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**The acronym FON, which is utilized on select Chapter 15 Figures, is defined as fraction of nominal.

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15.1-13	0	Pressurizer Pressure, Pressurizer Water Volume and Core Boron Concentration vs. Time for Failure of a Steam Generator Safety or Atmospheric Relief Valve	
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15.1-20a	0	Deleted	
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**The acronym FON, which is utilized on select Chapter 15 Figures, is defined as fraction of nominal.

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15.1-26	0	Steam Pressure and Core Flow vs. Time for SLB Transient at Hot Zero Power without Offsite Power Available	
15.1-27	0	Nuclear Power and Core Heat Flux vs. Time for SLB Transient at Hot Full Power	
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15.2-3	0	Nuclear Power and Pressure for Turbine Trip Event with Minimum Reactivity Feedback DNB Evaluation	
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15.2-5	0	Nuclear Power and Pressure for Turbine Trip Event with Minimum Reactivity Feedback SG Overpressure Evaluation	
15.2-6	0	Pressurizer Liquid Volume and RCS Temperature for Turbine Trip Event with Minimum Reactivity Feedback SG Overpressure Evaluation	
15.2-7	0	Deleted	
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15.2-9	0	Nuclear Power, Reactor Coolant Loop Flow Rate and Steam Generator Pressure Transients for Loss of AC Power	
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**The acronym fon, which is utilized on select Chapter 15 Figures, is defined as fraction of nominal.

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15.3-11	0	Hot Channel Heat Flux Transient for Four Loops in Operation, Four Pumps Coasting Down	
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15.4-3	0	Fuel Average and Clad Surface Temperatures at the Hot Spot for an Uncontrolled Rod Withdrawal from a Subcritical Condition	
15.4-4		Deleted	
15.4-4a	0	Uncontrolled RCCA Bank Withdrawal at Power (100% Power, Minimum Reactivity Feedback, 110 pcm/sec Reactivity Insertion Rate) Nuclear Power and Core Heat Flux	
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15.4-4c	0	Uncontrolled RCCA Bank Withdrawal at Power (100% Power, Minimum Reactivity Feedback, 110 pcm/sec Reactivity Insertion Rate) Vessel Average Temperature and DNBR	
15.4-4d	0	Deleted	
15.4-4e	0	Deleted	
15.4-4f	0	Deleted	
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15.4-5	0	Deleted	

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15.4-5c	0	Uncontrolled RCCA Bank Withdrawal at Power (100% Power, Minimum Reactivity Feedback, 1 pcm/sec Reactivity Insertion Rate) Vessel Average Temperature and DNBR	
15.4-6	0	Deleted	
15.4-6a	0	Uncontrolled RCCA Bank Withdrawal at Power (100% Power, Minimum and Maximum Reactivity Feedback) Minimum DNBR vs. Reactivity Insertion Rate	
15.4-6b	0	Uncontrolled RCCA Bank Withdrawal at Power (60% Power, Minimum and Maximum Reactivity Feedback) Minimum DNBR vs. Reactivity Insertion Rate	
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15.4-7	0	Deleted	
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15.4-7b	0	Uncontrolled RCCA Bank Withdrawal at Power (RCS Overpressure Limiting Case) Pressurizer Pressure and Water Volume	
15.4-7c	0	Uncontrolled RCCA Bank Withdrawal at Power (RCS Overpressure Limiting Case) Vessel Average Temperature and Peak RCS Pressure	
15.4-8	0	Removed	
15.4-9	0	Removed	
15.4-10	0	Nuclear Power Transient and Core Heat Flux Transient for Dropped Rod Cluster Control Assembly	
15.4-11a	0	Change in Pressurizer Pressure and Core Average Temperature for Dropped Rod Cluster Control Assembly	
15.4-11b	0	Deleted	
15.4-12	0	Nuclear Power Transient for Startup of an Inactive Reactor Coolant Loop	
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15.4-16	0	DNBR Transient for Startup of an Inactive Reactor Coolant Loop	
15.4-17	0	Deleted	
15.4-18	0	Deleted	
15.4-19	0	Deleted	
15.4-20	0	Deleted	
15.4-21	0	Deleted	
15.4-22	0	Nuclear Power Transient, BOL, HFP, Rod Ejection Accident	
15.4-23	0	Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, BOL, HFP, Rod Ejection Accident	
15.4-24	0	Nuclear Power Transient, EOL, HZP, Rod Ejection Accident	
15.4-25	0	Fuel Centerline, Fuel Average, and Cladding Surface Temperature Transients, EOL, HZP, Rod Ejection Accident	
15.4-26	0	Steam Generator Secondary Pressure Transient for Rod Ejection Analysis	
15.4-27	0	RCS Pressure Transient for Rod Ejection Analysis	
15.4-28	0	Deleted	
15.5-1	0	Nuclear Power and Vessel Average Temperature Transients for Inadvertent Operation of ECCS During Power Operation	
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15.5-6	0	Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Maximum Reactivity Feedback and No Pressurizer Spray	
15.5-7	0	Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Maximum Reactivity Feedback and No Pressurizer Spray	
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15.5-9	0	Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Minimum Reactivity Feedback and Pressurizer Spray	
15.5-10	0	Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Minimum Reactivity Feedback and No Pressurizer Spray	
15.5-11	0	Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Minimum Reactivity Feedback and No Pressurizer Spray	
15.6-1	0	Nuclear Power and Pressurizer Pressure Transients for Inadvertent Opening of a Pressurizer Safety Valve	
15.6-2	0	Reactor Coolant Average Temperature and DNBR Transients for Inadvertent Opening of a Pressurizer Safety Valve	
15.6-3A	0	SGTR Margin to Overfill Pressurizer Pressure	
15.6-3B	0	SGTR Margin to Overfill Ruptured Loop RCS Temperature	
15.6-3C	0	SGTR Margin to Overfill Intact Loop RCS Temperature	
15.6-3D	0	SGTR Margin to Overfill Steam Generator Pressure	
15.6-3E	0	Deleted	
15.6-3F	0	SGTR Margin to Overfill Pressurizer Level	
15.6-3G	0	SGTR Margin to Overfill Steam Generator Steam Releases	
15.6-3H	0	Deleted	
15.6-3I	0	SGTR Margin to Overfill Break Flow	
15.6-3J	0	SGTR Margin to Overfill Ruptured Steam Generator Water Volume	
15.6-3K	0	SGTR Thermal Hydraulic Pressurizer Pressure	
15.6-3L	0	SGTR Thermal Hydraulic Ruptured Loop RCS Temperature	
15.6-3M	0	SGTR Thermal Hydraulic Intact Loop RCS Temperature	
15.6-3N	0	SGTR Thermal Hydraulic Steam Generator Pressure	
15.6-3O	0	Deleted	
15.6-3P	0	SGTR Thermal Hydraulic Pressurizer Level	
15.6-3Q	0	SGTR Thermal Hydraulic Steam Generator Steam Releases	
15.6-3R	0	Deleted	
15.6-3S	0	SGTR Thermal Hydraulic Break Flow	
15.6-3T	0	SGTR Thermal Hydraulic Ruptured Steam Generator Water Volume	
15.6-3U	0	SGTR Thermal Hydraulic Break Flow Flashing Fraction	
15.6-3V	0	SGTR Thermal Hydraulic Total Flashed Break Flow	
15.6-3W	0	SGTR Thermal Hydraulic Ruptured Steam Generator Fluid Mass	

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*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

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15.6-3X	0	SGTR Thermal Hydraulic Intact Steam Generator Fluid Mass	
15.6-4	0	DELETED	
15.6-5	0	Code Scaling, Applicability, and Uncertainty (CSAU) Evaluation Methodology	
15.6-6	0	Code Interface Description for Small Break Model	
15.6-7	0	HOTSPOT, Cladding Temperature Transient and WC/T PCT for the Limiting PCT Case	
15.6-8	0	PCT Elevation for the Hot Rod for the Limiting PCT Case	
15.6-9		Break Flow for the Limiting PCT Case	
15.6-10	0	Void Fraction in the Reactor Coolant Pumps for the Limiting PCT Case	
15.6-11	0	Vapor Flow in the Hot Assembly Channel for the Limiting PCT Case	
15.6-12	0	Pressurizer Pressure for the Limiting PCT Case	
15.6-12a	0	DELETED	
15.6-12b	0	DELETED	
15.6-12c	0	DELETED	
15.6-12d	0	DELETED	
15.6-12e	0	DELETED	
15.6-12f	0	DELETED	
15.6-12g	0	DELETED	
15.6-12h	0	DELETED	
15.6-12i	0	DELETED	
15.6-13	0	Lower Plenum Collapsed Liquid Level for the Limiting PCT Case	
15.6-14	0	Accumulator Injection Flow for the Limiting PCT Case	
15.6-15	0	Safety Injection Flow (HHSI) for the Limiting PCT Case	
15.6-16	0	Safety Injection Flow (IHSI + RHR) for the Limiting PCT Case	
15.6-17	0	Downcomer Collapsed Liquid Level for the Limiting PCT Case	
15.6-18	0	Collapsed Liquid Level in Core Average Channel for the Limiting PCT Case	
15.6-19	0	Vessel liquid mass for the Limiting PCT Case	
15.6-20	0	Deleted	
15.6-21	0	Analysis vs. Calculated Containment Back Pressure	
15.6-22	0	Wolf Creek BELOCA Analysis Axial Power Shape Operating Space Envelope	
15.6-23	0	DELETED	
15.6-24a	0	DELETED	
15.6-24b	0	DELETED	
15.6-25	0	DELETED	
15.6-26	0	DELETED	
15.6-27	0	DELETED	

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*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

Figure #	Sheet	Title	Drawing #*
15.6-28	0	DELETED	
15.6-29	0	DELETED	
15.6-30	0	DELETED	
15.6-31	0	Reactor Coolant System Depressurization Transient (3 inch break)	
15.6-32	0	Core Mixture Level (3 inch break)	
15.6-33	0	Clad Temperature Transient Hot Spot (3 inch break)	
15.6-34	0	Steam Flow (3 inch break)	
15.6-35	0	Rod Film Heat Transfer Coefficient (3 inch break)	
15.6-36	0	Hot Spot Fluid Temperature (3 inch break)	
15.6-37	0	Reactor Coolant System Depressurization Transient (2 inch break)	
15.6-38	0	Reactor Coolant System Depressurization Transient (4 inch break)	
15.6-39	0	Reactor Coolant System Depressurization Transient (6 inch break)	
15.6-40	0	Core Mixture Level (2 inch break)	
15.6-41	0	Core Mixture Level (4 inch break)	
15.6-42	0	Core Mixture Level (6 inch break)	
15.6-43	0	Clad Temperature Transient Hot Rod (2 inch break)	
15.6-44	0	Clad Temperature Transient Hot Rod (4 inch break)	
15.6-45	0	Clad Temperature Transient Hot Rod (6 inch break)	
15.6-46	0	Safety Injection Flowrate	
15.6-47	0	Small Break Power Distribution	
15.6-48	0	Core Power(including Residual Fission) After Reactor Trip (Applies to All Small Breaks)	
15.6-49	0	Operator Ingress/Egress Path to Control Room Following a LOCA	
15.A-1	0	Release Pathways	

15.0 ACCIDENT ANALYSIS

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

Since 1970, the American Nuclear Society (ANS) classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

a. Condition I*: Normal operation and operational transients
b. Condition II*: Faults of moderate frequency
c. Condition III*: Infrequent faults
d. Condition IV*: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip and engineered safeguards system functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle, (i.e. only seismic Category I, Class IE, and IEEE qualified equipment, instrumentation, and components are used in the ultimate mitigation of the consequences of faulted conditions for Condition II, III and IV events). Step-by-step sequence-of-events diagrams are provided for each transient in Figures 15.0-8 through 15.0-31. Figure 15.0-7 provides the legend used in these diagrams. The accident analysis radiological consequences evaluation models and parameters are discussed in Appendix 15A.

15.0.1.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I events occur

* For the definition of Conditions I, II, III, and IV events, refer to ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," Section 5, 1973.

frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

A typical list of Condition I events is given below:

a. Steady state and shutdown operations

1. Power operation

2. Startup

3. Hot standby

4. Hot shutdown

5. Cold shutdown

6. Refueling

b. Operation with permissible deviations

Various deviations from normal operation but specifically allowed by the Technical Specifications which may occur during continued operation are considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service (such as an inoperable RCCA)

2. Leakage from fuel with limited clad defects

3. Excessive radioactivity in the reactor coolant

(a) Fission products
(b) Corrosion products
(c) Tritium

4. Operation with steam generator leaks

5. Testing

c. Operational transients

1. Plant heatup and cooldown

2. Step load changes (up to ± 10 percent)

3. Ramp load changes (up to 5 percent/minute)

4. Load rejection up to and including design full load rejection transient

15.0.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. These faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system or secondary system overpressurization. The following faults are included in this category:

a. Feedwater system malfunctions that result in a decrease in feedwater temperature (Section 15.1.1).

b. Feedwater system malfunctions that result in an increase in feedwater flow (Section 15.1.2).

c. Excessive increase in secondary steam flow (Section 15.1.3).

d. Inadvertent opening of a steam generator atmospheric relief or safety valve (Section 15.1.4).

e. Loss of external electrical load (Section 15.2.2).

f. Turbine trip (Section 15.2.3).

g. Inadvertent closure of main steam isolation valves (Section 15.2.4).

h. Loss of condenser vacuum and other events resulting in turbine trip (Section 15.2.5).

i. Loss of nonemergency ac power to the station auxiliaries (Section 15.2.6).

j. Loss of normal feedwater flow (Section 15.2.7).

k. Partial loss of forced reactor coolant flow (Section 15.3.1).

l. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Section 15.4.1).

m. Uncontrolled rod cluster control assembly bank withdrawal at power (Section 15.4.2).

n. Rod cluster control assembly misalignment (dropped RCCA, dropped RCCA bank, or statically misaligned RCCA) (Section 15.4.3).

o. Startup of an inactive reactor coolant pump at an incorrect temperature (Section 15.4.4).

p. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Section 15.4.6).

q. Inadvertent operation of the emergency core cooling system during power operation (Section 15.5.1).

r. Chemical and volume control system malfunction that increases reactor coolant inventory (Section 15.5.2).

s. Inadvertent opening of a pressurizer safety or relief valve (Section 15.6.1).

t. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate the containment (Section 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

Condition III events are faults which may occur very infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, per the guidelines of 10 CFR 50.67. A Condition III event will not, by itself, generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

a. Steam system piping failure (minor) (Sections 15.1.5 and 15.1.6).

b. Complete loss of forced reactor coolant flow (Section 15.3.2).

c. Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (Section 15.4.3).

d. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

e. Loss-of-coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (Section 15.6.5).

f. Radioactive gas waste system leak or failure (Section 15.7.1).

g. Radioactive liquid waste system leak or failure (Section 15.7.2).

h. Postulated radioactive releases due to liquid tank failures (Section 15.7.3).

i. Spent fuel cask drop accidents (Section 15.7.5).

15.0.1.4 Condition IV - Limiting Faults

Condition IV events are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV events are not to cause a fission product release to the environment resulting in doses in excess of guideline values of 10 CFR 50.67. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults have been classified in this category:

a. Steam system piping failure (major) (Sections 15.1.5 and 15.1.6).

b. Feedwater system pipe break (Section 15.2.8).

c. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3).

d. Reactor coolant pump shaft break (Section 15.3.4).

e. Spectrum of rod cluster control assembly ejection accidents (Section 15.4.8).

f. Steam generator tube rupture (Section 15.6.3).

g. Loss-of-coolant accidents, resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (Section 15.6.5).

h. Design basis fuel handling accidents (Section 15.7.4).

15.0.2 OPTIMIZATION OF CONTROL SYSTEMS

A control system setpoint study is performed prior to operation to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control and protection system which will automatically maintain prescribed conditions in the plant even under the most adverse set of anticipated plant operating transients with respect to both system stability and equipment performance. For each mode of plant operation, a group of optimum controller setpoints is determined. A consistent set of control system parameters is derived from this study that satisfies plant operational requirements throughout the core life and for various levels of power operation.

The study is comprised of an analysis of the following control systems: rod cluster control assembly, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed for the transients analyzed. Two ratings are given:

a. The guaranteed NSSS thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

b. The rated reactor core thermal power output is 3565 MWt.

All non-LOCA analyses have been performed assuming a conservatively high NSSS thermal power output of 3651 MWt which is intended to bound a core thermal power output of 3565 MWt plus a 2% power uncertainty, or an increased core thermal power and a corresponding power uncertainty that, when combined, do not exceed 3651 MWt minus the assumed contribution of the reactor coolant pumps.

Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The core thermal power used for each transient is given in Table 15.0-2.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most transients that are DNB-limited, nominal values of initial conditions are applied, and the steady-state error allowances on power, temperature, and pressure are statistically combined with other uncertainties to determine the DNBR limit as described in Reference 14. This analytical method is known as the Revised Thermal Design Procedure (RTDP). The reactor coolant Minimum Measured Flow is used in all RTDP transient analyses.

For transients that are not DNB-limited or for which the RTDP is not employed, the following steady-state errors are accounted for:

a.	Core power	Up to a +2 percent allowance for calorimetric error
b.	Average reactor coolant system temperature	+6.5 / -4.0°F allowance for controller deadband and measurement errors (Note that a positive value corresponds to an actual value that is greater than that indicated by the instrumentation. Conversely, a negative value corresponds to an actual value that is lower than that indicated by the instrumentation.)
c.	Pressurizer pressure	+50 / -30 psi allowance for steady state fluctuations and measurement errors (Note that a positive value corresponds to an actual value that is greater than that indicated by the instrumentation. Conversely, a negative value corresponds to an actual value that is lower than that indicated by the instrumentation.)

Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor (F_{DH}) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.0.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in F_{DH} is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a F_{DH} consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is the 1.70 chopped cosine, as discussed in Chapter 4 for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at non-full power or asymmetric RCCA conditions.

The radial and axial power distributions described above are input to the VIPRE code, as described in Chapter 4.

For transients which may be overpower limited, the total peaking factor (F_q) is of importance. All transients that may be over-power limited are assumed to begin with plant and reactor operating conditions consistent with the restrictions defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant, for example the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow, that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Chapter 4. For overpower transients that are fast with respect to the fuel rod thermal time constant, for example, the uncontrolled RCCA bank withdrawal from subcritical or low power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds, a detailed fuel transient heat transfer calculation must be performed.

15.0.4 REACTIVITY COEFFICIENTS ASSUMED IN THE ACCIDENT ANALYSES

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in detail in Section 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the reactor coolant system, do not depend on reactivity feedback effects. The values used are given in Table 15.0-2. Reference is made in that table to Figure 15.0-2, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life. For example, in a load increase transient it is conservative to use a small Doppler defect and a small moderator coefficient.

15.0.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

All accident analysis results contained herein are applicable to both Ag-In-Cd and hafnium control rods.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. The RCCA position versus time assumed in accident analyses is shown in Figure 15.0-3. The RCCA insertion time to dashpot entry is taken as 2.7 seconds unless otherwise noted in the discussion. The use of such a long insertion time provides conservative results for all accidents and is intended to be applicable to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the plant Technical Specifications.

Figure 15.0-4 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to all point kinetics

core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0-4 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity would have been inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0-5. The curve shown in this figure was obtained from Figures 15.0-3 and 15.0-4. A total negative reactivity insertion following a trip of 4-percent Δk is assumed in the transient analyses, except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available, as shown in Table 4.3-3. For Figures 15.0-3 and 15.0-4, the RCCA drop time is normalized to 2.7 seconds, unless otherwise noted for a particular event, in order to provide a bounding analysis for all RCCAs to be used in the WCGS core.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0-5) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code, and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0-3 is used as code input.

15.0.6 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and

begin to fall. The safety analysis trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0-1. Figure 15.0-1 shows the core thermal limits over the range of allowable operating pressures assuming a loop Minimum Measured Flow (MMF) of 92,750 gpm. The DNB limit lines correspond to a safety analysis limit DNBR value as determined by the WRB-2 CHF Correlation (see Section 4.4.1.1)(Reference 16). The Overtemperature and Overpower Delta T trip functions in conjunction with the steam generator safety valve lift settings are designed to protect the core thermal limits.

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7.0. Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The difference between the safety analysis trip setpoint assumed for the analysis and the nominal trip setpoint represents an allowance for instrumentation channel error. Nominal trip setpoints are specified in the plant Technical Specifications Bases. During plant startup tests, it was demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the plant Technical Specifications.

15.0.7 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.

The calorimetric error is the error assumed in the determination of core thermal power, as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

Two options are available for indication of secondary power. These are (1) a feedwater calorimetric or (2) a steam flow calorimetric. The feedwater calorimetric uses feedwater flow (feedwater venturi differential pressure), feedwater inlet temperature to the steam generators, and steam pressure (from which feedwater pressure is inferred). The steam flow calorimetric uses the same inputs but uses the pressure drop across the steam generator orifice instead of the feedwater flow. This pressure drop value is normalized to a precision feedwater flow measurement at the beginning of the cycle. High accuracy instruments provided measurements with accuracy tolerances much tighter than those which are required to normally control feedwater flow.

15.0.8 MAJOR PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR MITIGATION OF TRANSIENT AND ACCIDENT CONDIITONS

The plant is designed to afford protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and equipment listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.0. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53, in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are assumed to be energized during some of the Chapter 15.0 events.

15.0.9 FISSION PRODUCT INVENTORIES

15.0.9.1 Activities in the Core

The calculation of the core iodine fission product inventory is based on a core power level of 3565 MWt plus 2% uncertainty. The fission product inventories are calculated by the industry standard ORIGEN-S code. These inventories are given in Table 15A-3. The isotopes included in Table 15A-3 (Appendix 15A) are the dose-significant isotopes from the radionuclide groups identified in Regulatory Guide 1.183 that should be considered in design basis analyses.

15.0.9.2 Activities in the Fuel Pellet Clad Gap

The fuel-clad gap activities are determined, using the model given in Regulatory Guide 1.183, except for fuel handling accident, which used the more conservative gap fractions listed in Regulatory Guide 1.25 and modified by NUREG/CR-5009 (see Section 15.7.4). In accordance with Regulatory Guide 1.183, the amount of activity accumulated in the fuel clad gap is assumed to be 8 percent for I-131, 10 percent for Kr-85, 12 percent for alkali metals, and 5 percent for other noble gases and iodines, with certain exceptions. For the rod ejection accident, it is assumed to be 10% for iodines and noble gases and 12% for alkali metals. For LOCA, it is assumed to be 5% for these three groups of radionuclides.

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA per the requirements of Appendix K of 10 CFR 50.46, as described in Reference 4 for NOTRUMP SBLOCA. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For the ASTRUM LBLOCA from Section 15.6.5.3.1, the decay heat is calculated using the ANSI/ANS 5.1-1979 Standard model as noted in Table 1-10 of Reference 18. As indicated in the text for the individual impacted accident analyses, the core residual heat generation for the non-LOCA transients is calculated based on the 1979 version of ANSI/ANS-5.1 (Reference 19). ANSI/ANS-5.1-1979 is a conservative representation of the decay heat release rates.

15.0.10.2 Distribution of Decay Heat Following Loss-of-Coolant Accident

Per Section 3.3.1.1 of Reference 4, during a small break LOCA, the core is rapidly shut down by void formation or RCCA insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4 percent, which represents the fraction of heat generated within the clad and pellet, drops to 95 percent for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

The ASTRUM LBLOCA analysis documented in Section 15.6.5.3.1 utilizes a condition and time-dependent model to determine the distribution of decay heat following the loss-of-coolant accident (see Section 8 of Reference 18).

15.0.11 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to

simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (see Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0-2.

15.0.11.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad, using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

a. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

c. The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 5.

15.0.11.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program, which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR, based on the input from the core limits illustrated on Figure 15.0-1. The core limits represent the minimum value of DNBR, as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 6.

15.0.11.3 PARAGON/NEXUS

PARAGON is based on collision probability - interface current cell coupling methods. PARAGON provides flexibility in modeling including exact cell geometry representation, multiple rings and regions within the fuel pin and the moderator cell geometry, and variable cell pitch. The solution method permits flexibility in choosing the quality of the calculation through both increasing the number of regions modeled within the cell and the number of angular current directions tracked at the cell interfaces. The PARAGON cross section library is a 70-group library that is based on the ENDF/B basic nuclear data. The PARAGON qualification library has been improved through the addition of more explicit fission products and fission product chains; however it is designed to employ any number of energy groups. The new NEXUS cross-section generation system uses PARAGON as the lattice code.

PARAGON and NEXUS are further described in References 7 and 11.

15.0.11.4 ANC

ANC is a multidimensional nodal analysis program used to predict nuclear reactor core reactivity and assembly and rod distributions for normal and off-normal conditions. The code allows for the treatment of enthalpy, xenon, and Doppler feedback. A high degree of automation has been incorporated into the code to address fuel depletion, reactivity coefficients, control rod worths for nonuniform inlet temperature distribution.

ANC is further described in Reference 8.

15.0.11.5 TWINKLE

The TWINKLE program is a multidimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups, and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration,

control rod motion, and others. Various edits are provided, e.g., channel-wise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 9.

15.0.11.6 APOLLO

APOLLO is a two-group, one dimensional neutron diffusion code designed to characterize the axial behavior of the core during plant operation in situations where three dimensional analyses would be time consuming and cumbersome. Pertinent uses include evaluation of limiting power shapes and scoping the adequacy of reactor protection systems in order to maintain margin to overpower and DNB limits.

APOLLO is further described in Reference 10.

15.0.11.7 RETRAN

RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

RETRAN is further described in References 12 and 17.

15.0.11.8 VIPRE

The core thermal-hydraulic analysis code VIPRE is an open channel code designed to evaluate DNBR and coolant state for steady state operations and transients using subchannel analysis.

VIPRE is further described in References 13 and 15 and Section 4.4.4.5.

15.0.12 LIMITING SINGLE FAILURES

The most limiting single failure as described in Section 3.1 of safety-related equipment, where one exists, is identified in each analysis description, and the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 OPERATOR ACTIONS

For most of the events analyzed in Chapter 15.0 the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will, in fact, be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time at which these actions would occur, will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator safety valves may be used, both of which are safety grade systems. Although the auxiliary feed system may be started manually, it will be automatically actuated, if needed, by one of the signals shown on Figure 7.2-1, such as low-low steam generator water level. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for xenon decay and maintain shutdown margin.

Where a stabilized condition is reached automatically following a reactor trip and only actions typical of normal operation are required, this has been stated in the text of the Chapter 15.0 events. For several events involving breaks in the reactor coolant system or secondary system piping, additional requirements for operator action are identified.

Following the postulated MSLB, a steamline isolation signal will be generated almost immediately, causing the steamline isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves, or if one valve fails to close, the break will be isolated from three steam generators while the affected steam generator will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here, since the break followed by isolation of all steam generators will terminate the transient.

Steam pressure from the steam generators is relieved by the turbine bypass system, secondary system atmospheric relief valves, or secondary system safety valves. The operator is instructed to terminate auxiliary feedwater flow to the affected steam generator, as soon as he determines which steam generator is affected. As soon as an indicated water level returns to the pressurizer and pressure is no longer decreasing, the operator is instructed to terminate the charging pump flow to limit system repressurization.

For long-term cooling following a steamline break, the operator is instructed to use the intact steam generators for the purpose of removing decay heat and plant stored energy. This is done by feeding the steam generators with auxiliary feed-water to maintain an indicated water level in the steam generator narrow-range span.

A safety injection signal (generated a few seconds after the break on low steam line pressure) will cause main feedwater isolation to occur. A steam line break protection signal (Section 7.3.8.3.2) will also cause main feedwater isolation to occur. The only source of water available to the affected generator is then the auxiliary feedwater system. Following steamline isolation, steam pressure in the steamline with the affected steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining three steam lines. The indication of the different steam pressures will be available to the operator, within a few seconds of steamline isolation. This will provide the information necessary to identify the affected steam generator so that auxiliary feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the auxiliary feedwater pumps and for the control valves associated with the auxiliary feedwater system. The means for detecting the affected steam generator and isolating auxiliary feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long term (following the initial cooldown), using the remaining steam generators, requires only the auxiliary feedwater system as a water source and the secondary system safety valves to relieve steam.

The operator has available, in the control room, an indication of pressurizer water level from the reactor protection system instrumentation. Indicated water level eventually returns to the pressurizer following the steamline break. To maintain the indicated water level, the operator can start and stop the charging pumps as necessary. The pressurizer level instrumentation and manual controls for the operation of the charging pumps meet the required standards for safety systems.

As indicated, the information for terminating auxiliary feedwater to the affected steam generator is available to the operator within 1 minute of the break, while the information required for terminating the charging/SI flow becomes available a short time later. The requirements to terminate auxiliary feedwater flow to the affected steam generator can be met by switch actions by the operators, i.e., closing auxiliary feed discharge valve. Thus, the required actions to limit the cooldown can be recognized, planned, and performed within 10 minutes. After it is determined that the pressurizer level is restored and SI flow is no longer required, normal charging flow is established and the SI flow is eventually terminated to prevent repressurization of the RCS. For decay heat removal and plant cooldown, the operator has a considerably longer time period in which to respond because of the large initial cooldown associated with a steamline break transient.

For a feedwater line break, auxiliary feedwater flow is initiated automatically, as is safety injection. As in the steamline break, the operator terminates auxiliary feedwater flow to the affected steam generator as soon as he determines which unit is affected, using safety grade equipment. Where possible, the operator should also increase auxiliary feedwater flow to the intact steam generators in order to shorten the time until primary temperatures begin to decrease. The analysis presented in Section 15.2.8 does not assume these actions occur.

As soon as primary temperature begins to decrease, the operator can use the steam dump system or the steam generator atmospheric relief valves to begin a controlled cooldown. In addition, if the U-tubes of the intact steam generators are covered with water as indicated by post accident monitoring system (PAMS) steam generator water level instrumentation (see Chapter 7.0), the operator can modulate the high-head charging pumps, so that the primary pressure decreases while ensuring that voiding does not occur within the RCS. The primary pressure-temperature relationship can be monitored by the operator via the PAMS widerange RCS pressure and temperature instruments.

Using the above-mentioned PAMS indications, the operator can maintain the plant in a hot shutdown condition for an extended period of time, or can proceed to a cold shutdown condition as desired.

The safety-related indicators for steamline pressure and pressurizer water level noted above are further discussed in Section 7.5.

Tables 15.0-8 and 15.0-9 list the short term operator actions required to bring the plant to a stable condition for the LOCA and steam generator tube rupture (SGTR). Further information (including alarms which alert the operator) on operator action for these two accidents are given in Section 6.3.2.8 for the LOCA and Section 15.6.3 for the SGTR.

Process information available to the operator in the control room following either of these accidents (LOCA or SGTR) is given in Section 7.5.

Operator actions aimed at preventing the loss of available shutdown margin following a chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (inadvertent boron dilution) event are discussed in Section 15.4.6.

For the inadvertent operation of the ECCS event, timely operator actions required to be completed to keep the pressurizer from filling water-solid are: (1) using at least one secondary system atmospheric relief valve, control the temperature of the RCS cold legs to 557°F, (2) terminate safety injection flow to the cold legs, and (3) re-establish letdown flow. Credited times for these operator actions are provided in Section 15.5.1.

For the CVCS malfunction that increases reactor coolant inventory event, the analysis described in Section 15.5.2 demonstrates that plant operators have more than 8 minutes to respond to the event and terminate the reactor coolant inventory addition.

Instrumentation and controls provided to allow the operator to complete required manual actions are classified as Class IE. Electrical components are also classified as Class IE. Mechanical components are classified as Safety Class 1, 2, or 3.

Safety systems required for accident mitigation are designed to function after the occurrence of the worst postulated single failure. There are no adverse impacts as a result of these actions.

15.0.14 REFERENCES

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WOLF CREEK

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

NSSS thermal power output, MWt	3,579*
Nominal thermal power generated by the reactor coolant pumps, MWt	14
Rated reactor core thermal power output, MWt	3,565*

* All non-LOCA analyses have been performed assuming a conservatively high NSSS thermal power output of 3651 MWt which is intended to bound a core thermal power output of 3565 MWt plus a 2% power uncertainty, or an increased core thermal power and a corresponding power uncertainty that, when combined, do not exceed 3651 MWt minus the assumed contribution of the reactor coolant pumps.

TABLE 15.0-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

	Faults	Computer Codes Utilized	Reactivity Coefficients (a) Assumed: Moderator Density /Temp Coeff	Reactivity Coefficients (a) Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (b) (MWt)
15.1	Increase in heat removal by the secondary system				
	Feedwater system malfunctions that result in a decrease in feedwater temperature	RETRAN	0.47 Δk/gm/cc	Lower (a)	3,651
	Feedwater system malfunctions that result in an increase in feedwater flow	RETRAN, VIPRE	0.47 Δk/gm/cc	Lower (a)	0 and 3,651
	Excessive increase in secondary steam flow	RETRAN	0 pcm/°F and 0.47 Δk/gm/cc	Lower and upper (a)	3,651
	Inadvertent opening of a steam generator atmospheric relief or safety valve	RETRAN, VIPRE	Function of moderator density (see Section 15.1.4, Figure 15.1-11)	See Section 15.1.5	0 (Subcritical)
	Steam system piping failure at zero power	RETRAN, VIPRE	Function of moderator density (see Section 15.1.5, Figure 15.1-11)	See Section 15.1.5	0 (Subcritical)
	Steam system piping failure at full power	RETRAN, ANC, VIPRE	0.47 Δk/gm/cc	Lower (a)	3,651
15.2	Decrease in heat removal by the secondary system				
	Loss of external electrical load and/or turbine trip	RETRAN	0 pcm/°F	Lower (a)	3,651

TABLE 15.0-2 (Sheet 2)

	Loss of nonemergency ac power to the station auxiliaries	RETRAN	0 pcm/°F	Upper (a)	3,651
	Loss of normal feedwater flow	RETRAN	0 pcm/°F	Upper (a)	3,651
	Feedwater system pipe break	RETRAN	0 pcm/°F and 0.47 Δk/gm/cc	Lower and upper (a)	3,651
15.3	Decrease in reactor coolant system flow rate				
	Partial and complete loss of forced reactor coolant flow	RETRAN, VIPRE	0 pcm/°F	Upper (a)	3,651
	Reactor coolant pump shaft seizure (locked rotor)	RETRAN, VIPRE	0 pcm/°F	Upper (a)	3,651
15.4	Reactivity and power distribution anomalies				
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE	+6.0 pcm/°F	Least negative Doppler defect 1007 pcm	0
	Uncontrolled rod cluster control assembly bank withdrawal at power	RETRAN, VIPRE, LOFTRAN	0 pcm/°F (d) and 0.47 Δk/gm/cc	Lower and upper (a)	Varies from 365 to 3,651
	Rod cluster control assembly misalignment	PARAGON/NEXUS, ANC	NA	NA	3,561
	Startup of an inactive reactor coolant loop at an incorrect temperature (e)	LOFTRAN, VIPRE	0.43 Δk/gm/cc	Lower (a)	2,495
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	NA	0 and 3,637

TABLE 15.0-2 (Sheet 3)

	Inadvertent loading and operation of a fuel asembly in an improper position	ANC, PARGON/NEXUS	NA	NA	3,651
	Spectrum of rod cluster control assembly ejection accidents	TWINKLE, FACTRAN, ANC	Refer to Section 15.4.8 (c)	Least negative Doppler defect 1007 pcm (BOL) 925 pcm (EOL)	0 and 3,637 (core power)
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the ECCS during power operation	RETRAN	0.47 Δk/gm/cc	Upper (a)	3,651
	Chemical and volume control system malfunction that increases reactor coolant inventory	RETRAN	0.47 Δk/gm/cc and 0 pcm/°F	Upper and Lower (a)	3,651
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer safety or relief valve	RETRAN	0	Lower (a)	3,651
	Steam generator tube rupture	RETRAN	NA	NA	3,651
	Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	SB LOCA: NOTRUMP, LOCTA-IV. LB LOCA: WCOBRA/TRAC, COCO, HOTSPOT	See Section 15.6.5 references (c)	See Section 15.6.5, references	3,565

(a) See Figure 15.0-2.
(b) All non-LOCA analyses have been performed assuming a conservatively high NSSS thermal power output of 3651 MWt which is intended to bound a core thermal power output of 3565 MWt plus a 2% power uncertainty, or an increased core thermal power and a corresponding power uncertainty that, when combined, do not exceed 3651 MWt minus the assumed contribution of the reactor coolant pumps.
(c) The moderator density coefficients for these accidents were translated into moderator temperature coefficients.
(d) The MTC value presented corresponds to the full power cases. An MTC of +6 pcm/°F is modeled at part-power conditions.
(e) The values presented correspond to historical analyses, refer to Section 15.4.4 for additional information.

TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN THE ACCIDENT ANALYSES *

Thermal output of NSSS, MWt	See Table 15.0-2			
Core inlet temperature, °F	536.5	555.2	536.5	555.2
Vessel average temperature, °F	570.7	588.4	570.7	588.4
Reactor coolant system pressure, psia	2250	2250	2250	2250
Reactor coolant flow per loop, gpm				
Thermal Design Flow	90,300	90,300	90,300	90,300
Minimum Measured Flow	92,750	92,750	92,750	92,750
Steam flow from NSSS, 10^6lb/hr				
At Minimum Feedwater Temperature	15.16	15.24	15.15	15.23
At Maximum Feedwater Temperature	16.21	16.30	16.20	16.29
Steam pressure at steam generator outlet, psia	810	954	793	934
Maximum steam moisture content, %	0.25	0.25	0.25	0.25
Assumed feedwater temperature at steam generator inlet, °F				
Minimum	400.0	400.0	400.0	400.0
Maximum	448.6	448.6	448.6	448.6
Average core heat flux, Btu/hr-ft^2**	202,326	202,326	202,326	202,326
Steam generator tube plugging, %	0	0	10	10

* Steady-state errors discussed in Section 15.0.3.2 are added to these values to obtain initial conditions for transient analyses, except where discussed otherwise.

** The above footnote does not apply to the average core heat flux since it is based on the power level that includes uncertainty.

TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (Seconds)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	84%	0.5
Power range neutron flux high positive rate	6.9%/sec	1.0
Overtemperature ΔT	Variable see Figure 15.0-1	6.25*
Overpower ΔT	Variable see Figure 15.0-1	6.25*
High pressurizer pressure	2,425 psia	1.0
Low pressurizer pressure	1924.7 psia	2.0
Low reactor coolant flow (from loop flow detectors)	86.3 % indicated loop flow	1.0
Undervoltage trip	10308 VAC	1.5
Turbine trip	Not applicable	2.0
Low-low steam generator level	0% of narrow range level span	2.0
High steam generator level, trip of the feedwater pumps** and closure of feedwater system valves, and turbine trip***	100% of narrow range level span	2.0

* Total time delay (including combined RTD/thermowell response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loop exceeds the trip setpoint until the rods are free to fall. This is modeled in the analysis as a 4.0-second first-order lag delay plus a 2.25-second pure delay. This is the limiting value used for both Overtemperature ΔT and Overpower ΔT, which the plant is required to meet. Some analyses used a conservative total time delay of 7.0 seconds (3.0-second pure delay); however, a time delay of 6 seconds is maintained in the field.

** Not part of primary success path for accident mitigation.

*** No direct credit is taken for the turbine trip function in the accident analyses.

TABLE 15.0-5

DETERMINATION OF MAXIMUM OVERPOWER TRIP POINT - POWER RANGE NEUTRON FLUX CHANNEL - BASED ON NOMINAL SETPOINT CONSIDERING INHERENT INSTRUMENTATION ERRORS

Nominal setpoint (% of rated power) 109

Calorimetric errors in the measurement of secondary system thermal power:

Variable	Accuracy of Measurement of Variable (% error)	Estimated Effect on Thermal Power Determination (% error)
Assumed calorimetric error (% of rated power for both feedwater calorimetric and steam flow calorimetric measurements)		2 (a)*
Axial power distribution effects on total ion chamber current		
Estimated error (% of rated power)	3	
Assumed error (% of rated power)		5 (b)*
Instrumentation channel drift and setpoint reproducibility		
Estimated error (% of rated power)	1	

TABLE 15.0-5 (Sheet 2)

Variable	Accuracy of Measurement of Variable (% error)	Estimated Effect on Thermal Power Determination (% error)
Assumed error (% of rated power)		2 (c)*
* Total assumed error in setpoint (% of rated power) (a) + (b) + (c)		±9
Maximum overpower trip point, assuming all individual errors are simultaneously in the most adverse direction (% of rated power)		118

TABLE 15.0-6

MAJOR PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR MITIGATION OF 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	-	-	-	-
	Feedwater system malfunctions that result in a decrease in feedwater temperature or an increase in feedwater flow	Source range high flux, intermediate range high flux, power range high flux, low-low steam generator level, manual, overpower ΔT, over-temperature ΔT, turbine trip	SI on low pressurizer pressure produces feedwater isolation or High-high SG level produces feedwater isolation and turbine trip	Feedwater isolation valves	-
	Excessive increase in secondary steam flow	Power range high flux, over-temperature ΔT, overpower ΔT, manual	-	Pressurizer safety valves, steam generator safety valves	-
	Inadvertent opening of a steam generator atmospheric relief or safety valve	Manual, SIS, power range high flux, overpower ΔT	Low pressurizer pressure, low compensated steam line pressure, high negative steam pressure rate	Feedwater isolation valves, steam line isolation valves	Auxiliary feedwater system, safety injection system
	Steam system piping failure	Manual, SIS, power range high flux, overpower ΔT	Low pressurizer pressure, low compensated steam line pressure, hi-1 containment pressure, hi-2 containment pressure, high negative steam pressure rate, SIS, high-high SG level, low-low SG level, 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 load/ turbine trip	Low-low SG level, turbine trip, high pressurizer pressure, overtemperature ΔT, manual, high pressurizer water level	Low-low SG level, manual	Pressurizer safety valves, steam generator safety valves	-

TABLE 15.0-6 (Sheet 2)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
	Loss of non-emergency ac power to the station auxiliaries	Manual, low-low SG level, low flow	Low-low SG level, loss of offsite power, manual	Steam generator safety valves, pressurizer safety valves	Auxiliary feedwater system
	Loss of normal feedwater flow	Low-low SG level, manual, overtemperature ΔT	Low-low SG level, loss of offsite power, manual	Steam generator safety valves, pressurizer safety valves	Auxiliary feedwater system
	Feedwater system pipe break	Low-low SG level, high pressurizer pressure, SIS, manual, overtemperature ΔT	Hi-1 containment pressure, low-low SG level, low compensated steam line pressure SIS, loss of offsite power, manual	Steam line isolation valves, feedline isolation, pressurizer safety valves, steam generator safety valves	Auxiliary feedwater system, safety injection system
15.3	Decrease in reactor coolant system flow rate	-	-	-	-
	Partial and, complete loss of forced reactor coolant flow	Low flow, undervoltage underfrequency, manual	-		-
	Reactor coolant pump shaft seizure (locked rotor)	Low flow, high pressurizer pressure, manual	-	Pressurizer safety valves, steam generator safety valves	-
15.4	Reactivity and power distribution anomalies	-	-	-	-
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power start-up condition	Source range high flux, intermediate range high flux, power range high flux (high setpoint), power range high flux (low setpoint), manual, power range high flux rate	-	-	-

TABLE 15.0-6 (Sheet 3)

	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 water level, overpower ΔT, power range high positive flux rate	-	Pressurizer safety valves,steam generator safety valves	-
	Rod cluster control assebmly mis-alignment	Power range negative flux rate, overtemperature ΔT, manual	-	-	-
	Startup of an inactive reac-tor coolant loop at an in-correct temperature	Manual, power range high flux above P8 setpoint with low flow	-	Low insertion limit annun-ciators for boration	-
	Chemical and volume control, system mal-function that results in a decrease in boron concentrattion in the reactor coolant	Power range high flux, over-temperature ΔT, manual, over-power ΔT high pressurizer pressure, high pressurizer water level	-	Pressurizer safety valves, steam generator safety valves rod insertion limit alarms, position step counter, RT alarm, audible count rate from BF3 detector source range high flux alarm	-
	Spectrum of rod cluster con-trol assembly (RCCA) ejection accidents	Source range high flux, intermediate range high flux, power range high flux, high positive flux rate, manual	Low pressurizer pressure, Hi-1 containment pressure, Hi-3 containment pressure, manual.	Pressurizer safety valves	ESFAS
15.5	Increase in reactor coolant inventory	-	-	-	-
	Inadvertent operation of the ECCS dur-ing power oper-tion	Low pressurizer pressure, manual, safety injection trip	SIS, low-low SG level, manual	Secondary system atmospheric relief valves, steam generator safety valves, CVCS (letdown)	Auxiliary feedwater system
	CVCS malfunction that increases reactor coolant inventory	High pressure level, manual	-	Secondary system atmospheric relief valves, steam generator safety valves, CVCS (letdown)	-

TABLE 15.0-6 (Sheet 4)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.6	Decrease in reactor coolant inventory	-	-	-	-
	Inadvertent opening of a pressurizer, safety or relief valve	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure, manual	-	Safety injection system
	Steam generator tube rupture	Low pressurizer pressure, SIS, manual, overtemperaturer ΔT	Low pressurizer pressure, manual	Essential service water system, component cooling water system, steam generator shell side fluid operating system, steam generator safety and/or atmospheric relief valves, steam line isolation valves, pressurizer power-operated relief valves	Emergency core cooling system, auxiliary feed-water system, emergency power system containment isolation
	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Pressurizer low pressure, SIS, manual	Hi-1 containment pressure, low pressurizer pressure Hi-3 containment pressure, manual AFAS on loss of offsite power	Essential service water system, component cooling water system, steam gen-erator safety and/or atmospheric relief valves	Emergency core cooling system, auxiliary feedwater system, containment heat removal system, emergency power system containment isolation

TABLE 15.0-7

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Worst Failure Assumed
Feedwater temperature reduction	One protection train
Excessive feedwater flow	One protection train
Excessive steam flow	(1)
Inadvertent secondary depressurization	One safety injection train
Steam system piping failure at zero power	One safety injection train
Steam system piping failure at full power	One protection train
Steam pressure regulator malfunction	(2)
Loss of external load	One protection train
Turbine trip	One protection train
Inadvertent closure of MSIV	One protection train
Loss of condenser vacuum	One protection train
Loss of ac power	One auxiliary feedwater pump
Loss of normal feedwater	One auxiliary feedwater pump
Feedwater system pipe break	One auxiliary feedwater pump
Partial loss of forced reactor coolant flow	One protection train
Complete loss of forced reactor coolant flow	One protection train
RCP locked rotor	One protection train
RCP shaft break	One protection train
RCCA bank withdrawal from subcritical	One protection train
RCCA bank withdrawal at power	One protection train
Dropped RCCA, dropped RCCA bank	One nuclear instrumentation system channel
Statically misaligned RCCA	(3)
Single RCCA withdrawal	One protection train
Inactive RC pump startup	One protection train
Flow controller malfunction	(2)
Uncontrolled boron dilution	Standby charging pump is operating (4)
Improper fuel loading	(3)
RCCA ejection	One protection train
Inadvertent ECCS operation at power	One secondary system atmospheric relief valve fails to open
CVCS malfunction that increases RCS inventory	(5)
BWR transients	(2)
Inadvertent RCS depressurization	One protection train
Failure of small lines carrying primary coolant outside containment	(3)
Steam Generator Tube Rupture	AFW flow control valve to ruptured SG fails open, ARV on the ruptured SG fails open
Fuel Handling Accident	(3)
BWR piping failures	(2)
Spectrum of LOCA	
Small breaks	One safety injection train
Large breaks	One RHR pump

TABLE 15.0-7 (Sheet 2)

NOTES:
(1) No protection action required
(2) Not applicable to WCGS
(3) No transient analysis involved
(4) Applies to power and startup operations only. A single failure in the VCT level controller which results in the continuation of the dilution flow, even after the "stop auto makeup" setpoint is exceeded, is assumed for Mode 3, 4 and 5.
(5) A reactor trip on high pressurizer level would be expected to occur for this event. However, the transient is initiated by the pressurizer level channel that is used for control purposes failing low and, as a worst single failure, a second pressurizer level channel is then assumed to fail in an as-is condition or low, defeating the two-out-of-three high pressurizer level trip. In addition, there are several alarms available to alert plant operators of this event. The analysis credits a high pressurizer level alarm, which is not expected to be the first alarm signal generated. Thus, the failure of an earlier alarm to alert the operators is also assumed.

TABLE 15.0-8

OPERATOR ACTIONS (1) REQUIRED FOR SMALL AND LARGE LOCAS

Time	Operator Action
Reactor trip signal is actuated (2)	None
Safety injection signal is actuated (2)	None
Prior to generation of RWST low level signal	Reset safeguards actuation signal. Check sump water level indicator.
Generation of RWST low level signal	Verify completion of automatic switchover to ensure components have been properly realigned. Perform the additional valve alignments required for switchover to recirculation (3).
6.5 - 7.5 hours	Perform operations necessary to switch to simultaneous hot and cold leg recirculation.
To final stabilized condition	Monitor system pressure and temperature. Control pressurizer water level with safety injection system.

(1) Actions associated with primary system protection.
(2) These times can be found in the sequence of events tables in Section 15.6.5.
(3) See Section 6.3.2.8 for the manual actions required for completion of switchover. Operator actions associated with containment protection is discussed in Section 6.2.

TABLE 15.0-9

SHORT TERM OPERATOR ACTIONS REQUIRED FOR
STEAM GENERATOR TUBE RUPTURE

Time	System	Operator Action
Reactor trip signal actuated	Reactor trip system	None
Safety injection signal actuated	Safety injection system	None
Post SI signal generation to plant stabilization	Safety injection system, auxiliary feedwater system, steam dump system	Perform operator actions necessary to isolate the ruptured steam generator. [Isolate steam flow from and feed flow to the ruptured steam generator]
		Reduce RCS temperature to increase RCS subcooling. Depressurize RCS to equilibrate RCS and ruptured steam generator pressure. Terminate SI, establish normal charging/letdown flow.
		Proceed with normal plant cooldown, while monitoring reactor coolant system pressure, temperature, and boron concentration.

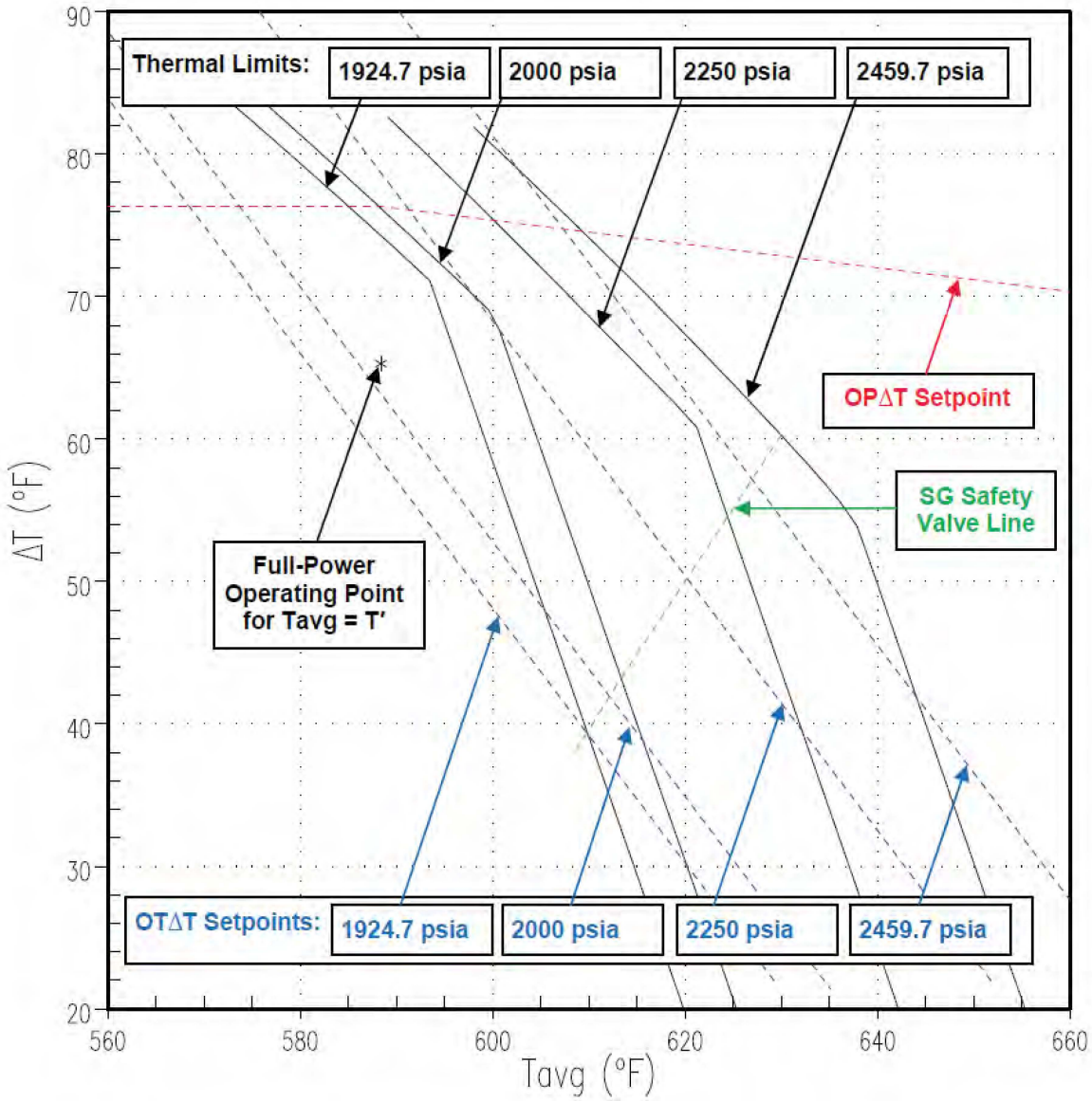

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FIGURE 15.0-1 ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER Δ T PROTECTION	

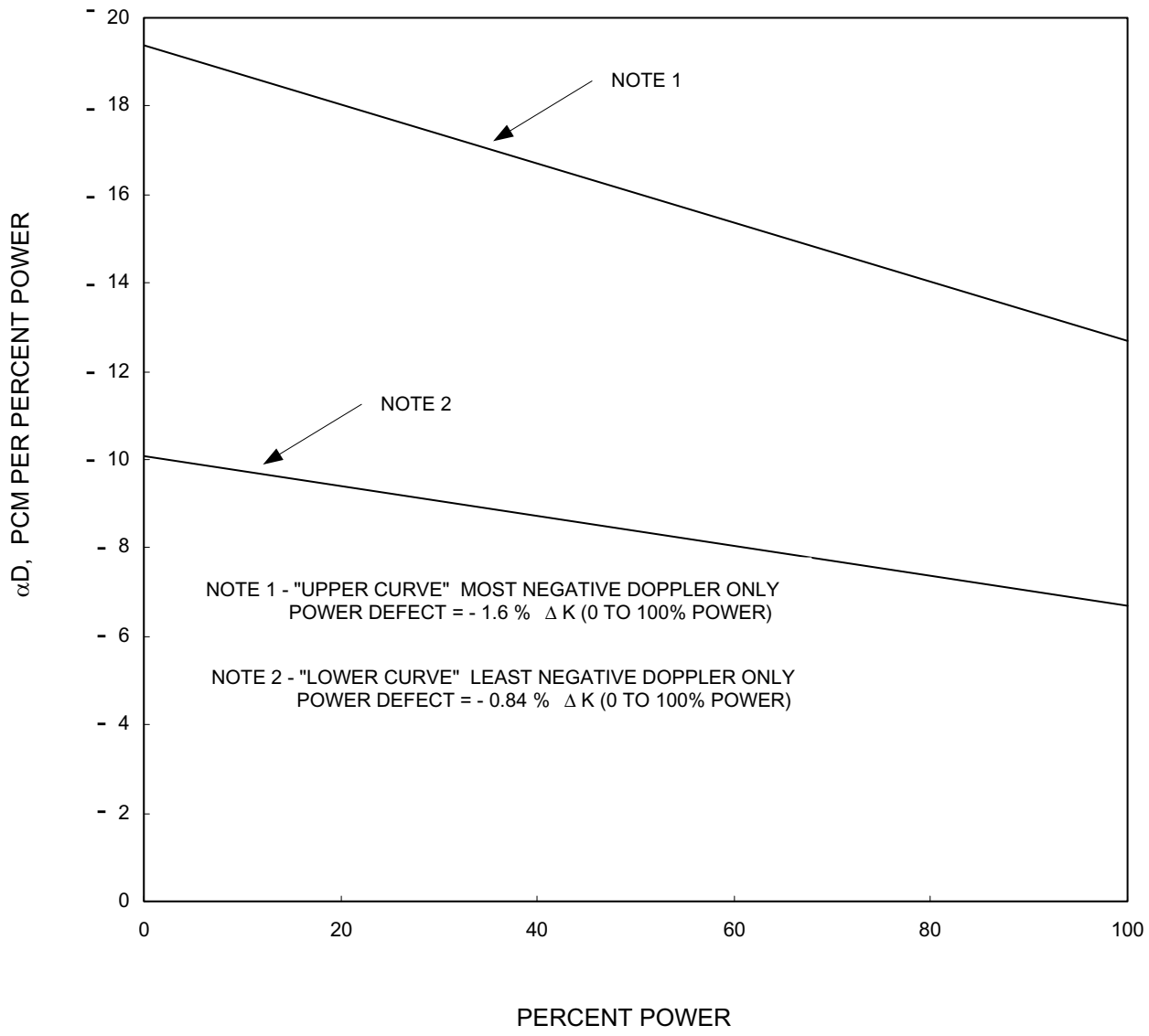

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FIGURE 15.0-2

DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS

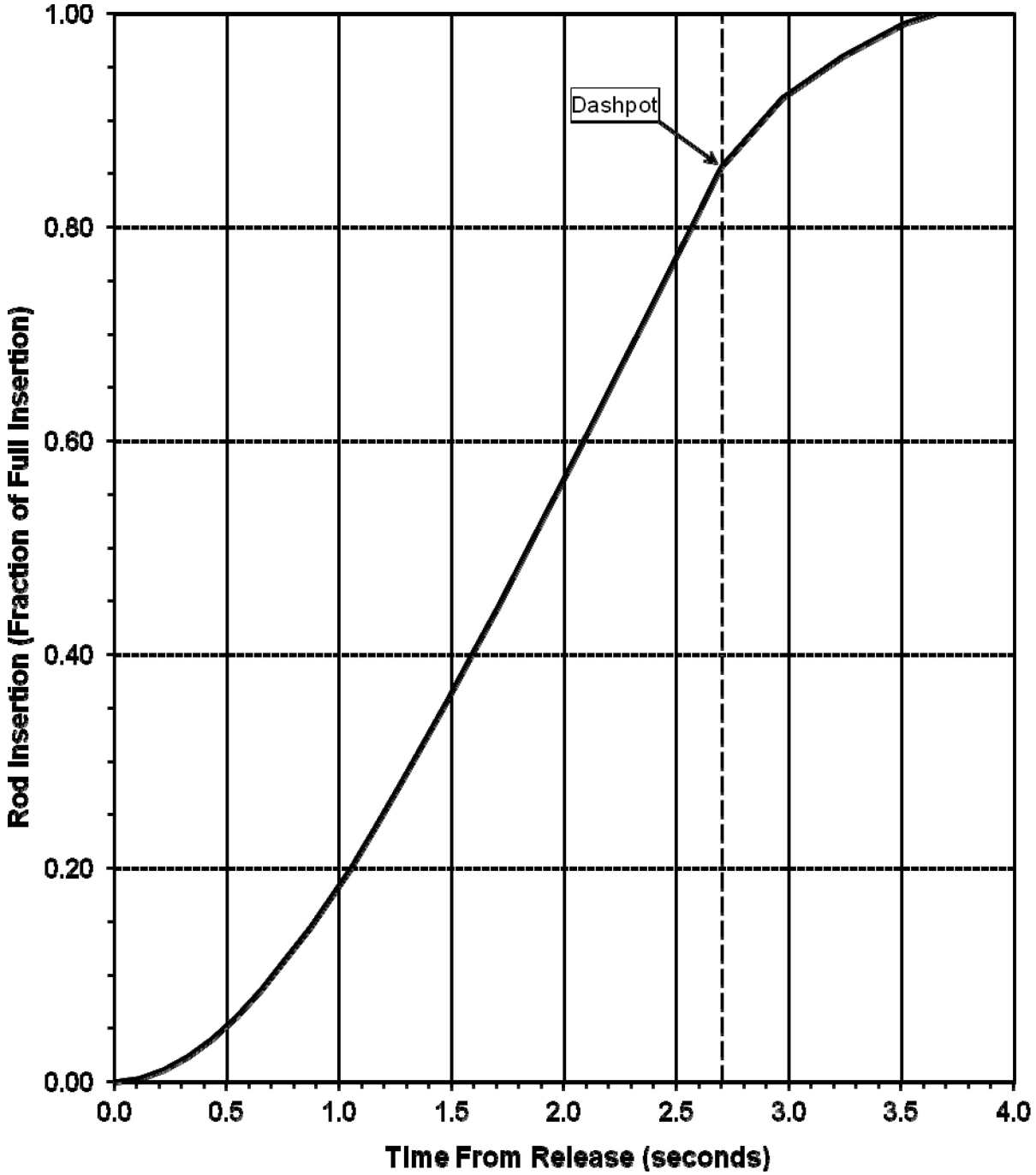

FIGURE 15.0-3
FRACTIONAL ROD INSERTION VERSUS TIME FROM RELEASE

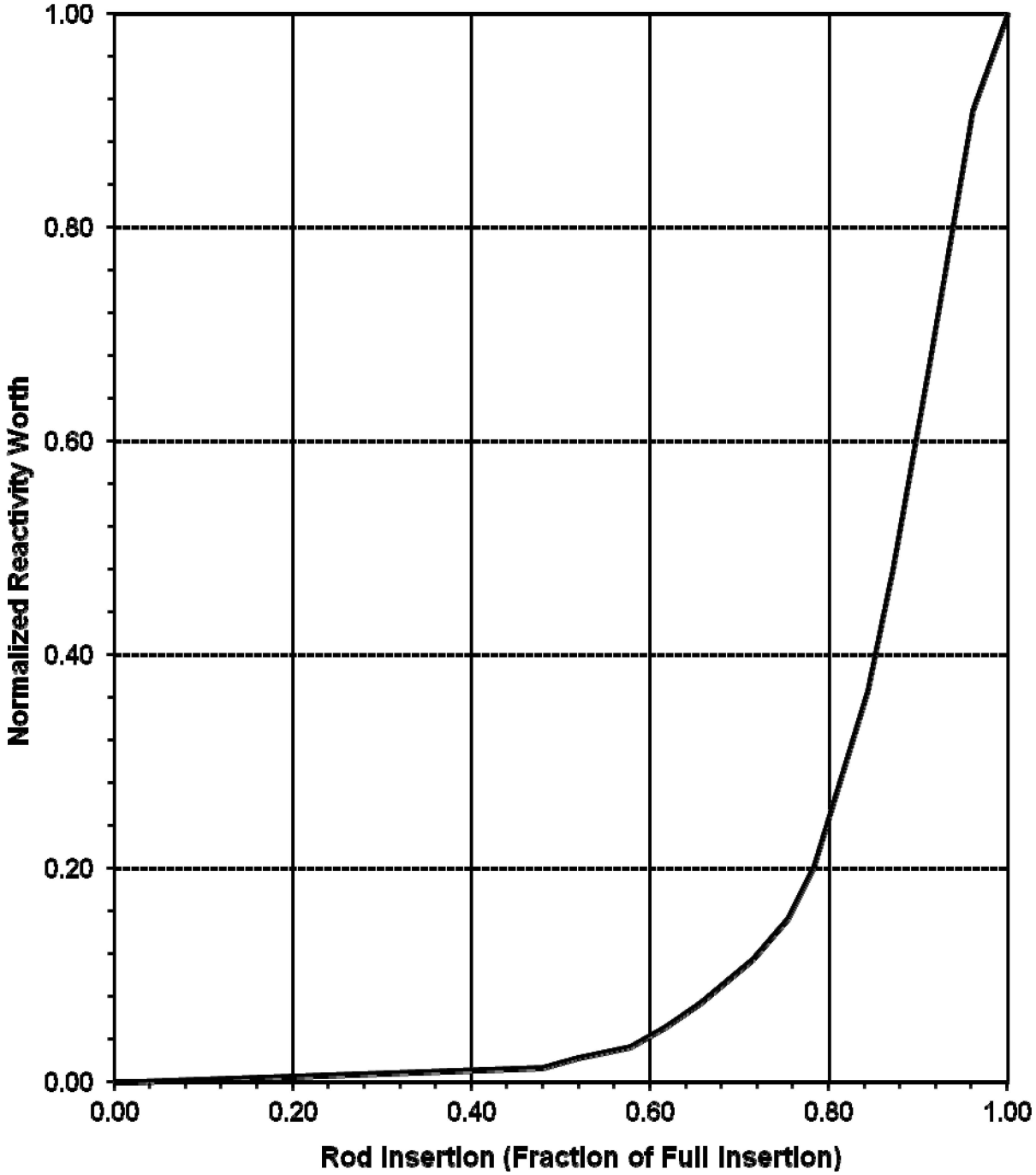

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FIGURE 15.0-4

NORMALIZED RCCA REACTIVITY WORTH VERSUS FRACTIONAL ROD INSERTION

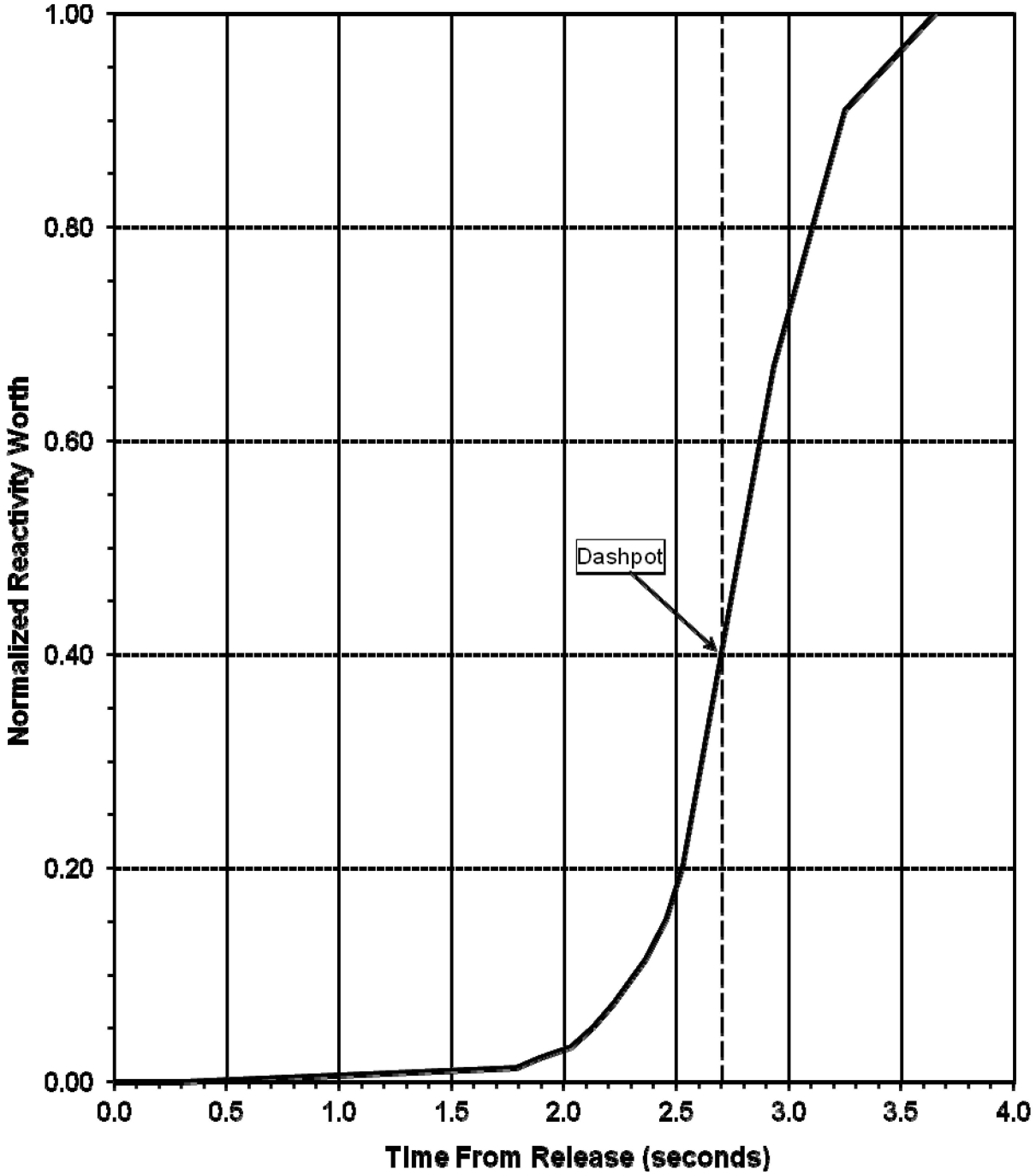

FIGURE 15.0-5
NORMALIZED RCCA REACTIVITY WORTH VERSUS TIME FROM RELEASE

FIGURE 15.0-6 HAS BEEN DELETED

ABBREVIATIONS USED:

AFWS	- AUXILIARY FEEDWATER SYSTEM
CVCS	- CHEMICAL AND VOLUME CONTROL SYSTEM
ESFAS	- ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
FW	- FEEDWATER
RTS	- REACTOR TRIP SYSTEM
SIS	- SAFETY INJECTION SYSTEM
SI	- SAFETY INJECTION
RT	- REACTOR TRIP
CS	- CONTIANMENT SPRAY
ESF	- ENGINEERED SAFETY FEATURE
ECCS	- EMERGENCY CORE COOLING SYSTEM
HL	- HOT LEG
CL	- COLD LEG
CCWS	- COMPONENT COOLING WATER SYSTEM
RCS	- REACTOR COOLANT SYSTEM
ESWS	- ESSENTIAL SERVICE WATER SYSTEM
HPI	- HIGH PRESSURE INJECTION
LPI	- LOW PRESSURE INJECTION
CI	- CONTAINMENT ISOLATION
SG	- STEAM GENERATOR
GWPS	- GASEOUS WASTE PROCESSING SYSTEM

NOTES;

1. FOR TRIP INITIATION AND SAFETY SYSTEM ACTUATION. MULTIPLE SIGNALS ARE SHOWN BUT ONLY A SINGLE SIGNAL IS REQUIRED. THE OTHER SIGNALS ARE BACKUPS.

2. NO TIMING SEQUENCE IS IMPLIED BY POSITION OF VARIOUS BRANCHES. REFER TO EVENT TIMING SEQUENCES PRESENTED IN TABULAR FORM IN PERTINENT ACCIDENT ANALYSIS SECTION OF CHAPTER 15.0 OF THE FSAR.

DIAGRAM SYMBOLS:

- EVENT TITLE

- BRANCH POINT FOR DIFFERENT PLANT CONDITIONS

- SAFETY SYSTEM

- SAFETY ACTION

- SYSTEM REQUIRED TO MEET SINGLE-FAILURE CRITERIA

- MANUAL ACTION REQUIRED DURING SYSTEM OPERATION

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FIGURE 15.0 7

ABBREVIATIONS AND SYMBOLS USED IN SEQUENCE DIAGRAMS

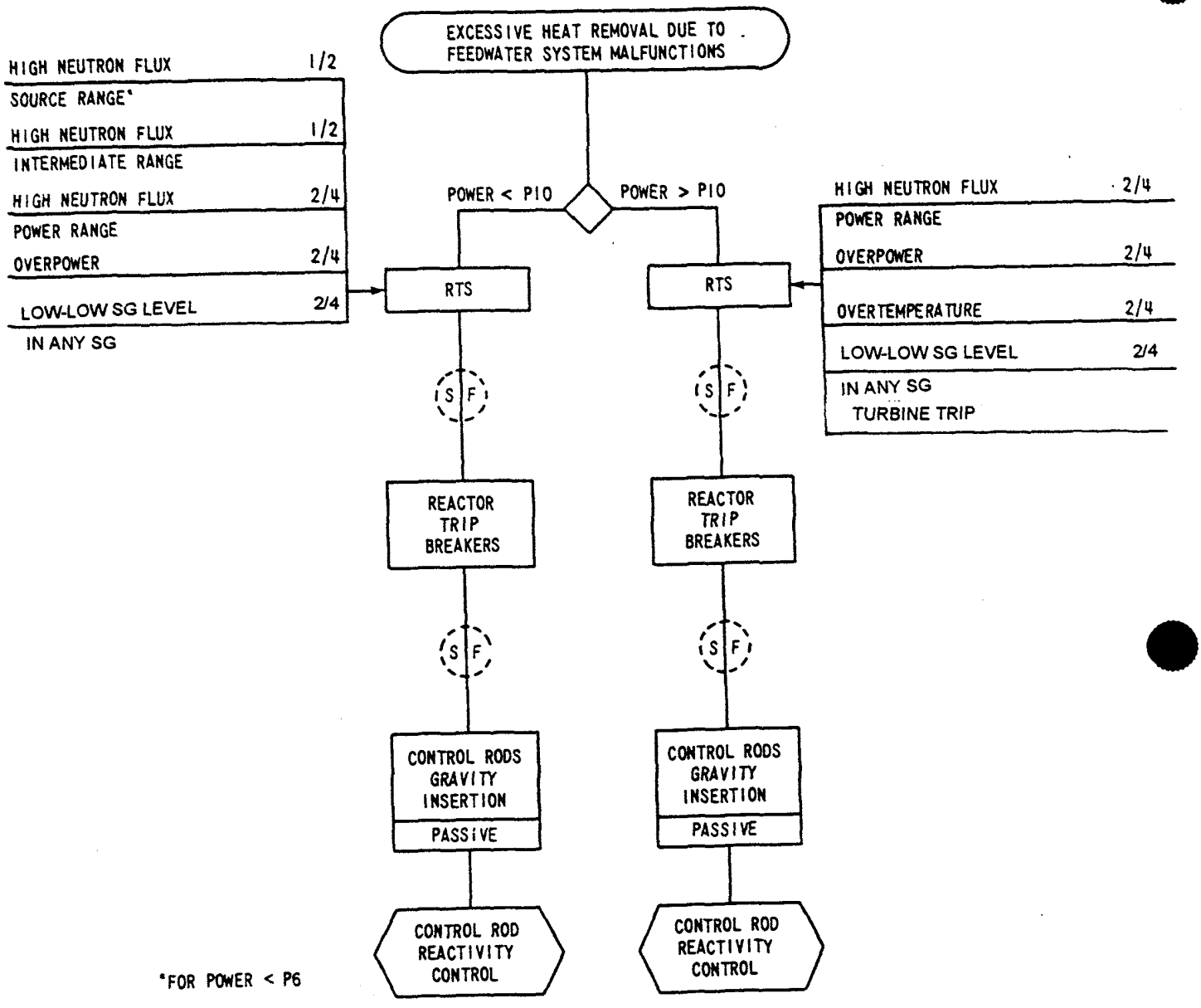

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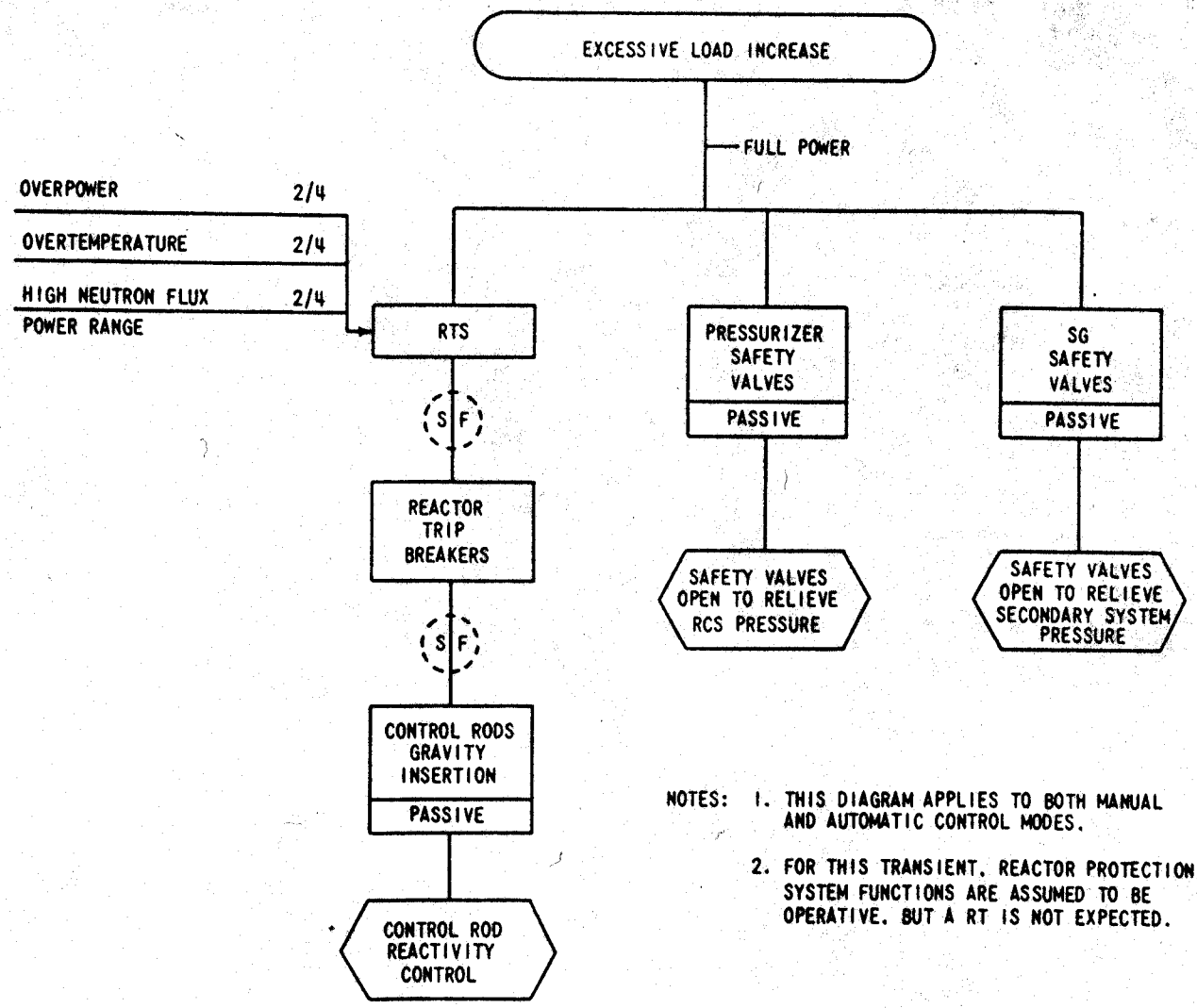
FIGURE 15.0-8

EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEMS MALFUNCTION



NOTES:
1. THIS DIAGRAM APPLIES TO BOTH MANUAL AND AUTOMATIC CONTROL MODES.
2. FOR THIS TRANSIENT, REACTOR PROTECTION SYSTEM FUNCTIONS ARE ASSUMED TO BE OPERATIVE, BUT A RT IS NOT EXPECTED.

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FIGURE 15.0-9

EXCESSIVE LOAD INCREASE

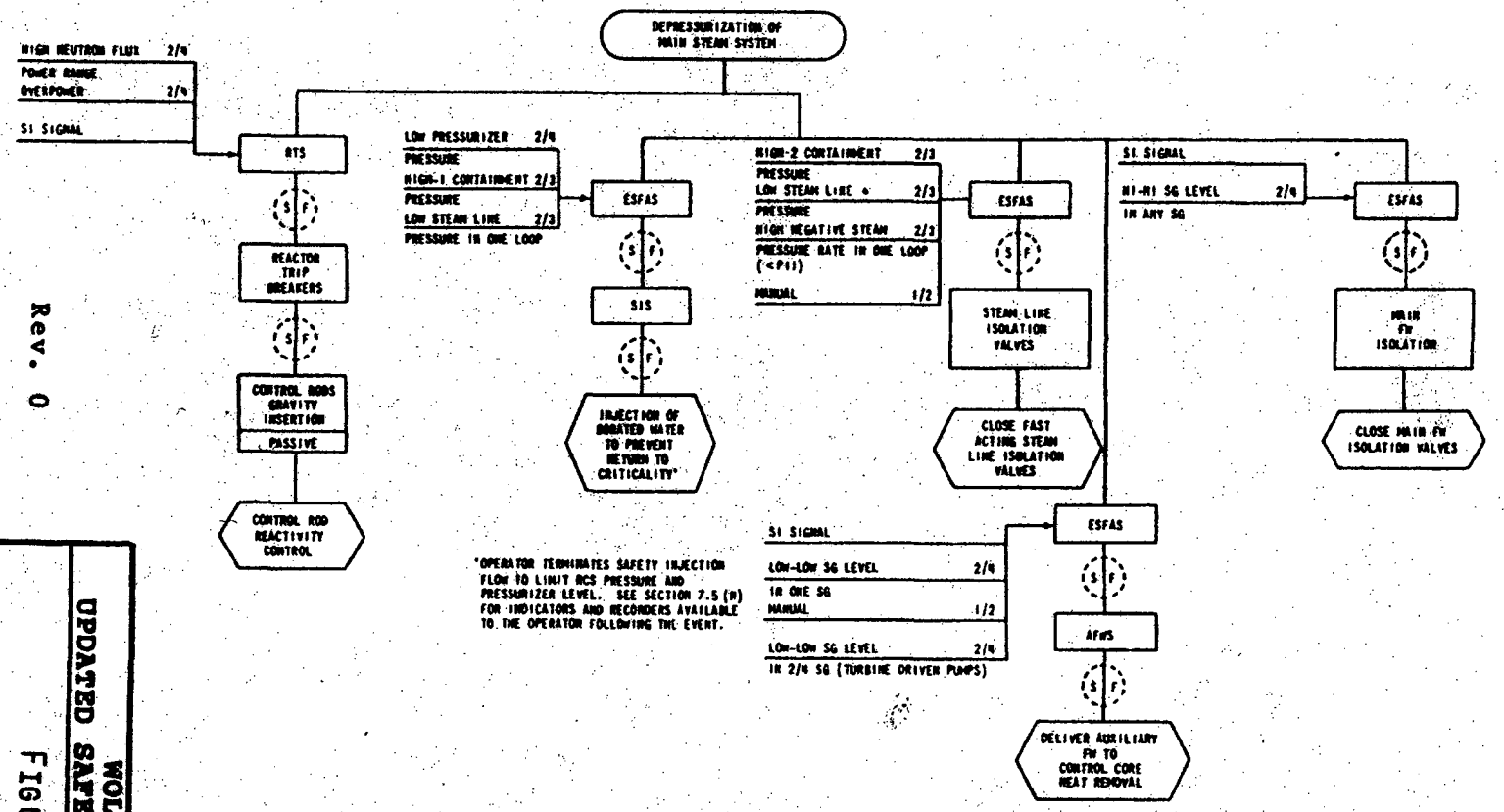

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FIGURE 15.0-10

DEPRESSURIZATION OF MAIN STEAM SYSTEM

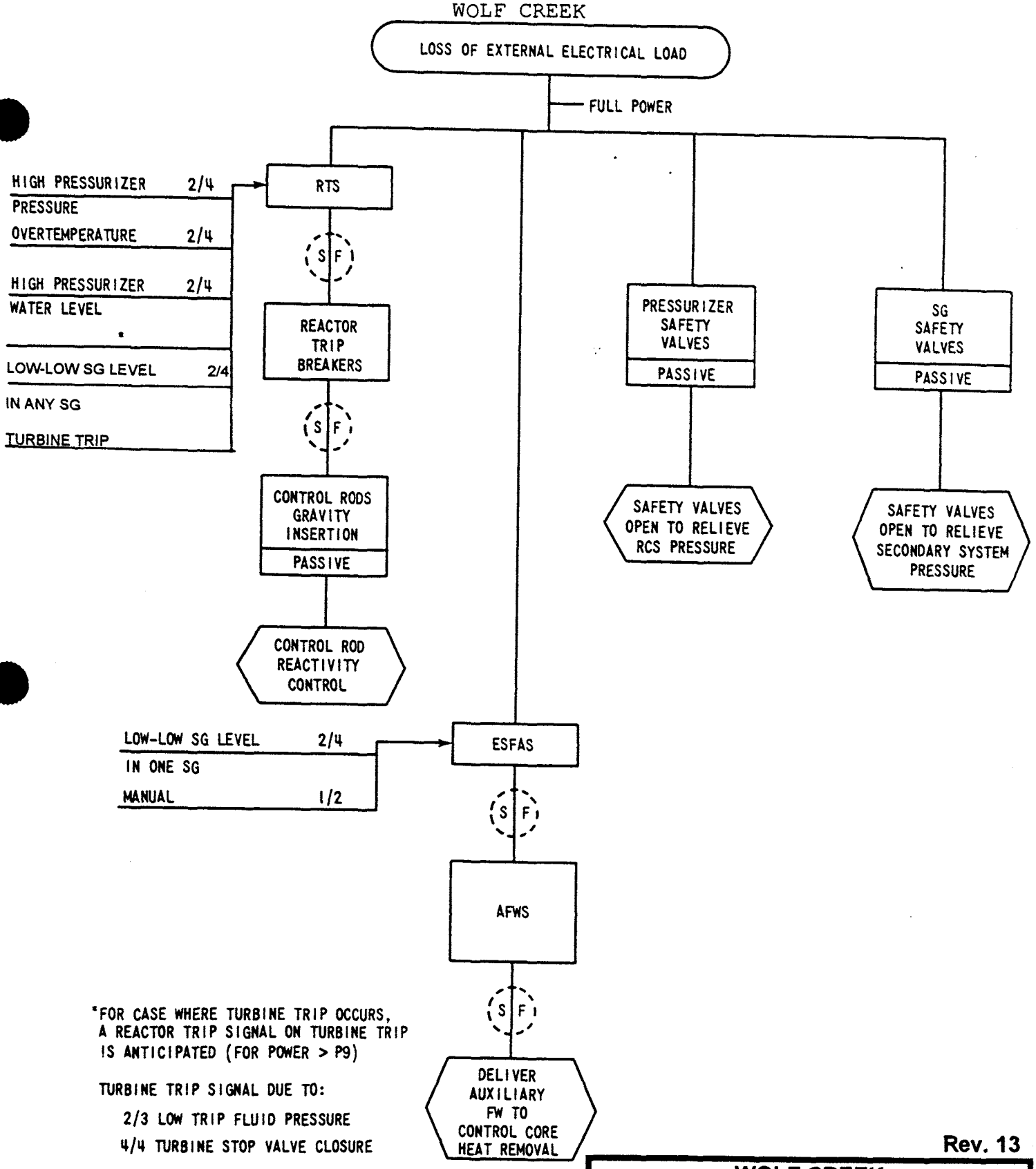

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FIGURE 15.0-11

LOSS OF EXTERNAL LOAD

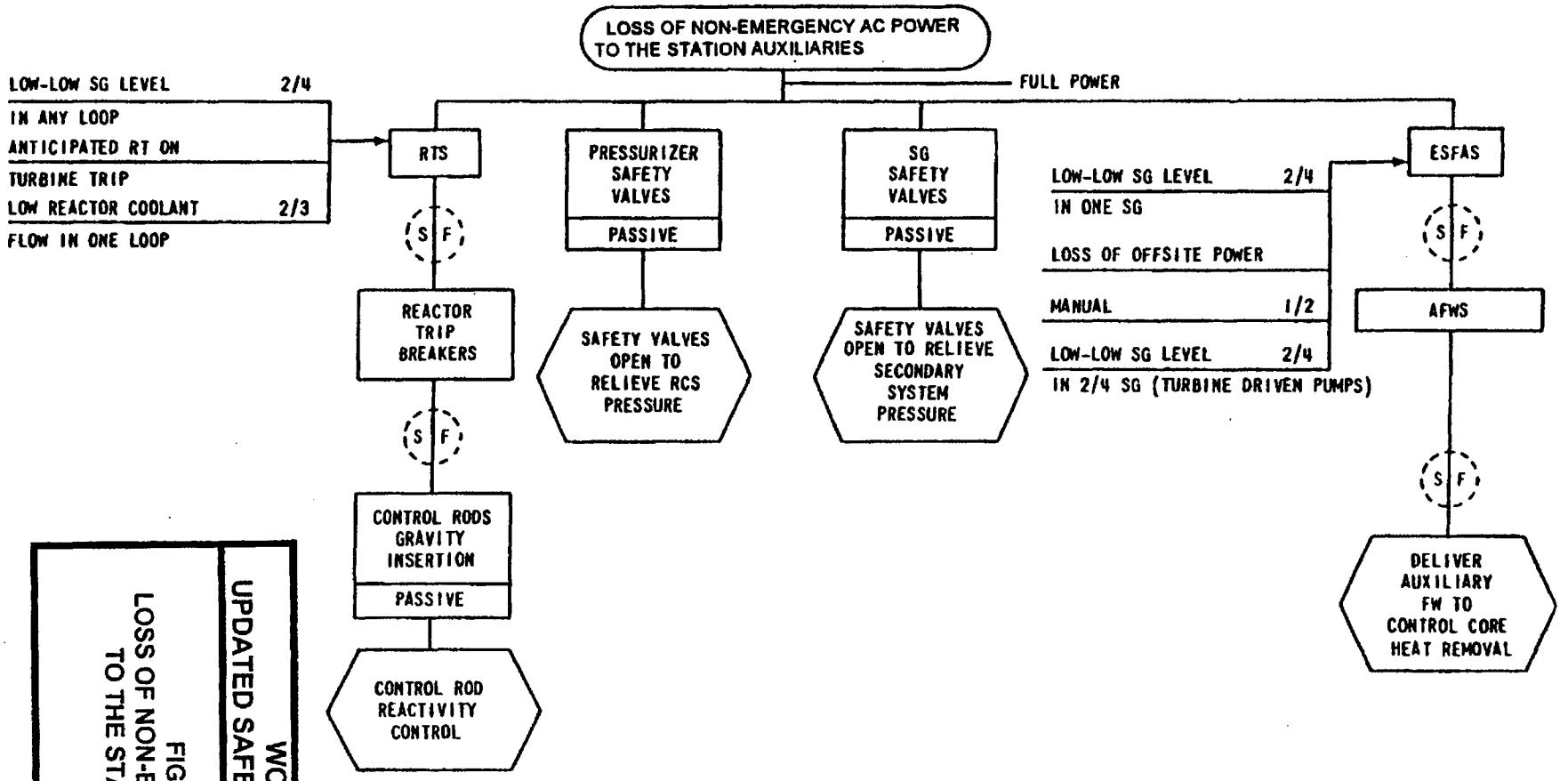

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FIGURE 15.0-12
LOSS OF NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

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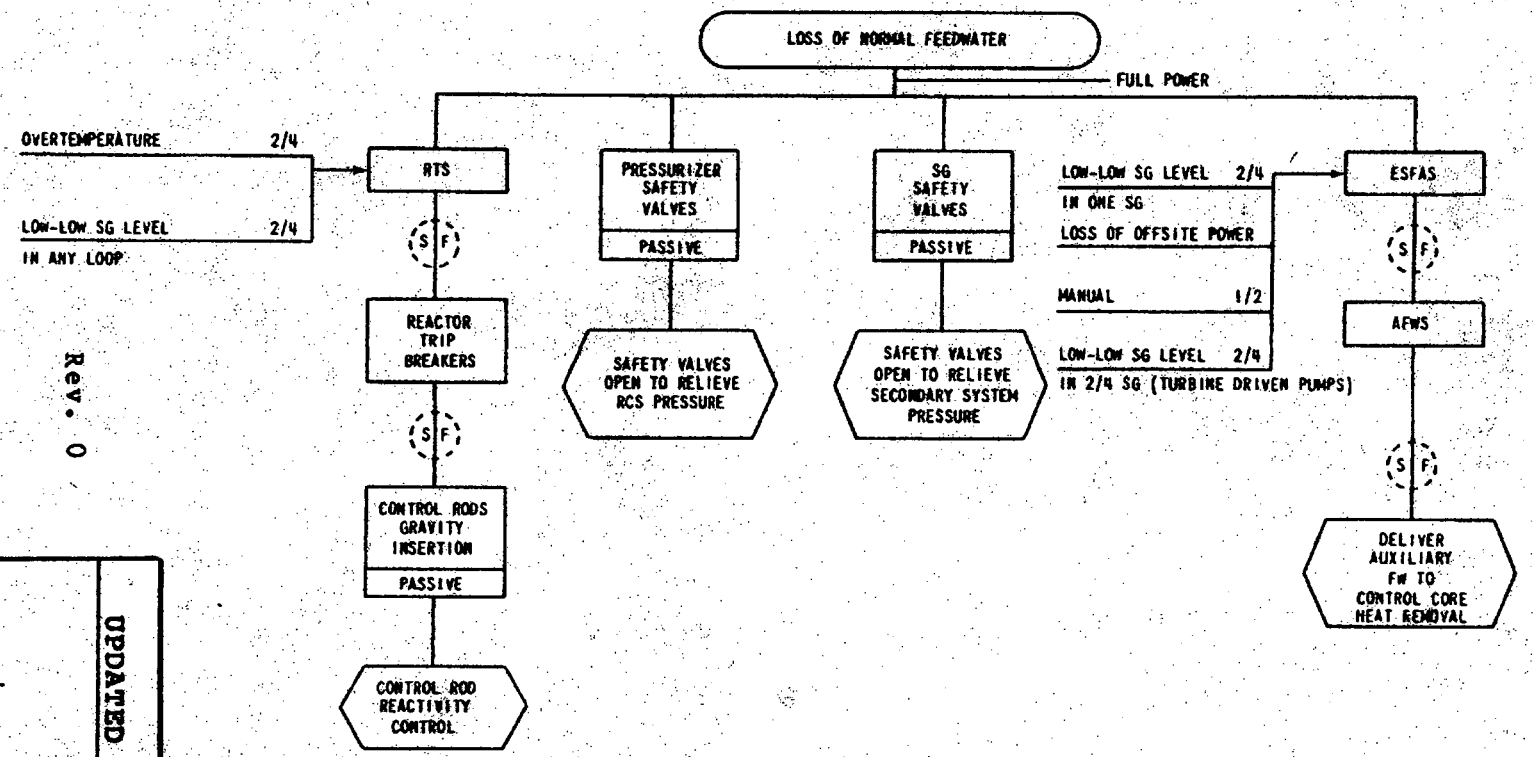

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FIGURE 15.0-13

LOSS OF NORMAL FEEDWATER

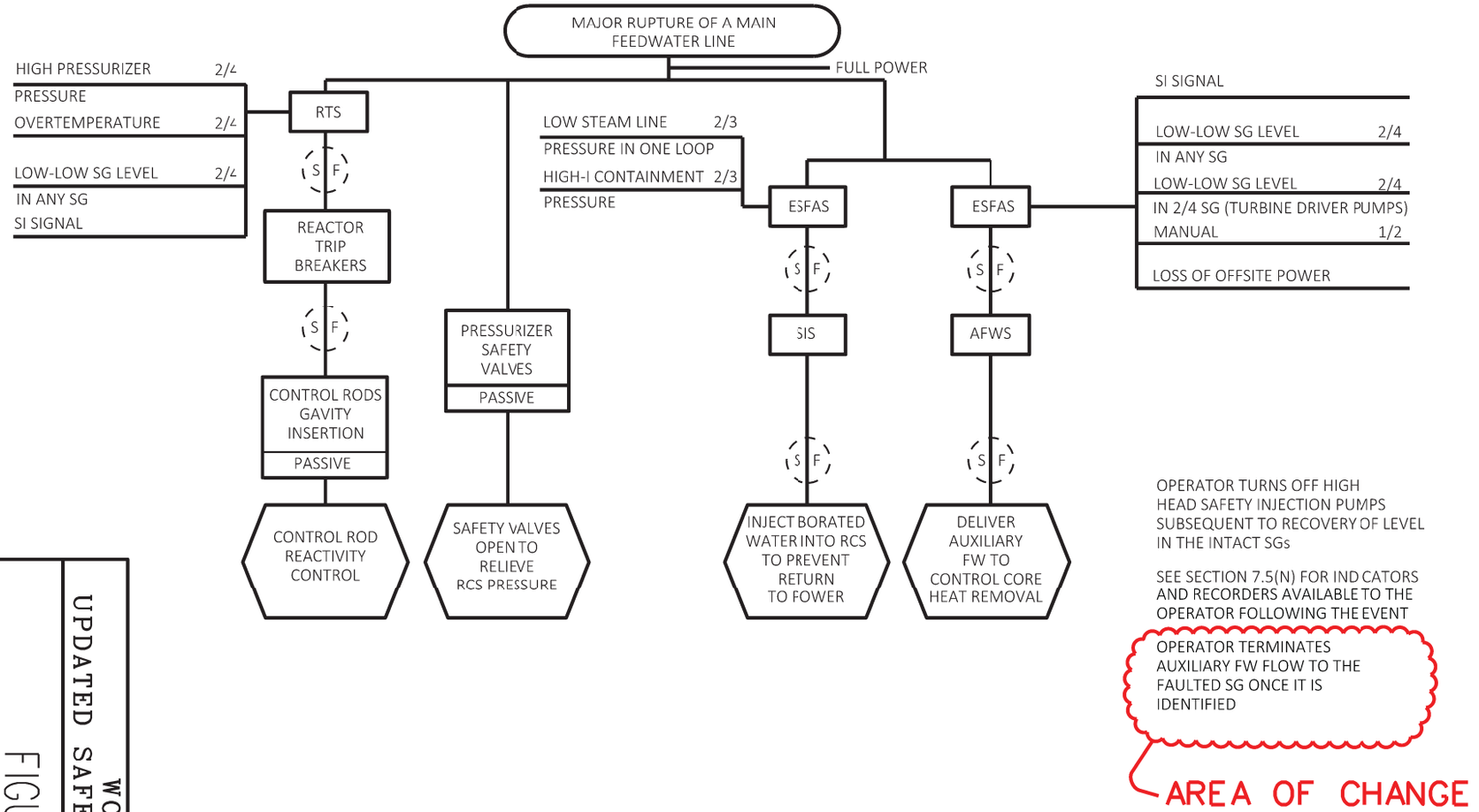

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FIGURE 15.0-14

MAJOR RUPTURE OF A MAIN FEEDWATER LINE

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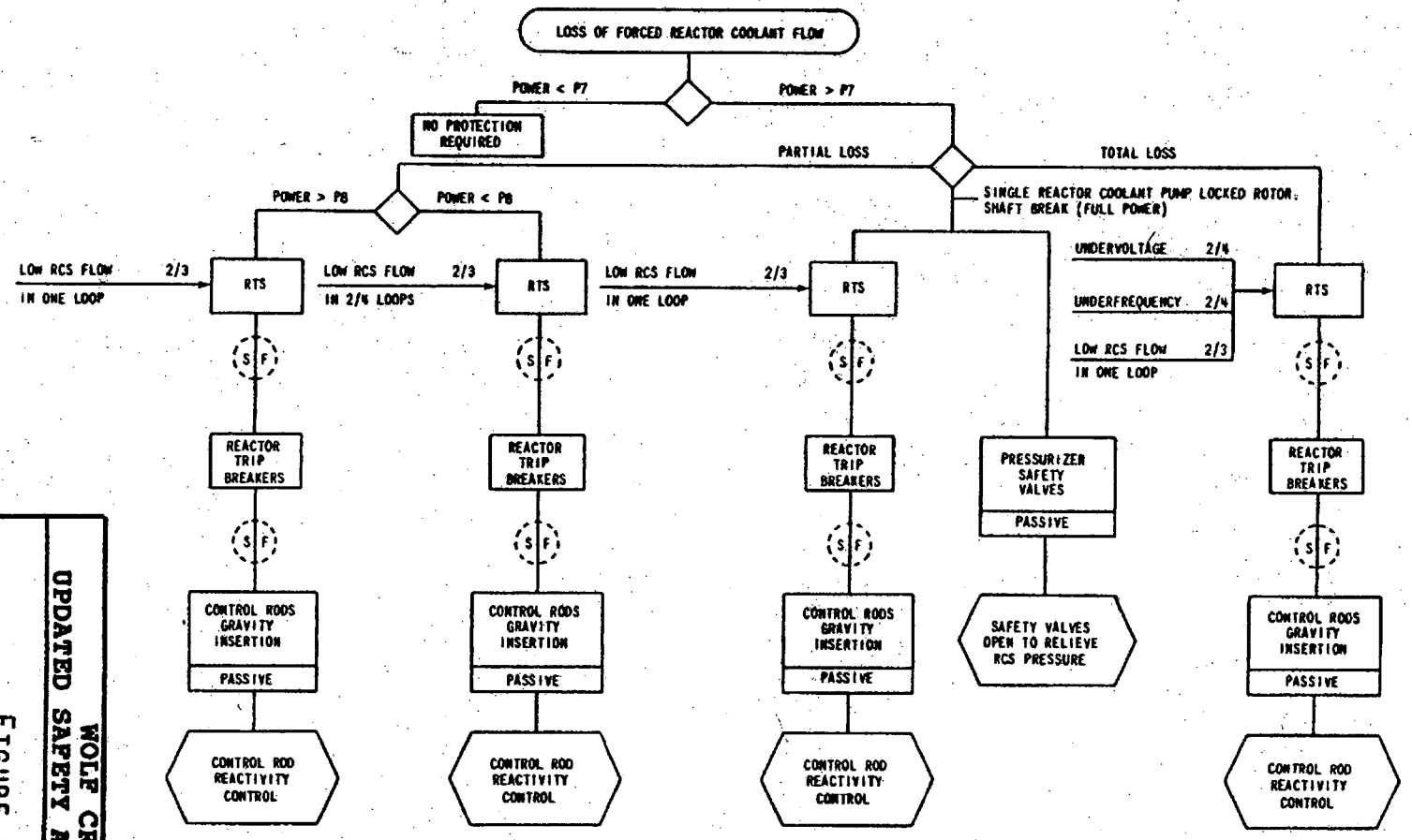

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FIGURE 15.0-15

LOSS OF FORCED REACTOR COOLANT FLOW

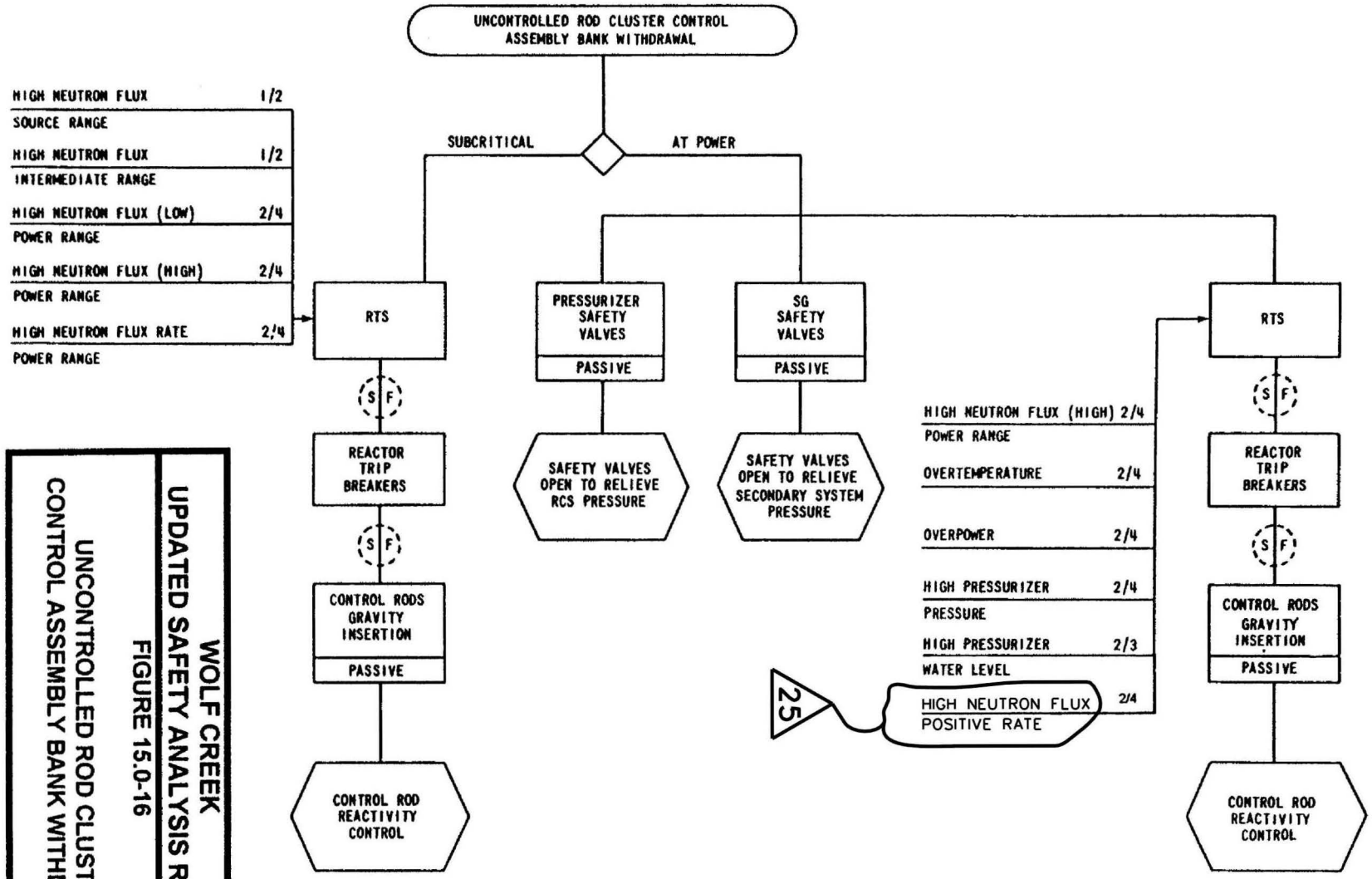

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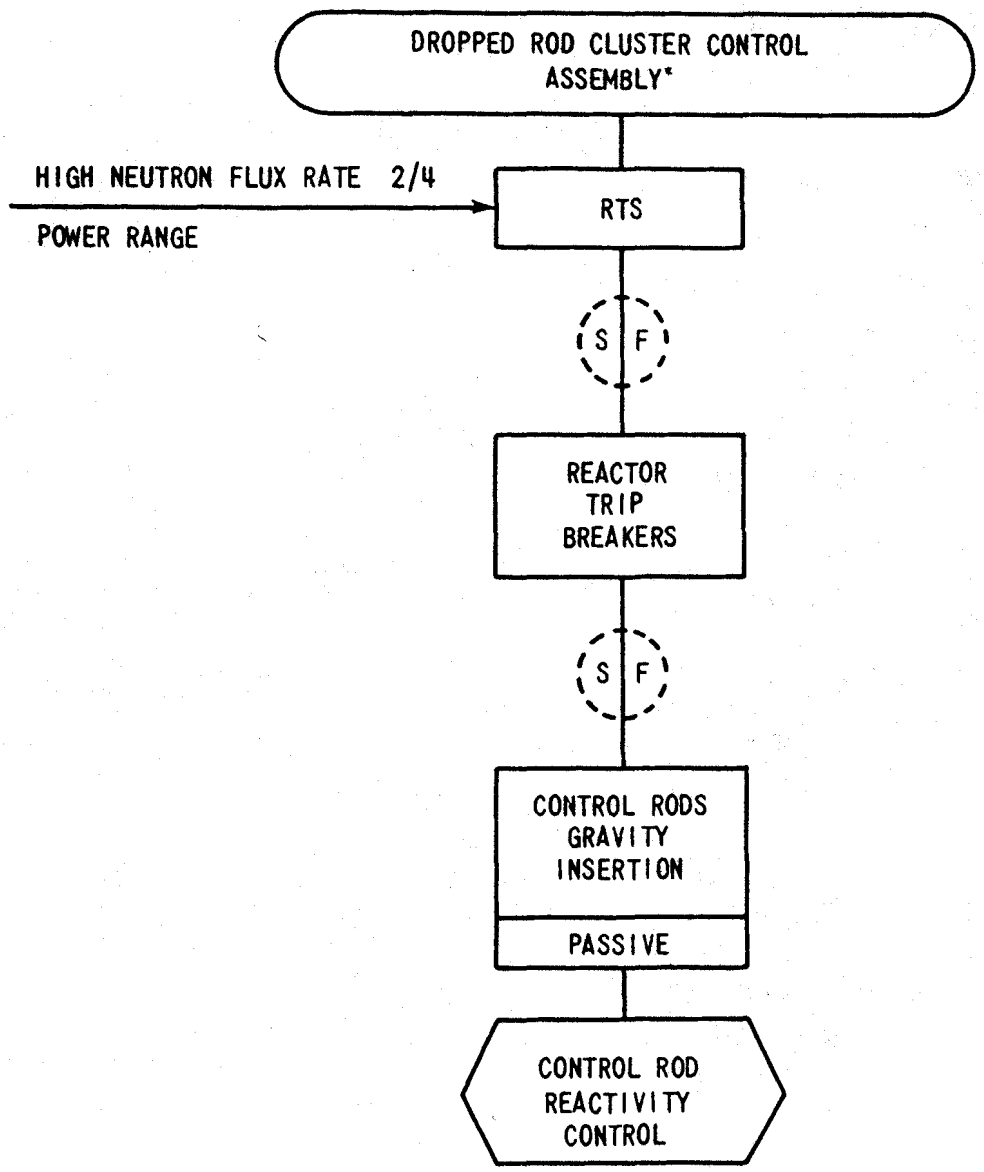
FIGURE 15.0-16
UNCONTROLLED ROD CLUSTER
CONTROL ASSEMBLY BANK WITHDRAWAL



*TRIP SEQUENCE OCCURS ONLY IF ROD WORTH IS GREATER THAN MINIMUM REQUIRED TO TRIGGER THE FLUX RATE TRIP. OR IF AN ENTIRE ROD CLUSTER CONTROL ASSEMBLY BANK DROPS. FOR OTHER ROD WORTH. OR FOR MISALIGNMENT. NO PROTECTION REQUIRED.

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FIGURE 15.0-17

DROPPED ROD CLUSTER CONTROL ASSEMBLY

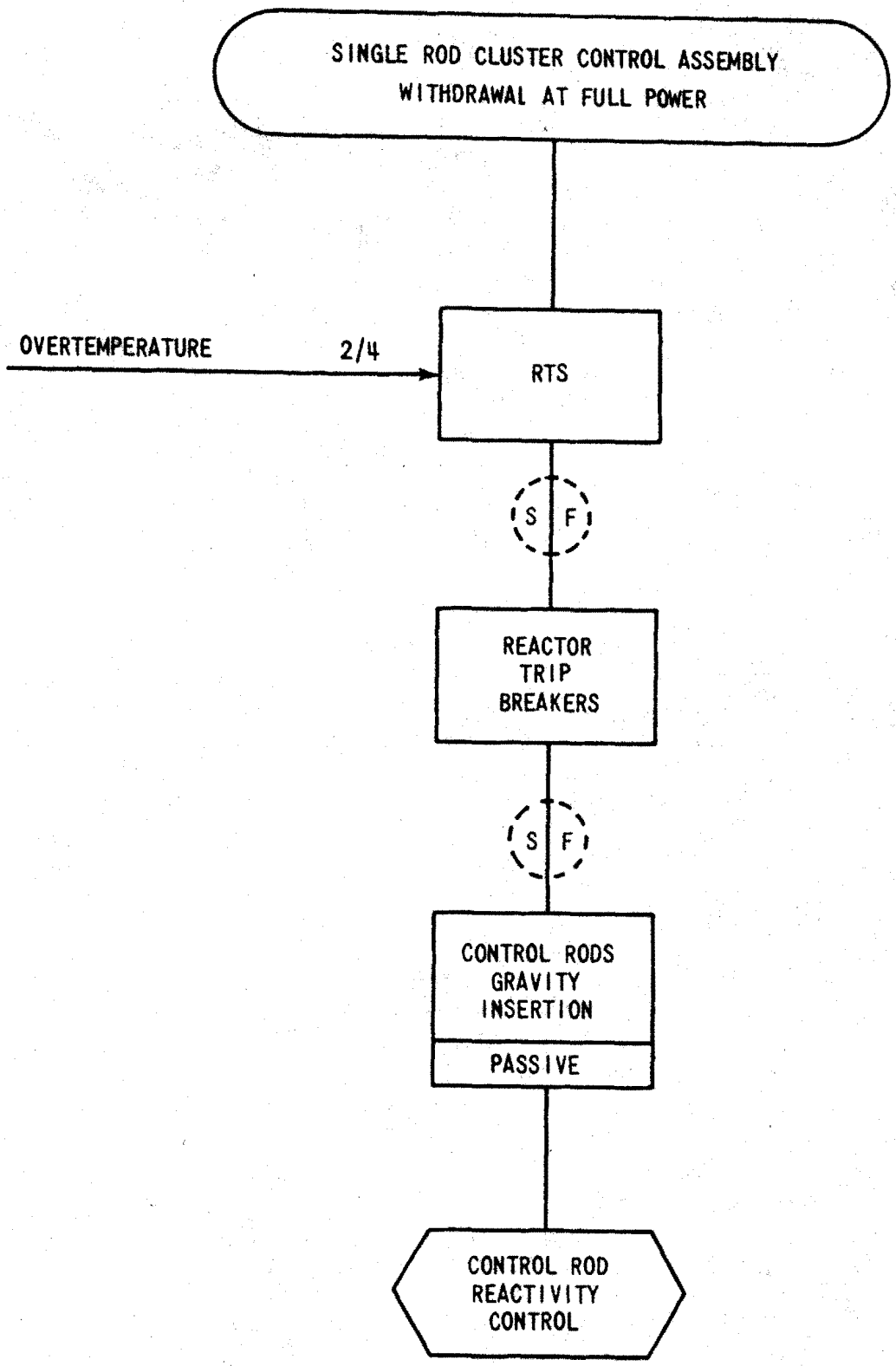

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FIGURE 15.0-18

SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER

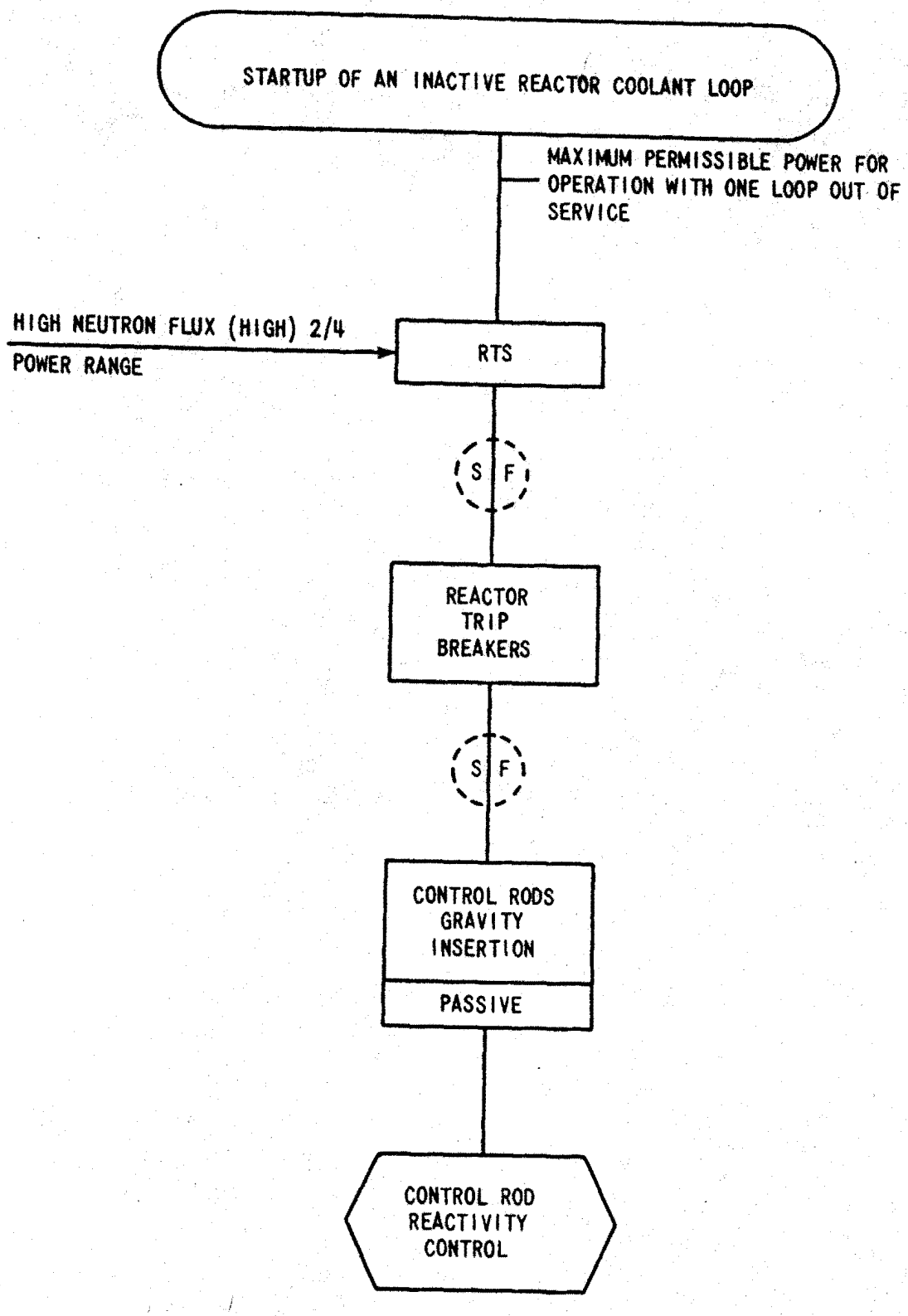

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FIGURE 15.0-19

STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

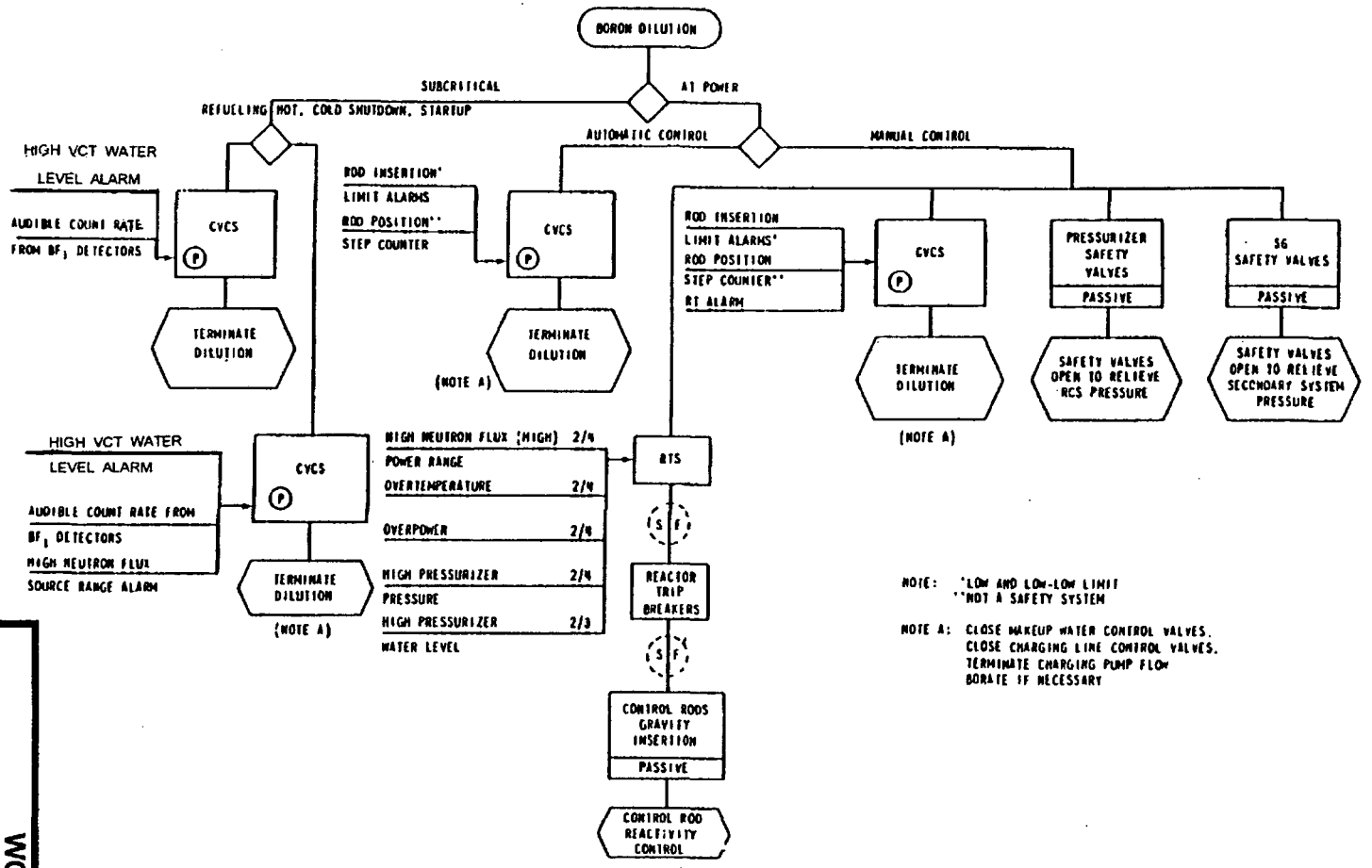

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FIGURE 15.0-20
BORON DILUTION

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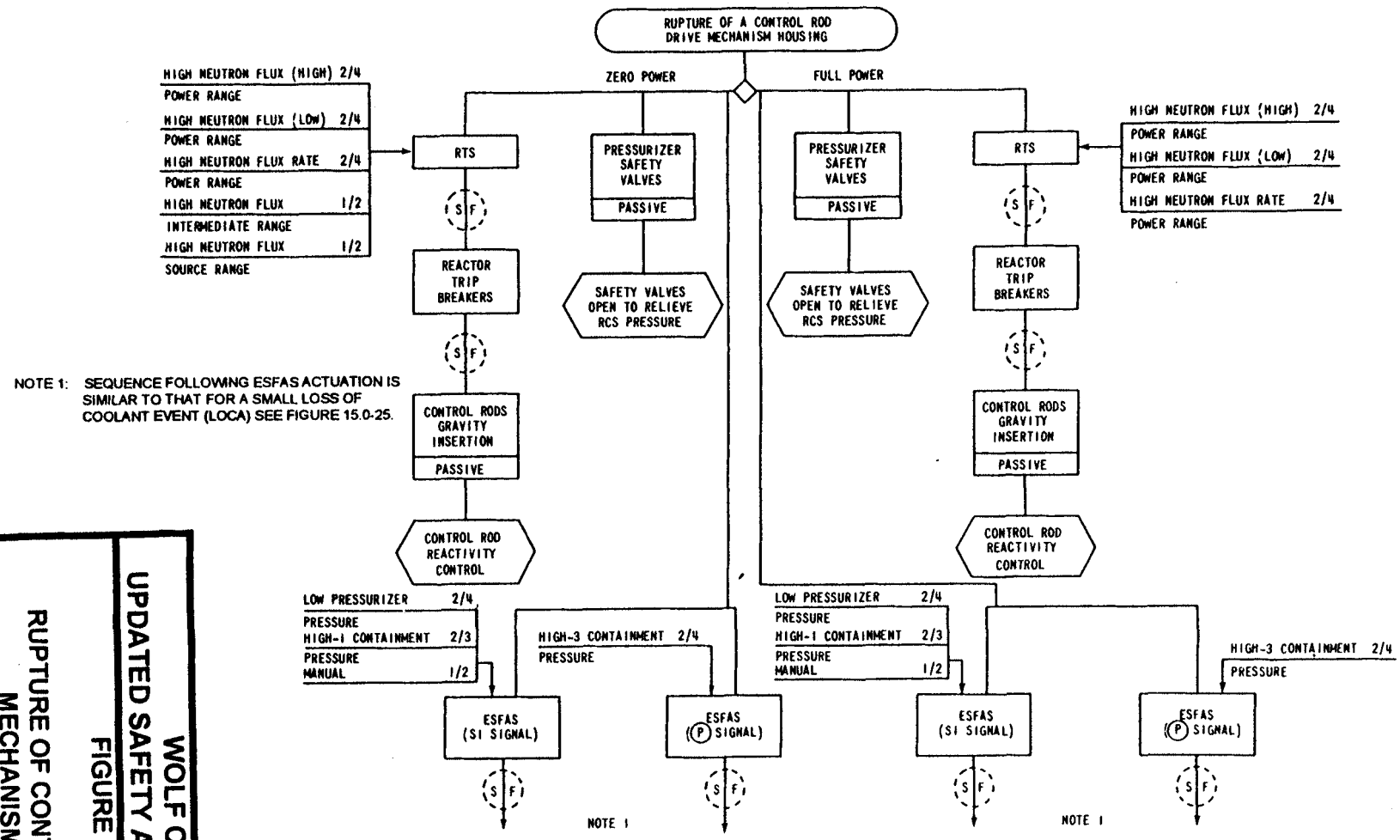

NOTE 1: SEQUENCE FOLLOWING ESFAS ACTUATION IS SIMILAR TO THAT FOR A SMALL LOSS OF COOLANT EVENT (LOCA) SEE FIGURE 15.0-25.

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FIGURE 15.0-21
RUPTURE OF CONTROL ROD DRIVE MECHANISM HOUSING

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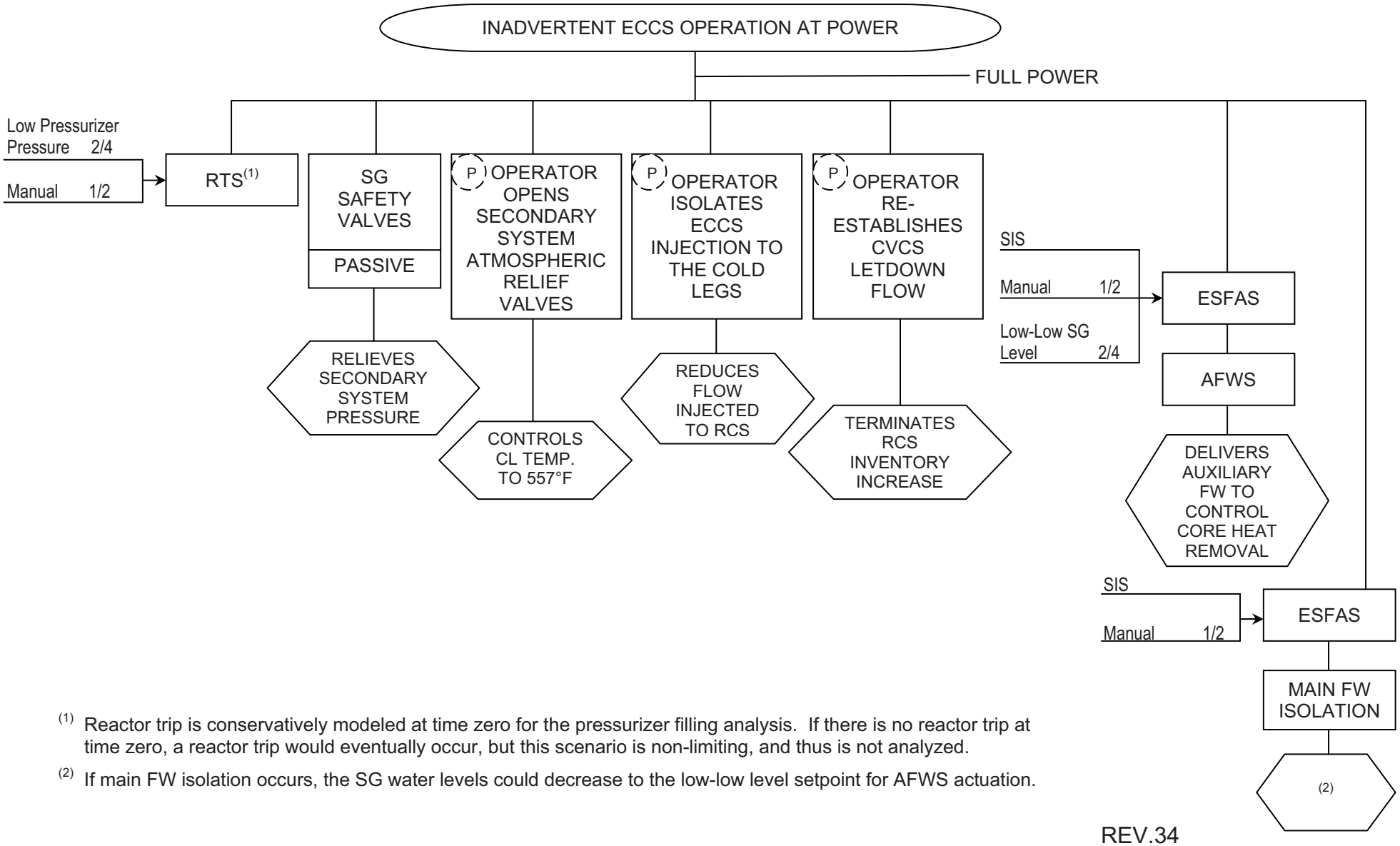

(1) Reactor trip is conservatively modeled at time zero for the pressurizer filling analysis. If there is no reactor trip at time zero, a reactor trip would eventually occur, but this scenario is non-limiting, and thus is not analyzed.

(2) If main FW isolation occurs, the SG water levels could decrease to the low-low level setpoint for AFWS actuation.

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Figure 15.0-22

INADVERTENT ECCS OPERATION AT POWER

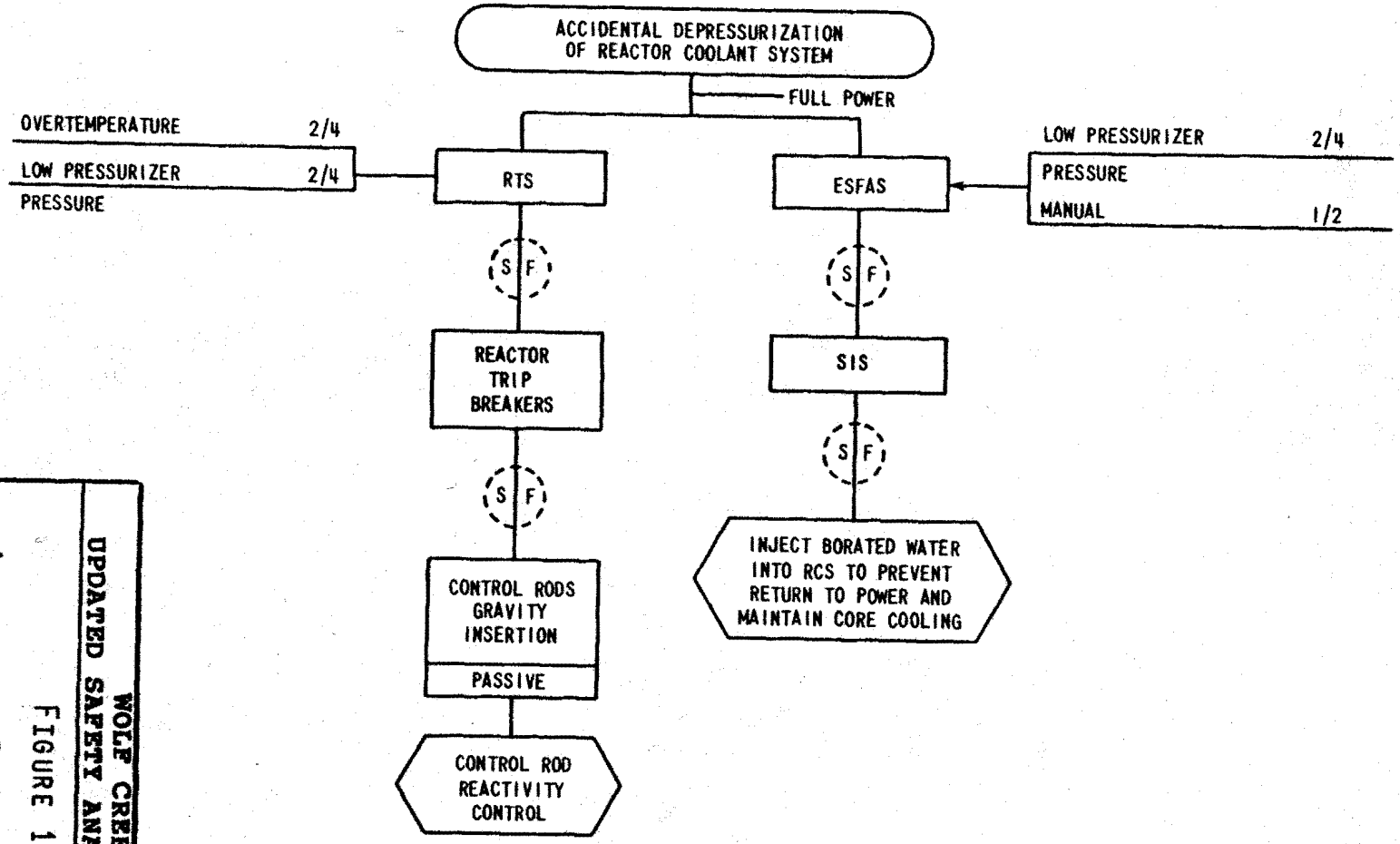

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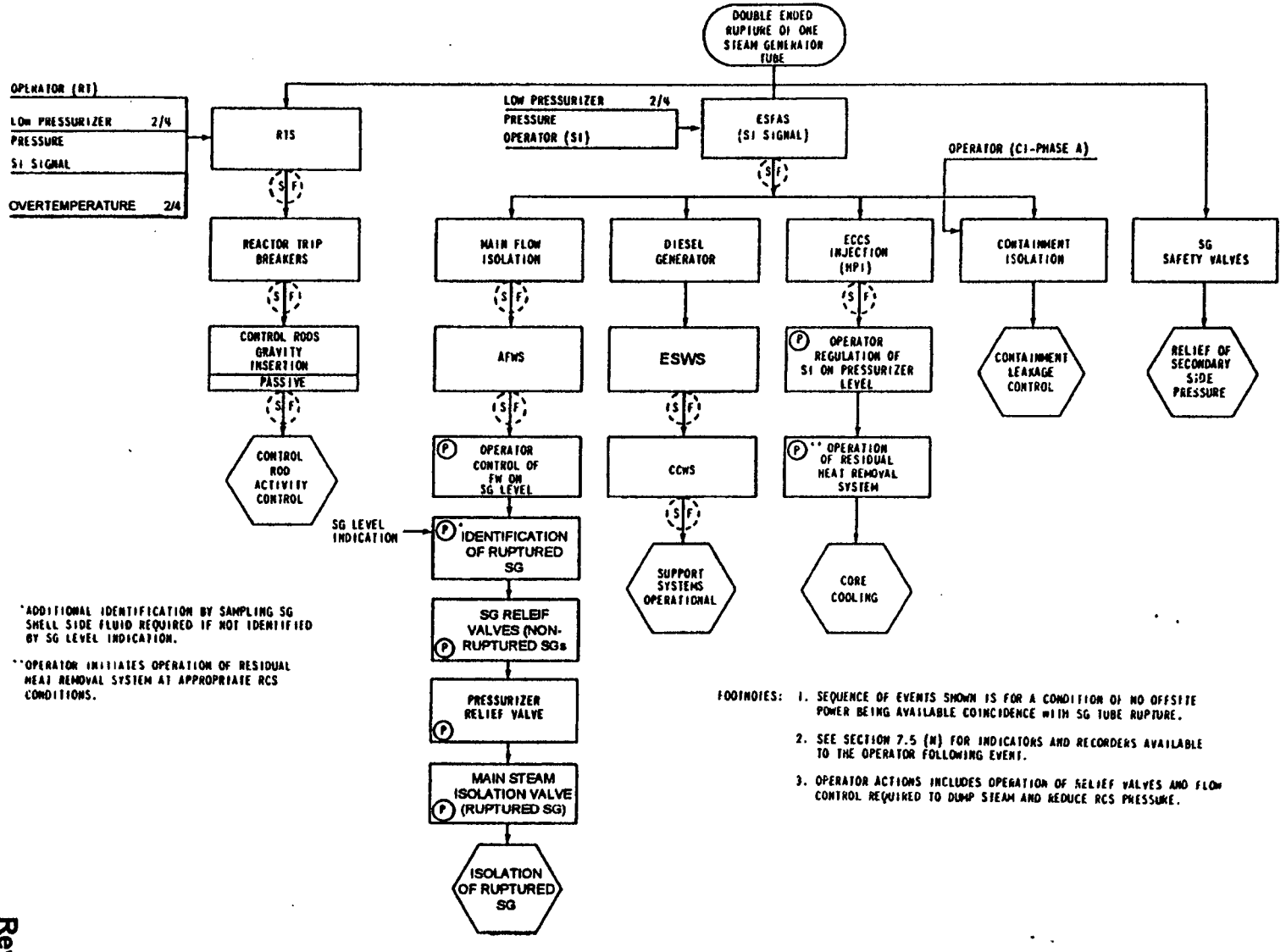
FIGURE 15.0-23

ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM



*ADDITIONAL IDENTIFICATION BY SAMPLING SG SHELL SIDE FLUID REQUIRED IF NOT IDENTIFIED BY SG LEVEL INDICATION.

**OPERATOR INITIATES OPERATION OF RESIDUAL HEAT REMOVAL SYSTEM AT APPROPRIATE RCS CONDITIONS.

FOOTNOTES:
1. SEQUENCE OF EVENTS SHOWN IS FOR A CONDITION OF NO OFFSITE POWER BEING AVAILABLE COINCIDENCE WITH SG TUBE RUPTURE.
2. SEE SECTION 7.5 (N) FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT.
3. OPERATOR ACTIONS INCLUDES OPERATION OF RELIEF VALVES AND FLOW CONTROL REQUIRED TO DUMP STEAM AND REDUCE RCS PRESSURE.

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FIGURE 15.0-24
STEAM GENERATOR TUBE RUPTURE

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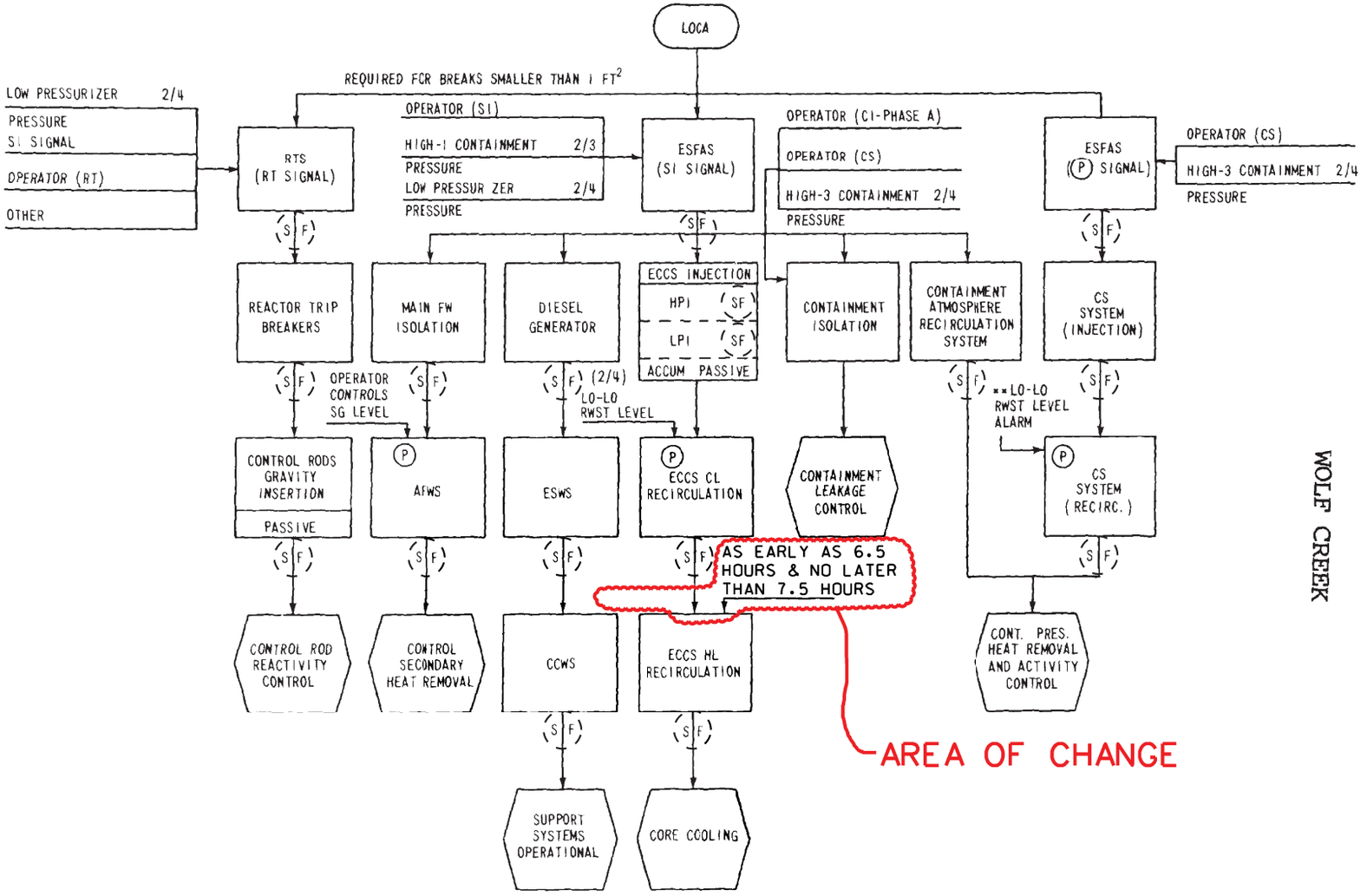

* OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION

** OPERATOR ALIGNS SPRAY PUMP SUCTION FOR RECIRCULATION TO CONTAINMENT SUMP ON LO LO RWST LEVEL ALARM

FOOTNOTE: SEE SECTION 7.5 (N) FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT

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FIGURE 15 0-25

LOSS OF COOLANT ACCIDENT

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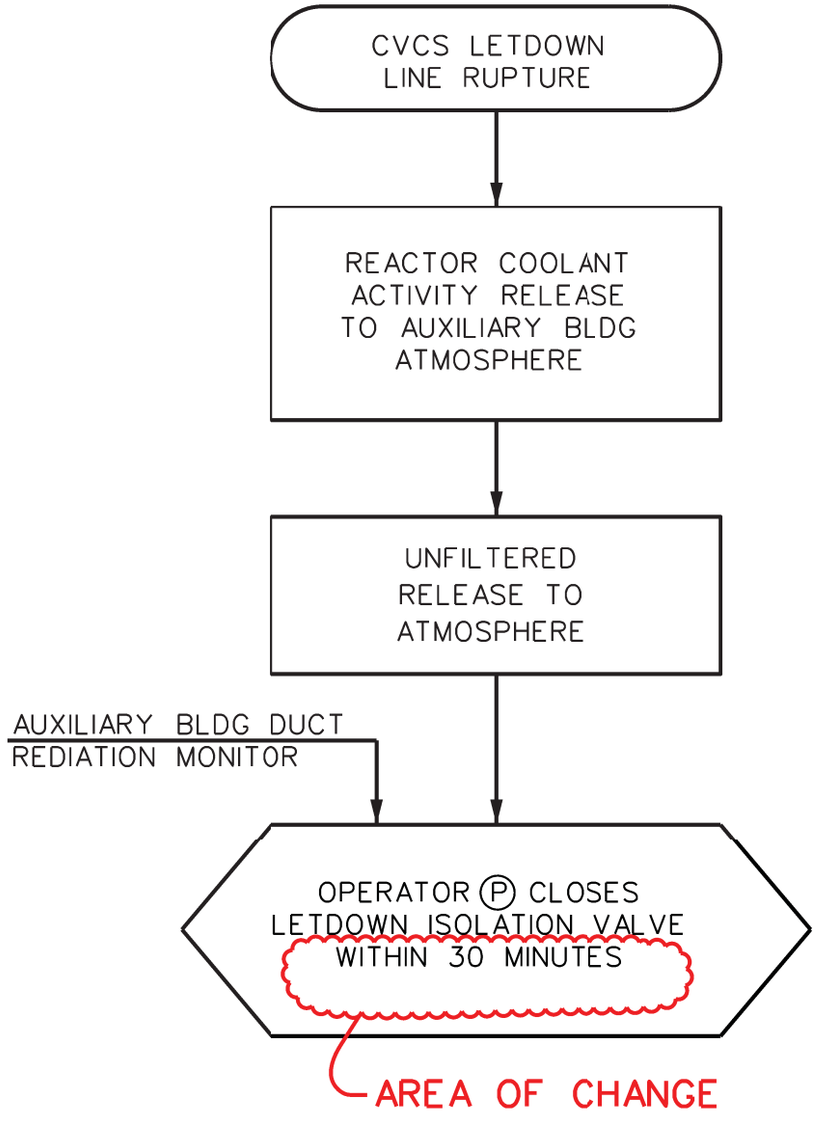

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FIGURE 15.0-26

CVCS LETDOWN LINE RUPTURE

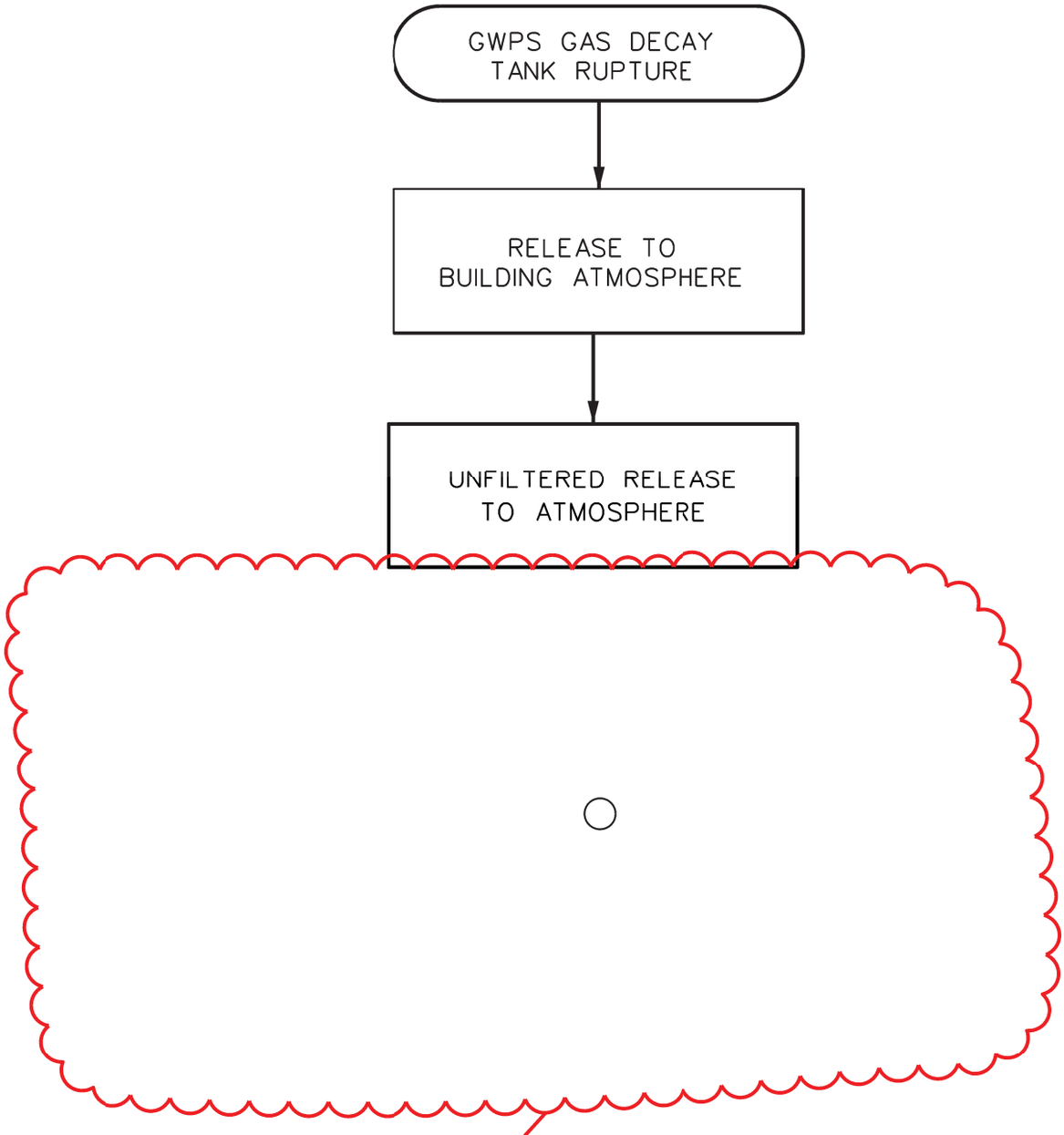

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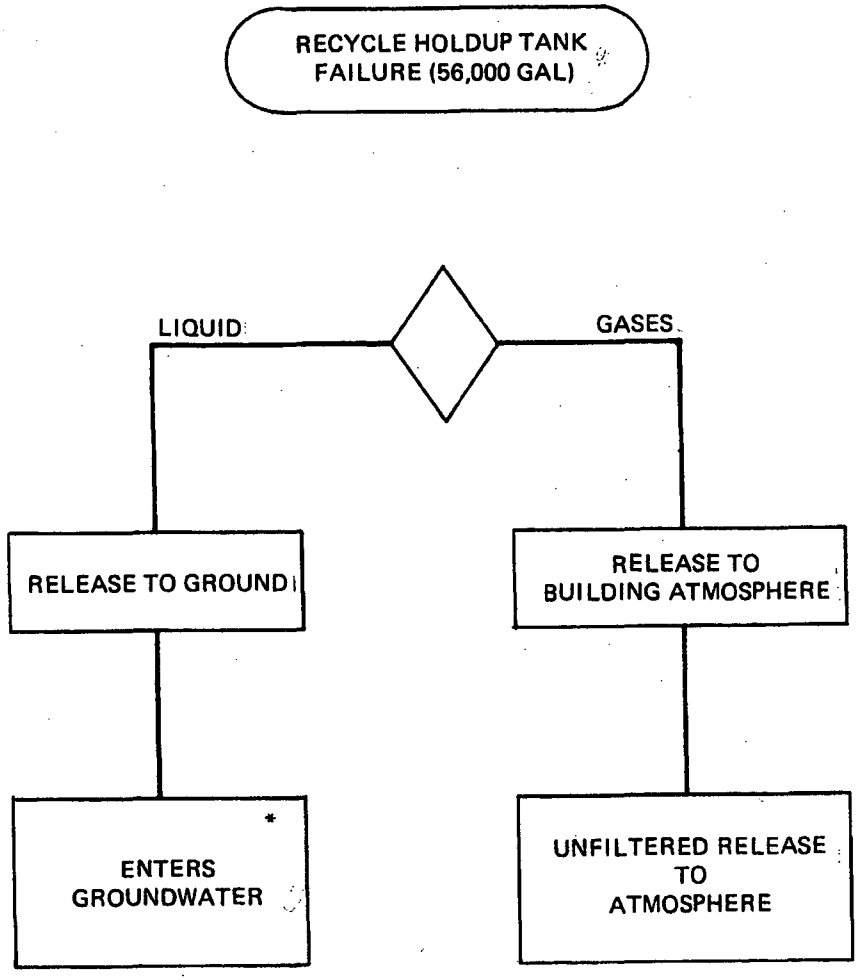
FIGURE 15.0-27

GWPS GAS DECAY TANK RUPTURE



*See Section 2.4.13

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FIGURE 15.0-28

FLOOR DRAIN TANK FAILURE

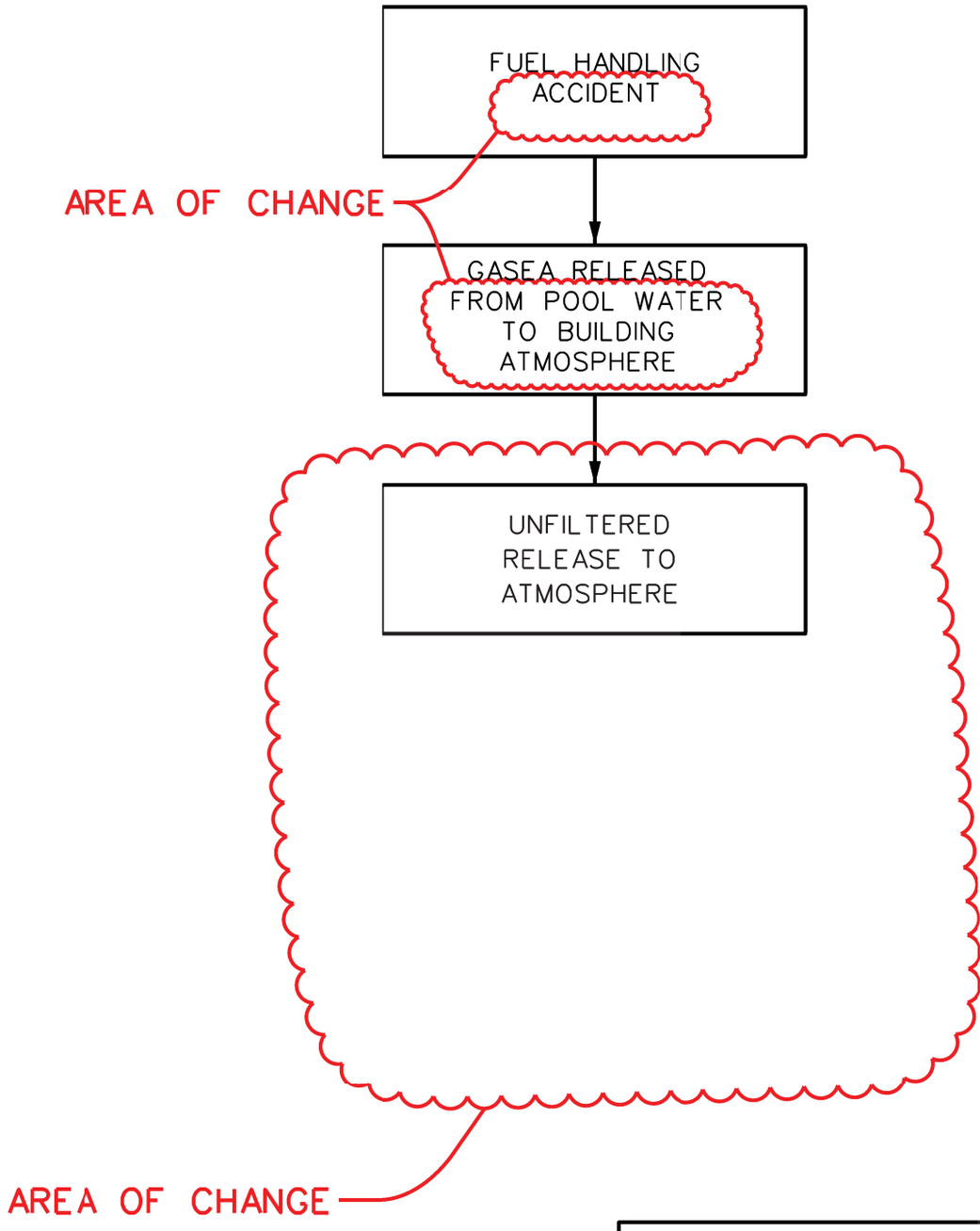

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FIGURE 15.0-29

FUEL HANDLING ACCIDENT

AREA OF CHANGE

FIGURE 15.0-30 HAS BEEN DELETED

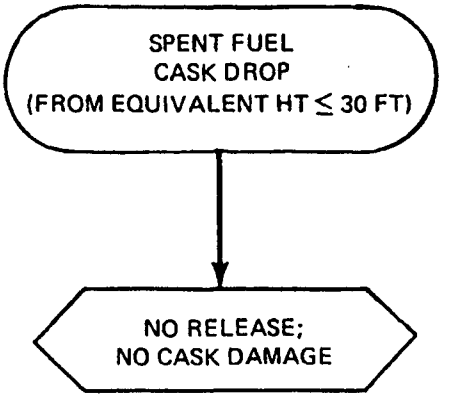

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FIGURE 15.0-31

SPENT FUEL CASK DROP ACCIDENT

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 the following events, which have been identified as the most limiting cases:

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 atmospheric relief or safety valve

e. Steam system piping failure

The above are considered to be ANS Condition II events, with the exception of steam system piping failures, which are considered to be ANS Condition III (minor) and Condition IV (major) events. Section 15.0.1 provides a discussion of ANS classifications and applicable acceptance criteria.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the main steam line break accident discussed in Section 15.1.5. Therefore, the radiological consequences are only reported for that limiting case.

