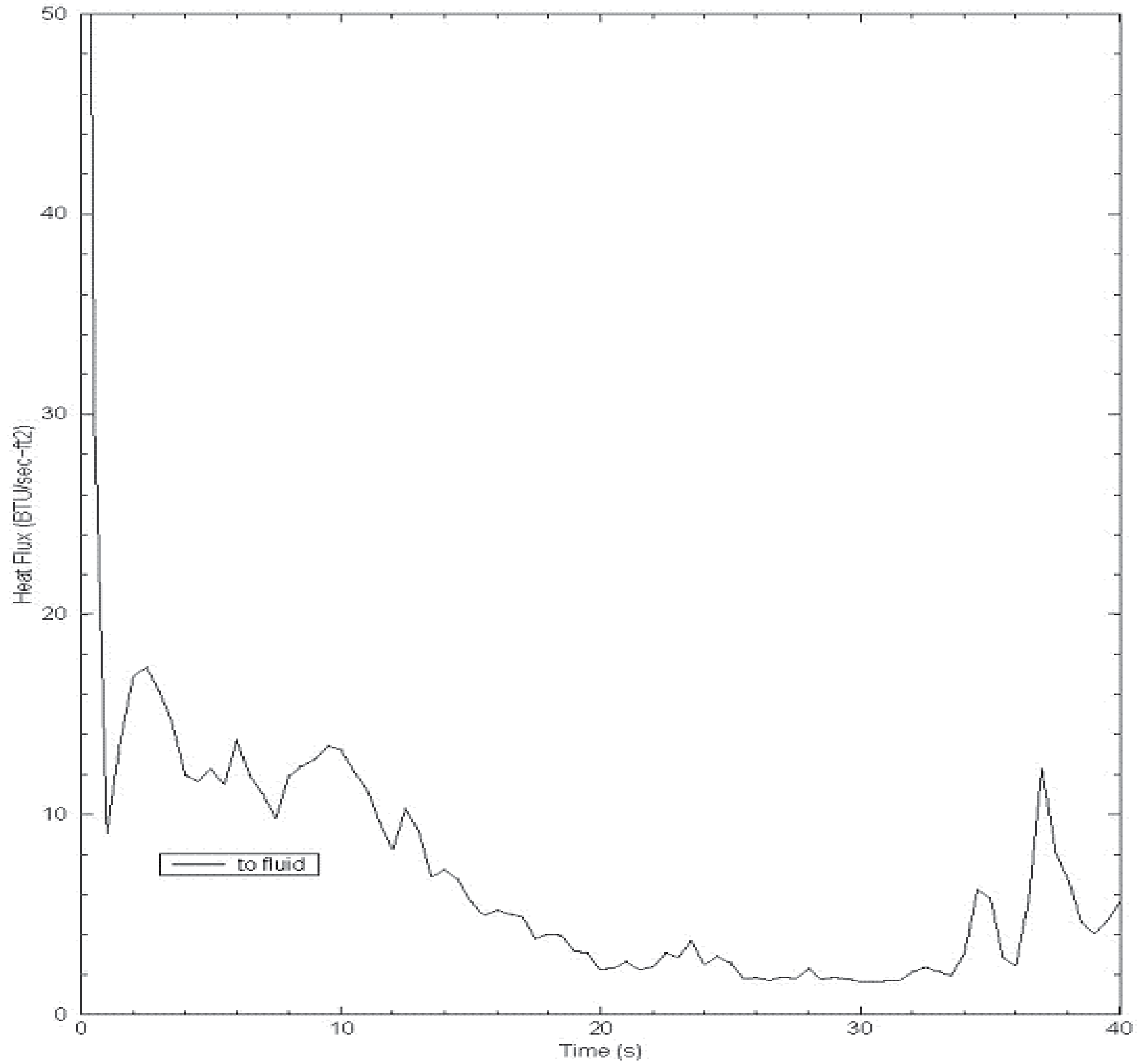
Figure 15.6.5-18 Deleted By Revision No. 17

Figure 15.6.5-19 Deleted By Revision No. 17



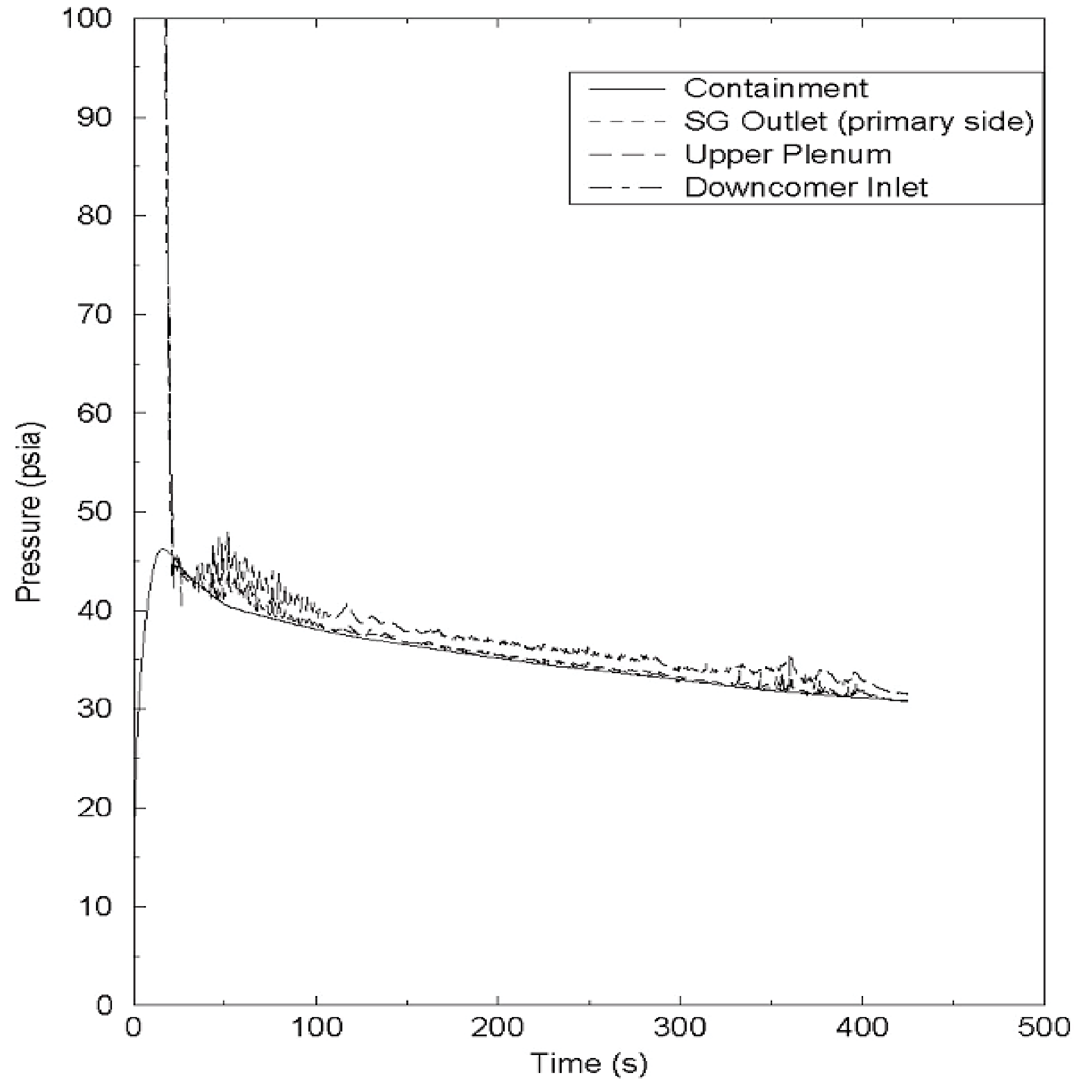
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PCT Node Heat Transfer Coefficient for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-20



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PCT Node Heat Flux for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-21

