CHAPTER 15

ACCIDENT ANALYSES

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Chapter 15

ACCIDENT ANALYSES

Since 1970, the American Nuclear Society (ANS) classification of plant conditions has been used to divide 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:

(1)	Condition I:	Normal Operation and Operational Transients (Initial Conditions)
(2)	Condition II:	Faults of Moderate Frequency
(3)	Condition III:	Infrequent Faults
(4)	Condition IV:	Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no 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 system and engineered safety features (ESFs) functioning is assumed, to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

In the evaluation of the radiological consequences associated with initiation of a spectrum of accident conditions, numerous assumptions must be postulated. In many instances these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

The specific accident sequences analyzed in this chapter include those required by Regulatory Guide 1.70, Revision 1, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, and others considered significant for the Diablo Canyon Power Plant (DCPP). Refer to Table 15.0-1 for a comparison between Table 15-1 of Regulatory Guide 1.70, Revision 1, and the corresponding section(s) where the conditions are discussed. Because the DCPP design differs from other plants, some of the representative types of events identified in Table 15-1 of Regulatory Guide 1.70, Revision 1 are not applicable to this plant. In addition, some events are analyzed or discussed in separate chapters. The location of the analysis for each event or reason the event is not applicable to DCPP is provided in Table 15.0-1.

This section describes the acceptance criteria, input assumptions, analysis techniques, equipment performance, and analysis results of the required accident analysis but does not include details on the set points, capacity or capabilities of mitigating equipment or operational limitations that determine the initial conditions for each analysis. For details of required reactor operational limitations and of the performance capabilities of the emergency equipment not covered in Chapter 15, refer to the following chapters:

- Reactor coefficients, power distribution, reactivity controls (refer to Chapter 4)
- Reactor coolant flow (refer to Chapter 5)
- Emergency core cooling system (ECCS), Auxiliary feed water, Containment systems (refer to Chapter 6)
- Reactor trips and permissives, ESFs actuation (refer to Chapter 7)
- Boration capabilities (refer to Chapter 9)

Additionally the availability, testing and performance criteria of the operational limits and mitigating systems are administratively controlled by the plant Technical Specifications described in Chapter 16 and Appendix A of the DCPP Unit 1 and Unit 2 Operating Licenses.

15.1 CONDITION I - NORMAL OPERATION AND OPERATIONAL TRANSIENTS (INITIAL CONDITIONS)

15.1.1 INTRODUCTION

Condition I occurrences are those that are expected frequently or regularly 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. Since Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions that can occur during Condition I operation.

Typical Condition I events are shown below:

(1) Steady state and shutdown operations

Mode 1 - Power operation (greater than 5 percent of rated thermal power [RTP])

Mode 2 - Startup ($k_{eff} \geq 0.99$, less than or equal to 5 percent of RTP)

Mode 3 - Hot standby (k_{eff} less than 0.99, T_{avg} greater than or equal to 350°F)

Mode 4 - Hot shutdown (subcritical, residual heat removal [RHR] system in operation, k_{eff} less than 0.99, 200°F less than T_{avg} less than 350°F)

Mode 5 - Cold shutdown (subcritical, RHR system in operation, k_{eff} less than 0.99, T_{avg} less than or equal to 200°F)

Mode 6 - Refueling (k_{eff} less than or equal to 0.95, T_{avg} less than or equal to 140°F)

(2) Operation with permissible deviations

Various deviations that may occur during continued operation as permitted by the plant Technical Specifications (Reference 1) must be considered in conjunction with other operational modes. These include:

(a) Operation with components or systems out of service

(b) Leakage from fuel with cladding defects

(c) Activity in the reactor coolant

1. Fission products

2. Corrosion products

3. Tritium

(d) Operation with steam generator (SG) leaks up to the maximum allowed by the Technical Specifications

(3) Normal Operational transients

Normal design transients which do not result in a reactor trip are listed below. Refer to Section 5.2.2.1.5.1 for additional details on these transients.

(a) Plant heatup and cooldown

(b) Step load changes (up to plus or minus 10 percent between 15 percent load and full load)

(c) Ramp load changes (up to 5 percent per minute between 15 percent load and full load)

(d) Turbine load reduction up to and including a 50 percent load rejection from full power

(e) Steady state fluctuations of the reactor coolant average temperature, for purposes of design, is assumed to increase or decrease at a maximum rate of 6°F in 1 minute.

15.1.2 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 the NOTRUMP code used in the analysis of the reactor coolant system (RCS) small pipe rupture (refer to Section 15.3.1), and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient event are listed in Table 15.1-4.

15.1.2.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metalclad UO_2 fuel rod (refer to Figure 15.1-8) and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density).

The code uses a fuel model that exhibits the following features simultaneously:

(1) A sufficiently large number of finite difference radial space increments to handle fast transients such as rod ejection accidents

(2) Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation

(3) The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, zirconium-water (Zr-H_2O) reaction, and partial melting of the materials

The gap heat transfer coefficient is calculated according to an elastic pellet model. The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded cladding, there is no fuel-cladding contact and the gap conductance is calculated on the basis of the thermal conductivity of the gas contained in the gap. If the pellet outside radius so calculated is larger than the cladding inside radius (negative gap), the pellet and the cladding are pictured as exerting upon each other a pressure sufficient to reduce the gap to zero by elastic deformation of both. This contact pressure determines the heat transfer coefficient.

FACTRAN is further discussed in the licensing topical report, Section 1.6.1, Item 44.

15.1.2.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by modeling the reactor core and vessel, hot and cold leg piping, SG (tube and shell-sides), pressurizer, and reactor coolant pumps (RCPs), with up to four reactor coolant loops (RCLs). 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 SG 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 neutron flux, overpower and overtemperature reactor coolant 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 safety injection system (SIS), including the accumulators, is also modeled.

LOFTRAN is a versatile program that 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 DNB based on the input from the core limits illustrated in Figure 15.1-1. The core limits represent the minimum value of departure from nucleate boiling ratio (DNBR) as calculated for a typical or thimble cell. LOFTRAN is further discussed in the licensing topical report, Section 1.6.1, Item 47.

15.1.2.3 PHOENIX- P

The PHOENIX-P computer code is a two-dimensional, multi-group, transport based lattice code and is capable of providing all necessary data for PWR analysis. Being a dimensional lattice code, PHOENIX-P does not rely on pre-determined spatial/spectral interaction assumptions for a heterogeneous fuel lattice. The PHOENIX-P computer code is approved by the U.S. Nuclear Regulatory Commission (NRC) as the lattice code for generating macroscopic and microscopic few group cross-sections for PWR analysis.

The PHOENIX-P computer code is described in more detail in Section 4.3.3.10.2 and is further discussed in the licensing topical report, Section 1.6.1, Item 60.

15.1.2.4 ANC

With the advent of VANTAGE 5 fuel and axial features such as axial blankets and part length burnable absorbers, the three dimensional nodal codes ANC (Advanced Nodal Code) has replaced the previous two group X-Y TURTLE code. The three dimensional nature of the nodal codes provides both the radial and axial power distributions, and also determines the critical boron concentrations and power distributions. The moderator coefficient is evaluated by varying the inlet temperature in the same calculations used for power distribution and reactivity predictions.

Axial calculations are used to determine differential control rod worth curves (reactivity versus rod insertion) and axial power shapes during steady state and transient xenon conditions. Group constants are obtained from three-dimensional nodal calculations homogenized by flux volume weighting.

The ANC computer code is described in more detail in Section 4.3.3.10.3 and is further discussed in the licensing topical reports, Section 1.6.1, Items 60 and 61.

15.1.2.5 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-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 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 provide channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, fuel temperatures, and so on.

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

TWINKLE is further described in the licensing topical report, Section 1.6.1, Item 50.

15.1.2.6 THINC

The Steady state and transient analysis using the THINC code (THINC-I, THINC-III and THINC-IV) is described in Section 4.4.3. THINC is further described in the licensing topical reports, Section 1.6.1, Item 28.

15.1.2.7 RETRAN-02

The Electric Power Research Institute (EPRI) RETRAN-02 program is used to perform the best estimate thermal-hydraulic analysis of operational and accident transients for light water reactor systems. The program is constructed with a highly flexible modeling technique that provides the RETRAN-02 program the capability to model the actual performance of the plant systems and equipment.

The main features of the RETRAN-02 program are:

(1) A one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the reactor cooling system

(2) A point neutron kinetics model for the reactor core

(3) Special auxiliary or component models (such as non-equilibrium pressurizer temperature transport delay)

(4) Control system models

(5) A consistent steady state initialization technique

The RETRAN-02 program is further discussed in Reference 21.

15.1.2.8 RETRAN-02W

The RETRAN-02W program is the Westinghouse version of the RETRAN-02 program. RETRAN-02W is used to determine plant transient response to selected accidents, as described in Sections 15.2 and 15.4.
RETRAN-02W is further described in the licensing topical report, Section 1.6.1, Item 58.

15.1.2.9 NOTRUMP

The NOTRUMP computer code is a state-of-the-art, one-dimensional general network code consisting of a number of advanced features. Among these features is 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. Additional features of the code are condensation heat transfer model applied in the SG region, loop seal model, core reflux model, flow regime mapping, etc. NOTRUMP is used to model the thermal-hydraulic behavior of the system and thereby obtain time-dependent values of various core region parameters, such as system pressure, temperature, fluid levels and flow rates, etc.

Small-Break loss-of-coolant accident (SBLOCA) analysis performed using the NOTRUMP code is further described in Section 15.3 and in the licensing topical reports, Section 1.6.1, Items 63 and 64.

15.1.2.10 SBLOCTA (LOCTA-IV)

The NOTRUMP topical report WCAP-10054-P-A makes reference to the LOCTA-IV code (WCAP-8301) and provides modifications to the LOCTA-IV code for use in SBLOCA analyses (i.e., Small Break LOCTA). Further modifications for an annular fuel pellet model were submitted and approved by the NRC in WCAP-14710-P-A, which states, “the revised model has been installed in the SBLOCTA code, which is one of a series of codes descended from the original LOCTA-IV code, and is specific to analyzing SBLOCA transients.” So, SBLOCTA is the actual computer code name, with base references of WCAP-8301 and WCAP-10054-P-A.

SBLOCA analysis performed using the LOCTA-IV code is further described in Section 15.3 and listed as Reference 4 in that section.

15.1.2.11 WCOBRA/TRAC

The thermal-hydraulic computer code (WCOBRA/TRAC, Version Mod 7A, Revision 1) that was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large-break LOCA (LBLOCA) in WCAP-12945-P-A, Volumes I through V is described in Section 15.4.1.3 and in the licensing topical report, Section 1.6.1, Item 62.

15.1.2.12 HOTSPOT

The use of HOTSPOT along with WCOBRA/TRAC to examine Unit 2 uncertainty using the ASTRUM methodology is discussed in Section 15.4.1.7B.

15.1.2.13 MONTECF

Unit 2 uncertainty evaluation calculations using the ASTRUM methodology was performed by applying a direct, random Monte Carlo sampling to generate the input for the WCOBRA/TRAC and HOTSPOT computer codes as discussed in Section 15.4.1.7B.

15.1.2.14 COCO

Containment pressure is calculated using the COCO code (WCAP-8327 and WCAP-8326) as discussed in Section 15.4.1.3 and listed as Reference 61 in that section.

15.1.3 OPTIMIZATION OF CONTROL SYSTEMS

Prior to initial startup, a setpoint study (Reference 2) was performed in order to simulate performance of the reactor control and protection systems. Emphasis was placed on the development of a control system that will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints was determined. In areas where the resultant setpoints were different, compromises based on the optimum overall performance were made and verified. A consistent set of control system parameters was derived satisfying plant operational requirements throughout the core life and for power levels between 15 and 100 percent. The study contained an analysis of the following control systems: rod cluster assembly control, steam dump, SG level, pressurizer pressure, and pressurizer level.

Since initial startup, setpoints and control system components have been maintained to optimize performance. Plant operability margin-to-trip analyses are performed on the nuclear steam supply system (NSSS) control systems for DCPP Unit 1 and Unit 2. The purpose of these analyses is to demonstrate that the margin to relevant reactor trip and engineered safety features actuation system setpoints is adequate. The NSSS control systems setpoints and time constants are analyzed to provide stable plant response during and following the operational (Condition I) transients:
:

- 50 percent load rejection from 100 percent power
- 10 percent step-load decrease from 100 percent power
- 10 percent step-load increase from 90 percent power
- Turbine trip without reactor trip from permissive P-9 setpoint

When changes are made, the accident analyses are reviewed and revised as necessary. The impact of maintaining pressurizer level between 22% and 35% during a shutdown to mode 3 and when power is ≤20%, has been evaluated as acceptable since it was determined that there is no adverse impact on any accident analyses (Reference 31). The impact of maintaining pressurizer level greater than or equal to 22 percent and less than or equal to 90 percent in Modes 3, 4, and 5 has been evaluated as acceptable because there is no adverse impact on any accident analyses (References 28 and 29).

The analysis for the 50 percent load reduction (References 33 and 35) shows that the DCPP control system is capable of controlling the SG water level so that a reactor trip on SG low-low level or turbine trip / feedwater isolation on SG high-high level does not occur. Specific analysis results show that the SG level is maintained within +/-20 percent of the nominal setpoint and all control system responses are smooth and have no sustained oscillations or divergence. To ensure that a load reduction transient presents no hazard to the integrity of the RCS or the main steam system (MSS), the Condition II analysis presented in Section 15.2.7 continues to assume a total loss of external electrical load without an immediate reactor trip.

15.1.4 INITIAL POWER CONDITIONS ASSUMED IN ACCIDENT ANALYSES

Reactor power-related initial conditions assumed in the accident analyses presented in this chapter are described in this section.

15.1.4.1 Power Rating

Table 15.1-1 lists the principal power rating values that are assumed in analyses performed in this section. Two ratings are given:

(1) The RTP output. The RTP is the total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.

(2) The NSSS thermal power output. This power output includes the RTP plus the thermal power generated by the RCPs.

(3) The ESFs design rating. The Westinghouse-supplied ESFs are designed for a thermal power higher than the NSSS value in order not to preclude realization of future potential power capability. This higher thermal power value is designated as the ESF design rating.

Where initial power operating conditions are assumed in accident analyses, the NSSS or core RTP output (plus allowance for errors in steady state power determination for some accidents) is assumed. Where demonstration of the adequacy of the ESF is concerned, the ESF design rating plus allowance for error is assumed. The thermal power values for each transient analyzed are given in Table 15.1-4.

15.1.4.2 Initial Conditions

For most accidents, which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in Reference 3. This procedure is known as the "Improved Thermal Design Procedure" (ITDP) and these accidents utilize the WRB-1 and WRB-2 DNB correlations (References 4 and 5). ITDP allowances may be more restrictive than non-ITDP allowances. The initial conditions for other key parameters are selected in such a manner to maximize the impact on DNBR. Minimum measured flow is used in all ITDP transients. The allowances on power, temperature, pressure, and flow that were evaluated for their effect on the ITDP analyses for a 24-month fuel cycle are reported in Reference 22. These allowances are conservatively applicable for shorter fuel cycle lengths.

For accident evaluations that are not DNB limited, or for which the ITDP is not employed, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered:

(1)	Core power	Plus or minus 2 percent allowance calorimetric error
(2)	Average RCS	Plus or minus 4.7°F allowance for deadband and measurement error temperature
(3)	Pressurizer pressure	Plus or minus 38 psi or plus or minus 60 psi allowance for steady state fluctuations and measurement error (refer to Note)

Note: Pressurizer pressure uncertainty is plus or minus 38 psi in analyses performed prior to 1993; however, NSAL 92-005 (Reference 17) indicates plus or minus 60 psi is a conservative value for future analyses. Reference 18 evaluates the acceptability of existing analyses, which use plus or minus 38 psi.

For some accident evaluations, an additional allowance has been conservatively added to the measurement error for the average RCS temperatures to account for SG fouling.

DCPP Unit 1 and Unit 2 are expected to operate at a RCS vessel average temperature (Tavg) over a range from 565 ºF to 577.3/577.6 ºF (Unit 1/Unit 2).

15.1.4.3 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, control rods, and by operation instructions. The power distribution may be characterized by the radial peaking factor $F_{\Delta H}$ and the total peaking factor F_q. The peaking factor limits are given in the Technical Specifications.

For transients that 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 $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.1-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.3.13.

For transients that may be overpower-limited, the total peaking factor F_q is of importance. The value of F_q may increase with decreasing power level so that the full power hot spot heat flux is not exceeded; i.e., F_q x Power = design hot spot heat flux. All transients that may be overpower-limited are assumed to begin with a value of F_q consistent with the initial power level as defined in the Technical Specifications.

The value of peak kW/ft can be directly related to fuel temperature as illustrated in Figures 4.4-1 and 4.4-2. For transients that are slow with respect to the fuel rod thermal time constant (approximately 5 seconds), the fuel temperatures are illustrated in Figures 4.4-1 and 4.4-2. For transients that are fast with respect to the fuel rod thermal time constant, (for example, rod ejection), a detailed heat transfer calculation is made.

15.1.5 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 (CRDMs). The loss of power to the mechanism coils causes the mechanism to release the rod cluster control assemblies (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. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.1-2. Reference is made in that table to the overtemperature and overpower ΔT trip shown in Figure 15.1-1. This figure presents the allowable RCL average temperature and ΔT for the design flow and the NSSS Design Thermal Power distribution 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 values (1.68 and 1.71 for V-5 thimble cell and typical cells, respectively) for analyses using the ITDP. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit values. 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 current fuel cycles for the DCPP Unit 1 and Unit 2 only use the Vantage 5 (V-5) fuel assembly type. However, the safety analyses performed in support of the transition to Vantage-5 fuel also considered the presence of the Standard type fuel assemblies. The DNBR values and transient results presented in the UFSAR continue to reflect the Standard limits, since they are limiting with respect to DNB margin in comparison to the Vantage-5 limits. Analyses performed subsequent to the transition to a full Vantage-5 core reflect only the Vantage-5 limits as described in Sections 15.2, 15.3, and 15.5. The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints); and by a line defining conditions at which the SG safety valves open.

The limit values, which were used as the DNBR limits for all accidents analyzed with the ITDP are conservative compared to the actual design DNBR values required to meet the DNB design basis.

The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error.

During startup tests, it is demonstrated that actual instrument errors and time delays are equal to or less than the assumed values.

Accident analyses that assume the SG low-low water level to initiate protection functions may be affected by the trip time delay (Reference 19) that was developed to reduce the incidence of unnecessary feedwater related reactor trips.

Refer to Section 7.2.2.1.5 for a discussion about the low-low SG water level trip, including the trip time delay.

15.1.6 CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX

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. The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the SGs, and steam pressure. High-accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those that would be required to control feedwater flow.

15.1.7 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCA and the variation in rod worth as a function of rod position.

With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The RCCA position versus time assumed in accident analyses is shown in Figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. This curve is used as input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed axial power distribution that would exist relatively infrequently. For cases other than those associated with xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial power distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.1-4. The curve shown in this figure was obtained from Figures 15.1-2 and 15.1-3. 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 Tables 4.3-2 and 4.3-3.

The normalized RCCA negative reactivity insertion versus time after trip curve for an axial power distribution skewed to the bottom (refer to Figure 15.1-4) is used in transient analyses.

Where special analyses require the use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.1-2 is used as code input.

15.1.8 REACTIVITY COEFFICIENTS

The transient response of the RCS is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.

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 RCS, do not depend on reactivity feedback effects. The values used are given in Table 15.1-4; reference is made in that table to Figure 15.1-5 that shows the upper and lower 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 is discussed on an event-by-event basis.

15.1.9 FISSION PRODUCT INVENTORIES

The fission product inventories existing in the core and fuel rod gaps are described in Section 15.5.3. The description of the models used for calculating fuel gap activities is included in Section 15.5.3.

15.1.10 RESIDUAL DECAY HEAT

Residual heat in a subcritical core consists of:

(1) Fission product decay energy

(2) Decay of neutron capture products

(3) Residual fissions due to the effect of delayed neutrons

These constituents are discussed separately in the following paragraphs.

15.1.10.1 Fission Product Decay

The heat generation rates from radioactive decay of fission products that have been assumed in the SBLOCA accident analyses are equal to 1.2 times the values for infinite operating time in the 1971 Draft ANS-5 Standard. (Reference 30)

The decay heat curve used for the best estimate large break LOCA (LBLOCA) analysis is based on the 1979 ANS decay heat curve as described in Section 8 of Reference 23. This curve with the 20 percent factor included is shown in Figure 15.1-6.The 1979 ANS decay heat curve (Reference 11) is used for the non-LOCA analyses. Figure 15.1-7 presents this curve as a function of time after shutdown.

15.1.10.2 Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5-minute half-life) and Np-239 (2.35-day half-life) contribute significantly to the heat generation after shutdown. The cross-sections for production of these isotopes and their decay schemes are relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_0 = \frac{(E_{\gamma 1} + E_{\beta 1})\, c\,(1+\alpha)}{200\ \text{MeV}}\, e^{-\lambda_1 t}\ \text{watts/watt} \qquad (15.1\text{-}1)$$

$$P_2/P_0 = \frac{(E_{\gamma 2} + E_{\beta 2})\, c\,(1+\alpha)}{200\ \text{MeV}} \left[\frac{\lambda_2}{\lambda_1 - \lambda_2} \left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t} \right] \text{watts/watt} \qquad (15.1\text{-}2)$$

where:

P_1/P_0 is the energy from U-239 decay
P_2/P_0 is the energy from Np-239 decay
t is the time after shutdown (seconds)
$c(1+\alpha)$ is the ratio of U-238 captures to total fissions = 0.6 (1 + 0.2)
λ_1 = the decay constant of U-239 = 4.91×10^{-4} per second
λ_2 = the decay constant of Np-239 decay = 3.41×10^{-6} per second
$E_{\gamma 1}$ = total γ-ray energy from U-239 decay = 0.06 MeV
$E_{\gamma 2}$ = total γ-ray energy from Np-239 decay = 0.30 MeV
$E_{\beta 1}$ = total β-ray energy from U-239 decay = 1/3[(a)] x 1.18 MeV
$E_{\beta 2}$ = total β-ray energy from Np-239 decay = 1/3[(a)] x 0.43 MeV

(a) Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.

For the SBLOCA, based on conservative modeling of the ratio of U-238 captures to total fissions, heavy element decay heat is calculated without applying further uncertainty correction (Reference 24). For the best estimate LOCA analysis, the heat from the radioactive decay of U-239 and Np-239 is calculated as described in Section 8 of Reference 23. The decay of other isotopes, produced by neutron reactions other than fission, is neglected. For the non-LOCA analysis, the decay of U-238 capture products is included as an integral part of the 1979 decay heat curve presented as Figure 15.1-7.

15.1.10.3 Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout a transient under consideration. Core average conditions are more conservative for the calculation of reactivity and power level than actual local conditions as they would exist in hot areas of the core. Thus, unless otherwise stated in the text, static power shapes have been assumed in the analysis and these are factored by the time behavior of core average fission power calculated by a point kinetics model calculation with six delayed neutron groups.

For the purpose of illustration, only one delayed neutron group calculation, with a constant shutdown reactivity of -4 percent Δk is shown in Figure 15.1-6.

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

During an SBLOCA the core is rapidly shut down by void formation or RCCA insertion, or both, and long-term shutdown is assured by the borated ECCS water. 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 that are important for the neutron dependent part of the heat generation do not apply to the gamma ray source contribution. The steady state factor of 97.4 percent that represents the fraction of heat generated within the cladding and pellet drops to 95 percent for the hot rod in a LOCA.
For example, 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.

For the best estimate LOCA analysis, the energy deposition modeling is performed as described in Section 8 of Reference 23.

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15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults result at worst in the reactor shutdown with the plant capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault; i.e., a Condition III or IV fault. In addition, Condition II events are not expected to result in fuel rod failures, RCS overpressurization, or MSS overpressurization. For the purposes of this report the following faults have been grouped into these categories:

(1) Uncontrolled RCCA bank withdrawal from a subcritical condition

(2) Uncontrolled RCCA bank withdrawal at power

(3) RCCA misoperation

(4) Uncontrolled boron dilution

(5) Partial loss of forced reactor coolant flow

(6) *Startup of an inactive RCL* (Historical)

(7) Loss of external electrical load and/or turbine trip

(8) Loss of normal feedwater

(9) Loss of offsite power (LOOP) to the station auxiliaries

(10) Excessive heat removal due to feedwater system malfunctions

(11) Sudden feedwater temperature reduction

(12) Excessive load increase incident

(13) Accidental depressurization of the RCS

(14) Accidental depressurization of the MSS

(15) Spurious operation of the SIS at power

Each of these faults of moderate frequency are analyzed in this section. In general, each analysis includes acceptance criteria, an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

An evaluation of the reliability of the reactor protection system actuation following initiation of Condition II events has been completed and is presented in Reference 1 for the relay protection logic. Standard reliability engineering techniques were used to assess the likelihood of the trip failure due to random component failures. Common-mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following the initiation of Condition II events is extremely small (2 x 10^{-7} derived for random component failures). The solid-state protection system design has been evaluated by the same methods as used for the relay system and the same order of magnitude of reliability is provided.

Hence, because of the high reliability of the protection system, no special provision is included in the design to cope with the consequences of Condition II events without trip.

The time sequence of events corresponding to the respective Condition II fault is shown in Table 15.2-1.

15.2.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION

15.2.1.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) Minimum DNBR is not less than the appropriate limit value at any time during the transient.

15.2.1.2 Identification of Causes and Accident Description

A RCCA withdrawal accident is defined as an uncontrolled increase in reactivity in 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 control rod drive systems. The Section 15.2.1 event occurs with the reactor at hot zero power (i.e., subcritical). The at-power case is discussed in Section 15.2.2.

Although the reactor can be brought to power from a subcritical condition by means of RCCA withdrawal, startup procedures following refueling also permit boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (refer to Section 15.2.4).

The RCCA drive mechanisms are wired into preselected bank configurations that are not altered during core reactor life. These circuits prevent the assemblies from being 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. 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 two control banks having the maximum combined worth at maximum speed.

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 burst 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.

15.2.1.2.1 Source Range High Neutron Flux Reactor Trip

The source range high neutron flux reactor trip is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when either 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.

15.2.1.2.2 Intermediate Range High Neutron Flux Reactor Trip

The intermediate range high neutron flux reactor trip is actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels give readings above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power below this value.

15.2.1.2.3 Power Range High Neutron Flux Reactor Trip (Low Setting)

The power range high neutron flux trip (low setting) is actuated when two-out-of-four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below 10 percent.

15.2.1.2.4 Power Range High Neutron Flux Reactor Trip (High Setting)

The power range high neutron flux reactor trip (high setting) is actuated when two-out-of-four power range channels indicate a power level above a preset setpoint. This trip function is always active.

15.2.1.2.5 Power Range High Positive Neutron Flux Rate Trip

The power range high positive neutron flux rate trip is actuated when the rate of change in power on two-out-of-four power range channels exceeds the preset setpoint. This trip function is always active.

15.2.1.3 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. The TWINKLE (Reference 2) code is used to calculate the reactivity transient and hence the nuclear power transient. The FACTRAN (Reference 3) code is then used to calculate the thermal heat flux transient based on the nuclear power transient calculated by the TWINKLE code. FACTRAN also calculates the fuel, cladding, and coolant temperatures. A detailed thermal and hydraulic computer code, THINC (refer to Section 1.6.1, Item 28 and Section 4.4.3) (Reference 9) is used to calculate the DNB.

The event is not analyzed with the ITDP since it is analyzed with reduced flow.

In order to give conservative results for a startup accident, the following assumptions are made concerning the initial reactor conditions:

(1) 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 coefficient, conservative values (low absolute magnitude) as a function of power are used (refer to Section 15.1.8 and Table 15.1-4).

(2) 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. The conservative value, given in Table 15.1-4, is used in the analysis to yield the maximum peak heat flux.

(3) The reactor is assumed to be at hot zero power. 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 is assumed to be 1 since this results in maximum neutron flux peaking.

(4) 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 increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 to 35 percent. Previous results, however, show that the rise in neutron flux is so rapid that the effect of error on this 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. Refer to Section 15.1.7 for RCCA insertion characteristics.

(5) The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 inches/minute). CRDM design is discussed in Section 4.2.3.

(6) The initial power level is assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

15.2.1.4 Results

Figures 15.2.1-1 through 15.2.1-3 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35 percent nominal power. This insertion rate is greater than that for the two highest worth control banks, both assumed to be in their highest incremental worth region.

Figure 15.2.1-1 shows the neutron flux transient. The neutron flux overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increase are relatively small. The thermal flux response, of interest for DNB considerations, is shown in Figure 15.2.1-2. The beneficial effect on the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value. The DNBR remains above the applicable safety analysis limit value at all times.

Figure 15.2.1-3 shows the response of the average fuel, cladding, and coolant temperatures at the hot spot.

15.2.1.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) Minimum DNBR remains above the appropriate limit value at any time during the transient.

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 above the limiting value.

15.2.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.2.2.1 Acceptance Criteria

The following are the relevant specific acceptance criteria.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

(2) The peak core average power (heat flux) does not exceed a value that would cause fuel centerline melt at any time during the transient (refer to Section 4.4.3.2.7).

(3) The RCS pressure does not exceed 110% of design pressure (2,750 psia) at any time during the transient.

(4) The pressurizer does not go water solid at any time during the transient.

15.2.2.2 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the SG lags behind the core power generation until the SG 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 would eventually result in DNB, an RCS overpressure condition, or the pressurizer filled with liquid. Therefore, the reactor protection system is designed to terminate any such transient before the DNBR falls below the safety analysis limit values, the RCS pressure exceeds 110 percent of the design value, or the pressurizer becomes filled with liquid.

The automatic features of the reactor protection system that ensure these limits are not exceeded following the postulated accident include the following:

(1) The power range neutron flux instrumentation actuates a reactor trip if two-out-of-four channels exceed a high flux or a positive flux rate high setpoint.

(2) The reactor trip is actuated if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint.

(3) The reactor trip is actuated if any two-out-of-four ΔT channels exceed an overpower ΔT setpoint.

(4) A high pressurizer pressure reactor trip actuated from any two-out-of-four pressure channels that are set at a fixed point.

(5) A high pressurizer water level reactor trip actuated from any two-out-of-three level channels that are set at a fixed point.

The positive flux rate trip provides adequate protection to ensure that the most limiting RCCA bank withdrawal event does not result in the peak RCS pressure exceeding 110 percent of the design limit. The positive flux rate trip setpoint and response time that are credited in the evaluation of this event are listed in Table 15.1-2. Various reactor trips (e.g., High Neutron Flux) may also be credited to prevent RCS overpressurization during an RCCA bank withdrawal event.

Reference 18 documents a generic and conservatively bounding evaluation that has been performed to ensure that pressurizer overfill conditions are not a concern for this event. The evaluation demonstrates that the pressurizer water level high trip prevents a pressurizer overfill condition for those RCCA bank withdrawal events that are very slow and do not generate any other automatic protection signal. The pressurizer water level high trip response time is listed as N/A with the note indicating that the evaluation results are insensitive to the assumed response time.

The manner in which the combination of overpower and overtemperature ΔT trips provide fuel cladding protection over the full range of RCS conditions is described in Chapter 7 and Section 15.1.5.

15.2.2.3 Analysis of Effects and Consequences

This transient is analyzed by the LOFTRAN (Reference 4) code. This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperatures (including 2.5°F for SG fouling) and nominal reactor coolant pressure are assumed.

In order to obtain conservative results, the following assumptions are made:

(1) Reactivity Coefficients - two cases are analyzed:

(a) Minimum reactivity feedback. A positive moderator coefficient of reactivity of +5 pcm/°F is assumed. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

(b) Maximum reactivity feedback. A conservatively large positive moderator density coefficient of 0.43 Δk/gm/cc is assumed. A large (in absolute magnitude) negative Doppler power coefficient is assumed.

(2) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

(3) The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

(4) The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

15.2.2.4 Results

Figures 15.2.2-1 and 15.2.2-2 show the response of neutron flux, pressure, average coolant temperature, and DNBR (thimble cell) due to a rapid RCCA withdrawal starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The response of neutron flux, pressure, average coolant temperature, and DNBR (thimble cell) for a slow control rod assembly withdrawal from full power is shown in Figures 15.2.2-3 and 15.2.2-4. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is never less than the safety analysis limit values.

Figure 15.2.2-5 shows the minimum DNBR (thimble cell) as a function of reactivity insertion rate from initial full power operation for the minimum and for the maximum reactivity feedbacks. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip channels. The minimum DNBR is never less than the safety analysis limit values.

Figures 15.2.2-6 and 15.2.2-7 show the minimum DNBR (thimble cell) as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power, respectively. The results are similar to the 100 percent power case, except that as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the safety analysis limit values.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip. Referring to Figure 15.2.2-7, for example, it is noted that:

(1) For reactivity insertion rates above ~30 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

(2) The Overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured RCS average temperature and pressure. It is important to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increase.

(3) For reactivity insertion rate below ~30 pcm/sec the Overtemperature ΔT trip terminates the transient.

For reactivity insertion rates between ~30 pcm/sec and ~7 pcm/sec the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

(4) For reactivity insertion rates less than ~7 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the SG safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the RCS, sharply decreases the rate of increase of RCS average temperature. This decrease in rate of increase of the average RCS temperature during the transient is accentuated by the lead-

lag compensation causing the Overtemperature ΔT trip setpoint to be reached later with a resulting lower minimum DNBR.

Figures 15.2.2-5 through 15.2.2-7 illustrate minimum DNBRs calculated for minimum and maximum reactivity feedback.

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 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will still 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 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB is not predicted to occur at any time during the RCCA withdrawal at power transient, the ability of the primary 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 this accident is shown in Table 15.2-1. 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.

15.2.2.5 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell). The accompanying DNBR figures for this event (refer to Figures 15.2.2-2 and 15.2.2-4 through 15.2.2-7) reflect the results for the more limiting Standard fuel (limit 1.48/1.44) previously in the core.

(2) The core heat flux is maintained below 118 percent of its nominal value. Thus the peak fuel centerline temperature will remain below the fuel melting temperature (refer to Section 4.4.3.2.7).

(3) The RCS pressure does not exceed 110 percent of design pressure (2,750 psia) at any time during the transient.

(4) The pressurizer does not become water solid during the event.

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; i.e., the minimum value of DNBR is always larger than the safety analysis limit values.

15.2.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION

This section discusses RCCA misoperation that can result either from system malfunction or operator error.

15.2.3.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

15.2.3.2 Identification of Causes and Accident Description

RCCA misoperation accidents include:

(1) One or more dropped RCCAs within the same group

(2) A dropped RCCA bank

(3) Statically misaligned RCCA

Each RCCA has a position indicator channel that 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 two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two 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 that would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA, or RCCA bank, is detected by:

(1) A sudden drop in the core power level as seen by the nuclear instrumentation system (NIS)

(2) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples

(3) Rod at bottom signal

(4) Rod deviation alarm

(5) Rod position indication

Misaligned RCCAs are detected by:

(1) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples

(2) Rod deviation alarm

(3) Rod position indicators

The deviation alarm alerts the operator whenever an individual rod position signal deviates from the other rods in the bank by a preset limit.

During time intervals when the Rod Position Deviation Monitor is inoperable:

(1) Each rod position indicator is determined to be operable by verifying that the demand position indication system and the digital rod position indication system agree within 12 steps at least once per four hours.

During time intervals when the rod insertion limit monitor is inoperable, the individual rod positions are verified to be within insertion limits at least once per four hours.

If one or more rod position indicator channels should be out of service, detailed operating instructions are followed to ensure the alignment of the nonindicated RCCAs. The operator is also required to take action as required by the Technical Specifications.

15.2.3.3 Analysis of Effects and Consequences

The accident is analyzed with the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperature and nominal reactor coolant pressure are assumed.

Method of Analysis

(1) One or More Dropped RCCAs from the Same Group

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

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 transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code (refer to Section 1.6.1, Item 28 and Section 4.4.3). The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 10.

(2) Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in Reference 10, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

(3) Statically Misaligned RCCA

Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC code (refer to Section 1.6.1, Item 28 and Section 4.4.3) to calculate the DNBR. The analysis examines the case of the worst rod withdrawn from control bank D inserted at the insertion limit with the reactor initially at full power. The analysis assumes this incident to occur at beginning of life (BOL) or the time in core life which this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.2.3.4 Results

(1) One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly.

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 15.2.3-1 and 15.2.3-2 show a typical transient response to a dropped RCCA(s) in automatic control. Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 10. In all cases, the minimum DNBR remains above the safety analysis limit value.

Following plant stabilization, the operator may manually retrieve the RCCA(s) by following approved operating procedures.

(2) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm. The core is not adversely affected during the insertion period since power is decreasing rapidly. The dropped RCCA bank transient will proceed as described in the previous section for one or more dropped RCCA(s), except the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

(3) 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 insertion limits in the Technical Specifications 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 the minimum DNBR and peaking factor limits. The full power insertion limits 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 safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values 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 safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, 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 rod 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.

15.2.3.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) For all cases of RCCA misoperation, the DNBR remains greater than the Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell); therefore, the DNB design criterion is met.

15.2.4 UNCONTROLLED BORON DILUTION

15.2.4.1 Acceptance Criteria

(1) There is ample/adequate time for the operator to mitigate a boron dilution event.

15.2.4.2 Identification of Causes and Accident Description

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

The opening of the primary water makeup control valves provides makeup to the RCS that can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

The rate of addition of unborated makeup water to the RCS when it is not at pressure is limited by the capacity of the primary water supply pumps. The maximum net addition rate in this case is 200 gpm, which is based on a conservative evaluation of two primary water pumps operating in parallel through a common flow path.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flowrates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required:

(1) The operator must change from the automatic makeup mode to the dilute mode

(2) The operator must select start to initiate system start

Excluding either step would prevent dilution.

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

Consistent with the DCPP licensing basis, the acceptance criteria of meeting a minimum time before loss of SDM from the start of dilution for Modes 2-5 is 15 minutes; for Mode 6, it is 30 minutes.

In order to meet the acceptance criteria for Mode 4 on RHR and Mode 5 (for both filled and mid-loop operation), the analysis defines a required minimum critical boron concentration ratio (Cbi/Cbc) that must be confirmed on a reload basis. This is a ratio of the initial boron concentration (Cbi) in the RCS to the boron concentration at which shutdown margin is lost (Cbc). Consistent with this approach, a minimum Cbi/Cbc ratio is also defined for Mode 3 and Mode 4 with one RCP in operation, in order to provide additional margin. These limits are evaluated for the core reloads of both units as part of the normal Restart Safety Analysis Checklist process. If analysis shows that these ratios will be violated for future reload cycles, administrative and/or operating procedures will need to be revised to ensure that these limits are maintained. In such case, the Core Operating Limits Report must be revised at that time to specify either an increased SDM requirement or the required minimum Cbi/Cbc ratio(s) directly.

15.2.4.3 Analysis of Effects and Consequences

15.2.4.3.1 Method of Analysis

To cover all phases of plant operation, boron dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation is considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident.

15.2.4.3.2 Dilution during Refueling

During refueling the following conditions exist:

(1) One RHR pump is operating to ensure continuous mixing in the reactor vessel.

(2) The seal injection water supply to the RCPs is typically isolated for the purpose of performing RCP maintenance.

(3) Boric acid supply to the suction of the charging pumps is available for the addition of boric acid to the RCS. Alternatively, boric acid supply may be lined up to the suction of the SI pumps when all the reactor vessel head bolts are fully detensioned.

(4) The boron concentration in the refueling water is greater than or equal to 2000 ppm, corresponding to a shutdown margin of at least 5 percent Δk with all RCCAs in; periodic sampling ensures that this concentration is maintained.

(5) Neutron sources are installed in the core and the source range detectors outside the reactor vessel are active and provide an audible count rate. *During initial core loading, BF_3 detectors are installed inside the reactor vessel and are connected to instrumentation giving audible count rates to provide direct monitoring of the core.* (Historical)

(6) A minimum water volume in the RCS of 3462 cubic feet is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the RHR loop.

(7) A maximum dilution flow of 200 gpm and uniform mixing are assumed.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation and the high flux at shutdown alarm in the control room. Count rate will increase with the subcritical multiplication factor during boron dilution.

If a SI pump is used for boration, it is aligned to take suction from the refueling water storage tank (RWST) and discharge to the cold legs of the RCS, and the boundary valves from the CVCS to the SIS are closed. These requirements ensure no new dilution flowpaths are introduced when using the SIS boration flowpath.

15.2.4.3.3 Dilution during Cold Startup

In this mode, the plant is being taken into or out of refueling or hot shutdown. Typically, the plant is maintained in the cold shutdown mode when reduced RCS inventory is necessary, ambient temperatures are required to address various plant issues, or as a result of a Technical Specification action statement. The water level can be dropped to the mid-plane of the hot leg for maintenance work that requires the SGs to be drained. The plant is maintained in cold shutdown at the beginning of the cycle for start-up testing of certain systems and components. Conditions used for the analysis are as follows:

(1) A maximum dilution flow of 200 gpm, limited by the capacity of two primary water makeup pumps operating in parallel through a common flow path, is considered.

(2) A minimum RCS water volume of 4690 cubic feet is used, corresponding to the active RCS volume excluding the pressurizer and the reactor vessel upper head. A minimum RCS water volume of 3462 cubic feet is also considered for mid-loop operation.

(3) The required operator action time for this mode is 15 minutes. To meet this requirement, a minimum critical boron concentration ratio must be maintained to ensure sufficient time is available from the initiation of the dilution event for the operators to act to prevent a loss of shutdown margin and preclude criticality.

The analysis determined that the minimum critical boron concentration ratio to meet the required operator action is 1.128 for the most limiting case, assuming mid-loop operation.

15.2.4.3.4 Dilution during Hot Shutdown

In this mode, the plant is being taken into or out of cold shutdown or hot standby. The plant is maintained in this mode at the beginning of the cycle for startup testing of certain systems and components. Throughout the cycle, the plant may enter hot shutdown if plant issues arise requiring a plant shutdown or as a result of a Technical Specification action statement. In hot shutdown, mixing of the RCS is provided by either the RHR system or a single RCP, depending on system pressure and temperature. Conditions used for the analysis are as follows:

(1) A maximum dilution flow of 200 gpm, limited by the capacity of two primary water makeup pumps operating in parallel through a common flow path, is considered.

(2) A minimum RCS water volume of 9365 cubic feet is used, corresponding to the active RCS volume excluding the pressurizer and the reactor vessel upper head with one RCP in operation. A reduced minimum RCS water volume of 4690 cubic feet is considered for the case with no RCPs operating and mixing provided by the RHR system.

(3) The required operator action time for this mode is 15 minutes. To meet this requirement, a minimum critical boron concentration ratio must be maintained to ensure sufficient time is available from the initiation of the dilution event for the operators to act to prevent a loss of shutdown margin and preclude criticality.

The analysis determined that the minimum critical boron concentration ratio to meet the required operator action time is 1.101 for the most limiting case, assuming RHR system operation only.

15.2.4.3.5 Dilution during Hot Standby

In this mode, the plant is being taken into or out of hot shutdown or startup. The plant is maintained in hot standby at the beginning of cycle for startup testing of certain systems and components, and to achieve plant heatup before entering the startup mode and going critical. During cycle operation, the plant will enter this mode following a reactor trip or as a result of a Technical Specification action statement. During hot standby, not all RCPs may be in operation. Rod control is in manual and the rods may be partially or completely withdrawn. The more limiting hot standby dilution scenario is with the control rods not withdrawn and the reactor shut down by boron to the Technical Specifications minimum requirement for this mode. Conditions used for the analysis are as follows:

(1) A maximum dilution flow of 200 gpm, limited by the capacity of two primary water makeup pumps operating in parallel through a common flow path, is considered.

(2) A minimum RCS water volume of 9365 cubic feet is used, corresponding to the active RCS volume excluding the pressurizer and the reactor vessel upper head with at least one RCP in operation.

(3) The required operator action time for this mode is 15 minutes. To meet this requirement, a minimum critical boron concentration ratio must be maintained to ensure sufficient time is available from the initiation of the dilution event for the operators to act to prevent a loss of shutdown margin and preclude criticality.

The analysis determined that the minimum critical boron concentration ratio to meet the required operator action time is 1.059.

15.2.4.3.6 Dilution during Startup

In this mode, the plant is being taken into or out of hot standby or power operation. The RCS is filled with borated water from the blender during vacuum fill. Conditions used for the analysis are as follows:

(1) The initial boron concentration is modeled as 2000 ppm boron, which is conservative.

(2) Core monitoring is by external BF_3 detectors.

(3) Mixing of the reactor coolant is accomplished by operation of all four RCPs.

(4) The High Flux at Shutdown, NIS Source Range High Flux ½ Reactor Trip, and Reactor Trip Initiated alarms are available to warn the operator of the transient.

(5) A maximum dilution flow of 200 gpm, limited by the capacity of the two primary water makeup pumps operating in parallel through a common flow path, is considered.

(6) The volume of the reactor coolant is 9883 cubic feet, which is the minimum active volume of the RCS excluding the pressurizer.

The analysis determined that to maintain a minimum critical boron concentration ratio (initial boron concentration at the most reactive time in core life to the maximum critical boron concentration) of 1.25, the operator action time is 61.6 minutes.

15.2.4.3.7 Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging pumps. The effective reactivity addition rate for the reactor at full power and for a boron dilution flow of 262 gpm is shown as a function of RCS boron concentration in Figure 15.2.4-1. This figure includes the effect of increasing boron worth with dilution. The reactivity rate used in the analysis is 1.752 x 10^{-5} Δk/sec based on a conservatively high value for the expected boron concentration (1600 ppm) at power.

15.2.4.4 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

For dilution during refueling and startup, the analysis assumes the following.
In refueling, cold shutdown, hot shutdown, hot standby, and startup, the reactor operators are relied upon to detect and recover from an inadvertent boron dilution event. Numerous alarms from the CVCS, the reactor makeup water system, and the NIS are available to provide assistance to the reactor operators in the detection of an inadvertent boron dilution event. Analysis has demonstrated that the reactor operators have at least 15 minutes from initiation of the dilution event in cold shutdown, hot shutdown, hot standby, and startup, and at least 30 minutes in refueling, to terminate the dilution event and initiate boration of the RCS prior to the loss of the available shutdown margin.

For dilution during full power operation:

(1) With the reactor in automatic control at full power, the power and temperature increase from boron dilution results in the insertion of the RCCAs and a decrease in shutdown margin. Continuation of dilution and RCCA insertion would cause the assemblies to reach the minimum limit of the rod insertion monitor in approximately 4.7 minutes, assuming the RCCAs to be initially at a position providing the maximum operational maneuvering band consistent with maintaining a minimum control band incremental rod worth. Before reaching this point, however, two alarms would be actuated to warn the operator of the accident condition. The first of these, the low insertion limit alarm, alerts the operator to initiate normal boration.

    The other, the low-low insertion limit alarm, alerts the operator to follow emergency boration procedures. The low alarm is set sufficiently above the low-low alarm to allow normal boration without the need for emergency procedures. If dilution continues after reaching the low-low alarm, it takes approximately 15.0 minutes after the low-low alarm before the total shutdown margin (assuming 1.6 percent, consistent with the Technical Specifications) is lost due to dilution. Therefore, adequate time is available following the alarms for the operator to determine the cause, isolate the primary grade water source, and initiate boration.

(2) With the reactor in manual control and if no operator action is taken, the power and temperature rise will cause the reactor to reach the high neutron flux or overtemperature ΔT trip setpoint. The boron dilution

accident in this case is essentially identical to a RCCA withdrawal accident at power. The maximum reactivity insertion rate for boron dilution is shown in Figure 15.2.4-1 and is seen to be within the range of insertion rates analyzed for a RCCA withdrawal accident. Reactor trip will occur approximately 40 seconds after event initiation. If dilution were to continue after the reactor trip, there would still be approximately 14.5 minutes left after a reactor trip for the operator to determine the cause of dilution, isolate the primary grade water sources, and initiate reboration before the reactor can return to criticality assuming a 1.6 percent shutdown margin at the beginning of dilution. Therefore, there is ample time available (approximately 40 seconds to reactor trip plus 14.5 minutes after a reactor trip).

15.2.5 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

During review of this accident, it was identified that the complete loss of flow reanalysis (Westinghouse calculation note CN-TA-12-29) without undervoltage and underfrequency reactor trips credits the RCL low flow reactor trip as protection. This reactor trip is also credited in the partial loss of flow event. Since the complete loss of flow reanalysis assumes all four RCPs coasting down and the partial loss of flow analysis assumes two out of four RCPs coast down, the complete loss of flow accident, as discussed in Section 15.3.4 is bounding.

15.2.5.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

15.2.5.2 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a RCP or from a fault in the power supply to the pump. 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 DNB with subsequent fuel damage if the reactor is not tripped promptly.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip that is actuated by two-out-of-three low flow signals in any RCL. Above approximately 35 percent power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 and 35 percent power

(Permissive 7 and Permissive 8), low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive 7.

A reactor trip on RCP breakers open is provided as a backup to the low flow signals. Above Permissive 7, a breaker open signal from any two pumps will actuate a reactor trip. Reactor trip on RCP breakers open is blocked below Permissive 7.

Normal power for the RCPs is supplied through buses connected through transformers to the generator. Two RCPs are on each bus. When a generator trip occurs, the buses are automatically transferred to a power source 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 or mechanical faults that require immediate tripping of the generator from the network, the generator remains connected to the network for approximately 30 seconds. The RCPs remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

15.2.5.3 Analysis of Effects and Consequences

15.2.5.3.1 Method of Analysis

The following case has been analyzed:

(1) All loops operating, two loops coasting down.

This transient is analyzed by three digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calculate the time of reactor trip, based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (refer to Section 1.6.1, Item 28 and Section 4.4.3) is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical and thimble cells for Standard fuel, which bound VANTAGE 5 fuel.

15.2.5.3.2 Initial Conditions

The accident is analyzed using the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperature (including 2.5 °F for SG fouling) and nominal reactor coolant pressure are assumed.

15.2.5.3.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (refer to Table 15.1-4). The total integrated Doppler reactivity from 0 to 100 percent power is assumed to be -0.016 Δk.

The most positive moderator temperature coefficient (+5 pcm/°F) is assumed since this results in the maximum hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

15.2.5.3.4 Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each RCL 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.

15.2.5.4 Results

The calculated sequence of events is shown in Table 15.2-1. Figures 15.2.5-1 through 15.2.5-4 show the core flow coastdown, the loop flow coastdown, the heat flux coastdown, and the nuclear power coastdown. The minimum DNBR is not less than the safety analysis limit value. A plot of DNBR vs. time is given in Figure 15.2.5-5 for the most limiting thimble cell for Standard fuel, which bounds VANTAGE 5 fuel.

15.2.5.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell). The accompanying DNBR figure for this event (refer to Figure 15.2.5-5) reflects the results for the more limiting Standard fuel (limit 1.48/1.44) previously in the core.

The analysis shows that the DNBR will not decrease below the safety analysis limiting values at any time during the transient. Thus no core safety limit is violated.

15.2.6 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

In accordance with the Technical Specifications, DCPP operation during startup and power operation with less than four loops is not permitted. This analysis is presented for completeness.

15.2.6.1 Identification of Causes and Accident Description

If a plant is operating with one pump out of service, there is reverse flow through the 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 SG in the inactive loop is not isolated, there is a temperature drop across the SG 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.

Starting of an idle RCP 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 causes a rapid reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined at the beginning of this chapter.

Should the startup of an inactive RCP at an incorrect temperature occur, the transient will be 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.

15.2.6.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. The LOFTRAN Code (Reference 4) is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Reference 3) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC Code (Reference 9) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

(1) Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. A 25 percent maximum steady state power level including appropriate allowances for calibration and instrument errors is assumed, however DCPP is not allowed to be at power with an inactive loop. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

(2) *Following the start of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value.*

(3) *A conservatively large (absolute value) negative moderator coefficient associated with the end of life (EOL).*

(4) *A conservatively low (absolute value) negative Doppler power coefficient is used.*

(5) *The initial RCL flows are at the appropriate values for one pump out of service.*

(6) *The reactor trip is assumed to occur on low coolant flow when the power range neutron flux exceeds the P-8 setpoint, which has been reset for N-1 loop operation. The P-8 setpoint is conservatively assumed to be 84 percent of rated power, which corresponds to the nominal N-1 loop operation setpoint plus 9 percent for nuclear instrumentation errors.*

15.2.6.3 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.2.6-1 through 15.2.6-5. 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 values.

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 a) the increase in reactor coolant flow, and b) 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 SG 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 in Figure 15.2.6-1.

The calculated sequence of events for this accident is shown in Table 15.2-1. The transient results illustrated in Figures 15.2.6-1 through 15.2.6-5 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.2.6.4 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the safety analysis DNBR limit values; thus, no fuel or cladding damage is predicted.

15.2.7 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP

15.2.7.1 Acceptance Criteria

The following are the relevant specific acceptance criteria.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

(2) The RCS pressure does not exceed 110 percent of design pressure (2,750 psia) at any time during the transient.

(3) The MSS pressure does not exceed 110 percent of design pressure (1,210 psia) at any time during the transient.

15.2.7.2 Identification of Causes and Accident Description

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the RCPs. The case of LOOP is analyzed in Section 15.2.9.

For a turbine trip, the reactor would be tripped directly (unless it is below the P-9 setpoint) from a signal derived from the turbine autostop oil pressure and turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, SG level would be maintained by the auxiliary feedwater (AFW) system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the electrical demand of plant auxiliaries.

In the event the 10 percent atmospheric dump valves fail to open following a large loss of load (LOL), the SG 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 SG shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves (PSVs) and SG safety valves are, however, sized to protect the RCS and SG against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves (PORVs), automatic RCCA control, or direct reactor trip on turbine trip.

The SG safety valve capacity is sized to remove the steam flow at the engineered safeguards design rating (105 percent of steam flow at rated power) from the SG without exceeding 110 percent of the steam system design pressure. The PSV capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the SG safety valves. The PSVs are then able to maintain the RCS pressure within 110 percent of the RCS design pressure without direct or immediate reactor trip action.

A more complete discussion of overpressure protection can be found in Reference 8.

15.2.7.3 Analysis of Effects and Consequences

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This is done to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump actuation. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for AFW (except for long-term recovery) to mitigate the consequences of the transient.

Total LOL transients are analyzed for DNB and overpressure concerns. The LOFTRAN computer program (refer to Section 15.1) is used to analyze the total LOL transients for the DNB concern. The RETRAN-02 computer program (refer to Section 15.1) is used to analyze the transients for the overpressure concern. Both programs simulate the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The programs compute pertinent variables, including temperatures, pressures, and power level.

The following assumptions are used in the LOFTRAN analysis for the DNB concern.

(1) Initial Operating Conditions

The accident is analyzed using the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperature (including 2.5°F for SG tube fouling) and nominal reactor coolant pressure are assumed.

(2) Moderator and Doppler Coefficients of Reactivity

The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback for EOL cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback for BOL cases assume a minimum moderator temperature coefficient and the least negative Doppler coefficient.