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

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing the reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the RCS. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit.

A reduction in feedwater temperature may be caused by a feedwater heater bypass valve failure or a spurious heater drain pump trip. In the event of one of these causes, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature, and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load transient is less severe than the full power case.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-8.

15.1.1.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater temperature reduction transient was analyzed with the RETRAN computer code (References 5 and 7). This code simulates a multi-loop RCS, core neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray and heaters, steam generators, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. RETRAN (Reference 5) is used to conservatively predict DNBR.

The system is analyzed to demonstrate plant behavior due to a feedwater temperature reduction of 200°F (step change) to all steam generators which occurs due to a spurious opening of the low pressure feedwater heater bypass valve. Two cases initiated from full power are analyzed, assuming a conservatively large negative moderator temperature coefficient:

a. feedwater temperature reduction with automatic rod control.

b. feedwater temperature reduction with manual rod control.

All feedwater temperature reduction analyses assume the following:

a. Plant initial power level is 3651 MWt, which includes all uncertainties.

b. An initial water level corresponding to the nominal level minus uncertainties is modeled in all four steam generators.

c. Pressurizer sprays and PORVs are modeled to reduce RCS pressure, resulting in a conservative evaluation of the margin to the DNBR safety analysis limit.

d. The OPΔT reactor trip function was modeled (credited) for this event.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

No credit is taken in the licensing basis analyses for a reactor trip on turbine trip. No single active failure will prevent operation of the reactor protection system.

Results

A comparison of both cases analyzed demonstrates that the limiting feedwater temperature reduction case corresponds to the case that models manual rod control. This case produces the largest reactivity feedback, and therefore results in the greatest power increase.

For both analyzed feedwater temperature reduction cases, the reactor trips on the OPΔT function, which then causes a consequential turbine trip. Minimum DNBR and peak core average thermal power are reached shortly after the reactor trip. The event is effectively terminated following feedwater isolation on a low pressurizer pressure safety injection (SI) signal.

Transient results presented in Figures 15.1-1A, 15.1-1B, and 15.1-1C for the limiting case show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. The DNBR, as shown in Figure 15.1-1C, does not drop below the safety analysis limit.

Since the power level rises during the feedwater temperature reduction incident, the fuel temperatures will initially rise. Subsequent fuel melting is precluded by ensuring that the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot. Power levels up to 121% of the applied initial full power value have been shown to meet this criterion. As shown in Figure 15.1-1A, transient power remains below 121% of the full power value, thereby confirming the fuel melting criterion is satisfied.

The calculated sequence of events for this accident is presented in Table 15.1-1.

15.1.1.3 Conclusions

The results of the analysis show that the minimum DNBR encountered for a feedwater temperature reduction incident remains above the safety analysis limit; in addition, the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot. The DNBR design basis is described in Section 4.4.

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

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater will cause an increase in core power by decreasing the reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in the RCS temperature and thus a reactivity insertion in the presence of a negative moderator temperature coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which initiates feedwater isolation. The feedwater isolation signal closes the feedwater isolation valves, the feedwater control valves, and trips the turbine and main feedwater pumps. However, trip of the main feedwater pumps is not part of the primary success path for accident mitigation, and no direct credit is taken for the turbine trip function.

An increase in the normal feedwater flow is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of ANS Condition II events.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-8.

15.1.2.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code RETRAN (Refs. 5 and 7). This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level. VIPRE (Ref. 6) is used to determine the core thermal limits to determine DNBR. RETRAN generated statepoints are used as VIPRE boundary conditions to perform the DNB analysis for the HZP cases. For the HFP cases, RETRAN is used to conservatively predict DNBR based on the core thermal limits.

The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition occurs due to a control system malfunction or operator error that allows a feedwater control valve to open fully. Six cases are analyzed, assuming a conservatively large negative moderator temperature coefficient:

a. Accidental opening of one feedwater control valve with the reactor in automatic rod control at full power.

b. Accidental opening of one feedwater control valve with the reactor in manual rod control at full power.

c. Accidental opening of two feedwater control valves with the reactor in automatic rod control at full power.

d. Accidental opening of two feedwater control valves with the reactor in manual rod control at full power.

e. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions.

f. Accidental opening of two feedwater control valves with the reactor just critical at zero load conditions.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. For the feedwater control valve accident at full power cases, a step increase to 200 percent of nominal feedwater flow is assumed to flow into one steam generator for the 1/4 loop malfunction cases, and into each of two steam generators for the 2/4 loop malfunction cases.

b. For the feedwater control valve accident at zero load conditions, an increase in flow from zero to 250 percent of the nominal full load value is assumed to flow into one steam generator for the 1/4 loop malfunction cases, and into each of two steam generators for the 2/4 loop malfunction cases.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 35 °F.

d. No credit is taken for the heat capacity of the RCS and steam generator thick metal structure in attenuating the resulting plant cooldown.

e. The feedwater flow is terminated by a steam generator high-high level trip signal, which initiates feedwater isolation. The feedwater isolation signal closes the feedwater isolation vales, the feedwater control valves, and trips the main feedwater pumps. However, trip of the main feedwater pumps is not part of the primary success path for accident mitigation.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Subsequent to feedwater isolation initiated by a steam generator high-high level trip, the reactor would continue to operate until the low-low steam generator level setpoint is reached. No credit is taken in the licensing basis analysis for a reactor trip on turbine trip. Furthermore, the reactor trip on low-low steam generator level is not modeled as it has no impact on the results. No single active failure will prevent operation of the reactor protection system. A discussion of anticipated transients without scram (ATWS) considerations is presented in Section 15.8.

Results

In the case of an accidental opening of one or two feedwater control valve(s) with the reactor at zero power and the above mentioned assumptions, the event was confirmed to be bounded by the steam system piping failure event described in Section 15.1.5.

At full power, the two-loop feedwater malfunction case with manual rod control resulted in the most limiting minimum DNBR and peak core average thermal power values, both meeting applicable safety analysis limits.

The feedwater flow resulting from one or two fully open control valve(s) is terminated by a steam generator high-high level trip signal, which initiates feedwater isolation. The feedwater isolation signal closes the feedwater isolation valves, the feedwater control valves, and trips the main feedwater pumps. However, trip of the main feedwater pumps is not part of the primary success path for accident mitigation. This prevents continuous addition of the feedwater. Once main feedwater is isolated, the reactor would continue to operate until the low-low steam generator level trip setpoint is reached.

When the low-low steam generator level setpoint is reached the reactor would trip and the control rods will fall into the core terminating the event. It should be noted that the analysis does not model a reactor trip based on steam generator low-low-level.

Transient results presented in Figures 15.1-2A, 15.1-2B, and 15.1-2C for the limiting full power case show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. The DNBR does not drop below the safety analysis limit.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will initially rise. Subsequent fuel melting is precluded by ensuring that the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot. Power levels up to 121% of the applied initial full power value have been shown to meet this criterion. As shown in Figure 15.1-2A, transient power remains below 121% of the full power value, thereby confirming the fuel melting criterion is satisfied.

The transient results show that DNB does not occur at any time during the excessive feedwater flow incident. Thus, the ability of the reactor coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

The calculated sequence of events for the limiting case (flow increased in two loops with manual rod control) is presented in Table 15.1-1.

15.1.2.3 Conclusions

The results of the analysis show that the minimum DNBR encountered for an excessive feedwater addition at power remains above the safety analysis limit; in addition, the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot. The DNBR design basis is described in Section 4.4. At zero power, the event is bounded by the steam system piping failure event described in Section 15.1.5.

15.1.3 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10-percent step load increase or a 5-percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. Steam flow increases greater than 10 percent are analyzed in Sections 15.1.4 and 15.1.5.

This accident could result from either an administrative violation, such as excessive loading by the operator, or an equipment malfunction in the turbine bypass control or turbine speed control.

During power operation, turbine bypass to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for turbine bypass. A single controller malfunction does not cause turbine bypass; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

a. Overpower ΔT
b. Overtemperature ΔT
c. Power range high neutron flux

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-9.

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using the RETRAN code (Ref. 5). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables, including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

a. Reactor control in manual with minimum moderator reactivity feedback

b. Reactor control in manual with maximum moderator reactivity feedback

c. Reactor control in automatic with minimum moderator reactivity feedback

d. Reactor control in automatic with maximum moderator reactivity feedback

For the minimum feedback cases, the core has the most positive moderator temperature coefficient of reactivity, therefore, reductions in coolant temperature will have the least impact on core power. For the maximum feedback cases, the moderator temperature coefficient is assumed at the most negative value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For the cases with automatic rod control, no credit was taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant would quickly stabilize.

A 10-percent step increase in steam demand is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at nominal values, consistent with steady-state full power operation. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure will prevent the reactor protection system from performing its intended function.

Results

Figures 15.1-3 through 15.1-6 illustrate the transient with the reactor in the manual control mode. As expected, for the minimum moderator feedback case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum moderator feedback manually controlled case there is a much larger increase in reactor power, due to the moderator feedback. A reduction in DNBR is experienced, but DNBR remains above the safety analysis limit.

Figures 15.1-7 through 15.1-10 illustrate the transient, assuming that the reactor is in the automatic control mode and no reactor trip signals occur. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the safety analysis limit.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power. Note that due to the measurement errors assumed in the setpoints, it is possible that reactor trip could actually occur for the automatic control cases. The plant would then reach a stabilized condition following the trip.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip may not occur for some of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow. Subsequent fuel melting is precluded by ensuring that the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot. Power levels up to 121% of the applied initial full power value have been shown to meet this criterion. As shown in Figures 15.1-3, 15.1-5, 15.1-7, and 15.1-9, transient power remains below 121% of the full power value, thereby confirming the fuel melting criterion is satisfied.

Since DNB does not occur at any time during the excessive load increase transients, the ability of the reactor coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase incident is shown in Table 15.1-1.

15.1.3.3 Conclusions

The analysis presented above shows that for a 10-percent step load increase, the DNBR remains above the safety analysis limit and the fuel melting criterion is satisfied; the design basis for DNBR is described in Section 4.4. The plant rapidly reaches a stabilized condition, following the load increase.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC RELIEF OR SAFETY VALVE

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single turbine bypass, atmospheric relief, or safety valve. The analyses performed, assuming a rupture of a main steam line, are given in Section 15.1.5.

The steam release, as a consequence of this accident, results in an initial increase in steam flow followed by a decrease in steam flow during the rest of the accident as the steam generator pressure decreases. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA, with offsite power available, and assuming a single failure in the engineered safety features system, there will be no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, atmospheric relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event. See Section 15.0.1 for a discussion of Condition II events.

The following systems provide the necessary protection against an accidental depressurization of the main steam system:

a. Safety injection system actuation from any of the following:

   1. Two out of four low pressurizer pressure signals

   2. Two out of three low steam line pressure signals in any one loop

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SIS

c. Redundant isolation of the main feedwater lines

Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, an SIS will rapidly close all feedwater control valves and back up feedwater isolation valves. The SIS will also trip the main feedwater pumps, and close the feedwater pump discharge valves; however, trip of the main feedwater pumps is not part of the primary success path for accident mitigation.

d. Trip of the fast-acting main steam line isolation valves (designed to close within the limits of Figure B 3.7.2-1, shown in Technical Specification Bases) on:

1. Safety injection system actuation derived from two out of three low steam line pressure signals in any one loop (above Permissive P-11)

2. Two out of three high negative steam pressure rates in any one loop (below Permissive P-11)

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-10.

15.1.4.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of a secondary system steam release are performed in the same manner as the analysis of the steam pipe rupture described in Section 15.1.5.2.

The following conditions are assumed to exist at the time of a secondary system steam release:

a. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks throughout core burnup during each operating cycle is restricted by the insertion limits so that addition of positive reactivity following a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

b. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The K_{eff} versus temperature at 1,110 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11.

c. Minimum capability for injection of RWST boron solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by one centrifugal charging pump delivering flow to the cold leg header and taking suction from the RWST. Reactor coolant seal injection flow is not included in the total core delivery. The volume downstream of the RWST must be swept prior to delivery of boric acid to the reactor coolant loops.

d. The case studied is a steam flow of approximately 240 pounds per second at 1,075 psia (corresponding to 268 pounds per second at 1,200 psia), with offsite power available. This is the maximum capacity of any single turbine bypass, atmospheric relief, or safety valve. Initial hot standby conditions (557° F average coolant temperature) at time zero are assumed, since this represents a conservative initial condition.

   Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam release before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are expected to be less for a steamline release occurring at power.

e. In computing the steam flow, the Moody Curve (Ref. 2) for fL/D = 0 is used.

f. Perfect moisture separation in the steam generator is assumed.

g. Offsite power is assumed, since this would maximize the cooldown.

h. Maximum cold auxiliary feedwater flow is assumed.

i. Four reactor coolant pumps are operating.

Results

The results presented are a conservative indication of the events which would occur, assuming a secondary system steam release, since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.1-12, 15.1-13, and 15.1-13a show the transient results for a steam flow of approximately 240 lb/sec at 1,075 psia (corresponding to 268 lb/sec at 1,200 psia).

The assumed steam release is greater than or equal to the capacity of any single steam dump, atmospheric relief, or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution from the RWST enters the RCS, providing sufficient negative reactivity to prevent core damage. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes.

Since the transient occurs over a period of about 700 seconds, the neglected stored energy is likely to have a significant effect in slowing the cooldown. The calculated time sequence of events for this accident is listed in Table 15.1-1.

15.1.4.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the DNB and fuel temperature design limits are not exceeded. This case is less limiting than the steamline rupture case described in Section 15.1.5.

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow that decreases during the accident as the steam generator pressure decreases.

The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the emergency core cooling system.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. Radiation doses do not exceed the guidelines of 10 CFR 50.67.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB and fuel temperature design bases are not exceeded for any rupture, assuming the most reactive control rod assembly stuck in its fully withdrawn position. The DNB and fuel temperature design bases are discussed in Section 4.4.

A major steam line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Section 15.0.1.3.

The historical methodology for analyzing the reactor core response to excessive secondary steam releases is documented in WCAP-9226 (Reference 3). This WCAP, known as the "Steamline Break Topical," examined the effect of power level (including full power cases), break size, and plant variations for typical three-loop and four-loop Westinghouse-designed PWRs. This WCAP concludes that the largest double-ended steamline rupture at end-of-life, hot zero-power (Mode 2) conditions, with the most reactive RCCA in the fully withdrawn position, bounds all other power levels and other Modes for the post-trip phase of the transient.

Note that the conclusion that the hot-zero power case is the limiting case was based on certain specific protection system performance characteristics credited for "at power" steamline break analyses. Since the WCAP was first issued, plant modifications have been made which might not be bounded by the generic assumptions of the WCAP, such as increases in the Overpower ΔT response times, lead/lag time constant changes, reduced Low Steam Pressure setpoints, etc. As a result, the full power steamline break event, which credits the Overpower ΔT reactor trip function, could be initiated from the plant conditions that are outside the limits of applicability.

A Wolf Creek specific analysis for steamline breaks occurring while the reactor is at power has been performed as described in Section 15.1.6 to confirm the conclusions documented in WCAP-9226 are valid, even with assumed parameters being outside the generic limits established by Westinghouse. The analysis demonstrates the adequacy of the protection systems by showing that with the appropriate actions of these systems, the DNB design basis is satisfied and fuel centerline melting is precluded.

As such, a detailed analysis of this transient with the largest double-ended rupture at the hot-zero power conditions, is presented here.

During startup or shutdown evolutions, when the operator manually blocks the safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than P-11 setpoint (i.e., 1970 psig), the steamline pressure-negative rate-high signal is automatically enabled to provide steamline isolation. For inside containment breaks, steamline isolation may also be provided by the containment pressure High-2 signal and safety injection would be actuated by the

containment pressure High-1 signal. For a steamline break occurring outside containment, an automatic actuation signal for safety injection would not be available. Since the steamline break could occur outside containment, it is possible to have a steamline break event below the P-11 interlock setpoint that does not generate a safety injection actuation of borated ECCS flow. However, prior to manually blocking the safety injection automatic actuation signal below P-11, the RCS and pressurizer are borated to meet shutdown margin requirements at 350°F. The available shutdown margin at 350°F ensures that an RCS cooldown from 350°F to 200°F would not result in a return to criticality; therefore, subcriticality is assured for the steamline break event initiated in Mode 3 with automatic safety injection blocked below P-11.

The following functions provide the protection for a steam line rupture:

a. Safety injection system actuation from any of the following:

   1. Two out of four low pressurizer pressure signals

   2. Two out of three high-1 containment pressure signals

   3. Two out of three low steam line pressure signals in any one loop

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SIS.

c. Redundant isolation of the main feedwater lines

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action, which will close the main feedwater valves following a reactor trip, an SIS will rapidly close all feedwater control valves and back up feedwater isolation valves. The SIS will also trip the main feedwater pumps, and close the feedwater pump discharge valves; however, trip of the main feedwater pumps is not part of the primary success path for accident mitigation.

d. Trip of the fast acting main steam line isolation valves (designed to close within the limits of Figure B 3.7.2-1, shown in Technical Specification Bases) on:

   1. High-2 containment pressure

   2. Safety injection system actuation derived from two out of three low steam line pressure signal in any one loop (above Permissive-11)

   3. Two out of three high negative steam pressure rate in any one loop (below Permissive-11)

Fast-acting isolation valves are provided in each steam line; these valves will fully close within the limits of Figure B 3.7.2-1, shown in Technical Specification Bases, upon receipt of an isolation actuation signal, following a large break in the steam line. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break in any location, no more than one steam generator would experience an uncontrolled blowdown, even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.0.

Flow restrictors are installed in the steam generator outlet nozzle, an integral part of the steam generator. The effective throat area of the nozzles is approximately 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1-2 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since the requirements will vary, depending upon postulated break size and location. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-10.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

a. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The RETRAN computer code (Ref. 5) has been used.

b. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, VIPRE (Ref. 6), has been used to determine if DNB occurs for the core conditions computed in item a above.

The following conditions were assumed to exist at the time of a main steamline break accident:

a. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks throughout core burnup during each operating cycle is restricted by the insertion limits so that addition of positive reactivity following a steam line break accident will not lead to a more adverse condition than the case analyzed.

b. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1,110 psia, corresponding to the negative moderator temperature coefficient used, is shown in Figure 15.1-11.

The effect of power generation in the core on overall reactivity (due to Doppler feedback) is shown in Figure 15.1-14.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. The stuck rod is assumed in the region of the core of lowest temperature.

To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions during the transient for the cases analyzed. This core analysis, performed with the ANC code, considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution, and nonuniform core inlet temperature effects. For cases in which surface boiling occurs in the high heat flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated, including the above local effects for the limiting conditions during the transient. These results verify conservatism, i.e., underprediction of negative reactivity feedback from power generation.

c. Minimum capability for injection of boron solution corresponding to the most restrictive single failure in the safety injection system. The emergency core cooling system consists of four subsystems: 1) the passive SI accumulators, 2) the residual heat removal system, 3) the SI system, and 4) the high head SI system (centrifugal charging pumps). Only the high head SI system (one CCP) is credited for delivery of boron solution to the core for the steam line break accident analysis.

The actual modeling of the safety injection system in RETRAN is as follows:
The flow corresponds to full flow (less seal injection flow) of one charging pump delivering to the RCS via the cold leg header. No credit has been taken for the borated water that must be swept from the lines downstream of the RWST prior to the delivery of the boron solution from the RWST to the reactor coolant loops. The RWST Boron concentration is assumed to be 2400 ppm.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is as follows. Following the generation of a safety injection system actuation signal two seconds after the appropriate setpoint has been reached (to account for the appropriate delays in instrumentation, logic, and signal transport), the appropriate valves begin to operate, and the high head safety injection pump starts. After an additional 25 seconds, the valves are assumed to be in their final position, and the pump is assumed to be at full speed, resulting in a total safety injection system response time of 27 seconds. The volume downstream of the RWST must then be swept prior to delivery of boric acid to the reactor coolant loops. This delay is also included in the calculations.

In cases where offsite power is not available, an additional 12-second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them, resulting in a total safety injection system response time of 39 seconds.

d. Since the steam generators are provided with integral flow restrictors with a throat area of approximately 1.4-square-foot, any rupture with a break area greater than that of the flow restrictor, regardless of location, would have the same effect on the NSSS as approximately a 1.4-square-foot break. The following cases have been considered in determining the core power and RCS transients:

   1. Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

   2. Case (1) above, but with loss of offsite power simultaneous with the steam line break. Loss of offsite power results in reactor coolant pump coastdown.

e. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

   The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

Both cases above assume initial hot standby conditions (557°F average coolant temperature) at time zero, since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis, which assumes no-load condition at time zero.

f. In computing the steam flow during a steam line break, the Moody Curve (Ref. 2) for fL/D = 0 is used.

g. AFW is activated immediately as the transient begins. Based upon the plant specific flow rate calculations, the AFW is assumed to be 1360.13 gpm to the faulted SG and 699.62 gpm to the intact SGs, for a total of 2059.75 gpm.

h. The actuation signal for closure of the main steam isolation valves (MSIVs) is modeled to occur two seconds after a low compensated steamline pressure setpoint is reached. Closure of the MSIVs is delayed for 15 seconds after the actuation signal, followed by an instantaneous closure.

i. The actuation signal for closure of the main feedwater isolation valves (MFIVs) is modeled to occur when the SI signal is initiated, which includes a two second time delay. Closure of the MFIVs is delayed for 15 seconds after the actuation signal, followed by an instantaneous closure.

Results

The calculated sequence of events for both cases analyzed is shown on Table 15.1-1.

The results presented are a conservative indication of the events which would occur, assuming a steam line rupture, since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 15.1-15 through 15.1-20 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (case 1).

Offsite power is assumed to be available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast-acting isolation valves in

the steam lines by high-2 containment pressure signals, low steam line pressure signals or by high negative steam pressure rate signals. Even with the failure of one valve, release is limited by main steam line isolation valve closure for the other steam generators while the one generator blows down. The main steam line isolation valves are designed to be fully closed in less than 15 seconds from receipt of a closure signal.

As shown in Figure 15.1-18, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before the RWST boron solution enters the RCS. A peak core power significantly lower than the nominal full power value is attained.

The calculation assumes that the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation, as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

Figures 15.1-21 through 15.1-26 show the salient parameters for case 2, which corresponds to case 1 discussed above, but with the additional loss of offsite power at the time the steam line break occurs. The safety injection system delay time for this case includes 12 seconds to start the diesels and to load the necessary safety injection equipment onto them, in addition to 25 seconds to start the safety injection pump and open the valves. Criticality is achieved later, and the core power increase is slower than in the similar case with offsite power available. The ability of the faulted steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full power value.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for the dissipation of decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the atmospheric relief or safety valves.

Fuel Damage Criteria

The DNB and fuel temperature design bases were evaluated for both of these cases. The results of the DNBR and kW/ft calculations for the limiting SLB case (with offsite power available) demonstrated that the minimum DNBR remains above the applicable limit value and the peak kW/ft value remains below the value that would cause fuel centerline melt. Thus, it is concluded that the DNB and fuel temperature design bases are met for the steam line break event initiated from zero power.

15.1.5.3 Radiological Consequences

15.1.5.3.1 Method Of Analysis

15.1.5.3.1.1 Physical Model

The radiological consequences of a MSLB inside the containment are less severe than the one outside the containment because the radioactivity released will be held up inside the containment, allowing decay and plateout of the radionuclides. To evaluate the radiological consequences due to a postulated MSLB (outside the containment), it is assumed that there is a complete severance of a main steam line outside the containment.

It is also assumed that there is a simultaneous loss of offsite power, resulting in reactor coolant pump coastdown. The safety injection system is actuated and the reactor trips.

The main steam line isolation valves, their bypass valves, and the steam line drain valves isolate the steam generators and the main steam lines upon a signal initiated by the engineered safety features actuation system under the conditions of high steam negative pressure rate or low steam line pressure. The main steam isolation valves are installed in the main steam lines from each steam generator downstream from the safety and atmospheric relief valves outside the containment. The break in the main steam line is assumed to occur outside of the containment. The affected steam generator (steam generator connected to a broken steam line) blows down completely. The steam is vented directly to the atmosphere.

Each of the steam generators incorporates integral flow restrictors, which are designed to limit the rate of steam blowdown from the steam generators following a rupture of the main steam line. This, in turn, reduces the cooling rate of the reactor coolant system to preclude departure from nucleate boiling (DNB).

In case of loss of offsite power, the remaining steam generators are available for dissipation of core decay heat by venting steam to the atmosphere via the atmospheric relief valves. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently so that the RHR system can be utilized to cool the reactor.

15.1.5.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.1-3 and 15A-1.

Tables 15B-1 and 15B-4 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

The assumptions used to determine the concentrations of radioactive isotopes within the secondary system for this accident are as follows:

a. The secondary system initial concentrations of radioactive isotopes are assumed to be the dose equivalent of 0.1 μCi/gm of I-131.

b. A primary-to-secondary leakage rate of 1 gpm is assumed to exist and is assumed to be in the faulted steam generator.

c. The reactor coolant concentration of radioactive isotopes is determined by two methods, and both cases are analyzed. These are:

   1. The initial reactor coolant concentrations of radioactive isotopes are assumed to be the dose equivalent of 1.0 μCi/gm of I-131 with an iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. This increased rate of transfer from the fuel into the coolant is assumed for 8 hours.

   2. An assumed reactor coolant concentration of radioactive isotopes with a dose equivalent of 60 μCi/gm of I-131 as a result of preaccident iodine spikes.

d. The reactor coolant concentrations of noble gas correspond to 500 micro-Ci/gm DE Xe-133.

e. Partition factors used to determine the secondary system activities are given in Table 15.1-3 for iodines and alkali metals.

f. The reactor coolant concentrations of alkali metals correspond to 1-percent failed fuel as provided in Table 11.1-5.

g. The secondary system initial concentration of alkali metals is 10% of the reactor coolant concentrations.

The following specific assumptions and parameters are used to calculate the activity release:

a. Offsite power is lost, resulting in reactor coolant pump coastdown.

b. No condenser air removal system release and no normal operating steam generator blowdown is assumed to occur during the course of the accident.

c. Twelve hours after the occurrence of the accident, the residual heat-removal system (RHRS) operation matches decay heat and steam releases are terminated.

d. After the accident, the primary-to-secondary leakage continues for 34 hours, at which time the reactor coolant system is cooled to 212°F such that there would be no flashing of the leaked fluid.

e. The faulted steam generator (steam generator connected to the broken steam line) is allowed to blow down completely.

f. Steam release to the atmosphere and the associated activity release from the safety and atmospheric relief valves on the intact steam generators are terminated 12 hours after the accident, when the RHRS is activated to complete cooldown.

g. The amount of noble gas activity released is equal to the amount present in the reactor coolant, which leaks to the secondary during the accident. The amount of iodine and alkali metal activity released is based on the activity present in the secondary system and the amount of leaked reactor coolant which is entrained in the steam that is discharged to the environment via the safety and atmospheric relief valves and the broken steam line. Partition factors used for the unaffected steam generators after the accident occurs are given in Table 15.1-3. An iodine partition factor of 1 is used for the faulted steam generator.

h. The activity released from the broken steam line and the safety and atmospheric relief valves during the accident is immediately vented to the atmosphere.

15.1.5.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.1.5.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to a postulated MSLB, the activity released from the faulted steam generator (steam generator connected to the broken steam line) is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and atmospheric relief valves up to the time initiation of the RHRS can be accomplished.

Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated MSLB.

15.1.5.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant iodine and noble gas activities are based on the Technical Specification limits which are significantly higher than the reactor coolant activities associated with normal operating conditions.

b. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation (150 gpd/SG). Furthermore, it was conservatively assumed that all 1 gpm leakage is to the faulted steam generator only and a total of 450 gpd is to the intact steam generators.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

d. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

15.1.5.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the more limiting LOCA analysis, as discussed in Section 15.6.5.4.3.1. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated MSLB releases.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed, using assumptions and models described. The TEDE doses have been analyzed for the limiting 2 hour duration dose at the exclusion area boundary, for the duration of the accident (0 to 34 hrs) at the low-population zone outer boundary, and in the control room for 30 days. The results are listed in Table 15.1-4. The resultant offsite doses are within the acceptance limits, a small fraction (10 percent) of exposure limits of 10CFR50.67 for the case of concurrent iodine spike and the exposure limits of 10CFR50.67 for the case of pre-accident iodine spike. The resultant control room doses are within the acceptance criteria limits of 10CFR50.67.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the DNB and fuel temperature design bases are met for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position.

15.1.6 STEAM SYSTEM PIPING FAILURE AT FULL POWER

15.1.6.1 Identification of Causes and Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the RCS via the steam generators. This results in a reduction in RCS temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of-cycle conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow. Depending on the break size, a reactor trip may occur due to overpower conditions or as a result of a steam line break protection function actuation.

The steam system piping failure accident analysis described in Section 15.1.5 is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod in the fully withdrawn position, out of the core. Such a condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin or after the plant has been tripped automatically by the reactor protection system or manually by the operator. For an at-power break, the analysis of Section 15.1.5 represents the limiting condition with respect to core protection for the time period following reactor trip. The purpose of this section is to describe the analysis of a steam system piping failure occurring from at-power initial conditions, to demonstrate that core protection is maintained prior to and immediately following reactor trip.

Depending on the size of the break, this event is classified as either an ANS Condition III (infrequent fault) or Condition IV (limiting fault), as defined in Section 15.0.1. The acceptance criteria for this event are defined in Section 15.1.5.

15.1.6.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe rupture was performed in the following stages:

a. The RETRAN computer code (Refs. 5 and 7) was used to calculate the transient core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break.

b. A detailed core analysis was performed using the ANC code (Ref. 4) to confirm the validity of the RETRAN predicted reactivity feedback model. The core models developed in ANC were also used to calculate the power peaking factors for input to the DNB analysis and the calculation of the peak linear heat rate (kW/ft).

c. The detailed thermal and hydraulic digital computer code VIPRE (Ref. 6) was used to calculate the DNBR for the limiting time in the transient.

This accident is analyzed with the Revised Thermal Design Procedure (Ref. 1). Plant characteristics and initial conditions are discussed in Section 15.0.3.

The following assumptions are made in the transient analysis:

a. Initial Conditions – The initial NSSS power was conservatively modeled to be 3651 MWt, which includes all applicable uncertainties. The full power condition is more limiting than part power. The initial reactor coolant temperature and RCS pressure are assumed to be at their nominal full power values. The RCS Minimum Measured Flow is used. The initial loop flows were assumed to be symmetric. Uncertainties for the initial conditions of pressurizer pressure, reactor vessel average temperature, and reactor coolant flow are statistically accounted for in the DNBR limit calculated using the Revised Thermal Design Procedure methodology (Ref. 1).

   The full power RCS average temperature range is from 570.7°F to 588.4°F. Since the full power steam line rupture event is primarily a DNB event, assuming a maximum RCS average temperature is limiting. Therefore, an initial RCS average temperature of 588.4°F was assumed.

   The main feedwater analytical temperature range is from 400°F to 448.6°F. A higher feedwater temperature is more limiting for this event. Thus, a feedwater temperature of 448.6°F was assumed.

   To maximize the primary to secondary heat transfer rate, 0 percent steam generator tube plugging is assumed.

b. Break size - A spectrum of break sizes was analyzed. Small breaks do not result in a reactor trip; in this case core power stabilizes at an increased level corresponding to the increased steam flow. Intermediate size breaks result in a reactor trip on overpower ΔT as a result of the increasing core power. Larger break sizes result in a reactor trip soon after the break from the safety injection (SI) signal actuated by low steam line pressure, which includes lead/lag dynamic compensation. The limiting break size is the largest break that does not trip on a low steam pressure SI signal.

c. Break flow - In computing the steam flow during a steam line break, the Moody curve (Ref. 2) for $fL/D = 0$ is used.

d. Reactivity Coefficients - Maximum moderator reactivity feedback and minimum Doppler power feedback are assumed to maximize the power increase following the break.

e. Protection System - The protection system features that mitigate the effects of a steam line break are described in Section 15.1.5. This analysis only considers the initial phase of the transient from at-power conditions. Protection in this phase of the transient is provided by reactor trip, if necessary. Section 15.1.5 presents the analysis of the bounding transient following reactor trip, where other protection system features are actuated to mitigate the effects of the steam line break.

f. Control Systems - In general, the results would be less severe as a result of normal control system operation. Therefore, the mitigation effects of control systems have been ignored in the analysis. However, the main feedwater control system is assumed to operate in that feedwater flow is assumed to match the steam flow prior to reactor trip.

Results

A spectrum of steam line break sizes was analyzed. The results show that for small break sizes (less than 0.44 ft^2) a reactor trip is not generated. In this case, the event is similar to an excessive load increase event as described in Section 15.1.3. The core reaches a new equilibrium condition at a higher power equivalent to the increased steam release. For increasing break sizes up to and including 1.04 ft^2 the power increase results in a reactor trip on overpower ΔT. For break sizes above 1.04 ft^2 a reactor trip is generated within a few seconds of the break on the lead/lag compensated low steam pressure safety injection actuation signal.

The limiting case for demonstrating DNB protection is the 1.04 ft^2 break, the largest break size that results in a trip on overpower ΔT. The time sequence of events for this case is shown in Table 15.1-1. Figures 15.1-27 through 15.1-30 show the transient response.

The results of the analysis of a major rupture of a main steam pipe event at full power demonstrate that the DNB design basis is met. The calculated minimum DNBR is well above the limit value. In addition, the peak linear heat generation rate (expressed in kW/ft) does not exceed the value that would cause fuel centerline melt. The pressure limits for the primary and secondary systems are not challenged for this accident because the pressures in these systems each decrease from their initial values during the transient. The steam pressure does not increase significantly following turbine trip. This event does not adversely affect the core or the RCS, and all applicable acceptance criteria are met.

15.1.6.3 Conclusions

A detailed DNB and kW/ft analysis is performed as part of each cycle-specific reload evaluation, using radial and axial core peaking factors that are dependent on the cycle-specific loading pattern. The analysis concludes that the DNB and fuel centerline temperature design bases are met for the limiting case. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the minimum DNBR remains above the limit value for any rupture occurring from an at-power condition prior to and immediately following a reactor trip.

15.1.7 References

1. Friedland, A. J., and Ray, S., “Revised Thermal Design Procedure,” WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-proprietary), April 1989.

2. Moody, F. J., “Transactions of the ASME, Journal of Heat Transfer,” Figure 3, page 134, February 1965.

3. Wood, D. C., and Hollingsworth, S. D., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226-P-A, Revision 1 (Proprietary), WCAP-9227-P-A, Revision 1 (Non-proprietary), February 1998.

4. Davidson, S. L. (Ed.), et al., “ANC: A Westinghouse Advanced Nodal Computer Code,” WCAP-10965-P-A (Proprietary) and WCAP-10966-A (Non-proprietary), September 1986.

5. Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), April 1999, and WCAP-15234-A (Non-proprietary), May 1999.

6. Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-proprietary), October 1999.

7. McFadden, J. H., et al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
Feedwater system malfunctions that result in a decrease in feedwater flow at full power		
	Low pressure feedwater heater bypass valves open	0.01
	OPΔT setpoint reached in two loops	35.2
	Reactor trip (rod motion starts) on OPΔT	38.2
	Minimum DNBR reached	38.5
	Low pressurizer pressure safety injection setpoint reached	83.1
	Feedwater isolation initiated	100.0
Feedwater system malfunctions that result in an increase in feedwater flow at full power		
	Two main feedwater control valves fail fully open	0.01
	Steam generator level reaches high-high setpoint - loop 1	36.9
	Turbine trip initiated on high-high steam generator level	39.3
	Minimum DNBR occurs	41.5
	Feedwater isolation due to high-high steam generator level initiated	53.8
Excessive increase in secondary steam flow		
1. Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Minimum DNBR reached	6.7

TABLE 15.1-1 (Sheet 2)

Accident	Event	Time (sec)
	Equilibrium conditions reached (approximate time only)	~300
	Peak heat flux reached	382.5
2. Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	~350
	Peak heat flux reached	362.8
	Minimum DNBR reached	395.7
3. Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Peak heat flux reached	265.6
	Equilibrium conditions reached (approximate time only)	~300
	Minimum DNBR reached	377.6
4. Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Peak heat flux reached	368.5
	Minimum DNBR reached	398.2
	Equilibrium conditions reached (approximate time only)	~400
Inadvertent opening of a steam generator atmospheric relief or safety valve		
	Inadvertent opening of one main steam safety or atmospheric relief valve	0.0

TABLE 15.1-1 (Sheet 3)

Accident	Event	Time (sec)
	Auxiliary feedwater actuation	0.0
	Low pressurizer pressure (SIS) setpoint reached	216.2
	SI signal generated	218.2
	Feedwater isolation complete	233.2
	SI flow initiated	243.2
	Criticality attained	306.5
	RWST boron solution reaches core	565.8
	Peak reactor power	569.3
	Core becomes subcritical	587.3
Steam system piping failure at zero power		
1. Case 1 (offsite power available)	Steam line ruptures	0.0
	Low steam line pressure setpoint reached	0.6
	SI signal generated	2.6
	Pressurizer empties	15.8
	Steam line isolation and feed line isolation complete	17.6
	Criticality attained	20.8
	SI flow initiated	27.6
	Accumulators begin to inject (unborated water)	116.5
	Peak core thermal power occurs	316.5
	RWST boron solution reaches core	316.8
	Core becomes subcritical	324.0

TABLE 15.1-1 (Sheet 4)

Accident	Event	Time (sec)
2. Case 2 (concurrent loss of offsite power)	Steam line ruptures	0.0
	Low steam line pressure setpoint reached	0.6
	SI signal generated	2.6
	Steam line isolation and feed line isolation complete	17.6
	Pressurizer empties	18.0
	Criticality attained	28.3
	SI flow initiated	39.7
	RWST boron solution reaches core	363.8
	Peak core thermal power occurs	376.8
	Core becomes subcritical	381.8
Steam system piping failure at full power (1.04 ft^2 break)		
	Steam line ruptures	0.0
	Overpower ΔT reactor trip setpoint reached (in two loops)	17.7
	Rods begin to drop	20.7
	Peak core heat flux occurs	21.5
	Minimum DNBR Occurs	21.5

TABLE 15.1-2

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAM LINE

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguard actuation channels, including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded)	Auxiliary feedwater system, including pumps, water supply steam generator atmospheric relief valves and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident)	Steam generator atmospheric relief valves Control for defeating automatic safety injection actuation during a cooldown and depressurization (i.e., SIS is reset)
Safety injection system, including the pumps, the refueling water storage tank, the boron injection tank, and the system valves and piping	Containment air coolers Capability for obtaining reactor coolant system sample	Residual heat removal system, including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the reactor coolant system in a cold shutdown condition
Containment spray system	Capability for boration to required hot standby concentration	Capability to depressurize the reactor coolant system to allow residual heat removal system operation
Diesel generators and emergency power distribution equipment		
Essential service water system		
Containment air coolers		

TABLE 15.1-2 (Sheet 2)

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Auxiliary feedwater system, including pumps, water supplies, piping and valves		
Main feedwater control valves (trip closed feature)		
Bypass feedwater control valves (trip closed feature)		
Primary and secondary safety valves		Capability to borate to cold shutdown concentration
Associated pump room coolers		
Main feedwater isolation valves (trip closed feature)		
Main steamline isolation valves (trip closed feature)		
Main steamline isolation valve bypass valves (trip closed feature)		
Steam generator blowdown isolation valves (automatic closure feature)		
Batteries (Class 1E)		

TABLE 15.1-2 (Sheet 3)

Short-Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Control room ventilation		
Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting		
Post-accident monitoring system*		

*See Section 7.5 for a discussion of the post-accident monitoring systems.

TABLE 15.1-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

I. Source Data:

a.	Core power level, Mwt	3637 (includes uncertainty)
b.	Steam generator tube leakage, gpm	1 to the faulted steam generator and 450 gpd total to the remaining steam generators
c.	Reactor coolant iodine activity:	
	1) Case 1	Dose equivalent of 1.0 μCi/gm of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500
	2) Case 2	An assumed pre-accident iodine spike which has resulted in the dose equivalent of 60 μCi/gm of I-131
d.	Reactor coolant noble gas activity:	
	1) Case 1	Dose equivalent of 500 μCi/gm of Xe-133
	2) Case 2	Dose equivalent of 500 μCi/gm of Xe-133
e.	Reactor coolant alkali metal activity:	
	1) Case 1	Based on 1-percent failed fuel as provided in Table 11.1-5
	2) Case 2	Based on 1-percent failed fuel as provided in Table 11.1-5
f.	Secondary system initial iodine activity	Dose equivalent of 0.1 μCi/gm of I-131
g.	Secondary system initial alkali metal activity	Based on the same ratio of primary-to-secondary system activity as iodine (i.e., 10% of primary side activity)
h.	Iodine partition factors	
	1) Faulted steam generator	1.0
	2) Intact steam generators	0.01
i.	Alkali metal partition factors	
	1) Faulted steam generator	1.0
	2) Intact steam generators (based on full power moisture carryover of 0.25%)	0.0025

TABLE 15.1-3 (Sheet 2)

	j. Reactor coolant mass, lbs	3.99E+5
	k. Steam generator mass	
	1) Faulted steam generator, lbs	165,000
	2) Each intact steam generator, lbs	82,333
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data:	
	a. Faulted steam generator	
	1) Initial steam release, 0-2 min, lbs	165,000
	2) Reactor coolant release, 0-34 hr, lbs	17,018
	b. Intact steam generators	
	1) Steam release, 0-2 hr, lbs	419,340
	2) Steam release, 2-12 hr, lbs	1,310,269
IV.	Control Room	
	a. Control room switchover to emergency mode, sec	See Table 15A-5

TABLE 15.1-4

RADIOLOGICAL CONSEQUENCES OF A
MAIN STEAM LINE BREAK

	Dose (rem TEDE)
CASE 1, 1.0 μCi/gm I-131 equivalent w/I spike	
Exclusion area boundary (7.8-9.8 hr)	5.8E-1
Low-population zone outer boundary (duration)	5.4E-1
Control Room (30 days)	4.8E+0
CASE 2, 60 μCi/gm I-131 equivalent	
Exclusion area boundary (0-2 hr)	2.0E-1
Low population zone outer boundary (duration)	1.2E-1
Control Room (30 days)	4.5E+0

FIGURE 15.1-1 HAS BEEN DELETED

FIGURE 15.1-1

NUCLEAR POWER, AND PRESSURIZER
PRESSURE TRANSIENTS
FOR FEEDWATER CONTROL VALVE
MALFUNCTION

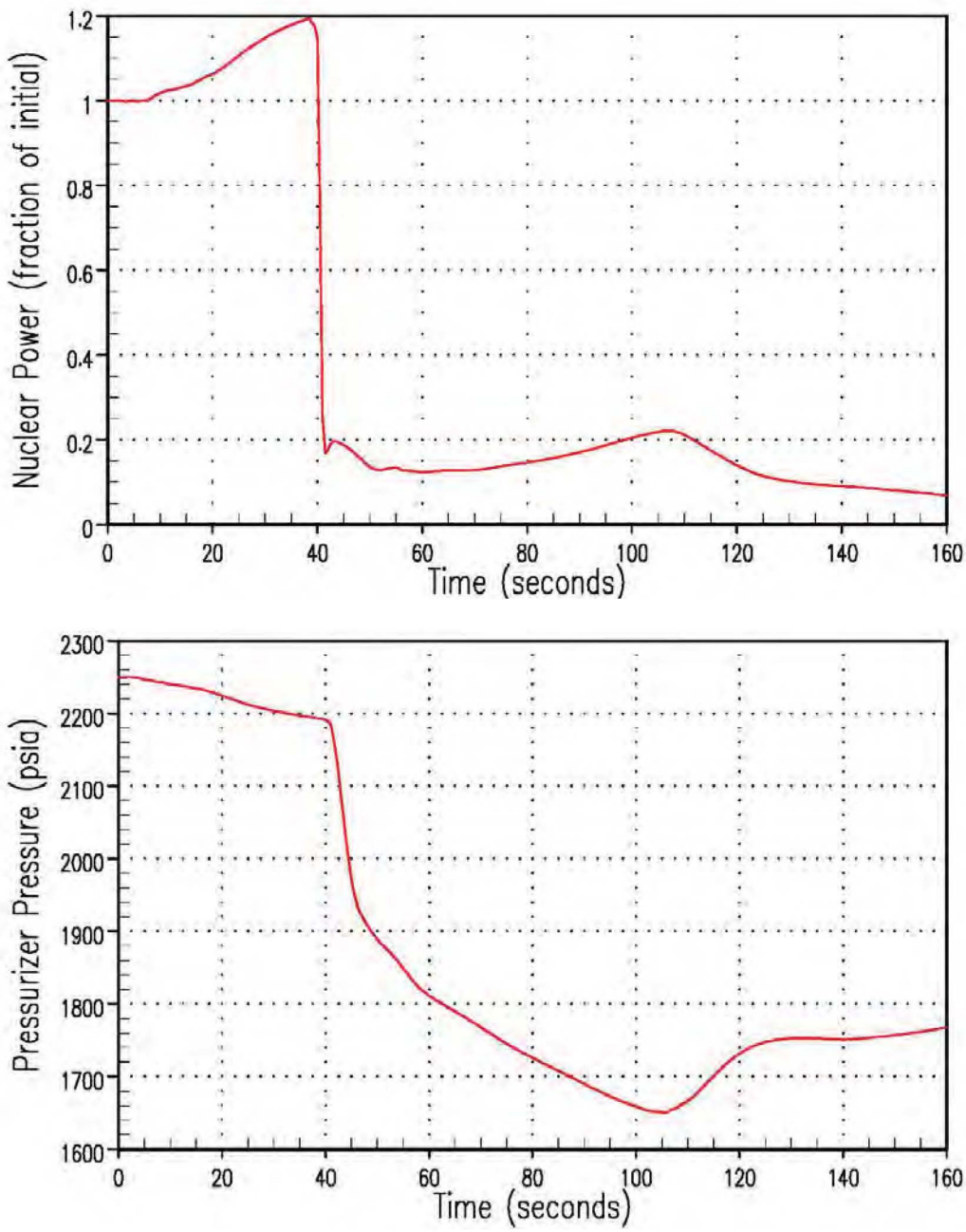

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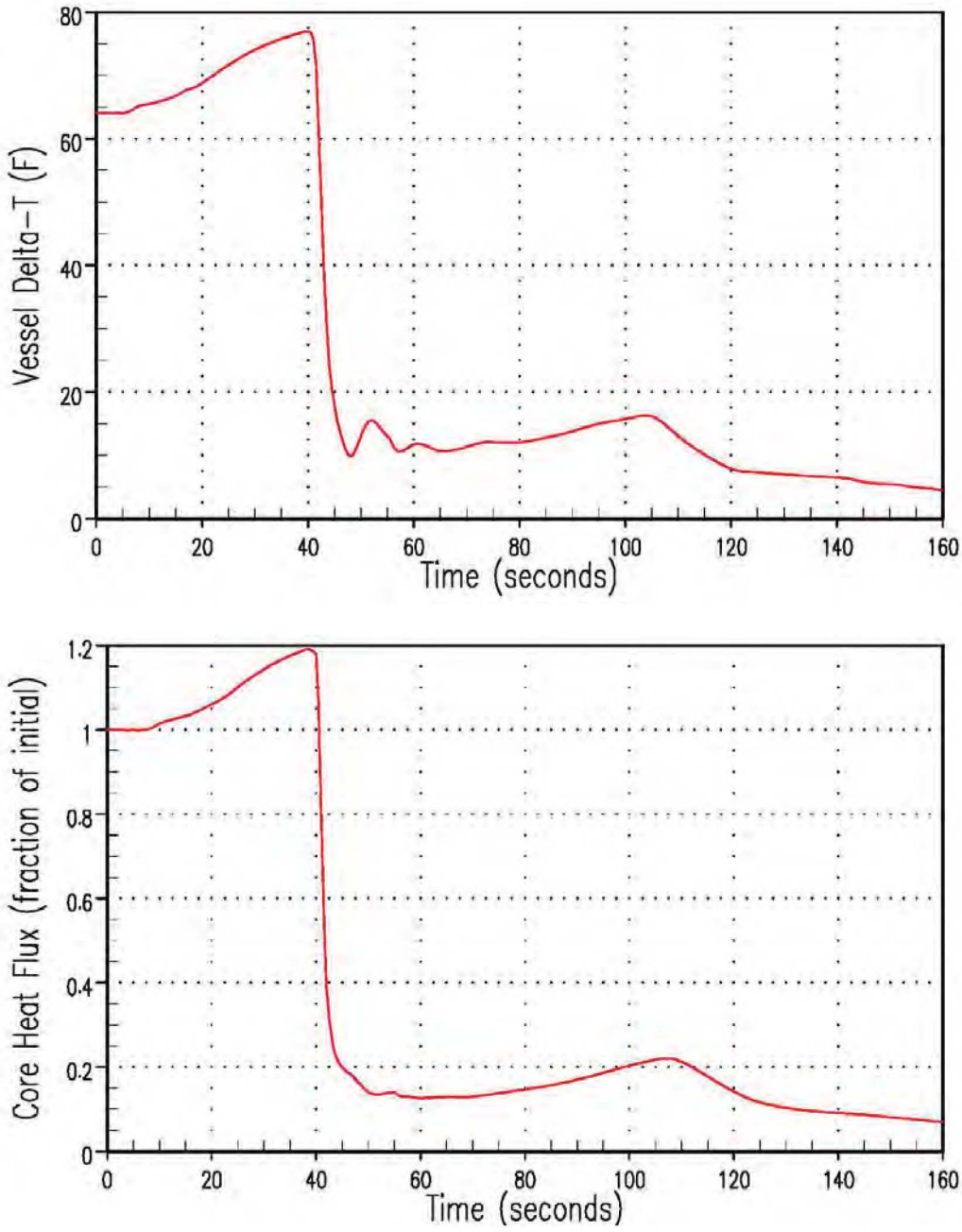
FIGURE 15.1-1A NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR FEEDWATER MALFUNCTION TEMPERATURE REDUCTION



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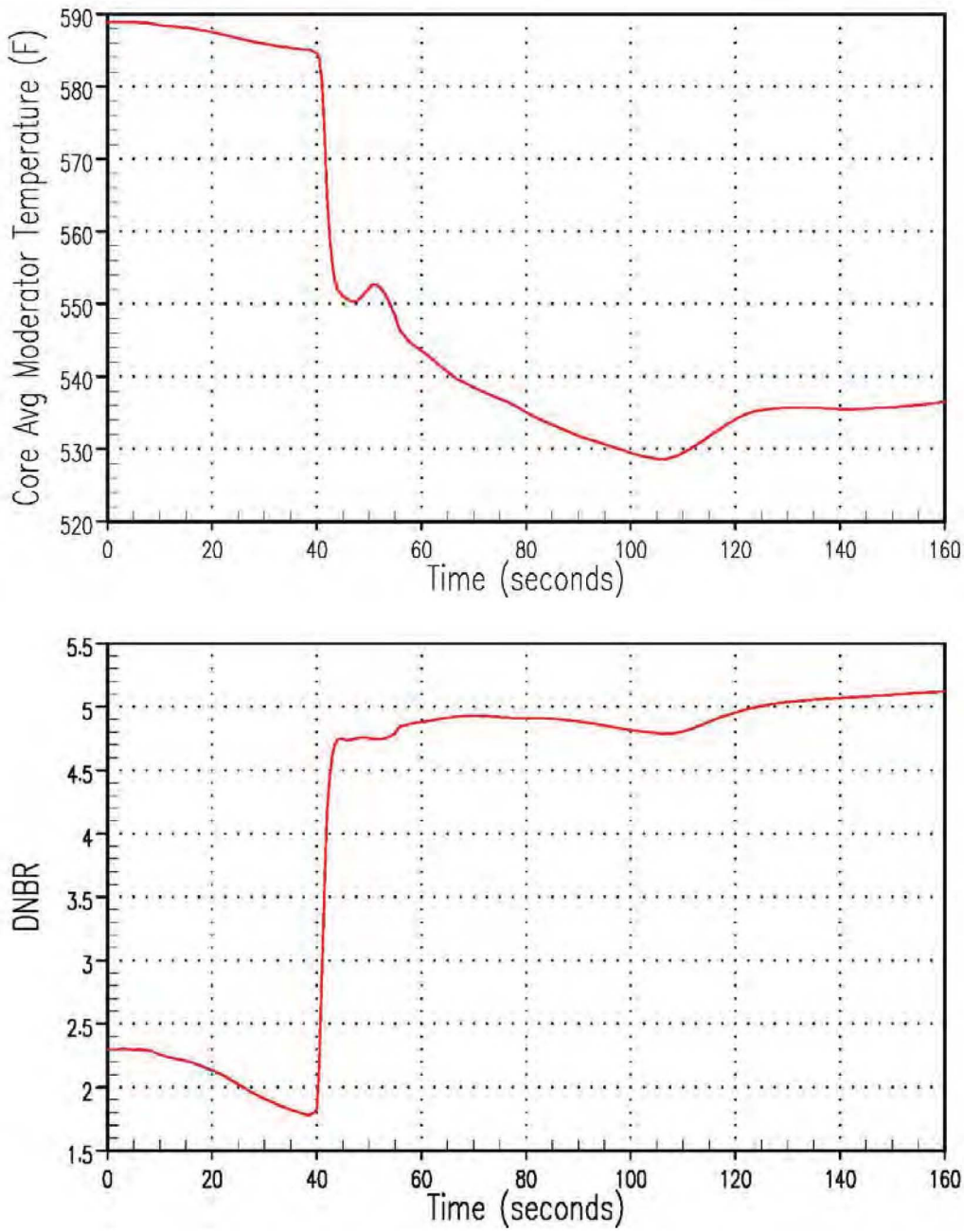
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-1B REACTOR COOLANT LOOP ΔT AND CORE HEAT FLUX TRANSIENTS FOR FEEDWATER MALFUNCTION TEMPERATURE REDUCTION



REV. 34

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.1-1C CORE AVERAGE TEMPERATURE AND DNBR TRANSIENTS FOR FEEDWATER MALFUNCTION TEMPERATURE REDUCTION

FIGURE 15.1-2 HAS BEEN DELETED

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FIGURE 15.1-2

REACTOR COOLANT LOOP ΔT,
AND CORE HEAT FLUX
TRANSIENTS FOR FEEDWATER CONTROL
VALVE MALFUNCTION

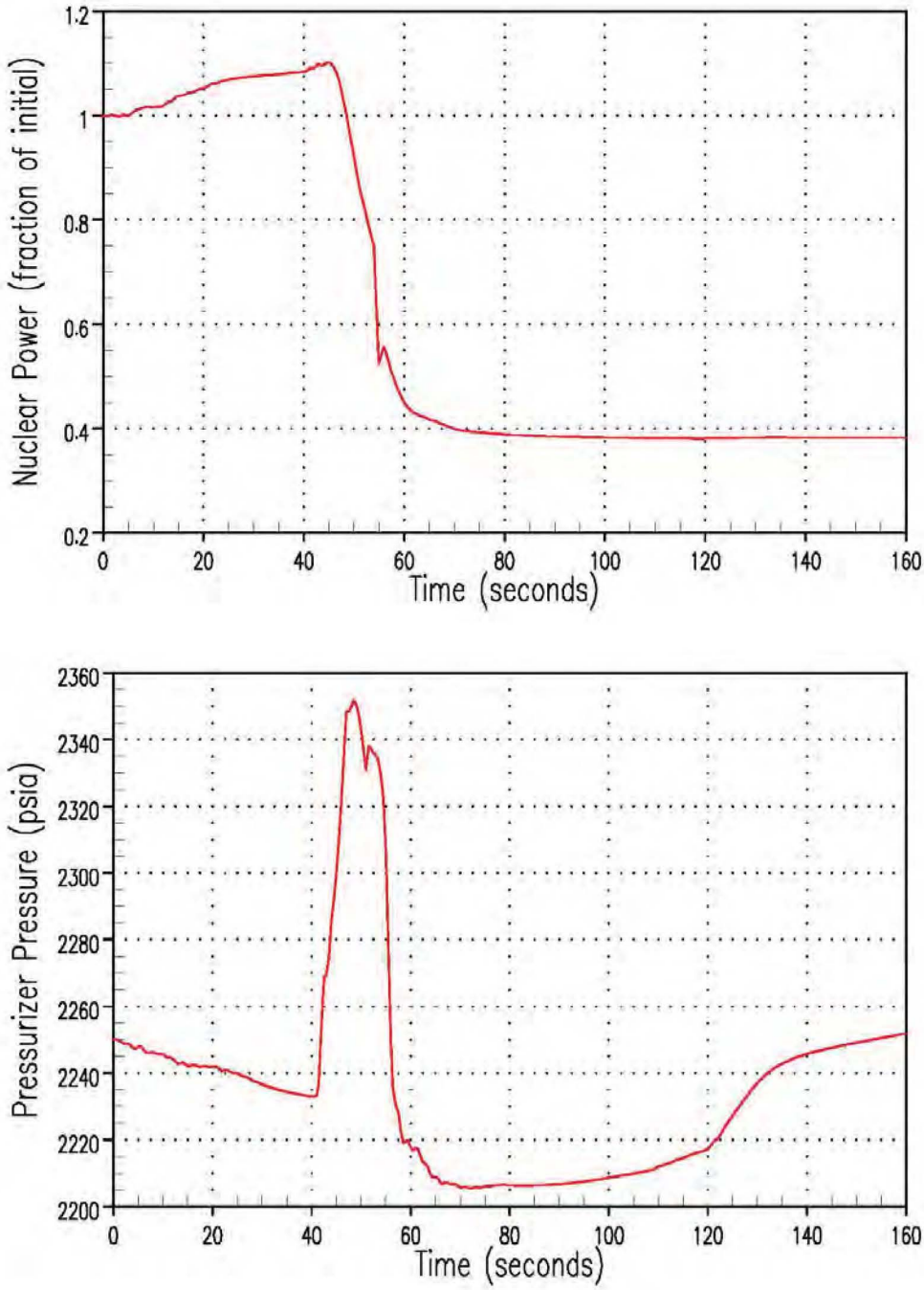

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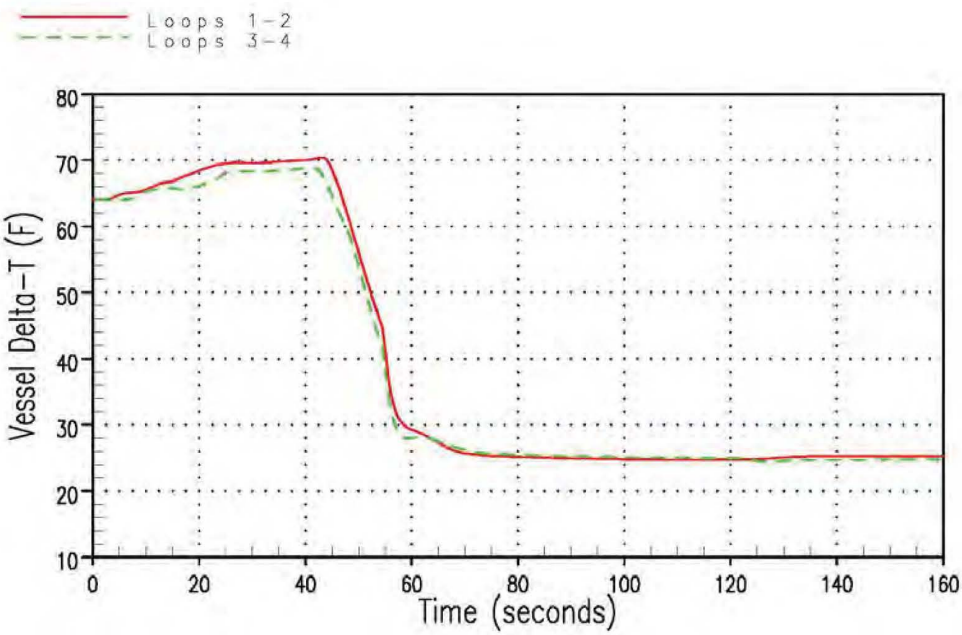
FIGURE 15.1-2A NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION



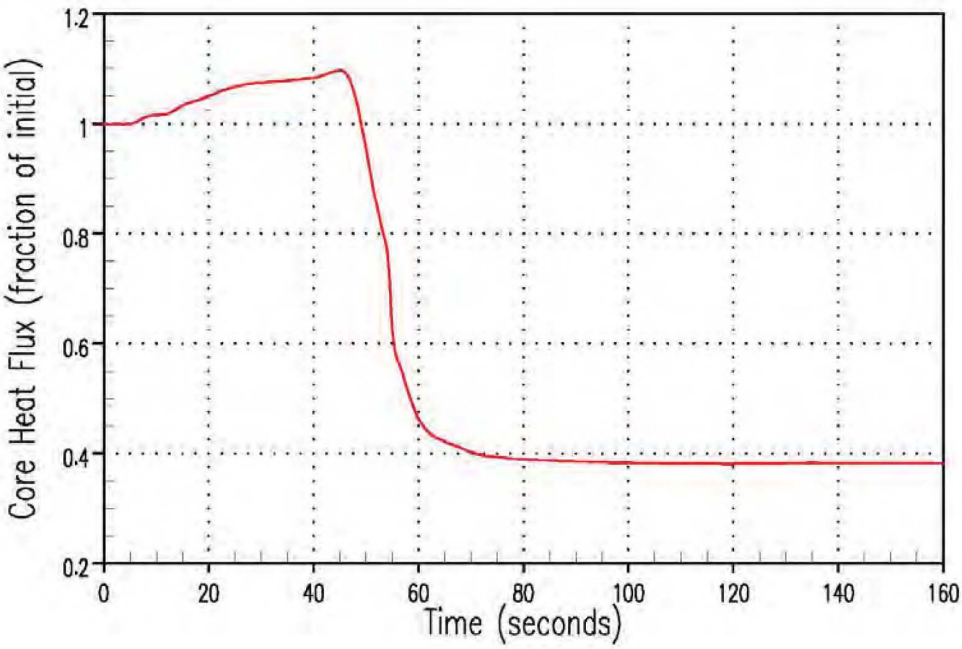


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FIGURE 15.1-2B

REACTOR COOLANT LOOP ΔT AND CORE HEAT FLUX TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

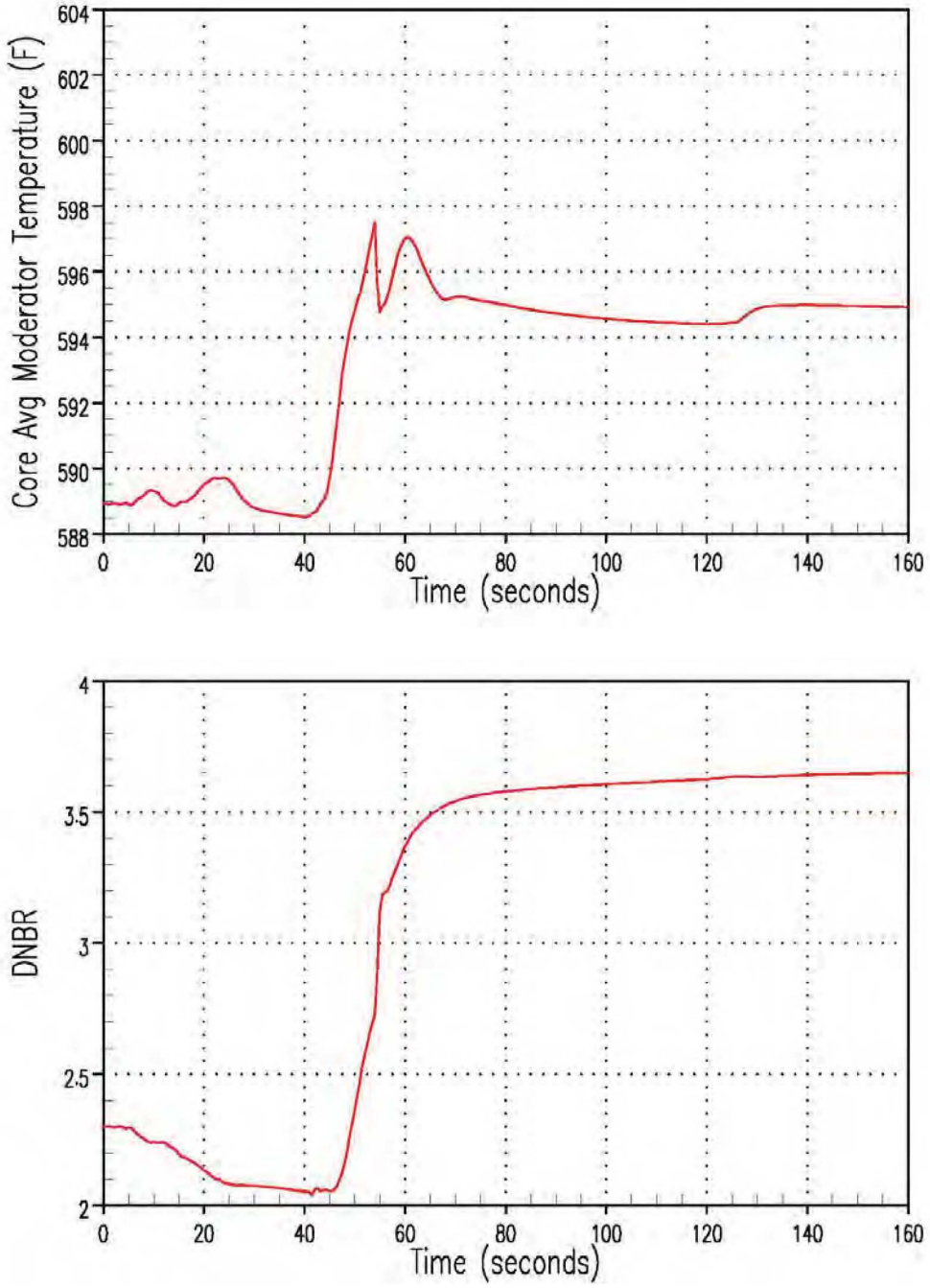

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FIGURE 15.1-2C

CORE AVERAGE TEMPERATURE AND DNBR TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

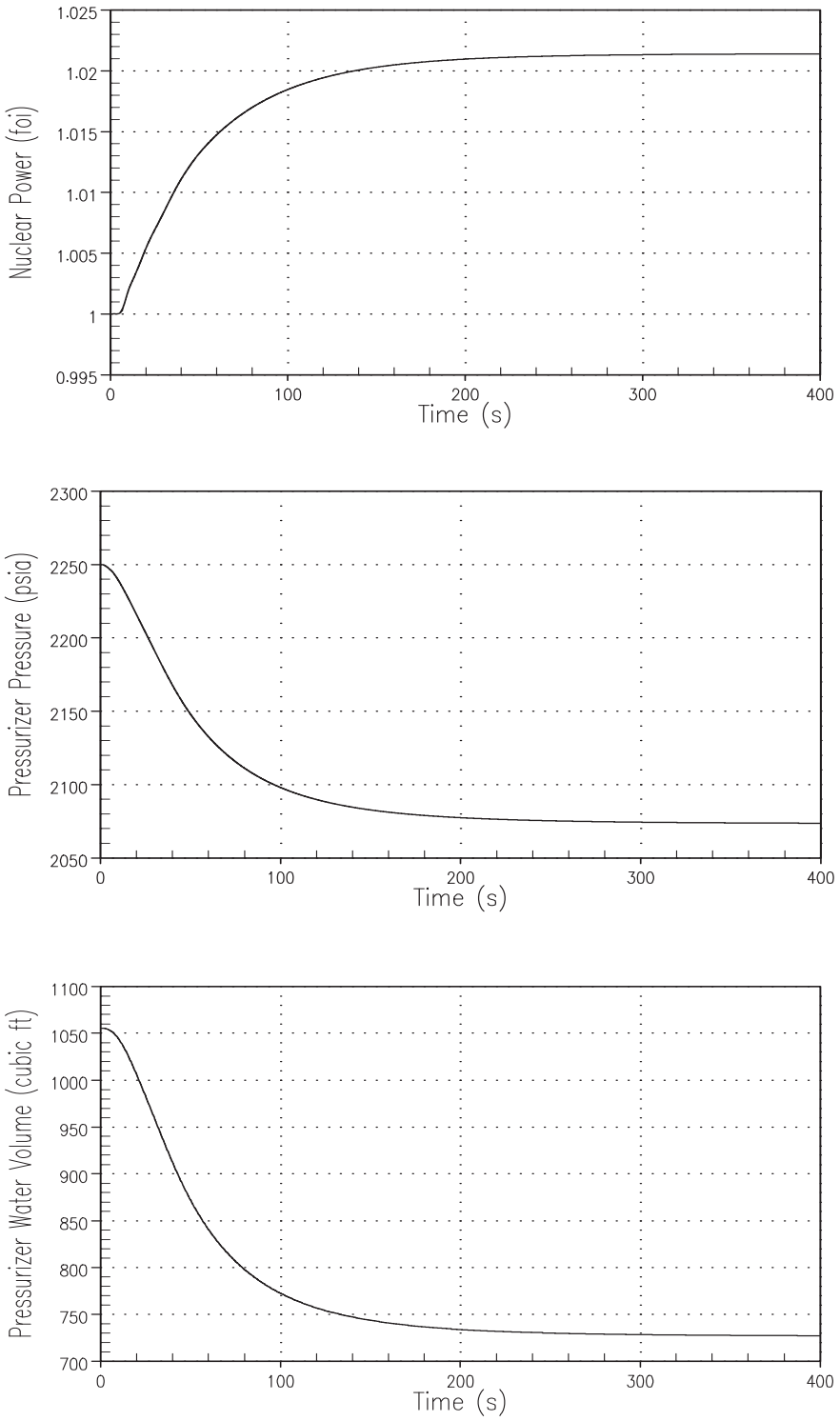

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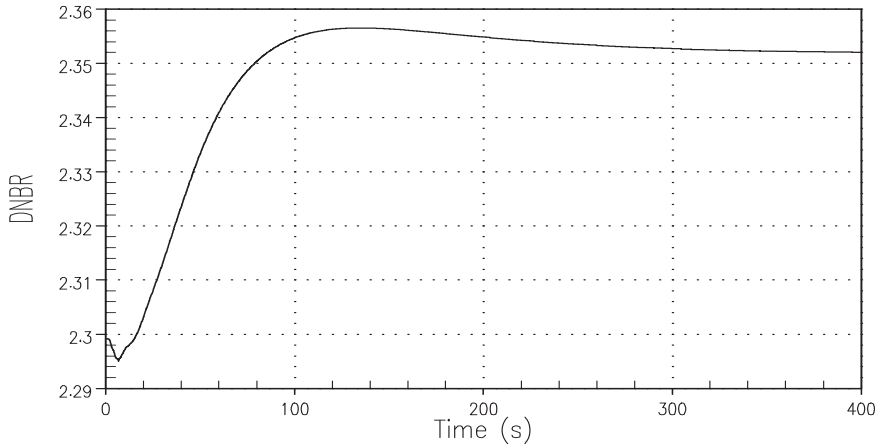
FIGURE 15.1-3

NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL



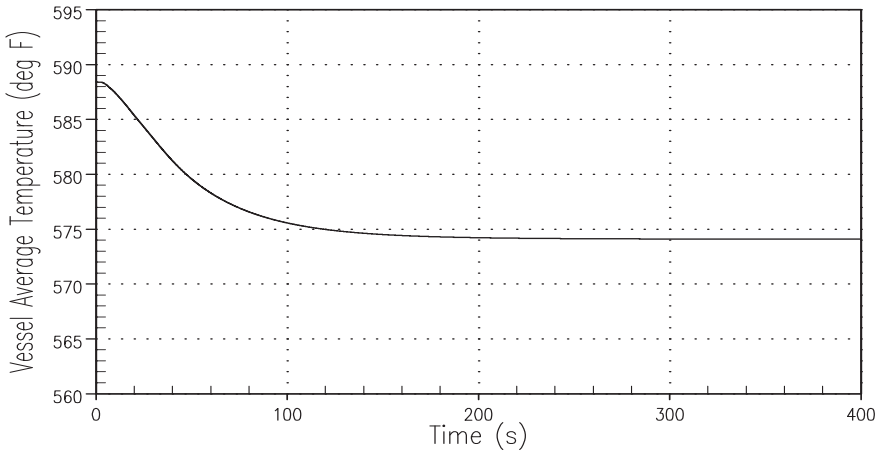


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FIGURE 15.1-4 DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL

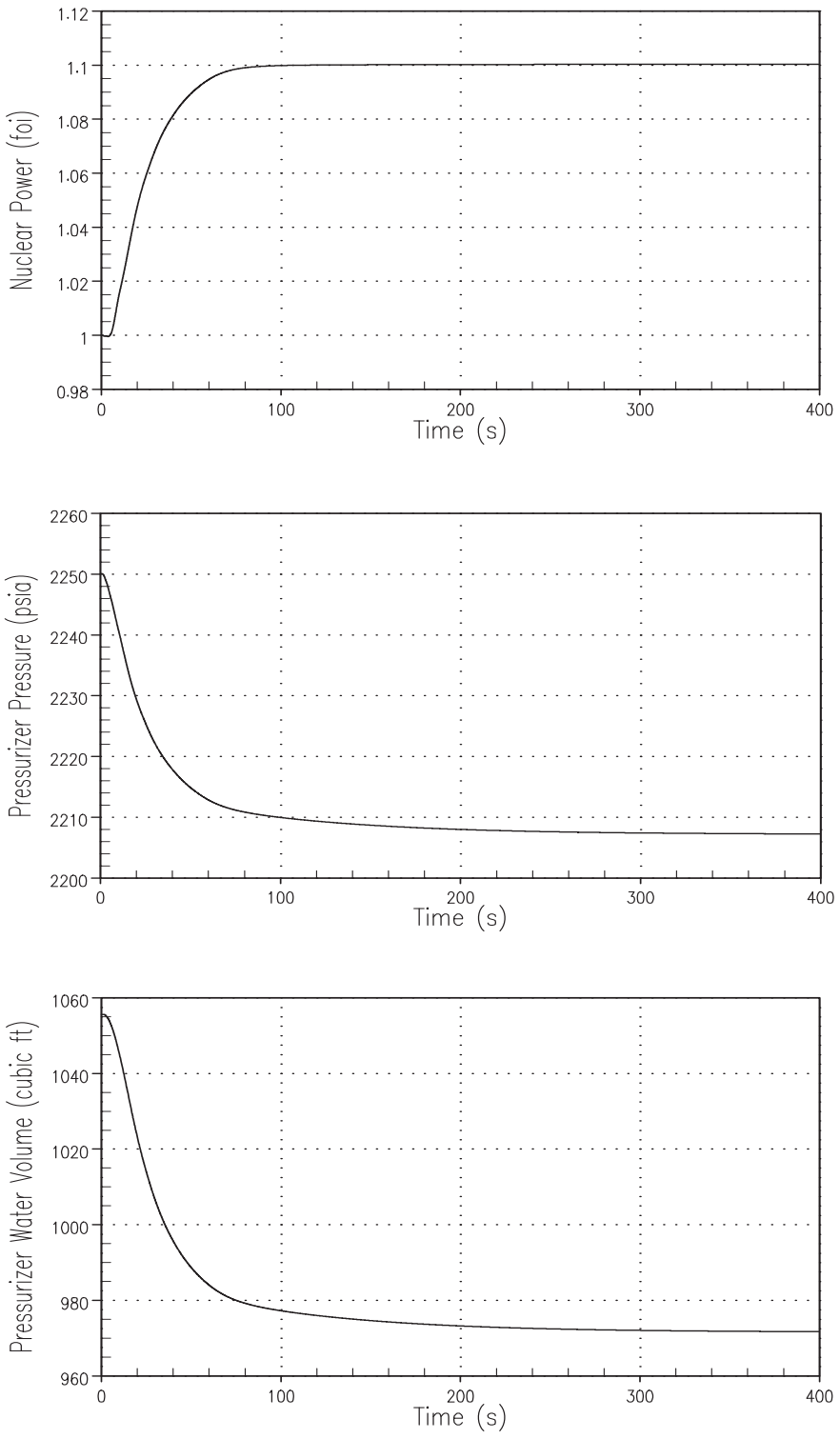

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FIGURE 15.1-5 NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MAXIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL

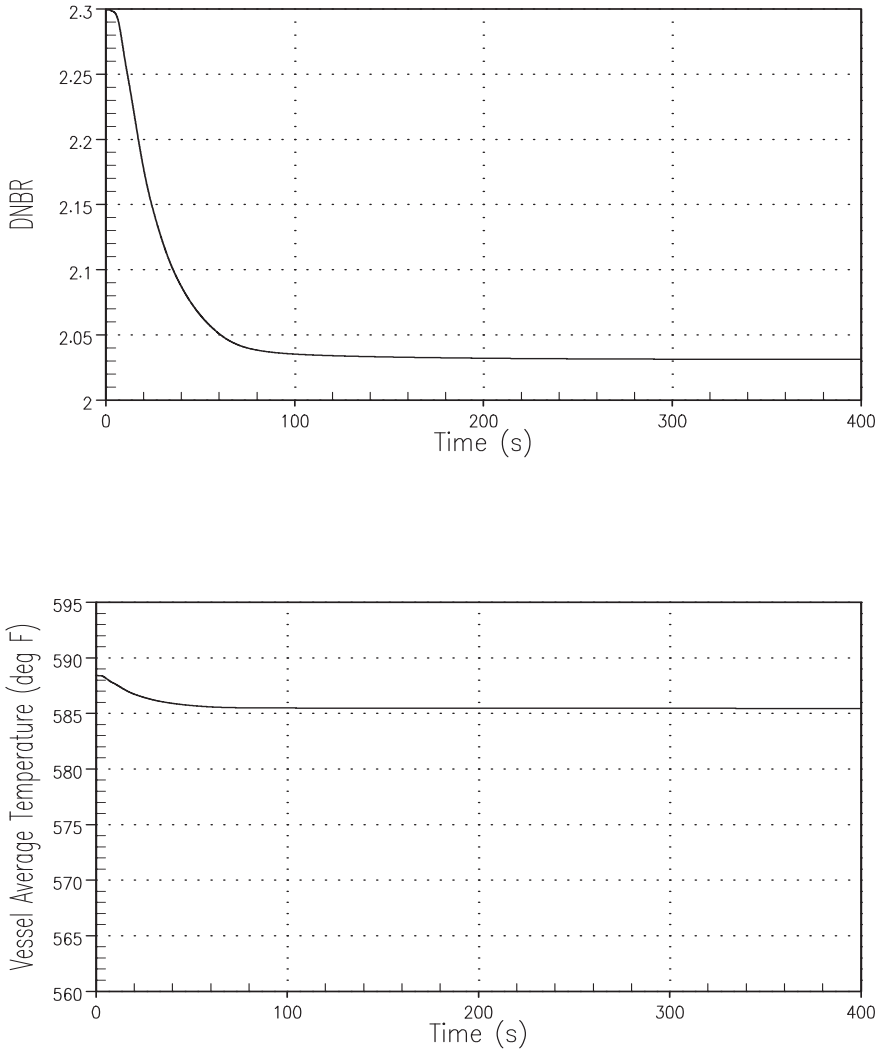

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FIGURE 15.1-6 DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MAXIMUM REACTIVITY FEEDBACK, MANUAL REACTOR CONTROL

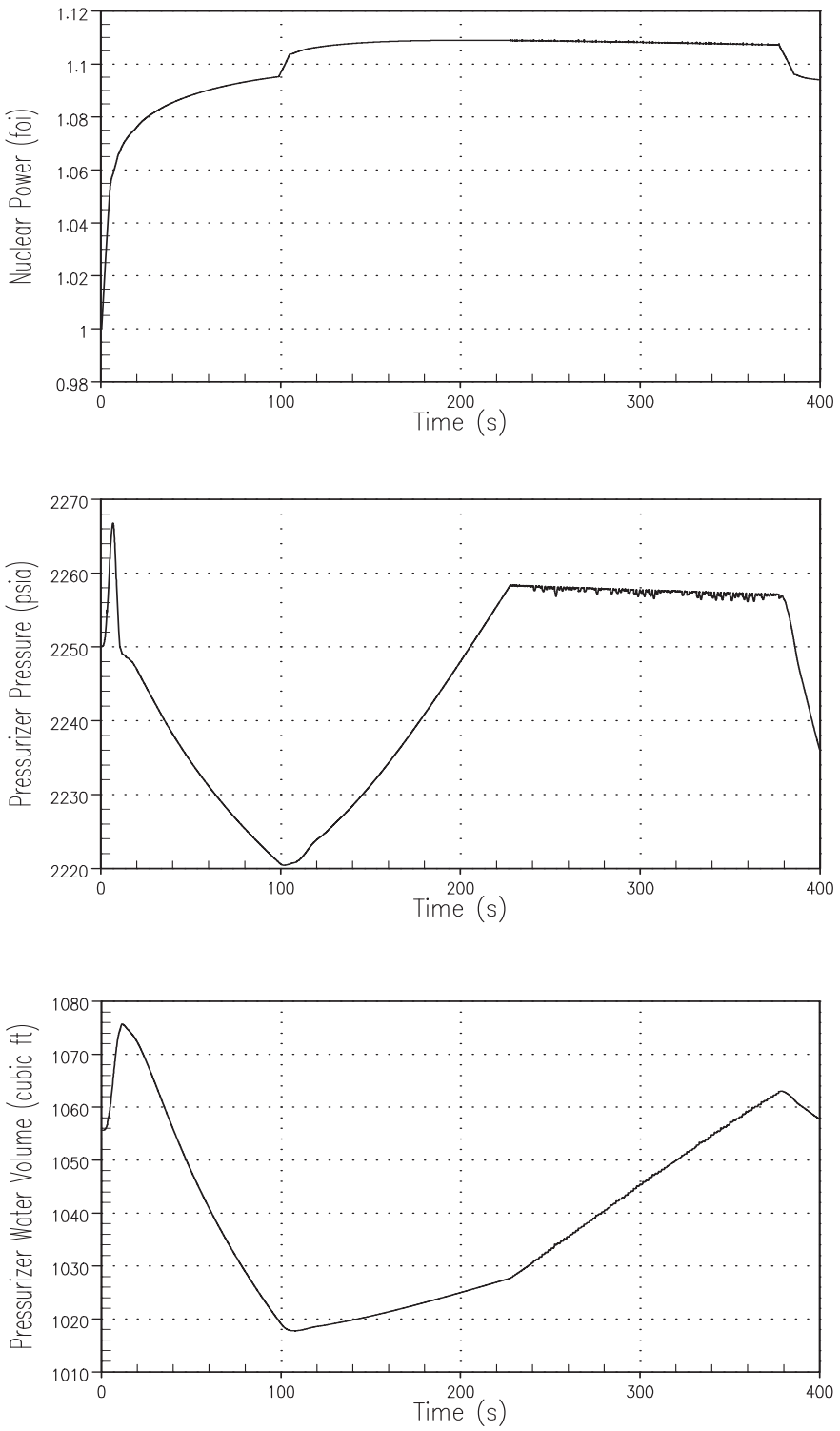

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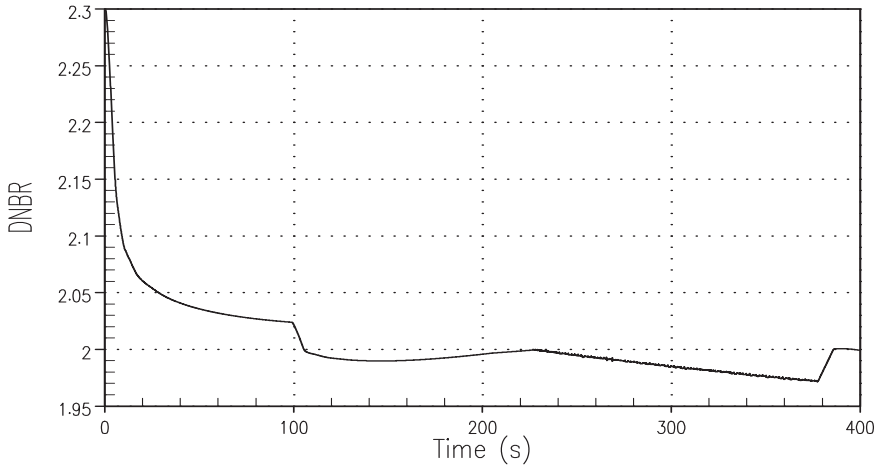
FIGURE 15.1-7

NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL



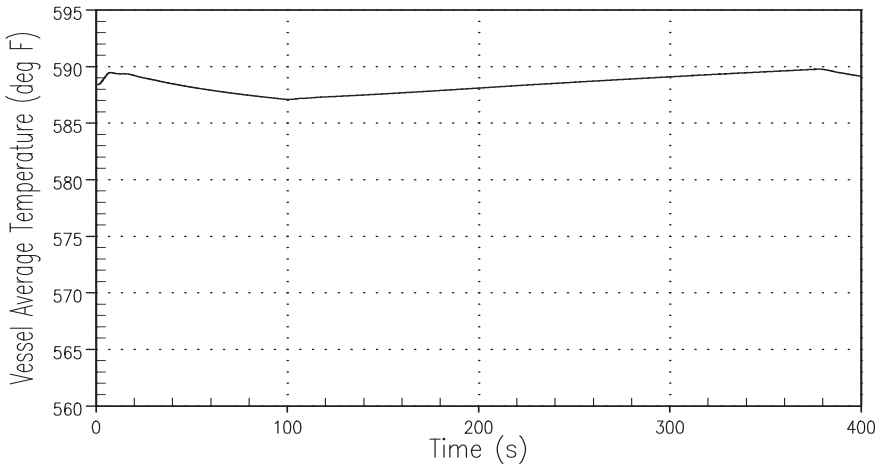


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FIGURE 15.1-8 DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MINIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL

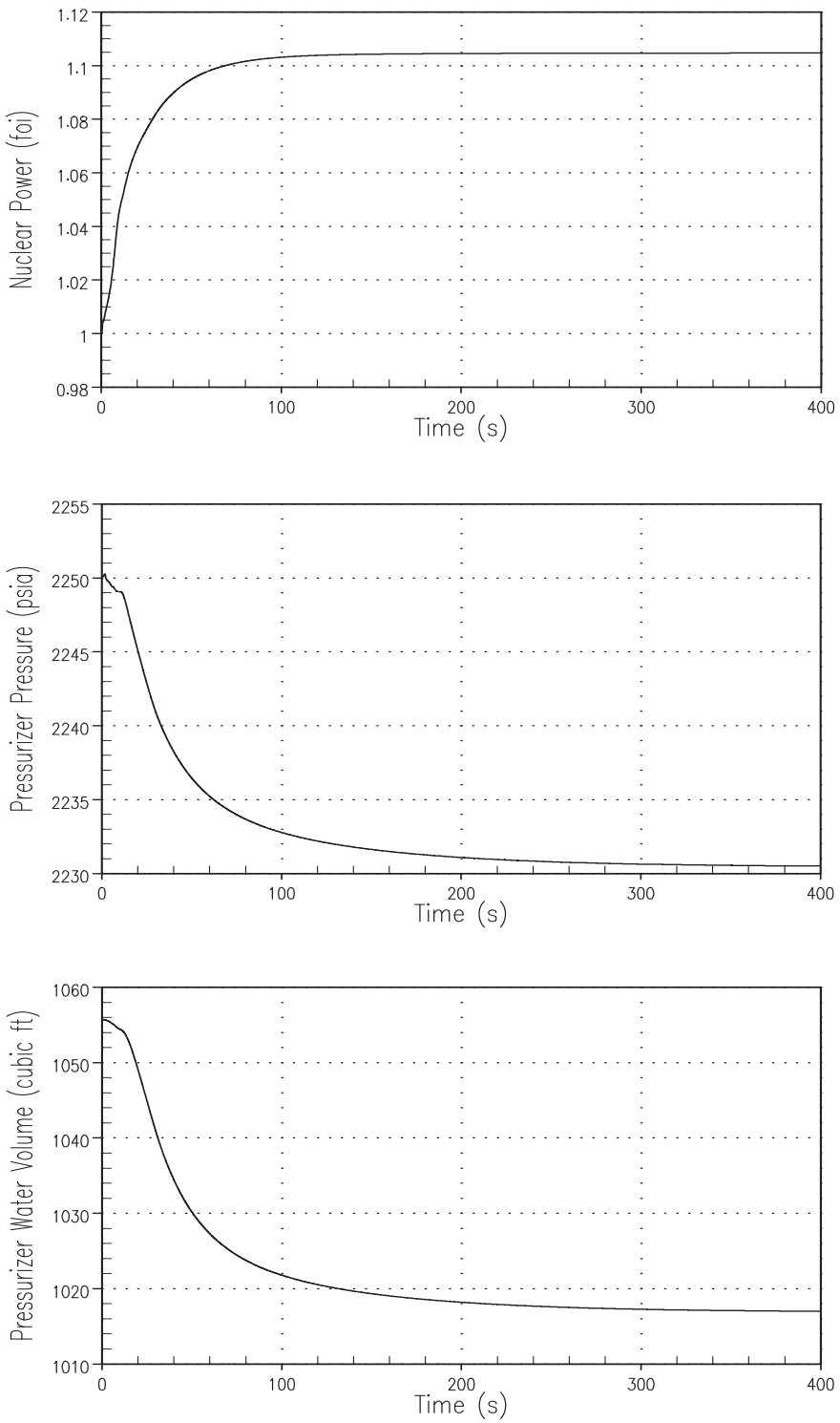

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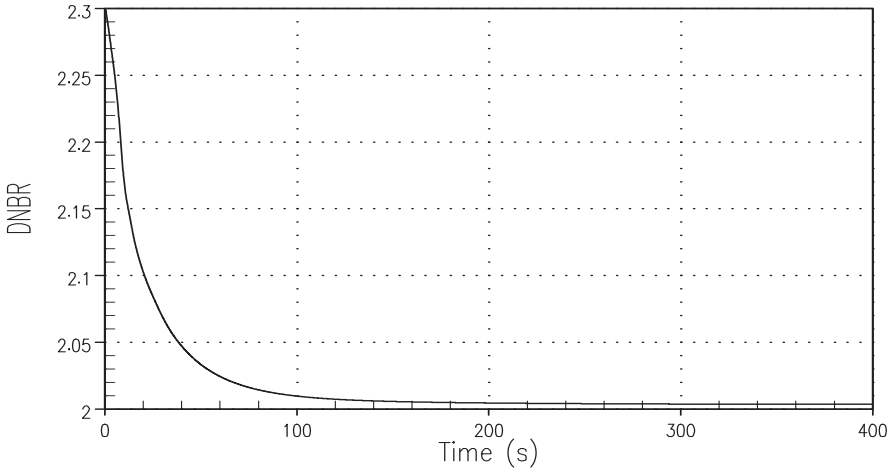
FIGURE 15.1-9 NUCLEAR POWER, PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL



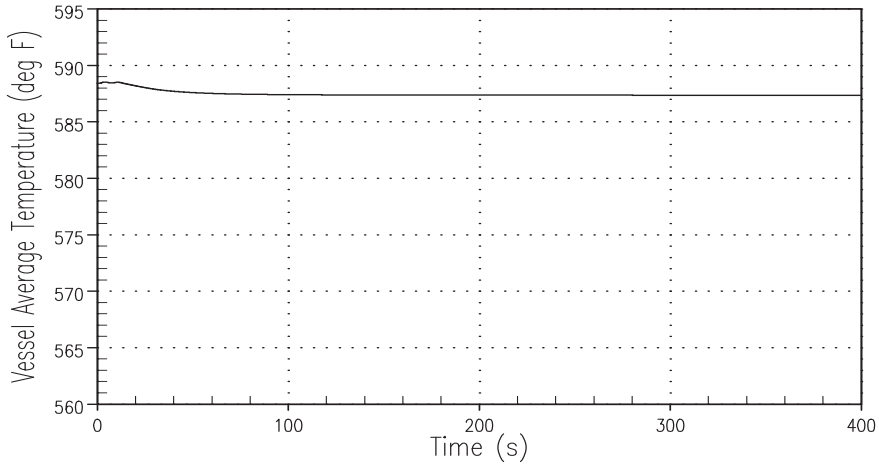


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FIGURE 15.1-10 DNBR AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR TEN PERCENT STEP LOAD INCREASE, MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC REACTOR CONTROL

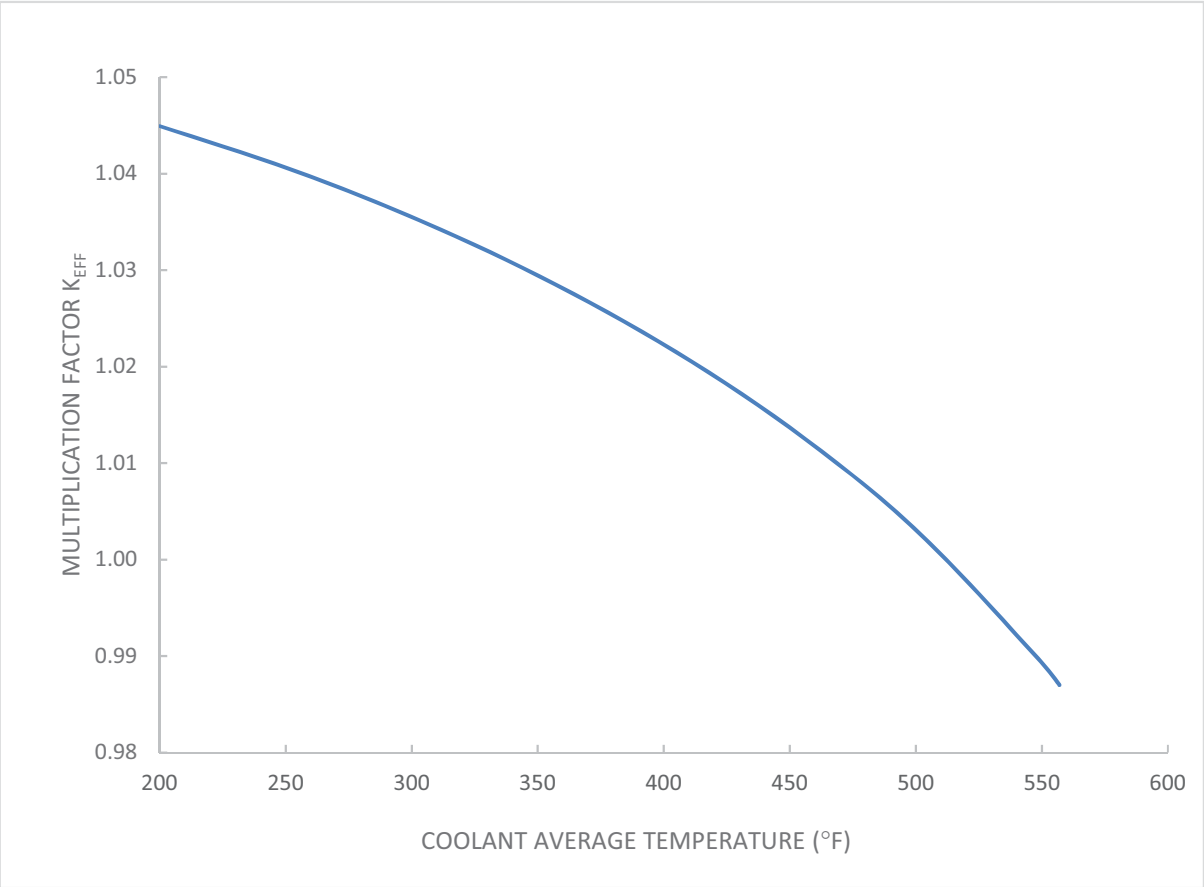

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Figure 15.1-11

K_{eff} versus Temperature

FIGURE 15.1-11a HAS BEEN DELETED

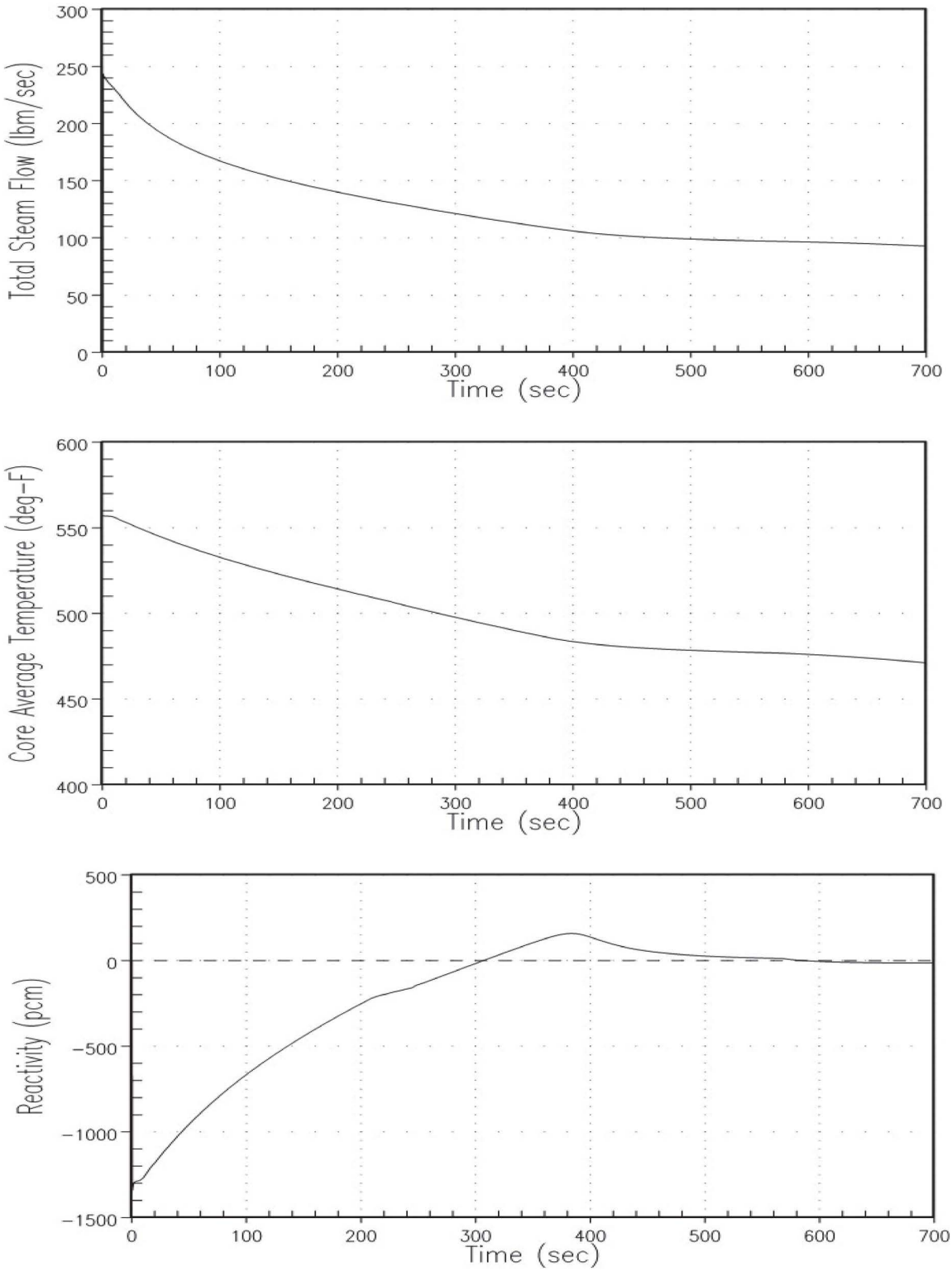

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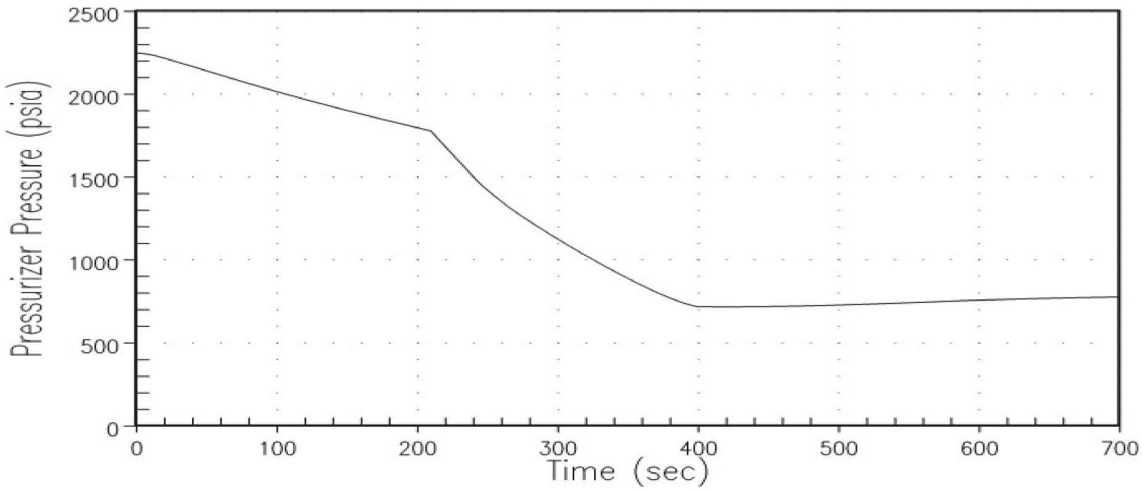
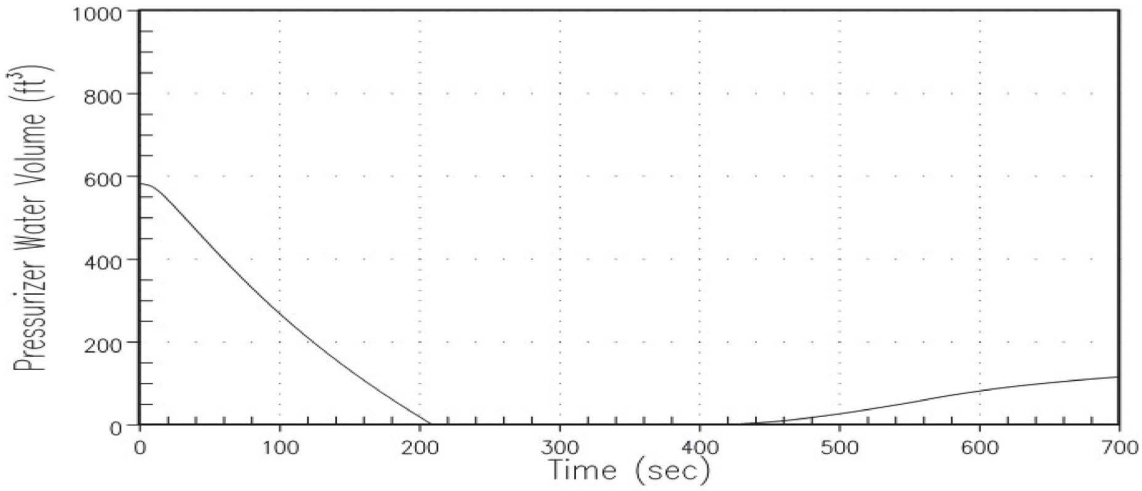
Figure 15.1-12
Failure of a Steam Generator Safety or Atmospheric Relief Valve
Total Steam Flow,
Core Average Temperature and
Reactivity vs. Time





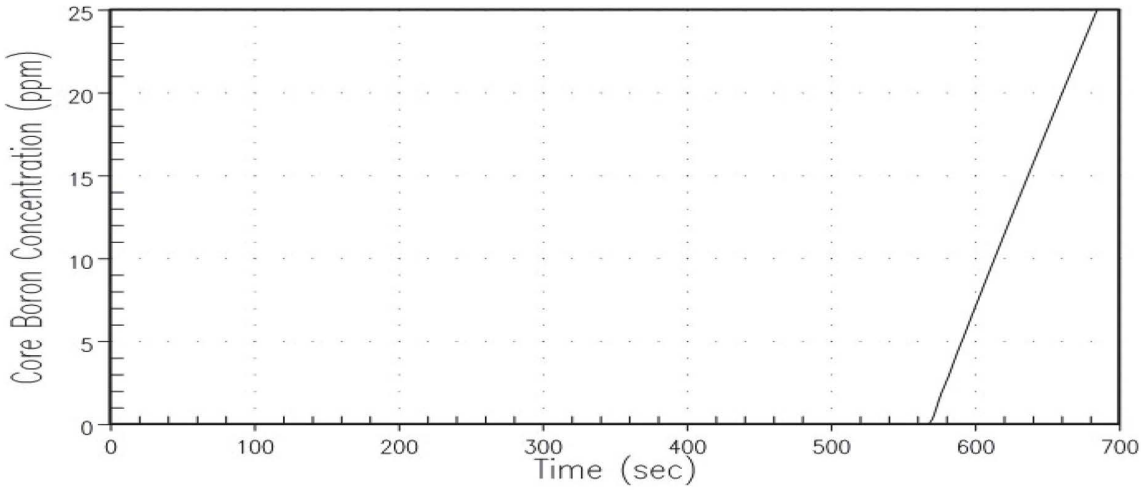


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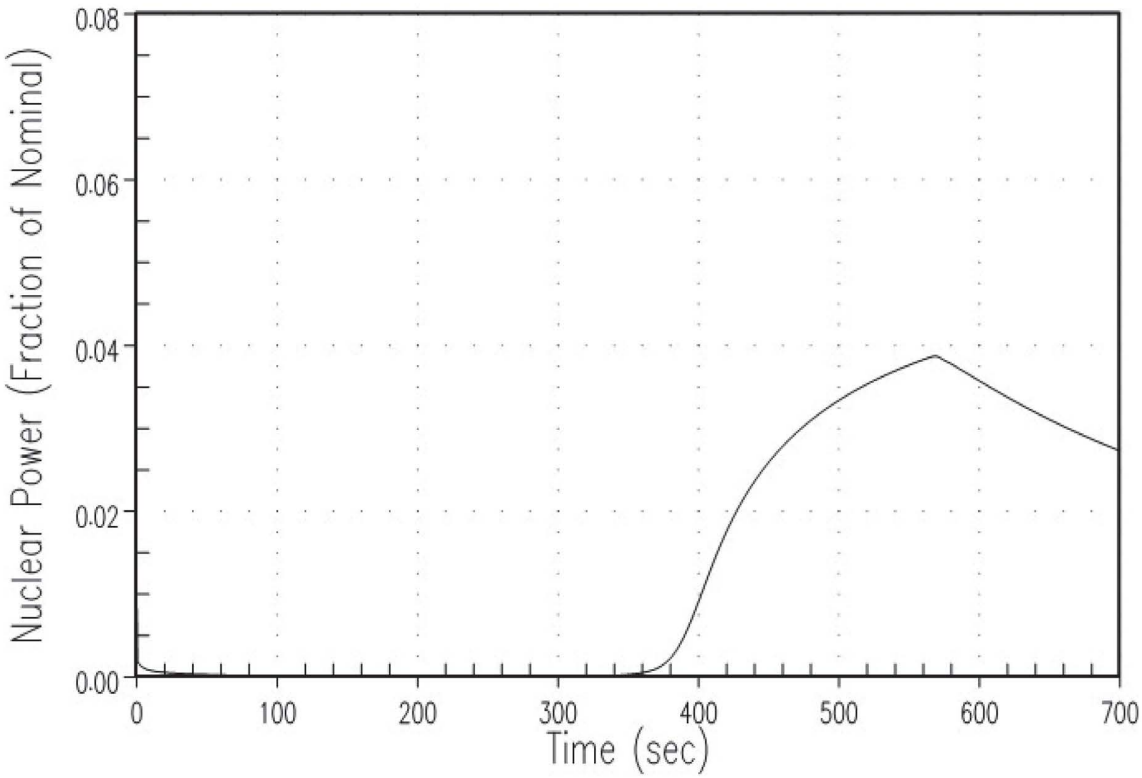
Figure 15.1-13
Failure of a Steam Generator Safety or Atmospheric Relief Valve Pressurizer Pressure, Pressurizer Water Volume and Core Boron Concentration vs. Time



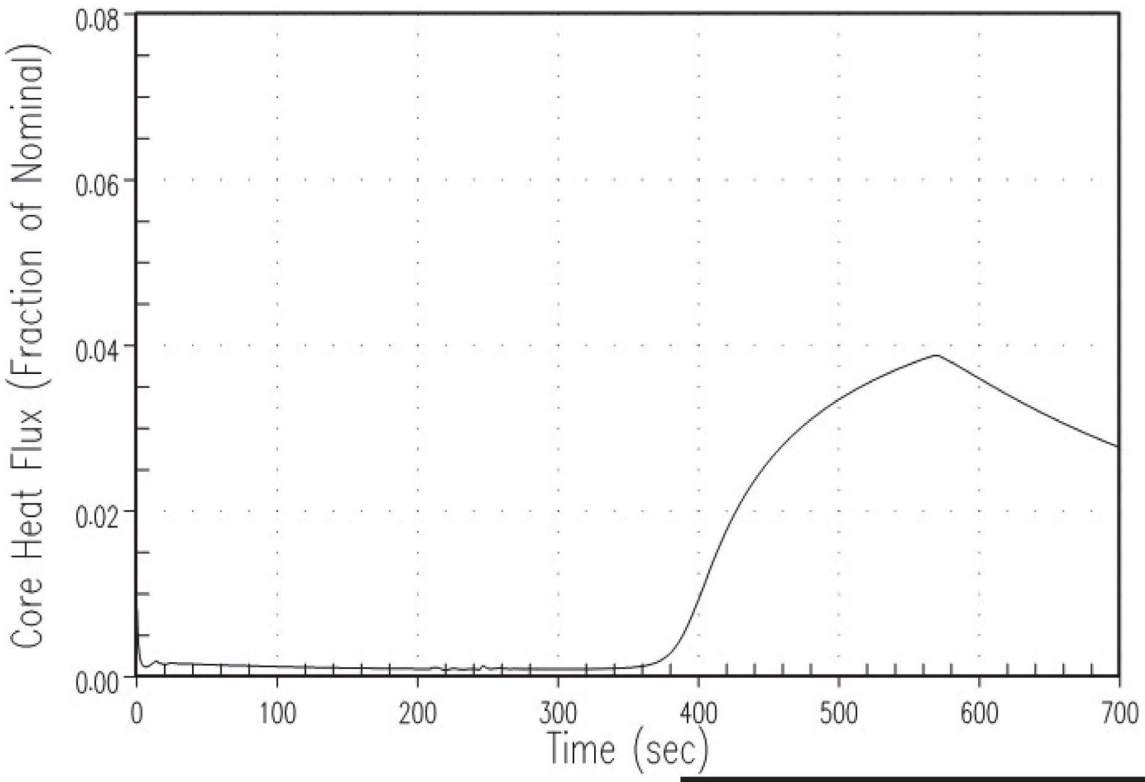


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Figure 15.1-13a
Failure of a Steam Generator Safety or Atmospheric Relief Valve Nuclear Power and Core Heat Flux vs. Time

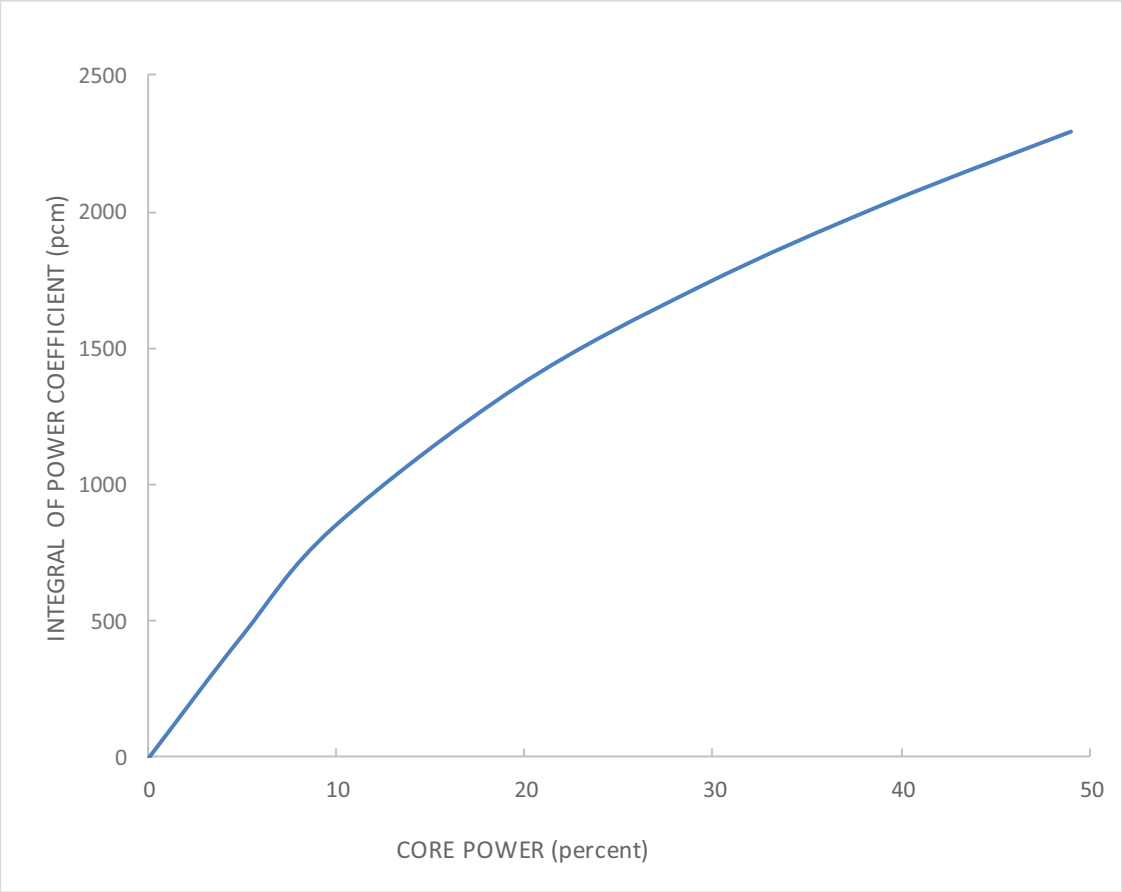

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Figure 15.1-14

Doppler Power Feedback

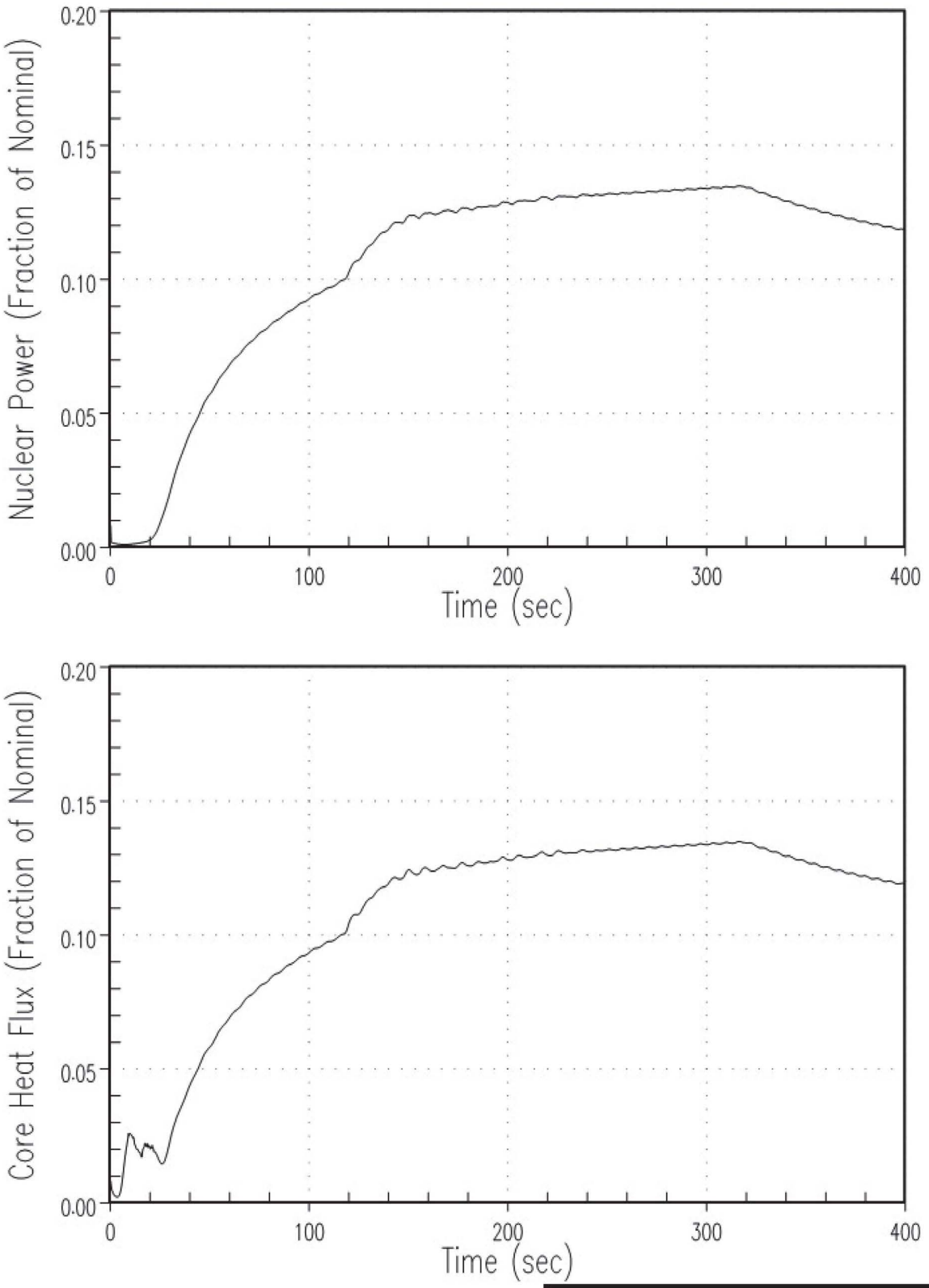

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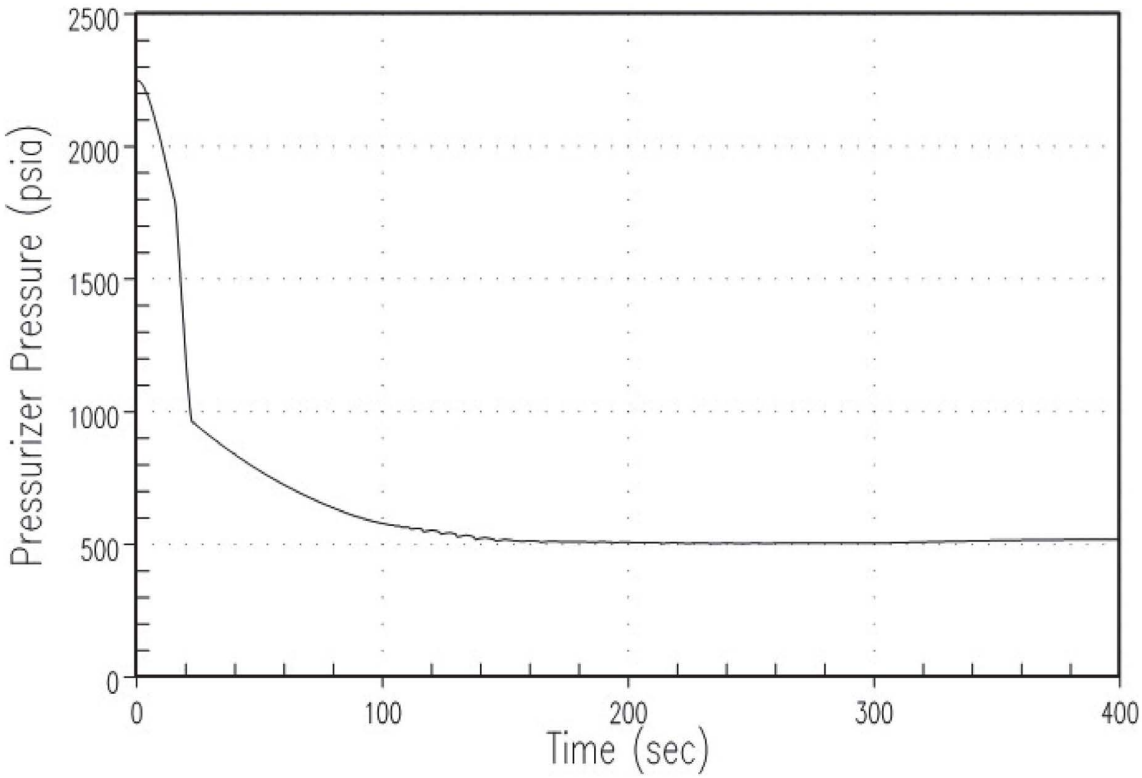
Figure 15.1-15
SLB Transient at Hot Zero Power with Offsite Power Available Nuclear Power and Core Heat Flux vs. Time



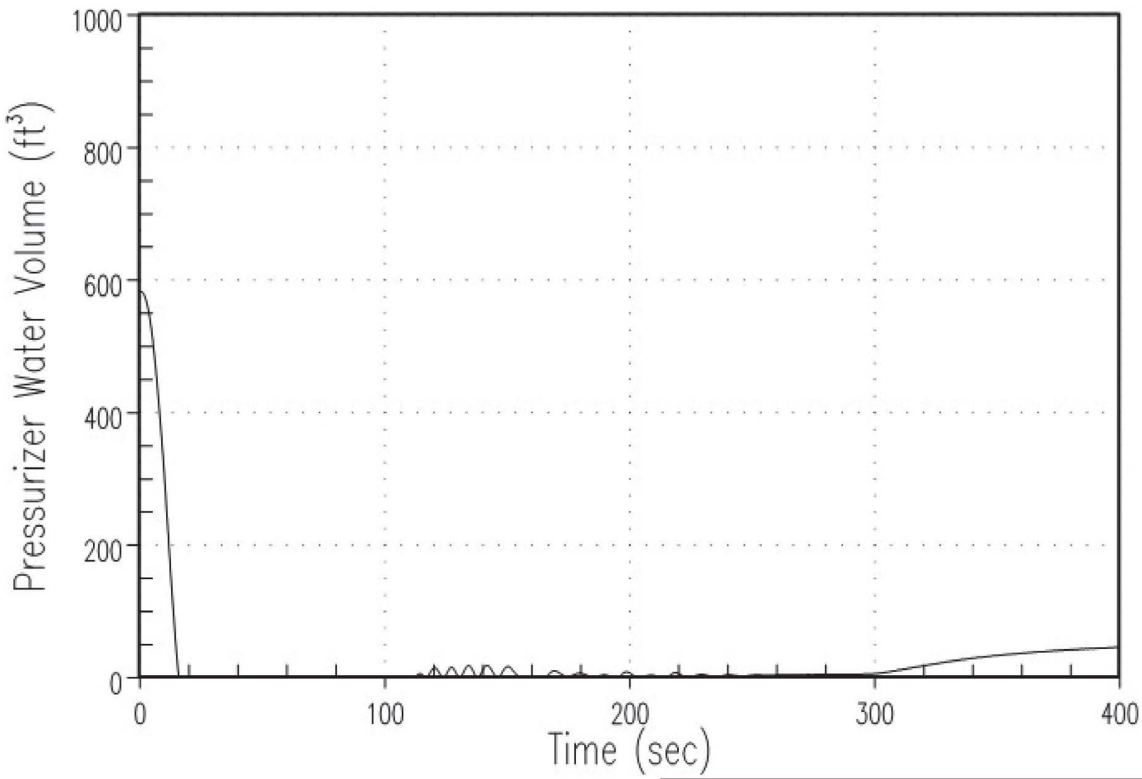


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Figure 15.1-16
SLB Transient at Hot Zero Power with Offsite Power Available Pressurizer Pressure and Pressurizer Water Volume vs. Time

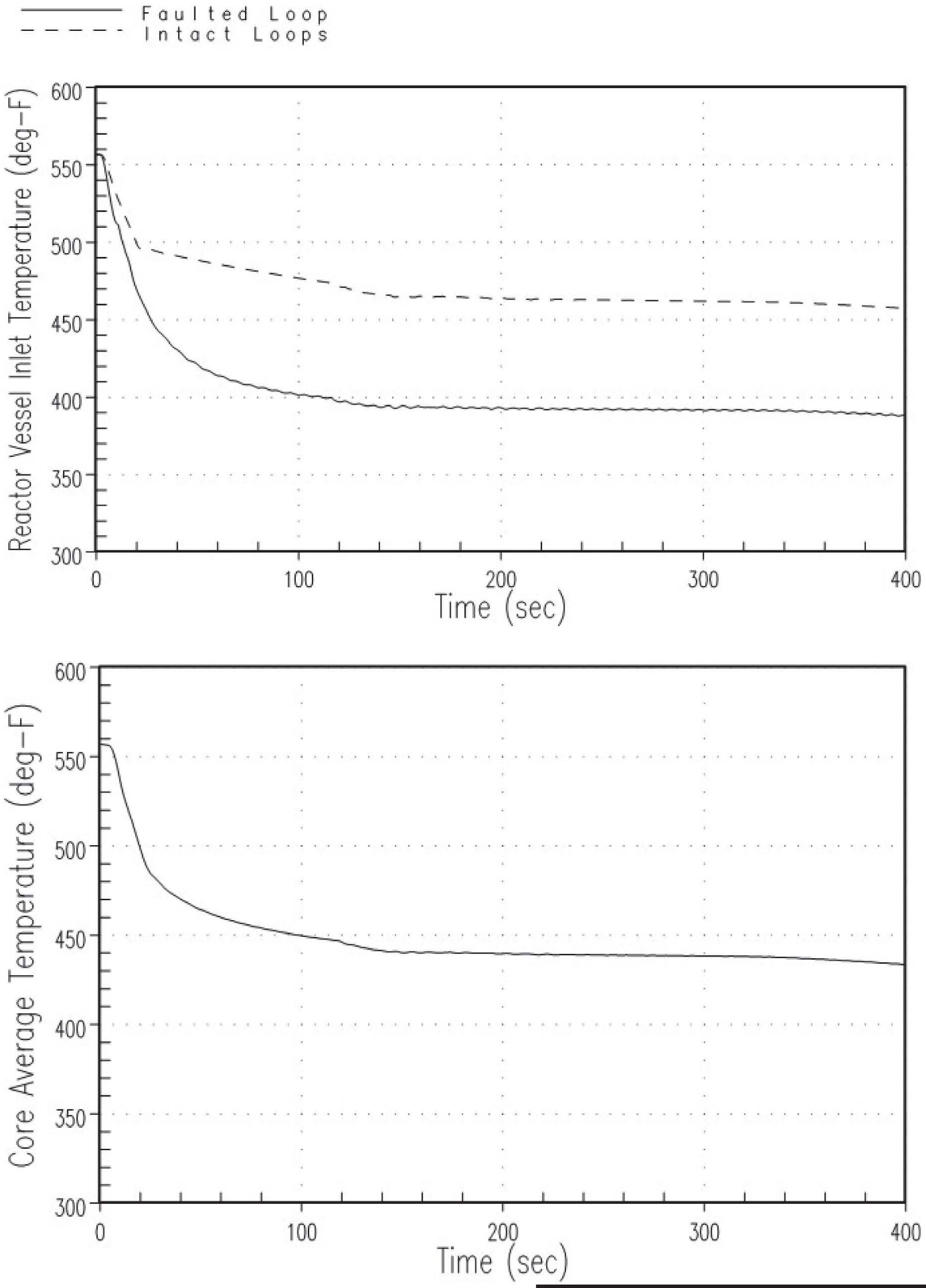

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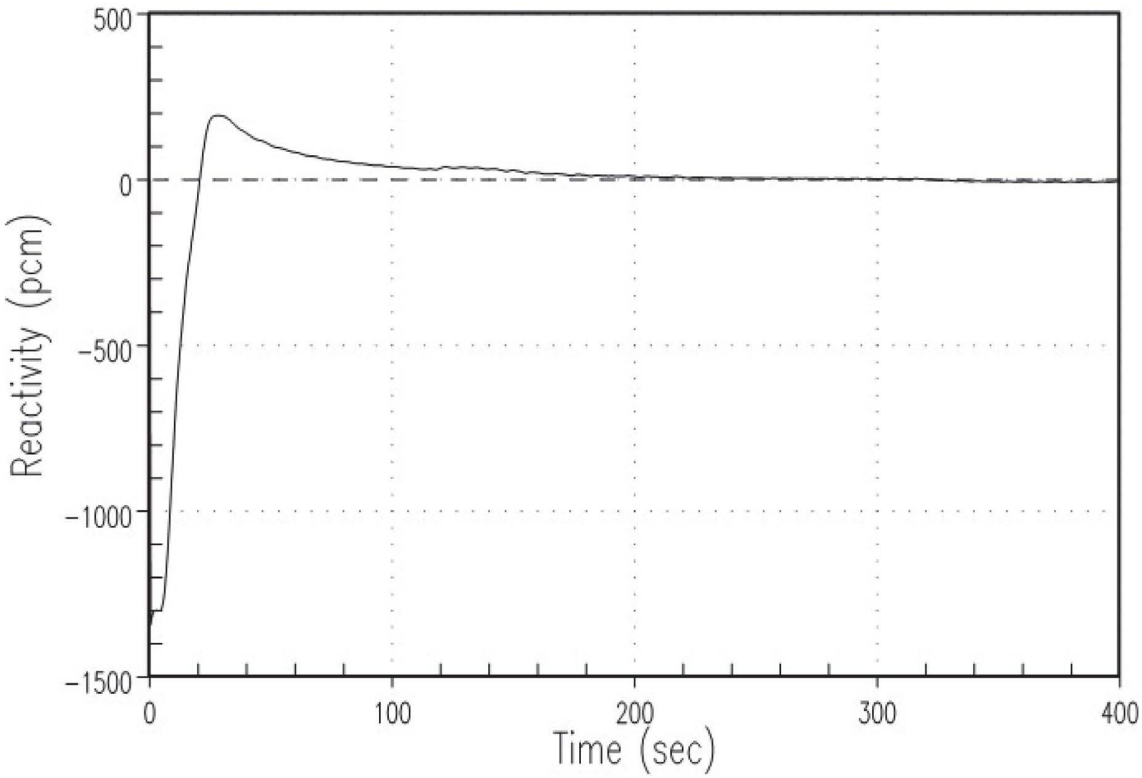
Figure 15.1-17
SLB Transient at Hot Zero Power with Offsite Power Available Reactor Vessel Inlet Temperature and Core Average Temperature vs. Time



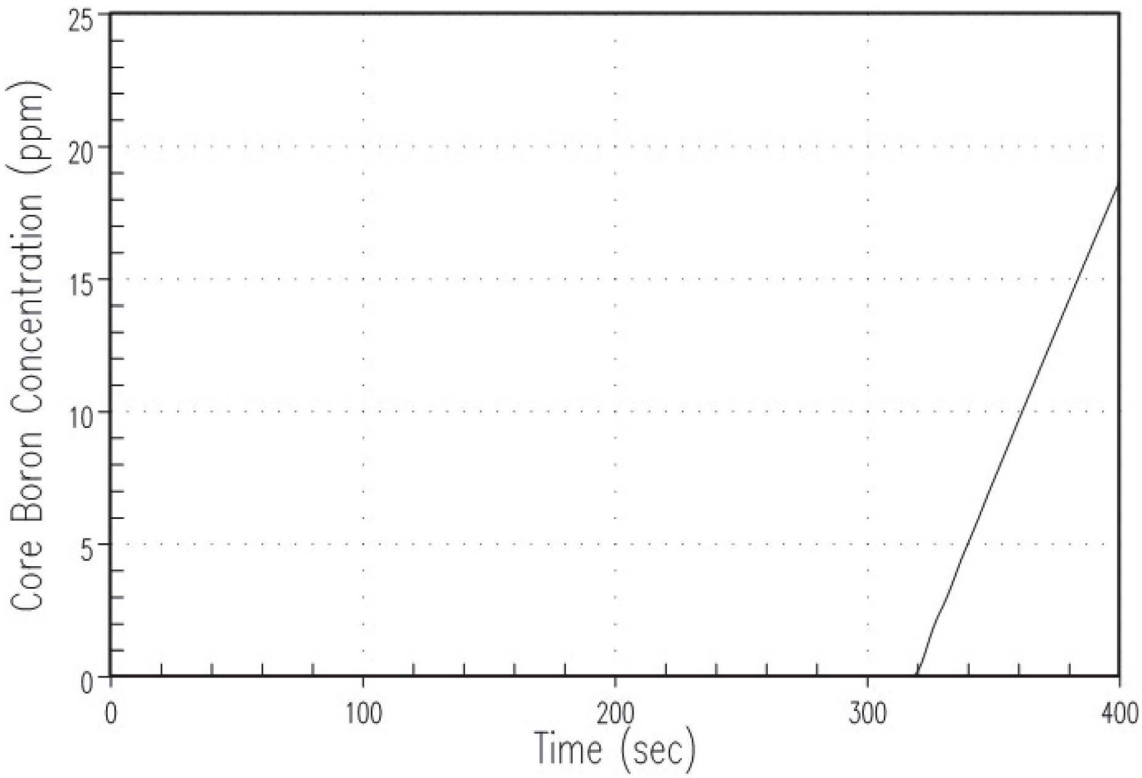


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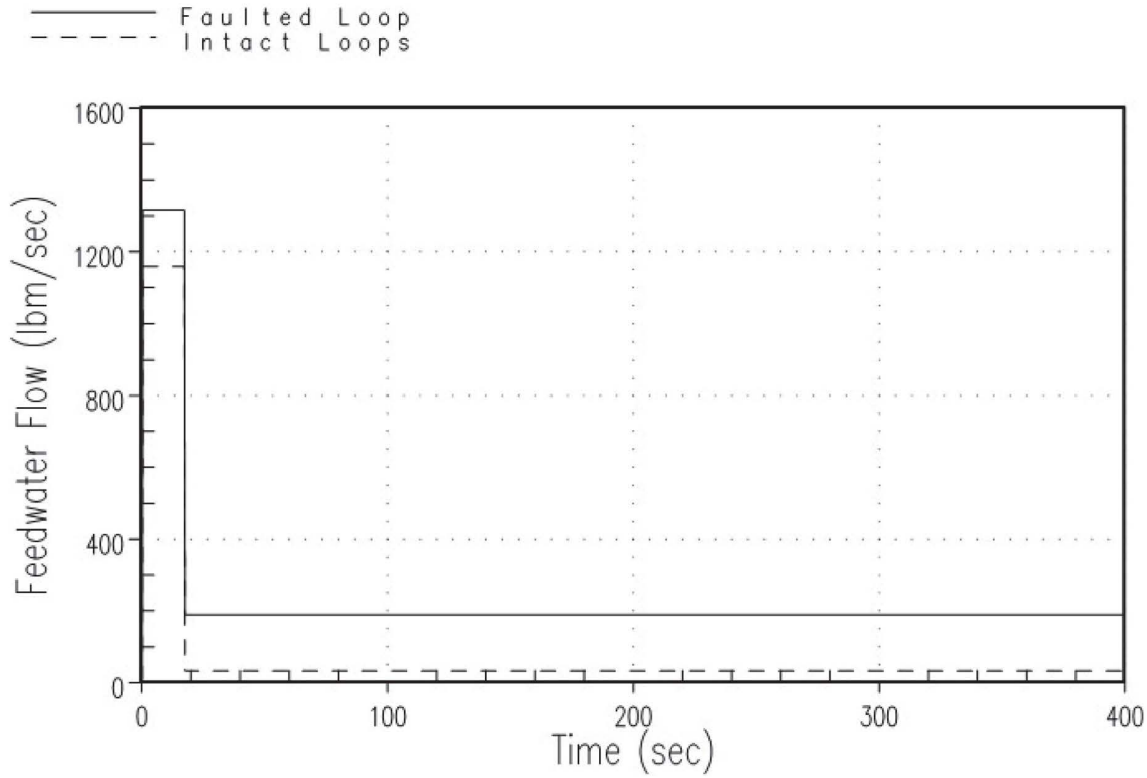
Figure 15.1-18
SLB Transient at Hot Zero Power with Offsite Power Available Reactivity and Core Boron Concentration vs. Time



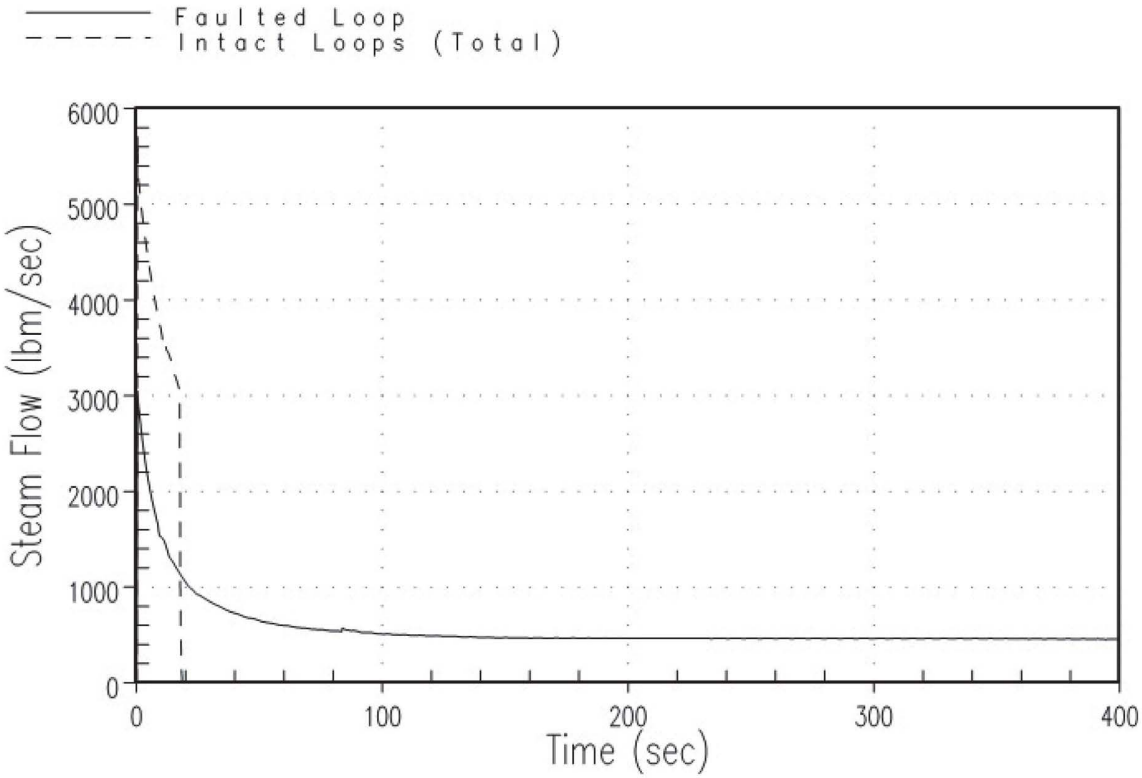


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Figure 15.1-19
SLB Transient at Hot Zero Power with Offsite Power Available Feedwater Flow and Steam Flow vs. Time