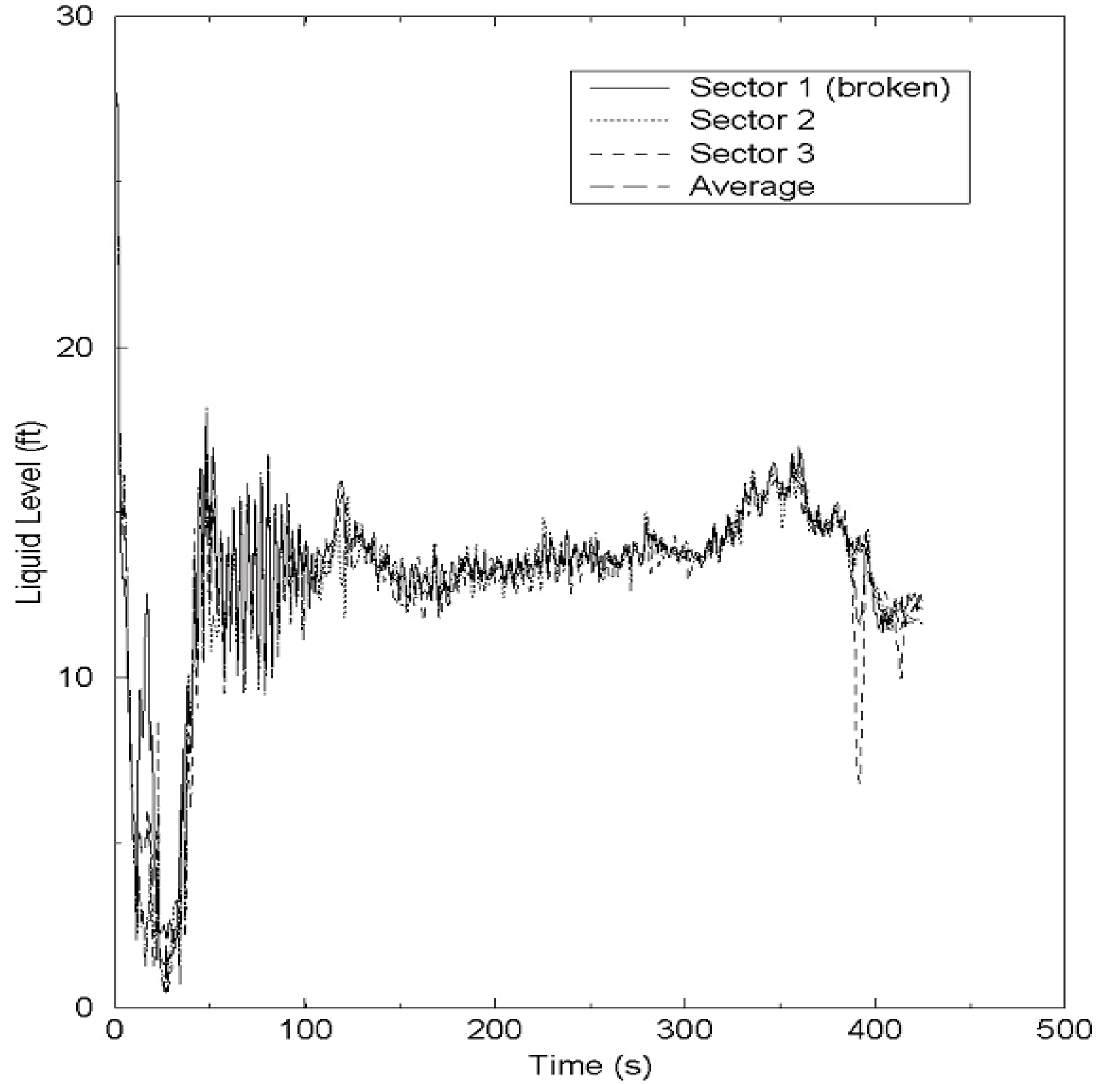


REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Containment and LOOP Pressures for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-22

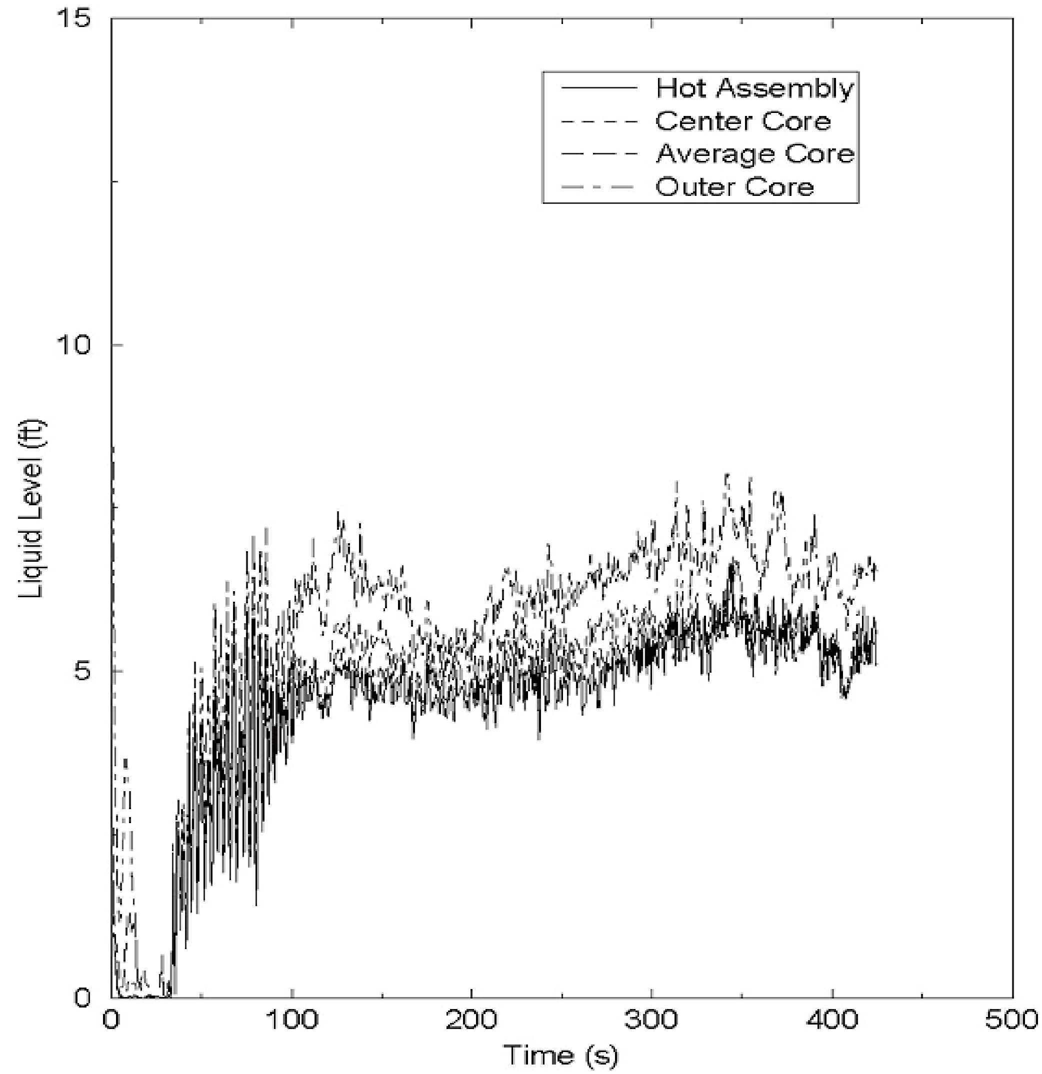
FIGURE 15.6.5-23 was deleted by Revision No. 22