(3) Reactor Control

From the standpoint of the maximum pressures attained, 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.

(4) Steam Release

No credit is taken for the operation of the steam dump system or SG power-operated relief valves. The SG pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

(5) Pressurizer Spray and PORVs

Two cases for both BOL and EOL are analyzed using the LOFTRAN computer program.

(a) Full credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure. Safety valves are also operable.

(b) No credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure. Safety valves are operable.

(6) Feedwater Flow

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

The following assumptions are used in the RETRAN-02 analysis for the overpressure concern only.

(1) Initial Operating Conditions

The accident analysis assumes: maximum core power; maximum Tavg, and minimum operating RCS pressure 2189.7 psia is used in the analysis, which includes 60 psi uncertainty.

(2) Moderator and Doppler Coefficients of Reactivity

BOL minimum reactivity feedback is modeled assuming the most positive moderator temperature coefficient and the least negative Doppler temperature coefficient.

(3) Steam Release

No credit is taken for secondary heat removal from the steam dump system. Only the PSVs and the main steam safety valves (MSSVs) are credited for overpressure protection.

(4) Pressurizer Pressure Control

Since the total LOL overpressure transients result in higher peak RCS and SG pressures at BOL, two cases are analyzed using the RETRAN-02 computer program for BOL only.

(a) For the peak secondary side pressure case assumes pressurizer pressure control. This delays the reactor trip and maximizes the heat transfer to the SGs. Safety valves are also operable.

(b) For the peak RCS pressure case, no credit is taken for pressurizer pressure control, which maximizes the peak RCS pressure for the event. Safety valves are operable.

(5) Feedwater Flow

The turbine stop valves and feedwater control valves are assumed to close instantaneously at the initiation of the event to maximize the duration of the primary to secondary heat imbalance. The AFW system is conservatively not credited and assumed unavailable for decay heat removal during the event.

(6) MSSVs

The MSSV setpoints are assumed to be at their maximum 3 percent drift values. The RETRAN MSSVs are modeled to provide zero to full flow as a linear function from the lift setpoint to the full open 3 percent accumulation value.

(7) PSV Water Loop Seal

All PSVs have been converted to a steam-seat design and condensate in the loop is now continuously drained back to the pressurizer, thereby eliminating the water loop seal. Even though the water loop seal has been eliminated, the resulting benefit is not credited in the analysis. The presence of a water loop seal delays the opening of the PSV. The loop seal water starts to leak out from the safety valve when the safety valve setpoint is reached. However, no pressure is relieved from the pressurizer until the loop seal water is completely purged, after which the safety valve pops full open in less than 0.1 second. The loop seal water purge time of 1.272 seconds was used in the analysis.

(8) Maximize Reactor Power

It is conservative to maximize the reactor power. Therefore, the reactor trip due to high neutron flux is not credited in the analysis.

In all cases for DNB and overpressure concerns 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.

15.2.7.4 Results

The transient responses for a total LOL from full power operation are shown for four cases the DNB concern is evaluated at BOL and EOL with pressure control since this condition bounds the cases without pressure control and overpressure concern is evaluated at BOL with and without pressure control (refer to Figures 15.2.7-1 through 15.2.7-4 and 15.2.7-9 through 15.2.7-12.

Figures 15.2.7-1 and 15.2.7-2 show the transient responses for the total loss of steam load at BOL, for the DNB concern, assuming full credit for the pressurizer spray and pressurizer PORVs. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip channel. The minimum DNBR is (thimble cell) well above the limit value.

Figures 15.2.7-3 and 15.2.7-4 show the responses for the total LOL at EOL, for the DNB concern, assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as in the above case. As a result of the maximum reactivity feedback at EOL, no reactor protection system trip setpoint is reached. Because main feedwater is assumed to be lost, the reactor is tripped by the low-low SG water level trip channel. The DNBR (thimble cell) increases throughout the transient and never drops below its initial value. The PSVs are not actuated in these transients.

Figures 15.2.7-9 and 15.2.7-10 show the typical transient responses for the total LOL at BOL for the RCS overpressure concern. No credit is taken for the pressurizer spray, pressurizer PORVs, or steam dump. The pressurizer and MSSVs are modeled as described in assumptions 6 and 7. The initial pressurizer pressure includes the pressurizer pressure uncertainty to maximize the peak pressure. The reactor is tripped on the high pressurizer pressure signal. This case results in the highest RCS peak pressure among all cases. The peak RCS pressure is below 110 percent of the design value.

Figures 15.2.7-11 and 15.2.7-12 show the typical transient responses for the total LOL at BOL for the secondary side overpressure concern, assuming full credit for the pressurizer spray and the pressurizer PORVs. No credit is taken for the steam dump. The models for the pressurizer and MSSVs and the initial pressurizer pressure are the same as those used in the above case. The reactor trip due to high neutron flux is not credited in order to maximize the peak SG pressure. The reactor is tripped on the high pressurizer pressure signal. This case results in the highest SG peak pressure among all cases. The peak SG pressure is below 110 percent of the design value.

Reference 8 presents additional results for a complete loss of heat sink including loss of main feedwater. This report shows the overpressure protection that is afforded by the pressurizer and SG safety valves.

Technical Specification 3.7.1 establishes reduced plant operating power limits for off normal conditions when one or more MSSVs are inoperable to ensure a LOL event does not result in overpressurization of the SGs. When two or more MSSVs are inoperable per SG loop, the reduced power limits are established using a conservative energy balance algorithm established in the Westinghouse Nuclear Safety Advisory Letter NSAL-94-001 as documented in Reference 21. To evaluate off normal plant operation with a single inoperable MSSV on one or more SG loops, an additional spectrum of LOL analyses are performed as documented in Reference 22. These analyses use the RETRAN-02W code to analyze the BOL LOL overpressure case as discussed in this section and which represents the limiting case for challenging the SG peak pressure limit. These analysis results, as summarized in the Technical Specification Bases 3.7.1, credit the overtemperature ΔT reactor trip to demonstrate that the specified reduced operating power limit ensures that the available relief capacity with one inoperable MSSV per loop maintains the peak SG pressure below 110 percent of the design value.

15.2.7.5 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell). The accompanying DNBR figures for this event (refer to Figures 15.2.7-1 and 15.2.7-3) reflect the results for the more limiting Standard fuel (limit 1.48/1.44) previously in the core.

(2) The calculated RCS pressure (2723 psia) does not exceed 110 percent of design pressure (2,750 psia) at any time during the transient.

(3) The calculated MSS peak pressure (1203 psia) does not exceed 110 percent of design pressure (1,210 psia) at any time during the transient.

Results of the analyses, including those in Reference 8, show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the MSS. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the DNBR will be maintained above the safety analysis limit values. Thus, no core safety limit will be violated.

15.2.8 LOSS OF NORMAL FEEDWATER

15.2.8.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The pressurizer does not go water solid at any time during the transient.

15.2.8.2 Identification of Causes and Accident Description

A loss of normal feedwater (resulting from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater were not supplied to the SGs, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. A significant loss of water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the SG heat transfer capability is reduced, the primary system conditions never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

(1) Reactor trip on low-low water SG level in any SG

(2) Two motor-driven AFW pumps (MDAFWPs) that are started on:

(a) Low-low SG water level in any SG

(b) Trip of both main feedwater pumps

(c) Any SI signal

(d) LOOP (automatic transfer to diesel generators)

(e) Manual actuation

(3) One turbine-driven auxiliary feedwater pump (TDAFWP) that is started on:

(a) Low-low SG water level in any two SGs

(b) Undervoltage on both RCP buses

(c) Manual actuation

The MDAFWPs are connected to Class 1E buses and are supplied by the diesels if a LOOP occurs. The turbine-driven pump utilizes steam from the secondary system and exhausts it to the atmosphere. The controls are designed to start both types of pumps within 1 minute even if a loss of all ac power occurs simultaneously with loss of normal feedwater. The AFW pumps take suction from the condensate storage tank for delivery to the SGs.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored energy and residual decay heat, and RCP heat thus preventing either overpressurization of the RCS or liquid relief through the pressurizer PORVs or safety valves.

15.2.8.3 Analysis of Effects and Consequences

A detailed analysis using the RETRAN-02W code (Reference 19) is performed in order to determine the plant transient following a loss of normal feedwater. The code describes the plant neutron kinetics, RCS including factors that influence the natural circulation, pressurizer, SGs, and feedwater system, and compute pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

(1) Reactor trip occurs on SG low-low level at 8 percent of narrow range span (NRS).

(2) The plant is initially operating at 102 percent of the NSSS rating, including a conservatively large RCP heat of 20 MWt.

(3) Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The ANSI/ANS-5.1-1979 + 2σ was used for calculation of residual decay heat levels.

(4) The AFW system is actuated by the low-low SG water level signal.

(5) The limiting single failure in the AFW system occurs (turbine-driven pump failure). The AFW system is assumed to supply a total of 600 gpm to all four SGs from the motor-driven pumps.

(6) The pressurizer sprays and heaters are assumed operable. This maximizes the peak transient pressurizer water volume. Sensitivity analyses determined that it is conservative to assume that the PORVs are inoperable (Reference 20).

(7) Secondary system steam relief is achieved through the self-actuated safety valves. The MSSVs are assumed to begin to lift 3 percent above the set pressure with a 5 psi accumulation to full open. Note that steam relief will, in fact, be through the 10 percent atmospheric dump valves or 40 percent condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

(8) The initial reactor coolant average temperature is 5.5°F lower than the nominal value. The initial pressurizer pressure is 60 psi above the nominal value.

(9) The minimum steam generator tube plugging (SGTP) of 0 percent was assumed.

(10) The initial feedwater temperature is assumed to be 435°F.

15.2.8.4 Results

Figures 15.2.8-1 through 15.2.8-3 show plant parameters following a loss of normal feedwater at the conditions associated with Unit 2, which were determined to be limiting when compared to Unit 1. Figure 15.2.8-2 shows the pressurizer pressure as a function of time.

Following the reactor and turbine trip from full load, the water level in the SGs will fall due to the reduction of SG void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low SG level trip, the MDAFWPs are automatically started, reducing the rate of water level decrease.

The capacity of the MDAFWPs combined with the available secondary inventory is capable of dissipating the core residual heat without liquid water relief from the RCS PORVs or safety valves.

From Figure 15.2.8-2 it can be seen that at no time is there liquid relief from the pressurizer. If the AFW delivered is greater than that of two motor-driven pumps, the initial reactor power is less than 102 percent of the NSSS rating, or the SG water level in one or more SGs is above the low-low level trip point at the time of trip, then the results for this transient will be less limiting.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.8-1 through 15.2.8-3, the plant approaches a stabilized condition following reactor trip and AFW initiation. Plant procedures may be followed to further cool down the plant.

15.2.8.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) The pressurizer does not become water solid during the event as shown in Figure 15.2.8-2.

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 AFW capacity is such that the pressurizer does not become water solid, which ultimately precludes reactor coolant liquid relief from the pressurizer relief or safety valves. This ensures a Condition III or IV event will not be generated.

15.2.9 LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES

15.2.9.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The pressurizer does not go water solid at any time during the transient.

15.2.9.2 Identification of Causes and Accident Description

During a complete LOOP and a turbine trip there will be loss of power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc.

The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below:

(1) Plant Instrument Class I instruments are supplied by emergency power sources.

(2) As the steam system pressure rises following the trip, the steam system PORVs are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the PORVs are not available, the SG self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

(3) As the no-load temperature is approached, the steam system PORVs (or the self-actuated safety valves, if the PORVs are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

(4) The emergency diesel generators (EDGs) started on loss of voltage on the plant emergency buses begin to supply plant Class 1E loads.

The AFW system is started automatically as discussed in the loss of normal feedwater analysis. The steam-driven AFW pump utilizes steam from the secondary system and exhausts to the atmosphere. The MDAFWPs are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the SGs.

Upon the loss of power to the RCPs, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the RCLs.

15.2.9.3 Analysis of Effects and Consequences

A detailed analysis using the RETRAN-02W code (Reference 19) is performed in order to determine the plant transient following LOOP. The code describes the plant neutron kinetics, RCS including factors that influence the natural circulation, pressurizer, SGs, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions differing from those in a loss of normal feedwater are:

(1) No credit is taken for immediate response of CRDMs caused by a LOOP.

(2) RCP coastdown to natural circulation conditions is assumed after reactor trip (i.e., rod motion), which is more limiting for long-term heat removal capability.

(3) The initial feedwater temperature is assumed to be 425°F.

(4) A nominal RCP heat input of 14 MWt.

15.2.9.4 Results

The time sequence of events for the accident at the conditions associated with Unit 2, which were determined to be limiting, is given in Table 15.2-1. This event is bounded by the complete-loss-of-flow analysis (refer to Section 15.3.4), in terms of minimum DNBR (Reference 23). Therefore, this event is not analyzed for DNB concerns, but rather, for the long-term heat removal capability. After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The RETRAN-02W code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

Figures 15.2.9-1 through 15.2.9-3 show plant parameters following a LOOP at the conditions associated with Unit 2, which were determined to be limiting. Figure 15.2.9-2 shows the pressurizer water volume as a function of time.

15.2.9.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) The pressurizer does not become water solid during the event as shown in Figure 15.2.9-2.

Results of the analysis show that, for the LOOP to the station auxiliaries event, all safety criteria are met. Since the DNBR remains above the safety analysis limit, the core is not adversely affected. AFW capacity is sufficient to prevent the pressurizer from becoming water solid, which ultimately precludes reactor coolant liquid relief from the pressurizer relief and safety valves; this assures that the RCS is not overpressurized. This ensures that a Condition III or IV event will not be generated.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long-term heat removal capability exists following RCP coastdown to prevent fuel or cladding damage.

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

15.2.10.1 Acceptance Criteria

The following are the relevant specific acceptance criteria:

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

(2) The peak linear heat generation rate does not exceed a value which would cause fuel centerline melt at any time during the transient (refer to Section 4.4.3.2.7).

15.2.10.2 Identification of Causes and Accident Description

Reductions in feedwater temperature or excessive feedwater additions are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (neutron high flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit.

One example of excessive feedwater flow would be a full opening of a main feedwater regulating valve (MFRV) 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 SG. 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. Continuous excessive feedwater addition is prevented by the SG high-high level feedwater isolation and turbine trip.

15.2.10.3 Analysis of Effects and Consequences

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the RETRAN-02W code. This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction. The accident is analyzed with the ITDP and initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperatures and nominal reactor coolant pressure are assumed.

Excessive feedwater addition due to a control system malfunction or operator error that allows a MFRV to open fully is considered. Two conditions are evaluated as follows:

(1) Accidental opening of one MFRV with the reactor just critical at zero load conditions. The feedwater flow increase event at hot zero power conditions is not limiting with respect to DNB concerns and is bounded by the full power event; therefore, the event has not been explicitly analyzed.

(2) Accidental opening of one MFRV at full power (with automatic and manual rod control).

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

(1) For the MFRV accident at full power, one MFRV is assumed to malfunction resulting in a step increase to 250 percent of nominal feedwater flow to one SG.

(2) Coincident with the feedwater flow increase in the faulted loop, the feedwater temperature in all loops decreases approximately 23°F from the nominal full power value. This accounts for the effect of the feedwater passing through the heaters at a higher velocity.

(3) The initial water level in all the SGs is at a conservatively low level.

(4) No credit is taken for the heat capacity of the RCS and SG thick metal in attenuating the resulting plant cooldown.

(5) The feedwater flow resulting from a fully open control valve is terminated by the SG high-high level signal that closes all MFRVs, closes all feedwater bypass valves, trips the main feedwater pumps, and shuts the main feedwater isolation valves (MFIVs).

15.2.10.4 Results

The full power case (EOL, with manual rod control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip and reactor trip is actuated when the SG level reaches the high-high level setpoint. Although turbine trip and subsequent reactor trip are assumed, the results show that the DNBR remains relatively constant prior to the time of reactor trip. This demonstrates that a reactor trip on turbine trip is not needed to protect against DNB, but is assumed as a means to terminate the transient.

Transient results (refer to Figures 15.2.10-1 through 15.2.10-3) show the core heat flux, pressurizer pressure, core T_{avg}, and DNBR (thimble cell), as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. SG level rises until the feedwater is terminated as a result of the high-high SG level trip. The DNBR does not drop below the limit safety analysis DNBR.

15.2.10.5 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) as shown on Figure 15.2.10-3.

(2) Figure 15.2.10-1 shows the total power is maintained below 118 percent of its nominal value. Thus the peak fuel centerline temperature will remain below the fuel melting temperature (refer to Section 4.4.3.2.7).

An excessive feedwater addition at no-load conditions is bounded by the analysis at full power. The DNBRs encountered for excessive feedwater addition at power are well above the safety analysis limit DNBR values.

15.2.11 SUDDEN FEEDWATER TEMPERATURE REDUCTION

15.2.11.1 Acceptance Criteria

The following are the relevant specific acceptance criteria.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

(2) The peak linear heat generation rate does not exceed a value that would cause fuel centerline melt at any time during the transient (refer to Section 4.4.3.2.7).

15.2.11.2 Identification of Causes and Accident Description

A concern was raised during the Unit 1 power ascension test program that an inadvertent actuation of the load transient bypass relay (LTBR) might initiate a transient that exceeds analyzed reactor operating limits. An evaluation performed showed that since the expected feedwater temperature decrease due to inadvertent actuation of the LTBR was significantly less than that of the net load trip, the consequences and events of inadvertent actuation of the LTBR were bounded by the feedwater temperature decrease event.

The automatic load transient bypass (LTB) feature has been eliminated for Unit 1 and Unit 2. Control of the feedwater heater bypass valve has been changed to manual only.

A reduction in feedwater temperature may be caused by an inadvertent manual opening of the feedwater heater bypass valve. This would divert flow around the low pressure feedwater heaters. A consequent maximum 70°F reduction in feedwater temperature to the SGs would occur.

Feedwater temperature may also be reduced during a load rejection trip. The feedwater transient data taken from a 100 percent net load trip test with LTB active showed that a maximum feedwater temperature decrease of 230°F occurred over a 400-second time period. The temperature decrease without LTB is significantly less.

Reductions in temperature of feedwater entering the SGs, if not accompanied by a corresponding reduction in steam flow, would result in an increase in core power and create a greater load demand on the RCS. The net effect on the RCS of a reduction in reactor coolant temperature is similar to the effect of increasing secondary steam flow. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The high neutron flux trip, overtemperature ΔT trip, and overpower ΔT trip act to prevent any power increase that could lead to a DNBR less than the limit value. The reactor may reach a new equilibrium condition at a power level corresponding to the new SG ΔT. A small temperature reduction results in only a small increase in reactor power and does not result in a reactor trip. A larger temperature reduction produces a larger increase in reactor power and may cause a power/temperature mismatch and a reactor trip.

15.2.11.3 Analysis of Effects and Consequences

The accident is analyzed with the ITDP and initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperatures and nominal reactor coolant pressure are assumed.

15.2.11.3.1 Temperature Drops Less than 73°F

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive increase in steam flow event, as discussed in Section 15.2.12. A step load increase of 10 percent from full load was analyzed, and the minimum DNBR for this event was found to be above the safety analysis limit values.

The increase in heat load resulting from a 10 percent increase in load is equivalent to a 73°F drop in feedwater temperature at the SG inlet. Thus a feedwater temperature transient that results in a feedwater temperature drop of 73°F or less at the SG inlets is less severe than the excessive load increase incident presented in Section 15.2.12 and as such does not exceed any safety limits.

15.2.11.3.2 Temperature Drops Greater than 73°F

To address feedwater temperature reductions that exceed 73°F, analyses were performed assuming instantaneous temperature drops of 175°F and 250°F at the SG, with corresponding steam load reductions of 50 percent and 100 percent, respectively. The maximum temperature drop of 250°F was chosen to bound the temperature decrease of 230°F experienced during the net load trip test when the LTBR was actuated in response to a load reduction. In this test, feedwater temperature dropped approximately 230°F over a time period of 400 seconds, which is significantly less severe than the instantaneous drop of 250°F assumed in the analysis. Since LTB has been eliminated, the feedwater temperature drop will be significantly less and is bounded by the instantaneous drop of 250°F assumed in the analysis.

15.2.11.4 Results

Both a 175°F feedwater temperature reduction concurrent with a 50 percent load reduction and a 250°F feedwater temperature reduction concurrent with a full (100 percent) load reduction were analyzed. The analysis shows that the cooldown effects of the large feedwater reduction are more than counteracted by the reduced heat removal resulting from the turbine load reduction, such that the transient causes a heatup of the RCS. As a result, the core power decreases and the DNBR increases during the transient. These cases do not challenge core thermal limits.

15.2.11.5 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell).

(2) The total power is maintained below 118 percent of its nominal value. Thus the peak fuel centerline temperature will remain below the fuel melting temperature (refer to Section 4.4.3.2.7).

All safety criteria are met for credible scenarios of sudden feedwater temperature reduction. Instantaneous feedwater temperature reductions up to 73°F result in an RCS cooldown that is bounded by the analysis of an excessive load increase incident presented in Section 15.2.12. This bounds the maximum feedwater temperature decrease of 70°F that could result from the inadvertent opening of a feedwater heater bypass valve. For feedwater temperature reductions during a load reduction transient, analyses conclude that these cases result in a net RCS heatup and core power decrease, with no significant challenge to the core thermal limits.

15.2.12 EXCESSIVE LOAD INCREASE INCIDENT

15.2.12.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

15.2.12.2 Identification of Causes and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the SG 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.

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

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

(1) Overpower ΔT

(2) Overtemperature ΔT

(3) Power range high neutron flux

15.2.12.3 Analysis of Effects and Consequences

This accident is analyzed using the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. 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:

(1) Reactor control in manual with BOL minimum moderator reactivity feedback

(2) Reactor control in manual with EOL maximum moderator reactivity feedback

(3) Reactor control in automatic with BOL minimum moderator reactivity feedback

(4) Reactor control in automatic with EOL maximum moderator reactivity feedback

For the BOL minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters.

The accident is analyzed using the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, reactor coolant average temperature (plus 2.5°F for SG fouling) and nominal reactor coolant pressure are assumed.

Plant characteristics and initial conditions are further discussed in Section 15.1.

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.

The cases, which assume automatic rod control, are analyzed to ensure that the worst case is presented. The automatic function is not required.

15.2.12.4 Results

The calculated sequence of events for the excessive load increase incident is shown in Table 15.2-1.

Figures 15.2.12-1 through 15.2.12-4 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL 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 EOL 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 limit value.

Figures 15.2.12-5 through 15.2.12-8 illustrate the transient assuming the reactor is in the automatic control mode. Both the BOL minimum and EOL 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 limit value.

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.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Since DNB is not predicted to occur at any time during the excessive load increase transients, the ability of the primary 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.

15.2.12.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell). The accompanying DNBR figures for this event (refer to Figures 15.2.12-2, 15.2.12-4, 15.2.12-6 and 15.2.12-8) reflect the results for the more limiting Standard fuel (limit 1.48/1.44) previously in the core. The figures represent the thimble cell results.

The analysis presented above shows that for a 10 percent step load increase, the DNBR remains above the safety analysis limit values, thereby precluding fuel or cladding damage. The plant reaches a stabilized condition rapidly, following the load increase.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

15.2.13.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient.

15.2.13.2 Identification of Causes and Accident Description

An accidental depressurization of the 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 flowrate 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 PSV. Initially, the event results in a rapidly decreasing RCS pressure, which could reach the hot leg saturation pressure if a reactor trip does not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature essentially constant until the reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor will be tripped by the following reactor protection system signals:

(1) Pressurizer low pressure

(2) Overtemperature ΔT

15.2.13.3 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperature (plus 2.5 °F for SG fouling) and nominal reactor coolant pressure are assumed.

In order to obtain conservative results, the following assumptions are made:

(1) A positive moderator temperature coefficient of reactivity for (+ 7 pcm/°F) BOL operation is assumed in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. These voids would tend to flatten the core power distribution.

(2) A low (absolute value) Doppler-only power coefficient of reactivity such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

15.2.13.4 Results

Figure 15.2.13-1 illustrates the nuclear power transient following the RCS depressurization accident. The nuclear power increases until the time reactor trip occurs on overtemperature ΔT, thus resulting in a rapid decrease in the nuclear power. The time of reactor trip is shown in Table 15.2-1. The pressure decay transient following the accident is given in Figure 15.2.13-2. The resulting DNBR (thimble cell) never goes below the safety analysis limit value as shown in Figure 15.2.13-1.

15.2.13.5 Conclusions

The analysis demonstrates that the acceptance criterion is met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) shown on Figure 15.2.13-1.

The pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against this accident, and the minimum DNBR remains in excess of the safety analysis limit value.

15.2.14 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.14.1 Acceptance Criteria

The following is the relevant specific acceptance criterion.

(1) Minimum DNBR is not less than the applicable Safety Analysis Limit of 1.45 (W-3 DNB correlation, for coolant pressure less than 1,000 psia) at any time during the transient.

15.2.14.2 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the MSS are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses, assuming a rupture of a main steam pipe, are discussed in Section 15.4.

The steam released as a consequence of this accident results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. 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 a reduction of core shutdown margin.

The analysis is performed to demonstrate that the following criterion is satisfied: Assuming a stuck RCCA and a single failure in the ESFs the limit DNBR value will be met after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

The following systems provide the necessary mitigation of an accidental depressurization of the MSS.

(1) SIS actuation from any of the following:

(a) Two-out-of-four low pressurizer pressure signals

(b) Two-out-of-three low steam line pressure signals on any one loop

(2) The overpower reactor trips (neutron flux and ΔT), the overtemperature ΔT reactor trip, and the reactor trip occurring in conjunction with receipt of the SI signal.

(3) Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, a SI signal will rapidly close all MFRVs, trip the main feedwater pumps, and close the MFIVs.

15.2.14.3 Analysis of Effects and Consequences

Due to the size of the break and the assumed initial conditions, an accidental depressurization of the MSS event is bounded by the Main Steam Line Rupture accident analyzed in Section 15.4.2.1. As such, no explicit analysis is performed for the accidental depressurization of the MSS. All applicable acceptance criteria are shown to be met via the results and conclusions in Section 15.4.2.1.

15.2.14.4 Conclusions

The applicable acceptance criterion is shown to be met via the results and conclusions in Section 15.4.2.1. The rupture of a main steam line at hot zero power is a Condition IV event that meets the minimum DNBR criterion of the Condition II accidental depressurization of the MSS. The Condition IV event models a large double-ended rupture, with more limiting results than the inadvertent opening of a SG PORV or dump valve, regardless of any differences in safety system actuations. Therefore, demonstrating that the Condition IV event meets the same Condition II criterion for minimum DNBR, as the calculated value is never below the limit value of 1.45 at any time during the transient, satisfies the acceptance criterion for the Condition II event discussed in Section 15.2.14.

15.2.15 SPURIOUS OPERATION OF THE SAFETY INJECTION SYSTEM AT POWER

15.2.15.1 Acceptance Criteria

The following are the relevant specific acceptance criteria.

The minimum DNBR must not go below the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell) (refer to Section 4.4.4.1) at any time during the transient. This criterion is discussed in Section 15.2.15.2.

The spurious safety injection (SSI) event is terminated prior to challenging the PSV liquid water (hereinafter referred to as water) relief capability. This criterion is discussed in Section 15.2.15.3.

The RCS overpressure concern is bounded by the Section 15.2.7 event.

15.2.15.2 Spurious Safety Injection Departure from Nucleate Boiling Ratio Analysis

15.2.15.2.1 Identification of Causes and Accident Description

SSI system operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal may originate from any of the SI actuation channels. Refer to Section 7.2 for a description of the actuation system.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the RWST. The charging injection valves between the charging pumps and the injection header open automatically. The charging pumps then pump RWST water through the header and injection line and into the cold legs of each loop. The SI pumps also start automatically but provide no flow when the RCS is at normal pressure. The passive injection system and the RHR system also provide no flow at normal RCS pressure.

The analyses of the potential for DNB, loss of fuel integrity, and excessive cooldown are presented in the discussions herein.

An SIS signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. Therefore, two different courses of events are considered.

Case A: Trip occurs at the same time spurious injection starts.

Case B: The reactor protection system produces a trip later in the transient.

For Case A, the operator should determine if the spurious signal was transient or steady state in nature; i.e., an occasional occurrence or a definite fault. The operator will determine this by following approved procedures. In the transient case, the operator would stop the SI. If the SIS must be disabled for repair, boration should continue and the plant brought to cold shutdown.

For Case B, the reactor protection system does not produce an immediate trip and the reactor experiences a negative reactivity excursion causing a decrease in the reactor power. The power unbalance causes a drop in T_{avg} and consequent coolant shrinkage, and pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load if the electrohydraulic governor fully opens the turbine throttle valve. 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-pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history that affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

Recovery from this incident for Case B is in the same manner as for Case A. The only difference is the lower T_{avg} and pressure associated with the power imbalance during this transient. The time at which reactor trip occurs is of no concern for this accident. At lighter loads coolant contraction will be slower resulting in a longer time to trip.

15.2.15.2.2 Analysis of Effects and Consequences

The spurious operation of the SIS is analyzed for DNBR with the LOFTRAN program. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, SG safety valves, and the effect of the SIS. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analyses of several cases show that the results are relatively independent of time to trip.

A typical transient is considered representing conditions at BOL. Results at EOL are similar except that moderator feedback effects result in a slower transient.

The accident is analyzed using the ITDP and the initial condition uncertainties are included in the limit DNBR as described in Reference 5. Therefore, initial conditions of nominal core power, nominal reactor coolant average temperature (plus 2.5°F for SG fouling) and nominal reactor coolant pressure are assumed.

The assumptions used in the analysis are:

(1) Moderator and Doppler Coefficients of Reactivity

A positive BOL moderator temperature coefficient was used. A low absolute value Doppler power coefficient was assumed.

(2) Reactor Control

The reactor was assumed to be in manual control.

(3) Pressurizer Heaters

Pressurizer heaters were assumed to be inoperative in order to increase the rate of pressure drop.

(4) Boron Injection

At time zero, two charging pumps (CCP1 and CCP2) begin injection and pump borated water through the SIS and into the cold leg of each loop.

(5) Turbine Load

Turbine load was assumed constant until the electrohydraulic governor drives the throttle valve wide open. Once the throttle valve is wide open, turbine load drops as the steam pressure decreases.

(6) Reactor Trip

Reactor trip was initiated by low pressure. The trip was conservatively assumed to be delayed until the pressure reached 1860 psia.

15.2.15.2.3 Results

The transient response for the minimum feedback case is shown in Figures 15.2.15-1 and 15.2.15-2. Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until 25 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes Tavg, pressurizer water level, and pressurizer pressure to drop. The low-pressure trip setpoint is reached at 23 seconds and rods start moving into the core at 25 seconds.

15.2.15.2.4 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) There is margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell).

Results of the DNBR analysis show that SSI with or without immediate reactor trip presents no hazard to the integrity of the RCS.

DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS.

If the reactor does not trip immediately, the low-pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

15.2.15.3 Spurious Safety Injection Pressurizer Filling Analysis

The purpose of the SSI pressurizer filling analysis is to demonstrate the SSI pressurizer filling event can be terminated before water is relieved through the pressurizer safety valves (PSVs). Reference 16 establishes that more than three cycles of water relief through the PSVs could potentially result in the PSVs failing open resulting in an unisolable reactor coolant pressure boundary breach. Terminating the SSI pressurizer filling event before water relief through the PSVs occurs prevents this potential failure.

15.2.15.3.1 Identification of Causes and Accident Description

The SSI pressurizer overfill event begins when a safety injection (SI) signal starts the two PG&E Design Class I charging pumps resulting in reactor coolant inventory addition. The SI signal also results in a Phase A containment isolation that causes loss of instrument air to containment, which is required to open the letdown valves to control pressurizer level. If the pressurizer subsequently fills, the pressurizer power-operated relief valves (PORVs) are available to relieve water inventory from the RCS, as long as nitrogen is available from the PG&E Design Class I backup nitrogen accumulators (since instrument air is not available). Since Technical Specifications define a PORV as operable with its block valve closed (if the PORV can be made available for automatic pressure relief), operators may need to take action to open the block valve to enable the PORV to provide water relief. Using the PORVs to relieve water from the RCS prevents water relief through the PSVs.

Mitigation of pressurizer filling is accomplished by operators taking action to control pressurizer level by isolating charging flow and initiating letdown by re-establishing instrument air to containment, before backup nitrogen to the PORVs is depleted.

The SSI pressurizer filling analysis models the long term plant response and the operator actions needed to mitigate pressurizer filling during a SSI and thus prevent water relief through the PSVs. The operator recovery actions credited for SSI mitigation at power are included in the plant emergency operating procedures (EOPs) and include (1) ensuring a PG&E Design Class I pressurizer PORV is available and (2) controlling pressurizer level using charging and letdown.

15.2.15.3.2 Analysis of Effects and Consequences

The SSI event is analyzed for pressurizer filling in accordance with the NRC approved methodology for a 4-loop plant (Reference 19) using the Westinghouse version of the RETRAN-02 computer code (RETRAN-02W). The code simulates the neutron kinetics, RCS, pressurizer, PORVs and PSVs pressurizer spray, SG, SG safety valves, and safety injection. The code computes pertinent plant variables including temperatures, pressures, and power level.

Separate cases to accommodate different limiting assumptions are analyzed to determine the time by which the operators would need to ensure a PG&E Design Class I PORV is available and the times by which the operators would need to achieve control of pressurizer level using charging and letdown. The beginning of each SSI case is essentially identical as follows.

The SI signal occurs at time zero. This generates a concurrent reactor trip signal from full power conditions followed by a turbine trip signal. The pressurizer pressure and pressurizer level initially decrease as the RCS power and temperature reduce from full power conditions to hot no load conditions. Typically, the initiation of the emergency core cooling system (ECCS) injection flow halts the post trip pressure decrease and then rapidly increases the pressure until the pressurizer spray valves open enough to maintain the pressurizer pressure relatively constant. For this analysis, it is assumed the pressurizer spray valves are not operable and the pressurizer level continues to increase due to ECCS injection flow until the pressurizer fills.

The assumptions for the pressurizer filling analysis are conservatively chosen to minimize the time to reach a water-solid condition and maximize the predicted number of pressurizer PORV relief open/close cycles. Sensitivity studies are performed for a number of parameters to determine the appropriate conservative assumptions. The key assumptions made in the analysis are described below.

(1) Initial Operating Conditions

The initial pressurizer pressure is assumed to be 2,190 psia, which is 60 psi lower than the nominal value. This lower initial RCS pressure results in increased ECCS injection flow during the transient and maximizes the challenges to the PSVs and PORVs.

Initial RCS Tavg values of 559.5ºF and 572.1ºF were analyzed, which correspond to the minimum and maximum values of the nominal Tavg window for Unit 2, less a bounding temperature uncertainty bias. This conservatively maximizes the initial RCS mass, and minimizes the RCS volumetric shrinkage after the reactor trip. For these initial Tavg values, the corresponding programmed initial pressurizer levels are50.8 and 66.8 percent span, respectively, which bound the pressurizer level uncertainty of 5.7 percent span.

(2) Pressurizer Heaters

Both the backup and proportional pressurizer heaters are assumed to remain available during the event, which conservatively maximizes the pressurizer water volume and minimizes the time to fill the pressurizer with water.

(3) Reactor Trip / Turbine Load

The reactor trip occurs coincident with the SI actuation, which results in an immediate turbine trip. There is no credit for heat removal from the steam dump system to the condenser or atmosphere. The MSSVs are assumed to be operable and were modeled to open at conservative set pressures that account for setpoint drift and accumulation. The minimum heat transfer from the primary coolant loop to the secondary system leads to a conservatively early pressurizer fill condition.

(4) Moderator and Doppler Coefficients of Reactivity

The pressurizer filling analysis assumes a positive EOL moderator density coefficient and large absolute value Doppler power coefficient. Since the reactor trip occurs immediately for the pressurizer filling event, these reactivity coefficients have a negligible impact on the results.

(5) Reactor Decay Heat

Conservative core residual heat generation is assumed based on long-term operation at the initial power level preceding the trip. The ANS 5.1-1979 decay heat model+ 2σ is used for calculation of residual decay heat levels.

(6) Pressurizer PORVs

No credit is taken for relief through the PORV that is actuated on a compensated pressurizer pressure deviation signal (i.e., the non-safety-grade, PG&E Design Class II PORV). However, relief through the PORVs that are actuated on the indicated (measured) pressurizer pressure signal (i.e., the safety-grade, PG&E Design Class I PORVs) has been modeled with assumptions that maximize the number of PORV opening cycles experienced. The number of safety-grade PORVs available for relief depends on the single failure being considered as discussed in the results section.

Since an SI signal causes Phase A containment isolation and the instrument air is a PG&E Design Class II (non-safety-grade) system, there is a loss of instrument air to containment due to this signal. Accordingly, the PG&E Design Class I backup nitrogen accumulators are needed to maintain functionality of the PG&E Design Class I PORVs. The backup nitrogen accumulators are each sized and leak tested to ensure at least 300 PORV cycles before the backup nitrogen supply is depleted, after which the PORV would be unavailable. Therefore, transient mitigation must be demonstrated to occur before 300 PORV cycles is reached.

(7) ECCS Injection Flow

Maximum SI flow rates were conservatively modeled with a flow profile that bounds the maximum flow from the two PG&E Design Class I high-head centrifugal charging pumps (CCP1 and CCP2), plus two intermediate-head SI pumps. The SI flow profile is shown in Figure 15.2.15-6. Full SI flow was conservatively assumed to occur immediately after the SI actuation signal. The RWST fluid temperature is assumed to be 35°F to maximize the ECCS fluid density.

(8) Pressurizer Sprays

The air-operated pressurizer spray valves are assumed to be inoperable, since instrument air to containment is lost on an SI signal and normal pressurizer spray flow is unavailable following coastdown of the RCPs. There are auxiliary spray lines that are equipped with backup nitrogen if the spray valves are unavailable; however, auxiliary spray requires a manual alignment that would not be completed until after the TCOAs necessary to mitigate this transient are complete.

15.2.15.3 Results

The sequences of events for the three cases evaluated for the pressurizer filling cases are listed in Table 15.2-1. Typical transient responses are shown in Figures 15.2.15-3 through 15.2.15-5.

<u>Case to establish the maximum time available to ensure a PG&E Design Class I pressurizer PORV is available</u>

This case determines the maximum time available for operations to ensure a PG&E Design Class I pressurizer PORV is available before water relief through the PSVs occurs. Both PG&E Design Class I pressurizer PORVs are assumed to be unavailable, with one failed and the other isolated by the block valve as allowed by Technical Specifications.

Because the PORVs are unavailable the RCS experiences a rapid pressure increase to the PSV lift setpoint. The pressurizer level continues to increase until the pressurizer fills and the first water relief through the PSV occurs at 904 seconds (~15.1 minutes). This establishes the maximum time available for the operators to unblock a pressurizer PORV to prevent water relief through the PSVs.

<u>Case to establish the minimum time to pressurizer filling</u>

The case to establish the minimum time to pressurizer filling assumes that both PG&E Design Class I PORVs are available, which results in the earliest filling of the pressurizer and earliest initiation of water relief through the pressurizer PORVs. For this scenario, the RCS pressure increases only to the PORV lift setpoint where it is maintained within a relatively constant range as the PORV continues to cycle and relieve steam until the pressurizer becomes water solid. The minimum time to pressurizer filling was determined to be 11.5 minutes.

<u>Case to establish the minimum time to deplete the backup nitrogen accumulators</u>

In the case to establish the minimum time to deplete the nitrogen accumulators, one PORV is assumed to be available. The RCS pressure increases only to the pressurizer PORV lift setpoint, at which time the PORV starts to cycle and relieve steam as the pressurizer level increases due to the ECCS injection flow. Since the RCS pressure is maintained near the PORV setpoint, rather than the PSV setpoint, the pressurizer fills earlier than the case to establish the maximum allowable time to make a PORV available. Once the pressurizer becomes water solid, the PORV begins relieving water. The PORV generates less of a setpoint undershoot while relieving water. As a result, the PORV cycles at a slightly faster rate than when relieving steam. The analysis determines the minimum time to reach 300 PORV cycles and deplete the backup nitrogen accumulator for a single PORV is 45.8 minutes. This establishes the maximum time available for operators to control pressurizer level before water relief through the PSVs occurs.

The results of the SSI analysis for pressurizer filling demonstrate that, if the pressurizer fills, the following TCOAs preclude water relief through the PSVs:

(1) Ensure a PG&E Design Class I pressurizer PORV is available within 15 minutes

If no pressurizer PORV relief is available at the start of the transient because a PG&E Design Class I PORV is not available, operator action is required to ensure a PG&E Design Class I PORV is available in time to prevent water relief through the PSVs. The analysis determined that the minimum time to lift the PSVs is 15.1 minutes; therefore, the operators must ensure a PG&E Design Class I PORV is available within 15 minutes after event initiation.

(2) Control pressurizer level using charging and letdown within 45 minutes

The EOPs direct operators to reduce charging flow by stopping charging pumps and to re-establish RCS letdown by restoring instrument air to containment. Once the pressurizer level is controlled, there is no longer a concern relative to water relief through the PSVs and transient mitigation is complete. The results of the limiting case determined the operators must control pressurizer level by reducing charging flow and establishing letdown within 45 minutes after event initiation.

15.2.15.4 Conclusions

The analysis demonstrates the acceptance criterion is met as follows:

(1) The SSI event is terminated prior to challenging the PSV water relief capability. The SSI pressurizer filling analysis demonstrates that with timely operator actions the SSI event can be terminated before water relief through the PSVs occurs, which prevents the potential generation of a more serious plant event. For the limiting SSI event, the operators have a maximum of about 15 minutes after event initiation to ensure a PG&E Design Class I pressurizer PORV is available before water relief through the PSVs occurs. In addition, with the worst-case control system operation, the operators must terminate the SSI event within 45 minutes to prevent depleting the PG&E Design Class I pressurizer PORV backup nitrogen accumulators.

15.2.16 REFERENCES

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15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults that may occur very infrequently during the life of the plant. They will be accompanied with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers. For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, that actuates the ECCS.

(2) Minor secondary system pipe breaks.

(3) Inadvertent loading of a fuel assembly into an improper position.

(4) Complete loss of forced reactor coolant flow.

(5) Single RCCA withdrawal at full power.

Each of these infrequent faults is analyzed in this section. In general, each analysis includes acceptance criteria, an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequences of events during four Condition III faults of type (1) above, SBLOCA, are shown in Table 15.3-1.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES THAT ACTUATES EMERGENCY CORE COOLING SYSTEM

15.3.1.1 Acceptance Criteria

15.3.1.1.1 10 CFR 50.46 - Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

(1) *Peak cladding temperature (PCT).* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam

shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. This reduces the potential for explosive hydrogen/oxygen mixtures inside containment.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.3.1.1.2 Radiological Criteria

The radiological consequences of a SBLOCA are within a small fraction of the applicable guidelines and limits specified in 10 CFR 50.67 detailed in Section 15.5.11.

15.3.1.2 Identification of Causes and Accident Description

A LOCA is defined as a rupture of the RCS piping or of any line connected to the system. This includes small pipe breaks, typically a 3/8-inch diameter opening (0.11 square inch), up to and including a break size of 1.0 square foot that results in flow that is greater than the makeup flow rate from either CCP1 or CCP2 (refer to Section 6.3.3.6.2.2). Refer to Section 3.6 for a more detailed description of the LOCA boundary limits. The coolant that would be released to the containment contains fission products.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging system flow capability when aligned for maximum charging at normal RCS pressure.

Should a larger break occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low-pressure trip setpoint is reached. The SIS is actuated when the appropriate pressurizer low-pressure setpoint is reached. Reactor trip and SIS actuation are also initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

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

(2) Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the AFW pumps. The SI signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting AFW pumps. The secondary flow aids in the reduction of RCS pressure. When the RCS depressurizes to below approximately 600 psia, the accumulators begin to inject water into the RCLs. The RCPs are assumed to be tripped at the beginning of the accident and the effects of pump coastdown are included in the blowdown analyses.

15.3.1.3 Analysis of Effects and Consequences

For LOCAs due to small breaks less than 1 square foot, the NOTRUMP (Reference 12) computer code is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a one-dimensional general network code with a number of features. Among these features are the calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP SBLOCA ECCS evaluation model was developed to determine the RCS response to design basis SBLOCAs 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 Plants."

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 the NOTRUMP code is provided in References 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 the 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.

SI flowrate to the RCS as a function of the system pressure is used as part of the input. The SIS was assumed to be delivering water to the RCS 27 seconds after the generation of a SI signal.

For the analysis, the SIS delivery considers pumped injection flow that is depicted in Figure 15.3-1 as a function of RCS pressure. This figure represents injection flow from the SIS pumps based on performance curves degraded 5 percent from the design head. The 27-second delay includes time required for diesel startup and loading of the SI pumps onto the emergency buses. The effect of RHR pump flow is not considered here since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also, minimum safeguards ECCS capability and operability have been assumed in these analyses.

PCT analyses are performed with the LOCTA IV (Reference 4) code that determines the RCS pressure, fuel rod power history, steam flow past the uncovered part to the core, and mixture height history.

15.3.1.4 Results

15.3.1.4.1 Reactor Coolant System Pipe Breaks

This section presents the results of a spectrum of small break sizes analyzed. The small break analysis was performed at 102 percent of the Rated Core Power (3411 MWt), a Total Peaking Factor (FQT) of 2.70, a Thermal Design Flow of 87,700 / 88,500 gpm/loop (Unit 1 / Unit 2) and a SGTP level of 10 percent. For Unit 1, the small-break analysis was performed for the Steam Generator (SG). For Unit 2, the small break analysis was performed for the upflow core barrel/baffle configuration, upper head temperature reduction and RSG.

The limiting small break size was shown to be a 3-inch diameter break in the cold leg. In the analysis of this limiting break, an RCS T_{avg} window of 577.3 / 577.6°F, +5°F, -4°F (Unit 1 / Unit 2) was considered. The high T_{avg} cases were shown to be more limiting than the Low T_{avg} cases and therefore are the subject of the remaining discussion. The time sequence of events and the fuel cladding results for the breaks analyzed are shown in Tables 15.3-1 and 15.3-2.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the RCPs through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rods and cladding to very near the coolant temperature as long as the core remains covered by a two-phase mixture. This effect is evident in the accompanying figures.

The depressurization transients for the limiting 3-inch breaks are shown in Figure 15.3-9. The extent to which the core is uncovered for these breaks is presented in Figure 15.3-11. The maximum hot spot cladding temperature reached during the transient, including the effects of fuel densification as described in Reference 3, is 1391 / 1288°F (Unit 1 / Unit 2). The PCT transients for the 3-inch breaks are shown in Figure 15.3-13. The top core node vapor temperatures for the 3-inch breaks are shown in Figure 15.3-33. When the mixture level drops below the top of the core, the top core node vapor temperature increases as the steam superheats along the exposed portion of the fuel. The rod film coefficients for this phase of the transients are given in Figure 15.3-34. The hot spot fluid temperatures are shown in Figure 15.3-35 and the break mass flows are shown in Figure 15.3-36.

The core power (dimensionless) transient following the accident (relative to reactor scram time) is shown in Figure 15.3-8. The reactor shutdown time (4.7 seconds) is equal to the reactor trip signal processing time (2.0 seconds) plus 2.7 seconds for complete rod insertion. During this rod insertion period, the reactor is conservatively assumed to operate at 102 percent rated power. The small break analyses considered 17x17 Vantage 5 fuel with intermediate flow mixer (IFM) grids, ZIRLO cladding, and an axial blanket. Fully enriched annular pellets, as part of an axial blanket core design, were modeled explicitly in this analysis. The results when modeling the enriched annular pellets were not significantly different than the results from solid pellet modeling.

Several figures are also presented for the additional break sizes analyzed. Figures 15.3-37, 15.3-2, and 15.3-40 present the RCS pressure transient for the 2-, 4-, and 6-inch breaks, respectively. Figures 15.3-38, 15.3-3, and 15.3-41 present the core mixture height plots for 2-, 4-, and 6-inch breaks, respectively. The PCT transients for the 2-inch breaks are shown in Figure 15.3-39. The PCT transients for the 4-inch breaks are shown in Figure 15.3-4. These results are not available for the 6-inch break because the core did not uncover for this transient.

The small break analysis was performed with the Westinghouse ECCS Small Break Evaluation Model (References 12 and 4) approved for this use by the NRC in May 1985. An approved cold leg SI condensation model, COSI (Reference 26), was utilized as part of the Evaluation Model.

15.3.1.5 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

15.3.1.5.1 10 CFR 50.46 - Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

(1) PCT. The calculated maximum fuel element cladding temperature does not exceed 2200°F, as shown in Table 15.3-2.

(2) Maximum cladding oxidation. The calculated total oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation, as shown in Table 15.3-2.

(3) Maximum hydrogen generation. Table 15.3-2 shows that the average cladding oxidation is less than 0.01 times the cladding thickness. Thus the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times

the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) & (5) Coolable Geometry and Long Term Cooling. The results associated with the SBLOCA analysis performed with the NOTRUMP Evaluation Model explicitly demonstrate compliance with Criteria 1 through 3. Because of the fuel rod burst and blockage models used in the LOCTA code, and modeling of the cold leg recirculation phase in NOTRUMP, SBLOCA analysis results also support the coolable geometry and long term cooling criteria. Since Criteria 1 through 3 are explicitly met, Criteria 4 and 5 are met as well. The SBLOCA phenomena and results are therefore in compliance with 10 CFR 50.46 acceptance criteria.

15.3.1.5.2 Radiological

The radiological consequences of a SBLOCA are within a small fraction of the applicable guidelines and limits specified in 10 CFR 50.67 detailed in Section 15.5.11.

15.3.2 MINOR SECONDARY SYSTEM PIPE BREAKS

15.3.2.1 Acceptance Criteria

(1) The minimum DNBR does not go below the safety analysis limit (refer to Sections 15.4.2.1.1 and 15.4.2.3.1) at any time during the transient to ensure that the core remains geometrically intact with no loss of core cooling capability.

(2) Any activity release must be such that the calculated doses at the site boundary are a small fraction of the applicable guidelines and limits specified in 10 CFR 50.67 as detailed in Section 15.5.12.

15.3.2.2 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-inch diameter break or smaller.

15.3.2.3 Analysis of Effects and Consequences

Minor secondary system pipe breaks must not result in more than the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analyses for minor secondary system pipe breaks is not required.
The analyses of the more probable accidental opening of a secondary system steam dump, relief, or safety valve is presented in Section 15.2.14. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.3.2.4 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The analysis presented in Section 15.4.2 demonstrates that the consequences of a minor secondary system pipe break are acceptable because a DNBR of less than the design basis values does not occur even for a more critical major secondary system pipe break.

(2) Section 15.5.12 demonstrates the potential radiological exposures to the public following a minor secondary system pipe rupture are within a small fraction of the applicable guidelines and limits specified in 10 CFR 50.67.

15.3.3 INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION

15.3.3.1 Acceptance Criteria

(1) In the event of a fuel loading error not identified until normal operation, the offsite dose consequences should be a small fraction of the applicable guidelines and limits specified in 10 CFR 50.67 as detailed in Section 15.5.1.

15.3.3.2 Identification of Causes and Accident Description

Fuel and core loading errors such as inadvertently loading one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or loading a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. The inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods is also included among possible core loading errors.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes that are more peaked than those calculated with the correct enrichments. The incore system of movable neutron flux detectors that is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error that causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. For each core loading, the identification number is checked to ensure proper core configuration. The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with movable incore neutron flux detectors. In addition to the flux detectors, thermocouples are located at the outlet of about one-third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation. A more detailed discussion of the flux detection capabilities may be found in Section 4.3.3.2.

15.3.3.3 Analysis of Effects and Consequences

Steady state power distributions in the x-y plane of the core are calculated with the TURTLE code (refer to Section 1.6.1, Item 49, and Section 4.3.3.8.5), based on macroscopic cross-sections calculated by the LEOPARD code (refer to Section 1.6.1, Item 48, and Section 4.3.3.10.2). A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distributions in the x-y plane for a correctly loaded core assembly are given in Chapter 4 based on enrichments given in that section.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (refer to Figures 15.3-15 through 15.3-19).

15.3.3.4 Results

The following core loading error cases have been analyzed:

(1) Case A

The case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (refer to Figure 15.3-15).

(2) Case B

The case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (refer to Figures 15.3-16 and 15.3-17).

In Case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the Region 2 assembly mistakenly loaded into Region 1.

In Case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded into the Region 2 position.

(3) Case C

Enrichment error: the case in which a Region 2 fuel assembly is loaded in the core central position (refer to Figure 15.3-18).

(4) Case D

The case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (refer to Figure 15.3-19).

15.3.3.5 Conclusions

In the event that a single rod or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures will be limited to the incorrectly loaded rod or rods.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the incore movable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

The analysis demonstrates the acceptance criterion is met as follows:

(1) No events leading to environmental radiological consequences are expected as a result of loading errors (refer to Section 15.5.13).

15.3.4 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.4.1 Acceptance Criteria

(1) Maintain the minimum DNBR greater than the safety analysis limit for fuel (refer to Section 4.4.3.1).

15.3.4.2 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all RCPs. 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 DNB with subsequent fuel damage if the reactor were not tripped promptly. The following reactor trips provide necessary protection against a loss of coolant flow accident:

(1) Low RCL flow (primary protection)

(2) Undervoltage or underfrequency on RCP power supply buses (backup protection to low RCL flow trip)

(3) Pump circuit breaker opening (also backup to low RCL flow trip)

The reactor trip on low RCL flow is provided as primary protection against loss-of-flow conditions. This function is generated by two-out-of-three low-flow signals per RCL. Above approximately 35 percent power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 and 35 percent power (Permissive 7 and Permissive 8), low-flow in any two loops will actuate a reactor trip.

The reactor trip on RCP bus undervoltage is provided as protection against conditions that can cause a loss of voltage to all RCPs (i.e., LOOP). This function serves as backup protection to the low RCL flow trip and is blocked below approximately 10 percent power (Permissive 7).

The reactor trip on RCP underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. Underfrequency also opens the RCP breakers that disengage the RCPs from the power grid so that the pumps flywheel kinetic energy is available for full coastdown. This function also serves as backup protection to the low RCL flow trip.

A reactor trip from opened pump breakers is also provided as a backup to the low-flow signals. Above Permissive 7 a breaker open signal from any 2 of 4 pumps will actuate a reactor trip. Reactor trip on RCP breakers open is blocked below Permissive 7.

Normal power for the RCPs is supplied through buses from a transformer connected to the generator. Two pumps are on each bus. When a generator trip occurs, the buses 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 or mechanical faults which require immediate tripping of the generator from the network, the generator remains connected to the network for approximately 30 seconds. The RCPs remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

15.3.4.3 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. First the LOFTRAN (Reference 8) code is used to calculate the loop and core flow during the transient. The LOFTRAN code is used to calculate the nuclear power transient. The FACTRAN (Reference 9) code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC-IV code (refer to Section 4.4.3.4) is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The transients presented represent the minimum of the typical and thimble cells.