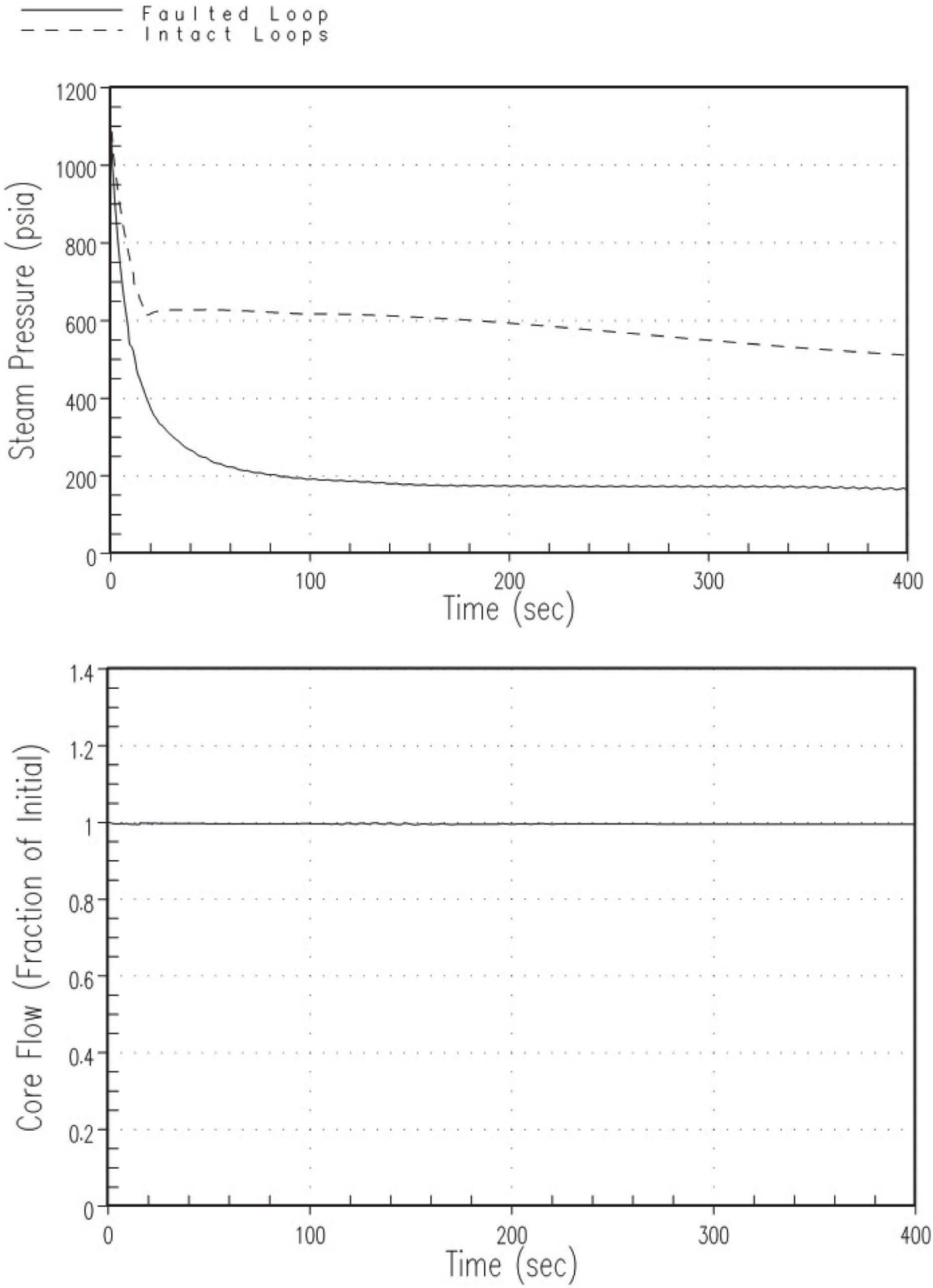

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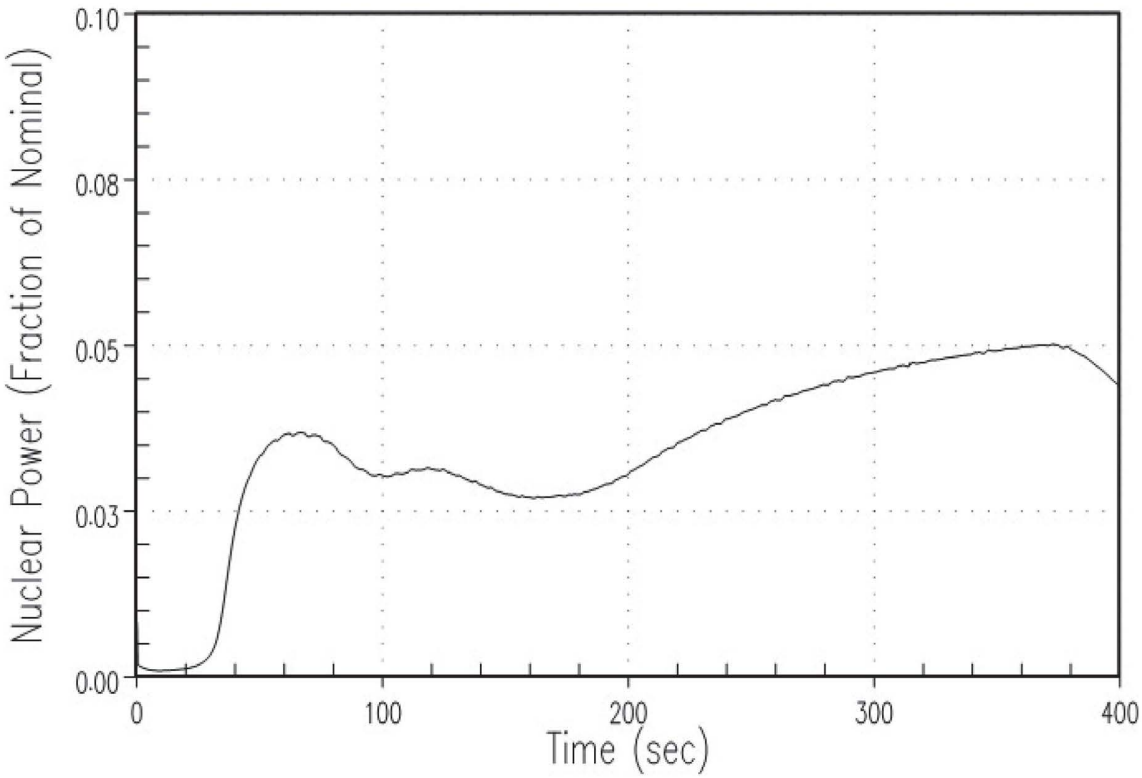
Figure 15.1-20
SLB Transient at Hot Zero Power with Offsite Power Available Steam Pressure and Core Flow vs. Time

FIGURE 15.1-20a HAS BEEN DELETED



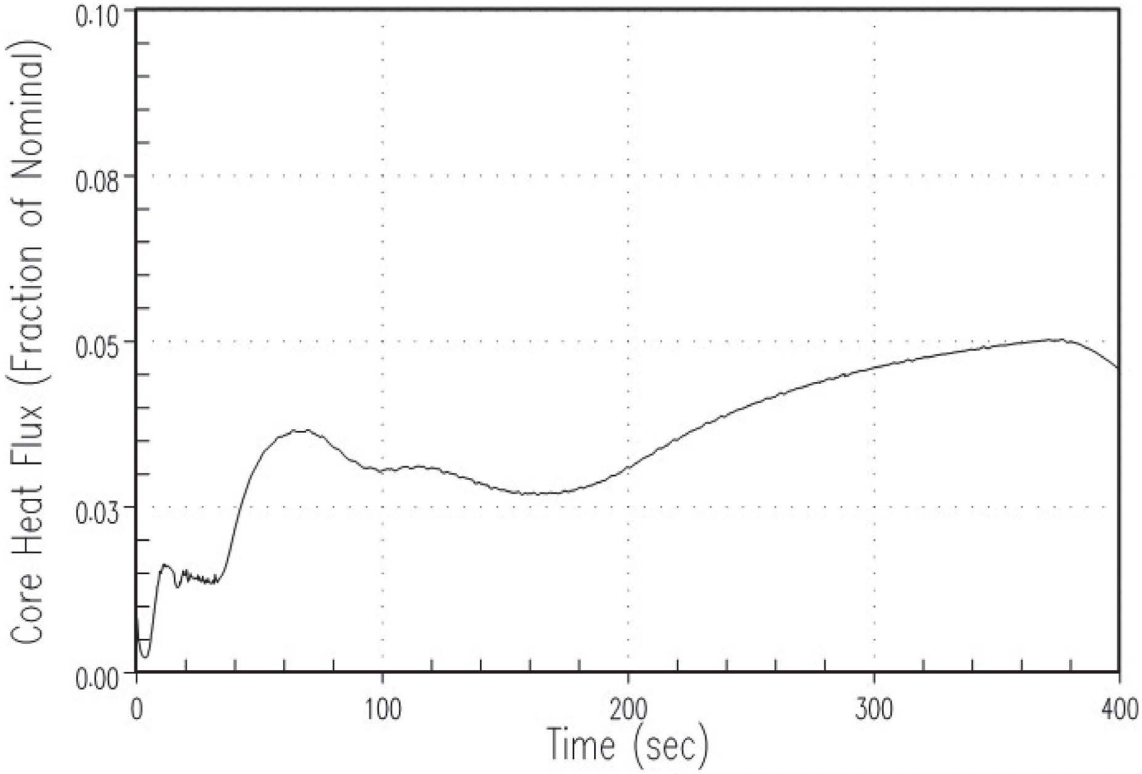


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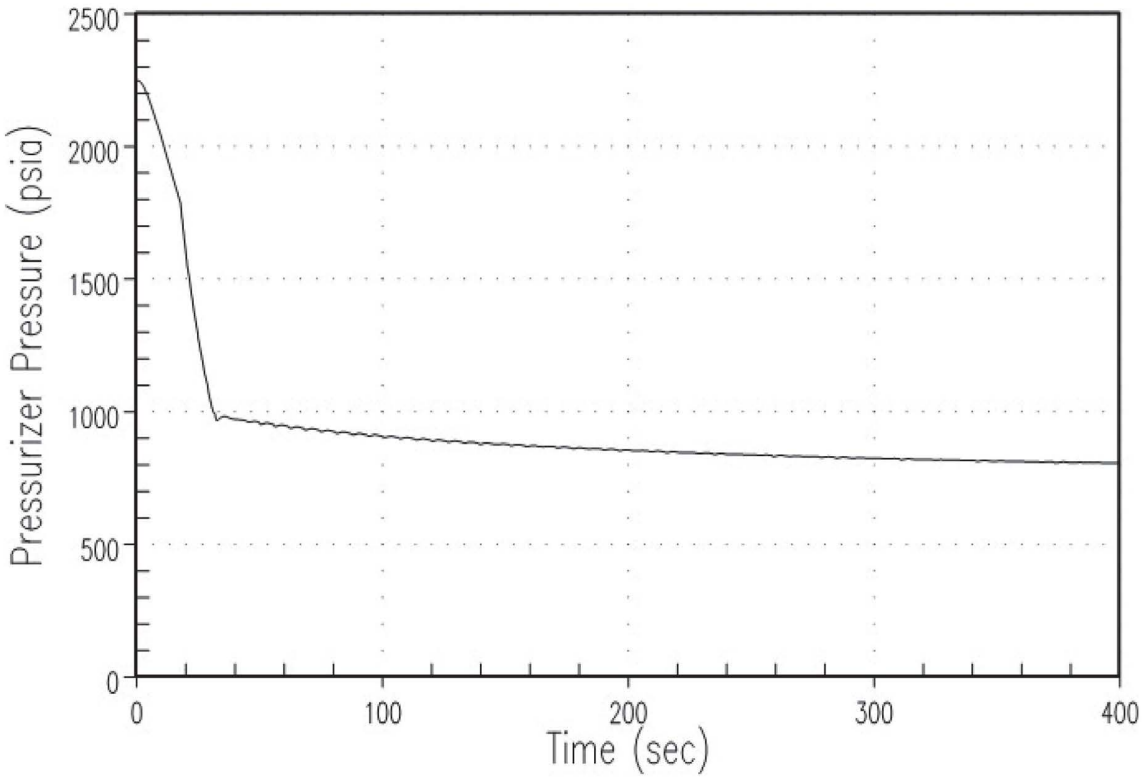
Figure 15.1-21
SLB Transient at Hot Zero Power without Offsite Power Available Nuclear Power and Core Heat Flux vs. Time



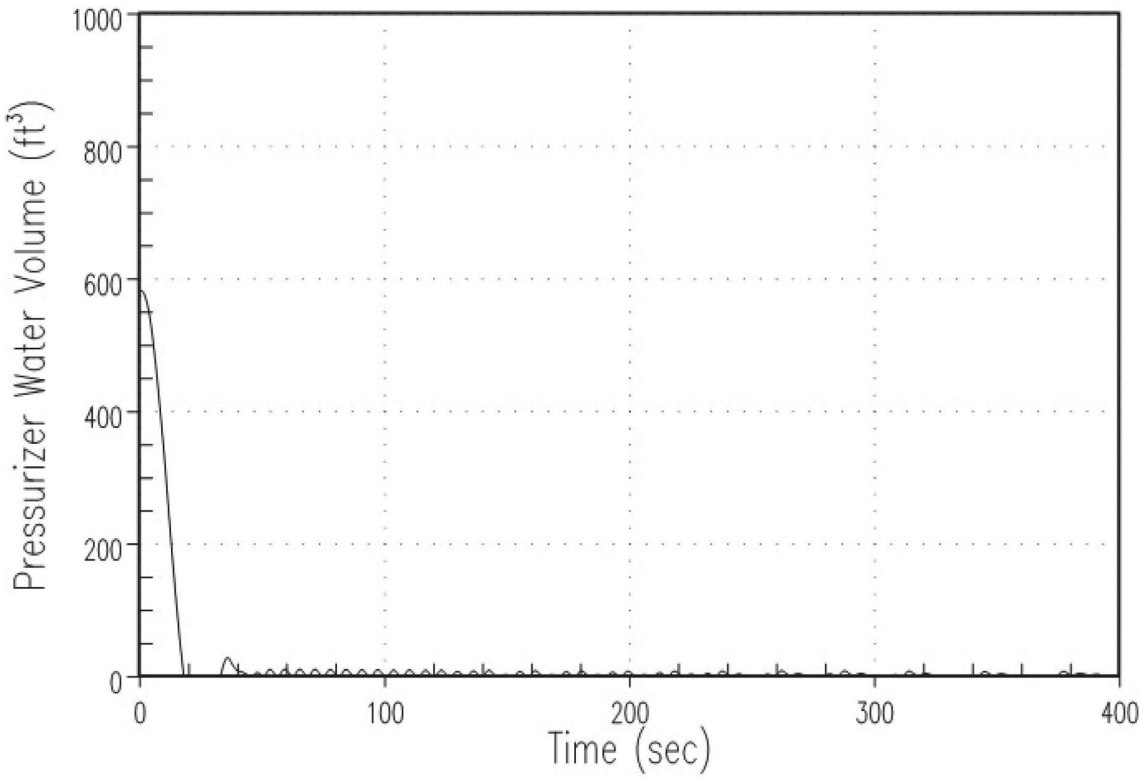


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Figure 15.1-22
SLB Transient at Hot Zero Power without Offsite Power Available Pressurizer Pressure and Pressurizer Water Volume vs. Time

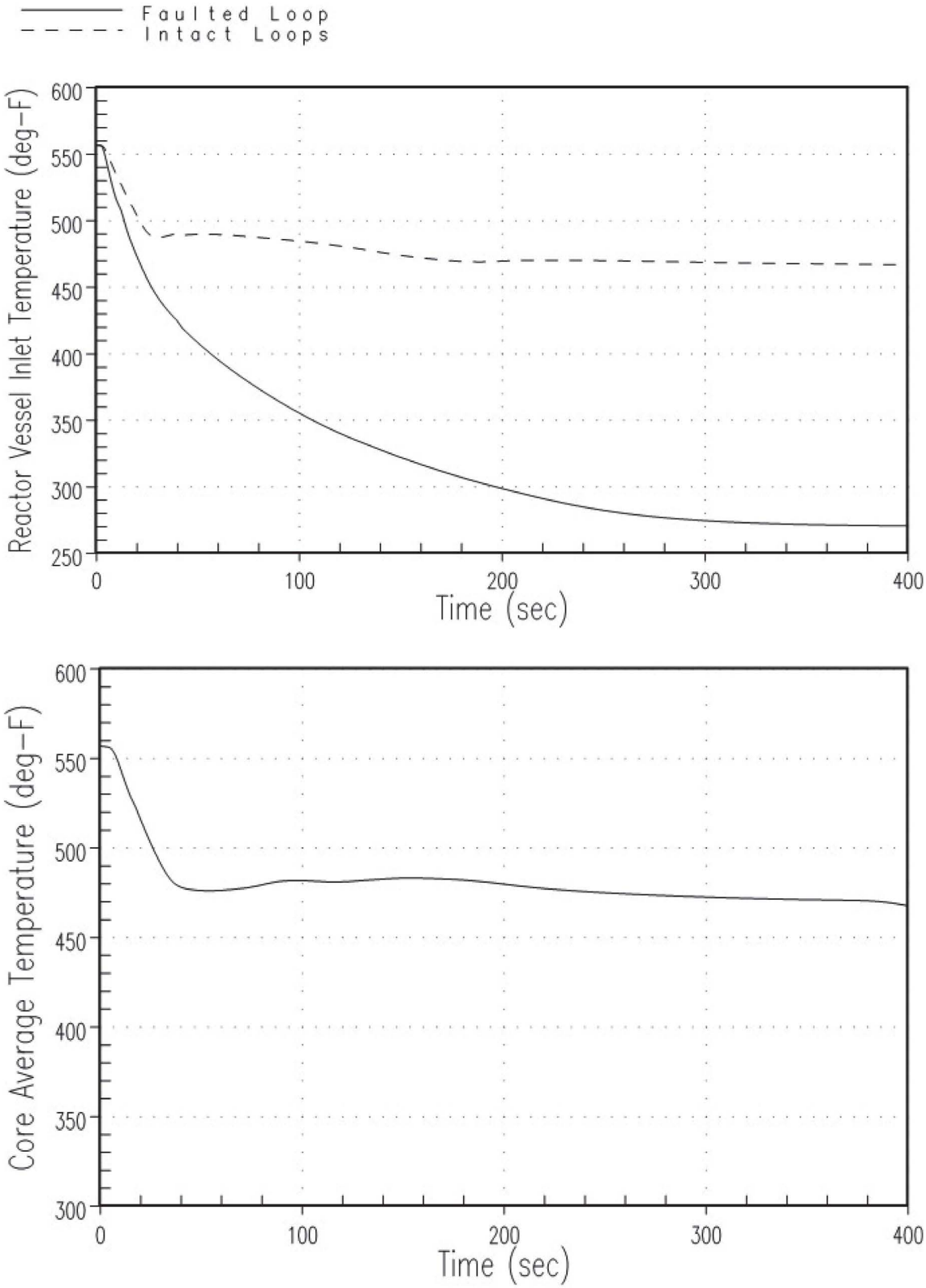

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Figure 15.1-23
SLB Transient at Hot Zero Power without Offsite Power Available Reactor Vessel Inlet Temperature and Core Average Temperature vs. Time

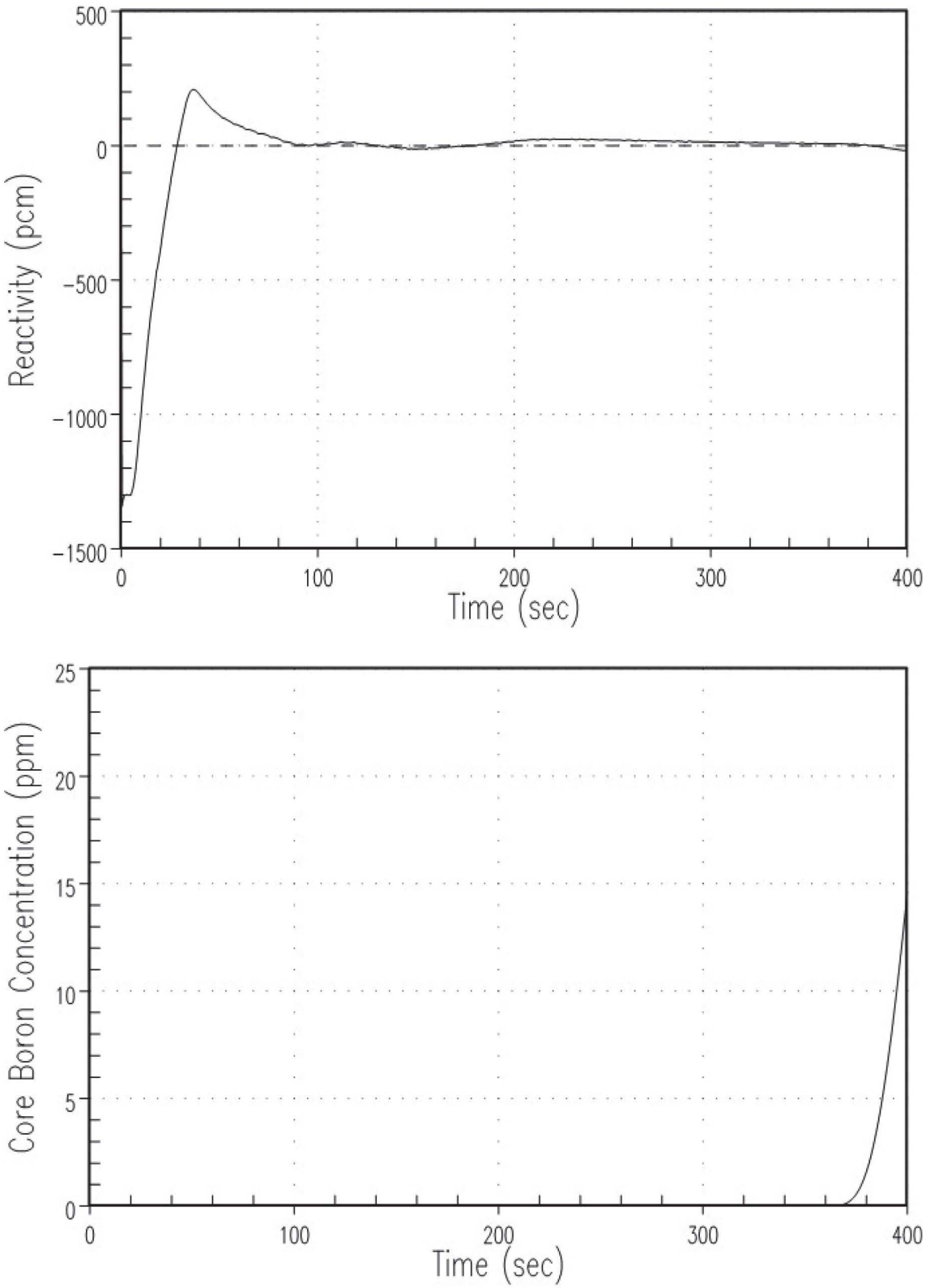

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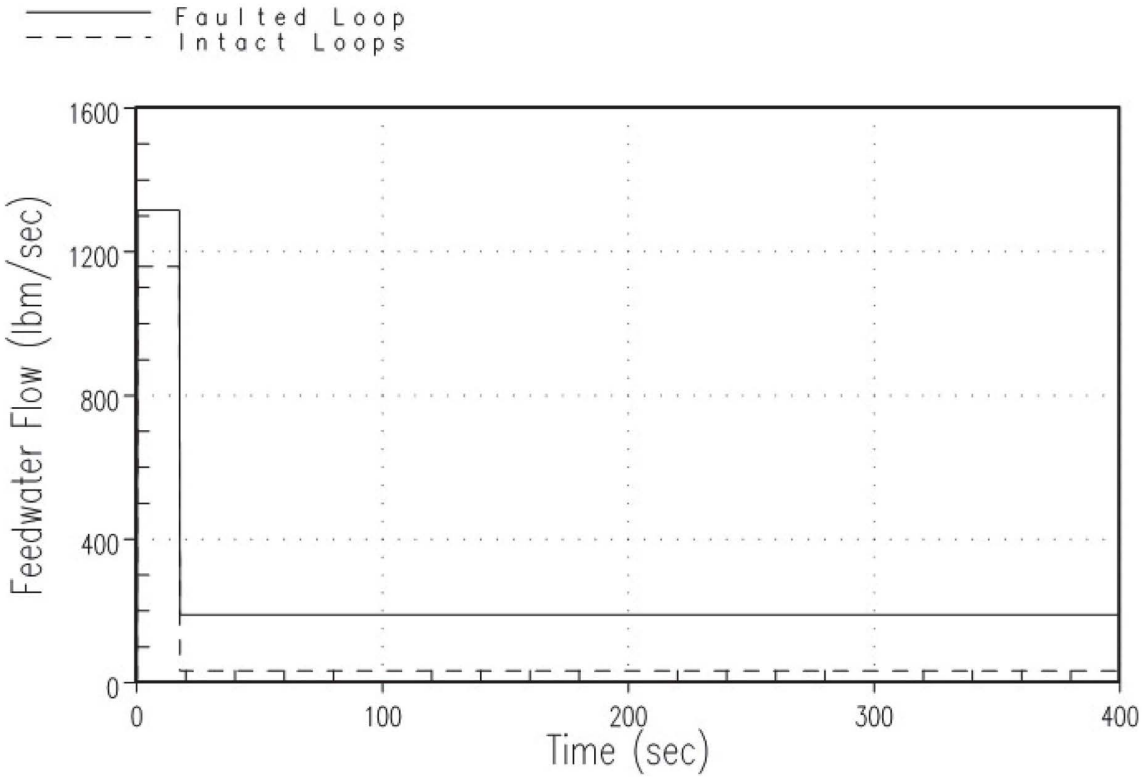
Figure 15.1-24
SLB Transient at Hot Zero Power without Offsite Power Available Reactivity and Core Boron Concentration vs. Time



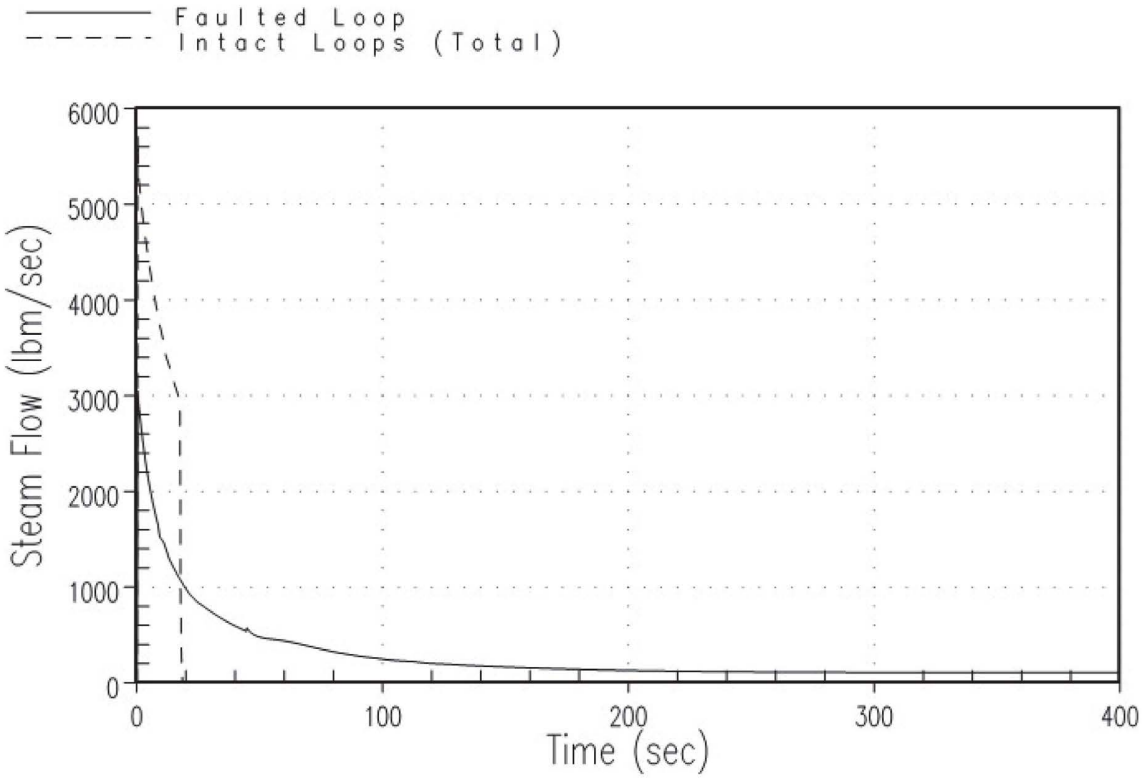


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Figure 15.1-25
SLB Transient at Hot Zero Power without Offsite Power Available Feedwater Flow and Steam Flow vs. Time

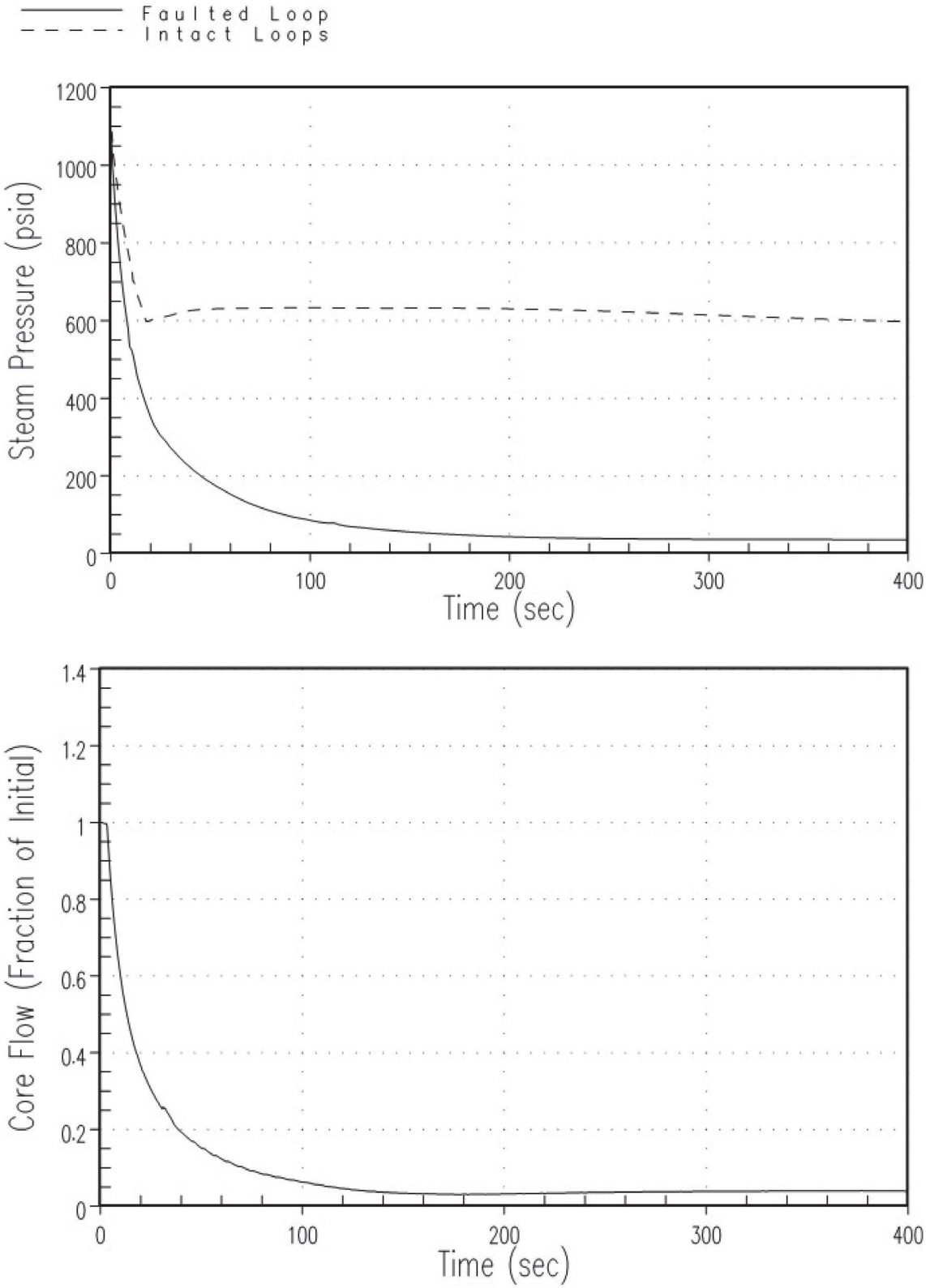

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Figure 15.1-26
SLB Transient at Hot Zero Power without Offsite Power Available Steam Pressure and Core Flow vs. Time

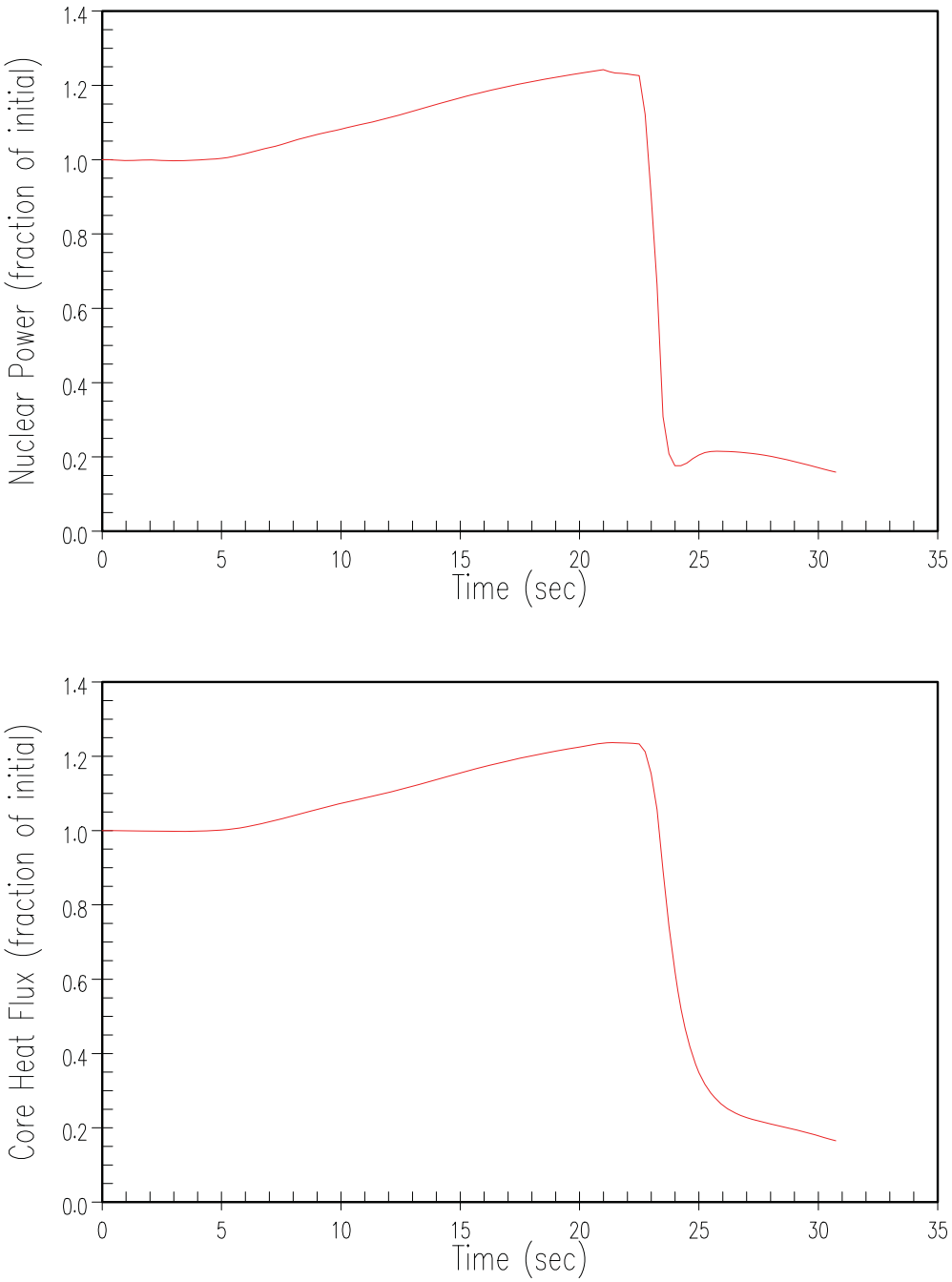

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Figure 15.1-27

Nuclear Power and Core Heat Flux vs. Time for SLB Transient at Hot Full Power

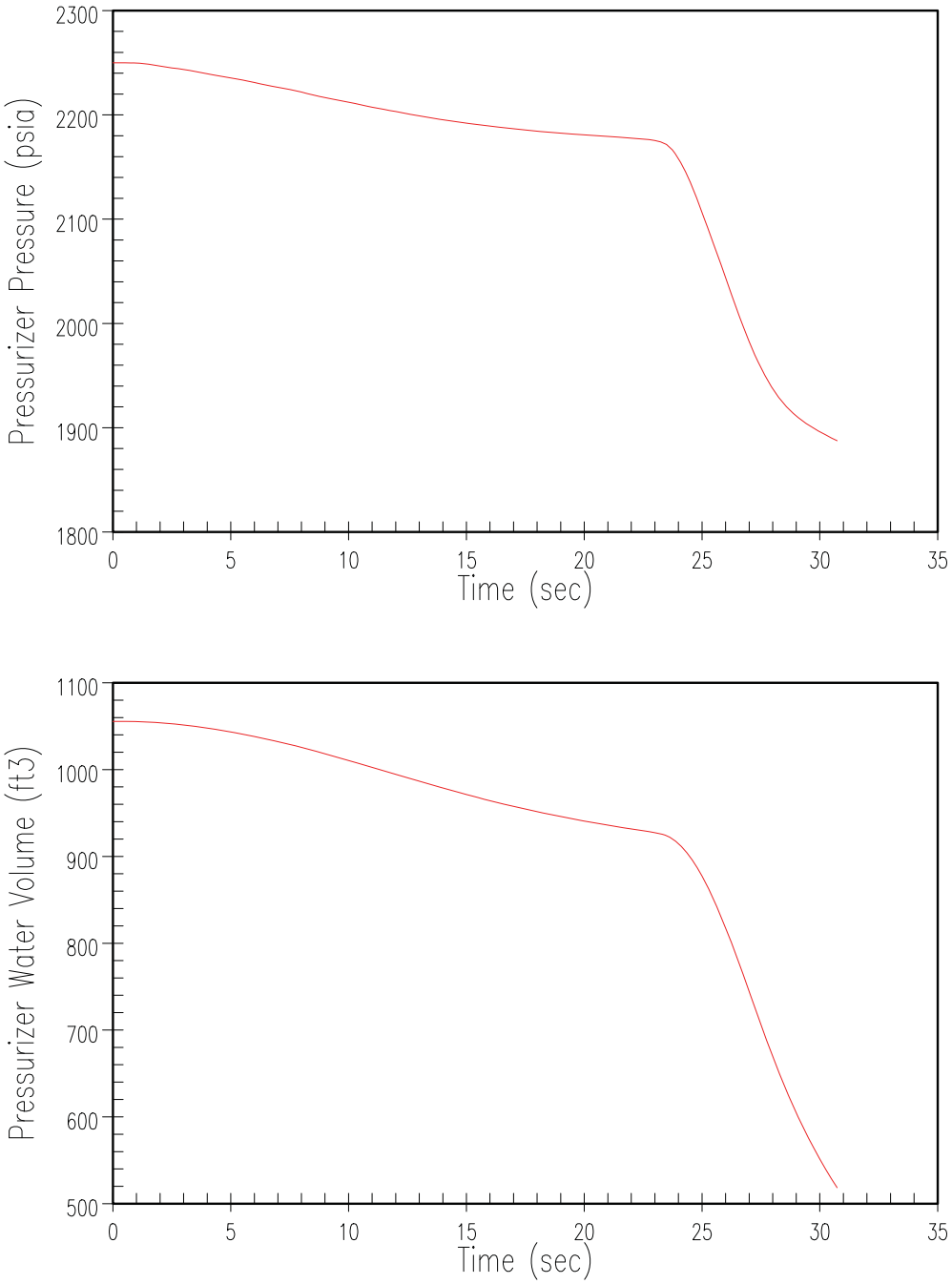

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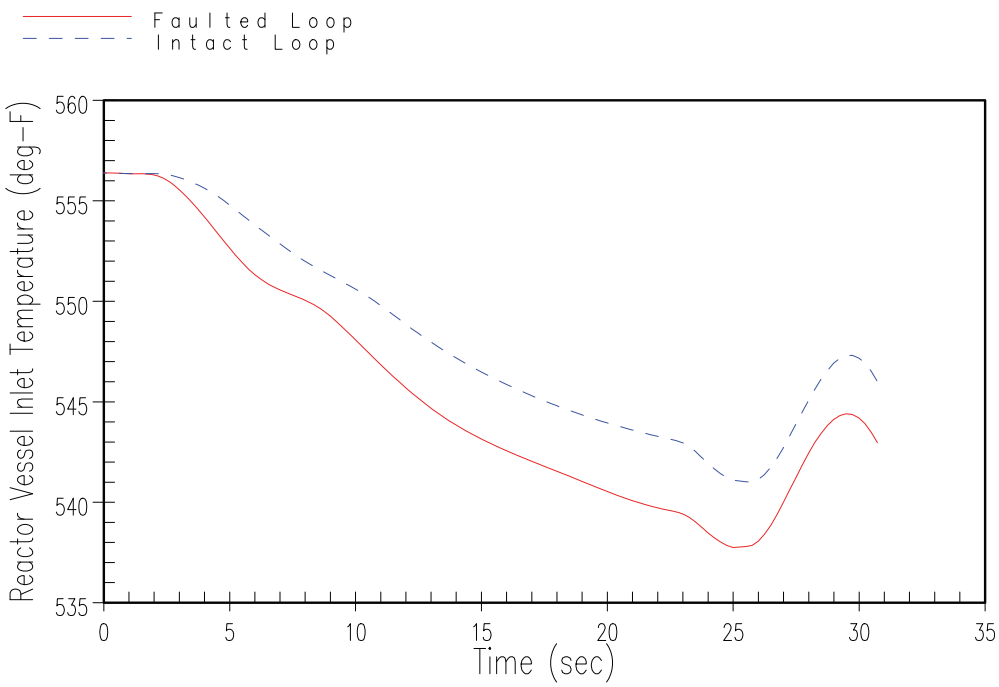
Figure 15.1-28

Pressurizer Pressure and Pressurizer Water Volume vs. Time for SLB Transient at Hot Full Power



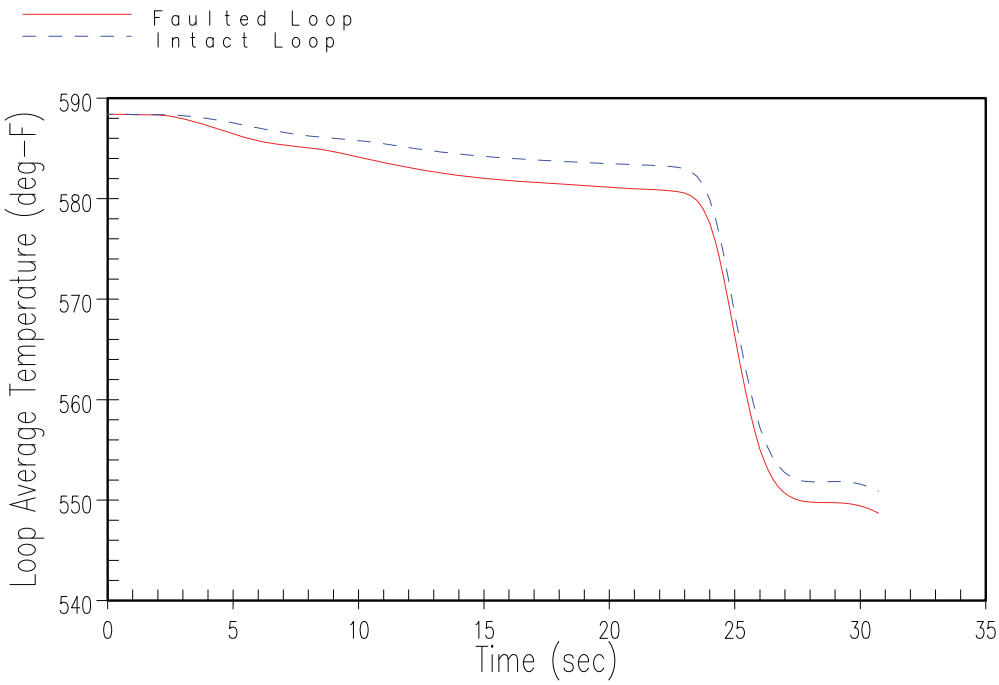


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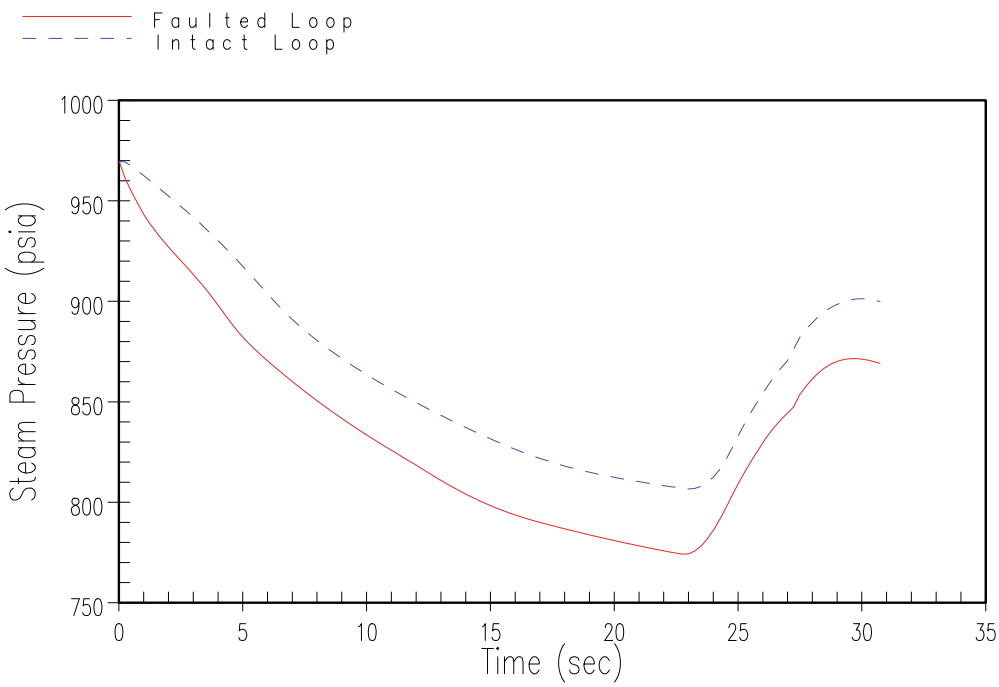
Figure 15.1-29

Reactor Vessel Inlet Temperature and Loop Average Temperature vs. Time for SLB Transient at Hot Full Power



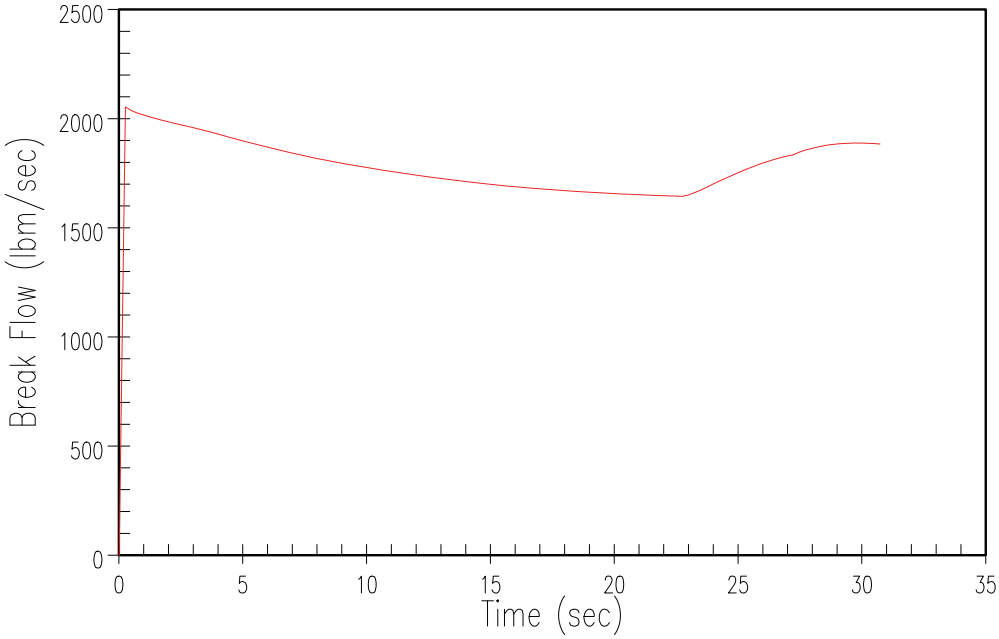


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Figure 15.1-30

Steam Pressure and Break Flow vs. Time for SLB Transient at Hot Full Power

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 the following events, which have been identified as more limiting than the others:

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

b. Loss of external electrical load

c. Turbine trip

d. Inadvertent closure of main steam isolation valves

e. Loss of condenser vacuum and other events resulting in turbine trip

f. Loss of nonemergency ac power to the station auxiliaries

g. Loss of normal feedwater flow

h. Feedwater system pipe break

The above items are considered to be ANS Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event. Section 15.0.1 contains a discussion of ANS classifications and applicable acceptance criteria.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the loss of ac power accident of Section 15.2.6. Therefore, the radiological consequences are only reported for that limiting case.

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

There are no steam pressure regulators in the Wolf Creek unit whose failure or malfunction could cause a steam flow transient. Therefore, this event is not applicable to the Wolf Creek Generating Station.

15.2.2 LOSS OF EXTERNAL ELECTRICAL LOAD

15.2.2.1 Identification of Causes and Accident Description

A major load loss on the plant can result from a loss of external electrical load due to some electrical system disturbance. Offsite ac power remains available to operate plant components, such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. The automatic turbine bypass system would accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser were not available, the excess steam generation would be relieved to the atmosphere. Additionally, main feedwater flow would be lost if the condenser were not available. For this situation, feedwater flow would be maintained by the auxiliary feedwater system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would trip from the reactor protection system if a safety limit were approached. A continued steam load of approximately 5 percent would exist after a total loss of external electrical load, because of the steam demand of plant auxiliaries.

Following the loss of load, protection would be provided by high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips should a safety limit be approached. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz. This resulting overfrequency is not expected to damage the voltage and frequency sensors in any way. Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. For a postulated loss of load and subsequent turbine-generator overspeed, any overfrequency condition is not seen by other safety-related pump motors, reactor protection system equipment, or other safety-related loads. Safety-related loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor protection system equipment is supplied from the 120-Volt ac instrument power supply system which, in turn, is supplied from the inverters; the inverters are supplied from a dc bus energized from batteries or by a rectified ac voltage from safety-related busses.

In the event that the turbine bypass valves fail to open following a large loss of load, the steam generator safety valves may lift, and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperature will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the turbine bypass system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly (RCCA) control, or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safety features rating (see section 5.2.2.1) from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

A more complete discussion of overpressure protection can be found in Reference 1.

A loss of external load is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

A loss of external load event results in a nuclear steam supply system transient that is bounded by the turbine trip event analyzed in Section 15.2.3. Therefore, a detailed transient analysis is not presented for the loss of external load event.

The primary side transient is caused by a decrease in heat transfer capability from primary to secondary, due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feedwater flow not be reduced, a larger heat sink would be available and the transient would be less severe). Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure (see section 10.2.2.2). The analysis performed for the turbine trip event assumes an instantaneous loss of steam flow, therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss of external load than for the turbine trip, due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-6.

15.2.2.2 Analysis of Effects and Consequences

Refer to Section 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis bound those expected for the loss of external load, as discussed in Section 15.2.2.1.

Plant systems and equipment which may be required to function to mitigate the effects of a complete loss of load are discussed in Section 15.0.8 and listed in Table 15.0-6.

The reactor protection system may be required to function to terminate core heat input and to prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. Normal reactor control systems and engineered safety systems are not required to function. The auxiliary feedwater system may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event (see Section 15.2.3) and considerations described in Section 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met.

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (Direct reactor trip on turbine trip is blocked below 50% power by the P-9 interlock) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (see section 10.2.2.2) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. Generator trip
b. Low condenser vacuum
c. Loss of lubricating oil
d. Turbine thrust bearing failure
e. Turbine overspeed
f. Manual trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass, and, if above 50-percent power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure, with a resultant primary system transient as described in Section 15.2.2.1 for the loss of external load event. A slightly more severe transient may occur for the turbine trip event, due to the more rapid loss of steam flow caused by the more rapid valve closure (see section 10.2.2.2), hence a more rapid loss of primary-to-secondary heat transfer.

The automatic turbine bypass system would accommodate up to 40 percent of rated steam flow. Reactor coolant temperatures and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control system are functioning properly. If the condenser were not available, the excess steam generation would be relieved to the atmosphere, and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the auxiliary feedwater system to ensure adequate residual and decay heat removal capability. Should the turbine bypass system fail to operate, the steam generator safety valves may lift to provide pressure control. See Section 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

A turbine trip is more limiting than a loss of external load, loss of condenser vacuum, and other events which result in a turbine trip as a result of the potential for a more rapid loss of steam flow during the turbine trip. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Section 15.2.3.2.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from the analyzed power level, without direct reactor trip, primarily to show the adequacy of the pressure relieving devices, and also to demonstrate core protection margins; that is, the turbine is assumed to trip

without actuating all the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program RETRAN with the model documented in Reference 6. This RETRAN model has been found acceptable by the NRC for use as a licensing basis safety analysis code. The RETRAN model uses a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

The turbine trip event is analyzed for overpressurization concerns assuming initial conditions that include allowances for measurement errors. DNBR is evaluated using Westinghouse RTDP methodology (Ref. 7) which assumes nominal initial conditions.

The major assumptions used in the analysis are summarized below:

a. Initial operating conditions (RCS Overpressure)
   Initial NSSS power is assumed to be 3651 MWt, which includes all applicable uncertainties. The initial RCS pressure is assumed at a minimum value consistent with steady state full power operation, including allowances for calibration and instrument errors. This is to maximize the heat up time prior to reactor trip and therefore, maximizing the peak RCS pressure and peak pressurizer level during the transient. The initial RCS temperature is assumed at the maximum value that results in actuation of the steam generator safety valves after the peak RCS pressure occurs.

b. Initial operating conditions (DNB)
   Initial NSSS power is assumed to be 3651 MWt, which includes all applicable uncertainties. The initial pressure and RCS temperatures are assumed at their nominal values consistent with steady state full power operation. Allowances for calibration and instrument errors are treated statistically by the DNBR evaluation code.

c. Initial operating conditions (SG Overpressure)
   Initial NSSS power is assumed to be 3651 MWt, which includes all applicable uncertainties. The initial RCS pressure is assumed at a minimum value consistent with steady state full power operation, including allowances for calibration and instrument errors. This is to maximize the heat up time prior to reactor trip and therefore, maximize the peak SG pressure during the transient. The initial RCS temperature is at the maximum value consistent with steady state full power operation, including allowances for calibration and instrument errors.

d. Reactivity coefficients:
The turbine trip is analyzed assuming minimum reactivity feedback. These cases assume a moderator temperature coefficient of 0 pcm/°F and the least negative Doppler power coefficient. Maximum reactivity feedback cases have been determined to be non-limiting with respect to both DNB and peak pressure concerns. These cases are no longer analyzed.

e. Reactor control
It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

f. Steam release
No credit is taken for the operation of the turbine bypass system or steam generator atmospheric relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

g. Pressurizer spray and power-operated relief valves

1. DNB
Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

2. RCS Overpressure
No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable. This case conservatively accounts for the effects of the pressurizer safety valve loop seals, as described in Reference 2.

3. SG overpressure
Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the primary coolant pressure, thereby delaying the time to reactor trip.

h. Feedwater flow
Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow, since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

i. Reactor trip is actuated by the first reactor protection system trip setpoint reached, with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, and low-low steam generator water level.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of a turbine trip event are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-11.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. Normal reactor control system and engineered safety systems are not required to function.

Results

The transient responses for a turbine trip from full power operation at the analyzed power level are shown for three cases in Figures 15.2-1 through 15.2-6. One case is presented without pressure control (i.e., PORVs, pressurizer spray) to ensure that 110% of the RCS design pressure (2750 psia) is not exceeded, one case using nominal initial plant conditions and pressure control is analyzed to ensure that the DNB design basis is satisfied, and one case is presented with pressure control to ensure that 110% of the SG design pressure is not exceeded. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-1 and 15.2-2 show the transient responses for the total loss of steam load RCS overpressure evaluation with minimum reactivity feedback, assuming no credit is taken for the steam bypass. The reactor is tripped on the high pressurizer pressure signal. In this case, the pressurizer safety valves are actuated, and maintain RCS pressure below 110 percent of the design value.

The DNB evaluation for the turbine trip event was analyzed assuming full pressure control and nominal initial conditions. The errors in initial conditions are treated statistically and included in the DNBR safety analysis limit. Figures 15.2-3 and 15.2-4 show the transient responses for the total loss of steam load with minimum reactivity feedback for the DNB evaluation. No credit is taken for the steam bypass. The reactor is tripped on the overtemperature ΔT trip signal. Minimum DNBR remains above the safety analysis limit.

Figures 15.2-5 and 15.2-6 show the transient responses for the total loss of steam load with minimum reactivity feedback for SG overpressurization. No credit is taken for the steam bypass. The reactor is tripped on the overtemperature ΔT signal. The steam generator safety valves are actuated and maintain SG pressure below 110 percent of the design value.

Reference 1 presents additional results of analysis for a complete loss of heat sink, including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Conclusions

Results of the analyses show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The analyses show that the DNBR will not decrease below the safety analysis limit at any time during the transient. Thus, the DNB design basis, as described in Section 4.4, is met.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

Inadvertent closure of the main steam isolation valves would result in a turbine trip with no credit taken for the turbine bypass system. Turbine trips are discussed in Section 15.2.3.

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

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser; however, since steam dump is assumed to be unavailable in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Section 15.2.3.1, are covered by Section 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Section 15.2.2.1, and are not a concern for this type of event.

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

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency ac power may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite ac distribution system.

The DNB transient is more severe than the turbine trip event analyzed in Section 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip: 1) due to turbine trip, 2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal, or 3) due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below will occur:

a. Plant vital instruments are supplied from emergency dc power sources.

b. As the steam system pressure rises following the trip, the steam generator atmospheric relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the atmospheric relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

c. As the no-load temperature is approached, the steam generator atmospheric relief valves (or safety valves, if the atmospheric relief relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

d. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The auxiliary feedwater system is started automatically, as follows:

a. Two motor-driven auxiliary feedwater pumps are started on any of the following:

   1. Low-low level in any steam generator

   2. Any safety injection signal

   3. Loss of offsite power

   4. Manual actuation

b. The turbine-driven auxiliary feedwater pump is started on any of the following:

   1. Low-low level in any two steam generators

   2. Loss of offsite power

   3. Manual actuation

The motor-driven auxiliary feedwater pumps are supplied power by the diesels, and the turbine-driven pump utilizes steam from the secondary system. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank (or the essential service water system, if the condensate storage tank is unavailable) for delivery to the steam generators.

A loss of nonemergency ac power to the station auxiliaries is classified as an ANS Condition II event, fault of moderate frequency. This event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power, which was analyzed in Section 15.2.3. However, a loss of ac power to the station auxiliaries, as postulated above, could also result in a loss of normal feedwater if the condensate pumps lose their power supply.

Following the reactor coolant pump coastdown caused by the loss of ac power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. The DNB transient for this event is bounded by the complete loss of flow event. Thus, the intent of the analysis presented here is to show that the natural circulation flow in the RCS following a loss of normal feedwater with a loss of ac power event is sufficient to remove residual heat from the core. Section 15.2.7 discusses the analysis of the loss of normal feedwater with ac power.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-12.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis of the loss of non-emergency AC transient has been performed using the RETRAN computer code (Ref 5). RETRAN has been found acceptable by the NRC for use as a licensing basis safety analysis code (Ref 6). RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

The assumptions used in the analysis are as follows:

a. The plant is initially operating at an NSSS power of 3651 MWt, which includes all applicable uncertainties.
b. A conservative core residual heat generation based upon long-term operation at the initial power level preceding the trip.

c. The auxiliary feedwater system is actuated by the low-low steam generator water level signal, assumed to occur at 0% of narrow range span. The analysis assumes a total of 880 gpm is evenly delivered to 4 steam generators. This is a conservative minimum value for the AFW flow and bounds either a single failure of the turbine-driven AFW pump or one motor-driven AFW pump (e.g., diesel generator failure). A delay of 60 seconds is assumed, which accounts for diesel generator and pump start, before auxiliary feedwater flow is initiated. The analysis conservatively accounts for the purging of the hotter main feedwater in the feedwater piping, which delays the delivery of the relatively cold auxiliary feedwater to the steam generators.
d. Reactor trip occurs when the steam generator level reaches 0.0% of narrow range span.
e. The pressurizer sprays and heaters are assumed operable. This maximizes the peak transient pressurizer water volume.
f. The steam dump system and steam generator atmospheric relief valves are assumed to be unavailable. Secondary system steam relief is assumed through the main steam safety valves (MSSVs) only.
g. Cases were analyzed to cover bounding combinations of the initial vessel average temperature (high/low plus and minus uncertainties), initial pressurizer pressure (nominal plus and minus uncertainties), initial feedwater temperature (high and low), steam generator tube plugging level (minimum and maximum), and with and without the pressurizer power-operated relief valves available.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Results

The transient response of the RCS following a loss of ac power is shown in Figures 15.2-9 and 15.2-10. The calculated sequence of events for this event is listed in Table 15.2-1.

The RETRAN code results demonstrate that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown. Table 15.2-4 presents historical information related to natural circulation flow test criteria that was used during the initial plant startup testing.

The analysis begins with a loss of normal feedwater resulting from a loss of non-emergency AC power. This causes the steam generator level to fall, the RCS temperature and pressurizer pressure to rise. When the reactor trips on low-low level, the reactor power drops toward decay heat levels, and the RCPs subsequently trip. Due to the reactor power reduction following reactor trip, the RCS temperature falls, as do the pressurizer pressure and level.

Following reactor trip, the focus of the analysis becomes the falling steam generator inventory. While the inventory is high, there will be sufficient heat transfer to remove the decay heat. However, the timely arrival of auxiliary feedwater is sufficient to reverse the steam generator inventory reduction at approximately 222 seconds.

Results of the analysis show that, for the loss of non-emergency AC to plant auxiliaries, all safety criteria are met. Auxiliary feedwater capacity is sufficient to prevent water relief through the pressurizer relief or safety valves; this assures that the transient does not propagate into a more serious transient. The calculated peak pressurizer water volume is approximately 1623 ft^3, which is less than the limit of 1800 ft^3.

15.2.6.3 Radiological Consequences

15.2.6.3.1 Method of Analysis

15.2.6.3.1.1 Physical Model

The dose calculation for loss of ac power assumed that heat removal from the nuclear steam supply system is achieved by venting the steam for 12 hours.

The reactor coolant is assumed to be contaminated by radioactive fission products introduced through fuel cladding defects. The secondary system is contaminated by the inleakage of reactor coolant through postulated steam generator tube leaks.

The radioactivity in the vented steam is dispersed in the atmosphere without any reduction due to plateout, fallout, filtering, etc.

15.2.6.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are found in Tables 15.2-2 and 15.A-1. The assumptions used to determine the activity released are as follows:

a. Two different cases are examined with regards to initial reactor coolant activity. Both cases assume an initial reactor coolant iodine activity equal to the dose equivalent of 1.0 microCi/gm of I-131. The first case, referred to as an accident-initiated iodine spike, assumes an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 500 immediately after the accident occurs. This increased escape rate is assumed for 8 hours. The second case, referred to as a pre-accident iodine spike, assumes a reactor transient occurs prior to the postulated LOAC event that has raised the iodine concentration to the maximum value allowed by TS LCO 3.4.16, 60 microCi/gm of DE I-131.

b. The reactor coolant activity assumed for noble gas is the Technical Specification limit of 500 microCi/gm Xe-133 dose equivalent.

c. The reactor coolant system activity assumed for alkali metals is based on 1% fuel defects, as provided in Table 11.1-5.

d. The initial steam generator iodine activity assumed is the Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent.

e. The initial steam generator alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity).

f. A 1-gpm steam generator primary-to-secondary leakage is assumed for the duration of steam venting.

g. For noble gases, the activity released is taken to be the activity introduced by reactor coolant inleakage without holdup in the steam system.

h. The iodine activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed with the iodine activity initially present in the steam generators. The iodine partition factor provided in Table 15.2-2 is utilized to determine the iodine activity released via steam venting from the steam generators.

i. The alkali metal activity present in the primary-to-secondary leakage is assumed to be homogeneously mixed with the alkali metal activity initially present in the steam generators. The full power moisture carryover provided in Table 15.2-2 is utilized to determine the alkali metal activity released via steam venting from the steam generators.

j. The atmospheric dispersion factors are given in Table 15A-2.

15.2.6.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated using the onsite meteorological measurement programs described in Section 2.3.

c. TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low population zone, and in the control room were analyzed using the models described in Appendix 15A.

15.2.6.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activities

Normal activity paths from the secondary system, such as the condenser air removal system and steam generator blowdown, cease during station blackout. The steam is released to the atmosphere through the:

a. Atmospheric relief valves

b. Main steam safety valves

Since all these paths are taken as direct to the atmosphere without any form of decontamination, they are all radiologically equivalent and need not be distinguished.

15.2.6.3.2 Identification of Uncertainties in, and Conservative Aspects of, the Analysis

The principal uncertainties in the dose calculation arise from the uncertainties in the accident circumstances, particularly the extent of steam contamination, the weather at the time, and delay before preferred ac power is restored. Each of these uncertainties is handled by making very conservative or worst-case assumptions.

a. Reactor coolant iodine and noble gas activities are based on the Technical Specification limits, which are significantly higher than the activities associated with normal operating conditions.

b. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

c. A 1-gpm steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation.

d. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the evaluated radiological consequences, based on the meteorological conditions assumed, will be conservative.

15.2.6.3.3 Conclusions

15.2.6.3.3.1 Filter Loadings

No filter serves to limit the release of radioactivity in this accident. There is no significant activity buildup on any filters as a consequence of loss of ac power.

15.2.6.3.3.2 Doses to Receptor at Exclusion Area Boundary, Low Population Zone Outer Boundary, and in the Control Room

The maximum doses to an individual who spends 2 hours during the most limiting time in the event after loss of ac power at the exclusion area boundary or at the outer boundary of the low-population zone and in the control room for 30 days are given in Table 15.2-3 for each case. These doses are within the guideline values of 10 CFR 50.67 for offsite locations and the 10 CFR 50.67 guideline value in the control room.

15.2.6.4 Conclusions

Results of the analysis show that, for the loss of non-emergency ac power to plant auxiliaries event, all safety criteria are met.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS.
The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The following occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

a. As the steam system pressure rises following the trip, the steam generator atmospheric relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the atmospheric relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

b. As the no-load temperature is approached, the steam generator atmospheric relief valves (or safety valves, if the atmospheric relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

The auxiliary feedwater system is started automatically, as discussed in Section 15.2.6.1. The turbine-driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven auxiliary feedwater pumps are supplied power by offsite power sources. The auxiliary feedwater pumps take suction directly from the condensate storage tank (or essential service water system, if the condensate storage tank is unavailable) for delivery to the steam generators.

An analysis of the system transient is presented below to show that, following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis of the loss of normal feedwater transient has been performed using the RETRAN computer code (Ref 5). RETRAN has been found acceptable by the NRC for use as a licensing basis safety analysis code (Ref 6). RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

The assumptions used in the analysis are as follows:

a. The plant is initially operating at an NSSS power of 3651 MWt, which includes all applicable uncertainties.

b. A conservative core residual heat generation, based upon long-term operation at the initial power level preceding the trip.

c. Reactor trip occurs on steam generator low-low level at 0.0% of narrow range span.

d. The auxiliary feedwater system is actuated by the low-low steam generator water level signal, assumed to occur at 0% of narrow range span. The analysis assumes a total of 880 gpm is evenly delivered to 4 steam generators. This is a conservative minimum value for the AFW flow and bounds either a single failure of the turbine-driven AFW pump or one motor-driven AFW pump (e.g., diesel generator failure). A delay of 60 seconds is assumed, which accounts for diesel generator and pump start, before auxiliary feedwater flow is initiated. The analysis conservatively accounts for the purging of the hotter main feedwater in the feedwater piping, which delays the delivery of the relatively cold auxiliary feedwater to the steam generators.

e. Secondary system steam relief is achieved through the steam generator safety valves.

f. The pressurizer sprays and heaters are assumed operable. This maximizes the peak transient pressurizer water volume.

g. Cases were analyzed to cover bounding combinations of the initial vessel average temperature (high/low plus and minus uncertainties), initial pressurizer pressure (nominal plus and minus uncertainties), initial feedwater temperature (high and low), steam generator tube plugging level (minimum and maximum), and with and without the pressurizer power-operated relief valves available.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the auxiliary feedwater system) in removing long-term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value, and the reactor trips via the low-low steam generator level trip. The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-13. Normal reactor control systems are not required to function. The reactor protection system is required to function following a loss of normal feedwater, as analyzed here. The auxiliary feedwater system is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of anticipated transients without scram (ATWS) considerations is presented in Section 15.8.

Results

Figures 15.2-11 through 15.2-12 show the significant plant parameters following a loss of normal feedwater.

The analysis begins with a loss of normal feedwater. This causes the steam generator level to fall, the RCS temperature to increase gradually and the pressurizer pressue and liquid level to rise. When the reactor trips on low-low steam generator level, the reactor power drops toward decay heat levels. Due to the reactor power reduction, the RCS temperature falls, as do the pressurizer pressure and level.

Following reactor trip, the focus of the transient analysis becomes the falling steam generator inventory. While the inventory is high, there will be sufficient heat transfer to remove the core decay heat. However, the timely arrival of auxiliary feedwater is sufficient to reverse the steam generator inventory reduction at approximately 431 seconds. The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators is sufficient to dissipate core residual heat without water relief through the pressurizer relief or safety valves. Figure 15.2-12 shows that at no time is there water relief from the pressurizer.

The results of the analysis show that, for the loss of normal feedwater event, all safety analysis criteria are met.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2-11 and 15.2-12, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system, since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves.

15.2.8 FEEDWATER SYSTEM PIPE BREAK

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Sections 15.2.6 and 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

a. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

b. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

c. The break may be large enough to prevent the addition of any main feedwater after trip.

An auxiliary feedwater system functions to ensure the availability of adequate feedwater so that:

a. No substantial overpressurization of the RCS and MSS occurs (less than 110 percent of design pressures); and

b. Sufficient liquid in the RCS is maintained so that the core remains in place and geometrically intact with no loss of core cooling capability.

A major feedwater line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

The severity of the feedwater line rupture transient depends on a number of system parameters, including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. Sensitivity studies presented in Reference 3 illustrate many of the limiting assumptions for the feedwater line rupture. In addition, the major assumptions pertinent to this analysis are defined below.

The main feedwater control system is assumed to fail due to an adverse environment. The water levels in all steam generators are assumed to decrease equally until the low-low steam generator level reactor trip setpoint is reached. After reactor trip, a double-ended rupture of the largest feedwater line is assumed. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analyses have been performed at full power, with and without loss of offsite power, with credit taken for the pressurizer power-operated relief valves but no SI actuation modeled. For the case without offsite power available, the power is assumed to be lost at the time of reactor trip. This is more conservative than the case where power is lost at the initiation of the event. These cases are analyzed below.

The following provides the protection for a main feedwater line rupture:

a. A reactor trip on any of the following conditions:

1. High pressurizer pressure
2. Overtemperature ΔT

3. Low-low steam generator water level in any steam generator
4. Safety injection signals from any of the following (See items b. and m. in Section 15.2.8.2): 1) two-out-of-three low steam line pressure in any one loop or 2) two-out-of-three high containment pressure (hi-1)

Refer to Chapter 7.0 for a description of the actuation system.

b. The auxiliary feedwater system provides an assured source of feedwater to the steam generators for decay heat removal. Refer to Section 10.4.9 for a description of the auxiliary feedwater system.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-14.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis of the feedwater line break event has been performed using the RETRAN computer code (Ref 5). RETRAN has been found acceptable by the NRC for use as a licensing basis safety analysis code (Ref 6). RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

Major assumptions used in the analysis are as follows:

a. The plant is initially operating at an NSSS power of 3651 MWt, which includes all applicable uncertainties. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

b. Operation of the pressurizer power operated relief valves is modeled in the feedwater line rupture analysis to minimize RCS pressure, which produces lower (more limiting) saturation temperatures within the RCS. To ensure the conservatism of this assumption, safety injection flow, which would increase with reduced RCS pressure, is set to zero. Pressurizer spray and heaters are both assumed to be inoperable.

c. Initial pressurizer level is at the nominal programmed value plus 7 percent (error); initial steam generator water level in all SGs is at the nominal value, plus 10 percent of the narrow range span. Since this transient is a heatup event which leads to a pressurization of the RCS, a higher SG water level tends to prolong the initiating event sequence prior to reactor trip on low-low SG level, increasing the initial RCS heatup.

d. Conservative core residual heat generation is based on ANSI/ANS-5.1-1979 (Reference 4), which is a conservative representation of the decay energy release rates.

e. Main feedwater is assumed to be lost to all SGs at event initiation due to a malfunction in the feedwater control system under an adverse environment. The feedwater line break, and subsequent reverse blowdown of the faulted SG, is conservatively assumed to occur when the SG inventory reaches 0% narrow range span. The combination of conditions described above yields the most severe feedwater line break transient, with control and protection interaction considered.

f. The worst possible break area is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.

g. Choked flow at the break is assumed.

h. Failure of one protection train is taken as the worst single failure and results in one AFW pump being inoperable. The total AFW flow delivered to the three intact SGs is assumed to be 594.4 gpm. This is a conservative minimum value for the AFW flow following a FLB and bounds either a single failure of the turbine-driven AFW pump or one motor-driven AFW pump. The distribution of the 594.4 gpm total AFW flow is 222.7 gpm to each of two intact SGs with the third intact SG receiving 149 gpm. No flow is modeled as being delivered to the SG in the faulted loop.

i. The auxiliary feedwater system is actuated by the low-low steam generator water level signal. A 60-second delay in the delivery of AFW flow was assumed following the low-low level signal to allow time for startup of the standby diesel generators and the auxiliary feedwater pumps. The analysis conservatively accounts for the purging of the hotter main feedwater in the feedwater piping, which delays delivery of the relatively cold auxiliary feedwater flow to the SGs.

j. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

k. No credit is taken for charging or letdown.

l. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

1. High pressurizer pressure
2. Overtemperature ΔT
3. High pressurizer level
4. High containment pressure

m. Although the analysis for the feedline rupture accident predicts actuation of the SI system, the analysis conservatively does not model any SI flow.

n. Both minimum and maximum reactivity feedback conditions have been considered. Consistent with the limiting conditions determined for each scenario, the case analyzed with offsite power available throughout the transient models maximum reactivity feedback while the case without offsite power models minimum reactivity feedback. The minimum reactivity feedback input includes a least-negative moderator temperature coefficient (MTC) value of 0 pcm/°F.

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor-driven auxiliary feedwater pumps, which in turn initiates auxiliary feedwater flow to the steam generators. The turbine-driven auxiliary feedwater pump is initiated if the low-low steam generator water level signal is reached in at least two steam generators. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes all main steam line isolation valves.

Emergency operating procedures following a feedwater system pipe rupture require the following actions to be taken by the reactor operator:

a. Isolate feedwater flow spilling from the ruptured feedwater line and align the system so that the level in the intact steam generators is recovered.

b. High head safety injection should be terminated in accordance with the emergency operating procedures.

Subsequent to terminating high head safety injection, plant operating procedures are followed in cooling the plant to a safe shutdown condition. However, these actions are taken during the reactor recovery phase and are not modeled in the safety analyses.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

The reactor protection system is required to function following a feedwater line rupture as described above. No single active failure will prevent operation of this system.

The engineered safety system assumed to function in the analysis is the auxiliary feedwater system. For the auxiliary feedwater system, the worst case configuration has been used, i.e., only three intact steam generators receive auxiliary feedwater following the break. A discharge flow control device, located on the auxiliary feedwater line to each steam generator, is assumed to regulate the flow from the motor-driven auxiliary feedwater pump to the faulted steam generator. The analysis models a total auxiliary feedwater flow of 594.4 gpm, which reflects the system alignment that produces the minimum auxiliary feedwater flow being provided to the three intact steam generators following a feedline rupture. The distribution of flow is 222.7 gpm to each of two intact steam generators with the third intact steam generator receiving 149 gpm. No flow is modeled as being delivered to the steam generator in the faulted loop.

Two separate cases are analyzed, i.e., with and without offsite power. Where offsite power is available, the reactor coolant pumps continue to operate after the reactor trip. In the case where offsite power is not available, the reactor coolant pumps coast down upon loss of offsite power which is assumed to occur at the reactor trip. For the case without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown (in Section 15.2.6) to be sufficient to remove core decay heat following reactor trip, for the loss of ac power transient. Pump coastdown characteristics are demonstrated in Sections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A detailed description and analysis of the safety injection system is provided in Section 6.3. The auxiliary feedwater system is described in Section 10.4.9.

Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2-13 through 15.2-22. Results for the case with offsite power available are presented in Figures 15.2-13 through 15.2-17. Results for the case where offsite power is lost are presented in Figures 15.2-18 through 15.2-22. The calculated sequence of events for both cases analyzed are listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. In both cases, the primary and secondary pressures increase prior to reactor trip. After reactor trip occurs on low-low SG water level, the pressure decreases sharply, due to the cooldown caused by the break, until steam line isolation occurs. The pressure in the faulted SG continues to decrease, while the pressure in the intact SGs and primary side begins to increase (cooldown ends, heat-up begins) until the safety valve settings are reached. Results presented in Figures 15.2-14 and 15.2-17 (with offsite power available), and Figures 15.2-19 and 15.2-22 (without offsite power) show the predicted pressures in the RCS and main steam system.

The primary temperatures are stable or increase slightly prior to reactor trip and decrease sharply during cooldown after reactor trip. Once the heat-up begins, the primary temperature increases until the heat removal capability of the intact SGs, with the inventory maintained by the AFW System, equals the decay heat generated in the core plus pump heat ("turnaround" time). The peak primary temperature remains below the saturation temperature although the margin to boiling is decreased. At the predicted time of turn around, the minimum margin to hot leg saturation for the case with offsite power is 65.3°F, and 40.5°F for the case without offsite power. Thus, there is no bulk boiling in the RCS.

In addition, the pressurizer does not fill due to thermal expansion nor does the pressurizer empty. Therefore, the reactor remains covered with water throughout the transient.

15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed auxiliary feedwater system capacity is adequate to remove decay heat and to prevent uncovering the reactor core. Radioactivity doses from the postulated feedwater lines rupture are less than those previously presented for the postulated steam line break. All applicable acceptance criteria are therefore met.

15.2.9 REFERENCES

1. Cooper, L., Miselis, V. and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June, 1972 (also letter NS-CE-622, dated April 16, 1975, C. Eicheldinger (Westinghouse) to D.B. Vassallo (NRC), additional information on WCAP-7769, Revision 1).

2. Barrett, G. O., et al., "Pressurizer Safety Valve Set Pressure Shift," WCAP-12910 Rev. 1-A, May 1993.

3. Lang, G. E. and Cunningham, J. P., "Report on the Consequences of a Postulated Main Feedline Rupture," WCAP-9230 (Proprietary) and WCAP-9231 (Non-Proprietary), January 1978.

4 ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

5. McFadden, J. H., et al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI-NP-1850-CCM-A, October 1984.

6. Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

7. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Fiedland, et al., April 1989.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
Turbine Trip		
1. RCS Overpressure evaluation without pressurizer pressure control (minimum feedback)	Turbine trip; Loss of main feedwater flow	0.0
	Initiation of steam release from pressurizer relief valve	NONE
	High pressurizer pressure reactor trip setpoint reached	6.6
	Rods begin to drop	7.6
	Initiation of steam release from pressurizer safety valve	8.2
	Peak RCS pressure occurs	9.7
	Initiation of steam release from steam generator safety valves	12.1
2. DNB evaluation with pressurizer pressure control (minimum feedback)	Turbine trip; Loss of main feedwater flow	0.0
	Initiation of steam release from pressurizer relief valve	2.8
	Initiation of steam release from pressurizer safety valve	9.7
	Initiation of steam release from steam generator safety valves	9.7
	Overtemperature ΔT reactor trip setpoint reached	17.0
	Minimum DNBR occurs	19.8
	Rods begin to drop	20.0

TABLE 15.2-1 (Sheet 2)

Accident	Event	Time (sec)
Turbine Trip		
3. SG overpressure evaluation with pressurizer pressure control (minimum feedback)	Turbine trip; Loss of main feedwater flow	0.0
	Initiation of steam release from pressurizer relief valve	1.7
	Initiation of steam release from steam generator safety valves	4.7
	Initiation of steam release from pressurizer safety valve	9.6
	Overtemperature ΔT reactor trip setpoint reached	15.8
	Rods begin to drop	18.8
	Peak SG pressure occurs	22.0
Loss of nonemergency ac power to the station auxiliaries	Main feedwater flow stops	0.0
	Low-low steam generator water level trip	37.7
	Rods begin to drop and turbine trip initiated	39.7
	Reactor coolant pumps begin to coast down	41.7
	On each loop, the steam generator safety valve with the lowest setting opens	73.2
	Flow from two motor-driven auxiliary feedwater pumps initiated	97.7
	Steam generator inventory reduction reverses	221.5
	Maximum pressurizer water volume occurs	2953.5

TABLE 15.2-1 (Sheet 3)

Accident	Event	Time (sec)
Loss of normal feedwater flow	Main feedwater flow stops	0.0
	Low-low steam generator water level trip	37.8
	Rods begin to drop and turbine trip initiated	39.8
	On each loop, the steam generator safety valve with the lowest setting opens	68.0
	Flow from two motor-driven auxiliary feedwater pumps initiated	97.8
	Steam generator inventory reduction reverses	431.0
	Maximum pressurizer water volume occurs	1372.0

TABLE 15.2-1 (Sheet 4)

Accident	Event	Time (sec)
Feedwater system pipe break with offsite power available	Actions/Events	
	Steady State Operation	0.0
	Feedwater Line Rupture Occurs Causing Loss of Feedwater to All SGs	0.0
	Pressurizer Power Operated Relief Valve Opens (First Occurrence)	13.9
	Low-Low SG Water Level Setpoint Reached	36.2
	Reactor Trip on Low-Low SG Water Level and Faulted SG Begins Discharging Fluid Directly out the Break	38.2
	Pressurizer Power Operated Relief Valve Closes	41.7
	AFW is Delivered to Intact SGs	96.2
	Low Steam Line Pressure SI Setpoint Reached in Faulted SG	132.7
	All MSIVs Closed	149.7
	SG Safety Valve #1 Opens (First Occurrence)	579.0
	Core Decay Heat plus RCP Heat Decrease to AFW Heat Removal Capacity	1700.0

TABLE 15.2-1 (Sheet 5)

Accident	Event	Time (sec)
Feedwater system pipe break without offsite power	Actions/Events	
	Steady State Operation	0.0
	Feedwater Line Rupture Occurs Causing Loss of Feedwater to All SGs	0.0
	Pressurizer Power Operated Relief Valve Opens (First Occurrence)	13.9
	Low-Low SG Water Level Setpoint Reached	36.3
	Reactor Trip on Low-Low SG Water Level and Faulted SG Begins Discharging Fluid Directly out of the Break	38.3
	Power Lost to RCPs	40.3
	Pressurizer Power Operated Relief Valve Closes	42.1
	AFW is Delivered to Intact SGs	96.3
	Low Steam Line Pressure SI Setpoint Reached in Faulted SG	105.6
	All MSIVs Closed	122.6
	SG Safety Valve #1 Opens (First Occurrence)	553.4
	Core Decay Heat Decreases to AFW Heat Removal Capacity	1100.0

TABLE 15.2-2

PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES OF LOSS OF NONEMERGENCY AC POWER

I.	Source Data		
	a.	Core power level, MWt	3637 (includes uncertainty)
	b.	Steam generator tube leakage, gpm	1
	c.	Reactor coolant initial iodine activity	1.0 μCi/gm of I-131 dose equivalent with either an assumed pre-accident iodine spike of 60 μCi/gm of DE I-131 or an accident-initiated iodine spike, which increases the rate of iodine release into the reactor coolant by a factor of 500. The increased rate is assumed for 8 hours.
	d.	Secondary system initial iodine activity	0.1 μCi/gm of I-131 dose equivalent
	e.	Reactor coolant initial noble gas activity	500 μCi/gm of Xe-133 dose equivalent
	f.	Reactor coolant initial alkali metal activity	Based on 1-percent failed fuel as provided in Table 11.1-5
	g.	Iodine partition factor for the steam generators	0.01
	h.	Alkali metal partition factor for steam generators (based on full power moisture carryover of 0.25%)	0.0025
	i.	Each steam generator water mass, lb	
		0-2 hrs	82,500
		>2 hrs (assumes water level reaches 0% narrow range span)	121,250

TABLE 15.2-2 (Sheet 2)

II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
a.	Total primary to secondary leakage 0-12 hr, lb	6006
b.	Steam release from all steam generators 0-2 hours, lb 2-12 hours, lb	 419,846 1,352,918

TABLE 15.2-3

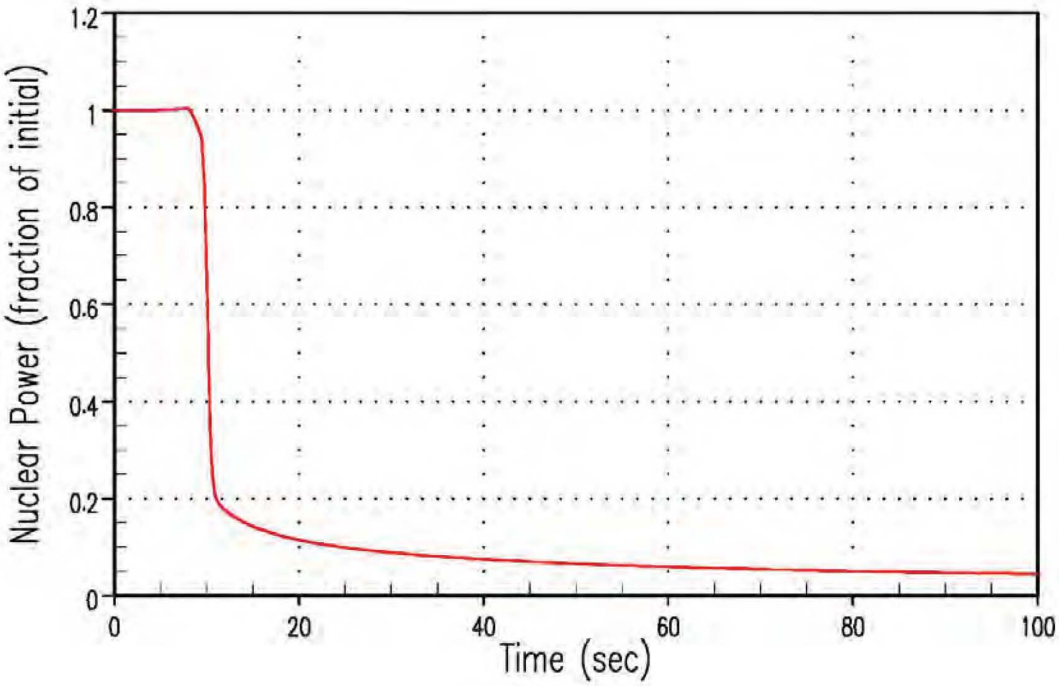
RADIOLOGICAL CONSEQUENCES OF LOSS OF NON-EMERGENCY AC POWER

CASE 1, 1.0 μCi/gm I-131 equivalent w/I spike	Wolf Creek Dose (rem TEDE)
Exclusion area boundary (Limiting 2 hour duration)	1.3E-2
Low-population zone, outer boundary (duration)	4.7E-3
Control Room (30 days)	2.5E+0
CASE 2, 60 μCi/gm I-131 equivalent	
Exclusion area boundary (Limiting 2 hour duration)	1.8E-3
Low-population zone, outer boundary (duration)	1.5E-3
Control Room (30 days)	8.6E-1

TABLE 15.2-4

NATURAL CIRCULATION FLOW

Core Power (percent)	RCS Natural Circulation Flow (percent)
5.0	5.9
4.0	5.4
3.0	4.8
2.0	4.1
1.0	3.2



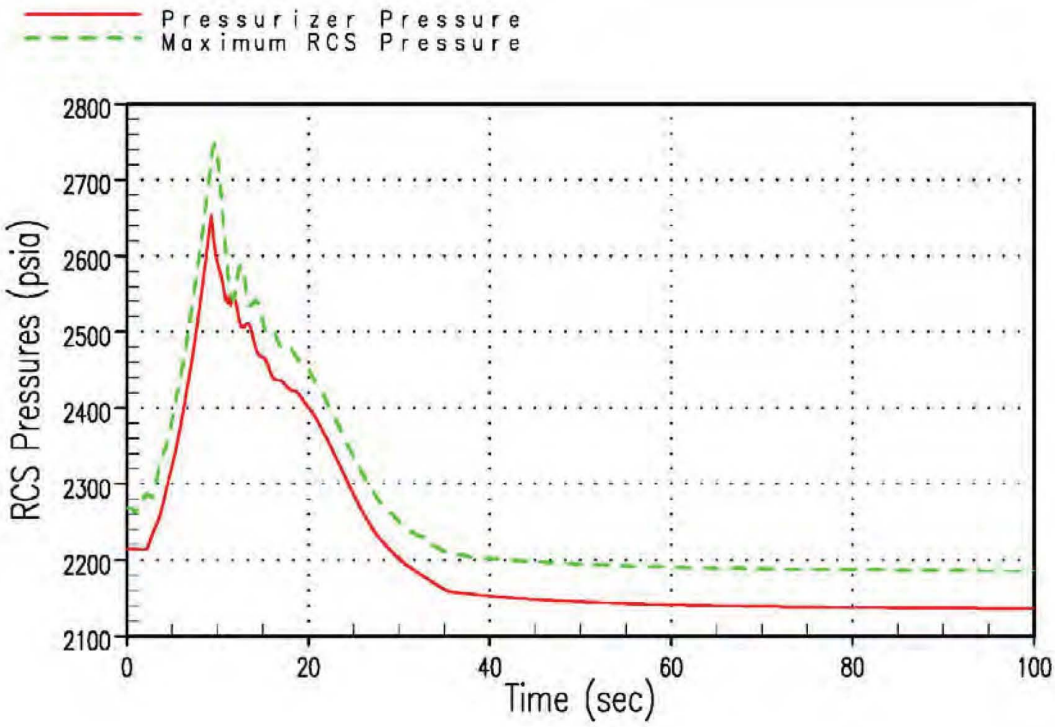


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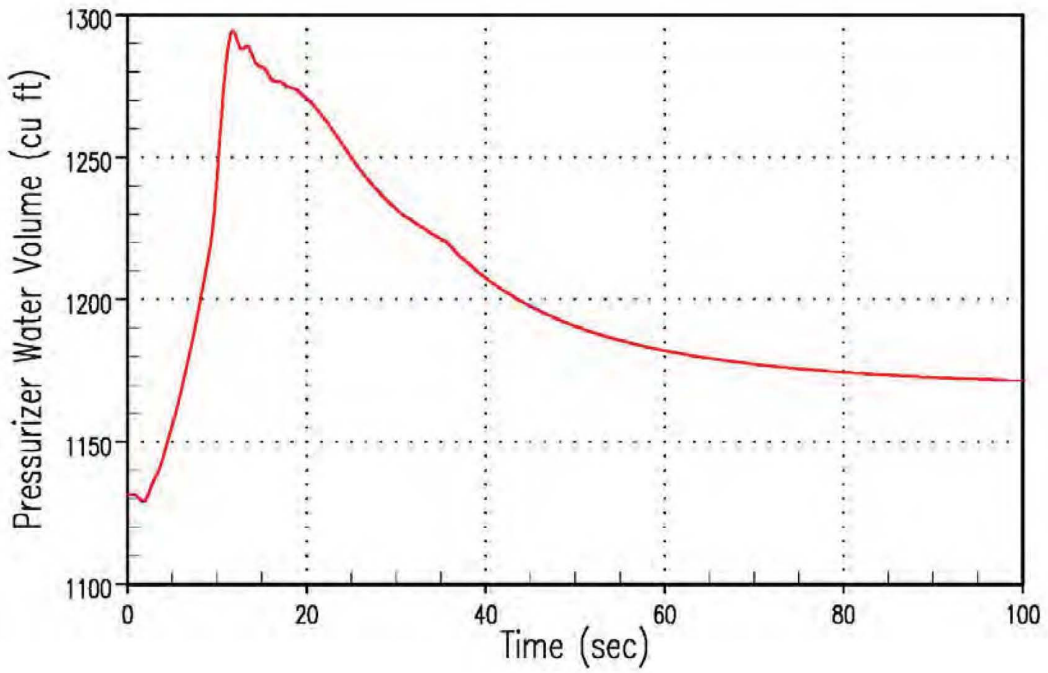
FIGURE 15.2-1 NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK RCS OVERPRESSURE EVALUATON



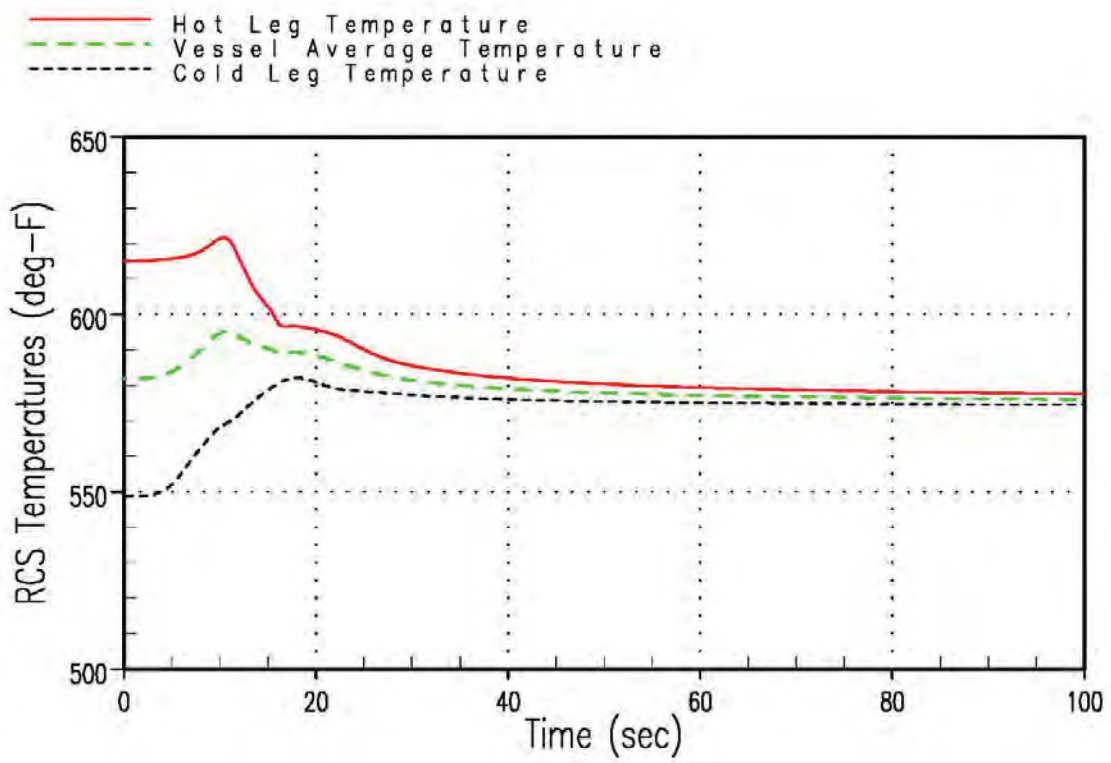


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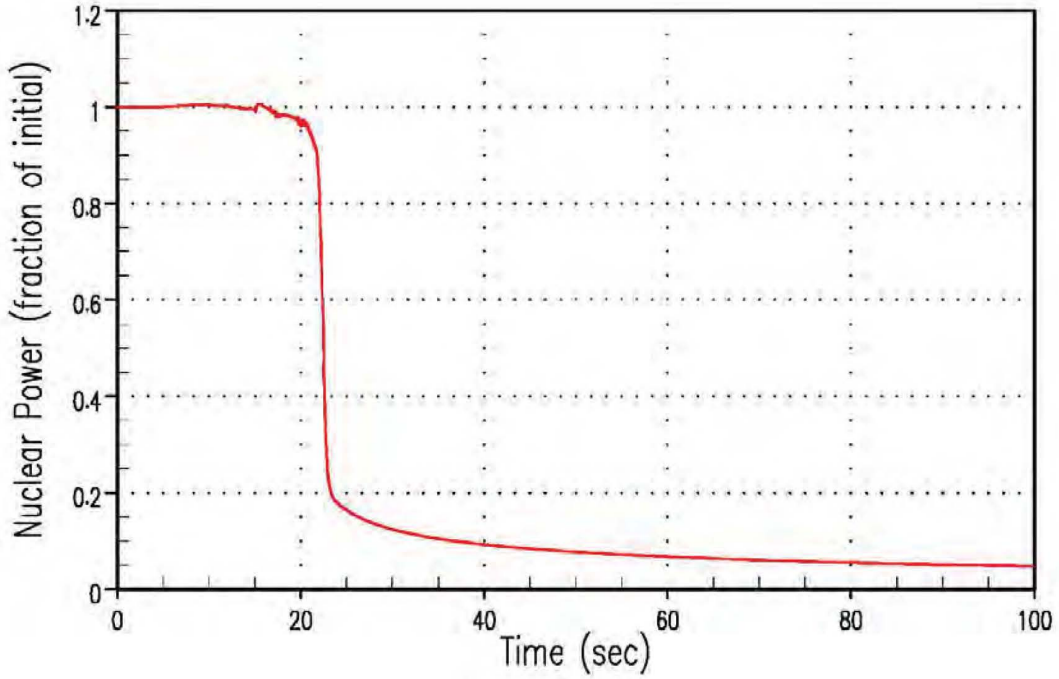
FIGURE 15.2-2
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK
RCS OVERPRESSURE EVALUATION



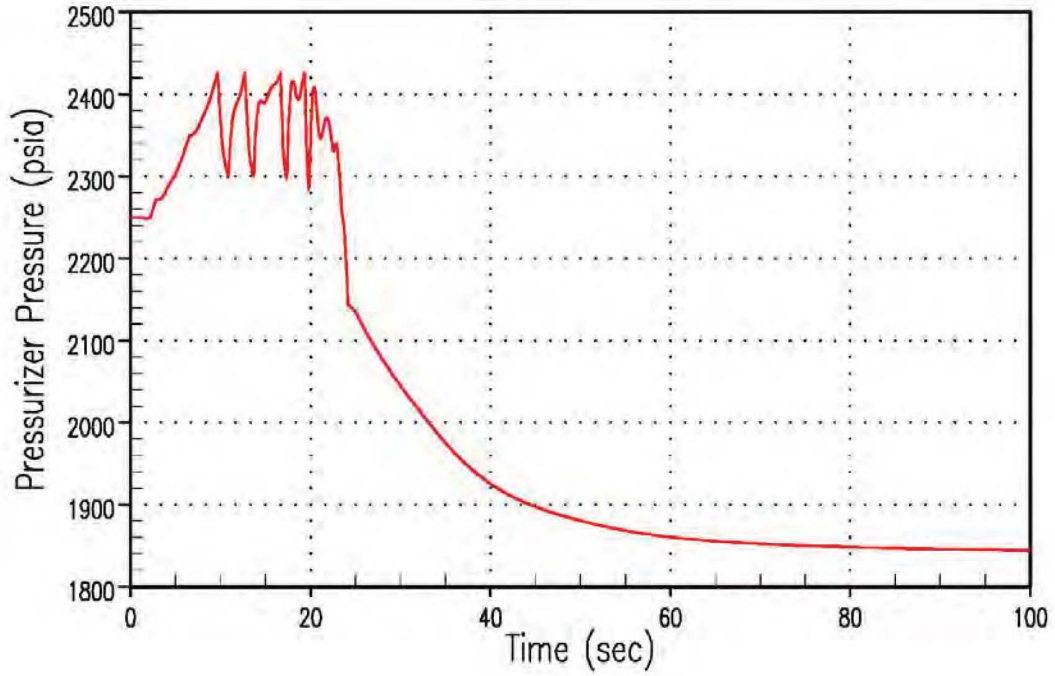


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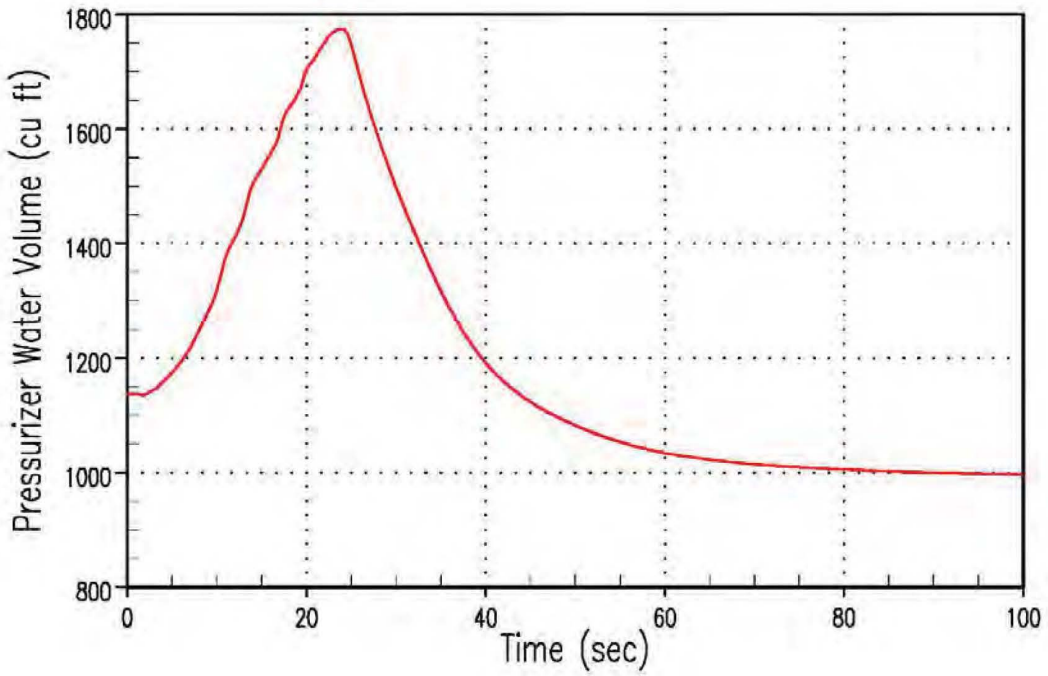
FIGURE 15.2-3 NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK DNB EVALUATION



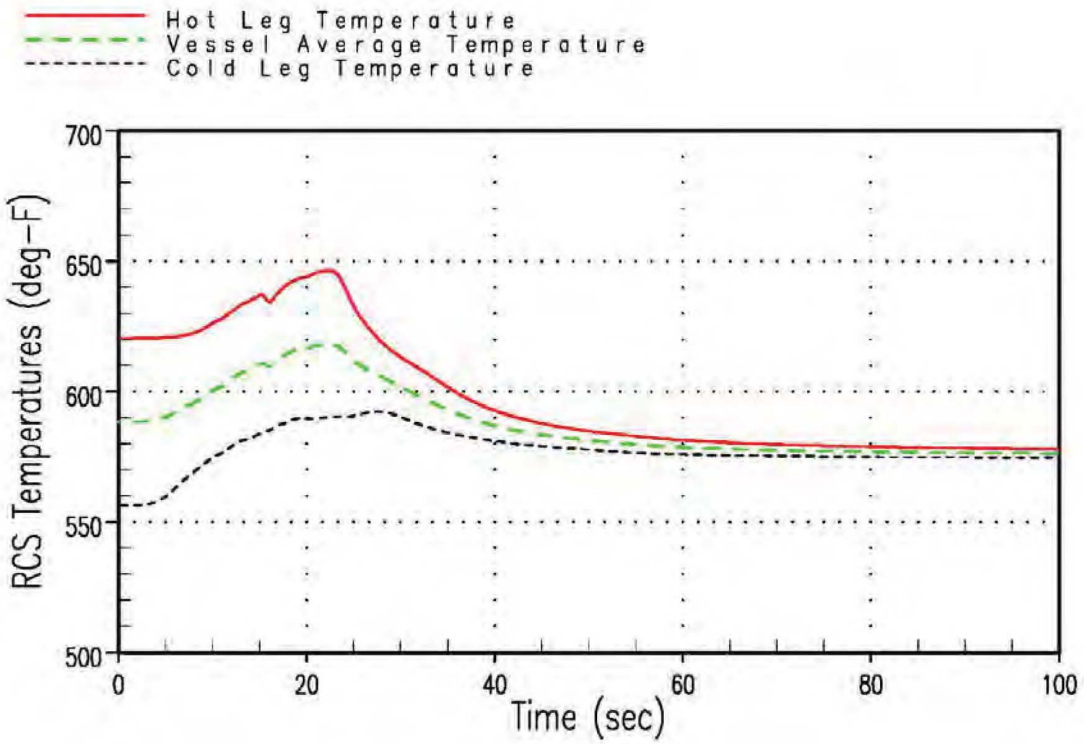


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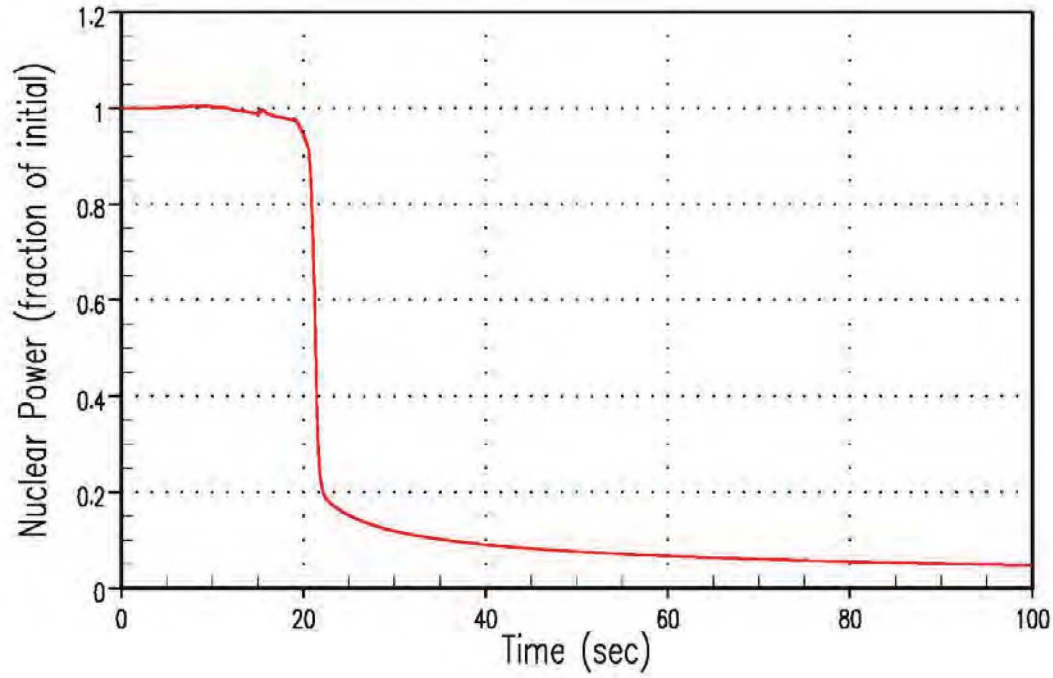
FIGURE 15.2-4 PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK DNB EVALUATION



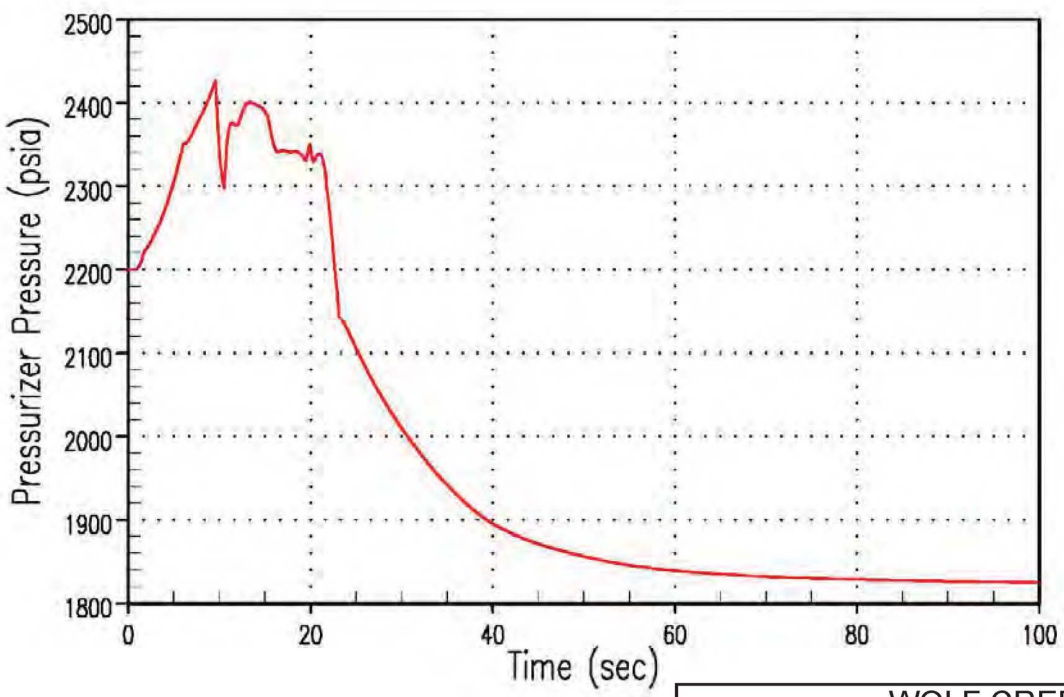


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FIGURE 15.2-5
NUCLEAR POWER AND PRESSURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK
SG OVERPRESSURE EVALUATON

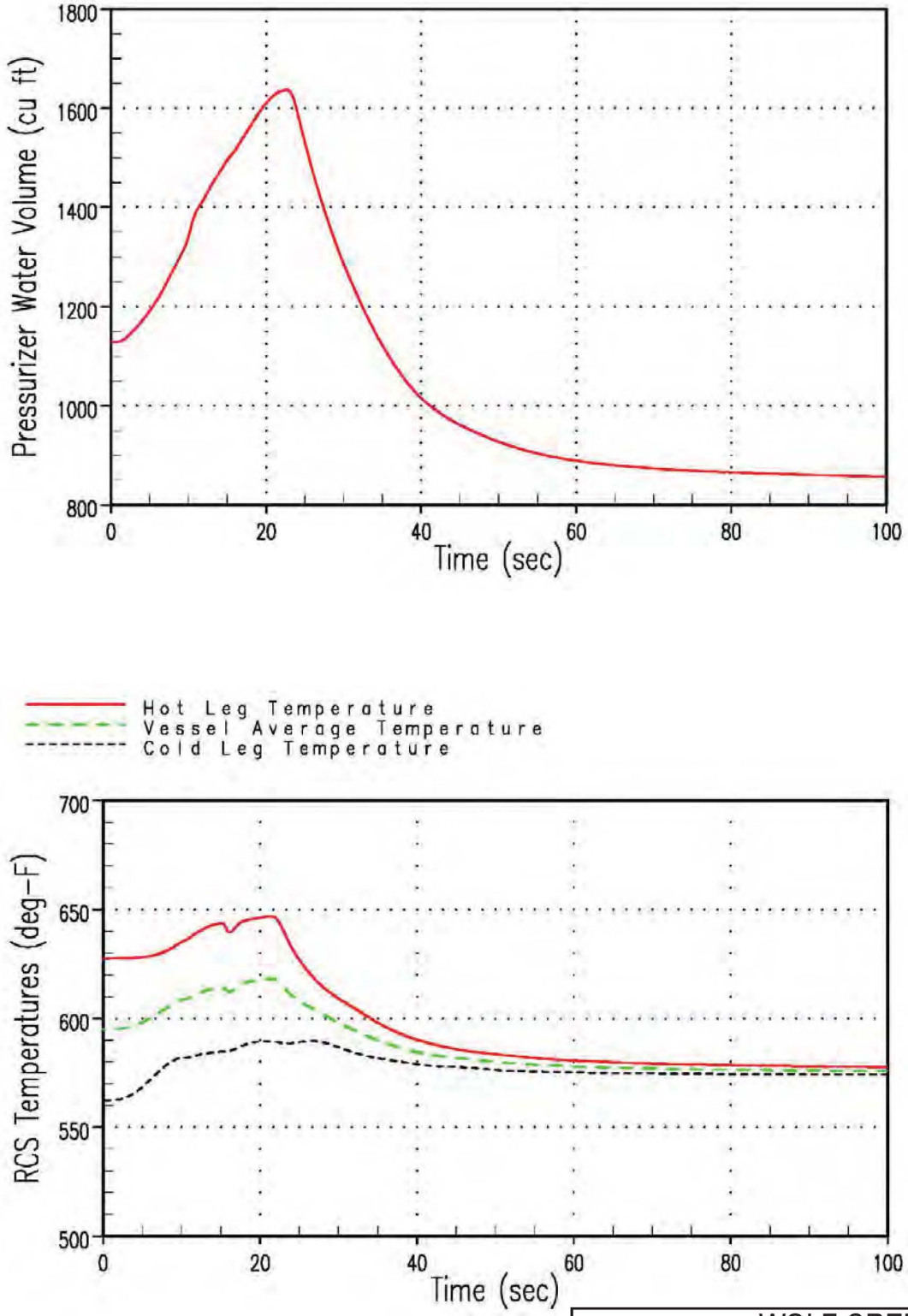

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FIGURE 15.2-6
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MINIMUM REACTIVITY FEEDBACK
SG OVERPRESSURE EVALUATION

FIGURE 15.2-7 HAS BEEN DELETED

FIGURE 15.2-8 HAS BEEN DELETED

FIGURE 15.2-8
PRESSURIZER LIQUID VOLUME AND RCS TEMPERATURE FOR TURBINE TRIP EVENT WITH MAXIMUM REACTIVITY FEEDBACK DNB EVALUATION

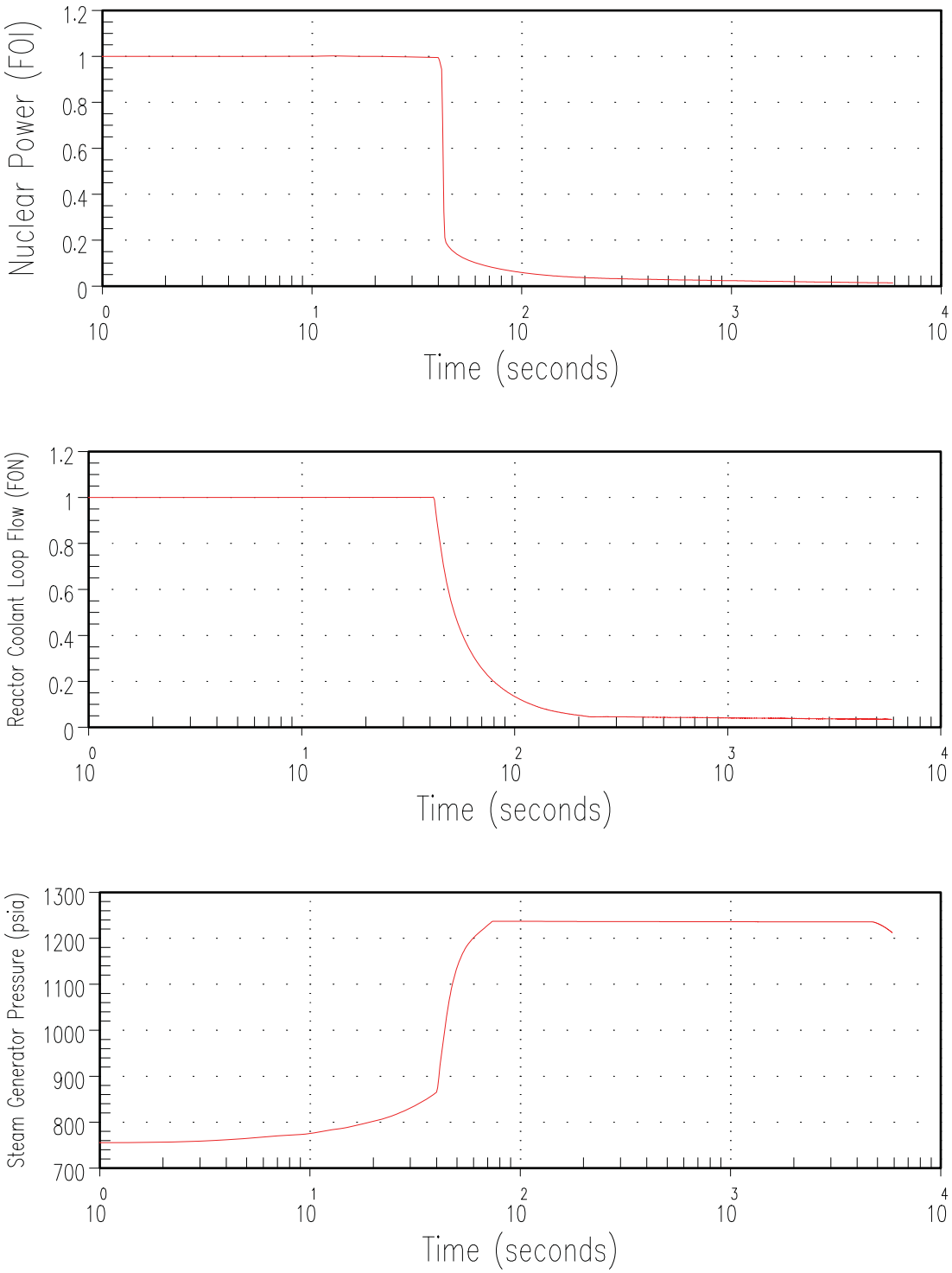

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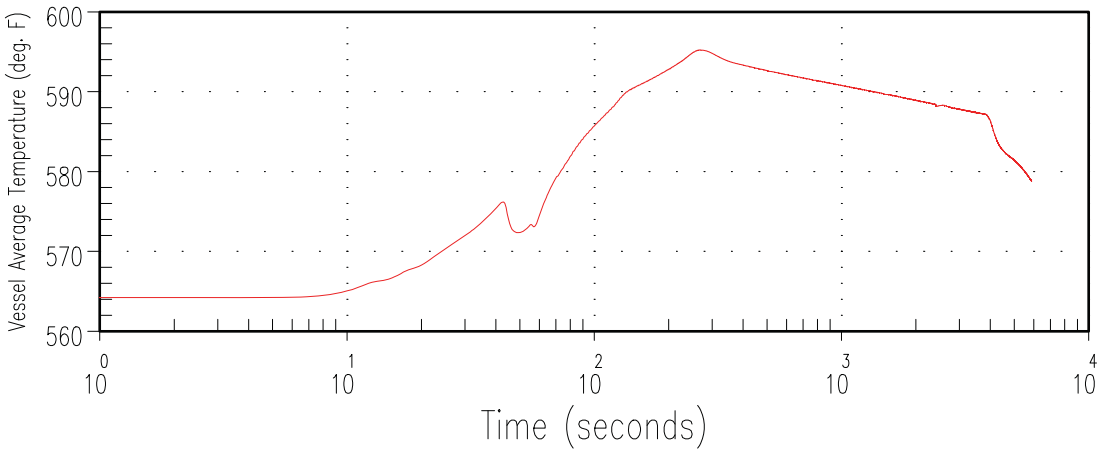
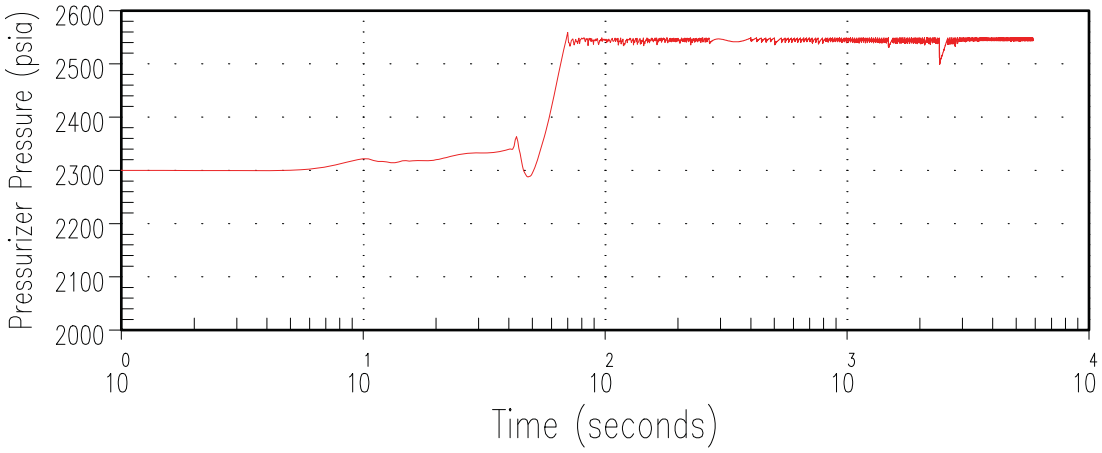
Figure 15.2-9
Nuclear Power, Reactor Coolant Loop Flow Rate and Steam Generator Pressure Transients for Loss of AC Power





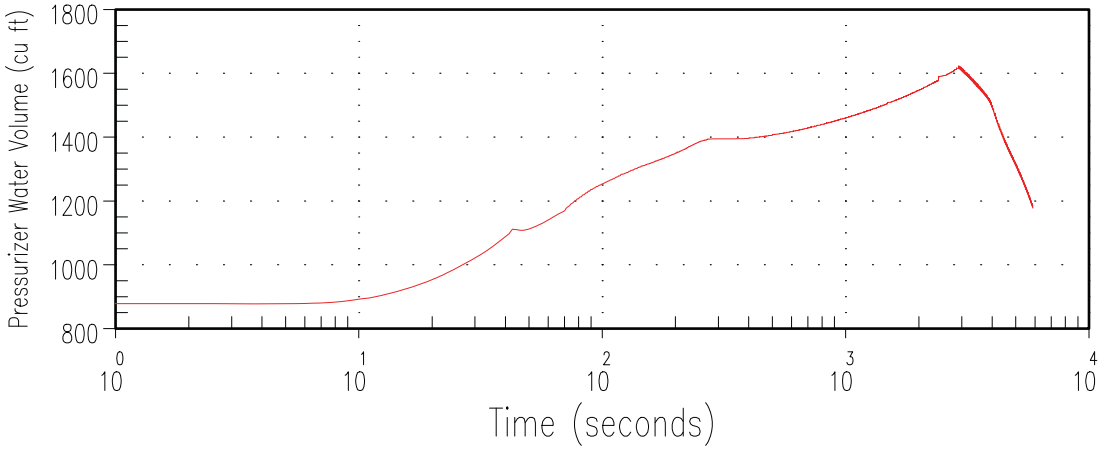


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Figure 15.2-10
Reactor Coolant Temperature, Pressurizer Pressure and Pressurizer Water Volume Transients for Loss of AC Power

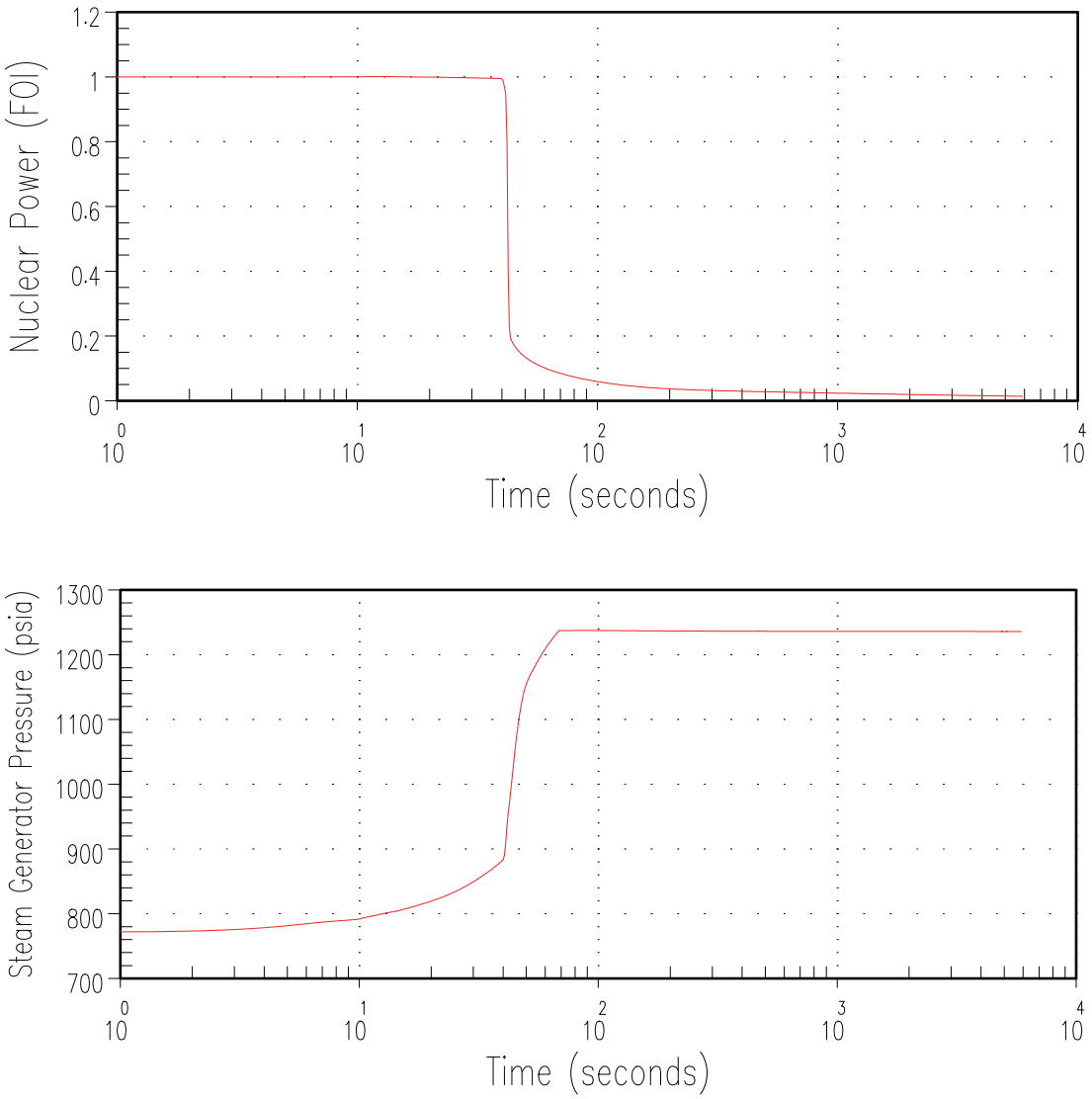

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Figure 15.2-11

Nuclear Power and Steam Generator Pressure Transients for Loss of Normal Feedwater

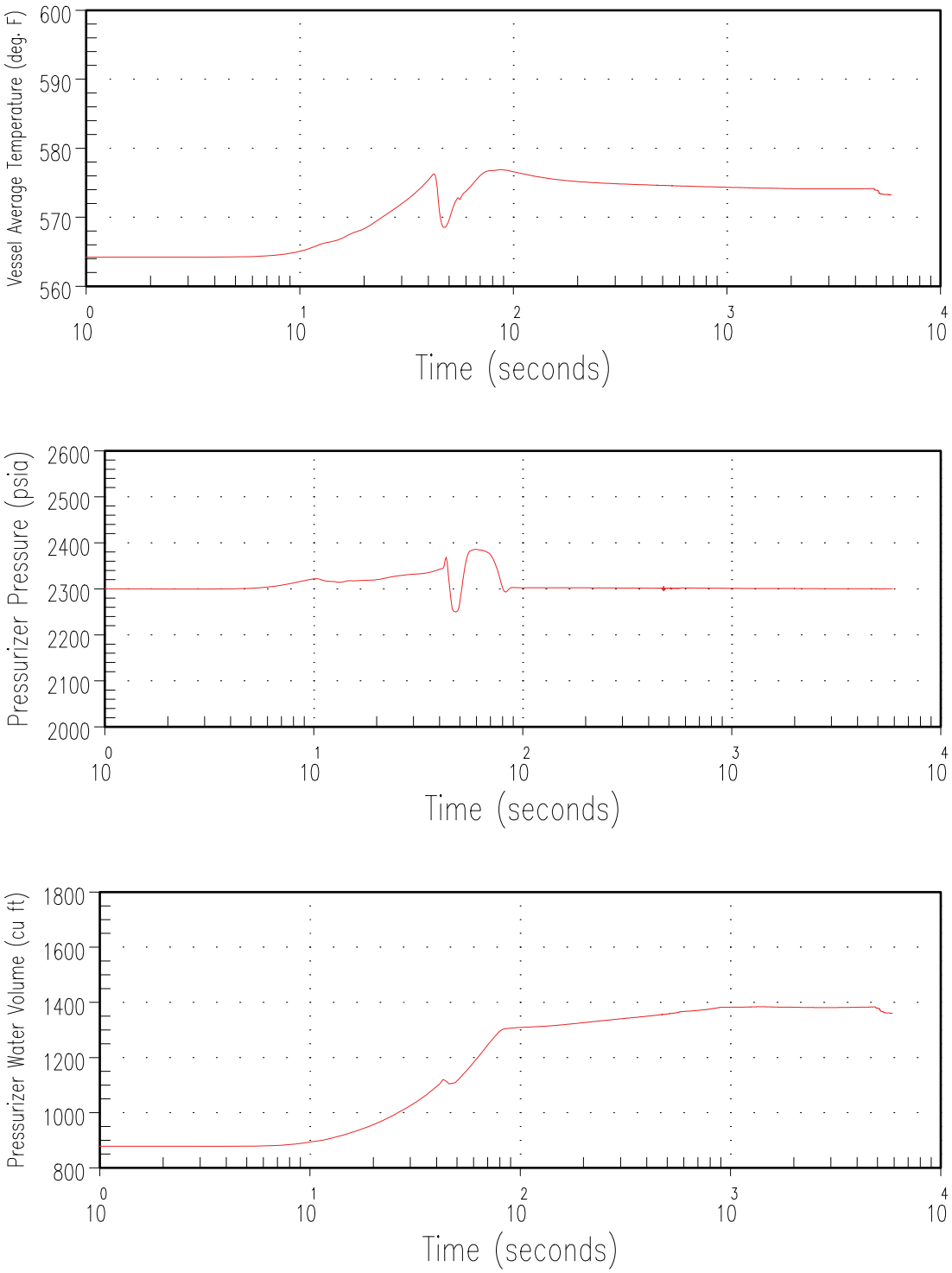

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Figure 15.2-12
Reactor Coolant Temperature, Pressurizer Pressure and Pressurizer Water Volume Transients for Loss of Normal Feedwater

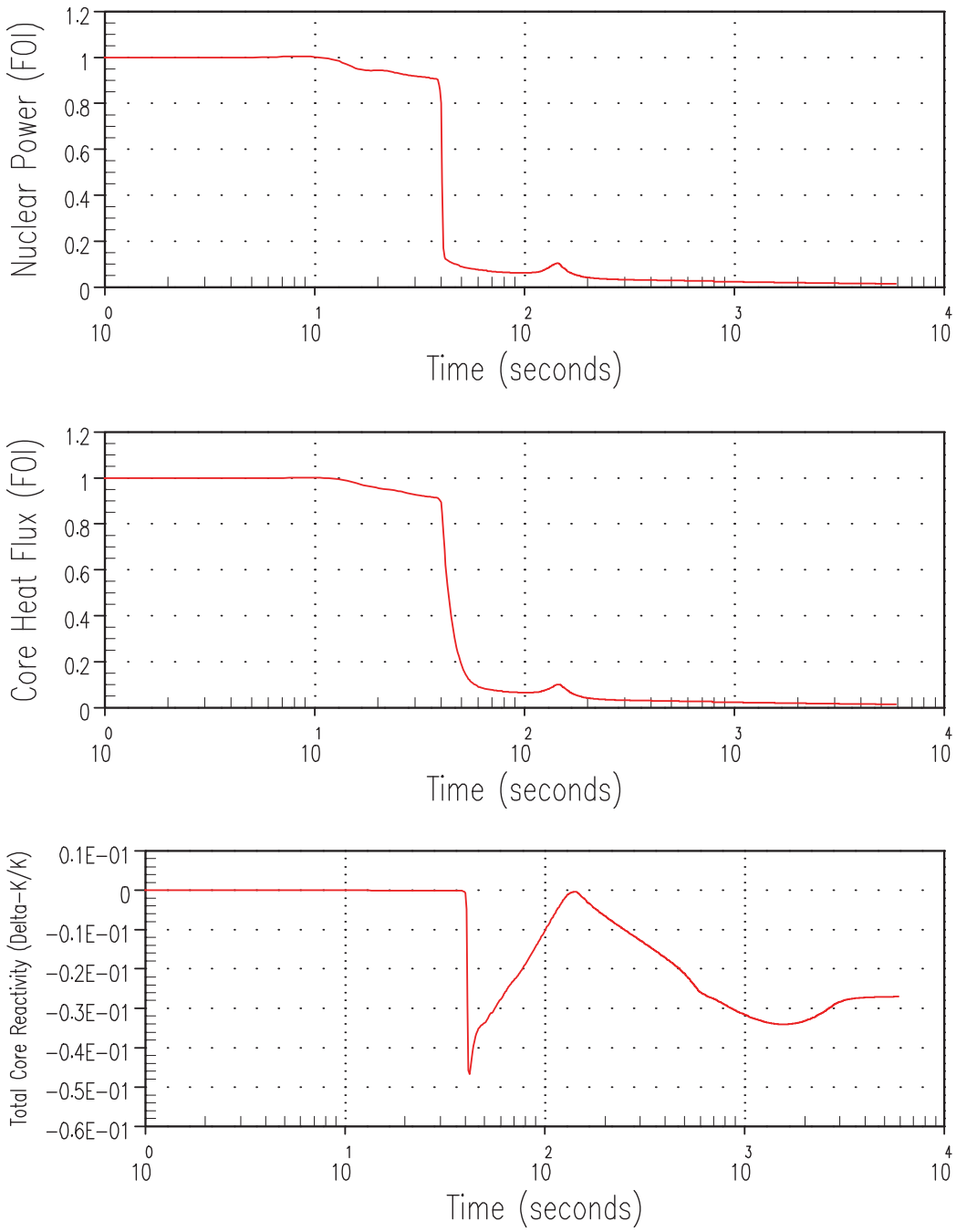

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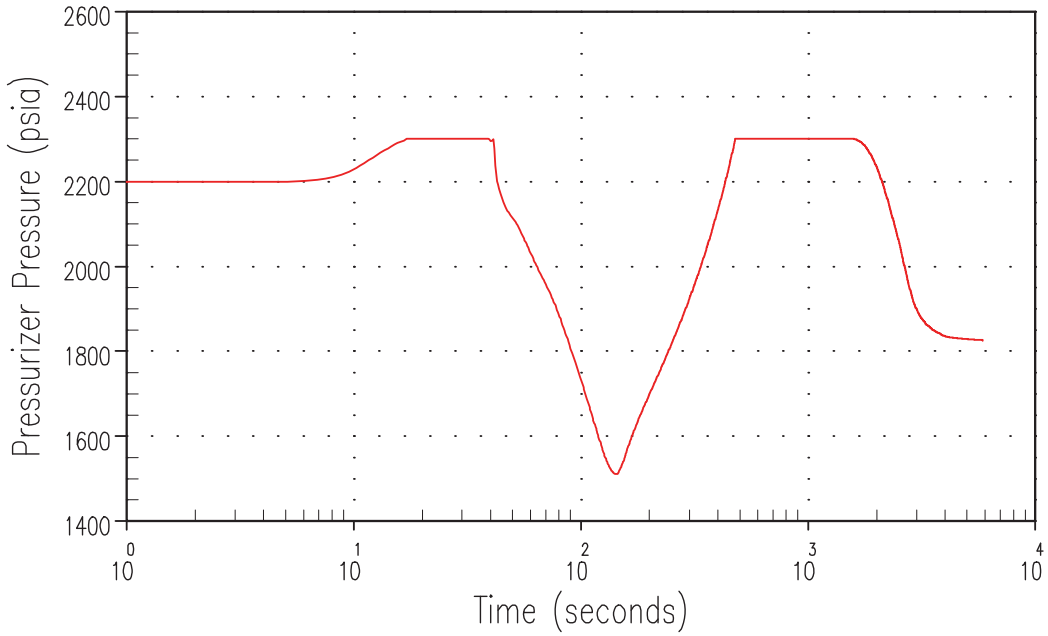
Figure 15.2-13
Nuclear Power, Core Heat Flux and Total Core Reactivity for Main Feedwater Line Rupture With Offsite Power Available



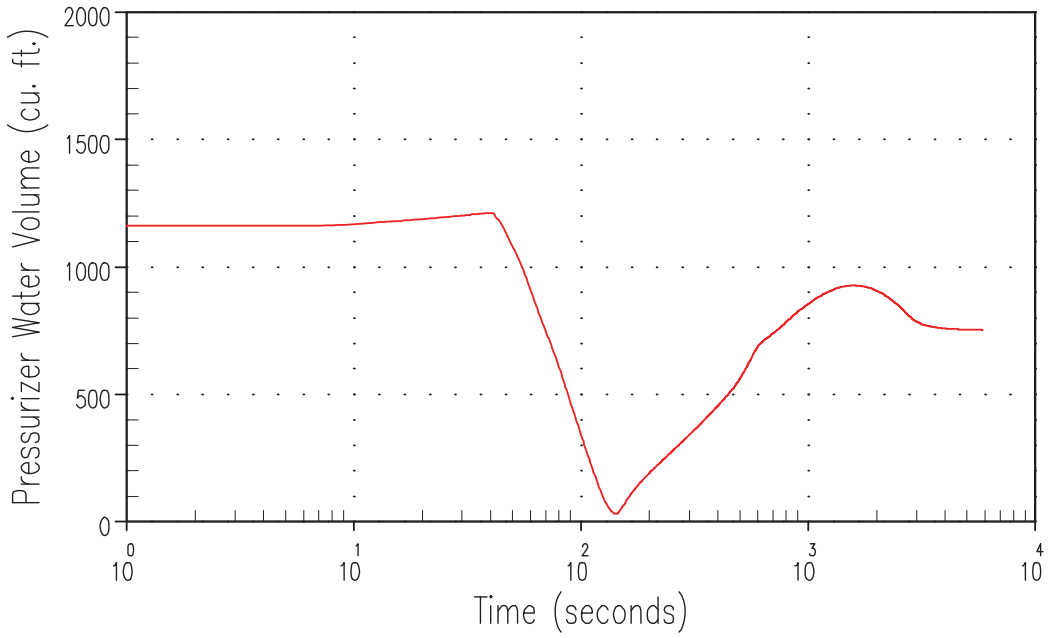


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Figure 15.2-14
Pressurizer Pressure and Pressurizer Water Volume for Main Feedwater Line Rupture With Offsite Power Available

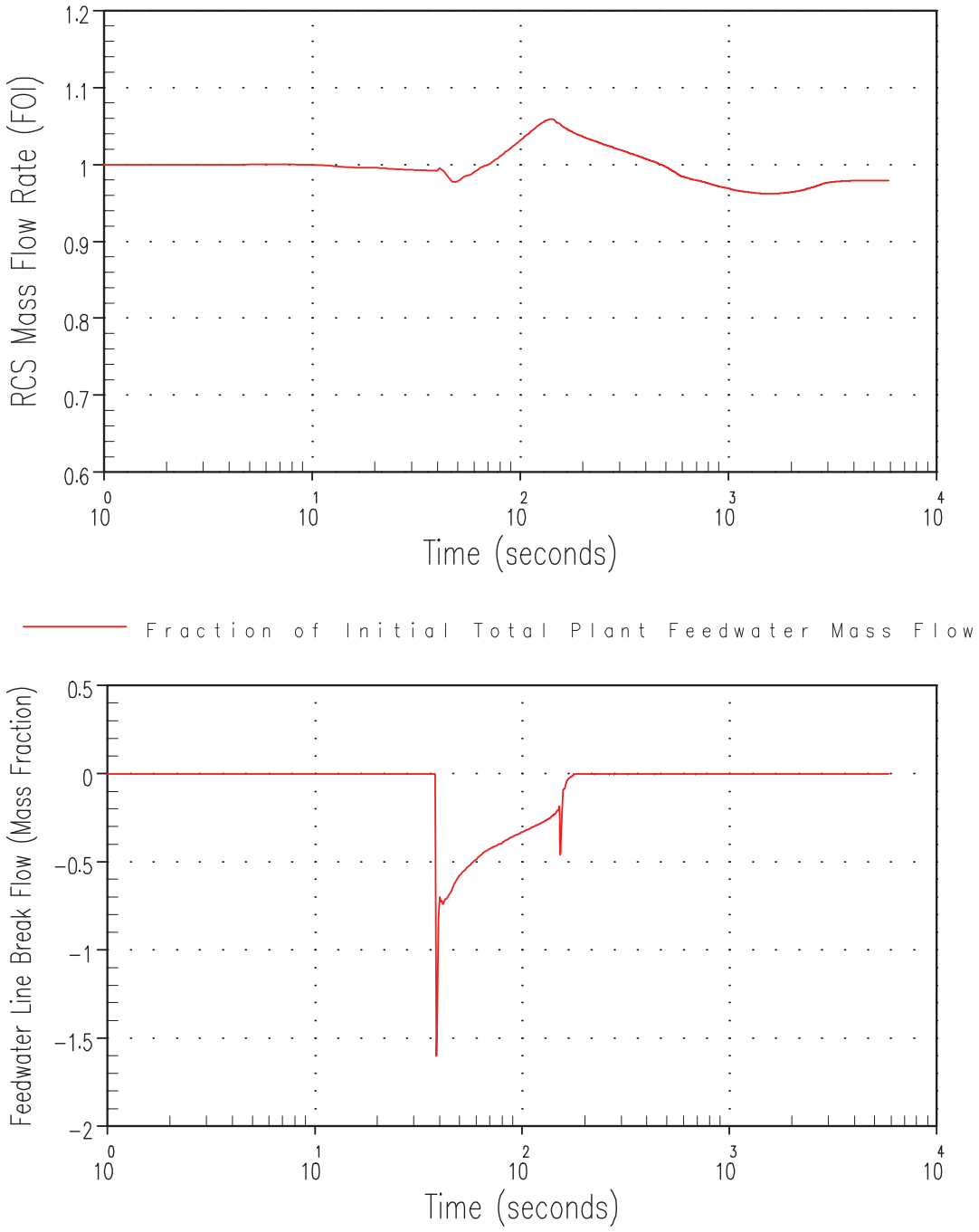

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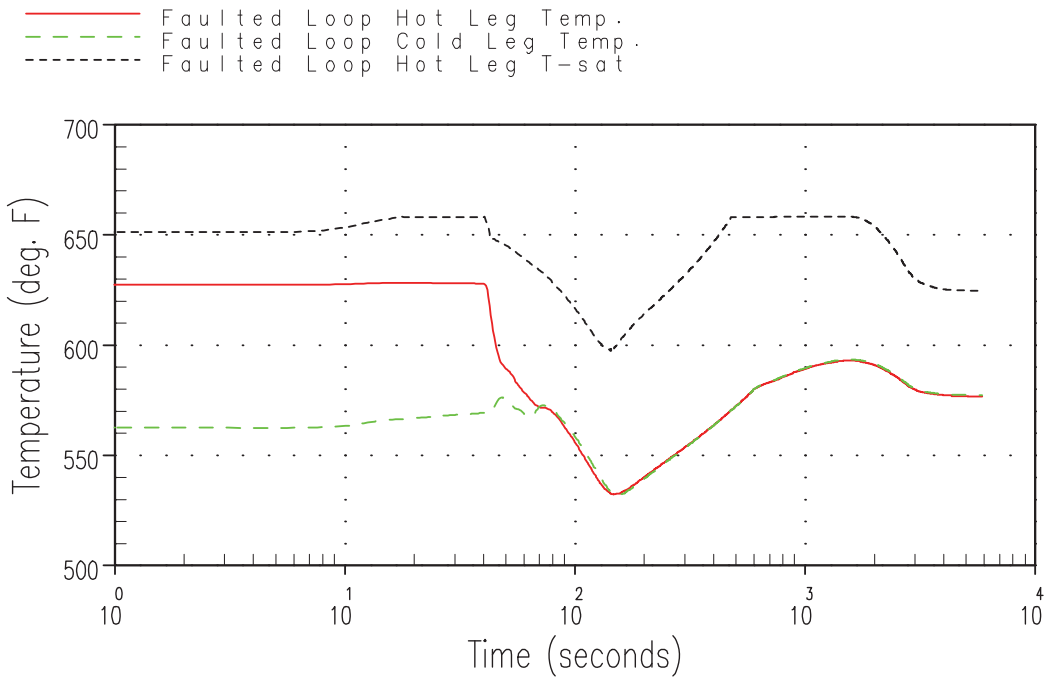
Figure 15.2-15
Reactor Coolant Flow and Feedwater Line Break Flow for Main Feedwater Line Rupture With Offsite Power Available



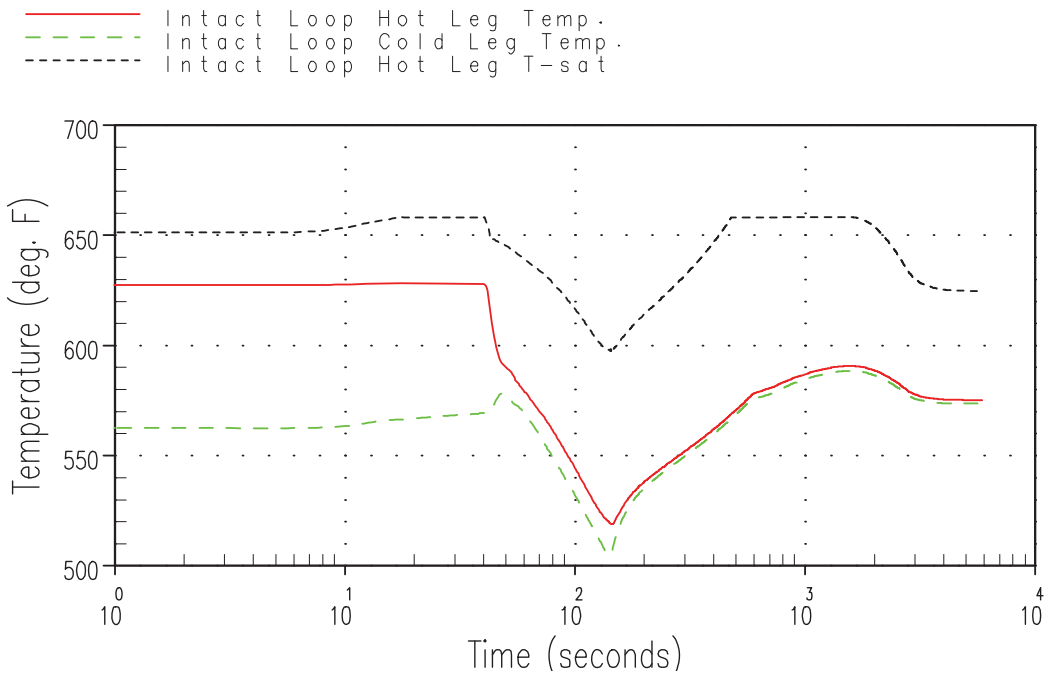


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Figure 15.2-16
Faulted Loop and Intact Loop Reactor Coolant Temperature for Main Feedwater Line Rupture With Offsite Power Available