FIGURE 15.6.5-24 was deleted by Revision No. 22



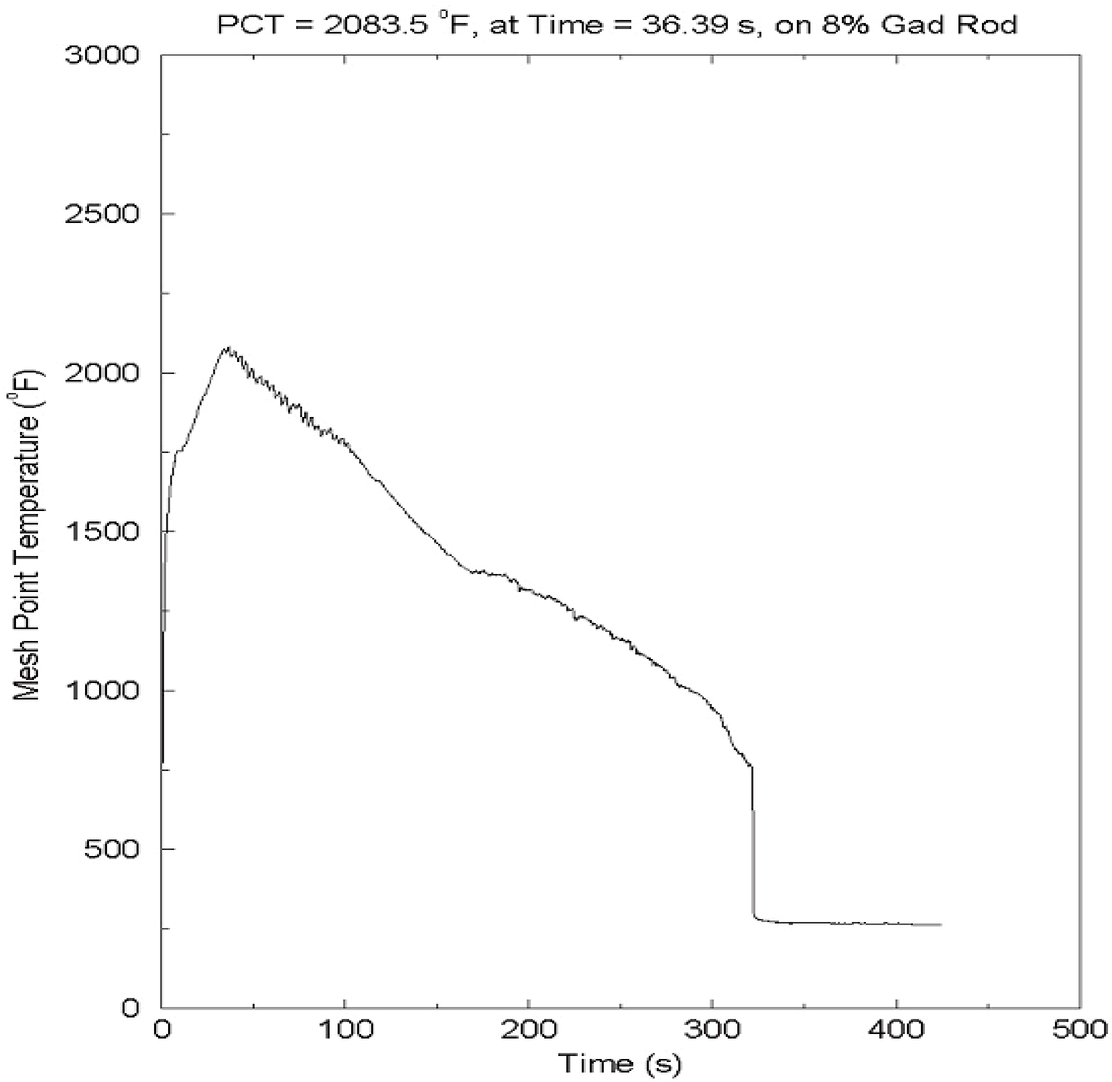
REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Downcomer Collapsed Liquid Level for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-25



REVISION NO. 24

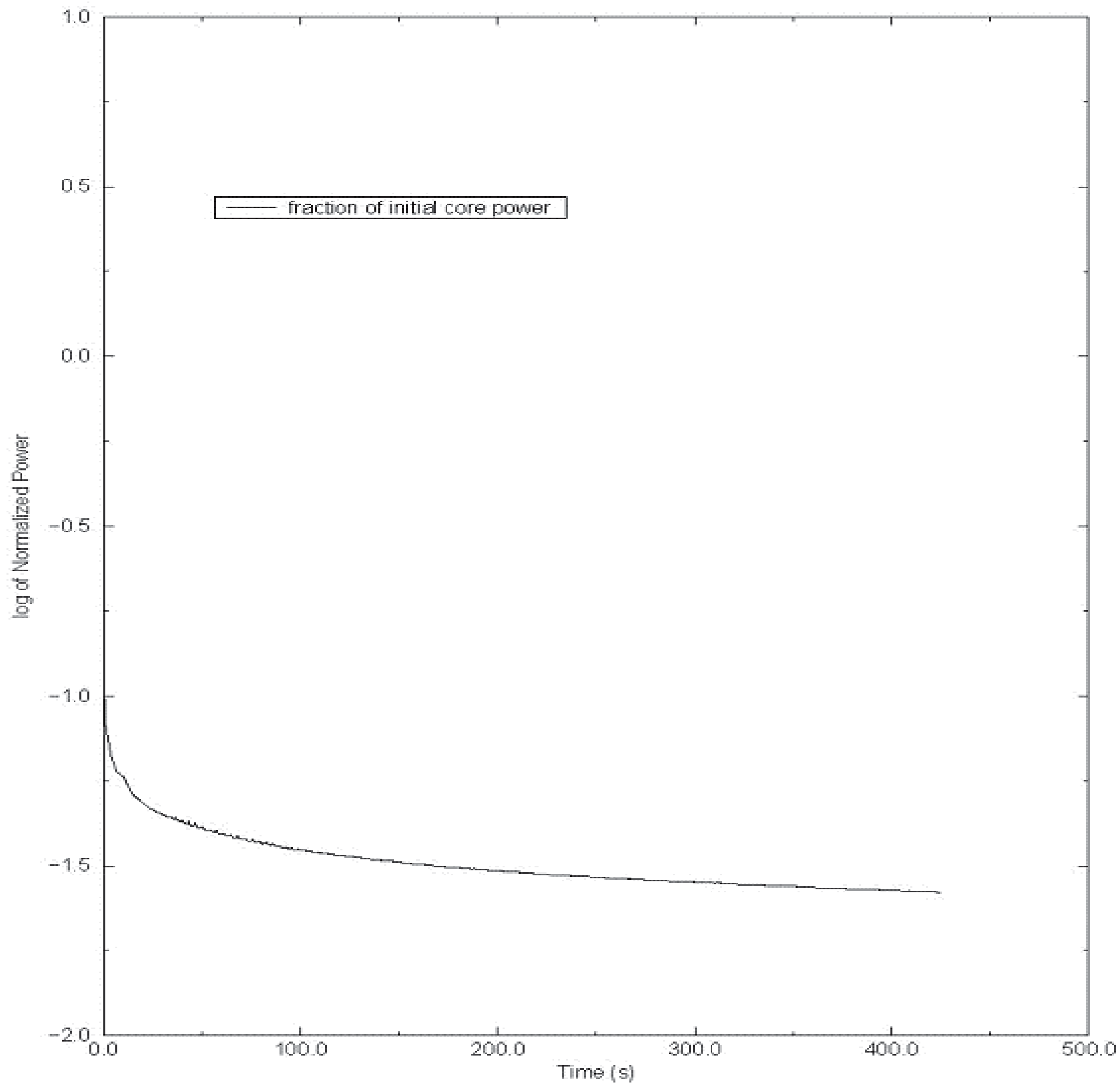
H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Core Collapsed Liquid Level for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-26



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Cladding Temperature (Independent of Elevation) for Realistic LBLOCA Limiting Case	FIGURE 15.6.5-27

Figures 15.6.5-28 through 15.6.5-32 have been deleted.



REVISION NO. 24

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized Power for LBLOCA	FIGURE 15.6.5-33

15.7 Radioactive Release From a Subsystem or Component

15.7.1 Radioactive Waste Gas System Leak or Failure

The Waste Gas Decay Tanks (WGDTs) receive the radioactive gases from the liquids processed by the waste disposal system and stripped from the reactor coolant. The maximum activity that can be stored in one tank is 19,000 dose equivalent Curies of Xe-133, in accordance with limits established in the Technical Requirements Manual (Reference 15.7.1-1). This 19,000 dose equivalent Curies of Xe-133 in the WGDT is greater than the design basis (operation with 1% defective fuel cladding) calculated noble gas activity in the Volume Control Tank or liquid waste holdup tanks. Therefore, evaluation of the radiological consequences of a failure of a WGDT bounds the consequences of the failure of the Volume Control Tank or liquid holdup tank.