The following cases have been analyzed:

(1) Four of four loops coasting down.

(2) RCPs power supply frequency decay at a maximum constant 3 Hz/sec rate (underfrequency).

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2 and Table 15.1-4, except that following the loss of electrical supply to all pumps at power, the reactor trip is actuated by low RCL flow rather than bus undervoltage or bus underfrequency.

15.3.4.4 Results

The calculated sequence of events is shown in Table 15.3-3. Figures 15.3.4-1 through 15.3.4-3 show the flow coastdown, the heat flux response, and the nuclear power response for the limiting complete loss of flow event(four-loop coastdown). For analysis purposes, the reactor is assumed to trip on the low RCL flow signal, as this trip signal is PG&E Design Class I, while the undervoltage and underfrequency trips are PG&E Design Class II. Because the low RCL flow trip responds to an actual condition (while the undervoltage/underfrequency trips are anticipatory), the reactor trip is delayed and the transient is more DNBR limiting. A plot of DNBR versus time is given in Figure 15.3.4-4. This plot represents the limiting (thimble) cell for the four-loop coastdown. Figures 15.3.4-1 through 15.3.4-4 present only the Unit 1 results; however, the Unit 2 results are nearly identical. Therefore, the figures are representative of Unit 2 response.

15.3.4.5 Conclusions

The safety analysis results described in Section 15.3.4.4 have demonstrated that for the complete loss of forced reactor coolant flow, the minimum DNBR is above the safety analysis limit values of 1.71/1.68 (typical cell/thimble cell) during the transient; therefore, no core safety limit is violated.

15.3.5 SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER

15.3.5.1 Acceptance Criteria

(1) No more than 5 percent of the fuel rods experience a DNBR less than the limit value.

15.3.5.2 Identification of Causes and Accident Description

By design, 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 deliberately withdraw a single RCCA in the control bank; this feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures that could result in single RCCA withdrawal, rod deviation and control rod urgent failure may be displayed on the plant annunciator, 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.

Each bank of RCCAs in the system is divided into two groups of four mechanisms each (except Group 2 of Bank D which consists of five mechanisms). The rods comprising a group operate in parallel through multiplexing thyristors. The two 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 and 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 four stationary grippers, movable grippers, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are either in the direction of insertion or immobility.

In the unlikely event of multiple failures that result in continuous withdrawal of a single RCCA, it is not possible, in all cases, to provide assurance of automatic reactor trip so that core safety limits are not violated. 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 covered by the RCCA.

15.3.5.3 Analysis of Effects and Consequences

Power distributions within the core are calculated by the ANC code based on macroscopic cross-sections generated by PHOENIX-P (refer to Section 4.3.3.10.2). The peaking factors calculated by ANC (refer to Section 4.3.3.10.3) are then used by THINC (refer to Section 1.6.1, Item 28, and Section 4.4.3) to calculate the minimum DNBR for the event. The plant was analyzed for the case of the worst rod withdrawn from Control Bank D inserted at the insertion limit, with the reactor initially at full power.

15.3.5.4 Results

Two cases have been considered as follows:

(1) If the reactor is in the automatic control mode, 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 2 described below. For such cases as above, a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety limit.

(2) 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 failed RCCA. In terms of the overall system response, this case is similar to those presented in Section 15.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR than for the withdrawn bank cases. Depending on 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 safety limit value. Evaluation of this case to determine the most limiting DNBR condition, which would occur at the power and coolant condition 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 safety limit value is 5 percent.

15.3.5.5 Conclusions

The analysis demonstrates the acceptance criterion is met as follows:

(1) For both cases of one RCCA fully withdrawn, with the reactor in either the automatic or manual control mode and initially operating at full power with Bank D at the insertion limit, 5 percent or less of the total fuel rods in the core will go below the minimum DNBR safety analysis limit.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before any DNB could occur. For Case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard. However, operator action is not required to meet the acceptance criteria.

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27. T. Q. Nguyen, et al., Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11596-P-A, June 1988.

28. S. L. Davidson, (Ed), et al., ANC: Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A, September 1986.

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. Condition IV faults shall not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 50.67 and Regulatory Guide 1.183, July 2000. A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the ECCS and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the RCS; i.e., LOCA

(2) Major secondary system pipe ruptures

(3) Steam generator tube rupture (SGTR)

(4) Single RCP locked rotor

(5) Fuel handling accident (FHA)

(6) Rupture of a control rod mechanism housing (RCCA ejection)

Each of these six limiting faults is analyzed in this section. In general, each analysis includes acceptance criteria, an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The analyses of dose consequences, resulting from events leading to fission product release, are presented in Section 15.5. The fission product inventories that form a basis for these calculations are presented in Chapter 11 and Section 15.5. Also included is a discussion of system interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

The LBLOCA analysis contained in Section 15.4.1 has been revised to incorporate separate best estimate LOCA analyses for Unit 1 and Unit 2. The general discussion of the best estimate LOCA transient in Sections 15.4.1.2 through 15.4.1.4 are applicable to Unit 1 and Unit 2. However, the statistical treatment methodologies are slightly different for Unit 1 and Unit 2. Statistical treatment methodologies for Unit 1 and Unit 2 are discussed in Sections 15.4.1.4A and 15.4.1.4B respectively.

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENT)

15.4.1.1 Acceptance Criteria

15.4.1.1.1 10 CFR - Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

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 acceptance criteria are listed below:

(1) Peak cladding temperature (PCT). The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

(2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.4.1.1.2 Radiological Criteria

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR Part 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.1.2 Background of Best Estimate Large-Break Loss-of-Coolant Accident

The analysis performed to comply with the requirements of 10 CFR 50.46 (Reference 1), and Revisions to the Acceptance Criteria (Reference 54) is presented in this section.

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 55).

A LOCA evaluation methodology for three- and four-loop 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. The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 56).

The time sequence of events during a nominal large double-ended cold leg guillotine (DECLG) break LOCA is shown in Tables 15.4.1-1A and 15.4.1-1B. The results of the LBLOCA analysis are shown in Tables 15.4.1-2A and 15.4.1-2B and show compliance with the acceptance criteria. The analytical techniques used for the LBLOCA analysis are in compliance with 10 CFR 50.46 (Reference 1) as amended in Reference 54, and are described in Reference 56. Due to the significant differences between the Unit 1 and Unit 2 reactor vessel internals, plant-specific vessel models were developed and evaluated. The significant differences between the units are summarized below:

Unit 1	Unit 2
"Top Hat"-Upper Support Plate	Flat Upper Support Plate
Domed Lower Support Plate	Flat Lower Support Plate
Thermal Shield	Neutron Pads
Diffuser Plate	No Diffuser Plate

An analysis of each unit was performed and a comparison determined that the Unit 1 vessel model resulted in more limiting PCT values. As a result, the best estimate base LBLOCA analysis (Reference 60) results were based on Unit 1 and were considered bounding for both Unit 1 and Unit 2. Recently, the Unit 1 best estimate LOCA was reanalyzed for Unit 1 using the approved reanalysis methodology established in Reference 56. In the process of performing the Unit 1 reanalysis (Reference 67), it was determined that the Unit 1 vessel model no longer consistently resulted in the limiting PCTs, and could not be considered bounding for Unit 2. Therefore, the reanalysis methodology (Reference 56) was only applied to Unit 1, and a new and separate best estimate LBLOCA analysis was performed for Unit 2 using an updated and slightly different methodology as described in Reference 69. Both Unit 1 and Unit 2 use the base best estimate LBLOCA analysis methodology and computer code as described in Reference 60 and described in Section 15.4.1.3, which is applicable to Unit 1 and Unit

2. Separate subsequent subsections describe the Unit 1 reanalysis methodology (Reference 67), the Unit 2 analysis methodology (Reference 69), and the respective results.

15.4.1.3 WCOBRA/TRAC Thermal-hydraulic Computer Code

The thermal-hydraulic computer code that was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a LBLOCA is WCOBRA/TRAC, Version Mod 7A, Revision 1 (Reference 56). A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in the PWR LBLOCA. Slightly different revisions to this computer code were used for the Unit 1 reanalysis and the separate Unit 2 analysis as described in later sections.

WCOBRA/TRAC combines two-fluid, three-field, 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, SGs, RCPs, 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 hydrodynamics 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 loop, major loop components, and SI 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. Special purpose components exist to model specific components such as the SG and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 61) and mass and energy releases from the WCOBRA/TRAC calculation.

15.4.1.4 Thermal Analysis

15.4.1.4.1 Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a LOCA including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident.

The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the acceptance criteria of 10 CFR 50.46.

15.4.1.4.2 Sequence of Events and Systems Operations

The sequence of events following a nominal large DECLG break LOCA is presented in Tables 15.4.1-1A and 15.4.1-1B for Unit 1 and Unit 2, respectively. Should a major 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 SI signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

(1) Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken during the LOCA transient for negative reactivity due to the boron concentration of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is not assumed in the large break analysis.

(2) Injection of borated water provides the fluid medium for heat transfer from the core and prevents excessive cladding temperatures.

For the present Westinghouse PWR design, the limiting single failure assumed for a LBLOCA is the loss of one train of ECCS pumps (one charging pump (CCP1 or CCP2), one high-head SI pump, and one RHR pump). One ECCS train delivers flow through the injection lines to each loop, with the least resistant branch injection line spilling to containment backpressure (refer to Figures 15.4.1-14A and 15.4.1-14B and Tables 15.4.1-7A and 15.4.1-7B). All EDGs are assumed to start in the modeling of the containment fan coolers and spray pumps. Modeling full operation of the containment heat removal system is required by Branch Technical Position CSB 6-1, and is a conservative assumption for the LBLOCA analysis.

15.4.1.4.3 Description of a Large-Break Loss-of-Coolant Accident Transient

Before the break occurs, the RCS 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 DECLG break is assumed to open almost instantaneously in one of the main RCS pipes. Calculations have demonstrated that the most severe transient results occur for a DECLG break between the pump and the reactor vessel.

The LBLOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the critical heat flux phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 15.4.1-1A to 15.4.1-12A for Unit 1 and Figures 15.4.1-1B to 15.4.1-12B for Unit 2.

(1) Critical Heat Flux Phase

Immediately following the cold leg rupture, the break discharge rate is subcooled and high. The regions of the RCS with the highest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses, and the fuel rods begin to go through DNB. The fuel cladding rapidly heats up while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

(2) Upward Core Flow Phase

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figures 15.4.1-4A and 15.4.1-4B show the void fraction for one intact loop pump and the broken loop pump for Unit 1 and Unit 2, respectively. The figures show that the intact loop remains in single-phase liquid flow for several seconds, 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

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of core vapor flow, as shown in Figures 15.4.1-5A and 15.4.1-5B for Unit 1 and Unit 2, respectively, best illustrate this phase of core cooling. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs. 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 consequently the downward core flow, are reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water.

(4) Refill Period

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water. 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 period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

(5) Reflood Period

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system repressurization, and the lower core region begins to quench. 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 repressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the SGs, and the RCPs before it is vented out 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 subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled and the early reflood period is terminated.

A second cladding heatup transient may occur due to boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel and vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer. The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations. Figures 15.4.1-12A and 15.4.1-12B show only a slight reduction in downcomer level which indicates that a late reflood heatup does not occur for either unit. However, the Unit 1 reanalysis methodology (Reference 67) still requires that both the early and late reflood PCT periods be considered, while the Unit 2 updated analysis methodology (Reference 69) has eliminated the need to evaluate the late reflood period for PCT. For the Unit 1 reanalysis, the first reflood peak is considered to be the maximum PCT, which occurs after the beginning of reflood, and before the beginning of gravity driven reflood. In Unit 1 Figure 15.4.1-1A, this corresponds to the maximum PCT between about 35 and 50 seconds after the break. The second reflood peak is then considered to be the maximum PCT, which occurs after the beginning of gravity driven reflood. This terminology for first and second reflood PCTs is only used in the further discussions of the Unit 1 best estimate LBLOCA reanalysis.

Continued operation of the ECCS pumps supplies water during the long-term cooling period. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. When low level is reached in the RWST, switchover to the recirculation phase is initiated. The RHR pumps are tripped, and the operator manually aligns the charging (CCP1 or CCP2) and SI pumps to the RHR pump discharge. Once the alignment is completed, all ECCS pumps recirculate containment recirculation sump water. The containment spray pumps continue to draw suction from the RWST until the low-low level is reached, at which time the containment spray pumps are tripped. If two RHR pumps are running, the containment spray valves can be aligned so that an RHR pump can be utilized to deliver recirculation water to the containment spray ring headers and spray nozzles for continued containment spray system (CSS) post-accident operation.

Approximately 7.0 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel. Long-term cooling also includes long-term criticality control. To achieve long-term criticality control, a mixed-mean sump boron concentration is determined and verified against core design margins to ensure core subcriticality, without credit for RCCA insertion. A mixed-mean sump boron concentration is calculated based on minimum volumes for boron sources and maximum volumes for dilution sources. The calculated mixed-mean sump boron concentration is verified against available core design margins on a cycle-specific basis. The analysis criteria for assessing subcriticality is Keff is <1 assuming all rods out, no xenon in the core, the most reactive time in life, and temperature between 68-212°F.

At the time hot leg switchover is performed, there is the potential following a cold leg LOCA that boron-diluted liquid from the containment sump will displace the boron-concentrated liquid in the core. The evaluation of subcriticality in the reload analysis at the beginning of cold leg recirculation adequately addresses potential recriticality due to sump dilution entering hot leg recirculation if the following factors are considered at hot leg recirculation: existing cycle-specific design margin, control rod insertion after a cold leg LOCA, and the boron worth of Xenon at the time of hot leg switchover.

15.4.1.4A Unit 1 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation Model

The thermal-hydraulic computer code that was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a LBLOCA is WCOBRA/TRAC, Version MOD7A Revision 1 (Reference 56). Modeling of the PWR introduces additional uncertainties that are identified and quantified for the plant-specific Unit 1 analysis (Reference 60). The final step of the best estimate analysis methodology is to combine all the uncertainties related to the code and plant parameters, and estimate the PCT at 95 percent probability. The steps taken to derive the PCT uncertainty estimate are summarized below

(1) Plant Model Development

In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

(2) Determination of Plant Operating Conditions

In this step, the expected or desired operating range of the plant to which the analysis applies is established. 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 for input 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. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the “reference transient.”

(3) PWR Sensitivity Calculations

A series of PWR transients is performed in which the initial fluid conditions and boundary conditions are ranged around the nominal condition used in the reference transient. The results of these calculations for DCPP form the basis for the determination of the initial condition bias and uncertainty discussed in Section 6 of Reference 60.

Next, a series of transients is performed that vary the power distribution, taking into account all possible power distributions during normal plant operation. The results of these calculations for DCPP form the basis for the determination of the power distribution bias and uncertainty discussed in Section 7 of Reference 60.

Finally, a series of transients is performed that vary parameters that affect the overall system response (“global” parameters) and local fuel rod response (“local” parameters). The results of these calculations for DCPP form the basis for the determination of the model bias and uncertainty discussed in Section 8 of Reference 60.

(4) Response Surface Calculations

Regression analyses are performed to derive PCT response surfaces from the results of the power distribution run matrix and the global model run matrix. The results of the initial conditions run matrix are used to generate a PCT uncertainty distribution.

(5) Uncertainty Evaluation

The total PCT uncertainty from the initial conditions, power distribution, and model calculations is derived using the approved methodology (Reference 56). The uncertainty calculations assume certain plant operating ranges that may be varied depending on the results obtained. These uncertainties are then combined to determine the initial estimate of the total PCT uncertainty distribution for the DECLG and split breaks. The results of these initial estimates of the total PCT uncertainty are compared to determine the limiting break type. If the split break is limiting, an additional set of split transients is performed that vary overall system response (“global” parameters) and local fuel rod response (“local” parameters). Finally, an additional series of runs is made to quantify the bias and uncertainty due to assuming that the above three uncertainty categories are independent. The final PCT uncertainty distribution is then calculated for the limiting break type, and the 95th percentile PCT is determined.

(6) 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.

There are three major uncertainty categories or elements:

(1) Initial condition bias and uncertainty

(2) Power distribution bias and uncertainty

(3) Model bias and uncertainty

Conceptually, these elements may be assumed to affect the reference transient PCT as shown below.

$$PCT_i = PCT_{REFi} + \Delta PCT_{ICi} + \Delta PCT_{PDi} + \Delta PCT_{MODi} \qquad (15.4.1\text{-}1)$$

where,

PCT_{REFi} = Reference transient PCT: The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 15.4.1-3A, for blowdown (i=1), first reflood (i=2), and second reflood (i=3).

ΔPCT_{ICi} = Initial condition bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes several nominal or average initial conditions, and the average PCT taking into account all possible values of the initial conditions. This bias takes into account plant variations that have a relatively small effect on PCT. The elements that make up this bias and its uncertainty are plant specific.

ΔPCT_{PDi} = Power distribution bias and uncertainty: This bias is the difference between the reference transient PCT, which assumes a nominal power distribution, and the average PCT taking into account all possible power distributions during normal plant operation. Elements that contribute to the uncertainty of this bias are calculational uncertainties, and variations due to transient operation of the reactor.

ΔPCT_{MODi} = Model bias and uncertainty: This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena that affect the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT taking into account all possible values of global and local parameters.

The separability of the uncertainty components in the manner described above is an approximation since the parameters in each element may be affected by parameters in other elements. The bias and uncertainty associated with this assumption are quantified as part of the overall uncertainty methodology and included in the final estimates of the 95-percentile PCT ($PCT^{95\%}$).

The application of the reanalysis methodology to Unit 1 first determines a new reference transient PCT. The bias and uncertainty associated with the initial conditions, power distributions, and models are assumed to remain unchanged. This assumption is assessed to determine that the fundamental LOCA transient characteristics remain unchanged from the new reference transient to that of the original analysis. If applicable, the uncertainty in applying the reanalysis methodology is determined when the superposition assumption is requantified (i.e., the assumption that the major uncertainty elements are independent), and the new bias and new uncertainty is calculated.

15.4.1.5A Unit 1 Containment Backpressure

A conservatively bounding minimum containment back pressure (refer to Figure 15.4.1-14A) is calculated using the methods and assumptions described in Reference 2, Appendix A. Containment back pressure is calculated using the COCO code (Reference 61) and mass and energy releases from the WCOBRA/TRAC calculation. Input parameters used for the Unit 1 containment backpressure calculation are presented in Table 15.4.1-5A. This minimum containment back pressure is modeled using a time dependent pressure table as a boundary condition for the best estimate LBLOCA analysis.

15.4.1.6A Unit 1 Reference Transient Description

A series of WCOBRA/TRAC calculations is performed to determine the PCT effect of variations in key LOCA parameters. An initial transient calculation is performed in which several parameters are set at their assumed bounding (most limiting) values in order to calculate a conservative PCT response to a LBLOCA. The results of these confirmatory runs, as well as the limiting plant determination runs, are incorporated into a final calculation that is referred to as the reference transient. The Unit 1 reference transient models a DECLG break that assumed the conditions listed in Table 15.4.1-3A and includes the LOOP assumption that was shown to produce more limiting PCT results than the offsite power available assumption. The reference transient calculation was performed with several parameters set at their bounding values in order to calculate a relatively high PCT. Single parameter variation studies based on the reference transient were performed to assess which parameters have a significant effect on the PCT results. The results of these studies are presented in Section 15.4.1.7A. The reference transient is the basis for the uncertainty calculations necessary to establish the Unit 1 $PCT^{95\%}$.

15.4.1.7A Unit 1 Sensitivity Studies

A large number of single parameter sensitivity calculations of key LOCA parameters was performed to determine the PCT effect on the LBLOCA transient. These calculations are required as part of the approved best estimate LOCA methodology (Reference 56) to develop data for use in the uncertainty evaluation. For each sensitivity study, a comparison between the reference transient results and the sensitivity transient results was made. These single parameter sensitivity calculations were determined to remain applicable for the Unit 1 reanalysis methodology, as applied (Reference 67).

The results of a small sample of these sensitivity studies performed for the original analysis (Reference 60) are summarized in Table 15.4.1-4A. The results of the entire array of sensitivity studies are included in Reference 60. The Unit 1 reanalysis is documented in Reference 67. The conclusions of the confirmatory cases were determined to remain the same (i.e., limiting direction of conservatism).

15.4.1.7A.1 Unit 1 Initial Condition Sensitivity Studies

Several calculations were performed to evaluate the PCT effect of changes in the initial conditions on the LBLOCA transient. These calculations modeled single parameter variations in key initial plant conditions over the expected ranges of operation, including T_{AVG}, RCS pressure, and ECCS temperatures, pressures, and volumes. The results of these studies are presented in Section 6 of Reference 60.

The results of these sensitivity studies were used to develop uncertainty distributions for the blowdown, first, and second reflood peaks. The uncertainty distributions resulting from the initial conditions, ΔPCT_{ICi}, are used in the overall PCT uncertainty evaluation to determine the final estimate of $PCT^{95\%}$.

15.4.1.7A.2 Unit 1 Power Distribution Sensitivity Studies

Several calculations were performed to evaluate the PCT effect of changes in power distributions on the LBLOCA transient. The approved methodology was used to develop a run matrix of peak linear heat rate relative to the core average, maximum relative rod power, relative power in the bottom third of the core, and relative power in the middle third of the core, as the power distribution parameters to be considered. These calculations modeled single parameter variations as well as multiple parameter variations. The results of these studies indicate that power distributions with peak powers skewed to the top of the core produced the most limiting PCTs. These results are presented in Section 7 of Reference 60.

The results of these sensitivity studies were used to develop response surfaces, which are used to predict the ΔPCT due to changes in power distributions for the blowdown, first, and second reflood peaks. The uncertainty distributions resulting from the power distributions, ΔPCT_{PDi}, are used in the overall PCT uncertainty evaluation to determine the final estimate of $PCT^{95\%}$.

15.4.1.7A.3 Unit 1 Global Model Sensitivity Studies

Several calculations were performed to evaluate the PCT effect of changes in global models on the LBLOCA transient. Reference 56 provides a run matrix of break discharge coefficient, broken cold leg resistance, and condensation rate as the global models to be considered for the double-ended guillotine break. These calculations modeled single parameter variations as well as multiple parameter variations. The limiting split break size was also identified using the approved methodology (Reference 56). These results are presented in Section 8 of Reference 60.

The results of these sensitivity studies were used to develop response surfaces, which are used to predict the ΔPCT due to changes in global models for the DECLG blowdown, first, and second reflood peaks. The uncertainty distribution resulting from the global models, ΔPCT_{MODi}, is used in the overall PCT uncertainty evaluation to determine the final estimate of $PCT^{95\%}$.

These single parameter sensitivity calculations were determined to remain applicable for the Unit 1 reanalysis methodology, as applied (Reference 67).

15.4.1.7A.4 Unit 1 Overall Peak Cladding Temperature Uncertainty Evaluation and Results

The equation used to initially estimate the 95 percentile PCT (PCT_i of Equation 15.4.1-1) was presented in Section 15.4.1.4A. Each of the uncertainty elements (ΔPCT_{ICi}, ΔPCT_{PDi}, ΔPCT_{MODi}) is considered to be independent of each other. Each element includes a correction or bias, which is added to PCT_{REFi} to move it closer to the expected, or average, PCT. The bias from each element has an uncertainty associated with the methods used to derive the bias.

Each bias component of the uncertainty elements is considered a random variable, whose uncertainty distribution is obtained directly, or is obtained from the uncertainty of the parameters of which the bias is a function. Since PCTi is the sum of these biases, it also becomes a random variable. Separate initial PCT frequency distributions are constructed as follows for the DECLG break and the limiting split break:

(1) Generate a random value of each uncertainty element (ΔPCT_{IC}, ΔPCT_{PD}, ΔPCT_{MOD})

(2) Calculate the resulting PCT using Equation 15.4.1-1

(3) Repeat the process many times to generate a histogram of PCTs

The results of this assessment showed the DECLG break to be the limiting break type.

A final verification step is performed to quantify the bias and uncertainty resulting from the superposition assumption (i.e., the assumption that the major uncertainty elements are independent). Several additional WCOBRA/TRAC calculations are performed in which variations in parameters from each of the three uncertainty elements are modeled for the DECLG break. These predictions are compared to the predictions based on Equation 15.4.1-1, and additional biases and uncertainties are applied where appropriate.

The superposition assumption verification step was performed for the Unit 1 reanalysis (Reference 67). These calculations resulted in an adjustment of the bias and uncertainty that is required for the reanalysis methodology.

The estimate of the PCT at 95 percent probability is determined by finding that PCT below which 95 percent of the calculated PCTs reside. This estimate is the licensing basis PCT, under the revised ECCS rule (10 CFR 50.46). The results of the best estimate LBLOCA analysis are presented in Table 15.4.1-2A. The difference between the 95 percentile PCT and the average PCT increases with each subsequent PCT period, due to propagation of uncertainties.

15.4.1.8A Unit 1 Additional Evaluations

Zircaloy Clad Fuel: An evaluation of Zircaloy clad fuel has shown that the Zircaloy clad fuel is bounded by the results of ZIRLO clad fuel analysis.

Integral fuel burnable absorber (IFBA) Fuel: An evaluation of IFBA fuel has shown that the IFBA fuel is bounded by the results of the non-IFBA fuel analysis.

T_{AVG} Coastdown: An end-of-cycle, full power T_{AVG} coastdown at 565°F evaluation was performed and concluded that there would be no adverse effect on the best estimate LBLOCA analysis as a T_{AVG} window between 565°F and 577.3°F was explicitly modeled in the best estimate LBLOCA analysis.

These evaluations have been shown to continue to apply for the Unit 1 reanalysis (Reference 67).

15.4.1.9A Unit 1 10 CFR 50.46 Results

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:

(1) There is a high level of probability that the PCT shall not exceed 2200°F. The 95th percentile PCT results presented in Table 15.4.1-2A indicate that this regulatory limit has been met.

(2) The local maximum oxidation (LMO) calculated in the original BELOCA analysis results (Reference 60) is based on a limiting PCT transient that is in excess of the Unit 1 reanalysis 95 percentile PCT and remains bounding for Unit 1. Based on this original conservative PCT transient, a LMO of 11 percent was calculated, which meets the 10 CFR 50.46 acceptance criterion (b)(2); i.e., "Local Maximum Oxidation of the cladding less than 17 percent," remains bounding for Unit 1, and is presented as an upper bound in Table 15.4.1-2A.

(3) The maximum core wide oxidation (CWO) determined in the original BELOCA analysis results (Reference 60) was based on limiting fuel temperatures that exceed those in the Unit 1 reanalysis and remain bounding for Unit 1. Based on these original conservative fuel temperatures, the total amount of hydrogen generated (i.e., CWO) , is 0.0089 times (0.89 percent) the maximum theoretical amount, which meets the 10 CFR 50.46 acceptance criterion (b)(3); i.e., "Core-Wide Oxidation less than 1 percent," remains bounding for Unit 1, and is presented as an upper bound in Table 15.4.1-2A.

(4) Criterion (b)(4) 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. The approved methodology (Reference 56) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low-power channel as defined in the DCPP WCOBRA/TRAC model. This situation has not been calculated to occur for DCPP Unit 1. Therefore, acceptance criterion (b)(4) is satisfied.

(5) The approved Westinghouse position on criterion (b)(5) is that this requirement is satisfied if a coolable geometry is maintained, and the core remains subcritical following the LOCA (Reference 56). This position is independent from and unaffected by the use of best estimate LOCA methodology.

15.4.1.10A Unit 1 Plant Operating Range

The expected PCT and associated uncertainty presented above for Unit 1 are valid for a range of plant operating conditions. Many parameters in the reference transient calculation are at nominal values. The range of variation of the operating parameters has been accounted for in the estimated PCT uncertainty. Table 15.4.1-7A summarizes the operating ranges for Unit 1. Note that Figure 15.4.1-15A illustrates the axial power distribution limits that were analyzed and are verified on a cycle-specific basis. Table 15.4.1-5A summarizes the LBLOCA containment data used for calculating containment back pressure. If plant operation is maintained within the plant operating ranges presented in Table 15.4.1-7A, the LOCA analyses presented in this section are considered to be valid.

15.4.1.4B Unit 2 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation Model

The thermal-hydraulic computer code, which was reviewed and approved for the calculation of fluid and thermal conditions in a PWR during a LBLOCA, is WCOBRA/TRAC Version MOD7A, Revision 1 (Reference 56). Westinghouse has since developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical Treatment of Uncertainty Method (Reference 69). This method is still based on the "Code Qualification Document" (CQD) methodology (Reference 56). 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 calculations (SER appended to Reference 69). The WCOBRA/TRAC MOD7A, Revision 6, is an evolution of Revision 1 that includes logic to facilitate the automation aspects of ASTRUM, user conveniences, and error corrections. WCOBRA/TRAC MOD7A, Revision 6, is documented in Reference 69.

A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons with experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR LBLOCA. Modeling of a PWR introduces additional uncertainties that are identified and quantified in the plant-specific analysis.

The final step in application of the best estimate methodology for Unit 2, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, LMO, and CWO at 95-percent probability, is described below.

(1) Plant Model Development

In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to ensure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences, such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

(2) Determination of Plant Operating Conditions

In this step, the expected or desired operating range of the plant to which the analysis applies is established. 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 for input 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 to as the confirmatory analysis. 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 uncertainties.

(3) Assessment of Uncertainty

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of Reference 69. 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, CWO). The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each 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.

(4) 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 or may be narrower for some parameters to gain additional margin.

15.4.1.5B Unit 2 Containment Backpressure

A conservatively bounding minimum containment back pressure (refer to Figure 15.4.1-14B) is calculated using the methods and assumptions described in Reference 2, Appendix A. Containment back pressure is calculated using the COCO code (Reference 61), the input parameters presented in Table 15.4.1-5B, mass and energy releases from the WCOBRA/TRAC calculation, and the structural heat sinks presented in Table 15.4.1-5A. Input parameters used for the Unit 2 containment backpressure calculation are presented in Table 15.4.1-5B. This minimum containment back pressure is modeled using a time dependent pressure table as a boundary condition for the best estimate LBLOCA analysis.

15.4.1.6B Unit 2 Confirmatory Studies

A few confirmatory studies were performed to establish the limiting conditions for the uncertainty evaluation. In the confirmatory studies performed, key LOCA parameters are varied over a range and the impact on the PCT is assessed.

The results for the confirmatory studies are summarized in Table 15.4.1-4B. In summary, the limiting conditions for the plant at the time the design basis accident (DBA) is postulated to occur are reflected in the final reference transient. These limiting conditions are:

(1) LOOP

(2) High RCS average temperature

(3) High SGTP of 15 percent

(4) High average power fraction in the assemblies on the core periphery (fraction of power in outer assemblies = 0.8)

15.4.1.7B Unit 2 Uncertainty Evaluation

The ASTRUM methodology (Reference 69) differs from the previously approved Westinghouse best estimate methodology (Reference 56) primarily in the statistical technique used to make a singular probabilistic statement with regard to the conformance of the system under analysis to the regulatory requirement of 10 CFR 50.46.

The ASTRUM methodology applies a non-parametric statistical technique to generate output; e.g., PCT, LMO, and CWO from a combination of WCOBRA/TRAC and HOTSPOT (Reference 68) calculations. These calculations are performed by applying a direct, random Monte Carlo sampling to generate the input for the WCOBRA/TRAC and HOTSPOT computer codes.

This approach allows the formulation of a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46. Based on the non-parametric statistical approach, the number of Monte Carlo runs is only a function of the tolerance interval and associated confidence level required to meet the desired level of safety.

15.4.1.8B Unit 2 Limiting Peak Cladding Temperature Transient Description

The DCPP Unit 2 PCT-limiting transient is a DECLG break which analyzes conditions that fall within those listed in Table 15.4.1-7B. The sequence of events following is presented in Table 15.4.1-1B. The PCT-limiting case was chosen to show a conservative representation of the response to a LBLOCA.

15.4.1.9B Unit 2 10 CFR 50.46 Results

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:

(1) Because the resulting PCT for the limiting case is 1872 °F, which represents a bounding estimate of the 95th percentile PCT at the 95-percent confidence level, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1); i.e., “Peak Cladding Temperature less than 2200 °F”, is met. The results are shown in Table 15.4.1-2B.

(2) Because the resulting LMO for the limiting case is 1.64 percent, which represents a bounding estimate of the 95th percentile LMO at the 95-percent confidence level, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2); i.e., “Local Maximum Oxidation of the cladding less than 17 percent,” is met. The results are shown in Table 15.4.1-2B.

(3) The limiting hot fuel assembly rod has a calculated maximum oxidation of 0.17 percent. Because this is the hottest fuel rod within the core, the calculated maximum oxidation for any other fuel rod would be less than this value. For the low power peripheral fuel assemblies, the calculated oxidation would be significantly less than this maximum value. The CWO is essentially the sum of all calculated maximum oxidation values for all of the fuel rods within the core. Therefore, a detailed CWO calculation is not needed because the calculated sum will always be less than 0.17 percent. Because the resulting CWO is conservatively assumed to be 0.17 percent,

which represents a bounding estimate of the 95th percentile CWO at the 95-percent confidence level, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3); i.e., “Core-Wide Oxidation less than 1 percent,” is met. The results are shown in Table 15.4.1-2B.

(4) Criterion (b)(4) 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. The approved methodology (Reference 56) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low-power channel as defined in the DCPP WCOBRA/TRAC model. This situation has not been calculated to occur for DCPP Unit 2. Therefore, acceptance criterion (b)(4) is satisfied.

(5) The approved Westinghouse position on Criterion (b)(5) is that this requirement is satisfied if a coolable geometry is maintained, and the core remains subcritical following the LOCA (Reference 56). This position is independent from and unaffected by the use of best estimate LOCA methodology.

15.4.1.10B Unit 2 Plant Operating Range

The accepted PCT and its uncertainty developed previously are valid for a range of Unit 2 plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 15.4.1-7B summarizes the operating ranges for DCPP Unit 2 as defined for the proposed operating conditions, which are supported by the best estimate LBLOCA analysis. Table 15.4.1-5B summarizes the LBLOCA containment data used for calculating containment back pressure. It should be noted that other non-LBLOCA analyses may not support these ranges. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T_{avg}).

15.4.1.11 Conclusions (Common)

15.4.1.11.1 10 CFR 50.46 Acceptance Criteria

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:

(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level such that the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1); i.e., “Peak Cladding Temperature less than 2200 ºF”, is demonstrated.

(2) 10 CFR 50.46 acceptance criterion (b)(2), requires that the maximum calculated reduction in fuel cladding thickness at any location in the core due to the Zr-H_2O reaction shall be less than 17 percent of the original cladding thickness. Because the Zr-H_2O reaction essentially oxidizes the fuel cladding and generates hydrogen as a by-product, the reduction in cladding thickness is evaluated based on the amount of H_2 generated (i.e., oxidation) at a given core location. The BELOCA methodology calculates the LMO, which corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level such that the analysis confirms that the 10 CFR 50.46 acceptance criterion (b)(2); i.e., “Local Maximum Oxidation of the Cladding Less than 17 percent,” is demonstrated.

(3) 10 CFR 50.46 acceptance criterion (b)(3) requires that the total quantity of fuel cladding oxidized due to the Zr-H_2O reaction shall be less than 1 percent, which is verified by ensuring the total calculated amount of H_2 generated is less than 1 percent of the theoretical maximum possible if all of the fuel cladding in the core was oxidized. The BELOCA methodology calculates the limiting CWO which corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level such that the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3); i.e., “Core-Wide Oxidation Less than 1 percent,” is demonstrated.

(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. The approved methodology (Reference 56) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless fuel grid crushing extends beyond the assemblies representing the low-power channel.

(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. The approved Westinghouse position on this criterion is that this requirement is satisfied if a coolable geometry is maintained, and the core remains subcritical following the LOCA (Reference 56). This position is independent from and unaffected by the use of best estimate LOCA methodology.

15.4.1.11.2 Radiological

Section 15.5.17 concludes that the resulting potential exposures have been found to be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.2 MAJOR SECONDARY SYSTEM PIPE RUPTURE

Three major secondary system pipe ruptures are analyzed in this section: rupture of a main steam line at hot zero power, rupture of a main feedwater pipe, and rupture of a main steam line at power. The time sequence of events for each of these events is provided in Table 15.4-8.

15.4.2.1 Rupture of a Main Steam Line at Hot Zero Power

15.4.2.1.1 Acceptance Criteria

The following limiting criteria are applicable for a main steam line rupture at hot zero power:

15.4.2.1.1.1 Fuel Damage Criteria

Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This is conservatively demonstrated by meeting the following criteria:

(1) DNB will not occur on the lead rod with at least a 95 percent probability at a 95 percent confidence level. The minimum DNBR must not go below the applicable limit value of 1.45 at any time during the transient.

15.4.2.1.1.2 Radiological Criteria

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.2.1.2 Identification of Causes and Accident Description

The steam release from a rupture of a main steam pipe would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. 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 a positive reactivity insertion and subsequent reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the SIS and accumulators.

In order to allow for routine plant heatups and cooldowns, plant procedures allow the SIS to be blocked per permissive P-11, provided that the RCS boron concentration is maintained at a value greater than or equal to the cold shutdown margin requirement. As discussed in Reference 63, this additional shutdown margin ensures that there would be no return to power for a steam pipe rupture such that the analysis of a rupture of a steam line at hot zero power remains bounding.

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

(1) Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the ESFs there is no consequential damage to the primary system and the core remains in place and intact.

(2) Energy release to containment from the worst steam pipe break does not cause failure of the containment structure (refer to Appendix 6.2D).

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is met for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide protection for a steam line rupture:

(1) SIS actuation from any of the following:

(a) Two-out-of-four low pressurizer pressure signals

(b) Two-out-of-three low steam line pressure signals in any one loop

(c) Two-out-of-three high containment pressure signals

(2) The overpower reactor trips (neutron flux and ΔT), the overtemperature ΔT reactor trip, and the reactor trip occurring in conjunction with receipt of the SI signal.

(3) Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, a SI signal will rapidly close all MFRVs, trip the main feedwater pumps, and close the MFIVs that backup the control valves.

(4) Closure of the fast acting main steam line isolation valves on: (refer to Figure 7.2-1 and the Technical Specifications [Reference 30])

(a) Two-out-of-three low steam line pressure signals in any one loop

(b) Two-out-of-four high-high containment pressure

(c) Two-out-of-three high negative steam line pressure rate signals in any one loop (used only during cooldown and heatup operations)

The fast-acting isolation valves are provided in each main steam line and will fully close within 10 seconds of a large steam line break. 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 SG would blow down even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

The effective throat area of the integral flow restrictors in the SGs is 1.388 ft^2, which is considerably smaller than the area of the main steam pipe. These restrictors serve to limit the maximum steam flow for any break at any location.

15.4.2.1.3 Analysis of Effects and Consequences

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

(1) The plant transient conditions, including core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The RETRAN-02W code (Reference 70) has been used.

(2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC (refer to Section 1.6.1, Item 28, and Section 4.4.3), has been used to determine if DNB occurs for the core conditions computed in (1) above.

The following conditions were assumed to exist at the time of a main steam line break (MSLB) accident.

(1) EOL shutdown margin at no-load, equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

(2) The negative moderator coefficient corresponds to the EOL rodded core with the most reactive rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1050 psia corresponding to the negative moderator temperature coefficient, plus the Doppler temperature effect used is shown in Figure 15.4.2-2. The effect of power generation in the core on overall reactivity is shown in Figure 15.4.2-1.

The core properties associated with the sector nearest the affected SG and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. To verify the conservatism of this method, the reactivity as well as the power distribution was checked with the advanced nodal code core model (refer to Section 4.3.3.10.3). These core analyses 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 non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was confirmed that the reactivity feedback model employed in the RETRAN-02W kinetics analysis was consistent with the core analysis and the overall analysis is conservative.

(3) The modeling of the SIS in RETRAN-02W is described in Reference 70. The minimum boric acid solution concentration of 2300 ppm in the RWST is assumed. The SIS piping downstream of the RWST isolation valves is assumed to contain no boron (0 ppm), which delays the delivery of boron to the RCLs from the RWST water. With this conservative assumption, the SIS and accumulators combine to limit the return to power. Cases were examined for both minimum and maximum SIS flow rates.

For the minimum SIS flow rate cases the most restrictive single failure in the SIS is considered. The SIS flow assumed conservatively corresponds to that delivered by only one high-head charging pump delivering full flow to the cold leg header. The charging pump (CCP1 or CCP2) is assumed to begin providing flow to the RCS at 25 seconds after receipt of the SI signal for the case in which offsite power is assumed available, and at 35 seconds for the case where offsite power is not available; the additional 10-second delay is assumed to start the diesels and load the necessary SI equipment onto them.

For the maximum SIS flow rate cases, a flow profile was assumed that bounds the maximum flow from two high-head charging pumps (CCP1 and CCP2) plus two intermediate-head SI pumps plus the nonsafety-related CVCS charging pump (CCP3). A 2-second signal delay was assumed.

For this analysis, it was determined that the maximum SIS flow rate assumption is conservative for the more limiting case with offsite power available, due to the effect of higher SIS flow on the timing of cold leg accumulator actuation. The cold leg accumulators provide an additional source of borated water to the core when the RCS pressure decreases below the actuation setpoint. The minimum accumulator boron concentration of 2200 ppm is assumed, along with a conservatively low actuation setpoint of 577.2 psia. Actuation of the accumulators causes a significant influx of boron, which rapidly shuts down the reactor. Assuming the maximum SIS flow rate slows down the rate of the RCS pressure decrease and thus delays the accumulator actuation. If the most reactive RCCA is assumed stuck in its fully withdrawn position after a reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that would exist assuming the most-reactive RCCA to be stuck in its fully withdrawn position. Therefore, the limiting case presented herein conservatively assumes a maximum SIS flow rate.

(4) Because the SGs are equipped with integral flow restrictors with a 1.388 ft^2 throat area, any rupture with a break greater than this size, regardless of the location, would have the same effect on the reactor as a 1.388 ft^2 break. The following two cases have been considered in determining the core power and RCS transients:

(a) Complete severance of a pipe with the plant initially at no-load conditions and with offsite power available. Full reactor coolant flow is maintained.

(b) Complete severance of a pipe with the plant initially at no-load conditions and with offsite power unavailable. LOOP results in RCP coastdown.

(5) Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at EOL. 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 control assembly 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 on the core power, operating history, temperature, pressure, and flow.

All the cases above assume initial hot shutdown conditions at time zero, because this represents the most limiting 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.

However, because the initial SG water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

(6) In computing the steam flow during a steam line break, the Moody Curve (Reference 16) for fl/D = 0 is used.

(7) Perfect moisture separation in the SG is assumed. This assumption leads to conservative results because, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core.

(8) To maximize the primary-to-secondary heat transfer rate, 0 percent SGTP is assumed.

(9) All main and AFW are assumed to be operating at full capacity when the rupture occurs. This assumption maximizes the cooldown. A conservatively high AFW flow rate of 1700 gpm at a minimum temperature of 60ºF is assumed to be delivered to the affected SG. Main feedwater is isolated 64 seconds following the SI signal by closure of the MFIVs. No credit is taken for the faster-closing MFRVs. AFW continues for the duration of the transient.

(10) The effect of heat transferred from thick metal in the RCS and the SGs is not included in the cases analyzed. The heat transferred from these sources would be a net benefit because it would slow the cooldown of the RCS.

15.4.2.1.4 Results

The double-ended rupture of a main steam line at zero power was analyzed for both Unit 1 and Unit 2; however, only the results from the slightly more limiting Unit 1 cases are presented. Unit 2 results are similar. The time sequence of events, both with and without offsite power available for Unit 1, are presented in Table 15.4-8.

Figures 15.4.2-4 through 15.4.2-6 show the plant response following a main steam pipe rupture. Offsite power is assumed to be available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one SG.

Figures 15.4.2-7 through 15.4.2-9 show the plant response for the case with a LOOP. This assumption results in a coastdown of the RCPs. In this case, the core power increases at a slower rate and reaches a lower peak value than in the case with offsite power available. The ability of the emptying SG to extract heat from the RCS is reduced by the decreased flow in the RCS.

It should be noted that following a steam line break only one SG blows down completely. Thus, the remaining SGs are still available for dissipation of decay heat after the initial transient is over. In the case with LOOP, this heat would be removed to the atmosphere via the MSSVs.

15.4.2.1.5 Conclusions

The analysis demonstrates the acceptance criteria are met as follows:

15.4.2.1.5.1 Fuel Limits

Based on the results of the analysis, the core will remain in place and intact with no loss of core cooling capability.

A DNB analysis was performed for the limiting steam line break case with offsite power available as described above. The analysis demonstrated that the minimum DNBR remains well above the limit value of 1.45. Therefore, the DNB design basis is met for the steam line break event initiated from zero power.

15.4.2.1.5.2 Radiological

Section 15.5.18 concludes that potential exposures from major steam line ruptures will be well below the guideline levels specified in 10 CFR Part 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Acceptance Criteria

The following limiting criteria are applicable for a main feedwater pipe rupture:

15.4.2.2.1.1 Fuel Damage Criteria

Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This is conservatively demonstrated by meeting the following criteria:

(1) With respect to fuel damage due to “dryout” where the water level in the vessel drops below the top of the core, criterion that no bulk boiling occurs in the primary coolant system prior to event “turnaround” is applied. Turnaround is defined as the point when the heat removal capability of the SGs, being fed by AFW, exceeds NSSS heat generation.

15.4.2.2.1.2 Maximum Reactor Coolant System and Main Steam System Pressure Requirements:

The maximum pressure in the RCS and MSS should be maintained below 110 percent of the design value, 2748.5 psia and 1208.5 psia, respectively.

15.4.2.2.1.3 Radiological Criteria

The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000 for the Main Steam Line Break.

15.4.2.2.2 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell-side fluid inventory in the SGs. If the break is postulated in a feedline between the check valve and the SG, fluid from the SG may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of AFW to the affected SG. (A break upstream of the feedline check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.8).

Depending on 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 15.4.2.1. Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

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

(1) Feedwater to the SGs is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip

(2) Liquid in the SG may be discharged through the break, and would then not be available for decay heat removal after trip

(3) The break may be large enough to prevent the addition of any main feedwater after trip

The following provide the necessary protection against a main feedwater line rupture:

(1) A reactor trip on any of the following conditions:

    (a) High pressurizer pressure

(b) Overtemperature ΔT

(c) Low-low SG water level in any SG

(d) SI signals from any of the following:

- Low steam line pressure
- High containment pressure

(Refer to Chapter 7 for a description of the actuation system)

(2) An AFW system to provide an assured source of feedwater to the SGs for decay heat removal (refer to Chapter 6 for a description of the AFW system)

15.4.2.2.3 Analysis of Effects and Consequences

The feedline break transient is analyzed using the RETRAN-02W computer code described in Reference 70. The RETRAN-02W model simulates the RCS, neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, SGs, feedwater system, and MSSVs. The code computes pertinent plant variables including SG mass, pressurizer water volume, reactor coolant average temperature, RCS pressure, and SG pressure.

The feedline rupture analysis methodology presented in Section 15.4.2.2 is not intended to minimize the predicted time to pressurizer filling, as this scenario is evaluated in Section 15.4.2.4.

Major assumptions are:

(1) The plant is initially operating at 102 percent of the NSSS rating, including a conservatively large RCP heat of 20 MWt for the case with offsite power available and a nominal (minimum guaranteed) RCP heat of 14 MWt for the case without offsite power available. These assumptions maximize the primary side heat that must be removed for each case.

(2) Initial reactor coolant average temperature is 5.0°F above the nominal value, and the initial pressurizer pressure is 60 psi above its nominal value.

(3) The initial pressurizer level is set to the nominal full power programmed level plus an uncertainty of +5.7 percent span for Diablo Canyon Unit 1 and Unit 2, resulting in an initial pressurizer level of 66.4 percent span and 66.8 percent span, respectively. Initial SG water level is at 75 percent narrow range span (NRS) in the faulted SG, and at 55 percent NRS in the intact SGs.

(4) No credit is taken for the pressurizer PORVs or pressurizer spray.

(5) No credit is taken for the high pressurizer pressure reactor trip.

(6) Main feed to all SGs is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

(7) The break discharge quality is calculated by RETRAN-02W as a function of pressure and temperature.

(8) Reactor trip is assumed to be initiated when the low-low level trip setpoint in the ruptured SG is reached. A low-low level setpoint of 0 percent NRS is assumed.

(9) A double-ended break area of 0.5184 ft^2 is assumed. A break area of 0.5184 ft^2 corresponds to the flow area of the reducer leading to the feedring, and is the largest effective area of flow out of the SGs for the feedline break event. This minimizes the SG fluid inventory available for removal of long-term decay heat and stored energy following reactor trip, and thereby maximizes the resultant heatup of the reactor coolant.

(10) No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

(11) No credit is taken for charging or letdown.

(12) The SG heat transfer correlation for the SG tubes is automatically adjusted by RETRAN-02W as the shell-side inventory decreases.

(13) Conservative core residual heat generation based on the ANSI/ANS-5.1-1979 (Reference 32) decay heat standard plus uncertainty was used for calculation of residual decay heat levels.

(14) The AFW is assumed to be initiated 10 minutes after the trip with a feed rate of 390 gpm

15.4.2.2.4 Results

Analyses were performed for both Unit 1 and Unit 2 separately; the most limiting case with offsite power and the corresponding case without offsite power are presented.

Results for two feedline break cases are presented. Results for a case in which offsite power is assumed to be available are presented in Section 15.4.2.2.4.1. Results for a case in which offsite power is assumed to be lost following reactor trip are presented in Section 15.4.2.2.4.2. The calculated sequence of events for both cases is listed in Table 15.4-8.

15.4.2.2.4.1 Feedline Rupture with Offsite Power Available

The system response following a feedwater line rupture, assuming offsite power is available, is presented in Figures 15.4.2-10 through 15.4.2-13. Results presented in Figures 15.4.2-11 and 15.4.2-13 show that pressures in the RCS and MSS remain below 110 percent of the design pressures, 2748.5 psia and 1208.5 psia, respectively. Pressurizer pressure decreases after reactor trip on low-low SG water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to the PSV setpoint. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the SGs. Figure 15.4.2-11 indicates a pressurizer water volume equivalent to a water-solid condition; however, this is not an acceptance criteria for the analysis. Pressurizer filling during a main feedwater pipe rupture event is evaluated in Section 15.4.2.4. At approximately 5900 seconds, decay heat generation decreases to a level such that the total RCS heat generation (decay heat plus pump heat) is less than AFW heat removal capability, and RCS pressure and temperature begin to decrease.

The results show that the core remains covered at all times and that no boiling occurs in the RCLs.

15.4.2.2.4.2 Feedline Rupture with Offsite Power Unavailable

The system response following a feedwater line rupture without offsite power available is similar to the case with offsite power available. However, as a result of the LOOP (assumed to occur at reactor trip), the RCPs coast down. This results in a reduction in total RCS heat generation by the amount produced by pump operation.

The reduction in total RCS heat generation produces a milder transient than in the case where offsite power is available. Results presented in Figures 15.4.2-14 through 15.4.2-17 show that pressure in the RCS and MSS remain below 110 percent of the design pressures, 2748.5 psia and 1208.5 psia, respectively. Pressurizer pressure decreases after reactor trip on low-low SG water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to a peak pressure of 2426 psia at 106 seconds. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the SGs. Figure 15.4.2-15 shows that the water volume in the pressurizer increases in response to the heatup, but does not fill the pressurizer. At approximately 2200 seconds, decay heat generation decreases to a level less than the AFW heat removal capability, and RCS temperature begins to decrease. The results show that the core remains covered at all times and that no boiling occurs in the RCLs.

15.4.2.2.5 Conclusions

Results of the analysis show that for the postulated feedline rupture, the assumed AFW system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. The analysis documents that the acceptance criteria for a postulated feedline rupture are met as follows:

15.4.2.2.5.1 Fuel Damage

Any fuel damage calculated to occur is of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This is conservatively demonstrated by Figures 15.4.2-12 and 15.4.2-16 that show no bulk boiling occurs in the primary coolant system prior to event “turnaround”.

15.4.2.2.5.2 Maximum Reactor Coolant System and Main Steam System Pressure

As shown in Figures 15.4.2-11 and 15.4.2-13, the maximum pressure in the RCS and MSS is maintained below 110 percent of the design value, 2748.5 psia and 1208.5 psia, respectively.

15.4.2.2.5.3 Radiological

Section 15.5.19 concludes that potential exposures from major feedwater line ruptures will be well below the guideline levels specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000 for the Main Steam Line Break, and that the occurrence of such ruptures would not result in undue risk to the public.

15.4.2.3 Rupture of a Main Steam Line at Full Power

15.4.2.3.1 Acceptance Criteria

The following limiting criteria are applicable for a main steam line rupture at full power:

15.4.2.3.1.1 Fuel Damage Criteria

Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This is conservatively demonstrated by meeting the following criteria:

(1) DNB will not occur on the lead rod with at least a 95 percent probability at a 95 percent confidence level. The minimum DNBR must not go below

the DNBR Safety Analysis Limit of 1.68/1.71 (refer to Section 4.4.4.1) at any time during the transient.

(2) The peak linear heat generation rate will not exceed a 22 kW/ft (refer to Section 4.4.4.2 and Figure 4.4-2) which would cause fuel centerline melt.

15.4.2.3.1.2 Radiological Criteria

The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.2.3.2 Identification of Causes and Accident Description

A rupture in the MSS piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the RCS via the SGs. 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 15.4.2.1, is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod, which remains fully withdrawn out of the core. This condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin, or after the plant has been tripped manually, or by the reactor protection system following a steam line break from an at-power condition. For an at-power break, the Section 15.4.2.1 analysis represents the limiting condition with respect to core protection for the time period following reactor trip. The analysis of a main steam pipe rupture at power is performed to demonstrate that the following criteria are satisfied:

(1) Assuming a stuck RCCA and a single failure in the ESFs, there is no damage to the primary system and the core remains in place and intact.

(2) Core protection is maintained prior to, and immediately following, a reactor trip, if one is required, such that the DNBR remains above the applicable limit value for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

Depending on the size of the break, this event is classified as either an ANS Condition III (infrequent fault) or Condition IV (limiting fault) event. The main steam pipe rupture at power is protected by the same reactor protection and ESF functions as the main steam pipe rupture at hot zero power. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the analysis shows that the calculated DNBR remains above the applicable DNBR limit value.

15.4.2.3.3 Analysis of Effects and Consequences

The analysis of the steam line rupture is performed in the following stages:

(1) The RETRAN-02W code (Reference 70) is used to calculate the nuclear power, core heat flux, and RCS temperature and pressure transients resulting from the cooldown following the steam line break.

(2) The core radial and axial peaking factors are determined using the thermal-hydraulic conditions from the transient analysis as input to the nuclear core models. The THINC-IV code (refer to Section 4.4.3) is then used to calculate the DNBR for the limiting time during the transient.

This accident is analyzed with the ITDP as described in Reference 62.