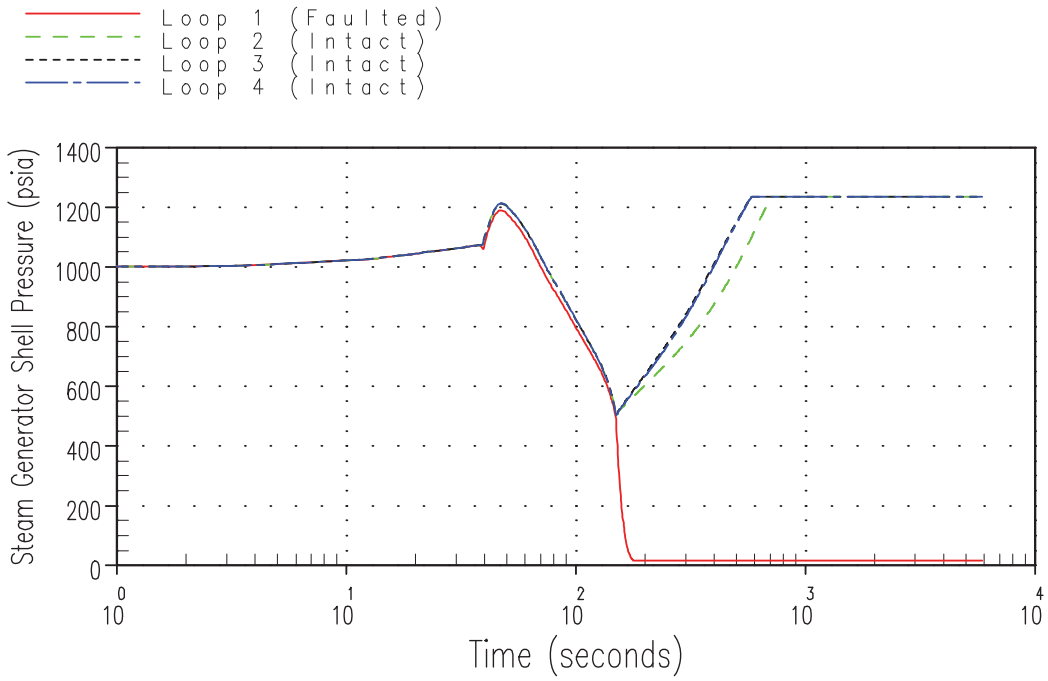

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Figure 15.2-17
Steam Generator Pressure for Main Feedwater Line Rupture With Offsite Power Available

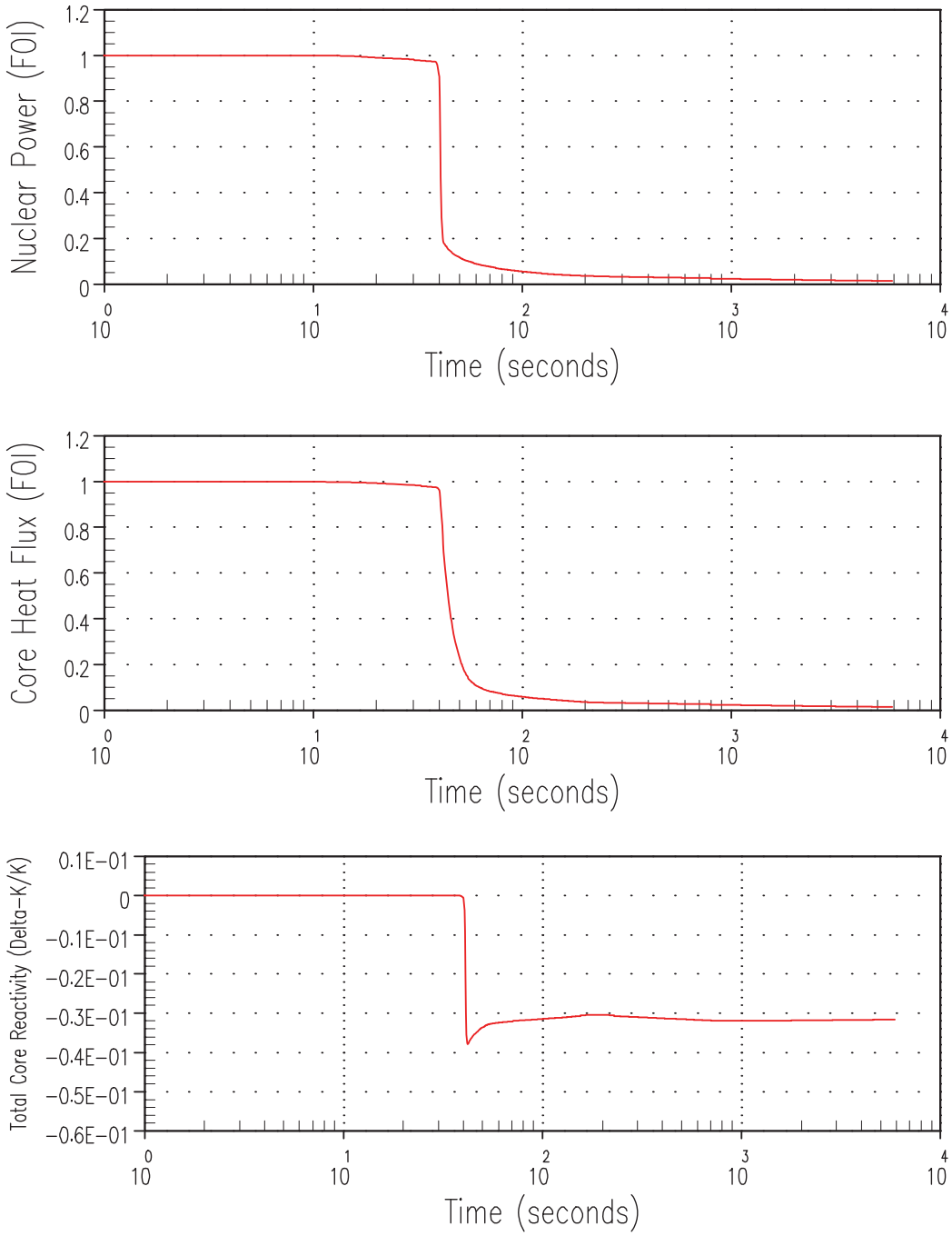

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Figure 15.2-18
Nuclear Power, Core Heat Flux and Total Core Reactivity for Main Feedwater Line Rupture Without Offsite Power

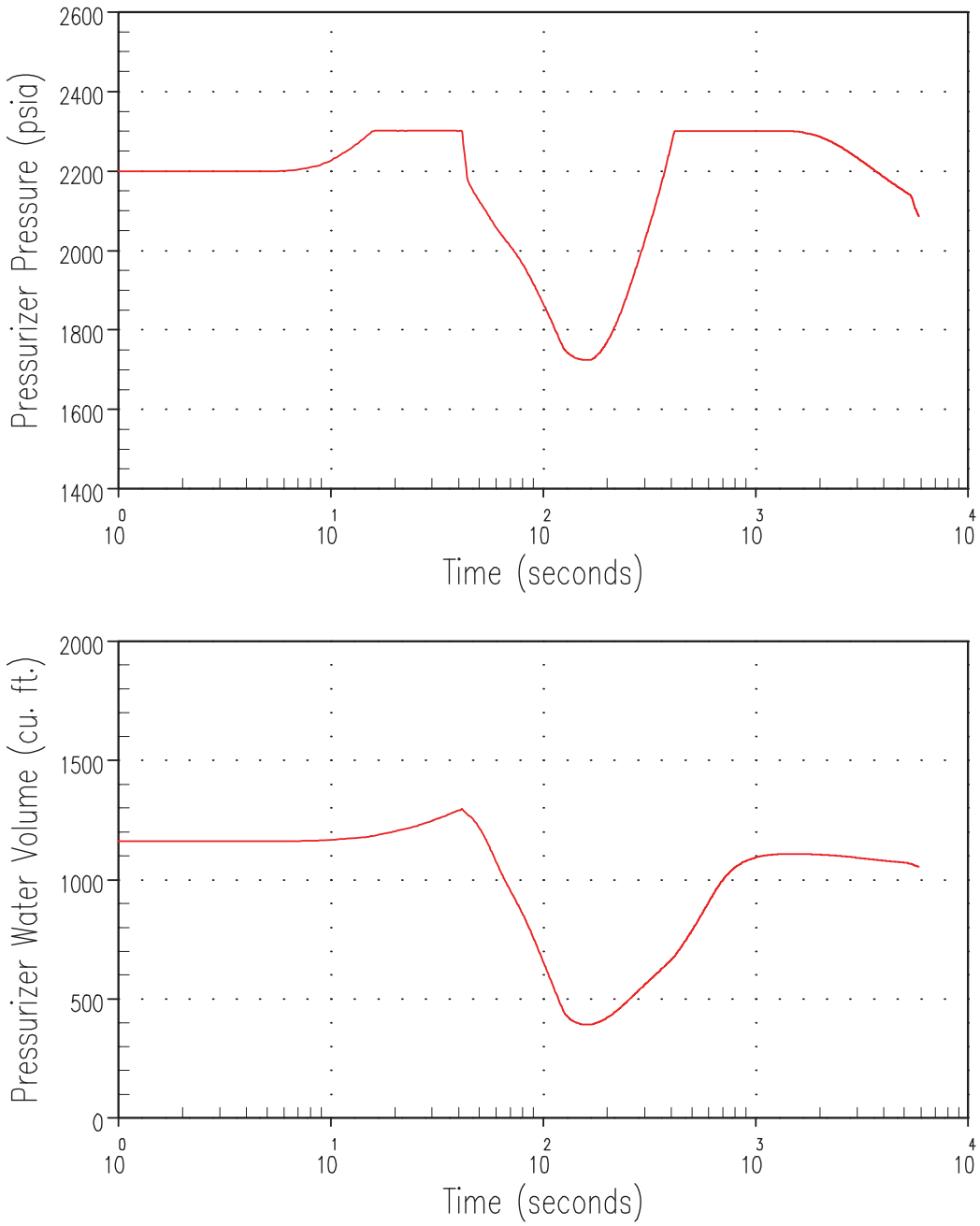

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Figure 15.2-19
Pressurizer Pressure and Pressurizer Water Volume for Main Feedwater Line Rupture Without Offsite Power

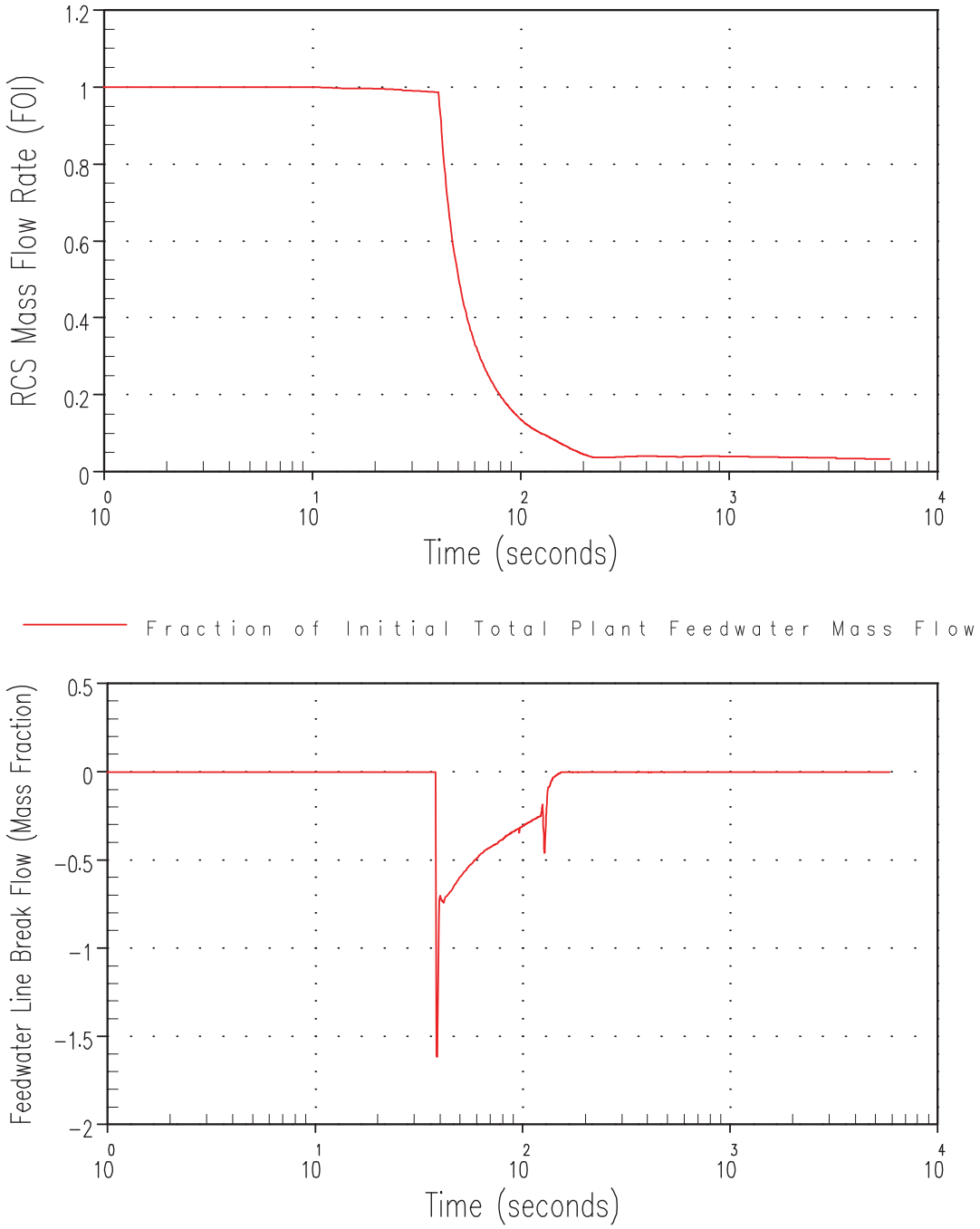

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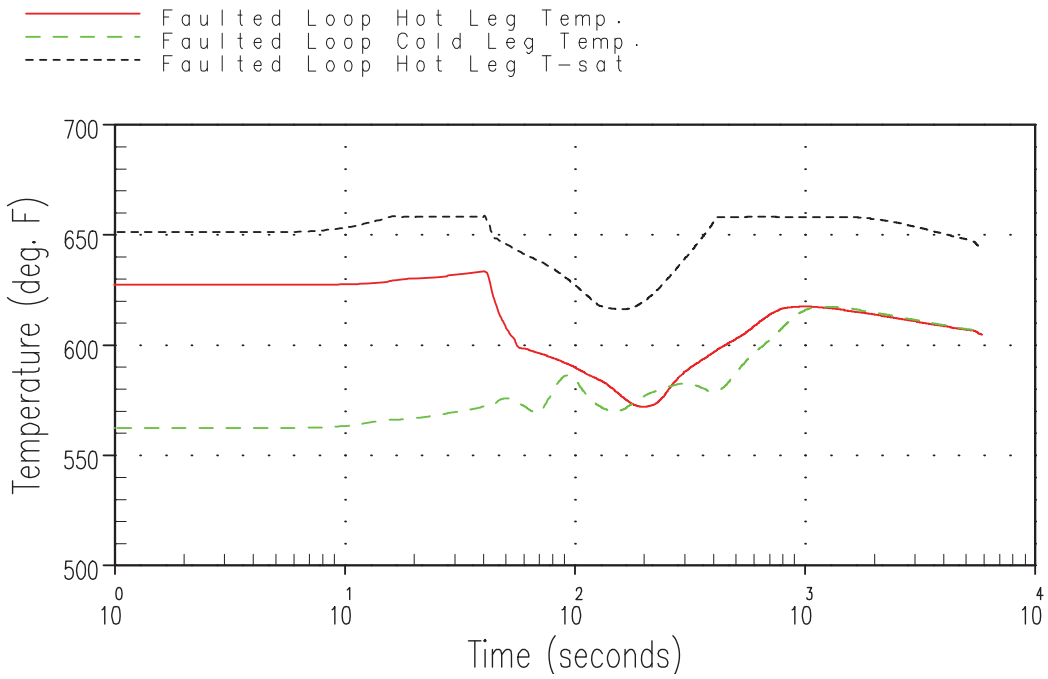
Figure 15.2-20
Reactor Coolant Flow and Feedwater Line Break Flow for Main Feedwater Line Rupture Without Offsite Power



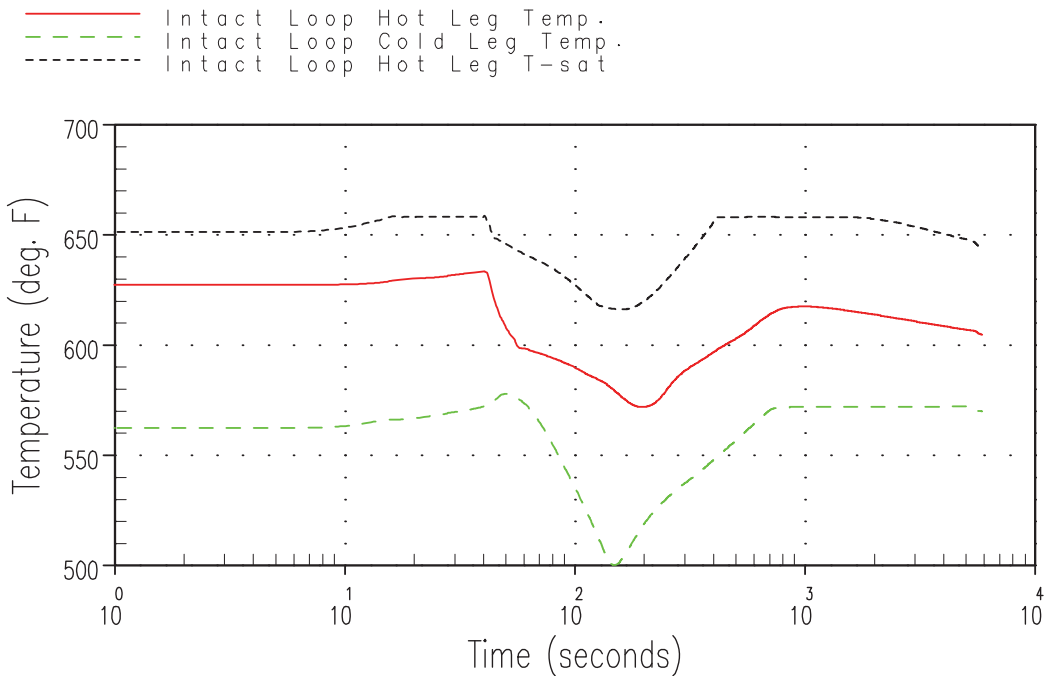


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Figure 15.2-21
Faulted Loop and Intact Loop Reactor Coolant Temperature for Main Feedwater Line Rupture Without Offsite Power

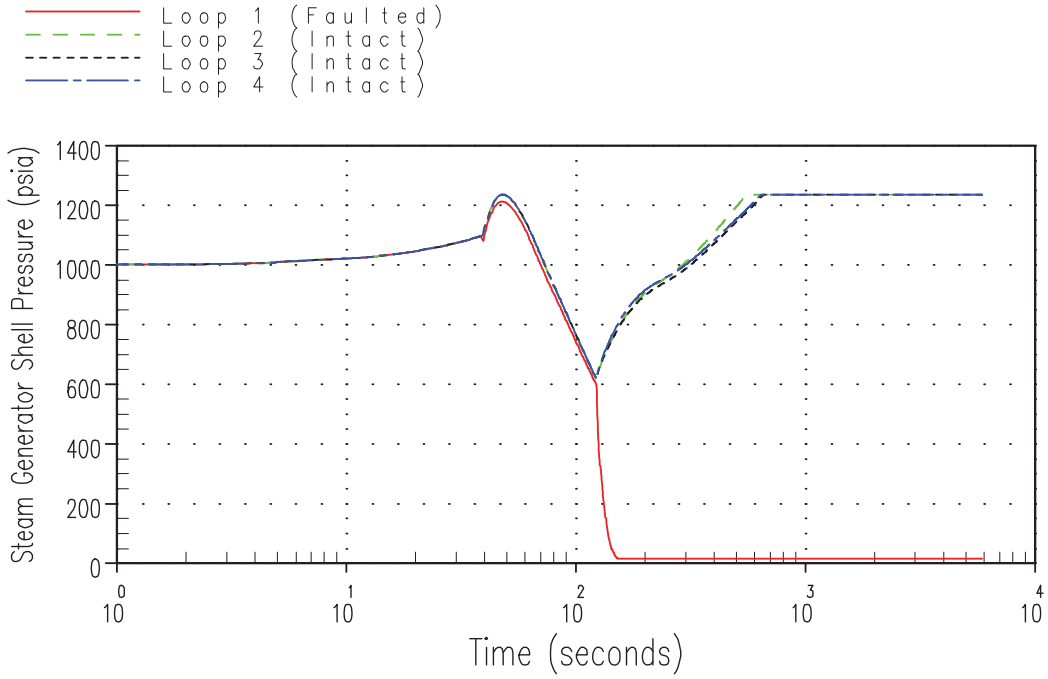

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Figure 15.2-22
Steam Generator Pressure for Main Feedwater Line Rupture Without Offsite Power

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults which could result in a decrease in the reactor coolant system (RCS) flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following flow decrease events:

a. Partial loss of forced reactor coolant flow
b. Complete loss of forced reactor coolant flow
c. Reactor coolant pump shaft seizure (locked rotor)
d. Reactor coolant pump shaft break

Item a above is considered to be an ANS Condition II event, item b an ANS Condition III event, and items c and d ANS Condition IV events. Section 15.0.1 contains a discussion of ANS classifications.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the reactor coolant pump shaft seizure accident of Section 15.3.3. Therefore, doses are reported only for that limiting case.

15.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Identification of Causes and Accident Description

A partial loss-of-coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual busses connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to operate. Following any turbine trip where there are no electrical faults or turbine faults which require immediate tripping of the generator from the network, the generator remains connected to the network for at least 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for at least 30 seconds after the reactor trip before any transfer is made.

A partial loss of coolant flow is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The necessary protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8 (see Section 7.2), flow in any loop below the trip setpoint will actuate a reactor trip. Between approximately 10-percent power (Permissive 7) and the power level corresponding to Permissive 8, flow in any two loops below the trip setpoint will actuate a reactor trip. Above Permissive 7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

The case for a loss of two reactor coolant pumps with four loops in operation is analyzed here.

This event is analyzed by employing the detailed digital computer program RETRAN (Refs. 1 and 2). RETRAN has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature. VIPRE (Ref. 5) is used to evaluate the core thermal limits to determine DNBR. RETRAN generated state points are used as VIPRE boundary conditions to perform a DNB analysis using the Revised Thermal Design Procedure (Ref. 4).

Initial Conditions

Plant characteristics and initial conditions are discussed in Section 15.0.3. This event is initiated at nominal operating conditions in accordance with Westinghouse RTDP methodology (Ref. 4) to perform the DNB analysis.

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0-2), and a least-negative moderator temperature coefficient is assumed since this results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

For this analysis, the curve of trip reactivity insertion versus time (Figure 15.0-5) was used.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics, and is based on high estimates of system pressure losses.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-15.

Results

Figures 15.3-1 through 15.3-5 show the transient response for the loss of two reactor coolant pumps with four loops in operation. Figure 15.3-6 shows the minimum DNBR to be always greater than the safety analysis limit.

The plant is tripped by the low flow trip rapidly enough to ensure that the ability of the reactor coolant to remove heat from the core is not greatly reduced for the cases analyzed. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events table for the cases analyzed is shown on Table 15.3-1. The affected reactor coolant pump(s) will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the two pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Conclusions

The analysis shows that the minimum DNBR will not decrease below the safety limit at any time during the transient. The DNBR analysis design basis is described in Section 4.4. All applicable acceptance criteria are met.

15.3.2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in a DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or turbine faults which require immediate tripping of the generator from the network, the generator remains connected to the network for at least 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring full flow for at least 30 seconds after the reactor trip before any transfer is made.

A complete loss of flow accident is classified as an ANS Condition III event (an infrequent fault), as defined in Section 15.0.1. The following signals provide protection against this event:

a. Reactor coolant pump power supply undervoltage or underfrequency
b. Low reactor coolant loop flow

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps. This function is blocked below approximately 10-percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference 3 provides analyses of grid frequency disturbances and the resulting NSSS protection requirements which are applicable to WCGS.

Reference 3 shows that the underfrequency trip of the reactor coolant pump breakers is not required for grid decay rates up to 5 Hz/sec. Grid stability and transient analyses for both the Wolf Creek and Callaway sites show maximum grid decay rates of less than 5 Hz/sec. Therefore, the reactor coolant pump breaker trip on under frequency (Figure 7.2-1, Sheet 5) is not a safety function in the WCGS design.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, flow in any loop below the trip setpoint will actuate a reactor trip. Between approximately 10-percent power (Permissive 7) and the power level corresponding to Permissive 8, flow in any two loops below the trip setpoint will actuate a reactor trip. At full power, if the maximum grid frequency decay rate is less than approximately 2.5 Hertz per second, this trip function will protect the core from underfrequency events. This effect is fully described in Reference 3.

15.3.2.2 Analysis of Effects and Consequences

Method of Analysis

The following case has been analyzed:

Loss of four pumps with four loops in operation

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.3.1, except that, following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

Results

Figures 15.3-7 through 15.3-11 show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-12 shows the minimum DNBR to be always greater than the safety analysis limit.

For the case analyzed, the plant is tripped by the undervoltage trip sufficiently fast to ensure that the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Section 15.2.6. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis performed has demonstrated that, for the complete loss of forced reactor coolant flow, the minimum DNBR does not decrease below the safety analysis limit at any time during the transient. The design basis for the DNBR is described in Section 4.4. All applicable acceptance criteria are met.

15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced - first, because the reduced flow results in a decreased tube side film coefficient,

and then, because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation, and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, are not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault), as defined in Section 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

Two digital computer codes are used to analyze this transient. The RETRAN code (Refs. 1 and 2) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the VIPRE code (Ref. 5).

Accident Scenario

Three locked rotor cases are analyzed. One case is analyzed to determine the percentage of fuel rods that experience DNB for consideration in the radiological dose analysis. A loss of offsite power is conservatively assumed for this case. The other two cases, one with offsite power and one without offsite power, are analyzed to determine the maximum RCS pressure and peak clad temperature (PCT).

The locked rotor transient is postulated to occur in the following manner:

a. Reactor coolant pump rotor locks and flow in that loop rapidly reduces. The shaft break incident described in Section 15.3.4 is accounted for by allowing reverse flow in the faulted loop.
b. The reactor is tripped on low RCS flow in one loop.
c. Turbine generator trips.
d. Offsite power is lost for the without offsite power case.

   NOTE: Grid stability analyses show that the grid will remain stable and offsite power will not be lost because of a unit trip from 100 percent power. Refer to Section 8.2.2. The following analysis assumes no time delay between reactor trip and loss of offsite power. This is a conservative assumption based on the grid stability analyses.

e. The loss of offsite power causes the three remaining reactor coolant pumps to coast down.

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., maximum guaranteed steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

For the peak pressure evaluation, the initial pressure is conservatively set at 50 psi above nominal pressure (2,250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The maximum RCS pressure is located at the reactor vessel lower plenum.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 86.3 percent of indicated loop flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

Evaluation of PCT and Zirconium-Water Reaction During the Accident

For the PCT cases, DNB is assumed to occur in the core at the initiation of the transient in order to conservatively predict the PCT, and an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

Film Boiling Coefficient

The film boiling coefficient is calculated with VIPRE (Ref. 5) using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, inlet enthalpy, and mass flow rate as a function of time are used as program input. For conservatism for the PCT cases, DNB was assumed to occur in the core at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial gap coefficient value is released to the clad at the initiation of the transient.

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1,800 F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45500}{1.986T}\right)$$

where:

w = amount reacted, mg/cm^2
t = time, sec
T = temperature, K

The reaction heat is 1,510 cal/gm.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-15. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results and Conclusions

Figures 15.3-13 through 15.3-19 show the transient results for the two locked rotor cases analyzed to determine the maximum RCS pressure and PCT. The results of these calculations are also summarized in Table 15.3-2. The peak RCS pressure reached during the transient is less than 110% of the design pressure. The DNB case demonstrated that the percent of rods-in-DNB is less than the limit assumed in the dose evaluation.

Also, the PCT is considerably less than 2,700 °F for Standard ZIRLO® fuel cladding or 2375 °F for Optimized ZIRLO™ fuel cladding. It should be noted that the clad temperature was conservatively calculated, assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the cases analyzed is shown on Table 15.3-1. Figures 15.3-14 and 15.3-15 shows that the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

15.3.3.3.1 Method of Analysis

15.3.3.3.1.1 Physical Model

The instantaneous seizure of a reactor coolant pump rotor results in a reactor trip on a low coolant flow signal. With the coincident loss of offsite power, the condensers are not available, so the excess heat is removed from the secondary system by a steam dump through the steam generator safety and atmospheric relief valves. Steam generator tube leakage is assumed to continue until the residual heat removal system can match decay heat and releases from the secondary system are terminated. The reactor coolant will contain the gap activities of the fraction of the fuel which undergoes DNB in addition to its assumed equilibrium activity.

15.3.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.3-3 and 15A-1 and summarized below. Tables 15B-1 and 15B-6 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

The assumption used to determine the initial concentrations of isotopes in the reactor coolant and secondary coolant prior to the accident are as follows:

a. The reactor coolant iodine activity is based on the dose equivalent of 1.0 μCi/gm of I-131.

b. The noble gas activity in the reactor coolant is based on the dose equivalent of 500 μCi/gm of Xe-133.

c. The reactor coolant alkali metal activity is based on 1-percent failed fuel as provided in Table 11.1-5.

d. The secondary coolant iodine activity is based on the dose equivalent of 0.1 μCi/gm of I-131.

e. The secondary coolant alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity).

The following conditions are used to calculate the activity released.

a. 5 percent of fuel rod gap activity is additionally released to the reactor coolant, i.e., 5 percent of the rods experienced DNB.

b. Offsite power is lost.

c. Following the incident, secondary steam is released to the environment for heat removal. The total quantity of steam released is given in Table 15.3-3.

d. Primary-to-secondary leakage continues after the accident for a period of 12 hours. At that time, the residual heat removal system has matched decay heat removal and releases from the secondary system are terminated. The leakage rate is assumed to be constant and equal to 1 gpm (500 lbs/hr).

e. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant.

f. The noble gas activity released is equal to the amount present in the reactor coolant which leaks into the secondary system after the accident.

g. The iodine activity present in the primary to secondary leakage is assumed to mix homogeneously with the iodine activity initially present in the steam generators.

h. A partition factor of 0.01 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

i. A full power moisture carryover of 0.25% is utilized to determine alkali metal releases to the environment via steam venting from the steam generators.

j. The activity released from the steam generators is immediately vented to the environment.

k. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite locations or the control room.

l. Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Tables 15A-2, 15A-1, and 15A-4, respectively.

15.3.3.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed using the models described in Appendix 15A.

15.3.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system atmospheric relief and safety valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system atmospheric relief and safety valves

15.3.3.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

a. A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

b. The coincident loss of offsite power with the occurrence of a reactor coolant pump locked rotor is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release to the environment.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.3.3.3.3 Conclusions

15.3.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the reactor coolant pump locked rotor accident is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated reactor coolant pump locked rotor accident releases.

15.3.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated reactor coolant pump locked rotor have been conservatively analyzed, using assumptions and models described in previous sections. Note that the radiological consequences for the rod ejection primary-to-secondary leakage (from Table 15.4-4) were shown to bound the locked rotor event and meet the lower locked rotor limits. Therefore, the rod ejection primary-to-secondary leakage dose results are presented in Table 15.3-4.

The TEDE doses have been analyzed for the limiting 2 hour dose at the exclusion area boundary, for the duration of the accident (0 to 12 hours) at the low-population zone outer boundary, and in the control room for 30 days. The results are listed in Table 15.3-4. The resultant doses are within a small fraction of the guideline values of 10 CFR 50.67 for offsite locations and the full 10 CFR 50.67 guideline value in the control room.

15.3.3.4 Conclusions

a. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

b. Since the PCT calculated for the hot spot during the worst transient remains considerably less than the more restrictive limit of 2375 °F (associated with Optimized ZIRLO™ fuel cladding), the core will remain in place and intact with no loss of core cooling capability.

c. Since the percent of rods-in-DNB is less than the limit assumed in the dose analysis, any radiological release will be within acceptable limits.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced - first, because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, are not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault), as defined in Section 15.0.1.

15.3.4.2 Conclusions

The consequences of a reactor coolant pump shaft break are addressed via the locked rotor analysis described in Section 15.3.3. With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as in a locked rotor incident. However, the locked rotor analysis described in Section 15.3.3 accounts for the shaft break incident by allowing the faulted loop's impeller to spin in the reverse direction.

15.3.5 REFERENCES

1. McFadden, J. H., et. al., "RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM-A, October 1984.

2. Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

3. Baldwin, M. S., Merrian, M. M., Schenkel, H. S. and VanDeWalle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May 1975.

4. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-PA (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

5. Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A, October 1999.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS
WHICH RESULT IN A DECREASE IN REACTOR COOLANT
SYSTEM FLOW RATE

Accident	Event	Time (sec)
Partial loss of forced reactor coolant flow		
1. Loss of two pumps with four loops in operation	Coastdown begins	0.0
	Low flow reactor trip	1.5
	Rods begin to drop	2.5
	Minimum DNBR occurs	4.3
Complete loss of forced reactor coolant flow		
1. Loss of four pumps with four loops in operation	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump undervoltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.5

TABLE 15.3-1 (Sheet 2)

Accident	Event	Time (sec)
Reactor coolant pump shaft seizure (locked rotor)		
1. PCT/Peak Pressure Case: one locked rotor with four loops in operation W/ offsite power	Rotor on one pump locks	0.0
	Low flow trip point reached	0.04
	Rods begin to drop	1.04
	Peak clad temperature	3.50
	Maximum RCS pressure occurs	4.08
2. PCT/Peak Pressure Case: one locked rotor with four loops in operation W/O offsite power	Rotor on one pump locks	0.0
	Low flow trip point reached	0.04
	Rods begin to drop	1.04
	Power lost to remaining pumps	1.04
	Peak clad temperature	3.90
	Maximum RCS pressure occurs	4.75

TABLE 15.3-1 (Sheet 3)

Accident	Event	Time (sec)
3. Percentage of Rods-in-DNB Case: One locked rotor with four loops in operation W/O offsite power	Rotor on one pump locks	0.0
	Low Flow trip setpoint reached	0.04
	Rods begin to drop	1.04
	Power lost to remaining pumps	1.05
	Minimum DNBR occurs	3.20

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

	With Offsite Power	Without Offsite Power
Maximum reactor coolant system pressure, psia	2565	2675
Maximum clad temperature, °F, core hot spot	1710	1787
Zr-H_2O reaction at core hot spot, percent by weight	0.18	0.29

TABLE 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

I.	Source Data	
a.	Power level, MWt	3637 (includes uncertainty)
b.	Steam generator tube leakage, gpm	1
c.	Core Inventories	see Table 15A-3
d.	Radial peaking factor	1.65
e.	Extent of core damage	5 percent of the fuel rods experience cladding failure
f.	Activity released to reactor coolant from fuel gap of failed fuel, percent of core inventory	
	1. Kr-85	10
	2. I-131	8
	3. Other noble gases	5
	4. Other iodines	5
	5. Alkali metals	12
g.	Iodine partition factor for steam generators	0.01
h.	Alkali metal partition factor for steam generators (based on full power moisture carryover of 0.25%)	0.0025
i.	Reactor coolant mass, lbs	3.99E+5
j.	Each steam generator water mass, lb,	
	0-2 hrs	82,500
	>2 hrs (assumes water level reaches 0% narrow range span)	121,250
II.	Atmospheric Dispersion Factors	see Table 15A-2

TABLE 15.3-3 (Sheet 2)

III.	Activity Release Data	
	a. Total primary to secondary leakage 0-12 hr, lb	6006
	b. Steam release from all steam generators	
	0-2 hr, lb	419,846
	2-12 hr, lb	1,352,918
IV.	Control Room	
	a. Control Room Switchover to Emergency Mode, sec	See Table 15A-5

TABLE 15.3-4

RADIOLOGICAL CONSEQUENCES OF A
LOCKED ROTOR ACCIDENT

	Wolf Creek Dose (rem TEDE)
Exclusion Area Boundary (limiting 2 hour duration)	3.8E-1
Low Population Zone Outer Boundary (duration)	3.2E-1
Control Room (30 days)	3.5E+0

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TABLE 15.3-5

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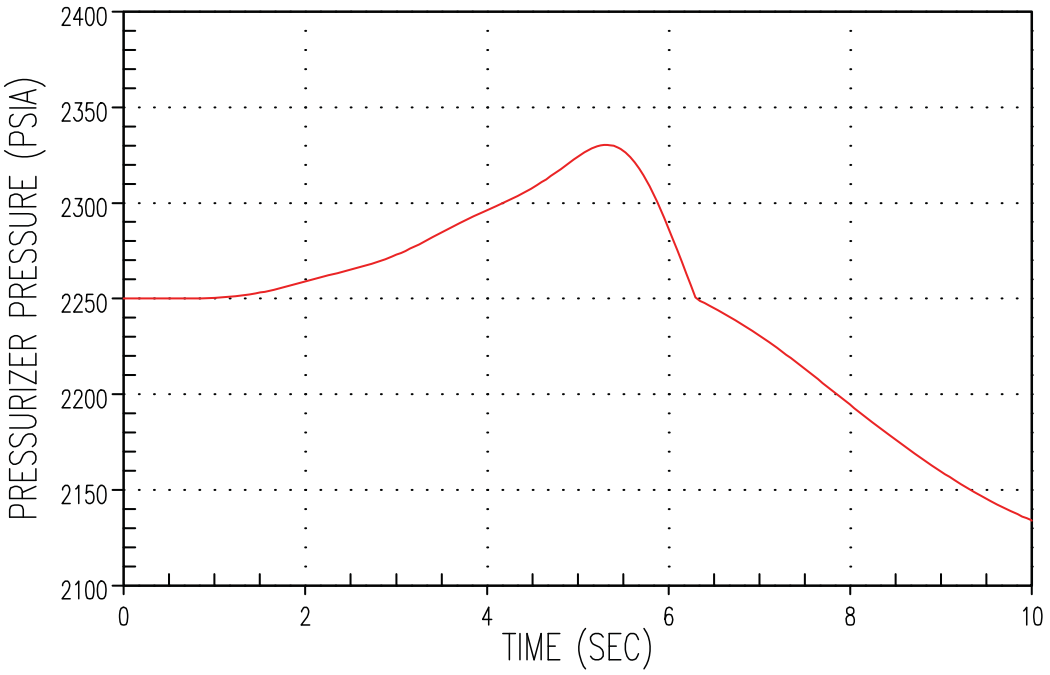

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FIGURE 15.3-1

PRESSURIZER PRESSURE TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

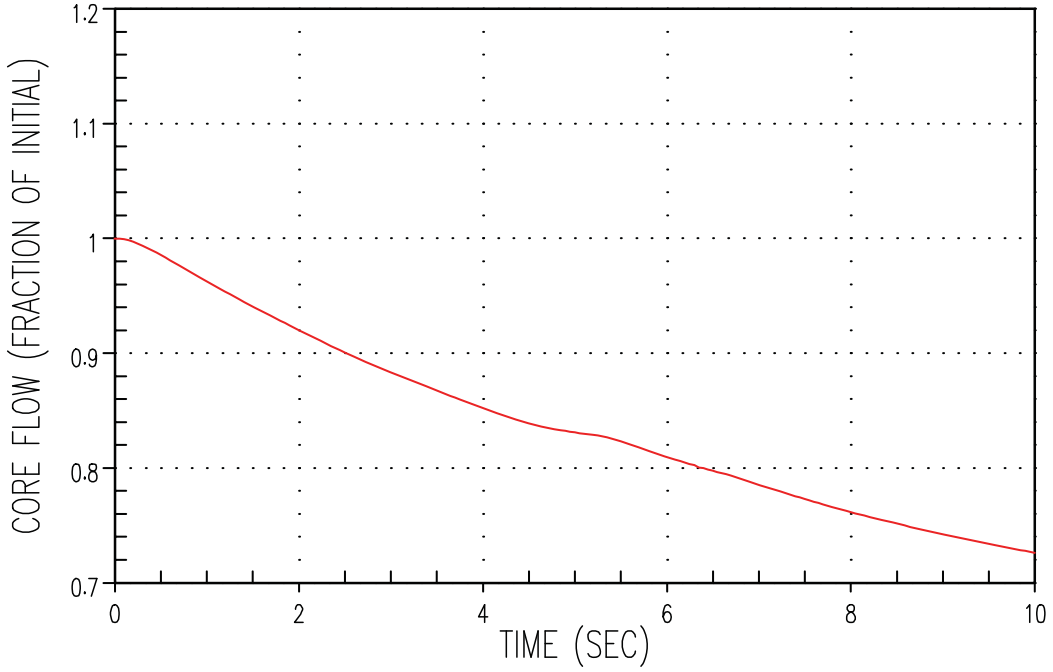

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FIGURE 15.3-2

TOTAL CORE FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

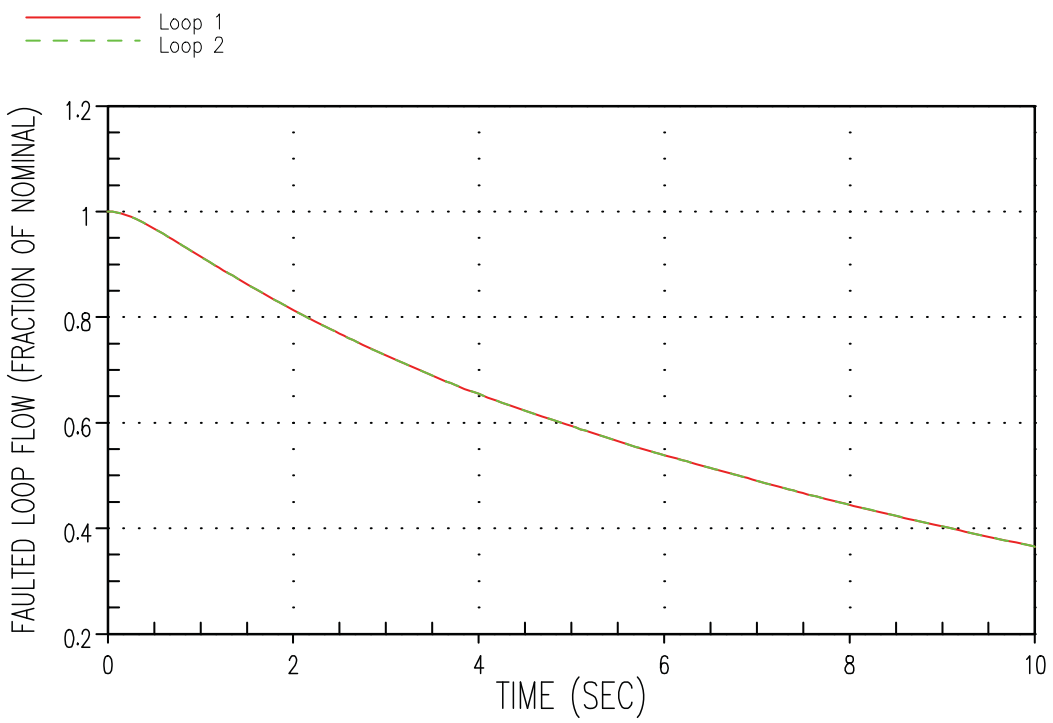

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FIGURE 15.3-3

FAULTED LOOP FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

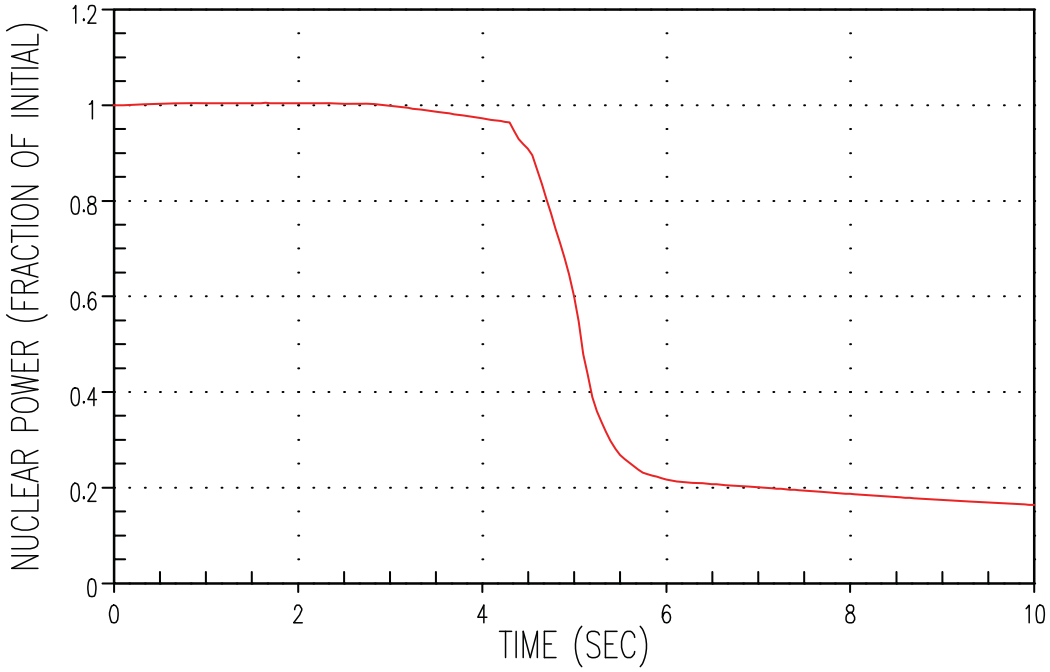

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FIGURE 15.3-4

NUCLEAR POWER TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

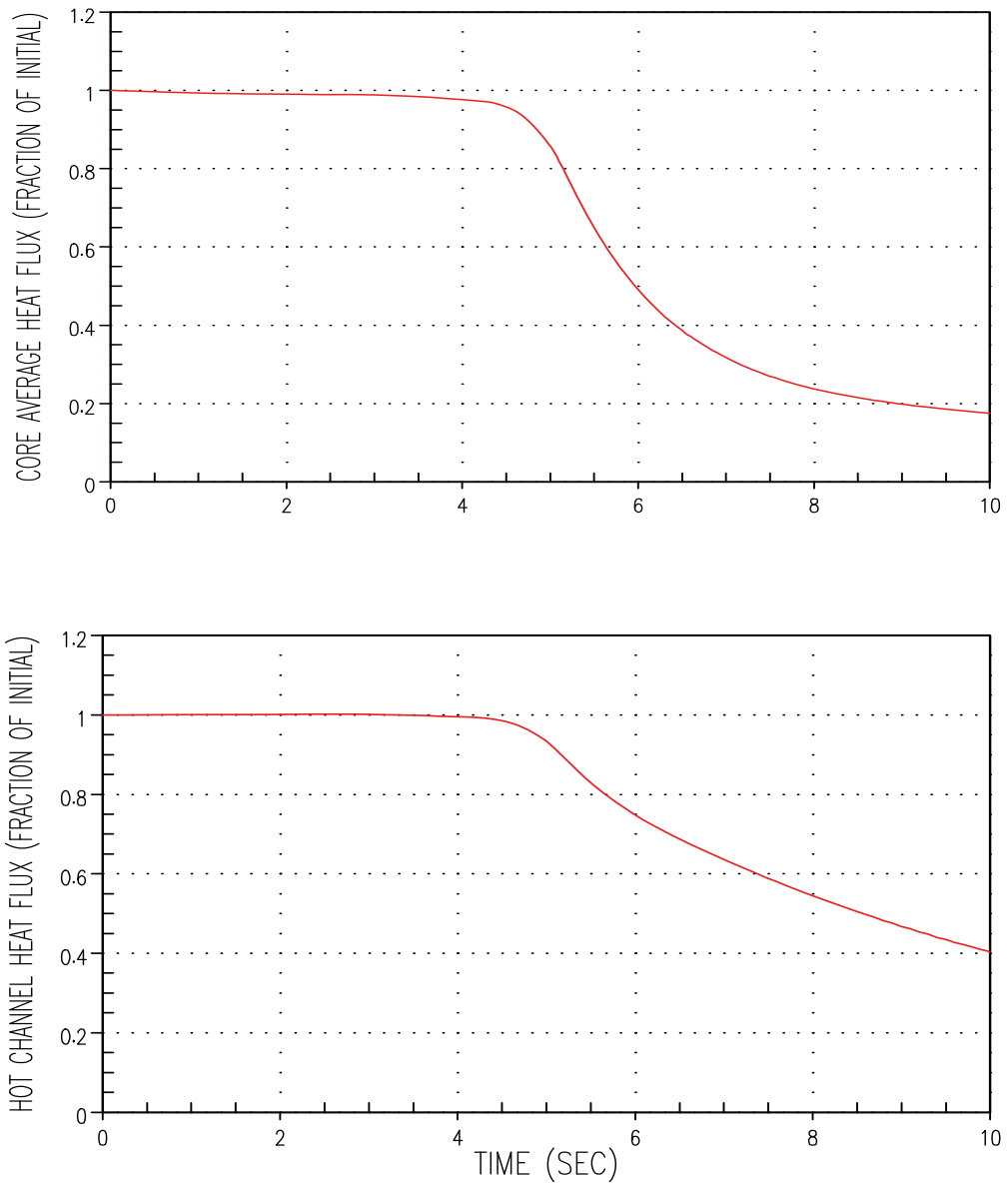

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FIGURE 15.3-5

CORE AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

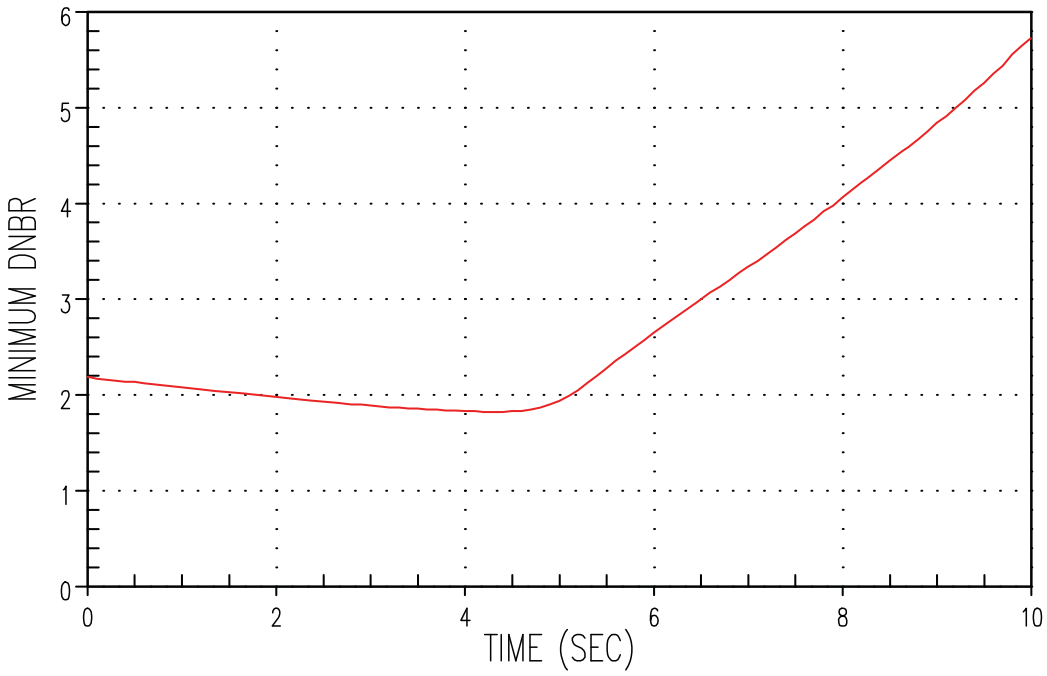

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FIGURE 15.3-6

DNBR TRANSIENT FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN

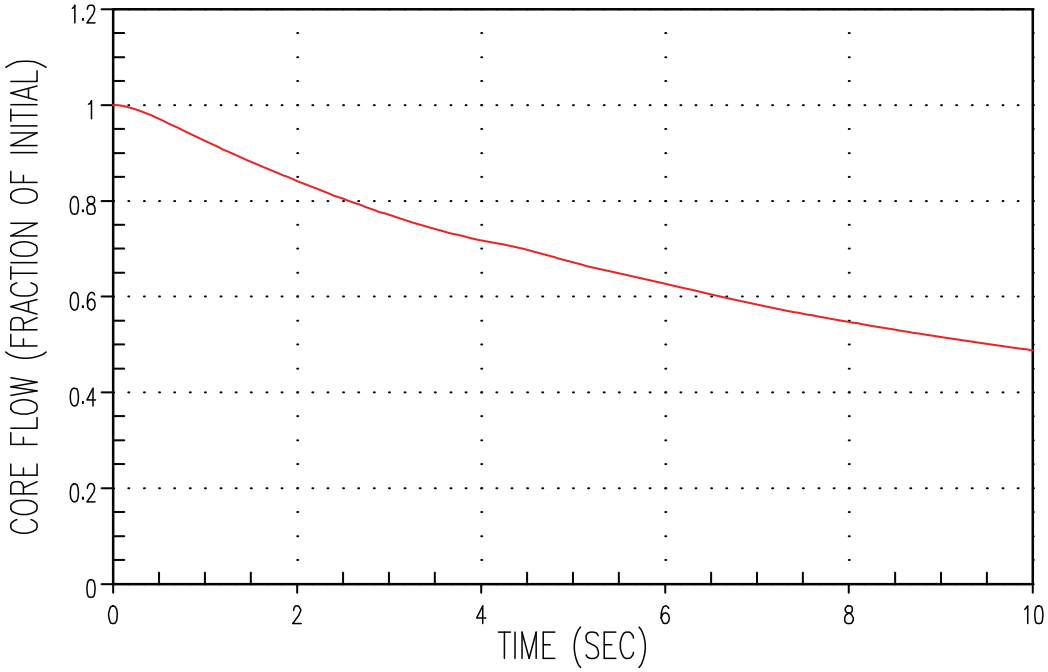

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FIGURE 15.3-7

TOTAL CORE FLOW TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

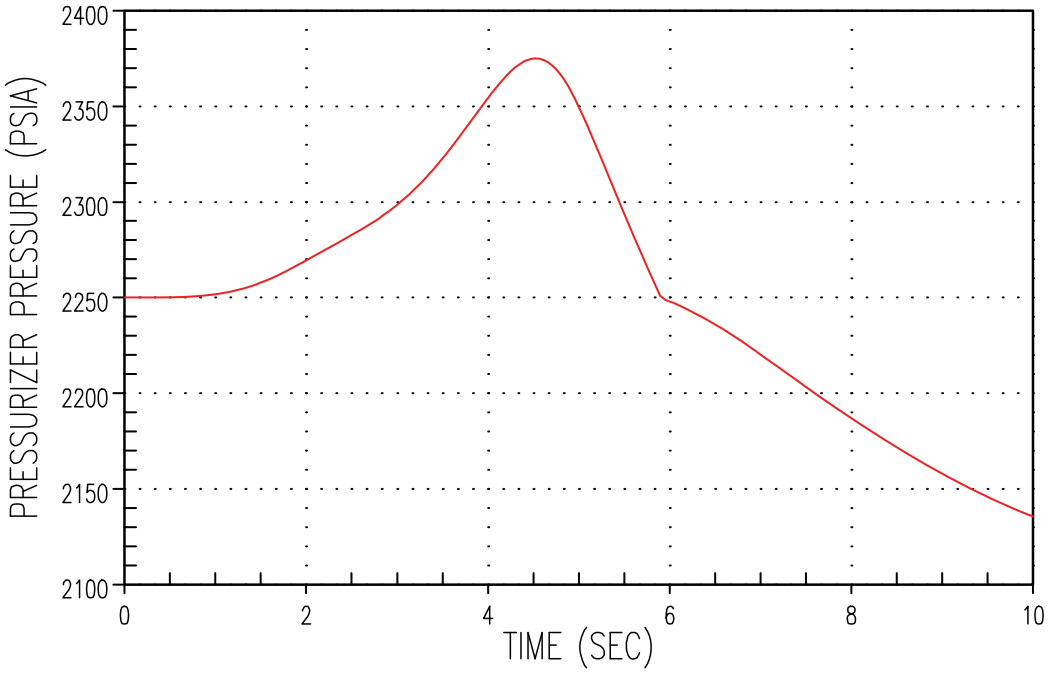

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FIGURE 15.3-8

PRESSURIZER PRESSURE TRANSIENT
FOR FOUR LOOPS IN OPERATION,
FOUR PUMPS COASTING DOWN

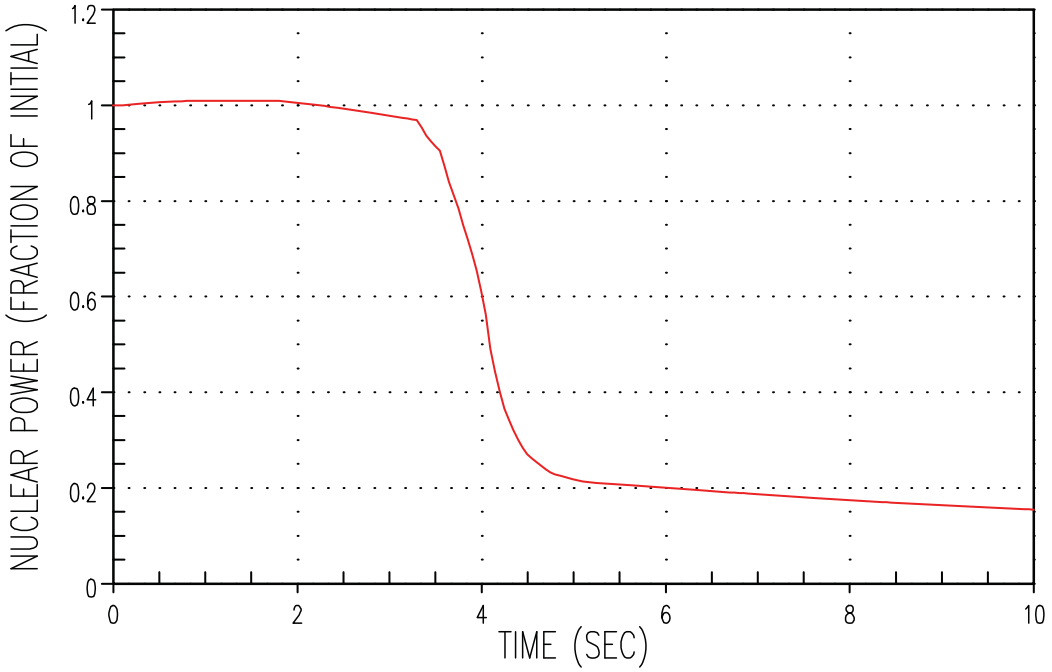

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FIGURE 15.3-9

NUCLEAR POWER TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

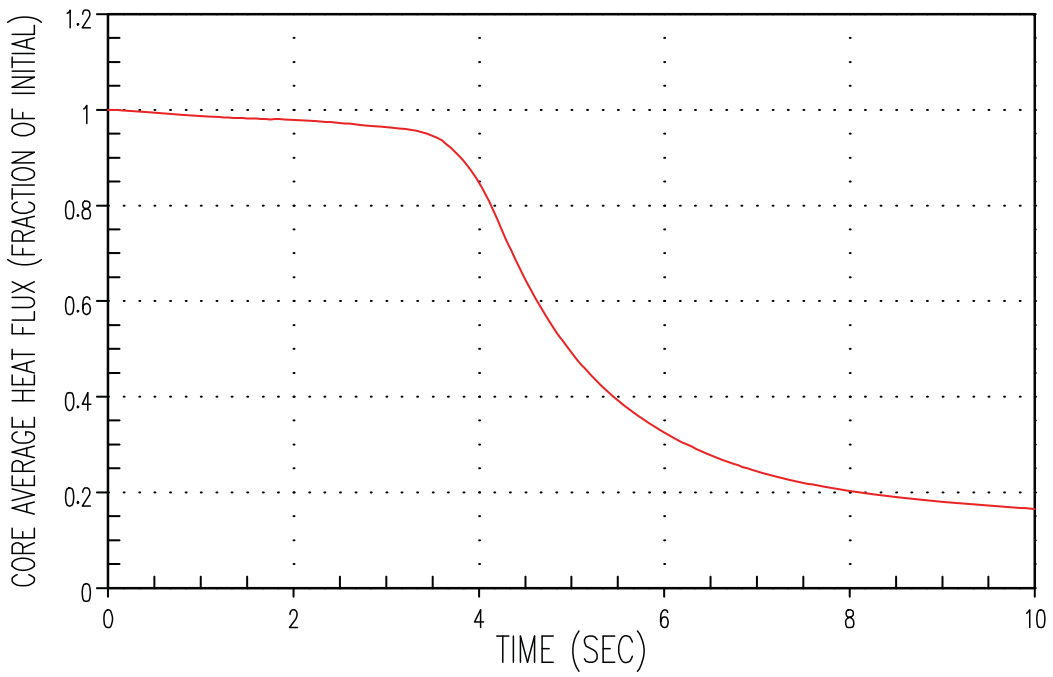

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FIGURE 15.3-10

CORE AVERAGE HEAT FLUX TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

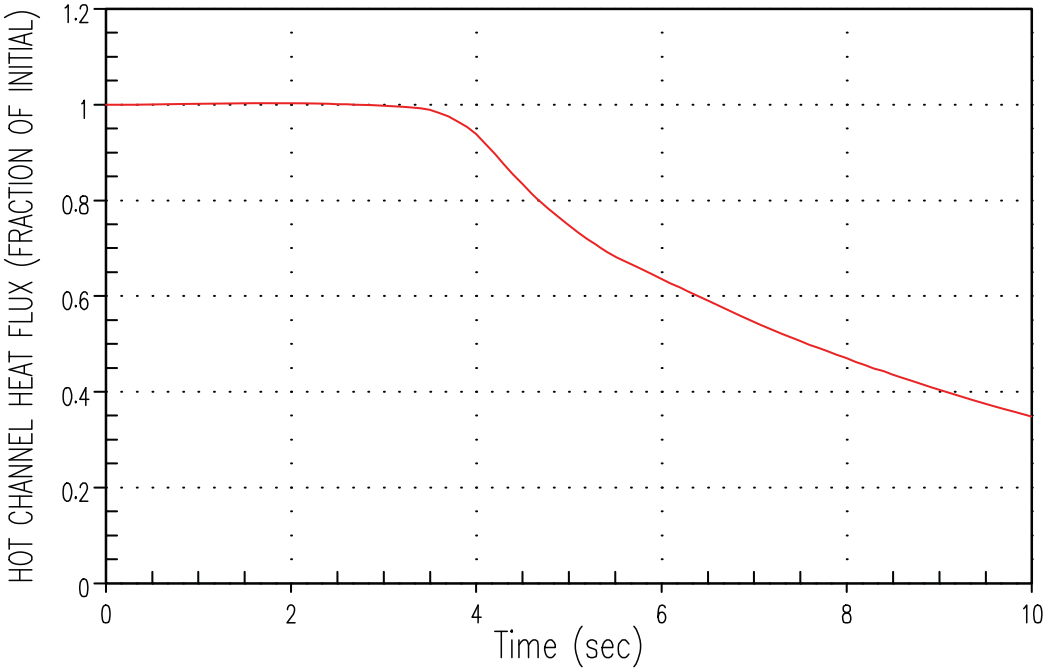

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FIGURE 15.3-11

HOT CHANNEL HEAT FLUX TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

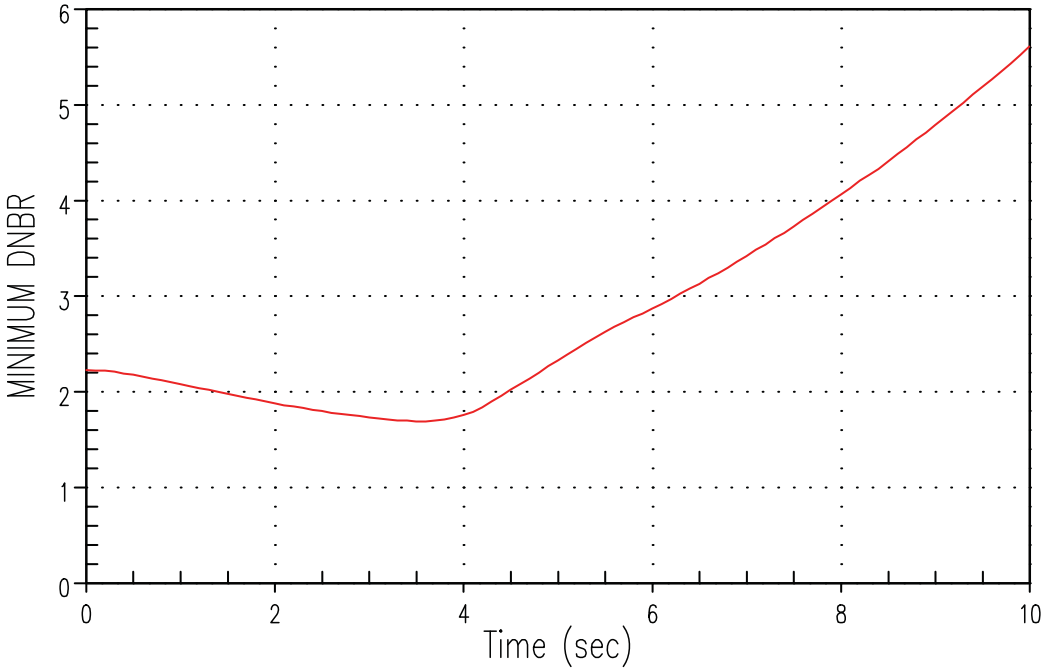

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FIGURE 15.3-12

DNBR TRANSIENT FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN

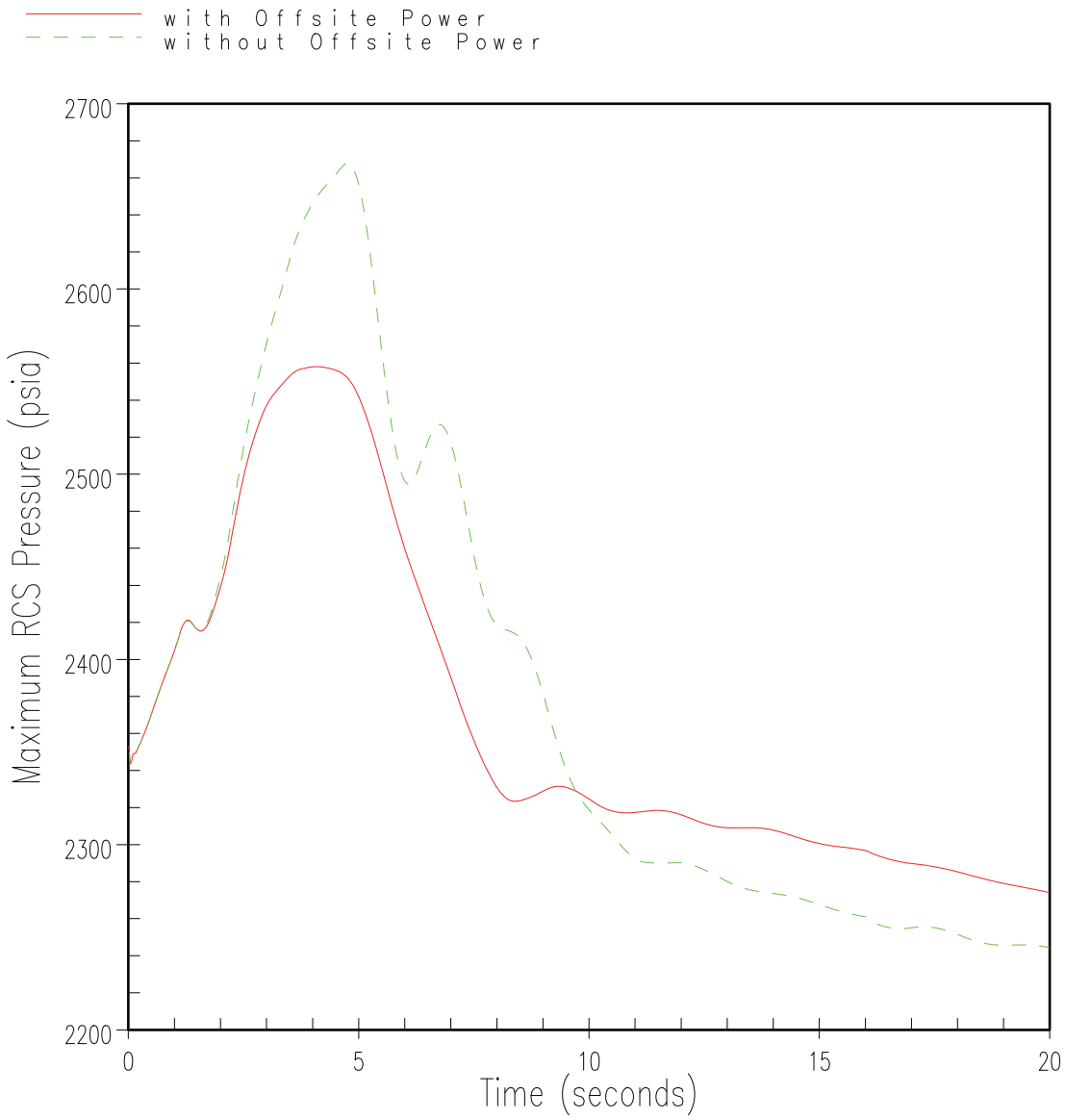

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FIGURE 15.3-13 PEAK RCS PRESSURE TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

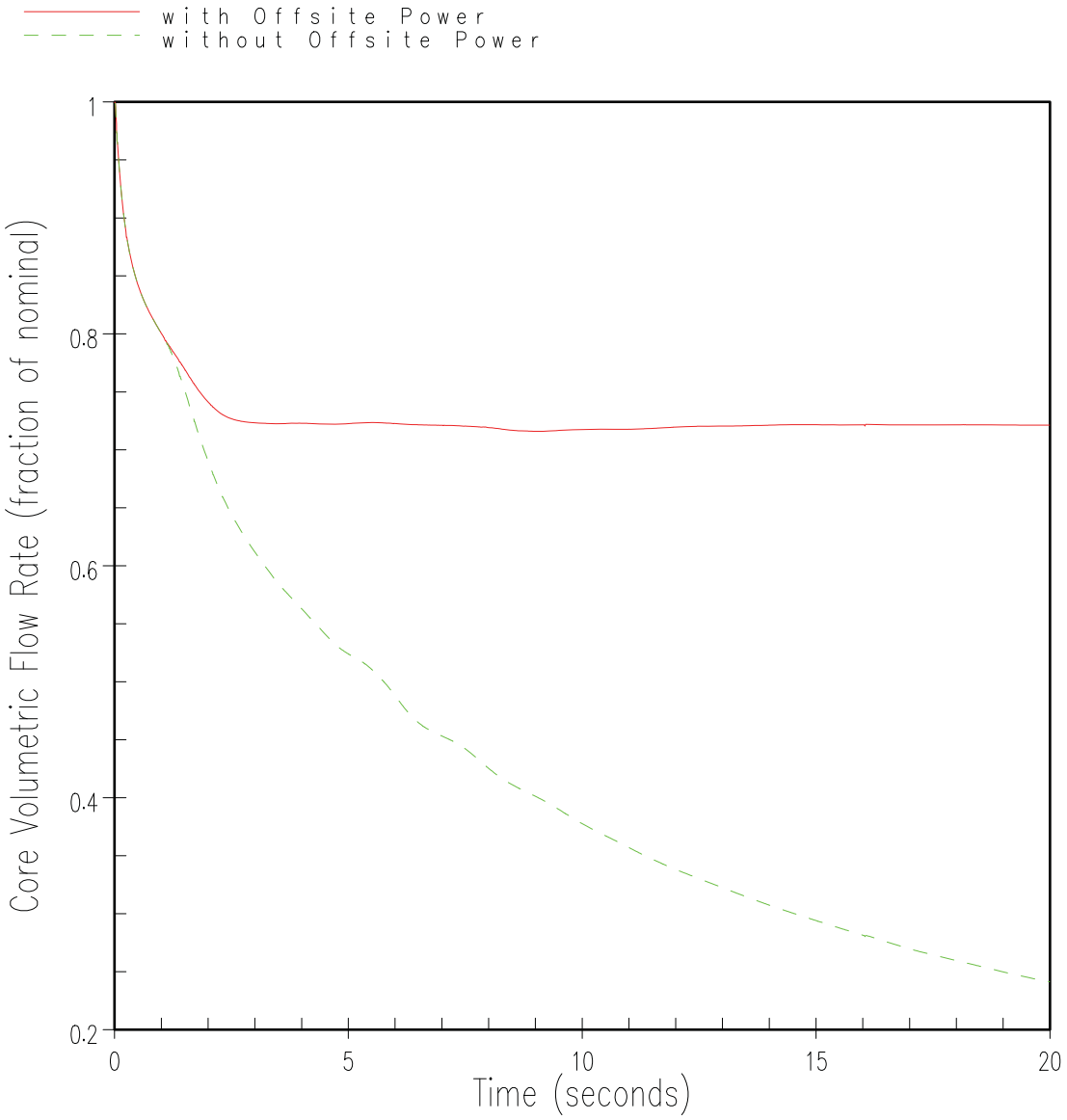

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FIGURE 15.3-14 CORE VOLUMETRIC FLOW RATE LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

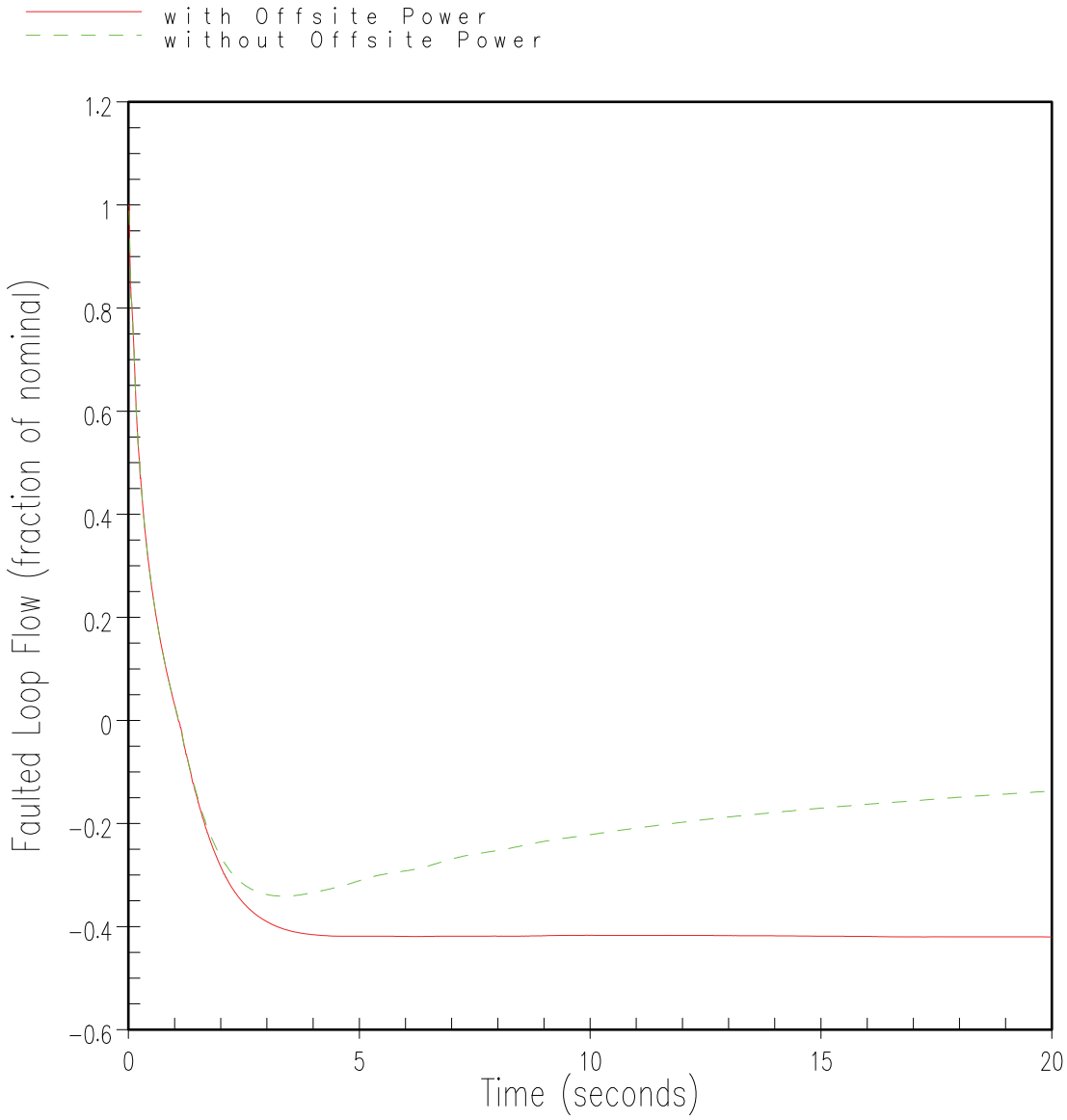

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FIGURE 15.3-15 FAULTED LOOP FLOW TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

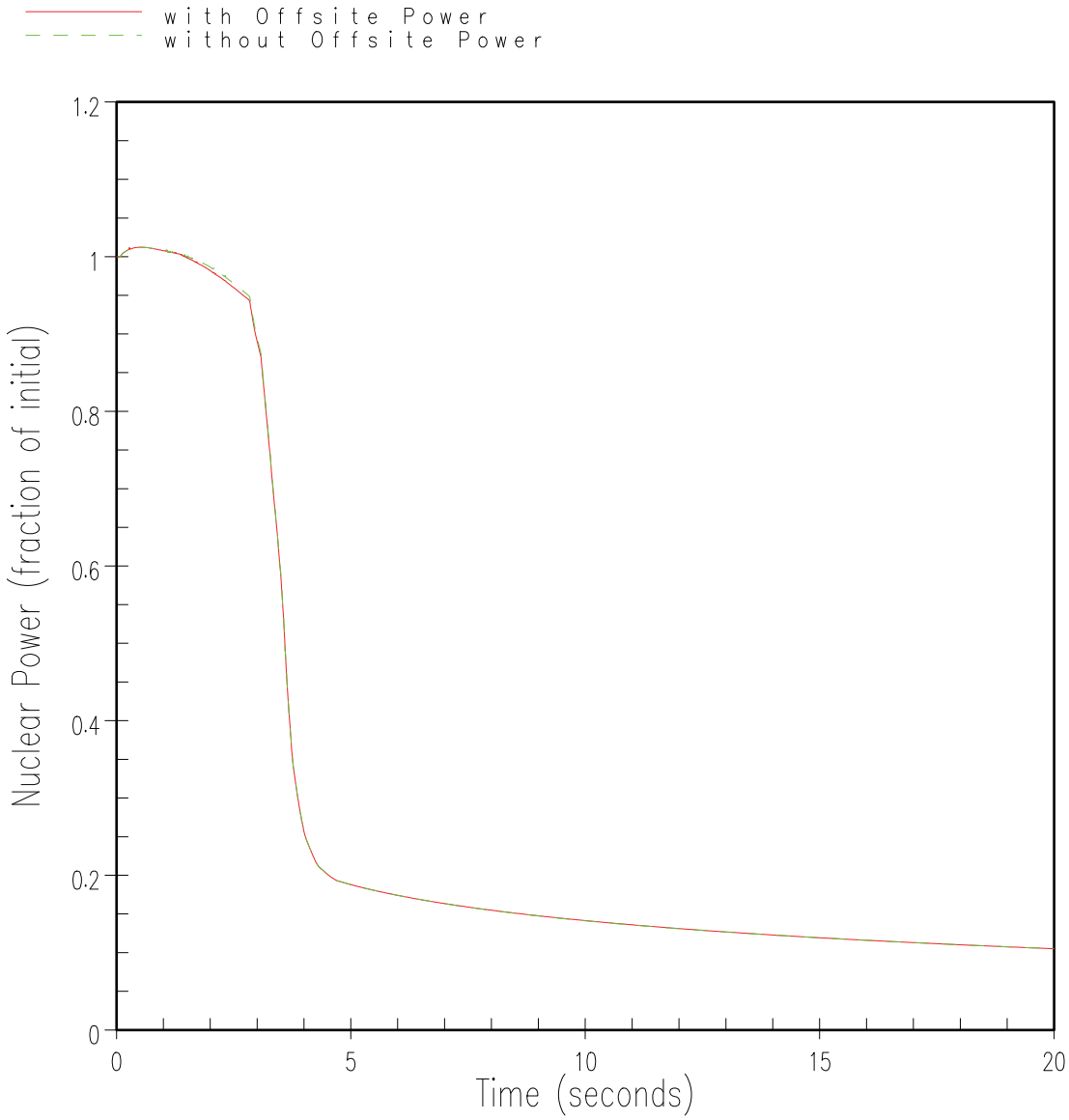

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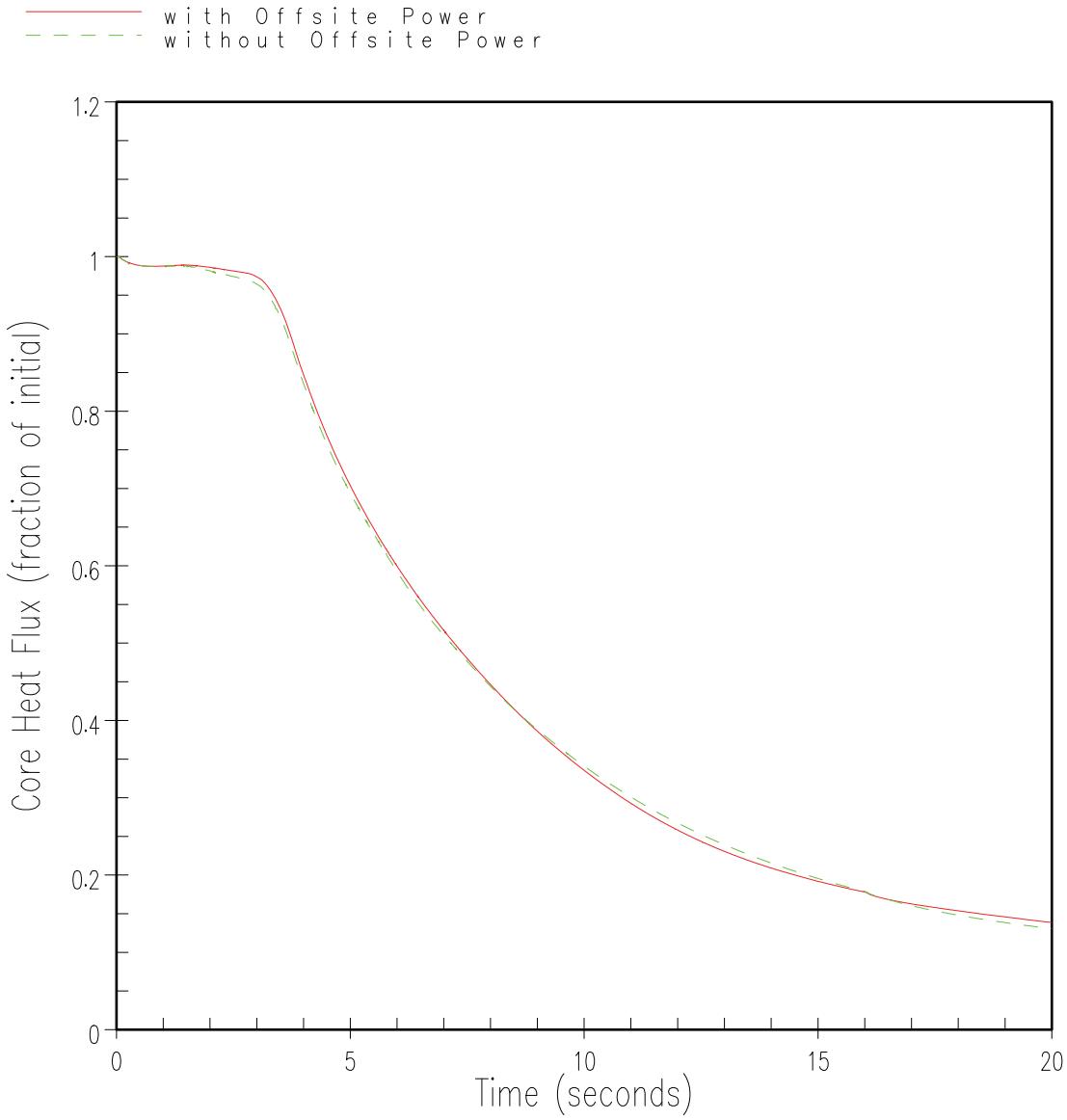
FIGURE 15.3-16 NORMALIZED POWER TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER



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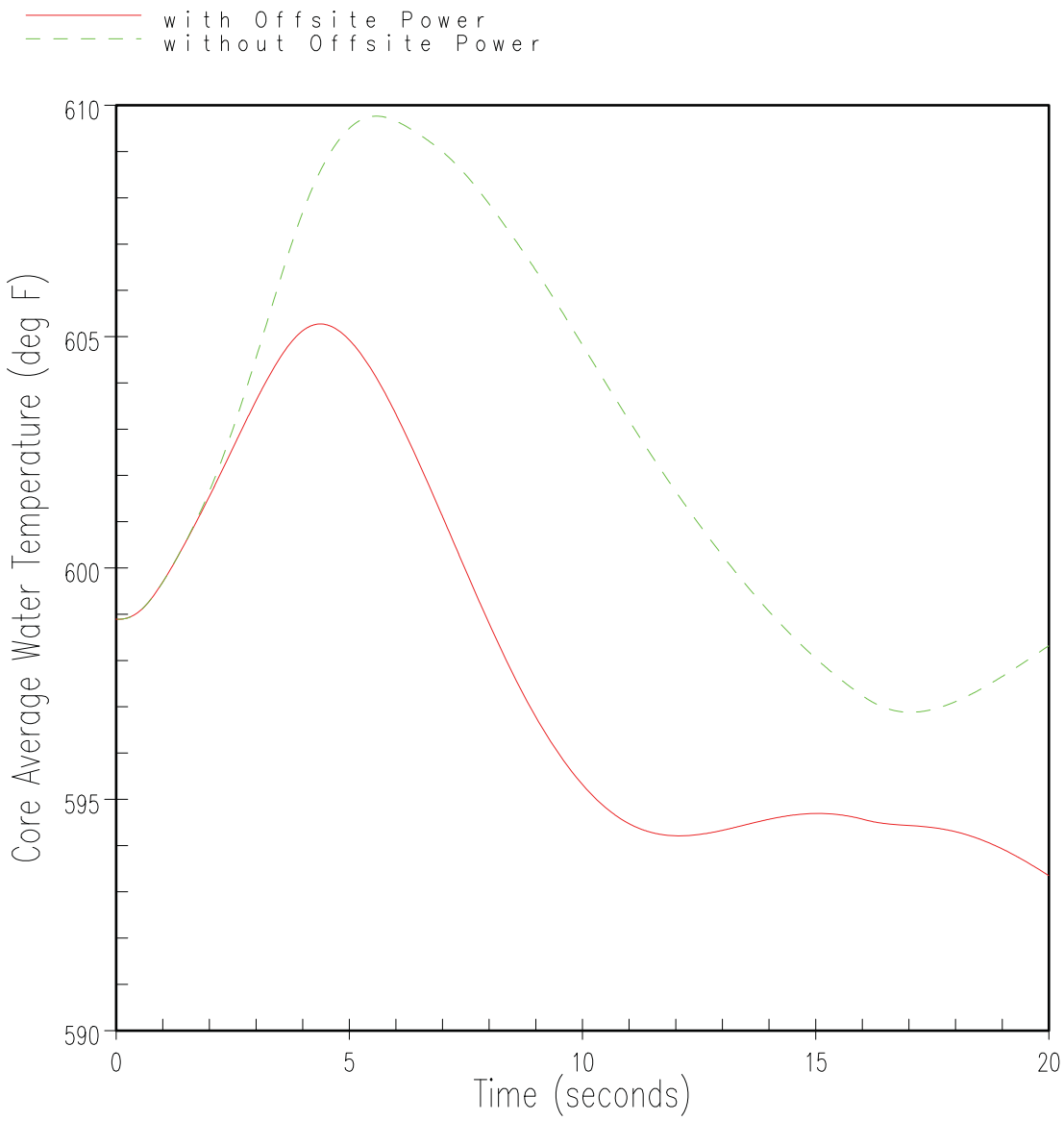
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.3-17 CORE HEAT FLUX TRANSIENT LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.3-18 CORE AVERAGE WATER TEMPERATURE, LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

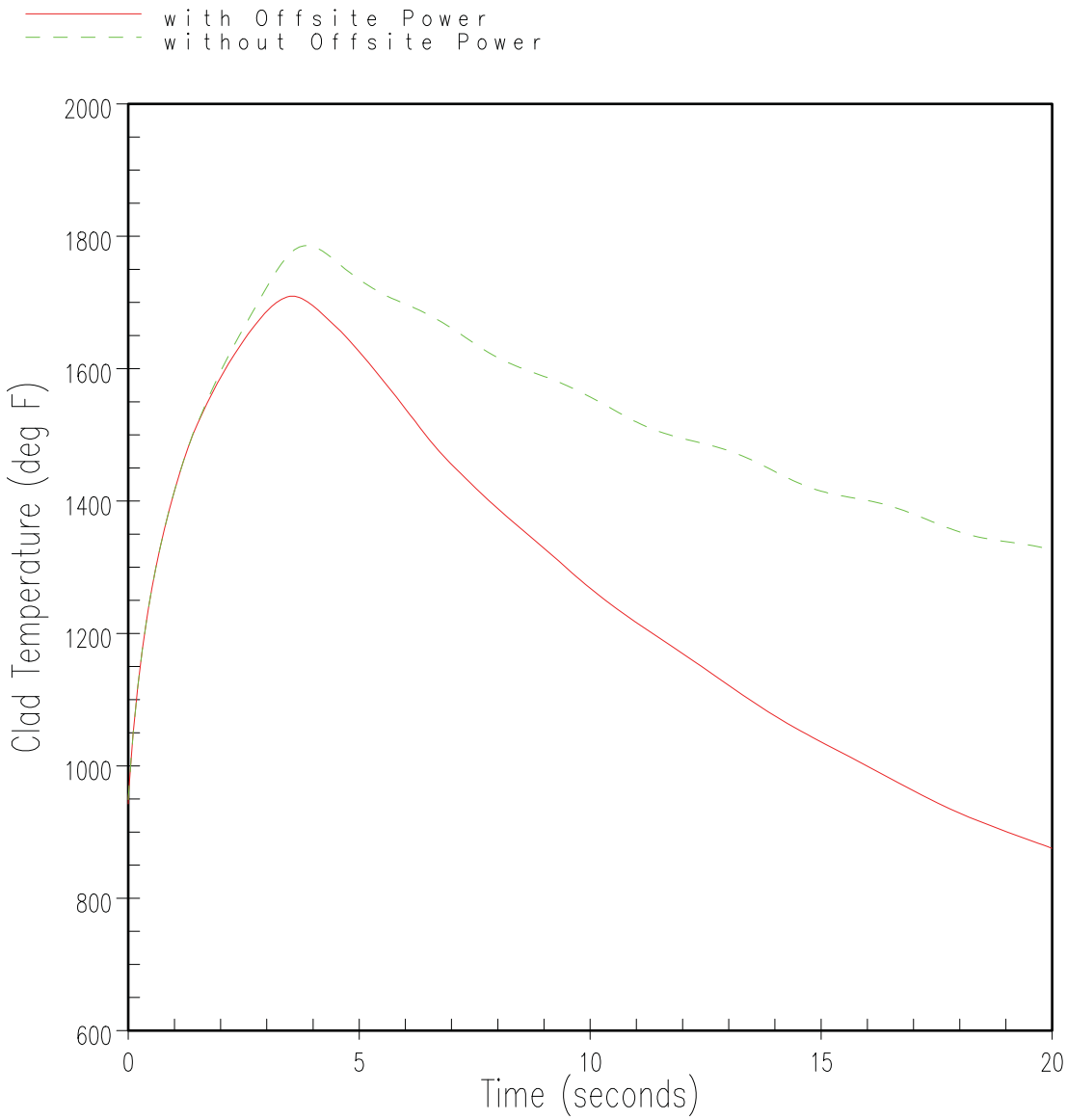

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FIGURE 15.3-19 PEAK CLAD TEMPERATURE, LOCKED ROTOR WITH AND WITHOUT OFFSITE POWER

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following events are presented in this section:

a. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition

b. Uncontrolled rod cluster control assembly bank withdrawal at power

c. Rod cluster control assembly misalignment

d. Startup of an inactive reactor coolant pump at an incorrect temperature

e. A malfunction or failure of the flow controller in a BWR recirculation loop that results in an increased reactor coolant flow rate (not applicable to Wolf Creek)

f. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

g. Inadvertent loading and operation of a fuel assembly in an improper position

h. Spectrum of rod cluster control assembly ejection accidents

Items a, b, d, and f above are considered to be ANS Condition II events; item g, an ANS Condition III event; and item h, an ANS Condition IV event. Item c entails both Conditions II and III events. Section 15.0.1 contains a discussion of ANS classifications.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing provided in Section 15.4.8. Therefore, radiological consequences are reported only for that limiting case.

15.4.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs, resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor subcritical, at hot zero power or at power. The "at power" case is discussed in Section 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.4.6).

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled so that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

This event is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance, since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

a. Source range high neutron flux reactor trip

This trip function is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

b. Intermediate range high neutron flux reactor trip

This trip function is actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. It may be manually bypassed only after two out of the four power range channels are reading above approximately 10 percent of full power, and is automatically reinstated when three out of the four channels indicate a power level below this value.

c. Power range high neutron flux reactor trip (low setting)

This trip function is actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power, and is automatically reinstated only after three out of the four channels indicate a power level below this value.

d. Power range high neutron flux reactor trip (high setting)

This trip function is actuated when two out of the four power range channels indicate a power level above a preset setpoint. It is always active.

e. High nuclear flux rate reactor trip

This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above the preset setpoint. It is always active.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed, using spatial neutron kinetics methods, TWINKLE (Ref. 1), to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Ref. 2). VIPRE (Ref 14) is used to evaluate the core thermal limits to determine DNBR.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To give conservative results for a startup accident, the following assumptions are made:

a. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler-only power coefficient, a conservatively low value is used. This does not correlate to Figure 15.0-2 since the TWINKLE computer code, on which the neutronics analysis is based, is a diffusion theory code rather than a point-kinetics approximation.

b. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux.

c. The reactor is assumed to be at hot zero power (557° F average coolant temperature). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect, thereby increasing the neutron flux peak. The initial effective multiplication factor (k_{eff}) is assumed to be 1.0, since this results in the worst nuclear power transient.

d. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent error increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.0.5 for RCCA insertion characteristics.

e. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.6.

f. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, are assumed in the DNB analysis.

g. The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

h. Two reactor coolant pumps are assumed to be in operation.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-16.

Results

Figures 15.4-1 through 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal accident, with the accident terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region.

Figure 15.4-1 shows the nuclear power transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient, since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the hot spot average fuel and inner cladding temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limiting value.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limiting value. The DNBR design basis is described in Section 4.4; applicable acceptance criteria have been met.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power (RWAP) results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB, overpressure, or a more severe accident. Therefore, to avert damage to the fuel clad or pressure boundary, the reactor protection system is designed to terminate any such transient before the DNB design basis is violated or the pressure boundary is threatened by over pressure.

This event is classified as an ANS Condition II incident (a fault of moderate frequency), as defined in Section 15.0.1.

The automatic features of the reactor protection system which prevent core damage and preclude RCS overpressurization during the postulated accident include the following:

a. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed any overpower setpoint.

b. A reactor trip actuated if any two out of four ΔT channels exceed the overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

c. A reactor trip actuated if any two out of four ΔT channels exceed the overpower ΔT setpoint. This setpoint is capable of being automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

d. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels which are set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

e. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent (Permissive 7).

f. A power range neutron flux high positive rate trip actuates a reactor trip if two out of four channels exceed any flux rate trip setpoint.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

a. High neutron flux (one out of four power range)

b. Overpower ΔT (two out of four)

c. Overtemperature ΔT (two out of four)

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed set-point); low pressure (fixed setpoint); neutron flux high positive flux rate trip (fixed rate setpoint) and overpower and overtemperature ΔT (variable setpoints).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

The RWAP transients are analyzed by employing the digital computer program RETRAN (Ref. 19).

VIPRE (Ref. 14) is used to evaluate the core thermal limits to determine DNBR. RETRAN generated state points for the limiting case (as determined by the RETRAN code) are used as VIPRE boundary conditions to perform a DNB analysis.

The LOFTRAN code (Ref. 3) was used to demonstrate that RCS overpressure safety limits were protected.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To obtain conservative results with respect to core damage for an uncontrolled RCCA bank withdrawal at power accident, the following assumptions are made:

a. The accident is analyzed with the Revised Thermal Design Procedure (Ref. 20). Uncertainties in initial conditions, with the exception of power, were included in the DNBR safety analysis limit (Ref. 20). The initial nuclear steam supply system (NSSS) power included all applicable uncertainties, while the initial pressurizer pressure and RCS temperatures were assumed to be at their nominal values (see Section 15.0.3).

b. Initial conditions corresponding to a spectrum of core powers of 10%, 60%, and 100% of the analyzed full power value were considered.

c. Reactivity coefficients - two conditions are analyzed:

1. Minimum reactivity feedback

Consistent with the beginning of core life, a positive moderator temperature coefficient of +6 pcm/°F is assumed for cases initiated at less than full power. A zero moderator temperature coefficient is assumed for cases initiated at full power. A variable Doppler power coefficient with core power is used in the analysis; a conservatively small (in absolute magnitude) value is assumed.

2. Maximum reactivity feedback

A conservatively large positive moderator density coefficient (0.47 dk/gm/CC) and a large (in absolute magnitude) Doppler power coefficient are assumed.

d. The reactor trip on high neutron flux was modeled to actuate at the safety analysis limit of 118 percent of the analyzed full power value. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

e. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

f. A range of reactivity insertion rates from 1 to 110 pcm/sec was examined. The maximum assumed positive reactivity insertion rate is greater than that which corresponds to the simultaneous withdrawal at maximum speed of the two control banks, having the maximum combined worth.

For the RCS overpressure analysis, the preceding assumptions still apply, with the following differences, in order to obtain conservative results with respect to RCS overpressurization:

a. The accident is analyzed with the Standard Thermal Design Procedure. Initial reactor power, pressure, and RCS temperature were assumed to be within their respective allowable operating ranges with uncertainties applied in the conservative directions (see Section 15.0.3).

b. Minimum reactivity feedback conditions were modeled.

c. The pressurizer sprays were not modeled because operation of the pressurizer spray valves would minimize the pressure increase during the transient.

d. A range of initial power levels from 8% to 100% of the analyzed full power value, and a range of reactivity insertion rates from 1 pcm/sec to 110 pcm/sec, were examined.

e. The high positive neutron flux rate trip was modeled with response lag, setpoint, and trip delay values of 2 seconds, 6.9%/sec, and 1 second, respectively.

f. The pressurizer safety valves were modeled with an opening setpoint that includes the maximum positive setpoint tolerance plus a positive set pressure shift to account for the effect of the water-filled loop seals (Ref. 15); a maximum opening delay was also modeled, and it includes the time required to purge the loop seal water.The pressurizer power-operated relief valves were not modeled as their operation is non-conservative for the overpressure transient.

g. The main steam safety valves were modeled with maximum setpoints that account for setpoint tolerance in order to conservatively prolong the mismatch between core heat generation and secondary heat removal capability.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-16.

Results

Figures 15.4-4a through 15.4-4c show the transient results for a rapid uncontrolled RCCA bank withdrawal incident (110 pcm/sec) starting from 100% power with minimum reactivity feedback. Figures 14.4-5a through 14.4-5c show the transient results for a slow uncontrolled RCCA bank withdrawal incident (1 pcm/sec) starting from 100% power with minimum reactivity feedback. The limiting DNB case was the 12 pcm/sec reactivity addition rate case initiated at 10% power with minimum reactivity feedback (beginning of life) conditions. Detailed DNB calculations performed using the VIPRE code (Ref. 14) demonstrated that the DNB design basis was met for this limiting case. Primary protection over the range of DNB cases analyzed is high neutron flux and overtemperature ΔT, with high flux providing protection for the faster transients (high reactivity addition rates) while the overtemperature ΔT provided protection for the slower transients (low reactivity addition rates). Margin to DNB is maintained for the limiting case based on the detailed thermal-hydraulic calculations performed by the VIPRE code. The design basis for DNBR is described in Section 4.4.

Figures 15.4-6a, 15.4-6b, and 15.4-6c show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 100, 60, and 10 percent power, respectively. The minimum DNBR values presented in the figures were calculated using the RETRAN code and are representative of the trends for minimum DNBR and not indicative of final calculated DNBR values. The detailed thermal-hydraulic code VIPRE was used to confirm that the limiting case satisfied the DNB design basis.

Figures 15.4-7a through 15.4-7c show the transient results of the RCS pressure analysis of the RWAP accident event. The limiting conditions are beginning of life, 74% rated thermal power and 21 pcm/sec reactivity addition rate. Peak pressure response is shown in Figure 15.4-7c with a peak pressure that is less than 110% of the design pressure. Primary protection over the range of RCS pressure cases analyzed is high pressurizer pressure, high neutron flux, and high neutron flux positive rate.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 121 percent of the analyzed full power value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 121 percent of the analyzed full power value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will remain below the fuel melting temperature.

The reactor is tripped sufficiently fast during the RCCA bank withdrawal at power transient to preclude DNB, and so the ability of the primary coolant to remove heat from the fuel rods is maintained. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate fuel integrity protection over the entire range of possible reactivity addition rates, i.e., the DNB design basis is met and core damage is precluded. The high neutron flux, high pressurizer pressure, and neutron flux high positive rate reactor trip functions, in conjunction with the pressurizer safety valves and steam generator safety valves, provide adequate RCS overpressure protection over the entire range of possible reactivity addition rates and initial conditions analyzed, i.e., the analysis demonstrates that the integrity of the RCS pressure boundary is maintained as the maximum transient pressure does not exceed the RCS pressure boundary safety limit.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

a. One or more dropped RCCAs within the same group, or a dropped RCCA bank

b. Statically misaligned RCCA

c. Withdrawal of a single RCCA

Each RCCA has a position indicator channel 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 at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into groups of four or five. The RCCAs comprising a group operate in parallel through multiplexing thyristors. The groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion or immobility.

The dropped RCCA assemblies, dropped RCCA assembly bank, and statically misaligned RCCA assembly events are classified as ANS Condition II incidents (incidents of moderate frequency), as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures [probability for single random failure is on the order of 10^{-4}/year (refer to Section 7.7.2.2)] or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low. The limiting consequences of such errors or failures may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . ."

This classification is in accordance with GDC-25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control <u>rods</u>." (Emphasis has been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC-25.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the nuclear instrumentation system

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

c. Rod at bottom signal

d. Rod deviation alarm (control rods only)

e. Rod position indication

Misaligned RCCAs are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

b. Rod deviation alarm (control rods only)

c. Rod position indication

The resolution of the rod position indicator channel is +12 steps (+7.5 inches). Deviation of any RCCA from its group by twice this distance (+24 steps or +15.0 inches) will not cause power distributions worse than the design limits. The deviation alarm (control rods only) alerts the operator to rod deviation with respect to the group position in excess of 12 steps. If the rod deviation alarm is not functional, the operator is required to take action required by the Technical Requirements Manual.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the nonindicated RCCAs. The operator is also required to take action, as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciators, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of the various control rod misoperations and discussed in Section 15.0.8 and listed in 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figures 15.0-17 & 18.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

Method of Analysis

a. One or more dropped RCCA from the same group:

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN computer code (Reference 3) with the methodology described in Reference 11.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transients and the hot channel factor from the nuclear analysis, the DNB basis can be shown to be met using dropped rod limit lines developed with the VIPRE computer code (Reference 14).

b. Statically misaligned RCCA:

No transient analysis is performed, rather steady state power distributions are analyzed using the ANC computer code (Reference 16). The peaking factors are then compared to peaking factor limits developed using the VIPRE computer code (Reference 14), which are based on meeting the DNB design criterion.

Results

a. One or more dropped RCCAs from the same group or bank:

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. If the worth of the dropped rod(s) is sufficiently large, and if the turbine load is constant, a reactor trip may occur on low pressurizer pressure due to the mismatch between the reactor power and the turbine power. The core is not adversely affected during this period since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs which do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.4-10 and 15.4-11a show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. In all cases, the minimum DNBR remains above the limit values.

b. Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion with an absolute value greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described in part a; however, the return to power will be less due to the greater worth of the entire bank.

c. Statically misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The Control Rod insertion limits in the Core Operating Limits Report (COLR) may vary from time to time, depending on a number of limiting criteria. The full power insertion limits on control bank D must be chosen to be above that position which meets minimum DNBR and peaking factors. The full power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle, depending on fuel arrangements.

For this RCCA misalignment with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperature are at their nominal values including uncertainties (as given in Subsection 15.0.3.2), but with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming that the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident, and thus, the ability of the primary coolant to remove heat from the fuel is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single RCCA Withdrawal

Method of Analysis

Power distributions within the core are calculated using the ANC computer code (Reference 16). The peaking factors are then used by the VIPRE computer code (Reference 14) to calculate the DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at the initial core power, pressurizer pressure and vessel average temperature at the respective nominal values, and the reactor coolant flow is at the minimum measured flow value. Uncertainties in the initial conditions are accounted for in the DNBR limit value as described in Reference 20.

Results

For the single rod withdrawal event, two cases have been considered, as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Section 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending upon initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the limit value. Evaluation of this case at the power and coolant conditions, at which the overtemperature ΔT trip would be expected to trip the plant, shows that an upper limit for the number of rods with a DNBR less than the limit value is five percent.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case 1 described above.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, the DNBR remains greater than the safety analysis limit values and, therefore, the DNB design basis is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

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

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Historically, administrative procedures had required that the unit be brought to a load of less than 10 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

Should the startup of an inactive reactor coolant pump accidentally occur, the ensuing transient is terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three-loop operation. As the plant Technical Specifications require all four reactor coolant system loops to be in operation while at power or in startup conditions (Modes 1 and 2), this event is administratively precluded, and thus no analysis is required.

Given that no analysis is required, the information presented in the remainder of this section is retained for historical purposes only.

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by using three digital computer codes. The LOFTRAN code (Ref. 3) is used to calculate the loop and core flow, nuclear power and core pressure, and temperature transients following the startup of an idle pump. VIPRE-01 (Ref 14) is used to evaluate the core thermal limits to determine DNBR. LOFTRAN generated state points are used as VIPRE-01 boundary conditions to perform the DNB analysis.

Plant characteristics and initial conditions are discussed in Section 15.0.3. To obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

a. Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. These values are consistent with maximum steady state power level allowed with three loops in operation. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

b. Following initiation of startup of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value in approximately 27 seconds.

c. A conservatively large negative moderator temperature coefficient.

d. A least negative Doppler only power coefficient (see Figure 15.0-2).

e. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

f. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0-8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-19.

Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4-12 through 15.4-16. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit. See Section 4.4 for a description of the DNBR design basis.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.4-12.

The calculated sequence of events for this accident is shown on Table 15.4-1. The transient results illustrated in Figures 15.4-12 through 15.4-16 indicate that a stabilized plant condition, with the reactor tripped, is rapidly attained. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Conclusions

The transient results show that the core is not adversely affected. There is a considerable margin to the limiting DNBR; thus, the DNB design basis, as described in Section 4.4, is met.

15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

This section is not applicable to Wolf Creek.

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 Accident Description

One of the two principal means of positive reactivity insertion to the core is the addition of unborated, primary grade water from the demineralized and reactor makeup water system (RMWS) into the

RCS through the reactor makeup portion of the chemical and volume control system (CVCS). Boron dilution with these systems is a manually initiated operation under strict administrative controls requiring close operator surveillance with procedures limiting the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup's boron concentration to that of the RCS during normal charging.

The means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The CVCS and RMWS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which, with indication by alarms and instrumentation, will allow sufficient time for automatic or operator response (depending on the mode of operation) to terminate the dilution. An inadvertent dilution from the RMWS may be terminated by closing the primary water makeup control valve. All expected sources of dilution may be terminated by closing isolation valves in the CVCS, BG-LCV-112B and C. The lost shutdown margin (SDM) may be regained by the opening of isolation valves to the RWST, BN-LCV-112D and E, thus allowing the addition of borated water to the RCS.

Generally, to dilute, the operator must perform two distinct actions:

a. Switch control of the makeup from the automatic makeup mode to the dilute mode

b. Momentarily place handswitch BGHS0026, RCS M/U CTRL, to RUN

Failure to carry out either of the above actions prevents initiation of dilution. Also, during normal operation the operator may add borated water to the RCS by blending boric acid from the boric acid storage tanks with primary grade water. This requires the operator to determine the concentration of the addition and to set the blended flow rate and the boric acid flow rate. The makeup controller will then limit the sum of the boric acid flow rate and primary grade water flow rate to the blended flow rate, i.e., the controller determines the primary grade water flow rate after the start button is depressed.

The status of the RCS makeup is continuously available to the operator by:

a. Indication of the boric acid and blended flow rates

b. CVCS and RMWS pump status lights

c. Deviation alarms, if the boric acid or blended flow rates deviate by more than 10 percent from the preset values

d. High charging flow rate alarm

e. Volume control tank high and high-high water level and high pressure alarms

f. Letdown divert valve position alarm

g. Source range neutron flux - when reactor is subcritical

1. High flux at shutdown alarm

2. Indicated source range neutron flux count rates

3. Audible source range neutron flux count rate

4. Source range neutron flux - doubling alarm

h. When the reactor is critical

1. Axial flux difference alarm (reactor power $\geq$50 percent RTP)

2. Control rod insertion limit low and low-low alarms

3. Overtemperature ΔT alarm (at power)

4. Overtemperature ΔT turbine runback (at power)

5. Overtemperature ΔT reactor trip

6. Power range neutron flux - high, both high and low setpoint reactor trips

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Section 15.0.1.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-20.

15.4.6.2 Analysis of Effects and Consequences

To cover all phases of plant operation, boron dilution during Cold Shutdown, Hot Shutdown, Hot Standby, Start-up, and Power modes of operation is considered in this analysis. Conservative values for necessary parameters were used, i.e., high RCS critical boron concentrations, high boron worths, minimum shutdown margins, and lower than actual RCS volumes. These assumptions result in conservative determinations of the time available for operator or system response after detection of a dilution transient in progress.

Dilution During Refueling

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Valves V-178 and V-601 (or V-602) in the CVCS will be locked closed during refueling operations. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the RWST by the RHR pumps or the CCPs if using the CCP to flood up the Refueling Pool.

Dilution During Cold Shutdown, Hot Shutdown and Hot Standby

The Technical Specifications require the reactor to be shutdown to a SDM value as specified in the COLR when the unit is operating in Mode 5, Cold Shutdown. When in Modes 4 or 3, (Hot Shutdown or Hot Standby), the Technical Specifications require the plant to be shutdown to a SDM value as specified in the COLR. The following conditions are assumed for the analysis of the inadvertent boron dilution event while in these operating modes:

a. An assumed dilution flow rate of 157.5 gpm is used. This value corresponds to the high charging flow rate alarm plus a 5% allowance for uncertainties.

b. A minimum RCS water volume of 8639 ft^3 is assumed. This volume is a conservative estimate of the active volume of the RCS when at least one reactor coolant pump is in operation.

c. If no reactor coolant pump is in operation, all dilution sources are isolated or under administrative control.

Dilution During Start-up

In this mode, the plant is being taken from one long-term mode of operation, Hot Standby, to another, Power. The plant is maintained in the Start-up mode only for the purpose of start-up testing at the beginning of each cycle. During this mode of operation, the plant is in manual control, i.e., T_{avg}/rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. The COLR and Technical Specifications require an available trip reactivity of 1.3 percent $\Delta K/K$ (in MODE 2 with $K_{eff} < 1.0$) and four reactor coolant pumps operating. Other conditions assumed are:

a. Dilution flow is the maximum capacity of two centrifugal charging pumps with the RCS at 2250 psia (approximately 245 gpm).

b. A minimum RCS water volume of 9810 ft^3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

Dilution During Full Power Operation

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The COLR specifies an available trip reactivity of 1.3 percent $\Delta K/K$ and four reactor coolant pumps operating. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the centrifugal charging pumps (analysis is performed assuming two charging pumps are in operation even though normal operation is with one pump). Conditions assumed for this mode are:

a. Dilution flow from two centrifugal charging pumps is at the maximum at an RCS pressure of 2250 psia (approximately 245 gpm) when the reactor is in manual control. When in automatic control, the dilution flow is the maximum letdown flow (approximately 120 gpm).

b. A minimum RCS water volume of 9810 ft^3. This is a conservative estimate of the active RCS volume, minus the pressurizer volume.

15.4.6.3 Conclusions

Dilution During Refueling

Dilution during this mode has been precluded through administrative control of valves in the possible dilution flow paths (see Section 15.4.6.2).

Dilution During Cold Shutdown, Hot Shutdown or Hot Standby

In Modes 3, 4 or 5, the reactor operators are relied upon to detect and recover from a inadvertent boron dilution event. Numerous alarms from the CVCS and RMWS and in the Nuclear Instrumentation System are available to provide assistance to the reactor operator in the detection of an inadvertent boron dilution event. In the analyses of the event initiated from Modes 3, 4 or 5, the VCT high water level alarm (at 70% span) is credited as the initial alarm. The time required to perform the VCT/RWST valve swap-over and to purge the charging lines of dilute water are included in the calculation of the time available to the reactor operators to initiate corrective actions. In Modes 3, 4 and 5, analyses have demonstrated that the reactor operators have at least 15 minutes in which to initiate actions to terminate the dilution and initiate boration of the RCS.

There may exist other dilution events not obviously bounded by the analysis scenario described above. For example, during small dilution flow rate situations, the time required to fill the VCT to the high VCT water level

setpoint may exceed the time required to dilute the RCS to the critical condition. However, the time intervals involved in this case are relatively long and alternate alarms, such as the concentrated boric acid flow and total makeup flow deviation alarms, would alert the reactor operator to a potential inadvertent boron dilution. For these relatively slow transients, an alternate event acceptance criterion is applied. This criterion requires that the time between the start of the event and the complete loss of shutdown margin (inadvertent criticality) be at least 30 minutes.

Dilution During Startup

This mode of operation is a transitory mode to go to power and is the operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the reactor is in manual rod control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications require that the operator determine the estimated critical

position of the control rods prior to approaching criticality thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the Source Range reactor trip after receiving P-6 from the Intermediate Range. Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the Source Range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

For Mode 2 - Startup, the reactor is assumed to trip when the flux level reaches the source range high flux trip setpoint. In the event of an inadvertent dilution, no credit is taken for operator action blocking the source range trip (permissive P-6). This reactor trip also functions as an alarm indicating a boron dilution transient is under way, as the control rods are not moving. The control rods are assumed to be at the insertion limits, minimizing the available shutdown margin; it is assumed that the RCS boron concentration at the time of reactor trip corresponds to the predicted critical boron concentration for rods at the insertion limits. Note: Due to the selection of boron concentration assumptions, the time available from the alarm is calculated to be identical to the total time from event initiation to loss of shutdown margin, i.e., the time of alarm will coincide with the analysis initial condition assumption. After reactor trip there is at least 15 minutes for operator action prior to return to criticality. The required operator action is the opening of valves BN-LCV-112D and E to initiate boration and the closing of valves BG-LCV-112B and C to terminate dilution.

Dilution During Full Power Operation

With the reactor under manual rod control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. After reactor trip there is at least 50 minutes for operator action prior to return to criticality. The required operator action is the opening of valves BN-LCV-112D and E and the closing of valves BG-LCV-112B and C. The boron dilution transient in this case is essentially equivalent to an uncontrolled rod withdraw at power. A reactor trip occurs when either the Hi Neutron Flux or the Overtemperature ΔT setpoint is reached. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be within the range of insertion rates analyzed for uncontrolled rod withdraw at power.

It should be noted that prior to reaching the Overtemperature ΔT reactor trip the operator will have received an alarm on Overtemperature ΔT and an Overtemperature ΔT turbine runback. With the reactor in automatic rod control, the pressurizer level controller will limit the dilution flow rate to the maximum letdown rate, approximately 120 gpm. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller will throttle charging flow down to match the letdown rate.

Thus, with the reactor in automatic rod control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in at least three alarms to the operator:

a. rod insertion limit - low level alarm.

b. rod insertion limit - low-low level alarm if insertion continued after (a) above, and

c. axial flux difference alarm (ΔT outside of the target band).

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. For example, the operator has at least 110 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at beginning-of-life. The time would be significantly longer at end-of-life, due to the low initial boron concentration, when shutdown margin is a concern.

The above results demonstrate that in all modes of operation an inadvertent boron dilution is precluded, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition with the operator regaining the required shutdown margin.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN IMPROPER POSITION

15.4.7.1 Acceptance Criteria

This event is classified as a Condition III event. Condition III events are defined as those events that do not cause more than a small fraction of fuel rods to fail, although sufficient fuel damage might occur to preclude immediate resumption of operation. The specific acceptance criteria for this event are as follows:

a. To meet the requirements of General Design Criteria 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

b. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

15.4.7.2 Identification of Causes and Accident Description

The Inadvertent Loading Event comprises core misloading scenarios such as the loading of one or more fuel assemblies into improper positions, the loading of a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. In addition to these scenarios, misloading events involving burnable absorbers are theoretically possible, scenarios such as the placement of a cluster of 20 burnable absorbers into a core location slated to have 24 burnable absorbers. All of these misloading scenarios potentially result in a core reactivity distribution that differs from the intended core reactivity

distribution. As a result, the core power distribution and peaking factors may differ from predictions. Specifically, misloading errors can lead to increased local power peaking at the location of the misloading if the misloading results in a local reactivity increase relative to the intended pattern. If the misloading results in a local reactivity decrease, power peaking increases away from the location of the misloading are possible due to unintended power tilts. These kinds of increases, however, are generally distributed over a large core volume and are small relative to those where the local reactivity is increased.

Fuel misloads are prevented by the manufacturing controls employed to build the fuel and the core loading controls used to assemble the core. The manufacturing controls include checks on fuel rod weight to confirm the uranium loading in the fuel rod, active and passive gamma scans of individual fuel rods to confirm fuel enrichments, pellet stack lengths, pellet types, and the absence of pellet gaps during fuel manufacturing, and bar coding of each fuel rod to confirm its proper placement in the fuel assembly.

To reduce the probability of core loading errors during fuel loading, each fuel assembly and core component is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification numbers are checked before each assembly is moved into the core. Identification numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed. These procedures make the likelihood of core misloadings very small.

The severity and detectability of fuel misloads are influenced by several factors: the local reactivity perturbation relative to the intended core loading pattern, the core position of the misload, the local environment of the misloaded fuel assembly, and the number of operable incore detector locations and their proximity to the misload location. Should misloadings occur, the incore system of movable flux detectors, which is used to verify power distributions during startup and throughout the operating cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce large numbers of fuel rod failures. In addition, thermocouples and excore detectors can provide additional indications of power distribution anomalies. This instrumentation, along with the startup testing performed each cycle, make the detection of severe misloadings highly likely.

15.4.7.3 Evaluation

The incore moveable detector system is used to search for potential fuel misloads at the start of each operating cycle. Following fuel loading and low power physics testing, an initial core power distribution measurement is made. The core power level of this initial flux map is typically between ~30% and ~50% of rated thermal power. This initial power distribution measurement is used to confirm that the measured power distribution is consistent with the predicted power distribution. Observed flux map deviations in excess of the flux map review criteria (see Table 15.4-5) would prompt an investigation of a possible core anomaly. This satisfies the first acceptance criterion given above.

In Reference 22, a large number of misloads were evaluated for representative core designs employing current fuel types and fuel features. The simulated misloads, involving one or two fuel assemblies, covered a wide range of local reactivity perturbations and core positions. The resulting hot full power

(HFP) $F_{\Delta H}$ peaking factors ranged from benign to very severe. Severe misloads with peaking factors that exceed the FΔH limit for DNB at normal operation conditions have the potential for fuel failure if they remain undetected. The simulated misloads were assessed with respect to severity and detectability.

The detectability assessments of Reference 22 demonstrated that the incore detector system is very robust with respect to detection of misloads severe enough to fail fuel during normal operation. By examining a large number of moveable detector thimble patterns, the detectability assessments considered the effect of inoperable moveable detector thimbles on misload detectability. Even when the minimum number of operable detector locations allowed per the plant licensing bases was assumed, the incore detector system was capable of reliably detecting misloads severe enough to fail fuel during normal operation.

Fuel misloads involving a single fuel rod or fuel pellet were not evaluated as part of Reference 22. Such misloads, in general, will not be detectable using the incore detector system due to the very small power distribution perturbation. In terms of increased peaking factors and reduced DNBR values, however, the consequences of such misloads will be very small and limited to the affected fuel rod and the immediately adjacent fuel rods.

Detection of fuel misloads is, in part, a function of the number of available incore detector locations. Reference 22 demonstrated that the flux map review criteria of Table 15.4-5 are effective in detecting fuel misloads that could lead to fuel failures during normal operation. To enhance the probability that significant misloads will be detected, tighter review criteria are employed when the number of available detector locations is reduced. These review criteria will be used for startup and subsequent at-power flux maps.

The detectability assessments of Reference 22 confirm that the moveable detector system can reliably detect fuel misloads that could fail fuel during normal operation when the Table 15.4-5 review criteria are employed. Specifically, Reference 22 demonstrated that only a small fraction of 1% of misloads severe enough to fail fuel during normal operation would be undetected at startup using these limited review criteria. Furthermore, it was judged that even these "undetected" misloads would very likely be detected if other attributes of the startup power distribution measurement (e.g., tilts and reaction rate error contours) were considered along with the results of low power physics testing. Given that detection of >99% of misloads severe enough to fail fuel is expected using these review criteria, a radiological consequences analysis is deemed unnecessary. Failures in fresh fuel during startup would have negligible radiological consequences since there is only a small fission product inventory. Following startup, any fuel rod failures would occur gradually and would be detected by coolant activity monitoring. Since the number of potential fuel rod failures due to a core misload would be extremely small and such failures would occur gradually, any coolant activity releases would initially be well within the cleanup capacity of the plant. Any trend in increased coolant activity would warrant further investigation and evaluation. Therefore, the second acceptance criterion for this event would be satisfied since failures would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidelines.

15.4.7.4 Conclusion

Fuel misloads are prevented by manufacturing controls and core loading controls. In the unlikely event that a fuel misload should occur, the incore moveable detector system is capable of reliably detecting misloads that could fail fuel at normal operation conditions. Exceeding the review criteria herein would initiate an investigation to identify potential core anomalies. Any failures associated with an undetected fuel misload would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidelines.

15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY 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 an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

Certain features in the WCGS pressurized water reactor are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative, mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

a. Each control rod drive mechanism housing is completely assembled and shop tested at 4,100 psi.

b. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed RCS.

c. Stress levels in the mechanism are not affected by anticipated system transients at power, or by the 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 1 components.

d. The latch mechanism housing and rode travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which are encountered.

A significant margin of strength in the elastic range, together with the large 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 head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds which are subject to periodic inspections.

Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated for by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger-than-normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at a low level alarm and shutdown margin verification or boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 6. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. The control rod drive mechanism is described in Section 3.9(N).4.

Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2,250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If a failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite

ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the separated section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the separated section upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield, it would dissipate its kinetic energy without penetrating the shield. The water jet from the break would continue to push the separated section against the missile shield.

If the separated section of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the separated section from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding separated section were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage (sufficient to cause failure of an adjacent housing).

Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident. See Section 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of an RCCA ejection accident, some fuel damage would be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Ref. 7).

Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ

significantly from the TREAT (Ref. 8) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are (Ref. 12):

a. Average fuel pellet enthalpy at the hot spot is below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

b. Peak reactor coolant pressure is less than that which could cause stresses to exceed the faulted condition stress limits.

c. Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion a above.

15.4.8.2 Analysis of Effects and Consequences

15.4.8.2.1 Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed, using spatial neutron kinetics methods, to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 6 and Reference 12.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Ref. 1), is used for the average core transient analysis. This code uses cross sections generated by LEOPARD (Ref. 5) to solve the two group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2,000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code, since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal value times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection is coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN (Ref. 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter (Ref. 17) or Jens-Lottes (Ref. 18) correlation to determine the film heat transfer coefficient before DNB, and the Bishop-Sandburg-Tong correlation (Ref. 9) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero bulk fluid quality. The DNBR is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 15.0.11.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient, taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing. The overpressure analysis is a generic analysis provided in Reference 6. The results of the generic analysis remain applicable to WCGS.

15.4.8.2.2 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-2 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated, using either three-dimensional static methods or by a synthesis method employing one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties. No allowance for nuclear power peaking due to densification is required, as is justified in Reference 21.

Power distributions before and after ejection for a "worst case" can be found in Reference 6. During initial plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature increases, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time, accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative, compared to three-dimensional analysis (Ref. 6).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life and end-of-life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative, compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level, using a one-dimensional steady state computer code with a Doppler weighting factor of 1.0. The Doppler defect used does not correlate to Figure 15.0-2 since the TWINKLE computer code, on which the neutronic analysis is based, is a diffusion-theory code rather than a point-kinetics approximation. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning-of-life and 0.50 percent at end-of-life for the first cycle. The accident is sensitive to if the ejected rod worth is equal to or greater than as in zero power transients. In order to allow for future cycles, conservative estimates of β of 0.49 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.7 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod and adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1 percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for a typical four-loop plant, assuming the maximum possible size break (2.75-inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1,200 psi) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary systems, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of

borated safety injection flow starting 1 minute after the break is sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

As discussed in Section 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-21.

Results

Cases are presented for both beginning and end-of-life at zero and full power.

a. Beginning of cycle, full power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.23 percent Δk and 6.6, respectively. The peak hotspot clad average temperature reached 2,270°F. The peak hot spot fuel center temperature reached melting at 4,900 °F. However, melting was restricted to less than 10 percent of the pellet.

b. Beginning of cycle, zero power

For this condition, control bank D was assumed to be fully inserted, and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.78 percent Δk and a hot channel factor of 13.0. The peak hotspot clad average temperature reached 2,498 °F. The peak hot spot fuel center temperature was 3,867 °F.

c. End of cycle, full power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.250 percent Δk and 7.1, respectively. The peak hotspot clad average temperature reached 2,191 °F. The peak hotspot fuel center temperature reached melting at 4,800°F. However, melting was restricted to less than 10 percent of the pellet.

d. End of cycle, zero power

The ejected rod worth and hot channel factor for this case were obtained, assuming control bank D to be fully inserted with banks B and C at their insertion limits. The results were 0.86 percent Δk and 21.0, respectively The peak hotspot clad average temperature reached 2,626 °F. The peak fuel center temperature was 3,869 °F.

A summary of the cases presented above is given in Table 15.4-2. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning-of-life, full power and end-of-life, zero power) are presented in Figures 15.4-22 through 15.4-25.

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-22 through 15.4-25, is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Section 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss-of-coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods experiencing DNB. In all cases considered, less than 10 percent of the rods experienced the DNB. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident. Parameters assumed in the radiological consequences analysis are summarized in Table 15.4-3.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Ref. 6). Since the

severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow of coolant away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

15.4.8.3.1 Method of Analysis

15.4.8.3.1.1 Physical Model

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a time sufficient to establish equilibrium levels of activity in the reactor coolant and secondary systems.

The RCCA ejection results in reactivity being inserted to the core which causes the local power to rise. In a conservative analysis, it is assumed that partial cladding failure and fuel melting occurs. The fuel pellet and gap activities are assumed to be immediately and uniformly released within the reactor coolant or containment, depending on which release path is being considered.

Two release paths to the environment exist which are analyzed separately and conservatively, as if all the activity is available for release from each path.

The activity released to the containment from the reactor coolant through the ruptured control rod mechanism pressure housing is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere. The only removal processes considered in the containment are radioactive decay and leakage from the containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the fraction of the activity leaking from the reactor coolant through the steam generator tubes. The leakage of reactor coolant to the secondary side of the steam generator continues until the residual heat removal system can match decay heat and steam releases from the steam generators are terminated. After this time, no more releases to the environment occur.

15.4.8.3.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.4-3 and 15A-1 and summarized below. The assumptions are consistent with Regulatory Guide 1.183. Tables 15B-1 and 15B-7 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

The following conditions are used to calculate the activity released and the offsite doses following a RCCA ejection accident.

a. 10 percent of the fuel rod gap activity for iodine and noble gas and 12 percent of the fuel rod gap activity for alkali metal is released to the reactor coolant.

b. 0.25 percent of the fuel is assumed to melt.

c. Following the incident until RHR operation to take over decay heat removal at 12 hours, secondary steam is released to the environment. The total quantity of steam released is given in Table 15.4.3.

d. The 1-gpm primary-to-secondary leakage is to the unaffected steam generators.

e. All noble gas activity in the reactor coolant which is transported to the secondary system via the primary-to-secondary leakage is assumed to be immediately released to the environment.

f. Fission products released from the fuel-cladding gap of the damaged fuel rods are assumed to be instantaneously and homogeneously mixed with the reactor coolant for the primary to secondary release pathway case and instantaneously and homogeneously mixed within containment for the containment leakage pathway case.

g. The iodine and alkali metal activity present in the primary to secondary leakage is assumed to mix homogeneously with the water in the steam generators.

h. A partition factor of 0.01 between the vapor and liquid phases for radioiodine in the steam generators is utilized to determine iodine releases to the environment via steam venting from the steam generators.

i. The full power moisture carryover of 0.25% is utilized to determine the alkali metal releases to the environment via steam venting from the steam generators.

j. The activity released from the steam generators is immediately vented to the environment.

k. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

l. No credit is taken for radioactive decay or ground deposition during radioactivity transport to offsite or control room locations.

m. Short-term accident atmospheric dispersion factors corresponding to ground level releases, breathing rates, and dose conversion factors are given in Tables 15A-2, 15A-1, and 15A-4, respectively.

n. Offsite power is assumed lost.

15.4.8.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and are provided in Table 15A-2.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.4.8.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The leakage pathways are:

a. Direct steam dump to the atmosphere through the secondary system relief valves for the secondary steam

b. Primary-to-secondary steam generator tube leakage and subsequent steam dump to the atmosphere through the secondary system relief valves

c. The resultant activity released to the containment is assumed available for leakage directly to the environment.

15.4.8.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

a. A 1-gpm steam generator primary-to-secondary leakage, which is significantly greater than that anticipated during normal operation, is assumed.

b. The coincident loss of offsite power with the occurrence of a RCCA ejection accident is a highly conservative assumption. In the event of the availability of offsite station power, the condenser steam dump valves will open, permitting steam dump to the condenser. Thus there is no direct release via that path to the environment.

c. It is assumed that none of the iodines released to the containment atmosphere are adsorbed (i.e. plate out) onto the internal surfaces of the containment or adheres to internal components. However, it is recognized that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses.

d. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressures associated with a RCCA ejection accident are considerably lower than that calculated for a LOCA. The pressure inside the containment also decreases considerably with time, with an expected decrease in leakage rates. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. 10).

e. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.4.8.3.3 Conclusions

15.4.8.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the RCCA ejection accident is the control room filtration system. Activity loadings on the control

room charcoal filter are based on the flow rate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

The activity in the control room filter as a function of time has been evaluated for the loss-of-coolant accident, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis loss-of-coolant accident fission product iodine loadings, more than adequate design margin is available with respect to postulated RCCA ejection accident releases.

15.4.8.3.3.2 Doses to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

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

The TEDE doses have been analyzed for the limiting 2 hour dose at the exclusion area boundary, for the duration of the accident at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.4-4. The resultant doses are well within the guideline values of 10 CFR 50.67 for offsite locations and the full 10 CFR 50.67 guideline value in the control room.

15.4.8.4 Conclusions

Even on a conservative basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10 percent.

15.4.9 REFERENCES

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TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	10.43
	Peak nuclear power occurs	10.57
	Rods begin to fall into core	10.93
	Peak heat flux occurs	12.73
	Minimum DNBR occurs	12.73
	Peak average clad temperature occurs	13.06
	Peak average fuel temperature occurs	13.26
	Peak fuel centerline temperature occurs	13.71
Uncontrolled RCCA bank withdrawal at Power (Limiting DNB Case)	Initiation of uncontrolled RCCA withdrawal at 12 pcm/sec, 10% power, BOL	0.0
	Overtemperature ΔT reactor trip setpoint reached	57.03
	Rods begin to drop	59.28
	Minimum DNBR occurs	59.90
Uncontrolled RCCA Bank withdrawal at Power (Rapid Reactivity Addition DNB Case)	Initiation of uncontrolled RCCA withdrawal at 110 pcm/sec, 100% power, BOL	0.0
	High neutron flux reactor trip setpoint reached	1.25
	Rods begin to drop	1.75
	Minimum DNBR occurs	3.05

TABLE 15.4-1 (Sheet 2)

Accident	Event	Time (sec)
Uncontrolled RCCA Bank withdrawal at Power (Slow Reactivity Addition DNB Case)	Initiation of uncontrolled RCCA withdrawal at 1 pcm/sec, 100% power, BOL	0.0
	Overtemperature ΔT reactor trip setpoint reached	99.3
	Rods begin to drop	101.6
	Minimum DNBR occurs	102.0
Uncontrolled RCCA Bank withdrawal at Power (Limiting RCS Pressure Case)	Initiation of uncontrolled RCCA withdrawal at 21 pcm/sec, 74% power, BOL	0.0
	Neutron flux high positive rate reactor trip setpoint reached	12.54
	Rods begin to drop	13.54
	Maximum RCS pressure	15.50
Startup of inactive reactor coolant loop at an incorrect temperature	Initiation of pump startup	1.0
	Power reaches P-8 trip setpoint	12.8
	Rods begin to drop	12.9
	Minimum DNBR occurs	12.9
Rod cluster control assembly ejection accident		
1. Beginning-of-life, full power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.22

TABLE 15.4-1 (Sheet 3)

Accident	Event	Time (sec)
	Peak clad temperature occurs	2.28
	Peak heat flux occurs	2.30
	Peak fuel centerline temperature	4.00
2. End-of-life, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.19
	Peak nuclear power occurs	0.22
	Rods begin to fall into core	0.69
	Peak clad temperature occurs	1.56
	Peak heat flux occurs	1.57
	Peak fuel average temperature occurs	1.82
	Peak centerline fuel temperature	2.87

TABLE 15.4-2

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP (Beginning)	HFP (Beginning)	HZP (End)	HFP (End)
Power level, %	0	100*	0	100*
Initial average coolant temperature, F	557.0	594.9	557.0	594.9
Ejected rod worth, %Dk	0.78	0.23	0.86	0.25
Delayed neutron fraction, %	0.49	0.49	0.44	0.44
Doppler weighting factor	2.309	1.433	3.078	1.499
Trip reactivity, %Dk	2.0	4.0	2.0	4.0
F_Q before rod ejection		2.50		2.50
F_Q after rod ejection	13.0	6.6	21.0	7.1
Number of operational pumps	2	4	2	4
Fuel Melt	0.0	4.62	0.0	3.96
Maximum fuel center temperature, F	3867	>4900	3869	>4800
Maximum clad average temperature, F	2498	2270	2626	2191
Maximum fuel stored energy, Btu/lbm	254.7	317.6	261.4	305.4

* The full power analyses were done at a core power of 3637 MWt. See Section 15.0.3.1 and Table 15.0-1 for additional information.

WOLF CREEK

TABLE 15.4-3

PARAMETERS USED IN EVALUATING THE
RCCA EJECTION ACCIDENT

I. Source Data

a.	Core power level, MWT	3637 (includes uncertainty)
b.	Core inventories	Table 15A-3
c.	Steam generator tube leakage, gpm	1
d.	Radial peaking factor	1.65
e.	Extent of core damage	10 percent of fuel rods experience cladding failure; 0.25 percent of fuel experiences melting
f.	Activity released to reactor coolant, percent of core activity	
	1. Cladding failure	
	(a) Noble gas gap activity	100
	(b) Iodine gap activity	10
	(c) Alkali metal gap activity	12
	2. Fuel melting	
	(a) Noble gas fuel activity	100
	(b) Iodine fuel activity	50
	(c) Alkali metal fuel activity	100
g.	Iodine carryover factor for steam generators	0.1
h.	Alkali metal partition factor for steam generators (based on full power moisture carryover of 0.25%)	0.0025
i.	Reactor coolant mass, lbs	3.99E+5
j.	Each steam generator water mass, lb	
	0-2 hrs	82,500
	>2 hrs (assumes water level reaches 0% narrow range span)	121,250

TABLE 15.4-3 (Sheet 2)

II.	Atmospheric Dispersion Factors	Table 15A-2
III.	Activity Release Data	
	a. Containment volume, ft^3	2.5E + 6
	b. Containment leak rate, volume percent/day	
	1. 0-24 hours	0.20
	2. 1-30 days	0.10
	c. Percent of containment leakage that is unfiltered	100
	d. Offsite power	Lost
	e. Steam release from all steam generators	
	0-2 hr, lb	419,846
	2-12 hr,lb	1,352,918
	f. Duration of release from relief valves, hr	12
IV.	Control Room	
	a. Control Room Switchover to Emergency Mode, sec	
	i. Containment Leakage	See Table 15A-5
	ii. Secondary Releases	See Table 15A-5

TABLE 15.4-4

RADIOLOGICAL CONSEQUENCES OF A
ROD-EJECTION ACCIDENT

	Wolf Creek Dose (rem TEDE)
Containment Leakage	
Exclusion Area Boundary (limiting 2 hour duration)	1.1E+0
Low Population Zone Outer Boundary (duration)	1.9E+0
Control Room (30 days)	2.6E+0
Secondary Releases	
Exclusion Area Boundary (limiting 2 hour duration)	3.8E-1
Low Population Zone Outer Boundary (duration)	3.2E-1
Control Room (30 days)	3.5E+0

TABLE 15.4-5

Flux Map Review Criteria for a 4-Loop Core

Number of Available Detector Locations	Measured vs. Predicted Detector Reaction Rate Comparison*	Symmetric Thimble Reaction Rate Comparison**
55 to 58	10%	7%
49 to 54	8%	5%
44 to 48	6%	5%

*The review criterion is the table value (%) or an absolute normalized reaction rate difference equal to the table value divided by 100 % (e.g., 10 % / 100 % = 0.1), whichever is greater.

**Applicable to symmetric thimbles with normalized reaction rates above 0.7. For even lattice cores (i.e., 14x14, 16x16), thimbles in symmetric assemblies may be used for the comparison. The review criterion is relative to the expected reaction rate difference.