The 19,000 dose equivalent Curies of Xe-133 are assumed to be instantaneously released to the Fuel Handling Building and subsequently to the environment over a two hour period. No mixing or dilution in the Fuel Handling Building is assumed. No filtration is assumed.

The dose analysis (Reference 15.7.1-2) results in a calculated 2-hour whole body dose of 0.19 rem at the Exclusion Area Boundary and 0.0097 rem at the Low Population Zone. The Control Room dose is 0.0033 rem whole body, using an assumed inleakage rate of 500 cfm. No thyroid dose was calculated as this event only involves the release of noble gases. The TEDE dose would be equivalent to the whole body dose. The offsite doses are within the acceptance criterion of 0.5 rem whole body as specified in References 15.7.1-1 and 15.7.1-3. The Control Room dose is well within the acceptance criterion of 5 rem TEDE specified in 10 CFR 50, Appendix A, General Design Criterion 19.

15.7.2 LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 Identification of Causes

Accidents that could result in the release of radioactive liquids from the Liquid Waste System may involve the rupture or leaking of system pipes, valves, pumps, instrumentation, or storage tanks. The Liquid Waste System has components located inside of the Auxiliary Building, Containment Building, Radwaste Building and Fuel Handling Building. The Liquid Waste System also has components located outside such as the pumps, valves, piping, and tanks associated with Waste Condensate Tanks "C," "D," and "E" and the Environmental & Radiation Control(E&RC) Building lab/waste sump. The Chemical and Volume Control System (CVCS) also has components containing liquid waste that are located outside, such as the piping, valves, instrumentation and tanks associated with Monitor Tanks "A" or "B."

Any liquid leakage from components located in the Auxiliary Building, Containment Building, Radwaste Building and Fuel Handling Building will be collected in the building sumps to be pumped back into the Liquid Waste System. Liquid Waste System piping running between these buildings is contained in pipe chases to contain any leakage.

Liquid leakage from components associated with Waste Condensate Tanks "C," "D," and "E" or Monitor Tanks "A" and "B" would result in a release of radioactive material to the environment. Waste Condensate Tanks "C," "D," and "E" are 11,250 gallon tanks that receive radioactive waste liquids that have been processed through the Waste Water Demineralization System (WWDS). Waste Condensate Tanks "C," "D," and "E" and their associated pumps, valves and piping were designed with the following features to minimize the potential for leaks or failures that would result in a release of radioactivity:

1) The number of valves and the amount of piping outside of the Auxiliary Building was minimized;

2) Diaphragm valves were used whenever possible instead of valves with stem packing to minimize the potential for leaks;

3) Vent and drain valves are capped and the drain valves locked closed;

4) Condensate Pumps "C" and "D" and the Recirculation Pump were enclosed in a metal building equipped with a floor weir and collection system that returns any leakage back to the Auxiliary Building floor drain system; and,

5) The Waste Condensate Tanks were provided with greater than full capacity overflow lines back to the Auxiliary Building floor drain system.

Monitor Tanks "A" and "B" are 10,000 gallon tanks that receive liquids from the CVCS Holdup Tanks that have been processed through the Base and Cation Demineralizers and the Evaporator Condensate Demineralizers. Monitor Tanks "A" and "B" and their associated valves and piping were designed with the following features to minimize the potential for leaks or failures that could result in a release of radioactivity:

1) The tanks were designed to prevent the release of contaminated liquids to the environment during the operational basis earthquake;
2) Quality control of the material and the installation of the CVCS valves and piping was provided in order to minimize leakage to the atmosphere;
3) Components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere; and
4) Diaphragm valves which have essentially zero leakage to atmosphere were used where the operating pressure and operating temperature permitted use of these valves.

Liquids flowing into and out of the Waste Condensate Tanks and Monitor Tanks are controlled by manual valve and pump operations that are governed by prescribed administrative procedures to ensure that only water that has been processed to reduce the levels of radioactivity are stored in these tanks. Technical Specification 5.5.12 requires a surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Waste Disposal System is < 10 Curies, excluding tritium and dissolved or entrained noble gases. Technical Requirements Manual (TRM) Specification 3.19 identifies the 10 Curie limit as applicable to Waste Condensate Tanks "C," "D," and "E" and Monitor Tanks "A" and "B." Administrative controls are provided in plant procedures to ensure that the Technical Specification limit on the quantity of radioactivity is met.

The E&RC Building lab/waste sump provides a holding tank for up to 1,122 gallons of diluted liquid chemical wastes from the E&RC Building. The E&RC Building lab/waste sump also has a pump and piping to transfer the wastes to the Liquid Waste System sump tank in the Fuel Handling Building. The E&RC Building lab/waste sump and the associated pump and piping were designed to meet the requirements of NRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 1, October, 1979. The following features to minimize the potential for leaks or failures that could result in a release of radioactivity were included in the design:

1) The lab/waste sump was designed to prevent the release of contaminated liquids to the environment during the operational basis earthquake;
2) The transfer line to the Liquid Waste System sump in the Fuel Handling Building has a low discharge pressure shutoff switch that will shut off the E&RC sump pump if system pressure drops to 5 psig to ensure that liquid is not released to the environment should a pipe fracture occur; and,
3) The lab/waste sump has a level switch that generates an alarm if the liquid level approaches the overflow point.

15.7.2.2 Analysis of Events and Consequences

The potential releases from a liquid waste system leak or failure that occurs in the Auxiliary Building, Containment Building, Radwaste Building or Fuel Handling Building would only involve airborne effluents since the liquid component would be contained within the building. The consequences from such a release would be bounded by the consequences of a waste gas system release as presented in Section 15.7.1.

The potential releases from a liquid waste system leak or failure involving Waste Condensate Tanks “C,” “D,” and “E”, Monitor Tanks “A” and “B” and the E&RC Building lab/waste sump and their associated components would be bounded by the consequences of a liquid tank failure as presented in Section 15.7.3.

15.7.3 POSTULATED RADIOACTIVITY RELEASES DUE TO LIQUID TANK FAILURE

15.7.3.1 Identification of Causes

There are 5 tanks located outside of the Auxiliary Building that contain radioactive liquid waste. Waste Condensate Tanks “C,” “D,” and “E” are 11,250 gallon tanks that receive radioactive waste liquids that have been processed through the Waste Water Demineralization System (WWDS). Monitor Tanks “A” and “B” are 10,000 gallon tanks that receive liquids from the Chemical and Volume Control System (CVCS) Holdup Tanks that have been processed through the Base and Cation Demineralizers and the Evaporator Condensate Demineralizers.

Failure of one of these tanks would result in radioactive liquids entering the storm drain system and flowing to the West Settling Pond (West Waste Retention Basin). The discharge from the West Settling Pond normally flows to the Discharge Canal where it mixes with Circulating Water and flows to Lake Robinson. An alternate discharge from the West Settling Pond directly to Black Creek downstream of the Lake Robinson Dam exists; however, this flow path is isolated by a locked closed valve and requires sampling/analysis of the Settling Pond for radioactivity and compliance with National Pollutant Discharge Elimination System (NPDES) limits and management approval prior to use. Based on this information, the flow path for a liquid tank failure would be via the storm drains to the West Settling Pond and then to Lake Robinson via the Discharge Canal.

A diffusion analysis was performed to determine the concentrations that would result in the lake if a release were assumed. Details of the analysis are presented in Section 2.4.6, under Diffusion of Short-Term Releases.

Technical Specifications Section 5.5.12 requires a surveillance program to ensure that the quantity of radioactivity contained in each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank’s contents and that does not have tank overflows and surrounding area drains connected to the Liquid Waste Disposal System is < 10 Curies, excluding tritium and dissolved or entrained noble gases. The purpose of the 10 Curie limit is to ensure that in the event of a tank rupture, the concentrations in the nearest potable water supply and the nearest surface water supply in an unrestricted area would not exceed the limits of 10 CFR Part 20, Appendix B, Table II that were in effect when the NRC approved the Radiological Effluent Technical Specifications (1984). Technical Requirements Manual (TRM) Specification 3.19 identifies the 10 Curie limit as applicable to Waste Condensate Tanks “C,” “D,” and “E” and Monitor Tanks “A” and “B.” Administrative controls are provided in plant procedures to ensure that the Technical Specification limit on the quantity of radioactivity is met.