To give conservative results in calculating the DNBR during the transient, the following assumptions are made:

(1) Initial Conditions - The initial core power, reactor coolant temperature, and RCS pressure are assumed to be at their nominal full-power values. The full power condition is more limiting than part-power with respect to DNBR. Uncertainties in initial conditions are included in the DNBR limit value, as described in Reference 62.

(2) Break size - A spectrum of break sizes is 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 may 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 SI signal actuated by low steam line pressure, which includes lead/lag dynamic compensation.

(3) Break flow - The steam flow out the pipe break is calculated using the Moody curve for an fL/D value of 0 (Reference 16).

(4) Reactivity Coefficients - The analysis assumes maximum EOL moderator reactivity feedback and minimum Doppler-only power reactivity feedback in order to maximize the power increase following the break.

(5) Protection System - The analysis only models those reactor protection system features that would be credited for at power conditions and up to the time a reactor trip is initiated. Section 15.4.2.1, presents the analysis

of the bounding transient following reactor trip, where ESFs are actuated to mitigate the effects of a steam line break.

(6) Control Systems - The results of a main steam pipe rupture at power would be made less severe as a result of control system actuation. Therefore, the mitigation effects of control systems have been ignored in the analysis.

15.4.2.3.4 Results

A spectrum of steam line break sizes was analyzed for each unit. The results show that for break sizes up to 0.49 ft^2 (Unit 1) and 0.50 ft^2 (Unit 2) a reactor trip is not generated. In this case, the event is similar to an excessive load increase event as described in Section 15.2.12. The core reaches a new equilibrium condition at a higher power equivalent to the increased steam flow. For break sizes larger than those noted above, a reactor trip is generated within a few seconds of the break on the SI signal from low steam line pressure.

The limiting case for demonstrating DNB protection is the 0.49 ft^2 (Unit 1) break, the largest break size that does not result in an early trip on low steam pressure SI actuation. The peak linear heat rate (kW/ft) remains below a value corresponding to fuel centerline melting. The time sequence of events for this case is shown in Table 15.4-8. Figures 15.4.2-18 through 15.4.2-21 show the transient response.

15.4.2.3.5 Conclusions

The analysis demonstrates the acceptance criteria are met as follows:

15.4.2.3.5.1 Fuel Damage

Any fuel damage calculated to occur is of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability. This is conservatively demonstrated by the following:

(1) The analysis demonstrates that there is a large margin to the DNBR Safety Analysis Limit of 1.71/1.68 (typical cell/thimble cell).

(2) The analysis calculates that the maximum linear power meets the fuel centerline melt limit of 22.0 kW/ft.

The analysis concludes that the DNB and fuel centerline 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 shows that the minimum DNBR remains above the safety analysis limit.

15.4.2.3.5.2 Radiological

Section 15.5.18 concludes that potential exposures from main steam line ruptures at full power will be well below the guideline levels specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000, and that the occurrence of such ruptures would not result in undue risk to the public.

15.4.2.4 Major Rupture of a Main Feedwater Pipe for Pressurizer Filling

15.4.2.4.1 Acceptance Criteria

The acceptance criterion is to ensure the major rupture of a main feedwater pipe (hereinafter referred to as feedwater line break [FLB]) for pressurizer filling event does not result in liquid water (hereinafter referred to as water) relief through the PSVs in order to prevent an unisolable reactor coolant pressure boundary breach due to a PSV failing open. This can be accomplished through appropriate operator actions and equipment design/response that mitigate the consequences of the event before water relief through the PSVs occurs.

15.4.2.4.2 Identification of Causes and Accident Description

The causes and accident description for the pressurizer filling analysis of the main feedwater pipe rupture described in this section are discussed generally in Section 15.4.2.2.2. The aspects that relate specifically to pressurizer filling follow.

Following a FLB accident, secondary water level decreases in the SGs until AFW flow is initiated, after which level will begin to recover in the SGs being fed with AFW flow (i.e., the intact SGs). Depending on the AFW flow available, there is the potential for an increase in reactor coolant temperatures in the early part of the post-trip transient, along with an increase in reactor coolant volume due to thermal expansion. Also, following initiation of the FLB accident, a low steam line pressure setpoint will be reached in the faulted loop, causing actuation of the SI signal and start of the two PG&E Design Class I charging pumps. The reactor coolant inventory addition from the charging flow and RCS thermal expansion contributes to pressurizer filling.

If pressurizer filling occurs, the pressurizer PORVs are available to relieve water inventory from the RCS, as long as an air supply is available from instrument air to containment or from the PG&E Design Class I backup nitrogen accumulators. Also, since Technical Specifications define a PORV as operable with its block valve closed if the PORV can be made available for automatic pressure relief, operators may need to take action to open the block valve to enable the PORV to provide water relief. Using the PORVs to relieve water from the RCS precludes water relief through the PSVs, which could render the PSVs inoperable.

Mitigation of the pressurizer filling condition is complete when (1) the heat removal capability of the SGs being fed by AFW exceeds NSSS heat generation and stops thermal expansion of the RCS and (2) operator actions are taken to isolate charging flow, and subsequently stop RCP seal injection flow, which terminates all remaining reactor coolant inventory addition.

The pressurizer filling analysis models the long term plant response to a FLB to demonstrate that operator actions, if taken in a timely manner, preclude water relief through the PSVs. The operator actions for mitigation of a FLB accident are included in the plant EOPs.

15.4.2.4.3 Analysis of Effects and Consequences

The FLB transient is analyzed for pressurizer filling in accordance with the NRC approved methodology for a 4-loop plant (Reference 70) using the Westinghouse version of the RETRAN-02 computer code (RETRAN-02W), which is also used for the analysis of the FLB transient described in Section 15.4.2.2.3.

Separate cases to accommodate different limiting assumptions were analyzed to determine the time by which the operators would need to ensure a PG&E Design Class I PORV is available and the times by which the operators would need to isolate charging flow and subsequently stop RCP seal injection flow. Cases were also analyzed with and without offsite power available to determine the more limiting condition.

The assumptions for the pressurizer filling analysis are conservatively chosen to minimize the time to reach a water-solid condition and maximize the number of pressurizer PORV relief open/close cycles predicted. Sensitivity studies were performed for a number of parameters to determine the appropriate conservative assumptions. Major assumptions are the same as described in Section 15.4.2.2.3 with the following changes:

(1) Initial reactor coolant average temperature is 5.5°F below the nominal value, and the initial pressurizer pressure is 60 psi below its nominal value.

(2) No credit is taken for relief through the PORV that is actuated on a compensated pressurizer pressure deviation signal (i.e., the non-safety-grade PORV). However, relief through the PORVs that are actuated on the indicated (measured) pressurizer pressure signal (i.e., the safety-grade, PG&E Design Class I PORVs) has been modeled with assumptions that maximize the number of PORV opening cycles experienced. The number of safety-grade PORVs available for relief (i.e., either one or both of the PG&E Design Class I PORVs) depends on the single failure being considered.

    Also, since an SI signal causes Phase A containment isolation and the instrument air is a PG&E Design Class II (non-safety-grade) system, there is a loss of instrument air to containment due to this signal. Accordingly, the PG&E Design Class I backup nitrogen accumulators are needed to maintain functionality of the PG&E Design Class I PORVs. The backup nitrogen accumulators are each sized and leak tested to ensure at least 300 PORV cycles before the backup nitrogen supply is depleted, after which the PORV would be unavailable. Therefore, transient mitigation must be demonstrated to occur before 300 PORV cycles is exceeded.

(3) No credit is taken for normal charging or letdown flow.

(4) The TDAFWP is aligned to all four SGs, whereas the MDAFWPs are each independently aligned to two of the four SGs. The AFW flow for each case analyzed depends on the assumed single failure.

With the single failure of the TDAFWP considered, it is assumed that 390 gpm total AFW flow will be delivered to two of the intact SGs at 1 minute after the trip and an additional 195 gpm of AFW flow will be delivered to the third intact SG at 10 minutes after the trip. The AFW flow initiated at 1 minute after the trip is delivered from the MDAFWP that is aligned to two intact SGs. All flow from the other MDAFWP aligned to both the third intact SG and the faulted SG is initially assumed to spill out the break. Subsequently, a time critical operator action (TCOA) is taken within 10 minutes to isolate the faulted SG and direct AFW flow from this MDAFWP to the third intact SG.

With the single failure of a PG&E Design Class I pressurizer PORV considered, the AFW flow from the MDAFWPs is the same as described above. However, with this scenario it is also assumed the TDAFWP will deliver an additional total 585 gpm of AFW flow to the three intact SGs at 10 minutes after the trip when the TCOA is taken to isolate the faulted SG.

(5) Maximum SI flow rates were conservatively modeled with a flow profile that bounds the maximum flow from the two PG&E Design Class I high-head CCPs (CCP1 and CCP2), plus the non-safety-related CVCS charging pump (CCP3), plus two intermediate-head SI pumps. Full SI flow was conservatively assumed to occur immediately after the SI actuation signal. The maximum SI flow profile, which includes RCP seal injection flow, is modeled until the TCOA is taken to isolate charging flow. Note that no flow is actually injected from the intermediate-head SI pumps, since RCS pressure remains above the shutoff head of these pumps during the transient.

(6) Maximum RCP seal injection flow was conservatively modeled until the TCOA is taken to stop it. A limiting FLB inside containment may cause the high-high containment pressure setpoint to be reached, resulting in Phase B isolation and a loss of component cooling water (CCW) to the RCPs. Accordingly, RCP seal injection flow must be maintained to ensure RCP cooling until operator action can be taken to reset the Phase B containment isolation and restore CCW flow to the RCPs.

(7) The air-operated pressurizer spray valves are assumed to be inoperable, since instrument air to containment is lost on an SI signal and normal pressurizer spray flow is unavailable following coastdown of the RCPs. There are auxiliary spray flow lines that are equipped with backup nitrogen if the spray valves are unavailable; however, auxiliary spray requires a manual alignment that would not be completed until after the TCOAs necessary to mitigate this transient are complete.

(8) For the cases with a LOOP, the pressurizer heaters are assumed to be inoperable, since they are not automatically loaded onto an EDG bus and will not be manually loaded onto the EDGs until after the TCOAs to mitigate this transient are complete. For cases with offsite power available, the pressurizer heaters are assumed to be operable.

(9) For cases with LOOP, the RCPs are assumed to trip automatically following reactor trip. For cases with offsite power, the RCPs continue to operate unless manually tripped by the operators. The EOPs direct the operators to trip the RCPs within 5 minutes following the Phase B containment isolation (to protect the RCP motors, which are cooled by CCW). For the case to determine time by which the operators need to ensure a pressurizer PORV is available, it is assumed the operators manually trip the RCPs at greater than 90 seconds after FLB initiation, because Phase B containment isolation from high-high containment pressure would not occur before this time. However, for the case analyzed to determine the times by which the operators would need to isolate charging flow and subsequently stop RCP seal injection flow, it was conservatively assumed that the RCPs are manually tripped following reactor trip.

15.4.2.4.4 Results

The results for Unit 2 were more limiting than those calculated for Unit 1.

With respect to the single failure scenarios, it was found the failure of the TDAFWP is limiting for the calculation of minimum time to pressurizer filling, unless one of the two PG&E Design Class I PORVs is blocked at the start of the transient. If a PORV is blocked, the failure of the other PG&E Design Class I PORV is limiting for pressurizer filling. The failure of a PG&E Design Class I PORV is also limiting for the calculation of the operator action times required to ensure that transient mitigation is complete before the maximum number of PORV cycles is reached.

For cases with offsite power available, pressurizer pressure is maintained after the RCP seal injection flow is stopped, since the pressurizer heaters (specifically, the backup heaters, actuated on high pressurizer level deviation) continue to operate. However, as a steam bubble forms again in the pressurizer and the pressurizer water volume begins decreasing, relief flow switches from water to steam. Because there is no longer a concern relative to water relief through the PSVs, transient mitigation is complete. For cases with a LOOP, pressurizer pressure decreases after the RCP seal injection flow is stopped. Because the pressurizer PORV and PSV setpoints are no longer challenged, transient mitigation is complete for these cases.

The results of the FLB analysis for pressurizer filling demonstrate that, if the pressurizer fills, the following TCOAs preclude water relief through the PSVs:

(1) Ensure a pressurizer PORV is available within 8.6 minutes

If no pressurizer PORV relief is available at the start of the transient because of a failure of one of the PG&E Design Class I PORVs and isolation of the other by its respective block valve, operator action is required to ensure a PG&E Design Class I PORV is available in time to prevent water relief through the PSVs. The analysis determined that the minimum time to pressurizer filling is 8.3 minutes and the minimum time to subsequently lift the PSVs is 8.6 minutes; therefore, the operators must ensure a PG&E Design Class I PORV is available within 8.6 minutes of event initiation.

The system response for the limiting case with no pressurizer PORV relief available at the start of the transient is presented in Figures 15.4.2-22 and 15.4.2-23. The calculated sequence of events is listed in Table 15.4-8.

(2) Isolate the faulted SG within 10 minutes

Similar to the main feedwater pipe rupture analysis discussed in Section 15.4.2.2, the operators are assumed to isolate the faulted SG within 10 minutes after the low-low SG water level setpoint is reached in accordance with operating procedures. This directs all available AFW flow to the intact SGs.

(3) Isolate charging flow within 25 minutes and

(4) Stop RCP seal injection flow within 45 minutes

The results of the limiting case determined that in order for transient mitigation to occur before the maximum number of PORV cycles is reached, the operators must isolate charging flow within 25 minutes after the low-low SG water level setpoint is reached and subsequently stop RCP seal injection flow within 45 minutes after the low-low SG water level setpoint is reached. These actions ensure that a steam bubble is formed in the pressurizer and the pressurizer water volume begins to decrease, causing relief flow to switch from water to steam before the capacity of the backup nitrogen accumulators is depleted. Once this occurs, there is no longer a concern relative to water relief through the PSVs and transient mitigation is complete for these cases.

The system response is presented in Figures 15.4.2-24 through 15.4.2-27.
Table 15.4-8, "Sequence of Events," indicates that a steam bubble forms again in the pressurizer at 6723 seconds. This occurs at cycle 295 before the maximum number of 300 PORV cycles is reached at 7137.6 seconds.

Summary of TCOAs

The TCOAs established for mitigation of the FLB event are summarized below. All TCOA times are from event initiation.

1. Ensure a PG&E Design Class I pressurizer PORV is available within 8.6 minutes
2. Isolate the faulted SG within 10 minutes
3. Isolate charging flow within 25 minutes
4. Stop RCP seal injection flow within 45 minutes

15.4.2.4.5 Conclusion

The results of the FLB analysis for pressurizer filling show that operator actions, when taken in a timely manner, will preclude water relief through the PSVs. Thus, the reactor coolant pressure boundary integrity is maintained.

15.4.3 STEAM GENERATOR TUBE RUPTURE

15.4.3.1 Acceptance Criteria

The following limiting criteria are applicable for a SGTR:

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in Section 15.5.20.

(2) There are no regulatory acceptance criteria associated with a SGTR margin to overfill (MTO) transient analysis. However, it will be demonstrated that there is sufficient margin to prevent overfill of the SG during an SGTR event. Overfill of the SG may result in significantly increased offsite dose consequences, along with damage to secondary components such as the turbine and the main steam line.

15.4.3.2 Identification of Causes and Accident Description

The accident examined is the complete severance of a single SG 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 amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident LOOP, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the SG PORVs (and safety valves if their setpoint is reached).

Although the SG tube material is thermally treated Inconel 690, a highly ductile material, it is assumed that complete severance could occur. 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 that exceeds the limits established in the Technical Specifications (Reference 30) is not permitted during the unit operation.

The operator is expected to determine that a SGTR has occurred, to identify and isolate the ruptured SG, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured unit. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale that ensures that break flow to the secondary system is terminated before water level in the affected SG rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

(1) Pressurizer low pressure and low-level alarms are actuated 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 affected SG is reduced due to the break flow that is now being supplied to that unit.

(2) The main steam line radiation monitors, the air ejector radiation monitor and/or the SG blowdown radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system, and SG blowdown will be automatically terminated.

(3) Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or overtemperature ΔT. An SI signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The SI signal automatically terminates normal feedwater supply and initiates AFW addition.

(4) The reactor trip automatically trips the turbine and, if offsite power is available, the 40 percent condenser dump valves open permitting steam dump to the condenser. In the event of a coincident LOOP, the 40 percent condenser dump valves would automatically close to protect the condenser. The SG pressure would rapidly increase resulting in steam discharge to the atmosphere through the SG PORVs and safety valves if their setpoint is reached.

(5) Following reactor trip and SI actuation, the continued action of AFW supply and borated SI flow (supplied from the RWST) provides a heat sink that absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of LOOP, steam relief to the atmosphere.

(6) SI flow results in stabilization of the RCS pressure and pressurizer water level, and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the EOPs (Reference 42). The major operator actions include identification and isolation of the ruptured SG, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below:

(1) Identify the ruptured SG.

High secondary side activity, as indicated by the main steam line radiation monitors, the air ejector radiation monitor, or SG blowdown radiation monitor typically will provide the first indication of an SGTR event. The ruptured SG can be identified by an unexpected increase in SG level, or a high radiation indication on the corresponding main steam line monitor, or from a radiation survey of the main steam lines. For an SGTR that results in a reactor trip at high power, the SG water level may decrease off-scale on the narrow range for all of the SGs. The AFW flow will begin to refill the SGs, distributing approximately equal flow to each of the SGs. Since primary to secondary leakage adds additional liquid inventory to the ruptured SG, the water level will return to the narrow range earlier in that SG and will continue to increase more rapidly. This response, as indicated by the SG water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured SG.

(2) Isolate the ruptured SG from the intact SGs and isolate feedwater to the ruptured SG.

Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured SG. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured SG with water by (a) minimizing the accumulation of feedwater flow and (b) enabling the operator to establish a pressure differential between the ruptured and intact SGs as a necessary step toward terminating primary to secondary leakage.

(3) Cool down the RCS using the intact SGs.

After isolation of the ruptured SG, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured SG pressure by dumping steam from only the intact SGs. This ensures adequate subcooling in the RCS after depressurization to the ruptured SG pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the PORVs on the intact SGs.

(4) Depressurize the RCS to restore reactor coolant inventory.

When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until the RCS and ruptured SG pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured SG.

The RCS depressurization is performed using normal pressurizer spray if the RCPs are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

(5) Terminate SI to stop primary to secondary leakage.

The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until the RCS and ruptured SG pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured SG.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured SG.

15.4.3.3 Analysis of Effects and Consequences

15.4.3.3.1 Steam Generator Tube Rupture Margin to Overfill Analysis

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the radiological consequences resulting from an SGTR are within allowable guidelines. Another concern for SGTR consequences is the possibility of SG overfill because this could potentially result in a significant increase in the radiological consequences. Overfill could result in water entering the main steam line. If water continues to leak into the main steam lines, the release of liquid through the SG safety valves could result in an increase in radiological doses. Therefore, an analysis was performed to demonstrate margin to SG overfill, assuming the limiting single failure relative to overfill. The results of this analysis demonstrate that there is margin to SG overfill for DCPP.

The overfill analysis is presented in Reference 72 and the major assumptions include:

(1) Complete severance of a single tube located at the top of the tube sheet on the outlet side of the SG, resulting in double ended flow

(2) Initiation of the event from full power

(3) A LOOP coincident with reactor trip

(4) Failure of an AFW control valve to close (limiting single failure)

(5) The PORVs on all three intact SGs are fully opened during the RCS cooldown

(6) Operator actions are consistent with the times shown in Table 15.4-12

The SGTR MTO analysis acceptance criterion is to maintain a positive MTO when the event is terminated. The limiting MTO analysis presented in Reference 72 demonstrates that the SG liquid volume is 30 cubic feet less than the total SG volume of 5800 cubic feet when the SGTR event is terminated. The SGTR MTO analysis sequence of events is listed in Table 15.4-13A and the transient responses are presented in Figures 15.4.3-1A through 15.4.3-4A and Figures 15.4.3-6A through 15.4.3-8A.

An analysis was also performed to determine the transient thermal hydraulic data for input into the radiological consequences analysis, assuming the limiting single failure relative to doses without SG overfill (as opposed to one that is relative to overfill). Because SG overfill does not occur, the radiation consequences (refer to Section 15.5.20) calculated using the results of this analysis represent the limiting consequences for an SGTR for DCPP. The thermal hydraulic results used by the radiological consequences (Dose) analysis are discussed below.

15.4.3.3.2 Steam Generator Tube Rupture Dose Input Analysis

A thermal and hydraulic analysis was performed to determine the plant response for a design basis SGTR, and to determine the integrated primary to secondary break flow and the mass releases from the ruptured and intact SGs to the condenser and to the atmosphere. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences. The thermal and hydraulic analysis discussed in this section is presented in Reference 41 and the results of the radiological consequences analysis are discussed in Section 15.5.20.

The plant response following an SGTR was analyzed with the RETRAN-02W program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

The accident modeled is a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold leg) side of the SG. However, as indicated subsequently, the break flow flashing fraction was conservatively calculated assuming that all of the break flow comes from the hot leg side of the SG. The combination of these conservative assumptions regarding the break flow location results in a very conservative calculation of the radiation doses. It was assumed that the reactor is operating at full power at the time of the accident and the secondary mass was assumed to correspond to operation at the SG nominal level with an allowance for uncertainties. It was also assumed that a LOOP occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The limiting single failure was assumed to be the failure of the PORV on the ruptured SG. Failure of this PORV in the open position will cause an uncontrolled depressurization of the ruptured SG which will increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured SG PORV fails open when the ruptured SG is isolated, and that the PORV was isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.4.3.2 and these operator actions were simulated in the analysis. The operator action times which were used for the analysis are presented in Table 15.4-12. It is noted that the PORV on the ruptured SG was assumed to fail open at the time the ruptured SG was isolated. It was assumed that the operators isolate the failed open PORV by locally closing the associated block valve to complete the isolation of the ruptured SG before proceeding with the subsequent recovery operations. It was assumed that the ruptured SG PORV was isolated at 30 minutes after the valve was assumed to fail open. After the ruptured SG PORV was isolated, an additional delay time of 5 minutes (refer to Table 15.4-12) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The RETRAN-02W (Reference 70) analysis results are described below. The sequence of events for this transient is presented in Table 15.4-13B.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured SG since the primary pressure is greater than the SG pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 15.4.3-1B. The pressurizer pressure also decreases as shown in Figure 15.4.3-2B as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an overtemperature ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the 40 percent condenser dump valves remain closed due to the loss of condenser vacuum resulting from the assumed LOOP at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the SG PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.4.3-3B. The main feedwater flow will be terminated and AFW flow will be automatically initiated following reactor trip and the LOOP.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the SI flow rate maintains the reactor coolant inventory and the pressurizer level begins to stabilize. The RCS pressure also trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Because offsite power was assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (refer to Figures 15.4.3-4B and 15.4.3-5B), however, the temperature differential subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the SG temperature as the fluid residence time in the tube region increases. The intact SG loop temperatures slowly decrease due to the continued AFW flow until operator actions are taken to control the AFW flow to maintain the specified level in the intact SGs. The ruptured SG loop temperatures also continue to slowly decrease until the ruptured SG is isolated, at which time the PORV is assumed to fail open.

Major Operator Actions

(1) Identify and Isolate the Ruptured SG

As indicated in Table 15.4-12, it was assumed that the ruptured SG is identified and isolated at 10 minutes after the initiation of the SGTR or when the narrow range level reaches 38 percent, whichever time is longer. Since the time to reach 38 percent narrow range level was 953 seconds, it was assumed that the actions to isolate the ruptured SG are performed at this time.

The ruptured SG PORV was also assumed to fail open at this time, and the failure was simulated at 953 seconds. The failure causes the ruptured SG to rapidly depressurize, which results in an increase in primary to secondary leakage. The depressurization of the ruptured SG increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure 15.4.3-5B. As noted previously, the intact SG loop temperatures also decrease, as shown in Figure 15.4.3-4B, until the AFW flow to the intact SGs is throttled. These effects result in a decrease in the RCS pressure and pressurizer level, until the failed open PORV is isolated.

It was assumed that the time required for the operator to identify that the ruptured SG PORV is open and to locally close the associated block valve is 30 minutes. Thus, the isolation of the ruptured SG was completed at 2753 seconds, and the depressurization of the ruptured SG was terminated. At this time, the ruptured SG pressure increases rapidly and the primary to secondary break flow begins to decrease.

(2) Cool Down the RCS to establish Subcooling Margin

After the ruptured SG PORV block valve was closed, a 5 minute operator action time was imposed prior to initiation of cooldown. The depressurization of the ruptured SG affects the RCS cooldown target temperature because the temperature is dependent upon the pressure in the ruptured SG. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the intact SG PORVs. The cooldown was continued until RCS was subcooled 36°F including an allowance for instrument uncertainty. Because the pressure in the ruptured SG continued to decrease during the cooldown, the associated temperature the RCS was less than the initial target temperature, which had the net effect of extending the time for cooldown. The cooldown was initiated at 3053 seconds and was completed at 4424 seconds.

The reduction in the intact SG pressures required to accomplish the cooldown is shown in Figure 15.4.3-3B, and the effect of the cooldown on the RCS temperature is shown in Figure 15.4.3-4B. The pressurizer level and pressurizer pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in Figures 15.4.3-1B and 15.4.3-2B, respectively.

(3) Depressurize to Restore Inventory

After the RCS cooldown, a 4-minute operator action time was included prior to depressurization. The RCS depressurization was initiated at 4664 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS was depressurized by opening a pressurizer PORV. The depressurization was continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured SG pressure and pressurizer level is greater than the allowance of 12 percent for pressurizer level uncertainty, or pressurizer level is greater than 74 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. The RCS depressurization reduces the break flow as shown in Figure 15.4.3-6B, and increases SI flow to refill the pressurizer as shown in Figure 15.4.3-1B.

(4) Terminate SI to Stop Primary to Secondary Leakage

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated after a delay to allow for operator response if RCS subcooling is greater than the 20°F allowance for uncertainty, minimum AFW flow is available or at least one intact SG level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 12 percent allowance for uncertainty.

After depressurization was completed, an operator action time of 2 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination was performed at this time. After SI termination, the pressurizer pressure decreases as shown in Figure 15.4.3-2B. Figure 15.4.3-6B shows that the primary to secondary leakage continues after the SI flow was stopped until the RCS and ruptured SG pressures equalize.

The ruptured SG water volume for the radiological consequences analysis is shown in Figure 15.4.3-7B. The mass of water in the ruptured SG is also shown as a function of time in Figure 15.4.3-8B.

Mass Releases

The mass releases were determined for use in evaluating the exclusion area boundary (EAB) and low population zone (LPZ) radiation exposure. The steam releases from the ruptured and intact SGs, the feedwater flows to the ruptured and intact SGs, and primary to secondary break flow into the ruptured SG were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours were used to calculate the radiation doses at the EAB for a 2-hour exposure, and the releases for 0-8 hours were used to calculate the radiation doses at the LPZ for the duration of the accident.

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage were simulated in the RETRAN-02W analysis. Thus, the steam releases from the ruptured and intact SGs, the feedwater flows to the ruptured and intact SGs, and the primary to secondary leakage into the ruptured SG were determined from the RETRAN-02W results for the period from the initiation of the accident until the leakage was terminated.

Following the termination of leakage, it was assumed that the actions are taken to cool down the plant to cold shutdown conditions. The PORVs for the intact SGs were assumed to be used to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The steam releases and the feedwater flows for the intact SG for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact SG conditions at the time of leakage termination and at 2 hours. The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 350°F is reached. Depressurization of the ruptured SG was then assumed to be performed to the RHR in-service pressure of 405 psia via steam release from the ruptured SG PORV. The RCS pressure was also assumed to be reduced concurrently as the ruptured SG is depressurized. It was assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact SGs from a mass and energy balance using conditions at 2 hours and at the RHR system in-service conditions. The steam released from the ruptured SG from 2 to 8 hours was determined based on a mass and energy balance for the ruptured SG using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

After 8 hours, it was assumed that further plant cooldown to cold shut down as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

During the time period from initiation of the accident until leakage termination, the releases were determined from the RETRAN-02W results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip would be through the condenser air ejector and/or the condenser vacuum pump exhaust (if in operation). After reactor trip, the releases to the atmosphere were assumed to be via the SG PORVs. The mass release rates to the atmosphere from the RETRAN-02W analysis are presented in Figures 15.4.3-9 and 15.4.3-10 for the ruptured and intact SGs, respectively, for the time period until leakage termination. The total flashed break flow from the RETRAN-02W analysis is presented in Figure 15.4.3-11. The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours were also assumed to be released to the atmosphere via the SG PORVs. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals are presented in Table 15.4-14.

15.4.3.4 Conclusions

The analysis demonstrates the acceptance criteria are met as follows:

15.4.3.4.1 Overfill Analysis

The SGTR MTO analysis acceptance criteria are to maintain a positive MTO when the event is terminated. Therefore, the limiting MTO analysis demonstrates that the SG liquid volume is less than the total SG volume of 5800 cubic feet when the SGTR event is terminated.

15.4.3.4.2 Radiological

Section 15.5.20 demonstrates that the acceptance criteria for Dose Consequences of a SGTR are met. Table 15.5-64 provides the offsite and control room radiation doses from the release of airborne activity following a SGTR accident.

15.4.4 SINGLE REACTOR COOLANT PUMP LOCKED ROTOR

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor.

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 SGs 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 SGs 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 PORVs, and opens the PSVs in that sequence. The three PORVs 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 is not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

Three digital computer codes are used to analyze this transient. The LOFTRAN (Reference 26) code is used to calculate the resulting loop and core coolant flow following the pump seizure. The LOFTRAN code is also used to calculate the time of reactor trip based on the calculated flow, 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 using the FACTRAN (Reference 17) code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient. The THINC (Reference 31) code (refer to Section 4.4.3) is used to calculate the DNBR during the transient based on flow calculated by LOFTRAN and heat flux calculated by FACTRAN.

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the RCPs is assumed to seize, for the DNB evaluation the plant is assumed to be under steady state operating conditions consistent with use of the ITDP (Reference 62).

When the peak pressure is evaluated, the initial power is assumed as 2 percent above nominal full power, the initial coolant average temperature is assumed 5°F above nominal, and the initial pressure is conservatively assumed as 60 psi above nominal pressure (2250 psia) to allow for uncertainties. This is done to obtain the highest possible rise in the coolant pressure during the transient. The pressure response for the point in the RCS having the maximum pressure is shown in Figure 15.4.4-1.

The analysis accounts for the potential effect of asymmetric steam generator tube plugging, which results in a loop-to-loop flow asymmetry. The loop with the locked rotor is assumed to have the highest initial flow rate, which conservatively minimizes the core flow during the transient.

15.4.4.2.1 Evaluation of the Pressure Transient

After pump seizure and reactor trip, the neutron flux is rapidly reduced by the effect of control rod insertion. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 85 percent of nominal 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.

The pressurizer safety valve model includes a +3 percent opening tolerance plus 5 psi accumulation above the nominal setpoint of 2500 psia. A purge delay of 1.272 seconds was also included to account for the presence of water-filled loop seals (Reference 75). The analysis conservatively assumes an additional +1% shift in the opening setpoint. Note that all PSVs have been concerted to a steam seat design and condensate in the loop is now continuously drained back to the pressurizer, thereby eliminating the water loop seal. Even though the water loop seal has been eliminated, the resulting benefit is not credited in the analysis.

15.4.4.2.2 Evaluation of the Effects of Departure from Nucleate Boiling in the Core During the Accident

For this accident, DNB is assumed to occur in the core and, therefore, 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 cladding temperature and ZR-H_2O reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be greater than or equal to 2.7 (i.e., $F_Q \geq 2.7$) at the initial core power level.

15.4.4.2.3 Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code 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 on the actual heat transfer conditions at the time. The neutron flux and mass flowrate, as a function of time, are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to cladding temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

15.4.4.2.4 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with the initial fuel temperature to 10,000 Btu/hr-ft^2-°F within 0.5 seconds after the initiation of the transient. This assumption causes energy stored in the fuel to be released to the cladding at the initiation of the transient and maximizes the cladding temperature during the transient.

15.4.4.2.5 Zirconium-steam Reaction

The zirconium-steam reaction can become significant above 1800°F (cladding temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left[\frac{-45{,}500}{1.986T}\right] \quad (15.4\text{-}1)$$

where:

w = amount reacted, mg/cm^2
t = time, sec
T = temperature, °K
and the reaction heat is 1510 cal/gm.

15.4.4.3 Results

Transient plots of maximum RCS pressure, flow coastdown, hot channel heat flux, and neutron flux are shown in Figures 15.4.4-1 and 15.4.4-3 through 15.4.4-5. Maximum RCS pressure, maximum cladding temperature, and amount of Zr-H_2O reaction are contained in Table 15.4-10. Figure 15.4.4-2 shows the cladding temperature transient for the worst case.

15.4.4.4 Conclusions

(1) Because 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 maintained. For the MSS, the maximum pressure is bounded by the analysis of the loss of external electrical load/turbine trip event (refer to Section 15.2.7).

(2) Because the peak cladding average temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F and

the amount of $Zr\text{-}H_2O$ reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

(3) The results of the transient analysis show that less than 10 percent of the fuel rods will have DNBRs below the safety analysis limit values.

15.4.5 FUEL HANDLING ACCIDENT

15.4.5.1 Acceptance Criteria

The following limiting criterion is applicable for a FHA:

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.5.2 Identification of Causes and Accident Description

15.4.5.2.1 Fuel Handling Procedures

One major task that must be performed routinely as part of the operation of a nuclear power plant is the handling of the reactor fuel. The bulk of this fuel handling occurs during refueling outages, which occur every one to two years, and all of these operations are carried out with the fuel under water. A typical refueling outage would include the following major operations:

(1) Shutdown of the reactor and cooldown to ambient conditions

(2) Removal and storage of pressure vessel head

(3) Filling of refueling cavity above the pressure vessel with water to provide shielding from radioactive fuel

(4) Transfer of the reactor fuel assemblies from the reactor itself to underwater storage racks in the spent fuel pool

(5) Performance of outage tasks appropriate to the “core off-load” window

(6) Return of the appropriate number of partially burned and new fuel assemblies to the reactor

Fuel handling operations within the containment building and the fuel handling area are accomplished with overhead cranes, specially designed fuel grapples, and miscellaneous other equipment. To facilitate the transfer of the fuel between the two buildings, an underwater penetration called the transfer tube is provided through the

walls where the buildings adjoin. A conveyer cart is used to transport the fuel from one building to the other through this penetration. A more detailed description of the equipment used in fuel handling operations can be found in Chapter 9.
Spent fuel remains in storage in the spent fuel pool until placed in a cask for transport to the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) or for shipment from the site.

15.4.5.2.2 Probability of Activity Release

In the above operations, there exists the remote possibility that one or more fuel assemblies will sustain some mechanical damage. There exists an even more remote possibility that this damage will be severe enough to breach the cladding and release some of the radioactive fission products contained therein.

Both the fuel handling procedure and the fuel handling equipment design adhere to the following safety criteria:

(1) Fuel handling operations must not commence before short-lived core activity has decayed, leaving only relatively long-lived activity. Equipment control guidelines (ECGs) for refueling operations specify the minimum waiting time.

(2) Fuel handling operations must preclude any critical configuration of the core, spent fuel, or new fuel.

(3) The fuel handling system design must ensure an adequate water depth for radiation shielding of operating personnel.

(4) Active components of the fuel handling systems must be designed such that loss-of-function failures will terminate in stable modes.

(5) The design of fuel handling equipment must minimize the possibility of accidental impact of a moving fuel assembly with any structure.

(6) The design of fuel handling equipment and procedures must minimize the possibility of any massive object damaging a stationary fuel assembly.

(7) Fuel assembly design must minimize the possibility of damage in the event that portable or hand tools come into contact with a fuel assembly.

(8) The design of structures around the fuel handling system must minimize the possibility of the structures themselves failing in the event of a Design Earthquake (DE), Double Design Earthquake (DDE), or Hosgri Earthquake (HE). Furthermore, the structures must minimize the possibility of any external missile from reaching fuel assemblies.

(9) Fuel handling equipment must be capable of supporting maximum loads under seismic conditions. Furthermore, fuel handling equipment must not generate missiles during seismic conditions. The earthquake loading of the fuel handling equipment is evaluated in accordance with the seismic considerations addressed in Sections 9.1.4.3.1 and 9.1.4.3.9.

Implementation of the above safety criteria into the fuel handling system design is discussed in greater detail in Chapter 9.

Because of the above design, the probability of breaching the fuel cladding and releasing radioactive fission products is very small.

15.4.5.2.3 Accident Description

In order to assess the probable extent of fuel cladding damage from a FHA, it is necessary to look more closely at specific FHAs that might realistically occur.

Multiple assemblies are loaded into the multi-purpose canister/transfer cask assembly for movement to the ISFSI, as described in Section 9.1.4.2.6. The multi-purpose canister is subsequently drained, evacuated, backfilled with helium, and sealed. However, extensive design and analysis along with application of the ISFSI Technical Specifications ensure temperatures remain within the design basis and no fuel cladding damage occurs.

The possibility of damaging fuel cladding by overheating during fuel handling operations was considered. Because irradiated fuel is always handled under water, overheating would require draining either the refueling cavity or the spent fuel pool while irradiated fuel was located within them. Consideration has been given in design of the cavity and pool to prevent either of these possibilities. The probability of losing coolant while an assembly is in the transfer tube is also extremely small in view of the fact that the tube is open on one end to the reactor cavity and on the other end to the pool. There is no realistic occurrence that would simultaneously block off both ends of the tube. Therefore, it is expected that there will be no radiological consequences over the lifetime of the plant that results from overheating during fuel handling operations.

The possibility of dropping a foreign object of sufficient size to produce cladding rupture onto irradiated fuel located either in the reactor or the pool is extremely remote because the design of the plant is such that only rarely are objects of this size transported over locations containing irradiated fuel. The three large objects that are routinely handled in the vicinity of irradiated fuel are the reactor head, upper internals package, which must be removed and reinstalled from the pressure vessel at each refueling outage, and the spent fuel shipment cask, which must be placed in the pool for loading. As discussed in Section 9.1.4.2.5, load drop analyses were performed for the reactor head and upper internals and are summarized in the PG&E NUREG-0612 submittal. It is not necessary to lift the cask over the fuel racks in moving it to or from the pool. Protection of nuclear fuel assemblies from overhead load handling is a key element of the control of heavy loads program described in Section 9.1.4.3.10.

The possibility has also been considered of one of the bridge cranes falling into the reactor or the pool as a result of an earthquake. However, both of these cranes are seismically qualified for the DE, DDE, and HE. Therefore, it is expected that there will be no radiological consequences over the lifetime of the plant that result from dropping objects onto radiated fuel.

If a fuel assembly were to strike an object, it is possible that the object might damage the fuel rods with which it comes into contact. If a fuel assembly were to strike against a flat, plane-like object or a linear, edge-like object, impact loads would be distributed across several fuel rods, and no cladding damage would be expected. If a fuel assembly were to strike against a sharp, corner-like object, impact loads would be concentrated, and cladding damage might occur. Thus, there is a very remote possibility that impact loads would be severe enough to rupture fuel cladding.

Analyses have been made by Westinghouse of the effects that would result from dropping a fuel assembly from an initial vertical orientation onto a flat surface, the core, or a loaded fuel rack. Westinghouse has also analyzed the case where an assembly in the holder on the conveyor car falls from the vertical to the horizontal position. The results of these analyses indicate there is only a very remote possibility of fuel cladding rupture.

The above discussion indicates that the unlikely event of a fuel cladding integrity failure would most likely result from a fuel assembly striking a sharp object or dropping a fuel assembly.

15.4.5.3 Results

15.4.5.3.1 Containment Building Accident

During fuel handling operations, the containment ventilation penetrations to the outside atmosphere are maintained in a closed or automatically isolable condition. Isolation is automatically actuated if either of the containment purge exhaust monitors, RM-44A or RM-44B, alarms due to a concentration of radioactivity in the containment purge exhaust duct that exceeds the alarm setpoint. However, these penetrations are also allowed to be open under administrative controls, which provide the capability of closure within approximately 30 minutes.

Other containment penetrations, such as the personnel airlock and equipment hatch are allowed to be open during fuel handling operations. These penetrations are capable of manual closure and will be closed in accordance with plant procedures should a FHA occur.

In addition to the functions of the above mentioned monitors, fixed area radiation monitors are located in the containment. Should a fuel assembly be dropped and release activity above a prescribed level, the area monitors would sound an audible alarm. Personnel would exit the containment and containment closure would be initiated immediately per administrative procedures.

Because of containment isolation and closure capabilities, activity released from damaged fuel rods will be managed such that both the onsite and offsite exposures are minimized. The containment iodine removal system (refer to Section 9.4.5) can be used to remove any radioactive iodine from the containment atmosphere, but is not credited for iodine removal in the radiological analysis (refer to Section 15.5.22), and controlled containment venting can be initiated with offshore winds. Thus, there is a reasonable probability that only limited onshore exposures will result from a containment FHA.

15.4.5.3.2 Fuel Handling Area Accident

A fuel assembly could be damaged in the transfer canal or the spent fuel pit in the fuel handling area. Supply air for the spent fuel pit area is swept across the fuel pit and transfer canal and exhausted through the vent. An area radiation monitor is located on the bridge over the spent fuel pit. Doors in the fuel handling area are closed to maintain controlled leakage characteristics in the spent fuel pit region during refueling operations involving irradiated fuel. Should a fuel assembly be damaged in the canal or in the pit and release radioactivity above a prescribed level, the radiation monitors sound an alarm and the spent fuel pit ventilation exhaust through charcoal filters will remove most of the halogens prior to discharging it to the atmosphere. If the discharge is greater than the prescribed levels, an alarm sounds and the supply and exhaust ventilation systems servicing the spent fuel pit area can be manually shut down from the control room, limiting the leakage to the atmosphere.

The analysis of the radiological effects of this accident is contained in Section 15.5.22.1.

15.4.5.4 Conclusions

The analysis demonstrates the acceptance criteria are met as follows:

(1) Section 15.5.22 concludes that all potential exposures from a fuel handling accident will be well below the guideline levels specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000, and that the occurrence of such accidents would not result in undue risk to the public. Table 15.5-47 provides a summary of doses from a fuel handling accident in the fuel handling area from a fuel handling accident inside containment.

15.4.6 RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING (ROD CLUSTER CONTROL ASSEMBLY EJECTION)

15.4.6.1 Acceptance Criteria

Conservative 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:

15.4.6.1.1 Fuel Damage Criteria

(1) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel

(2) Average cladding temperature at the hot spot below the temperature at which cladding embrittlement may be expected (2700°F)

(3) Fuel melting will be 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 (1) above

15.4.6.1.2 Maximum Reactor Coolant System Pressure Criteria

(1) Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits

15.4.6.1.3 Radiological Criteria

(1) The resulting potential exposures to individual members of the public and to the general population shall be lower than the applicable guidelines and limits specified in 10 CFR 50.67 and Section 4.4, Table 6 of Regulatory Guide 1.183, July 2000.

15.4.6.2 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.

15.4.6.2.1 Design Precautions and Protection

Certain features of the DCPP 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 that lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

15.4.6.2.2 Mechanical Design

The mechanical design is discussed in Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

(1) Each full length CRDM housing is completely assembled and shop tested at 3107 psig.

(2) Pressure housings were individually hydrotested. The lower latch housing to nozzle connection is hydrotested during hydrotest of the completed reactor vessel closure head (RVCH).

(3) 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 DE, DDE, or HE can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class I components.

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

(5) The CRDM housing plug is an integral part of the rod travel housing.

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 rod travel housing are threaded joints reinforced by canopy-type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

15.4.6.2.3 Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation of a plant 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 by boron changes. Further, the location and grouping of control rod 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. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at low-level alarm and emergency boration at the low-low alarm.

15.4.6.2.4 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 18. The protection for this accident is provided by the power range 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.

15.4.6.2.5 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 is not expected to cause damage to adjacent housings leading to increased severity of the initial accident.

15.4.6.2.6 Limiting Criteria

Due to the extremely low probability of an RCCA ejection accident, limited fuel damage is 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 (Reference 19). Extensive tests of zirconium-clad UO_2 fuel rods representative of those in PWR-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 (Reference 20) 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 cladding 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.

15.4.6.3 Analysis of Effects and Consequences

The analysis of the RCCA ejection accident is performed in two stages: (a) an average core nuclear power transient calculation and (b) a hot spot heat transfer 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 in 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 pessimistically assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference 21.

15.4.6.3.1 Average Core Analysis

The spatial kinetics computer code, TWINKLE (refer to Section 1.6.1, Item 50 and Section 15.1.2.5) is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equations in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multi-region, transient fuel-clad-coolant heat transfer model for calculating 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 and the elimination of axial feedback weighting factors. 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. A further description of TWINKLE appears in Section 15.1.2.5.

15.4.6.3.2 Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and cladding transient heat transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross-section of a metalclad 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 Zr-H_2O reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter (Reference 28) or Jens-Lottes (Reference 29) correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 23) to determine the film boiling coefficient after DNB. The

DNB heat flux 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 that predicted by design fuel heat transfer codes.

For full power cases, the design initial hot channel factor (F_Q) is input to the code. The hot channel factor during the transient is assumed to increase from the steady state design value to the maximum transient value in 0.1 seconds, and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel. Further description of FACTRAN appears in Section 15.1.2.1.

15.4.6.3.3 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 system, heat transfer to the SGs, and the action of the pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing (Reference 21).

15.4.6.3.4 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of calculated values for this type of core. The more important parameters are discussed below. Table 15.4-11 presents the parameters used in this analysis. A summary of the values used in the reload analysis process is also provided in Table 15.4-11.

15.4.6.3.5 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three-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 calculations.

The total transient hot channel factor F_Q is then obtained by combining the axial and radial factors.

15.4.6.3.6 Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of regions is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis. Physics calculations were 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 that, 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 used. 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 were shown to be conservative compared to three-dimensional analysis.

15.4.6.3.7 Moderator and Doppler Coefficient

The critical boron concentrations at the BOL and EOL 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 the one-dimensional steady state computer code with a Doppler weighting factor of 1. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1 (approximately 1.3), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above.

15.4.6.3.8 Delayed Neutron Fraction

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of 0.70 percent at BOL and 0.50 percent at EOL for the first cycle. The accident is sensitive to β_{eff} if the ejected rod worth is nearly equal to or greater than β_{eff} as in zero power transients. In order to allow for future fuel cycles, pessimistic estimates of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

15.4.6.3.9 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-11 and includes the effect of one stuck rod. 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 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open, and 0.15 seconds for the coil to release the rods. The analyses presented are applicable for a rod insertion time of 2.7 seconds from coil release to entrance to the dashpot, although measurements indicate that this value should be closer to 1.8 seconds.

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 rod insertion versus time is described in Section 15.1.4.

15.4.6.4 Results

Typical reload values of the parameters used in the VANTAGE 5 analysis, as well as the results of the analysis, are presented in Table 15.4-11 and discussed below. Actual values vary slightly from reload to reload.

15.4.6.4.1 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 assumed to be 0.20 percent Δk and 6.70, respectively. The peak hot spot cladding average temperature was 2434°F. The peak hot spot fuel center temperature exceeded the BOL melting temperature of 4900°F; however, melting was restricted to less than 10 percent of the pellet.

15.4.6.4.2 Beginning of Cycle, Zero Power

For this condition, control Bank D was assumed to be fully inserted and C was at its insertion limit. The worst ejected rod is located in control Bank D and was conservatively assumed to have a worth of 0.785 percent Δk and a hot channel factor of 13. The peak hot spot cladding average temperature reached only 2660°F.

15.4.6.4.3 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 assumed to be 0.21 percent Δk and 6.50, respectively. This resulted in an average PCT of 2218°F. The peak hot spot fuel center temperature exceeded the EOL melting temperature of 4800°F. However, melting was restricted to less than 10 percent of the pellet.

15.4.6.4.4 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 and Bank C at its insertion limit. The results were 0.85 percent Δk and 21.5, respectively. The peak cladding average and fuel center temperatures were 2632°F and 3849°F, respectively.

A summary of the cases presented above is given in Table 15.4-11. The nuclear power and hot spot fuel cladding temperature transients for these representative BOL full power and EOL zero power cases are presented in Figures 15.4.6-1 through 15.4.6-4.

15.4.6.4.5 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three-dimensional THINC analysis. 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.

15.4.6.4.6 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 force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. 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 away from that region. However, the heat from 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 analyses.

15.4.6.5 Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and cladding 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 show that less than 10 percent of the fuel rods enter DNB. Even in the portion of the core which does reach DNB, there will be no excessive release of fission product activity if the limiting hot channel factors are not exceeded (Reference 21).

The analysis shows the acceptance criteria for a RCCA Ejection Accident has been met as follows:

15.4.6.5.1 Fuel Damage

(1) Table 15.4-11 shows the average fuel pellet enthalpy at the hot spot (maximum fuel stored energy) below 225 cal/gm for non-irradiated fuel and 200 cal/gm (360 Btu/lb) for irradiated fuel.

(2) Table 15.4-11 shows the average clad temperature at the hot spot (maximum cladding average temperature) below 2700°F, the temperature above which clad embrittlement may be expected.

(3) Table 15.4-11 shows the fuel melting limited to less than the innermost 10 percent of the fuel pellet at the hot spot.

15.4.6.5.2 Maximum Reactor Coolant System Pressure

(1) A detailed calculation of the pressure surge for an ejection worth of one dollar reactivity insertion at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Because the severity of the present analysis does not exceed this worst case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

15.4.6.5.3 Radiological

(1) Section 15.5.23 concludes that offsite exposures from a RCCA ejection accident is below the guideline levels specified in 10 CFR 50.67 and Section 4.4, Table 6 of the Regulatory Guide 1.183, July 2000, and that the occurrence of such accidents would not result in undue risk to the public. Table 15.5-52 provides a summary of offsite doses from a rod ejection accident.

15.4.7 REFERENCES

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15.5 RADIOLOGICAL CONSEQUENCES OF PLANT ACCIDENTS

The purposes of this section are: (a) to identify accidental events that could cause radiological consequences, (b) to provide an assessment of the consequences of these accidents, and (c) to demonstrate that the potential consequences of these occurrences are within the limits, guidelines, and regulations established by the NRC.

An accident is an unexpected chain of events; that is, a process, rather than a single event. In the analyses reported in this section, the basic events involved in various possible plant accidents are identified and studied with regard to the performance of the ESFs. The full spectrum of plant conditions has been divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public. The four categories as defined above are as follows:

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency

Condition III: Infrequent Faults

Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of these conditions is that the most frequent occurrences must yield little or no 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.

These categories and principles were developed by the ANS (Reference 1). Similar, though not identical, categories have been defined in the guide to the Preparation of Environmental Reports (Reference 3). While some differences exist in the manner of sorting the different accidents into categories in these documents, the basic principles are the same.

It should also be noted that the range of plant operating parameters included in the Condition I category, and some of those in the Condition II category, fall in the range of normal operation. For this reason, the radioactive releases and radiological exposures associated with these conditions are analyzed in Chapter 11 and are not discussed separately in this chapter. The analyses of the variations in system parameters associated with Condition I occurrences or operating modes are discussed in Chapter 7 since these states are not accident conditions. In addition, some of the events identified as potential accidents in Regulatory Guide 1.70, Revision 1 (Reference 2), have no significant radiological consequences, or result in minor releases within the range of normal releases, and are thus not analyzed separately in this chapter.

15.5.1 DESIGN BASES

The following regulatory requirements and guidance are applicable to the DCPP radiological consequence analyses presented in this chapter. They form the bases of the acceptance criteria and methodologies as described in the following sections:

(1) 10 CFR Part 100, "Reactor Site Criteria"*

(2) 10 CFR 50.67, "Accident Source Term"

(3) General Design Criterion 19, 1999 "Control Room"

(4) Regulatory Guide 1.4, Revision 1, " Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"

(5) Regulatory Guide 1.183, July 2000, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

*The licensing basis for acceptability of the dose consequences of the tank rupture events; i.e., the accidental release of radioactivity accumulated in tanks during normal plant operations, is 10 CFR Part 100. However, in accordance with DCPP UFSAR Section 3.2.2.1.2, the more restrictive acceptance criteria associated with a failure in a PG&E Design Class II system has been determined as applicable to the dose consequences at the site boundary for the Rupture of a Gas Decay Tank (GDT) and the Rupture of a Liquid Holdup Tank (LHUT).

In accordance with UFSAR Section 3.2.2.1.2, fluid systems and fluid system components that contain or may contain radioactive material, but whose failure would not result in calculated potential exposures in excess of 0.5 rem whole body (or its equivalent to parts of the body) at the site boundary may be classified as PG&E Design Class II.

Consistent with the design classification of the GDT and LHUT, and in accordance with the above more restrictive acceptance criteria, the description of these accident evaluations have been relocated from Section 15.5 to Chapter 11. The Rupture of the LHUT and GDT is discussed in Sections 11.2.3.12 and 11.3.2.2 respectively. It is noted that the Rupture of the Volume Control Tank (VCT) has also been relocated to Section 11.2. Since the VCT is PG&E Design Class I, the more restrictive acceptance criteria applicable to the failure in a PG&E Design Class II system is not applicable, however, PG&E has elected to use this more limiting acceptance criteria for all tank ruptures. The VCT rupture is dicscussed in Section 11.2.3.12.

15.5.1.1 List of Analyzed Accidents

The following table summarizes the accident events that have been evaluated for radiological consequences. The table identifies the applicable section describing the analysis and results for each event, the offsite/onsite locations and applicable dose limits, and the radiological analysis and isotopic core inventory codes used.