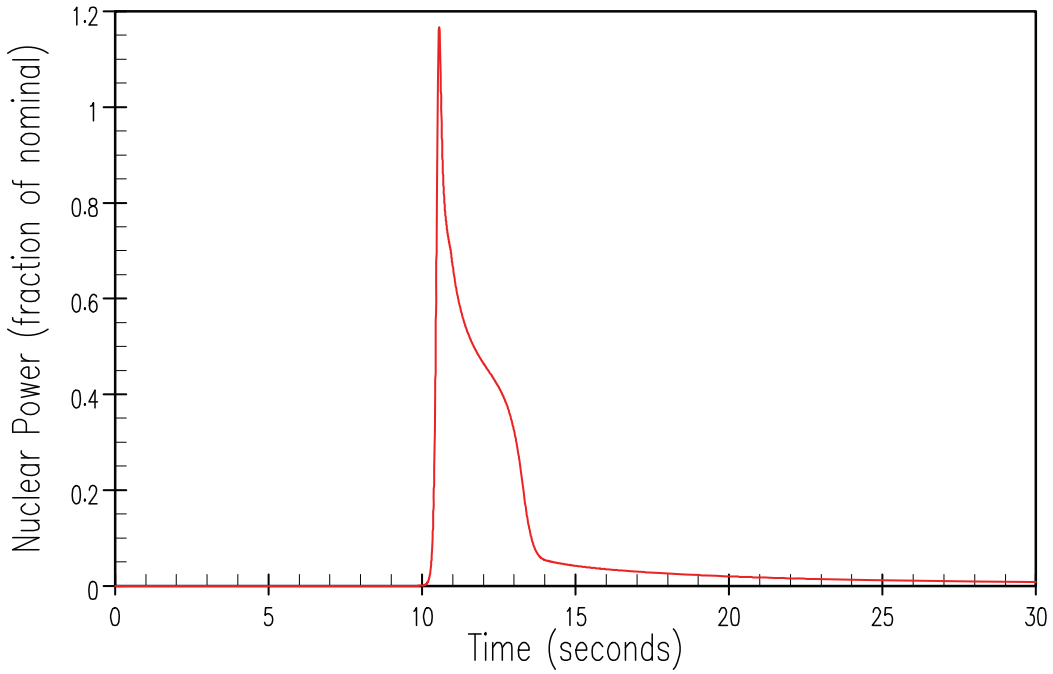

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FIGURE 15.4-1

Nuclear Power Transient for an Uncontrolled Rod Withdrawal from a Subcritical Condition

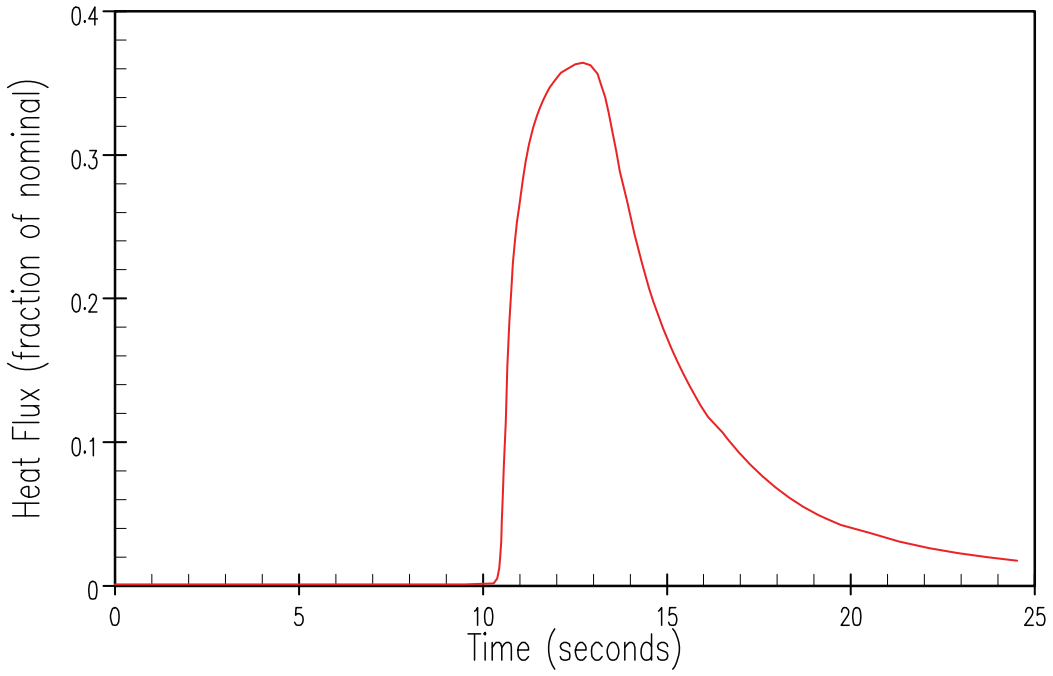

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FIGURE 15.4-2

Heat Flux Transient for an Uncontrolled Rod Withdrawal from a Subcritical Condition

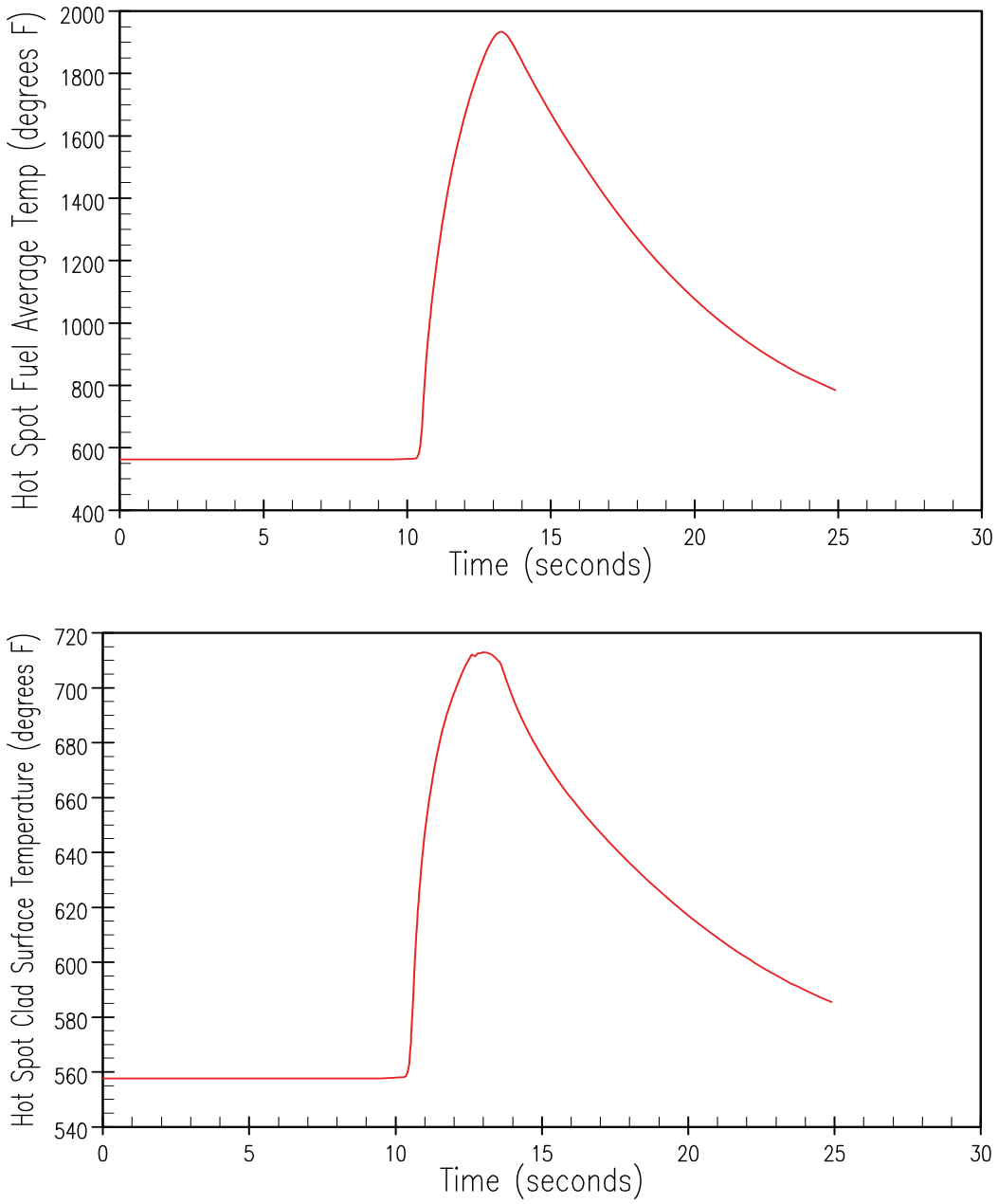

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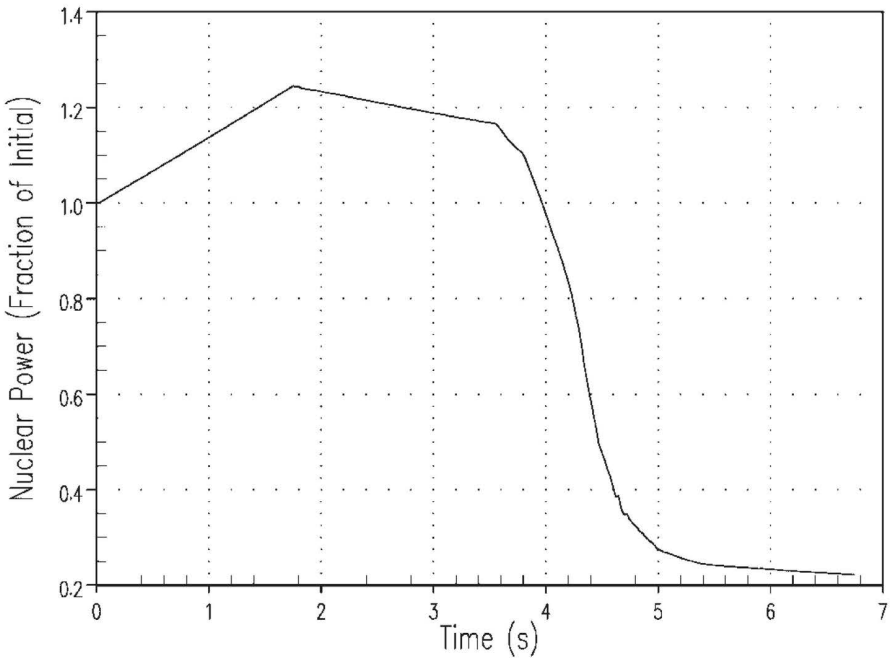
FIGURE 15.4-3

Fuel Average and Clad Surface Temperatures at the Hot Spot for an Uncontrolled Rod Withdrawal from a Subcritical Condition

FIGURE 15.4-4 HAS BEEN DELETED

FIGURE 15.4-4
UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER WITH
MINIMUM REACTIVITY FEEDBACK
(5 pcm/sec WITHDRAWAL RATE)



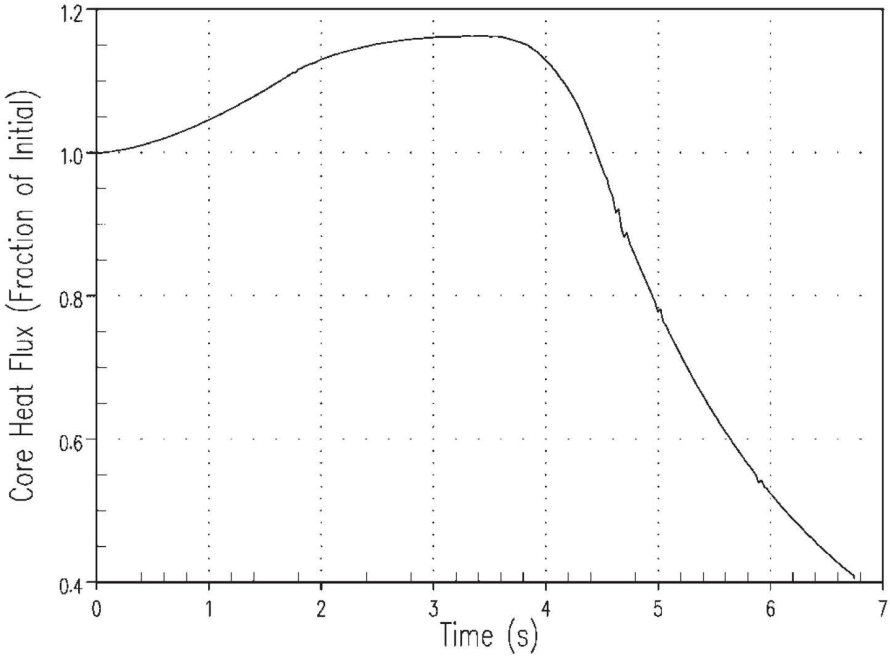


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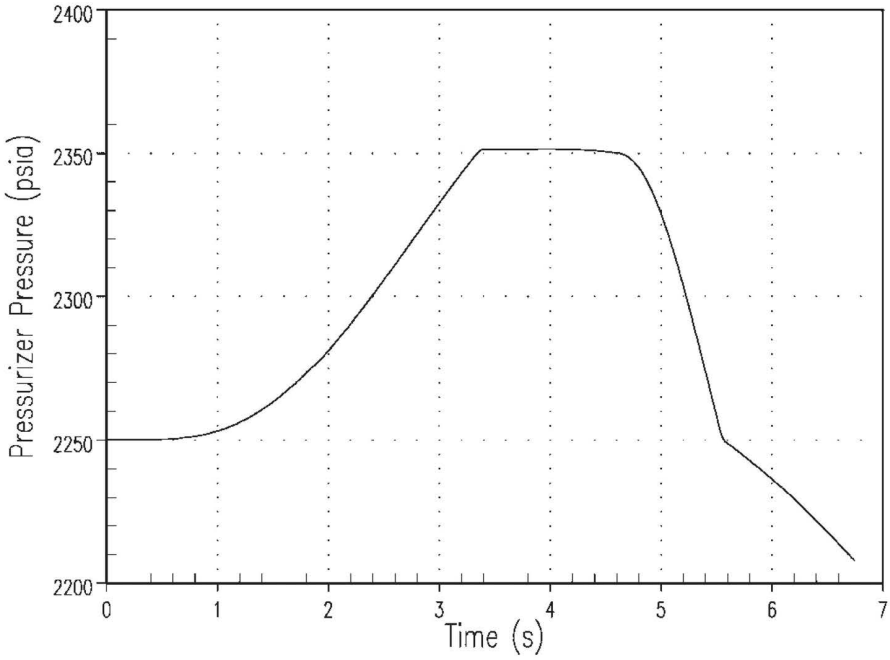
Figure 15.4-4a

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 110 pcm/sec Reactivity Insertion Rate)

Nuclear Power and Core Heat Flux



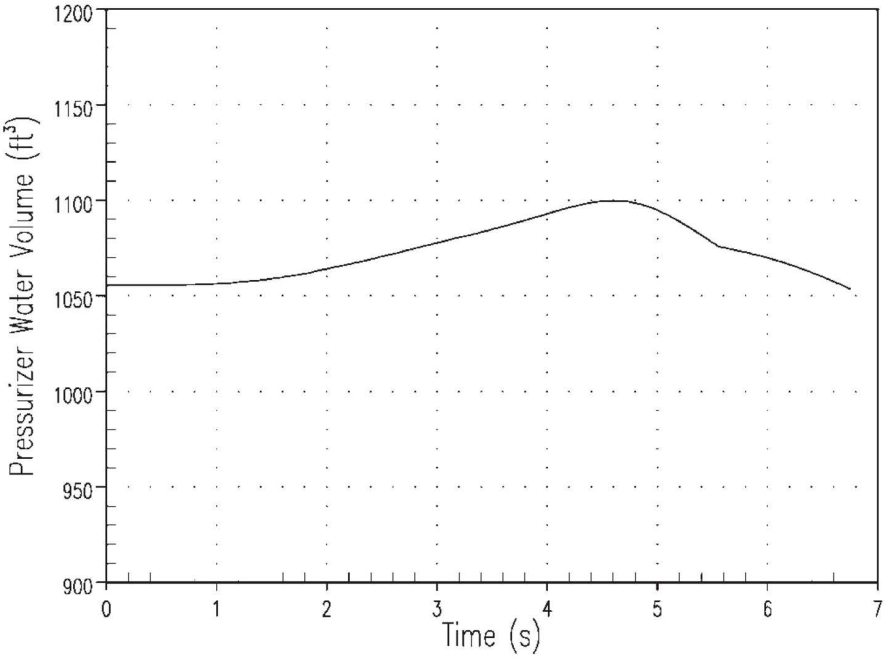


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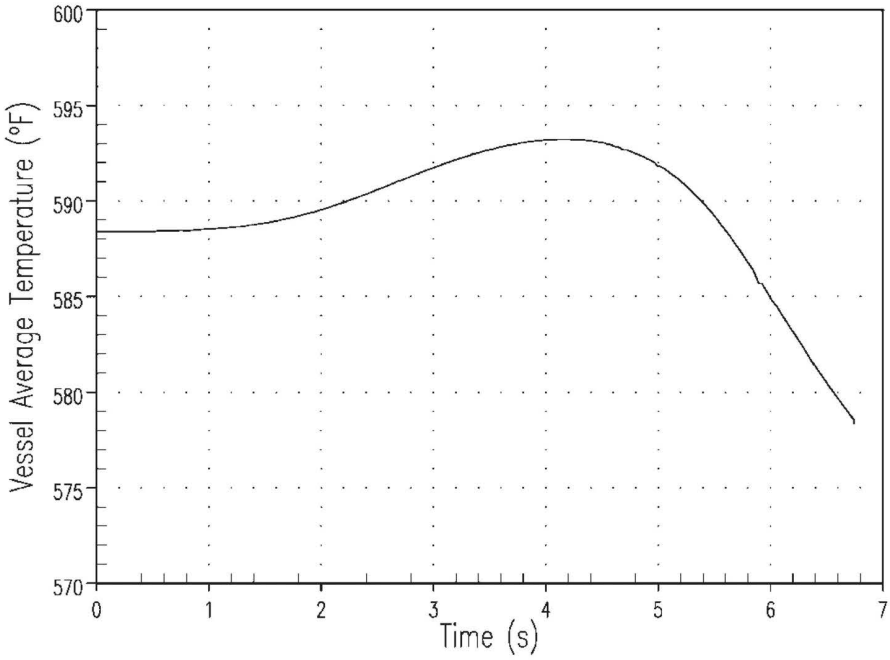
Figure 15.4-4b

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 110 pcm/sec Reactivity Insertion Rate)

Pressurizer Pressure and Water Volume



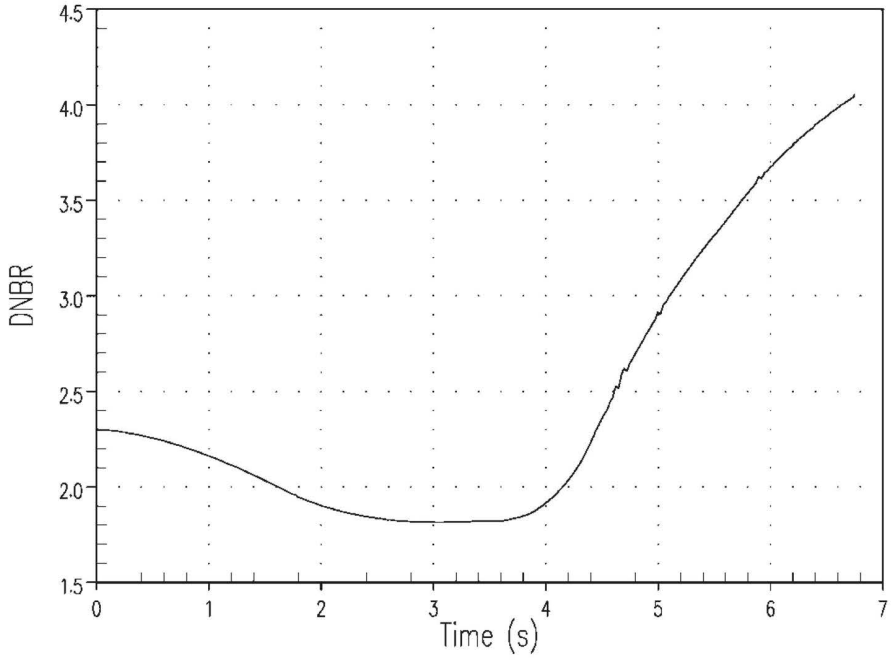


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Figure 15.4-4c

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 110 pcm/sec Reactivity Insertion Rate)

Vessel Average Temperature and DNBR

FIGURE 15.4-4d HAS BEEN DELETED

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FIGURE 15.4-4d
FLUX RATIO
UNCONTROLLED RCCA BANK
WITHDRAWAL AT POWER
(DNB LIMITING CASE)

FIGURE 15.4-4e HAS BEEN DELETED

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FIGURE 15.4-4e
CORE EXIT ENTHALPY
UNCONTROLLED RCCA BANK
WITHDRAWAL AT POWER
(DNB LIMITING CASE)

FIGURE 15.4-4f HAS BEEN DELETED

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FIGURE 15.4-4f
CORE INLET FLOW
UNCONTROLLED RCCA BANK
WITHDRAWAL AT POWER
(DNB LIMITING CASE)

FIGURE 15.4-4g HAS BEEN DELETED

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FIGURE 15.4-4g
MINIMUM DNBR
UNCONTROLLED RCCA BANK
WITHDRAWAL AT POWER
(DNB LIMITING CASE)

FIGURE 15.4-5 HAS BEEN DELETED

FIGURE 15.4-5
UNCONTROLLED RCCA BANK
WITHDRAWAL FROM FULL POWER WITH
MINIMUM REACTIVITY FEEDBACK
(5 pcm/sec WITHDRAWAL RATE)

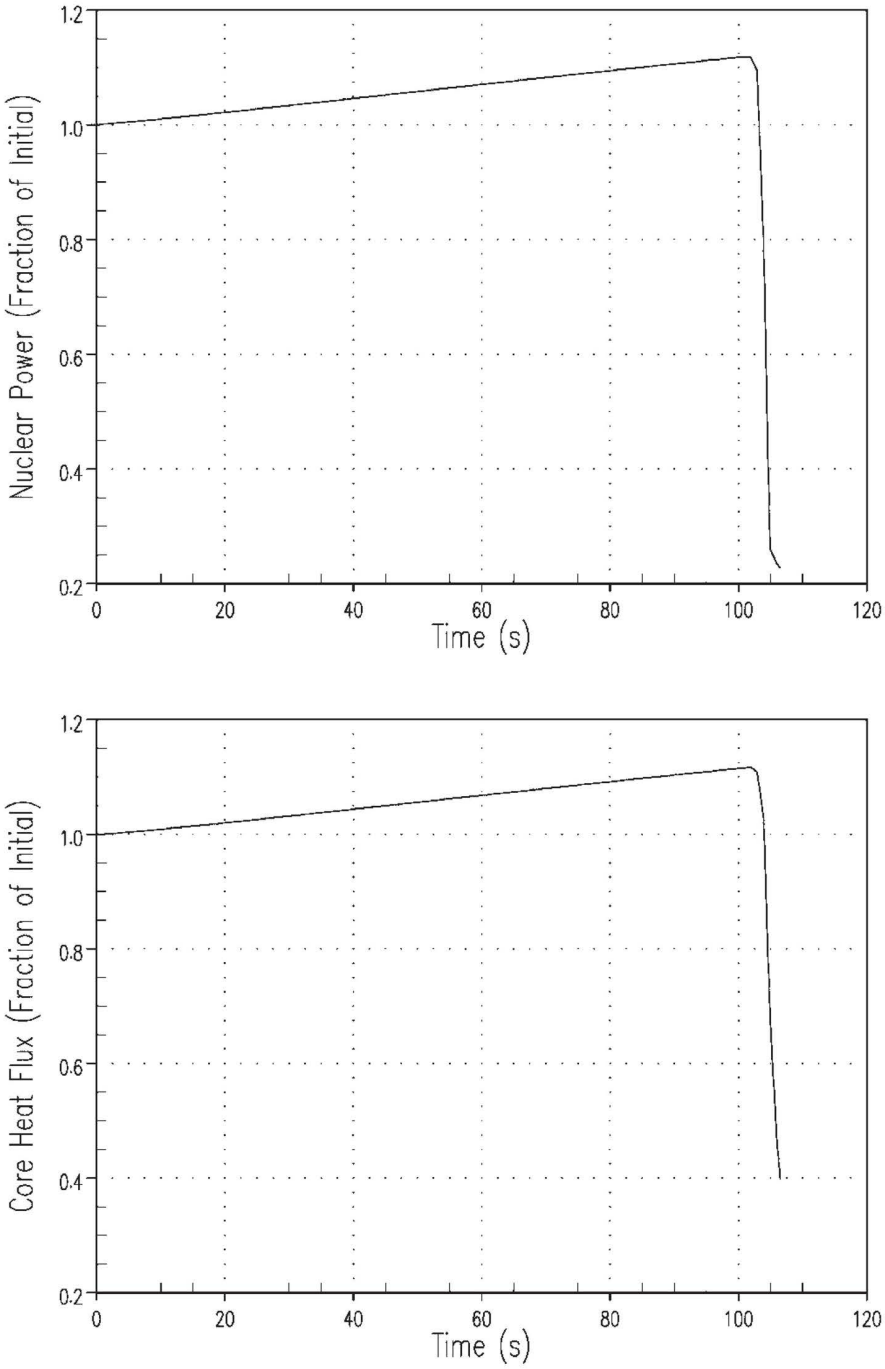

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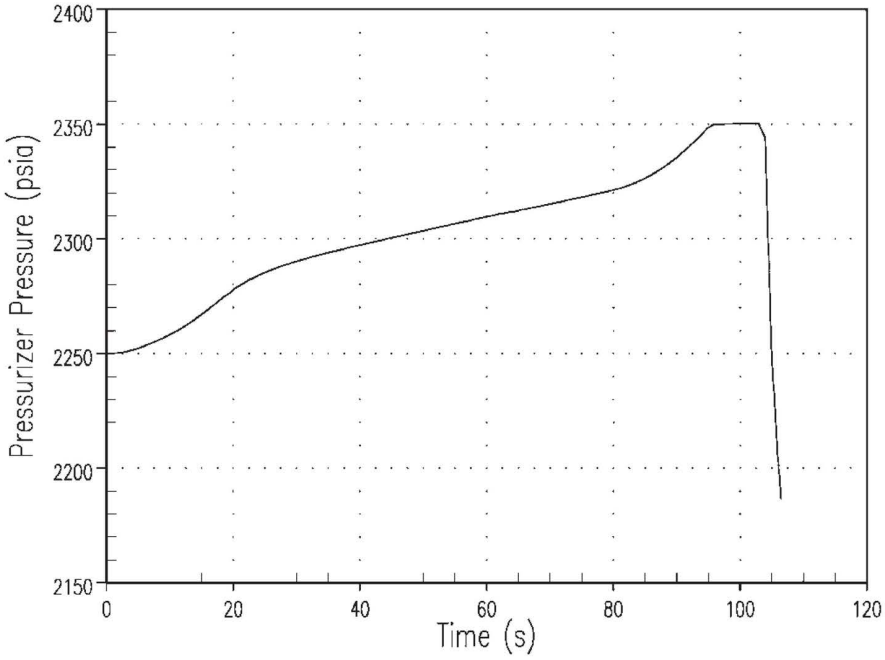
Figure 15.4-5a

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 1 pcm/sec Reactivity Insertion Rate)

Nuclear Power and Core Heat Flux



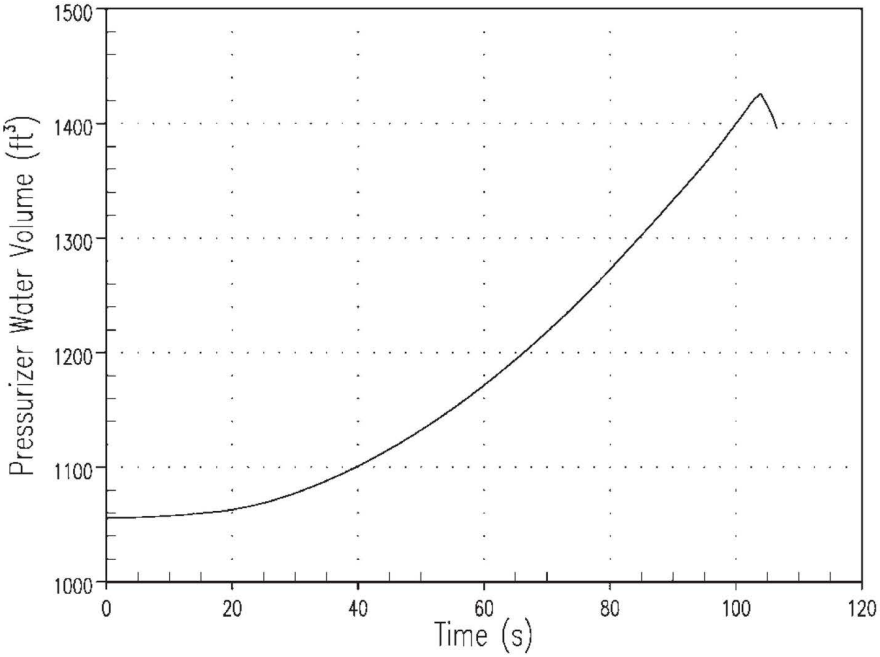


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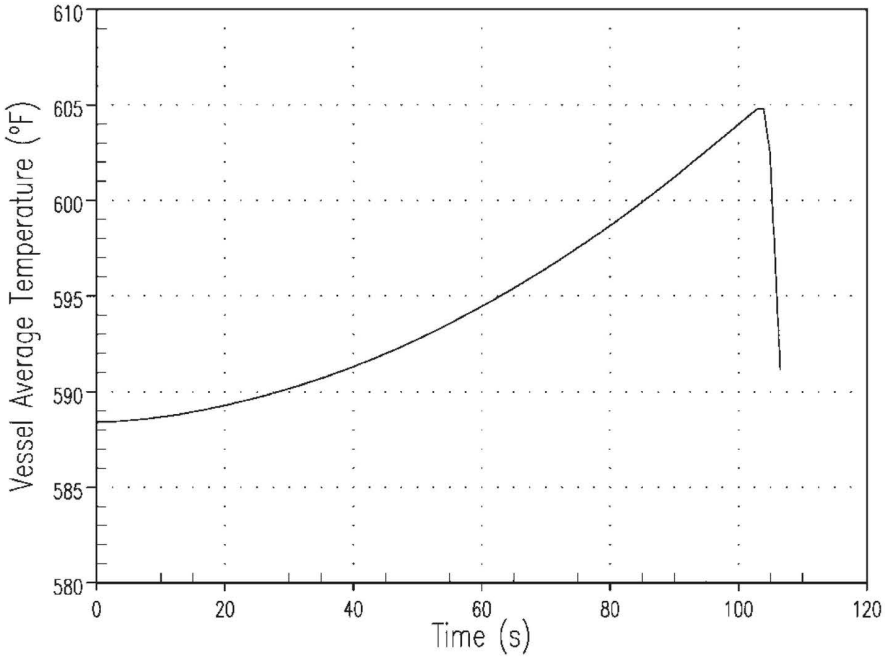
Figure 15.4-5b

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 1 pcm/sec Reactivity Insertion Rate)

Pressurizer Pressure and Water Volume



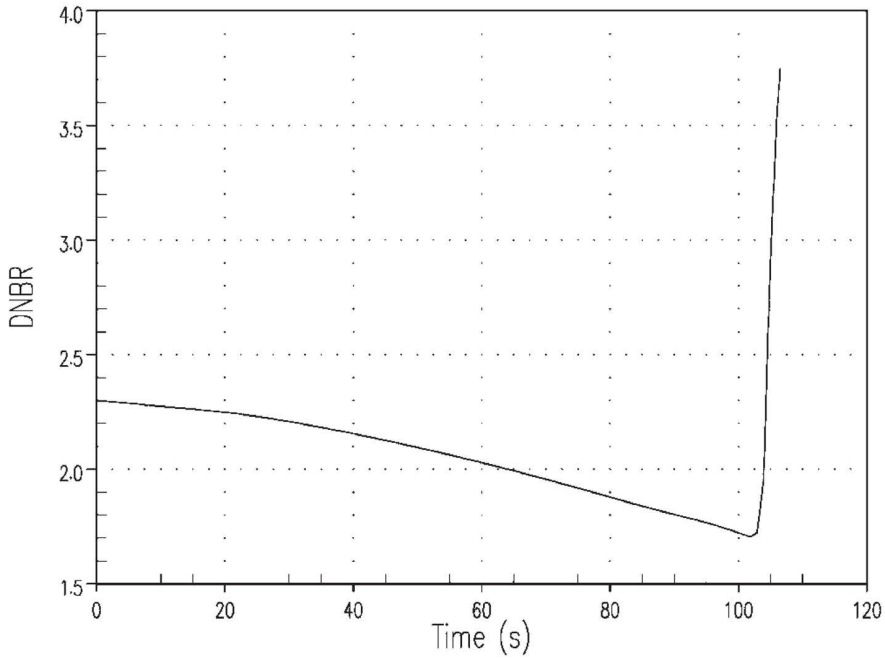


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Figure 15.4-5c

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum Reactivity Feedback, 1 pcm/sec Reactivity Insertion Rate)

Vessel Average Temperature and DNBR

FIGURE 15.4-6 HAS BEEN DELETED

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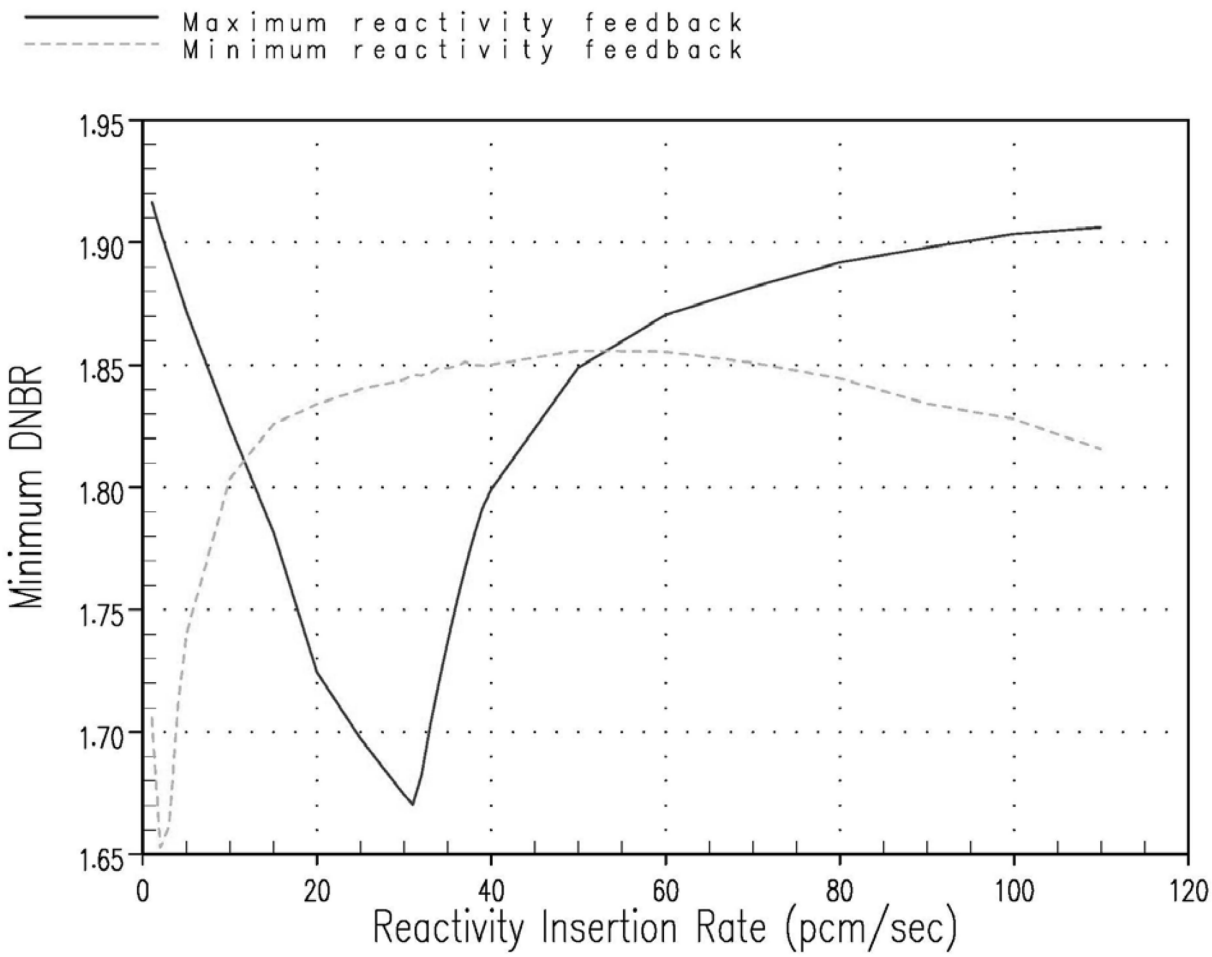
FIGURE 15.4-6
WITHDRAWAL FROM FULL POWER WITH BEGINNING-OF-LIFE REACTIVITY
(5 pcm/sec WITHDRAWAL RATE)



Note: The minimum DNBR values presented were calculated using the RETRAN code and are representative of the trends for minimum DNBR and not indicative of final calculated DNBR values. The detailed thermal-hydraulic code VIPRE was used to confirm that the limiting case (10 percent power, minimum reactivity feedback, 12 pcm/sec reactivity insertion rate) satisfied the DNB design basis.

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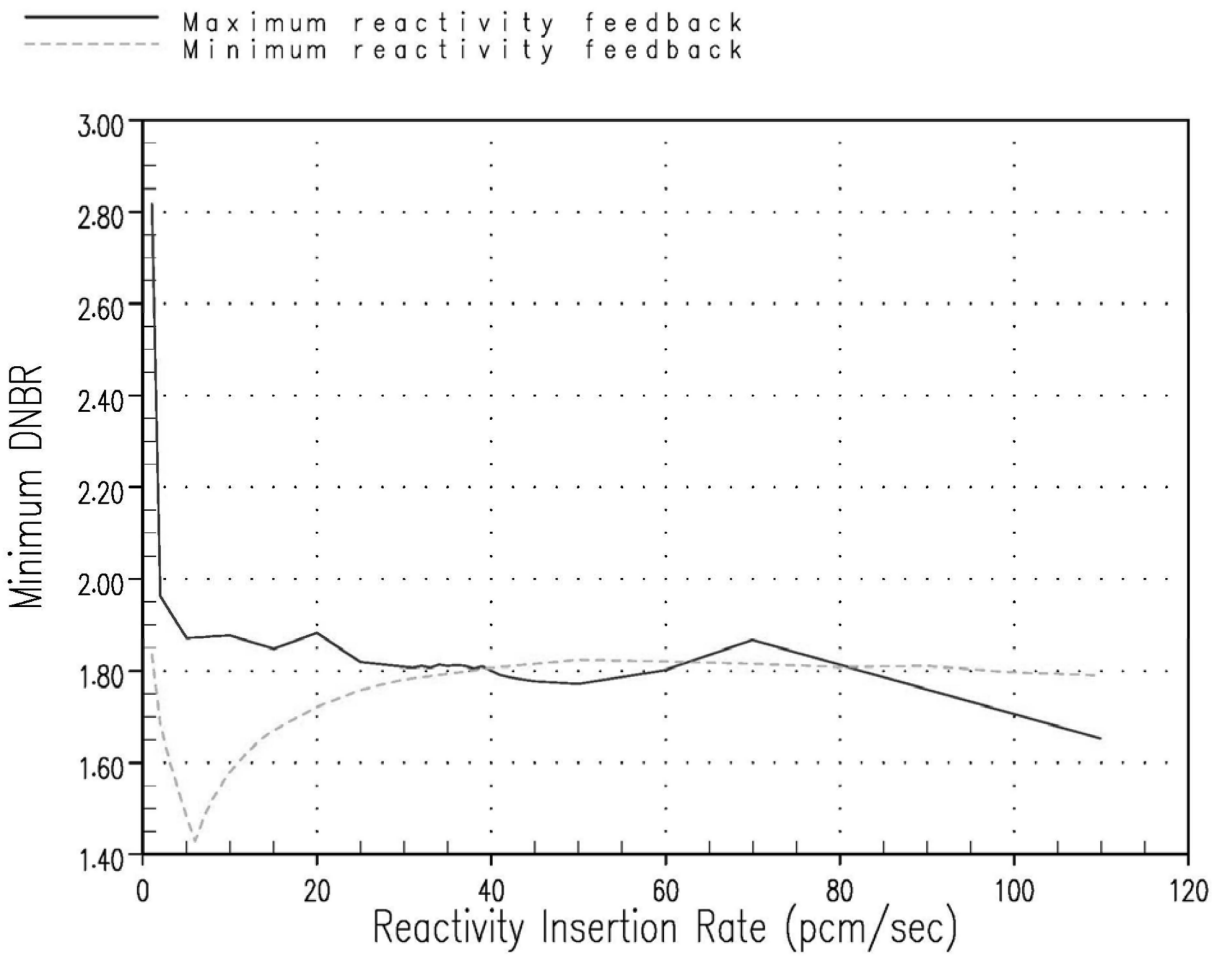
Figure 15.4-6a

Uncontrolled RCCA Bank Withdrawal at Power

(100% Power, Minimum and Maximum Reactivity Feedback)

Minimum DNBR vs. Reactivity Insertion Rate



Note: The minimum DNBR values presented were calculated using the RETRAN code and are representative of the trends for minimum DNBR and not indicative of final calculated DNBR values. The detailed thermal-hydraulic code VIPRE was used to confirm that the limiting case (10 percent power, minimum reactivity feedback, 12 pcm/sec reactivity insertion rate) satisfied the DNB design basis.

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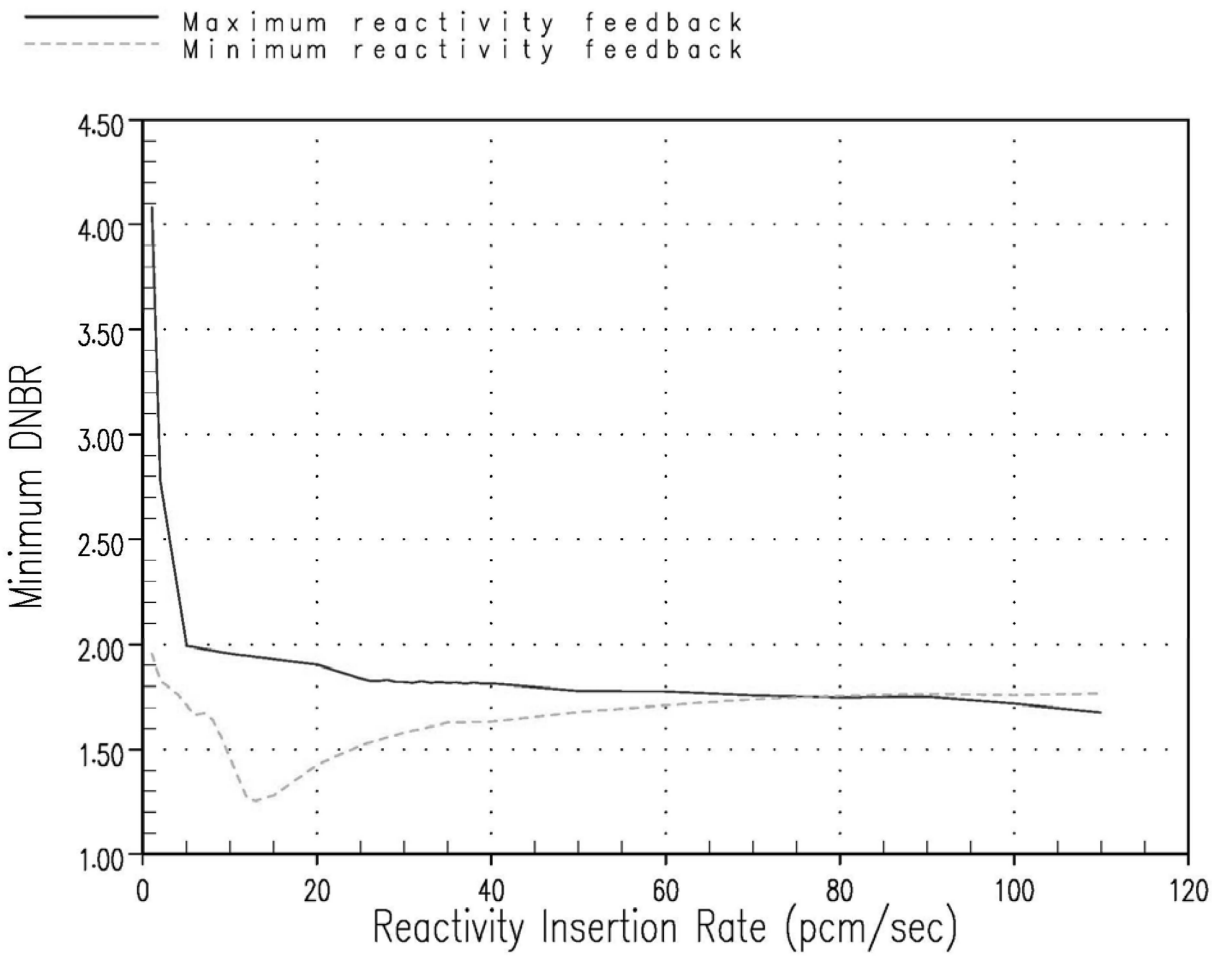
Figure 15.4-6b

Uncontrolled RCCA Bank Withdrawal at Power

(60% Power, Minimum and Maximum Reactivity Feedback)

Minimum DNBR vs. Reactivity Insertion Rate



Note: The minimum DNBR values presented were calculated using the RETRAN code and are representative of the trends for minimum DNBR and not indicative of final calculated DNBR values. The detailed thermal-hydraulic code VIPRE was used to confirm that the limiting case (10 percent power, minimum reactivity feedback, 12 pcm/sec reactivity insertion rate) satisfied the DNB design basis.

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Figure 15.4-6c

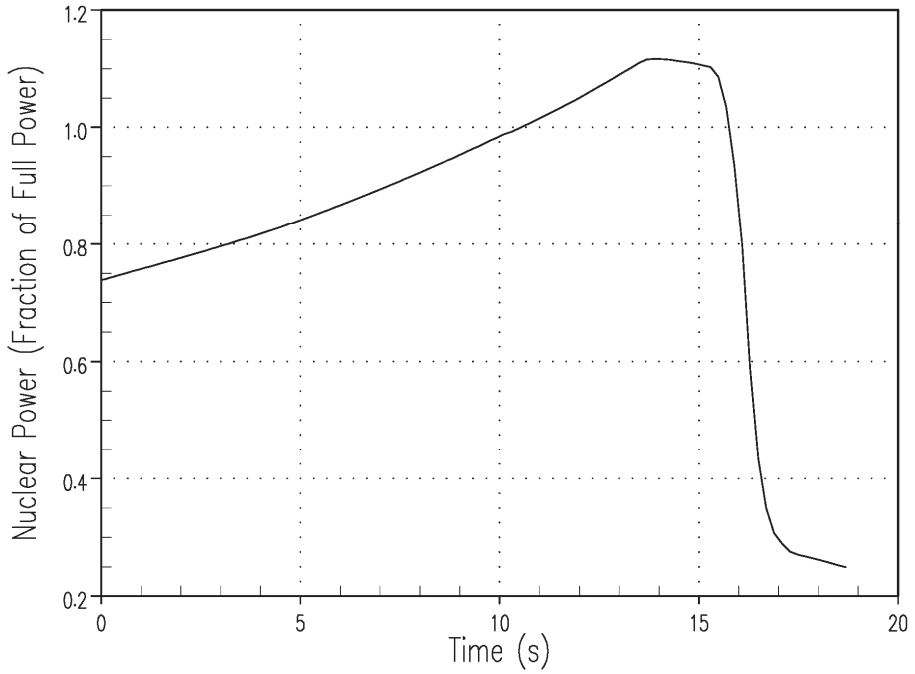
Uncontrolled RCCA Bank Withdrawal at Power

(10% Power, Minimum and Maximum Reactivity Feedback)

Minimum DNBR vs. Reactivity Insertion Rate

FIGURE 15.4-7 HAS BEEN DELETED

FIGURE 15.4-7
MINIMUM DNBR Vs. REACTIVITY
INSERTION RATE, ROD WITHDRAWAL
FROM 100% POWER



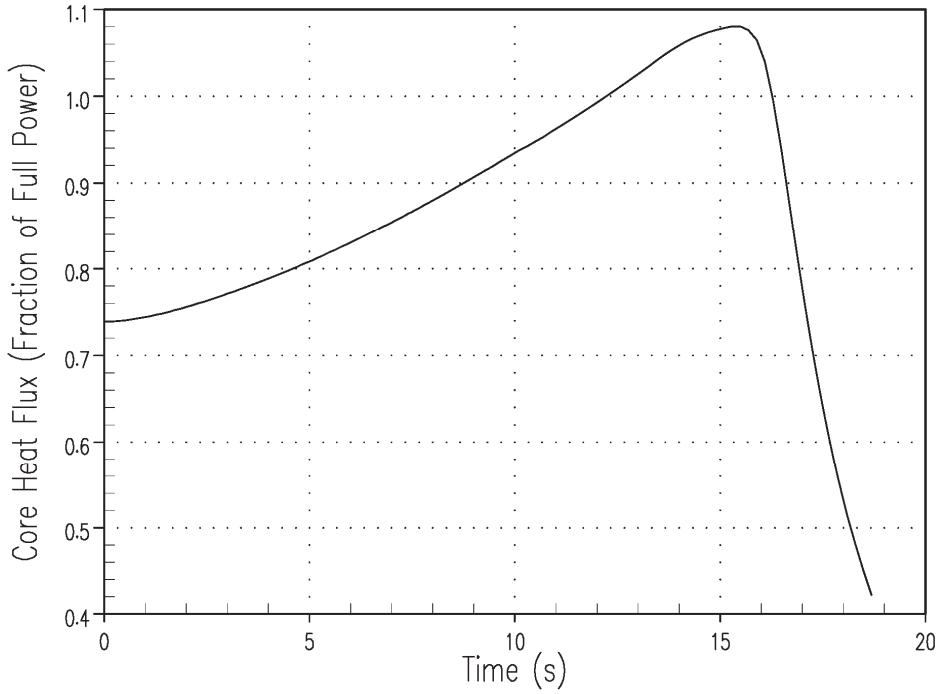


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Figure 15.4-7a

Uncontrolled RCCA Bank Withdrawal at Power

(RCS Overpressure Limiting Case)

Nuclear Power and Core Heat Flux

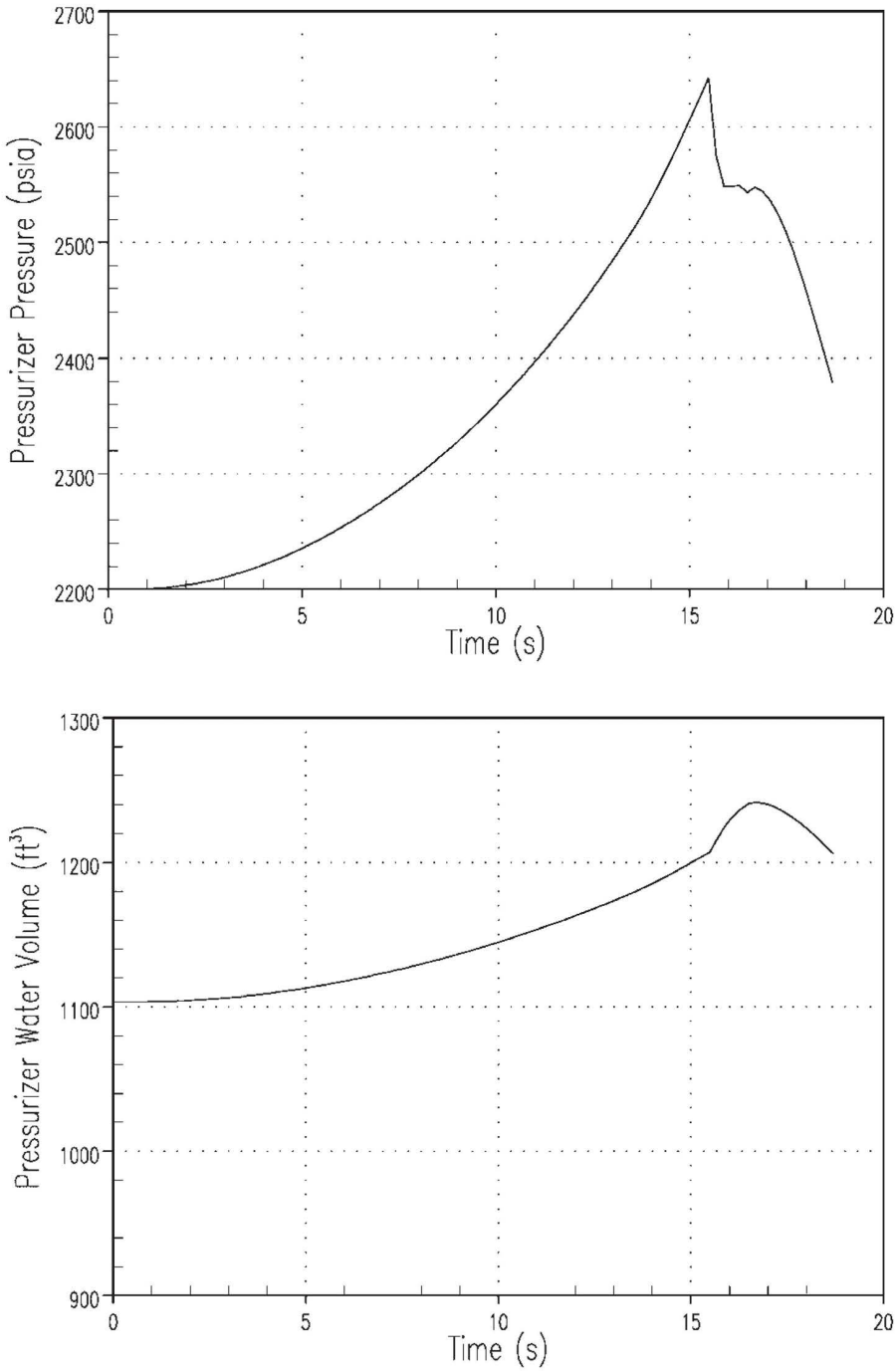

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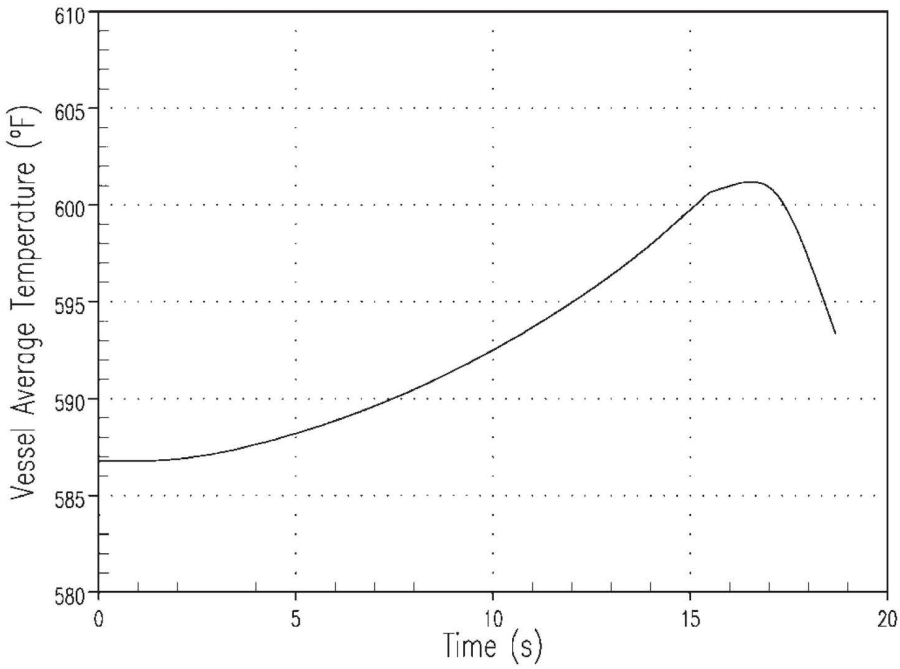
Figure 15.4-7b

Uncontrolled RCCA Bank Withdrawal at Power

(RCS Overpressure Limiting Case)

Pressurizer Pressure and Water Volume



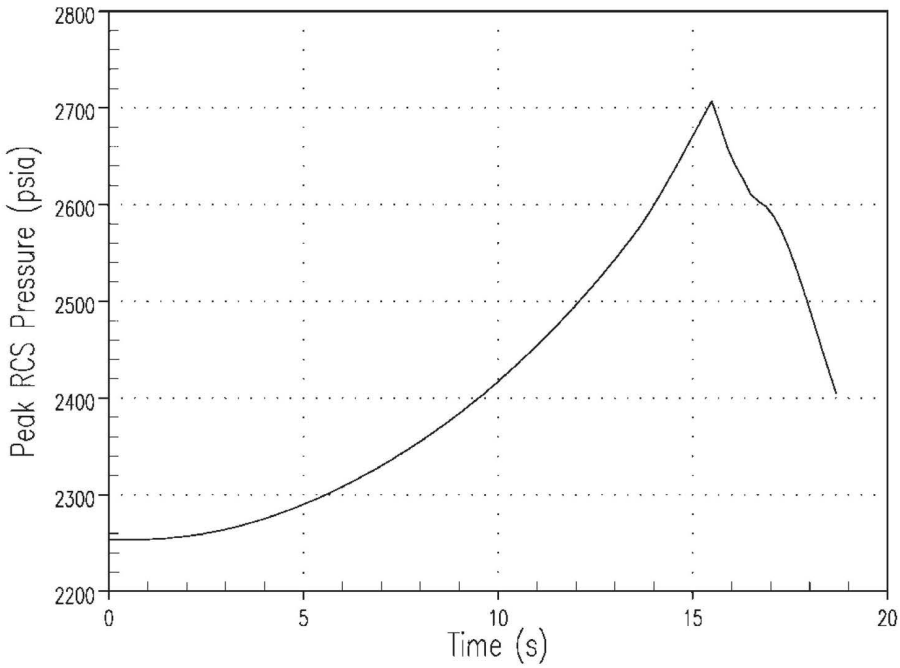


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Figure 15.4-7c

Uncontrolled RCCA Bank Withdrawal at Power

(RCS Overpressure Limiting Case)

Vessel Average Temperature and
Peak RCS Pressure

"This Figure has been deleted."

FIGURE 15.4-8

Minimum DNBR Vs. Reactivity Insertion Rate, Rod Withdrawal from 60% Power

"This Figure has been deleted."

FIGURE 15.4-9

Minimum DNBR Vs. Reactivity Insertion Rate, Rod Withdrawal from 10% Power

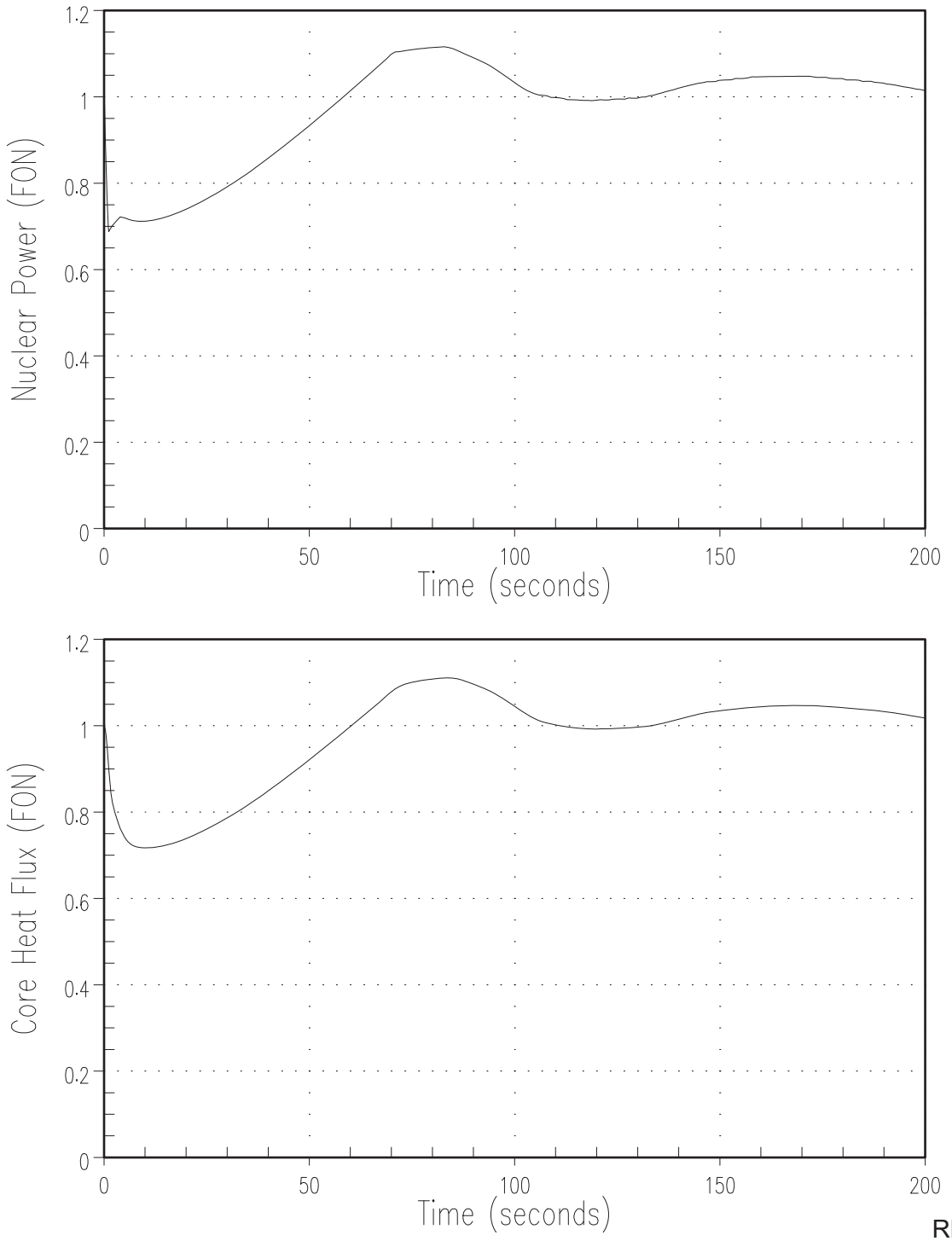

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Figure 15.4-10
Nuclear Power Transient and Core Heat Flux Transient For Dropped Rod Cluster Control Assembly

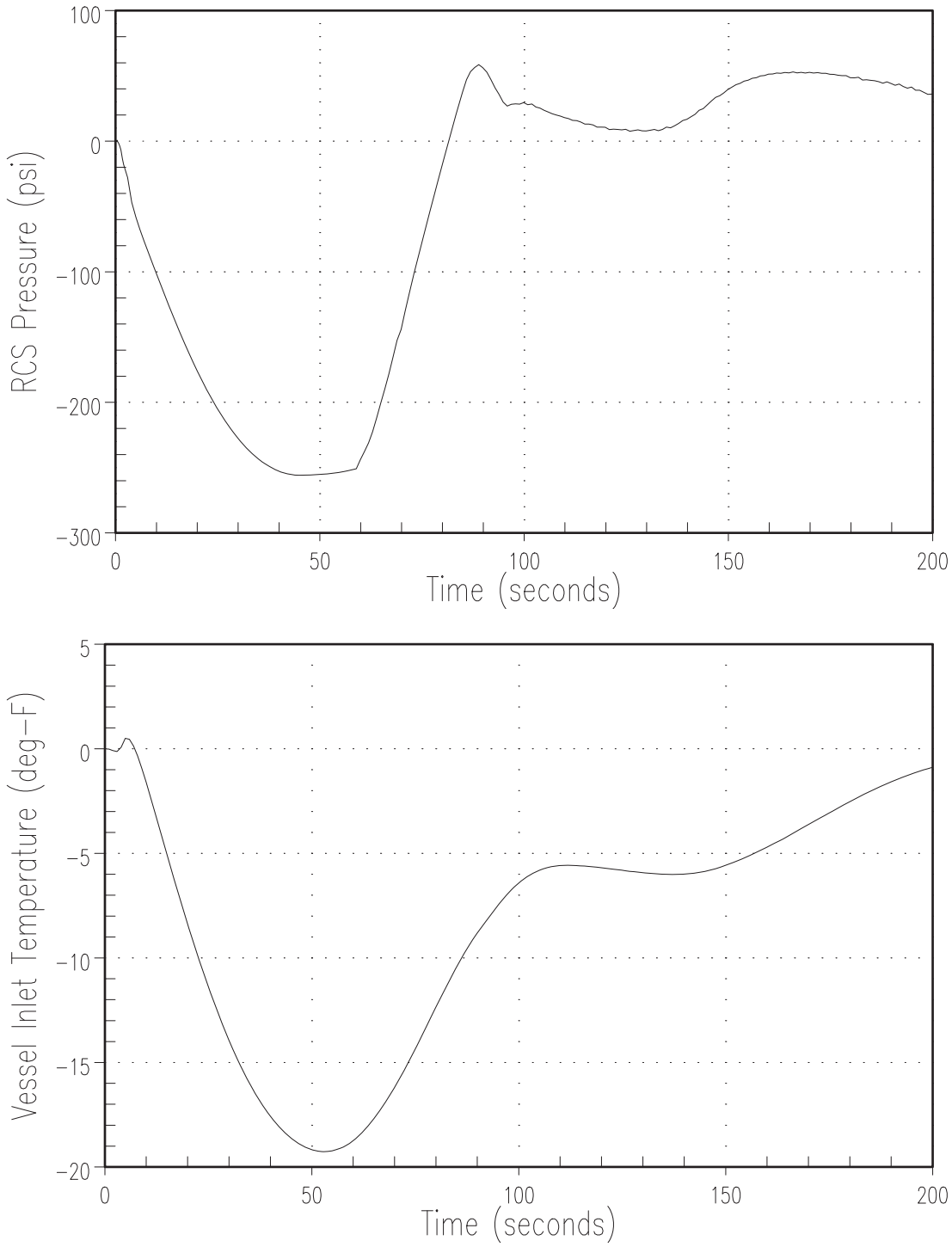

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Figure 15.4-11a
Change in Pressurizer Pressure and Core Average Temperature for Dropped Rod Cluster Control Assembly

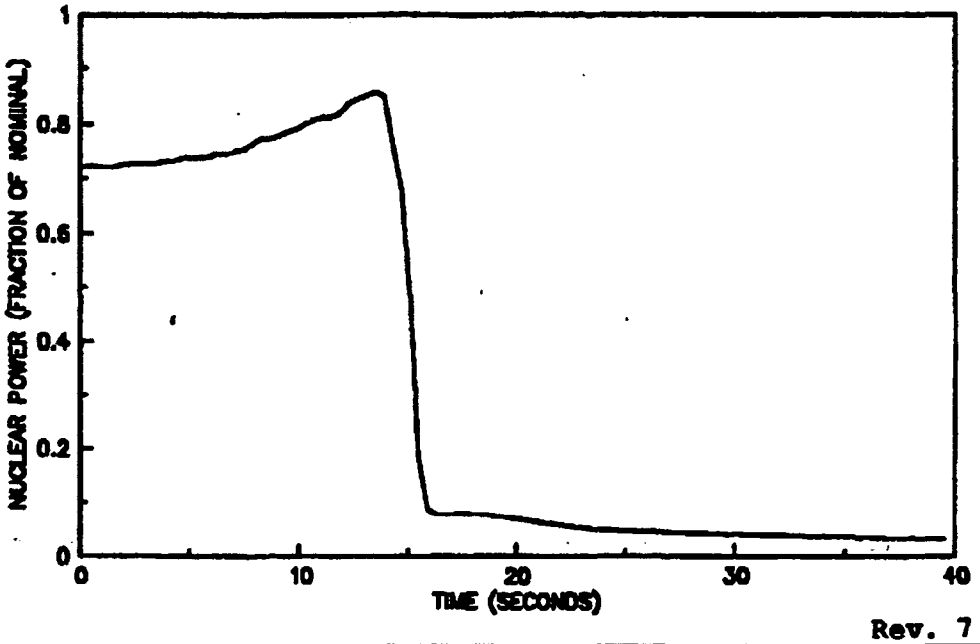

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FIGURE 15.4-12

NUCLEAR POWER TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

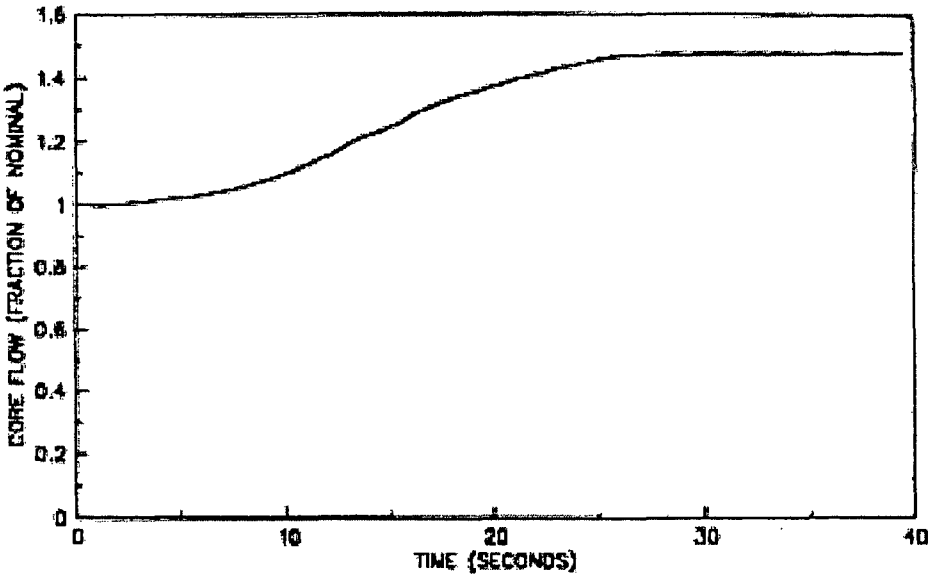

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FIGURE 15.4-13

CORE FLOW TRANSIENT FOR
STARTUP OF AN INACTIVE REACTOR
COOLANT LOOP

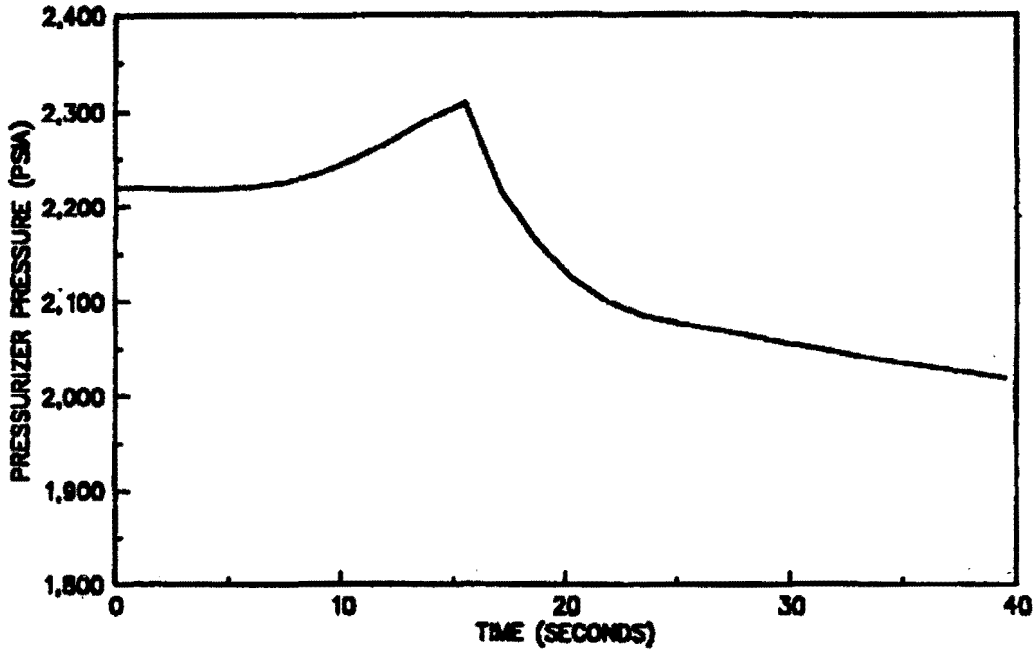

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FIGURE 15.4-14 PRESSURIZER PRESSURE TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

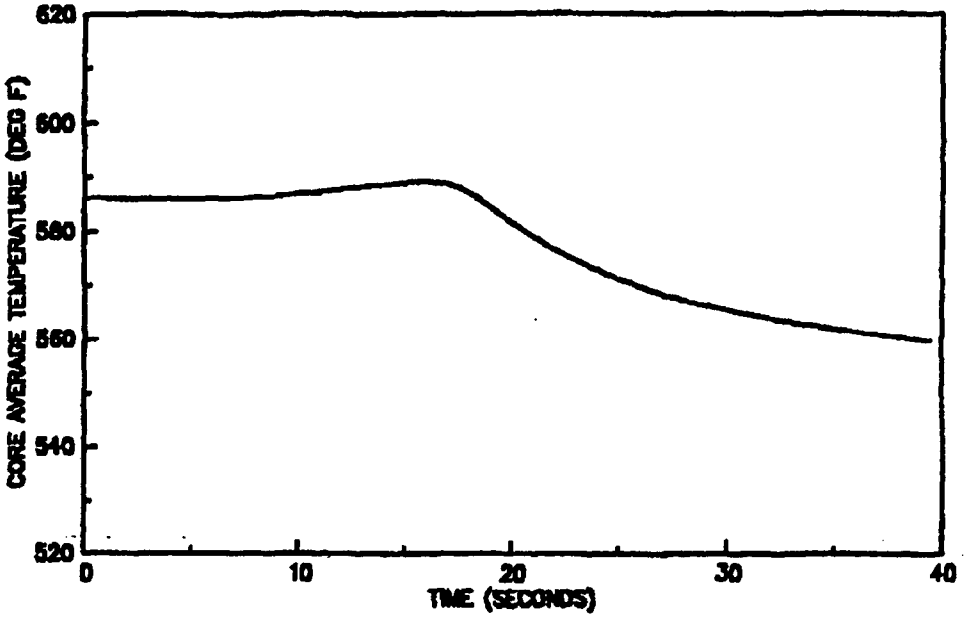

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FIGURE 15.4-15 CORE AVERAGE TEMPERATURE TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

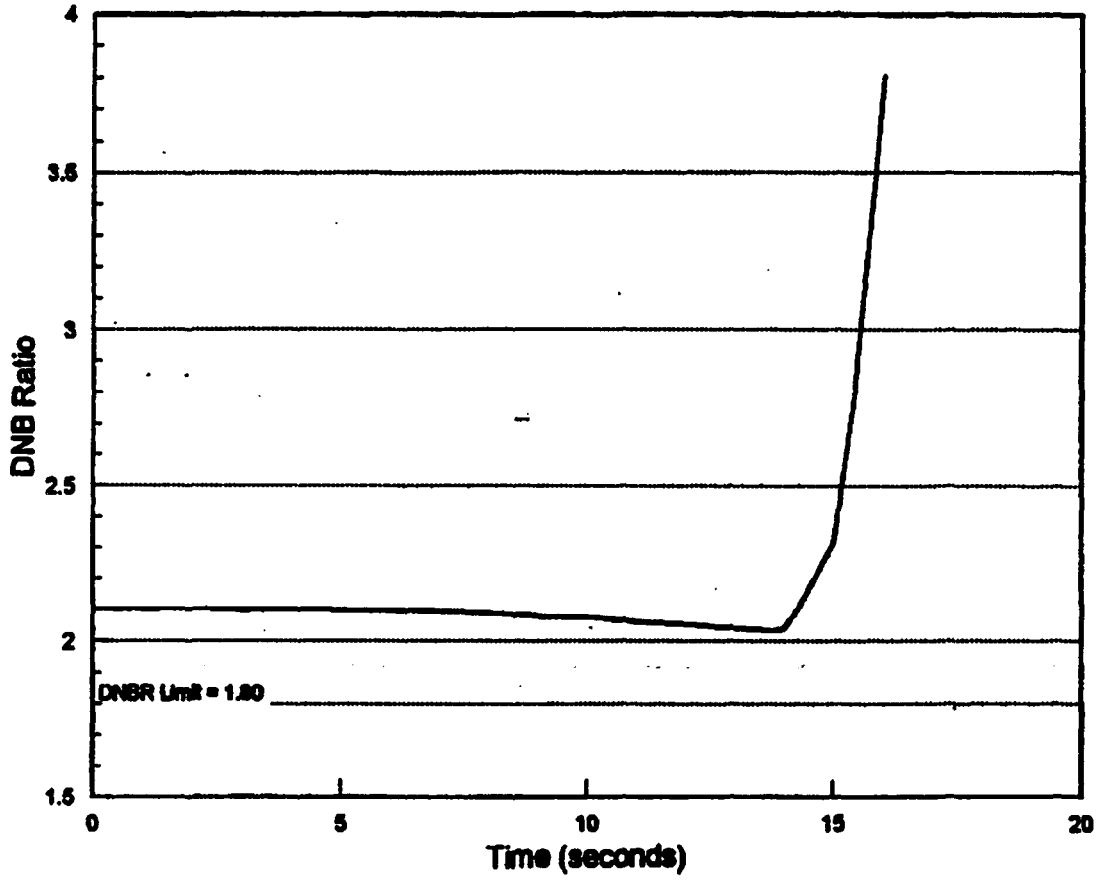

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FIGURE 15.4-16

DNBR TRANSIENT FOR STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

FIGURE 15.4-17 HAS BEEN DELETED

FIGURE 15.4-17
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 3 ASSEMBLY

FIGURE 15.4-18 HAS BEEN DELETED

FIGURE 15.4-18
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 2 ASSEMBLY WITH THE BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY

FIGURE 15.4-19 HAS BEEN DELETED

FIGURE 15.4-19
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN INTERCHANGE BETWEEN A REGION 1 AND A REGION 2 ASSEMBLY WITH THE BURNABLE POISON RODS BEING TRANSFERRED TO THE REGION 1 ASSEMBLY

FIGURE 15.4-20 HAS BEEN DELETED

FIGURE 15.4-20
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR AN ENRICHMENT ERROR (A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION)

FIGURE 15.4-21 HAS BEEN DELETED

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FIGURE 15.4-21
REPRESENTATIVE PERCENT CHANGE IN LOCAL ASSEMBLY AVERAGE POWER FOR LOADING A REGION 2 ASSEMBLY INTO A REGION 1 POSITION NEAR THE CORE PERIPHERY

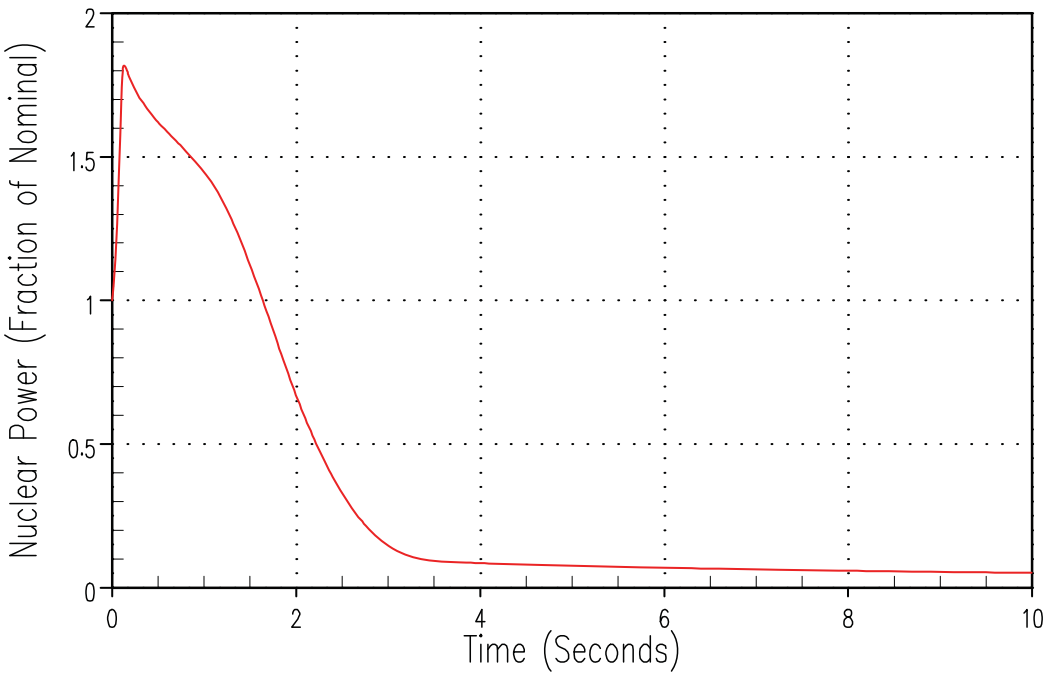

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FIGURE 15.4-22

Nuclear Power Transient, BOL,
HFP, Rod Ejection Accident

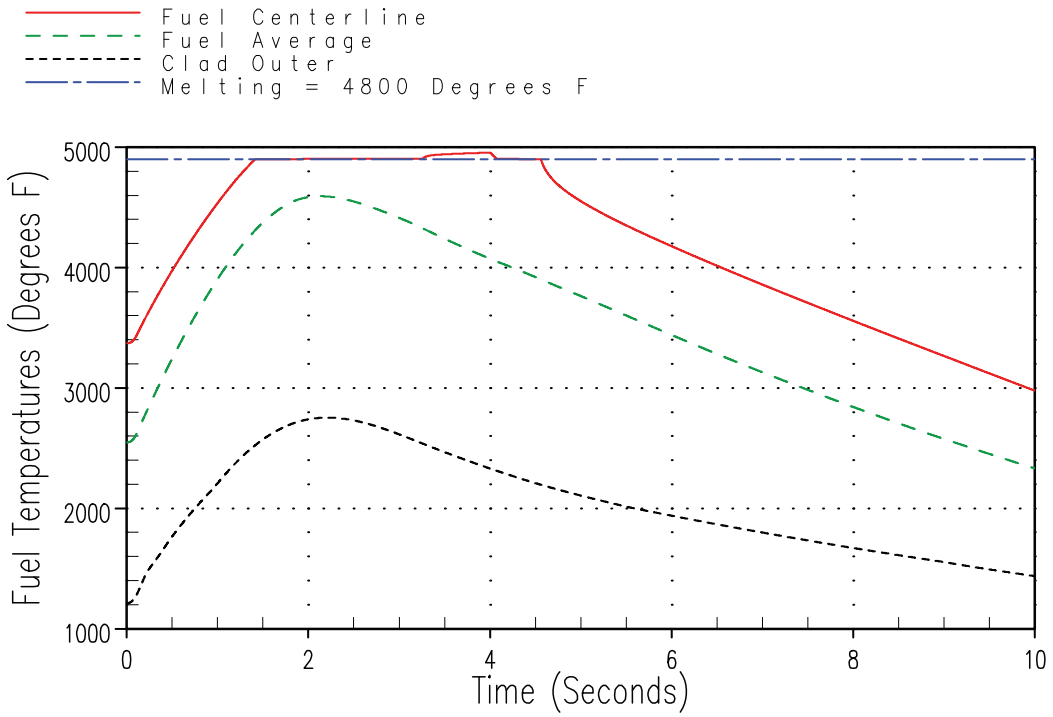

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FIGURE 15.4-23

Fuel Centerline, Fuel Average,
and Cladding Surface Temperature
Transients, BOL,HFP,
Rod Ejection Accident

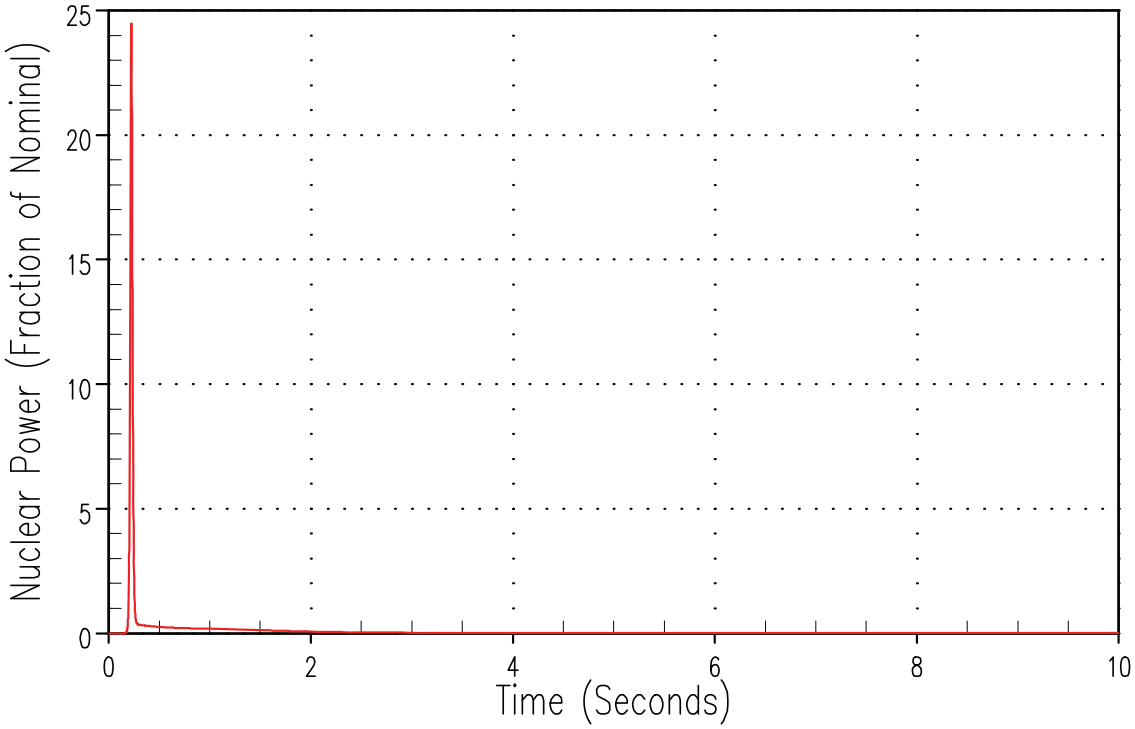

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FIGURE 15.4-24

Nuclear Power Transient, EOL,
HZP, Rod Ejection Accident

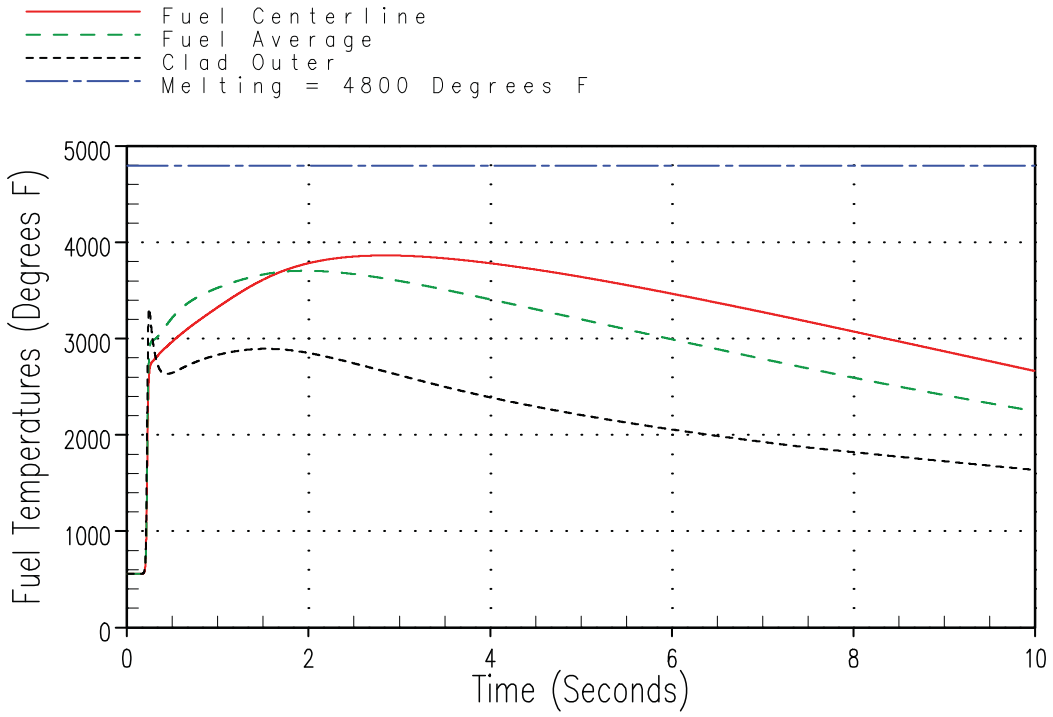

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FIGURE 15.4-25

Fuel Centerline, Fuel Average,
and Cladding Surface Temperature
Transients, EOL, HZP,
Rod Ejection Accident

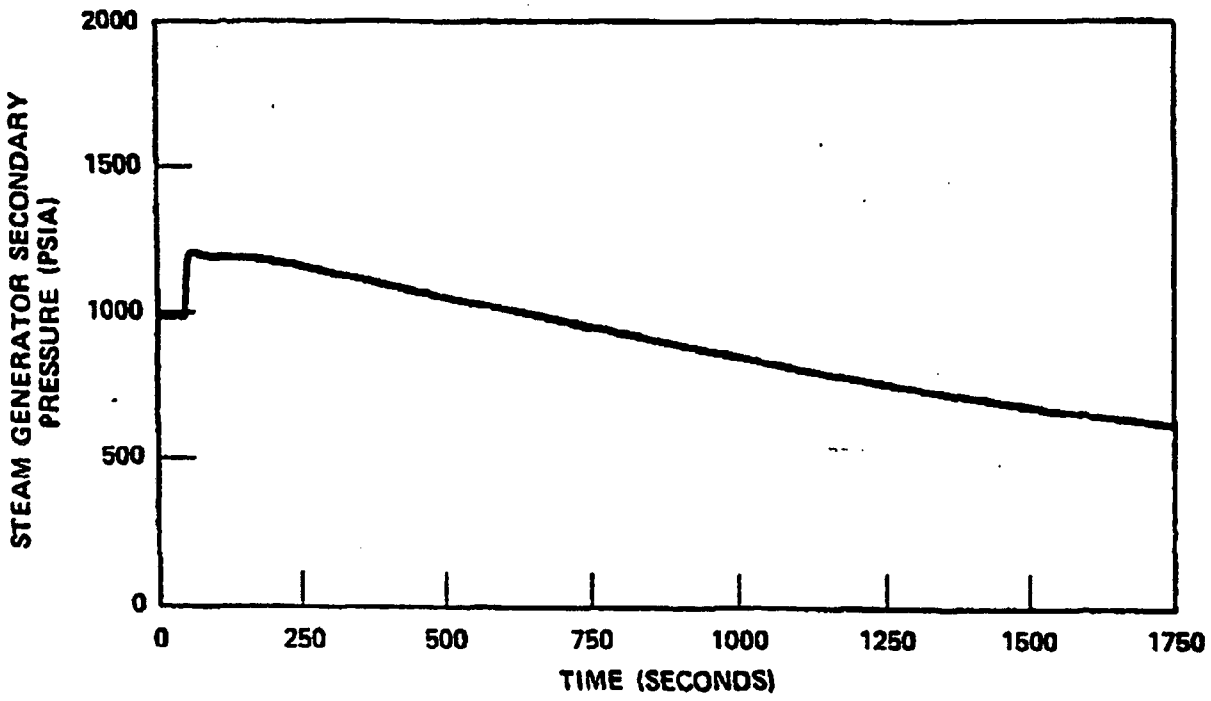

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Figure 15.4-26

STEAM GENERATOR SECONDARY PRESSURE TRANSIENT FOR ROD EJECTION ANALYSIS

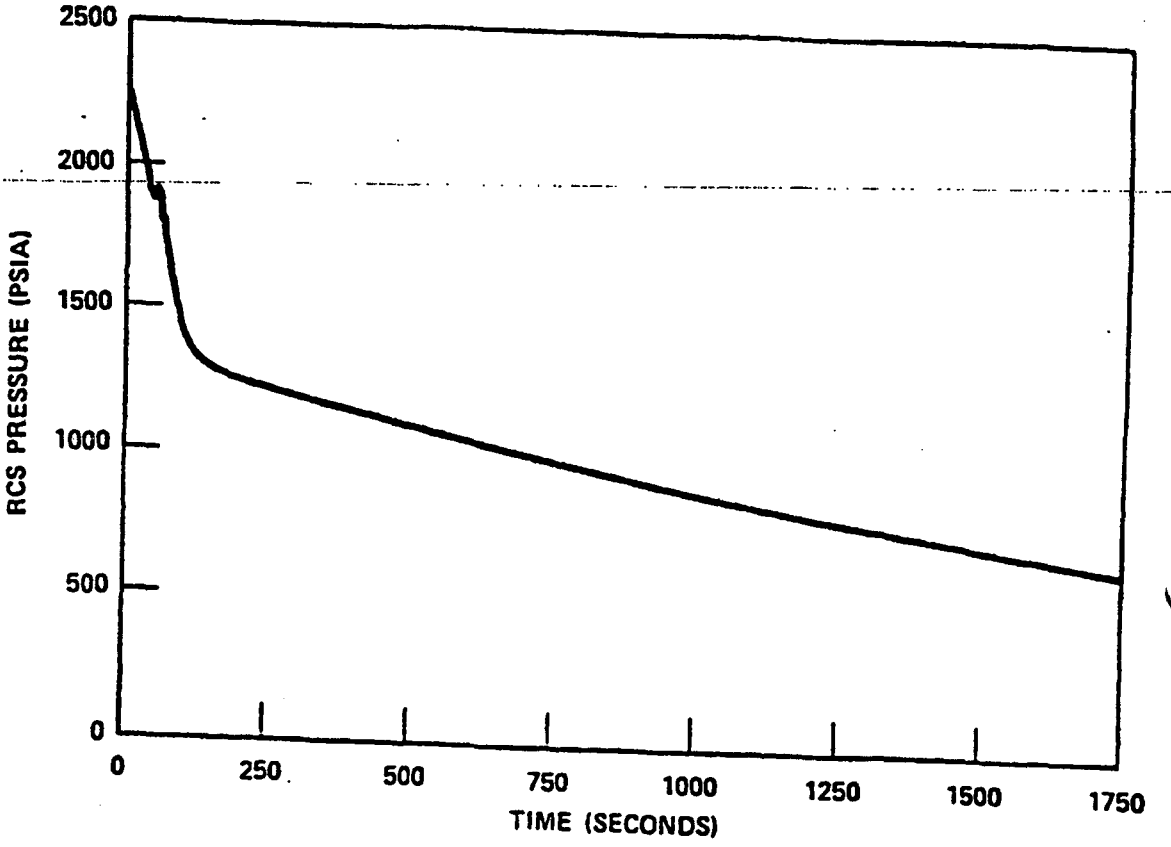

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FIGURE 15.4-27

RCS PRESSURE TRANSIENT FOR
ROD EJECTION ANALYSIS

FIGURE 15.4-28 HAS BEEN DELETED

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FIGURE 15.4-28

REACTOR COOLANT SYSTEM
INTEGRATED BREAK FLOW FOLLOWING
A ROD EJECTION ACCIDENT

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events are presented in this section:

a. Inadvertent operation of the emergency core cooling system during power operation.

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

c. A number of BWR transients. (Not applicable to WCGS).

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Section 15.0.1 contains a discussion of ANS classifications.

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 Identification of Causes and Accident Description

Spurious emergency core cooling system (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels, as described in Section 7.3.

Following the actuation signal, the suction of the charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the boron injection tank from the centrifugal charging pumps and the valves isolating the boron injection tank from the injection header then automatically open. The centrifugal charging pumps then inject RWST boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the reactor coolant system (RCS) is at normal pressure. The passive, accumulator safety injection system and the low head, residual heat removal system also provide no flow at normal RCS pressure.

A safety injection signal (SIS) normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates ECCS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS, the operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the SIS should be blocked. For a spurious occurrence, the operator

would terminate ECCS and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, subsequent plant operation would be in accordance with the Technical Specifications.

If the reactor protection system does not produce an immediate trip as a result of the spurious SIS, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power. The power mismatch causes a drop in Tavg and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load will decrease due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual reactor trip.

The time to trip is affected by initial operating conditions, including core burnup history, which affects initial boron concentration, rate of change of boron concentration, Doppler, and moderator coefficients.

Recovery from the case without an immediate reactor trip is made in the same manner as described for the case where the SIS results directly in a reactor trip. The main difference is that the Tavg and RCS pressure decrease to lower values, as a result of the power mismatch, and thus the pressurizer level at the time of reactor trip is at a lower value. Tripping the reactor coincident with event initiation is conservative for pressurizer filling because the pressurizer level is greater at the time of reactor trip. At lower loads, coolant contraction will be slower, resulting in a longer time to trip.

15.5.1.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the ECCS is analyzed by employing the detailed digital computer program RETRAN (Refs. 1 and 2). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Based on its expected frequency of occurrence, the inadvertent ECCS actuation at power is considered to be a Condition II event, a fault of moderate frequency. The specific criteria established for Condition II events include the following:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,
b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs,
c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

With respect to overpressurization, the inadvertent ECCS actuation at power event is bounded by the Loss of Load/Turbine trip events, in which assumptions are made to conservatively calculate the RCS and main steam system pressure transients. For the inadvertent ECCS actuation at power event, turbine trip occurs following reactor trip, whereas for the Loss of Load/Turbine Trip event, the turbine trip is the initiating fault. Therefore, the primary to secondary power mismatch and resultant RCS and main steam system heatup and pressurization transient are always more severe for the Loss of Load/Turbine Trip event. For this reason, it is not necessary to calculate the maximum RCS or main steam system pressure for the inadvertent ECCS actuation at power.

Based on historical precedence, the inadvertent ECCS actuation at power event does not lead to a serious challenge of the DNB design basis. The decrease in core power and RCS average temperature more than offset the decrease in RCS pressure such that the minimum calculated DNBR occurs at the start of the transient. As such, no explicit analysis of the event has been performed to address DNB concerns.

The primary concern that results from an inadvertent actuation of the ECCS at power is that associated with pressurizer overfill. The pressurizer water volume increases as a result of the safety injection flow. This may eventually lead to filling of the pressurizer and subsequent water relief through the safety or relief valves. The passing of liquid through the pressurizer safety or relief valves could result in rupture of the pressurizer relief tank rupture disks, spilling radioactive coolant into the containment building, thereby escalating a Condition II event to a Condition III or IV event. Historically, Criterion c has been shown to be met by demonstrating that the pressurizer will not become water-solid. To prevent the pressurizer from becoming water-solid, operator action is ultimately required to terminate safety injection or mitigate the consequences of this event. Therefore, the event is analyzed to demonstrate that sufficient time is available for the appropriate operator actions to be taken to prevent filling the pressurizer.

A bounding pressurizer overfill transient is presented representing maximum reactivity feedback. For calculational simplicity, zero injection line purge volume was assumed in this analysis, thus the boration transient begins immediately when the appropriate valves are opened. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

The assumptions are as follows:

a. Initial operating conditions
   The plant is initially operating at an NSSS power of 3651 MWt, which includes all applicable uncertainties. The initial vessel average temperature corresponds to the low full power nominal value (570.7°F) minus uncertainties (4.0°F), and the initial pressurizer pressure corresponds to the nominal value (2250 psia) minus uncertainties (35 psi). The lower initial temperature results in a higher RCS coolant mass, and the lower initial pressure allows more ECCS flow to be injected into the RCS, causing a more severe pressurizer water volume transient.
b. Reactivity feedback
   A large (absolute value) negative moderator temperature coefficient and a most-negative Doppler power coefficient are assumed.
c. Reactor control
   The reactor was assumed to trip at the time of the SI signal. Thus, the reactor control mode is of no consequence.

d. Pressurizer pressure control
   Pressurizer proportional heaters are assumed to be operable to maximize the thermal expansion of the water within the pressurizer. The backup heaters are not modeled because the SI signal causes them to be shed from their electric power supply, and they are not loaded onto another power supply automatically or manually until letdown flow is re-established. Pressurizer spray is assumed to be available in order to minimize RCS pressure, which allows more ECCS flow to be injected into the RCS. In addition, modeling the pressurizer spray increases the rate of the pressurizer level increase due to the density effects between the spray fluid, which is taken from the cold leg, and the pressurizer inventory.
e. Boron Injection
   At the initiation of the event, two ECCS centrifugal charging pumps (CCPs) and a normal charging pump (NCP) inject RWST boron solution into the cold leg of each loop.
f. Turbine load
   The reactor and turbine both trip at the time of SI actuation with the turbine load dropping to zero simultaneously with the steam flow.
g. ECCS injection flowrates
   Maximum safeguards ECCS flow is delivered from two CCPs, plus flow from an NCP used for normal charging prior to the occurrence of the event-initiating SI signal. All letdown is assumed isolated at the initiation of the event.
h. Reactor Trip
   An immediate reactor trip on the initiating SI signal to maximize pressurizer inventory is assumed.
i. Required Operator Actions
   Three operator actions are assumed. The first operator action is the opening of one of the four secondary system atmospheric relief valves (ARVs) within 6 minutes after initiation of the event to control the cold leg temperatures to 557°F. The second operator action is the termination of the safety injection flow to the cold legs within 8 minutes after event initiation. The third operator action is the re-establishment of letdown flow within 29.5 minutes after event initiation. These operator actions are required to preclude the possibility of a pressurizer water-solid condition and are supported by plant operations through use of an emergency operating procedure.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-22. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident.

Results

The calculated sequence of events is shown on Table 15.5-1.

Figures 15.5-1 through 15.5-3 show the transient response to inadvertent operation of ECCS during power operation.

Reactor trip occurs at event initiation followed by a rapid cooldown of the RCS. Coolant contraction due to the decrease in reactor power and RCS temperature results in a short-term reduction in pressurizer pressure and water level. The temperature drop is mainly caused by the decrease in power but is also affected by the addition of the cold SI flow. The combination of the RCS heatup, due to residual heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then begins to increase. At six minutes after event initiation, it is assumed that the plant operators initiate the opening of one of the four secondary system ARVs to control T_{cold} to 557°F. This results in a cooldown of the secondary side and a subsequent cooldown of the primary side, resulting in a thermal shrinking of the RCS water inventory. The continued injection of water into the RCS causes a continued increase in the pressurizer level. At eight minutes after event initiation, it is assumed that the plant operators take action to terminate the safety injection flow to the cold legs. This significantly attenuates the RCS inventory increase and leaves RCP seal injection flow as the only source of water injection into the RCS. At approximately 26 minutes after event initiation, the pressurizer pressure reaches a value that causes spray flow initiation, which adds to the increase of the pressurizer water volume inventory. At 29.5 minutes after event initiation, it is assumed the operators have re-established letdown flow, which terminates the increase in RCS water inventory. The cumulative effect of the assumed operator actions keeps the pressurizer from reaching a water-solid condition. As the pressurizer does not reach a water-solid condition, no water relief occurs through the pressurizer safety or relief valves.

Recovery from this accident is discussed in Section 15.5.1.1.

15.5.1.3 Conclusions

Results of the analysis show that the spurious ECCS operation with immediate reactor trip meets all acceptance criteria for an ANS Condition II event.

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

15.5.2.1 Identification of Causes and Accident Description

Increases in reactor coolant inventory caused by the chemical and volume control system may be postulated to result from operator error or a false electrical signal. Transients examined in this section are characterized by increasing pressurizer level, increasing pressurizer pressure, and constant boron concentration. The transients analyzed in this section are done to demonstrate that there is adequate time for the operator to take corrective action to prevent filling the pressurizer. An increase in reactor coolant inventory, which results from the addition of cold, unborated water to the RCS, is analyzed in Section 15.4.6, CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1, INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION.

Transients postulated as a result of operator error or failure of the charging pump controller which increase primary side inventory would cause a high pressurizer level reactor trip before the pressurizer can be filled, thus these are not the worst cases.

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow would be isolated. The worst single failure for this event would be another pressurizer level channel failing in an as is condition or a low condition. This will defeat the reactor trip on two out of three high pressurizer level channels. To prevent filling the pressurizer the operator must be relied upon to terminate charging.

15.5.2.2 Analysis of Effects and Consequences

Method of Analysis

The charging malfunction is analyzed by employing the detailed digital computer program RETRAN (Refs. 1 and 2). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Four cases were analyzed;

a. Minimum reactivity feedback with automatic pressurizer spray

b. Minimum reactivity feedback without automatic pressurizer spray

c. Maximum reactivity feedback with automatic pressurizer spray

d. Maximum reactivity feedback without automatic pressurizer spray

The assumptions incorporated in the analyses were as follows;

a. Initial Operating Conditions

The plant is initially operating at an NSSS power of 3651 MWt, which includes all applicable uncertainties. The initial vessel average temperature corresponds to the low full power nominal value (570.7°F) minus uncertainties (6.5°F), and the initial pressurizer pressure corresponds to the nominal value (2250 psia) minus uncertainties (50 psi).

b. Reactivity Coefficients

1. Minimum Reactivity Feedback Case

A least-negative moderator temperature coefficient and a least negative Doppler-only power coefficient.

2. Maximum Reactivity Feedback Case

A conservatively large (absolute value) negative moderator temperature coefficient and a most negative Doppler-only power coefficient.

c. Reactor Control

Cases with and without automatic rod control were examined. Since there is little or no change in core power and Tavg the effects of rod control are relatively negligible.

d. Charging System

Maximum charging system flow based on RCS back pressure from one centrifugal charging pump is delivered to the RCS. The charging flow is assumed to have the same boron concentration as the RCS.

e. Reactor Trip

A reactor trip on high pressurizer level would be expected to occur for this event. However, the transient is initiated by the pressurizer level channel which is used for control purposes failing low. As a worst single failure, a second pressurizer level channel is then assumed to fail in an as-is condition or low, defeating the two out of three high pressurizer level trip. No reactor trips are modeled.

Results

In all cases analyzed, the core power and RCS temperatures remain relatively constant. Cases with and without automatic rod control were examined. Since there was little or no change in core power and RCS average temperatures, the results showed that automatic rod control has no impact on the cases that model maximum reactivity feedback conditions and a relatively negligible effect on the cases that model minimum reactivity feedback conditions. Figures 15.5-4 through 15.5-11 show the transient responses for the cases with automatic rod control turned on.

The cases that model maximum reactivity feedback show that the pressurizer level increases at a relatively constant rate; whereas, the cases that model minimum reactivity feedback show that the pressurizer level increases at a somewhat varying rate. This is because the reactivity feedback in the maximum feedback cases is of a magnitude that it is able to maintain Tavg within the temperature deadband for the automatic rod control system and no rod movement is predicted in these cases, but rod movement is predicted in the minimum feedback cases, resulting in slight variations in Tavg and ultimately in the pressurizer level increase in these cases.

The rate of increase in the pressurizer level is slightly faster in the cases where the pressurizer spray is modeled operable, as compared to the cases in which the pressurizer sprays are modeled inoperable, because spray actuation tends to keep the RCS pressure lower for several minutes, which allows the charging pumps to deliver more flow to the RCS. However, pressurizer pressure does eventually increase enough to open the relief valves in the cases with the pressurizer spray modeled operable.

Times at which the operator would receive alarms are listed in Table 15.5-1.

15.5.2.3 Conclusions

Results show none of the operating conditions during the transient approach core limits. The high pressurizer level trip has been defeated by failures. The transient is ultimately terminated by the plant operators. The sequence of events presented in Table 15.5-1 show that the operators have sufficient time to take action necessary to prevent a pressurizer water-solid condition.

15.5.3 A NUMBER OF BWR TRANSIENTS

This section is not applicable to WCGS.

15.5.4 REFERENCES

1. Huegel, D. S., et al. "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

2. McFadden, J. H., et al., “RETRAN-02 - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,” EPRI NP-1850-CCM-A, October 1984.

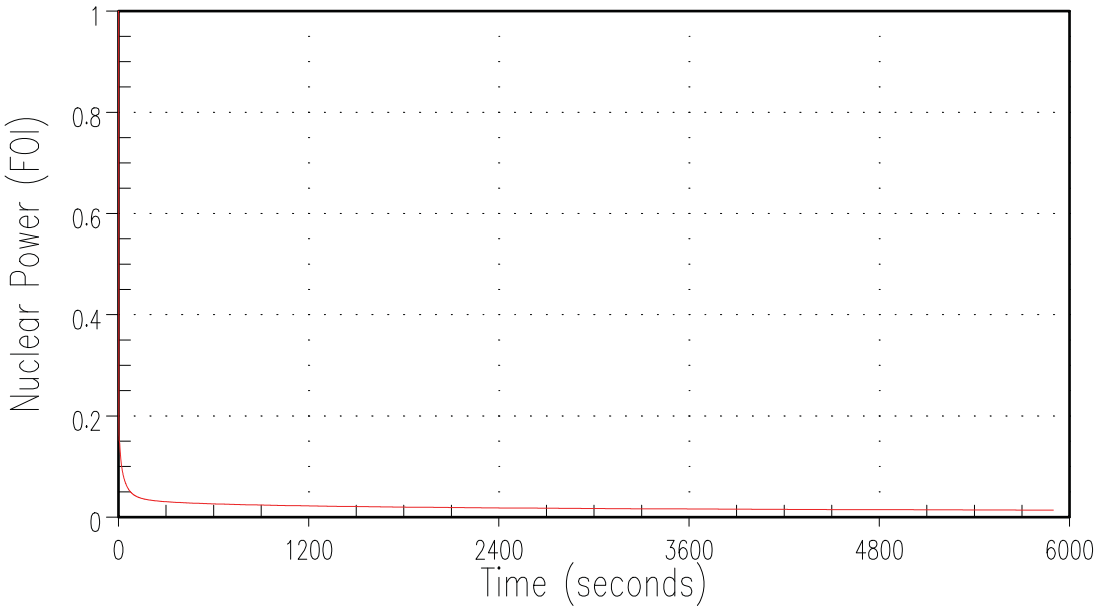
TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (seconds)
Inadvertent Actuation of the ECCS During Power Operation	Inadvertent ECCS/SI Signal Actuation	0.0
	Reactor Trip due to SI Signal	0.0
	Turbine Trip from Reactor Trip	0.0
	Letdown Isolation	0.0
	On Each Loop, the MSSV with the Lowest Setting Opens	236.4
	One ARV Begins to Open (1st Operator Action)	360.0
	All MSSVs Closed	373.5
	SI Flow to Cold Legs Terminated (2nd Operator Action)	480.0
	Letdown Flow Re-established (3rd Operator Action)	1770.0
	Maximum Pressurizer Water Volume Reached	2450.0
Chemical and Volume Control System Malfunction, Maximum Reactivity Feedback, With Pressurizer Spray	Two Pressurizer Level Channels Fail Low	0.0
	Maximum Charging Flow from One Centrifugal Charging Pump Begins	0.0
	Letdown is Isolated	0.0
	Low-Low Pressurizer Level Alarm	0.0
	High Pressurizer Level Alarm	514.6
	Pressurizer Fills	1049.5
	Pressurizer Relief Valve Setpoint Reached	1054.3

TABLE 15.5-1 (Sheet 2)

Accident	Event	Time (seconds)
Chemical and Volume Control System Malfunction, Maximum Reactivity Feedback, Without Pressurizer Spray	Two Pressurizer Level Channels Fail Low	0.0
	Maximum Charging Flow from One Centrifugal Charging Pump Begins	0.0
	Letdown is Isolated	0.0
	Low-Low Pressurizer Level Alarm	0.0
	Pressurizer Relief Valve Setpoint Reached	15.4
	High Pressurizer Level Alarm	609.1
	Pressurizer Fills	1365.7
Chemical and Volume Control System Malfunction, Minimum Reactivity Feedback, With Pressurizer Spray	Two Pressurizer Level Channels Fail Low	0.0
	Maximum Charging Flow from One Centrifugal Charging Pump Begins	0.0
	Letdown is Isolated	0.0
	Low-Low Pressurizer Level Alarm	0.0
	High Pressurizer Level Alarm	529.3
	Pressurizer Fills	1036.4
	Pressurizer Relief Valve Setpoint Reached	1239.8
Chemical and Volume Control System Malfunction, Minimum Reactivity Feedback, Without Pressurizer Spray	Two Pressurizer Level Channels Fail Low	0.0
	Maximum Charging Flow from One Centrifugal Charging Pump Begins	0.0
	Letdown is Isolated	0.0
	Low-Low Pressurizer Level Alarm	0.0
	Pressurizer Relief Valve Setpoint Reached	24.5
	High Pressurizer Level Alarm	597.6
	Pressurizer Fills	1345.7



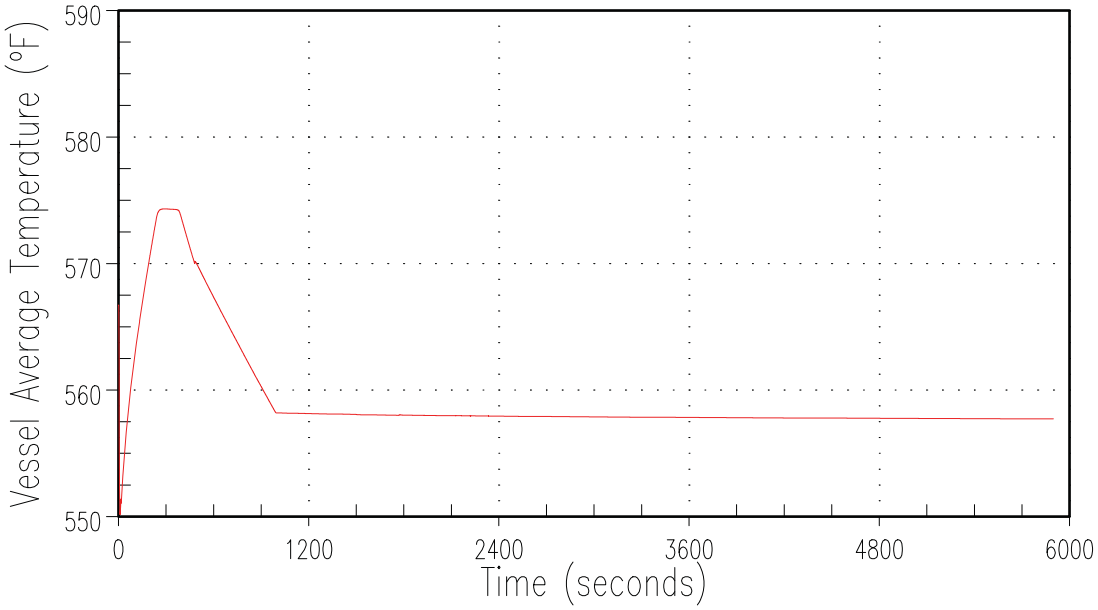


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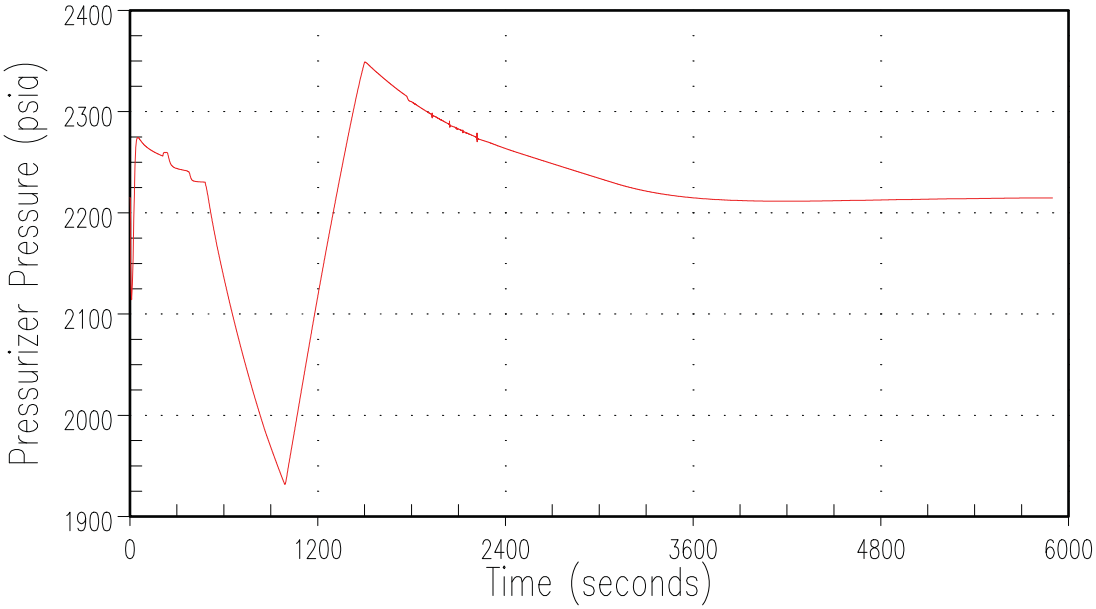
Figure 15.5-1
Nuclear Power and Vessel Average Temperature Transients for Inadvertent Operation of ECCS During Power Operation



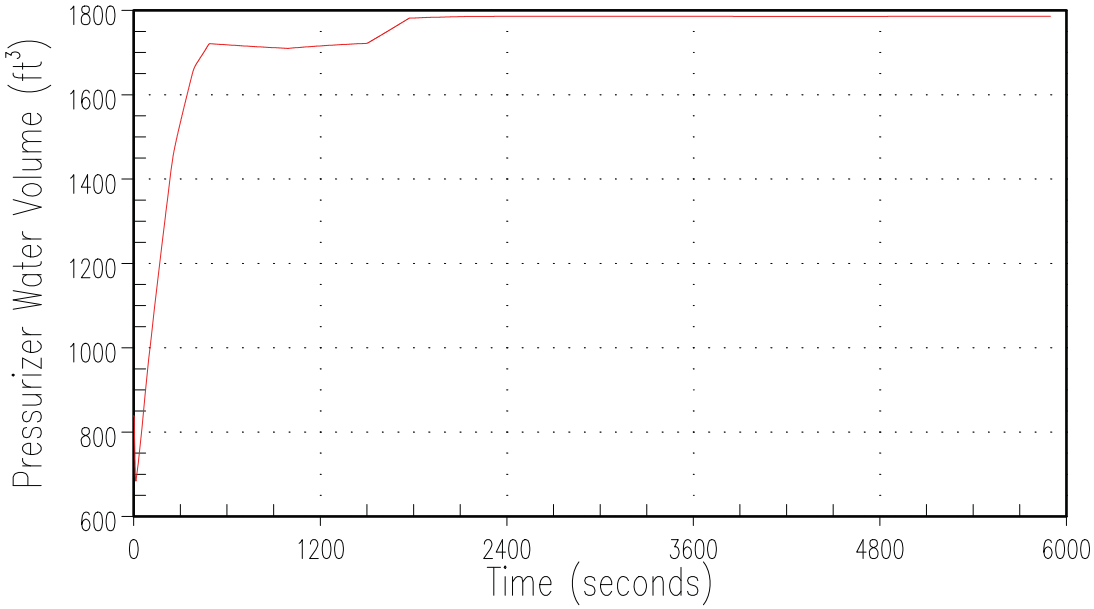


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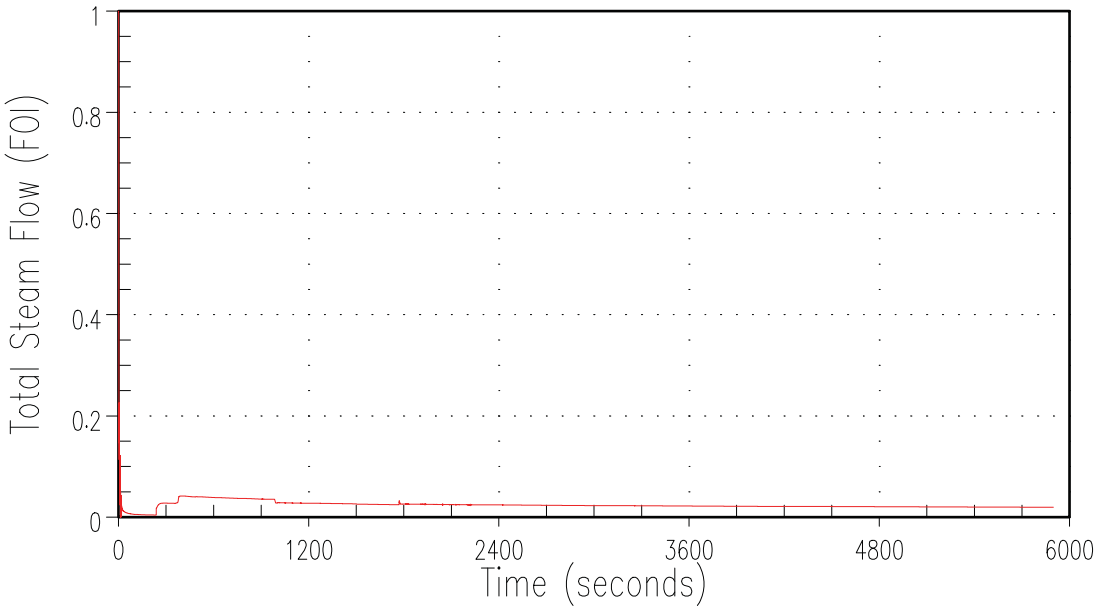
Figure 15.5-2
Pressurizer Pressure and Pressurizer Water Volume Transients for Inadvertent Operation of ECCS During Power Operation



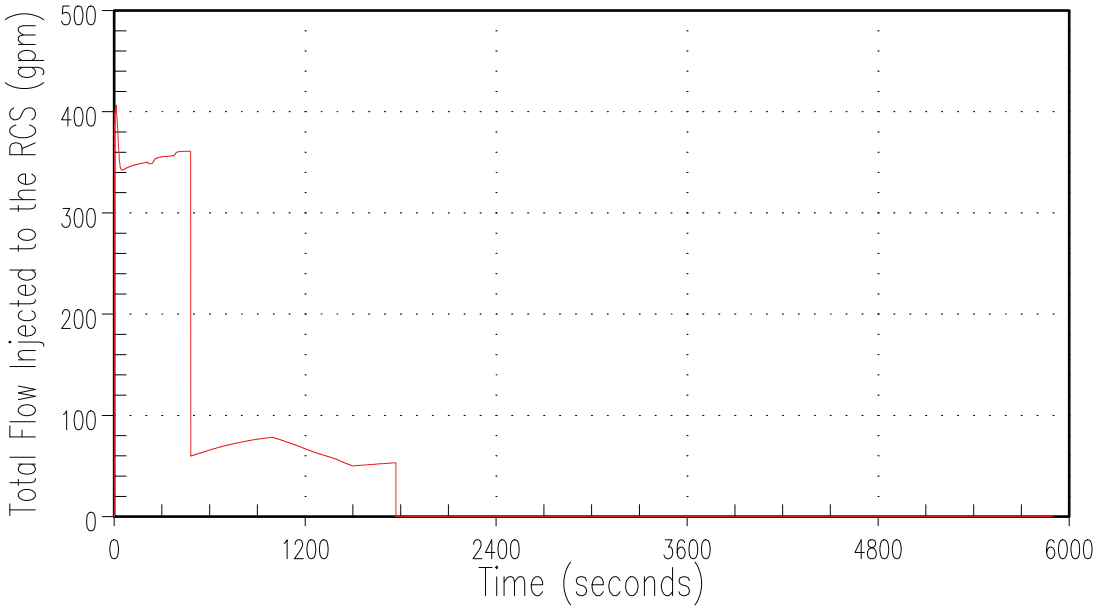


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Figure 15.5-3
Total Steam Flow and Total Flow Injected to the RCS Transients for Inadvertent Operation of ECCS During Power Operation

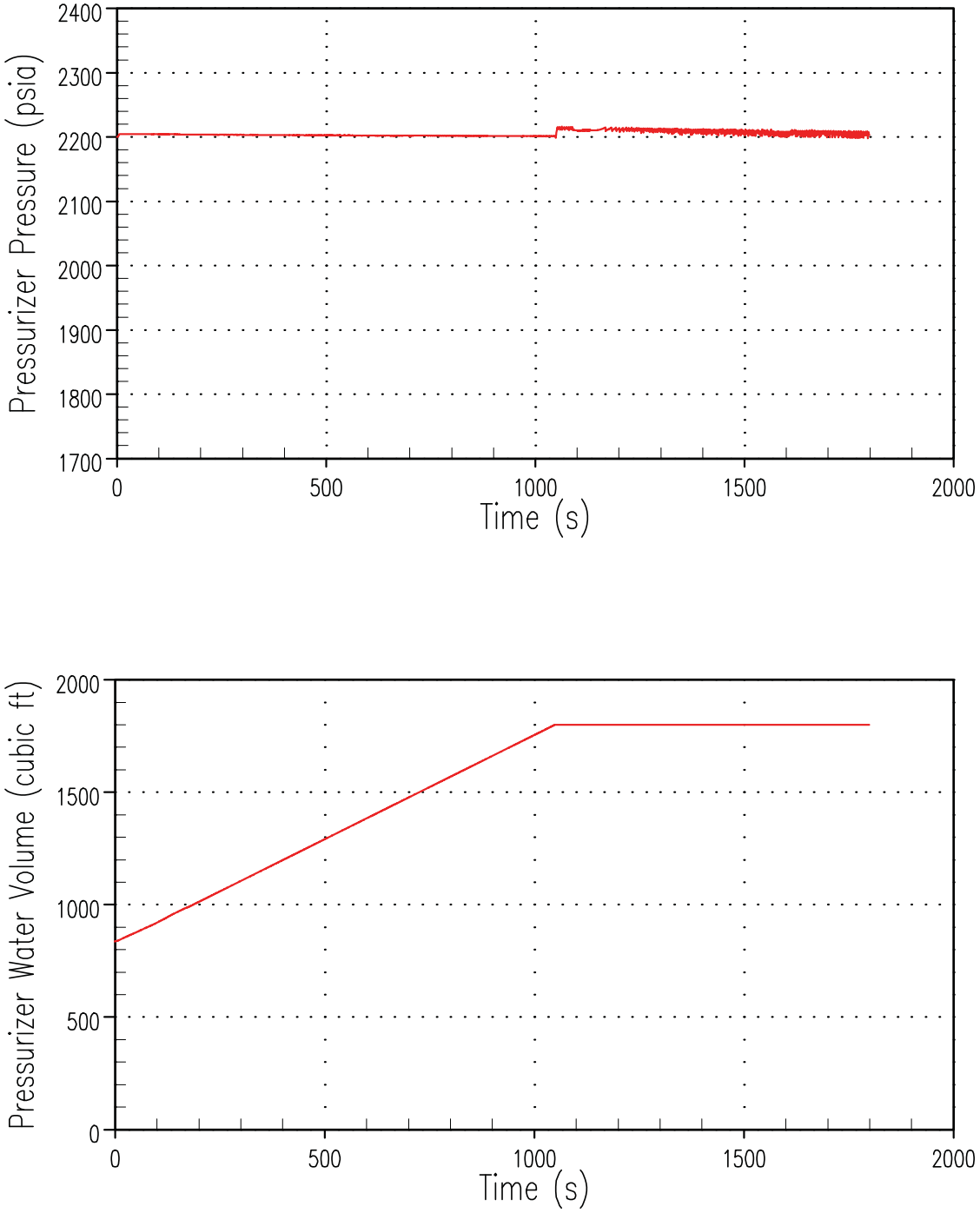

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Figure 15.5-4
Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Maximum Reactivity Feedback and Pressurizer Spray

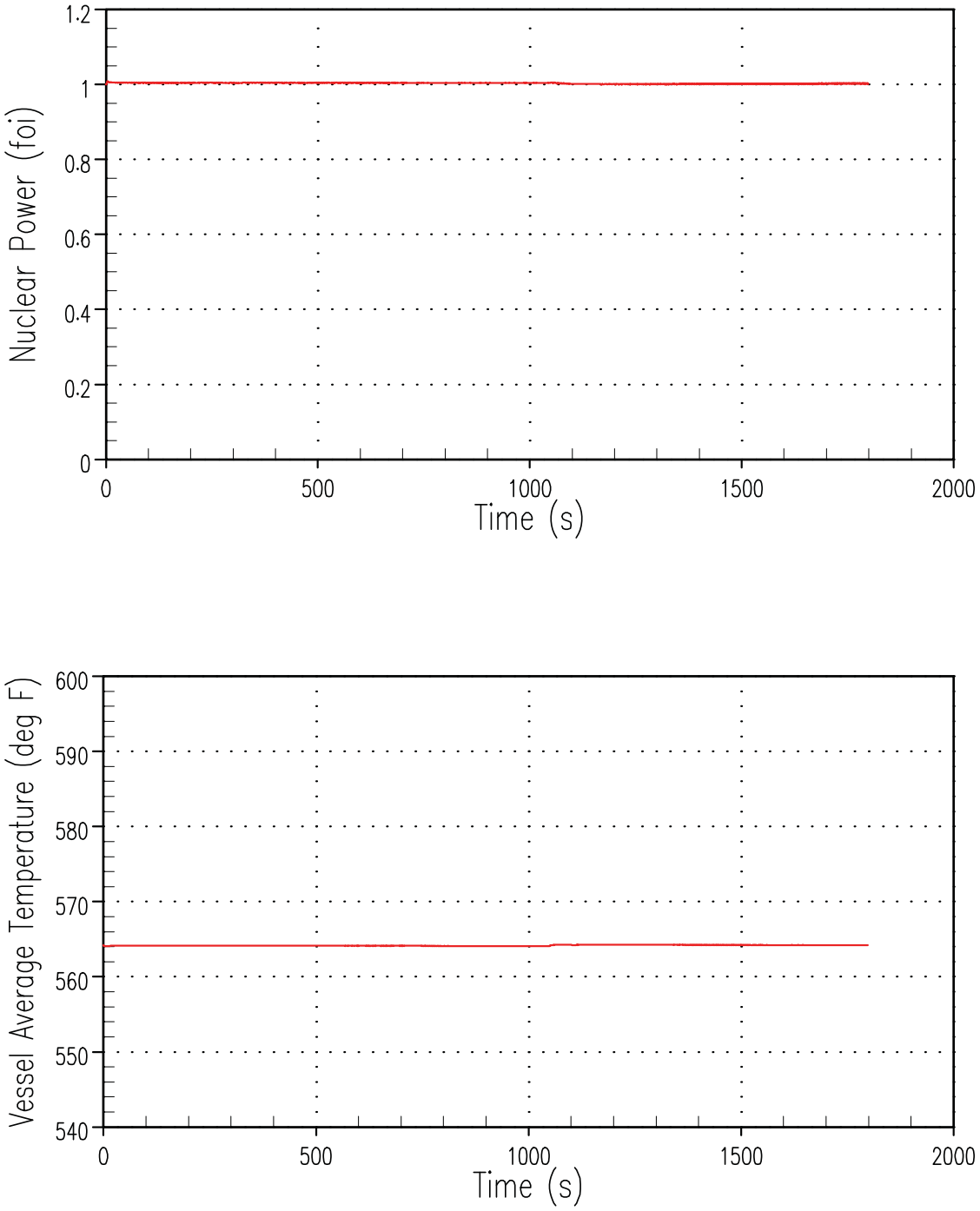

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Figure 15.5-5
Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Maximum Reactivity Feedback and Pressurizer Spray

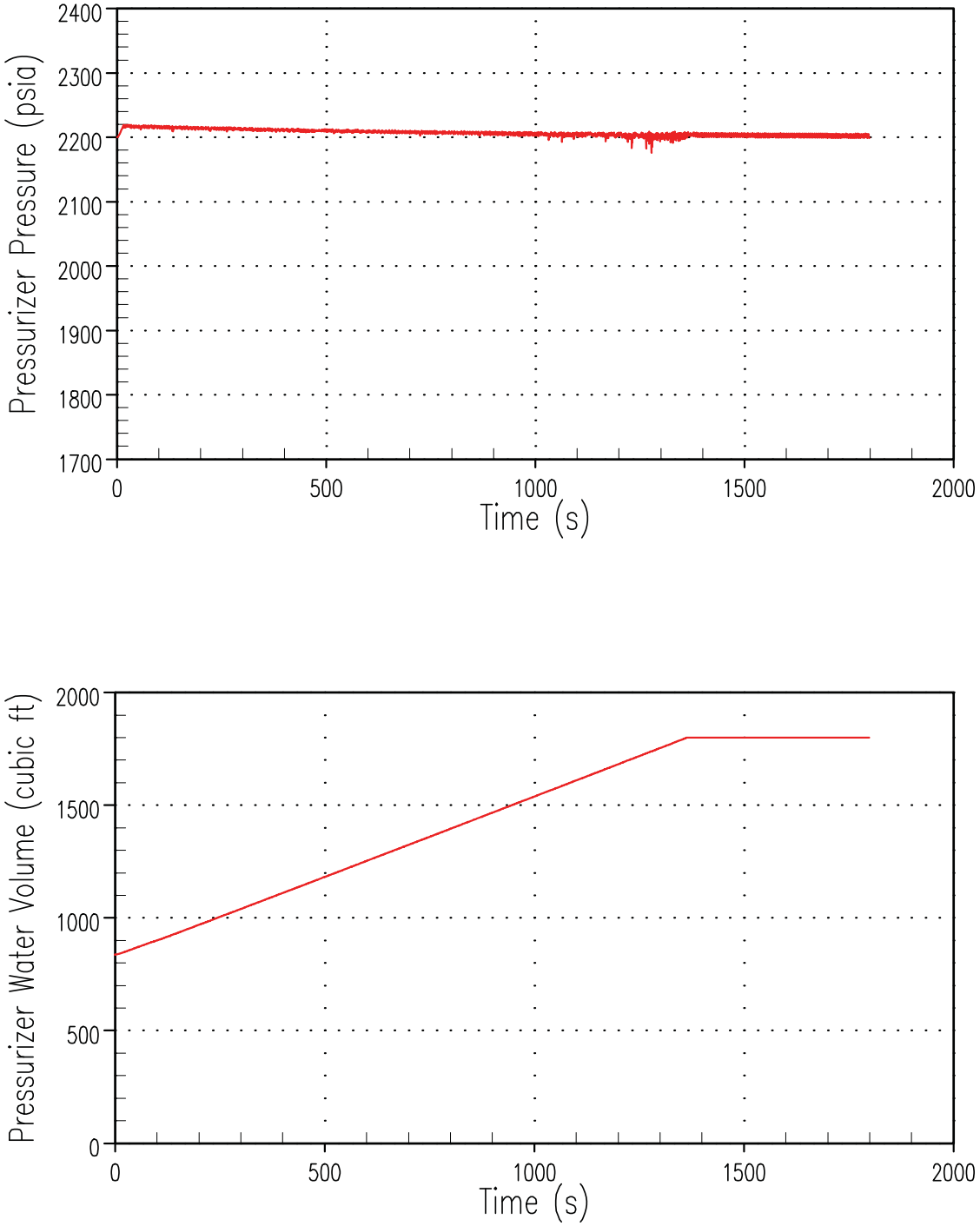

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Figure 15.5-6
Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Maximum Reactivity Feedback and No Pressurizer Spray

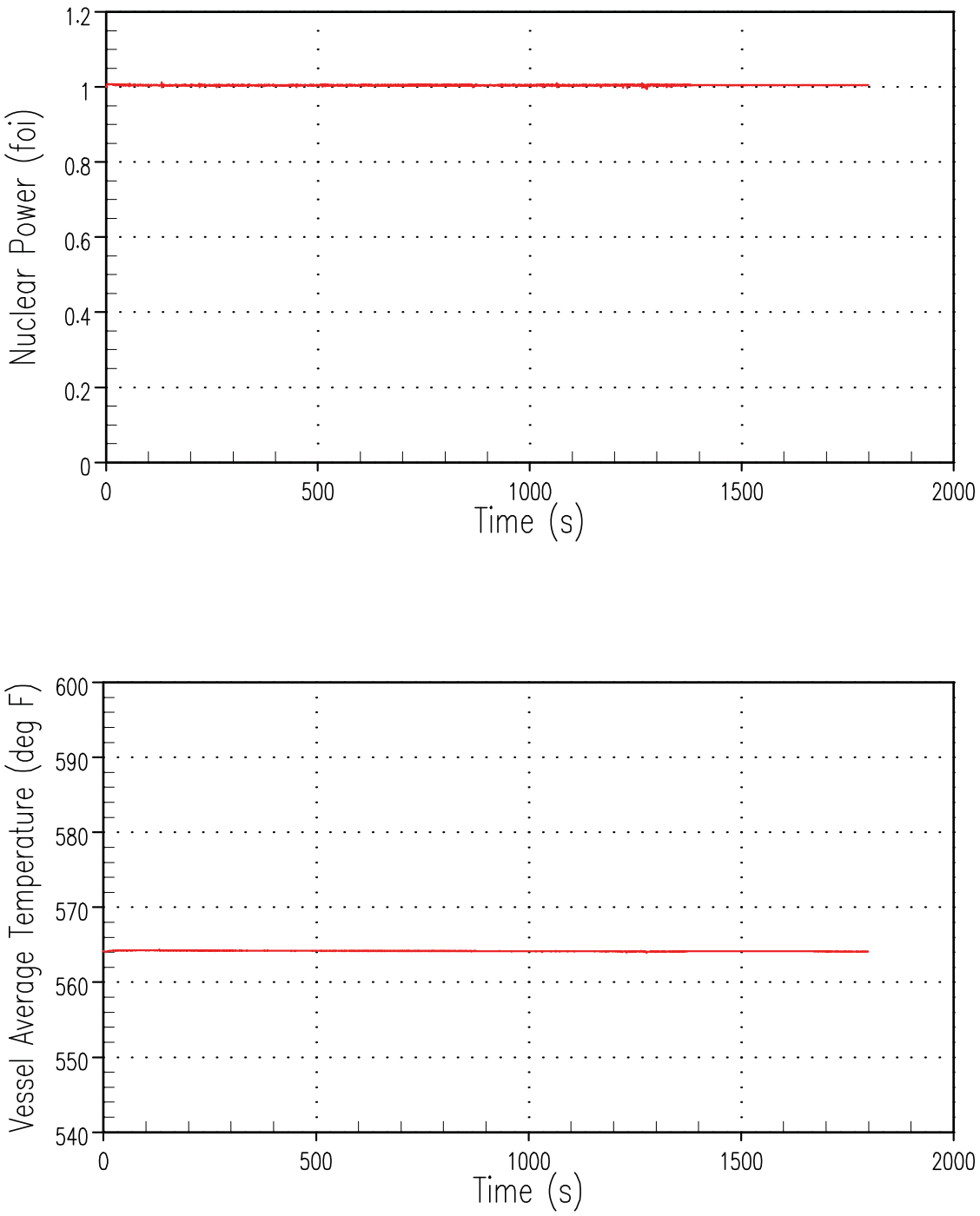

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Figure 15.5-7
Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Maximum Reactivity Feedback and No Pressurizer Spray

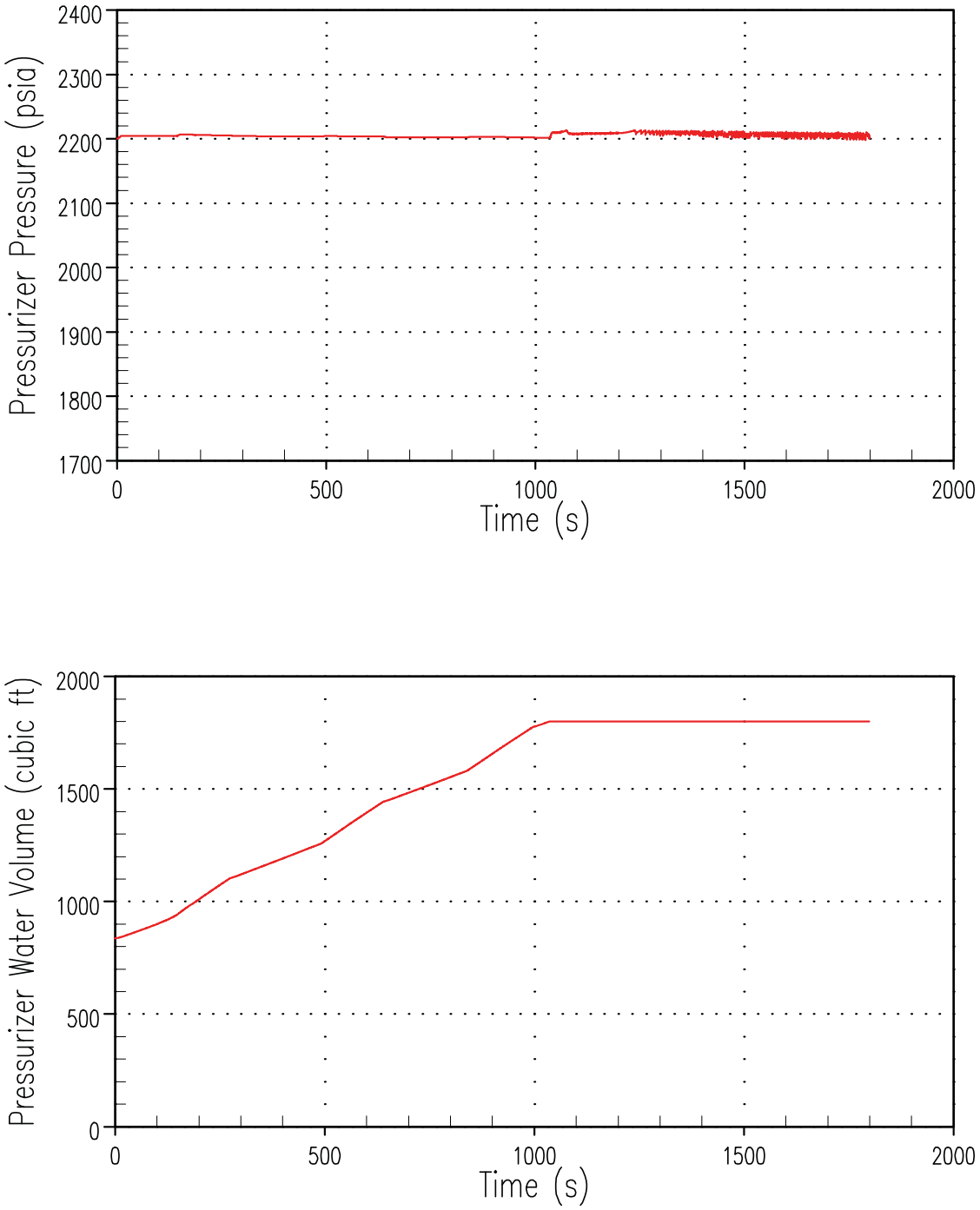

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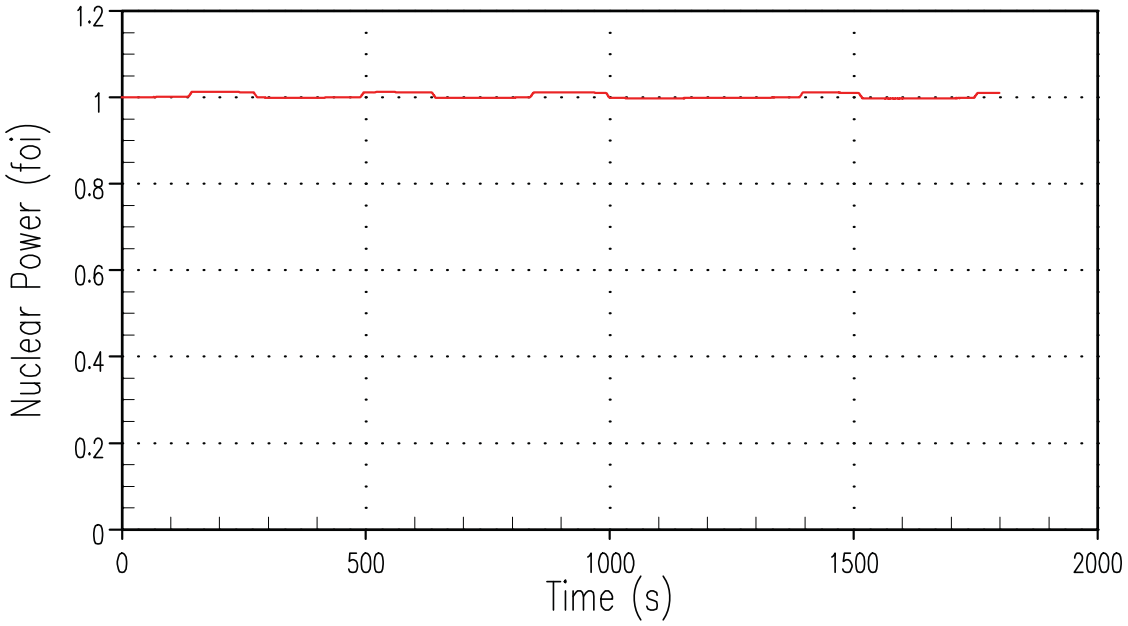
Figure 15.5-8
Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Minimum Reactivity Feedback and Pressurizer Spray



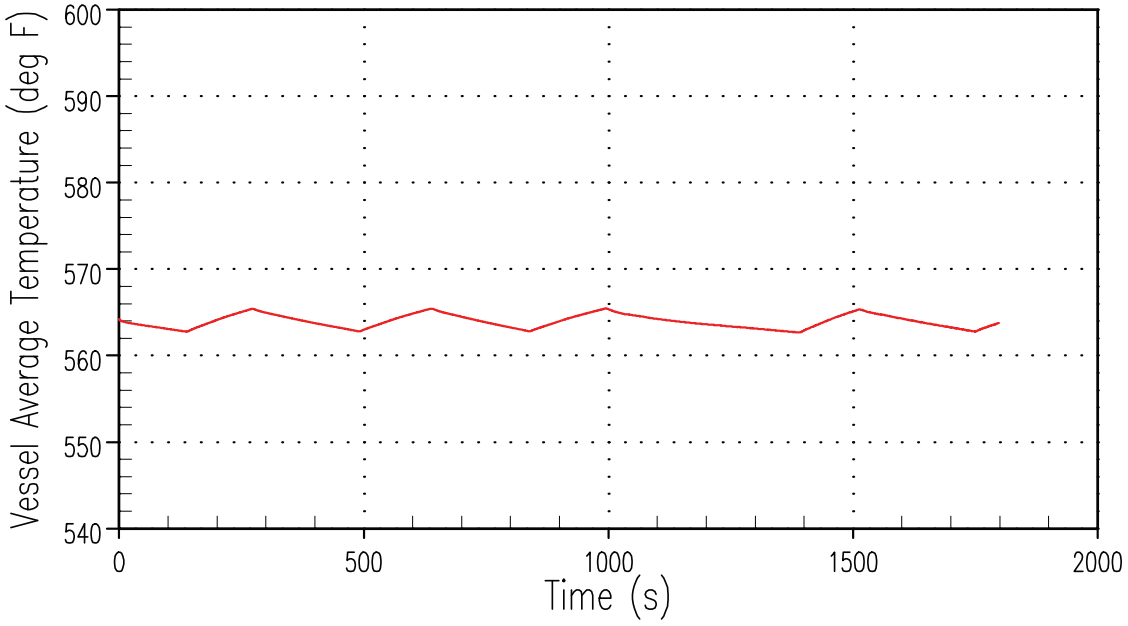


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Figure 15.5-9
Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Minimum Reactivity Feedback and Pressurizer Spray

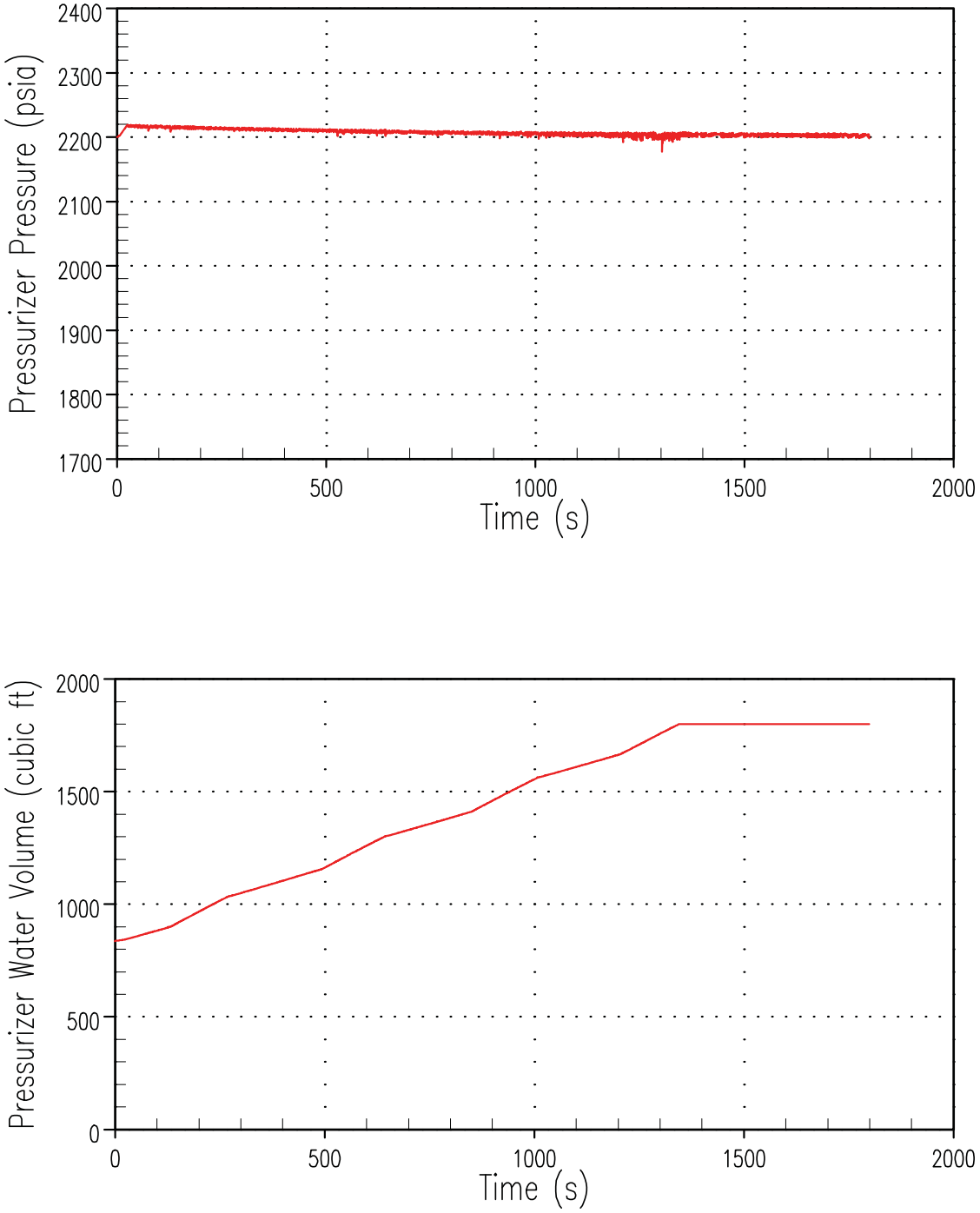

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Figure 15.5-10
Pressurizer Pressure and Pressurizer Water Volume Transients for CVCS Malfunction with Minimum Reactivity Feedback and No Pressurizer Spray

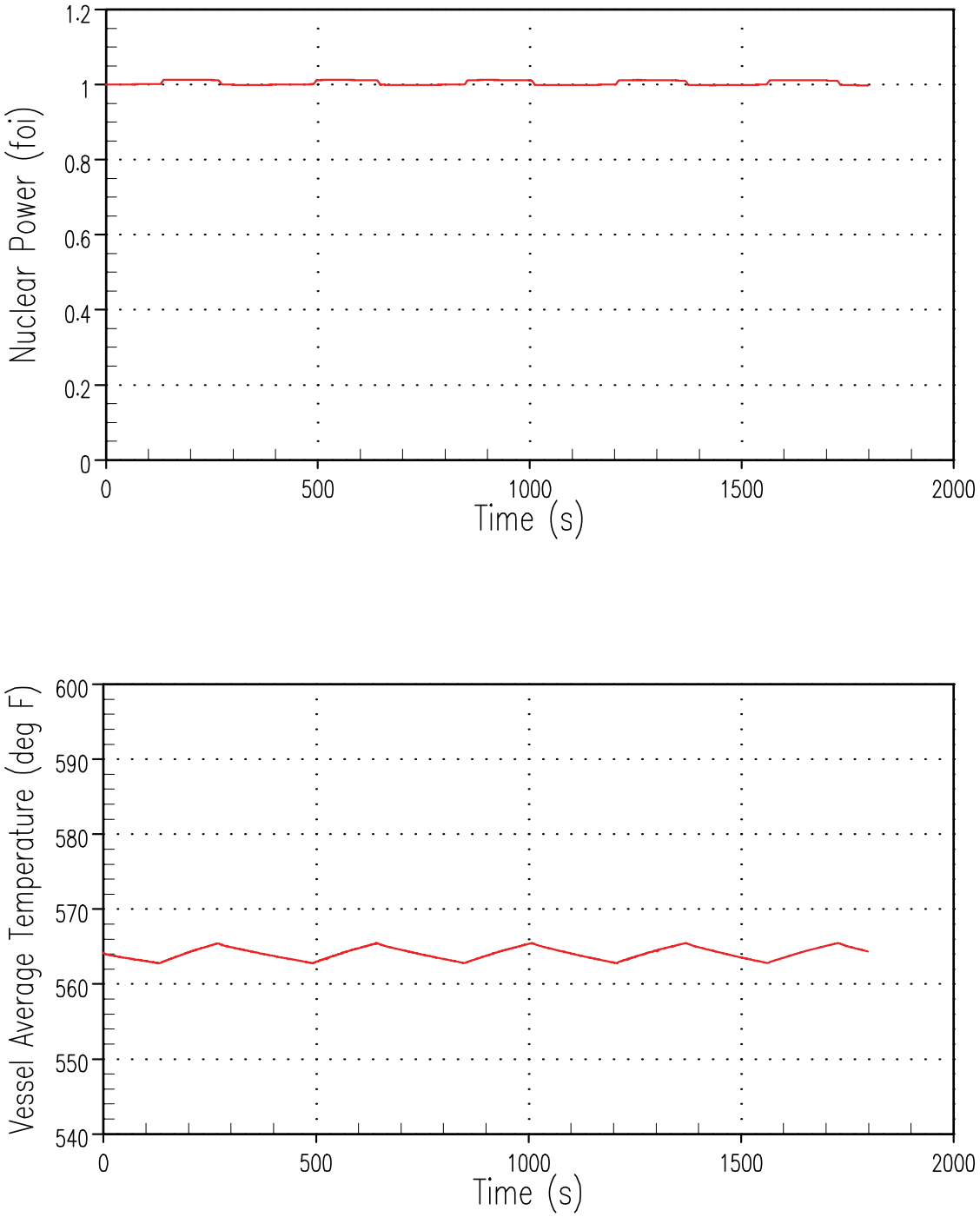

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Figure 15.5-11
Nuclear Power and Vessel Average Temperature Transients for CVCS Malfunction with Minimum Reactivity Feedback and No Pressurizer Spray

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

a. Inadvertent opening of a pressurizer safety or relief valve

b. Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment

c. Steam generator tube rupture (SGTR)

d. Spectrum of boiling water reactor (BWR) steam system piping failures outside of the containment (Not applicable to WCGS)

e. Loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

f. A number of BWR transients (Not applicable to WCGS)

All of the applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences will result from the major LOCA of Section 15.6.5. Therefore, the LOCA is the design basis accident. The LOCA CVCS letdown line break outside the containment and the SGTR accident have been analyzed radiologically. All other accidents in this section are bounded by these accidents.