15.7.3.2 Analysis of Events and Consequences

The limit of < 10 Curies, excluding tritium and dissolved or entrained noble gases, ensures that failure of a radioactive liquid-containing tank will not result in concentrations in the unrestricted area exceeding the limits of 10 CFR Part 20, Appendix B, Table II that were in effect when the NRC approved the Radiological Effluent Technical Specifications (1984).

15.7.4 Design Basis Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description

The following fuel handling accidents were evaluated to ensure that no hazards are created:

1. A fuel assembly becomes stuck inside the reactor vessel

2. A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pit

3. A fuel assembly becomes stuck in the penetration valve, and

4. A fuel assembly becomes stuck in the conveyor car or the conveyor itself becomes stuck.

The possibility of a fuel handling incident is remote because of the administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during these operations. Also, before any refueling operations begin, verification of complete rod cluster control assembly insertion is obtained by tripping the rods or tripping each rod individually to obtain indication of rod drop and disengagement from the control rod drive mechanisms. Boron concentration in the coolant is raised to the refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all rod cluster assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the drive shafts are free in the mechanism housing.

After the vessel head is removed, the rod cluster control drive shafts are removed from their respective assemblies using the containment crane and the drive shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

a) Fuel at rest is positioned by positive restraints in a safe, subcritical, geometrical array, with no credit for boric acid in the water

b) Fuel can be manipulated only one assembly at a time

c) Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality, and

d) Crane facilities do not permit the handling of heavy objects, such as a spent fuel shipping container, above the fuel racks.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Should a spent fuel assembly become stuck in the transfer tube, natural convection will maintain adequate cooling. The fuel handling equipment is described in detail in Section 9.1.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and activate a horn and illuminate a light in the plant Control Room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with ≥ 6% Δk/k shutdown margin with all rod cluster control assemblies inserted. At this boron concentration, the core would also be subcritical with all control rods withdrawn satisfying requirements of the Post LOCA Sub-Criticality event. The refueling cavity is filled with water meeting the same boric acid specification.

All these safety features make the probability of a fuel handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and that of accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pit and during installation in the reactor. All irradiated fuel handling operations are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. Also, the design of the facility is such that it is not possible to carry heavy objects, such as a spent fuel transfer cask, over the fuel assemblies in the storage racks. In addition, the design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes which move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or Fuel Storage Building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

If, during handling, the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no damage would be expected in any fuel rods.

If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it comes in contact, but breaching of the cladding is not expected. It is on this basis that the assumption of the failure of all the fuel rods in an assembly is a very conservative upper limit.

A dropped fuel assembly may result in a leak in the permanent cavity seal plate (PCSP) that requires make-up water be provided to the Reactor Cavity through a make-up source to ensure a water level of 23 feet above the RV flange is maintained.

15.7.4.2 Method of Analysis

Analyses have been made assuming the extremely remote situation where a fuel assembly is dropped and strikes a flat surface, where one assembly is dropped on another, and where one assembly strikes a sharp object. The analysis of a fuel assembly assumed to be dropped and to strike a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grid clip supports. The results showed that the buckling load at the bottom section of the fuel rod, which would receive the highest loading, would be below the critical buckling load and the stresses would be relatively low and below the yield stress.

The end plates and guide thimbles would absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. The results of this analysis indicated that the buckling load on the fuel rods would be below the critical buckling loads, and the stresses in the cladding would be relatively low and below yield.

The refueling operation experience that was obtained with Westinghouse reactors prior to operation of HBR 2 and during HBR 2 operation has verified the fact that no fuel cladding integrity failures are expected to occur during any fuel handling operations.

For the assumed accident, there would be a sudden release of the gaseous fission products held in the fuel rod plenum and in the voids between the pellets and cladding of all 204 fuel rods. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Halogen release is also greatly minimized due to their low volatility at these temperatures. The strong tendency for iodine in vapor and particulate form to be scrubbed out of gas bubbles during their ascent to the water surface further alleviates the inhalation hazard.

The HBR 2 Fuel Handling Accidents (FHA) have been analyzed in accordance with Regulatory Guide 1.183 (Reference 15.7.4-3) using the Alternative Source Term methodology and limits endorsed by 10 CFR 50.67 (Reference 15.7.4-4). These references provide detailed descriptions of acceptable modeling and analysis methods. In general, the HBR 2 FHA dose analysis complies with those specified methods. In the following paragraphs, the HBR 2 specific conditions and modeling assumptions which differ from the generically approved models in the References will be discussed.

HBR 2 specific conditions in the Spent Fuel Pit (References 15.4.7-1 and -2) required that a minimum of 21 feet of water coverage be evaluated against the generic decontamination factors (DF) approved in these references, which was 23 feet of coverage. Reference 15.7.4-3 provides guidance that Reference 15.7.4-5 may be used to evaluate DF for coverage less than the generic 23 feet. In order to apply the equations for evaluating the overall DF in Reference 15.7.4-5, it was necessary to derive an elemental iodine DF from the overall DF of 200 that is specified in Reference 15.7.4-3. The elemental DF was re-evaluated, in accordance with the relationships defined in Reference 15.7.4-5, for 21 feet of coverage. Elemental DF was conservatively reduced from the generic, 23 foot coverage value of DF=285 to a 21 foot coverage value of DF=174. Organic DF (the other component of the overall DF) was specified in Reference 15.7.4-3 to be DF=1, which means that this chemical species is not reduced at all by the water coverage. Since this component is already at its most conservative value, it is not affected by the change in water coverage from the generic 23 feet to 21 feet. Recombining the elemental and organic components using the overall DF equation from Reference 15.7.4-5, the generic overall DF from Reference 15.7.4-3 is reduced from DF=200 to DF=138.

The activity could be released either in the containment or in the Auxiliary (Fuel Storage) Building. Ventilation systems in both areas are in operation under administrative control during refueling. In evaluating doses inside the structures, the assumption is made that the release is drawn directly into the ventilation system before substantial mixing occurs. Radioactivity monitors would immediately indicate and alarm the increased activity level. Activity in the containment would automatically close the purge ducts, although the offsite dose was conservatively evaluated assuming that the entire radionuclide inventory would be released before containment isolation. In evaluating the dose to refueling personnel inside the containment, or inside the Fuel Storage Building, the lack of substantial mixing and existence of alarms that would cause a prompt evacuation, lead to the conclusion that the total personnel dose would be small. Following evacuation, re-entry to the buildings would be delayed or otherwise planned using indicated radiation levels, and would factor in any airborne radioactivity cleanup that may have occurred.

The Alternative Source Term methodology of Reference 15.7.4-3 is used to calculate offsite exposures using the RADTRAD Version 3.02 computer code (Reference 15.7.4-7). The analysis of the accident occurring in the containment does not credit containment ventilation filtration systems, in order to conservatively bound the consequences of the event which might occur with containment openings, such as the equipment hatch, personnel airlock, or other penetrations, not sealed. In accordance with Reference 15.7.4-3, all activity released to the containment atmosphere is released to the environment over a two hour period. The analysis of the accident occurring in the fuel handling building does apply credit for the fuel handling building ventilation and filtration systems. All activity which is released to the fuel handling building atmosphere is released over a two hour period through the ventilation and filtration systems to the environment. Conservatively, both the containment and fuel handling building releases are modeled as ground level releases for offsite dose analysis purposes.

HBR 2 specific Atmospheric Dispersion Factors (known as X/Q factors) for offsite dose consequence evaluations were developed from meteorological data gathered at HBR 2 during the 9 year time period of 1988 through 1996. The data was evaluated using the PAVAN computer code (described in Reference 15.7.4-9), which implements Regulatory Guide 1.145 methods (Reference 15.7.4-10). For on-site receptor locations (including such locations at the Control Room), the meteorological data from this same time period was evaluated using the ARCON96 computer code (described in Reference 15.7.4-6). Minor changes to ARCON96 default values were made to implement draft NRC guidance contained in Reference 15.7.4-8.