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
CONDITION II					
Loss of Electrical Load	15.5.10	EAB and LPZ Control Room TSC	2.5 rem TEDE 5 rem TEDE 5 rem TEDE	RADTRAD 3.03	SAS2/ORIGEN-S
CONDITION III					
Small Break LOCA (SBLOCA)	15.5.11	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.23	N/A Refer to Section 15.5.23
Minor Secondary System Pipe Breaks	15.5.12	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.18	N/A Refer to Section 15.5.18
Inadvertent Loading of a Fuel Assembly	15.5.13	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.13	N/A Refer to Section 15.5.13
Complete Loss of Forced Reactor Coolant Flow	15.5.14	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.10	N/A Refer to Section 15.5.10
Under-Frequency	15.5.15	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.10	N/A Refer to Section 15.5.10
Single Rod Cluster Control Assembly Withdrawal	15.5.16	EAB and LPZ Control Room	2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.23	N/A Refer to Section 15.5.23
CONDITION IV					
Large Break LOCA (LBLOCA)	15.5.17	EAB and LPZ Control Room TSC	25 rem TEDE 5 rem TEDE 5 rem TEDE	RADTRAD 3.03 PERC2 SW-QADCGGP	SAS2/ORIGEN-S

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Main Steam Line Break (MSLB)	15.5.18	EAB and LPZ Pre-Accident Iodine Spike Accident-initiated Iodine Spike Control Room TSC	 25 rem TEDE 2.5 rem TEDE 5 rem TEDE 5 rem TEDE	RADTRAD 3.03	SAS2/ORIGEN-S
Main Feedwater Line Break (FWLB)	15.5.19	EAB and LPZ Pre-Accident Iodine Spike Accident-initiated Iodine Spike Control Room	 25 rem TEDE 2.5 rem TEDE 5 rem TEDE	N/A Refer to Section 15.5.18	N/A Refer to Section 15.5.18
Steam Generator Tube Rupture (SGTR)	15.5.20	EAB and LPZ Pre-Accident Iodine Spike Accident-initiated Iodine Spike Control Room TSC	 25 rem 2.5 rem 5 rem 5 rem TEDE	RADTRAD 3.03	SAS2/ORIGEN-S
Locked Rotor Accident (LRA)	15.5.21	EAB and LPZ Control Room TSC	 2.5 rem TEDE 5 rem TEDE	RADTRAD 3.03	SAS2/ORIGEN-S
Fuel Handling-Accident (FHA)	15.5.22	EAB and LPZ Control Room TSC	6.3 rem TEDE 5 rem TEDE 5 rem TEDE	RADTRAD 3.03 PERC2 SW-QADCGGP	SAS2/ORIGEN-S

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Control Rod Ejection Accident (CREA)	15.5.23	EAB and LPZ Control Room TSC	6.3 rem TEDE 5 rem TEDE 5 rem TEDE	RADTRAD 3.03	SAS2/ORIGEN-S

15.5.1.2 Assumptions associated with Loss of Offsite Power

The assumptions regarding the occurrence and timing of a Loss of Offsite Power (LOOP) during an accident are selected with the intent of maximizing the dose consequences. A LOOP is assumed for events that have the potential to cause grid perturbation.

i. The dose consequences of the LOCA, MSLB, SGTR, LRA, CREA, and LOL event are evaluated with the assumption of a LOOP concurrent with reactor trip.

ii. The assumption of a LOOP related to a postulated design basis accident which leads to a reactor trip does not directly correlate to an FHA. Specifically, a FHA does not directly cause a reactor trip and a subsequent LOOP due to grid instability; nor can a LOOP be the initiator of a FHA. Thus, the FHA dose consequence analyses are evaluated without the assumption of a LOOP.

In addition, in accordance with current DCPP licensing basis, the non-accident unit is assumed unaffected by the LOOP.

15.5.2 APPROACH TO ANALYSES OF RADIOLOGICAL EFFECTS OF ACCIDENTS

15.5.2.1 Introduction

The potential radiological effects of plant accidents are analyzed by the evaluation of all physical factors involved in each chain of events which might result in radiation exposures to humans. These factors include the meteorological conditions existing at the time of the accident, the radionuclide uptake rates, exposure times and distances, as well as the many factors which depend on the plant design and mode of operation. In these analyses, the factors affecting the consequences of each accident are identified and evaluated, and uncertainties in their values are discussed. Because some degree of uncertainty always exists in the prediction of these factors, it has become general practice to assume conservative values in making calculated estimates of radiation doses. For example, it is customarily assumed that the accident occurs at a time when very unfavorable weather conditions exist, and that the performance of the plant engineered safety systems is degraded by unexpected failures. The use of these unfavorable values for the various factors involved in the analysis provides assurance that each safety system has been designed adequately; that is, with sufficient capacity to cover the full range of effects to which each system could be subjected. For this reason, these conservative values for each factor have been called design basis values.

In a similar way, the specific chain of events in which all unfavorable factors are coincidentally assumed to occur has been called a DBA. The calculated doses for the DBA provide a basis for determination of the design adequacy of the plant safety systems. In the process of safety review and licensing, the radiation exposure levels calculated for the DBA are compared to the regulatory limits established in 10 CFR 50.67 (for accidents analyzed using AST methodology) including acceptance criteria proposed in regulatory guidance, and if these calculated exposures fall below the regulatory guideline limits, the plant safety systems are judged to be adequate.

As noted in Section III.2.a of Standard Review Plan Section 15.0.1, Revision 0, (Reference 59), a full implementation of AST addresses a) all the characteristics of AST (i.e., the radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these nuclides), b) replaces the previous accident source term used in all design basis radiological analyses, and c) incorporates the Total Effective Dose Equivalent (TEDE) criteria of 10 CFR 50.67, and Section II of Standard Review :Plan 15.0.1, Revision 0.

The dose consequences of the following accidents have been re-evaluated using AST in accordance with Regulatory Guide 1.183, July 2000.

1. Loss of Coolant Accident (LOCA) – Section 15.5.17
2. Fuel Handling Accident (FHA) – Section 15.5.22
3. Locked Rotor Accident (LRA) – Section 15.5.21
4. Control Rod Ejection Accident (CREA) – Section 15.5.23
5. Main Steam Line Break (MSLB) – Section 15.5.18
6. Steam Generator Tube Rupture (SGTR) – Section 15.5.20
7. Loss-of Load (LOL) Event – Section 15.5.10

The dose consequences for the remaining accidents are addressed by qualitative comparison to the seven accidents listed above (with the exception of the tank rupture events).

Note reference to Regulatory Guide 1.183, July 2000 is used extensively within this section, as a result any reference to “Regulatory Guide 1.183” within Section 15.5 refers to Regulatory Guide 1.183, July 2000.

The methodology used to assess the dose consequences of the DBAs, including the specific values of all important parameters, data, and assumptions used in the radiological exposure calculations are listed in the following sections. The computer programs used to assess the dose consequences of the DBAs are described briefly in Section 15.5.8.

15.5.2.2 Dose Acceptance Criteria

<u>EAB and LPZ Dose</u>

The acceptance criteria for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) Dose are based on 10 CFR 50.67, and Section 4.4, Table 6 of Regulatory Guide 1.183:

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, shall not receive a radiation dose in excess of the accident-specific TEDE value noted in Reference 55, Table 6.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a radiation dose in excess of the accident-specific TEDE value noted in Reference 55, Table 6.

EAB and LPZ Dose Acceptance Criteria - Condition II and Condition III events:

Regulatory Guide 1.183 does not specifically address Condition II and Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

<u>Control Room Dose</u>

The acceptance criterion for the control room dose is based on 10 CFR 50.67.

Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

This criteria ensures that the dose criteria of GDC 19, 1999 and NUREG-0737, November 1980, Item III.D.3.4 (refer to Section 6.4.1) is met.

<u>Technical Support Center Dose</u>

The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, Section 1.2.1, which allows dose acceptance in accordance with 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.2.3 Dose Calculation Methodology

15.5.2.3.1 Inhalation and Submersion Doses from Airborne Radioactivity

Computer Code RADTRAD 3.03 is used to calculate the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from submersion due to airborne radioactivity at offsite locations and in the control room. The summation of CEDE and EDE is reported as TEDE, in accordance with Section 4.1.4 of Regulatory Guide 1.183.

The CEDE is calculated using the inhalation dose conversion factors provided in Table 2.1 of Federal Guidance Report 11 (Reference 41).

The submersion EDE is calculated using the air submersion dose coefficients provided in Table III.1 of Federal Guidance Report 12 (Reference 42). The dose coefficients are derived based on a semi-infinite cloud model. The submersion EDE is reported as the whole body dose in the RADTRAD 3.03 output.

RADTRAD 3.03 includes models for a variety of processes that can attenuate and/or transport radionuclides. It can model the effect of sprays and natural deposition that reduce the quantity of radionuclides suspended in the containment or other compartments. In addition, it can model the flow of radionuclides between compartments within a building, from buildings into the environment, and from the environment into a control room. These flows can be through filters, piping, or simply due to air leakage. RADTRAD 3.03 can also model radioactive decay and in-growth of daughters. Ultimately the program calculates the whole body dose, the thyroid dose, and the TEDE dose (rem) to the public located offsite, and to onsite personnel located in the control room due to inhalation and submersion in airborne radioactivity based on user specified, fuel inventory, nuclear data, dispersion coefficients, and dose conversion factors. Note that the code uses a numerical solution approach to solve coupled ordinary differential equations. The basic equation for radionuclide transport and removal is the same for all compartments. The program breaks its processing into 2 parts a) radioactive transport and b) radioactive decay and daughter in-growth.

Computer Code PERC2 is used to calculate the CEDE from inhalation and the EDE from submersion due to airborne radioactivity in the TSC. PERC2 is a multiple compartment activity transport code with the dose model consistent with Regulatory Guide 1.183. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included. The CEDE is calculated using the Federal Guidance Report No.11 (Reference 41) dose conversion factors. The EDE in the TSC is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the TSC. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes the integrated activity in the TSC air space, the photon energy release rates per energy group from activity airborne in the TSC, and the ANSI/ANS 6.1.1-1991 neutron and gamma-ray fluence-to-dose factors (Reference 84).

Offsite Dose

In accordance with Regulatory Guide 1.183, for the first 8 hours, the breathing rate of the public located offsite is assumed to be $3.5x10^{-4}$ m^3/sec. From 8 to 24 hours following the accident, the breathing rate is assumed to be $1.8x10^{-4}$ m^3/sec. After that and until the end of the accident, the rate is assumed to be $2.3x10^{-4}$ m^3/sec. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is calculated and used in determining compliance with the dose criteria in 10 CFR 50.67. The LPZ TEDE is determined for the most limiting receptor at the outer boundary of the low population zone and is calculated for the entire accident duration.

Control Room Dose

The control room inhalation CEDE is calculated assuming a breathing rate of $3.5x10^{-4}$ m^3/sec for the duration of the event. The following occupancy factors are credited in determining the control room TEDE: 1.0 during the first 24 hours after the event, 0.6 between 1 and 4 days, and 0.4 from 4 days to 30 days. The submersion EDE is corrected for the difference in the finite cloud geometry in the control room and the semi-infinite cloud model used in calculating the dose coefficients. The following expression obtained from Regulatory Guide 1.183 is used in RADTRAD 3.03 to correct the semi-infinite cloud dose, EDE_∞, to a finite cloud dose, EDE_{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. As allowed in Section 4.1.4 of RG 1.183, since the submersion exposure is uniform to the whole body, the EDE is used in lieu of the deep dose equivalent in determining the contribution of the submersion dose to the TEDE.

$$Submersion - EDE_{finite} = \frac{\mathrm{EDE}_\infty \ \mathrm{V}^{0.338}}{1173} \qquad (15.5\text{-}1)$$

Technical Support Center Dose

The TSC inhalation CEDE is calculated by computer code PERC2 assuming the same breathing rate and occupancy factors as those used in determining the control room dose. The submersion EDE developed by PERC2 (which computes the photon fluence at the center of TSC and utilizes the ANSI/ANS 6.1.1-1991 fluence to effective dose conversion factors), is a close approximation of the dose determined using Table III.1 of Federal Guidance Report No. 12 (Reference 42) (refer to Section 4.1.4 Regulatory

Guide 1.183) and adjusted by the finite volume correction factor given in Regulatory Guide 1.183, Section 4.2.7.

15.5.2.3.2 Direct Shine Dose from External and Contained Sources

Computer program SW-QADCGGP is used to calculate the deep dose equivalent in the control room, TSC and at the EAB due to external and contained sources following a LOCA. The calculated deep dose equivalent is added to the inhalation (CEDE) and the submersion (EDE) dose due to airborne radioactivity to develop the final TEDE. Conservative build-up factors are used and the geometry models are prepared to ensure that un-accounted streaming/scattering paths were eliminated. The dose albedo method with conservative albedo values is used to estimate the scatter dose in situations where the scattering contributions are potentially significant. ANSI/ANS 6.1.1-1977 (Reference 83) is used to convert the gamma flux to the dose equivalent rate.

15.5.3 ACTIVITY INVENTORIES IN THE PLANT PRIOR TO ACCIDENTS

The fission product inventories in the reactor core, the fuel rod gaps, and the primary coolant prior to an accident have been conservatively calculated based on plant operation at 105% of the current licensed rated thermal power of 3411 MWth, with current licensed values of fuel enrichment and fuel burnup.

15.5.3.1 Core Activity Inventory

In accordance with Section 3.1 of Regulatory Guide 1.183, the inventory of fission products in the reactor core available for release to the containment following an accident should reflect maximum full power operation of the core with the 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 in the 10 CFR 50 Appendix K analysis (typically 1.02).

The equilibrium core inventory is calculated using computer code ORIGEN-S. The calculation is performed using the Control Module SAS2 of the SCALE 4.3 computer code package. The SAS2 control module provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. It calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions with materials in the core. It calculates the neutron-activated products, the actinides and the fission products in a reactor core.

The reactor core consists of 193 fuel assemblies with various Uranium-235 enrichments. Per control imposed by DCPP core-reload design documentation, the peak rod burnup limit at the end of cycle is not allowed to exceed 62,000 MWD/MTU. The current licensed maximum value for fuel enrichment is 5.0%. To account for variation of U-235 enrichment in fresh fuel, the radionuclide inventories were calculated for a 4.2% average enriched core (representing minimum enrichment at DCPP), and 5% average enriched core (representing maximum enrichment). The higher activity for each isotope from the above two enrichment cases is chosen to represent the inventory of that isotope in the equilibrium core.

The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles and 1/3 of the core to three fuel cycles. This approach has been demonstrated to develop an isotopic core inventory that is a reasonable and conservative approximation of a core inventory developed using DCPP specific fuel management history data. Minor variations in fuel irradiation time and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these inventory changes will have an insignificant impact on the radiological consequences of postulated accidents. A 4% margin has been included in the final isotopic radioactive inventories in support of bounding analyses and to address minor changes in future fuel management schemes.

A 19 month fuel cycle length was utilized in the analysis. The 19-month average fuel cycle is an artifact of the current DCPP fuel management scheme which specifies 3 fuel cycles every 5 years and refueling outages in Spring or Fall.

In summary, the equilibrium isotopic core average inventory is based on:

i. A power level of 3580 MWth inclusive of power uncertainty.

ii. A range of enrichment of 4.2 to 5.0 w % U-235. Use of a few assemblies with lower enrichment is a common industry practice when replacing assemblies previously irradiated but proven unsuitable for continued irradiation. As these assemblies are designed to replace higher enrichment assemblies with ones of similar reactivity for the remainder of the fuel cycle, their inventory is enveloped by the isotopic core average inventory developed to support the dose consequence analyses.

iii. A maximum core average burnup of 50 GWD/MTU.

The core inventory developed by ORIGEN-S using the above methodology includes over 800 isotopes. The DCPP equilibrium core fission product inventory of dose significant isotopes relative to LWR accidents is presented in Table 15.5-77.

15.5.3.2 Coolant Activity Inventory

1. <u>Design Basis Primary and Secondary Coolant Activity Concentrations</u>

Computer code, ACTIVITY2, is used to calculate the design basis primary coolant activity concentrations for both DCPP Unit 1 and Unit 2 based on the core inventory developed using ORIGEN-S and discussed in Section 15.5.3.1. The source terms for the primary coolant fission product activity include leakage from 1% fuel defects and the decay of parent and second parent isotopes. The depletion terms of the primary coolant fission product activity include radioactive decay, purification of the letdown flow and neutron absorption when the coolant passes the reactor core. The nuclear library includes 3rd order decay chains of approximately 200 isotopes.

Computer code, IONEXCHANGER, is used to calculate the design basis halogen and remainder activity concentrations in the secondary side liquid. The source terms for the secondary side activity include the primary-to-secondary leakage in steam generators and the decay products of parent and second parent isotopes. The depletion terms of the secondary side liquid activity include radioactive decay, and purification due to the steam generator blowdown flow, and continuous condensate polishing.

The design basis noble gas concentrations in the secondary steam are calculated by dividing the appearance rate (µCi/sec) by the steam flow rate (gm/sec). The noble gas appearance rate in the steam generator steam space includes the primary-to-secondary leak contribution and the noble gas generation due to decay of halogens in the SG liquid. The activity concentrations of the other isotopes in the steam are determined by the SG liquid concentrations and the partition coefficients recommended in NUREG 0017, Revision 1 (Reference 56).

2. <u>Technical Specification Primary and Secondary Coolant Activity Concentrations</u>

In accordance with Technical Specifications the primary coolant Technical Specification activities for iodines and noble gases are based on 1.0 µCi/gm Dose Equivalent (DE) I-131 and 270 µCi/gm DE Xe-133, respectively.

The Technical Specification based primary coolant isotopic activity reflects the following:

a. Isotopic compositions based on the design basis primary coolant equilibrium concentrations at 1% fuel defects.

b. Iodine concentrations based on the thyroid inhalation weighting factors for I-131, I-132, I-133, I-134, and I-135 obtained from Federal Guidance Report 11 (Reference 41).

c. Noble gas concentrations based on the submersion weighting factors for Xe-133, Xe-133m, Xe-135m, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88 obtained from Federal Guidance Report 12 (Reference 42)

The Technical Specification 1 µCi/gm DE I-131 concentrations per nuclide in the primary coolant are calculated with the following equation:

$$DEI_{131}(i)(\mu Ci/gm) = \frac{\text{C(i) x CT}_{\text{tot}}}{\sum\{F(i)xC(i)\}} \qquad (15.5\text{-}2)$$

Where:

F(i) = DCF(i) / DCF $_{I-131}$

DCF(i)= Federal Guidance Report-11, Table 2-1 (Reference 41) Thyroid Dose Conversion Factor per Nuclide (Rem/Ci)

C(i) = design basis primary coolant equilibrium iodine concentration per nuclide (µCi/gm)

CT_{tot}= primary coolant total (DE I-131) Technical Specification iodine concentration (µCi/gm).

The CT_{tot} for the pre-accident iodine spike is 60 µCi/gm (transient Technical Specification limit for full power operation), or 60 times the primary coolant total iodine Technical Specification concentration.

The accident initiated iodine spike activities are based on an accident dependent multiplier, times the equilibrium iodine appearance rate. The equilibrium appearance rates are conservatively calculated based on the technical specification reactor coolant activities, along with the maximum design letdown rate, maximum Technical Specification based allowed primary coolant leakage, and an assumed ion-exchanger iodine efficiency of 100%.

The Technical Specification secondary liquid iodine concentration is determined using methodology similar to that described above for the primary coolant where CTtot is 0.1 µCi/gm DE I-131, and C(i) is the design basis secondary coolant equilibrium concentrations per nuclide.

The Technical Specification noble gas concentrations for the primary coolant are based on 270 µCi/gm DE Xe-133. The DE Xe-133 for noble gases is calculated as follows:

$$DEX_{133} = \Sigma\{F(i) \times C(i)\} \qquad (15.5\text{-}3)$$

Where:

F(i) = DCF(i) / DCF Xe-133

DCF(i) = EPA Federal Guidance Report No. 12 (Reference 42) Table III.1, Dose Coefficient per Nuclide [(rem-m^3)/(Ci-sec)]

C(i) = design basis primary coolant equilibrium noble gas concentration per nuclide (µCi/gm)

The noble gas and halogen primary and secondary coolant Technical Specification Activity Concentrations for Unit 1 and Unit 2 are presented in Table 15.5-78. The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop accident initiated iodine spike values), are presented in Table 15.5-79.

15.5.3.3 Gap Fractions for Non-LOCA Events

Regulatory Guide 1.183, July 2000, Table 3 provides the gap fractions for Non-LOCA events that are postulated to result in fuel damage for AST applications. The referenced gap fractions are contingent upon meeting Note 11 of Table 3 of Regulatory Guide 1.183. Note 11 indicates that the release fractions listed in Table 3 are "acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU." The burnup criterion associated with the maximum allowable linear heat generation rate is applicable to the peak rod average burnup in any assembly and is not limited to assemblies with an average burnup that exceeds 54 GWD/MTU.

DCPP has three design basis non-LOCA accidents that are postulated to result in fuel damage, i.e., the Locked Rotor Accident (LRA), the Fuel Handling Accident (FHA) and the Control Rod Ejection Accident (CREA).

To support flexibility of fuel management, and establish dose consequences that take into consideration fuel rods that may exceed the Regulatory Guide 1.183, Table 3, Note 11 linear heat generation criteria, the fuel gap fractions provided in Table 3 of Draft Guide (DG)-1199 (Reference 62) for all Non-LOCA events that are postulated to result in fuel damage with the exception of the CREA. This approach is acceptable (i.e., in lieu of developing plant specific fission gas release calculations using NRC approved methods and bounding power history to establish the gap fractions), since DCPP falls within, and intends to operate within, the maximum allowable power operating envelop for PWRs shown in Figure 1 of DG-1199.

In summary, the fuel gap activity fractions used to assess the dose consequences of the FHA and LRA are as follows:

Nuclide Group	FHA /LRA (based on DG-1199)
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

In accordance with Regulatory Guide 1.183 (Appendix H and Note 11 of Table 3), the gap fraction associated with the CREA is as follows:

Noble Gases:	10%
Halogens:	10%

Refer to Tables 15.5-80 for the isotopic concentrations in the gap assumed for the LRA and CREA. The isotopic concentrations assumed for the FHA are presented in Table 15.5-47C.

15.5.4 DELETED

15.5.5 POST-ACCIDENT METEOROLOGICAL CONDITIONS

The EAB and the LPZ atmospheric dispersion factors (χ/Q) utilized in the dose consequence analyses have been developed using Regulatory Guide 1.145, Revision 1 methodology and a continuous, temporally representative 5-year period of hourly meteorological data from the DCPP onsite meteorological tower; January 1, 2007 through December 31, 2011. Refer to Section 2.3.5.2.1 and Table 2.3-145.

Using the same hourly meteorological data, the χ/Q values applicable to on-site locations such as the control room and TSC, have been calculated using the "Atmospheric Relative CONcentrations in Building Wakes" (ARCON96) methodology (Reference 61). Refer to Section 2.3.5.2.2.

All of the release point and receptor locations are provided in Figure 2.3-5, while Tables 2.3-146 and 2.3-146A provide information on the release point / receptor combinations that were evaluated. Tables 2.3-147 and 2.3-148 provide the control room χ/Q values for the individual release point-receptor combinations for Unit 1 and Unit 2, respectively. Table 2.3-149 presents the χ/Q values for the individual post-LOCA release point - TSC receptor combinations for Unit 1 and Unit 2 applicable to the TSC normal intake and the center of the TSC boundary at roof level (considered an average value for potential TSC unfiltered in-leakage locations around the envelope). The Unit 1 and Unit 2 control room pressurization air intakes also serve the TSC during the emergency mode. Thus, the χ/Qs presented in Tables 2.3-147 and 2.3-148 for the control room pressurization intakes inclusive of the credit for dual intake design and ability to select the more favorable intake are also applicable to the TSC.

Note that the specific control room χ/Q values used in each of the accident analyses (and the specific TSC χ/Q values used for the LOCA) are presented in the accident-specific tables presented in Chapter 15.5. The χ/Q values selected for use in the dose consequence analyses are intended to support bounding analyses for an accident that occurs at either unit. They take into consideration the various release points-receptors applicable to each accident in order to identify the bounding χ/Q values and reflect the allowable adjustments and reductions in the values as discussed earlier and further summarized in the notes of Tables 2.3-147 through 2.3-149.

15.5.6 RATES OF ISOTOPE INHALATION

The breathing rates used in the calculations of inhalation doses are listed in Table 15.5-7A. These values are based on the average daily breathing rates provided in Section 4.1.3 of Regulatory Guide 1.183.

The active breathing rates are used for all onsite dose calculations, which are based on expected exposure times.

15.5.7 DELETED

15.5.8 RADIOLOGICAL ANALYSIS PROGRAMS

15.5.8.1 EMERALD (Revision I)

The EMERALD program (Reference 4) is designed for the calculation of radiation releases and exposures resulting from abnormal operation of a large PWR. The approach used in EMERALD is similar to an analog simulation of a real system. Each component or volume in the plant that contains a radioactive material is represented by a subroutine, which keeps track of the production, transfer, decay, and absorption of radioactivity in that volume. During the course of the analysis of an accident, activity is transferred from subroutine to subroutine in the program as it would be transferred from place to place in the plant. The rates of transfer, leakage, production, cleanup, decay, and release are read in as input to the program.

Subroutines are also included that calculate the onsite and offsite radiation exposures at various distances for individual isotopes and sums of isotopes. The program contains a library of physical data for 25 isotopes of most interest in licensing calculations, and other isotopes can be added or substituted. Because of the flexible nature of the simulation approach, the EMERALD program can be used for most calculations involving the production and release of radioactive materials, including design, operational and licensing studies. The complete description of the program, including models and equations, is contained in Reference 4.

The egress-ingress thyroid and whole body exposures from airborne activity following a LOCA are functions of containment activity, containment leakage, atmospheric dispersion, and excursion time. As part of original licensing basis, the EMERALD computer code was used to calculate the post-LOCA airborne activity concentrations, and then conventional exposure equations from Regulatory Guide 1.4, Revision 1, were used to calculate gamma, beta, and thyroid exposures (Reference 6).

15.5.8.2 DELETED

15.5.8.3 DELETED

15.5.8.4 DELETED

15.5.8.5 ISOSHLD II

ISOSHLD II (Reference 11) is a shielding code that is principally intended for use in calculating the radiation dose, at a field point, from bremsstrahlung and/or decay gamma rays emitted by radioisotope sources. This program, with the newly-added bremsstrahlung mode, is an extension of the earlier version (ISOSHLD). Five shield regions can be handled with up to twenty materials per shield; the source is considered to be the first shield region (i.e., bremsstrahlung and decay gamma rays are produced only in the source). Point kernel integration (over the source region) is used to calculate the radiation dose at a field point.

ISOSHLD II is used to determine the dose to the control room operator due to direct shine from the airborne activity inside the containment following a LOCA during daily egress-ingress for the duration of the accident.

15.5.8.6 DELETED

15.5.8.7 SAS2 / ORIGEN-S

ORIGEN-S is part of the SCALE 4.3 suite of codes which was developed by Oak Ridge National Laboratory (ORNL) for the NRC to perform standardized computer analyses for licensing evaluations. SAS2 is a control module that provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. SAS2 / ORIGEN-S (Reference 64) calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It properly accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions. It can calculate accurately the neutron-activated products, the actinides and the fission products in a reactor core. SAS2/ORIGEN-S is used to develop the equilibrium core activity inventory and the decayed fuel inventories after shutdown utilized to assess the design basis accidents, excluding tank ruptures.

ACTIVITY2

ACTIVITY2 (Reference 65) calculates the concentration of fission products in the fuel, coolant, waste gas decay tanks, ion exchangers, miscellaneous tanks, and release lines to the atmosphere for a pressurized water reactor system. The program uses a library of properties of more than 100 significant fission products and may be modified to include as many as 200 nuclides. The program output presents the activity and energy spectrum at the selected part of the system for any specified operating time.

ACTIVITY2 is used to develop the reactor coolant activity inventory (design and as limited by the plant Technical Specifications) utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.8 IONEXCHANGER

IONEXCHANGER (Reference 66) calculates the activity of nuclides in an ion exchanger or tank of a nuclear reactor plant by solving the appropriate growth-decay-purification equations. Based on a known feed rate of primary coolant or other fluid with known radionuclide activities, it calculates the activity of each nuclide and its products in the ion exchanger or tank at some later time. The program also calculates the specific gamma activity for each of the seven fixed energy groups.

IONEXCHANGER is used to develop the secondary coolant activity inventory (design and as limited by the plant Technical Specifications) utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.9 EN 113, Atmospheric Dispersion Factors

EN-113 Atmospheric Dispersion Factors (Reference 73) calculates χ/Q values at the EAB and LPZ following the methodology and logic outlined in Regulatory Guide 1.145, Revision 1. The program can handle single or multiple release points for a specified time period and set of site-specific and plant-specific parameters. A release point can be identified as either of two types of release (i.e., ground or elevated), time periods for which sliding averages are calculated (i.e., 1 to 624 hours and/or annual average), applicable short-term building wake effect, meandering plume, long-term building height wake effect, and a wind speed value to be assigned to calm conditions. Downwind distances can be assigned for each of the sixteen 22.5-degree sectors for two irregular boundaries and for ten additional concentric boundaries used only in the annual average calculation. EN-113 performs the same calculations as the NRC PAVAN code except that EN-113 calculates χ/Q values for the various averaging periods directly using hourly meteorological data whereas PAVAN uses a joint frequency distribution of wind speed, wind direction, and stability class.

EN-113 is used to develop the DCPP site boundary atmospheric dispersion factors utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.10 ARCON96

ARCON96 (Reference 74) was developed by Pacific Northwest National Laboratory (PNNL) for the NRC to calculate relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. ARCON96 has the ability to evaluate ground-level, vent, and elevated stack releases; it implements a straight-line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects. The methodology is also able to evaluate diffuse and area source releases using the virtual point source technique, wherein initial values of the dispersion coefficients are assigned based on the size of the diffuse or area source. Hourly, normalized concentrations (χ/Q) are calculated from hourly meteorological data. The hourly values are averaged to form χ/Qs for periods ranging from 2 to 720 hours in duration. The calculated values for each period are used to form cumulative frequency distributions.

ARCON96 is used to develop the control room and TSC atmospheric dispersion factors utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.11 SWNAUA

SWNAUA (Reference 67) is a derivative of industry computer code NAUA/Mod 4 which was originally developed in Germany and was based on experimental data. NAUA/Mod 4 addressed particulate aerosol transport and removal following a LOCA at an LWR. It developed removal coefficients to address physical phenomena such as gravitational settling (also called gravitational sedimentation), diffusion, particle growth due to agglomeration, etc using time-dependent airborne aerosol mass. NAUA4 (included in the NRC Source Term Code Package) was used by NRC during the initial evaluations of post-TMI data. NAUA/Mod 4 was modified to include spray removal and diffusiophoretic effects suitable for design basis accident analyses. A version of SWNAUA (SWNAUA-HYGRO) was proven to be the most reliable of more than a dozen international entries, in making predictions of aerosol removal for the LWR Aerosol Containment Experiments (LACE) series.

SWNAUA is used to develop the time dependent post LOCA particulate aerosol removal coefficients in the sprayed and unsprayed regions of containment.

15.5.8.12 RADTRAD 3.03

RADTRAD 3.03 (Reference 68) is a NRC sponsored program, developed by Sandia National Labs (SNL). It can be used to calculate radiological doses to the public, plant operators and emergency personnel due to environmental releases that resulting from postulated design basis accidents at light water reactor (LWR) power plants. The RADTRAD 3.03 (GUI Interface Mode) includes models for a variety of processes that can attenuate and/or transport radionuclides. It can model sprays and natural deposition that reduce the quantity of radionuclides suspended in the containment or other compartments. It can model the flow of radionuclides between compartments within a building, from buildings into the environment, and from the environment into a control room). These flows can be through filters, piping, or simply due to air leakage. RADTRAD 3.03 can also model radioactive decay and in-growth of daughters. Ultimately the program calculates the Thyroid and TEDE dose (rem) to the public located offsite and to onsite personnel located in the control room due to inhalation and submersion in airborne radioactivity based on user specified, fuel inventory, nuclear data, dispersion coefficients, and dose conversion factors.

RADTRAD is used to develop the TEDE dose to the public located offsite and to onsite personnel located in the control room due to inhalation and submersion in airborne radioactivity following design basis accidents excluding tank ruptures

15.5.8.13 PERC2

PERC2 (Reference 69) is a multi-region activity transport and radiological dose consequence program. It includes the following major features:

(1) Provision of time-dependent releases from the reactor coolant system to the containment atmosphere.

(2) Provision for airborne radionuclides for both TID and AST release assumptions, including daughter in growth.

(3) Provision for calculating the CEDE to individual organs as well as EDE from inhalation, deep dose equivalent and beta from submersion, and TEDE.

(4) Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model.

(5) Provision for calculating instantaneous and integrated gamma radiation source strengths as well as activities for the inventoried radionuclides to permit direct assessment of the dose from contained / or external sources for equipment qualification, vital area access and control room and EAB direct shine dose estimates.

PERC2 is used to calculate the accident energy release rates and integrated gamma energy releases versus time for the various post-LOCA external and contained radiation sources. This source term information is input into SW_QADCGGP to develop the direct shine dose to the control room. PERC2 is also used to develop the decay heat in the RWST and MEDT and develop the TEDE dose to personnel located in the TSC due to inhalation and submersion in airborne radioactivity following LOCA.

15.5.8.14 SW-QADCGGP

SW-QADCGGP (Reference 70) is a variant of the QAD point kernel shielding program originally written at the Los Alamos Scientific Laboratory by R. E. Malenfant. The QADCGGP version implements combinatorial geometry and the geometric progression build-up factor algorithm. The SW-QADCGGP implements a graphical indication of the status of the computation process.

SW-QADCGGP is used to develop the direct shine dose to the operator in the control room, TSC, and EAB.

15.5.8.15 GOTHIC

GOTHIC (Reference 71) is developed and maintained by Numerical Applications Incorporated (NAI) and an integrated, general purpose thermal-hydraulics software package for design, licensing, safety, and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC solves the conservation equations for mass, momentum, and energy for multicomponent, multi-phase flow in lumped parameter and/or multi-dimensional geometries. The phase balance equations are coupled by mechanistic models for interface mass, energy, and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal non equilibrium between phases and unequal phase velocities, including countercurrent flow. Other phenomena include models for commonly available safety equipment, heat transfer to structures, hydrogen burn, and isotope transport.

GOTHIC is used to estimate the containment and sump pressure and temperature response with recirculation spray, the temperature transient in the RWST / MEDT gas and liquid due to incoming sump water leakage / inflow / decay heat from the RWST / MEDT fission product inventory, and the volumetric release fraction transient from the RWST / MEDT gas space to the environment.

15.5.9 CONTROL ROOM DESIGN AND TRANSPORT MODEL

The control room serves both units and is located at El 140’ of the Auxiliary Building. The walls facing the Unit 1 and Unit 2 containments (i.e., the north and south walls) are made of 3’-0” concrete, whereas the control room east and west walls are made up of 2’-0” concrete. The floor and ceiling thickness / material reflect a minimum of 2’-0” and 3’-4” of concrete, respectively. The control room Mechanical Equipment and HVAC room is located adjacent to the control room (east side), at El 154’-6”.

The control room has a normal intake per unit (each located on opposite sides the auxiliary building; i.e. north and south), and a pressurization flow intake per unit (each located on either side of the turbine building; i.e. north and south). The control room pressurization air intakes have dual ventilation outside air intake design as defined by Regulatory Position C.3.3.2 of Regulatory Guide 1.194, June 2003 (refer to Section 2.3.5.2.2).

During normal operation (CRVS Mode 1), both control room normal intakes are operational. Redundant PG&E Design Class I radiation monitors located at each control room normal intake have the capability of isolating the control room normal intakes on detection of high radiation and switching the control room ventilation system (CRVS) to Mode 4 operation (i.e., control room filtered intake and pressurization).

CRVS Mode 4 operation utilizes redundant PG&E Design Class I radiation monitors located at each control room pressurization air intake and the provisions of acceptable control logic to automatically select the least contaminated inlet at the beginning of the accident, and manually select the least contaminated inlet during the course of the accident in accordance with Regulatory Guide 1.194, June 2003. Thus, during Mode 4 operation the dose consequence analyses can utilize the χ/Q values for the more favorable pressurization air intake reduced by a factor of 4 to credit the “dual intake” design (refer to Section 2.3.5.2.2).

Other signals that initiate CRVS Mode 4 operation include the safety injection signal (SIS) and Containment Isolation Phase A. The SIS does not directly initiate CRVS Mode 4; however, it initiates Containment Isolation Phase A which initiates Mode 4 operation.

During normal operations, unfiltered air is drawn into the control room envelope (refer to Table 15.5-81) from the Unit 1 and Unit 2 normal intakes. In response to a control room radiation monitor or SIS, the control room switches to CRVS Mode 4 operation, and control logic ensures that the CRVS pressurization fan of the non-accident unit is initiated and air is taken from the less contaminated of the Unit 1 or Unit 2 control room pressurization air intakes. The control room pressurization flowrate used in the dose consequence analyses is selected to maximize the estimated dose in the control room. With the exception of 100 cfm which is unfiltered due to backdraft damper leakage, all pressurization flow is filtered.

The allowable methyl iodide penetration and filter bypass for the CRVS Mode 4 Charcoal Filter is controlled by Technical Specifications and the ventilation filter testing program (VFTP), and is 2.5% and <1%, respectively. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses (refer to Section 9.4.1 and Table 9.4-2). Thus, the control room charcoal filter efficiency for elemental and organic iodine used in the DCPP safety analyses is 100% - [(2.5% + 1%) x 2] = 93%. The acceptance criteria for the in-place test of the high efficiency particulate air (HEPA) filters in Technical Specifications is a “penetration plus system bypass” < 1.0%. Similar to the charcoal filters, the HEPA filter efficiency for particulates used in the DCPP safety analyses is 100% - [(1%) x 2] = 98%.

During Mode 4 operation, the control room air is also recirculated and a portion of the recirculation flow filtered through the same filtration unit as the pressurization flow. Refer to Table 15.5-81 for a summary of recirculation flow rates.

Unfiltered inleakage into the control room during Mode 1 and Mode 4 is assumed to be 70 cfm (includes 10 cfm for inleakage due to egress-ingress based on the guidance provided in SRP 6.4).

For purposes of estimating the post-accident dose consequences, the control room is modeled as a single region. When in CRVS Mode 4, the Mode 1 intakes are isolated and outside air is a) drawn into the control room through the filtered emergency intakes; b) enters the control room as infiltration, c) enters the control room during operator egress-ingress, and d) enters the control room as unfiltered leakage via the emergency intake back draft dampers. The direction of flow uncertainty on the CRVS ventilation intake flowrates (normal as well as accident), are selected to maximize the dose consequence in the control room.

The dose consequence analyses for the LOCA, MSLB, SGTR, and the CREA, assume a LOOP concurrent with reactor trip.

In addition, and as noted in Section 15.5.1.2, in accordance with current licensing basis the non-accident unit is assumed unaffected by the LOOP. Thus, to address the effect of a LOOP, and taking into consideration the fact that the time of receipt of the signal to switchover from CRVS Mode 1 to Mode 4 is accident specific:

a. Automatic isolation of the control room normal intake of the "non-accident" unit is delayed by 12 seconds from receipt of the signal, to switch to CRVS Mode 4. This delay takes into account a 2 second SIS processing time and a 10 second damper closure time.
b. Automatic isolation of the control room normal intake of the accident unit, and credit for CRVS Mode 4 operation is delayed by 38.2 seconds from receipt of the signal to switch to CRVS Mode 4. This delay takes into account a) 28.2 seconds for the diesel generator to become fully operational including sequencing delays, and b) 10 seconds for the control room ventilation dampers to re-align. The 2 second SIS processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay. In addition, and as discussed earlier, the CRVS system design ensures that upon receipt of a signal to switch to Mode 4, the control room pressurization fans of the non-accident unit is initiated; thus fan ramp-up is assumed to occur well within the 38.2 seconds delay discussed above, unhampered by a LOOP.

The dose consequence analyses for the LRA and the LOL event assume that the control room remains in normal operation mode and do not credit CRVS Mode 4 operation.

Table 15.5-81 lists key assumptions / parameters associated with control room design.

15.5.10 RADIOLOGICAL CONSEQUENCES OF CONDITION II FAULTS

15.5.10.1 Acceptance Criteria

The radiological consequences of accidents analyzed in Section 15.2 (or from other events involving insignificant core damage, but requiring atmospheric steam releases) shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000, as outlined below:

EAB and LPZ Dose Criteria

Regulatory Guide 1.183 does not specifically address Condition II scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of

Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Technical Support Center Dose Criteria

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.10.2 Identification of Causes and Accident Description

15.5.10.2.1 Activity Release Pathways

As reported in Section 15.2, Condition II faults are not expected to cause breach of any of the fission product barriers, thus preventing fission product release from the core or plant. Under some conditions, however, small amounts of radioactive isotopes could be released to the atmosphere following Condition II events as a result of atmospheric steam dumps required for plant cooldown. The particular Condition II events that are expected to result in some atmospheric steam release are:

(1) Loss of electrical load and/or turbine trip
(2) Loss of normal feedwater
(3) LOOP to the station auxiliaries
(4) Accidental depressurization of the MSS

The amount of steam released following these events depends on the time relief valves remain open and the availability of condenser bypass cooling capacity.

The mass of environmental steam releases for the Loss of Load Event bound all Condition II events.

A LOL event is different from the Loss of Alternating Current (AC) power condition, in that offsite AC power remains available to support station auxiliaries (e.g., reactor coolant pumps). The Loss of AC power condition results in the condenser being unavailable and reactor cooldown being achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling.

In-keeping with the concept of developing steam releases that bound all Condition II events and encompass the LRA and CREA, the analysis performed to determine the mass of steam released following a LOL event incorporates the assumption of Loss of offsite power to the station auxiliaries.

Although Regulatory Guide 1.183 does not provide specific guidance with respect to scenarios to be assumed to determine radiological dose consequences from Condition II events, the scenario outlined below for the LOL analysis is based on the conservative assumptions outlined in Regulatory Guide 1.183 for the MSLB, and was analyzed to bound all Condition II events that result in environmental releases.

Table 15.5-9A lists the key assumptions / parameters utilized to develop the radiological consequences following a LOL event. The conservative assumptions utilized to assess the dose consequences ensure that it represents the Limiting Condition II event.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOL event.

15.5.10.2.2 Activity Release Transport Model

No melt or clad breach is postulated for the LOL (refer to Section 15.2.7). Thus, and in accordance with Regulatory Guide 1.183, Appendix E, Item 2, the activity released is based on the maximum coolant activity allowed by the plant Technical Specifications, which focus on the noble gases and iodines. In accordance with Regulatory Guide 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical

Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is the Technical Specification limit of 0.1 μCi/gm DE I-131.

Plant Technical Specification limits primary to secondary steam generator (SG) tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the LOL dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The entire primary-to-secondary tube leakage of 0.75 gpm (maximum leak rate at STP conditions; total for all 4 SGs) is leaked into an effective SG. In accordance with Regulatory Guide 1.183, the pre-existing iodine activity in the secondary coolant and iodine activity due to reactor coolant leakage into the 4 SGs is assumed to be homogeneously mixed in the bulk secondary coolant. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant. Therefore, per Regulatory Guide 1.183, the iodines are released to the environment via the via the main steam safety valves (MSSVs) and 10% atmospheric dump valves (ADVs) in proportion to the steaming rate and the inverse of a partition coefficient of 100. The iodine releases from the SG are assumed to be 97% elemental and 3% organic. The noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a LOL event is discharged to the environment from the steam generators via the MSSVs / 10% ADVs. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the Residual Heat Removal (RHR) system and environmental releases are terminated.

15.5.10.2.3 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the LOL event, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a LOL event at either unit is presented in Table 15.5-9.

15.5.10.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. A summary of the critical assumptions associated with control room response and activity transport for the LOL event is provided below:

Control Room Ventilation

The LOL event does not initiate any signal which could automatically start the control room pressurization air ventilation. Thus the dose consequence analysis for the LOL event assumes that the control room remains in normal operation mode.

Control Room Atmospheric Dispersion Factors

Due to the proximity of the MSSVs/10% ADVs to the control room normal intake of the affected unit and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the affected unit (closest to the release point) will be insignificant. Therefore, only the unaffected unit's control room normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs (refer to Section 2.3.5.2.2 for detail).

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an LOL event at either unit are provided in Table 15.5-9B. The χ/Q values presented in Table 15.5-9B take into consideration the various release points-receptors applicable to the LOL to identify the bounding χ/Q values applicable to a LOL event at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding Control Room dose following a LOL event at either unit is presented in Table 15.5-9.

15.5.10.3 Conclusions

It can be concluded from the results discussed that the occurrence of any of the events analyzed in Section 15.2 (or from other events involving insignificant core damage, but requiring atmospheric steam releases) will result in insignificant radiation exposures and are bounded by the LOL event.

Additionally, the analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for the 2-hour period following the onset of the postulated fission product release is within 0.025 Sv (2.5 rem) TEDE as shown in Table 15.5-9.

(2) The radiation dose to an individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.02 Sv (2.5 rem) TEDE as shown in Table 15.5-9.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-9.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the LOL is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

15.5.11 RADIOLOGICAL CONSEQUENCES OF A SMALL-BREAK LOSS-OF-COOLANT ACCIDENT

15.5.11.1 Acceptance Criteria

The radiological consequences of a SBLOCA shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000, as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.11.2 Identification of Causes and Accident Description

As discussed in Section 15.3.1, a SBLOCA (defined in UFSAR Chapter 15.3.1 as a break that is large enough to actuate the emergency core cooling system), is not expected to cause fuel cladding failure. For this reason, the only activity release to the containment will be the dissolved noble gases and iodine in the reactor coolant water expelled from the pipe rupture. Some of this activity could be released to the containment atmosphere as the water flashes, and some of this amount could leak from the containment as a result of a rise in containment pressure.

The possible radiological consequence of this event is expected to be bounded by the “containment release” scenario of the CREA discussed in Section 15.5.23.

The dose consequences following a SBLOCA will be significantly less than a CREA since the CREA is postulated to result in 10% fuel damage, whereas the SBLOCA has no fuel damage.

As demonstrated in Table 15.5-52, the dose consequences at the EAB and LPZ and in the Control Room following a CREA is within the acceptance criteria applicable to the SBLOCA.

15.5.11.3 Conclusions

On the basis of this conservative comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following a SBLOCA will remain within the acceptance criteria listed in Section 15.5.11.1.

15.5.12 RADIOLOGICAL CONSEQUENCES OF MINOR SECONDARY SYSTEM PIPE BREAKS

15.5.12.1 Acceptance Criteria

The radiological consequences of accidents analyzed in Section 15.3 such as minor secondary system pipe breaks shall not exceed the dose limits of 10 CFR 50.67, and will meet the does acceptance criteria of Regulatory Guide 1.183, July 2000, as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

<u>Control Room Dose Criteria</u>

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.12.2 Identification of Causes and Accident Description

The effects on the core of sudden depressurization of the secondary system caused by an accidental opening of a steam dump, relief or safety valve were described in Section 15.2 and apply also to the case of minor secondary system pipe breaks. As shown in that analysis, no core damage or fuel rod failure is expected to occur. In Section 15.4.2, analyses are presented that show the effects on the core of a major steam line break, and, in this case also, no fuel rod failures are expected to occur.

The analyses presented in Section 15.3.2 demonstrate that a DNBR of less than the safety analysis limit will not occur anywhere in the core in the event of a minor secondary system pipe rupture.

The steam releases following a minor secondary line break is expected to be significantly less than that associated with a main line steam break.

As demonstrated in Table 15.5-34, the dose consequences at the EAB and LPZ and in the Control Room following a MSLB is within the acceptance criteria applicable to the minor secondary line break.

15.5.12.3 Conclusions

On the basis of this conservative comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following a minor secondary system pipe rupture will remain within the acceptance criteria listed in Section 15.5.12.1.

15.5.13 RADIOLOGICAL CONSEQUENCES OF INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION

15.5.13.1 Acceptance Criteria

Fuel assembly loading errors shall be prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses supporting Section 15.3.3 shall confirm that no events leading to radiological consequences shall occur as a result of loading errors.

15.5.13.2 Identification of Causes and Accident Description

Fuel and core loading errors such as inadvertently loading one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or loading a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. The inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods is also included among possible core loading errors. Because of margins present, as discussed in detail in Section 15.3.3, no events leading to radiological consequences are expected as a result of loading errors.

15.5.13.3 Conclusions

Because of margins present, as discussed in detail in Section 15.3.3, no events leading to radiological consequences are expected as a result of loading errors.

15.5.14 RADIOLOGICAL CONSEQUENCES OF COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.5.14.1 Acceptance Criteria

The radiological consequences of small amounts of radioactive isotopes that could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown following a complete loss of forced reactor coolant flow shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

<u>Control Room Dose Criteria</u>

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.14.2 Identification of Causes and Accident Description

As discussed in Section 15.3.4, a complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all RCPs. 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.

The analysis performed and reported in Section 15.3.4 has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit during the transient, and thus there is no cladding damage or release of fission products to the RCS.

The possible radiological consequence of a complete loss of forced reactor coolant flow is expected to be bounded by the conservative Loss-of-Load scenario with a coincident Loss of offsite power described in Section 15.5.10.

As demonstrated in Table 15.5-9, the dose consequences at the EAB and LPZ and in the Control Room following a Loss of Load is within the acceptance criteria applicable to the complete loss of forced reactor coolant flow.

15.5.14.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following a complete loss of forced reactor coolant flow will remain with the acceptance criteria listed in Section 15.5.14.1.

15.5.15 RADIOLOGICAL CONSEQUENCES OF AN UNDERFREQUENCY ACCIDENT

15.5.15.1 Acceptance Criteria

The radiological consequences of small amounts of radioactive isotopes that could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown following an underfrequency accident shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population

zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.15.2 Identification of Causes and Accident Description

A transient analysis for this unlikely event is discussed in Section 15.3.4. The analysis demonstrates that for an underfrequency accident, the DNBR does not decrease below the safety analysis limit during the transient, and thus there is no cladding damage or release of fission products to the RCS. However, small amounts of radioactive isotopes could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown.

The possible radiological consequence of this event is expected to be bounded by the conservative Loss-of-Load scenario with a coincident Loss of offsite power described in Section 15.5.10.

As demonstrated in Table 15.5-9, the dose consequences at the EAB and LPZ and in the Control Room following a Loss of Load is within the acceptance criteria applicable to an underfrequency accident.

15.5.15.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following an underfrequency event will remain within the acceptance criteria listed in Section 15.5.15.1.

15.5.16 RADIOLOGICAL CONSEQUENCES OF A SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER

15.5.16.1 Acceptance Criteria

The radiological consequences of a single RCCA withdrawal shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.16.2 Identification of Causes and Accident Description

A complete transient analysis of this accident is presented in Section 15.3.5. For the condition of one RCCA fully withdrawn with the rest of the bank fully inserted, at full power, an upper bound of the number of fuel rods experiencing DNBR less than the safety analysis limit is 5 percent of the total fuel rods in the core.

The possible radiological consequence of this event is expected to be bounded by the CREA discussed in Section 15.5.23.

The dose consequences following a single rod cluster control assembly withdrawal will be less than a CREA since the CREA is postulated to result in 10% fuel damage, whereas the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power has only 5% fuel damage.

As demonstrated in Table 15.5-52, the dose consequences at the EAB and LPZ and in the Control Room following a CREA is within the acceptance criteria applicable to the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power.

15.5.16.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power will remain within the acceptance criteria listed in Section 15.5.16.1.

15.5.17 RADIOLOGICAL CONSEQUENCES OF MAJOR RUPTURE OF PRIMARY COOLANT PIPES

15.5.17.1 Acceptance Criteria

The radiological consequences of a LOCA shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below:

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.25 Sv (25 rem) TEDE.

<u>Control Room Dose Criteria</u>

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

<u>Technical Support Center Dose Criteria</u>

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.17.2 Identification of Causes and Accident Description

15.5.17.2.1 Activity Release Pathways

The accidental rupture of a main coolant pipe is the event assumed to initiate a LBLOCA. Analyses of the response of the reactor system, including the ECCS, to ruptures of various sizes have been presented in Sections 15.3.1 and 15.4.1. As demonstrated in these analyses, the ECCS, using emergency power, is designed to keep cladding temperatures well below melting and to limit Zr-H_2O reactions to an insignificant level. As a result of the increase in cladding temperature and the rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Following the cladding failure, some activity would be released to the primary coolant and subsequently to the inside of the containment building. Active mechanisms include radioactive particulate and iodine removal by the containment sprays inclusive of the containment air mixing provided by the CFCUs. Section 6.2 describes the design and operation of the CSS and the CFCUs.

Regulatory Guide 1.183, Appendix A, identifies the large break LOCA as the design basis case of the spectrum of break sizes for evaluating performance of release mitigation systems and the containment, and for facility siting relative to radiological consequences.

DCPP has identified six activity release paths following a LOCA:

1. Release via the Containment Pressure / Vacuum Relief pathway to the environment until the containment isolation valves are closed.

2. Containment leakage to the environment after containment isolation is achieved.

3. Sump water leakage from ESF systems that recirculate sump water outside containment.

4. Failure of the RHR pump seal at T=24 hrs resulting in a 50 gpm leak of sump water for 30 mins.

5. Releases to the environment from the Miscellaneous Equipment Drain Tank (MEDT) which collects component leakage hard-piped to the MEDT. The collected fluid includes both post-LOCA sump water and other non-radioactive fluid.

6. Releases to the environment via the refueling water storage tank (RWST) vent due to post-LOCA sump fluid back-leakage into the RWST via the mini-flow recirculation lines connecting the high head and low head safety injection pump discharge piping to the RWST.

The LOCA dose consequence analysis follows the guidance provided in the pertinent sections of Regulatory Guide 1.183 including Appendix A. Table 15.5-23A lists the key assumptions / parameters utilized to develop the radiological consequences following a LOCA at either unit.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOCA.

15.5.17.2.2 Activity Release Transport Model

15.5.17.2.2.1 Containment Pressure /Vacuum Relief Line Release

In accordance with Regulatory Guide 1.183, Appendix A, Section 3.8, for containments such as DCPP that are routinely purged during normal operations, the dose consequence analysis must assume that 100% of the radionuclide inventory in the primary coolant is released to the containment at the initiation of the LOCA. The inventory of the release from containment should be based on Technical Specifications primary coolant equilibrium activity (refer to Table 15.5-78). Iodine spikes need not be considered.

Thus, in accordance with the above guidance, the 12 inch containment vacuum / over pressure relief valves are assumed to be open to the extent allowed by Technical Specifications (i.e., blocked to prevent opening beyond 50 degrees), at the initiation of the LOCA, and the release via this pathway terminated as part of containment isolation. The analysis assumes that 100% of the radionuclide inventory in the primary coolant, assumed to be at Technical Specification levels, is released to the containment at T= 0 hours. It is conservatively assumed that 40% of release flashes and is instantaneously and homogeneously mixed in the containment atmosphere and that the activity associated with the volatiles, i.e., 100% of the noble gases and 40% of the iodine in the reactor coolant is available for release to the environment via this pathway.

Containment pressurization (due to the RCS mass and energy release), combined with the relief line cross-sectional area, results in a 218 acfs release of containment air to the environment for a conservatively estimated period of 13 seconds. Credit is taken for pressure boundary integrity of the containment pressure / vacuum relief system ductwork which is classified as PG&E Design Class II, and seismically qualified; thus, environmental releases are via the Plant Vent.

Since the release is isolated within 13 seconds after LOCA, i.e., before the onset of the gap phase release, releases associated with fuel damage are not postulated. The chemical form of the iodine released from the RCS to the environment is assumed to be 97% elemental and 3% organic.

15.5.17.2.2.2 Containment Leakage

The inventory of fission products in the reactor core available for release into the containment following a LOCA is provided in Table 15.5-77 which represents a conservative equilibrium reactor core inventory of the dose significant isotopes, assuming maximum full power operation at 1.05 times the current licensed thermal power, and taking into consideration fuel enrichment and burnup. The notes provided at the bottom of Table 15.5-77 provide information on isotopes used to estimate the inhalation and submersion doses following a LOCA, vs. isotopes that are considered to estimate the post-LOCA direct shine dose.

Per Regulatory Guide 1.183, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core.