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY OR RELIEF VALVE

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure without reactor protection system intervention. The effect of the pressure decrease would be to

decrease power via the moderator density feedback.

The reactor may be tripped by the following reactor protection system signals:

a. Overtemperature ΔT
b. Pressurizer low pressure

An inadvertent opening of a pressurizer relief valve is classified as an ANS Condition II event, a fault of moderate frequency. The failure of a pressurizer safety valve is classified as an ANS Condition III event. The analysis performed conservatively bounds the more limiting failure while still applying the more restrictive Condition II acceptance criterion of ensuring that the DNB design basis is met. See Section 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

Method of Analysis

The inadvertent opening of a pressurizer safety valve transient is analyzed by employing the detailed digital computer program RETRAN (Ref. 3). RETRAN has been found acceptable by the NRC for use as a licensing basis safety analysis code. RETRAN is a thermal-hydraulic systems analysis code employing a one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the RCS, a point kinetics model for the reactor core, special component and auxiliary models (ex., pumps, temperature transport, non-equilibrium pressurizer) and control system models. The code computes pertinent plant transient information including core power level, RCS pressure and temperature.

Plant characteristics and initial conditions are discussed in Section 15.0.3. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made:

a. The accident is analyzed with the Revised Thermal Design Procedure (Ref. 9). The event is analyzed at the conservatively high initial core power level listed in Table 15.0-2. The initial pressurizer pressure and vessel average temperature are assumed to be at the respective nominal values, and the reactor coolant flow is at the minimum measured flow value. Uncertainties in the initial conditions are accounted for in the DNBR limit value as described in Reference 9.
b. Minimum reactivity feedback is assumed to minimize the resulting DNBR, including a moderator temperature coefficient of zero and the least-negative Doppler coefficient. The spatial effect of the void due to local or subcooled boiling, which tends to flatten the core power distribution, is not considered in the analysis with respect to reactivity feedback or core power shape.

Major plant systems and equipment available for mitigation of transient and accident conditions which may be required to function to mitigate the effects of RCS depressurization caused by inadvertent safety valve opening are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-23.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst case assumption; if the reactor were in manual control, a trip could occur earlier on overtemperature ΔT or low pressurizer pressure. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

Results

The system response to an inadvertent opening of a pressurizer safety valve is shown on Figures 15.6-1 and 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on overtemperature ΔT. The pressure decay transient and average temperature transient following the accident are given in Figure 15.6-1 and Figure 15.6-2, respectively. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-2. The DNBR remains above the safety analysis limit throughout the transient. The DNBR design basis is described in Section 4.4.

The calculated sequence of events for the inadvertent opening of a pressurizer safety or relief valve incident is shown on Table 15.6-1.

15.6.1.3 Conclusions

The results of the analysis show that the pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against the RCS depressurization event.

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

There are no instrument lines connected to the RCS that penetrate the containment. There are, however, the grab sample lines from the hot legs of reactor coolant loops 1 and 3, from the steam and liquid space of the pressurizer, and from the 3-inch chemical and

volume control system letdown line penetrating the containment. The grab sample lines are provided with normal closed isolation valves on both sides of the containment wall and are designed in accordance with the requirements of GDC-55.

The most severe pipe rupture with regard to radioactivity release during normal plant operation is rupture of the chemical and volume control system letdown line at a point outside of the containment. For such a break, the reactor coolant letdown flow would have passed sequentially from the cold leg and through the regenerative heat exchanger and letdown orifices. The letdown orifice reduces the letdown line pressure from 2,235 psig to less than 600 psig outside containment during normal plant operation when letdown flow is maintained at a nominal 120 gpm. Increase in flow will occur due to a rupture of the letdown line downstream of the orifices. It has been determined that the occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of 222 gpm.

Since the reactor makeup water transfer pumps are designed to deliver 120 gpm to the boric acid blending tee, the capability of the reactor makeup system can not maintain VCT level. The imbalance of the VCT outflow and inflow would eventually result in water level dropping to VCT level Lo/Refueling Water Sequence setpoint (5%) and the suction of the charging pump would automatically be shifted from the VCT to the RWST. In addition, the calculated releases rate is beyond the capacity of a single charging pump, which is capable of delivering 150 gpm flow to the RCS under normal operating conditions, if the RCP seal leakoff and any identified leakage are accounted for. The control room operators would be alerted of the failure by a high charging flow alarm and/or continuous VCT makeup and a slowly decreasing VCT and pressurizer level. The high charging flow alarm procedure in conjunction with the plant off-normal procedure for high RCS leakage would require letdown isolation inside the containment which would terminate the coolant loss.

15.6.2.1 Radiological Consequences

15.6.2.1.1 Method of Analysis

15.6.2.1.1.1 Physical Model

The volatile fractions of the spilled reactor coolant are assumed to be available for immediate release to the environment.

15.6.2.1.1.2 Assumptions and Conditions

The major assumptions and parameters used in the analysis are provided in Table 15.6-2 and summarized below:

a. Two different cases are examined with regards to initial reactor coolant activity. Both cases assume an initial reactor coolant iodine activity equal to the dose equivalent of 1.0 microCi/gm of I-131. The first case, referred to as a pre-accident iodine spike, assumes a reactor transient occurs prior to the postulated LLB event that has raised the iodine concentration to the maximum value allowed by TS LCO 3.4.16, 60 microCi/gm of DE I-131. The second case, referred to as an accident initiated iodine spike, assumes an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 500 immediately after the accident occurs. This increased escape rate is assumed for 8 hours.

b. The noble gas activity in the reactor coolant is based on 500 microCi/gm of Xe-133 dose equivalent.

c. The alkali metal activity in the reactor coolant is based on 1-percent failed fuel as provided in Table 11.1-5.

d. A flow rate of 444 gpm of reactor coolant is spilled onto the floor of the auxiliary building. (Based on doubling the maximum flowrate of 222 gpm to account for backflow over a thirty minute release, followed by a ten second valve closure period).

e. All of the noble gases in the spilled reactor coolant are released to the environment.

f. Eighteen percent of the spill is assumed to flash based on letdown line fluid conditions of 380°F and 600 psig and saturation at atmospheric pressure. All of the iodine and alkali metal activity in the flashed fraction of the spill is assumed to be released.

g. No credit is taken for mixing and holdup of the releases within the auxiliary building, nor are the auxiliary building normal exhaust filters credited with reducing the release. That is, the release is modeled as being direct to the environment.

h. Control room isolation (i.e. the initiation of emergency mode HVAC flows and filtration) was credited following a high radiation signal.

15.6.2.1.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3 and provided in Table 15A-2.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.6.2.1.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The reactor coolant spilled in the auxiliary building will collect in the floor drain sumps. From there, it will be pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant.

Normally, gases released in the auxiliary building mix with the building atmosphere and are gradually exhausted through the filtered building ventilation system. The charcoal filters normally remove a very large fraction of the airborne iodine in the building atmosphere. However, the ventilation system is not designed to mitigate the consequences of an accident (e.g., it

might not survive an earthquake more severe than the operating-basis earthquake), nor can the possibility of unplanned leakages from the auxiliary building be eliminated; hence, no credit is taken for these effects reducing the released activity.

The evaporated radionuclides are assumed to be available immediately to the outside atmosphere.

15.6.2.1.2 Identification of Uncertainties and Conservatisms in the Analysis

The principal uncertainties in the calculation of doses following a letdown line rupture arise from the unknown extent of reactor coolant contamination by radionuclides, the quantity of coolant spilled, the fraction of the spilled activity that escapes the auxiliary building, and the environmental conditions at the time. Each of these uncertainties is treated by taking worst-case or extremely conservative assumptions.

The extent of coolant contamination assumed greatly exceeds the levels expected in practice. The rupture is postulated in a seismic Category I, ASME Section III, Class 2 piping system. It is assumed that the leak goes undetected for 30 minutes. It is expected that considerable holdup and filtration occurs in the auxiliary building, but no credit is assumed.

The purpose of all these conservatisms is to place an upper bound on doses.

15.6.2.1.3 Conclusions

15.6.2.1.3.1 Filter Loadings

No filter is credited with the collection of radionuclides in this accident analysis. The buildup on these filters (auxiliary building and control building charcoal filters) that may be expected due to the adsorption of some of the iodine and alkali metal is very small compared with the design capacity of these filters.

15.6.2.1.3.2 Dose to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated letdown line rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, for the duration of the accident at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.6-3. The resultant doses are within a small fraction of the guideline values of 10 CFR 50.67 for the offsite locations and the full 10 CFR 50.67 guideline value in the control room.

15.6.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

The steam generator tube rupture (SGTR) examined is the complete severance of one single steam generator tube which results in the leakage of reactor coolant into the secondary side of the steam generator. The consequences of SGTR depend largely upon the ability of the operator to take the necessary actions to terminate the primary to secondary leakage.

Two SGTR scenarios are evaluated in order to ensure that operators can respond to the accident in a timely manner so as to minimize the resulting offsite

releases and prevent overfilling of the effected steam generator. Those scenarios are described below:

Margin to Overfill Scenario:

The margin to overfill SGTR scenario assumes failures in the auxiliary feedwater (AFW) system such that if operators do not respond quickly enough to terminate feedwater, then the affected steam line may fill with water. The specific failure involves the control valve on the discharge side of the motor driven feedwater pump feeding the ruptured steam generator. To maximize flow to the ruptured steam generator, it is assumed that this valve fails in the wide-open position. Failure of this valve, coupled with the contribution from the turbine driven AFW pump, has the potential for overfilling the ruptured steam generator and subsequently, relieve water via a safety valve.

Stuck-open ARV Scenario:

The stuck-open ARV scenario assumes that the ruptured steam generator's atmospheric relief valve (ARV) fails open upon ruptured steam generator isolation and then remains stuck open for 30 minutes. The basis for this time period includes consideration of the time the valve is stuck open until the time that an operator manually closes the associated block valve. Failure of the ruptured steam generator's ARV thus maximizes offsite release by assuming a direct path to the atmosphere.

The above cases are considered to be ANS Condition IV events, a limiting fault (USAR Section 15.0.1). It has been determined that the most severe radiological consequences will result from the scenario with a stuck-open ARV on the ruptured steam generator. Therefore, the radiological consequences are reported for that limiting case only.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-24.

15.6.3.1 STEAM GENERATOR TUBE RUPTURE MARGIN TO OVERFILL

15.6.3.1.1 Identification of Cause and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to an increase in the contamination of the secondary system due to the leakage of radioactive coolant from the RCS. Loss of off-site power is assumed to occur coincident with reactor trip. Discharge of activity to the atmosphere takes place via the steam generator safety and/or atmospheric relief valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

Following the occurrence of the SG tube rupture, the primary to secondary leakage causes the pressurizer level and the RCS pressure to decrease. As the RCS pressure continues to decrease, automatic reactor trip occurs on low pressurizer pressure or over-temperature delta-T(OTΔT) signal. Because of the assumed loss of offsite power, the steam dump system will not be available, and the secondary side pressure increases rapidly after reactor trip until the steam generator ARVs and/or SV lift to dissipate the energy. After reactor trip, the RCS pressure continues to decrease and the safety injection is automatically initiated on low pressurizer pressure signal. Due to the assumed loss of offsite power at the reactor trip, normal feedwater flow is terminated and the AFW is initiated.

The analysis assumes failure of the AFW control valve on the discharge side of the motor-driven AFW pump feeding the ruptured steam generator. It is assumed that this valve fails in the wide-open position to maximize the flow to the ruptured steam generator. Failure of this valve coupled with the contribution from the turbine-driven AFW pump has a greater potential for overfilling the ruptured steam generator.

The operator is expected to determine that a SGTR has occurred and to identify and isolate the ruptured steam generator on a restricted time scale to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured steam generator. The recovery procedure then can be carried out on a time scale to ensure that break flow to the secondary system is terminated.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed so that pressure equalization between primary and secondary side of the ruptured steam generator can be achieved to stop the break flow.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a design basis tube rupture:

a. Pressurizer low pressure and low level alarms are activated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side, there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the ruptured steam generator is reduced due to the additional break flow being supplied to that loop.

b. Decrease in RCS pressure (Figure 15.6-3a) due to continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or over-temperature Delta T. It is conservatively postulated that loss of offsite power (LOOP) occurs when the reactor trips.

The LOOP signal initiates auxiliary feedwater addition. Resultant plant cooldown (Figures 15.6-3b and 15.6-3c) following reactor trip leads to a rapid change of pressurizer level (Figure 15.6-3f), and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. With the loss of offsite power occurring at the reactor trip, normal feedwater is terminated and the auxiliary feedwater is being supplied to the steam generators.

c. The steam generator blowdown liquid monitor alarm and/or the condenser air discharge radiation monitor alarm will activate, indicating a sharp increase in radioactivity in the secondary system. The alarms automatically cause termination of steam generator blowdown.

d. The reactor trip automatically trips the turbine, and if offsite power is available the steam dump valves open, permitting steam dump to the condenser. In the event of a coincident loss of offsite power, as assumed in the transient presented in this section, the steam dump valves would automatically close to protect the condenser. The steam generator pressure (Figure 15.6-3d) would increase, resulting in steam discharge to the atmosphere through the steam generator safety/atmospheric relief valves.

e. Following reactor trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs the decay heat.

f. Safety injection flow results in increasing the pressurizer water level (Figure 15.6-3f); the rate of which depends upon the amount of operating auxiliary equipment.

15.6.3.1.2 Analysis of Effects and Consequences

Method of Analysis

An analysis was performed to determine the thermal and hydraulic transient for a postulated SGTR, which results in margin to steam generator overfill. The analysis was performed using the RETRAN-02 computer code (Reference 3) following the methodology of WCAP-10698-P-A (Reference 10).

In estimating the margin to ruptured SG overfill, the following assumptions are made:

a. Reactor trip occurs automatically as a result of low pressurizer or overtemperature Delta T. Loss of offsite power occurs at reactor trip.

b. Auxiliary feedwater flow rate is allowed to vary with the fluctuation in the ruptured steam generator pressure. This results in higher AFW flow when the ruptured steam generator pressure decreases. Once the ruptured steam generator level reaches 15% NRS, the AFW flow from the turbine driven AFW pump to the ruptured steam generator is terminated. Thirty seconds after the AFW flow from the turbine driven AFW pump is isolated to the ruptured steam generator, the AFW flow from the motor-driven AFW pump to the ruptured steam generator is terminated. AFW flow to the intact steam generators maintains the narrow range level between 6% and 50% as indicated in the Emergency Operating Procedure EMG E-3.

c. Cooldown of the RCS is initiated at 23 minutes following reactor trip. It is assumed that steam is released through the remaining three operable steam generator ARVs in the intact loops until the RCS temperature of the core exit thermal couples corresponds to the ruptured SG pressure as listed in the EMG E-3 procedure. Note that the analysis assumes that the operators will continue to maintain the plant at that temperature for the duration of the transient, consistent with the EMG E-3 procedure. Technical Specification LCO 3.7.4 requires that all four steam generator ARVs shall be operable. With one of the required ARVs unavailable due to its association with the ruptured steam generator, the remaining three ARVs are available to ensure that subcooling can be achieved for the RCS.

d. Following termination of the RCS cooldown, the RCS is depressurized by opening a pressurizer PORV to assure an adequate coolant inventory prior to terminating SI flow. Primary depressurization is initiated at 5 minutes following the termination of the RCS cooldown and continues until the RCS pressure is less than the ruptured steam generator pressure.

e. Following the depressurization, termination of SI is delayed to ensure enough liquid enters the pressurizer to maintain inventory and preserve subcooling margin. It is assumed that 4 minutes following the termination of the RCS depressurization that the safety injection flow is terminated.

The above assumptions are conservatively made to minimize the margin to ruptured SG overfill.

Key Recovery Sequence

The recovery sequence to be followed consists of the following major operator actions:

a. Identification of the ruptured steam generator

b. Isolation of the ruptured steam generator

c. Assuring subcooling of the RCS fluid to approximately 50°F below the ruptured steam generator temperature

d. Depressurization of the RCS to a value equal to the ruptured steam generator pressure

e. Subsequent termination of safety injection flow

Results

In Table 15.6-1, the sequence of event are presented. These events include postulated operator response times and normal plant response to the normal plant setpoints. Primary and secondary system parameters are plotted as a function of time in Figures 15.6-3a through 15.6-3j. Margin to steam generator overfill was demonstrated.

15.6.3.2 STEAM GENERATOR TUBE RUPTURE THERMAL & HYDRAULIC

15.6.3.2.1 Identification of Causes and Accident Description

The accident description for this SGTR is identical to that discussed in Section 15.6.3.1.1 with the exception that instead of the motor-driven AFW pump control valve failing to close, the ARV for the ruptured steam generator is assumed to fail open for 30 minutes.

In this SGTR scenario, the operator is expected to determine that a SGTR has occurred, to recognize the failure of the ARV, to dispatch personnel to manually isolate the ARV, and then to isolate the affected steam generator. In this accident the failure of the ARV allows a longer period for radionuclide release than otherwise expected.

The accident involves the complete severance of a single steam generator tube with Loss of Offsite Power coincident with reactor trip. For a discussion of normal operation of plant control systems to a design basis SGTR refer to Section 15.6.3.1.1.

15.6.3.2.2 Analysis of Effects and Consequences

Methods of Analysis

An analysis was performed to determine the thermal and hydraulic transient for a postulated SGTR, which assumes failure of the ruptured steam generator's ARV. The amount of primary to secondary leakage and the mass releases to the atmosphere from both the ruptured and intact steam generators were determined from the occurrence of the tube rupture until break flow termination. The SGTR recovery actions to cooldown and depressurize the plant to cold shutdown were also simulated in the analysis. The analysis was performed using the RETRAN-02 computer code (Reference 3) following the methodology of Supplement 1 to WCAP-10698-P-A (Reference 11).

In estimating the mass transfer from the RCS through the broken tube, the following assumptions are made:

a. Reactor trip occurs automatically as a result of low pressurizer pressure or overtemperature Delta T. Loss of offsite power occurs at reactor trip.

b. As pressures rise on the secondary side, the steam generator atmospheric relief valves (ARVs) open to release excess secondary pressure. The ARV for the ruptured steam generator is assumed to fail open and steam release to continue for 30 minutes until the ARV block valve is manually closed.

c. AFW is initially delivered at a rate of 285 gpm to each steam generator. AFW is maintained to assure that narrow range level in each steam generator exceeds 6%.

d. Following ruptured steam generator isolation, cooldown is initiated 10 minutes from closure of the failed open ARV block valve. Cooldown continues until the temperature of the core exit thermocouples corresponds to the ruptured SG pressure as listed in the EMG E-3 procedure.

e. Reactor coolant system depressurization is initiated 5 minutes after completion of cooldown.

f. After primary side depressurization is completed and SI termination criteria are met, a 4 minute time delay is assumed prior to SI termination.

g. Following SI termination, the operators equalize pressure in the RCS and ruptured SG. During this time break flow in the ruptured SG continues. After pressures are equalized, it is conservatively assumed that the transition to cold shutdown is made utilizing steam release to the atmosphere from the ruptured SG.

Key Recovery Sequence

The recovery sequence to be followed consists of the following major operator actions:

a. Identification of the ruptured steam generator

b. Identification and manual isolation of the stuck-open atmospheric relief valve

c. Isolation of the ruptured steam generator

d. Assuring subcooling of the RCS fluid

e. Depressurization of the RCS to a value equal to the ruptured steam generator pressure

f. Subsequent termination of safety injection flow

g. Further cooldown and depressurization of the RCS to conditions suitable for RHR initiation.

Results

In Table 15.6-1, the sequence of events are presented. These events include postulated operator response times and normal plant responses to the normal plant setpoints. Primary and secondary system parameters are plotted as a function of time in Figures 15.6-3k through 15.6-3x.

15.6.3.3 Radiological Consequences

15.6.3 3.1 Method of Analysis

The analysis of the radiological consequences of an SGTR considers the most severe release of secondary activity, as well as primary system activity leaked from the tube break. The inventory of iodine, alkali metals, and noble gas fission product activity available for release to the environment depends on the tube rupture break flow and intact steam generator primary to secondary coolant leakage rate, activity initially in the primary and secondary systems, and the mass of steam discharged to the environment.

The analysis of the radiological consequences of a postulated SGTR uses the guidelines provided in Regulatory Guide 1.183 (Reference 7) for assumptions and methods to calculate the offsite and control room dose consequences.

15.6.3.3.1.1 Physical Model

The evaluation of the radiological consequences of a postulated steam generator tube rupture (SGTR) utilizes the results of the RETRAN analyses to calculate releases of radioactive iodines, alkali metals, and noble gases to the atmosphere to the time of RHR cut-in conditions.

Concentrations of radioactivity in the RCS water and in the ruptured and intact steam generators are calculated utilizing release rates from the fuel, calculated mass flows and conventionally used partitioning coefficients between the liquid and steam phases. These radioactivity concentrations and the calculated releases of mass to the atmosphere yield the released activity. Radiological consequences are calculated using atmospheric dispersion coefficients, breathing rates, and other aspects of conventional radiological consequence calculations.

15.6.3.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6-4 and 15A-1 and are summarized below.

Tables 15B-1 and 15B-5 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

a. The assumed reactor coolant iodine activity is determined for the following two cases:

Case 1 - An initial reactor coolant iodine activity equal to the dose equivalent of 60.0 μCi/gm of I-131 due to a pre-accident iodine spike caused by RCS transients prior to the SGTR.

Case 2 - An initial reactor coolant iodine activity equal to the dose equivalent of 1.0 μCi/gm of I-131 with an iodine spike that increases the escape rate from the fuel into the coolant by a factor of 335 immediately after the accident. This increased escape rate is assumed for 8 hours.

b. The noble gas activity in the reactor coolant is based on the dose equivalent of 500 microCi/gm of Xe-133.

c. The initial secondary coolant iodine activity is based on the dose equivalent of 0.1 microCi/gm of I-131.

d. The alkali metal activity in the reactor coolant is based on 1-percent failed fuel as provided in Table 11.1-5.

e. The initial secondary coolant alkali metal activity is based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity).

The following assumptions and parameters are used to calculate the activity released and the offsite and control room doses following an SGTR:

a. The amount of discharge or reactor coolant in the secondary system as a function of time is as calculated by RETRAN analysis. The total break flow to the ruptured steam generator is listed in Table 15.6-4.

b. It is assumed that all of the iodine and alkali metals in the fraction of reactor coolant that flashes to steam upon reaching the secondary side isreleased to the steam phase. No credit is taken for scrubbing.

c. A 1-gpm primary-to-secondary leakage is assumed to occur to the intact steam generators, through the accident sequence.

d. All noble gas activity in the reactor coolant that is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is released to the atmosphere.

e. The iodine partition fraction between the liquid and steam in the steam generator is assumed to be 0.01.

f. The full power moisture carryover of 0.25% is utilized to determine the alkali metal releases to the environment via steam venting from the steam generators.

g. The steam releases from the steam generators to the atmosphere are as calculated by RETRAN analysis and given in Table 15.6-4.

h. Radioactivity releases to the atmosphere are based on the concentrations of radioactivity in the steam phase times the calculated amounts of steam release.

i. No additional radioactivity releases occur after the initiation of RHR system cooling.

j. Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered.

k. Short-term accident atmospheric dispersion factors, breathing rates, and dose conversion factors are provided in Tables 15A-2, 15A-1, and 15.6-4, respectively.

l. Control room isolation is credited to occur at 60 seconds.

15.6.3.3.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.
b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurements program, as described in USAR Section 2.3, and are provided in Table 15A-2.
c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.6.3.3.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

The activity released from the ruptured steam generator, is released directly to the environment by the atmospheric relief valves. The intact steam generators discharge steam and entrained activity via the safety and atmospheric relief valves until the time that initiation of the RHR system can be accomplished. Since the activity is released directly to the environment with no credit for plateout or fall out, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resulting radiological consequences represent a conservative estimate of the potential integrated dose to the postulated SGTR.

15.6.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

a. Reactor coolant activities based on extreme iodine spiking effects are conservatively high.

b. The assumed 1-gpm steam generator primary-to-secondary leakage is greater than that anticipated during normal operation.

c. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption, since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one or more minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceeds the limits established in the technical specifications would lead to reactor shutdown. Therefore, it is unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

d. The coincident loss of offsite power with the occurrence of an SGTR is a conservative assumption. In the event of the availability of offsite power, the condenser dump valves will open, permitting steam dump to the condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the intact steam generators.

e. The radiological consequences have been based on a worst-case scenario, i.e., a stuck open ARV on the ruptured steam generator.

f. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

g. Reactor coolant iodine and noble gas activities are based on the Technical Specification limits which are significantly higher than the reactor coolant activities associated with normal operating conditions.

15.6.3.3.3 Conclusions

15.6.3.3.3.1 Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the steam generator tube rupture is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the LOCA, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated SGTR accident releases.

15.6.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated steam generator tube rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses have been analyzed for the 0-2 hour dose at the exclusion area boundary, for the duration of the accident (0 to 12 hours) at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.6-5. The resultant offsite doses are within the guideline values of 10 CFR 50.67 for the case with an assumed pre-accident iodine spike and are within a small fraction of the guideline values (i.e., 10%) of 10 CFR 50.67 for the case with an accident-initiated iodine spike. The resultant control room doses are within the guideline value of 10 CFR 50.67.

15.6.3.4 Conclusions

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneously loss of offsite power.

15.6.3.5 REFERENCES

1. Not Used

2 Not Used

3. Huegel, D. S., et. al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

4. Not Used.

5. Stewart, C. W., et. al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle, Pacific Northwest Laboratories, EPRI NP-2511-CCM-A, August 1989.

6. Not Used

7. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

8. Not Used

9. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

10. Lewis, R. N., et al., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A, August 1987.

11. Lewis R. N., et al., "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10698-P-A, March 1986.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to WCGS.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analysis reported here, a small break is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. A small break LOCA is classified as an ANS Condition III event (an infrequent fault), as defined in Section 15.0. A major break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the life of a plant but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 as follows:

A. The calculated peak fuel element clad temperature shall not exceed the requirement of 2200°F.

B. The amount of fuel element cladding that reacts chemically with water or steam to generator hydrogen, shall not exceed 1 percent of the total amount of Zircaloy in the fuel cladding.

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

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 a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975) Reference 2) presents a study in regards to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (Less than 1.0 $ft.^2$) yield results with more margin to the Acceptance Criteria Limits than large breaks.

Major plant systems and equipment available for mitigation of transient and accident conditions are discussed in Section 15.0.8 and listed in Table 15.0-6. The actual parameters assumed and results are given in the respective section/subsection. Table 15.0-6 incorporates events shown in the sequence diagram, Figure 15.0-25.

15.6.5.2 Sequence of Events and Systems Operations

Should a major pipe break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated and verified on a cycle-specific basis to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis, but is credited in the post-LOCA subcriticality evaluation. Details on the method for assuring post-LOCA subcriticality is discussed in Section 15.6.5.3.4.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of the Large Break LOCA Transient

Before the break occurs, the RCS (Reactor Coolant System) is assumed to be operating normally at full power in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. A large break is assumed to open instantaneously in one of the main RCS cold leg pipes.

Immediately following the cold leg break, a rapid system depressurization occurs along with a core flow reversal due to a high discharge of sub-cooled fluid into the broken cold leg and out of the break. The fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up, while the core power decreases due to voiding in the core. The hot water in the core, upper plenum, and upper head flashes to steam, and subsequently the cooler water in the lower plenum and downcomer begins to flash. Once the system has depressurized to the accumulator pressure, the accumulator begins to inject cold borated water into the intact cold legs. During the blowdown period, a portion of the injected Emergency Core Cooling System (ECCS) water is calculated to be bypassed around the downcomer and out of the break. The bypass period ends as the system pressure continues to decrease and approaches the containment pressure, resulting in reduced break flow and consequently, reduced core flow.

As the refill period begins, the core continues to heat up as the vessel begins to fill with ECCS water. This phase continues until the lower plenum is filled, the bottom of the core begins to reflood, and entrainment begins.

During the reflood period, the core flow is oscillatory as ECCS water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. The pumped cold leg injection ECCS water aids in the filling of the vessel and downcomer, which subsequently supplies water to maintain the core and downcomer water levels and complete the reflood period.

The time sequence of events for the PCT limiting Large Break LOCA case is given in Table 15.6-10.

Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for a small-break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in *10 CFR 50.46 and Appendix K of 10 CFR 50 (Reference 10).*

Large Break LOCA Evaluation Model

- General:

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in 10 CFR 50.46 (Reference 10), both the Nuclear Regulatory Commission (NRC) and the industry recognized that stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling System (ECCS) would be conservatively underestimated, resulting in predicted peak cladding temperatures (PCT's) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models required in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed.
2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 (Reference 15). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models", to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the LOCA analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 (Reference 16).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 24). This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis (Figure 15.6-5).

A LOCA evaluation methodology for three and four loop pressurized water reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and has been approved by the NRC (Reference 6).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical Treatment of Uncertainty Method (Reference 9). This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing applications in WCAP-16009-P-A (Reference 9). The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile (with 95% confidence level) of the Peak Cladding Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) to satisfy the 10 CFR 50.46 criteria.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A (Reference 9) as applicable to the ASTRUM methodology. Section 13-3 of WCAP-16009-P-A (Reference 9) was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report (SER) appended to this Topical Report. The Best-Estimate LBLOCA analysis and associated model for Wolf Creek is plant-specific.

The Wolf Creek ASTRUM LBLOCA uses a plant-specific adaptation of the ASTRUM methodology that includes explicit modeling of fuel thermal conductivity degradation (TCD), as well as a larger range for rod internal pressure (RIP) uncertainty. Additionally the ASTRUM LBLOCA analysis was performed using heat transfer multipliers different than those used in the as-approved ASTRUM methodology. These heat transfer multiplier distribution changes have been reported to the NRC in Reference 29.

- Development of WCOBRA/TRAC Computer Code:

The COBRA/TRAC code was originally developed at Pacific Northwest Laboratory (Reference 1) by combining the COBRA-TF code (Reference 8) and the TRAC-PD2 codes (Reference 3). The COBRA-TF code, which has the capability to model three-dimensional flow behavior in a reactor vessel, was incorporated into

TRAC-PD2 to replace its vessel model. TRAC-PD2 is a systems transient code designed to model all major components in the primary system. Westinghouse continued the development and validation of COBRA/TRAC through the FLECHT-SEASET program (Reference 4). To allow the code to model phenomena specific to plants with upper plenum injection, Westinghouse made modifications and improvements to COBRA/TRAC and renamed the new version WCOBRA/TRAC Mod 4 (Reference 5). Westinghouse continued the development of WCOBRA/TRAC to extend its application to three- and four-loop Westinghouse Pressurized Water Reactors (PWR's) with cold-leg injection with version MOD7A, Revision 1. This computer code is described in detail in a WCAP-12945-P-A, referred to as the Code Qualification Document (CQD) (Reference 6) as well as WCAP-16009-P-A, which describes the ASTRUM methodology (Reference 9).

WCAP-16009-P-A (Reference 9) states that the ASTRUM methodology is based on the frozen version of WCOBRA/TRAC MOD7A, Revision 6. WCOBRA/TRAC MOD7A, Revision 8 was used for the execution of the ASTRUM Uncertainty Studies for Wolf Creek. The confirmatory analysis (paragraph "Determination of Plant Operating Conditions") was executed with WCOBRA/TRAC MOD7A Revision 7.

The Nuclear Regulatory Commission (NRC) approved Westinghouse Best-Estimate Loss-of-Coolant Accident (BELOCA) ASTRUM methodology (Reference 9) based on the PAD 4.0 fuel performance code (Reference 28). PAD 4.0 was licensed without explicitly considering fuel thermal conductivity degradation (TCD) with burnup. Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures (pellet radial average temperature) as well as other fuel performance related effects beyond beginning-of-life. Since PAD provides input into the large-break LOCA analysis, this will tend to increase the stored energy at the beginning of the simulated large-break LOCA event. This in turn leads to an increase in Peak Cladding Temperature (PCT) if there is no provision to credit off-setting effects. In addition, a different fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT was used to more accurately model the fuel temperature profile when accounting for TCD.

In order to mitigate the impact of the increasing effect of pellet TCD with burnup, the large-break LOCA evaluation of second/third cycle fuel utilizing reduced peaking factors from those shown in USAR Table 15.6-9. The reduced peaking factors are limited to the following application: burndown credit for the hot rod and hot assembly is taken for higher burnup fuel in the second/third cycle of operation. The Wolf Creek peaking factor values utilized in this analysis are shown in Table 15.6-9B. Note that the beginning to middle of life values are the Table 15.6-9 values.

Evaluation of the fuel in its second/third cycle of irradiation is beyond the first cycle considered in the approved ASTRUM Evaluation Model (EM), but was considered in the analysis when explicitly modeling TCD to demonstrate that conformance to the acceptance criteria is met for the second/third cycle fuel.

In addition to the standard uncertainty calculations, the Wolf Creek LBLOCA analysis sampled a larger rod internal pressure (RIP) uncertainty than originally included in the ASTRUM methodology (Reference 9). It was discovered that the as-approved sampling range did not bound the plant-specific rod internal pressure uncertainties for Wolf Creek. Therefore, the approved sampling range was expanded to bound the Wolf Creek plant-specific data.

WCOBRA/TRAC combines two-fluid, three-field and multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1. Ability to model transient three-dimensional flows in different geometries inside the vessel
2. Ability to model thermal and mechanical non-equilibrium between phases
3. Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes

4. Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ. Dividing the liquid phase into two fields is a convenient and physically accurate way of handling flows where the liquid can appear in both film and droplet form. The droplet field permits more accurate modeling of thermal-hydraulic phenomena, such as entrainment, de-entrainment, fallback, liquid pooling, and flooding.

WCOBRA/TRAC also features a two-phase, one-dimensional hydrodynamic formulation. In this model, the effect of phase slip is modeled indirectly via a constitutive relationship that provides the phase relative velocity as a function of fluid conditions. Separate mass and energy conservation equations exist for the two-phase mixture and for the vapor.

The reactor vessel is modeled with the three-dimensional, three-field model, while the loops, major loop components, and safety injection points are modeled with the one-dimensional model.

All geometries modeled using the three-dimensional model are represented as a matrix of cells. The number of mesh cells used depends on the degree of detail required to resolve the flow field, the phenomena being modeled, and practical restrictions such as computing costs and core storage limitations.

The equations for the flow field in the three-dimensional model are solved using a staggered difference scheme on the Eulerian mesh. The velocities are obtained at mesh cell faces, and the state variables (e.g., pressure, density, enthalpy, and phasic volume fractions) are obtained at the cell center. This cell is the control volume for the scalar continuity and energy equations. The momentum equations are solved on a staggered mesh with the momentum cell centered on the scalar cell face.

The basic building block for the mesh is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

One-dimensional components are connected to the vessel. The basic scheme used also employs the staggered mesh cell. The one-dimensional components consist of all the major components in the primary system, such as pipes, pumps, valves, steam generators, and the pressurizer. The one-dimensional components are represented by a two-phase, five-equation, drift flux model. This formulation consists of two equations for the conservation of mass, two equations for the conservation of energy, and a single equation for the conservation of momentum. Closure for the field equations requires specification of the interphase relative velocities, interfacial heat and mass transfer, and other thermodynamic and constitutive relationships.

- Methodology:

WCAP-16009-P-A (Reference 9) provides ASTRUM methodology and also includes a description of the code models and their implementation. Volumes II and III of the CQD (Reference 6) presented a detailed assessment of the computer code WCOBRA/TRAC through comparisons to experimental data. From this assessment, a quantitative estimate was obtained of the code's ability to predict peak clad temperatures (PCT's) in a PWR large-break loss-of-coolant accident (LOCA). Modeling of a PWR introduced additional uncertainties, which were identified and discussed in Section 21 of the CQD Volume IV (Reference 6). A list of key

LOCA parameters was compiled as a result of these studies. Models of several PWRs were used to perform sensitivity studies and establish the relative importance of these parameters. The final step of the best-estimate methodology, in which all the important uncertainties of the LOCA parameters are accounted for to estimate PCT, local maximum oxidation (LMO), and core-wide oxidation (CWO) at 95-percent probability. The methodology is summarized below:

- Plant Model Development

  In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used to ensure an accurate simulation of the transient. Specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, with some plant-specific modeling dictated by hardware differences such as in the upper plenum of the reactor vessel or the Emergency Core Cooling System (ECCS) injection configuration.

- Determination of Plant Operating Conditions

  In this step, the expected or desired operating range of the plant to which the analysis is to be applied is established using information supplied by the utility. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient." Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred as the confirmatory analysis. The typical four sets of parameters evaluated in the confirmatory analysis are: High average temperature (Tavg) versus Low Tavg, High Steam Generator Tube Plugging (SGTP) level versus Low SGTP, High relative power in the low power peripheral assemblies (PLOW) versus Low PLOW, and Loss of Offsite Power (LOOP) versus no-LOOP. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainty. The confirmatory configuration analysis was performed previous to the ASTRUM uncertainty analysis prior to the identification of the TCD issue and associated PAD data. However, since the effects of TCD, analysis input changes (i.e., safety injection flow), and code version change (i.e., WC/T MOD7A, Revision 7 to WC/T MOD7A, Revision 8) were judged to have a negligible impact on the confirmatory study results, the limiting plant configuration (referred to as the Reference Transient) was judged to remain the same.

- Assessment of Uncertainty

  The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of WCAP-16009-P-A (Reference 9). The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, and CWO).

  The uncertainty contributors are sampled randomly from their respective distribution for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each WCOBRA/TRAC calculation, includes initial conditions, power distributions, and model uncertainties. The time in the cycle, break type (split or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level.

- Plant Operating Range

  The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range established in step 2, or may be narrower for some parameters to gain additional margin.

Small Break LOCA Evaluation Model

The NOTRUMP and LOCTA-IV computer codes are used in the analysis of LOCA due to small breaks in the RCS. The NOTRUMP computer code is a one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Also, safety injection into the broken loop is modeled using the COSI condensation model (Reference 13). The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plant." In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. the broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in Reference 11, 12 and 13.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Cladding thermal analyses are performed with the LOCTA-IV (Reference 7) code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

Figure 15.6-47 presents the hot rod axial power shape utilized to perform the small-break analysis. This power shape represents a distribution of power versus core height and was chosen because it maximizes the integral power at the top half of the core where peak cladding temperature (PCT) occurs and minimizes the power in the bottom half of the core, which reduces liquid swell due to void formation thus providing a deeper core uncovery. This power shape is skewed to the top of the core with the peak local power occurring at about the 10-foot core elevation.

This is limiting for the small-break analysis, because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two-phase mixture height, remain low. The peak clad temperature occurs above 10 feet.

Schematic representation of the computer code interface is given in Figure 15.6-6.

The small-break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model with NOTRUMP (References 7, 11, 12, and 13).

15.6.5.3.2 Input Parameters and Initial Conditions

Large Break Analysis Input Parameters:

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 15.6-9 summarizes the operating ranges for Wolf Creek as defined for the proposed operating conditions, which are supported by the Best-Estimate LBLOCA analysis. Table 6.2.1-65 summarizes the LBLOCA containment data used for calculating containment pressure. If operation is maintained within these ranges, the LBLOCA results developed in this report are considered to be valid. Note that some of these parameters vary over their range during normal operation within a fuel cycle (e.g., accumulator temperature) and other parameters are typically fixed during normal operation within a fuel cycle (full-power Tavg).

Small Break Specific Input Parameters and Initial Conditions

Significant input parameters are given in Table 15.6-9A.

For the small break LOCA (SBLOCA) analysis, loss of offsite power (LOOP) is assumed, which results in the limiting single failure assumption of the loss of one Emergency Diesel Generator (EDG) and a subsequent loss of one train of pumped Emergency Core Cooling System (ECCS). The SBLOCA analysis assumes that reactor trip occurs coincident with the LOOP, which results in the following: (a) Reactor Coolant Pump (RCP) trip and coastdown and (b) Steam Dump System being inoperable.

The SBLOCA analysis is performed with a high nominal vessel average temperature (Tavg) value of 588.4°F and minimum auxiliary feedwater (AFW) flow rate of 210 gpm/SG. The high Tavg value chosen is evaluated to be applicable over the range of nominal vessel average temperature values (570.7°F to 588.4°F) and the minimum AFW flow rate is evaluated to be applicable over a range of AFW flow rates (210-250 gpm/SG).

The safety injection (SI) system consists of gas pressurized accumulator tanks and pumped injection systems. The accumulators are modeled to inject borated water into the RCS when the RCS depressurizes to 583 psia. Minimum ECCS availability is assumed for the analysis from one centrifugal charging pump (CCP), one SI pump and one RHR pump. Assumed pumped SI characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in Table 15.6-14 and corresponding Figure 15.6-46 for break sizes less than 8.75 inches and Table 15.6-14A and corresponding Figure 15.6-46A for the 8.75 inch break case. For the break sizes less than 8.75 inches the broken loop safety injection flow is assumed to spill to RCS pressure and for the 8.75 inch break case the broken loop safety injection flow is assumed to spill to the containment back pressure of 0 psig. The ECCS was assumed to be delivering to the RCS 39 seconds after the generation of the SI signal. This delay includes time required for sensor response, diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with an accident.

15.6.5.3.3 Results

Large Break LOCA Results

The Wolf Creek PCT/LMO/CWO-limiting transients are double-ended cold leg guillotine break which analyzes conditions that fall within those listed in Table 15.6-9. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (WCAP-12945-P-A) (Reference 6).

- Engineering Inspection:

The Large Break LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. For a typical large break, the blowdown period can be divided into the critical heat flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood and long term cooling phases. The important phenomena occurring during each of these phases are discussed as follows. Plot results for the PCT limiting Transient are shown in Figures 15.6-7 through 15.6-21.

1. Critical Heat Flux (CHF) Phase:

Immediately following the cold leg rupture, the fluid discharged at the break is subcooled and high break flow rates are observed. During this early period the core flow reverses, the fuel rods go through Departure from Nucleate Boiling (DNB) and the cladding rapidly heats up while voiding in the core rapidly causes the fission power to drop. Figure 15.6-7 shows the HOTSPOT maximum cladding temperature and WC/T PCT in the core as a function of time, while Figure 15.6-8 provides the location of the peak cladding temperatures for the hot rod versus time. The regions of the RCS with the highest initial temperatures (core, upper plenum, and hot legs) begin to flash to steam during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 15.6-13 and 15.6-17, respectively).

2. Upward Core Flow Phase:

Heat transfer is improved as the lower plenum fluid is pulled into the core. This effect may be enhanced if the pumps are not degraded (thus for cases with offsite power available), or the break discharge rate is reduced because the fluid is saturated at the break. Figure 15.6-9 shows the break flow rates for the vessel side and pump side. If pumps are highly degraded (as is the case for this PCT/CWO limiting Transient since a Loss-of-Offsite power (LOOP) is modeled at beginning of transient), or the break flow is large, the cooling effect due to upward flow may not be significant. The intact loop remains in single-phase liquid flow for a few seconds (Figure 15.6-10), resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase and the pump head degrades.

3. Downward Core Flow Phase:

Downward flow in the core increases as the pump head continues to degrade (due to increased voiding and pump coastdown). The break flow then begins to dominate and pull flow down through the core, up the downcomer to the broken cold leg, and out the break. Figure 15.6-11 shows the vapor flow near the top of the core of the Hot Assembly Channel. While liquid and entrained liquid flows provide core cooling, the vapor flow near the top core best illustrates this phase of core cooling. This period may be enhanced by flow from the upper head. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs (Figure 15.6-14). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow and the downward core flow are reduced. The system pressure (Figure 15.6-12) reaches the containment pressure (Figure 15.6-21) at the end of this last period of the blowdown phase, around 21 seconds. During this phase, the core begins to heat up as the system reaches containment pressure and the vessel begins to fill with Emergency Core Cooling System (ECCS) water.

4. Refill Phase:

Following the blowdown phase, the core experiences a nearly adiabatic heat-up as the lower plenum fills with ECCS water (Figure 15.6-13). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This phase ends when the ECCS water fills the lower plenum (Figure 15.6-13) and enters the

core. At this time, a quench front is established and advances along the core and entrainment begins, with a resulting improvement in heat transfer. Figures 15.6-14, 15.6-15 and 15.6-16 show the liquid flows from the accumulator and the safety injection for an intact loop.

5. Reflood Phase:

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system (Figure 15.6-14). The nitrogen surge forces water into the core, which then boils, causing system re-pressurization and the lower core region begins to quench. The repressurization is illustrated by the increase in downcomer liquid level (Figure 15.6-17). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and continue throughout the reflood period. Figures 15.6-17 and 15.6-18 show the downcomer and core liquid levels, respectively. Figure 15.6-19 shows the vessel fluid mass. As the quench front progresses further into the core, the PCT location moves higher towards the top core region. Figure 15.6-8 shows the movement of the PCT location. At approximately 90 seconds, after the accumulator injection is terminated, the stored energy in the vessel structures causes the downcomer and lower plenum water to start boiling, as illustrated, for example, by the lower plenum collapsed liquid level in Figure 15.6-13. The boiling of the downcomer and liquid plenum water causes a reduction in the downcomer, core levels, and vessel mass as the two-phase mixture level swell pushes water out of the break (see Figure 15.6-9). The downcomer boiling phenomenon is heavily affected by the containment pressure (which influences the steam density and the saturation temperature of the downcomer and lower plenum). The effect of downcomer boiling can be noticed on the PCT transient, although the general trend of decreasing cladding temperature is maintained, and the cladding temperatures remain lower than the limiting values observed earlier during the reflood period. As the transient progresses, continued injection of ECCS water refills the downcomer and core, removes the vessel metal mass stored energy, and leads to an increase in the vessel fluid mass. The quench time is approximately 310 seconds (Figure 15.6-7).

- <u>Post Analysis of Record Evaluation:</u>

An evaluation of IFBA fuel including the effects of pellet TCD was performed, and shows that non-IFBA fuel bounds IFBA fuel.

In addition to the analyses presented in this section, evaluations and reanalyses may be performed in the future as needed to address computer code errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the Peak Cladding Temperature (PCT) is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT for the large break LOCA. The current PCT presented in Table 15.6-11 is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200 °F.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes to or errors in the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the USAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model results in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC.

Finally, the criteria of 10 CFR 50.46 require that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owner's Group (PWROG), has developed an approach for compliance with the reporting requirements. This approach is documented in WCAP-13451 (Reference 27), *Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting*. Wolf Creek Nuclear Operating Corporation provides the NRC with annual and 30-day reports, as applicable. Wolf Creek Nuclear Operating Corporation intends to provide future reports required by 10 CFR 50.46 consistent with the approach described in WCAP-13451.

- Conclusions:

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 1900 $^{\circ}$F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., "Peak Cladding Temperature less than 2200 $^{\circ}$F", is demonstrated. The results are shown in Table 15.6-11.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. Since the resulting LMO for the limiting case is 4.21 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., "Local Maximum Oxidation of the cladding less than 17 percent of the total cladding thickness before oxidation", is demonstrated. The results are shown in Table 15.6-11.

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.1352 percent. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than the HAR value. Since the resulting CWO is 0.1352 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., "Core-Wide Oxidation less than 1 percent", is demonstrated. The results are shown in Table 15.6-11.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria (b)(1) and (b)(2) remain in effect for BE LOCA applications. The approved methodology (WCAP-12945-P-A) (Reference 6) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation remains unchanged with the application of the ASTRUM methodology (Reference 9).

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The manual or automatic actions that are currently in place to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 9).

Based on the ASTRUM Analysis results (Table 15.6-11), it is concluded that Wolf Creek continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

The maximum calculated peak clad temperature for all small breaks analyzed is 936°F. The results of the analyses given in Table 15.6-12 are well below all Acceptance Criteria limits of 10 CFR 50.46, and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.3.4 Post-LOCA Long-Term Core Cooling/Subcriticality

10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" paragraph (b) item (5) sets forth the requirements for post-LOCA long-term core cooling. To satisfy the requirements, the core is maintained in a shutdown state solely by the soluble boron contained in the ECCS water after a LOCA because credit for the shutdown provided by the control rods was not taken for cold leg breaks ≥ 3.0 ft^2, or hot leg breaks ≥ 1.0 ft^2. Since safety injection flow is drawn from the sump following switchover from the RWST, the containment sump post-LOCA boron concentration must be higher than the boron concentration required to ensure subcritical conditions.

To determine if the requirements for post-LOCA long-term core cooling subcriticality are met, a calculation is performed for each reload to determine the boron concentration required to keep the core subcritical (K_{eff} < 1.0) and the mixed mean boron concentration (MMBC) of the post-LOCA sump water. This calculation, documented in the cycle-specific Reload Safety Analysis Checklist (RSAC), confirms that the post-LOCA sump MMBC exceeds the core critical boron concentration, thereby ensuring the reload core remains subcritical. Note: The post-LOCA long-term core cooling critical boron concentration is determined at the most reactive time in life, assuming an all rods out (ARO) no Xenon condition and a post-LOCA fluid temperature range of 68-212°F. All sources of water that may eventually reside in the containment sump at cold leg recirculation switchover time and their respective pre-accident boron concentrations are considered.

Westinghouse has identified a potential safety issue concerning core recriticality following a large break cold leg break LOCA (Reference 19). The potential safety issue is that during hot leg switchover the core will be flushed with a diluted sump solution, which may cause the core to return to criticality. The sump solution would become diluted as boron accumulates in the core during the core leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from being displaced to the sump which leads to a diluted sump solution.

However, a generic assessment (Reference 19) concludes that for any given plant and fuel cycle, the boron worth of the inserted control rods plus the equivalent boron worth due to the presence of Xenon at the time of hot leg switchover, would offset any reasonable calculation of sump dilution. As documented in WCAP-15704 (Reference 20), it was demonstrated that control rods will insert following a licensing basis cold leg LOCA for 3-loop and 4-loop Westinghouse plant designs. Thus, the negative reactivity credit associated with control rods can be applied when evaluating recriticality at the time of switchover to hot leg ECCS recirculation.

Consequently, post-LOCA subcriticality will continue to be confirmed using sump boron calculations that do not consider sump dilution, but also use the conservative assumptions of ARO and no Xenon at the most reactive time in life, with an assumed post-LOCA core/fluid temperature in the range of 68-212°F.

The post-LOCA phase of the accident begins at the transfer to CL recirculation. Prior to sump recirculation, core cooling is addressed for the full spectrum of break sizes by the large break LOCA (LBLOCA) and small break LOCA (SBLOCA) analyses areas. This satisfies the 10CFR50.46 acceptance criteria pertaining to peak cladding temperature, maximum cladding oxidation, and maximum hydrogen generation.

Decay heat removal checks are performed for the transfer to CL recirculation at transient times based upon full ECCS injection for the injection phase. Full ECCS injection for the injection phase consists of two residual heat removal (RHR) pumps, two intermediate head safety injection (IHSI) pumps, two high head centrifugal charging pumps (HHCCP), and two containment spray (CS) pumps. Maximizing the flow in the RWST drain down calculation conservatively bounds entry to CL recirculation. The earliest entry to CL recirculation for the LBLOCA scenario was determined to be 13 minutes. The earliest entry to CL recirculation for the SBLOCA scenario was determined to be 25 minutes.

The adequacy of the CL recirculation flow is checked at the earliest entry to CL recirculation. Minimum flows are generated assuming the failure of a diesel generator. One RHR pump takes suction from the containment sump while one IHSI pump and one HHCCP pump take suction from the RHR discharge. The RHR, IHSI, and HHCCP inject to all four cold legs. The spilling line is assumed to be at atmospheric pressure to conservatively minimize the ECCS available for core cooling.

The latest acceptable time to enter HL recirculation is determined by the calculated incipient boric acid precipitation time. The earliest acceptable time to enter HL recirculation is determined by providing the operators one hour of margin to establish HL recirculation. The adequacy of the earliest acceptable time to prevent core uncovery and to effectively remove the decay heat generated in the core is then confirmed. The earliest acceptable time to transfer to HL recirculation is also confirmed against the entrainment threshold.

The transfer to HL recirculation procedure instructs the operators to transfer the discharge of the IHSI pump from four cold legs to two hot legs. The IHSI pump is stopped to perform this transfer. The limiting flows for performing HL recirculation flow checks with respect to DHR occur when the HL recirculation transfer process is complete. This is due to the RHR pump providing boosting to the IHSI pump.

The limiting scenario for post-LOCA LTC for HL recirculation consists of one RHR pump taking suction from the containment sump and discharging to all four cold legs. The HHCCP continues to take suction from the RHR discharge and inject to all four cold legs. The IHSI pump takes suction from the RHR discharge and injects to two of the four hot legs. The adequacy to effectively remove decay heat at the early entry to HL recirculation was checked for both a CL and HL break. The adequacy of the HL flow to halt and reverse the concentration of boric acid in the core was checked at the late entry to HL recirculation for a CL break. An HL break is not a concern with respect to the concentration of boric acid in the core. This is due to CL recirculation flow always being present and providing a forward flushing path.

For small breaks, emergency procedures instruct operators to take action to depressurize and cool down the RCS. Although this depressurization and cooldown process typically begins within one hour after the event, the LTC analysis makes no specific assumptions regarding time to depressurize. However, the post-LOCA LTC safety analyses do assume the rate of cooldown is limited by the operating procedures to 100°F/hr. Depressurization to 120 psia (the threshold for boric acid precipitation concerns) may occur before or after the prescribed HLSO time.

An incipient boric acid concentration time was calculated. This value is conservatively rounded down to the nearest half hour to determine the latest acceptable time to complete the transfer to HL recirculation. The latest acceptable time to complete the transfer to HL recirculation was determined to be 7.5 hours from the initiation of the accident.

One hour of margin was provided to the operators to complete the steps necessary to transfer from CL recirculation to HL recirculation. The adequacy of the ECCS to effectively remove decay heat at an early entry to HL recirculation of 6.5 hours from the initiation of the accident was checked. Hot leg recirculation flow was shown to be adequate to effectively remove decay heat for a CL break while CL recirculation flow was shown to be adequate to effectively remove decay heat for an HL break. The earliest HL recirculation time of 6.5 hours is well after the entrainment threshold.

15.6.5.4 Radiological Consequences

15.6.5.4.1 Method of Analysis

The analysis of the radiological consequences of a postulated LOCA uses the recommended dose conversion factors (DCFs) as follows:

(1) The analysis uses committed effective dose equivalent dose conversion factors (DCFs) for inhalation of radionuclides based on the data provided in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion".
(2) The effective dose equivalent dose conversion factors, provided in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", are used to calculate the external exposure doses.

15.6.5.4.1.1 Containment Leakage Contribution

PHYSICAL MODEL - Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of a nuclear power plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

It is assumed that the equilibrium core saturation fission product inventory is released in phases to the containment atmosphere. The activity released from the core during each of the release phases is assumed to be immediately available for leakage from the containment.

Once the fission product activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment. The removal mechanisms include radioactive decay, containment sprays, and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage.

a. Radioactive Decay - Credit for radioactive decay for fission product concentrations located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit for radioactive decay or deposition is taken.

b. Containment Sprays - The containment spray system is designed to absorb airborne iodine fission products within the containment atmosphere. To enhance the iodine-removal and retention capability of the containment sprays, sodium hydroxide is added to the spray solution. The spray effectiveness for the removal of iodine is dependent on the iodine chemical form.

c. Containment Leakage - The containment leaks at a rate of 0.2 volume percent/day as incorporated as a Technical Specification requirement at peak calculated internal containment pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. The containment leakage is assumed to be directly to the environment.

ASSUMPTIONS AND CONDITIONS - The major assumptions and parameters assumed in the analysis are itemized in Tables 15A-1 and 15.6-6.

In the evaluation of a LOCA, all the fission product release assumptions of Regulatory Guide 1.183 have been followed. The following specific assumptions were used in the analysis. Tables 15B-1 and 15B-2 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183.

a. The reactor core equilibrium inventories are based on long-term operation at the ultimate core power level of 3,565 MWt plus 2% uncertainty.

b. The nuclide groups and their release fractions are presented in Table 15.6-6.

c. Of the iodine fission product inventory released to the containment, 4.85 percent is in the form of elemental iodine, 95 percent is in the form of particulate iodine, and 0.15 percent is in the form of organic iodine.

d. Credit for removal of iodine and particulates by the containment spray system is taken, starting 2 minutes after event initiation and continuing until a decontamination factor of 200 for the elemental species has been achieved. The credit for the particulate species removal is continued for the duration of sprays but is reduced by a factor of 10 after a decontamination factor of 50 is achieved.

e. The following removal constants for the containment spray system are assumed in the analysis:

Elemental iodine	10.0 hr^{-1}
Organic iodine	0.0 hr^{-1}
Particulates	5.0 hr^{-1} prior to DF of 50
	0.5 hr^{-1} after DF of 50

f. The following parameters were used in the two-region spray model:

Fraction of containment sprayed - 0.85
Fraction of containment unsprayed - 0.15
Mixing rate (cfm) between sprayed and unsprayed regions - 69,400
Time (min) to achieve full fan cooler flow - 2

Section 6.5 contains a detailed analysis of the sprayed and unsprayed volumes and Section 6.2 includes an explanation of the mixing rate between the sprayed and unsprayed regions.

g. The containment is assumed to leak at 0.2 volume percent/day during the first 24 hours immediately following the accident and 0.1 volume percent/day thereafter.

h. The containment leakage is assumed to be direct unfiltered to the environment.

MATHEMATICAL MODELS USED IN THE ANALYSIS - Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Section 15A.2.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurements program described in Section 2.3 and are provided in Table 15A-2.

c. The TEDE doses to a receptor exposed at the exclusion area boundary and the outer boundary of the low population zone were analyzed, using the models described in Appendix 15A.

d. Buildup of activity in the control room and the integrated doses to the control room personnel are analyzed, based on models described in Section 15A.3.

IDENTIFICATION OF LEAKAGE PATHWAYS AND RESULTANT LEAKAGE ACTIVITY - For evaluating the radiological consequences of a postulated LOCA, the resultant activity released to the containment atmosphere is assumed to leak directly to the environment.

No credit is taken for ground deposition or radioactive decay during transit to the exclusion area boundary, LPZ outer boundary, or control room.

15.6.5.4.1.2 Radioactive Releases Due to Leakage from ECCS and Containment Spray Recirculation Lines

Subsequent to the injection phase of ESF system operation, the water in the containment recirculation sumps is recirculated by the residual heat removal, centrifugal charging and safety injection pumps, and the containment spray pumps. Due to the operation of the ECCS and the containment spray system, most of the radioiodine released from the core would be contained in the containment sump. It is conservatively assumed that a leakage rate of 2 gpm from the ECCS and containment spray recirculation lines exists for the duration of the LOCA. This leakage would occur inside the containment as well as inside the auxiliary building. For this analysis, all the leakage is assumed to occur inside the auxiliary building. Only trace quantities of radioiodine are expected to be airborne within the auxiliary building due to the temperature and pH level of the recirculated water. However, 10 percent of the radioiodine in the leaked water is assumed to become airborne. This airborne iodine is assumed to be released immediately to the environment from the unit vent, via the safety grade filters associated with the auxiliary building emergency exhaust system. No credit is taken for holdup or mixing in the auxiliary building; however, mixing and holdup in the containment sumps are included in the determination of radioactive material releases and radioiodine removal through radioactive decay for this leakage pathway.

Radiological Consequences of ECCS/CS Recirculation Line Leakage - The assumptions used to calculate the amount of radioiodine released to the environment are given in Table 15.6-6. The dose models are presented in Section 15.A. The offsite doses at the site boundary and LPZ and the doses to control room personnel from this pathway are given in Table 15.6-8.

15.6.5.4.1.3 Radioactive Releases Due to Operation of Containment Mini-Purge System

The containment mini-purge is designed to reduce the containment noble gas concentration. The containment mini-purge system will be operated during power operation if access to the containment is desired. The containment mini-purge isolation valves are automatically closed upon a containment purge isolation signal should a LOCA occur during containment purging when the reactor is at power. The radioactive release via containment mini-purge will exist until the containment isolation signal is received and the valves can be closed. Exhaust from the containment is processed through the containment purge exhaust system filter absorber train prior to discharge through the unit vent.

The maximum time for the purge valve closure is limited to ten seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Therefore, the source terms used in the radiological consequences calculation is based on the fission product activity in the primary coolant. The containment mini-pure system is assumed to be isolated within 10 seconds following the initiation of the accident. The release rate from the containment mini-purge system is assumed at 4680 cfm. Filter removal of iodine and particulates is not assumed, nor is credit for removal by the containment spray system in the analysis for activity release via containment mini-purge system.

15.6.5.4.1.4 Radioactive Releases Due to Leakage from the Containment Sump to the RWST

The leakage pathway is from the containment recirculation sump through ECCS boundary valves back to the RWST, which is vented to the atmosphere. It is assumed that the activity released to the holdup system (in this case, the containment recirculation sump) instantaneously diffuses to uniformly occupy the sump volume. Removal mechanisms from the sump include decay and release (i.e., leakage) to the RWST.

It is assumed that 10% of the radioiodine leaked to the RWST becomes airborne, mixes with the RWST air volume, and is released to the environment. Credit is taken for radioactive decay in the RWST. The leakage rate from containment sump to the RWST liquid is assumed at 3.8 gpm for the duration of the accident. The release from the RWST is assumed to match the volumetric flow into RWST liquid (i.e., 3.8 gpm). The RWST air volume when leakage starts is assumed to be 354,000 gallons. Other assumptions used to calculate the amount of radioiodine released to the environment are the same as in the calculation for ECCS recirculation leakage inside the auxiliary building.

15.6.5.4.1.5 Operator Dose due to Ingress and Egress to the Control Room

PHYSICAL MODEL - 10 CFR 50.67(b)(2) requires that the LOCA analysis demonstrate with reasonable assurance that adequate radiation protection is provided not only to permit occupancy of the control room under accident conditions, but also permit ingress and egress without personnel receiving radiation exposures in excess of applicable dose limits for the duration of the accident.

It is assumed that the operator travels to the site, parks in the parking lot, and walks from the parking lot to the control building via the turbine building entrance without any supplemental radiation protection. The operator leaves using the same path. The scenario is not the most likely scenario, as it ignores the principal of ALARA, but the modeling and assumptions used are consistent with a design basis accident.

ASSUMPTIONS AND CONDITIONS – The major assumptions and parameters assumed in the analysis of operator transit dose are itemized in Tables 15A-1 and 15.6-6.

The following specific assumptions were used in this analysis:

a. Walking distance from the parking lot to the entrance of the turbine building is conservatively assumed to be 1550 feet.

b. Operator walking speed is assumed to be 3 miles per hour. The resulting transit time is conservatively modeled as 6.2 minutes.

c. The operators will be confined to the control room for the first 24 hours (100% occupancy factor) following the LOCA. The first ingress/egress will begin at 24 hours following the event and be repeated every 12 hours for the length of the accident (30 days, 59 trips total)

d. Operator respiration is conservatively taken as double the respiration rate for the control room (i.e. the RG 1.183 breathing rate of 3.5E-4 m^3/sec is doubled to 7.0E-4 m^3/sec).

e. For the ground shine dose, deposition velocities of 1.0E-2 m/sec for elemental iodine, 1.0E-4 m/sec for organic iodine, and 1.0E-3 m/sec for particulates are modeled, consistent with NUREG/CR-3332. Noble gases are not assumed to deposit on the ground.

f. The dose point for the operator walking along the ingress/egress path is 5 feet off the ground.

g. Shielding provided by intervening equipment, plant buildings, or other materials is ignored (except for the containment concrete and liner when determining the containment shine dose).

MATHEMATICAL MODELS USED IN THE ANALYSIS - Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Section 15A.2.

b. Nine points along the ingress/egress path are used to determine dose rates (See Table 15.6-6 and Figure 15.6-49 for the location of these points). Atmospheric dispersion factors (χ/Qs) are calculated for these dose points and are given in Table 15A-2. A weighted χ/Q for the entire length of the pathway is modeled for each transit based on the transit segment distance.

c. The TEDE dose to an operator in transit from airborne and deposited activity are calculated by the models described in Section 15A.1 for various locations along the operator transit route.

d. The overall Control Room TEDE dose, including TEDE dose accrued during ingress/egress, is given in Table 15.6-8.

15.6.5.4.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a LOCA result principally from assumptions made involving the amount of the gaseous fission products available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. The ECCS is designed to prevent fuel cladding damage that would allow the release of the fission products contained in the fuel to the reactor coolant. Severe degradation of the ECCS (i.e., to the unlikely extent of simultaneous failure of redundant components) would be necessary in order for the release of fission products to occur of the magnitude assumed in the analysis.

b. It is assumed that none of the iodines released to the containment atmosphere is plated-out onto the internal surfaces of the containment or adheres to internal components; however, it is recognized that the removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling could reduce the resultant doses.

c. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.2-volume percent/day for the first 24 hours and 0.1-volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressure within the containment actually decreases with time. Taking into account that the containment leak rate is a function of pressure, the resultant doses could be reduced by a factor of 5 to 10 (Ref. 21).

d. The meteorological conditions assumed to be present at the site during the course of the accident are based on X/Q values, which are expected to be exceeded 5 percent of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.5.4.3 Conclusions

15.6.5.4.3.1 Filter Loadings

No recirculating or single-pass filters are used for fission product cleanup and control within the containment following a postulated LOCA. The only ESF filtration systems expected to be operating under post-LOCA conditions are the control room HVAC system and the auxiliary building emergency exhaust filtration system.

Activity loadings on the control room charcoal adsorbers are based on the flowrate through the adsorber, the concentration of activity at the adsorber inlet, and the adsorber efficiency. Based on the release assumptions previously described and the assumption that the charcoal adsorber is 90 percent efficient for iodine and particulates, the calculated dose to control room operators from filter loadings is 0.0749 rem TEDE.

15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses have been analyzed for the limiting 2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-5. The resultant doses are within the guideline values of 10 CFR 50.67.

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a TEDE dose. This dose has been analyzed, and is provided in Table 15.6-8. The resultant dose is within the limit established by 10 CFR 50.67.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to WCGS.

15.6.7 REFERENCES

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TABLE 15.6-1

TIME SEQUENCE OF EVENTS FOR INCIDENT WHICH RESULTS IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	21.4
	Rods begin to drop	24.4
	Minimum DNBR occurs	25.0
Steam generator tube rupture: Margin to Overfill		
	Initiate Break	100
	Reactor Trip (OTDT)	145
	Turbine trip	145
	MFW Coastdown Begins	145
	AFW Initiation	145
	RCP Trip	147
	SI Actuation (Low PZR Pressure)	304
	TDAFW Isolation	368
	MDAFW Isolation	398
	Steamline Isolation (All Loops)	625
	Start RCS Cooldown	1525
	End RCS Cooldown	2149
	Start RCS Depressurization	2449
	End RCS Depressurization	2527
	Isolate SI	2767
	Break Flow Terminates	4231

TABLE 15.6-1 (sheet 2)

Accident	Event	Time (sec)
Steam generator tube rupture: Thermal & Hydraulic		
	Initiate Break	100
	Reactor Trip (OTDT)	152
	Turbine trip	152
	RCP Trip	154
	AFW Initiation	212
	SI Actuation (Low PZR Pressure)	425
	Steamline Isolation (All Loops)	632
	Ruptured SG AFW Flow Isolated	1202
	Ruptured SG PORV Fails Open	1202
	Ruptured SG PORV Isolated	3002
	Start RCS Cooldown	3602
	Break Flow Flashing Terminated	3946
	End RCS Cooldown	4955
	Start RCS Depressurization	5255
	End RCS Depressurization	5377
	Isolate SI	5616
	Break Flow Terminated	7627
	RHRS Takes Over Cooling	43300

TABLE 15.6-2

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCE OF
THE CVCS LETDOWN LINE RUPTURE OUTSIDE OF CONTAINMENT

I.	Source Data	
	a. Core power level, MWt	3637 (includes uncertainty)
	b. Reactor coolant iodine activity	Dose equivalent of 1.0 µCi/gm of I-131 with either an assumed pre-accident iodine spike of 60 µCi/gm of DE I-131 or an accident-initiated iodine spike, which increases the rate of iodine release into the reactor coolant by a factor of 500. The increased rate is assumed for 8 hours.
	c. Reactor coolant noble gas activity	Dose equivalent of 500 microCi/gm of Xe-133
	d. Reactor coolant initial alkali metal activity	Based on 1-percent failed fuel as provided in Table 11.1-5
II.	Atmospheric Dispersion Factors	See Table 15A-2.
III.	Activity Release	
	a. Break flow rate, gpm	444
	b. Duration, secs	1810
	c. Fraction of iodine and alkali metal activity in the spill that is airborne	0.18

TABLE 15.6-3

RADIOLOGICAL CONSEQUENCES OF A
CVCS LETDOWN LINE BREAK OUTSIDE
OF CONTAINMENT

	Doses (rem TEDE)
CASE 1, 1.0 µCi/gm I-131 equivalent w/I spike	
Exclusion Area Boundary (0-2 hr)	3.5E-1
Low Population Zone Outer Boundary (duration)	1.2E-1
Control Room (30 days)	3.7E-1
CASE 2, 60 µCi/gm I-131 equivalent	
Exclusion Area Boundary (duration)	5.7E-1
Low-population zone, outer boundary (duration)	1.9E-1
Control Room (30 days)	1.5E+0

Table 15.6-4

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

I.	Source Data	
a.	Core power level, MWt	3637[1]
b.	Steam generator tube leakage, gpm	1[2]
c.	Reactor coolant iodine activity:	
	1. Case 1	Pre-accident iodine spike per Regulatory Guide 1.183. A pre-accident reactor transient has occurred which raised the primary coolant I-131 dose equivalent concentration to the maximum value permitted by Technical Specifications (60 μCi/gm).
	2. Case 2	Accident initiated iodine spike per Regulatory Guide 1.183. An iodine spike occurs concurrently with reactor trip where the release rate from the fuel rods increases to a value 335 time greater than the release rate corresponding to the I-131 dose equivalent concentration at the equilibrium value (1 μCi/gm) stated in the Technical Specifications.
d.	Reactor coolant initial alkali metal activity	Based on 1-percent failed fuel as provided in Table 11.1-5.
e.	Reactor coolant noble gas activity	Dose equivalent of 500 microCi/gm of Xe-133.
f.	Secondary system initial iodine activity	Dose equivalent of 0.1 μCi/gm of I-131
g.	Secondary system initial alkali metal activity	Based on the same ratio of primary to secondary system activity as iodine (i.e., 10% of primary side activity)
h.	Reactor coolant mass, lbs	3.99E+5

Table 15.6-4 (Sheet 2)

i.	Steam generator mass (each), lbs	
	1. Ruptured	70,000
	2. Intact	65,000
j.	Offsite power	Lost at event initiation
II.	Atmospheric Dispersion Factor	The atmospheric dispersion factors used are the same as those listed in USAR Table 15A-2.
III.	Activity Release Data	
a.	Ruptured steam generator	
	1. Reactor coolant discharged to steam generator, lbs	Figure 15.6-3S
	2. Flashed reactor coolant	Figure 15.6-3V
	3. Iodine and alkali metal partition factor for flashed fraction of reactor coolant	1.0
	4. Steam release to atmosphere, lbs	
	0-break flow termination	Figure 15.6-3Q
	break flow termination - RHR cut-in conditions, lbs	2530
	5. Carryover factors for the non-flashed fraction of reactor coolant that mixes with the initial iodine activity in the steam generator	
	Iodine	0.01
	Alkali Metal	0.0025
b.	Intact steam generators	
	1. Primary-to-secondary leakage, lbs	6012
	2. Flashed reactor coolant, percent	0
	3. Steam release to atmosphere,	
	0-break flow termination	Figure 15.6-3Q
	break flow termination-RHR cut-in conditions, lbs	1,645,930
	4. Iodine carryover factor	0.01

Table 15.6-4 (sheet 3)

	5. Alkali Metal carryover factor	0.0025
c.	Condenser	
	1. Steam release to atmosphere, lbs	750,750
	2. Iodine and alkali metal partition factor	0.01
IV	Dose Conversion Factors	
	See Table 15A-4	
V	Control Room	
a.	Control Room Switchover to Emergency Mode, sec	See Table 15A-5
b.	Control Room Switchover to Emergency Mode Completion, sec	See Table 15A-5

Table 15.6-4 (sheet 4)

Notes:

1. This value is 102% of the guaranteed core thermal power output. This power level was chosen to maximize steam generator releases, and thus the radiological consequences were calculated from the results of this model.

2. A constant 1 gpm primary-to-secondary leakage with an assumed density of 62.4 lbm/ft^3, is assumed to occur to the intact steam generators until the RHR cut-in conditions were achieved at 12 hours.

TABLE 15.6-5

RADIOLOGICAL CONSEQUENCES OF A
STEAM GENERATOR TUBE RUPTURE

Case 1	Dose (rem TEDE)
Exclusion Area Boundary (0-2 hr)	9.9E-1
Low Population Zone (duration)	3.2E-1
Control Room (30 days)	4.2E+0
Case 2	
Exclusion Area Boundary (0-2 hr)	8.0E-1
Low Population Zone (duration)	2.6E-1
Control Room (30 days)	1.1E+0

Case 1 - Pre-accident iodine spike per RG 1.183

Case 2 - Concurrent iodine spike per RG 1.183

TABLE 15.6-6

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
LOSS-OF-COOLANT-ACCIDENT

I. Source Data

a. Core power level, MWt — 3637 (includes uncertainty)

b. Burnup, full power days per cycle — 510

c. Release fraction and timing of core activity in the containment

	Gap Release Phase Fraction (30 sec – 0.5 hr)	Early In-vessel Phase Fraction (0.5 hr – 1.8 hr)
1. Noble gas	0.05	0.95
2. Iodine	0.05	0.35
3. Alkali metal	0.05	0.25
4. Tellurium	0.00	0.05
5. Barium and strontium	0.00	0.02
6. Noble metal	0.00	0.0025
7. Cerium	0.00	0.0005
8. Lanthanide	0.00	0.0002

d. Percent of core activity in containment sump

1. Noble gases and Particulates	0
2. Iodine	same release fraction and timing as Item c.2 above

e. Core inventories — Table 15A-3

f. Iodine distribution, percent Containment Leakage Release

	Containment Leakage Release	ECCS and mini-purge Release
1. Elemental	4.85	97
2. Organic	0.15	0
3. Particulate	95	3

g. Minimum sump pH — 7.0

h. Reactor Coolant Activity (mini-purge only)

1. Iodine	Dose equivalent of 1.0 µCi/gm of I-131
2. Noble gas	Dose equivalent of 500 µCi/gm of Xe-133

TABLE 15.6-6 (Sheet 2)

	3. Alkali metal	Based on 1-percent failed fuel as provided in Table 11.1-5
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
	a. Containment leak rate, volume percent/day	
	1. 0-24 hours	0.20
	2. 1-30 days	0.10
	b. Percent of containment leakage that is unfiltered	100
	c. Credit for containment sprays	
	1. Spray removal constants (per hour)	
	a. Elemental iodine	10.0
	b. Organic iodine	0.0
	c. Particulates	5.0 hr^{-1} prior to DF limit 0.5 hr^{-1} thereafter
	2. Decontamination factors for the containment atmosphere	
	a. Elemental iodine	200
	b. Organic iodine	0
	c. Particulates	50
	3. Sprayed volume, percent	85
	4. Unsprayed volume, percent	15
	5. Sprayed-unsprayed mixing rate, CFM	69,400
	6. Containment volume, ft^3	2.7E+6
	d. ECCS recirculation leakage	
	1. Leak rate to auxiliary building (0 hours-30 day), gpm	2.0
	2. Leak rate to RWST (0 hours-30 day), gpm	3.8 (1)

TABLE 15.6-6 (Sheet 3)

3. Sump volume, gal. 460,000
4. Fraction iodine airborne 0.1
5. ESF filter efficiency, % 90.0
6. RWST Initial Gas Volume, gal 354,000[1]

IV. Control room parameters Tables 15A-1 and 15A-2

a. Control Room Switchover to Emergency Mode, sec See Table 15A-5

V. Operator Ingress/Egress parameters

a. Transit Distance See Figure 15.6-49 for segment locations

Transit Segment	Starting Location	End Location	Conservative Segment Distance (ft)
1	Point 1	Point 2	200
2	Point 2	Point 3	100
3	Point 3	Point 4	125
4	Point 4	Point 5	150
5	Point 5	Point 6	100
6	Point 6	Point 7	150
7	Point 7	Point 8	250
8	Point 8	Point 9	275
9	Point 9	Parking Lot	200
		Total	1550

b. Transit Speed, mph 3
c. Occupancy Factor 100% for initial 24 hours, ingress/egress every subsequent 12 hours
d. Respiration Rate, m^3/sec 7.0E-4
e. Deposition Velocities, m/sec
   1. Elemental Iodine 1.0E-2
   2. Organic Iodine 1.0E-4
   3. Particulates 1.0E-3
f. Operator Dose Point Height, ft 5

Note:
1. The release from the RWST is assumed to match the ECCS leak rate, i.e., the volumetric flow into and out of the tank are equal.

TABLE 15.6-7

Not Used

TABLE 15.6-8

Radiological Consequences of a
Loss-of-Coolant-Accident

		Total Reported Doses (rem TEDE)	Regulatory Limits (rem TEDE)
I.	Exclusion Area Boundary (0.5-2.5 hr)	9.0	25
II.	Low Population Zone Outer Boundary(0-30 day)	14.0	25
III.	Control Room (0-30 day)	3.6	5

TABLE 15.6-9

Major Plant Parameter Assumptions Used in the BELOCA Analysis for Wolf Creek

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤ 10%
Plant Initial Operating Conditions	
• Reactor Power	≤ 3565 MWt (+ 2% uncertainty)
• Peaking Factors	See Table 15.6-9B
• Axial Power Distribution	See Figure 15.6-22
Fluid Conditions	
• T_{AVG}	570.7 – 4.0 °F ≤ T_{AVG} ≤ 588.4 + 4.0 °F
• Pressurizer Pressure	2250 – 50 psia ≤ P_{RCS} ≤ 2250 + 50 psia
• Reactor Coolant Flow	≥ 90,324 gpm/loop
• Accumulator Temperature	50 °F ≤ T_{ACC} ≤ 120 °F
• Accumulator Pressure	582.8 psia ≤ P_{ACC} ≤ 696.6 psia
• Accumulator Water Volume	818 ft^3 ≤ V_{ACC} ≤ 881 ft^3
• Accumulator Boron Concentration	≥ 2300 ppm
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	See Table 15.6-13
• Safety Injection Temperature	37 °F ≤ TSI ≤ 120 °F
• Safety Injection Initiation Delay Time	27 sec. (with offsite power) 39 sec. (without offsite power)
• Containment Pressure	Bounded (minimum); see Figure 15.6-21

TABLE 15.6-9A

Input Parameters Used in the Small LOCA Analysis

License Core Power[1] (100% power) (MWth)	3565
Peak Linear Power[1] (KW/ft)	13.909
Total Peaking Factor [F_Q^T]	2.50
Axial Peaking Factor [F_z]	1.5151
Hot Channel Enthalpy Rise Factor [$F_{\Delta H}$]	1.65
Maximum Assembly Average Power[P_{HA}]	1.469
Power Shape	Figure 15.6-47
Fuel Type/Assembly Array	17x17 Standard RFA-2 Zirlo+2, 1.5X IFBA, IFM(3)
Accumulator Water Volume (ft^3/accumulator)	850
Accumulator Tank Volume (ft^3/accumulator)	1350
Accumulator Gas Pressure, Minimum (psia)	583
Safety Injection Pumped Flow	Figures 15.6-46 & 15.6-46A, Tables 15.6-14 & 15.6-14A
Containment Parameters	See Section 6.2.1.5
Initial Loop Flow (gpm/loop)	90324
Nominal Vessel Average Temperture (F)	570.7 –588.4
Reactor Coolant Pressure[2] (psia)	2300
Steam Pressure (psia)	938.89
Steam Generator Tube Plugging Level (%)	10
AFW Flow Rate (gpm/steam generator)	210 -250

Notes:
(1) Two percent is added to this power to account for calorimetric uncertainty. Reactor coolant pump heat is not modeled in LOCA Analysis.
(2) This pressure includes 50 psi for measurement uncertainty.
(3) The SBLOCA analysis explicitly modeled ZIRLO® cladding. Optimized ZIRLO™ cladding was qualitatively evaluated as an acceptable fuel cladding.

TABLE 15.6-9B

Summary of Peaking Factor Burndown Analyzed by the Best-Estimate Large-Break LOCA Analysis for Wolf Creek

Hot Rod Burnup (MWD/MTU)	**FdH (with uncertainties)**	**FQ Transient (with uncertainties)**	**FQ Steady-state (without uncertainties)**
0	1.65	2.50	2.121
30000	1.65	2.50	2.121
60000	1.525	2.25	1.906
62000	1.525	2.25	1.906
Note: Full Table is used for 2nd Cycle, values at Burnup=0 were utilized for 1st Cycle			

TABLE 15.6-10

Wolf Creek Best Estimate Large Break Sequence of Events for the Limiting PCT Case in seconds

Start of Transient	0.0
Safety Injection Signal	6.0
Accumulator Injection Begins	10.0
End of Blowdown	21.0
Bottom of Core Recovery	30.0
Accumulator Begins to Empty	34.0
Safety Injection Begins	45.0
PCT Occurs	70.0
Quench Time (WCOBRA/TRAC)	310.0
End of Transient	350.0

TABLE 15.6-10 (Sheet 2)

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TABLE 15.6-10A

Time Sequence of Events for Small Break LOCA

Break Size	2-inch	3-inch	4-inch	6-inch	8.75-inch
Break Initiation	0	0	0	0	0
Reactor Trip Signal	112.02	40.29	22.75	9.71	7.16
Safety Injection Signal	121.62	48.85	30.40	15.08	8.77
Safety Injection Begins(1)	160.62	87.85	69.40	54.08	47.77
Loop Seal Clearing Occurs(2)	1166	418	248	55	17
Accumulator Injection	N/A	N/A	954	385	175
Top of Core Uncovered	N/A	875	713	422	N/A
Peak Clad Temperature Occurs	N/A	1236.6	1033.5	462.6	N/A
Top of Core Recovered	(3)	2233	1401	470	(3)
RWST Low Level (4)	1980	1943	1908	1793	1547

Notes:

(1) Safety Injection (SI) is assumed to begin 39 seconds after the SI signal.

(2) Loop seal clearing is considered to occur when the broken loop seal vapor flow rate is sustained above 1 lbm/s

(3) For the cases where core recovery is greater than the transient time, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(4) The analysis assumes minimum usable RWST volume (239, 324 gal) before the low-1 RWST water level signal for switchover to cold leg recirculation is reached.

WOLF CREEK

TABLE 15.6-11

Wolf Creek Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	**Value**	**Criteria**
95/95 PCT (°F)	1900	< 2200
95/95 LMO(1) (%)	4.21	< 17
95/95 CWO(2) (%)	0.1352	< 1

Notes: (1) LMO – Local Maximum Oxidation
(2) CWO – Core Wide Oxidation

TABLE 15.6-12

SMALL BREAK LOCA RESULTS FUEL CLADDING DATA

Break Size (in)	2[1]	3	4	4 (w/Annular Pellets)	6	8.75[3]
PCT (°F)	N/A	895.8	935.5	935.6	621.7	N/A
PCT Elevation (ft)		10.75	11.0	11.0	11.5	
Hot Rod Burst Time (sec)		No Burst	No Burst	No Burst	No Burst	
Hot Rod Burst Elevation (ft)[2]		N/A	N/A	N/A	N/A	
Max. Local transient ZrO_2 Reaction (%)		0.01	0.01	0.01	0.01	
Max. Local ZrO_2 Elev. (ft)		11.00	11.00	11.00	11.50	
Core-Wide Avg. ZrO_2 (%)		<1.0	<1.0	<1.0	<1.0	

Notes:

(1) There is no core uncovery for the 2-inch case.

(2) None of the SBLOCTA calculations exhibited rod burst (hot rod or hot assembly average rod).

(3) The core only uncovers for a minimal time and does not warrant SBLOCTA calculations.

TABLE 15.6-13

Safety Injection Flow for BE Large Break LOCA Analysis

RCS Pressure (psig)	HHSI		RCS Pressure (psig)	IHSI and RHR Combined	
	Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop
0	287	103	0	3563	1209
20	285	104	20	3285	1295
40	283	105	40	2992	1379
60	281	106	60	2679	1462
80	279	107	80	2340	1543
100	277	108	100	1965	1625
120	274	109	120	1532	1709
140	272	110	140	989	1800
160	270	111	160	397	1875
180	268	112	180	393	1876
200	266	113	200	389	1876
300	255	117	300	369	1880
400	244	122	400	347	1883
500	232	126	500	325	1887
600	217	190	600	302	1890
700	197	200	700	277	1894
800	177	210	800	250	1898
900	156	219	900	220	1937
1000	135	229	1000	180	1943
1100	113	238	1100	135	1949
1200	90	247	1200	81	1956
1300	67	256	1300	6	1964
1400	42	265	1400	0	1964
1500	15	274	-	-	-
1600	0	280	-	-	-

TABLE 15.6-14

Pumped Safety Injection (SI) Flow Rate
Assumed for 2 - 6 Inch Small Breaks

Broken Loop SI Flows (Spill to RCS Pressure)

RCS Pressure (psig)	CCP *		RCS Pressure (psig)	SIP*		RCS Pressure (psig)	RHRP*	
	Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop
0	317.48	112.59	0	444.82	154.97	0	2834.19	1075.76
100	309.44	109.74	100	429.33	149.57	20	2660.73	1009.97
200	301.32	106.87	200	413.26	143.97	40	2477.16	940.33
300	293.09	103.95	300	396.55	138.15	60	2280.95	865.91
400	284.76	101.00	400	379.11	132.07	80	2068.49	785.32
500	276.27	97.99	500	360.82	125.69	100	1834.31	696.50
600	267.61	94.92	600	341.54	118.98	120	1569.20	595.91
700	258.77	91.79	700	321.09	111.85	140	1254.38	476.48
800	249.72	88.58	800	299.20	104.22	150	1065.81	404.92
900	240.41	85.29	900	275.52	95.97	155	959.33	364.53
1000	230.86	81.90	1000	249.49	86.90	160	839.61	319.10
1100	221.00	78.41	1100	220.21	76.69	170	534.36	203.24
1200	210.79	74.79	1200	185.98	64.77	175	295.67	112.61
1300	200.18	71.03	1300	142.84	49.74	180	0.00	0.00
1400	189.13	67.12	1400	72.76	25.32			
1500	177.55	63.01	1420	40.00	13.91			
1600	165.35	58.69	1440	0.00	0.00			
1800	138.49	49.17	1500	0.00	0.00			
2000	106.72	37.91						
2235	55.68	19.82						
2300	34.70	12.38						
2350	11.76	4.22						
2400	0.00	0.00						

*1 train operating

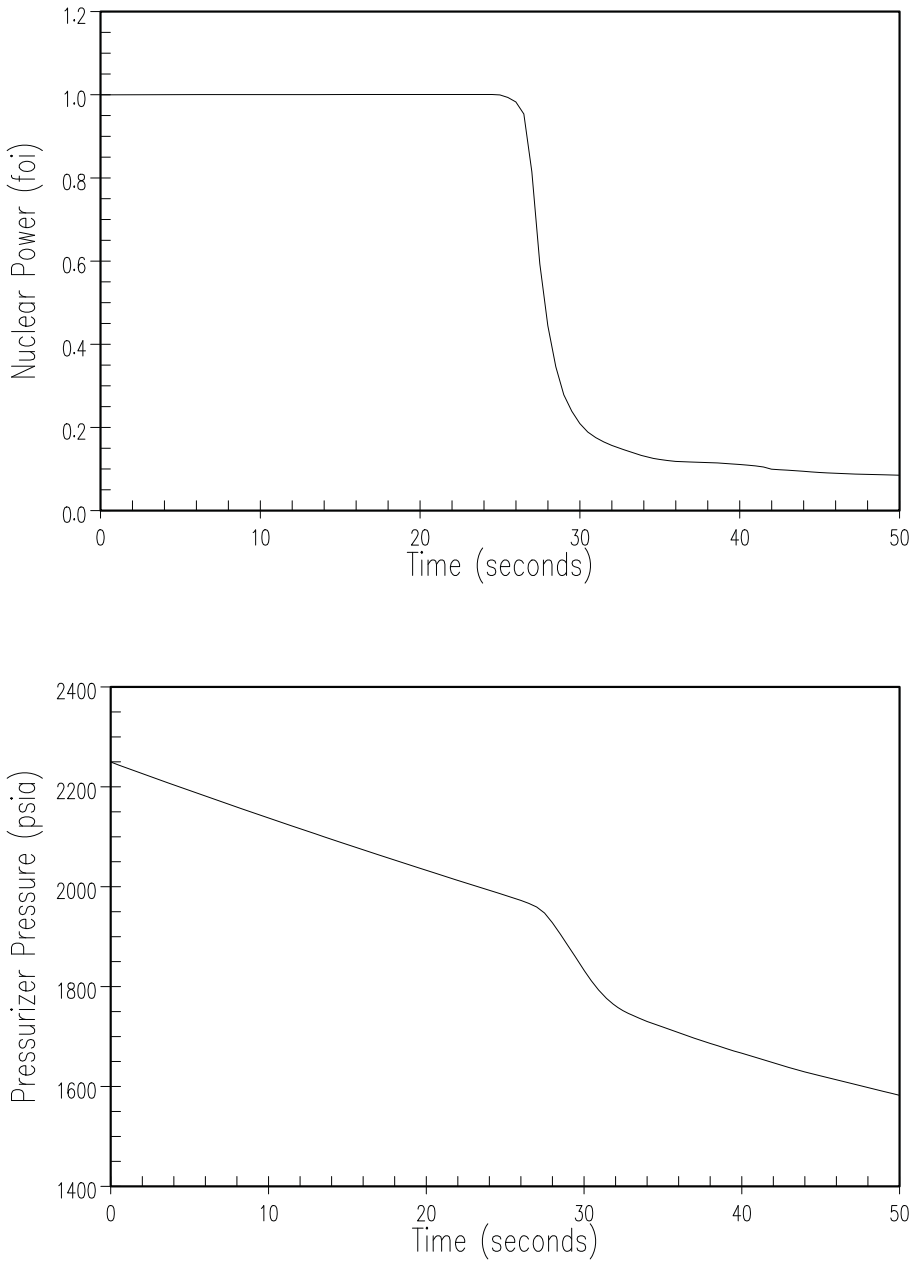
TABLE 15.6-14A

Pumped Safety Injection (SI) Flow Rate
Assumed for 8.75 inch Accumulator Line Break

Broken Loop SI Flows (Spill to Containment Pressure, 0 psig)

RCS Pressure (psig)	CCP *		RCS Pressure (psig)	SIP*		RCS Pressure (psig)	RHRP*	
	Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop
0	317.48	112.59	0	444.82	154.97	0	2834.19	1075.76
100	301.07	122.40	100	424.60	159.61	20	2579.31	1163.27
200	284.72	131.75	200	403.66	164.22	40	2310.43	1248.78
300	268.39	140.72	300	381.93	168.80	60	2023.36	1333.04
400	252.01	149.36	400	359.28	173.35	80	1712.00	1416.88
500	235.50	157.73	500	335.54	177.90	100	1366.47	1501.38
600	218.82	165.85	600	310.53	182.44	120	967.94	1588.36
700	201.92	173.76	700	283.99	187.00	140	464.71	1683.57
800	184.73	181.49	800	255.56	191.59	150	186.62	1726.84
900	167.18	189.06	900	224.71	196.24	155	7.15	1751.62
1000	149.19	196.49	1000	190.63	200.98	160	0.00	1752.56
1100	130.67	203.80	1100	151.88	205.88	170	0.00	1752.56
1200	111.45	211.03	1200	105.54	211.07	175	0.00	1752.56
1300	91.40	218.18	1300	43.25	216.89	180	0.00	1752.56
1400	70.27	225.28	1380	0.00	220.15			
1500	47.68	232.37	1400	0.00	220.15			
1600	23.08	239.49						
1650	9.73	243.09						
1700	0.00	246.13						
1800	0.00	250.38						
2000	0.00	251.94						

*1 train operating



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Wolf Creek UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-1 NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE

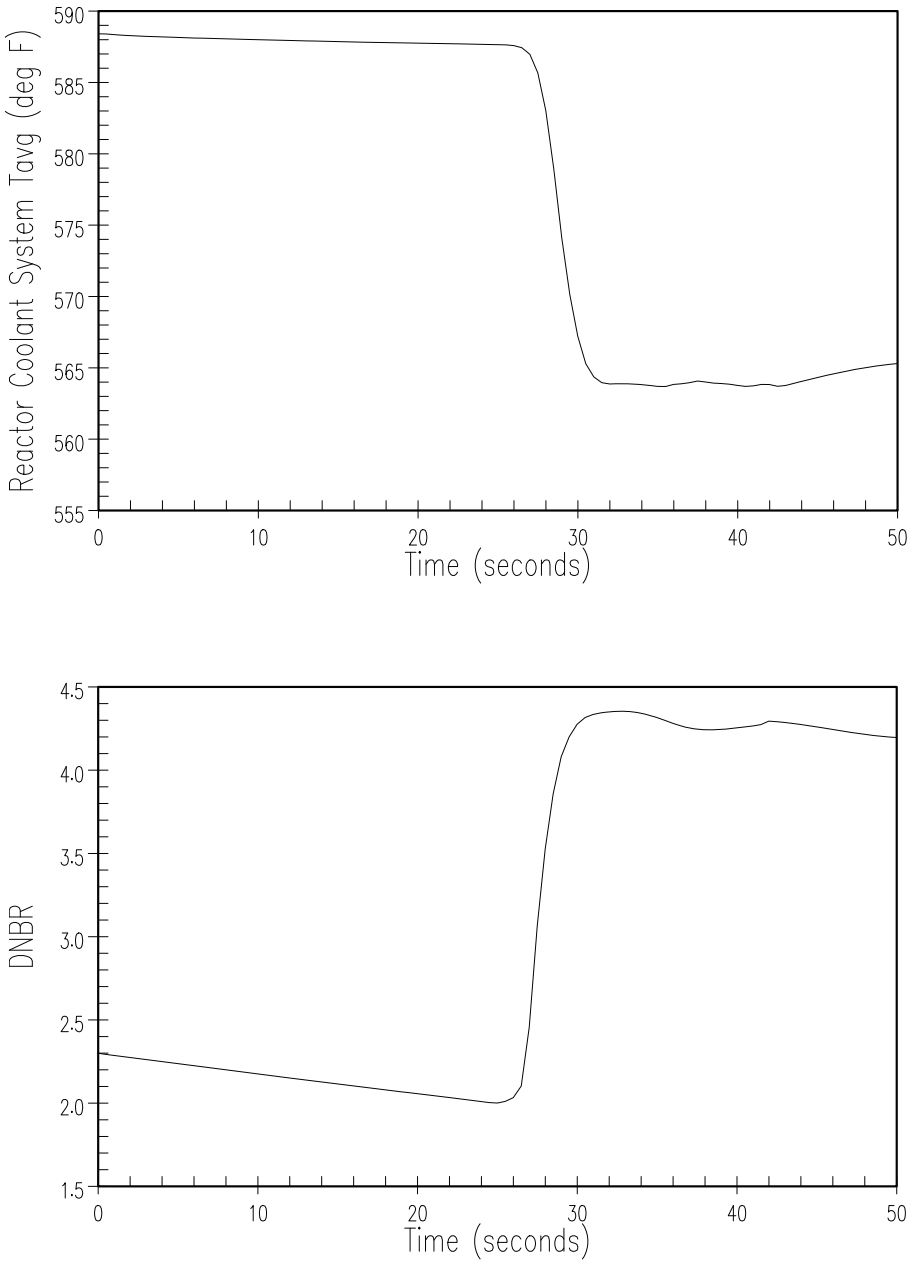

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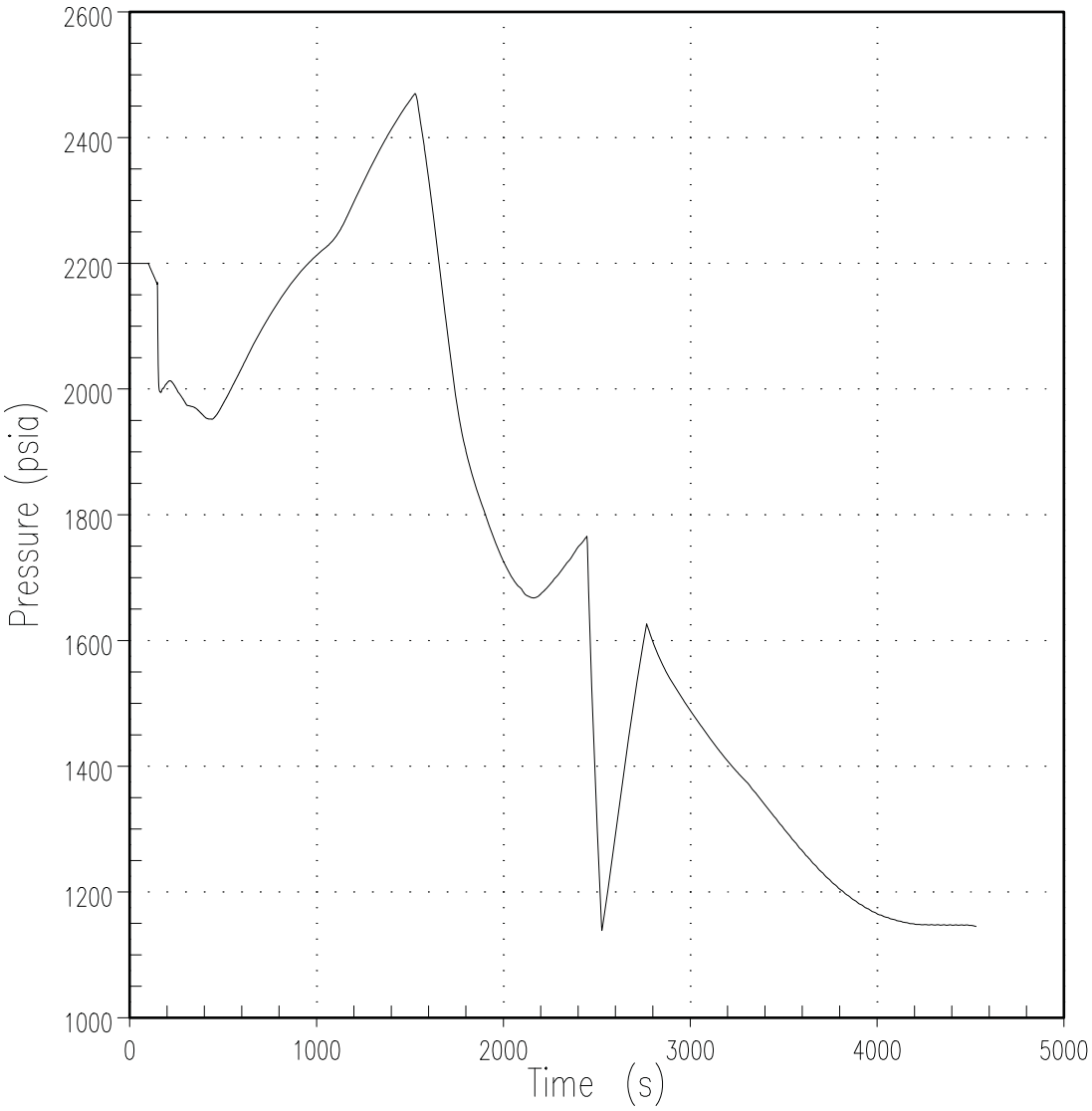
FIGURE 15.6-2 REACTOR COOLANT AVERAGE TEMPERATURE AND DNBR TRANSIENTS FOR INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3A: SGTR MARGIN TO OVERFILL PRESSURIZER PRESSURE

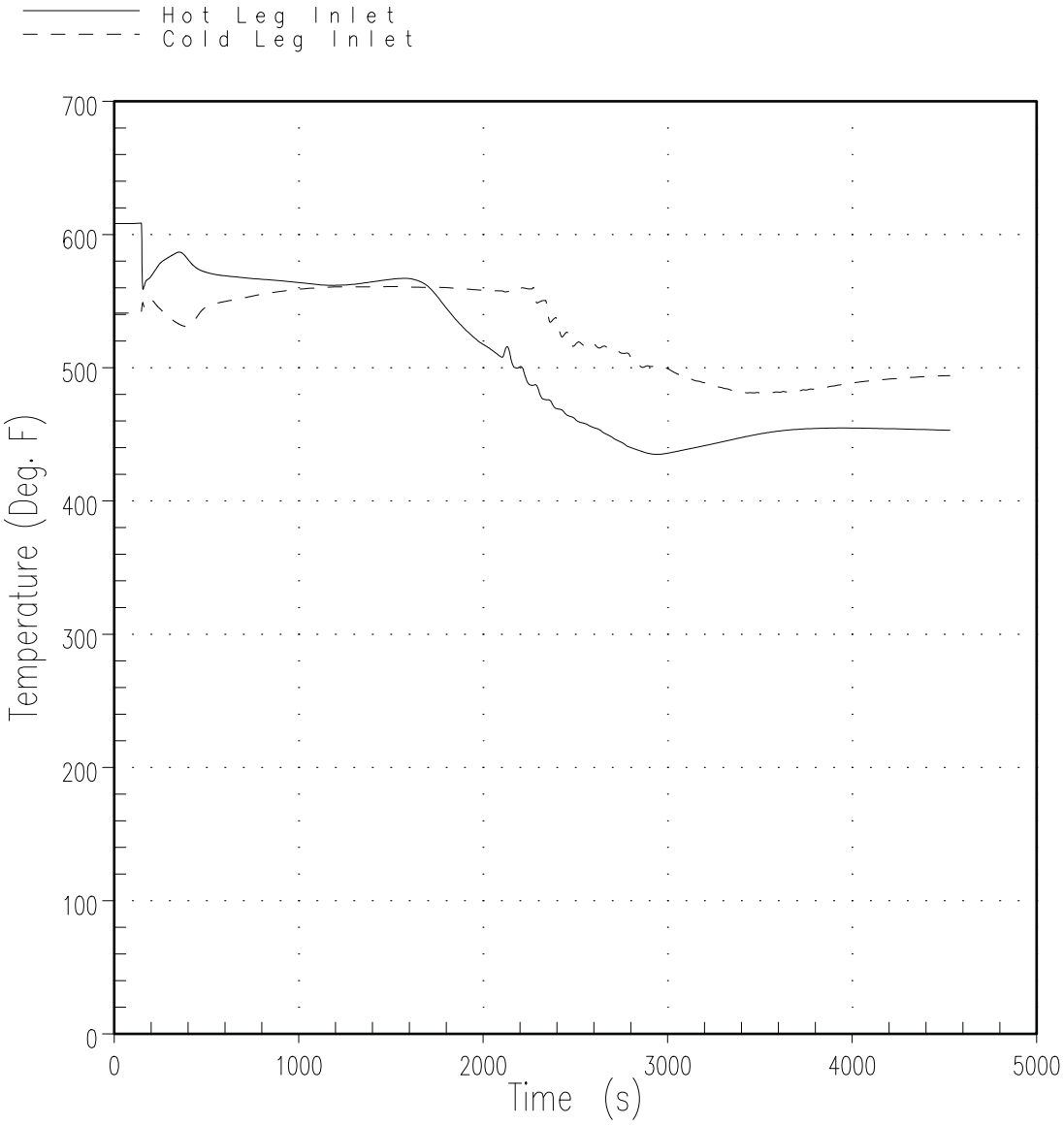

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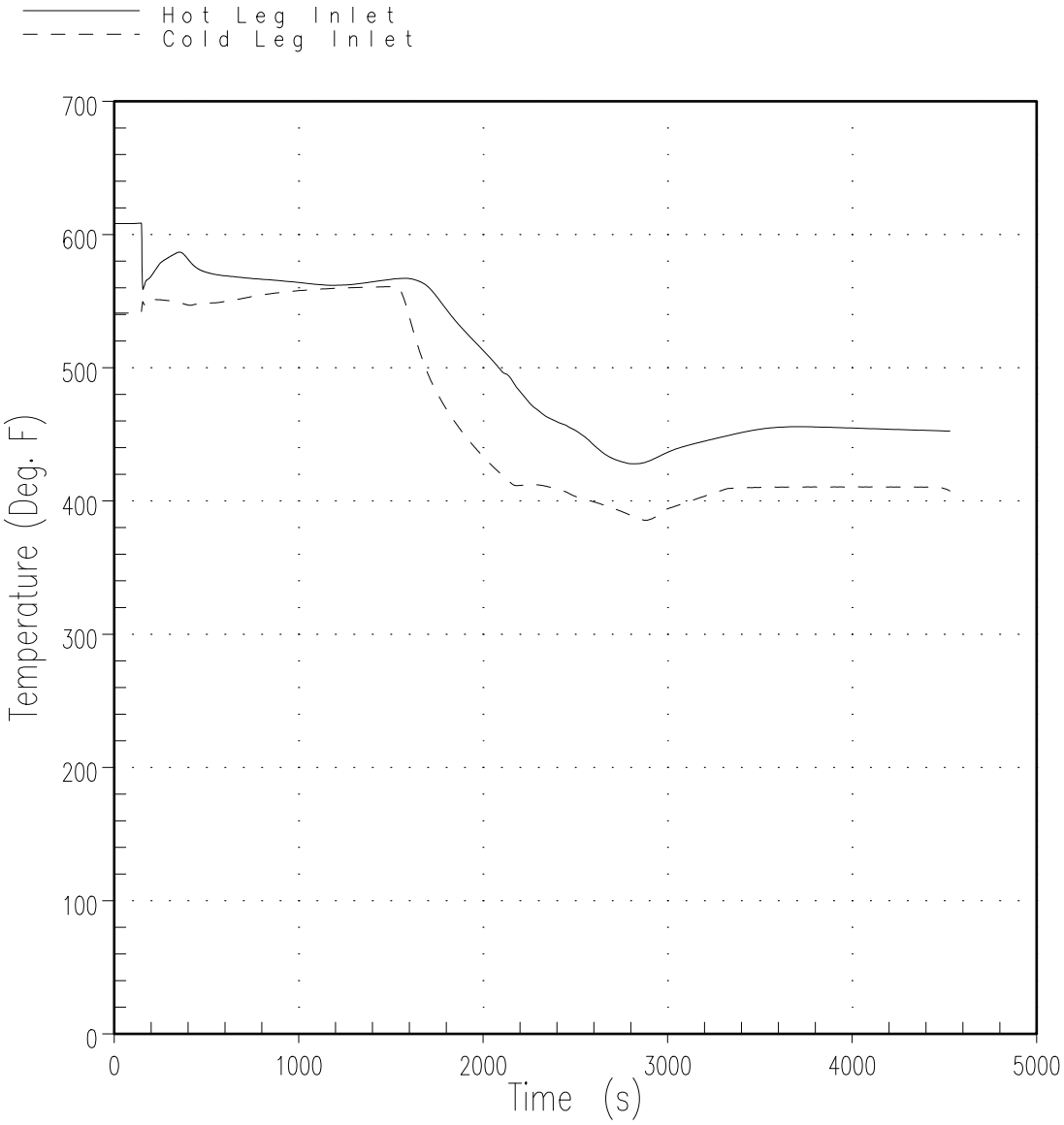
FIGURE 15.6-3B: SGTR MARGIN TO OVERFILL RUPTURED LOOP RCS TEMPERATURE



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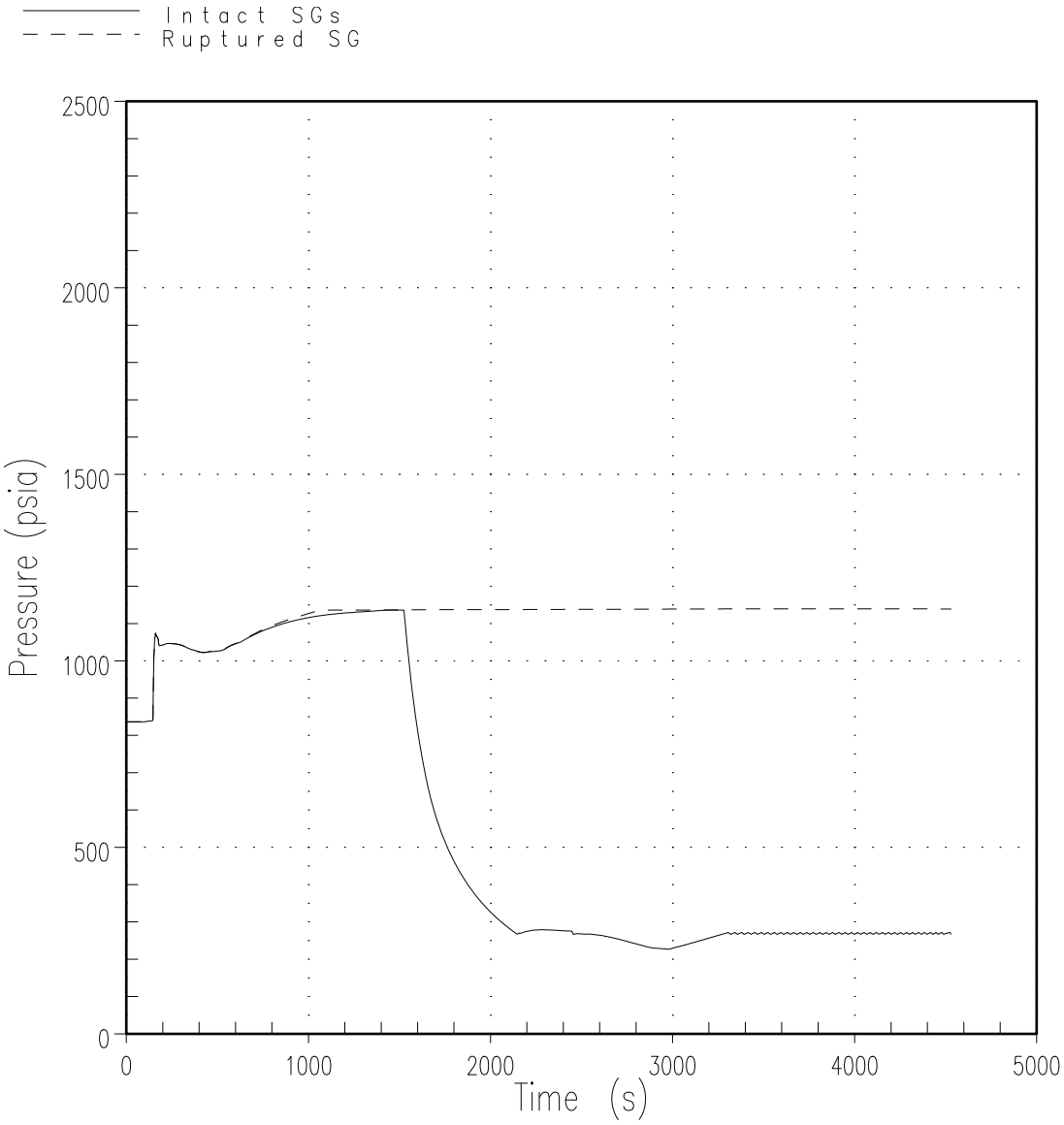
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3C: SGTR MARGIN TO OVERFILL INTACT LOOP RCS TEMPERATURE



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3D: SGTR MARGIN TO OVERFILL STEAM GENERATOR PRESSURE

FIGURE 15.6-3E HAS BEEN DELETED

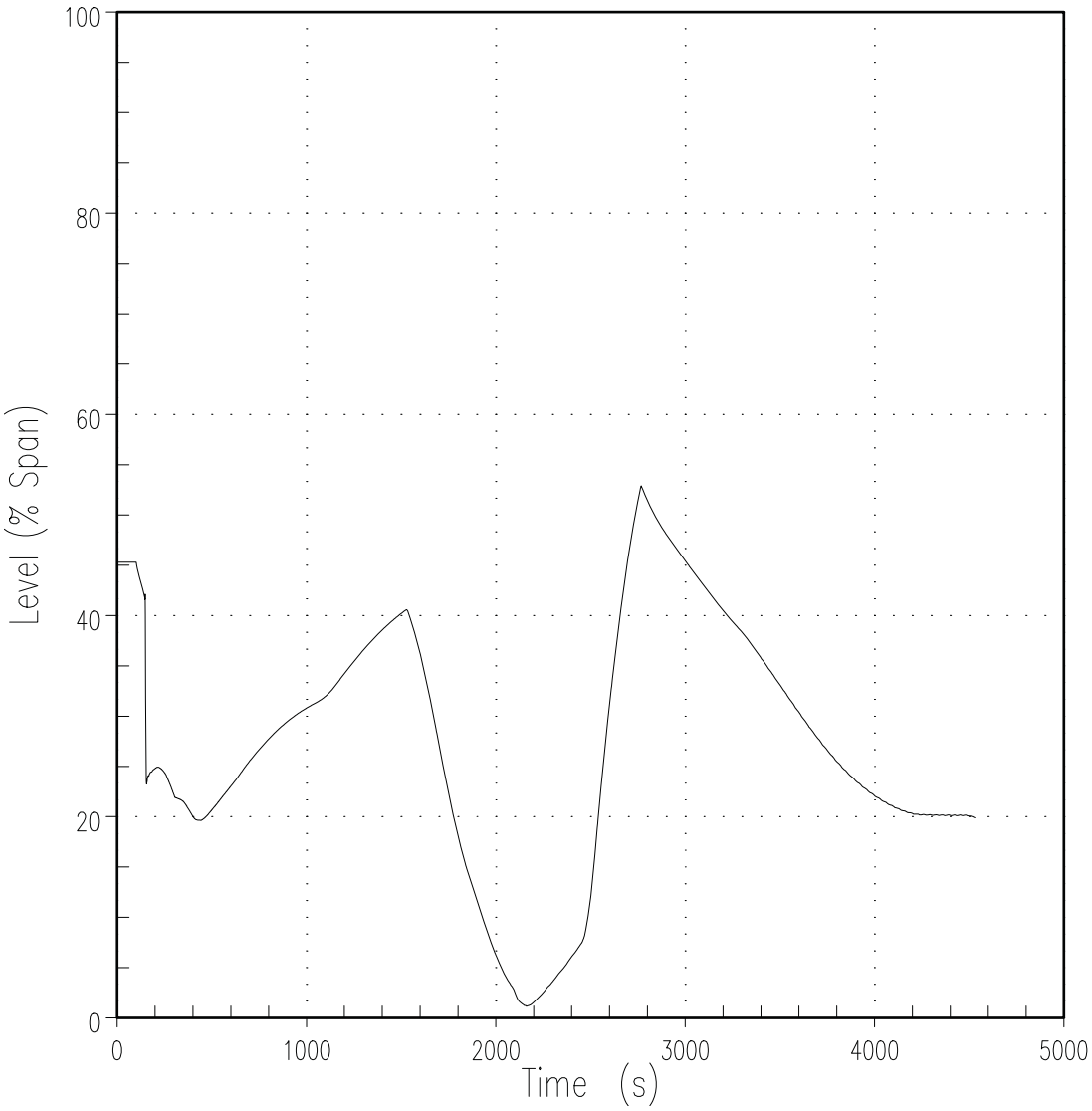

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FIGURE 15.6-3F: SGTR MARGIN TO OVERFILL PRESSURIZER LEVEL

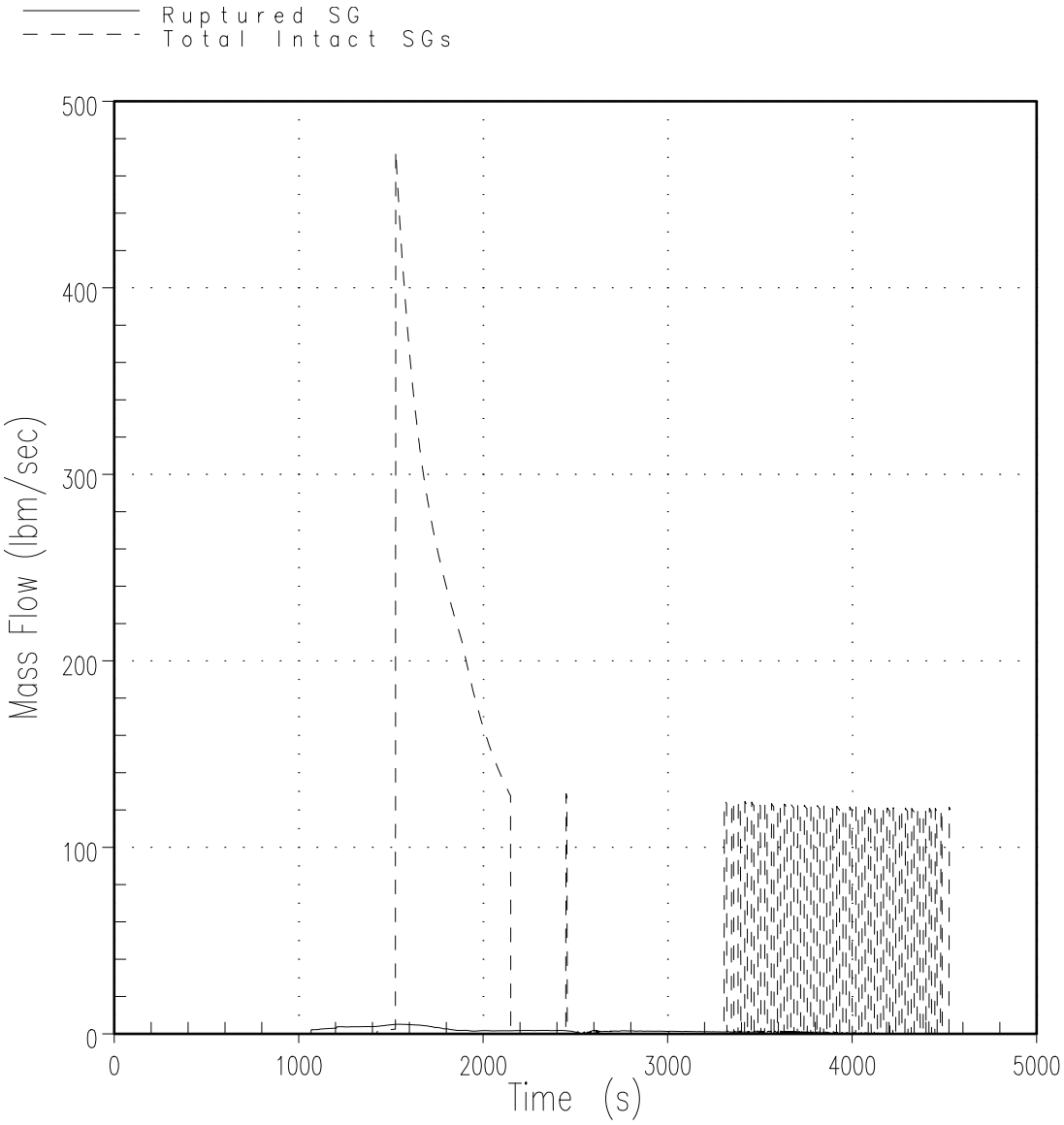

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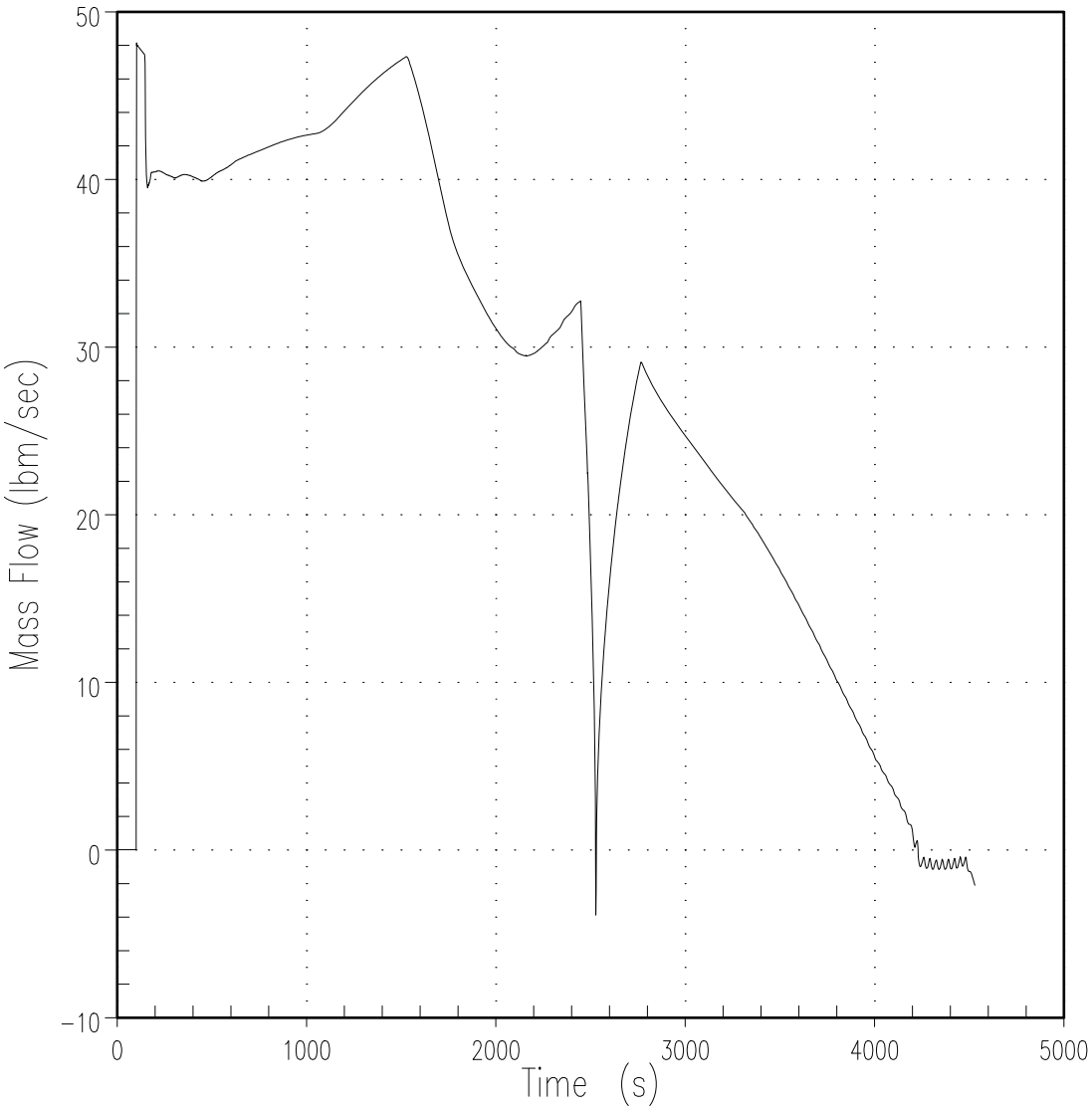
FIGURE 15.6-3G: SGTR MARGIN TO OVERFILL STEAM GENERATOR STEAM RELEASES

FIGURE 15.6-3H HAS BEEN DELETED



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3I: SGTR MARGIN TO OVERFILL BREAK FLOW

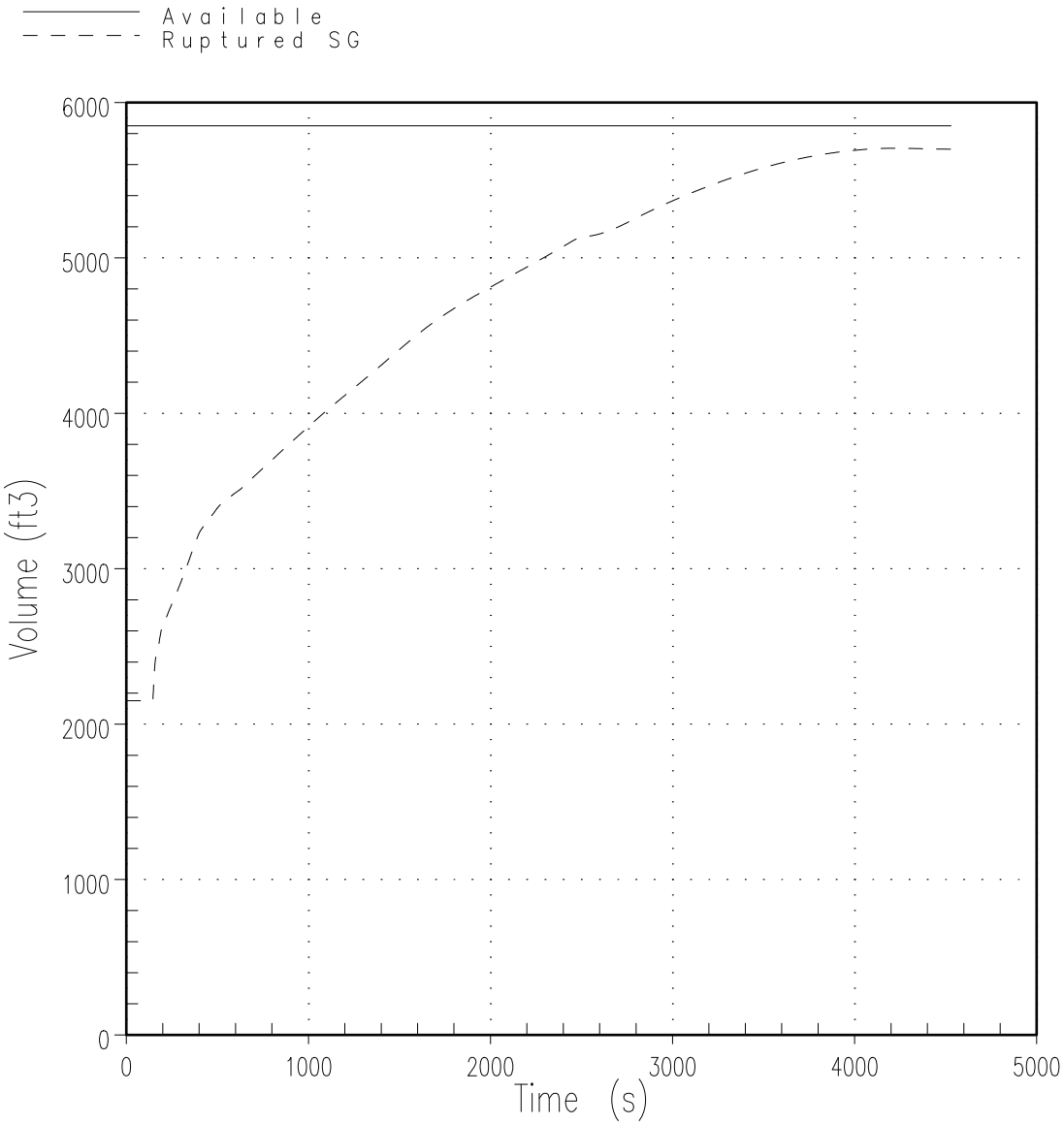

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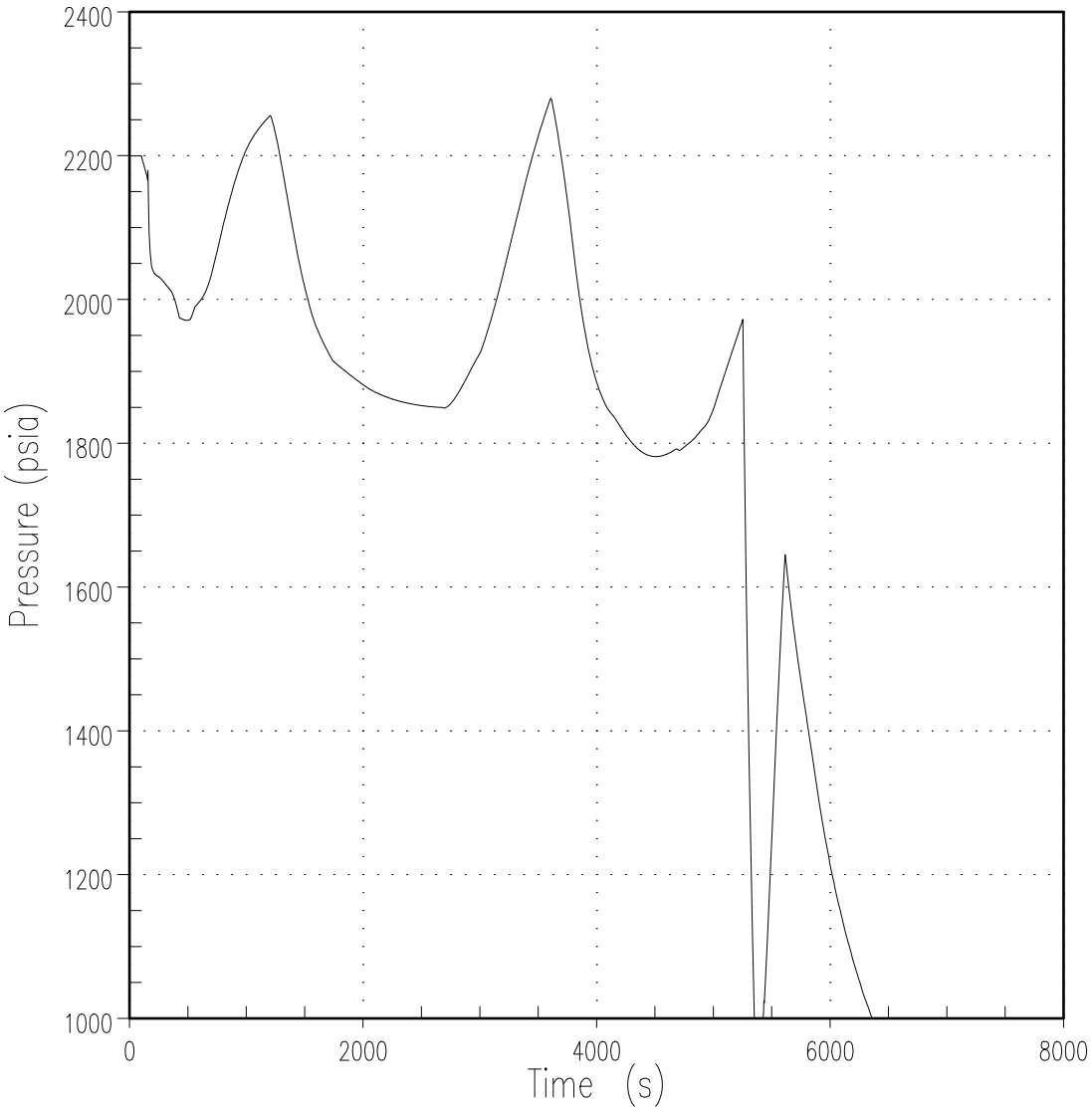
FIGURE 15.6-3J: SGTR MARGIN TO OVERFILL RUPTURED STEAM GENERATOR WATER VOLUME



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3K: SGTR THERMAL HYDRAULIC PRESSURIZER PRESSURE

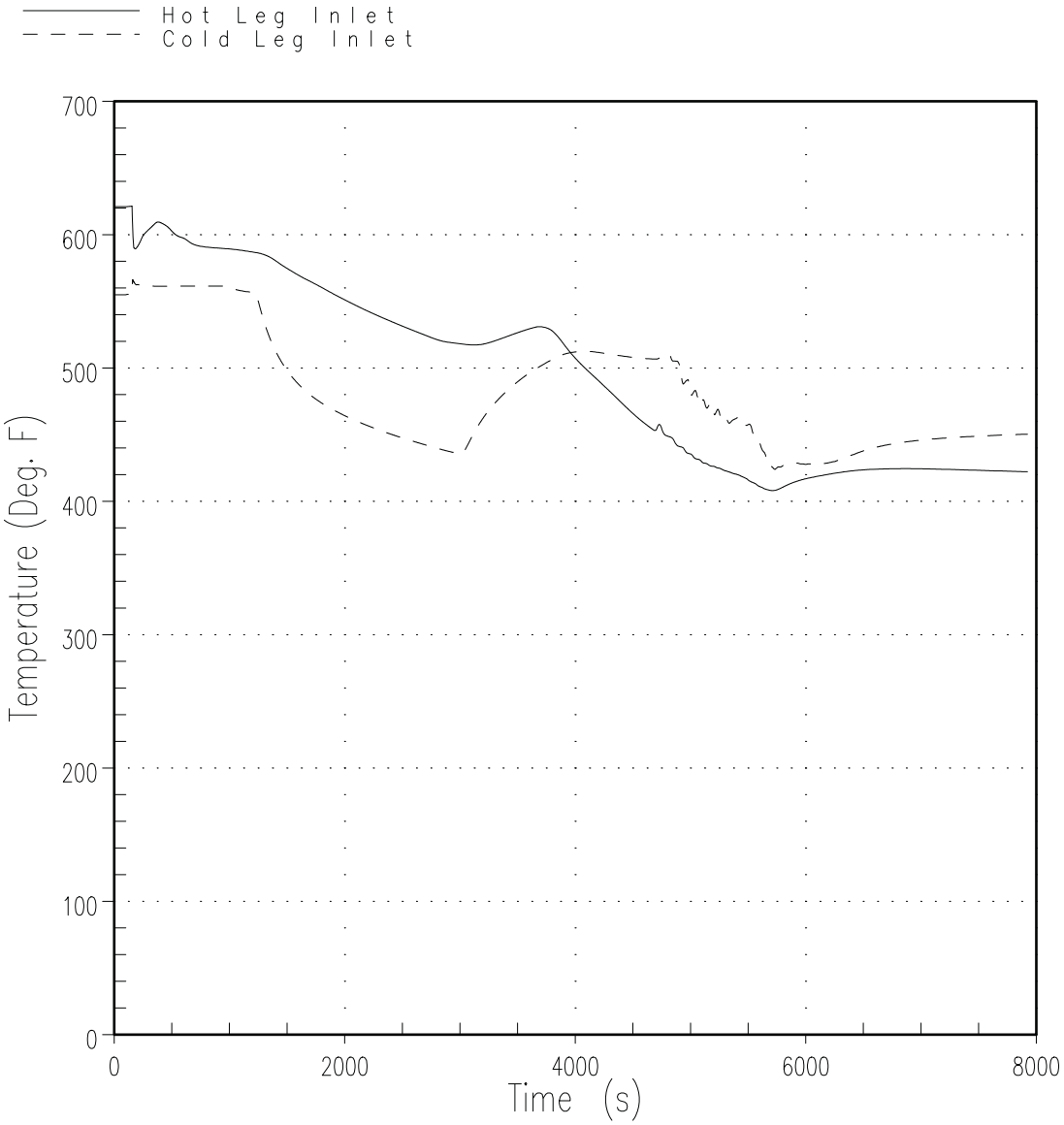

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FIGURE 15.6-3L: SGTR THERMAL HYDRAULIC RUPTURED LOOP RCS TEMPERATURE

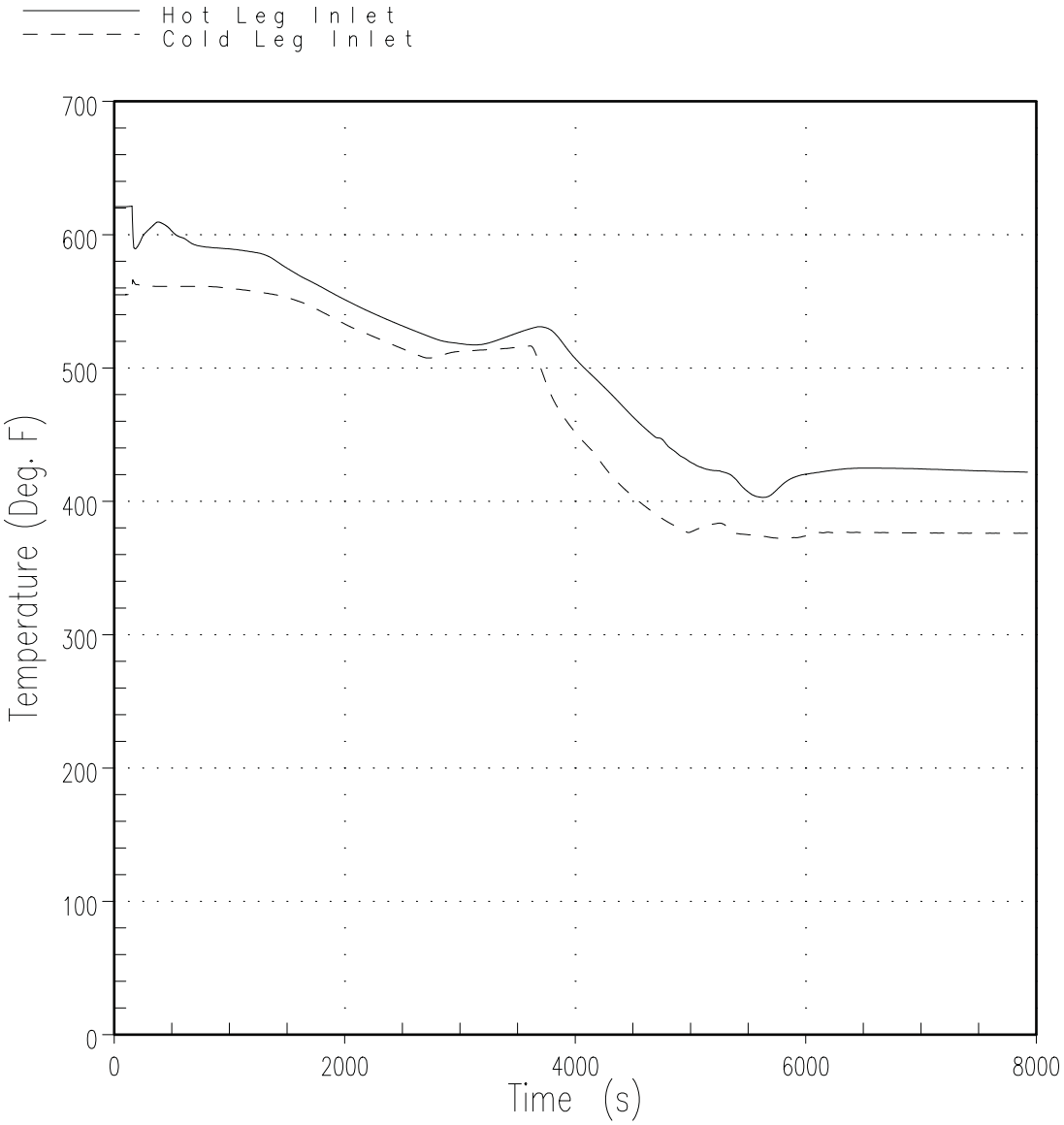

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FIGURE 15.6-3M: SGTR THERMAL HYDRAULIC INTACT LOOP RCS TEMPERATURE

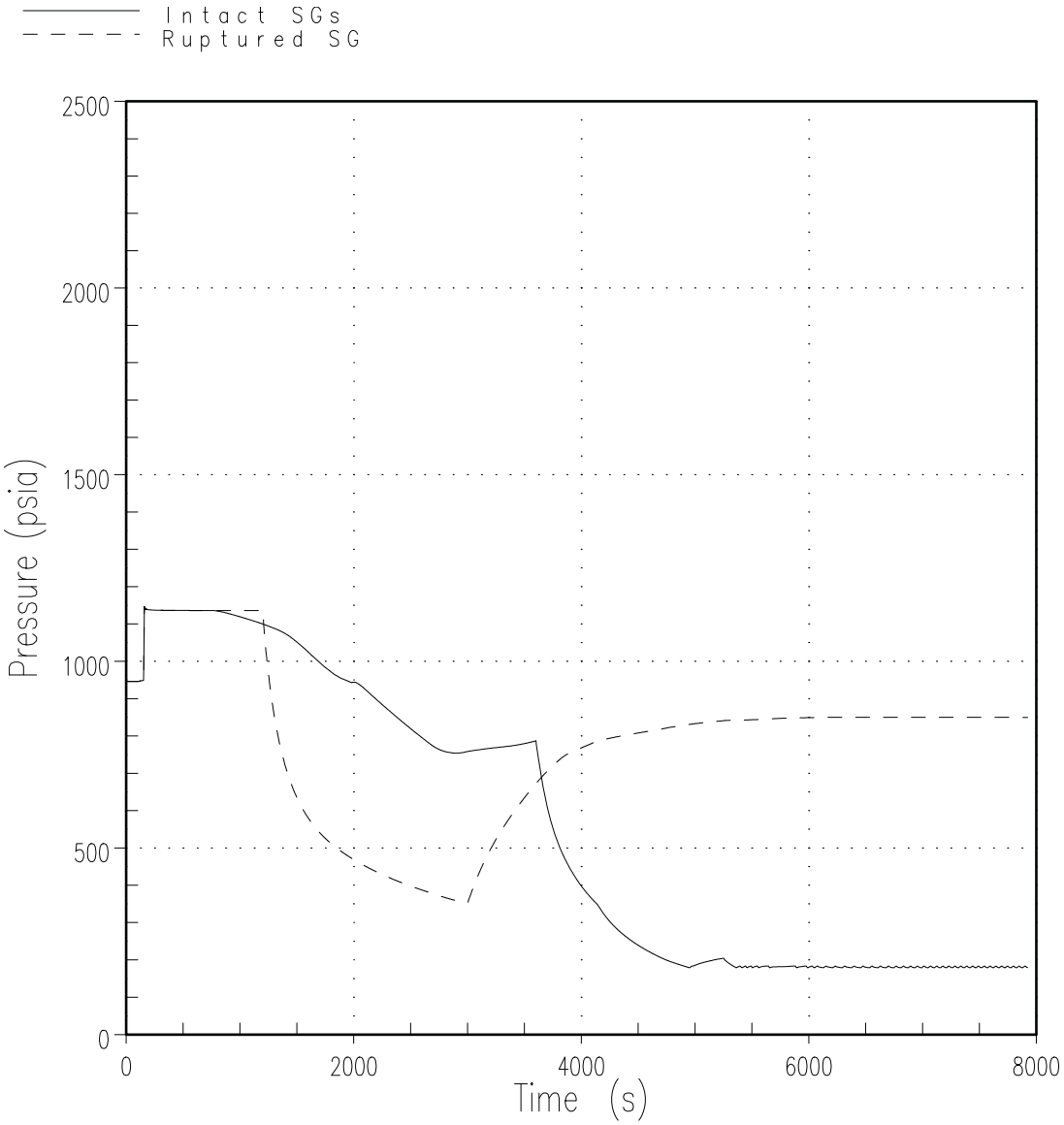

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FIGURE 15.6-3N: SGTR THERMAL HYDRAULIC STEAM GENERATOR PRESSURE

FIGURE 15.6-3O HAS BEEN DELETED

FIGURE 15.6-3O
STEAM GENERATOR TEMPERATURE
TRANSIENT SGTR W/ STUCK-OPEN
ATMOSPHERIC RELIEF VALVE

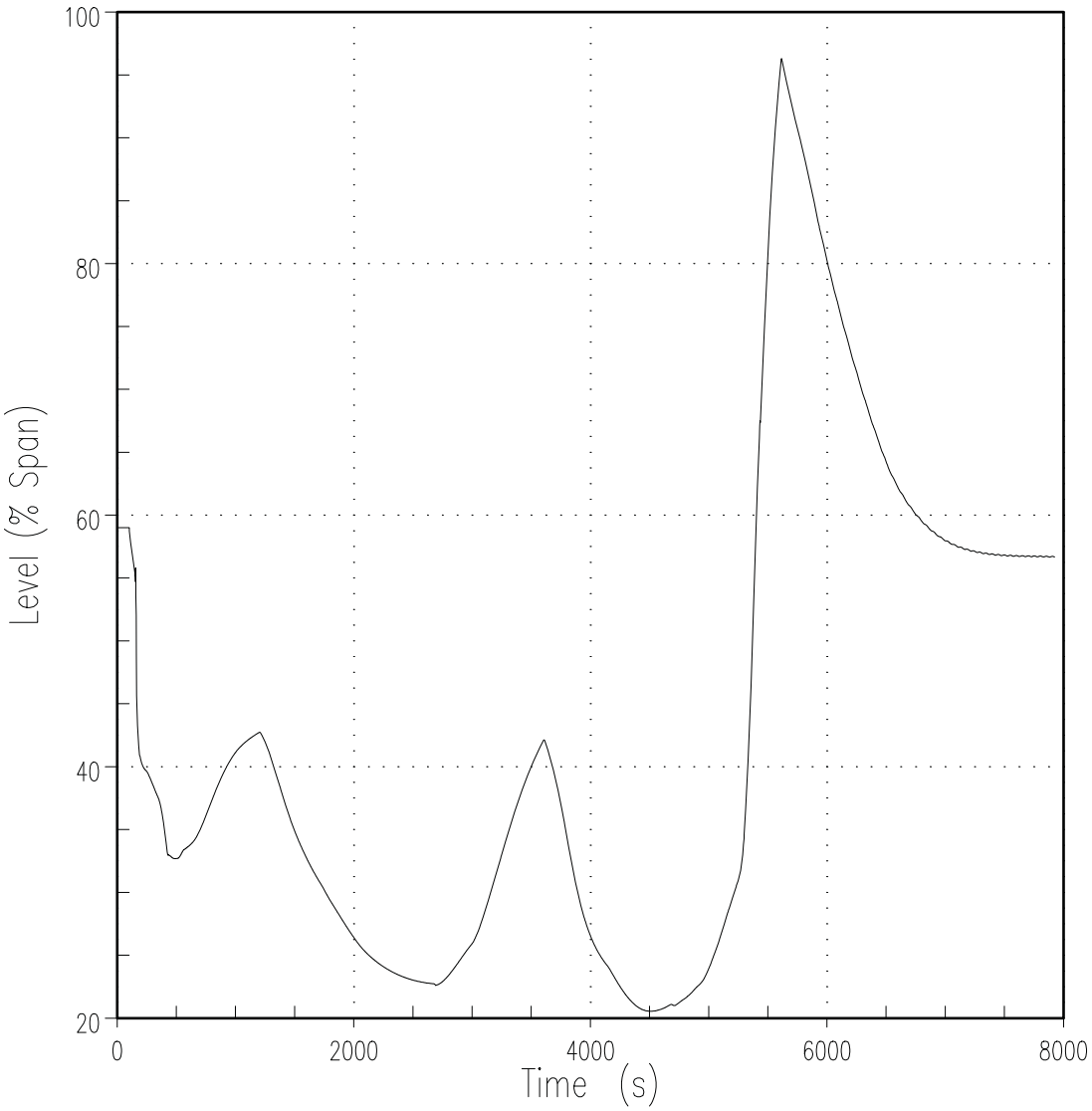

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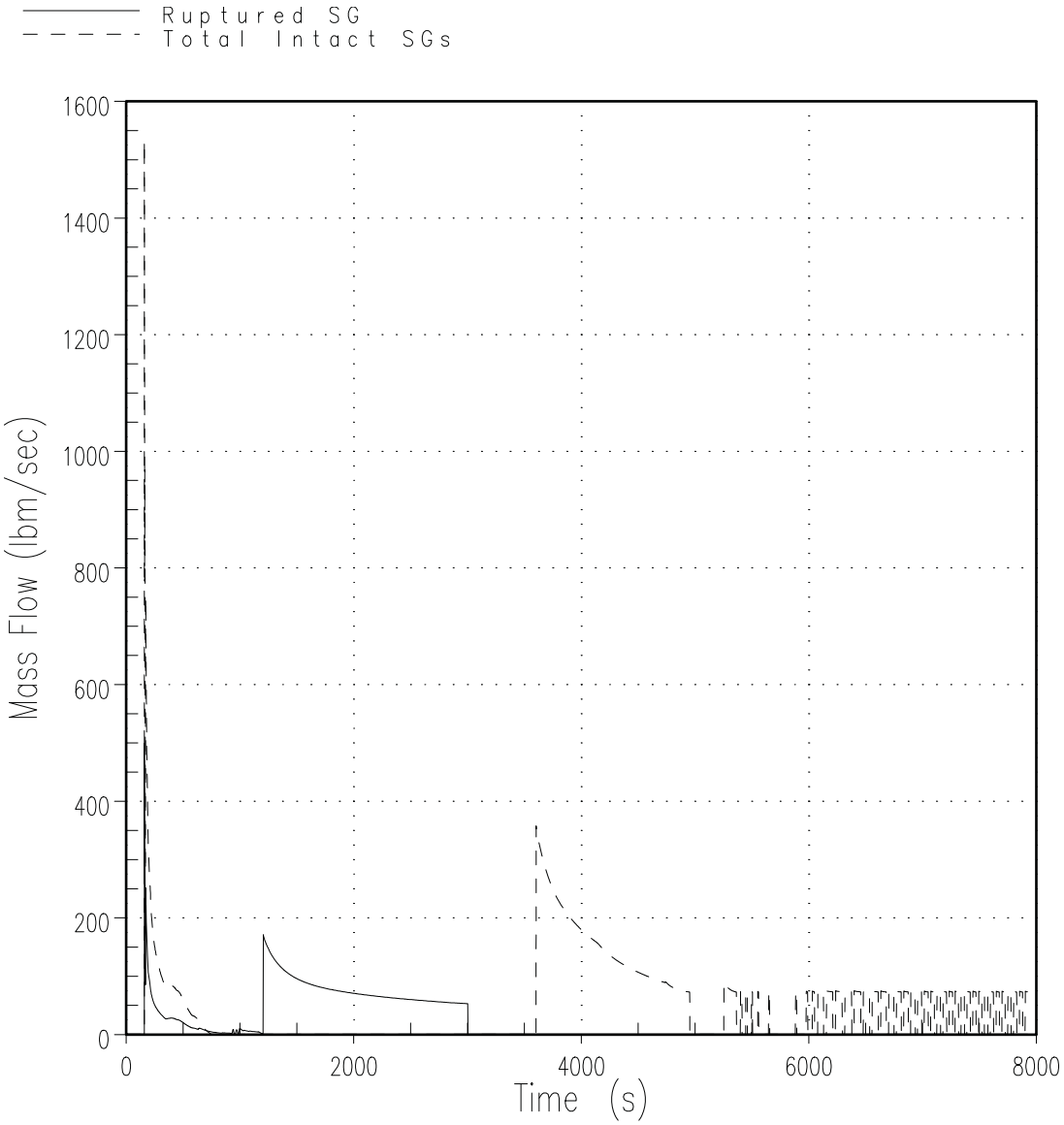
FIGURE 15.6-3P: SGTR THERMAL HYDRAULIC PRESSURIZER LEVEL