HBR-2 specific isotopic source terms were developed using a bounding approach. The ORIGEN-S computer code was used to develop isotopics for a variety of burnups, enrichments, and burnup rates (power levels). Sensitivity studies were run with various combinations of burnups and enrichments to identify a bounding single assembly isotopic source term. The assembly source term was multiplied by 1.02 to reflect operation prior to shutdown at 102% of rated power (102% of 2300 MWth). A bounding local peaking factor of 1.8 was then applied to simulate the effect of this accident for the assembly containing the peak fission product inventory.

The dose to the operator in the Control Room was evaluated for the implementation of the Reference 15.7.4-3 methodology. Table 15.7.4-3 provides the Control Room modeling parameters used as input for this evaluation. Control Room unfiltered air inleakage for these dose analyses was conservatively evaluated for a total of 300 cfm. Automatic switchover, backed up by confirmatory operator action, is credited at one hour to switch the Control Room ventilation from Normal alignment to Emergency Pressurization mode.

15.7.4.3 Radiological Consequences

15.7.4.3.1 Postulated fuel handling accident in the fuel handling building

Using the assumptions listed in Table 15.7.4-1, and the Reference 15.7.4-3 analysis methods, the offsite and HBR 2 Control Room TEDE doses due to the FHA in the Fuel Handling Building are shown in Table 15.7.4-4 to meet the specified acceptance criteria

15.7.4.3.2 Postulated fuel handling accident inside containment

Using the assumptions listed in Table 15.7.4-2, and the Reference 15.7.4-3 analysis methods, the offsite and HBR 2 Control Room TEDE doses due to the FHA in the Containment are shown in Table 15.7.4-4 to meet the specified acceptance criteria.

TABLE 15.7.4-1

FUEL HANDLING ACCIDENT IN FUEL HANDLING BUILDING

ASSUMPTIONS

1. Accident occurs at least 56 hours after reactor shutdown

2. Plant thermal power (prior to shutdown) is 2346 MWth

3. All rods in one assembly rupture, releasing their gap activity

4. Burnup in affected assembly is bounded up to 60,000 MWD/MTU

5. Enrichment in affected assembly is bounded up to a nominal 4.95 weight percent U-235. Higher enrichments up to 5.0 weight percent are bounded by sensitivity study results

6. Assembly Peaking Factor: 1.8

7. Fraction of assembly activity in gap:

I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

8. Iodine form split:

	In Clad Gap	Above Pool
Aerosol (CsI)	95%	N/A
Elemental	4.85%	57%
Organic	0.15%	43%

9. Pool DF (for 21 foot coverage):

Elemental Iodine	174
Organic Iodine	1
Effective (Iodine)	138
Noble Gases	1

10. 157 assemblies in the core

11. All activity released from the pool is exhausted as a ground level release over two hours to the environment through the FHB air handling and filtration system.

12. FHB air handling system filter efficiencies for iodine removal:

Elemental Iodine	90%
Organic Iodine	70%

TABLE 15.7.4-1 (Continued)

FUEL HANDLING ACCIDENT IN FUEL HANDLING BUILDING

ASSUMPTIONS

13. Breathing Rate: (m^3/sec)

Time Period	EAB/LPZ[(1)]	Control Room
0-8 hr	3.5E-04	3.5E-04
8-24 hr	1.8E-04	3.5E-04
1-30 days	2.3E-04	3.5E-04

14. Atmospheric Dispersion Factors (X/Q): (sec/m^3)

Time Period	EAB	LPZ	Control Room
0 – 2 hours	1.77E-03	8.92E-05	1.24E-03

15. Full Core Isotopic Activity:

Nuclide	Curies at T=0
I-131	6.20E+07
I-132	9.02E+07
I-133	1.28E+08
I-134	1.41E+08
I-135	1.21E+08
Kr-85	7.30E+05
Kr-85m	1.51E+07
Kr-87	3.03E+07
Kr-88	4.20E+07
Xe-133	1.28E+08
Xe-135	3.68E+07

(1) - EAB = Exclusion Area Boundary
LPZ = Low Population Zone

TABLE 15.7.4-2

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

ASSUMPTIONS

1. Accident occurs at least 56 hours after reactor shutdown

2. Plant thermal power (prior to shutdown) is 2346 MWth

3. All rods in one assembly rupture, releasing their gap activity

4. Burnup in affected assembly is bounded up to 60,000 MWD/MTU

5. Enrichment in affected assembly is bounded up to a nominal 4.95 weight percent U-235. Higher enrichments up to 5.0 weight percent are bounded by sensitivity study results

6. Assembly Peaking Factor: 1.8

7. Fraction of assembly activity in gap:

I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

8. Iodine form split:

	In Clad Gap	Above Pool
Aerosol (CsI)	95%	N/A
Elemental	4.85%	57%
Organic	0.15%	43%

9. Cavity DF (for 23 foot coverage):

Elemental Iodine	500
Organic Iodine	1
Effective (Iodine)	200
Noble Gases	1

10. 157 assemblies in the core

11. All activity released from the cavity is exhausted as a ground level release over two hours to the environment

12. Containment air handling system and containment closure are not credited

13. Breathing Rate (m^3/sec):

Time Period	EAB/LPZ[(1)]	Control Room
0-8 hr	3.5E-04	3.5E-04
8-24 hr	1.8E-04	3.5E-04
1-30 days	2.3E-04	3.5E-04

TABLE 15.7.4-2 (Continued)

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

ASSUMPTIONS

14. Atmospheric Dispersion Factors (X/Q): (sec/m^3)

Time Period	EAB	LPZ	Control Room
0 – 2 hours	1.77E-03	8.92E-05	4.15E-03

15. Full Core Isotopic Activity:

Nuclide	Curies at T=0
I-131	6.20E+07
I-132	9.02E+07
I-133	1.28E+08
I-134	1.41E+08
I-135	1.21E+08
Kr-85	7.30E+05
Kr-85m	1.51E+07
Kr-87	3.03E+07
Kr-88	4.20E+07
Xe-133	1.28E+08
Xe-135	3.68E+07

(1) – EAB = Exclusion Area Boundary
LPZ = Low Population Zone

TABLE 15.7.4-3

FUEL HANDLING ACCIDENT DOSE ANALYSIS

CONTROL ROOM PARAMETERS

Volume (ft^3)	20,124
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	400.0
Unfiltered Inleakage (Total)	300.0
Unfiltered Recirculation Flow Rate	(Not modeled)
Pressurization Mode Flow Rates (cfm)	
Filtered Makeup Air Flow Rate	400.0
Filtered Recirculation Flow Rate	2600.0
Unfiltered Inleakage (Total, Initial)	300.0
Hagan Room Unfiltered Air Inleakage (Terminates after 1 hour)	70.0
Unfiltered Recirculation Flow Rate	(Not modeled)
Filter Efficiencies (%)	
Elemental	95
Organic	95
Particulate	99
Automatic Switchover from Normal to Pressurization Mode	1 hour

TABLE 15.7.4-4

FUEL HANDLING ACCIDENT DOSE ANALYSIS RESULTS

Event	EAB[1] (rem TEDE)	LPZ[2] (rem TEDE)	CR Dose[3] (rem TEDE)
FHA Inside Containment	5.96	0.30	4.46
FHA Inside FHB	5.70	0.29	0.55
Regulatory Limit	6.3	6.3	5

1. Worst 2-hour integrated dose at Exclusion Area Boundary.
2. 30-day integrated dose at Low Population Zone.
3. Assumes a conservative unfiltered inleakage of 300 cfm for the first hour and 230 cfm thereafter.

15.7.5 Spent Fuel Cask Drop Accidents

A postulated cask drop which could occur at the pool edge and result in the cask being deflected into spent fuel has been eliminated as a credible accident by design considerations. Redundancy has been incorporated in the design of the spent fuel cask lifting lugs, redundant lifting yoke, and the replacement 125-ton spent fuel cask handling crane to eliminate any risk to public health and safety from this postulated accident.

A postulated cask drop could occur while the cask is being lifted with the redundant yoke between the spent fuel building and the decontamination facility. Redundancy has been incorporated in the design of the spent fuel cask lifting lugs, the redundant yoke and the replacement 125 ton spent fuel cask handling crane to eliminate any risk to the public health and safety from this pstulated accident.