In accordance with Regulatory Guide 1.183:

a. Two fuel release phases are considered for DBA analyses: (a) the gap release, which begins 30 seconds after the LOCA and continues to t=30 mins and (b) the early In-Vessel release phase which begins 30 minutes into the accident and continues for 1.3 hours (i.e., t=1.8 hrs).
b. The core inventory release fractions, by radionuclide groups, for the gap and early in-vessel damage are as follows:

Group	**Gap Release Phase**	**Early In-Vessel Release Phase**
Noble gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Group	-	0.05
Ba, Sr	-	0.02
Noble Metals	-	0.0025
Cerium Group	-	0.0005
Lanthanides	-	0.0002

Note: Footnote 10 criterion in Section 3.2 of RG 1.183 is met in that peak fuel rod burnup is limited to 62,000 MWD/MTU.

The elements in each radionuclide group released to the containment following a LOCA are assumed to be as follows (note that the groupings were expanded from that in Regulatory Guide 1.183 to address isotopes in the core with similar characteristics; the added isotopes are in bold font):

The elements in each radionuclide group released to the containment following a LOCA are assumed to be as follows (note that the groupings were expanded from that in Regulatory Guide 1.183 to address isotopes in the core with similar characteristics; the added isotopes are in bold font):

Noble gases: Xe, Kr
Halogens: I, Br
Alkali Metals: Cs Rb
Tellurium Grp: Te, Sb, Se, **Sn, In, Ge, Ga, Cd, As, Ag**
Ba,Sr: Ba, Sr
Noble Metals: Ru, Rh, Pd, Mo, Tc, Co
Cerium Grp: Ce, Pu, Np, **Th**
Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, **Gd, Ho, Tb**

As discussed in Section 6.2.3.3.7, the design includes chemical addition into the containment spray system which ensures a long term sump pH equal to or greater than 7.0. Thus, the chemical form of the radioiodine released from the fuel is assumed to be 95% particulate (cesium iodide (CsI)), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

Isotopic decay, containment leakage, selected natural removal mechanisms and spray removal are credited to deplete the inventory of fission products airborne in containment.

Isotopic decay, containment leakage, selected natural removal mechanisms and spray removal are credited to deplete the inventory of fission products airborne in containment.

Containment spray in the injection and recirculation mode is utilized as one of the primary means of fission product cleanup following a LOCA. Mixing of the effectively sprayed volume of containment, with the unsprayed volume of the containment is enhanced by operation of the PG&E Design Class I containment fan coolers. In order to quantify the effectiveness of the containment spray system, both the volume fraction of containment that is sprayed, and the mixing rate between the sprayed and unsprayed volumes are quantified.

The LOCA analysis is based on an assumed worst case single failure of loss of one ESF train. A single train ESF consists of one train of ECCS, one train of CSS, and two Containment Fan Cooling Units (CFCUs). A single train scenario is selected to be consistent with the use of reduced iodine and particulate removal coefficients associated with single train operation.

a. <u>Containment Spray Duration</u>: Containment Spray in the injection mode is initiated at 111 seconds after the LOCA and terminated at 3798 seconds. Manual operation is credited to initiate containment recirculation spray within twelve (12) minutes after injection spray is terminated. Thus, based on single train operation, containment spray in the recirculation mode is initiated at 4518 seconds, and terminated 5 hours later at 22,518 seconds. In summary, containment spray operation (injection plus recirculation) is credited for 6.25 hrs post-LOCA, with a twelve minute gap after injection spray is terminated.

b. Containment Spray Coverage: As discussed in Section 6.2.3.3.7.1, the containment sprays are estimated to effectively cover 82.5% of the containment free volume during the containment spray injection as well as spray recirculation mode.

c. Mixing between Sprayed and Unsprayed Regions of Containment: The containment mixing rate between the sprayed and unsprayed regions following a LOCA is determined to be 9.13 turnovers of the unsprayed regions per hour. This mixing rate is based on the operation of two CFCU with a total volumetric flow rate that addresses surveillance margins and uncertainty, between the unsprayed regions and sprayed regions. Review of the layout and arrangement of the intake and exhaust registers of the CFCUs indicate that the air intakes are all located above the operating floor (sprayed region) and the air discharge registers are all located below the operating floor in the unsprayed region. Additional review of the containment configuration including the location of the major openings in the containment structure, and various active and passive mixing mechanisms, results in the conclusion that following a LOCA, credit can be taken for a) the entire flowrate provided by each operating CFCU to support mixing between the sprayed and unsprayed regions, and b) homogeneous mixing within the sprayed and unsprayed regions, of the volume of air transferred from one region to the other due to CFCU operation. In accordance with Regulatory Guide 1.183, Appendix A, Section 3.3, prior to CFCU initiation, the dose consequence model assumes a mixing rate attributable to natural convection between the sprayed and unsprayed regions of 2 turnovers of the unsprayed region per hour.

d. Fission Product Removal: The fission product removal coefficients developed for the LOCA reflect the following guidance documents:

   i. Elemental iodine removal coefficients are calculated using guidance provided in Standard Review Plan Section 6.5.2, Revision 4 (Reference 80), which is invoked by Regulatory Guide 1.183, Appendix A, Section 3.3.

   ii. Time dependent particulate aerosol removal coefficients are estimated using Regulatory Guide 1.183, Appendix A, Section 3.3, which permits the use of time-dependent particulate aerosol removal coefficients by invoking NUREG/CR 5966, June 1993 (Reference 81), and indicates that no reduction in particulate aerosol removal coefficients is required when a DF of 50 is reached, if the removal rates are based on the calculated time-dependent airborne aerosol mass. There are several aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. For DCPP, agglomeration of the aerosol is considered in both sprayed and unsprayed regions. In the sprayed region, the particulate removal calculation takes credit for the removal effectiveness of sprays and diffusiophoresis (aerosol removal due to steam condensation). Computer program SWNAUA is used to develop the time dependent particulate aerosol removal coefficients which reflect the effect of diffusiophoresis and sprays. Gravitational settling is considered only in the unsprayed region.

The methodology used to develop the elemental iodine and particulate removal coefficients in the sprayed and unsprayed region of the containment is discussed in Section 6.2.3.3.7.2. The total elemental iodine and particulate removal coefficients in the sprayed and unsprayed region of the containment as a function of time are summarized in Table 6.2-32.

In summary, the activity transport model takes credit for aerosol removal due to steam condensation and via containment spray based on spray flowrates associated with minimum ESF during the containment spray injection and recirculation mode. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol removal lambdas, and isotopic in-growth due to decay.
During spray operation in the injection mode, the elemental iodine removal rate for the sprays exceeds 20 hr-1, the maximum value permitted by NUREG-0800, Standard Review Plan Section 6.5.2; thus the elemental iodine removal rate attributable to sprays is limited to 20 hr-1. During recirculation spray operation, the elemental removal rate for the sprays is 19.34 hr-1. As discussed in 6.2.3.3.7.2, the wall deposition removal coefficient for elemental iodine has been calculated with the model provided in NUREG-0800, SRP Section 6.5.2. In sprayed and unsprayed regions, prior to spray actuation, the wall deposition removal coefficient is estimated to be 2.74 hr-1, while during spray operation, and in the sprayed region only, the wall deposition removal coefficient is estimated to be 0.57 hr-1.

In the unsprayed region, the aerosol removal lambdas reflect gravitational settling. No credit is taken for elemental iodine removal in the unsprayed region.

Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the DF for particulate iodine. The maximum DF for elemental iodine is based on Standard Review Plan Section 6.5.2 and is limited to a DF of 200.

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment at the containment technical specification leak rate for the first day, and half that leak rate for the remaining duration of the accident (i.e., 29 days). To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point, the plant Vent, the Containment Penetration Area GE (EL 140’) and the Containment Penetration Areas GW/FW (EL 140’).

15.5.17.2.2.3 ESF System Leakage Outside Containment

The fluid that collects in the containment recirculation sump after a LOCA (i.e., the fluid contents of reactor coolant system, the RWST, the NaOH tank and the accumulators) contain radioactive fission products that has been released from the core as a result of the LOCA.

The containment recirculation sump water is circulated by the RHR pumps, cooled via the RHR heat exchangers, returned to the containment via the RHR system piping and the CSS piping, passed through the RCS and the containment spray nozzles, and finally returned to the containment recirculation sump. In the event of circulation loop leakage in the auxiliary building, post-LOCA activity has a pathway to the atmosphere.

The complete RHR system and CSS descriptions; including detection of leakage, equipment isolation, and corrective maintenance, are contained in Sections 5.5.6 and 6.2.2, respectively.

In accordance with Regulatory Guide 1.183, with the exception of noble gases, all the fission products released from the core during the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment recirculation sump water at the time of release from the fuel. A minimum sump water volume of 480,015 gallons is utilized in this analysis.

In accordance with Regulatory Guide 1.183, the ESF systems that recirculate sump fluids outside containment are analyzed to leak at twice the sum of the administratively controlled total allowable leakage applicable to all components in the ESF recirculation systems. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

ESF leakage is assumed to occur at initiation of the recirculation mode for safety injection. Since the maximum temperature of the recirculation fluid supports a flash fraction less than 10%, per Regulatory Guide 1.183, ten percent (10%) of the halogens associated with this leakage are assumed to be airborne and are exhausted (without mixing and without holdup) to the environment. The iodine release from the core is 95% particulate (CsI), 4.85% elemental and 0.15% organic; however, after interactions with sump water the environmental release is assumed to be 97% elemental and 3% organic.

The environmental release of ESF system leakage can occur via the 2 pathways listed below.

a. <u>Environmental release of ESF System leakage via the plant vent</u>: The sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the auxiliary building is limited to 120 cc/min. Thus, and in accordance with the guidance provided in Regulatory Guide 1.183, the analysis addresses an ESF leakage of 240 cc/min in the auxiliary building. The areas where these components are located are covered by the PG&E Design Class I ABVS which discharges to the environment out of the plant vent. Only selected portions of the Auxiliary Building ventilation system are processed through the PG&E Design Class I AB ventilation filters. For purposes of estimating the dose consequences, it is assumed that with the exception of the RHR pump rooms (refer to Section 15.5.17.2.2.4), this release pathway bypasses the PG&E Design Class I AB ventilation filters.

b. Environmental release of ESF System leakage via Containment Penetration Area GE and Areas GW & FW: The sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the containment penetration areas is limited to 6 cc/min. Thus, and in accordance with the guidance provided in Regulatory Guide 1.183, the analysis addresses an ESF leakage of 12 cc/min in the containment penetration areas. The ventilation system covering this area is not PG&E Design Class I, thus the release path to the environment is unfiltered and could occur via the Plant Vent or via the closest structural opening in the Containment Penetration Areas GE and Areas GW & FW.

15.5.17.2.2.4 RHR Pump Seal Failure

Failure of an RHR pump seal was assumed to be the worst case single failure to be tolerated without loss of the required functioning of the RHR system, as was required by the following clauses in the addendum to the ANS Standard N18.2 proposed at the time of original license:

> "Fluid systems provided to mitigate the consequences of Condition III and Condition IV events shall be designed to tolerate a single failure in addition to the incident which requires their function, without loss of the function to the unit.
>
> "A single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions when called upon. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against a single failure if neither (a) a single failure of any active component (assuming passive components function properly); nor (b) a single failure of a passive component (assuming active components function properly) results in a loss of the safety function to the nuclear steam electric generating unit.
>
> "An active failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand.

"Examples of active failures include the failure of a valve or a check valve to move to its correct position, or the failure of a pump, fan or diesel generator to start.

"Spurious action of a powered component originating within its actuation system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action.

"A passive failure is a breach of the fluid pressure boundary or blockage of a process flowpath."

The failure of auxiliary building charcoal filters, a second failure, was not assumed, in accordance with the standard.

A review of the equipment in the RHR system loop and the CSS loop indicates that the largest leakage would result from the failure of an RHR pump seal. Evaluation of RHR pump seal leakage rate, assuming only the presence of a seal retention ring around the pump shaft, shows that flows less than 50 gpm would result (refer to Section 3.1 and Chapter 6). Circulation loop piping leaks, valve packing leaks, and flange gasket leaks are much smaller and less severe than an RHR pump seal failure leak. On this basis, a 50 gpm leakrate was assumed for LOCA.

For the DBA LOCA pump seal leakage was assumed to commence 24 hours after the start of the LOCA. This assumption is consistent with the discussion in Sections 3.1.1.1 and 6.3.3.5.3, and with the guidance in SRP 15.6.5, Appendix B (Reference 77). In this context, the limiting recirculation loop long term passive failure is 50 gpm leakage at 24 hours after the start of the LBLOCA.

Evaluation of an RHR pump seal failure shows that the failure could be detected and the pump isolated well within 30 minutes (refer to Chapter 6). Thus a leakage duration of 30 minutes is conservatively assumed for the DBA LOCA.

In summary, the RHR pump seal failure resulting in a filtered release via the plant vent is DCPP's licensing basis with respect to the worst case passive single failure in the RHR system. Therefore, the RHR pump Seal Failure is retained as a release pathway for the AST LOCA dose consequence analysis.

The activity transport model is based on a 50 gpm leak of sump water activity for 30 minutes that occurs 24 hours after the LOCA. The temperature of the recirculation fluid is conservatively assumed to remain at the maximum temperature of 259.9oF. Thus as discussed above in Section 15.5.17.2.2.3 under ESF system leakage, the amount of iodine that becomes airborne is assumed to be 10% of the total iodine activity in the leaked fluid.

The ventilation exhaust from the RHR pump rooms is covered by the PG&E Design Class I Auxiliary Building ventilation system and processed through the PG&E Design

Class I AB ventilation filters. Thus, credit for filtration of the release of a RHR pump seal failure by the Auxiliary Building Ventilation system is taken in determining the dose consequences to the public at the EAB and LPZ, to the operator in the control room, and to personnel in the technical support center.

The efficiency of the auxiliary building charcoal filters is determined using methodology similar to that documented in Section 15.5.9 for the CRVS Mode 4 ventilation filters. The allowable methyl iodide penetration / filter bypass for the auxiliary building charcoal filter is controlled by DCPP Technical Specification 5.5.11; and are 5% and <1%, respectively. Based on the above, an efficiency of 88% is assigned to the charcoal filters in the AB ventilation system prior to environmental release via the plant vent. Similar to the ESF system leakage, the environmental release of iodine is assumed to be 97% elemental and 3% organic.

15.5.17.2.2.5 Refueling Water Storage Tank Back Leakage

The safety injection and containment spray systems function to provide reactor core cooling and mitigate the containment pressure and temperature rise, respectively, in the event of a LOCA. Both systems initially take suction from the RWST. Once the RWST water supply is depleted, both the containment spray and safety injection systems are supplied by the RHR System. The RHR pumps take suction from the containment recirculation sump water. Under LOCA conditions, the recirculation sump water is assumed to be radioactively contaminated by fission products, of which the main contributors to airborne dose are the various isotopes of iodine.

As discussed in NRC Information Notice 91-56, September 1991 during containment sump water recirculation, there is the potential for leakage from the mini-flow recirculation lines connecting the high head and low head safety injection pump discharge piping to the RWST. Since the RWST is vented to the atmosphere, this presents a pathway for iodine release to the atmosphere. The acceptance criteria in the DCPP administrative test procedures ensure that the total as-tested back leakage into the RWST from the containment recirculation sump is less than or equal to 1 gpm.

Dose consequences of RWST back-leakage assumes that leakage starts at the switchover to recirculation following the LOCA and continues for 30 days. Per regulatory guidance, a safety factor of 2 is applied to the leak rate, i.e., a 2-gpm leakage rate is assumed for the full duration of the event, which is two times the allowable leakage of 1 gpm. With the exception of noble gases, all fission products released from the fuel to the containment are instantaneously and homogeneously mixed in the sump water at the time of release. Only iodine and their daughter products are released through RWST back-leakage since the particulates would remain in the sump water.

A significant portion of the iodine associated with sump water back-leakage into the RWST is retained within the RWST fluid due to the equilibrium iodine distribution balance between the RWST gas and liquid phases. The time dependent iodine partition coefficient takes into consideration the temperature and pH of the RWST liquid and sump fluid, the RWST liquid and gas volumes, and the temperature, pH and volume of the incoming leakage. The iodines that evolve into the RWST gas space as a result of the equilibrium iodine distribution balance, and the noble gas daughters of iodines, are released to the environment via the RWST vent, at a vent rate established by the temperature transient in the RWST (which includes the effect of decay heat), the increase in the liquid inventory of the RWST due to the incoming leakage, the gases evolving out of incoming leakage, and the environmental conditions outside the RWST.

The average time-dependent RWST iodine release fractions along with the fractional RWST gas venting rates (may be applied to the noble gas daughters of iodines) to the atmosphere from the Unit 1 and Unit 2 RWSTs due to RWST back-leakage following switchover to the sump water recirculation mode of operation is summarized in Table 15.5-23C. As discussed earlier, the release fractions / rates presented in Table 15.5- 23C reflect a safety factor of 2 on the leak rates, i.e., are developed based on a RWST back-leakage of 2 gpm. The iodine released to the environment is assumed to be 97% elemental and 3% organic.

The equilibrium iodine concentration in the RWST gas space utilized to develop Table 15.5-23C is based on the iodine mass in the sump fluid entering the RWST vapor space as back-leakage or the total iodine mass contained in the RWST liquid, whichever results in higher RWST vapor phase concentrations. The RWST maximum venting rate averaged over an interval is primarily based on RWST back-leakage entering the RWST gas space and thermally equilibrating, and is used in conjunction with the higher RWST gas space iodine concentration to calculate an iodine mass release rate as a function of time. An interval based averaging approach is utilized in preparing Table 15.5-23C to reduce the number of input values to the dose analysis while preserving the boundaries for the time periods used for atmospheric dispersion; the actual iodine release calculated in an interval is normalized to the iodine mass leaking into the RWST during that time interval.

Examination of the average gas space venting rates indicate that after the first day, the noble gases formed by decay of iodine will primarily remain in the RWST during the 30 day period of evaluation and not be released. However, the dose consequence analysis conservatively releases the noble gases formed by decay of iodine, directly to the environment without taking any credit for tank holdup.

15.5.17.2.2.6 Miscellaneous Equipment Drain Tank (MEDT) Leakage

The DCPP Unit 1 and Unit 2 MEDT is a covered rectangular stainless steel lined concrete tank located in the auxiliary building below El 60 foot. The MEDT tank vent is hard-piped to the auxiliary building ventilation ductwork; thus the airborne releases from the MEDT are ultimately discharged to the environment via the plant vent (refer to Section 9.4.2).

Following a LOCA, the MEDT will receive both post-LOCA sump fluids as well as non-radioactive fluids (i.e., ESF system leakage from the accident unit as well as non-radioactive fluids from equipment drains / RWST leakage from the non-accident unit) which are hard-piped to the MEDT. The acceptance criteria in the DCPP administrative test procedures ensure the total as-tested flow hard piped to the MEDT is less than 950 cc/min of ESF system leakage and 484 cc/min of non-radioactive fluid leakage.

Similar to the RWST back-leakage model, dose consequences due to releases from the MEDT assumes that leakage starts at the switchover to recirculation (829 second following the LOCA) and continues for 30 days. Per Regulatory Guide 1.183, a safety factor of 2 is applied to the leak rate, i.e., 1900 cc/min of ESF system leakage and 968 cc/min of non-radioactive fluids into the MEDT is assumed for the full duration of the event, which is two times the allowable leakage. With the exception of noble gases, all fission products released from the fuel to the containment are instantaneously and homogeneously mixed in the sump water at the time of release. Only iodine and their daughter products are released through MEDT leakage since the particulates would remain in the sump water.

The methodology used to determine the post-LOCA iodine and noble gas releases via the MEDT vent and Plant Vent is similar to that used to address RWST back-leakage. Adaptation of the methodology to address overflows/room ventilation releases is straightforward with the room ventilation rate being treated as the tank exhaust rate.

The transport model utilized to determine airborne releases from the MEDT takes into account the fact that the MEDT is a small tank with an auto-transfer capability which is PG&E Design Class II. Consequently, and for purposes of conservatism, it is assumed that a) the LOCA occurs when the MEDT water level is at the normal maximum setpoint to initiate auto transfer, b) the auto-transfer capability is not initiated because it is not a safety function, and c) the MEDT contents will spill over into the Equipment Drain Receiver Tank (EDRT) Room after the tank is full. Thus, for the post-LOCA scenario, the MEDT is conservatively assumed to overflow via its manway into the EDRT Room. The EDRT room drains into the auxiliary building sump, which ultimately overflows into the Unit 1/Unit 2 pipe tunnels. The auxiliary building sump is also a covered rectangular stainless steel lined concrete tank with a vent that is hard-piped to the auxiliary building ventilation system (ABVS) with a PG&E Design Class II auto transfer capability. The auxiliary building sump is located adjacent to the MEDT.

The bounding transient release of iodine along with the gas venting rate to the atmosphere as a result of post-LOCA leakage of radioactive and non-radioactive fluid hard-piped into the MEDT is developed in 2 parts: a) prior to MEDT overflow and b) post MEDT overflow.

a) Prior to MEDT overflow - The iodines evolve into the MEDT gas space as a result of the equilibrium iodine distribution balance between the MEDT gas and liquid phases (either the MEDT liquid inventory or the incoming leakage), and are released to the environment via the plant vent, at a vent rate established by the temperature transient in the MEDT (including the effect of decay heat), the increase in the liquid inventory of the MEDT due to the incoming leakage, and the gases evolving out of the incoming leakage.

b) After MEDT overflow – The equilibrium iodine distribution balance is conservatively assumed to be between the iodine concentrations in the MEDT overflow liquid and the EDRT room (or Unit 1/Unit 2 pipe tunnels) ventilation flow (rather than the average concentration in the EDRT room (or Unit 1/Unit 2 pipe tunnels) free volume). This maximizes the iodine release rate. Thus, the iodines released are a sum total of the following:

   i) the iodines that evolve into the EDRT room air space as a result of the equilibrium iodine distribution balance between the spilled liquid from the MEDT (at the temperature of the MEDT), and the EDRT room ventilation flow, and is released to the environment via the plant vent, at the vent rate established by the EDRT room ventilation system, and

   ii) the iodines that evolve into the Unit 1/Unit 2 Pipe Tunnel air space as a result of the equilibrium iodine distribution balance between the spilled liquid from the MEDT (at the maximum temperature of the Unit 1/Unit 2 Pipe Tunnel), and the U1/U2 Pipe Tunnel ventilation flow, and is released to the environment via the plant vent, at the vent rate established by the U1/U2 Pipe Tunnel ventilation system.

The exhaust fans servicing the EDRT room and pipe tunnel are PG&E Design Class I. There is also a potential that the non-LOCA unit's ABVS will be operating with the flow exhausting to its unit specific plant vent. Thus, it is conservatively assumed that the non-LOCA unit's ABVS is also operating, and together with the accident units' exhaust fans, are providing the motive force to exhaust the airborne releases to the respective unit vents.

The average time-dependent MEDT iodine release fractions, along with the fractional MEDT gas venting rates (which may be applied to the noble gas daughters of iodines prior to MEDT overflow) to the atmosphere following switchover to the sump water recirculation mode of operation, is summarized in Table 15.5-23D. As discussed earlier, the release fractions / rates presented in Table 15.5-23D reflect a safety factor of 2 on the leak rates, i.e., are developed based on an input of 1900 cc/min of ESF system leakage and 968 cc/min of non-radioactive fluids into the MEDT. Through the use of extremely conservative assumptions, the calculated iodine release fractions / gas venting rates presented in Table 15.5-23D when used in combination with the analyzed ESF system leak rate, bound the iodine releases of all combinations of radioactive and non- radioactive leakages less than or equal to the leak rates analyzed. The iodine released to the ventilation system is assumed to be 97% elemental and 3% organic, and is released to the environment via the plant vent. In addition, the dose consequence analysis conservatively releases the noble gases formed by decay of iodine, directly to the environment without taking any credit for tank holdup.

15.5.17.2.3 Offsite Dose Assessment

Due to the delayed post-LOCA fuel release sequence of an AST model, and the rate at which aerosols and elemental iodine are removed from the containment, the maximum 2-hour EAB dose for a PWR LOCA typically occurs between 0.5 hrs to 2.5 hrs.

To establish the “worst case 2-hour release window” for the DCPP EAB dose, the integrated dose versus time for each of the six pathways discussed in Section 15.5.17.2.2 was evaluated. The 0-2 hr EAB Atmospheric Dispersion Factor from Table 2.3-145 was utilized for all cases.

The analysis demonstrated that for DCPP the maximum 2 hour EAB dose will occur, as a result of the RHR pump seal failure, between T=24 hrs to T=26 hrs, and is unrelated to the post-LOCA fuel release sequence associated with AST.

The direct shine dose at the EAB due to a) the airborne activity inside containment, and b) the sump water collected in the RWST due to RWST back-leakage, was also evaluated. Based on the results of the EAB evaluation which determined that the dose contribution due to direct shine was minimal (<0.01 rem), the dose at the LPZ due to direct shine is deemed negligible.

The bounding EAB and LPZ dose following a LOCA at either unit is presented in Table 15.5-23.

15.5.17.2.4 Post LOCA Control Room Operator Exposure

The design basis for control room ventilation, shielding, and administration is to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE.

The control room shielding, described in Section 12.1 is designed to attenuate gamma radiation from post-accident sources to levels consistent with the requirements of GDC 19, 1999 and 10 CFR 50.67.

The control room ventilation system (CRVS) is described in Section 9.4.1. It is designed to limit the concentration of post-accident activity in the control room air to levels consistent with requirements of GDC 19, 1999 and 10 CFR 50.67.

The control room post-accident administration is described in the DCPP Manual. It is to limit post-accident control room personnel exposures to levels consistent with requirements of GDC 19, 1999 and 10 CFR 50.67.

Exposures to control room personnel during post-LOCA occupancy have been estimated for a design basis LOCA to evaluate the adequacy of the control room shielding, the adequacy of the CRVS, and the adequacy of the control room administration in limiting exposures to the specified limits.

Radiation exposures to personnel in the control room could result from the following sources:

(1) Airborne activity, which infiltrates into the control room

(2) Direct gamma radiation from the external cloud and contained sources.

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical LOCA-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

i. An SIS will be generated at t = 6 sec following a LOCA.

ii. The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The control room dampers are fully closed 10 secs later, or at t=44.2 secs (i.e., 6 + 28.2 + 10). The 2 second SIS processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=18 secs (i.e., 6 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors:

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a LOCA at either unit are provided in Table 15.5-23B. The χ/Q values presented in Table 15.5-23B take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to a LOCA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Table 2.3-147 and Table 2.3-148 for Unit 1 and Unit 2, respectively.

Direct Shine from External and Contained Sources

The direct shine dose to an operator in the control room due to contained or external sources resulting from a postulated LOCA is calculated using point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rates (MeV/sec) and integrated gamma energy release (MeV-hr/sec) in the various external sources are developed using computer program PERC2.

The LOCA sources that could potentially impact the control room operator dose due to direct shine are identified below.

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through one of the main steam line penetrations and the Personnel Hatch facing the control room.
2. Direct shine from the contaminated cloud outside the control room pressure boundary resulting from containment leakage, ESF system leakage, RHR pump seal leakage, RWST back- leakage, MEDT leakage - shine occurs through the control room walls, via wall penetrations such as control room doors to the outside, and from the airborne activity in cable spreading room below via control room floor penetrations.
3. Dose due to scattered gamma radiation through wall penetrations from the CRVS filters located in the adjacent mechanical equipment room.
4. Direct shine from the sump fluid that is postulated to collect in the RWST.

Cloud shine through control room doorways was found to be the most significant of all the identified contained or external post-LOCA radiation sources listed above, followed by the dose contribution through the control room floor penetrations. Note that other radiation sources were identified and deemed insignificant due to the presence of significant shielding between the operator and the radiation sources. Examples of these dose contributors include most of the large and small electrical and pipe penetrations in the Containment outer wall that faces the control room, and the ESF system piping and components located in the Auxiliary Building.

The direct shine dose estimate in the control room takes into consideration the function of Room 506 (which serves as a control room foyer adjacent to the Shift Supervisor's office), where occupancy is deemed to be minimal; i.e., conservatively estimated at less than 5% of the total time spent daily in the control room. The above "occupancy adjustment" is utilized to determine the maximum 30-day integrated dose in control room (i.e., the total direct shine dose in the control room includes the 30-day dose in Room 506 adjusted by the referenced occupancy factor).

Control Room Operator Dose during Access

Diablo Canyon assumes that the dose received by the operator during routine access to the control room for the 30 day period following the LOCA is minimal. Thus, as long as some reasonable margin exists between the regulatory limit and the estimated dose to the operator during control room occupancy, the additional dose due to egress-ingress can be accommodated.

This approach is consistent with the approach used by other licensees, and is reasonable since a) transit to and from the control room is only expected after the first 24 hours following the accident by which time the airborne levels inside containment has reduced significantly due to the use of active fission product removal mechanisms such as containment sprays, and radioactive decay, and b) the operator is protected from radioactive ESF fluids by the shielding provided by the buildings that house such equipment. In addition, it is expected that during a postulated event, access to the control room will be controlled by Health Physics and the Emergency Plan based on real time data, with the purpose of minimizing personnel dose.

It is also noted that the dose received by the operator during transit outside the control room is not a measure of the “habitability” of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post- LOCA access to the control room is addressed separately from the control room occupancy dose which is used for the demonstration of control room habitability.

in accordance with DCPP original licensing basis radiation exposures to personnel during egress and ingress (i.e., during routine access to the control room for the duration of the accident) could result from the following sources:

(1) Airborne activity in the containment leakage plume

(2) Direct gamma radiation from fission products in the containment structure

Post-accident egress-ingress exposures were based on 27 outbound excursions, from the control room to the site boundary, and 26 inbound excursions, from the site boundary to the control room. It was estimated that each excursion would take 5 minutes, and no credit was taken for breathing apparatus or special whole body shielding.

Egress-ingress thyroid and whole body exposures from airborne activity are functions of containment activity, containment leakage, atmospheric dispersion, and excursion time. The EMERALD computer code was used to calculate the airborne activity concentrations, and then conventional exposure equations from Regulatory Guide 1.4 Revision 1, were used to calculate gamma, beta, and thyroid exposures (Reference 6). The exposure from betas was calculated on the basis of an infinite uniform cloud, and exposure from gammas was calculated on the basis of a semi-infinite cloud.

Because of the containment shielding and short excursion time, egress-ingress containment shine exposures were estimate to be small. Egress-ingress containment shine exposures were calculated using ISOSHLD-II. The shine model assumes a cylindrical radiation source having the same radius and height as the containment structure with a 3.5-foot-thick concrete shield surrounding it. The receptor point is assumed to be a distance of 10 meters from the outer surface of the containment wall.

The estimated egress-ingress exposures developed in support of DCPP original licensing basis are listed in Table 15.5-33 and summarized below.

a. The dose to control room personnel during egress-ingress from airborne fission products in the containment leakage plume: 0.0066 rem gamma, 0.0243 rem beta, and 4.72 rem thyroid.
b. The dose to control room personnel during egress-ingress as a result of direct radiation shine from the fission products in the containment structure is 0.022 rem.

Subsequent to the original licensing basis assessment described above, DCPP has identified additional post-LOCA fission product release pathways, as discussed in Section 15.5.17.2.1. The postulated effect of these additional radioactivity release paths, as well as the implementation of AST, on the estimated dose to control room personnel during routine egress-ingress takes into consideration the following:

a. The transport models used to develop the dose to the control room operator during occupancy address a control room occupancy factor of 1.0 till t=24 hours after the accident. This implies that during the first 24 hours the control room operator stays in the control room. This is also reflected in the DCPP original licensing basis which addresses one more outbound trip than the inbound trips.

b. Routine egress-ingress to the control room during the 30 day period following a LOCA falls into the mission dose category as discussed in NUREG 0737, November 1980, Item II.B.2.

c. In accordance with NUREG 0737, November 1980, Item II.B.2 leakage of systems outside containment need not be considered as potential sources.

Based on the above considerations, the dose consequences of the additional activity release paths addressed in Section 15.5.17.2.1 (and listed below), in addition to Regulatory Guide 1.183 is addressed as follows:

i. Containment Pressure /Vacuum relief release - this release occurs at accident initiation (before t=24hr), so there is no dose contribution to the control room operator during routine egress-ingress during the 30 day period following the accident.

ii. Containment leakage:

   a. The airborne activity in the containment after t=24 hours with an AST source term is primarily 100% of the core noble gases and 0.06% of the core iodines that were released to containment.

      Note: The iodine source term at t=24 hrs is essentially the organic iodines released to the containment which are not affected by sprays, and which per Regulatory Guide 1.183, represent 0.06% of the core iodines (i.e., 0.15% of the 40% core iodines released to containment atmosphere at accident initiation). Also, the essentially particulate nature of the radioactivity release associated with an AST source term, and the effectiveness of particulate removal by sprays / settling makes the dose contribution from the particulate source minimal after t=24 hours.

b. The corresponding airborne activity in the containment after t=24 hours for a TID-14844 source term is 100% of the core noble gases and 1% of the core iodines.

   Note: Per Regulatory Guide 1.4, Revision 1, the organic iodines released to the containment is 4% of the 25% iodines released to containment atmosphere at accident initiation.

c. Based on the above it is concluded that after t=24hrs:

   - o Dose consequences due to containment leakage based on a TID- 14844 based scenario will bound the dose consequences based on an AST scenario.

   - o Thyroid dose is primarily due to iodines, the associated dose to the operator will vary proportionately to the amount of iodine airborne in containment. Thus, the thyroid dose to the operator during egress-ingress for an AST scenario may be estimated by adjusting the TID-14844 based dose by the ratio of the iodine estimated to be airborne in containment for each of the scenarios. As noted earlier, the current licensing basis thyroid dose to the operator during egress-ingress is 4.72 rem. The corresponding thyroid dose based on an AST scenario is estimated to be 4.72 x 0.06 = 0.28 rem thyroid.

iii. The RHR Pump Seal Failure, ESF System Leakage, RWST back leakage and MEDT leakage – All of these releases are based on leakage of systems outside containment. In accordance with NUREG 0737, November 1980, Item II.B.2, the dose contribution due to these sources need not be considered for access calculations.

To address the TEDE dose acceptance criteria applicable to use of AST, the original licensing basis egress-ingress exposures have been updated as noted below in accordance with 10 CFR 20.1003.

Title 10 CFR 20.1003 defines TEDE as the sum of the deep dose equivalent for external exposures (i.e., external whole body exposure) and the committed effective dose equivalent for internal exposures (i.e., sum of the product of the weighting factor applicable to each organ irradiated and the dose to that organ). Per 10 CFR 20.1003, the weighting factor for the whole body is 1.0 and for the thyroid is 0.03. While the weighting factor for beta radiation is undefined, the contribution of the beta dose to the total effective dose equivalent is expected to be insignificant. Therefore,

a. Radiation from airborne fission products in the containment leakage plume to the control room personnel during egress-ingress is approximately 0.0066 rem + 0.28 x 0.03 rem, i.e., 0.015 rem TEDE.

b. Direct radiation from the fission products in the containment structure to control room personnel during egress-ingress is 0.022 rem TEDE.

Thus, the total dose to the control room operator during access is estimated to be 0.037 rem TEDE. This value is 1% of the estimated operator dose due to control room occupancy following a LOCA (Refer to Table 15.5-23) and is therefore considered to be minimal.

15.5.17.2.5 Post-LOCA Technical Support Center Operator Exposure

In accordance with NUREG-0737, Supplement 1, January 1983, Section 8.2.1(f) the TSC design has been evaluated for the LOCA.

Computer code PERC2 is used to calculate the dose to TSC personnel due to airborne radioactivity releases following a LOCA. The direct shine dose to an operator in the TSC due to contained or external sources resulting from a postulated LOCA is calculated using point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rates (MeV/sec) and integrated gamma energy release (MeV-hr/sec) in the various external sources are developed with computer program PERC2.

The TSC serves both units and is located at El 104' on the south-west side of the Unit 2 turbine building and is shared between Unit 1 and Unit 2.

The nominal TSC air intake flowrate during normal operations is 500 cfm. The air inflow is filtered through a HEPA filter and drawn into the TSC envelope. The TSC normal intake is isolated and the TSC ventilation placed into filtered / pressurized (CRVS Mode 4) operation by manual operator action within 2 hours of the LOCA.

The post-accident pressurization flow to the TSC is provided via the CRVS Mode 4 pressurization intakes (i.e., 1 per unit, each located on either side of the Turbine Building). As noted in Section 15.5.9, the control room pressurization air intakes have dual ventilation outside air intake design. The nominal air intake flowrate during the TSC pressurization mode is 500 cfm.

As discussed in Section 15.5.9, CRVS Mode 4 operation utilizes redundant PG&E Design Class I radiation monitors located at each pressurization air intake and has the provisions of acceptable control logic to automatically select the least contaminated inlet at the beginning of the accident, and manually select the least contaminated inlet during the course of the accident. Thus, during Mode 4 operation the TSC dose consequence analysis can utilize the χ/Q values for the more favorable pressurization air intake reduced by a factor of 4 to credit the "dual intake" design (refer to Section 2.3.5.2.2 for additional details).

The allowable methyl iodide penetration and filter bypass for the TSC Mode 4 Charcoal Filter is <2.5% and <1%, respectively. Thus, in accordance with Generic Letter 99-02, June 1999, the TSC charcoal filter efficiency for elemental and organic iodine used in the TSC dose analysis is 93%. The acceptance criteria for the TSC normal operation and Mode 4 HEPA filters is "penetration plus system bypass" < 1.0%. Thus, using methodology similar to the charcoal filters, the HEPA filter efficiency for particulates used in the TSC dose analysis is 98%.

During TSC Mode 4 operation, the TSC air is also recirculated through the same filtration unit as the pressurization flow (refer to Section 9.4.11). The air flow allowable through the pressurization charcoal / HEPA filter and minimum filtered recirculation flow for the TSC is provided in Table 15.5-82.

Unfiltered inleakage into the TSC during normal operation and Mode 4 is assumed to be 60 cfm (includes 10 cfm for egress-ingress based on the guidance provided in NUREG 0800, SRP 6.4).

For purposes of estimating the post-LOCA dose consequences, the TSC is modeled as a single region. When in TSC Mode 4, the Mode 1 intakes are isolated and outside air is a) drawn into the TSC through the filtered emergency intakes; b) enters the TSC as infiltration, and c) enters the TSC during operator egress-ingress.

The dose assessment model utilizes nominal values for the ventilation intake flowrates since the intake pathways (normal as well as accident) are filtered; thus, the controlling dose contributor is the unfiltered inleakage. The effect of intake flow uncertainty on the TSC dose is expected to be insignificant.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / TSC receptors applicable to a LOCA at either unit are provided in Table 15.5-23E. The χ/Q values presented take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to a LOCA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2.

The direct shine dose into the TSC due to the external cloud and contained sources is calculated in a manner similar to that described for the control room in Section 15.5.17.2.4. The LOCA sources that could potentially impact the TSC operator dose due to direct shine are identified below.

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through the Personnel Hatch facing the TSC.
2. Direct shine from the contaminated cloud outside the TSC pressure boundary

resulting from containment leakage, ESF system leakage, RWST back-leakage, RHR pump seal leakage, MEDT leakage - shine occurs through the TSC walls, and via wall penetrations such as TSC doors to the outside.

3. Dose due to scattered gamma radiation through wall penetrations from the TSC filters located in the adjacent mechanical equipment room and scatter past labyrinths provided for selected doors.

Note that other radiation sources were identified and deemed insignificant due to the presence of significant shielding between the operator in the TSC and the radiation sources.

Table 15.5-82 lists key assumptions / parameters associated with DCPP TSC design. The bounding TSC operator dose following a LOCA at either unit is presented in Table 15.5-23.

15.5.17.2.6 Summary

In the preceding sections, the potential exposures from a major primary system pipe rupture have been calculated for various possible mechanisms:

(1) Containment Pressure / Vacuum Relief

(2) Containment leakage

(3) ESF System Leakage

(4) RHR pump seal Failure

(5) RWST Back-Leakage

(6) MEDT Leakage

(7) Shine from Contained and External Sources (e.g., Contained Containment shine, RWST Shine, external clouds due to the various leakage sources, etc.)

The analyses have been carried out using the models and assumptions specified in Regulatory Guide 1.183 and other regulatory guidance identified above. In all analyses, the resulting potential exposures to plant personnel, to individual members of the public, and to the general population have been found to be lower than the applicable guidelines and limits specified in 10 CFR Part 50.67and Regulatory Guide 1.183.

15.5.17.3 Conclusions

Based on the results discussed, the occurrence of a major pipe rupture in the primary system of a DCPP unit would not constitute an undue risk to the health and safety of the public. In addition, the ESF provided for the mitigation of the consequences of a LBLOCA are adequately designed.

Additionally, the analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.25 Sv (25 rem) TEDE as shown in Table 15.5-23.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.25 Sv (25 rem) TEDE as shown in Table 15.5-23.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-23. The dose received by the operator during transit outside the control room is not a measure of the "habitability" of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, and in accordance with DCPP current licensing basis, the dose contribution to the operator during routine access to control room for the duration of the accident (0.037 rem TEDE), is not included with the control room occupancy dose for the demonstration of control room habitability.

(4) The radiation dose to an individual in the TSC for the duration of the accident is within 1.5 Sv (5 rem) TEDE as shown in Table 15.5-23.

15.5.18 RADIOLOGICAL CONSEQUENCES OF A MAJOR STEAM PIPE RUPTURE

15.5.18.1 Acceptance Criteria

The radiological consequences of a MSLB shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below.

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

Control Room Dose Criteria

(3) Adequate radiation protection is provide to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Technical Support Center Dose Criteria

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.18.2 Identification of Causes and Accident Description

15.5.18.2.1 Activity Release Pathways

As reported in Section 15.4.2, a major steam line rupture is not expected to cause cladding damage, and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the steam escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected SGs is necessitated by unavailability of condenser capacity, additional activity will be released.

This event consists of a double-ended break of one main steam line. The analysis focuses on a MSLB outside the containment since a MSLB inside containment will clearly result in a lesser dose to a control room operator or to the offsite public due to hold-up of activity in the containment.

Following a MSLB, the affected SG rapidly depressurizes and releases the initial contents to the environment via the break. Based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs / 10% ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling. The activity in the RCS leaks into the faulted and intact steam generators via SG tube leakage and is released to the environment from the break point, and from the MSSVs 10% ADVs, respectively.

Regulatory guidance provided for the MSLB in pertinent sections of Regulatory Guide 1.183 including Appendix E is used to develop the dose consequence model. Table 15.5-34A lists the key assumptions / parameters utilized to develop the radiological consequences following a MSLB.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a MSLB

15.5.18.2.2 Activity Release Transport Model

In accordance with Regulatory Guide 1.183, Appendix E, Item 2, since no melt or clad breach is postulated for the DCPP MSLB event, the activity released is based on the maximum coolant activity allowed by the plant technical specifications. The plant technical specifications focus on the noble gases and iodines. In addition, and per Regulatory Guide 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131. Technical Specifications limit primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the MSLB dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

Following a MSLB, the primary and secondary reactor coolant activity is released to the environment via two pathways.

<u>Faulted Steam Generator</u>

The release from the faulted SG occurs via the postulated break point of the main-steam line. The faulted SG is estimated to dry-out almost instantaneously following the MSLB (within 10 seconds), releasing all of the iodine in the secondary coolant (at Technical Specification concentrations) that was initially contained in the steam generator. The EAB and LPZ dose to the public is calculated using an instantaneous release of the iodine inventory (Ci) in the SG liquid in the faulted SG. The secondary steam activity initially contained in the faulted steam generator is also released; however, the associated dose contribution is not included in this analysis since it is considered insignificant.

To maximize the control room and offsite doses following a MSLB, the maximum allowable primary to secondary SG tube leakage for all SGs (0.75 gpm or 1080 gpd at Standard Temperature and Pressure (STP) conditions), is conservatively assumed to occur in the faulted SG. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination. The primary to secondary SG tube leakage is assumed to go on until the RCS reaches 212°F, which based on minimum heat transfer rates, is conservatively estimated to occur 30 hours after the event.

Intact Steam Generators

The initial iodine activities in the secondary coolant at Technical Specification levels are released to the environment in proportion to the steaming rate and the inverse of the partition coefficient (limited to 100) defined in Regulatory Guide 1.183. The noble gases are released freely to the environment without retention in the steam generators. However, there is no primary to secondary leakage into the intact SG as all primary to secondary leakage (1080 gpd or 0.75 gpm) is assumed to be occurring in the faulted SG.

The iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The condenser is assumed unavailable due to the loss of offsite power. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

15.5.18.2.3 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose.

a. The Source/Release for the Pre-incident Spike Case is at its maximum levels between 0 and 2 hours.
b. The Source/Release for the Accident-Initiated Spike Case is at its maximum levels towards the end of the spiking period.

Regardless of the starting point of the “Worst 2-hr Window,” the 0-2 hrs χ/Q is utilized.

The bounding EAB and LPZ dose following a MSLB at either unit for both scenarios are presented in Table 15.5-34.

15.5.18.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical MSLB-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

An SIS will be generated at t = 0.6 sec following a MSLB.

i. An SIS will be generated at t = 0.6 sec following a MSLB.

ii. The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The control room dampers are fully closed within 10 seconds at t=38.8 secs (i.e., 0.6 + 28.2 + 10). The 2 second SIS processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=12.6 secs (i.e., 0.6 + 2 secs signal processing time + 10 sec damper closure time).

Transport of Radioactivity from the Break Location

Since the normal operation (CRVS Mode 1) control room intake of the faulted unit is in such close proximity to the break point, an atmospheric dispersion factor (χ/Q) cannot be accurately determined. Thus, atmospheric dispersion is not credited when determining the control room operator dose from the secondary coolant discharge or the primary to secondary SG tube leakage released from the faulted SG via the break point.

Secondary Coolant Discharge: The radioactivity release due to the almost immediate dry-out of the faulted SG following a MSLB is based on a) the radioactivity concentration of the iodine in a finite cloud created by the secondary coolant liquid flash at the break point; b) conservation of total iodine activity in the SG liquid. The activity concentration at the release point is conservatively based on saturated steam at a density of 5.98E-04 gm/cm^3, (i.e., at 1 atmosphere and 212°F). The activity concentration entering the control room is assumed to be the same as the concentration at the break point until the control room normal ventilation is isolated and the CRVS re-aligned to Mode 4 Pressurization.

Primary to Secondary SG Tube Leakage: Due to the close proximity of the normal operation control room intake of the faulted unit and MSL break release point and consequent unavailability of viable atmospheric dispersion factors, the primary to secondary SG tube leakage into the faulted SG is conservatively assumed to be piped directly into the control room. This model is reasonable since the relatively small plume of steam created by the ~0.485 gallon *{i.e. (0.75 gallon/min)(38.8 s) / 60 s/min}* of reactor coolant released due to SG tube leakage via the MSL break point could easily be swept into the control room due to the close proximity of the control room normal intake to the break point.

<u>Control Room Atmospheric Dispersion Factors</u>

As noted in Section 2.3.5.2.2, because of the proximity of the MSSVs/10% ADVs to the control room normal intake of the affected unit, and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the affected unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's control room normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a MSLB at either unit are provided in Table 15.5-34B. The χ/Q values presented in Table 15.5-34B take into consideration the various release points-receptors applicable to the MSLB to identify the bounding χ/Q values applicable to a MSLB at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding control room dose following a MSLB at either unit is presented in Table 15.5-34.

15.5.18.3 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-34.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.25 Sv (25 rem)

TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-34.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-34.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the MSLB is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

15.5.19 RADIOLOGICAL CONSEQUENCES OF A MAJOR RUPTURE OF A MAIN FEEDWATER PIPE

15.5.19.1 Acceptance Criteria

The radiological consequences of a major rupture of a main feedwater pipe (referred to herein as a feedwater line break (FWLB)) shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183 July 2000 as outlined below:

EAB and LPZ Dose Criteria

(1) An individual located at any point on the boundary of the exclusion area for the any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE, for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

(2) An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

Control Room Dose Criteria

(3) Adequate radiation protection is provide to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

15.5.19.2 Identification of Causes and Accident Description

As reported in Section 15.4.2, a major feedwater line rupture is not expected to cause cladding damage, and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the feedwater escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected SGs is necessitated by unavailability of condenser capacity, additional activity will be released.

Per Standard Review Plan 15.2.8, Section III, Item 6 (Reference 86), the evaluation of the radiological consequences of a design basis FWLB may be based on a qualitative comparison to the results of the design basis MSLB.

The dose consequences following a FWLB will be bounded by a MSLB since the airborne environmental release via the break point is expected to be less than the MSLB.

As demonstrated in Table 15.5-34, the dose consequences at the EAB and LPZ and in the Control Room following a MSLB is within the acceptance criteria applicable to the FWLB.

15.5.19.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ and in the Control Room following a feedwater line break will remain within the acceptance criteria listed in Section 15.5.19.1.

15.5.20 RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

15.5.20.1 Acceptance Criteria

The radiological consequences of a SGTR shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below.

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the

10 CFR 50.67 limit for the accident initiated iodine spike case.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Technical Support Center Dose Criteria

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.20.2 Identification of Causes and Accident Description

15.5.20.2.1 Activity Release Pathways

This event is caused by the instantaneous rupture of a SG tube with a resultant release of primary coolant into the lower pressure secondary system. No melt or clad breach is postulated for the SGTR event. The calculation assumes a stuck-open 10% ADV of the ruptured steam generator for 30 minutes. Based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs / 10% ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling. A portion of the primary coolant break flow in the ruptured SG flashes and is released a) to the condenser before reactor trip and b) directly to the environment after reactor trip, via the MSSVs and 10% ADVs. The remaining break flow mixes with the secondary side liquid, and is released to the environment via steam releases through MSSVs and 10% ADVs. The activity in the RCS also leaks into the intact steam generators via SG tube leakage and is released to the environment from the MSSVs / 10% ADVs.

Regulatory guidance provided for the SGTR in pertinent sections of Regulatory Guide 1.183 including Appendix F is used to develop the dose consequence model. Table 15.5-64A lists the key assumptions / parameters utilized to develop the radiological consequences following a SGTR. Table 15.5-64C provides the time dependent steam flow from the Ruptured and Intact SGs and the flashed and unflashed break flow in the Ruptured SG.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a SGTR.

15.5.20.2.2 Activity Release Transport Model

No melt or clad breach is postulated for the SGTR. Thus, and in accordance with Regulatory Guide 1.183, Appendix F, item 2, the activity released is based on the maximum coolant activity allowed by the plant technical specifications. The plant technical specifications focus on the noble gases and iodines. In addition, and per Regulatory Guide 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the SGTR dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd). To maximize the dose consequences, the analysis conservatively assumes that all of the 0.75 gpm SG tube leakage occurs in the intact SGs.

Following a SGTR, the primary and secondary reactor coolant activity is released to the environment via two pathways.

Ruptured Steam Generator

A SGTR will result in a large amount of primary coolant being released to the ruptured steam generator via the break location with a significant portion of it flashed to the steam space.

In accordance with the guidance provided in Regulatory Guide 1.183, the noble gases in the entire break flow and the iodine in the flashed portion of the break flow are assumed to be immediately available for release from the steam generator. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and the inverse of the allowable partition coefficient of 100. The iodine releases from the SGs are assumed to be 97% elemental and 3% organic.

Before the reactor trip the radioactivity in the steam is released to the environment from the air ejector which discharges into the plant vent. All noble gases and organic iodines in the steam are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returns to both the intact steam generators and the ruptured steam generator and will be available for future steaming releases.

After the reactor trip, the radioactivity in the steam is released to the environment from the MSSVs/10% ADVs, due to the assumption of LOOP. To isolate the ruptured steam loop, the auxiliary feed water to the ruptured SG is secured. The calculation assumes the 10% ADV of the ruptured SG fails open for 30 minutes. The fail-open 10% ADV is isolated at t = 2653 seconds at which time the ruptured steam loop is isolated. The break flow continues until the primary system is in equilibrium with the secondary side of the ruptured SG. The iodines in the flashed break flow and the noble gases in the entire break flow is bottled up in the steam space of the ruptured SG and released to the environment during the manual depressurization of the ruptured SG after t = 2 hours.

Intact Steam Generators

The radioactivity released from the intact steam generators includes two components (a) portion of the break flow activity that is transferred to the intact steam generators via the condenser before reactor trip, and (b) due to SG tube leakage.

Approximately 75% (3 intact SGs vs 1 ruptured SG) of the flashed break flow activity that is transported and retained in the condenser before reactor trip will be transferred to the intact steam generators and released to the environment during the cool-down phase.

The total primary-to-secondary tube leak rate in the 3 intact SGs is conservatively assumed to be 0.75 gpm. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant. Thus, all leaked primary coolant iodine activities are assumed to mix uniformly with the steam generator liquid and are released in proportion to the steaming rate and the inverse of the partition coefficient. Before the reactor trip, the activity in the main steam is released from the plant vent via the air ejector/ condenser. After the reactor trip, the steam is released from the MSSVs/10% ADVs. The reactor coolant noble gases that enter the intact steam generator are released directly to the environment without holdup. The iodine releases from the SGs are assumed to be 97% elemental and 3% organic. The intact SG steam release continues until shutdown cooling (SDC) is initiated at t = 10.73 hours.

Initial Secondary Coolant Activity Release

The initial iodine activities in the secondary coolant are released to the environment in proportion to the steaming rate and the inverse of the partition coefficient from the ruptured and intact SGs. Twenty five percent of the initial secondary coolant iodine inventory is in the ruptured SG and 75% of the initial secondary coolant iodine inventory is in the 3 intact SGs.

15.5.20.2.3 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose.

For the SGTR, the EAB dose is controlled by the release of the flashed break flow in the ruptured SG which stops at 3402 seconds. The break flow stops at 5872 seconds and the ruptured SG is manually depressurized 2 hours after the accident. Therefore the maximum EAB dose occurs during the 0-2hr period for both the pre-accident and accident initiated iodine spike cases.

Regardless of the starting point of the "Worst 2-hr Window," the 0-2 hrs χ/Q is utilized.

The bounding EAB and LPZ dose following a SGTR at either unit for both scenarios are presented in Table 15.5-64.

15.5.20.2.3 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical SGTR-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

i. An SIS will be generated at t = 219 sec following a SGTR.

ii. The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The control room dampers are fully closed 10 secs later, or at t=257.2 secs (i.e., 219 + 28.2 + 10). The 2 second SIS processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=231 secs (i.e., 219 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors

As noted in Section 2.3.5.2.2, because of the proximity of the MSSVs/10% ADVs to the control room normal intake of the affected unit, and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the affected unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's control room normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a SGTR at either unit are provided in Table 15.5-64B. The χ/Q values presented in Table 15.5-64B take into consideration the various release points-receptors applicable to the SGTR to identify the bounding χ/Q values applicable to a SGTR at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding control room dose following a SGTR at either unit is presented in Table 15.5-64.