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3Q: SGTR THERMAL HYDRAULIC STEAM GENERATOR STEAM RELEASES

FIGURE 15.6-3R HAS BEEN DELETED

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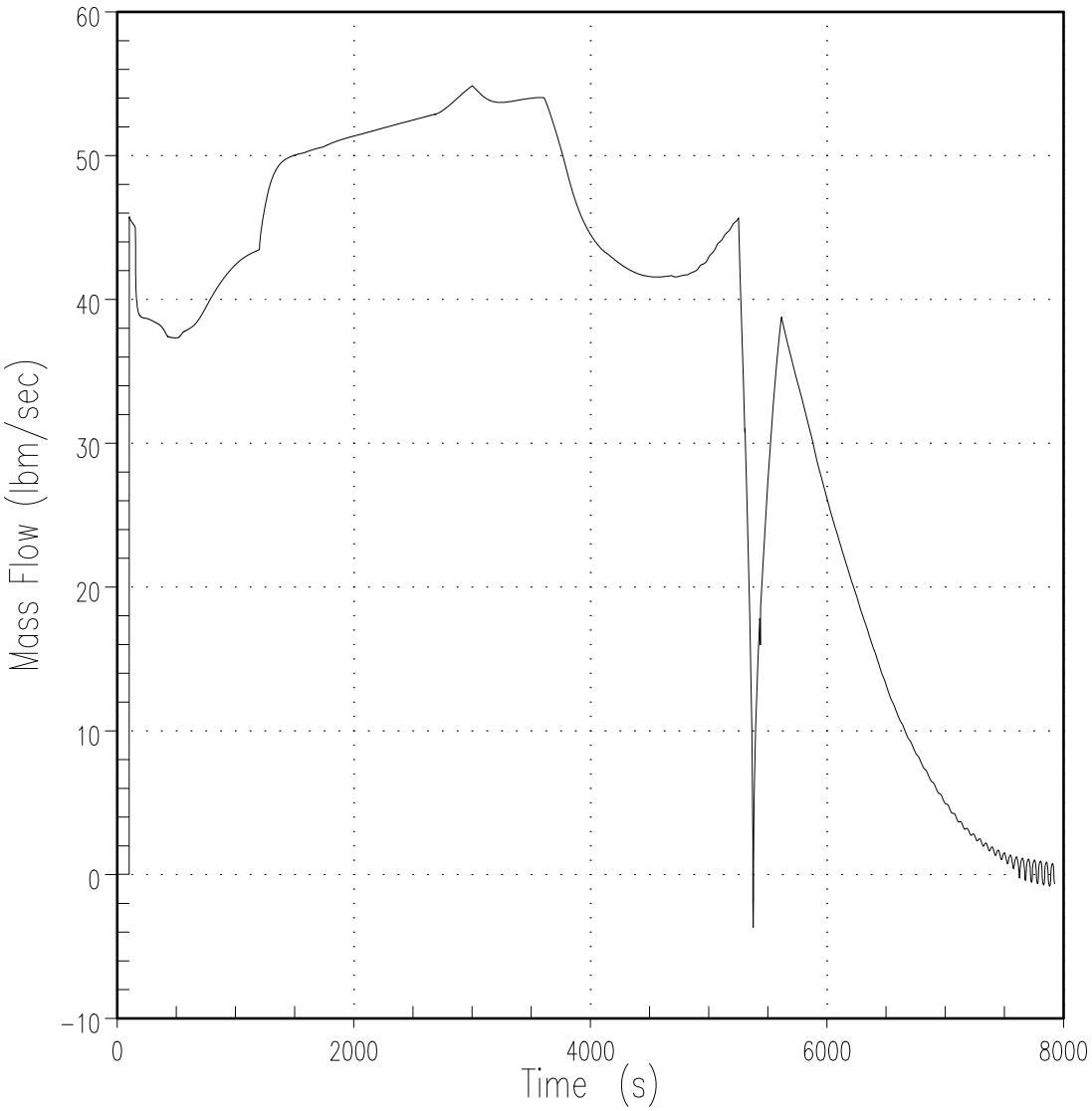
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.6-3R
FAULTED STEAM GENERATOR AFW AND NR LEVEL TRANSIENT SGTR W/ STUCK-OPEN ATMOSPHERIC RELIEF VALVE



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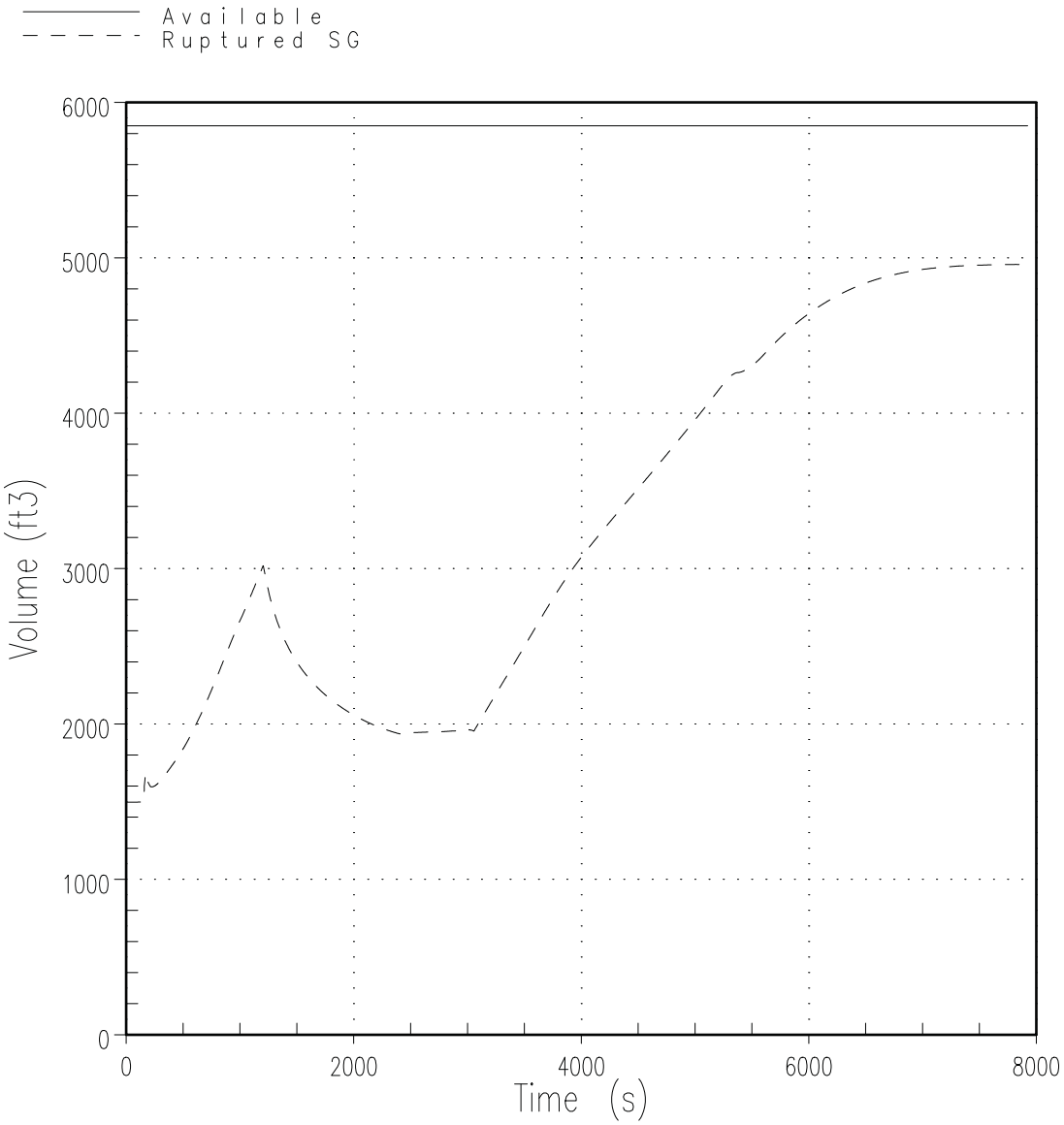
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3S: SGTR THERMAL HYDRAULIC BREAK FLOW



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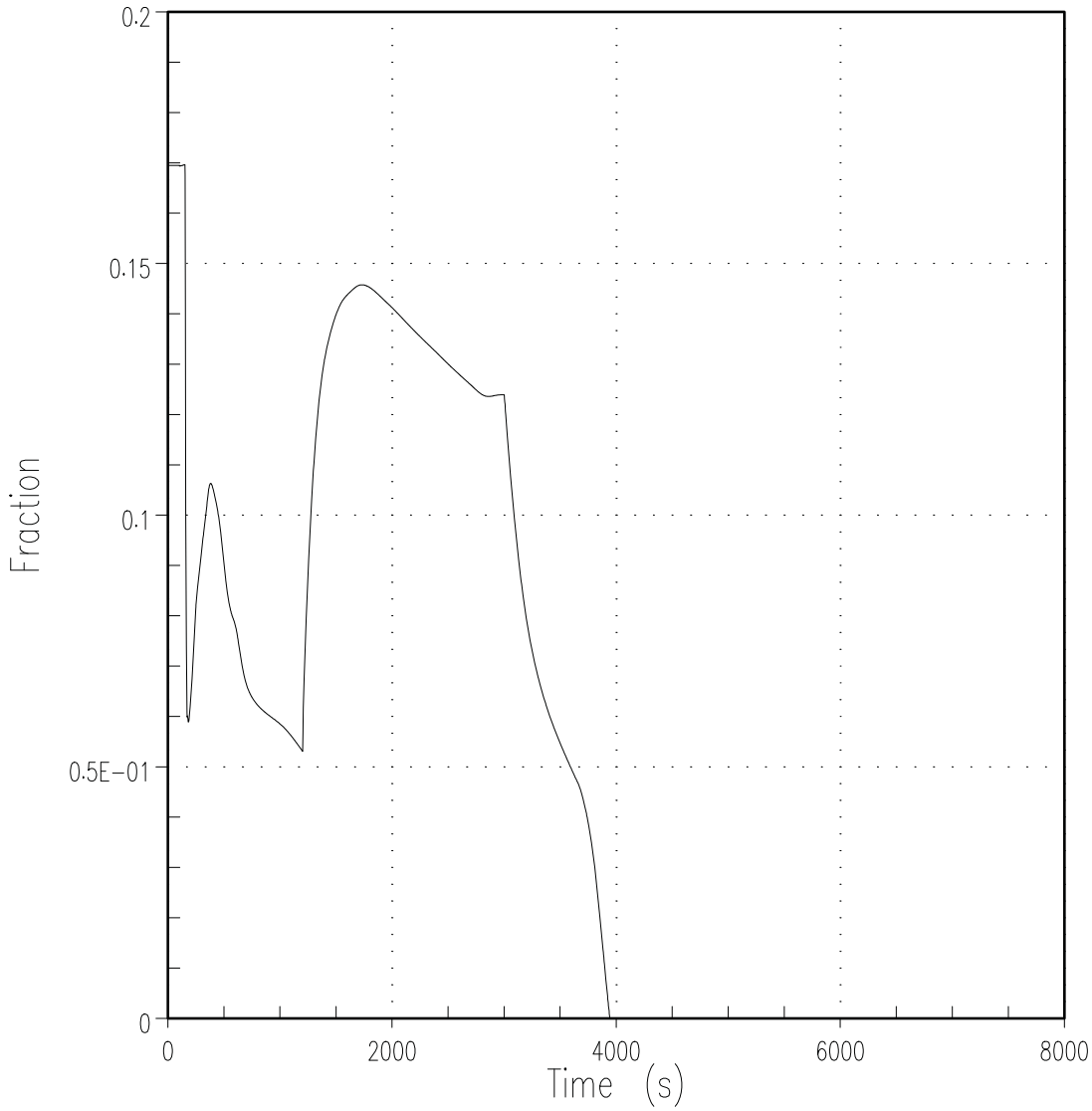
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3T: SGTR THERMAL HYDRAULIC RUPTURED STEAM GENERATOR WATER VOLUME



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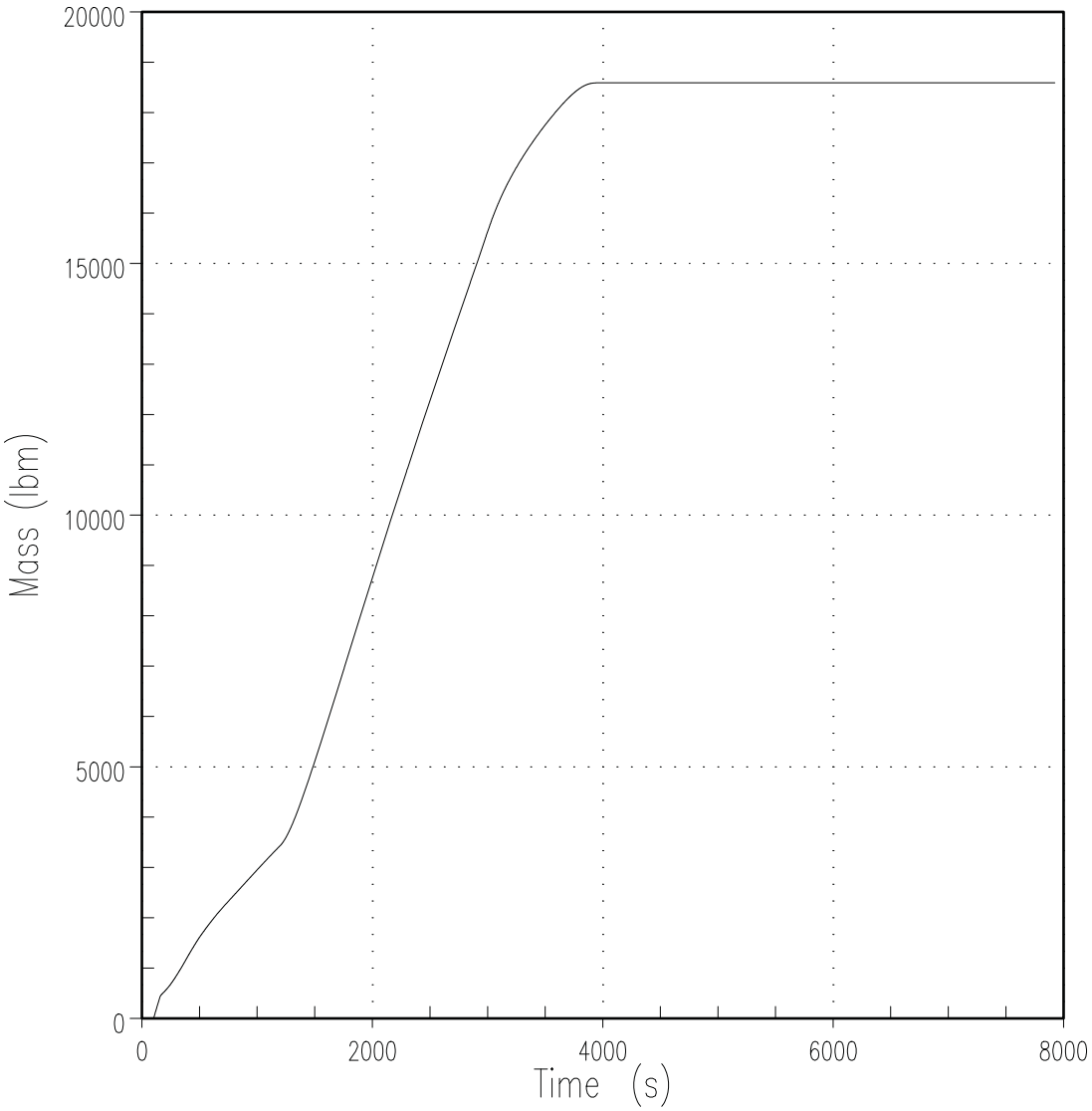
WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3U: SGTR THERMAL HYDRAULIC BREAK FLOW FLASHING FRACTION



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 15.6-3V: SGTR THERMAL HYDRAULIC TOTAL FLASHED BREAK FLOW

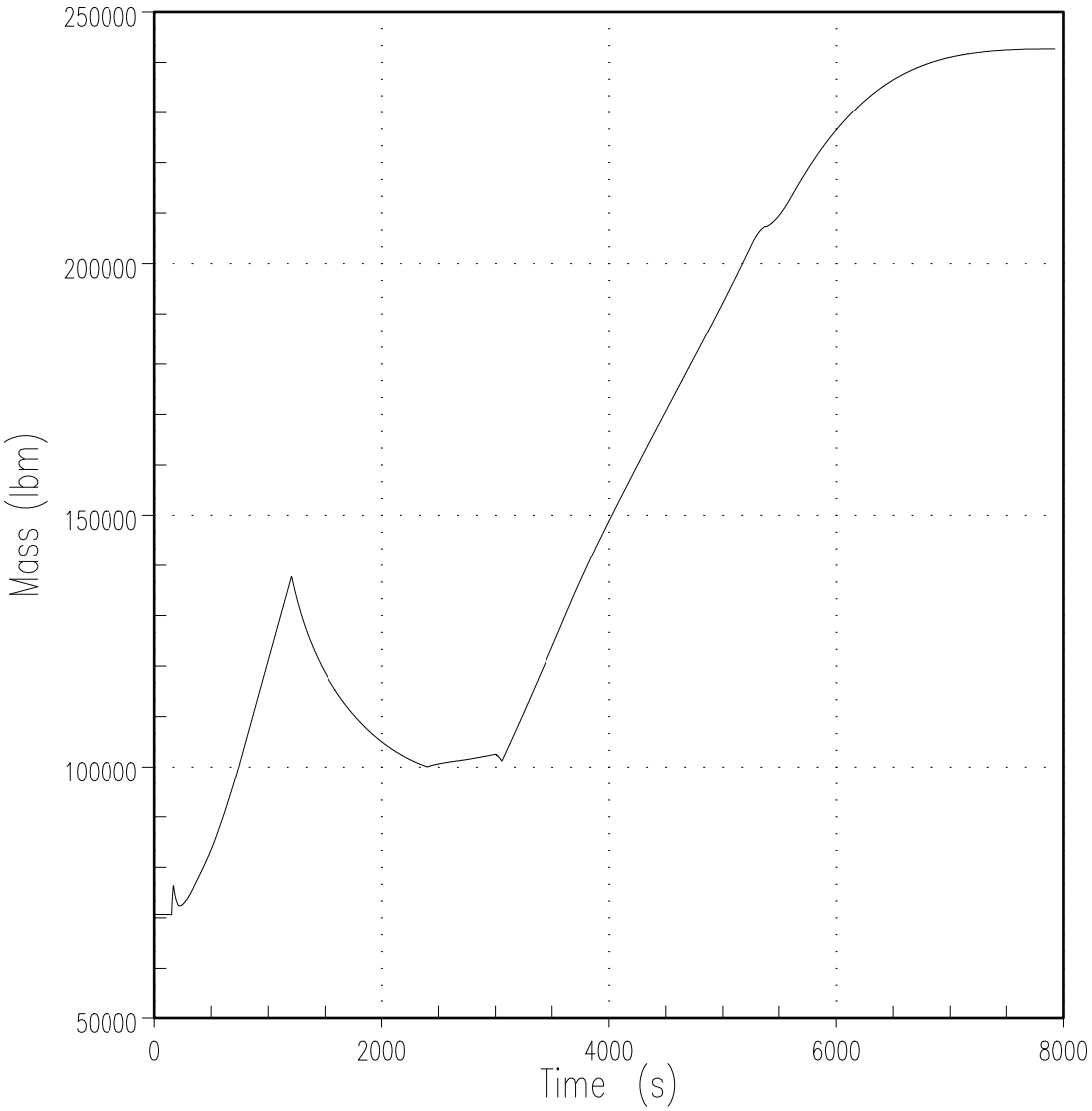

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FIGURE 15.6-3W: SGTR THERMAL HYDRAULIC RUPTURED STEAM GENERATOR FLUID MASS

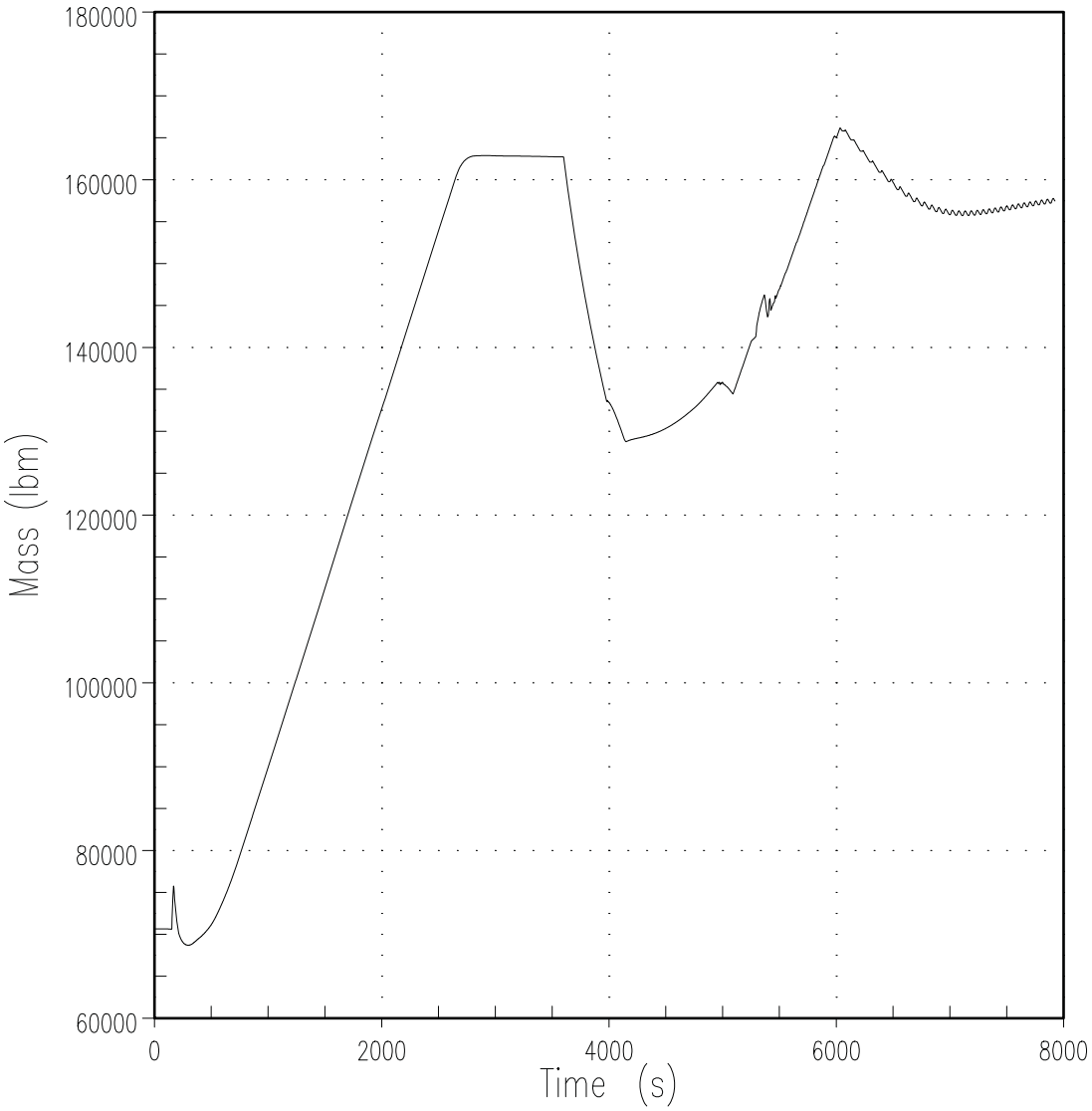

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FIGURE 15.6-3X: SGTR THERMAL HYDRAULIC INTACT STEAM GENERATOR FLUID MASS

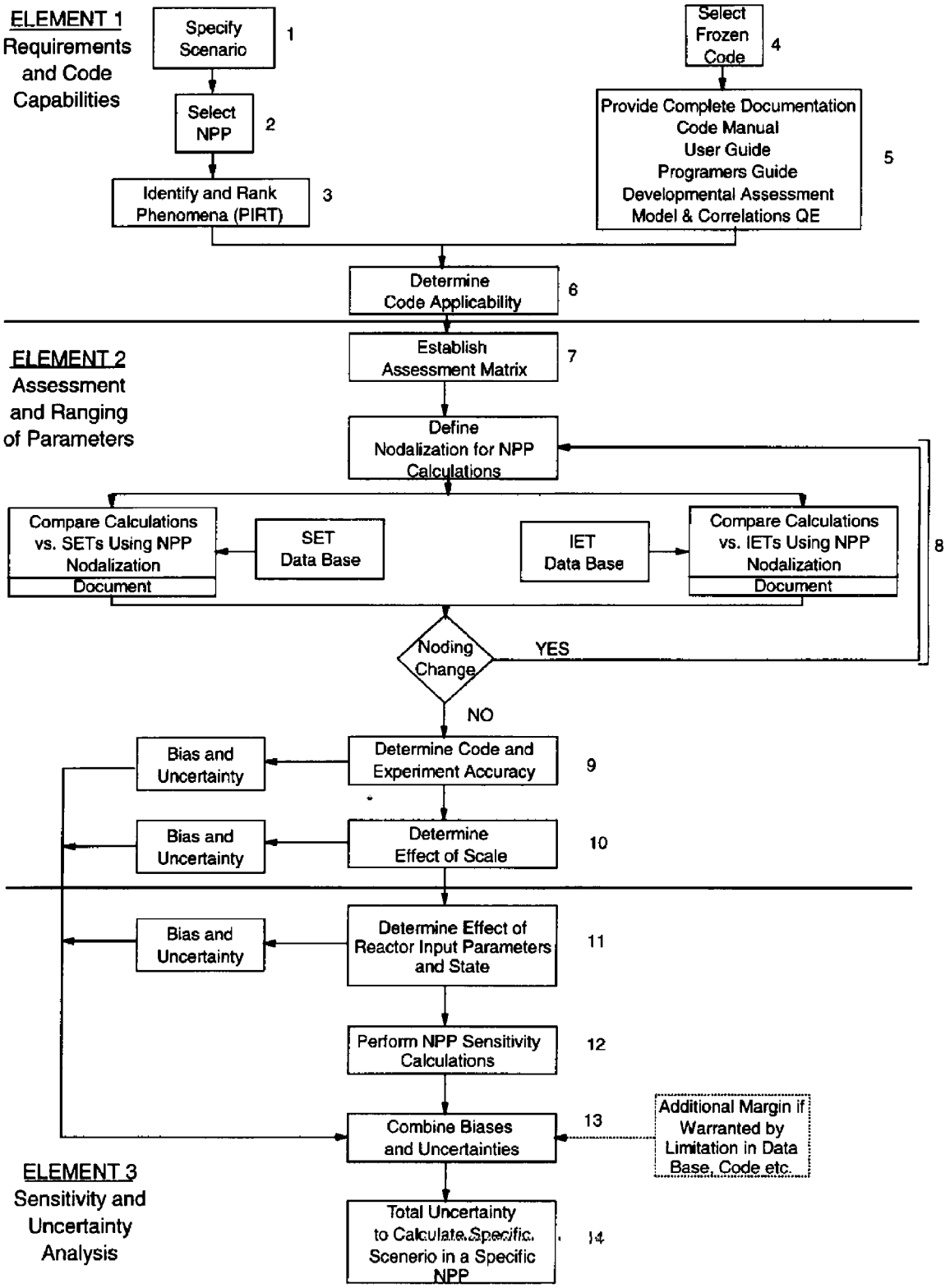

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FIGURE 15.6-5

CODE SCALING, APPLICABILITY, AND UNCERTAINTY (CSAU) EVALUATION METHODOLOGY

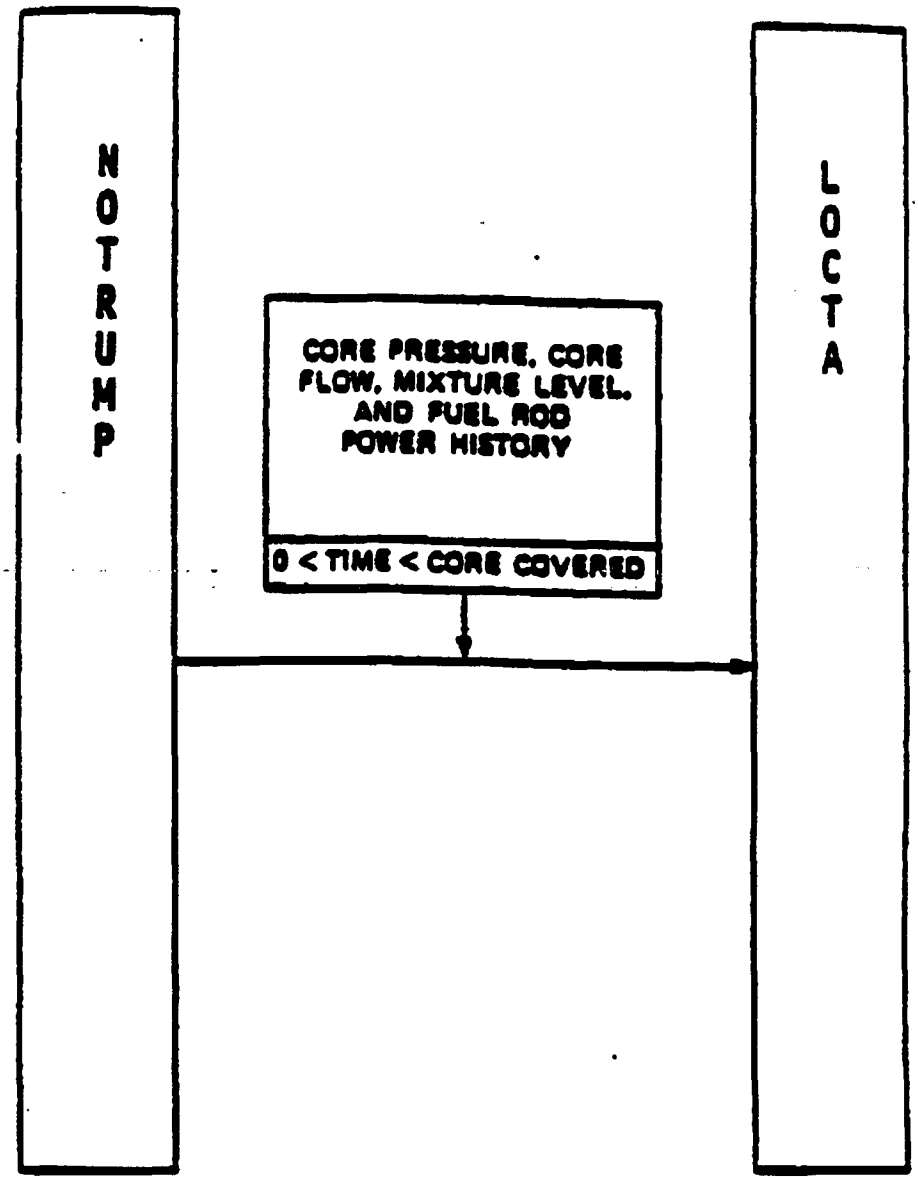

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FIGURE 15.6-6
CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

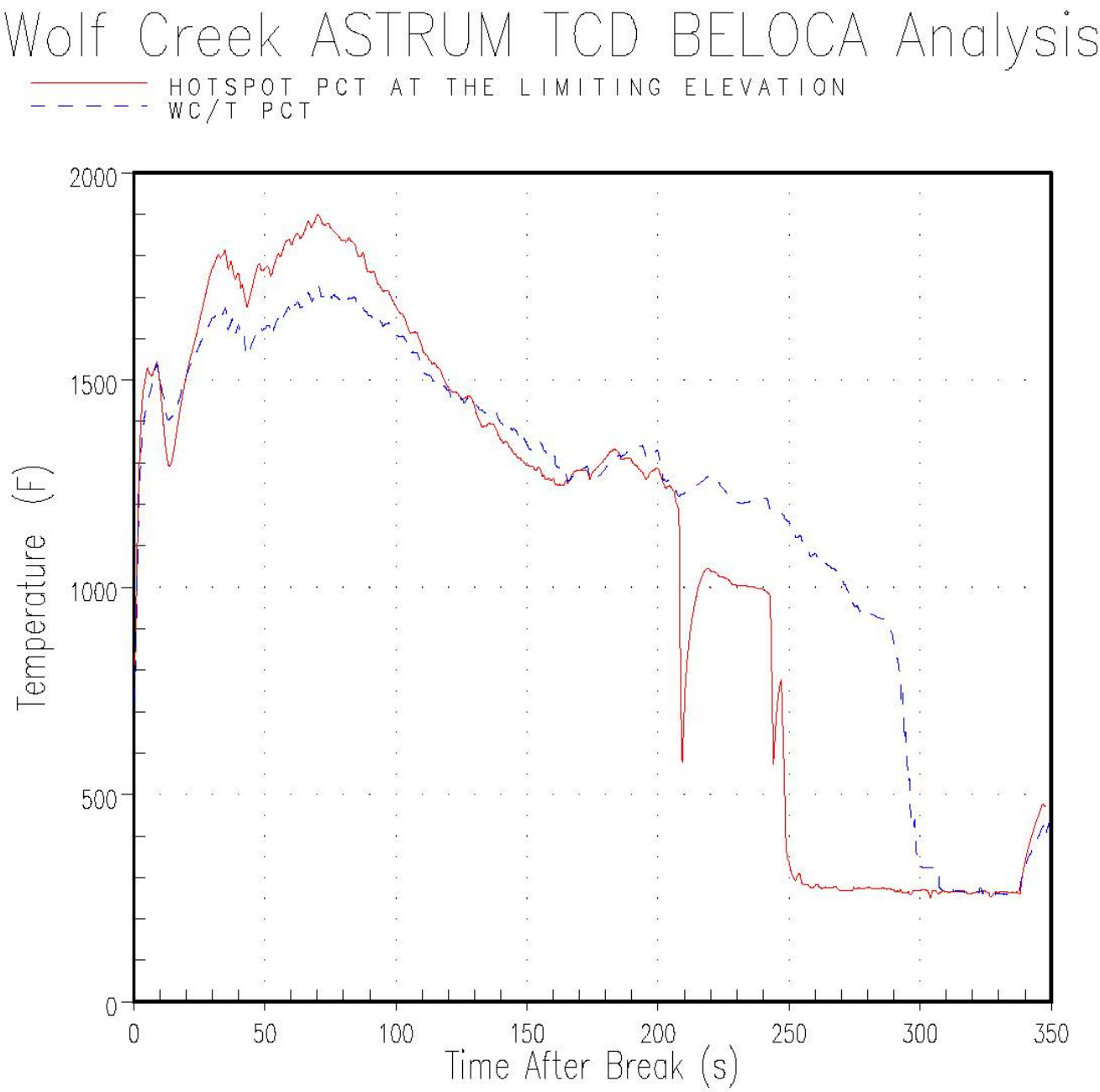

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FIGURE 15.6-7 HOTSPOT Cladding Temperature Transient and WC/T PCT for the Limiting PCT Case

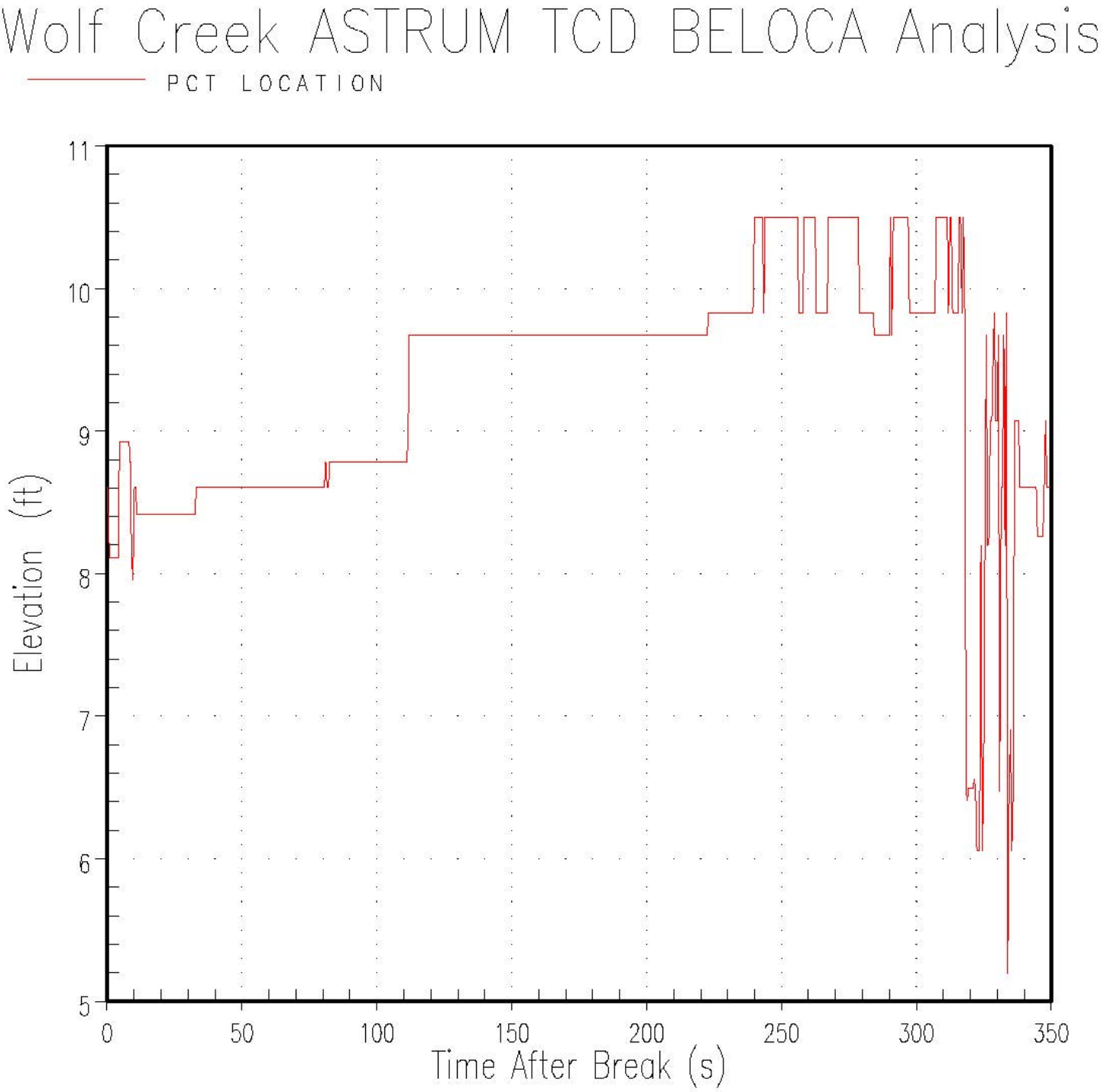

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FIGURE 15.6-8 PCT Elevation for the Hot Rod for the Limiting RCT Case

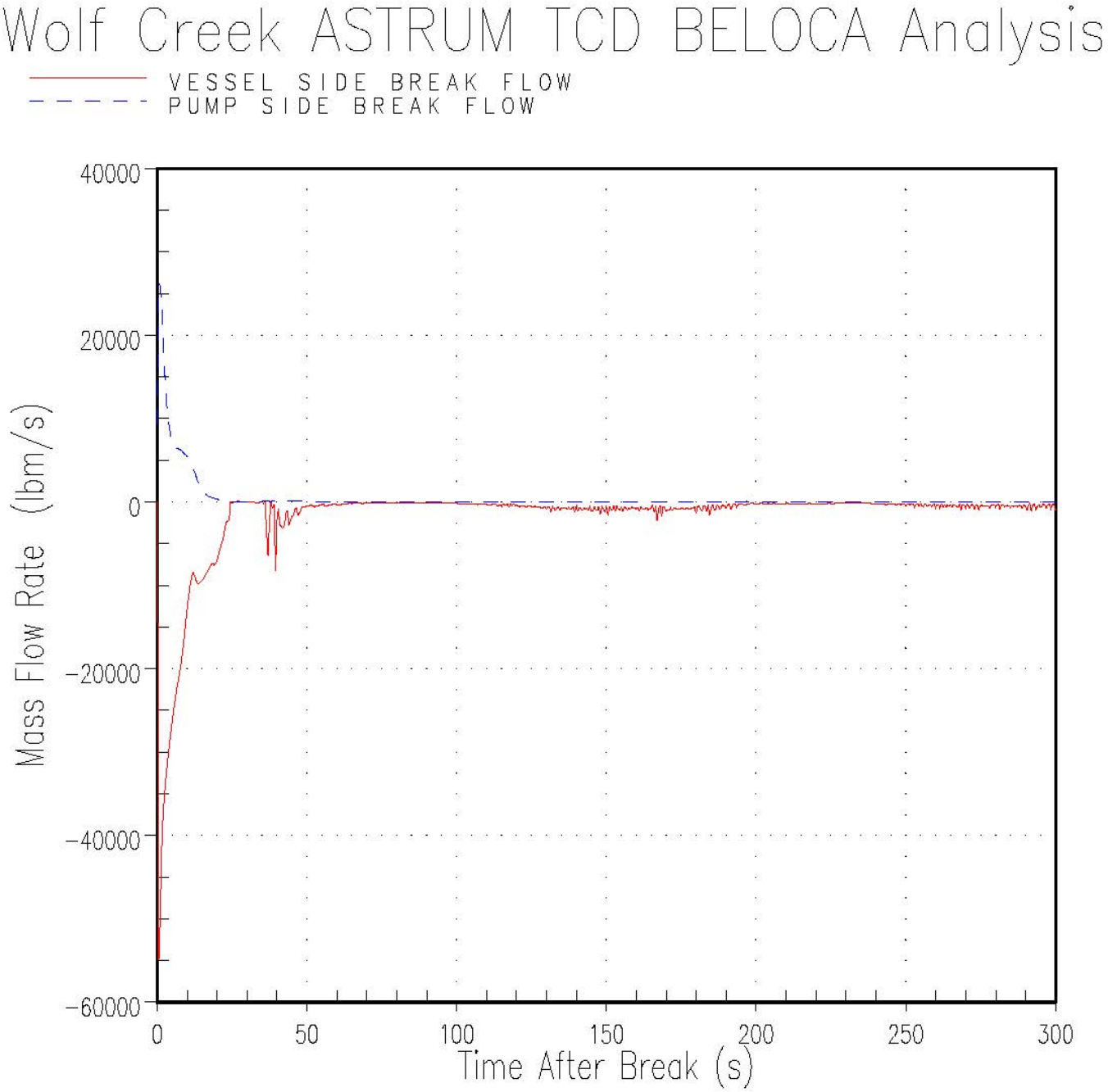

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FIGURE 15.6-9 Break Flow for the Limiting PCT Case

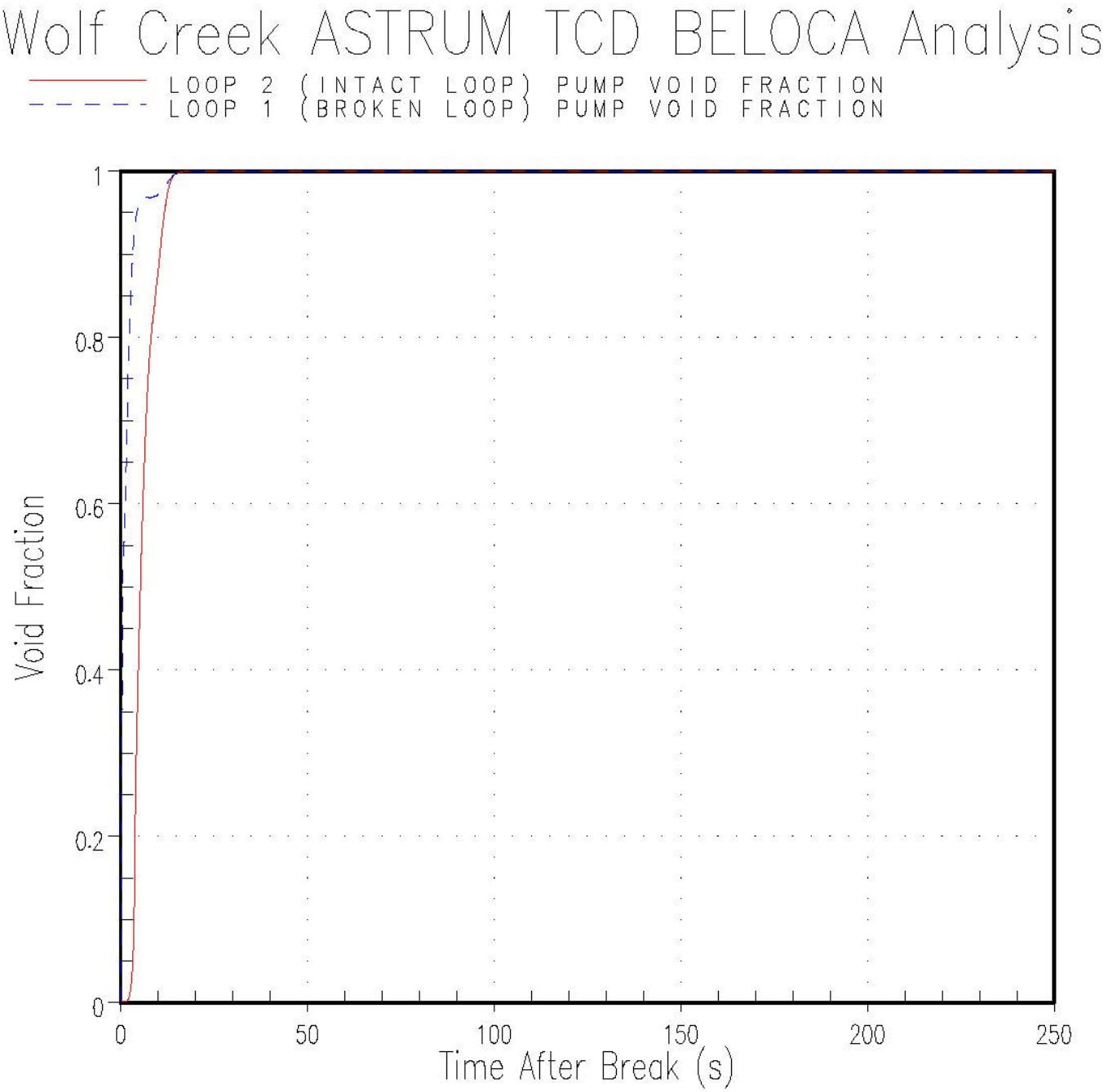

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FIGURE 15.6-10 Void Fraction in the Reactor Coolant Pumps for the Limiting PCT Case

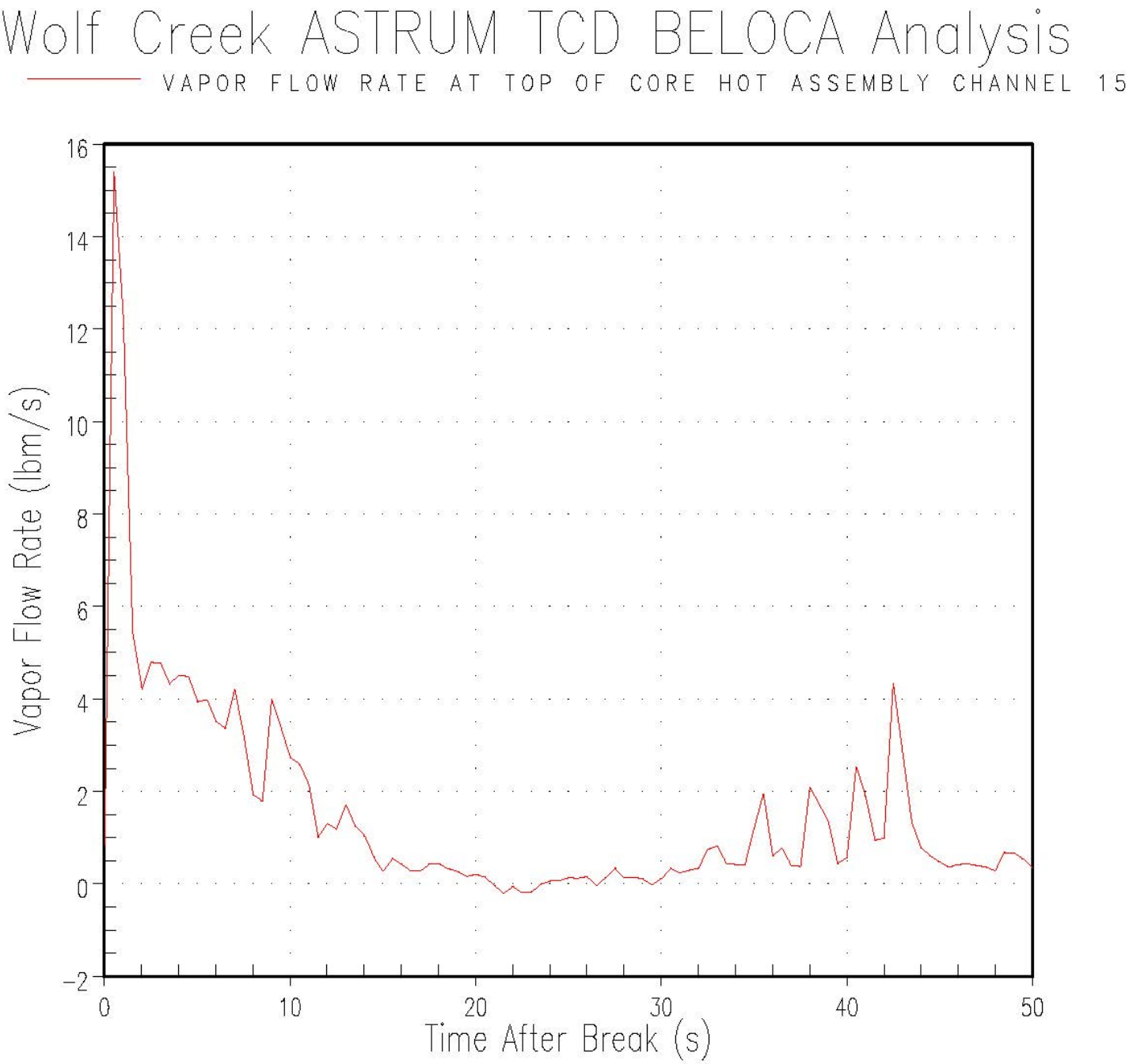

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FIGURE 15.6-11 Vapor Flow in the Hot Assembly Channel for the Limiting PCT Case

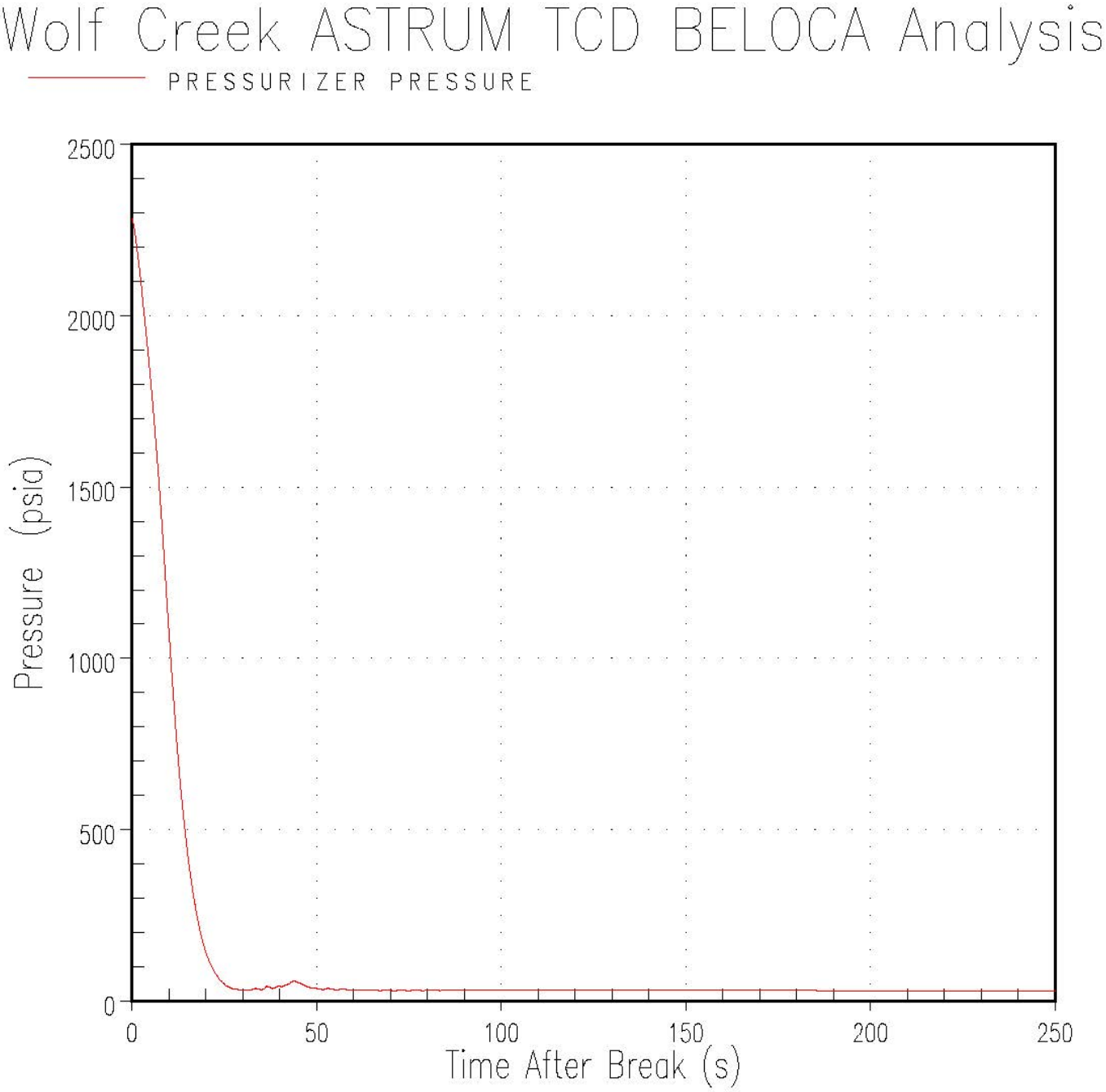

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FIGURE 15.6-12
Pressurizer Pressure for the Limiting PCT Case

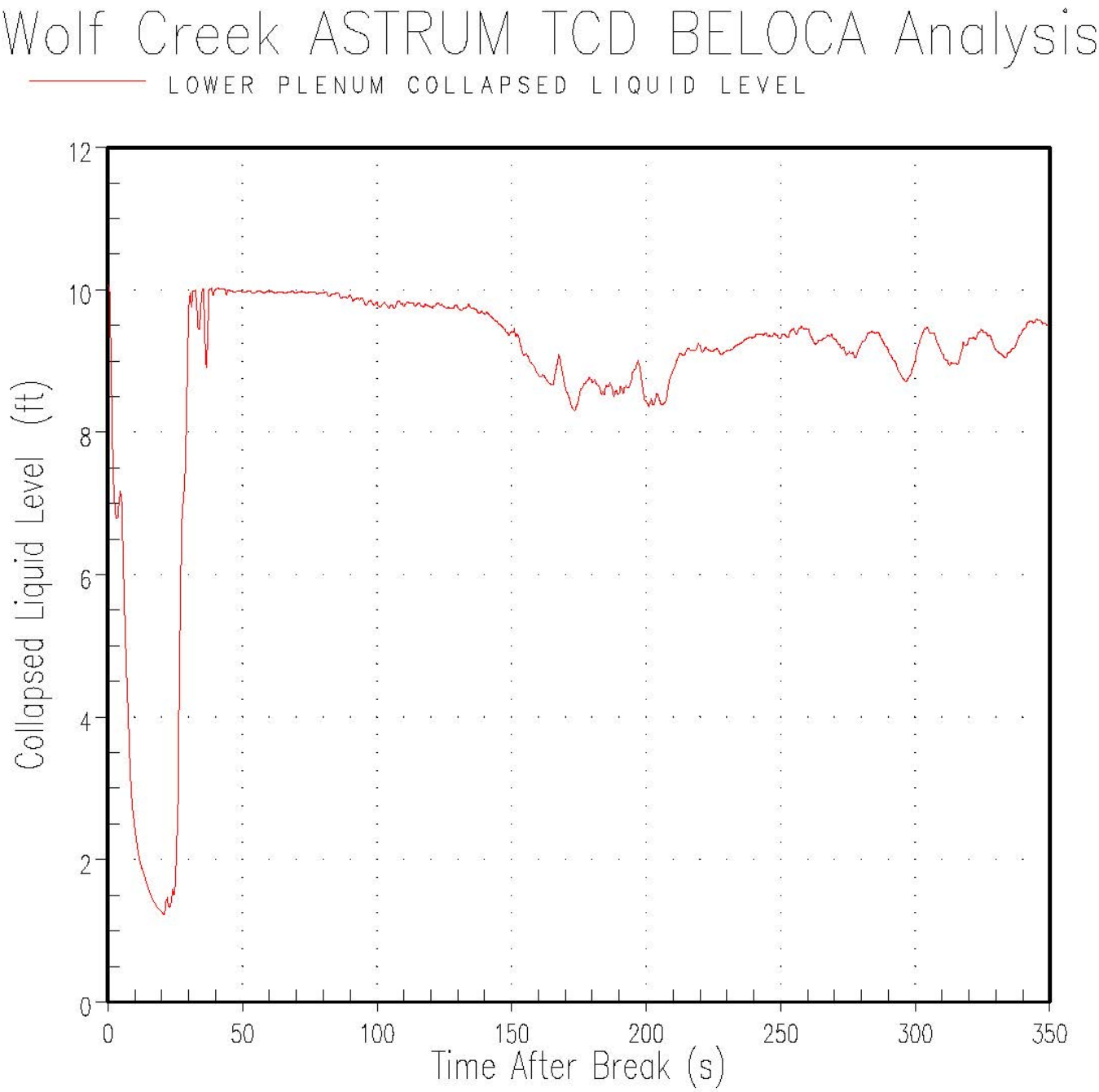

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FIGURE 15.6-13 Lower Plenum Collapsed Liquid Level for the Limiting PCT Case

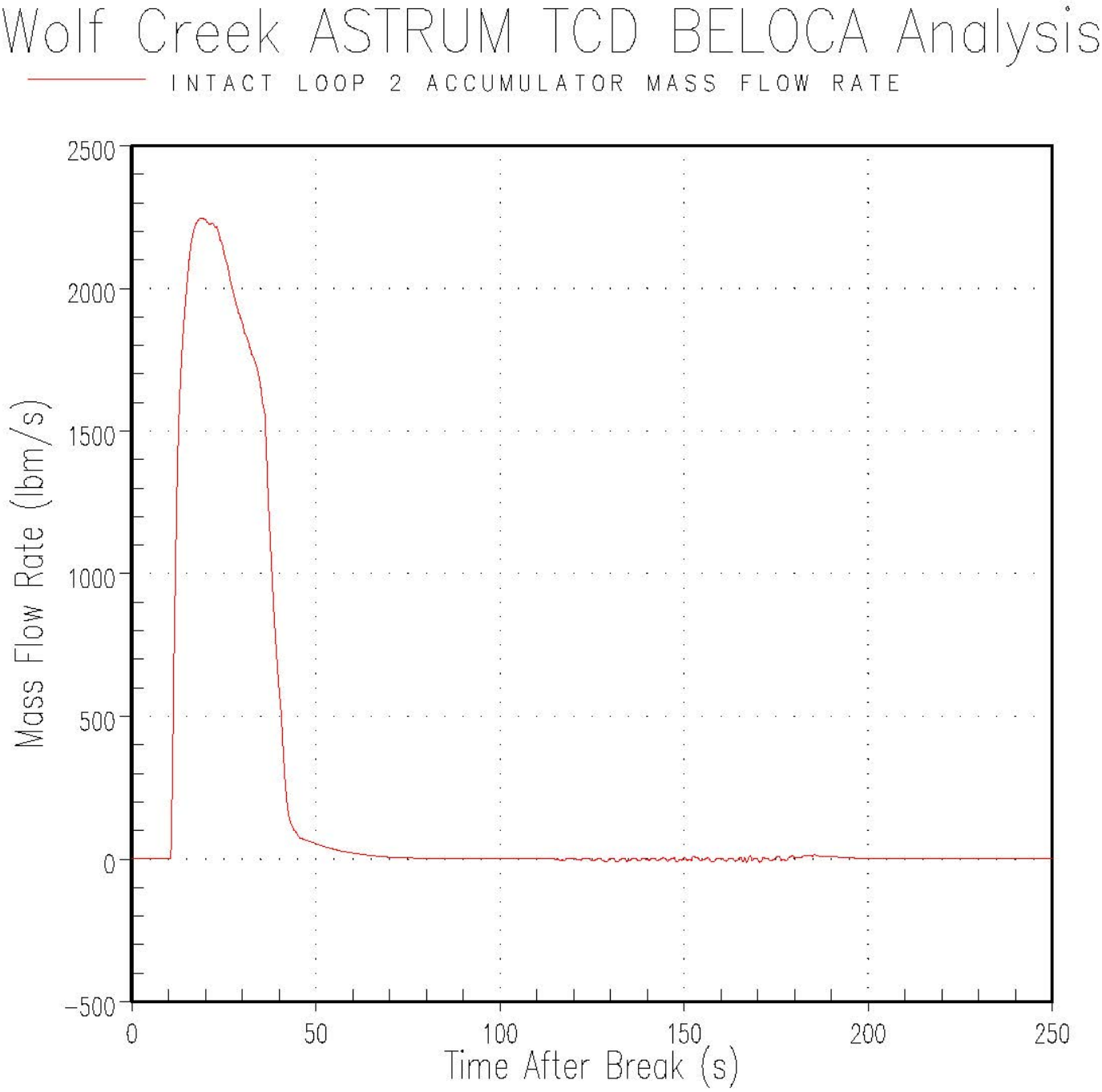

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FIGURE 15.6-14 Accumulator Injection Flow for the Limiting PCT Case

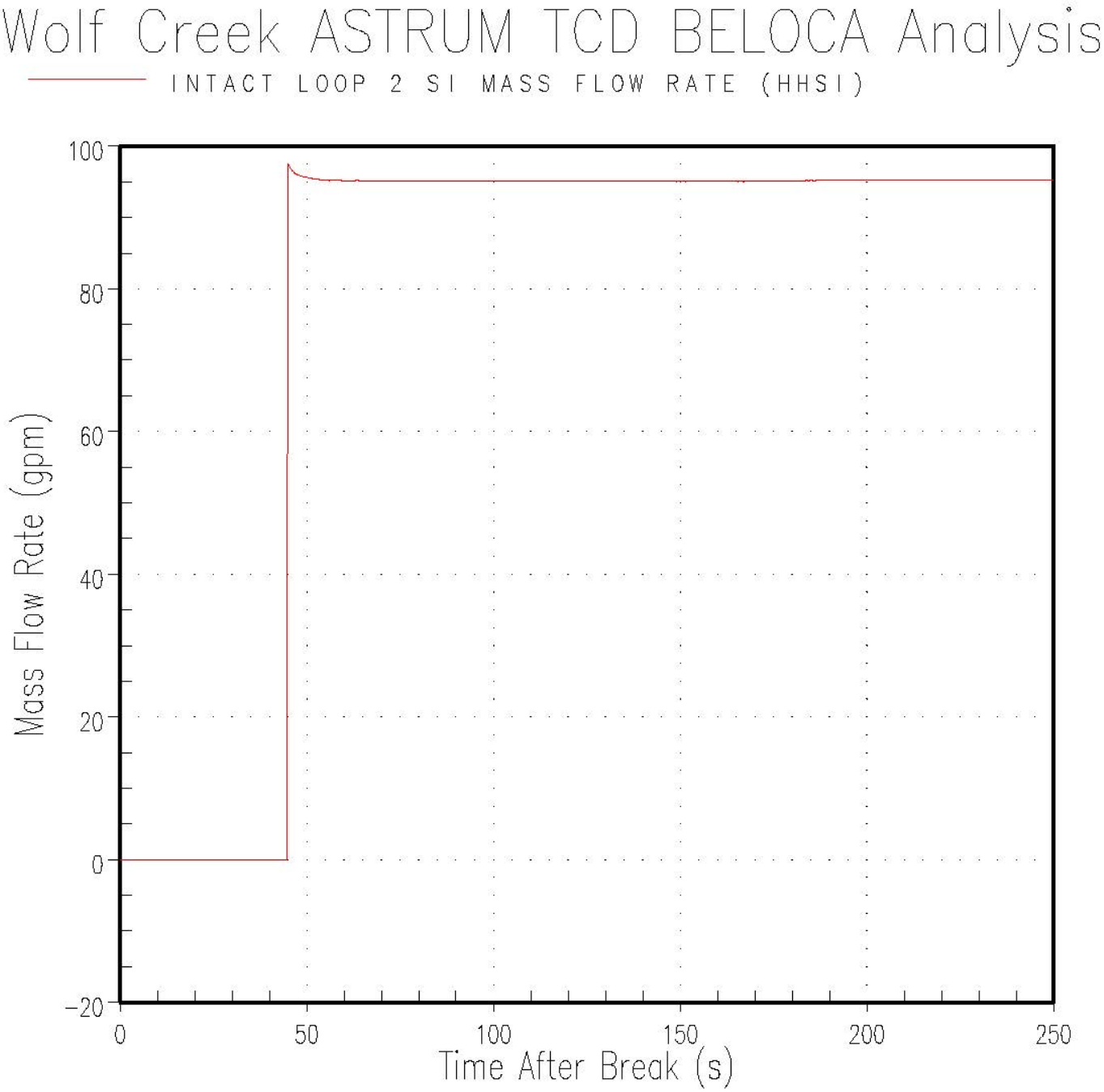

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FIGURE 15.6-15 Safety Injection Flow (HHSI) for the Limiting PCT Case

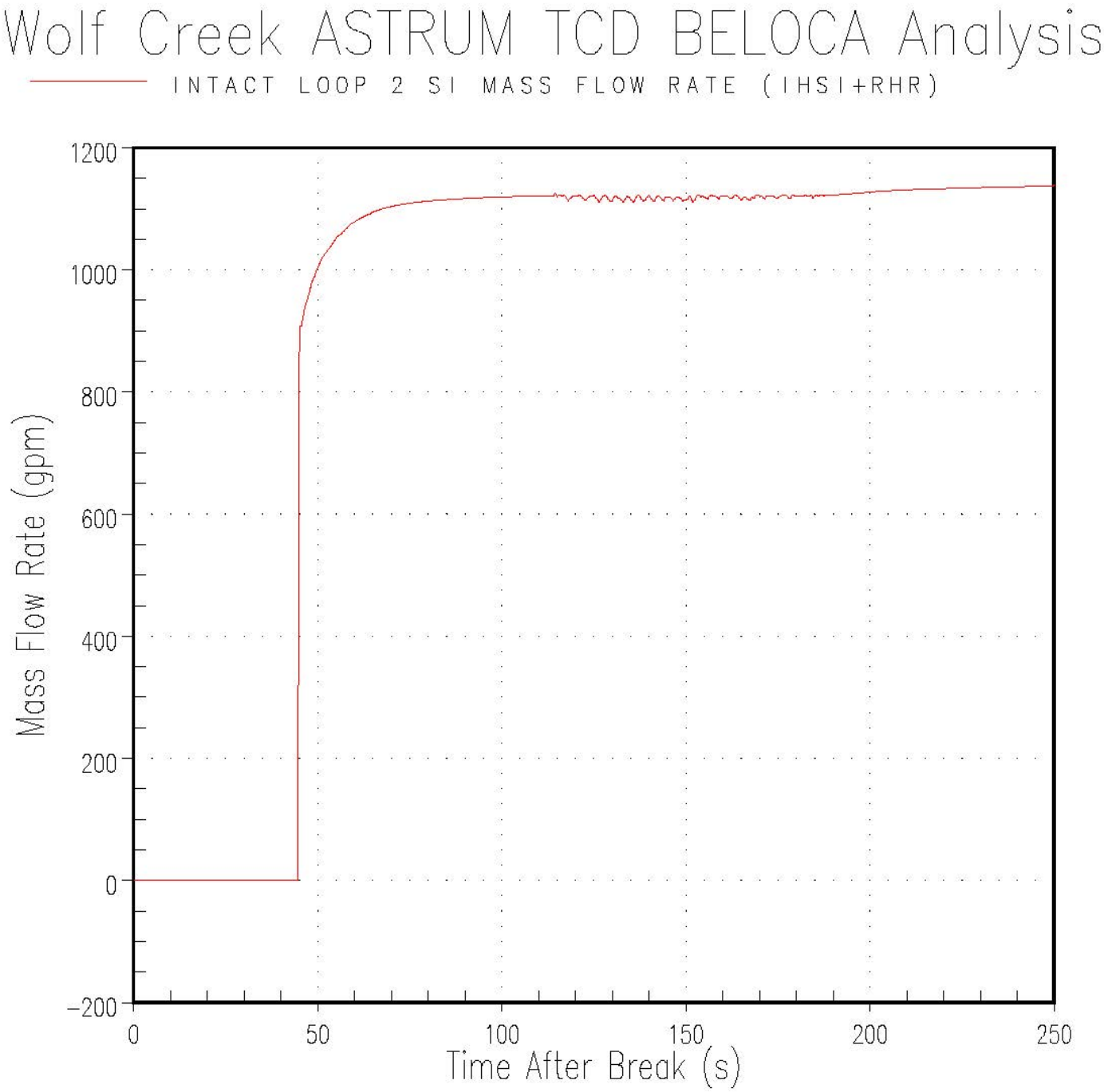

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FIGURE 15.6-16 Safety Injection Flow (IHSI+RHR) for the Limiting PCT Case

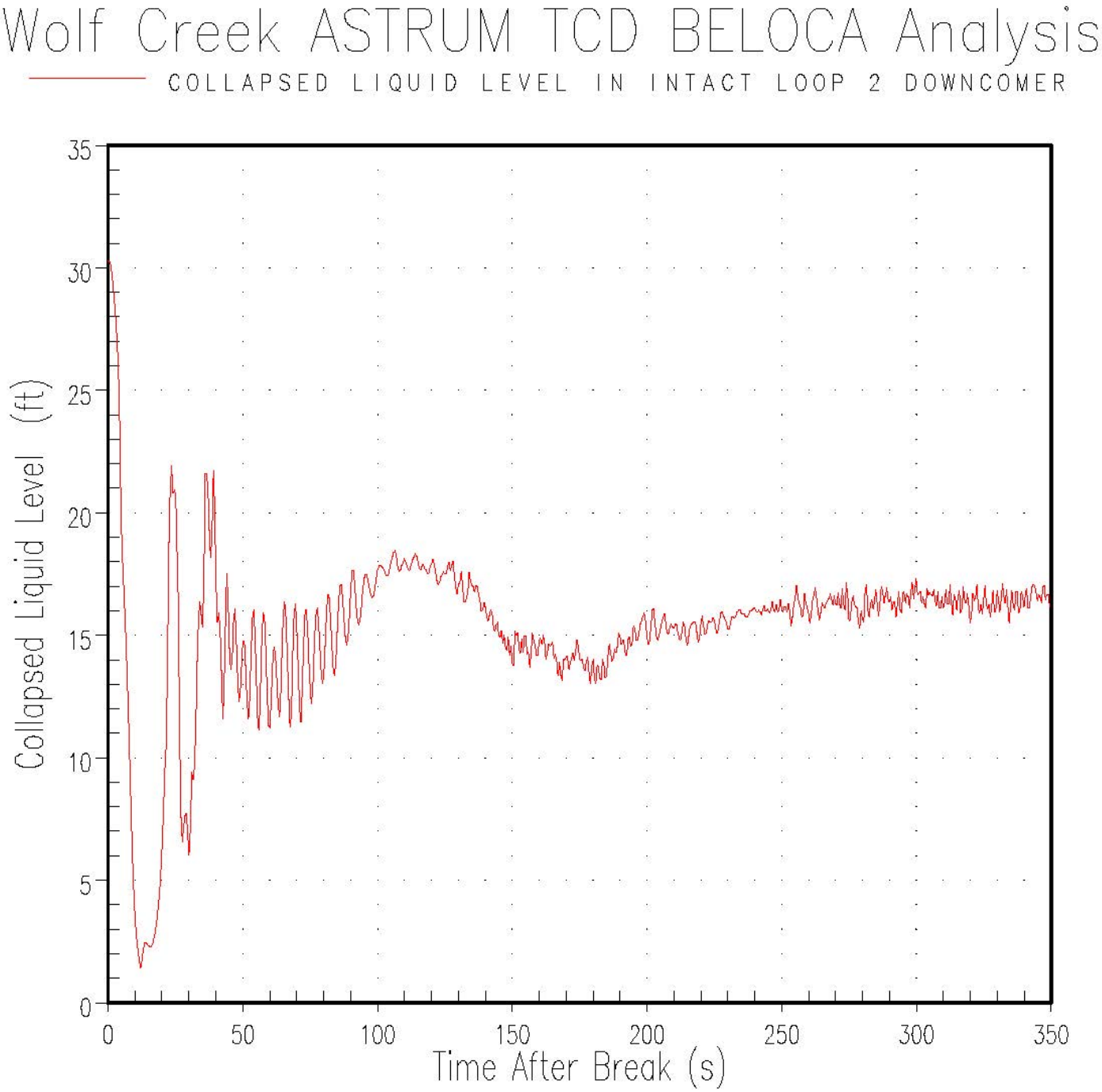

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FIGURE 15.6-17 Downcomer Collapsed Liquid Level for the Limiting PCT Case

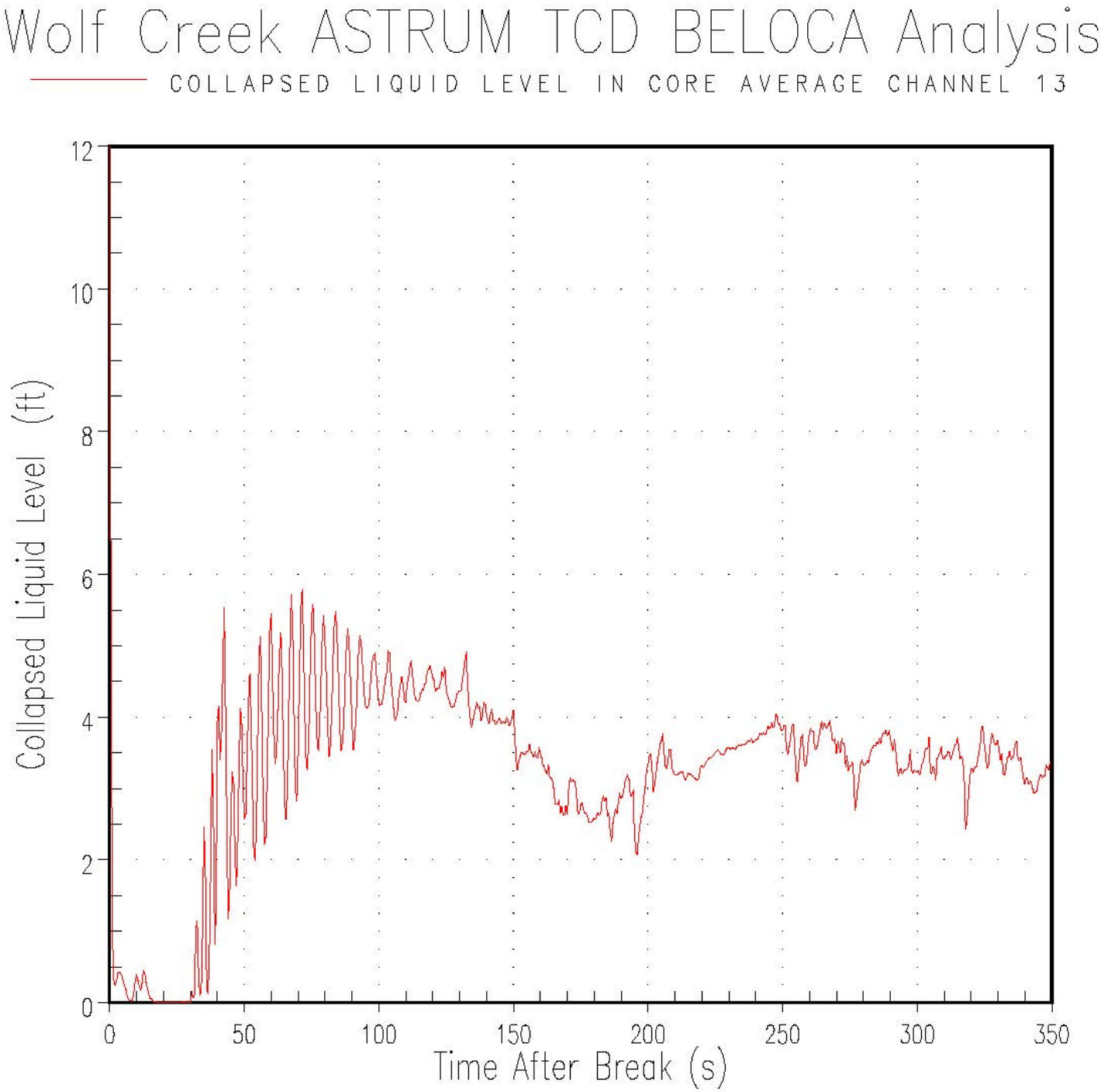

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FIGURE 15.6-18 Collapsed Liquid Level in Core Average Channel for the Limiting PCT Case

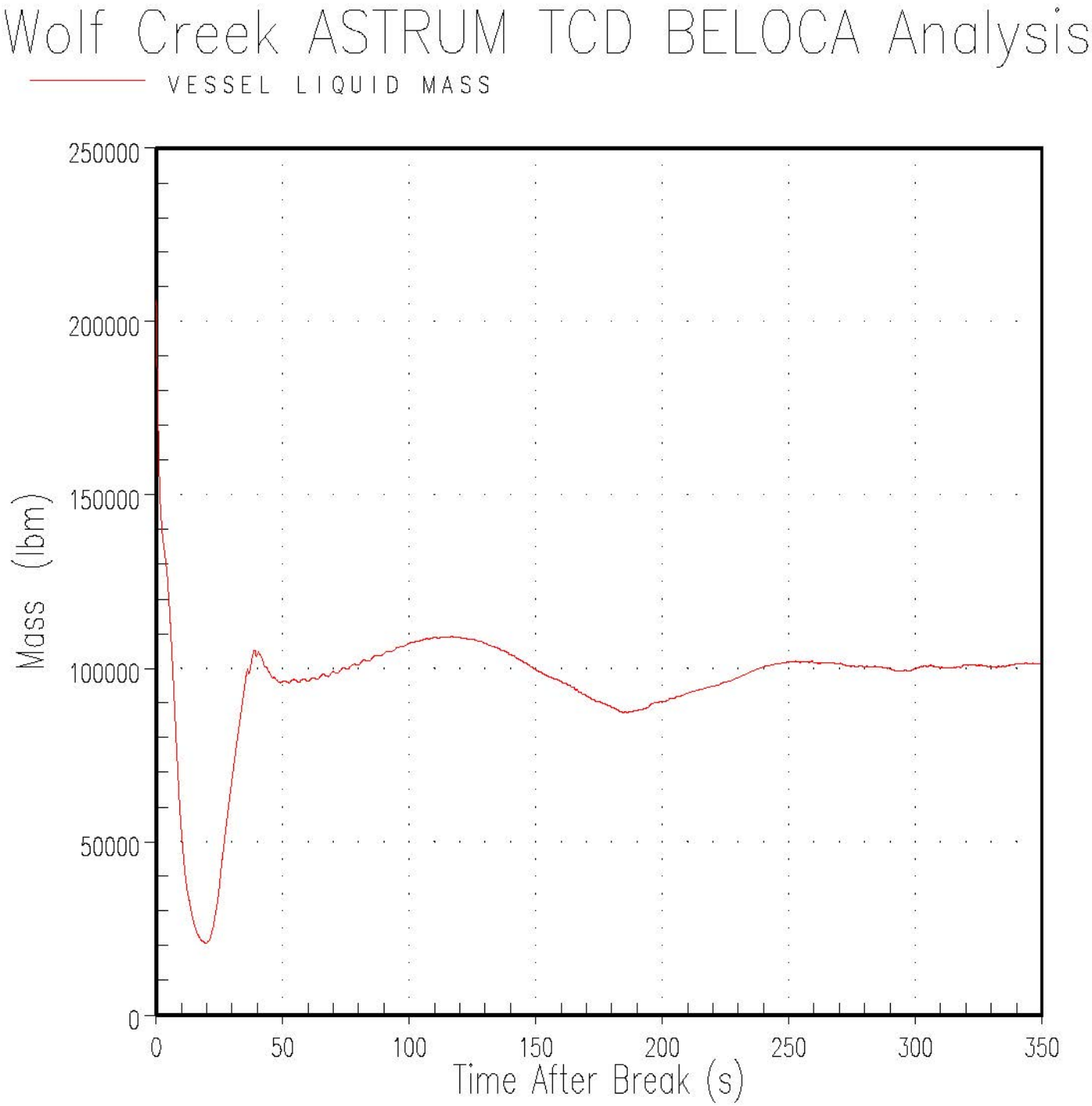

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FIGURE 15.6-19 Vessel Liquid Mass for the Limiting PCT Case

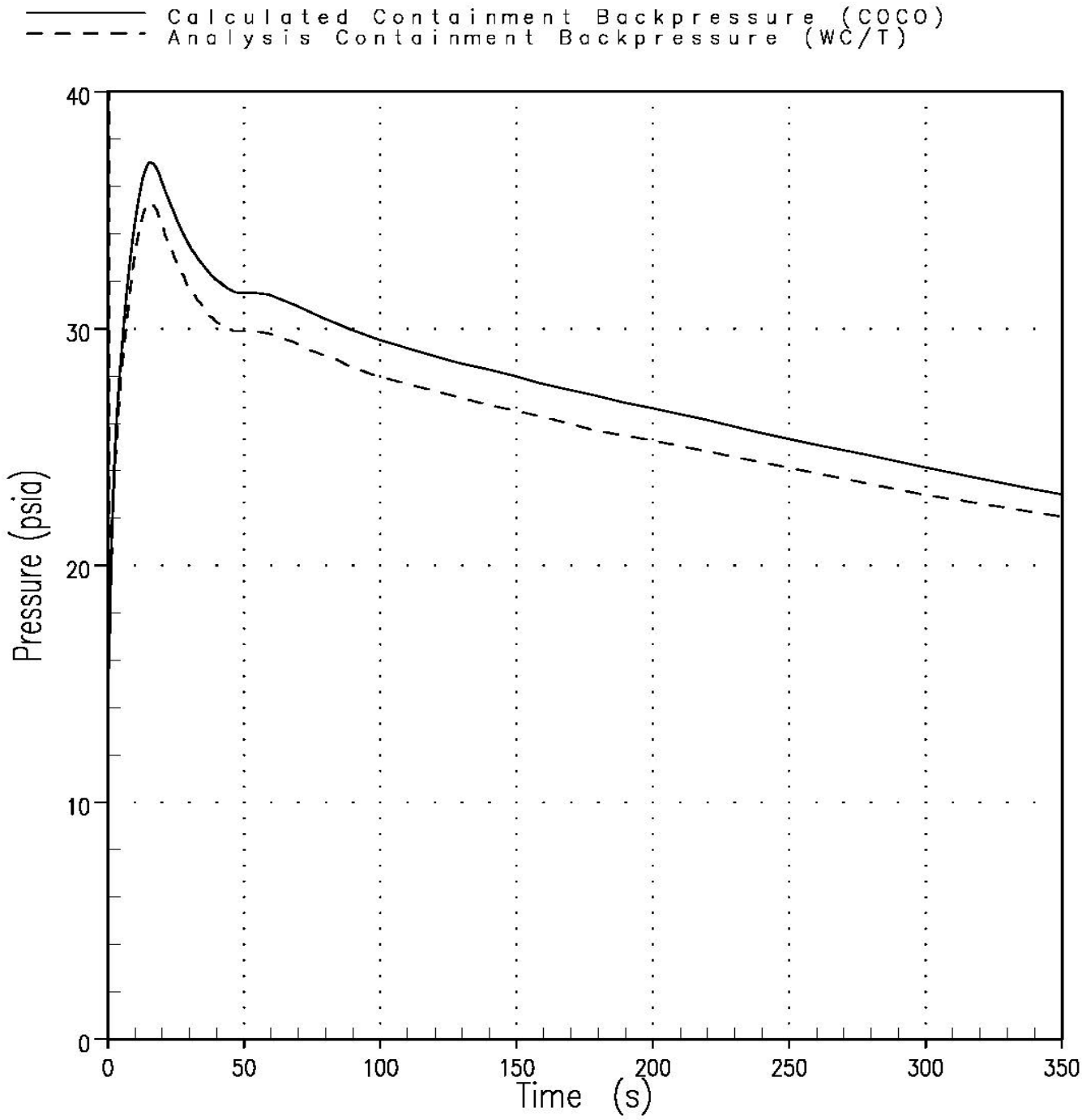

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FIGURE 15.6-21
Analysis vs. Calculated
Containment Backpressure

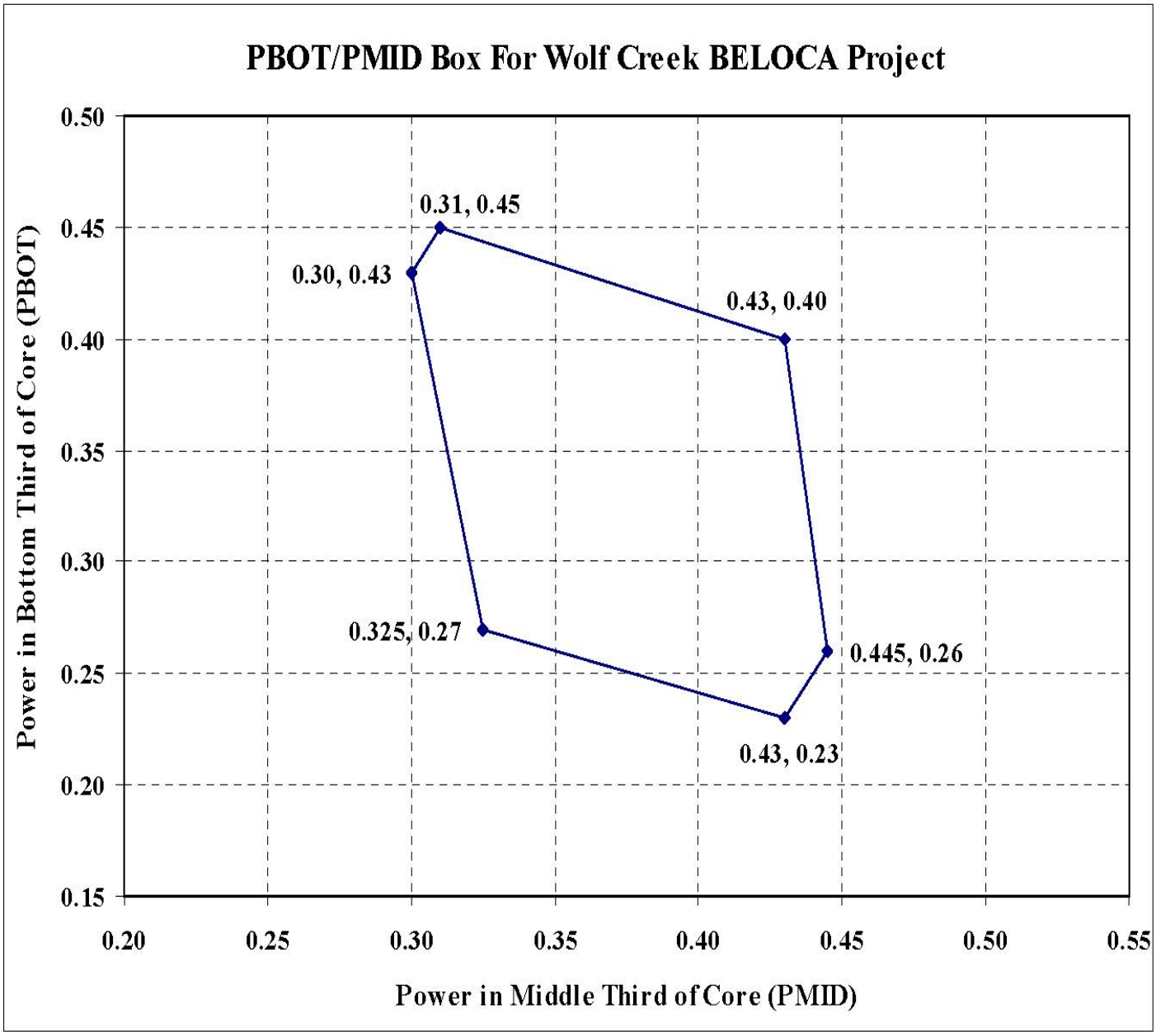

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FIGURE 15.6-22
Wolf Creek BELOCA Analysis Axial Power Shape Operating Space Envelope

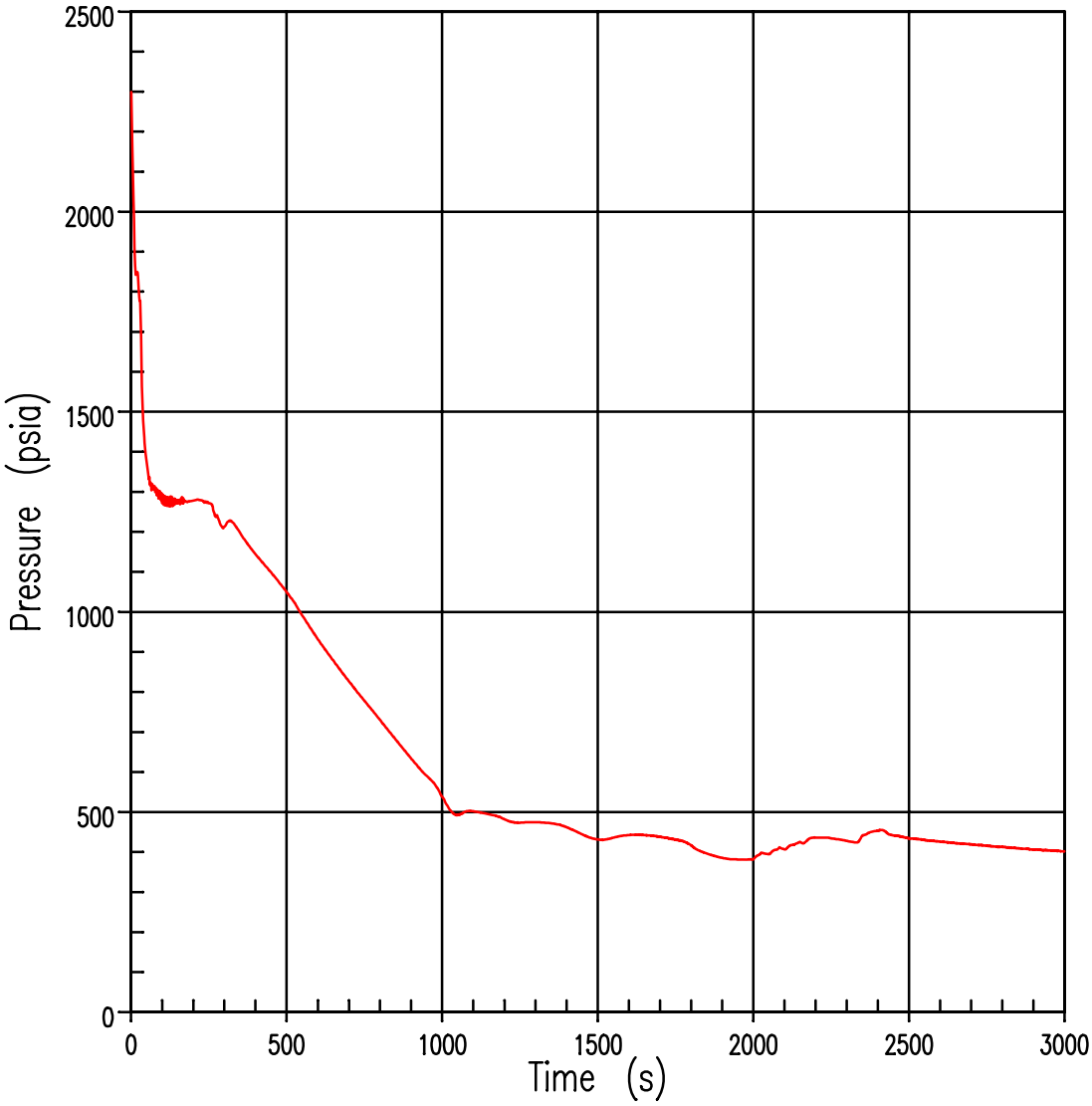

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FIGURE 15.6-31

REACTOR COOLANT SYSTEM PRESSURE
(4-INCH BREAK)

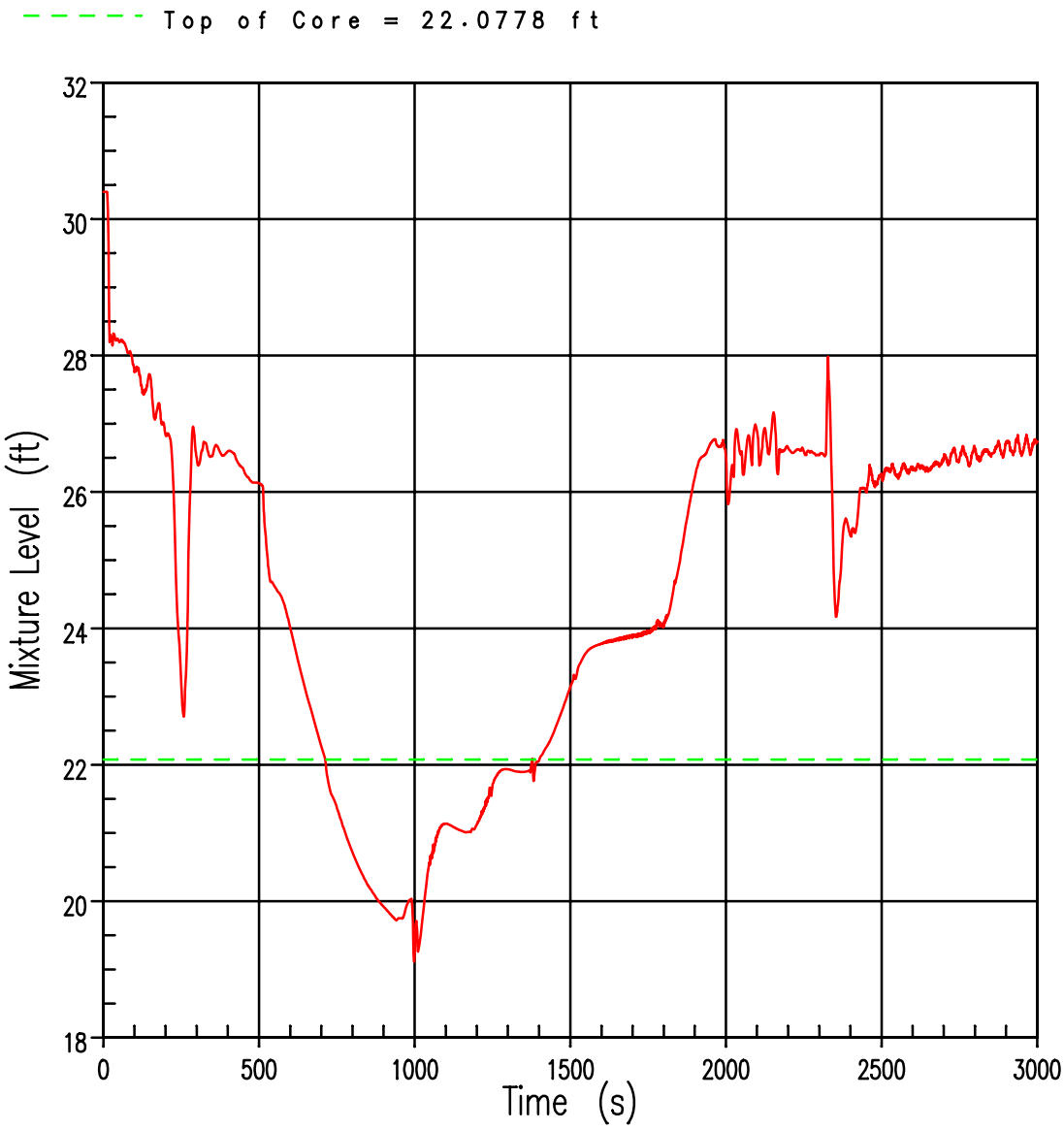

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FIGURE 15.6-32

CORE MIXTURE HEIGHT
(4-INCH BREAK)

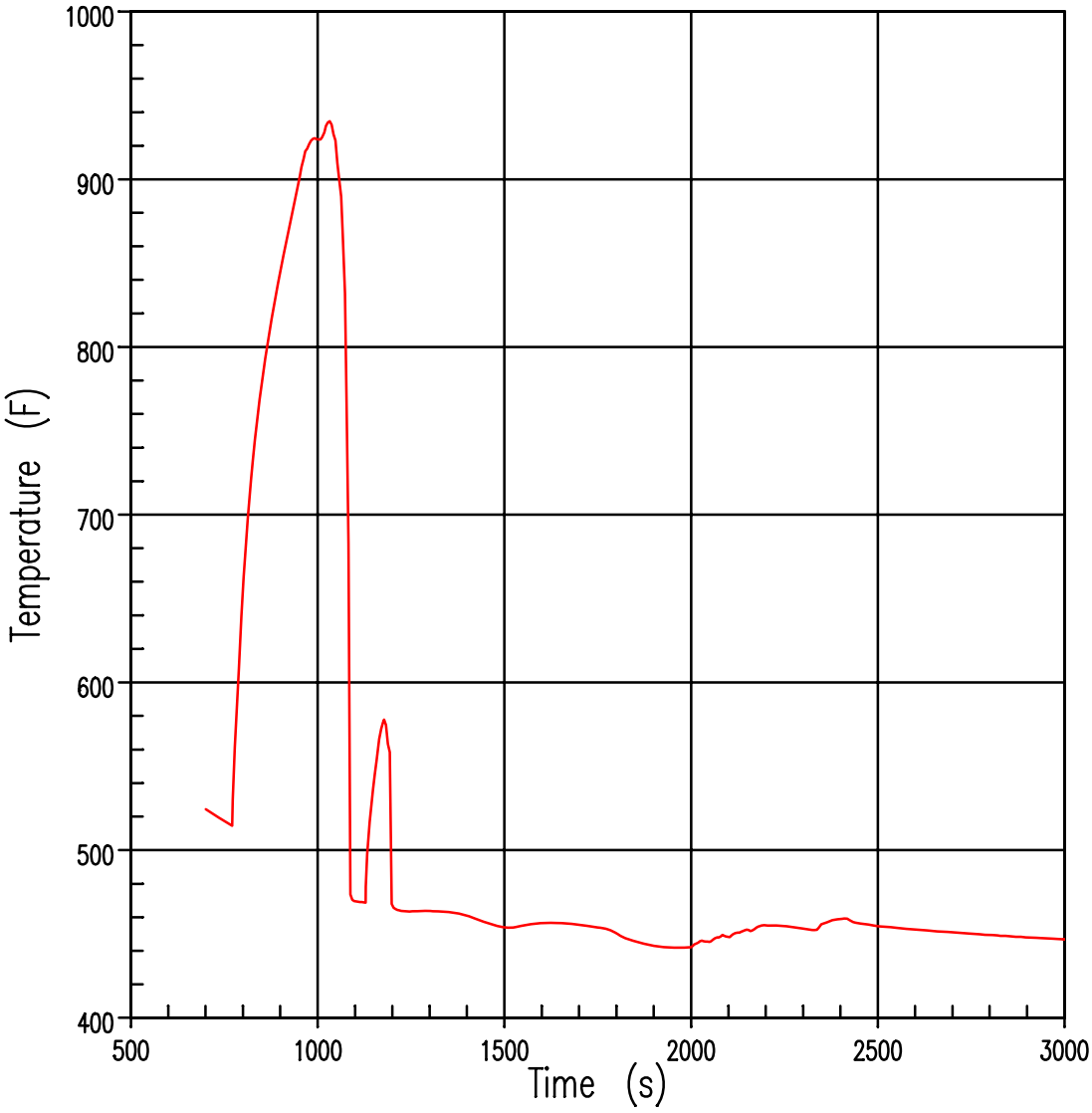

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FIGURE 15.6-33

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (4-INCH BREAK)

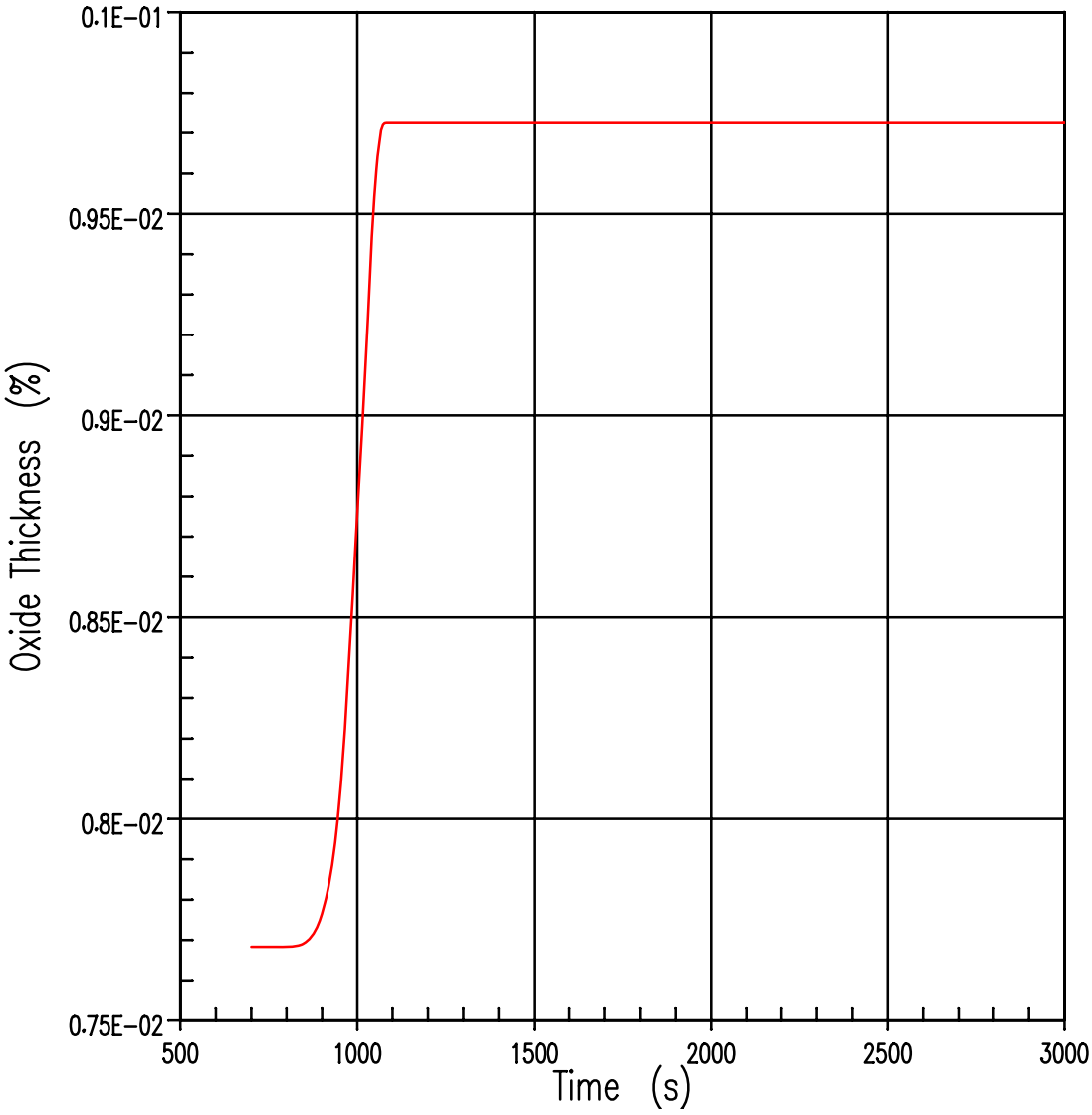

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FIGURE 15.6-33A

MAXIMUM LOCAL ZRO_2 THICKNESS (%)
(4-INCH BREAK)

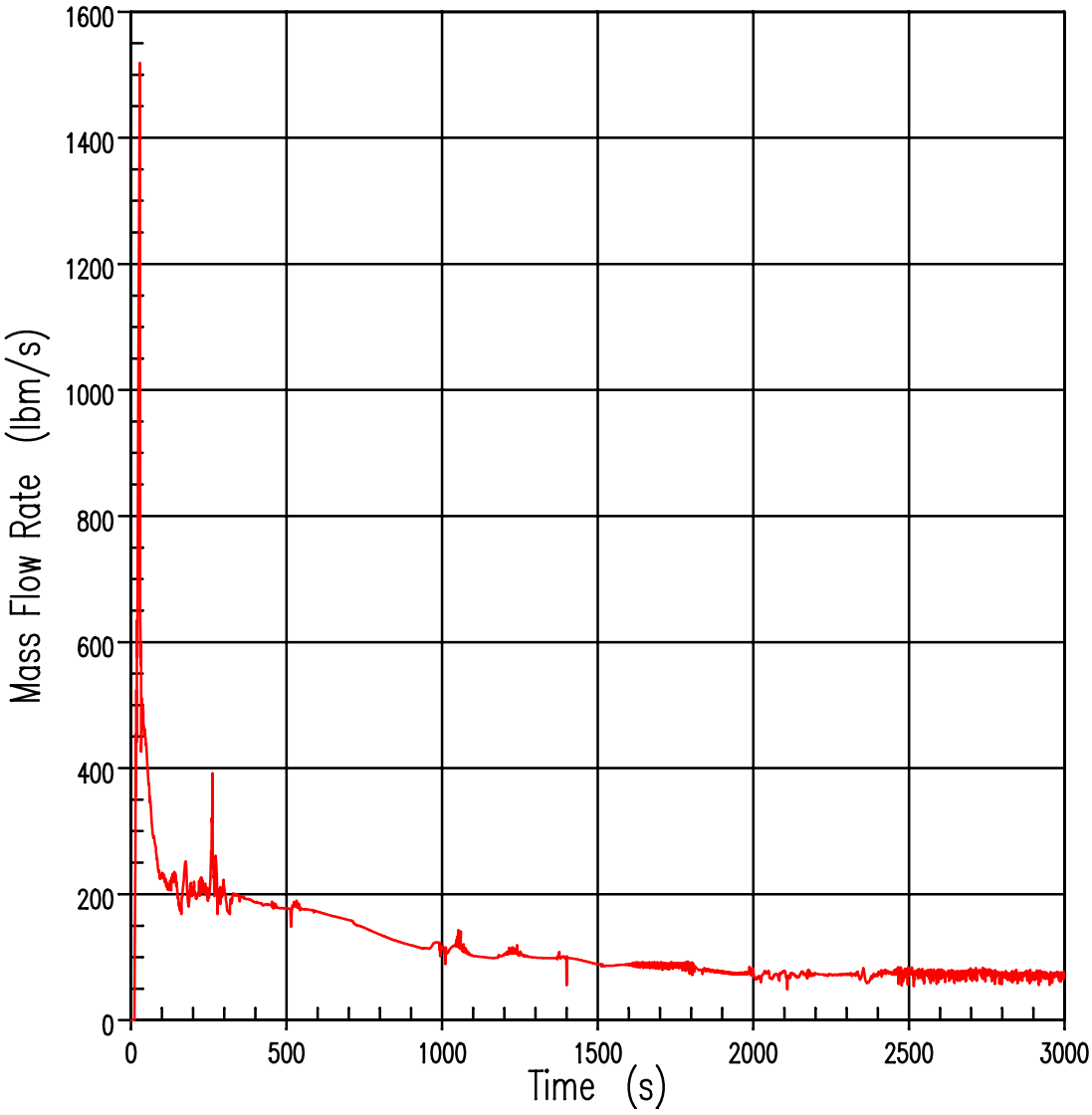

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FIGURE 15.6-34

STEAM MASS FLOW RATE OUT TOP OF CORE
(4-INCH BREAK)

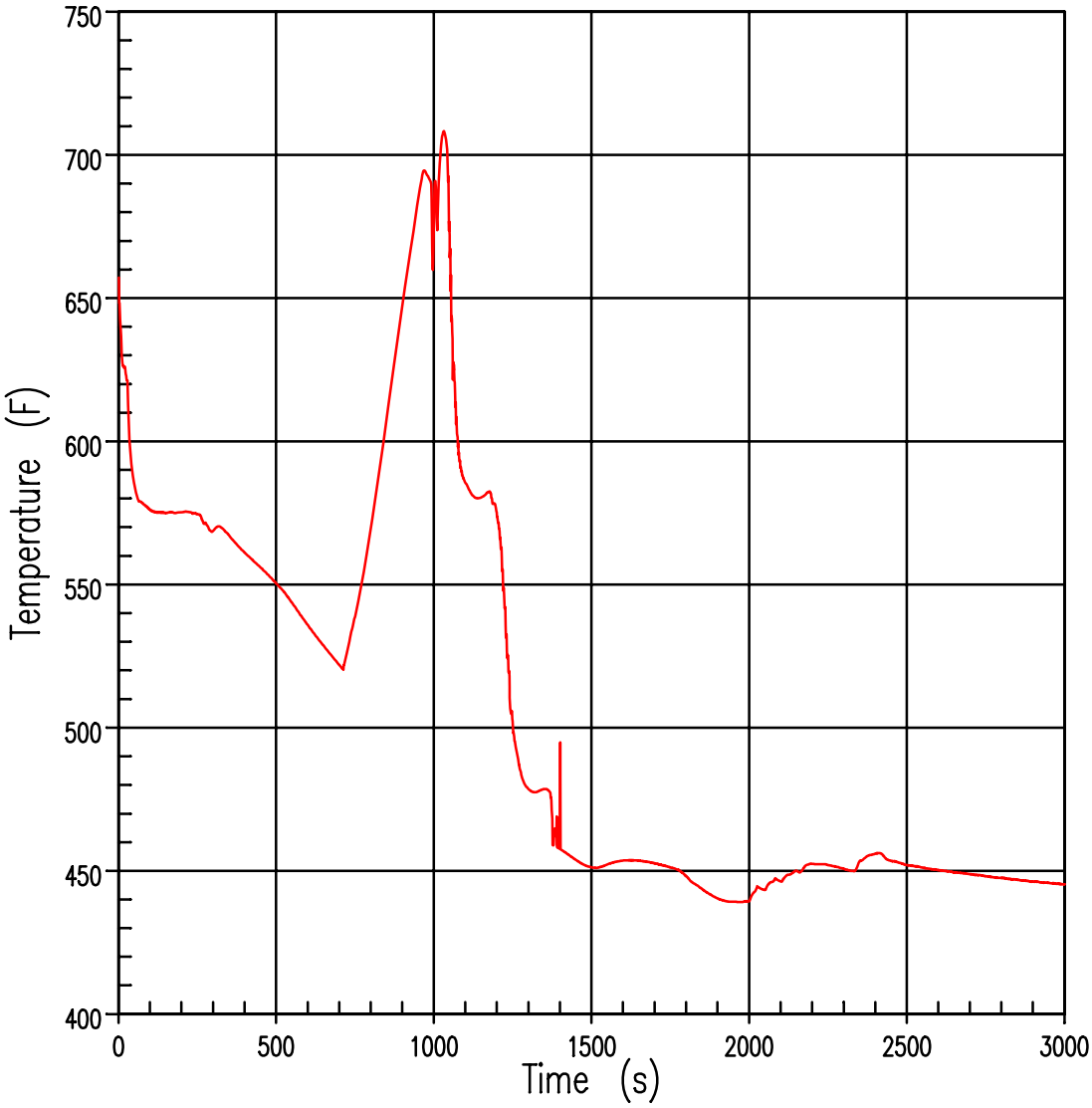

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FIGURE 15.6-34A

TOP CORE EXIT VAPOR TEMPERATURE
(4-INCH BREAK)

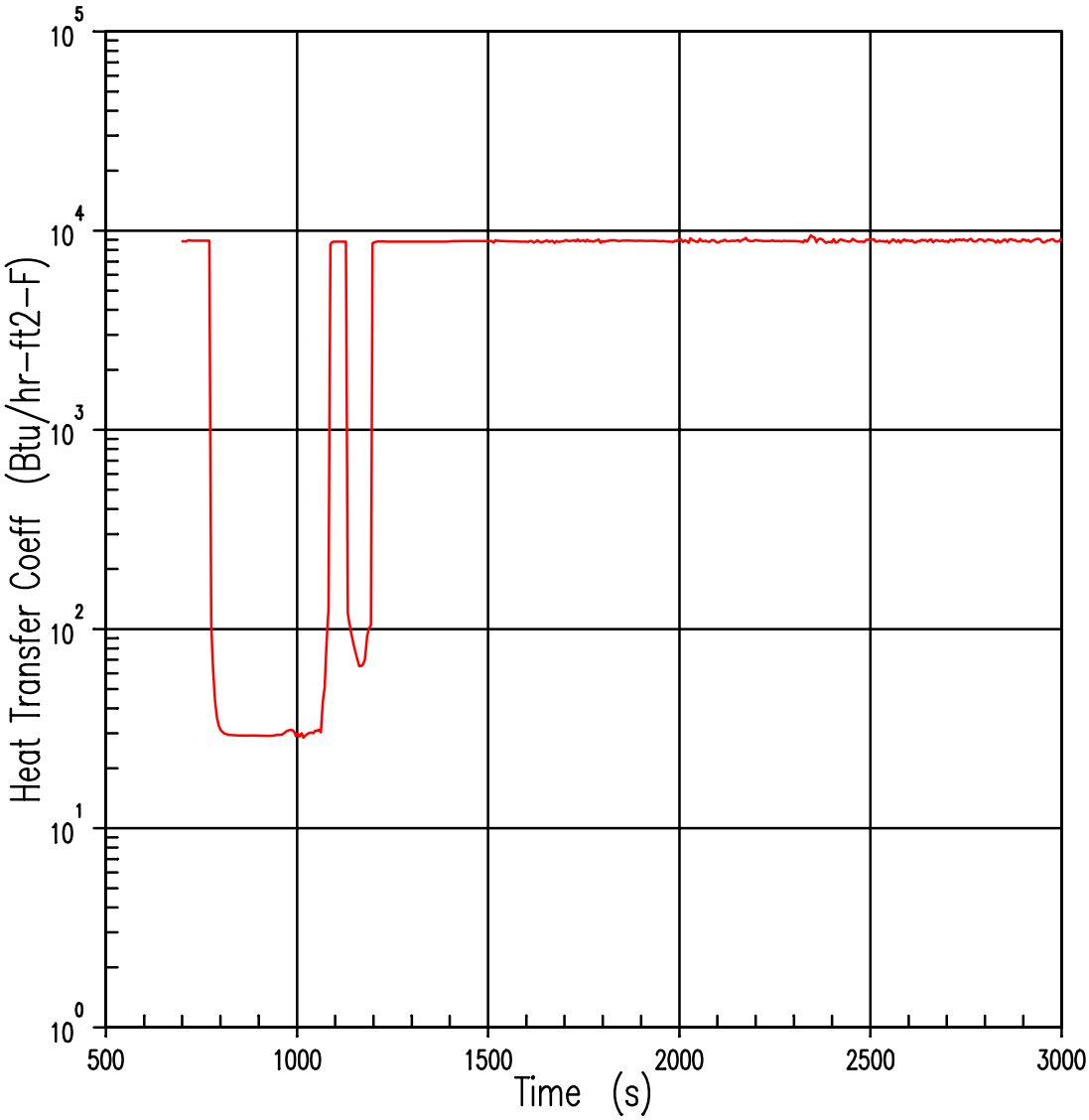

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FIGURE 15.6-35

CLAD SURFACE HEAT TRANSFER COEFFICIENT AT PEAK TEMPERATURE ELEVATION
(4-INCH BREAK)

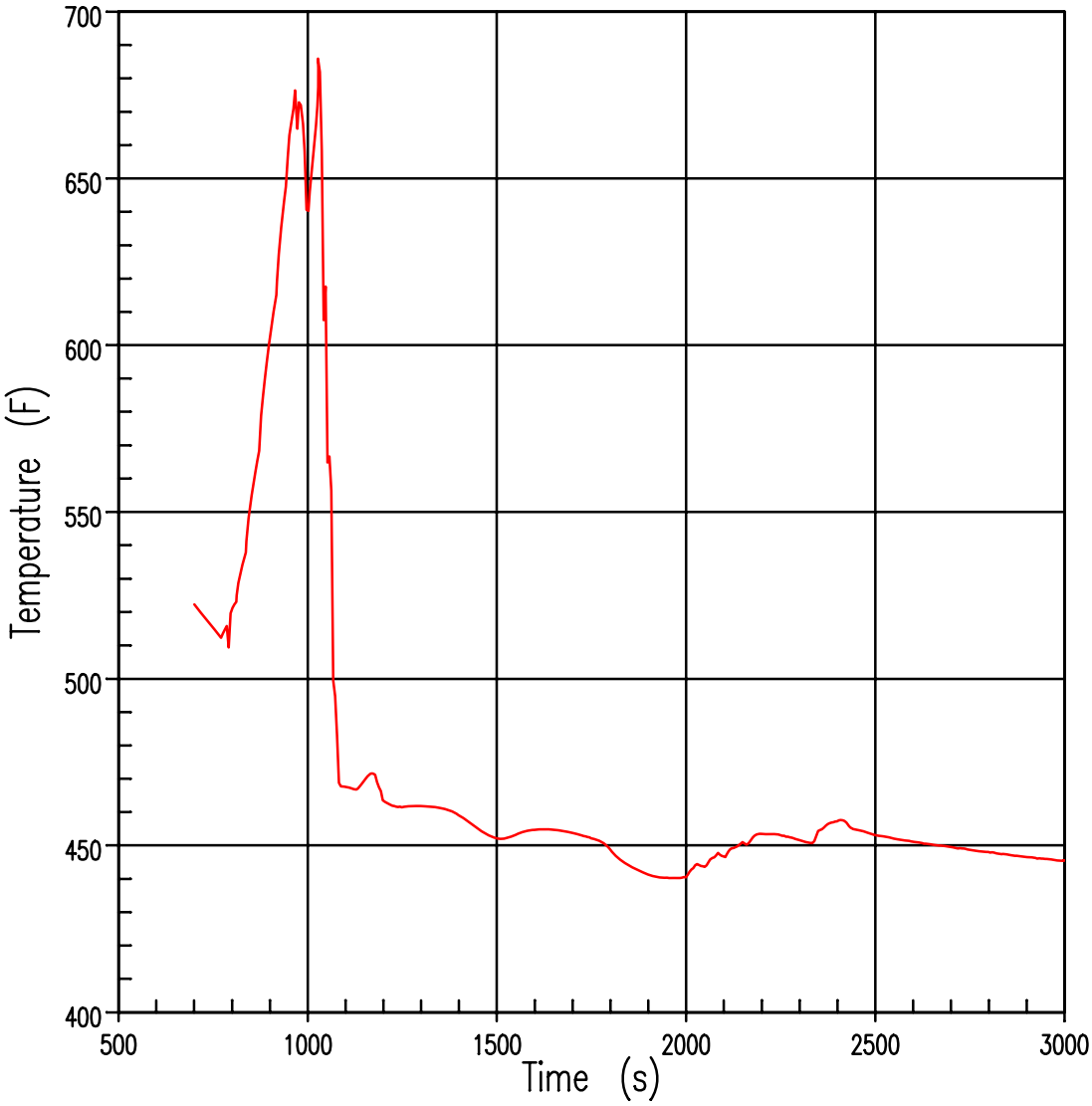

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FIGURE 15.6-36

FLUID TEMPERATURE AT PEAK CLAD TEMPERATURE ELEVATION (4-INCH BREAK)

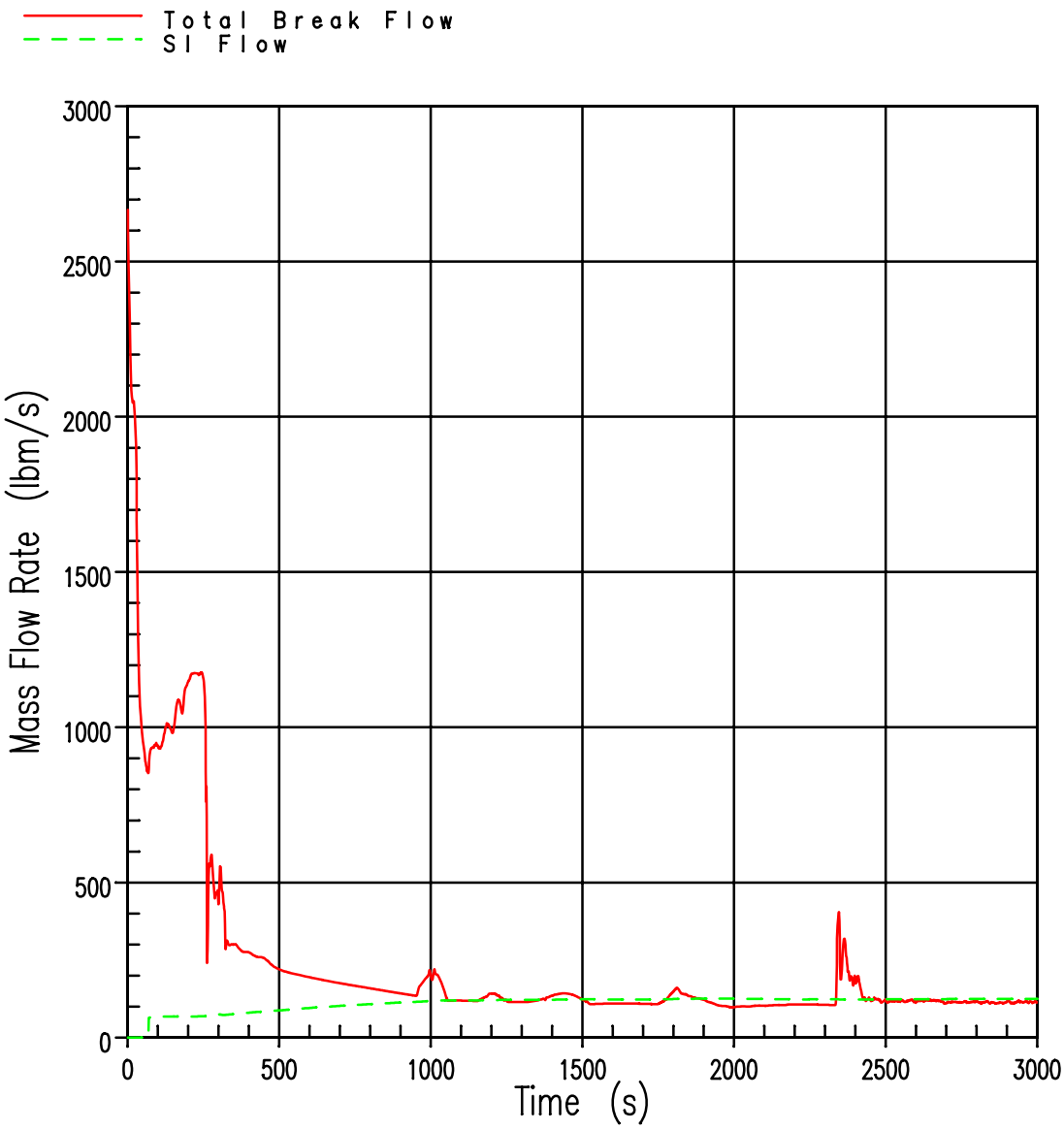

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FIGURE 15.6-36A

TOTAL BREAK FLOW & SAFETY INJECTION FLOW (4-INCH BREAK)

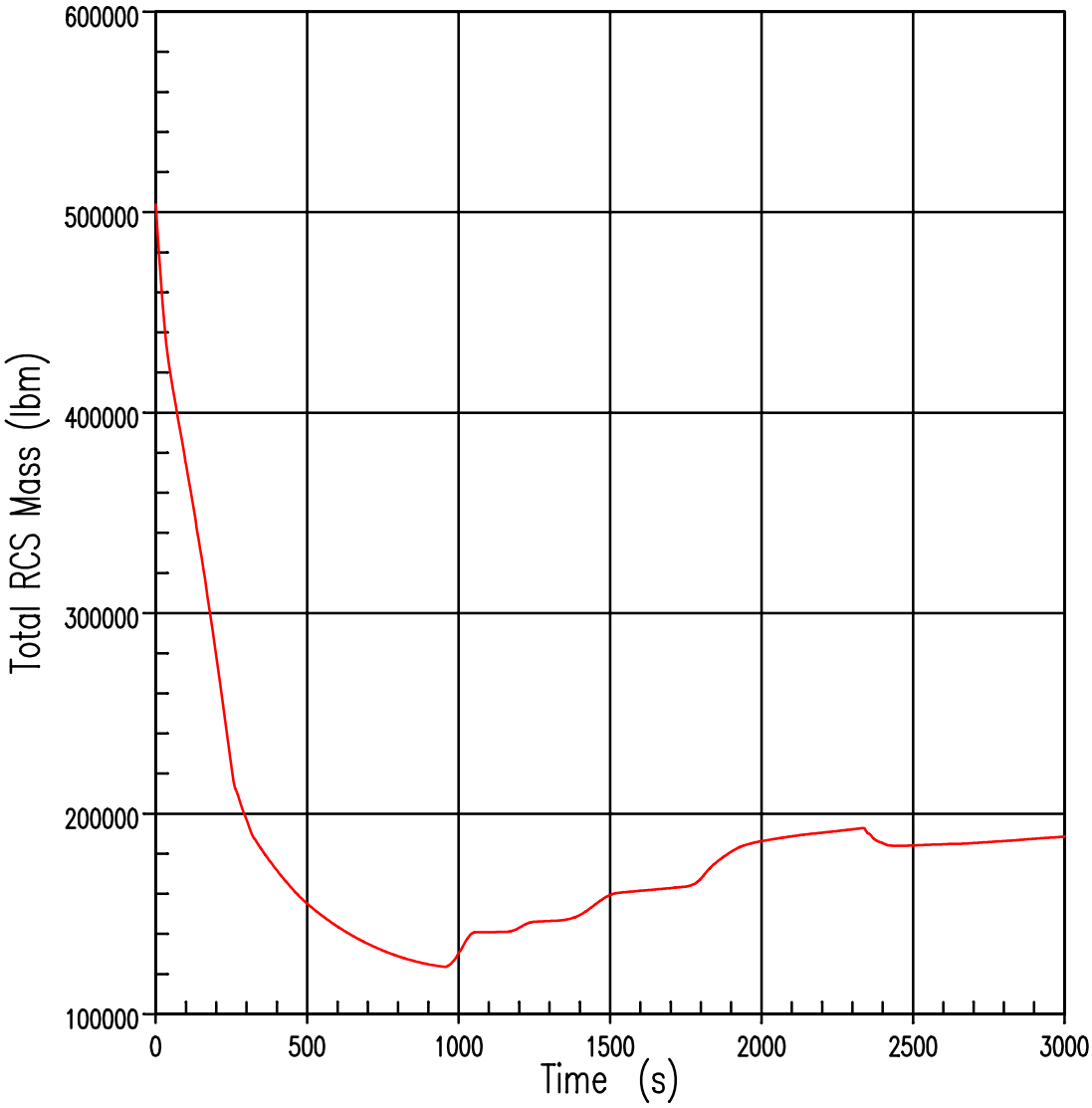

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FIGURE 15.6-36B

TOTAL REACTOR COOLANT SYSTEM MASS
(4-INCH BREAK)

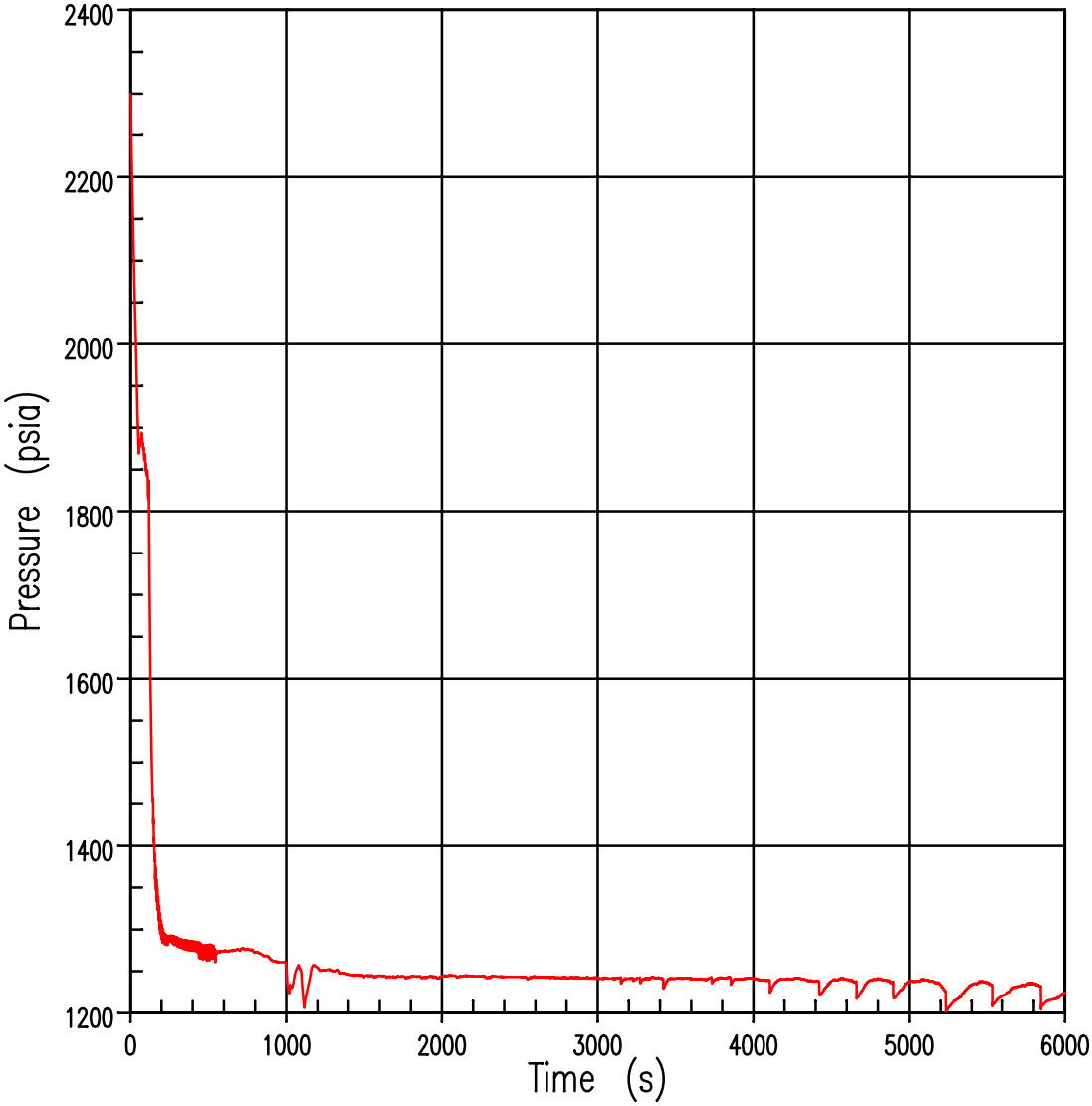

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FIGURE 15.6-37

REACTOR COOLANT SYSTEM PRESSURE
(2-INCH BREAK)

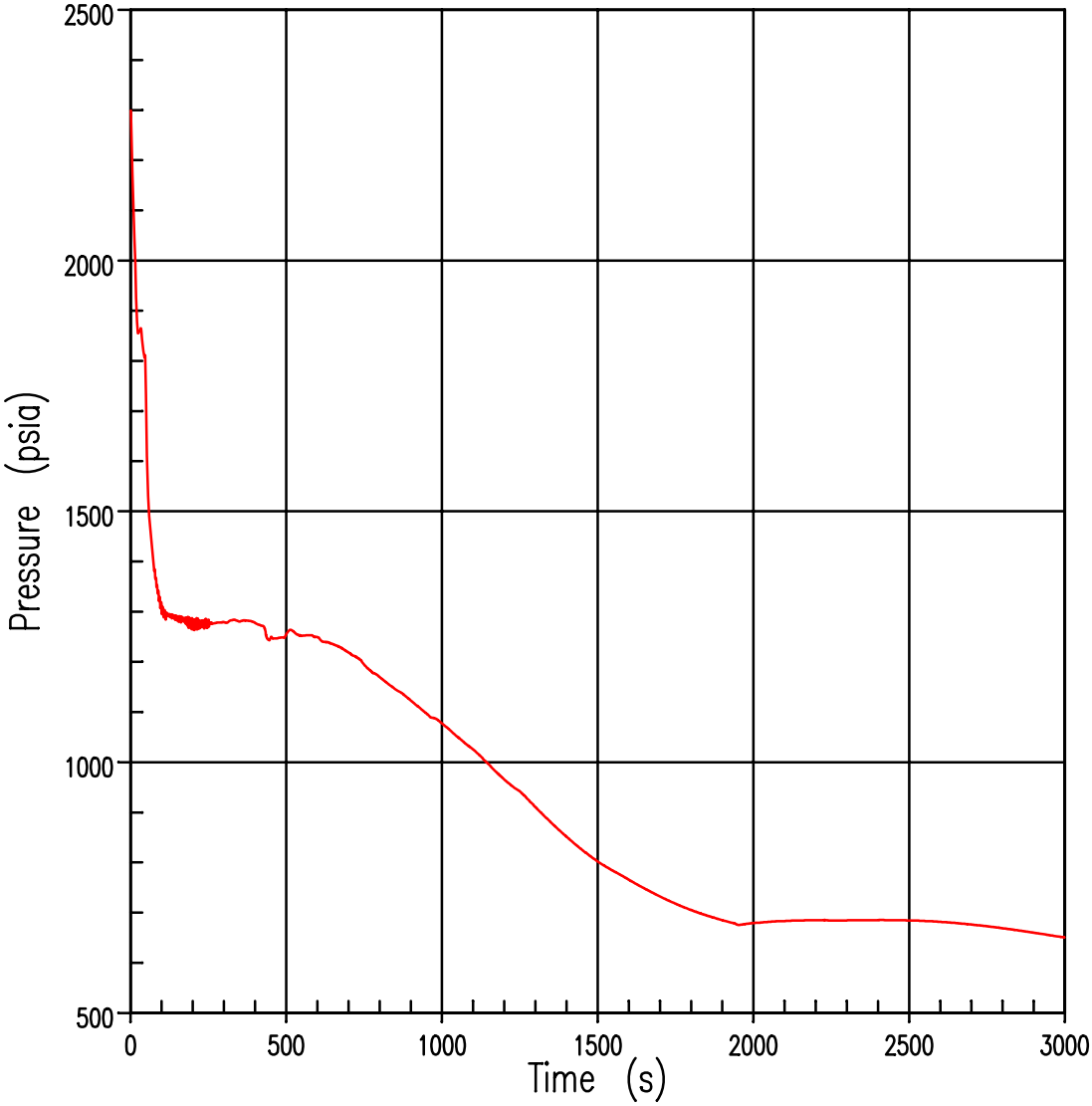

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FIGURE 15.6-38

REACTOR COOLANT SYSTEM PRESSURE
(3-INCH BREAK)

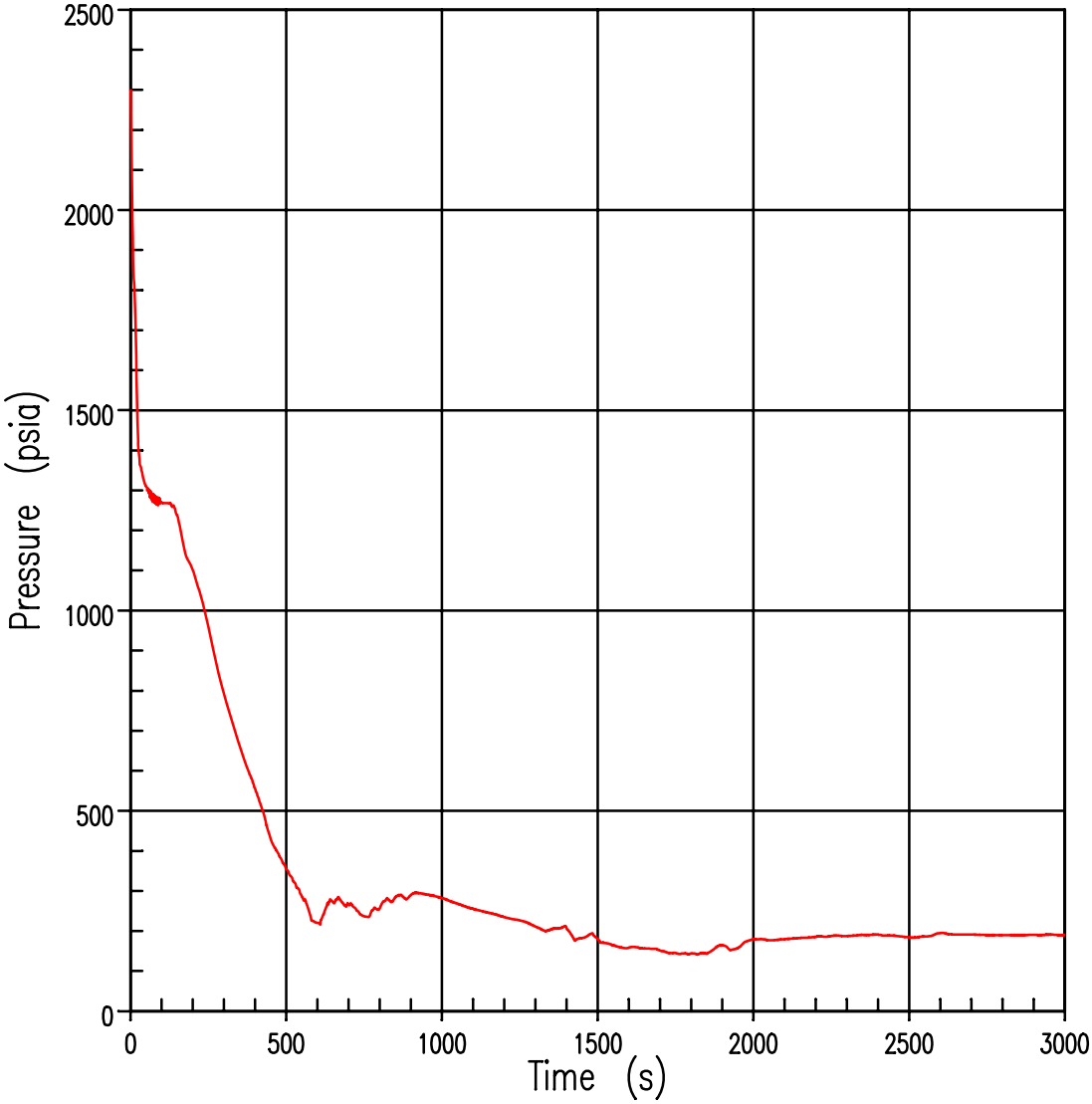

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FIGURE 15.6-39

REACTOR COOLANT SYSTEM PRESSURE
(6-INCH BREAK)

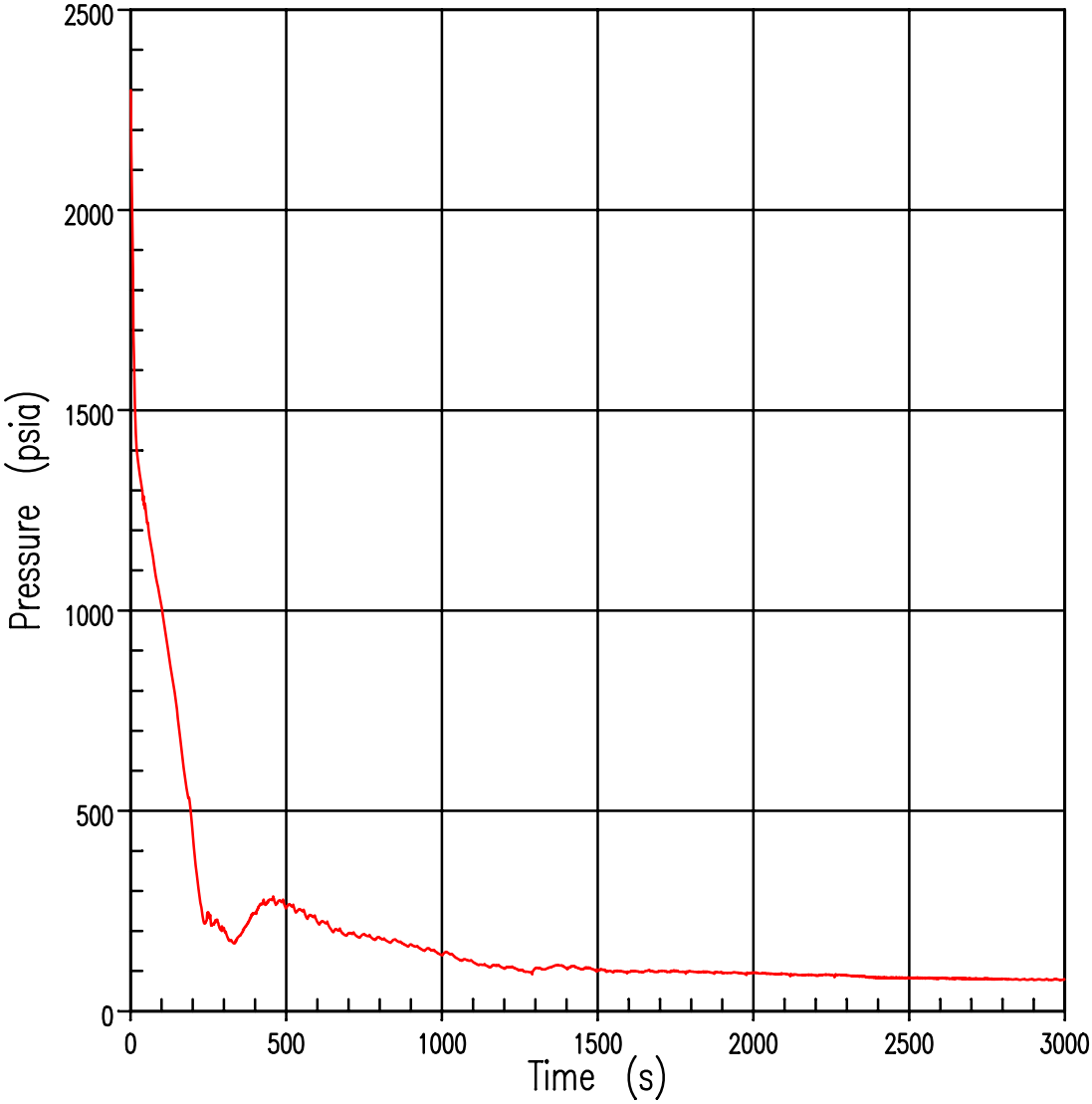

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FIGURE 15.6-39A

REACTOR COOLANT SYSTEM PRESSURE
(8.75-INCH BREAK)

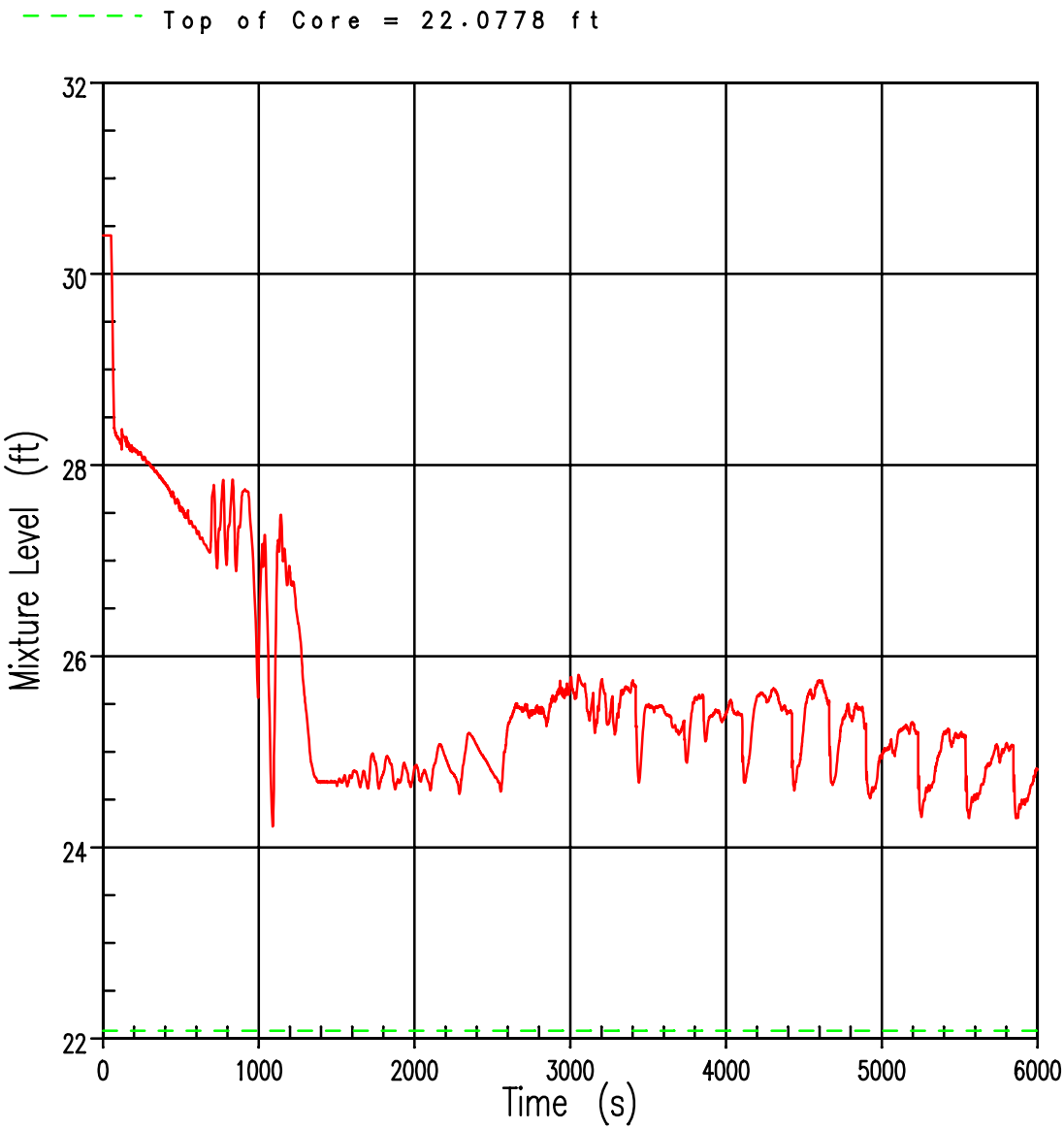

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FIGURE 15.6-40

CORE MIXTURE HEIGHT
(2-INCH BREAK)