A postulated cask drop could occur while the cask is being lifted with the non-redundant yoke between the decontamination facility and the shipping railcar. Administrative controls described in Section 15.7.5.3 limit the cask lift height to less than 30 feet (per 10 CFR 71.73) In accordance with the operating instructions supplied by the IF-300 cask vendor, the lift of the loaded cask from the decontamination facility to the railcar is performed with the head bolts fully tensioned drain/vent valves closed and the cask tested for leakage. The drain/vent valve covers are not installed, however, so the 30 foot drop analysis performed per 10 CFR 71.73 does not bound this situation. An evaluation of the 30 foot drop during movement from the decontamination facility to the railcar was performed and indicated that, while fuel components would be retained in the cask, the IF-300 cask vent/drain valves may be damaged, and thus not gas tight. A release of noble gas and iodine gap activity to the environment could occur. Using the maximum activity loading for the IF-300 cask, this type of release has been evaluated (Reference 15.7.5-1) and the whole body and thyroid doses which could result are a small fraction of those previously analyzed for the fuel handling accident in Section 15.7.4. The path of this lift does not go over any safety related structures, systems or components (SSC's).

A detailed discussion of the safety features of each component is given below to demonstrate sufficient redundancy that dropping the spent fuel cask is not a credible accident.

15.7.5.1 Spent Fuel Cask Non-Redundant and Redundant Lifting Yoke

A non-redundant lifting yoke was supplied with the spent fuel shipping cask for lifting the cask where the redundant yoke was not needed, or not possible to use because of the configuration of HBRSEP equipment or buildings. The design of the non-redundant lifting yoke meets the same criteria listed below for the redundant lifting yoke, except for the additional criteria describing redundancy.

The redundant lifting yoke supplied for the spent fuel shipping cask is furnished as part of a package which includes the shipping cask and its special transport vehicle. Specific details of the redundant lifting yoke conform to the following criteria. The design and fabrication of the shipping cask, transport vehicle, and handling equipment conform to all the applicable regulations of the NRC (10 CFR 71) and the DOT (49 CFR 170-178). The shipping cask redundant lifting yoke is of all steel construction and composed entirely of structural members. In addition to the above criteria the redundant lifting yoke was designed for protection against single failure in that both the primary and secondary parts of the yoke will alone support 300 percent of the fully loaded cask weight without exceeding the yield strength of the material. The secondary yoke is connected to the sister hook of the crane and the primary yoke is independently attached to the lifting eye of the crane.

Before using for shipments, both the primary and secondary parts of the redundant yoke were proof-load tested (200 percent of rated capacity) to assure compliance with the single failure criteria, and non-destructively tested i.e., magnetic particle or dye penetrant, and examined to ensure that no permanent deformations and/or other damage occurred. This design and testing eliminates the redundant lifting yoke as a factor contributing to a cask drop accident.

15.7.5.2 Cask Handling Crane (125 Ton Capacity, 110 Ton Maximum Critical Load)

The Whiting Redundant Hoist System consists of a dual load path through the hoist gear train, the reeving system, and the hoist load block along with restraints at critical points to provide load retention and minimize uncontrolled motions of the load upon failure of any single hoist component. The system includes two complete gear trains connecting the single hoist motor to the hoist drum.

Each gear train is designed to accept full motor torque at rated load capacity along with peak strength ratings adequate to absorb shock loadings within the yield strength of the component materials. Separate motor brakes are included with wheels mounted on an extension of each motor pinion input shaft.

The hoist drum and its shafts and bearings were designed to accept the forces and moments produced by full load tension on either half of its grooving or reeving and in addition is provided with close clearance retainers at its hubs to support the drum and prevent loss of pinion mesh in case of shaft or bearing failure at either or both ends of the drum.

Reeving consists of a sufficient number of parts of rope sized as commercially available to provide a minimum static factor of safety of 7.4 with all parts of rope effective, and based on the ultimate rope strength and the static rated load as defined by Crane Manufacturers Association of America, Inc. (CMAA) #70 specifications. This rope is furnished as two separate pieces, each of which is fastened at one end to the drum in conventional manner, reeved through the upper and lower blocks of the trolley as described below, and the other end adjustably attached to a specially damped equalizer assembly. Each rope is 1 1/8 in. diameter 6 x 37 Type 304 stainless steel with a breaking strength of 59.3 tons. This equalizer assembly is also provided with special retainers to assure its continued support of the load in case of pivot pin failure. Hydraulic dampers and mechanical stops are also provided on this assembly to define its maximum rate and extent of rotation about its pivot pin in either direction. If either piece of rope should fail, the equalizer assembly dampens the forces developed in the remaining rope caused by its increase in strain in order to continue support of the load. This damping system, however, does not interfere with the normally small and slow oscillations of the equalizer during rope tension equalizing functions while all parts of rope are effectively supporting the load. A special limit switch system is also supplied on this equalizer assembly which can either stop the hoist or provide a warning to the operator if unequal rope stretch or other causes have moved either end of the assembly to the danger point where insufficient damping travel remains for proper damping action in case of rope failure in the longer rope. This signal implies that an adjustment of either or both rope anchors at the equalizer should be made prior to critical load handling.

Upper and lower block sheaves are a minimum of 24 rope diameters and are uniquely arranged in upper and lower blocks so that the total sustaining force of all effective ropes remain nearly coaxial and concentric with the vertical axis of the hook shanks whether either or both pieces of rope are supporting the load. Each sheave in both the upper and lower block is also provided with vertical and lateral restraints that will assure continued rope tension in its ropes in case of sheave shaft or bearing failures.

The load block provides a dual concentric pair of load connecting devices to carry the load into and through the block housing and sheaves, either of which has the ability to sustain the full load, while still providing normal load rotation capabilities. The normal load path is to the lower connector consisting of an "eye" similar to that found on the ordinary crane hook and capable of connection to any existing handling devices designed for such load connection. The upper device consists of a sister type crane hook also capable of supporting the full load which will accommodate secondary load connections such as slings or pinned links connecting to the load or handling device.

All structural, i.e. load bearing members have been fabricated of ASTM A-36 steel in accordance with CMAA Specification #70. In addition, the crane was proof-load tested to 125 percent of its rated capacity prior to being put in service. All applicable portions of OSHA 1910.179 and ANSI B30.2.0 have been incorporated in the crane design.

The design criteria for the crane support structure (fuel handling building superstructure) are described in Section 9.1.4.1.1.

The allowable unit stresses are less than those permitted by AISC. Stresses are based on combined loads determined in accordance with the design criteria listed above.

The crane is equipped with a single trolley which contains a 125 ton capacity main hoist with redundant features as described above and a 5 ton capacity auxiliary hoist. The speeds and motor requirements are listed as follows:

	Main Hoist	Auxiliary Hoist	Trolley Traverse	Bridge Travel
Speeds (ft/min)				
Full Load	3	25	40	60
No Load	3	25	40	60
Motor Horsepower	40	10	5	15
Motor Speed (rpm)	900	900	1200	1200

All motors are controlled by magnetic controllers to provide variable speed for each crane motion by means of variance of motor secondary currents through resistors. Each motor has five steps of control with each step providing a reduction of approximately 20% in speed.

The bridge span for the 125 ton crane is 37 ft. The vertical travel of the main hook is 77 ft 10 3/4 in. and the auxiliary hook is 87 ft 9 in.

The maximum allowable stress for a combined load is 17,600 psi. Based on the yield point for ASTM-A-36 of 36,000 psi, a minimum factor of safety of two will exist for welded box girders in the bridge. The minimum safety factor for any non-redundant load-bearing parts, except structural members and ropes, is 3.5 based on yield strength for the maximum critical load.

The minimum safety factor for one path of redundant load carry parts, except structural members and ropes, is 2.5 (or 5 for both paths) based on yield strength for the maximum critical load.

Limit switches have been provided to limit the horizontal movement of the trolley and of the bridge. The switches have been located so that movement of the loaded crane over the spent fuel pool is prevented. Limit switches have also been provided to prevent overloads and critical elevations of the block assembly.