15.5.20.3 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-64.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-64.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-64.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the SGTR is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

15.5.21 RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

15.5.21.1 Acceptance Criteria

The radiological consequences of a LRA shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below:

EAB and LPZ Dose Criteria

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Technical Support Center Dose Criteria

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.21.2 Identification of Causes and Accident Description

15.5.21.2.1 Activity Release Pathways

This event is caused by an instantaneous seizure of a primary reactor coolant pump (RCP) rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage is predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system via the 10% ADVs and MSSVs. Following reactor trip, and based on an assumption of a LOOP coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling. DCPP has established that the LOL event generates the maximum primary to secondary heat transfer and the LRA assumes these same conservatively bounding secondary steam releases.

Regulatory guidance provided for the LRA in pertinent sections of Regulatory Guide 1.183 including Appendix G is used to develop the dose consequence model.

The LRA is postulated to result in 10% fuel failure resulting in the release of the associated gap activity. As discussed in Section 15.5.3.1.3, the core gap activity is assumed to be comprised of 8% of the core I-131 inventory, 23% of the core I-132 inventory, 35% of the core Kr-85 inventory, 4% of the remaining core noble gas inventory, 5% of the remaining core halogen inventory, and 46% of the core alkali metal (Cesium and Rubidium) inventory. Table 15.5-42A lists the key assumptions / parameters utilized to develop the radiological consequences following a LRA.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LRA.

15.5.21.2.2 Activity Release Transport Model

In accordance with Regulatory Guide 1.183, the activity released from the fuel is assumed to be released instantaneously and mixed homogenously through the primary coolant mass and transmitted to the secondary side via primary to secondary SG tube leakage. A radial peaking factor of 1.65 is applied to the activity release from the fuel gap. The activity associated with the release of the primary to secondary leakage of normal operation RCS, (at Technical Specification levels) via the MSSVs/10% ADVs are insignificant compared to the failed fuel release and are therefore not included in this assessment.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the LRA dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The chemical form of the iodines in the gap are assumed to be 95% particulate (CsI), 4.85% elemental and 0.15% organic. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, the gap iodines are assumed to have a partition coefficient of 100 in the SG. The iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are released freely to the environment without retention in the SG whereas theparticulates are assumed to be carried over in accordance with the design basis SG moisture carryover fraction.

The condenser is assumed unavailable due to the loss of offsite power. Consequently, the radioactivity release resulting from a LRA is discharged to the environment from all steam generators via the MSSVs and the 10% ADVs. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

15.5.21.2.3 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the LRA, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a LRA at either unit is presented in Table 15.5-42.

15.5.21.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical LRA-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CRVS Mode 4 (if applicable)</u>:

The LRA does not initiate any signal which could automatically start the control room emergency ventilation. Thus, the dose consequence analysis for the LRA assumes that the control room remains in normal operation mode.

<u>Control Room Atmospheric Dispersion Factors</u>

As noted in Section 2.3.5.2.2, because of the proximity of the MSSV/10% ADVs to the control room normal intake of the affected unit and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the faulted unit (closest to the release point) will be insignificant. Therefore, only the unaffected unit’s control room normal intake is assumed to be contaminated by a release from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an LRA at either unit are provided in Table 15.5-42B. The χ/Q values presented in Table 15.5-42B take into consideration the various release points-receptors applicable to the LRA to identify the bounding χ/Q values applicable to a LRA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding control room dose following a LRA at either unit is presented in Table 15.5-42.

15.5.21.3 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.025 Sv (2.5 rem) TEDE as shown in Table 15.5-42.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.025 Sv (2.5 rem) TEDE as shown in Table 15.5- 42.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-42.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the LRA is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

15.5.22 RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

The procedures used in handling fuel in the containment and fuel handling area are described in detail in Section 15.4.5. In addition, design and procedural measures provided to prevent FHAs are also described in that section, along with a discussion of past experience in fuel handling operations. The basic events that could be involved in a FHA are discussed in that section, and the following discussion evaluates the potential radiological consequences of such an accident.

The assumption of a LOOP related to a postulated design basis accident which leads to a reactor trip does not directly correlate to an FHA. Specifically, a FHA does not directly cause a reactor trip and a subsequent LOOP due to grid instability; nor can a LOOP be the initiator of a FHA. Thus, the FHA dose consequence analyses are evaluated without the assumption of a LOOP.

15.5.22.1 Acceptance Criteria

The radiological consequences of a FHA in the Fuel Handling Building (FHB) or in the Containment shall not exceed the dose limits of 10 CFR 50.67, as modified by Regulatory Guide 1.183, July 2000 and outlined below:

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.063 Sv (6.3 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.063 Sv (6.3 rem) TEDE.

<u>Control Room Dose Criteria (10 CFR 50.67)</u>

(3) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

<u>Technical Support Center Dose Criteria</u>

(4) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.22.2 Identification of Causes and Accident Description

15.5.22.2.1 Activity Release Pathways

This event postulates that a spent fuel assembly is dropped during refueling in the Spent Fuel Pool (SFP) located in the FHB, or in the reactor cavity located in the Containment. All of the fuel rods (264 rods) in the dropped fuel assembly are assumed to be damaged; thus, all of the activity in the fuel gap of the dropped assembly is assumed to be instantaneously released into the SFP or into the reactor cavity. As documented in the NRC SER for Amendments 8 and 6 to DCPP Facility Operating License Nos. DPR 80 and DPR-82, respectively (Reference 87), the assumption that all fuel rods in one assembly rupture is conservative because the kinetic energy available for causing damage to a fuel assembly dropped through water is fixed by the drop distance. The kinetic energy associated with the maximum drop height for a fuel handling accident is not considered sufficient to rupture the equivalent number of fuel rods of one assembly in both the dropped assembly and the impacted assembly.

During fuel handling operations, containment closure is not required. Generally, the containment ventilation purge system is operational and exhausts air from the containment through two 48-inch containment isolation valves. These two valves are connected in series. This flow of air from the containment is discharged to the environment via the plant vent.

This exhaust stream is monitored for activity by monitors in the plant vent. In the event of a postulated fuel handling accident, the plant vent monitors will alarm and result in the automatic closure of containment ventilation isolation valves. This activity release may result in offsite radiological exposures.

In addition to radiation monitor indications, a fuel handling accident would immediately be known to refueling personnel at the scene of the accident. These personnel would initiate containment closure actions and are required by an Equipment Control Guideline to be in constant communication with control room personnel. The plant intercom system is described in Section 9.5.2.

Containment penetrations are allowed to be open during fuel handling operations. The most prominent of these penetrations are the equipment hatch and the personnel airlock. Closure of these penetrations is achieved by manual means as discussed in Section 15.4.5. The closure of these penetrations is not credited in the design-basis fuel handling accident inside containment.

Following manual containment closure after the fuel handling accident, activity can be removed from the containment atmosphere by the redundant PG&E Design Class II Iodine Removal System (two trains at 12,000 cfm per train), which consists of HEPA/charcoal filters. This system is described in Section 9.4.5. There are no Technical Specification requirements for this filtration system.

The containment can also be purged to the atmosphere at a controlled rate of up to 300 cfm per train through the HEPA/charcoal filters of the hydrogen purge system. This system is described in Section 6.2.5.

In the very unlikely event of a serious FHA and in combination with the conservative assumptions discussed above, containment building or fuel handling area activity concentrations may be quite high. High activity concentrations necessitate the evacuation of fuel handling areas in order to limit exposures to fuel handling personnel. Upon indication of a serious FHA, the fuel handling area will be evacuated until the extent of the fuel damage and activity levels in the area can be determined. Any serious FHA would be both visually and audibly detectable via radiation monitors in the fuel handling areas that locally alarm in the event of high activity levels and would alert personnel to evacuate.

The fuel handling area has the additional safety feature of ventilation air flow that sweeps the surface of the spent fuel pool carrying any activity away from fuel handling personnel. This sweeping of the spent fuel pool is expected to considerably lower activity levels in the fuel handling area in the event of a serious FHA.

Spent fuel cask accidents in the fuel handling area causing fuel damage are precluded due to crane travel limits and design and operating features as described in Sections 9.1.4.3.9 and 9.1.4.2.6. Spent FHAs in the fuel handling area would not jeopardize the health and safety of the public.

The FHA dose assessment follows the guidance provided for the FHA in pertinent sections of Regulatory Guide 1.183 including Appendix B. As discussed in Section 15.5.3.1.3, the core gap activity is assumed to be comprised of 8% of the core I-131 inventory, 23% of the core I-132 inventory, 35% of the core Kr-85 inventory, 4% of the remaining core noble gas inventory, 5% of the remaining core halogen inventory and halogen isotopes, and 46% of the core alkali metal (Cesium and Rubidium) inventory. Table 15.5-47A lists the key assumptions / parameters utilized to develop the radiological consequences following an FHA at either location and at either unit.

DCPP procedures prohibit movement of recently irradiated fuel which is defined as fuel that has occupied part of a critical reactor core within the previous 100 hours. Table 15.5-47C provides the gap activity inventory of the noble gases, iodines, and alkali metals in a single fuel assembly at 72 hrs post reactor shutdown. This inventory conservatively bounds that associated with the 100-hour procedural fuel restriction for movement.

DCPP Technical Specification 3.7.15 requires the SFP water level to be ≥23 feet over the top of irradiated fuel assemblies seated in the storage racks. Technical Specification 3.9.7 requires the refueling cavity water level to be maintained ≥23 feet above the top of the reactor vessel flange. Additional margin is provided through operating procedures.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a FHA.

15.5.22.2.2 Activity Release Transport Model

The fission product inventory in the fuel rod gap of all the rods in the damaged assembly are assumed to be instantaneously released into the spent fuel pool or reactor cavity, both of which have a minimum of 23 ft of water above the damaged fuel assembly. A radial peaking factor of 1.65 is applied to the activity release.

Per Regulatory Guide 1.183, the radioiodine released from the fuel gap is assumed to be 95% particulate (CsI), 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the fuel pool (pH less than 7), the CsI is assumed to immediately disassociate and re-evolve as elemental iodine; thus, changing the chemical form of iodine to 99.85% elemental and 0.15% organic. In addition, and per Regulatory Guide 1.183, an iodine decontamination factor of 200 is assumed for the SFP / reactor cavity. Noble gases and unscrubbed iodines rise to the water surface where they are mixed in the available air space. All of the alkali metals released from the gap are retained in the pool. In accordance with Regulatory Guide 1.183, the chemical form of the iodines above the pool is 57% elemental and 43% organic.

Per Regulatory Guide 1.183, the activity released due to an FHA is assumed to be discharged to the environment in a period of 2 hrs (or less if the ventilation system promotes a faster release rate).

<u>FHA in the FHB</u>

The radioactivity release pathways following an FHA in the FHB are established taking into consideration the following Administration Controls:

During fuel movement in the FHB:

a. The movable wall is put in place and secured
b. No exit door is propped open
c. One FHBVS exhaust fan is operating (The supply fan flow (if operating) has been confirmed by design to have less flow than the exhaust fan)

Operation of the Fuel Handling Building Ventilation system (FHBVS) with a minimum of 1 exhaust fan operating and all significant openings administratively closed will ensure negative pressure in the FHB which will result in post-accident environmental release of radioactivity occurring via the Plant Vent. The activity release due to the FHA in the FHB is assumed to be discharged to the environment as follows:

a. A maximum release rate of 46,000 cfm via the Plant Vent due to operation of the FHBVS with a closed FHB configuration.

b. A maximum conservatively assumed outleakage of 500 cfm occurring from the closest edge of the FHB to the control room normal intake (i.e., 30 cfm outleakage is assumed for egress-ingress; 470 cfm is assumed for outleakage from miscellaneous gaps/openings in the FHB structure).

It has been determined that for the FHA in the FHB, the actual release rate lambda based on the FHBVS exhaust (i.e., 8.7 hr-1) is larger than the release rate applicable to “a 2-hr release” per Regulatory Guide 1.183 (i.e., 3.45 hr-1). Thus, the larger exhaust rate lambda associated with FHBVS operation plus the exhaust rate lambda for the 500 cfm outleakage is utilized in the analysis.

FHA in the Containment

The potential radioactivity release pathways following a FHA in the containment are established taking into consideration

a. Operation of the containment purge system which would result in radioactivity release via the plant vent

b. Plant Technical Specification Section 3.9.4 that allows for an "open containment" during fuel movement in containment during offload or reload. The most significant containment opening closest to the Control room normal operation intake is the equipment hatch. The equipment hatch is an approximately 20-ft wide circular opening in containment. In the event the containment purge system ceased to operate (a viable scenario since it is single train and has non-vital power), the density driven convective flow out of the equipment hatch (due to the thermal gradient between inside and outside containment conditions), could be significant.

It has been determined that for the FHA in the Containment, the release rate assuming a regulatory based 2 hr release is larger than that dictated by the containment purge ventilation system, or convective flow out of the equipment hatch. Thus, the regulatory based release rate (i.e., 3.45 hr-1), is utilized for this analysis. Review of the atmospheric dispersion factors associated with the plant vent vs the equipment hatch indicates that dose consequences due to releases via the equipment hatch will be bounding.

15.5.22.2.3 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. Since the FHA is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

The bounding EAB and LPZ dose following a FHA at either location and at either unit is presented in Table 15.5-47.

15.5.22.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical FHA-specific assumptions associated with control room response and activity transport.

Design Basis FHA (occurs at t=72 hours after reactor shutdown, which is conservative)

Credit is taken for PG&E Design Class I area radiation monitors located at the control room normal intakes (1-RE-25/26, 2-RE-25/26) to initiate CRVS Mode 4 (filtered /

pressurized accident ventilation) upon detection of high radiation levels at the control room normal intakes as a result of an FHA.

An analytical safety limit of 1 mR/hr for the gamma radiation environment at the control room normal operation air intakes has been used in the FHA analyses to initiate CRVS Mode 4. Note that the actual monitor trip setpoint is lower to include the instrument loop uncertainty.

The radiation monitor response time is primarily dependent on the type of monitor, the setpoint, the background radiation levels and the magnitude of increase in the radiation environment at the detector location.

For a monitor with an instrument time constant of “τ” (2 seconds) and a background of 0.05 mR/hr, the response time “t” to a high alarm Setpoint (HASP < 1 mr/hr), for a step increase of radiation level DR (mR/hr) is determined by solving the following equation that represents the monitor reading approaching the final reading exponentially.

$$HASP = 0.05 + DR(1 - e^{-\frac{t}{\tau}}) \qquad (15.5\text{-}5)$$

It is determined that a DBA FHA (i.e., occurs at 72 hrs post shutdown) will result in a radiation environment at the control room normal operation intakes that greatly exceed the analytical limit of 1 mR/hr for initiating CRVS Mode 4. This will result in an almost instantaneous generation of a radiation monitor signal to initiate CRVS Mode 4 (radiation monitor response time is estimated to be < 1 sec). For purposes of conservatism, and since the delay in isolation of the normal intake has a significant impact on the estimated dose consequences, the analysis conservatively assumes a monitor response time to the HASP of 20 secs.

As discussed in Section 15.5.1.2, when crediting CRVS Mode 4, the FHA dose consequence analyses is not required to address the potential effects of a LOOP. Thus, delays associated with diesel generator sequencing are not addressed.

Therefore, the time delay between the arrival of radioactivity released due to a DBA FHA at both the control room normal Intakes (assumed to be instantaneous) and CRVS Mode 4 operation is estimated to be the sum total of the monitor response time (20 secs), the signal processing time (2 secs) and the damper closure time (10 secs) for a total delay of 32 seconds.

<u>Delayed FHA:</u>

It is recognized that the response time for radiation monitors are dependent on the magnitude of the radiation level / energy spectrum of the airborne cloud at the location of the detectors, which in turn are dependent on the fuel assembly decay time. Thus, an additional case is considered for each of the two FHA scenarios described above (i.e., a FHA in the FHB and a FHA in Containment) when determining the dose to the control room operator; i.e., a case that reflects a delayed FHA at Fuel Offload or a FHA during Reload, occurring at a time when the fuel has decayed to such an extent that the radiation environment at the control room normal intake radiation monitors is just below the setpoint; thus, the control room remains in normal operation mode and CRVS Mode 4 is not initiated.

The analyses determined that the dose consequences of a DBA FHA bound that associated with the delayed FHA for both the FHA in the FHB and the FHA in the containment.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an FHA at either location, and at either unit, are provided in Table 15.5-47B. The χ/Q values presented in Table 15.5-47B take into consideration the various release points-receptors applicable to the FHA to identify the bounding χ/Q values applicable to a FHA at either unit and at either location, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding control room dose following a FHA at either location and at either unit is presented in Table 15.5-47.

15.5.22.3 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5-47.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5- 47.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-47.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the FHA is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

15.5.23 RADIOLOGICAL CONSEQUENCES OF A CONTROL ROD EJECTION ACCIDENT

15.5.23.1 Acceptance Criteria

The radiological consequences of a CREA shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below:

<u>EAB and LPZ Dose Criteria</u>

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.063 Sv (6.3 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.063 Sv (6.3 rem) TEDE.

<u>Control Room Dose Criteria</u>

(1) Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

<u>Technical Support Center Dose Criteria</u>

(1) The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

15.5.23.2 Identification of Causes and Accident Description

As discussed in Section 15.4.6, this event consists of an uncontrolled withdrawal of a control rod from the reactor core. The CREA results in reactivity insertion that leads to a core power level increase, and under adverse combinations of circumstances, fuel failure, and a subsequent reactor trip. In this case, some of the activity in the fuel rod gaps would be released to the coolant and in turn to the inside of the containment building. As a result of pressurization of the containment, some of this activity could leak to the environment.

Following reactor trip, and based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling. DCPP has established that the LOL event generates the maximum primary to secondary heat transfer and the CREA assumes these same conservatively bounding secondary steam releases.

Regulatory guidance provided for the CREA in pertinent sections of Regulatory Guide 1.183 including Appendix H is used to develop the dose consequence model. Table 15.5-52A lists the key assumptions / parameters utilized to develop the radiological consequences following a CREA.

The CREA is postulated to result in 10% fuel failure resulting in the release of the associated gap activity. Per Regulatory Guide 1.183, the core gap activity is assumed to be comprised of 10% of the core noble gases and halogens. A radial peaking factor of 1.65 is applied to the activity release from the fuel gap.

In accordance with the guidance provided in Regulatory Guide 1.183, two independent release paths to the environment are analyzed: first, via containment leakage of the fission products released due to the event from the primary system to containment, assuming that the containment pathway is the only one available; and second, via releases from the secondary system, outside containment, following primary-to-secondary leakage in the steam generators, assuming that the latter pathway is the only one available.

The actual doses resulting from a postulated CREA would be a composite of doses resulting from portions of the release going out via the containment building and, portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a CREA.

15.5.23.2.1 Activity Release Transport Model

The CREA dose consequence analysis evaluates the following two scenarios.

<u>Scenario 1</u>: The failed fuel resulting from a postulated CREA is released into the RCS, which is released in its entirety into the containment via the faulted control rod drive mechanism housing, is mixed in the free volume of the containment, and then released to the environment at the containment technical specification leak rate for the first 24 hrs and at half that value for the remaining 29 days.

<u>Scenario 2</u>: The failed fuel resulting from a postulated CREA is released into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental

releases occur from the steam generators via the MSSVs and 10% ADVs.

The chemical composition of the iodine in the gap is assumed to be 95% particulate (CsI), 4.85% elemental and 0.15% organic. However, because the sump pH is not controlled following a CREA, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e.; it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport From Containment

The failed fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of 0.10% per day for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen or noble gas concentrations airborne in the containment. Per Regulatory Guide 1.183, the chemical composition of the iodine in the gap fuel is 95% particulate (CsI), 4.85% elemental and 0.15% organic. However, since no credit is taken for the actuation of sprays or pH control, the iodine released via containment leakage pathway is assumed to have the same composition as iodine activity released to the environment from the secondary coolant; i.e.; 97% elemental and 3% organic. Environmental releases due to containment leakage can occur unfiltered as a diffuse source from the containment wall, and as a point source via the containment penetration areas or the Plant Vent. The dose consequences are estimated based on the worst case atmospheric dispersion factors, i.e., an assumed environmental release via the containment penetration areas.

Scenario 2: Transport from Secondary System

The failed fuel activity released due to a CREA into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary SG tube leakage. The activity associated with the release of the initial inventory in secondary steam/liquid, and primary to secondary leakage of normal operation RCS, (both at Technical Specification levels) via the MSSVs/10% ADVs are insignificant compared to the failed fuel release, and are therefore not included in this assessment.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the CREA dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant; therefore, the gap iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to the loss of offsite power. Consequently, the radioactivity release resulting from a CREA is discharged to the environment from steam generators via the MSSVs and the 10% ADVs. Per Regulatory Guide 1.183, 97% of all halogens available for release to the environment via the Secondary System are elemental, while the remaining 3% are organic. The SG releases continue until shutdown cooling is initiated via operation of the RHR system (10.73 hours after the accident) and environmental releases are terminated.

15.5.23.2.2 Offsite Dose Assessment

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For Scenario 1 (release via Containment leakage), the worst case 2-hour period occurs during the first 2 hours). For Scenario 2 (release via secondary side), the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a CREA at either unit for both scenarios are presented in Table 15.5-52.

15.5.23.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. Provided below are the critical CREA-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CRVS Mode 4</u>:

The time to generate a signal to switch CRVS operation from Mode 1 to Mode 4 is based on the containment pressure response following a 2 inch small-break LOCA (SBLOCA), and the fact that at DCPP, a Containment High Pressure signal will initiate a SIS which will automatically initiate CRVS Mode 4 pressurization. The containment pressure response analysis for a 2 inch SBLOCA shows that the 5 psig setpoint for Containment High Pressure is reached in 150 seconds after the SBLOCA. As indicated earlier, releases to the containment following a CREA are through a faulted control rod drive mechanism housing. The control rod shaft diameter is 1.840 inches and the

RCCA housing penetration opening is 4 inches in diameter. Based on the above and for the purposes of conservatism, the time to generate the Containment High Pressure SIS following a CREA is assumed to be double the value applicable to the 2 inch SBLOCA, or 300 seconds.

Based on the above, following a CREA,

a. An SIS will be generated at t = 300 sec following a CREA.
b. The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The control room dampers are fully closed 10 secs later, or at t=338.2 secs (i.e., 300 + 28.2 + 10). The 2 second SIS processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.
c. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=312 secs (i.e., 300 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors:

As noted in Section 2.3.5.2.2, because of the proximity of the MSSV/10% ADVs to the control room normal intake of the affected unit and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the faulted unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's control room normal intake is assumed to be contaminated by a release from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a CREA at either unit are provided in Table 15.5-52B. The χ/Q values presented in Table 15.5-52B take into consideration the various release points-receptors applicable to the CREA to identify the bounding χ/Q values applicable to a CREA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding control room dose following a CREA at either unit is presented in Table 15.5-52.

15.5.23.3 Conclusions

The analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release is within 0.063 Sv (6.3 rem) TEDE as shown in

Table 15.5-52.

(2) The radiation dose to an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), is within 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5- 52.

(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-52.

(4) In accordance with the current licensing basis, the TSC design has been evaluated for the LOCA. The dose consequences in the TSC due to airborne radioactivity releases from the CREA is bounded by the dose reported for the LOCA in Section 15.5.17. The atmospheric dispersion factors applicable to the TSC are presented in Table 15.5-83.

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TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
1	Uncontrolled control rod assembly withdrawal from a sub-critical condition (assuming the most unfavorable reactive conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling.	15.2.1	UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION					
2	Uncontrolled control rod assembly withdrawal at the critical power (assuming the most unfavorable reactive conditions of the core and reactor coolant system) which yields the most severe results (hot at zero power, full power, etc).	15.2.2	UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER					
3	Control rod misoperation or sequence of misoperations.	15.2.3	ROD CLUSTER CONTROL ASSEMBLY MISOPERATION					
4	Chemical and volume control system malfunction.	15.2.4	UNCONTROLLED BORON DILUTION					
5	Partial and total loss of reactor coolant flow force including trip of pumps and pump shaft seizures.	15.2.5	PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW	15.3.4	COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW			

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
6	Start-up of an inactive reactor coolant loop or recirculating loop at incorrect temperature.	15.2.6	STARTUP OF AN INACTIVE REACTOR COOLANT LOOP					Precluded in Modes 1 and 2 due to Tech Spec 3.4.4
7	Loss of external electrical load and/or turbine stop valve closure, including, for BWRs closure of main steam isolation valve.	15.2.7	LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP					
8	Loss of normal and/or emergency feedwater flow.	15.2.8	LOSS OF NORMAL FEEDWATER					
9	Loss of all a-c power to the station auxiliaries (station blackout).	15.2.9	LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES					Station Blackout is beyond design basis. Refer to UFSAR Section 8.3.1.6
10	Heat removal greater than heat generation due to (1) feedwater system malfunctions, (2) a pressure regulator failure, or inadvertent opening of a relief valve or safety valve, and (3) a regulating instrument failure.	15.2.10 15.2.12	EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS EXCESSIVE LOAD INCREASE INCIDENT					Regarding (2) and (3): There are no pressure regulators or regulating instruments in the Westinghouse pressurized water reactor (PWR) design whose failure could cause heat removal greater than heat generation.

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
11	Failure of the regulating instrumentation, causing for example, a power-coolant mismatch. Include reactor coolant flow controller failure resulting in increasing flow.	N/A	N/A	N/A	N/A	N/A	N/A	Reactor coolant flow controller is not a feature of the Westinghouse PWR design. Treatment of the performance of the reactivity controller in a number of accident conditions is offered in this chapter. (Chapter 15)

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
12	Internal and external events such as major and minor fires, flood, storms or earthquakes.	N/A	N/A	N/A	N/A	N/A	N/A	Refer to the following Sections: 3.3 - Wind & Tornado Loadings 3.4 - Water level (flood) design 3.5 - Missile protection 3.7 - Seismic design 3.8 - Design of Class I structures 9.5.1 - Fire protection system

TABLE 15.0-1
REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
13	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary and relief and safety valve blowdowns.	15.2.13	ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM	15.3.1	LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES THAT ACTUATE EMERGENCY CORE COOLING SYSTEM	15.4.1	MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOCA)	
14	Spectrum of postulated steam and feedwater system piping breaks inside and outside containment.	15.2.14	ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM	15.3.2	MINOR SECONDARY SYSTEM PIPE BREAKS	15.4.2	MAJOR SECONDARY SYSTEM PIPE RUPTURE	
15	Inadvertent loading and operation of a fuel assembly into an improper position.			15.3.3	INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION			
16	Waste gas decay tank leakage or rupture.					N/A	Refer to Section 11.3.3	
17	Failure of air ejector lines (BWR).	N/A	N/A	N/A	N/A	N/A	N/A	This applies to BWR plants only
18	Steam generator tube rupture (PWR).					15.4.3	STEAM GENERATOR TUBE RUPTURE (SGTR)	
19	Failure of charcoal or cryogenic system (BWR).	N/A	N/A	N/A	N/A	N/A	N/A	This applies to BWR plants only

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
20	The spectrum of rod ejection accidents (PWR).					15.4.6	RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING (ROD CLUSTER CONTROL ASSEMBLY EJECTION)	
21	The spectrum of rod drop accidents (BWR).	N/A	N/A	N/A	N/A	N/A	N/A	This applies to BWR plants only
22	Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment.	N/A	N/A	N/A	N/A	N/A	N/A	No instrument lines from the RCS boundary in the DCPP design penetrate the containment. (For definition of the RCS boundary, refer to the 1972 issue of ANS N18.2, Nuclear Safety Criteria for the Design of Stationary PWR Plants.)
23	Fuel handling accident.					15.4.5	FUEL HANDLING ACCIDENT	
24	Small spills or leaks of radioactive material outside containment.	N/A	N/A	N/A	N/A	N/A	N/A	The analysis of the consequences of such small spills and leaks is included within the cases evaluated in Chapter 11, and larger leaks and spills are analyzed in Section 15.5.

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
25	Fuel cladding failure (BWR, PWR) combined with steam generator leak (PWR).	N/A	N/A	N/A	N/A	N/A	N/A	The radiological consequences of this event are analyzed in Chapter 11, for the case of "Anticipated Operational Occurrences."
26	Control room uninhabitabi1ity.	N/A	N/A	N/A	N/A	N/A	N/A	Habitability of the control room following accident conditions is discussed in Chapter 6, and potential radiological exposures are reported in Section 15.5. In addition, Chapter 7 contains an analysis showing that the plant can be brought to, and maintained in, Mode 3 from outside the control room.
27	Failure or overpressurization of low pressure residual heat removal system.	N/A	N/A	N/A	N/A	N/A	N/A	Overpressurization of the residual heat removal system (RHRS) is considered extremely unlikely. PG&E reviewed possible RHRS overpressure scenarios and qualified the system for all credible high pressure transients in DCPP design change package N-049118.

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
28	Loss of condenser vacuum.	N/A	N/A	N/A	N/A	N/A	N/A	This event is covered by the analyses of Section 15.2.7. Separate event analysis is not required.
29	Turbine trip with coincident failure of turbine bypass valves to open.	N/A	N/A	N/A	N/A	N/A	N/A	This event is covered by the analyses of Section 15.2.7. Separate event analysis is not required.
30	Loss of service water system.	N/A	N/A	N/A	N/A	N/A	N/A	Malfunctions of auxiliary saltwater system and component cooling water system (CCWS) are discussed in Chapter 9, Sections 9.2.7 and 9.2.2 respectively.
31	Loss of one (redundant) d-c system.	N/A	N/A	N/A	N/A	N/A	N/A	There are no significant safety-related consequences of this event.
32	Inadvertent operation of ECCS during power operation.	15.2.15	SPURIOUS OPERATION OF THE SAFETY INJECTION SYSTEM AT POWER					
33	Turbine trip with failure of generator breaker to open.	N/A	N/A	N/A	N/A	N/A	N/A	The effects of turbine trip on the RCS are presented in Section 15.2.7. Separate event analysis is not required.
34	Loss of instrument air system.	N/A	N/A	N/A	N/A	N/A	N/A	Malfunctions of this system are discussed in Section 9.3.2.

TABLE 15.0-1

REGULATORY GUIDE 1.70 REVISION 1, APPLICABILITY MATRIX

Event	REG GUIDE 1.70, TABLE 15-1, REPRESENTATIVE TYPES OF EVENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR	15.2 Section	UFSAR SECTION 15.2 CONDITION II: FAULTS OF MODERATE FREQUENCY	15.3 Section	UFSAR SECTION 15.3 CONDITION III: INFREQUENT FAULTS	15.4 Section	UFSAR SECTION 15.4 CONDITION IV: LIMITING FAULTS	Location of Analyses or Reason Why Not Applicable
35	Malfunction of turbine gland sealing system.	N/A	N/A	N/A	N/A	N/A	N/A	The radiological effects of this event are not significant for PWR plants. Minor leakages are within the scope of the analysis cases presented in Chapter 11.

TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Core Rated Thermal Power	3411
Thermal power generated by the reactor coolant pumps minus heat losses to containment and letdown system [(b)]	14
Nuclear steam supply system (NSSS) thermal power output[(b)]	3425
The engineered safety features design rating (maximum calculated turbine rating)[(a)]	3570

(a) The units will not be operated at this rating because it exceeds the license ratings.
(b) As noted on Table 15.1-4, some analyses assumed a full power NSSS thermal output of 3,423 MWt, based on the previous net reactor coolant pump heat of 12 MWt. An evaluation concluded that the effect of an additional 2 MWt for NSSS is negligible such that analyses based on 3,423 MWt remain valid.

TABLE 15.1-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting TripPoint Assumed In Analyses	Time Delay, sec
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Power range high positive nuclear power rate	7% / 2 sec	0.65
Overtemperature ΔT	Variable, see Figure 15.1-1	7[(a)]
Overpower ΔT	Variable, see Figure 15.1-1	7[(a)]
High pressurizer pressure	2460 psia	2
Low pressurizer pressure	1860 psia	2
High pressurizer water level	100%	N/A[(f)]
Low reactor coolant flow (from loop flow detectors)	85% loop flow[(b, d)]	1
Undervoltage trip	[(b)]	N/A[(b)]
Low-low steam generator level	8.2% of narrow range level span	2[(c)]
High steam generator level trip of the feedwater pumps and closure of feedwater system valves and turbine trips	100% of narrow range level span[(e)]	2

(a) Total time delay consists of a maximum 5-second RTD lag time constant and a maximum 2-second electronics delay

(b) Complete loss of flow analysis assumes that reactor trips, on low reactor coolant loop flow, not undervoltage.underfrequency.

(c) When below 50% power, a variable trip time delay is utilized as discussed in Section 7.2.2.1.5.

(d) Value used in the analysis of the Locked Rotor event (Section 15.4.4) for RCS pressure and maximum cladding Temperature. Westinghouse letter PGE-96-582, Diablo Canyon Units 1 & 2 Evaluation of Revised Low Reactor Coolant Flow Reactor Trip Setpoint, June 27, 1996, concludes that a safety analysis setpoint of 85% loop flow is acceptable for the Locked Rotor event (Section 15.4.4) and the Partial Loss of Flow event (Section 15.2.5), for which 87% was assumed in the analysis.

(e) The analysis assumed 100% narrow range level span for conservatism. The plant setpoint analytical limit is 98.8% narrow range level span for Model Delta 54 steam generators due to void effects. Although the turbine trip is modeled for completeness it is not needed for DNBR analysis.

(f) Westinghouse Letter PGE-02-72, Diablo Canyon Units 1 and 2 Evaluation of Reactor Trip Functions for Uncontrolled RCCA Bank Withdrawal at Power, December 13, 2002, documents that a specific response time is not assumed since it is not a sensitive parameter for the generic evaluation results.

TABLE 15.1-4

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II					
Uncontrolled RCCA bank withdrawal from a subcritical condition	TWINKLE, THINC, FACTRAN	+5	-	Least negative defect - 954 pcm	0
Uncontrolled RCCA bank withdrawal at power	LOFTRAN	+5	0.43	Lower and Upper	3,423
RCCA misoperation	THINC, ANC, LOFTRAN	-	-	Lower	3,425
Uncontrolled boron dilution					0 and 3,423
Partial loss of forced reactor coolant flow	LOFTRAN, THINC, FACTRAN	+5	-	Upper	3,423
Startup of an inactive reactor coolant loop	LOFTRAN, FACTRAN, THINC	-	0.43	Lower	2,396
Loss of external electrical load and/or turbine trip-DNBR	LOFTRAN, RETRAN-02	+5	0.43	Lower and Upper	3,423
Loss of external electrical load and/or turbine trip - Overpressure	RETRAN-02	+5	-	Lower	3,425

TABLE 15.1-4

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II (Cont'd)					
Loss of normal feedwater	RETRAN-02W	0	-	Upper	3,425
Loss of offsite power to the plant auxiliaries	RETRAN-02W	0	-	Upper	3,425
Excessive heat removal due to feedwater system malfunctions	RETRAN-02W	-	0.43	Lower	3,425
Excessive load increase	LOFTRAN	-	0 and 0.43	Lower and Upper	3,423
Accidental depressurization of the reactor coolant system	LOFTRAN	+7	-	Lower	3,425
Inadvertent operation of ECCS during power operation - DNBR	LOFTRAN	+5	0.43	Lower and Upper	3,423
Inadvertent operation of ECCS during power operation – Pressurizer Overfill	RETRAN-02W	-	0.43	Upper	3,425
CONDITION III					
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	NOTRUMP SBLOCTA	-	-	-	3,479
Inadvertent loading of a fuel assembly into an improper position	PHOENIX-P, ANC	-	-	-	3,483
Complete loss of forced reactor coolant flow	LOFTRAN, THINC, FACTRAN	0	-	Upper	3,425

TABLE 15.1-4

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION III (Cont'd)					
Single RCCA withdrawal at full power	ANC, THINC, PHOENIX-P	-	-	-	3,425[e]
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident)	WCOBRA/TRAC HOTSPOT MONTECF	Function of moderator density. See Sec. 15.4.1	0	Function of fuel temp.	3,479
Major secondary system pipe rupture up to and including double-ended rupture (rupture of a steam pipe)	RETRAN-02W, ANC,THINC	-	Function of moderator density. See Figure 15.4.2-2.	See Figure 15.4.2-1	0.0 (Subcritical)
Major rupture of a main feedwater pipe	RETRAN-02W		0.0	Lower	3,425
Rupture of a main steam line at power	RETRAN-02W, ANC, THINC-IV		0.43	Lower	3,425
	-	-	-	-	
Steam generator tube rupture	RETRAN-02W	-	0.0	Lower and Upper	3,425
CONDITION IV (Cont'd)					
Single reactor coolant pump locked rotor	LOFTRAN,	+5 (rods in	0.0 (pressure	Upper	3,425

TABLE 15.1-4

Events	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Assumed Reactivity Coefficients: Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
	THINC, FACTRAN	DNB case)	case)		
Fuel handling accident					3,577
Rupture of a control rod mechanism housing (RCCA ejection)	TWINKLE, FACTRAN PHOENIX-P	+5.2 BOL -23.EOL	-	Least negative defect. See Table 15.4-11.	0 and 3,423

(a) Only one is used in analysis, i.e., either moderator temperature or moderator density coefficient.
(b) Reference Figure 15.1-5.
(c) Two percent calorimetric error considered where applicable.
(d) Pcm means percent mille. See footnote Table 4.3-1.
(e) Analysis only models core thermal power of 3411 MWt

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal 7.5×10^{-4} Δk/sec reactivity insertion rate from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	9.6
	Peak nuclear power occurs	9.8
	Rods begin to fall into core	10.1
	Peak heat flux occurs	11.9
	Peak hot spot average cladding temperature occurs	12.3
Uncontrolled RCCA Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at reactivity insertion rate of7.5 x 10^{-4} Δk/sec	0.0
	Power range high neutron flux high trip point reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.0
2. Case B	Initiation of uncontrolled RCCA withdrawal at a reactivity insertion rate of3.0 x 10^{-5} Δk/sec	0.0

Accident	Event	Time, sec
	Overtemperature ΔT reactor trip signal initiated	31.8
	Rods begin to fall into core	33.8
	Minimum DNBR occurs	34.2
Uncontrolled Boron Dilution		
1Dilution during refueling	Dilution begins	0.0
	Shutdown margin lost	~11884
2. Dilution during cold shutdown		
a. RCS filled	Dilution begins	0.0
	Shutdown margin lost	>900
b. RCS drained	Dilution begins	0.0
	Shutdown margin lost	>900
3. Dilution during hot shutdown		
a. One RCP operating	Dilution begins	0.0
	Shutdown margin lost	>900
b. RHR operating	Dilution begins	0.0
	Shutdown margin lost	>900
4. Dilution during hot standby	Dilution begins	0.0
	Shutdown margin lost	>900
5. Dilution during startup	Dilution begins	0.0
	Shutdown margin lost	~3696
6. Dilution during full power operation		
a. Automatic reactor control	1.6 % shutdown margin lost	~1180
b. Manual reactor control	Dilution begins	0.0

Accident	Event	Time, sec
	Reactor trip setpoint reached for high neutron flux	40
	Rods begin to fall into core	40.5
	1.6 % shutdown is lost (if dilution continues after trip)	~ 900
Historical *Partial Loss of Forced Reactor Coolant Flow*		
1. All loops operating, two pumps coasting down	*Coastdown begins*	*0.0*
	Low-flow reactor trip[(b)]	*1.43*
	Rods begin to drop	*2.43*
	Minimum DNBR occurs	*3.9*
Historical *Startup of an Inactive Reactor Coolant Loop*	*Initiation of pump startup*	*0.0*
	Power reaches high nuclear flux trip	*3.2*
	Rods begin to drop	*3.7*
	Minimum DNBR occurs	*4*

Accident	Event	Time, sec
Loss of External Electrical Load-DNBR		
1. With pressurizer control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	11.9
	Initiation of steam release from steam generator safety valves	12.0
	Rods begin to drop	13.9
	Peak pressurizer pressure occurs	14.5
	Minimum DNBR occurs	15
2. With pressurizer control (EOL)	Loss of electrical load	0.0
	Peak pressurizer pressure occurs	9.0
	Initiation of steam release from steam generator safety valves	12.5
	Low-low steam generator water level reactor trip	57
	Rods begin to drop	59
	Minimum DNBR occurs	(a)
3. Without pressurizer control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	6.1
	Rods begin to drop	8.1
	Minimum DNBR occurs	(a)

Accident	Event	Time, sec
	Peak pressurizer pressure occurs	9.5
	Initiation of steam release from steam generator safety valves	12.0
4. Without pressurizer control (EOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	6
	Rods begin to drop	8
	Minimum DNBR occurs	(a)
	Peak pressurizer pressure occurs	8.5
	Initiation of steam release from steam generator safety valves	12.5
Loss of External Electrical Load-Overpressure (Peak RCS Pressure)		
1. With no pressurizer control (BOL)	Reactor Trip	9.0
	PSVs Open	9.1
	Peak RCS Pressure	9.5
	MSSVs Open	9.8
	Peak Secondary Side Pressure	16.0
Overpressure (Peak Secondary Side Pressure)		
2. With pressurizer control (BOL)	PORVs Open	3.6
	MSSVs Open	9.1
	Reactor Trip	15.1
	PSVs Open	16.3
	Peak RCS Pressure	16.5

Accident	Event	Time, sec	
	Peak Secondary Side pressure	20.0	
		W/Power	W/O Power
Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries	Main feedwater flow stops	0.0	0.0
	Low-low steam generator water level reactor trip	52.7	54.2
	Rods begin to drop	54.7	56.2
	Reactor coolant pumps begin to coast down	-	58.2
	Four SGs begin to receive aux feed from both motor-driven AFW pumps	112.7	114.2
	Peak water level in pressurizer occurs (post-trip)	1294	2030

Accident	Event	Time, sec
Excessive Feedwater at Full Load	One main feedwater regulating valve fails full open	0.0
	High-high steam generator water level is reached	33.6
	Turbine trip signal (from high-high steam generator level, turbine stop valve fully closed 0.1second later	36.0
	Reactor trip occurs from turbine trip (rod motion begins)	38.1
	Minimum DNBR occurs	39.0
	Initial pressurizer PORV opens (all PORVs closed 1.3 seconds later)	39.7
	Main feedwater isolation valves closed in all four loops (from high-high steam generator level)	99.6
Excessive Load Increase		
1. Manual reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	240
2. Manual reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	64
3. Automatic reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	150

Accident	Event	Time, sec
4. Automatic reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	150
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one pressurizer safety valve	0.0
	Overtemperature ΔT reactor trip setpoint reached	27.5
	Rods begin to drop	29.5
	Minimum DNBR occurs	29.8
Inadvertent Operation of ECCS During Power Operation - DNBR	Charging pumps begin injecting borated water	0.0
	Low-pressure trip point reached	23
	Rods begin to drop	25
Inadvertent Operation of ECCS During Power Operation - Pressurizer Overfill		
Case to establish the maximum time available to ensure a PG&E Design Class I PORV is available	Reactor Trip/Safety injection	0
	Pressurizer fills	904
	First PSV opens after pressurizer fills	904
Case to establish the minimum time to pressurizer filling	Reactor Trip/Safety Injection	0
	PORV opens	41.2
	Pressurizer fills	690

Accident	Event	Time, sec
Case to establish the minimum time to deplete the backup nitrogen accumulators	Reactor Trip/Safety Injection	0
	PORV opens	43.3
	Pressurizer fills	764
	Final available PORV cycle (300th cycle)	2748

(a) DNBR does not decrease below its initial value.
(b) Analysis assumed low flow setpoint of 87 percent loop flow. An evaluation concludes that 85 percent loop flow is acceptable. Refer to Table 15.1-2, footnote (d).

TABLE 15.3-1

TIME SEQUENCE OF EVENTS - SMALL BREAK LOCA

Unit 1

	2-inch	3-inch	4-inch	6-inch
Transient Initiated, sec	0	0	0	0
Reactor Trip Signal, sec	43.58	18.32	10.55	5.9
Safety Injection Signal, sec	58	26.8	16.57	8.58
Safety Injection Begins(1), sec	85	53.8	43.57	35.58
Loop Seal Clearing Occurs(2), sec	1197	514	300	110
Top of Core Uncovered(3), sec	1796	941	635	N/A
Accumulator Injection Begins, sec	N/A	1984	885	385
Top of Core Recovered, sec	6500	3170	2545	N/A
RWST Low Level, sec	1709	1689	1664	1640

Unit 2

	2-inch	3-inch	4-inch	6-inch
Transient Initiated, sec	0	0	0	0
Reactor Trip Signal, sec	44.72	18.78	10.82	6.11
Safety Injection Signal, sec	59.45	27.41	16.68	9
Safety Injection Begins(1), sec	86.45	54.41	43.68	36
Loop Seal Clearing Occurs(2), sec	1360	575	290	120
Top of Core Uncovered(3), sec	3200	722	770	N/A
Accumulator Injection Begins, sec	N/A	3050	985	400
Top of Core Recovered, sec	N/A	3215	1630	N/A
RWST Low Level, sec	1708	1690	1666	1641

(1) Safety Injection begins 27.0 seconds (SI delay time) after the safety injection signal is reached.
(2) Loop seal clearing is considered to occur when the broken loop seal vapor flow rate is sustained above 1 lbm/s.
(3) Top of core uncovery time is taken as the time when the core mixture level is sustained below the top of the core elevation.

TABLE 15.3-2

FUEL CLADDING RESULTS - SMALL BREAK LOCA

Unit 1

	2-inch	3-inch	4-inch
PCT (°F)	907	1391	1241
PCT Time (s)	2173.3	1891.7	975.8
PCT Elevation (ft)	10.75	11.25	11.00
Burst Time (s) (1)	N/A	N/A	N/A
Burst Elevation (ft) (1)	N/A	N/A	N/A
Maximum Hot Rod Transient ZrO2 (%)	0.01	0.38	0.07
Maximum Hot Rod Transient ZrO2 Elev. (ft)	10.75	11.25	10.75
Hot Rod Average Transient ZrO2 (%)	0.01	0.06	0.01

Unit 2

	2-inch	3-inch	4-inch
PCT (°F)	814	1288	1004
PCT Time (s)	4838.3	1961.8	1079.2
PCT Elevation (ft)	11.00	11.25	10.75
Burst Time (s) (1)	N/A	N/A	N/A
Burst Elevation (ft) (1)	N/A	N/A	N/A
Maximum Hot Rod Transient ZrO2 (%)	0.01	0.18	0.01
Maximum Hot Rod Transient ZrO2 Elev. (ft)	11.00	11.25	10.75
Hot Rod Average Transient ZrO2 (%)	0	0.03	0.01

(1) Burst was not predicted to occur for any break size.

TABLE 15.3-3

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time sec[a]
Complete Loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down	Coastdown begins	0.0
	Rod motion begins	2.85
	Minimum DNBR occurs	4.8

a) Event times are Unit 1; Unit 2 is 0.04 seconds later.

TABLE 15.4.1-1A

UNIT 1 BEST ESTIMATE LARGE BREAK LOCA TIME SEQUENCE OF EVENTS
FOR THE REFERENCE TRANSIENT

Event	**Time (sec)**
Start of Transient	0.0
Safety Injection Signal	6.0
Accumulator Injection Begins	11.0
End of Blowdown	29.0
Safety Injection Begins	33.0
Bottom of Core Recovery	37.0
Accumulator Empty	50.0
PCT Occurs	39.0
Hot Rod Quench	>300.0
End of Transient	500.0

TABLE 15.4.1-1B

UNIT 2 BEST ESTIMATE LARGE BREAK SEQUENCE OF EVENTS
FOR LIMITING PCT CASE

Event	**Time (sec)**
Start of Transient	0.0
Safety Injection Signal	6.0
Accumulator Injection Begins	13.0
End of Blowdown	29.0
Safety Injection Begins	33.0
Bottom of Core Recovery	37.0
Accumulator Empty	48.0
PCT Occurs	110.0
Hot Rod Quench	285.0
End of Transient	500.0

TABLE 15.4.1-2A

UNIT 1 BEST ESTIMATE LARGE BREAK LOCA ANALYSIS RESULTS

Component	Blowdown Peak	First Reflood Peak	Second Reflood Peak
$PCT^{average}$	<1485°F	<1621°F	<1486°F
$PCT^{95\%}$	<1744°F	<1900°F	<1860°F
Maximum Oxidation		<11%	
Total Oxidation		<0.89%	

TABLE 15.4.1-2B

UNIT 2 BEST ESTIMATE LARGE BREAK LOCA ANALYSIS RESULTS

	Result	Criterion
95/95 PCT	1,872°F	< 2,200°F
95/95 LMO	1.64%	< 17%
95/95 CWO	0.17%	< 1%

PCT – Peak Cladding Temperature
LMO – Local Maximum Oxidation
CWO – Core Wide Oxidation

TABLE 15.4.1-3A

UNIT 1 KEY BEST ESTIMATE LARGE BREAK LOCA PARAMETERS
AND REFERENCE TRANSIENT ASSUMPTIONS

Parameter	Reference Transient	Uncertainty or Bias
1.0 Plant Physical Description		
a. Dimensions	Nominal	ΔPCT_{MOD}
b. Flow resistance	Nominal	ΔPCT_{MOD}
c. Pressurizer location	Opposite broken loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	17x17 V5 w/ZIRLO clad	Bounded
f. SG tube plugging level	High (15%)	Bounded[a]
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate	Nominal - 100% of uprated power (3411 MWt)	ΔPCT_{PD}
b. Peak linear heat rate (PLHR)	Derived from desired Technical Specifications (TS) limit and maximum baseload	ΔPCT_{PD}
c. Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$	ΔPCT_{PD}

TABLE 15.4.1-3A

Parameter	Reference Transient	Uncertainty or Bias
d. Hot assembly average heat rate	HRFLUX/1.04	ΔPCT_{PD}
e. Hot assembly peak heat rate	PLHR/1.04	ΔPCT_{PD}
f. Axial power distribution (PBOT, PMID)	Figure 3-2-10 of Reference 60	ΔPCT_{PD}
g. Low power region relative power (PLOW)	0.3	Bounded[(a)]
h. Hot assembly burnup	BOL	Bounded
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	TS Maximum (0)	Bounded
k. HFP boron	800 ppm	Generic
2.2 Fluid Conditions		
a. T_{avg}	Max. nominal T_{avg} = 577.3°F	Nominal is bounded, uncertainty is in ΔPCT_{IC}
b. Pressurizer pressure	Nominal (2250.0 psia)	ΔPCT_{IC}
c. Loop flow	85000 gpm	ΔPCT_{MOD}[(b)]
d. T_{UH}	Best Estimate	0
e. Pressurizer level	Nominal (1080 ft^3)	0
f. Accumulator temperature	Nominal (102.5°F)	ΔPCT_{IC}
g. Accumulator pressure	Nominal (636.2 psia)	ΔPCT_{IC}

TABLE 15.4.1-3A

Parameter	Reference Transient	Uncertainty or Bias
h. Accumulator liquid volume	Nominal (850 ft^3)	ΔPCT_{IC}
i. Accumulator line resistance	Nominal	ΔPCT_{IC}
j. Accumulator boron	Minimum	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine	ΔPCT_{MOD}
c. Break size	Nominal (cold leg area)	ΔPCT_{MOD}
d. Offsite power	Off (RCS pumps tripped)	Bounded[(a)]
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	Nominal (68°F)	ΔPCT_{IC}
g. Safety injection delay	Max delay (27.0 sec, with loss of offsite power)	Bounded
h. Containment pressure	Minimum based on WC/T M&E	Bounded
i. Single failure	ECCS: Loss of 1 SI train	Bounded
j. Control rod drop time	No control rods	Bounded
4.0 Model Parameters		
a. Critical Flow	Nominal (as coded)	ΔPCT_{MOD}

TABLE 15.4.1-3A

Parameter	Reference Transient	Uncertainty or Bias
b. Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}
d. Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Noncondensable gases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	ΔPCT_{MOD}

(a) Confirmed by plant-specific analysis.
(b) Assumed to be result of loop resistance uncertainity.

Notes:

1. ΔPCT_{MOD} indicates this uncertainty is part of code and global model uncertainty.

2. ΔPCTPD indicates this uncertainty is part of power distribution uncertainty.

3. ΔPCTIC indicates this uncertainty is part of initial condition uncertainty.