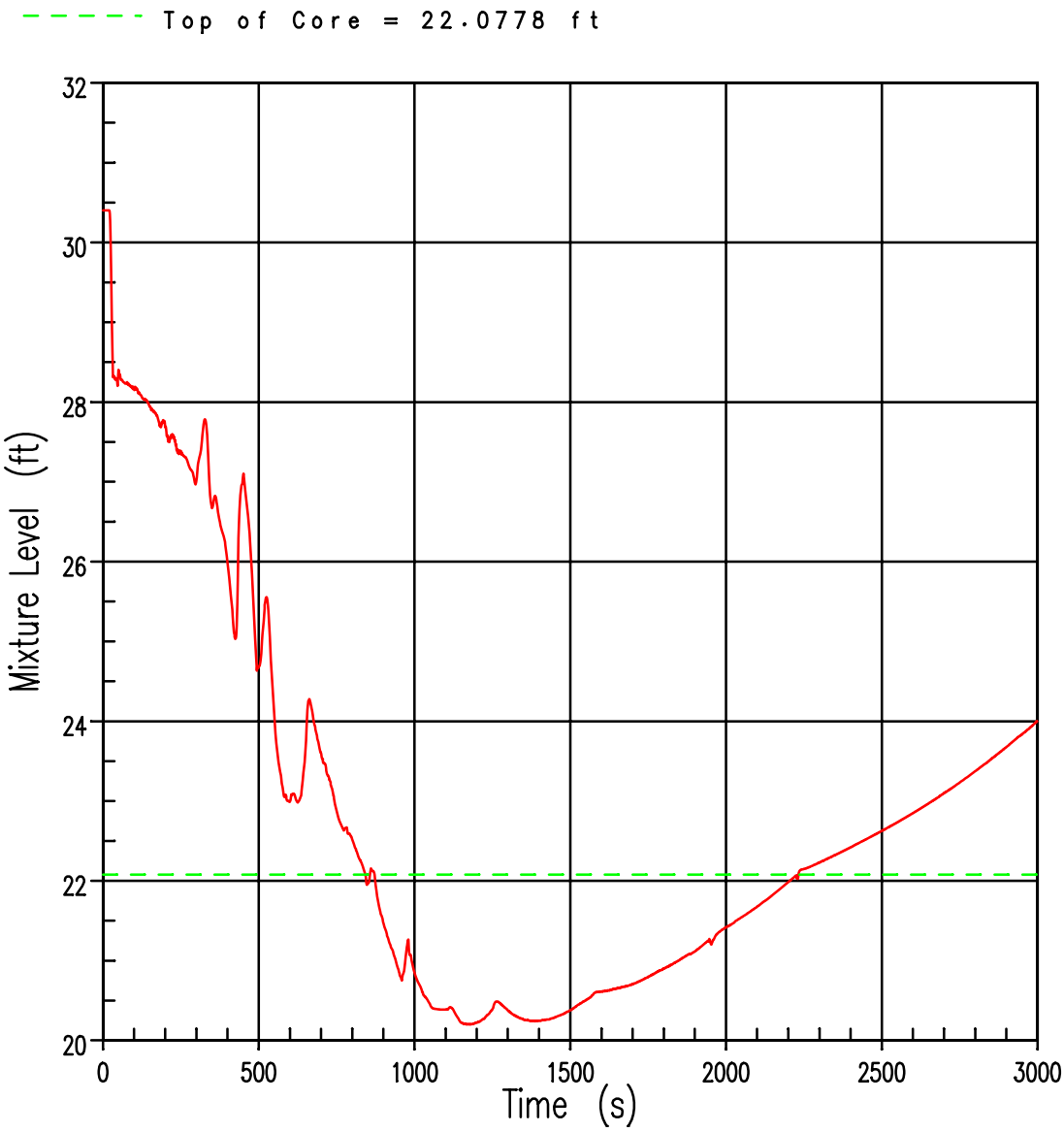

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FIGURE 15.6-41

CORE MIXTURE HEIGHT
(3-INCH BREAK)

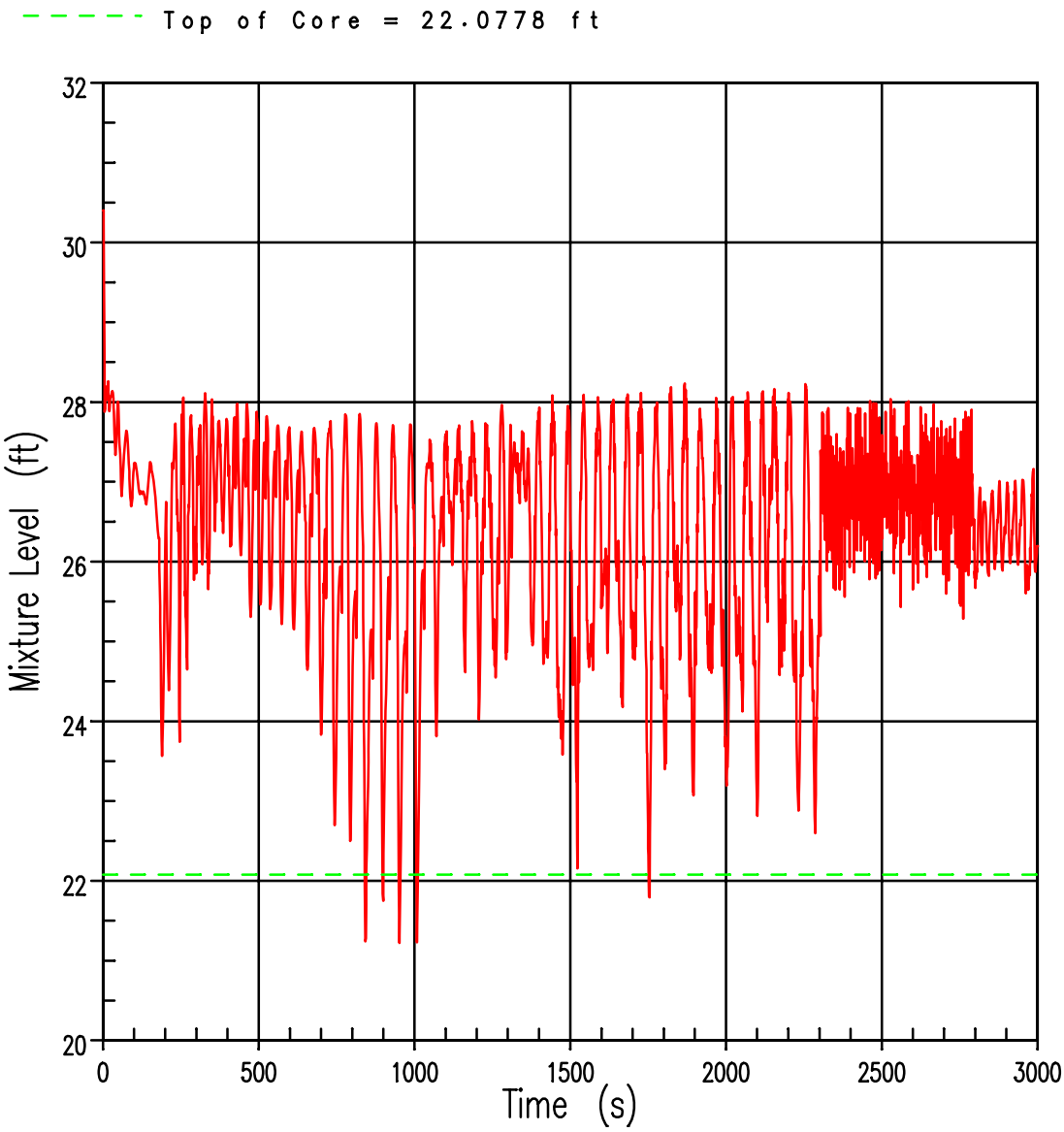

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FIGURE 15.6-42

CORE MIXTURE HEIGHT
(6-INCH BREAK)

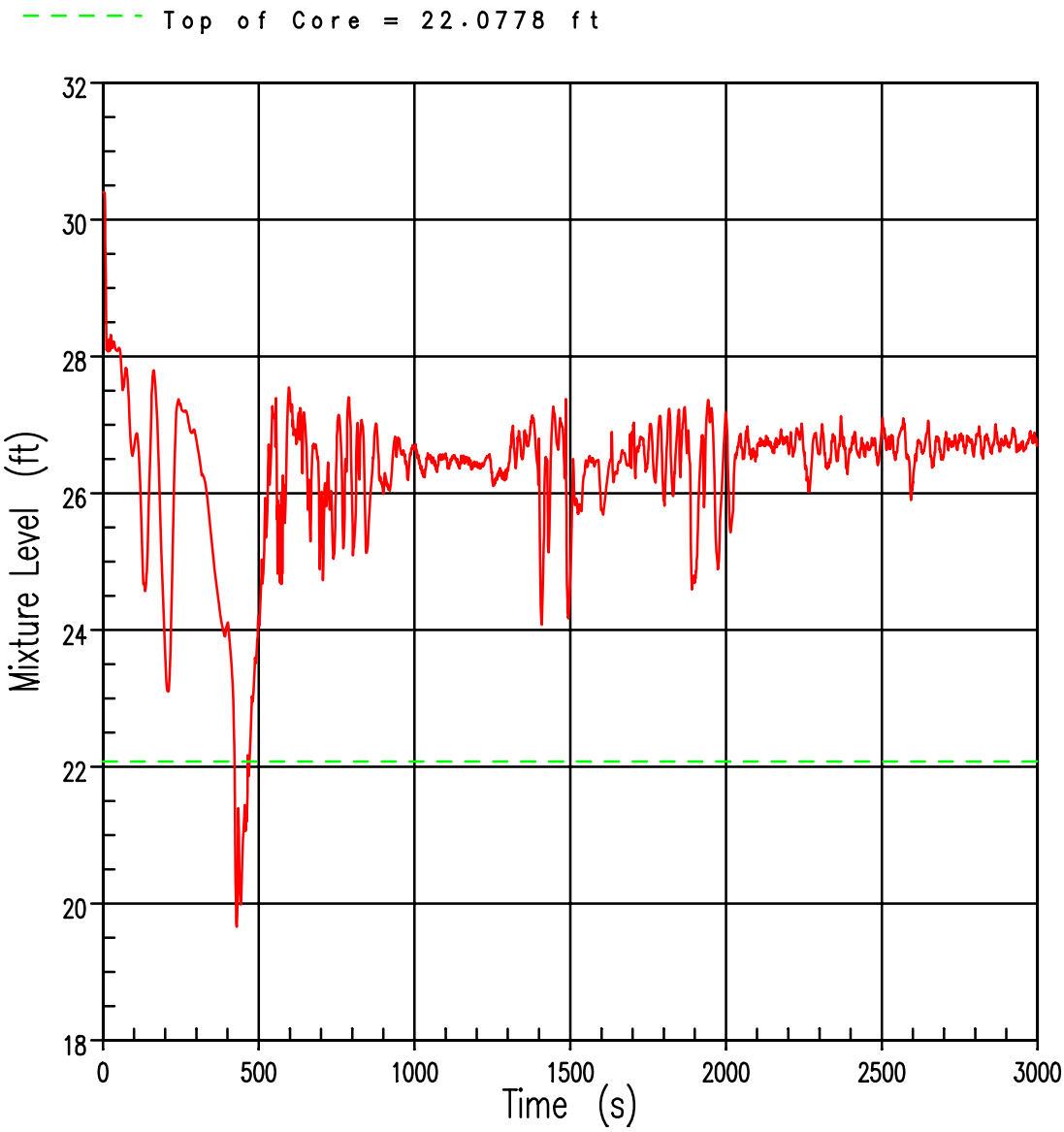

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FIGURE 15.6-42A

CORE MIXTURE HEIGHT
(8.75-INCH BREAK)

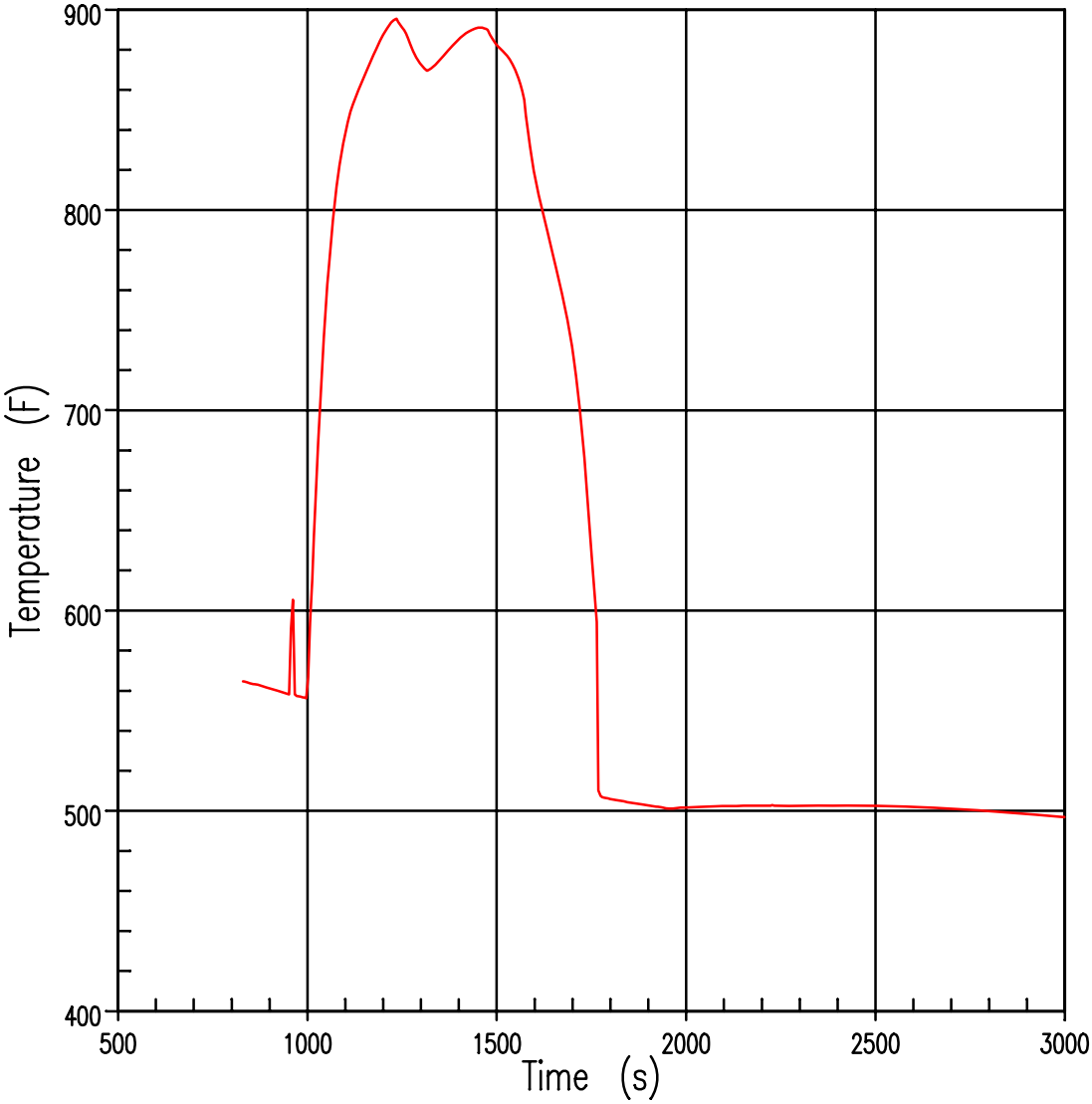

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FIGURE 15.6-44

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (3-INCH BREAK)

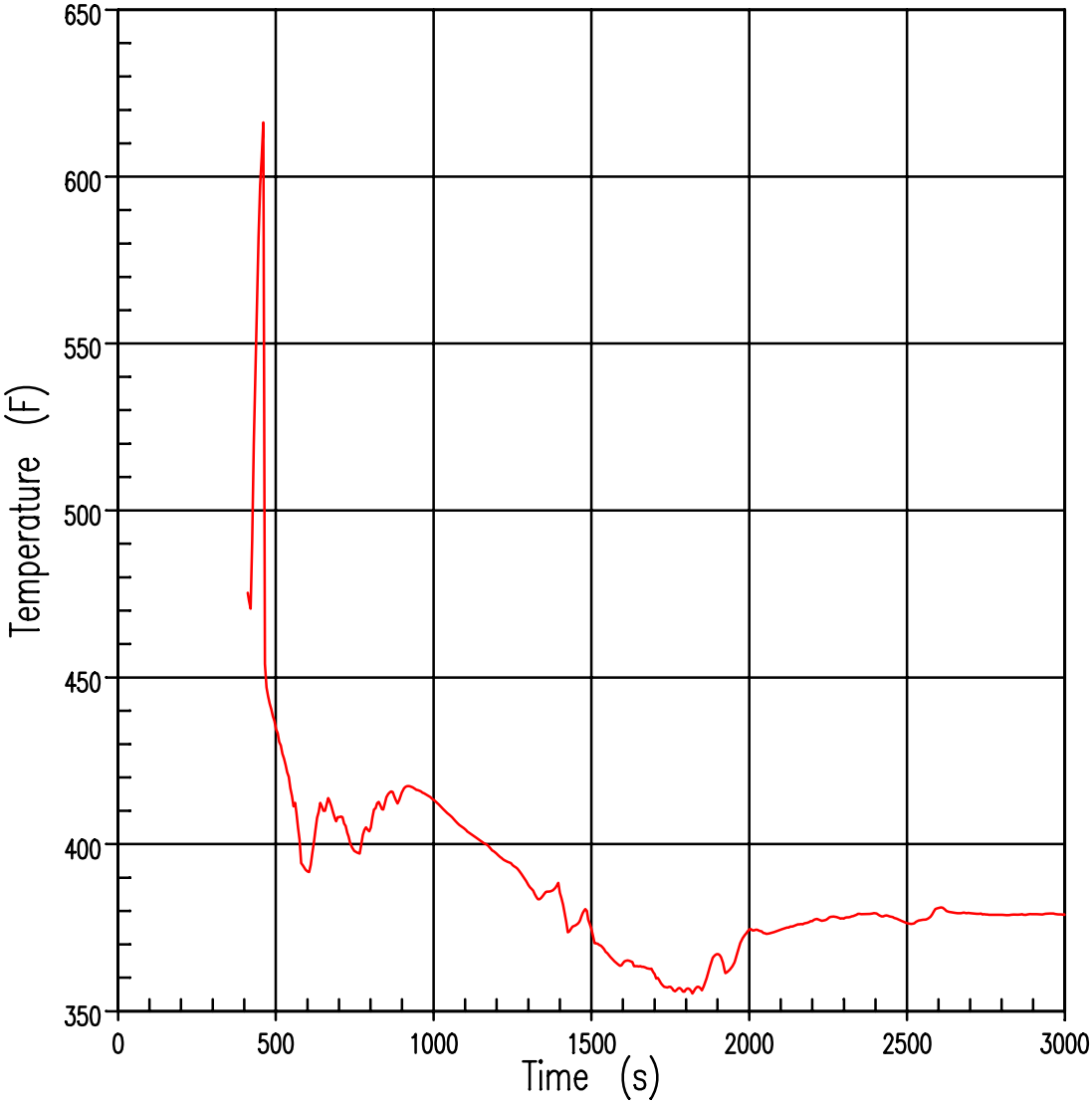

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FIGURE 15.6-45

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (6-INCH BREAK)

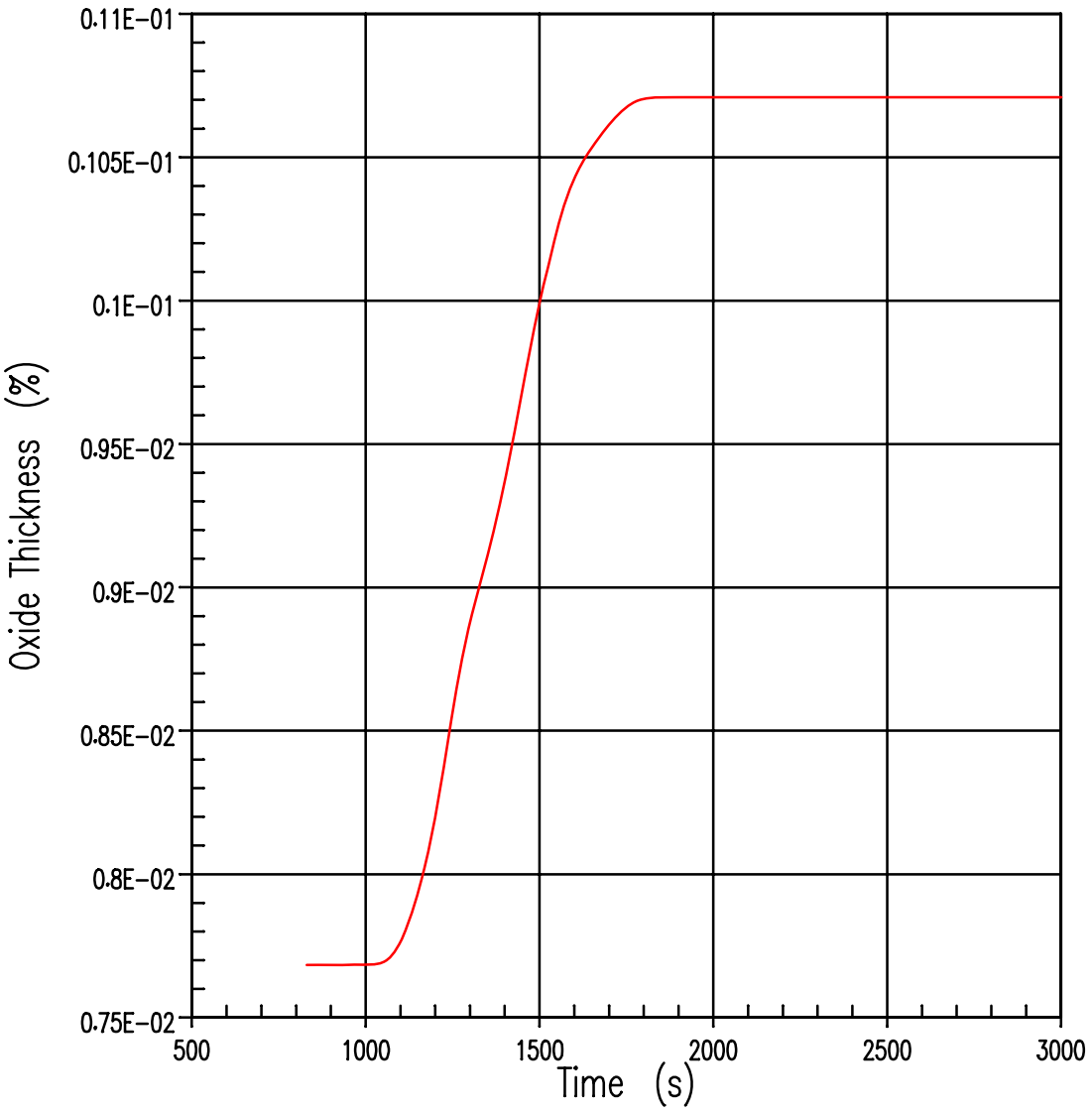

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FIGURE 15.6-45A

MAXIMUM LOCAL ZRO_2 THICKNESS (%)
(3-INCH BREAK)

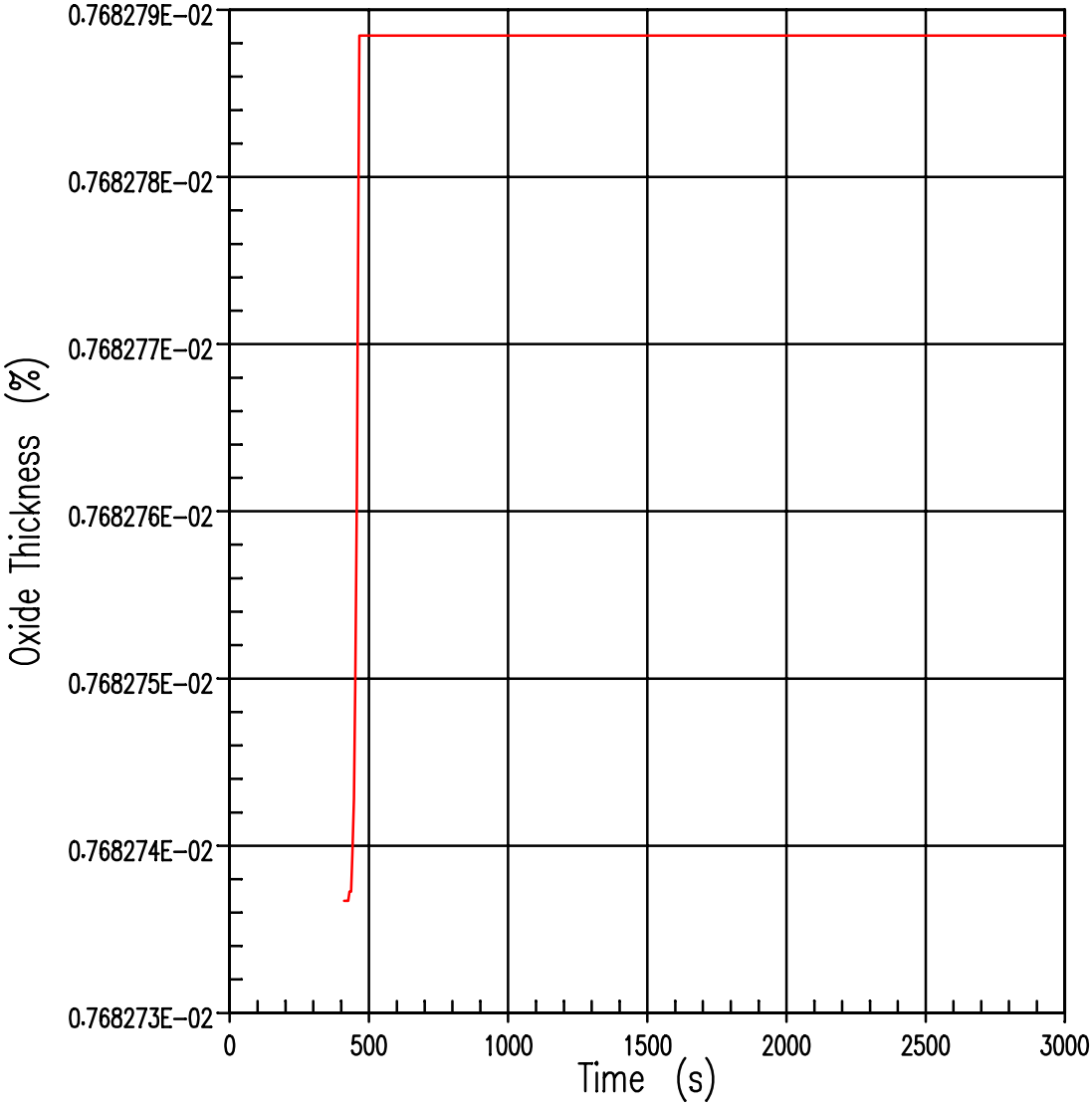

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FIGURE 15.6-45B

MAXIMUM LOCAL ZRO_2 THICKNESS (%)
(6-INCH BREAK)

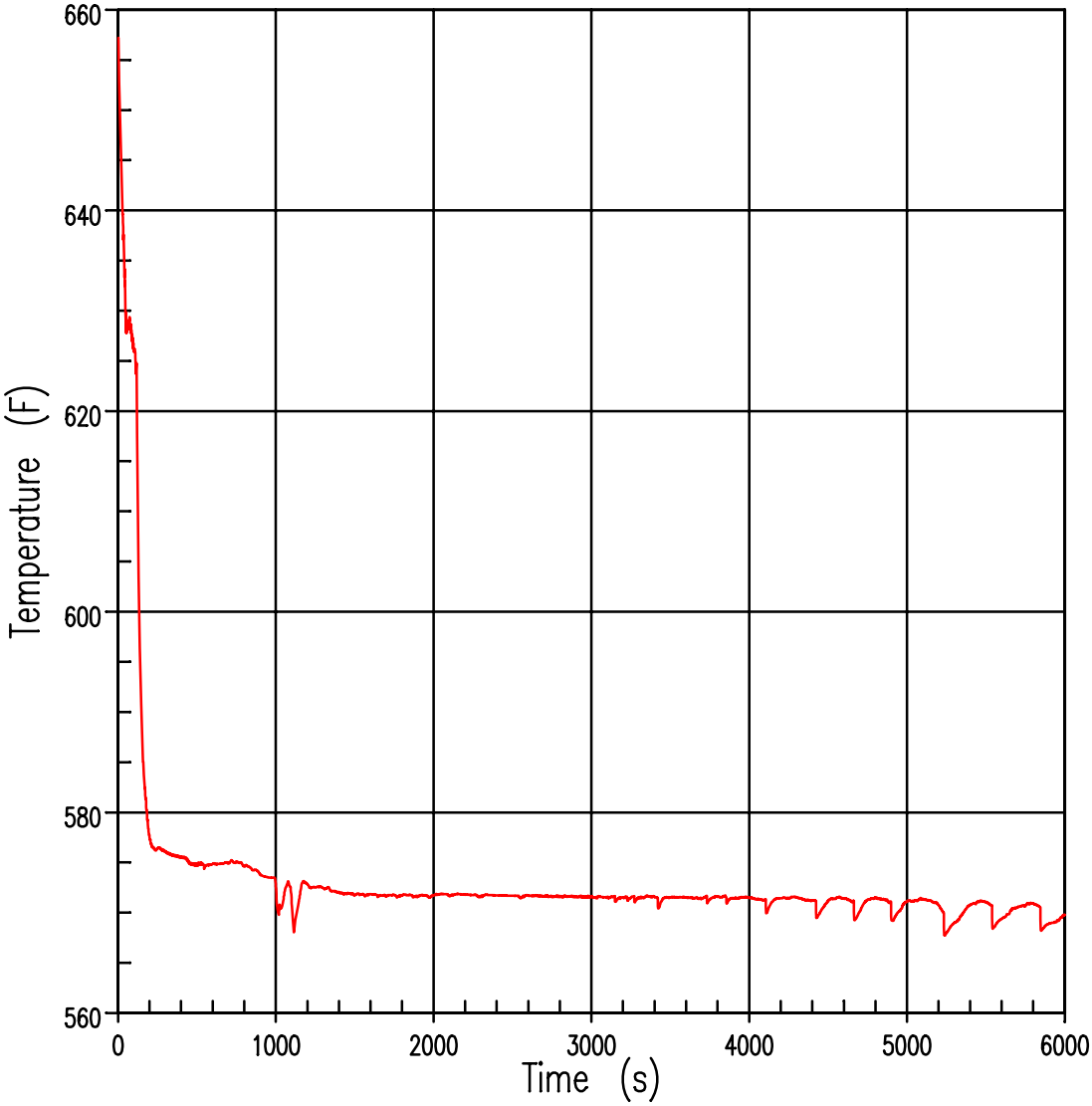

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FIGURE 15.6-45C

TOP CORE EXIT VAPOR TEMPERATURE
(2-INCH BREAK)

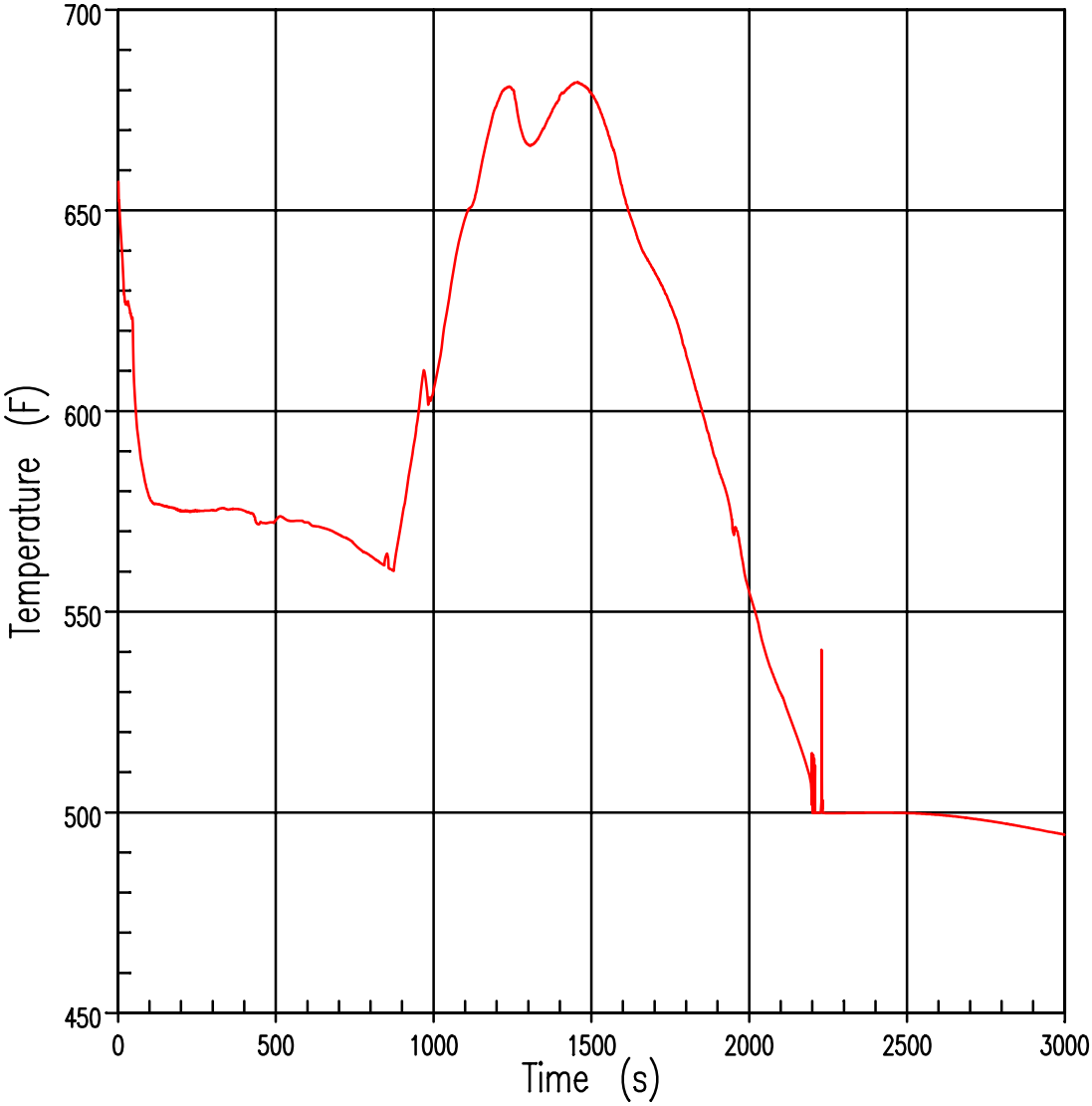

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FIGURE 15.6-45D

TOP CORE EXIT VAPOR TEMPERATURE
(3-INCH BREAK)

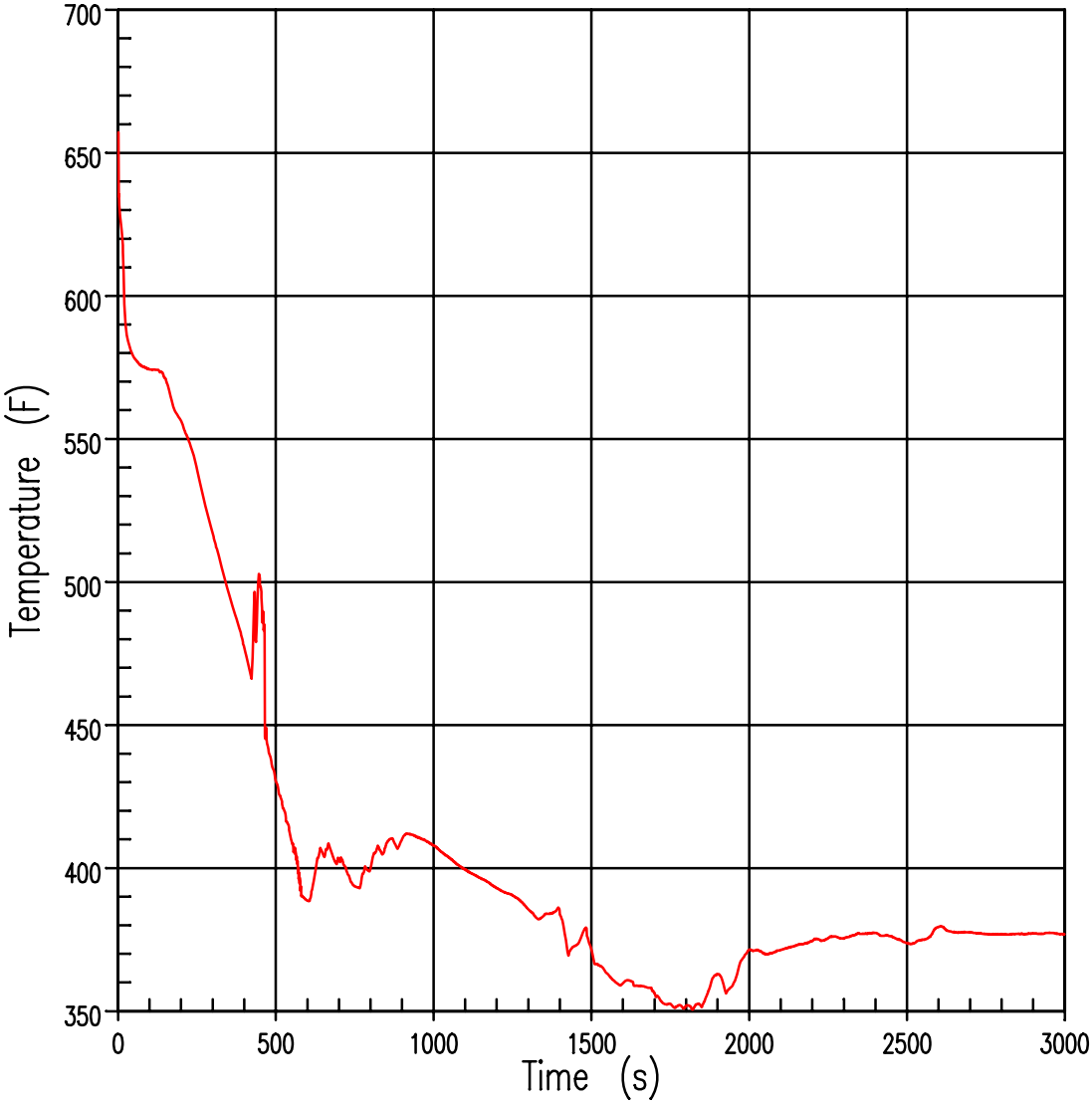

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FIGURE 15.6-45E

TOP CORE EXIT VAPOR TEMPERATURE
(6-INCH BREAK)

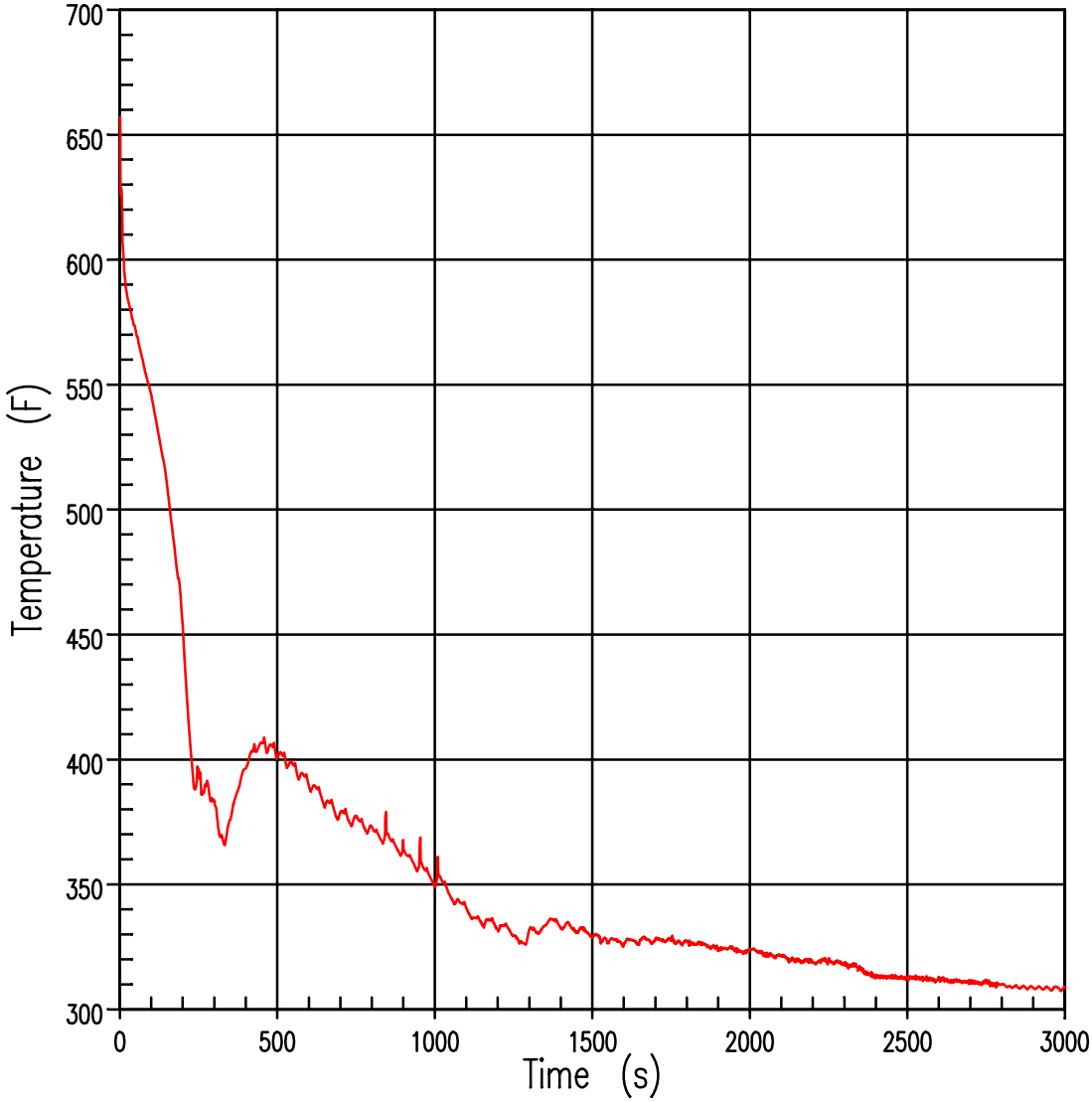

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FIGURE 15.6-45F

TOP CORE EXIT VAPOR TEMPERATURE
(8.75-INCH BREAK)

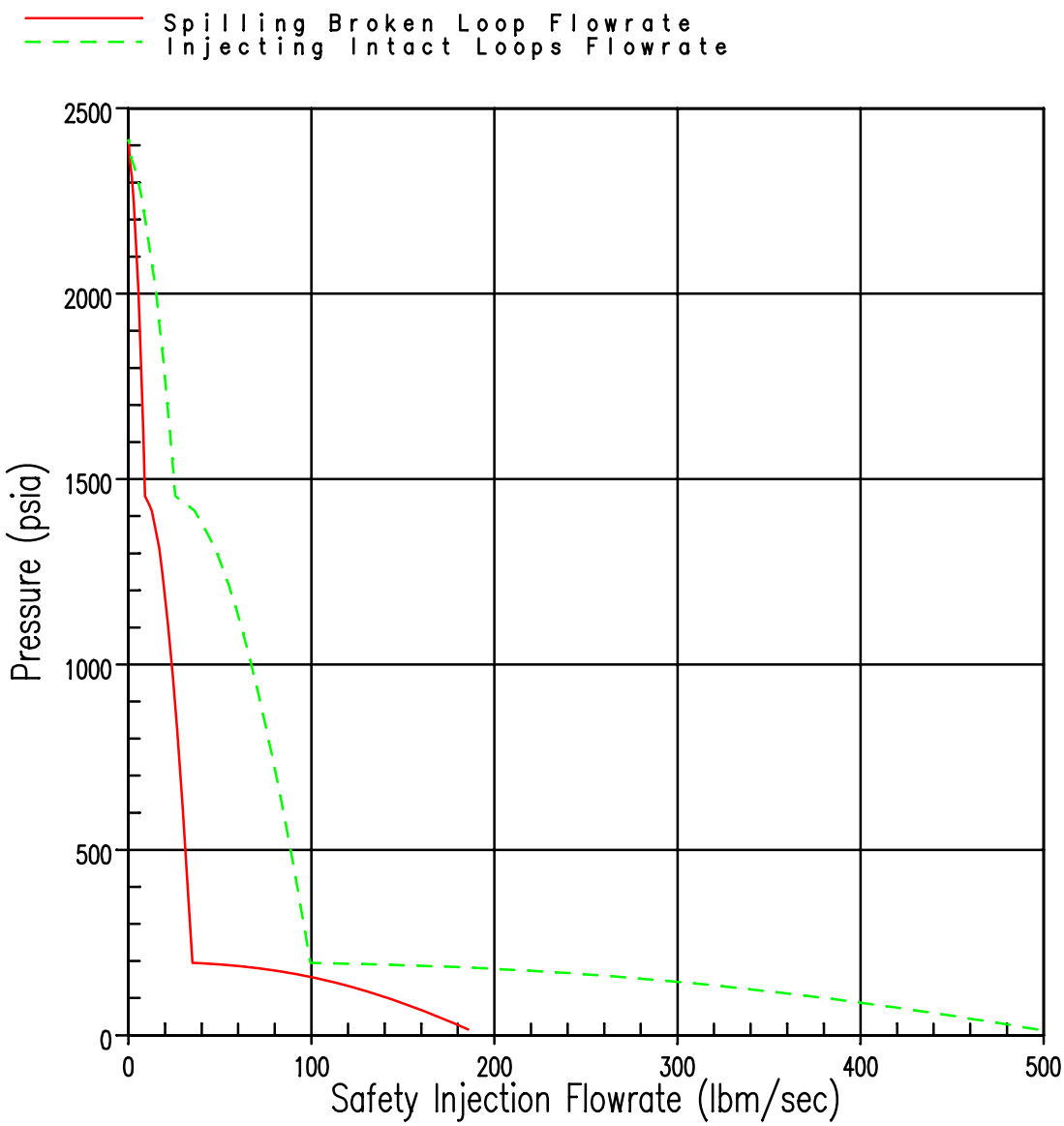

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FIGURE 15.6-46

PUMPED SAFETY INJECTION FLOW RATE, FAULTED LOOP INJECTS TO RCS PRESSURE
(2-6 INCH BREAKS)

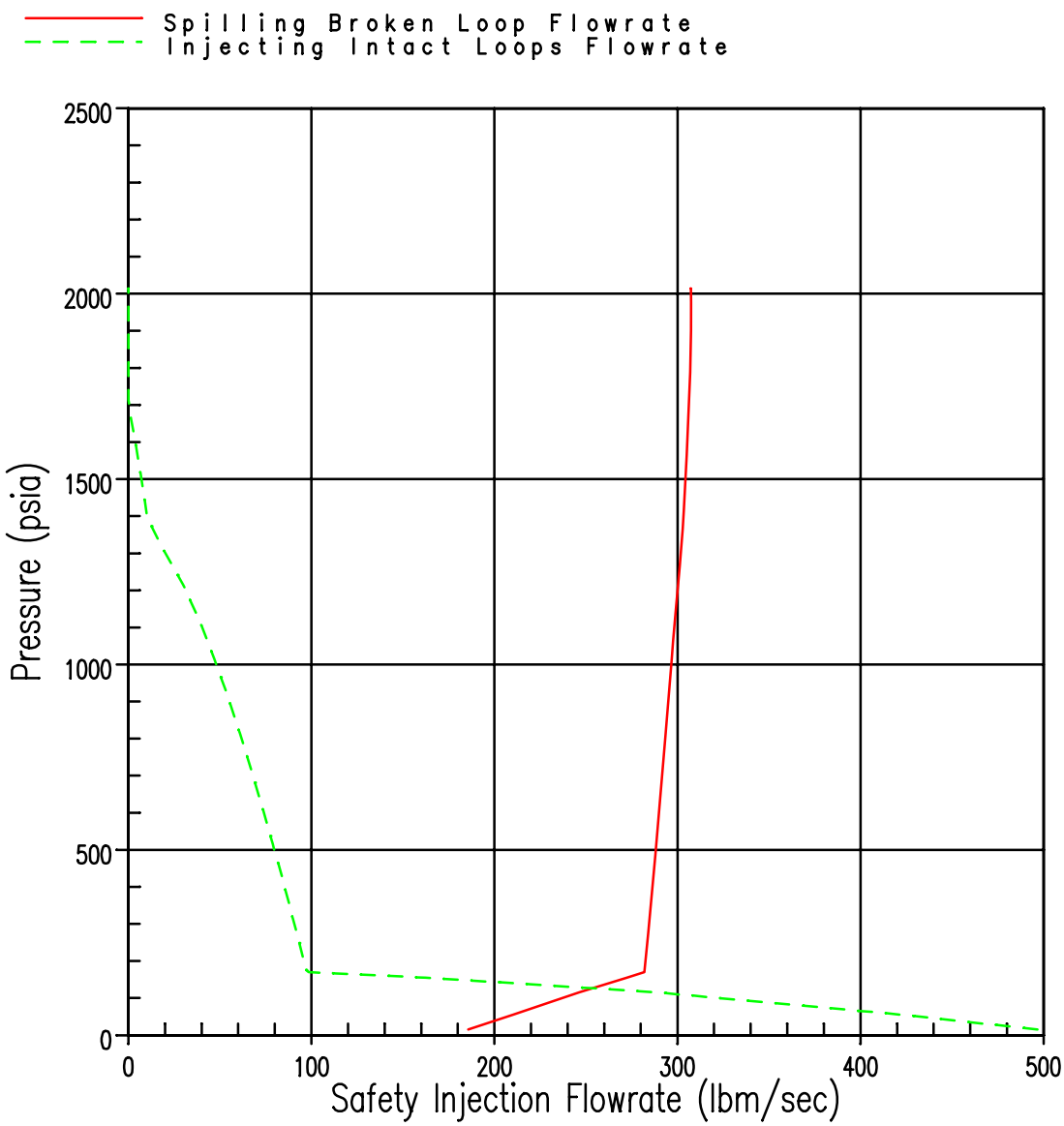

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FIGURE 15.6-46A

PUMPED SAFETY INJECTION FLOW RATE, FAULTED LOOP SPILL TO CONTAINMENT PRESSURE (0 PSIG) (2-6 INCH BREAKS)

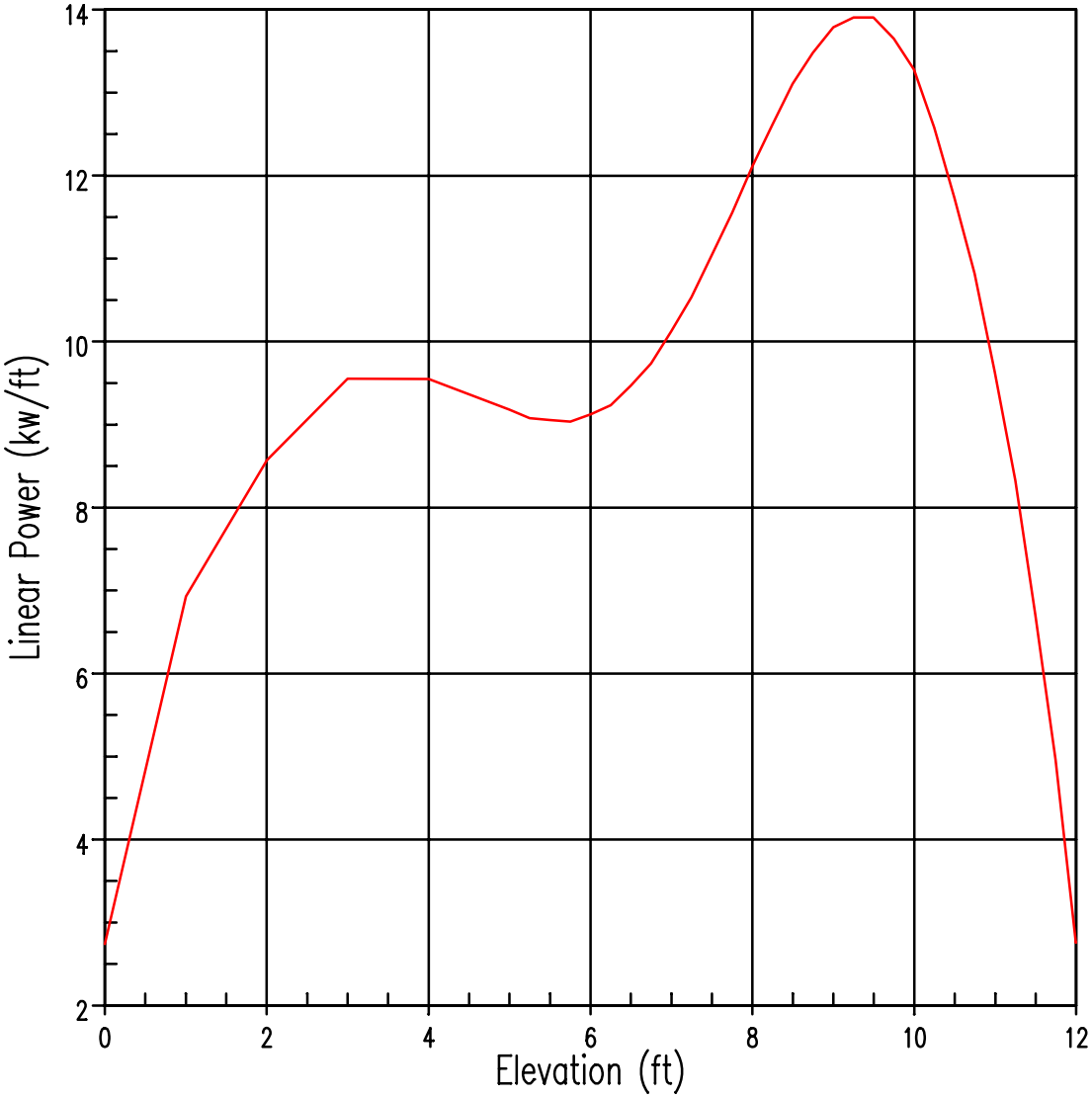

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FIGURE 15.6-47

HOT ROD AXIAL POWER SHAPE

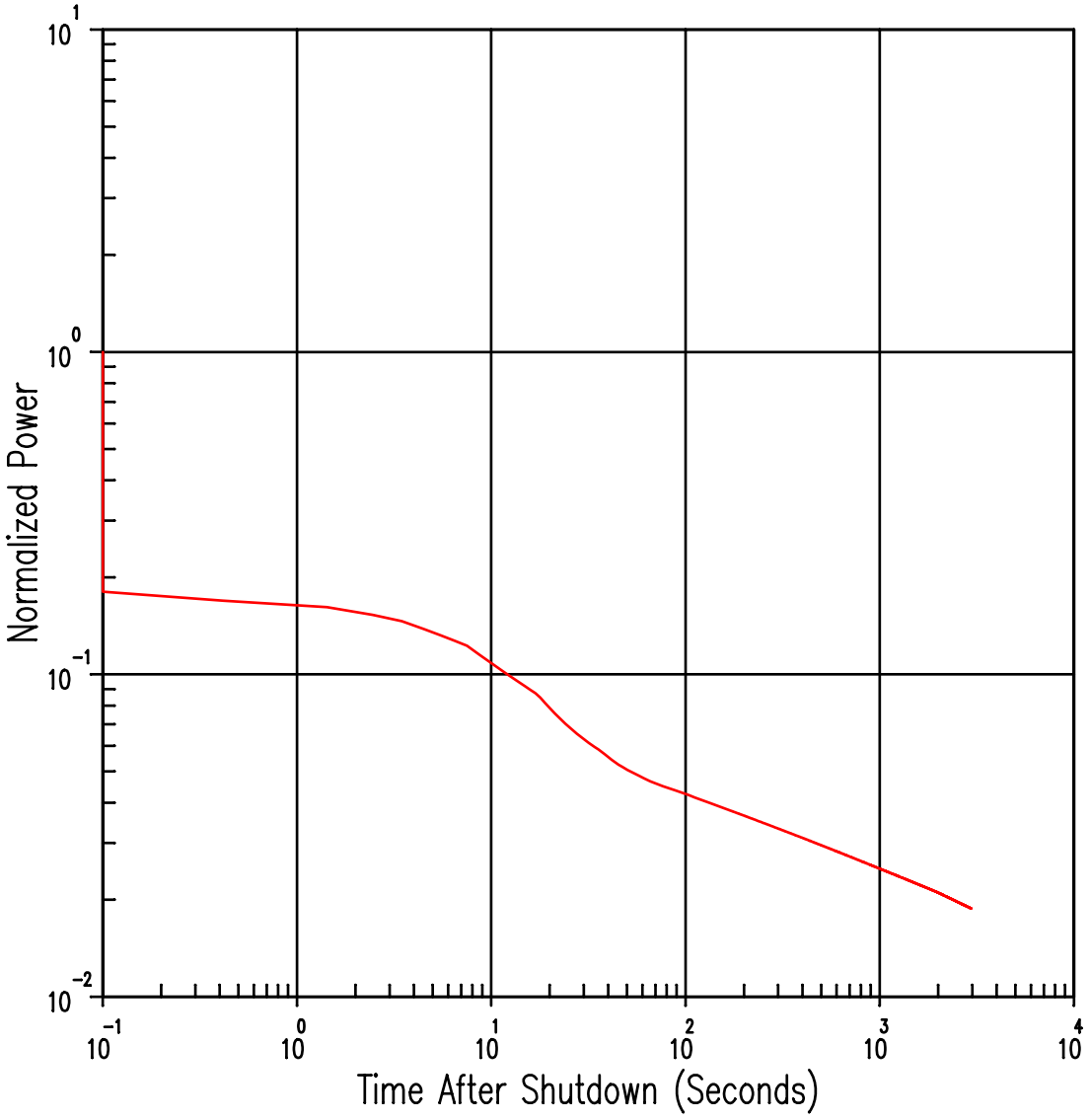

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FIGURE 15.6-48

CORE POWER (INCLUDING RESIDUAL FISSION) AFTER REACTOR TRIP (APPLIES TO ALL SMALL BREAKS)

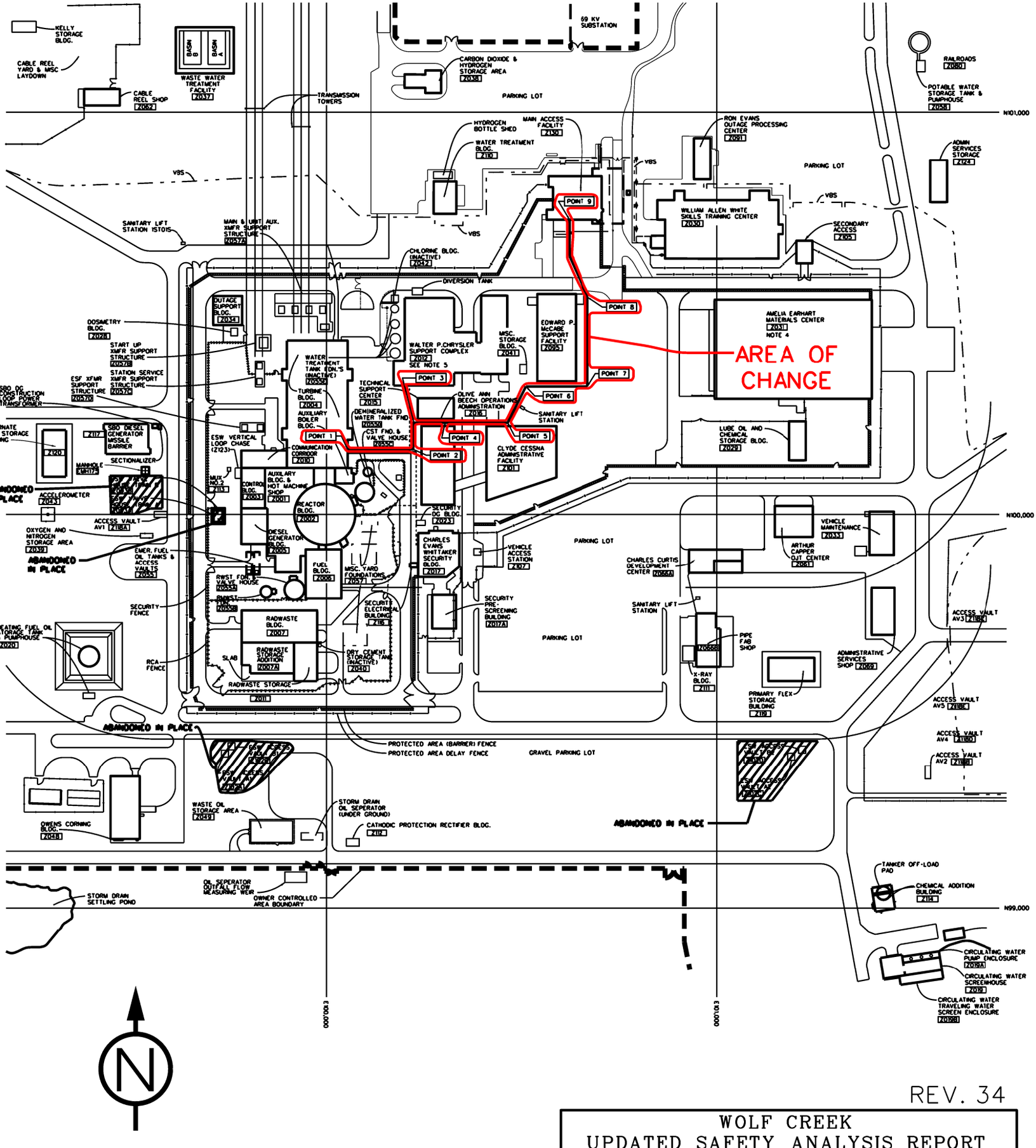

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FIGURE 15.6-49

OPERATOR INGRESS/EGRESS PATH TO CONTROL ROOM FOLLOWING A LOCA

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This class of accident can be caused by any of the following events:

a. Radioactive gas waste system leak or failure - this is an ANS Condition III event.

b. Radioactive liquid waste system leak or failure - this is an ANS Condition III event.

c. Postulated radioactive release due to liquid tank failures - this is an ANS Condition IV event.

d. Fuel handling accident - this is an ANS Condition IV event.

e. Spent fuel cask drop accidents - this is an ANS Condition III event.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the fuel handling accident analyzed in Section 15.7.4.

15.7.1 RADIOACTIVE WASTE GAS DECAY TANK FAILURE

15.7.1.1 Identification of Causes

This accident is an infrequent fault. Its consequences are considered in this section. The accident is defined as an unexpected and uncontrolled release of radioactive iodine, xenon, and krypton fission product gases stored in a waste gas decay tank as a consequence of a failure of a single gas tank or associated piping.

15.7.1.2 Sequence of Events and System Operations

During a refueling shutdown, the radioactive gases are stripped from the primary coolant and are stored in the gas decay tanks. After the transfer has been completed, the tank is assumed to fail. This releases all of the contents of the tank to the radwaste building. Also, since the tanks are isolated from each other, the only radioactivity released is from the failed tank. For conservatism, the gas decay tank inventory for all radionuclides except Kr-85 is assumed to be the maximum activity for each nuclide during the degassing operation. Further, the Kr-85 activity inventory is assumed to be the total activity released to the primary coolant during the fuel cycle.

15.7.1.3 Core and System Performance

This accident occurs when the reactor is in the shutdown condition. There is no impact on the core or its system performance.

15.7.1.4 Barrier Performance

The only barrier between the released activity and the environment is the radwaste building. During the course of this accident, the radwaste building is assumed to remain intact. This means that the only method of release is through the radwaste building ventilation system.

15.7.1.5 Radiological Consequences

15.7.1.5.1 Method of Analysis

15.7.1.5.1.1 Physical Model

Radioactive waste gas decay tanks are used in the design to permit the decay of radioactive gases as a means of reducing or preventing the release of radioactive materials to the atmosphere. To evaluate the radiological consequences of the gaseous waste processing system, it is postulated that there is an accidental release of the contents of one of the waste gas decay tanks resulting from a rupture of the tank or from another cause, such as operator error or valve malfunction. The gaseous waste processing system is so designed that the tanks are isolated from each other during use, limiting the quantity of gas released in the event of an accident by preventing the flow of radioactive gas between the tanks.

The principal radioactive components of the waste gas decay tanks are the noble gases krypton and xenon and trace quantities of halogens. The maximum amount of waste gases stored in any one tank occurs after a refueling shutdown, at which time the waste gas decay tanks store the radioactive gases stripped from the reactor coolant.

The maximum content of a gas decay tank which is conservatively assumed for the purpose of computing the noble gas and iodine inventory available for release is given in Table 15.7-3. Rupture of the waste gas decay tank is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building. For the purposes of evaluating the accident, it is assumed that all the activity is released directly to the environment during the 2-hour period immediately following the accident.

15.7.1.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.7-3.

In the evaluation of the waste gas decay tank rupture, the release assumptions of Regulatory Guide 1.24 have been used. Table 15.7-1 provides a comparison of the assumptions used in the analysis to those of Regulatory Guide 1.24. The assumptions related to the release of radioactive gases from the postulated rupture of a waste gas decay tank are:

a. The reactor has been operating at full core power with 1 percent defective fuel, and a shutdown to cold condition has been conducted prior to the accident.

b. All noble gas and gaseous iodine activity has been removed from the reactor coolant system via the CVCS volume control tank and transferred to the gas decay tank that is assumed to fail.

c. The maximum content of the waste gas decay tank was conservatively assumed to calculate the isotopic activities given in Table 15.7-3. The source term determination takes into account degassing of the reactor coolant system at shutdown.

d. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the radwaste building.

e. The dose is calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident, with no holdup. Credit for radioactive decay within the radwaste building is taken. Once the activity is released to the environment, no credit for radioactive decay is taken.

f. The control building and control room ventilation systems are configured in the normal power generation lineup at the time of the accident and remain in this configuration for the duration of a postulated tank failure event. That is, the control room ventilation isolation signal will not be actuated and consequently, the control room emergency ventilation system is not assumed to operate following the initiation of the accident.

15.7.1.5.1.3 Mathematical Models Used in the Analysis

The mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in Section 2.3.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and control room were analyzed, using the models described in Appendix 15A.

15.7.1.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated waste gas decay tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.1.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a waste gas decay tank rupture result from assumptions made involving the release of the waste gas from the decay tank and the meteorology present at the site during the course of the accident.

a. The accumulated activity following plant shutdown in the waste gas decay tank is based on 1 percent failed fuel, which is greater than that assumed fuel, which is greater than that assumed under normal operating conditions.

b. It is assumed that the waste gas decay tank fails immediately after the transfer of radioactive gases from the reactor coolant to the waste gas decay tank is complete. These assumptions result in the greatest amount of activity available for release to the environment.

c. The activity contained in the ruptured waste gas decay tank was assumed to be released over a 2-hour period immediately following the accident. This is a conservative assumption. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the decay tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would

be gained. This reduces the amount of activity released to the environment due to natural decay. Also no credit for iodine removal by the non-safety grade radwaste building HVAC charcoal adsorbers has been taken.

d. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, will be conservative.

15.7.1.5.3 Conclusions

15.7.1.5.3.1 Filter Loading

Since the accumulated iodine activity in the waste gas decay tanks is negligible, filter loading due to a waste gas decay tank rupture does not challenge the necessary design margin for the radwaste building exhaust or the control room intake filters. Hence, the respective filter loadings were not evaluated.

15.7.1.5.3.2 Dose to Receptor at the Exclusion Area Boundary, the Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated waste gas decay tank rupture have been conservatively analyzed using the RATRAD code, based on assumptions and models described in previous sections.

The analysis results show that the radiological consequences of the postulated waste gas decay tank failure do not exceed the limit set in 10 CFR 20 for offsite doses. The resultant control room dose is less than the limit set forth in 10 CFR 50.67, even with no credit taken for the CREVS and its actuation instrumentation. The results are listed in Table 15.7-4

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 Identification of Causes

This is an infrequent fault because, although it is unlikely to happen, the potential for release of significant amounts of radioactivity is present. The accident may be caused by an equipment malfunction or tank failure or rupture.

15.7.2.2 Sequence of Events and System Operation

The radioactive liquid tank is assumed to fail. This releases a maximum of 80 percent of the tank capacity to the equipment compartment.

15.7.2.3 Core and System Performance

This accident does not affect the core or the core system performance.

15.7.2.4 Barrier Performance

There are no barriers to the release of radioactivity from the radwaste building.

15.7.2.5 Radiological Consequences

15.7.2.5.1 Method of Analysis

15.7.2.5.1.1 Physical Model

The liquid radwaste tanks are used as a means of collecting waste to be: 1) processed through the liquid radwaste system, 2) pumped to the Solid Radwaste System, or 3) discharged from the plant. To evaluate the radiological consequences of the liquid waste processing system, it is postulated that there is an accidental release of the contents of one of the tanks.

In the analyses, it is assumed that the liquid contents of a tank are released to the radwaste building and, subsequently, the airborne activity is released to the environment during the 2-hour period immediately following the tank failure.

Two tanks have been analyzed for this accident, and the radiological consequences for both tanks are provided. The boron recycle holdup tank was selected because it contained the maximum total inventory. In addition a hypothetical tank containing the maximum possible amount of iodine was analyzed. Although the primary spent resin tank contains the highest inventory of airborne and soluble iodine, it is considered extremely unlikely that all the iodine activity will become airborne in case the tank fails. The assumptions, conditions, and mathematical models described in this section are identical for both tanks, except as stated.

15.7.2.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in this analysis are listed below and in Tables 15.7-5 and 15A-1:

a. The isotopic inventories of the ruptured tanks are given in Table 15.7-5 and are based on 1-percent failed fuel. The isotopic inventory of the ruptured hypothetical tank was calculated assuming that all of the iodine in the streams entering the liquid radwaste system are concentrated in a hypothetical tank where the only means of depletion is radioactive decay. These input streams are shown on USAR Figure 11.1A-2 sheets 2, 3, and 4.

b. The tank failure is assumed to occur when the contents of the tank are at a maximum.

c. The doses are calculated as if the release were from the radwaste building at ground level during the 2-hour period immediately following the accident. Credit is taken for radioactive decay during holdup in the tank. Once the activity is released to the environment, no credit for radioactive decay is taken in the transit to the site boundary, outer boundary of the low population zone, or control room.

d. For the boron recycle holdup tank 100 percent of all noble gas and iodine activity in the tank is released as airborne activity. 100 percent of all iodine activity is released from the hypothetical liquid waste tank.

e. Credit for iodine removal by non-safety grade radwaste building HVAC charcoal adsorber is not taken.

15.7.2.5.1.3 Mathematical Models Used in the Analysis

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors used in the analysis were calculated, based on the onsite meteorological measurement programs described in Section 2.3, and are provided in Table 15A-2.

c. The TEDE doses to a receptor at the exclusion area boundary, outer boundary of the low-population zone, and in the control room were analyzed, using the models described in Appendix 15A.

15.7.2.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to the postulated liquid radwaste tank rupture, the resultant activity is conservatively assumed to be released directly to the environment during the 2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual building exhaust ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.2.5.2 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of the liquid radwaste tank rupture result from assumptions made involving the release of the radioactivity from the tanks and the meteorology assumed for the site.

a. It was assumed that the liquid radwaste tank fails when the inventory in the tank is a maximum. This assumption results in the greatest amount of activity available for release to the environment.

b. The contents of the ruptured tank are assumed to be released over a 2-hour period immediately following the accident. If the contents of the tank were assumed to mix uniformly with the volume of air within the radwaste building where the tanks are located, then, using the actual building exhaust ventilation rate, a considerable amount of holdup time would be gained. This reduces the amount of activity released to the environment due to the natural decay. Also, no credit for iodine removal by the radwaste building HVAC charcoal adsorbers is taken.

c. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time.

d. A tank is assumed to have collected liquid waste based on operation at 100-percent power with 1 percent failed fuel for an extended period of time, which is higher than under normal operating conditions.

15.7.2.5.3 Conclusions

15.7.2.5.3.1 Filter Loadings

The filter loading due to a liquid radwaste tank rupture does not challenge the necessary design margin for the control room intake filters. Thus, the filter loading was not evaluated.

15.7.2.5.3.2 Doses to Receptor at the Exclusion Area Boundary, the Low-Population Zone Outer Boundary, and in the Control Room

The radiological consequences resulting from the occurrence of a postulated liquid radwaste tank rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The TEDE doses have been analyzed for the 0-2-hour dose at the exclusion area boundary, for the duration of the accident at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.7-6. The resultant offsite doses are less than the limit set in 10 CFR 20. The resultant control room dose is less than the limit set forth in 10 CFR 50.67.

15.7.3 POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES

This analysis is presented in Section 2.4.13.3.

15.7.4 FUEL HANDLING ACCIDENTS

The postulated fuel handling accident has been analyzed for two cases: Case 1, a fuel handling accident outside the containment, and Case 2, a fuel handling accident inside the reactor building.

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as the dropping of a spent fuel assembly while in transit between the spent fuel pool and reactor vessel, resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the achievement of plant cold safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer after shutdown is 76 hours.

The postulated fuel handling accident is assumed to occur during a core offload at least 76 hours after shutdown in either the reactor containment building, or in the fuel building subsequent to the transfer of a fuel assembly through the fuel storage pool transfer gate and prior to placement in a fuel storage pool storage rack designated location.

A fuel handling accident is also possible in the fuel building during fuel loading operations associated with the ISFSI, where spent fuel assemblies are being moved from the spent fuel pool to the dry shielded canister housed within the transfer cask located in the cask loading pool which would be water flooded during this event (i.e., transfer slot gate removed). This event would typically occur during normal reactor operation. The radiological consequences of this event are bounded by the fuel assembly drop in the spent fuel pool at a time period of 76 hours after reactor shutdown where it is assumed that cladding rupture occurs in all the fuel rods in the dropped fuel assembly.

15.7.4.3 Core and System Performance

As fuel damage can only occur inside the reactor vessel when the reactor is in the shutdown condition, a postulated fuel handling accident does not impair the safe operation of the reactor or its associated systems.

15.7.4.4 Barrier Performance

A barrier between the released activity and the environment is the reactor building and the fuel building. Since these buildings are designed seismic Category I, it is safe to assume that during the course of a fuel handling accident their integrity is maintained. This means that the pathway for release of radioactivity for a postulated accident in the fuel building is initially via auxiliary/fuel building normal exhaust system. After it is isolated on a high radiation signal, the release pathway is via the ESF emergency filtration system. For a postulated accident in the reactor building, since the containment personnel airlock (PAL) doors and equipment hatch are allowed to be open during core alterations or movement of irradiated fuels, portion of the gaseous effluent escaping from the refueling water pool in the Reactor Containment Building could be released to the environment via the open personnel and equipment hatches. The fuel storage pool and the refueling pool provide minimum decontamination factors of 170 for iodine.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model

The possibility of a fuel-handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel-handling operations (refer to Section 9.1.4). All refueling operations and spent fuel handling operations associated with the ISFSI are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the fuel storage pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to that in the fuel storage pool, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies were withdrawn. After the reactor head and rod cluster control drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the refueling pool, placed horizontally in a conveyor car and pulled through the transfer tube and canal, upended and transferred through the fuel storage pool transfer gate, then lowered into steel racks for storage in the fuel storage pool in a pattern which precludes any possibility of a criticality accident.

During ISFSI fuel loading operations, when transferring irradiated fuel from the racks in the spent fuel pool to the transfer cask located in the cask loading pool for transfer to the ISFSI, the spent fuel pool and cask loading pool are filled with borated water at a boron concentration in accordance with the NUHOMS® EOS System Technical Specifications, which ensures subcritical conditions are maintained. Irradiated fuel assemblies are lifted out of the storage racks in the spent fuel pool, transferred vertically through the transfer slot gate to the cask loading pool, then lowered into a dry storage canister housed within the transfer cask in a configuration which precludes any possibility of a criticality accident.

Fuel-handling manipulators and hoists are designed so that the fuel cannot be raised above a position that provides an adequate water shield depth for radiation protection of operating personnel.

The containment, fuel building, refueling cavity, refueling pool, fuel storage pool and cask loading pool including the associated cask pedestal are designed to seismic Category I requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The spent fuel storage racks and transfer cask when located on the pedestal in the cask loading pool are also designed to prevent any credible external missile from reaching the stored irradiated fuel. The fuel-handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel assemblies precludes damage to the fuel should portable or hand tools drop on an assembly.

The only time the postulated fuel-handling accident could occur is during the transfer of a fuel assembly from the core to its storage position in the fuel storage pool, or during fuel loading activities associated with the ISFSI. The facility is designed so that heavy objects, such as the spent fuel shipping cask or the ISFSI transfer cask, cannot be carried over or tipped over onto the irradiated fuel stored in the fuel storage pool. For hoisting activities over the cask loading pool while irradiated fuel assemblies are present in the transfer cask, the design of the cask handling crane main hoist and rigging for ISFSI equipment are such that rigging failures or crane drop events are precluded, since these components satisfy the single failure proof requirements of NUREG-0612 and ANSI N14.6. Only one fuel assembly is handled at a time by the refueling machine, transfer system, or spent fuel pool bridge crane. Movement of equipment handling the fuel is kept at low speeds while exercising caution that the fuel assembly does not strike another object or structure during transfer from the core to its storage position or from the storage racks in the fuel storage pool to the transfer cask in the cask loading pool. In the unlikely event that an assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling.

a. Reactor Building Accident

During fuel-handling operations, the containment is kept in an isolatable condition, with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from one of the redundant radiation monitors, indicating that radioactivity is above the prescribed limits.

In addition to the area radiation monitors in the containment, portable monitors capable of sounding audible alarms are to be located in the fuel-handling area. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors would sound an audible alarm, the containment would be isolated, and personnel would be evacuated.

During movement of irradiated fuel and core alterations, one door in the emergency air lock must be closed. The other emergency air lock door may be open provided it is capable of being closed. The equipment hatch may be open provided it is capable of being closed, and penetration flow paths providing direct access may be unisolated under administrative controls. Administrative controls are imposed for the closure of a personnel air lock door, closure of the equipment hatch, and to isolate open penetration flow paths in the event of a fuel handling accident.

b. Fuel Building Accident

In the fuel building, a fuel assembly could be dropped in the transfer canal, in the fuel storage pool, or in the cask loading pool.

In addition to the area radiation monitor located on the bridge over the fuel storage pool, portable radiation monitors capable of emitting audible alarms are located in this area during fuel-handling operations. The doors in the fuel building are closed to maintain controlled leakage characteristics in the fuel storage pool region during operations involving irradiated fuel. Should a fuel assembly be dropped in the canal, in the fuel storage pool, or in the cask loading pool and release radioactivity above a prescribed level, the radiation monitors sound an alarm.

If one of the redundant discharge vent radiation monitors indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the auxiliary/fuel building normal exhaust is switched to the ESF Emergency Exhaust system to allow the spent fuel pool ventilation to exhaust through the ESF charcoal filters to remove most of the halogens prior to discharging to the atmosphere via the unit vent. The supply ventilation system servicing the fuel storage pool area is automatically shut down, thus ensuring controlled leakage to the atmosphere through charcoal adsorbers (refer to Section 9.4).

The probability of a fuel-handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility, as previously mentioned.

15.7.4.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.7-7 and 15A-1.

In the evaluation of the fuel-handling accident, all the fission product release assumptions of Regulatory Guide 1.183 have been followed. Tables 15B-1 and 15B-3 provide a comparison of the analysis to the guidelines of Regulatory Guide 1.183. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, were used in the analyses:

a. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

b. The assembly fission product inventories are based on a radial peaking factor of 1.65.

c. The accident occurs 76 hours after shutdown, which is the earliest time fuel-handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

d. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation was assumed to be available for immediate release to the water following clad damage.

e. The gap activity released to the fuel pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine other than I-131, 12 percent of the I-131, contained in the fuel rods at the time of the accident. The gap fractions are obtained for high burnup fuel from Regulatory Guide 1.25 as modified by NUREG/CR-5009 to support the conservative assumption that 100 percent of the rods do not meet the burnup and kW/ft limits set forth in Footnote 11 of Regulatory Guide 1.183.

f. The pool decontamination factor is 1.0 for noble gases.

g. The effective pool decontamination factor is 170 for iodine assuming at least 23 feet of water above the top of the damaged fuel assemblies is maintained in the pool. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

h. The iodine above the fuel pool is assumed to be composed of 70 percent inorganic and 30 percent organic species.

i. The activity which escapes from the pool is assumed to be available for release to the environment over a time period of 2 hours.

j. No credit for decay or depletion during transit to the site boundary, outer boundary of the low-population zone, or in the control room is assumed.

k. No credit is taken for mixing or holdup in the fuel building atmosphere.

l. The activity released is assumed to be discharged directly to the environment with no credit for filtration or dilution.

m. For the inside the reactor building case, the containment personnel airlock doors and equipment hatch are assumed to be open at the time of the accident. For added conservatism, the gaseous effluent escaping from the refueling water pool in the reactor containment building is assumed to be released immediately to the environment through the equipment hatch without mixing in the surrounding atmosphere. For the gaseous effluent escaping through the personnel hatch, mixing with 50% of the containment volume is assumed to occur. The Auxiliary Building atmosphere is normally exhausted through filter absorbers designed to remove iodine. However, no credit is taken for iodine removal by the atmosphere filtration system filters. It is also assumed that no containment coolers or hydrogen mixing fans are operating and 100% of the activity escaping from the pool to the containment building is released to the environment over a two-hour period following the accident.

The following assumptions are made regarding the release pathway from the containment personnel hatch to the Control Room when considering a Fuel Handling accident in the Reactor building:

n. Containment is not pressurized during a Fuel Handling accident; therefore, the driving force for activity is assumed to be the Auxiliary building HVAC in normal and emergency modes. The HVAC exhausting to the environment from the Auxiliary building would bring an equivalent amount of flow from the Containment to the Auxiliary building, which would then be available for leakage into the Control Room equipment room and Control Room.

o. Wind pressurization from the open equipment hatch is assumed to drive penetration inleakage to the Control Room Envelope. A wind speed bounding 95% of that seen at the site was assumed for the determination of this parameter.

15.7.4.5.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A.

b. The atmospheric dispersion factors are calculated, based on the onsite meteorological measurements programs described in Section 2.3 and are provided in Table 15A-2.

c. The TEDE doses to a receptor located at the exclusion area boundary, outer boundary of the low population zone, and in the control room are described in Appendix 15A.

d. Bernoulli's theorem is used in conjunction with Assumption n to determine the Control Room envelope penetration pressurization considered in the analysis.

e. The methodology outlined in NAA-SR-10100, "Conventional Buildings for Reactor Containment," was utilized in order to calculate the unfiltered inleakage through the considered Control Room envelope penetrations.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to the postulated fuel-handling accident in the fuel building, the resultant activity is conservatively assumed to be released to the environment during the 0-2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available. For evaluating the radiological consequences due to the postulated fuel handling accident in the reactor building, both the containment personnel airlock doors and equipment hatch are assumed to be open at the time of the accident. Therefore, to ensure that the most conservative release pathway is considered, two cases are analyzed. The first case considers a release that is entirely exhausted to the environment via the open equipment hatch. The second case considers a release pathway through the personnel hatch to auxiliary building, then from the Auxiliary building to the Control Room equipment room, and finally to the Control room itself.

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel-handling accident result from assumptions made involving the amount of fission product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption since, transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

b. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. It has been conservatively assumed that the core has been operating at 100 percent for the entire burnup period.

c. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the fuel storage and refueling water pools. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of activity greater than anticipated.

The gaseous effluent escaping from the refueling water pool in the reactor containment building is assumed to be released immediately to the environment through the equipment hatch without mixing in the surrounding atmosphere. For the gaseous effluent escaping through the personnel hatch, mixing with 50% of the containment volume is assumed to occur. Radioactive material is assumed to be released from the refueling pool or from the spent fuel pool over a two-hour time period. The fuel building atmosphere filtration system has absorbers which normally filter any exhaust; however, credit for this system is not taken.

d. The containment purge exhaust system has charcoal adsorber units which filter any containment purge release. However, no credit has been taken for its capability (90-percent efficiency, minimum) since these units are not specifically designed to seismic Category I criteria. It is expected that for any event which would produce a catastrophic failure of the charcoal adsorber unit to the extent that its filtering capability would be negated would also result in the purge exhaust fan becoming inoperable. Therefore, failure within the purge exhaust system would terminate any high volume release from the containment. In fact, the purge exhaust fan is considerably more likely to be inoperable following any postulated event than the failure of a passive charcoal adsorber unit. Thus, although no credit in the analysis has been given for the normal purge exhaust filters, any release prior to containment isolation would be filtered.

e. There is also conservatism in the time to first fuel transfer. Despite the fact that fuel could be transferred at 76 hours, it is probable that fuel handling will begin sometime later.

f. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.7.4.5.2.1 Filter Loadings

The only ESF filtration systems considered in the analysis which limits the consequences of a fuel-handling accident is the control room filtration system.

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident, Section 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel-handling accident releases.

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary, Low-Population Zone Outer Boundary, and in the Control Room

The potential radiological consequences resulting from the occurrence of a postulated fuel-handling accident occurring in the fuel building and in the reactor building have been conservatively analyzed, using assumptions and models described in previous sections. The TEDE doses have been analyzed for the 0-2-hour dose at the exclusion area boundary, for the duration of the accident (0 to 2 hours) at the low-population zone outer boundary, and for 30 days in the control room. The results are listed in Table 15.7-8. The resultant doses are well within the guideline values of 10 CFR 50.67.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

Since the cask handling crane is designed, fabricated, and installed as single failure proof in accordance with NUREG 0554 and NUREG 0612, the cask drop accidents associated with the cask handling crane main hoist are not applicable. Therefore, no cask rupture will occur and thus no radioactivity will be released since drops of a spent fuel cask when rigged from the cask handling crane main hoist are precluded, provided that the intervening rigging equipment and attachment points to the cask are also single failure proof per the requirements of ANSI N14.6. However, during ISFSI transfer operations, it is postulated that the transfer cask is dropped off the transfer trailer while in the horizontal configuration. As described in the NUHOMS® EOS System UFSAR, Docket 72-1042, this event results in no radioactive release since the dry shielded canister contained within the transfer cask remains leak tight during this event. Refer to Section 9.1.4 for a description of the spent fuel shipping procedures.

TABLE 15.7-1

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.24 "ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.24 Position	Design
1. The assumptions related to the release of radioactive gases from the postulated failure of a gaseous waste storage tank are:	
a. The reactor has been operating at full power with one percent defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the primary cooling system and transferred to the gas decay tank that is assumed to fail.	1.a Complies.
b. The maximum content of the decay tank assumed to fail should be used for the purpose of computing the noble gas inventory in the tank. Radiological decay may be taken into account in the computation only for the minimum time period required to transfer the gases from the primary system to the decay tank.	1.b Complies.
c. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the building. The assumption of the release of the noble gas inventory from only a single tank is based on the premise that all gas decay tanks will be isolated from each other whenever they are in use.	1.c Complies.

TABLE 15.7-1 (Sheet 2)

Regulatory Guide 1.24 Position	Design
d. All of the noble gases are assumed to leak out of the building at ground level over a 2-hour time period.	1.d Complies.
2. The atmospheric diffusion assumptions for ground level releases are:	2. Short-term atmospheric dispersion factors corresponding to a ground level release and accident conditions were calculated based on on-site meteorological measurement programs described in Section 2.3. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.145 and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.
a. The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma^z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3) Q = amount of material released (curies/sec) u = windspeed (meters/sec) σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	

TABLE 15.7-1 (Sheet 3)

Regulatory Guide 1.24 Position	Design
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	
b. For ground level releases, atmospheric diffusion factors[1] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions: (1) windspeed of 1 meter/sec; (2) uniform wind direction (3) Pasquill diffusion category F.	
c. Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a. above under the meteorological conditions given in regulatory position 2.b. above.	
3. The following assumptions and equations may be used to obtain conservative approximations of external whole body dose from radioactive clouds:	3. Analyses followed Regulatory Guide 1.183 for calculating total effective dose equivalent doses.
a. External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel.	

TABLE 15.7-1 (Sheet 4)

Regulatory Guide 1.24 Position	Design
The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance.	
For an infinite uniform cloud containing curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[2]	
$\beta^{D'\infty} = 0.457\ \overline{E}_{\beta}\chi$	
Where:	
$\beta^{D'\infty}$ = beta dose rate from an infinite cloud (rad/sec)	
$\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis)	
χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)	
Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or:	
$\beta^{D'\infty} = 0.23\ \overline{E}_{\beta}\chi$	
For gamma emitting material the dose rate in air at the cloud center is:	
$\gamma^{D'\infty} = 0.507\overline{E}_{\gamma}\chi$	

TABLE 15.7-1 (Sheet 5)

Regulatory Guide 1.24 Position	Design
Where: $\gamma^{D'\infty}$ = gamma dose rate from an infinite cloud (rad/sec) $\overline{E}_\gamma$ = average gamma energy per disintegration (Mev/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $\gamma^{D'} = 0.25\ \overline{E}_\gamma \chi$ Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $\beta^{D\infty} = 0.23\ \overline{E}_\beta \Psi$ or $\gamma^{D\infty} = 0.25\ \overline{E}_\gamma \Psi$ Where is the concentration time integral for the cloud (curie sec/m3).	
b. The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,3 are averaged and used according to the methods described in ICRP Publication 2.	

TABLE 15.7-1 (Sheet 6)

Regulatory Guide 1.24 Position	Design

[1]These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to insure a conservative estimate of potential offsite exposures.

[2]Meteorology and Atomic Energy - 1968, Chapter 7.

[3]C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc., 1967).

WOLF CREEK

TABLE 15.7-2

Not Used

TABLE 15.7-3

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

I. Source Data

a.	Core power level, MWt	3637 (includes uncertainty)
b.	Failed fuel, percent	1

II. Atmospheric Dispersion Factors — See Table 15A-2

III. Activity Release Data

Isotope	0-2 hr (Ci)
I-131	3.99E+0
I-132	2.30E+0
I-133	4.28E+0
I-134	4.96E-1
I-135	1.94E+0
Xe-131m	1.07E+3
Xe-133m	1.27E+3
Xe-133	8.12E+4
Xe-135m	5.97E+1
Xe-135	1.02E+3
Xe-138	4.06E+0
Kr-85m	1.49E+2
Kr-85	5.52E+3
Kr-87	3.00E+1
Kr-88	1.79E+2

TABLE 15.7-4

RADIOLOGICAL CONSEQUENCES OF A
WASTE GAS DECAY TANK RUPTURE

	Doses (rem TEDE)
Exclusion Area Boundary (0-2 hr)	9.0E-2
Low Population Zone Outer Boundary (duration)	2.9E-2
Control Room (30 Days)	5.7E-2

TABLE 15.7-5

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LIQUID RADWASTE TANK FAILURE

I. Source Data

a.	Core power level, MWt	3637 (Includes Uncertainty)
b.	Failed fuel, percent	1

II. Atmospheric Dispersion Factors — See Table 15A-2

III. Activity Release Data

a.	Noble gas activity, percent of tank contents	100
b.	Iodine gas activity, percent of tank contents	
	1. Boron recycle holdup tank	100
	2. Hypothetical Liquid Waste tank	100
c.	Activity released to the environment	
	1. Boron recycle holdup tank	

Isotope	0-2 hr (Ci)
I-130	6.36E-3
I-131	5.83E+0
I-132	8.52E-2
I-133	1.15E+0
I-134	7.04E-3
I-135	2.07E-1
Xe-131m	3.67E+2
Xe-133m	1.44E+2
Xe-133	1.79E+4
Xe-135m	1.09E-1
Xe-135	3.64E+1
Xe-138	8.74E-2
Kr-85m	4.00E+0
Kr-85	1.69E+3
Kr-87	7.34E-1
Kr-88	4.49E+0

TABLE 15.7-5 (Sheet 2)

2. Hypothetical liquid waste tank

Isotope	0-2 hr (Ci)
I-130	2.75E-3
I-131	2.75E+1
I-132	7.10E-3
I-133	8.07E-1
I-134	2.26E-4
I-135	4.88E-2
Xe-131m	0.000E+00
Xe-133m	0.000E+00
Xe-133	0.000E+00
Xe-135m	0.000E+00
Xe-135	0.000E+00
Xe-138	0.000E+00
Kr-85m	0.000E+00
Kr-85	0.000E+00
Kr-87	0.000E+00
Kr-88	0.000E+00

TABLE 15.7-6

RADIOLOGICAL CONSEQUENCES OF A
LIQUID RADWASTE TANK FAILURE

	Dose (rem TEDE)
Boron Recycle Tank	
Exclusion Area Boundary (0-2 hr)	2.5E-2
Low Population Zone Outer Boundary (duration)	8.03E-3
Control Room (30 days)	5.3E-2
Hypothetical Liquid Waste Tank	
Exclusion Area Boundary (0-2 hr)	4.5E-2
Low Population Zone Outer Boundary (duration)	1.5E-2
Control Room (30 days)	2.3E-1

TABLE 15.7-7

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF A
FUEL-HANDLING ACCIDENT*

I.	Source Data	
	a. Core power level, MWt	3637 (includes uncertainty)
	b. Radial peaking factor	1.65
	c. Decay time, hours	76
	d. Number of fuel assemblies affected	1.2
	e. Fraction of fission product gases contained in the gap region of the fuel assembly	Per R.G. 1.25 and NUREG/CR-5009
II.	Atmospheric Dispersion Factors	See Table 15A-2
III.	Activity Release Data	
	a. Percent of affected fuel assemblies gap activity released	100
	b. Pool decontamination factors	
	1. Iodine	170
	2. Noble gas	1
	c. Building mixing volumes assumed, percent of total volume	50**
	d. Activity release period, hrs	2

*Note: The parameter values are consistent to bound the release pathway from both the fuel building and reactor building unless otherwise clarified.

**Note: This parameter applies to the containment FHA to personnel hatch to Aux Building case. The mixing volume assumed for the containment FHA release through equipment hatch case is 0%.

TABLE 15.7-7 (Sheet 2)

IV.	Control Room	
	a. Control Room Switchover to Emergency Mode, sec	See Table 15A-5
V.	FHA in Reactor Building, Aux Building Release Pathway Case Parameters	
	a. Volume of Auxiliary Building Credited, ft^3	70000
	b. Volume of Control Room Equipment Room Credited, ft^3	30000
	c. HVAC Exhaust Flow Rate from Containment to Auxiliary Building, cfm	6750
	d. Unfiltered inleakage from the Auxiliary Building to the Equipment Room prior to a CRVIS, cfm	100
	e. Unfiltered inleakage from the Auxiliary Building to the Equipment Room following a CRVIS, cfm	10
	f. Unfiltered inleakage from the Equipment Room to the Control Room prior to a CRVIS, cfm	100
	g. Unfiltered inleakage from the Equipment Room to the Control Room following a CRVIS. Cfm	50

TABLE 15.7-8

RADIOLOGICAL CONSEQUENCES OF A
FUEL HANDLING ACCIDENT

	Dose (rem TEDE)
Exclusion Area Boundary (0-2 hr)	1.2E+0
Low Population Zone Outer Boundary (duration)	3.9E-1
Control Room (30 days)	1.1E+0
Auxiliary Building Release	
Control Room (30 days)	3.1E+0

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An ATWS is an anticipated operational occurrence (such as loss of normal feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of AC power, and loss of condenser vacuum) that is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor. The most severe ATWS scenarios are those in which there was a complete loss of normal feedwater and the loss of load. The primary safety concern for these transients is the potential for high pressures within the RCS. Based on analyses, acceptable consequences would result, that is, RCS pressures less than the pressure (3200 psig) corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criteria, provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. Normally, these features would be actuated by the RPS. However, if a common mode failure in the RPS incapacitates AFW flow initiation and/or turbine trip (in addition to prohibiting a scram), then an alternate method of providing AFW flow and turbine trip is required. These two functions are provided by AMSAC, which is required by the final ATWS Rule in 10CFR50.62 (c)(1) for Westinghouse PWRs, as discussed below.

The ATWS Rule (10CFR50.62(b)) requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the probability of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event. 10CFR50.62 "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," was effective July 26, 1984. Paragraph (c)(6) of the ATWS Rule requires that detailed information to demonstrate compliance with the requirements be submitted to the Director, Office of Nuclear Reactor Regulation (NRR). In accordance with paragraph (c)(6) of the ATWS Rule, the Operating Agent provided information to the NRC by letter, dated March 20, 1987. The letter forwarded the detailed description of the ATWS Mitigation System Actuation Circuitry (AMSAC) proposed for installation at the Wolf Creek Generating Station (WCGS). Supplemental information on the AMSAC design was also submitted to the NRC in letters dated April 16, 1987 and October 5, 1987, for review. WCGS installed equipment (AMSAC) from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS.

On December 16, 1987, the NRC notified the Operating Agent by letter that the staff had completed its review of the submittal and concluded that the AMSAC design for the WCGS was acceptable and in compliance with the ATWS Rule, 10CFR50.62.

APPENDIX 15A

ACCIDENT ANALYSIS RADIOLOGICAL CONSEQUENCES EVALUATION MODELS AND PARAMETERS

15A.1 GENERAL ACCIDENT PARAMETERS

This section contains the parameters used in analyzing the radiological consequences of postulated accidents. Table 15A-1 contains the general parameters used in all the accident analyses. For parameters specific only to particular accidents, refer to that accident parameter section. The site specific, ground-level release, short-term dispersion factors (for accidents, ground-level releases are assumed) are based on Regulatory Guide 1.145 (Ref. 1) methodology and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst-sector meteorology and these are given in Table 15A-2 (see Section 2.3.4 for additional details on meteorology). The core inventories are given in Table 15A-3. The equations for dose calculations are consistent with those presented in References 4, 5, and 6 for the RADTRAD computer code. The TEDE dose factors from CEDE (via inhalation) and EDE (via external exposure) based on References 2 and 3 are given in Table 15A-4.

15A.2 OFFSITE RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

This section presents the models used for calculating the integrated activity released to the environment and the accident flow paths.

15A.2.1 ACCIDENT RELEASE PATHWAYS

The release pathways for the major accidents are given in Figure 15A-1. The accidents and their pathways are as follows:

LOCA: Following a postulated loss-of-coolant accident (LOCA), the release of radioactivity from the containment is to the environment with the containment spray and ESF systems brought into operation. The release in this case is calculated which takes into account a two-region spray model within the containment. The release of radioactivity to the environment due to assumed ESF system leakages in the auxiliary building will be conservatively calculated without credit for filtration. The release of radioactivity to the environment due to assumed ECCS boundary valves leakage through RWST is calculated. In addition, the release of radioactivity to the environment due to assumed containment mini-purge is calculated.

WGDTR: The activity release to the environment due to waste gas decay tank rupture (WGDTR) will be direct and unfiltered, with no holdup. The release pathway is A'-D. The total activity release in this case is therefore assumed to be the initial source activity itself.

FHA: The release to the environment due to a fuel handling accident (FHA) in the fuel building is via filters following the actuation of the emergency exhaust system. The release pathway is B-C-D. Since the release is calculated without any credit for holdup in the fuel building, the total release will be the unfiltered release over a two-hour period. The release of radioactivity to the environment due to FHA inside the containment is direct and unfiltered, via the A’-D pathway, and occurs over a two-hour period (actually, the release is via the non-safety graded filters).

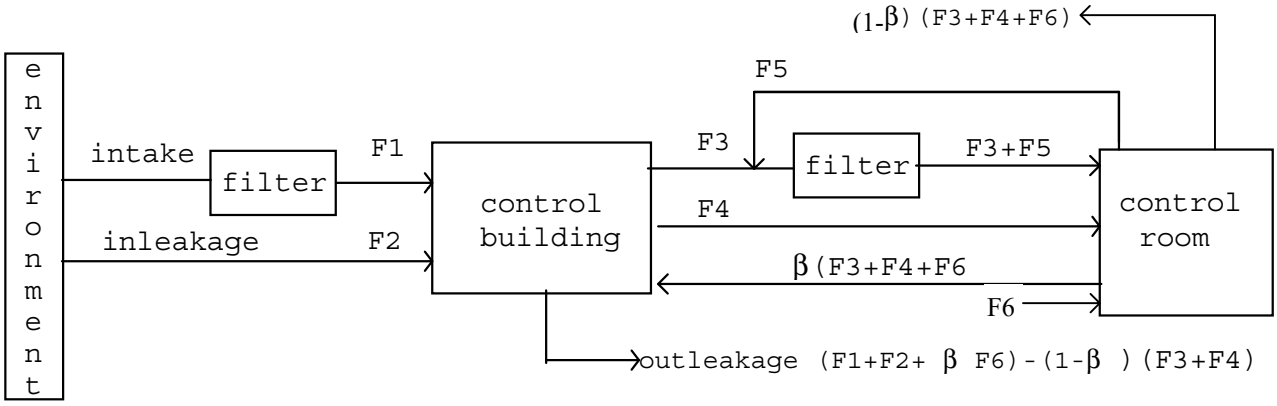
CAE: Radioactivity release from the containment to the environment due to the control assembly ejection (CAE) accident is direct and unfiltered (spray removal is not assumed). The releases from the primary system are calculated considering holdup in the primary system (excluding noble gases); the secondary (steam) releases via the relief valves are calculated with holdup in the steam generators. Releases of noble gases are direct and unfiltered. The pathways for these releases are A-D and A'-D.

MSLB, SGTR: Radioactivity releases from the affected SG to the environment due to main steam line break (MSLB) or steam generator tube rupture (SGTR) accidents are direct and unfiltered with no holdup via the A'-D pathway. The releases from the primary system into the intact steam generators are calculated considering holdup in the primary system (excluding noble gases); the secondary (steam) releases via the relief valves are calculated with holdup in the steam generators. Releases of noble gases are direct and unfiltered. The activity release calculations for these accidents are complex, involving spiking effects, time-dependent flashing fractions, and scrubbing of flashed activities; the release calculations are described in those sections that address these accidents.

Locked Rotor: The releases from the primary system into the steam generator due to locked rotor are calculated considering holdup in the primary system (excluding noble gases); the secondary (steam) releases via the relief valves are calculated with holdup in the steam generators. Releases of noble gases are direct and unfiltered.

15A.3 CONTROL ROOM RADIOLOGICAL CONSEQUENCES CALCULATIONAL MODELS

The control room ventilation is initially in normal mode until an isolation signal is received based on either a safety injection signal or control room radiation monitor. Outside air is brought into the control building and control room via normal ventilation flows with no filtration. After switchover to emergency mode ventilation, make-up air is brought into the control room via the control room filtration system which draws in air from the control building. Outside air is brought into the control building through safety grade filters via the control room pressurization fan. Some unfiltered air also may leak into the control building via an assumed inleakage rate. The activity concentrations at the control building intake for each time interval are found by multiplying the activity release to the environment by the appropriate X/Q for that time interval. The control room isolation is modeled in the dose analyses in two parts: actuation of the emergency mode filtration (in both the control building and the control room) and closure of the normal HVAC intake damper. The actuation of the emergency mode filtration occurs following receipt of an isolation signal (e.g. high radiation, safety injection or manual action). In the analyses, emergency mode filtration is actuated after a delay of at least 60 seconds to account for instrumentation delays and damper movement following an automatic isolation signal. Closure of the normal HVAC intake damper occurs on a safety injection signal. It would also occur on manual action but this is not credited in the analyses. Prior to the closure of the normal HVAC intake damper, the unfiltered inleakage is associated with the normal HVAC intake X/Q. After closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake and the communications corridor intake (associated with ingress/egress). The X/Qs associated with the communications corridor that are modeled in the analyses are presented in Table 15A-2. The flow path model is shown below.



Once activity is brought into the control building, mixing within the control building is afforded by the control room pressurization fan. The control room filtration system fan takes air from the control building and the control room (recirculation) and discharges to the control room through the control room filtration safety grade filters. The radiological analysis input parameters are provided in Table 15A-1.

The control room ventilation isolation signal (CRVIS) starts both trains of the control room pressurization system and the control room filtration system. For the determination of dose to control room personnel, the worst single failure has been ascertained to be the failure of the filtration fan in one of the two filtration system trains.

Prior to operator action, a potential pathway would exist allowing air from the control building to enter the control room, bypassing the control room filtration filters. Operator action is required to ensure no bypass pathways then exist for unfiltered air to enter the control room.

Owning to this single failure of the control room filtration fan, the assumed failure of one of the two containment spray (CS) trains, and two of the four hydrogen mixing subsystem fans, inherent in the LOCA analysis parameters given in Table 15.6-6 should not be applied in this analysis. With both trains of CS and four hydrogen mixing fans operating, more volumetric coverage of the containment spray and more mixing between the new sprayed and unsprayed regions would be expected, thereby giving much greater removal of iodine and particulates within the containment atmosphere.

15A.3.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room

This dose is calculated based on the semi-infinite cloud model which is modified using protection factors to account for the control room walls.

15A.4 REFERENCES

1. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," August 1979.

2. Eckerman, K.F., et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, September 1988.

3. Eckerman, K.F. and Ryman, J.C., "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, September 1993.

4. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Humphreys, S.L., et al., December 1997.

5. Supplement 1 to NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Bixler, N.E. and Erickson, C.M., June 1999.

6. Supplement 2 to NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport And Removal And Dose Estimation," Arcieri, W.C., October 2002.

TABLE 15A-1

PARAMETERS USED IN ACCIDENT ANALYSIS

I.	General	
	1. Core power level, Mwt	3637 (includes uncertainty)
	2. Full-power operation, days per cycle	511
	3. Number of fuel assemblies in the core	193
	4. Maximum radial peaking factor	1.65
	5. Percentage of failed fuel	1.0
	6. Steam generator tube leak, lb/hr	500
II.	Sources	
	1. Core inventories, Ci	Table 15A-3
	2. Primary coolant specific activities, Ci/gm	Table 11.1-5
	3. Primary coolant activity, technical specification limit for noble gas - Xe-133 dose equivalent, microCi/gm	500
	4. Primary coolant activity, technical specification limit for iodines - I-131 dose equivalent, µCi/gm	1.0
	5. Secondary coolant activity technical specification limit for iodines - I-131 dose equivalent, µCi/gm	0.1
III.	Activity Release Parameters	
	1. Free volume of containment, ft^3	2.5×10^6
	2. Containment leak rate	
	i. 0-24 hours, % per day	0.2
	ii. after 24 hrs, % per day	0.1
IV.	Control Room Dose Analysis (for LOCA)	
	1.Control building	
	i. Mixing volume, cfm	239,000
	ii. Emergency mode filtered intake, cfm	
	Prior to operator action (0-1.5 hours)	1350
	After operator action (1.5 hours-720 hours)	675
	iii. Emergency mode unfiltered inleakage, cfm	400
	iv. Emergency mode filter efficiency (all forms of iodine), %	95
	v. Normal mode unfiltered intake, cfm	14,360

TABLE 15A-1 (Sheet 2)

2. Control room	
i. Volume, cf	100,000
ii. Emergency mode filtered flow from control building, cfm	550(*)
iii. Emergency mode unfiltered flow from control building, cfm	
Prior to operator action	550(*)
(0-1.5 hours) After operator action (1.5 hours - 720 hours)	0
iv. Emergency mode filtered recirculation, cfm	1250
v. Emergency mode filter efficiency (all forms of iodine), %	95
vi. Normal mode unfiltered intake, cfm	2195
vii. Normal mode unfiltered inleakage, cfm	50
3. Control Room Radiation Monitor Setpoint, microCi/cc Xe-133	2.12E-3

(*)NOTE: Flows possible per train with two trains in operation. Each train is balanced for 400 cfm.

V. Miscellaneous

1.Atmospheric dispersion factors, χ/Q sec/m^3	Table 15A-2
2.Dose conversion factors	Table 15A-4
3.Breathing rates, $meter^3$/sec	
i. control room at all times	3.5×10^{-4}
ii. offsite	
0-8 hrs	3.5×10^{-4}
8-24 hrs	1.8×10^{-4}
24-720 hrs	2.3×10^{-4}
4.Control room occupancy fractions	
0-24 hrs	1.0
24-96 hrs	0.6
96-720 hrs	0.4
5.Control Room Isolation Modeling	Table 15A-5

TABLE 15A-2

LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS
(χ/Qs) FOR ACCIDENT ANALYSIS
(sec/meter^3)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)
Site boundary	
0-720	1.40E-4
Low-population zone	
0-2	4.50E-5
2-8	2.39E-5
8-24	1.29E-5
24-96	5.49E-6
96-720	1.61E-6
Control room	
Normal Mode Ventilation	
Equipment Hatch (FHA in Containment)	
0 – 2	1.38E-3
2 – 8	1.14E-3
8 – 24	4.22E-4
24 – 96	3.48E-4
96 – 720	2.67E-4
Unit Vent Exhaust (LOCA Containment Leakage, Mini-Purge, & ECCS Leakage, CAE Containment Leakage, MSLB Faulted SG, Letdown Line Break, FHA in Fuel Building)	
0 – 2	2.11E-3
2 – 8	1.66E-3
8 – 24	6.67E-4
24 – 96	5.15E-4
96 – 720	4.20E-4
RWST Vent (LOCA RWST Backleakage)	
0 – 2	1.03E-3
2 – 8	8.70E-4
8 – 24	3.34E-4
24 – 96	2.87E-4
96 – 720	2.09E-4

TABLE 15A-2 (Sheet 2)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)
MSSV (MSLB Intact SGs, Loss of AC Power, Locked Rotor, CAE Primary-to-Secondary Leakage, SGTR)	
0 – 2	2.55E-2
2 – 8	1.84E-2
8 – 24	7.46E-3
24 – 96	4.72E-3
96 – 720	3.43E-3
Radwaste Building (WGDTR, Liquid Waste Tank Failure)	
0 – 2	7.17E-4
2 – 8	6.00E-4
8 – 24	2.32E-4
24 – 96	1.97E-4
96 – 720	1.44E-4
Emergency Mode Ventilation	
Equipment Hatch (FHA in Containment)	
0 – 2	5.44E-4
2 – 8	4.35E-4
8 – 24	1.62E-4
24 – 96	1.22E-4
96 – 720	8.70E-5
Unit Vent Exhaust (LOCA Containment Leakage, Mini-Purge, & ECCS Leakage, CAE Containment Leakage, MSLB Faulted SG, Letdown Line Break, FHA in Fuel Building)	
0 – 2	6.12E-4
2 – 8	4.38E-4
8 – 24	1.79E-4
24 – 96	1.14E-4
96 – 720	8.94E-5
RWST Vent (LOCA RWST Backleakage)	
0 – 2	6.80E-4
2 – 8	6.19E-4
8 – 24	2.27E-4
24 – 96	1.96E-4
96 – 720	1.53E-4

TABLE 15A-2 (Sheet 3)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)
MSSV (MSLB Intact SGs, Loss of AC Power, Locked Rotor, CAE Primary-to-Secondary Leakage)	
0 – 2	1.04E-3
2 – 8	7.46E-4
8 – 24	3.03E-4
24 – 96	1.90E-4
96 – 720	1.39E-4
Radwaste Building (WGDTR, Liquid Waste Tank Failure)	
0 – 2	6.92E-4
2 – 8	6.06E-4
8 – 24	2.39E-4
24 – 96	2.00E-4
96 – 720	1.47E-4
Reactor Building Wall (LOCA Containment Leakage, CAE Containment Leakage)	
0 – 2	6.02E-4
2 – 8	4.35E-4
8 – 24	1.81E-4
24 – 96	1.29E-4
96 – 720	9.65E-5
Communication Corridor*	
Unit Vent Exhaust	
0 – 2	1.32E-3
2 – 8	1.02E-3
8 – 24	4.08E-4
24 – 96	2.99E-4
96 – 720	2.37E-4
MSSVs	
0 – 2	5.10E-3
2 – 8	3.77E-3
8 – 24	1.43E-3
24 – 96	1.00E-3
96 – 720	7.34E-4

TABLE 15A-2 (Sheet 4)

Location Type/ Time Interval (hrs)	χ/Q (Sec/Meters^3)
RWST	
0 – 2	7.90E-4
2 – 8	6.75E-4
8 – 24	2.59E-4
24 – 96	2.20E-4
96 – 720	1.67E-4

*: The Unit Vent Exhaust χ/Qs are used for the LOCA and Rod Ejection Containment Leakage releases, LOCA Containment Purge, LOCA ECCS leakage to the auxiliary building releases, and MSLB faulted SG releases. The MSSV χ/Qs are used for the MSLB intact SG releases and the SGTR ruptured and intact SG releases. The RWST χ/Qs are used for the LOCA RWST back-leakage release. The other analyses do not credit closure of the normal HVAC intake damper.

Operator Transit

Time	Point 1	Point 2	Point 3	Point 4	Point 5	Point 6	Point 7	Point 8	Point 9
Containment Leakage (Maximum of Unit Vent, Equipment Hatch, and Reactor Building Wall)									
0-2 hours	1.80E-03	9.36E-04	6.50E-04	4.22E-04	2.62E-04	1.99E-04	1.44E-04	1.08E-04	7.91E-05
2-8 hours	1.49E-03	6.28E-04	4.81E-04	2.72E-04	1.52E-04	1.17E-04	7.51E-05	7.14E-05	5.61E-05
8-24 hours	5.73E-04	2.30E-04	1.84E-04	9.75E-05	5.42E-05	4.22E-05	2.88E-05	2.49E-05	2.12E-05
24-96 hours	4.90E-04	1.68E-04	1.33E-04	7.05E-05	3.65E-05	2.82E-05	1.88E-05	1.82E-05	1.52E-05
96-720 hours	3.62E-04	1.29E-04	1.06E-04	5.12E-05	2.89E-05	2.19E-05	1.45E-05	1.33E-05	1.20E-05
ECCS Leakage (Unit Vent)									
0-2 hours	8.94E-04	4.96E-04	4.16E-04	2.89E-04	1.98E-04	1.58E-04	1.20E-04	9.40E-05	7.15E-05
2-8 hours	6.86E-04	3.02E-04	2.87E-04	1.76E-04	1.13E-04	9.19E-05	6.76E-05	6.11E-05	4.97E-05
8-24 hours	2.68E-04	1.10E-04	1.01E-04	6.42E-05	4.27E-05	3.40E-05	2.55E-05	2.15E-05	1.82E-05
24-96 hours	2.11E-04	7.03E-05	7.32E-05	4.14E-05	2.66E-05	2.12E-05	1.59E-05	1.46E-05	1.32E-05
96-720 hours	1.74E-04	5.08E-05	5.62E-05	2.97E-05	1.98E-05	1.59E-05	1.21E-05	1.07E-05	1.04E-05
RWST Back-Leakage (RWST)									
0-2 hours	5.15E-04	3.16E-04	2.54E-04	1.99E-04	1.49E-04	1.18E-04	9.56E-05	7.57E-05	5.88E-05
2-8 hours	4.05E-04	2.10E-04	1.85E-04	1.27E-04	9.02E-05	7.48E-05	5.42E-05	5.01E-05	4.05E-05
8-24 hours	1.55E-04	7.71E-05	7.00E-05	4.61E-05	3.18E-05	2.65E-05	1.93E-05	1.74E-05	1.52E-05
24-96 hours	1.24E-04	5.65E-05	5.01E-05	3.31E-05	2.21E-05	1.80E-05	1.30E-05	1.27E-05	1.08E-05
96-720 hours	9.25E-05	4.37E-05	4.00E-05	2.47E-05	1.66E-05	1.37E-05	1.01E-05	9.60E-06	8.71E-06

TABLE 15A-3

Core Inventory (Ci)

Isotope	Core Activity (Ci)	Isotope	Core Activity (Ci)
Kr-85m	2.69E+07	Sr-92	1.33E+08
Kr-85	1.10E+06	Ba-139	1.87E+08
Kr-87	5.30E+07	Ba-140	1.79E+08
Kr-88	7.12E+07	Ru-103	1.56E+08
Xe-131m	1.05E+06	Ru-105	1.08E+08
Xe-133m	6.06E+06	Ru-106	4.79E+07
Xe-133	2.01E+08	Rh-105	1.00E+08
Xe-135m	4.39E+07	Mo-99	1.91E+08
Xe-135	4.06E+07	Tc-99m	1.69E+08
Xe-138	1.80E+08	Ce-141	1.69E+08
I-130	1.98E+06	Ce-143	1.59E+08
I-131	1.01E+08	Ce-144	1.30E+08
I-132	1.49E+08	Pu-238	2.10E+05
I-133	2.10E+08	Pu-239	2.70E+04
I-134	2.36E+08	Pu-240	4.20E+04
I-135	2.00E+08	Pu-241	1.06E+07
Cs-134	1.65E+07	Np-239	1.94E+09
Cs-136	3.95E+06	Y-90	8.88E+06
Cs-137	1.11E+07	Y-91	1.30E+08
Cs-138	1.96E+08	Y-92	1.35E+08
Rb-86	1.86E+05	Y-93	1.52E+08
Te-127m	1.49E+06	Nb-95	1.76E+08
Te-127	9.09E+06	Zr-95	1.74E+08
Te-129m	5.04E+06	Zr-97	1.75E+08
Te-129	2.66E+07	La-140	1.85E+08
Te-131m	1.99E+07	La-142	1.64E+08
Te-132	1.45E+08	Nd-147	6.57E+07
Sb-127	9.23E+06	Pr-143	1.55E+08
Sb-129	2.85E+07	Am-241	9.94E+03
Sr-89	9.98E+07	Cm-242	2.82E+06
Sr-90	8.51E+06	Cm-244	2.11E+05
Sr-91	1.25E+08		

TABLE 15A-4

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	CEDE (Sv/Bq)	EDE (Sv-m^3/Bq-sec)
Kr-85m	NA	7.48E-15
Kr-85	NA	1.19E-16
Kr-87	NA	4.12E-14
Kr-88	NA	1.02E-13
Xe-131m	NA	3.89E-16
Xe-133m	NA	1.37E-15
Xe-133	NA	1.56E-15
Xe-135m	NA	2.04E-14
Xe-135	NA	1.19E-14
Xe-138	NA	5.77E-14
I-130	7.14E-10	1.04E-13
I-131	8.89E-9	1.82E-14
I-132	1.03E-10	1.12E-13
I-133	1.58E-9	2.94E-14
I-134	3.55E-11	1.30E-13
I-135	3.32E-10	7.98E-14
Cs-134	1.25E-8	7.57E-14
Cs-136	1.98E-9	1.06E-13
Cs-137	8.63E-9	2.88E-14
Cs-138	2.74E-11	1.21E-13
Rb-86	1.79E-9	4.81E-15
Te-127m	5.81E-9	1.47E-16
Te-127	8.60E-11	2.42E-16
Te-129m	6.47E-9	1.55E-15
Te-129	2.42E-11	2.75E-15
Te-131m	1.73E-9	7.01E-14
Te-132	2.55E-9	1.03E-14
Sb-127	1.63E-9	3.33E-14
Sb-129	1.74E-10	7.14E-14
Sr-89	1.12E-8	7.73E-17
Sr-90	3.51E-7	7.53E-18
Sr-91	4.49E-10	3.45E-14
Sr-92	2.18E-10	6.79E-14
Ba-139	4.64E-11	2.17E-15
Ba-140	1.01E-9	8.58E-15

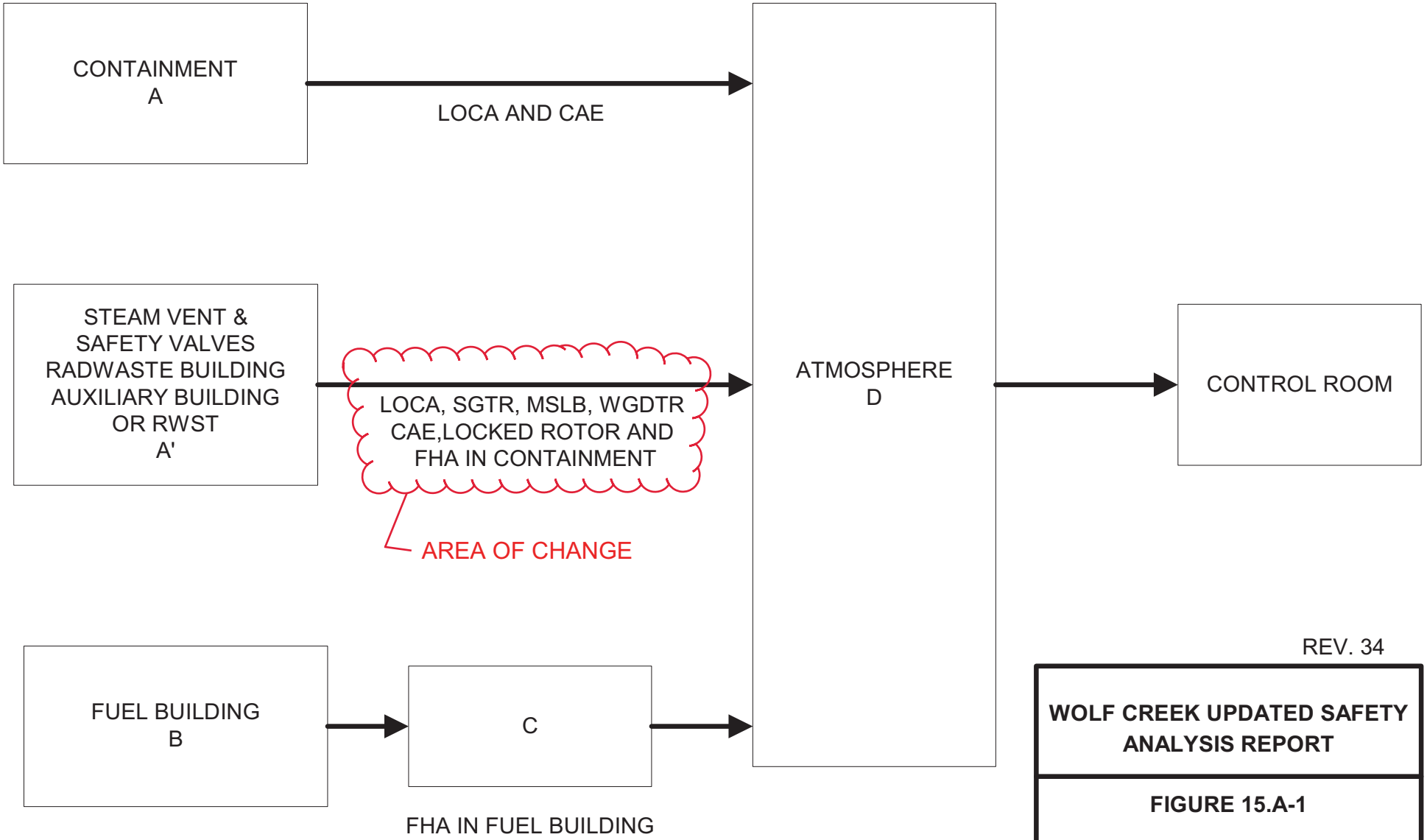
TABLE 15A-4 (Sheet 2)

Nuclide	CEDE (Sv/Bq)	EDE (Sv-m^3/Bq-sec)
Ru-103	2.42E-9	2.25E-14
Ru-105	1.23E-10	3.81E-14
Ru-106	1.29E-7	0
Rh-105	2.58E-10	3.72E-15
Mo-99	1.07E-9	7.28E-15
Tc-99m	8.80E-12	5.89E-15
Ce-141	2.42E-9	3.43E-15
Ce-143	9.16E-10	1.29E-14
Ce-144	1.01E-7	8.53E-16
Pu-238	1.06E-4	4.88E-18
Pu-239	1.16E-4	4.24E-18
Pu-240	1.16E-4	4.75E-18
Pu-241	2.23E-6	7.25E-20
Np-239	6.78E-10	7.69E-15
Y-90	2.28E-9	1.90E-16
Y-91	1.32E-8	2.60E-16
Y-92	2.11E-10	1.30E-14
Y-93	5.82E-10	4.80E-15
Nb-95	1.57E-9	3.74E-14
Zr-95	6.39E-9	3.60E-14
Zr-97	1.17E-9	9.02E-15
La-140	1.31E-9	1.17E-13
La-142	6.84E-11	1.44E-13
Nd-147	1.85E-9	6.19E-15
Pr-143	2.19E-9	2.10E-17
Am-241	1.20E-4	8.18E-16
Cm-242	4.67E-6	5.69E-18
Cm-244	6.70E-5	4.91E-18

TABLE 15A-5

CONTROL ROOM ISOLATION MODELING

Event/Scenario	High Radiation Signal, Time from event initiation	SI Signal Generation, Time from event initiation	Emergency Mode Actuation Credited	Normal HVAC Intake Damper Closure
Main Steamline Break, both iodine spikes	N/A	Immediate	120 seconds	120 seconds
Loss of AC Power, both iodine spikes	N/A	N/A	N/A	N/A
Locked Rotor	Immediate	N/A	120 seconds	N/A
Control Rod Ejection - Containment Leakage	N/A	<150 seconds	210 seconds	210 seconds
Control Rod Ejection - Primary to Secondary Leakage	Immediate	N/A	120 seconds	N/A
Letdown Line Break, both iodine spikes	Immediate	N/A	120 seconds	N/A
SGTR, both iodine spikes	Immediate	325 seconds	60 seconds	600 seconds
LOCA	N/A	Immediate	120 seconds	120 seconds
Tank Ruptures	N/A	N/A	N/A	N/A
Fuel Handling Accident	Immediate	N/A	120 seconds	N/A
Fuel Handling Accident - Auxiliary Building Releases	N/A	N/A	30 minutes (operator action)	N/A



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WOLF CREEK UPDATED SAFETY ANALYSIS REPORT

FIGURE 15.A-1

RELEASE PATHWAYS

APPENDIX 15B

Regulatory Guide 1.183, Revision 0 Conformance Tables

NOTE: In Tables 15B-1 to 15B-7, the text shown in the "Regulatory Guide 1.183 Position" columns is taken from Regulatory Guide 1.183, Revision 0. Therefore, references to footnotes, tables, and numbered references may be found in the regulatory guide.

TABLE 15B-1 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 MAIN SECTIONS

RG Section	RG Position	Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Reference 17) or ORIGEN-ARP (Reference 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	The inventory of fission products in the reactor core and available for release to the containment was based on the maximum full power operation with a core thermal power of 3637 MWt (102% of 3565 MWt nominal power). Core design parameters (enrichment, burnup, and MTU loading) are based on the cycle 19 core design. Margin is added to the EOC core inventory, calculated with ORIGEN-S, to account for potential core design differences in future cycles. The magnitude of this margin is based on sensitivity studies that consider variations in enrichment and burnup.
	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	For the DBA LOCA, all fuel assemblies were assumed to be affected and the core average inventory was used. A peaking factor of 1.65 was used for DBA events that do not involve the entire core (fuel handling accident, rod ejection, locked rotor), with fission product inventories for damages fuel rods determined by multiplying the total core inventory by the fraction of damaged rods.
	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power were made in any analysis. For the fuel handling accident, 76 hours of radioactive decay after shutdown was modeled.

TABLE 15B-1 (Sheet 2)

RG Section	RG Position	Analysis	Comments
3.2	The core inventory release fractions[10], by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 2** **PWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.35 0.4 Alkali Metals 0.05 0.25 0.3 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002	Conforms	For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized.

TABLE 15B-1 (Sheet 3)

RG Section	RG Position	Analysis	Comments
3.2 (cont.)	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3[11]** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12	Conforms	For non-LOCA events, the fraction of the core inventory assumed to be in the gap by radionuclide group in Table 3 were utilized in conjunction with the maximum core radial peaking factor of 1.65. The control rod ejection accident was evaluated per Footnote 11 of RG 1.183 (the gap fractions are assumed to be 10% for iodines and noble gases). To account for possible damage to an assembly with high burnup and rod power and to address Footnote 11, the fuel handling accident used conservatively high gap fractions of 12% for I-131, 30% for Kr-85, and 10% for all other iodines and noble gases. These gap fractions were obtained from NUREG/CR-5009. NUREG/CR-5009 determined that the RG 1.25 gap fractions may be used for extended burnup fuel with the exception of I-131, which may be up to 20% higher.

TABLE 15B-1 (Sheet 4)

RG Section	RG Position	Analysis	Comments
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. **Table 4** **LOCA Release Phases PWRs BWRs** **Phase** / **Onset** / **Duration** / **Onset** / **Duration** Gap Release / 30 sec / 0.5 hr / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.3 hr / 0.5 hr / 1.5 hr	Conforms	The Table 4 PWR onset and durations for the DBA LOCA releases were utilized in the analysis. Note that the gap release was modeled beginning at 30 seconds and ending in the first half hour in order to model the early in-vessel release beginning at 0.5 hr.
	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	No additional delays in gap release were assumed for the DBA analyses.

TABLE 15B-1 (Sheet 5)

RG Section	RG Position	Analysis	Comments
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The Table 5 elements in each radionuclide group were utilized in DBA analyses. Note that since RADTRAD is limited to modeling 63 nuclides, certain nuclides which were deemed to be insignificant from a dose perspective were not included.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95% of the iodine released should be assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	For releases from the reactor coolant system (RCS) to the containment, 95% of the iodine released was assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Fission products were assumed to be in particulate form with the exception of elemental and organic iodine and noble gases.

TABLE 15B-1 (Sheet 6)

RG Section	RG Position	Analysis	Comments
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The amount of fuel damage caused by non-LOCA design basis events was analyzed. The conservatively calculated values were reflected in the rod ejection and locked rotor DBA analyses.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]	Conforms	The dose calculations determine the TEDE and consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Reference 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Reference 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	CEDE Conversion factors for isotopes were taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The breathing rates provided were utilized to calculate the offsite dose consequences. For determining a limiting 2 hour EAB dose, a constant breathing rate of 3.5×10^{-4} cubic meters per second was used.

TABLE 15B-1 (Sheet 7)

RG Section	RG Position	Analysis	Comments
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Reference 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	EDE conversion factors for isotopes were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	The TEDE was determined for the most limiting person at the EAB. The maximum two hour TEDE was determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two hour periods. This was performed by the RADTRAD computer code with constant inputs for atmospheric dispersion factors and breathing rates.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	The TEDE was determined for the most limiting receptor at the outer boundary of the low population zone (LPZ).
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No correction was made for the depletion of the effluent plume by deposition on the ground.

TABLE 15B-1 (Sheet 8)

RG Section	RG Position	Analysis	Comments
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The TEDE analysis considered all significant sources of radiation that would cause exposure to Control Room personnel. For WCGS, the limiting Control Room dose included: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from the Control Building, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in Control Room recirculation filters and radioactive material in the Control Building.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in nonconservative results for the control room.	Conforms	The radioactive material releases and radiation levels used in the Control Room dose analyses were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.
4.2.3	The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	The models used to transport radioactive material into and through the Control Room, and the shielding models used to determine radiation dose rates from external sources, were developed to provide suitably conservative estimates of the exposure to Control Room personnel.

TABLE 15B-1 (Sheet 9)

RG Section	RG Position	Analysis	Comments
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Reference 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Reference 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	Conforms	Credit for engineered safety features that mitigate airborne radioactive material within the Control Room and Control Building were assumed as appropriate.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Credit was not taken for the use of personnel protective equipment or prophylactic drugs.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	The occupancy factors and breathing rate were utilized to determine the doses to the hypothetical maximum exposed individual who is present in the Control Room. Control Room χ/Q values were determined utilizing the ARCON96 computer code which does not incorporate occupancy factors. Occupancy factors were included in the RADTRAD computer code for the dose evaluations.

TABLE 15B-1 (Sheet 10)

RG Section	RG Position	Analysis	Comments
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Reference 22). $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	Conforms	The DDE from photons was corrected for the difference between finite cloud geometry in the Control Room and the semi-infinite cloud assumption used in calculating the dose conversion factors by the given equation. This correction was performed by the RADTRAD computer code.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Reference 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	Exception – The current TID-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than Control Room and Technical Support Center doses. Full implementation of AST at WCGS does not impact the assumptions or inputs to the current TID based analyses.

TABLE 15B-1 (Sheet 11)

<table>
<tr><th>RG Section</th><th>RG Position</th><th>Analysis</th><th>Comments</th></tr>
<tr><td>4.4</td><td>The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

The acceptance criteria for the various NUREG-0737 (Reference 2) items generally reference General Design Criteria (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

<table>
<tr><th colspan="3">Table 6[17]
Accident Dose Criteria</th></tr>
<tr><th>Accident or Case</th><th>EAB and LPZ Dose Criteria</th><th>Analysis Release Duration</th></tr>
<tr><td>LOCA</td><td>25 rem TEDE</td><td>30 days for containment, ECCS, and MSIV (BWR) leakage</td></tr>
<tr><td>BWR Main Steam Line Break</td><td></td><td>Instantaneous puff</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Equilibrium Iodine Activity</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>BWR Rod Drop Accident</td><td>6.3 rem TEDE</td><td>24 hours</td></tr>
<tr><td>PWR Steam Generator Tube Rupture</td><td></td><td>Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Coincident Iodine Spike</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>PWR Main Steam Line Break</td><td></td><td>Until cold shutdown is established</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Coincident Iodine Spike</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>PWR Locked Rotor Accident</td><td>2.5 rem TEDE</td><td>Until cold shutdown is established</td></tr>
<tr><td>PWR Rod Ejection Accident</td><td>6.3 rem TEDE</td><td>30 days for containment pathway; until cold shutdown is established for secondary pathway</td></tr>
<tr><td>Fuel Handling Accident</td><td>6.3 rem TEDE</td><td>2 hours</td></tr>
</table>

The column labeled “Analysis Release Duration’ is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.</td><td>Conforms</td><td>The DBAs were updated for consistency with the TEDE criterion in Table 6 for offsite doses and in 10 CFR 50.67(b)(2)(iii) for the Control Room and Technical Support Center doses.</td></tr>
</table>

TABLE 15B-1 (Sheet 12)

RG Section	RG Position	Analysis	Comments
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	The DBA analyses were prepared, reviewed, and maintained per 10 CFR 50 Appendix B and the guidance consistent with RG 1.183.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Credit was taken for Engineered Safeguard Features with failure assumptions to maximize the calculated doses. Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose. For a range of values, the value that resulted in a conservative postulated dose was used.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	Licensee has ensured that analysis assumptions and methods are compatible with the AST and the TEDE criteria.

TABLE 15B-1 (Sheet 13)

RG Section	RG Position	Analysis	Comments
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Reference 29) implements Regulatory Guide 1.145 (Reference 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Reference 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Reference 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.	Conforms	The re-calculation of atmospheric dispersion factors was performed for the EAB and LPZ using the NRC computer code PAVAN according to the guidance of RG 1.145 and for the control room and TSC intakes with new release points using the NRC computer code ARCON96 according to the guidance of RG 1.194. The meteorological data used in the calculation were collected in accordance with WCGS site-specific measurements program and RG 1.23.

TABLE 15B-2 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX A (LOSS-OF-COOLANT-ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The sump pH is controlled at values of 7 or greater.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The radioactivity released from the fuel was assumed to mix instantaneously and homogeneously throughout the region of containment not impacted by sprays as it was released. Recirculation fans provide a mechanism for mixing between the sprayed and unsprayed portions of containment.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Reference A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Reference A-2). The latter model is incorporated into the analysis code RADTRAD (Reference A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	No natural deposition was assumed for elemental or organic iodine. A sedimentation removal coefficient of 0.1 hr^{-1} for particulates was credited. This is consistent with previously approved submittals, including: • Point Beach Units 1 & 2 – April 2011 (ADAMS Accession Number ML110240054)

TABLE 15B-2 (Sheet 2)

RG Section	RG Position	Analysis	Comments
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Reference A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"[1] (Reference A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	Conforms	A spray system is available in containment. The spray removal model from Chapter 6.5.2 of NUREG-0800 was used in the analysis. The containment sprays cover 85% of containment. The mixing within containment is promoted by fan coolers which begin operation 2 minutes into the accident and are assumed on for the duration of the event. The elemental iodine removal coefficient was limited to 10 hr^{-1} and removal was terminated when a DF of 200 was reached. A particulate removal coefficient of 5 hr^{-1} was modeled until a DF of 50 was reached, at which time the coefficient was reduced to 0.5 hr^{-1} until sprays were terminated at 5 hours.

TABLE 15B-2 (Sheet 3)

RG Section	RG Position	Analysis	Comments
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	In-containment recirculation filters were not credited.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Reference 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Not Applicable	Suppression pool scrubbing was not applicable to the WCGS design.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Reference A-1).	Not Applicable	No credit was taken for reduction of airborne radioactivity in the containment by retention in ice condensers or other engineering safety features not addressed above.

TABLE 15B-2 (Sheet 4)

RG Section	RG Position	Analysis	Comments
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	Releases via the containment mini-purge system were calculated assuming 100% of the RCS activity (based on the Technical Specifications) was released to containment. The mini-purge release is isolated before the onset of the gap release phase.

TABLE 15B-2 (Sheet 5)

RG Section	RG Position	Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Not Applicable	A dual containment is not applicable to the WCGS design.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Not Applicable	A dual containment is not applicable to the WCGS design.

TABLE 15B-2 (Sheet 6)

RG Section	RG Position	Analysis	Comments
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Not Applicable	A dual containment is not applicable to the WCGS design.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Reference A-5) and Generic Letter 99-02 (Reference A-6).	Not Applicable	A dual containment is not applicable to the WCGS design.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Table 1 of RG 1.183) were assumed to instantaneously and homogenously mix in the primary containment sump.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Reference A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The ESF leakage to the Plant Auxiliary Building was doubled in the analysis and was assumed to conservatively begin at the start of the event. The dose from ESF leakage to the refueling water storage tank was also included in the analysis.

TABLE 15B-2 (Sheet 7)

RG Section	RG Position	Analysis	Comments
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	Iodines were the only nuclides modeled to be released from the liquid.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$ where, hf1 is the enthalpy of liquid at system design temperature and pressure; hf2 is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and hfg is the heat of vaporization at 212°F.	Conforms	The calculated flashing fraction based on the maximum sump temperature was less than 0.10.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	An airborne fraction of 10% was assumed for the iodine in the liquid.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Reference A-5) and Generic Letter 99-02 (Reference A-6).	Conforms	The radioiodine that was postulated to be available for release to the environment was assumed to be 97% elemental and 3% organic. No reductions due to dilution or holdup were assumed. No credit was taken for the ESF ventilation filtration system in the Plant Auxiliary Building (PAB) exhaust.

TABLE 15B-2 (Sheet 8)

RG Section	RG Position	Analysis	Comments
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Reference A-5) and Generic Letter 99-02 (Reference A-6).	Not Applicable	The containment is not purged post-LOCA.

TABLE 15B-3 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX B (FUEL HANDLING ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	For the postulated fuel handling accident, one entire assembly and 20% of an adjacent assembly were assumed to be damaged as a result of this event.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	The fission product release from the breached fuel was based on Regulatory Position 3.2 of RG 1.183 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods was assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All particulate iodine released to the spent fuel pool was assumed to dissociate and instantaneously re-evolve as elemental iodine.

TABLE 15B-3 (Sheet 2)

RG Section	RG Position	Analysis	Comments
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Reference B-1).	Conforms	The minimum water depth over the reactor core when handling fuel and over the spent fuel in the fuel handling building is 23 feet, so the overall DF of 200 was applied.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	The analysis modeled a noble gas DF of 1 and an infinite DF for particulates.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release of radioactive material was modeled as a linear release over a 2-hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system1 should be determined and accounted for in the radioactivity release analyses.	Not Applicable	Filtration from the engineered safety features were not credited for the releases from the fuel building.

TABLE 15B-3 (Sheet 3)

RG Section	RG Position	Analysis	Comments
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	No credit was taken for mixing or dilution in the fuel building.
5.1	If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.	Not Applicable	No credit was taken for containment isolation in the analysis.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	The containment was assumed to be open and the release of radioactive material was modeled as a linear release over a 2-hour period.
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]	Not Applicable	No engineered safety features were credited for the releases from containment.

TABLE 15B-3 (Sheet 4)

RG Section	RG Position	Analysis	Comments
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	No credit was taken for mixing or dilution in containment.

TABLE 15B-4 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX E (PWR MAIN STEAM LINE BREAK)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Not Applicable	No fuel damage was postulated to occur during the MSLB.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 µCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 µCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 500 and spike duration of 8 hours. The initial activity was based on 1.0 µCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

TABLE 15B-4 (Sheet 2)

RG Section	RG Position	Analysis	Comments
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment should be 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.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	For facilities that have not implemented alternative repair criteria (see Reference E-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be entirely to the faulted steam generator and the normal operation Technical Specification primary-to-secondary leakage of 150 gpd/SG was assumed to be evenly apportioned between the intact steam generators.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated at 34 hours when the reactor coolant system was cooled to 212°F. The release of radioactivity from the unaffected steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

TABLE 15B-4 (Sheet 3)

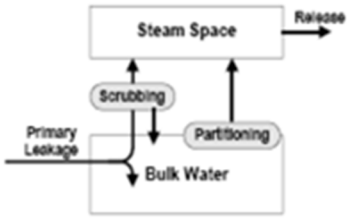
RG Section	RG Position	Analysis	Comments
5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below: 	Conforms	The transport model was utilized in the analysis.
5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.	Conforms	The faulted steam generator was assumed to blowdown to dryout conditions and any primary-to-secondary leakage was modeled as a release to the environment with no mitigation. Leakage to the intact steam generators was modeled to mix with the secondary water without flashing since the steam generator tubes are submerged.
5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Reference E-2), during periods of total submergence of the tubes.	Not Applicable	See Position 5.5.1 above.
5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Not Applicable	See Position 5.5.1 above.

TABLE 15B-4 (Sheet 4)

RG Section	RG Position	Analysis	Comments
5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	A partition coefficient of 100 was modeled for iodine and the particulate radionuclides release was limited by the moisture carryover of 0.25%.
5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Reference E-3). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Not Applicable	The issue of tube uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible and the issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor System Branch, to Lawrence A. Walsh, Chairman of the WOG. The letter states "… the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue to be resolved." Consistent with this position, the MSLB dose analysis did not model tube uncovery in the intact steam generators.

TABLE 15B-5 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX F (PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage was postulated to occur during the SGTR.
2	If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.	Conforms	Since no fuel damage occurs, two cases of iodine spiking (pre-accident and accident-initiated) were modeled.
2.1	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 (typically 60 µCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).	Conforms	The pre-accident iodine spike was modeled with a primary coolant iodine concentration of 60 µCi/gm DE I-131, consistent with the Technical Specification limit.
2.2	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 using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.	Conforms	The accident-initiated iodine spike was modeled with a spike factor of 335 and spike duration of 8 hours. The initial activity was based on 1.0 µCi/gm DE I-131, consistent with the Technical Specification limit.
3	The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.

TABLE 15B-5 (Sheet 2)

RG Section	RG Position	Analysis	Comments
4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned between the intact steam generators. This leakage modeling was used since the leakage is negligible compared to the flow through the ruptured tube.
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary break flow was terminated when the reactor coolant system pressure equalized with the ruptured steam generator secondary side pressure. The release of radioactivity from the ruptured and intact steam generators was terminated at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed coincident with reactor trip.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.

TABLE 15B-5 (Sheet 3)

RG Section	RG Position	Analysis	Comments
5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the SGTR. In addition, flashing of break flow in the ruptured steam generator with a time dependent flashing fraction was considered and all activity in the flashed break flow was released to the environment with no mitigation, dilution, or credit for scrubbing.

TABLE 15B-6 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX G (PWR LOCKED ROTOR ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled the fraction of the core inventory assumed to be in the gap by radionuclide group consistent with Table 3 in Regulatory Position 3.2. The assumed number of fuel rods breached was 5% of the core and the radial peaking factor of 1.65 was applied.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity was modeled to be released instantaneously and homogenously throughout the primary coolant.
4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment should be 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.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators.

TABLE 15B-6 (Sheet 2)

RG Section	RG Position	Analysis	Comments
5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The primary-to-secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.
5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	A loss of offsite power was assumed.
5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
5.6	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the locked rotor event.

TABLE 15B-7 (Sheet 1)

CONFORMANCE WITH REGULATORY GUIDE 1.183 APPENDIX H (PWR ROD EJECTION ACCIDENT)

RG Section	RG Position	Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel were obtained from Regulatory Position 3 of RG 1.183. The analysis modeled gap release fractions of 10% for iodines and noble gases and 12% for alkali metals. The assumed number of fuel rods breached was 10% of the core and the radial peaking factor of 1.65 was applied. The analysis also modeled fuel melting release fractions of 100% of noble gases and 50% of iodines and alkali metals in the melted fuel rods. Note that the assumption of 50% iodine and alkali metal release from melted fuel was conservatively modeled for each pathway.
2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Not Applicable	Fuel damage was postulated; therefore, an analysis was performed.
3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	Both cases were considered separately in the analysis.
4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The iodine chemical fractions for releases to containment were modeled as 95% CsI, 4.85% elemental, and 0.15% organic. All fission products, with the exception of elemental and organic iodine and noble gases, were assumed to be in particulate form. No removal processes were modeled in containment besides leakage and decay, so sump pH has no impact.

TABLE 15B-7 (Sheet 2)

RG Section	RG Position	Analysis	Comments
5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	Iodine chemical fractions for releases to the environment (97% elemental and 3% organic) were modeled in the analysis.
6	Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.	Conforms	See below.
6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Not Applicable	No removal processes were modeled in containment besides leakage and decay.
6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.	Conforms	The primary containment peak pressure leak rate was defined as 0.2% by weight of containment air. This leak rate was reduced to 0.1% after the first 24 hours of the accident.
7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	The accident-induced Technical Specification Bases leakage of 1 gpm was assumed to be evenly apportioned to the steam generators. The primary-to- secondary leakage was terminated when the release of radioactivity from the steam generators was terminated. This occurred at 12 hours when the residual heat removal system was in service and removing all decay heat.

TABLE 15B-7 (Sheet 3)

RG Section	RG Position	Analysis	Comments
7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	A density of 1.0 gm/cc (62.4 lbm/ft^3) was used.
7.3	All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.	Conforms	No reduction or mitigation of noble gas activity in releases from the primary system was modeled.
7.4	The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the rod ejection event.