A two blocking situation which would result from raising the load too high is prevented by a paddle type limit switch which opens the hoist motor circuit as the hook reaches the upper limit of travel. The operation of the limit switch does not prevent lowering the hook. Operational procedures prohibit using this upper limit switch as a routine operational limit. A special limit switch system on the reeving equalizer assembly warns the operator if unequal rope stretch or an unbalanced condition occurs in the assembly. A slack cable condition which would result from overhoisting is prevented by a limit switch which opens the hoist motor circuit as the cable becomes slack. An overload limit device is incorporated into the design of the main hoist and interrupts power to the main hoist motor if the load exceeds rated capacity.

The brake system consists of the following:

a) One disc mechanical load brake for each hoist gear case, a total of two for the crane

b) One electric shoe brake on each motor shaft of each hoist gear case, a total of two for the crane

c) A service (electric shoe) brake is provided on the trolley, and

d) A hydraulic brake with electric stopping feature is provided on the bridge.

This redundant breaking system together with the other redundant components of the crane insures that dropping the spent fuel cask is not a credible accident.

The main hoist has two electric stopping and holding brakes (Whiting Type 13 in. SESA) mounted on the motor shaft on each side of the motor and two mechanical control brakes which are built into each main hoist reduction gear (Whiting Type #25). Each brake is capable of stopping and holding 150 percent of the rated load. A solenoid is energized to release each electric brake thereby releasing the brake shoes from the wheels. Brakes are applied by opening the circuit to the solenoids allowing a compression spring to extend and force the brake arms to the set position. Electric brakes always maintain a safe condition by automatically setting and holding the load in case of power interruption. Brake solenoids are connected across two phases of the main hoist motor which results in the brakes being released when the motor is energized and being set when the motor is deenergized. The major components of the mechanical control brakes are located on a brake shaft in each main hoist reduction gear case. Situated between the brake gear and pinion on each brake shaft is a ratchet wheel which is governed by a pawl actuated by a friction sleeve on the motor drive shaft. The pinion on the motor drive shaft transmits its torque to the brake gear. The ratchet wheel is provided with two friction washers and is free to idle on the brake shaft but is held stationary when engaged by the pawl. The brake gear is not keyed to the brake shaft but transmits its torque to the shaft, through a brake nut which turns on a screw that is an integral part of the shaft. The brake pinion transmits its torque to the gear on the output shaft.

The starting of the hoisting cycle causes the brake nut to advance along the screw in the direction of the ratchet wheel until the friction washers are engaged, at which point the entire assembly operates as if it were simply a shaft with a gear and pinion keyed to it.

When the motor is reversed to lower, the pawl actuated by the motor drive shaft promptly engages the ratchet wheel and holds it stationary. The continued turning of the brake gear backs the brake nut off the screw, thereby loosening the entire assembly and allowing the load to lower. Should the load begin to drop faster than the speed for which the motor controller is set, the brake immediately tightens up and retards the load to the controller speed. At the same time that the lowering load is tightening the brake, the torque of the motor is being used to keep the brake loose, resulting in an alternate tightening and loosening that occurs in rapid succession. Hence, the load is lowered smoothly, without exceeding the speed for which the controller is set.

The auxiliary hoist has one electric stopping and holding brake (Whiting Type 13 in. SESA) mounted on a shaft extended from the first pinion shaft and one mechanical control brake which is built into the auxiliary hoist reduction gear (Whiting Type #10). These brakes operate in the same manner as described above for the main hoist brakes.

The trolley has one electric stopping and holding brake (Whiting Type 6 in. SESA). Operation of the brake is the same as described above for the electric brake for the main hoist. The solenoid for this brake is connected across two phases of the trolley drive motor.

The bridge has one hydraulic drum type brake mounted on the bridge drive shaft with an automatic electric parking brake feature (Wagner Type 10 in. HM). For stopping duty it operates by pressing on a foot pedal in the cab which transmits hydraulic pressure to the brake unit which closes the brake shoes onto the brake drum thereby bringing the crane to a stop. Whenever electric power to the crane is broken, whether due to power failure, or opening of the main line switch, the brake automatically sets, bringing the crane to an emergency stop. The brake remains set until electric power is restored.

15.7.5.3 Administrative Considerations

The spent fuel cask cannot be handled over critical safety systems or equipment within the coverage provided by the 125 ton spent fuel cask handling crane.

The spent fuel cask can be positioned during handling operations so that the vertical distance between it and a flat essentially unyielding horizontal surface exceeds the hypothetical accident condition of a 30 ft drop (per 10CFR71.36). This will occur when the cask is moved between the cask decontamination room and the entrance to the spent fuel building. The cask decontamination room floor is approximately 50 ft below the entrance to spent fuel building. As explained above, the cask drop during this move has been eliminated as a credible accident due to redundant safety features designed into the crane, cask lifting lugs, and redundant lifting yoke.

The spent fuel cask is lifted with the non-redundant lifting yoke between the decontamination facility and the shipping railcar. Administrative controls are implemented in plant procedures to limit the vertical distance between the cask and a flat essentially unyielding horizontal surface to less than 30 feet (per 10CFR71.73). An evaluation of the 30 foot drop during movement from the decontamination facility to the railcar was performed and indicated that, while fuel components would be retained in the cask, the IF-300 cask vent/drain valves may be damaged, and thus not gas tight. A release of noble gas and iodine gap activity to the environment could occur. Using the maximum activity loading for the IF-300 cask, this type of release has been evaluated (Reference 15.7.5-1) and the whole body and thyroid doses which could result are a small fraction of those previously analyzed for the fuel handling accident in Section 15.7.4. The path of this lift does not go over any safety related structures, systems or components (SSC's). Therefore, any risk to the public health and safety due to a cask drop during this non-redundant lift has been eliminated.

15.7.6 SPENT FUEL PIT WATER LOSS

Loss of water from the spent fuel pit is expected only by means of evaporation. The spent fuel pit is a seismic Class I concrete structure with a stainless steel liner of all welded construction. All welds are liquid penetrant and vacuum box tested.

The evaporative losses are replenished by primary demineralized water from the 150,000 gal primary water storage tank. A redundant supply of makeup is provided by the fire hoses in the vicinity of the spent fuel pit.

Leak detection is achieved by 10 one-inch diameter pipes imbedded in the concrete along the bottom of the pit where the walls join the floor approximately one inch below the liner plates. These leak detectors are valved and piped to an open floor drain in an area which is accessible at all times.

Analyses have also been performed of abnormal boron dilution accidents to confirm that the loss of soluble boron concentration would be readily detected in ample time for corrective action.

REFERENCES: SECTION 15.7

15.7.1-1 Technical Requirements Manual, Section 3.21

15.7.1-2 Calculation RNP-M/MECH-1735, “Radiological Consequence Analysis of the Waste Gas Decay Tank Rupture”

15.7.1-3 Technical Specifications, Section 5.5.12

15.7.4-1 CP&L letter to NRC RNP-RA/97-0190, "Request for Approval of Proposed Change to Updated Final Safety Analysis Report that Involves an Unreviewed Safety Question."

15.7.4-2 NRC letter dated January 27, 1998. Issuance of Amendment No. 177 to Facility Operating License No. DPR-23 regarding revision to Spent Fuel Handling Accident Analysis, H. B. Robinson Steam Electric Plant, Unit No. 2 (TAC No. 99822)

15.7.4-3 USNRC Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000

15.7.4-4 0 CFR 50.67, “Accident Source Term”

15.7.4-5 “Evaluation of Fission Product Release and Transport for a Fuel Handling Accident,” G. Burley, NRC Staff Technical Paper, October 5, 1971 (NRC Accession Number 8402080322 in ADAMS or PARS).

15.7.4-6 “Atmospheric Relative Concentrations in Building Wakes,” NUREG/CR-6331, Rev. 1, May 1997. ARCON96, RSICC Computer Code Collection No. CCC-664

15.7.4-7 “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation”, NUREG/CR-6604, April 1998 and Supplement 1, June 8, 1999

15.7.4-8 USNRC Draft Regulatory Guide DG-1111, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” December 2001

15.7.4-9 “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations,” NUREG/CR-2858, November 1982. RSICC Computer Code Collection No. CCC-445

15.7.4-10 USNRC Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments of Nuclear Power Plants,” Rev. 2

15.7.5-1 CP&L letters: RNP-RA/97-0128, August 28, 1997; RNP-RA/98-0113, June 17, 1998; RNP-RA/99-0033, February 11, 1999.
NRC letter dated April 7, 1999