TABLE 15.4.1-3B

UNIT 2 KEY BEST ESTIMATE LARGE BREAK LOCA PARAMETERS AND INITIAL TRANSIENT ASSUMPTIONS

Parameter	Initial Transient	Range/Uncertainty
1.0 Plant Physical Description		
a. Dimensions	Nominal	Sampled
b. Flow resistance	Nominal	Sampled
c. Pressurizer location	Opposite broken loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	17x17 V5 + with ZIRLO™ cladding, Non-IFBA	Bounded
f. Steam generator tube plugging level	High (15%)	Bounded[a]
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal – Based on 100% thermal power (3468 MWt)	Sampled
b. Hot rod peak linear heat rate (PLHR)	Derived from desired Technical Specification limit F_Q = 2.7 and maximum baseload F_Q = 2.1	Sampled
c. Hot rod average linear heat rate (HRFLUX)	Derived from Technical Specification $F_{\Delta H}$ = 1.7	Sampled
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	Sampled
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	Sampled
f. Axial power distribution (PBOT, PMID)	Figure 15.4.1-15B	Sampled
g. Low power region relative power (PLOW)	0.3	Bounded[a]
h. Cycle burnup	~2000 MWD/MTU	Sampled
i. Prior operating history	Equilibrium decay heat	Bounded

TABLE 15.4.1-3B

Parameter	Initial Transient	Range/Uncertainty
2.0 Plant Initial Operating Conditions (continued)		
j. Moderator temperature coefficient	Technical Specification Maximum (0)	Bounded
k. HFP boron	800 ppm	Generic
2.2 Fluid Conditions		
a. T_{avg}	High Nominal T_{avg} = 577.6°F	Bounded[(a)], Sampled
b. Pressurizer pressure	Nominal (2250.0 psia)	Sampled
c. Loop flow	85,000 gpm	Bounded
d. Upper head fluid temperature	T_{cold}	0
e. Pressurizer level	Nominal	0
f. Accumulator temperature	Nominal (102.5°F)	Sampled
g. Accumulator pressure	Nominal (636.2 psia)	Sampled
h. Accumulator liquid volume	Nominal (850 ft^3)	Sampled
i. Accumulator line resistance	Nominal	Sampled
j. Accumulator boron	Minimum (2200 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine (DECLG)	Sampled
c. Break size	Nominal (cold leg area)	Sampled
d. Offsite power	Loss of offsite power	Bounded[(a)]
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	Nominal (68°F)	Sampled
g. Safety injection delay	Maximum delay (27.0 sec, with loss of offsite power)	Bounded

TABLE 15.4.1-3B

Parameter	Initial Transient	Range/Uncertainty
3.0 Accident Boundary Conditions (continued)		
h. Containment pressure	Bounded – Lower (conservative) than pressure curve shown in Figure 15.4.1-14B.	Bounded
i. Single failure	ECCS: Loss of one safety injection train; Containment pressure: all trains operational	Bounded
j. Control rod drop time	No control rods	Bounded
4.0 Model Parameters		
a. Critical flow	Nominal (CD = 1.0)	Sampled
b. Resistance uncertainties in broken loop	Nominal (as coded)	Sampled
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	Sampled
d. Core heat transfer	Nominal (as coded)	Sampled
e. Delivery and bypassing of emergency core coolant	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Noncondensable gases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	Sampled

(a) Per Confirmatory Study results (Section 15.4.1.1.2.5)

TABLE 15.4.1-4A

UNIT 1 SAMPLE OF BEST ESTIMATE SENSITIVITY ANALYSIS RESULTS
FOR ORIGINAL ANALYSIS (Reference 60)

Type of Study	Parameter Varied	Value	PCT Results (°F)		
			Blowdown	Reflood 1	Reflood 2
Reference Transient		See Table 15.4-3	1600	1852	1984
Confirmatory Cases	Steam Generator Tube Plugging	0%	1569	1798	1878
	Offsite Power Assumption	Available	1500	1685	1781
	Normalized Power in Outer Assemblies	0.8	1611	1805	1939
	Vessel Average Temperature	565°F	1573	1843	1871
Initial Condition	Accumulator Volume	+50 ft^3	1601	1856	1823
		–50 ft^3	1599	1863	2182
Global Models	DECLG, CD	1.0	1600	1852	1984
	SPLIT, CD	1.4	-	1596	1637
		1.6	-	1784	1799
		1.8	-	1790	1738
		2.0	-	1765	1804

TABLE 15.4.1-4B

UNIT 2 RESULTS FROM CONFIRMATORY STUDIES

Transient Description	PCT (°F) Reflood
Initial Transient (High T_{avg}, High SGTP, Low PLOW, LOOP)	1595
SGTP Confirmatory Transient (High T_{avg}, Low SGTP, Low PLOW, LOOP)	1576
T_{avg}, Confirmatory Transient (Low T_{avg}, High SGTP, Low PLOW, LOOP)	1536
PLOW Confirmatory Transient (High T_{avg}, High SGTP, High PLOW, LOOP)	1657
LOOP Confirmatory Transient (High T_{avg}, High SGTP, Low PLOW, no-LOOP)	1425
Reference Transient (High T_{avg}, High SGTP, High PLOW, LOOP)	1657

TABLE 15.4.1-5A

UNIT 1 CONTAINMENT BACK PRESSURE ANALYSIS INPUT PARAMETERS USED FOR BEST ESTIMATE LOCA ANALYSIS

Net Free Volume, cu ft	2,630,000
Initial Conditions	
Pressure, psia	14.7
Temperature, °F	85
RWST temperature, °F	35
Service water temperature, °F	45
Outside temperature, °F	33
Spray System	
Number of pumps operating	2
Runout flowrate per pump, gpm	3400
Actuation time, sec	40.8
Safeguards Fan Coolers	
Number of fan coolers operating	5
Fastest post-accident initiation of fan coolers, sec	0
Structural Heat Sinks	
Thickness, in.	Area, ft^2
42.0 concrete	65,749
12.0 concrete	24,054
24.0 concrete	14,313
12.0 concrete	48,183
12.0 concrete	15,725
108.0 concrete	20,493
30.0 concrete	33,867
1.68 steel	8,525
1.92 steel	4,015

Structural Heat Sinks (continued)

Thickness, in.	Area, ft^2
6.99 steel	1,771
0.5656 steel	43,396
0.088 steel	24,090
0.22 steel	10,597
0.088 steel	8,470
0.102 steel	23,438
0.071 steel	20,266
0.708 steel	26,050
0.127 steel	33,000
0.773 steel	11,004
0.375 steel	99,616
1.596 steel	1,530
1.098 steel	21,022
0.745 steel	6,755
0.96 steel	792
0.144 stainless steel	9,737
0.654 stainless steel	943
0.642 steel	1,373
3.0 steel	575
0.75 steel	17,542

TABLE 15.4.1-5B

UNIT 2 CONTAINMENT BACK PRESSURE ANALYSIS INPUT PARAMETERS USED FOR BEST ESTIMATE LBLOCA ANALYSIS

Net Free Volume	2,630,000 ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	85.0°F
RWST temperature	35.0°F
Service water temperature	48.0°F
Temperature outside containment	33.0°F
Initial spray temperature	35.0°F
Spray System	
Number of spray pumps operating	2
Post-accident spray system initiation delay	40.8 sec
Maximum spray system flow from all pumps	6,800 gal/min.
Containment Fan Coolers	
Post-accident initiation fan coolers	0.0 sec[a]
Number of fan coolers operating	5

[a] Bounds delay with and without LOOP

TABLE 15.4.1-7A

UNIT 1 PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter		Operating Range
1.0	Plant Physical Description	
	a. Dimensions	No in-board assembly grid deformation assumed due to LOCA+DDE or LOCA + Hosgri
	b. Flow resistance	N/A
	c. Pressurizer location	N/A
	d. Hot assembly location	Anywhere in core
	e. Hot assembly type	Fresh 17X17 V5, ZIRLO, or Zircaloy cladding, 1.5X IFBA or non-IFBA
	f. SG tube plugging level	≤15%
	g. Fuel assembly type	Vantage 5, ZIRLO, or Zircaloy cladding, 1.5X IFBA or non-IFBA
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	a. Core average linear heat rate	Core power ≤ 102% of 3411 MWt
	b. Peak linear heat rate	$F_Q \leq 2.7$
	c. Hot rod average linear heat rate	$F_{\Delta H} \leq 1.7$
	d. Hot assembly average linear heat rate	$\overline{P}_{HA} \leq 1.57$

TABLE 15.4.1-7A

Parameter		Operating Range
	e. Hot assembly peak linear heat rate	$F_{QHA} \leq 2.7/1.04$
	f. Axial power distribution (PBOT, PMID)	Figure 15.4.1-15A
	g. Low power region relative power (PLOW)	$0.3 \leq$ PLOW ≤ 0.8
	h. Hot assembly burnup	$\leq$ 75,000 MWD/MTU, lead rod
	i. Prior operating history	All normal operating histories
	j. MTC	≤ 0 at HFP
	k. HFP boron	Normal letdown
	2.2 Fluid Conditions	
	a. Tavg	$560.0 \leq T_{ave} \leq 582.3$°F
	b. Pressurizer pressure	$2190 \leq P_{RCS} \leq 2310$ psia
	c. Loop flow	$\geq$ 85,000 gpm/loop
	d. T_{UH}	Current upper internals
	e. Pressurizer level	Normal level, automatic control
	f. Accumulator temperature	$85 \leq$ accumulator temperature ≤ 120°F
	g. Accumulator pressure	$579 \leq P_{ACC} \leq 664$ psig
	h. Accumulator volume	$814 \leq V_{acc} \leq 886$ ft^3

TABLE 15.4.1-7A

Parameter		Operating Range
	i. Accumulator fL/D	Current line configuration
	j. Minimum accumulator boron	≥ 2200 ppm
3.0	Accident Boundary Conditions	
	a. Break location	N/A
	b. Break type	N/A
	c. Break size	N/A
	d. Offsite power	Available or LOOP
	e. Safety injection flow	Figure 15.4.1-13A
	f. Safety injection temperature	46 ≤ SI Temperature ≤ 90°F
	g. Safety injection delay	≤17 seconds (with offsite power) ≤ 27 seconds (with LOOP)
	h. Containment pressure	Bounded - see Figure 15.4.1-14A
	i. Single failure	Loss of one train
	j. Control rod drop time	N/A

TABLE 15.4.1-7B

UNIT 2 PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

	Parameter	Operating Range
1.0	**Plant Physical Description**	
	a) Dimensions	No in-board assembly grid deformation during LOCA+DDE or LOCA + Hosgri
	b) Flow resistance	N/A
	c) Pressurizer location	N/A
	d) Hot assembly location	Anywhere in core interior (149 locations)[(a)]
	e) Hot assembly type	Fresh 17x17 V5+ fuel with ZIRLO™ cladding
	f) Steam generator tube plugging level	$\leq$ 15%
	g) Fuel assembly type	17x17 V5+ fuel with ZIRLO™ cladding, non-IFBA or IFBA
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core average linear heat rate	Core power $\leq$ 100.3% of 3,468 MWt
	b) Peak linear heat rate	$F_Q \leq 2.7$
	c) Hot rod average linear heat rate	$F_{\Delta H} \leq 1.7$
	d) Hot assembly average linear heat rate	$\overline{P}_{HA} \leq 1.7/1.04$
	e) Hot assembly peak linear heat rate	$F_{QHA} \leq 2.7/1.04$
	f) Axial power distribution (PBOT, PMID)	See Figure 15.4.1-15B.
	g) Low power region relative power (PLOW)	$0.3 \leq PLOW \leq 0.8$
	h) Hot assembly burnup	$\leq$ 75,000 MWD/MTU, lead rod[(a)]
	i) Prior operating history	All normal operating histories
	j) Moderator temperature coefficient	$\leq$ 0 at HFP
	k) HFP boron (minimum)	800 ppm (at BOL)
	2.2 Fluid Conditions	
	a) T_{avg}	$565 - 5°F \leq T_{avg} \leq 577.6 + 5°F$

TABLE 15.4.1-7B

	Parameter	Operating Range
	b) Pressurizer pressure	2250 - 60 psia ≤ P_{RCS} ≤ 2250 + 60 psia
	c) Loop flow	≥ 85,000 gpm/loop
	d) T_{UH}	Converted upper internals, T_{COLD} UH
	e) Pressurizer level	Nominal level, automatic control
	f) Accumulator temperature	85°F ≤ T_{ACC} ≤ 120°F
	g) Accumulator pressure	579 psia ≤ P_{ACC} ≤ 664 psia
	h) Accumulator liquid volume	814 ft^3 ≤ V_{ACC} ≤ 886 ft^3
	i) Accumulator fL/D	Current line configuration
	j) Minimum accumulator boron	≥ 2200 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	N/A
	b) Break type	N/A
	c) Break size	N/A
	d) Offsite power	Available or LOOP
	e) Safety injection flow	See Figure 15.4.1-13B.
	f) Safety injection temperature	46°F ≤ SI Temp ≤ 90°F
	g) Safety injection delay	≤ 17 seconds (with offsite power) ≤ 27 seconds (with LOOP)
	h) Containment pressure	See Figure 15.4.1-14B and raw data in Table 15.4.1-5B.
	i) Single failure	All trains operable[(b)]
	j) Control rod drop time	N/A

(a) 44 peripheral locations will not physically be lead power assembly.
(b) Analysis considers loss of one train of pumped ECCS.

TABLE 15.4-8

TIME SEQUENCE OF EVENTS FOR
MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time, sec
Steam Line Rupture @ HZP		
1. With Offsite Power Available	Main steam line ruptures	0.0
	Low steam line pressure setpoint reached	0.6
	SIS flow begins(maximum flow assumed)	2.6
	Steam line isolation occurs	8.6
	Criticality attained	36.5
	Borated water from the RWST reaches the core	~40
	Main feedwater isolation occurs	64.6
	Accumulators inject	79.0
	Peak core heat flux, minimum DNBR occurs	90.5
2. Without Offsite Power Available	Main steam line ruptures	0.0
	Low steam line pressure setpoint reached	0.6
	SIS flow begins (maximum flow assumed)	2.6
	RCPs begin to coast down	3.0
	Steam line isolation occurs	8.6
	Criticality attained	44.4
	Borated water from the RWST reaches the core	~50

TABLE 15.4-8 Sheet 2 of 5

Accident	Event	Time, sec
	Main feedwater isolation occurs	64.6
	Peak core heat flux, minimum DNBR occurs	123.4
	Accumulators inject	129.7
Rupture of Main Feedwater Pipe (Offsite Power Available)	Feedline rupture occurs	20
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	32
	Rods begin to drop	34
	Auxiliary feedwater is started	623
	Pressurizer liquid water relief begins if operator action is not assumed	2053
	Total RCS heat generation (decay heat + pump heat) decreases to auxiliary feedwater heat removal capability	5900
Rupture of Main Feedwater Pipe (Offsite Power Unavailable)	Feedline rupture occurs	20
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	32
	Rods begin to drop	34
	Reactor coolant pump coastdown	36
	Auxiliary feedwater is started	632
	Peak pressurizer level after initial outsurge reached	2091

TABLE 15.4-8

Accident	Event	Time, sec
	Total RCS heat generation decreases to auxiliary feedwater heat removal capability	2200
Steam Line Rupture at Power (0.49 ft2)	Steam line ruptures	0.0
	Peak core heat flux occurs	53.1
Rupture of a Main Feedwater Pipe for Pressurizer Filling	Feedwater line rupture occurs	0.0
(Unblock Pressurizer PORV)	Low-low SG water level reactor trip setpoint (0% NRS) reached in faulted SG	15.9
	Rods begin to drop	17.9
	Turbine trip occurs	18.4
	Steam line check valve closes in loop with faulted SG	18.5
	Reactor coolant pumps begin to coast down (from loss of offsite power)	19.9
	Low steam line pressure setpoint reached in loop with faulted SG	26.7
	Safety injection actuation signal generated	28.7
	Safety injection flow initation occurs	28.8
	Steam line isolation occurs on low steam line pressure safety injection signal	34.7
	Main steam safety valve relief begins	36.1
	PSV steam relief begins	56.0
	AFW flow initiation (390 gpm from a motor-driven AFW pump) occurs to intact SGs not connected to the faulted SG	75.9
	Pressurizer reaches a water-solid condition	501.0
	PSV water relief begins (maximum time for operator action to ensure a PORV is available)	518.8

TABLE 15.4-8 Sheet 4 of 5

Accident	Event	Time, sec
	Operator action to isolate faulted SG to direct all available AFW flow to intact SGs	615.9
	AFW flow initiation (195 gpm from the turbine-driven AFW pump and 195 gpm from the other motor-driven AFW pump) occurs to intact SG connected to the faulted SG	615.9
	AFW flow addition (390 gpm from the turbine-driven AFW pump) occurs to intact SGs not connected to the faulted SG	615.9
Rupture of a Main Feedwater Pipe for Pressurizer Filling	Feedline rupture occurs	0.0
(Isolate Charging Flow and Stop RCP Seal Injection Flow)	Pressurizer backup heater actuation on level deviation	13.5
	Low-low SG water level reactor trip setpoint (0% NRS) reached in faulted SG	15.9
	Rods begin to drop	17.9
	Turbine trip occurs	18.4
	Steam line check valve closes in loop with faulted SG	18.5
	Reactor coolant pumps begin to coast down (from manual trip)	19.9
	Pressurizer PORV steam relief begins	20.9
	Low steam line pressure setpoint reached in loop with faulted SG	26.8
	Safety injection actuation signal generated	28.8
	Safety injection flow initation occurs	28.9
	Steam line isolation occurs on low steam line pressure safety injection signal	34.8
	Main steam safety valve relief begins	40.1
	AFW flow initiation (390 gpm from a motor-driven AFW pump) occurs to intact SGs not connected to the faulted SG	75.9
	Pressurizer reaches a water-solid condition	408.5
	Operator action to isolate faulted SG to direct all available AFW flow to intact SGs	615.9
	AFW flow initiation (195 gpm from the	615.9

TABLE 15.4-8

Accident	Event	Time, sec
	turbine-driven AFW pump and 195 gpm from the other motor-driven AFW pump) occurs to intact SG connected to the faulted SG	
	AFW flow addition (390 gpm from the turbine-driven AFW pump) occurs to intact SGs not connected to the faulted SG	615.9
	Operator action to isolate charging/SI flow	1516.0
	Operator action to stop RCP seal injection flow	2715.9
	Steam bubble forms again in pressurizer	6723.0
	Maximum number of PORV cycles reached (capacity of backup nitrogen accumulators depleted)	7137.6

TABLE 15.4-10

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

	4 Loops Operating Initially 1 Locked Rotor
Maximum RCS pressure, psia	2729[(1)]
Maximum clad average temperature, °F core hot spot	1963
Amount of Zr - H_2O at core hot spot, % by weight	0.53%

(1) The locked rotor transient peak pressure of 2729 psia includes a conservative penalty of 41 psi, determined by Westinghouse in an evaluation to address Nuclear Safety Advisory Letter NSAL-09-2. "Locked Rotor Analysis for Reactor Coolant System Overpressure." May 7, 2009.

TABLE 15.4-11

TYPICAL PARAMETERS USED IN THE VANTAGE 5 RELOAD ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End
Generic Vantage 5 Reload Analysis Values				
Power level, %	102	0.0	102	0.0
Ejected rod worth, % Δk	0.20	0.785	0.21	0.85
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.071	1.30	3.55
Doppler - only power defect, pcm	-955	-954	-829	-788
Trip reactivity, % Δk	4	2	4	2
F_q before rod ejection	2.60	-	2.60	-
F_q after rod ejection	6.70	13	6.50	21.50
Number of operating pumps	4	2	4	2
Generic Vantage 5 Reload Analysis Results				
Maximum fuel pellet average temperature, °F	4154	3509	3812	3408
Maximum fuel center temperature, °F	>4900 [(a)]	4025	>4800 [(a)]	3849
Maximum cladding average temperature, °F	2434	2660	2218	2632
Maximum fuel stored energy, cal/gm	183	149	165	144
Reload Analysis Evaluation Values				
Power level, %	102	0.0	102	0.0
Ejected rod worth, % Δk	0.20	0.785	0.21	0.83
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.071	1.30	3.55
Doppler – only power defect, pcm	-995	-954	-829	-788
Trip reactivity, % Δk	4	2	4	2
F_q before rod ejection	2.60	-	2.60	-
F_q after rod ejection	6.70	13	6.50	22.50
Number of operating pumps	4	2	4	2

(a) Less than 10% fuel pellet melt (at hot spot)

TABLE 15.4-12

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

Action	Time (min)
Identify and isolate ruptured SG	10 min or RETRAN-02W calculated time to reach 38% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	5
Cooldown	Calculated by RETRAN-02W
Operator action time to initiate depressurization	4
Depressurization	Calculated by RETRAN-02W
Operator action time to initiate SI termination	2
SI termination and pressure equalization	Calculated time for SI termination and equalization of RCS and ruptured SG pressures

TABLE 15.4-13A

TIMED SEQUENCE OF EVENTS - SGTR MTO ANALYSIS

Event	Time (sec)
SG Tube Rupture	100
Reactor Trip	274
SI Actuated	380
Turbine Driven AFW Pump Flow Isolated	700
Ruptured SG Steamline Isolation	700
Ruptured SG MDAFW Pump Flow Isolated	820
RCS Cooldown Initiated	1120
RCS Cooldown Terminated	1706
RCS Depressurization Initiated	1946
RCS Depressurization Terminated	2072
SI Terminated	2192
Break Flow Terminated	3475

TABLE 15.4-13B

TIMED SEQUENCE OF EVENTS - SGTR DOSE ANALYSIS

Event	Time (sec)
SG Tube Rupture	100
Reactor Trip	279
SI Actuated	315
Ruptured SG Isolated	953
Ruptured SG PORV Fails Open	953
Ruptured SG PORV Block Valve Closed	2753
RCS Cooldown Initiated	3053
RCS Cooldown Terminated	4424
RCS Depressurization Initiated	4664
RCS Depressurization Terminated	4839
SI Terminated	4959
Break Flow Terminated	5972

TABLE 15.4-14

MASS RELEASE RESULTS - SGTR

	0 - 2 Hrs, lbm	2 - 8 Hrs, lbm
Ruptured SG		
- Condenser	294,500	0
- Atmosphere	140,200	27,000
- Feedwater	288,700	0
Intact SGs		
- Condenser	878,100	0
- Atmosphere	367,100	922,600
- Feedwater	1,476,800	961,700
Break Flow	262,200	0
Flashed Break Flow	18,150	0

Note: The 0-2 hour releases to the condenser and feedwater flows include 100 seconds of steady state operation.

TABLE 15.5-3 (Deleted)

TABLE 15.5-7

TABLE 15.5-7A

BREATHING RATES[a] ASSUMED IN ANALYSIS

Period	Offsite	Onsite
0-8 hrs	3.5×10^{-4}	3.5×10^{-4}
8-24 hrs	1.8×10^{-4}	3.5×10^{-4}
1-30 days	2.3×10^{-4}	3.5×10^{-4}

(a) All breathing rates are expressed in m^3/sec. Values taken from Reference 55.

TABLE 15.5-9

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES
LOSS OF ELECTRICAL LOAD

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]		
- Pre-incident iodine spike	<0.1	2.5
- Accident-initiated iodine spike	<0.1	2.5
30-day Integrated Low Population Zone Dose		
- Pre-incident iodine spike	<0.1	2.5
- Accident-initiated iodine spike	<0.1	2.5
30-day Integrated Control Room Occupancy Dose		
- Pre-incident iodine spike	<0.1	5
- Accident-initiated iodine spike	<0.1	5

Note:

1. The maximum 2-hour EAB dose occurs during the following time period :

   - Pre-incident iodine Spike 0 - 2 hours
   - Accident-Initiated Iodine Spike 8.73 – 10.73 hours

TABLE 15.5-9A
LOSS OF ELECTRICAL LOAD
ANALYSIS ASSUMPTIONS & KEY PARAMETER VALUES

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Primary to Secondary SG tube leakage	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3
Failed/Melted Fuel Percentage	0%
RCS Technical Specification Iodine Levels	Table 15.5-78 (1 µCi/gm DE I-131)
RCS Technical Specification Noble Gas Levels	Table 15.5-78 (270 µCi/gm DE Xe-133)
RCS Equilibrium Iodine Appearance Rates	Table 15.5-79 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentration	Table 15.5-79 (60 µCi/gm DE I-131)
Accident-Initiated Iodine Spike Appearance Rate	500 times TS equilibrium appearance rate
Duration of Accident-Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131 (Table 15.5-78)
Initial and Minimum SG Liquid Mass	92,301 lbm/SG
Time period of tubes uncovered	insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2- 8 hrs
Iodine Partition Coefficient in SGs	100
Iodine Species Released to Environment	97% elemental; 3% organic
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of releases from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation : Initiation Signal/Timing	Control Room is assumed to remain on normal ventilation for duration of the accident.
Control Room Atmospheric Dispersion Factors	Table 15.5-9B

TABLE 15.5-9B
LOSS OF ELECTRICAL LOAD
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release point and receptor	0-2 hr	2-8 hr	8-10.73 hr
MSSVs/10% ADVs to CR NOP Intake (Note 1)	8.12E-04	5.32E-04	5.32E-04
MSSVs/10% ADVs to CR Inleakage (CR Centerline)	2.46E-03	1.59E-03	1.59E-03

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. Thus, the χ/Qs presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit.

- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10% ADVs to the Unit 2 CR intake.
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10% ADV releases for the 0-2 hrs time period, and Unit 2 10% ADV releases for the 2-10.73 hrs time period.

TABLE 15.5-23

SUMMARY OF OFFSITE, CONTROL ROOM & TECHNICAL SUPPORT CENTER DOSES
LOSS OF COOLANT ACCIDENT[3]

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]	5.6	25
30-day Integrated Low Population Zone Dose	1	25
30-day Integrated Control Room Occupancy Dose[2]	3.7 (0.7)	5
30-day Integrated TSC Occupancy Dose[2]	4.1 (1.3)	5

Note:

1. The maximum 2 hr EAB dose is based on the assumed RHR pump seal failure resulting in a 50 gpm leak of sump water occurring at t=24 hr for 30 mins. This release pathway is considered a part of DCPP licensing basis with respect to passive system failure. If this assumed release pathway were not included, the maximum 2 hr dose at the EAB would occur between t=0.5 hrs to t=2.5 hrs (i.e., during the post-LOCA ex-vessel release phase and would be 3.4 rem.

2. The dose presented represents the operator dose due to occupancy. Value shown in parenthesis represents that portion of the total dose reported that is the contribution of direct shine from contained sources/external cloud.

3. The dose received by the operator during transit outside the control room is not a measure of the “habitability” of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post-LOCA access to the control room is addressed separately from the control room occupancy dose and is not included with the control room occupancy dose for the demonstration of control room habitability. As demonstrated in Section 15.5.17.2.4, the dose contribution to the operator during routine access to control room for the duration of the LOCA is minimal (~1% of the occupancy dose).

TABLE 15.5-23A
LOSS OF COOLANT ACCIDENT
Assumptions & Key Parameter Values

Parameter	Value
Core Power Level (105% of the rated power of 3411 MWth)	3580 MWt
Fuel Activity Release Fractions	Per Reg. Guide 1.183 (See Section 15.5.17.2.2.2)
Fuel Release Timing (gap)	Onset: 30 sec
	Duration: 0.5 hr
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr
	Duration: 1.3 hr
Core Activity	Table 15.5-77
Chemical Form of Iodine released from fuel to containment atmosphere	4.85% elemental 95% particulate 0.15% organic
Chemical Form of Iodine Released from RCS and sump water	97% elemental 3% organic
<u>Containment Vacuum/Pressure Relief Parameters</u>	
Minimum Containment Free Volume:	2.550E+06 ft^3
Primary Coolant Tech Spec Activity	Table 15.5-78
Chemical Form of Iodine Released	97% elemental; 3% organic
Maximum RCS flash fraction after LOCA Noble Gases Halogens	 100% 40%
Maximum containment pressure relief line air flow rate	218 actual cubic feet per second (acfs)
Maximum duration of release via containment pressure relief line	13 sec
Release Point	Plant Vent

TABLE 15.5-23A
LOSS OF COOLANT ACCIDENT
Assumptions & Key Parameter Values

Parameter	Value
Containment Leakage Parameters	
Containment Spray Coverage – Injection Spray and Recirculation Spray Modes: Sprayed Volume Unsprayed Volume	82.5% (sprayed fraction) 2.103E+06 ft^3 4.470E+05 ft^3
Minimum mixing flow rate from unsprayed to sprayed region: Before actuation of CFCUs After actuation of CFCUs	 2 unsprayed regions/hr 9.13 unsprayed regions/hr
Minimum duration of mixing via CFCUs	Start = 86 sec End = 30 days
Containment spray in injection mode Initiation time Termination time	 111 sec 3798 sec
Maximum delay between end of injection spray and initiation of recirculation spray	12 min (based on manual operator action)
Containment spray in recirculation mode Initiation time Termination time	 4518 sec 22,518 sec
Long-term Sump Water pH	≥ 7.5
Maximum allowable DF for fission product removal	Elemental Iodine: 200 Others: not applicable
Elemental iodine and particulate spray removal coefficients in sprayed region during both injection spray and recirculation spray modes	See Table 6.2-32
Elemental iodine removal coefficients due to wall deposition	See Table 6.2-32
Particulate removal coefficients in unsprayed region due to gravitational settling	See Table 6.2-32

TABLE 15.5-23A
LOSS OF COOLANT ACCIDENT
Assumptions & Key Parameter Values

Parameter	Value
Containment Leak rate (0-24 hr)	0.1% weight fraction per day
Containment Leak rate (1-30 day)	0.05% weight fraction per day
Containment Leakage Release Point (Unfiltered)	From the worst case release point of the following: Diffuse source via the containment wall Via Plant Vent Via Containment Pen Area GE Via Containment Pen Areas GW & FW
ESF System Environmental Leakage Parameters	
Minimum post-LOCA containment water volume sources	480,015 gal.
Minimum time after LOCA when recirculation is initiated	829 sec
Duration of leakage	30 days
Maximum ECCS fluid temperature after initiation of recirculation	259.9 ºF
Maximum ECCS leak rate (including safety factor of 2)	Unfiltered via plant vent = 240 cc/min Unfiltered via Containment Penetration Areas GE or GW & FW = 12 cc/min
RHR pump seal failure	Filtered(1) via plant vent 50 gpm starting at t = 24 hrs for 30 min
Iodine Airborne Release Fraction	10%
Auxiliary Building ESF Ventilation System filter efficiency	Elemental iodine: 88% Organic iodine: 88%
Refueling Water Storage Tank (RWST) Back-Leakage Parameters	

TABLE 15.5-23A
LOSS OF COOLANT ACCIDENT
Assumptions & Key Parameter Values

Parameter	Value
Earliest initiation time of RWST back-leakage	829 sec
Maximum ECCS / sump water back-leakage rate to RWST (includes safety factor of 2)	2 gpm
RWST back-leakage iodine release fractions	See Table 15.5-23C
RWST back-leakage noble gas, as iodine daughters, release rate from the RWST vent	See Table 15.5-23C
Miscellaneous Equipment Drain Tank (MEDT) Leakage Parameters	
MEDT inflow rate (includes safety factor of 2)	1900 cc/min
MEDT leakage Iodine release fractions	See Table 15.5-23D
MEDT leakage noble gas, as iodine daughters release rate from plant vent	See Table 15.5-23D
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	SI signal generated: 6 sec Non-Affected Unit NOP Intake Isolated: 18 sec Affected Unit NOP Intake Isolated and CRVS Mode 4 in full Operation: 44.2 sec
Bounding Control Room Atmospheric Dispersion Factors for LOCA	Table 15.5-23B

Note:
Releases from the RHR Pump Seal failure are filtered for CR dose evaluation and filtered for Site Boundary Dose Evaluation.

TABLE 15.5-23B
LOSS OF COOLANT ACCIDENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location/Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
Control Room Normal Intakes					
Plant Vent Release					
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----
Containment Penetration Areas					
- Affected Unit Intake	6.60E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	2.08E-03	-----	-----	-----	-----
Control Room Infiltration					
Plant Vent	1.25E-03	9.08E-04	3.61E-04	3.65E-04	3.17E-04
Containment Penetration Areas	3.09E-03	1.83E-03	7.22E-04	7.13E-04	6.50E-04
RWST Vent	1.05E-03	5.55E-04	2.12E-04	2.12E-04	1.72E-04
Control Room Pressurization Intake					
Plant Vent	5.55E-05	3.68E-05	1.36E-05	1.38E-05	1.11E-05
Containment Penetration Areas	6.00E-05	3.98E-05	1.63E-05	1.37E-05	1.10E-05
RWST Vent	4.75E-05	3.23E-05	1.25E-05	1.14E-05	8.73E-06

Note 1: Release from the Containment penetration areas (i.e., areas GE or GW & FW): applicable to containment leakage and ESF system leakage that occurs in the Containment Penetration Area

Note 2: Release from Plant Vent: applicable to ESF system leakage that occurs in the Auxiliary building, MEDT releases, RHR Pump Seal Failure Release and Containment Vacuum/Pressure Relief Line Release

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the Plant Vent to the CR Normal intakes (affected and non-affected unit) are based on Unit 1 releases.
- Releases from the containment penetration areas to the CR Normal intakes (affected and non-affected unit) are based on Unit 2 GE area releases.
- Releases from the Plant Vent to the CR Center (i.e., for CR Inleakage) are based on Unit 1 releases.
- Releases from the containment penetration areas to the CR Center are based on Unit 2 GE area releases for the 0-24 hour period and on the Unit 1 GW/FW area for the 1-30 day time period.
- Releases from the RWST vent to the CR Center are based on Unit 2 releases.
- Releases from the Plant Vent to the CR pressurization intakes are based on Unit 1 releases to the U2 CR intake.
- Releases from the containment penetration areas to the CR pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period.
- Releases from the RWST vent to the CR pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

TABLE 15.5-23C
LOSS OF COOLANT ACCIDENT
RWST Iodine Releases Fraction and Gas Venting Rate to Atmosphere

From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Sec	Sec	Fraction $I_{released}$ / $I_{entering}$	Fraction V_{rwst} / day
829	7200	9.451E-05	2.610E+00
7200	28,800	6.357E-05	7.291E-01
28,800	86,400	8.796E-06	7.375E-02
86,400	345,600	4.560E-07	9.955E-03
345,600	471,600	6.347E-07	1.311E-02
471,600	1,011,600	8.231E-07	1.489E-02
1,011,600	2,048,400	1.114E-06	1.547E-02
2,048,400	2,592,000	1.483E-06	1.702E-02

Where:

$I_{released}$ = Total Iodine mass released to atmosphere during specified time interval, gm
$I_{entering}$ = Total Iodine mass entering to the RWST during specified time interval, gm
Frac. V_{rwst} = Rate of Fractional RWST gas volume vented during specified time interval

TABLE 15.5-23D
LOSS OF COOLANT ACCIDENT
MEDT Iodine Release Fraction and Gaseous Venting Rate to Atmosphere

From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Sec	Sec	Fraction $I_{released}$ / $I_{entering}$	Fraction V_{MEDT} / day
829	7200	4.521E-07	5.024E+00
7200	28,800	1.386E-08	3.024E-02
28,800	86,400	2.362E-07	3.324E-01
86,400	345,600	3.950E-07	6.497E+00
345,600	471,600	1.236E-02 (Note 2)	(Note 1)
471,600	1,011,600	2.028E-02 (Note 2)	(Note 1)
1,011,600	2,048,400	2.390E-02 (Note 2)	(Note 1)
2,048,400	2,592,000	2.166E-02 (Note 2)	(Note 1)

Where:
$I_{released}$ = Total Iodine mass released to atmosphere during specified time interval, gm
$I_{entering}$ = Total Iodine mass entering to the MEDT during specified time interval, gm
Frac. V_{MEDT} = Rate of Fractional MEDT gas volume vented during specified time interval

Note 1: After the MEDT overflows at t = 183,289 sec, the gas venting rates are 2640 cfm from the EDRT room, and 1760 cfm from the U1/U2 Pipe Tunnels (i.e., the exhaust ventilation rate from the respective rooms + 10%). To be consistent with the methodology used to determine the iodine release fractions after spillover, the noble gases generated by decay of iodines in the tank and spilled liquid after overflow occurs, should also be released instantaneously to the environment without hold-up.

Note 2: The room ventilation flows addressed in Note 1 (utilized as clean in-coming air) are incorporated into the determination of the iodine equilibrium concentration in the EDRT room and U1/U2 Pipe Tunnels air space, respectively. The bounding iodine release fractions presented above after spillover assume instantaneous release of iodines to the environment without hold-up in the room.

TABLE 15.5-23E
LOSS OF COOLANT ACCIDENT
TSC Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location/Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
<u>TSC Normal Intakes</u>					
Plant Vent Release	5.47E-04	-----	-----	-----	-----
Containment Penetration Areas	1.71E-03	-----	-----	-----	-----
RWST Vent	3.52E-04	-----	-----	-----	-----
<u>TSC Infiltration</u>					
Plant Vent	5.41E-04	2.09E-04	9.67E-05	7.95E-05	6.43E-05
Containment Penetration Areas	1.76E-03	7.16E-04	3.01E-04	2.84E-04	2.28E-04
RWST Vent	3.61E-04	1.48E-04	6.30E-05	5.80E-05	4.69E-05
<u>CR/TSC Pressurization Intake</u>					
Plant Vent	-----	3.68E-05	1.36E-05	1.38E-05	1.11E-05
Containment Penetration Areas	-----	3.98E-05	1.63E-05	1.37E-05	1.10E-05
RWST Vent	-----	2.93E-05	1.13E-05	1.08E-05	8.50E-06

<u>Note 1</u>: Release from the Containment penetration areas (i.e., areas GE or GW & FW): applicable to containment leakage and ESF system leakage that occurs in the Containment Penetration Area

<u>Note 2</u>: Release from Plant Vent: applicable to ESF system leakage that occurs in the Auxiliary building, MEDT releases, RHR Pump Seal Failure Release and Containment Vacuum/Pressure Relief Line Release

<u>Note 3</u>: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the Plant Vent to the TSC Normal intake are based on Unit 2 releases.
- Releases from the containment penetration areas to the TSC Normal intakes are based on Unit 2 GW/FW area releases.
- Releases from the RWST vent to the TSC Normal intake are based on Unit 2 releases.
- Releases from the Plant vent to the TSC Center (i.e., for TSC Inleakage) are based on Unit 2 releases.
- Releases from the containment penetration areas to the TSC Center are based on Unit 2 GW/FW area releases.
- Releases from the RWST vent to the TSC Center are based on Unit 2 releases.
- Releases from the Plant Vent to the CR/TSC pressurization intakes are based on Unit 1 releases to the U2 CR intake.
- Releases from the containment penetration areas to the CR/TSC pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period.
- Releases from the RWST vent to the CR/TSC pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

TABLE 15.5-33

ESTIMATED POST LOCA EXPOSURE TO CONTROL ROOM PERSONNEL
DURING EGRESS-INGRESS (HISTORICAL)

	LOCA		
Radiation Source	Gamma Exposure, rem	Beta Exposure, rem	Thyroid Exposure, rem
1. Not Used			
2. Not Used			
3. Not Used			
4. Radiation from airborne fission products in the containment leakage plume to control room personnel during egress-ingress	0.0066	0.0243	4.72
5. Direct radiation from fission products in the containment structure to control room personnel during egress-ingress (53 5-minute trips)	0.022	0	0

TABLE 15.5-34

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES
MAIN STEAM LINE BREAK

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose		
- Pre-Incident Iodine Spike	0.1	25
- Accident-Initiated Iodine Spike	0.7	2.5
30-day Integrated Low Population Zone Dose		
- Pre-Incident Iodine Spike	<0.1	25
- Accident-Initiated Iodine Spike	0.2	2.5
30-day Integrated Control Room Occupancy Dose		
- Pre-Incident Iodine Spike	2.0	5
- Accident-Initiated Iodine Spike	4.1	5

Note:

1. The maximum 2-hour EAB dose occurs during the following time period:

- Pre-incident Iodine Spike	0 - 2 hours
- Accident-Initiated Iodine Spike	7.6 – 9.6 hours

TABLE 15.5-34A
MAIN STEAM LINE BREAK
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446.486 lbm
Leak rate to Faulted Steam Generator	0.75 gpm (conservative assumption); leakage density 62.4 lbm/ft^3
Leak rate to Intact Steam Generators	0 gpm (all leakage assumed into faulted SG)
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Conc.	Table 15.5-78 (1 µCi/gm DE I-131)
RCS Tech Spec Noble Gas Conc.	Table 15.5-78 (270 µCi/gm DE Xe-133)
RCS Equilibrium. Iodine Appearance Rates	Table 15.5-79 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentrations	Table 15.5-79 (60 µCi/gm DE I-131)
Accident-Initiated Iodine Spike Appearance Rate	500 times equilibrium appearance rate
Duration of Accident- Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	Table 15.5-78 (0.1 µCi/gm DE I-131)
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Fraction of Iodine Released from Faulted SG	1.0 (Released to Environ without holdup)
Fraction of Noble Gas Released from Faulted SG	1.0 (Released to Environ without holdup)

TABLE 15.5-34A
MAIN STEAM LINE BREAK
Analysis Assumptions & Key Parameter Values

Parameter	Value
Liquid mass in each SG	Faulted: 182,544 lbm (max.) Intact: 92,301 lbm (min. and initial)
Release Rate of SG liquid activity from Faulted SG	Dryout within 10 seconds
Time period when tubes not totally submerged (intact SG)	Insignificant
Steam Releases from intact SGs	0-2 hrs: 384,000 lbm 2-8 hrs: 893,000 lbm 8-10.73 hrs: Same release rate as that for 2-8 hrs
Iodine Partition Coefficient in Intact SG	100 (SGs fully covered)
Termination of release (0.75 gpm leak): Faulted SG	30 hrs when RCS reaches 212 °F
Termination of release from Intact SG	10.73 hours
Release Point: Faulted SG	Outside containment, at the steam line break location
Release Point: Intact SG	MSSVs/10% ADVs
<u>CR Emergency Ventilation: Initiation Signal/Timing</u>	
Initiation (signal)	SIS
Unaffected Unit CRVS inlet damper fully closed	Within 12.6 seconds
Affected Unit CRVS inlet dampers fully closed	Within 38.8 seconds
<u>Control Room Atmospheric Dispersion Factors</u>	Table 15.5-34B

TABLE 15.5-34B
MAIN STEAM LINE BREAK
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Receptor – Release Point	0-2 hr	2-8 hr	8-10.73 hr	10.73-30 hr
CR NOP Intake – Faulted SG (Break Location)	Note 1			
CR NOP Intake – Intact SG (MSSVs/10% ADVs) – Note 2	8.12E-04			
CR InLeakage – Faulted SG (Break Location)	1.14E-02	7.22E-03	3.00E-03	3.00E-03
CR InLeakage – Intact SG (MSSVs/10% ADVs)	2.46E-03	1.59E-03	1.59E-03	-----
CR Emergency Intake & Bypass Faulted SG (Break Location)	6.85E-05	4.70E-05	1.85E-05	1.85E-05
CR Emergency Intake & Bypass - Intact SG (MSSVs/10% ADVs)	1.40E-05	9.40E-06	9.40E-06	-----

Notes:

1. ARCON96 based χ/Qs are not applicable for these cases given that the horizontal distance from the source to the receptor is 1.5 meters (which is much less than the 10 meters required by ARCON96 methodology).

2. Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. Thus, the χ/Qs presented reflect those applicable to the CR intake of the unaffected unit.

3. The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:
   - Releases from the MSSVs/10% ADVs of the Intact SG to the CR Normal intake of the non-affected unit are based on Unit 1 10% ADVs releases to the Unit 2 CR intake.
   - Releases from the MSL break point of the faulted SG to the CR Center (i.e., for CR Inleakage) are based on Unit 1 for the 0-2 hour time period and Unit 2 releases for the 2-30 hour time period.
   - Releases from the MSSVs/10% ADVs of the intact SG to the CR Center are based on Unit 1 10% ADV releases for the 0-2 hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.
   - Releases from the MSL break point of the faulted SG to the CR pressurization intakes are based on Unit 1 releases to the Unit 2 CR intake for the 0-2 hour time period and Unit 2 releases to the Unit 1 CR intake for the 2-30 hour time period.
   - Releases from the MSSVs/10% ADVs of the Intact SG to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10% ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

TABLE 15.5-42

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES
LOCKED ROTOR ACCIDENT

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]	0.5	2.5
30-day Integrated Low Population Zone Dose	0.1	2.5
30-day Integrated Control Room Occupancy Dose	1.7	5

Note:

1. The maximum 2-hour EAB dose occurs between 8.73 – 10.73 hours

TABLE 15.5-42A
LOCKED ROTOR ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Primary to Secondary SG tube leakage	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3)
Melted Fuel Percentage	0%
Failed Fuel Percentage	10%
Equilibrium Core Activity	Table 15.5-77
Radial Peaking Factor	1.65
Fraction of Core Inventory in Fuel Gap	I-131: 8% I-132: 23% Kr-85: 35% Other Noble Gases: 4% Other Halogens: 5% Alkali Metals: 46%
Isotopic Inventory in Fuel Gap	Table 15.5-80
Iodine Chemical Form in Gap	4.85% elemental
	95% particulate
	0.15% organic
Secondary Side Parameters	
Initial and Minimum SG Liquid Mass	92,301 lbm/SG
Iodine Species Released to Environment	97% elemental; 3% organic
Time period when tubes not totally submerged	insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs

TABLE 15.5-42A
LOCKED ROTOR ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Iodine Partition Coefficient in SGs	100
Particulate Carry-Over Fraction in SGs	0.0005 by weight
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of releases from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation: Initiation Signal/Timing	
	Control Room is assumed to remain on normal ventilation (CRVS Mode 1) for duration of the accident.
Control Room Atmospheric Dispersion Factors	Table 15.5-42B

TABLE 15.5-42B
LOCKED ROTOR ACCIDENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release point and receptor	0-2 hr	2-8 hr	8-10.73 hr
MSSVs/10% ADVs to CR NOP Intake (Note 1)	8.12E-04	5.32E-04	5.32E-04
MSSVs/10% ADVs to CR In-leakage (CR Centerline)	2.46E-03	1.59E-03	1.59E-03

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. Thus, the χ/Qs presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs to the Unit 2 CR intake.
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10%ADV releases for the 0-2 hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.

TABLE 15.5-47

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES
FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING OR CONTAINMENT

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]		
- FHA in Fuel Handling Building	1.0	6.3
- FHA in Containment	1.0	6.3
30-day Integrated Low Population Zone Dose		
- FHA in Fuel Handling Building	0.1	6.3
- FHA in Containment	0.1	6.3
30-day Integrated Control Room Occupancy Dose		
- FHA in Fuel Handling Building	1.0	5
- FHA in Containment	4.3	5

Note:

1. The maximum 2-hour EAB dose occurs between 0 – 2 hours.

TABLE 15.5-47A
FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING OR CONTAINMENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Number of Damaged Fuel Assemblies	1
Total Number of Fuel Assemblies	264
Decay Time Prior to Fuel Movement	72 hours
Radial Peaking Factor	1.65
Fraction of Core Inventory in gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%)
Isotopic Inventory in Fuel Gap (Decayed 72 hours)	Table 15.5-47C
Iodine form of gap release before scrubbing	99.85% elemental
	0.15% Organic
Iodine form of gap release after scrubbing	57% elemental
	43% Organic
Scrubbing Decontamination Factors	Iodine (200, effective)
	Noble Gas (1)
	Particulates (∞)
Rate of Release from Fuel	Puff
Environmental Release Rate	All airborne activity released within a 2 hour period (or less if the ventilation system promotes a faster release rate)

TABLE 15.5-47A
FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING OR CONTAINMENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Environmental Release Points and Rates	
Accident in SFP in the FHB – Release flow rates	-Plant Vent – 46,000 cfm FHB Outleakage -Egress-ingress locations – 30 cfm -Miscellaneous gaps/openings – 470 cfm
Minimum free volume in FHB above SFP	317,000 ft^3
Accident in Containment - Release flow rates	-Open Equipment Hatch – All airborne activity released in 2 hrs
Minimum Free Volume in Containment above Operating Floor	2,013,000 ft^3
CR Emergency Ventilation: Initiation Signal/Timing	
Signal(s) available to switch the Control Room Ventilation System (CRVS) from normal operation (NOP) Ventilation (Mode 1) to Pressurized Filtered Ventilation (Mode 4) following a FHA	Radiation signals from gamma sensitive intake monitors that initiate closure of the CR normal intake dampers and switch the Control Room Ventilation System from normal operation Ventilation Mode 1 to Pressurized Filtered Ventilation Mode
Radiation Monitor Analytical Safety Limit	1 mR/hr
Delay time for CRVS Mode 4 operation, including monitor response, signal processing, and damper closure time	32 seconds (see below)
Radiation Monitor Response Time	20 seconds (conservative assumption) - (Refer to Section 15.5.22.2.4)
Radiation monitor signal processing time	2 seconds
NOP Ventilation Damper Closure Time	10 seconds
Bounding Control Room Atmospheric Dispersion Factors for FHA	Table 15.5-47B

TABLE 15.5-47B
FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING OR CONTAINMENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location/Receptor	0-22 sec	22 sec–2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
Control Room Normal Intakes						
Containment Hatch Release						
- Affected Unit Intake	2.48E-02	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.67E-03	-----	-----	-----	-----	-----
Plant Vent Release						
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----	-----
FHB Out-leakage points						
- Affected Unit Intake	6.68E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.69E-03	-----	-----	-----	-----	-----
Control Room Infiltration						
Containment Hatch Release	5.09E-03	5.09E-03	-----	-----	-----	-----
Plant Vent	1.25E-03	1.25E-03	-----	-----	-----	-----
FHB Out-leakage points	3.61E-03	3.61E-03	-----	-----	-----	-----
Control Room Pressurization Intake						
Containment Hatch Release	-----	6.15E-05	-----	-----	-----	-----
Plant Vent	-----	5.55E-05	-----	-----	-----	-----
FHB Out-leakage points	-----	6.13E-05	-----	-----	-----	-----

Note 1: Release from the Containment Hatch: applicable to FHA in Containment

Note 2: Release from Plant Vent / FHB Out-leakage: applicable to FHA in FHB

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the Containment Hatch to the CR Normal intake of the affected and non-affected unit are based on Unit 2 and Unit 1 releases, respectively.
- Releases from the Plant Vent to the CR Normal intake of the affected and non-affected unit are based on Unit 1 releases
- Releases from the FHB to the CR Normal intake of the affected and non-affected unit are based on Unit 1 releases
- Releases from the Containment Hatch to the CR Center (i.e., for CR Inleakage) are based on Unit 2 releases.
- Releases from the Plant Vent to the CR Center are based on Unit 1 releases.
- Releases from the FHB to the CR Center are based on Unit 2 releases.
- Releases from the Containment Hatch to the CR pressurization intakes are based on Unit 2 releases to the U1 CR intake
- Releases from the Plant Vent to the CR pressurization intakes are based on Unit 1 releases to the Unit 2 CR intake
- Releases from the FHB to the CR pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

TABLE 15.5-47C
ISOTOPIC GAP ACTIVITY – FUEL HANDLING ACCIDENT
Single Fuel Assembly (Decayed 72 hours)

Nuclide	Activity Per Assembly (Ci)	Gap Fraction	Gap Activity per Assembly (w/o Peaking Factor)
I-129	2.07E-02	0.05	1.04E-03
I-130	3.29E+02	0.05	1.65E+01
I-131	4.09E+05	0.08	3.27E+04
I-132	3.99E+05	0.23	9.18E+04
I-133	9.73E+04	0.05	4.87E+03
I-135	5.01E+02	0.05	2.51E+01
KR-83M	2.51E-04	0.04	1.00E-05
KR-85	5.75E+03	0.35	2.01E+03
KR-85M	1.77E+00	0.04	7.08E-02
KR-88	7.73E-03	0.04	3.09E-04
XE-127	9.64E-02	0.04	3.86E-03
XE-129M	5.28E+01	0.04	2.11E+00
XE-131M	6.96E+03	0.04	2.78E+02
XE-133	8.31E+05	0.04	3.32E+04
XE-133M	1.88E+04	0.04	7.52E+02
XE-135	1.07E+04	0.04	4.28E+02
XE-135M	8.18E+01	0.04	3.27E+00
CS-132	2.16E+01	0.46	9.94E+00
CS-134	1.25E+05	0.46	5.75E+04
CS-134M	1.04E-03	0.46	4.78E-04
CS-135	3.01E-01	0.46	1.38E-01
CS-136	3.10E+04	0.46	1.43E+04
CS-137	7.10E+04	0.46	3.27E+04

RB-86	1.16E+03	0.46	5.34E+02
RB-87	1.37E-05	0.46	6.30E-06
RB-88	8.63E-03	0.46	3.97E-03

TABLE 15.5-52

SUMMARY OF OFFSITE DOSES AND CONTROL ROOM DOSES
CONTROL ROD EJECTION ACCIDENT

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]		
- Containment Release	0.7	6.3
- Secondary Side Release	0.7	6.3
30-day Integrated Low Population Zone Dose		
- Containment Release	0.3	6.3
- Secondary Side Release	0.2	6.3
30-day Integrated Control Room Occupancy Dose		
- Containment Release	3.4	5
- Secondary Side Release	0.5	5

Note:

1. The maximum 2-hour EAB dose occurs between 0 – 2 hours.

TABLE 15.5-52A
CONTROL ROD EJECTION ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameters	Value
Containment Leakage Pathway	
Power Level	3580 MWt
Free Volume	2.550E+06 ft^3
Containment leak rate (0 -24 hr)	0.1% vol. fraction per day
Containment leak rate(1-30 day)	0.05% vol. fraction per day
Failed Fuel Percentage	10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	0%
Chemical Form of Iodine in Failed fuel	4.85% elemental 95% particulate 0.15% organic
Radial Peaking Factor	1.65
Core Activity Release Timing	Puff
Form of Failed Iodine in the Containment Atmosphere	97% elemental 3% organic
Equilibrium Core Activity	Table 15.5-77
Termination of Containment Release	30 days
Environmental Release Point	Same as LOCA Containment Leakage pathway
Secondary Side Pathway	
Reactor Coolant Mass	446,486 lbm

TABLE 15.5-52A
CONTROL ROD EJECTION ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameters	Value
Primary-to-Secondary Leak rate	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3
Failed Fuel Percentage	Same as containment leakage pathway
Percentage of Core Inventory in Fuel Gap	Same as containment leakage pathway
Minimum Post-Accident SG Liquid Mass	92,301 lbm / SG
Iodine Species released to Environment	97% elemental 3% organic
Time period when tubes not totally submerged	Insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs.
Iodine Partition Coefficient in SGs	100
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of Release from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	300 sec (SIS Generated) 312 sec (Non-Affected Unit NOP Intake fully Closed) 338.2 sec (Affected Unit NOP Intake fully Closed with full Mode 4 Emergency Ventilation Operation).
Control Room Atmospheric Dispersion Factors	Table 15.5-52B

TABLE 15.5-52B
CONTROL ROD EJECTION ACCIDENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-2hr	2-8hr	8-10.73hr	10.73-24hr	24-96hr	96-720hr
CR Normal Intakes						
Containment leakage						
- Affected Unit Intake	6.60E-03	-----		-----	-----	-----
- Non-Affected Unit Intake	2.08E-03	-----		-----	-----	-----
*MSSVs/*10% ADVs						
- Affected Unit Intake	Note 3	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	8.12E-04	-----	-----	-----	-----	-----
CR Infiltration						
Containment leakage	3.09E-03	1.83E-03	7.22E-04	7.22E-04	7.13E-04	6.50E-04
MSSVs/10% ADVs	2.46E-03	1.59E-03	1.59E-03	-----	-----	-----
CR Pressurization Intake						
Containment leakage	6.00E-05	3.98E-05	1.63E-05	1.63E-05	1.37E-05	1.10E-05
MSSVs/10% ADVs	1.40E-05	9.40E-06	9.40E-06	-----	-----	-----

Note 1: Containment leakage: Used for Containment release scenario; based on Containment penetration area release point.

Note 2: MSSV /10% ADVs: Used for Secondary System Release Scenario;

Note 3: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit.

Note 4: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the containment penetration areas to the CR Normal intake of the affected and non-affected units are based on Unit 2 GE area releases.
- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10% ADVs to the Unit 2 CR intake.
- Releases from the containment penetration areas to the CR Center are based on Unit 2 GE area releases for the 0-24 hour period and on the Unit 1 GW/FW area for the 1-30 day time period.
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10% ADV releases for the 0-2 hrs time period, and Unit 2 10% ADV releases for the 2-10.73 hrs time period.
- Releases from the containment penetration areas to the CR pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period.
- Releases from the MSSVs/10% ADVs to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10% ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

TABLE 15.5-53 (Deleted)

TABLE 15.5-56 (Deleted)

TABLE 15.5-57 (Deleted)

TABLE 15.5-62
(DELETED)

TABLE 15.5-64

SUMMARY OF OFFSITE DOSES AND CONTROL ROOM DOSES
STEAM GENERATOR TUBE RUPTURE

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]		
- Pre-incident iodine Spike	1.3	25
- Accident-Initiated Iodine Spike	0.7	2.5
30-day Integrated Low Population Zone Dose		
- Pre-incident iodine Spike	0.1	25
- Accident-Initiated Iodine Spike	<0.1	2.5
30-day Integrated Control Room Occupancy Dose		
- Pre-incident iodine Spike	0.6	5
- Accident-Initiated Iodine Spike	0.3	5

Note:

1. The maximum 2-hour EAB dose occurs between 0 – 2 hours.

TABLE 15.5-64A
STEAM GENERATOR TUBE RUPTURE ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Time of Reactor Trip	179.0 sec
Time of isolation of stuck-open 10% ADV on the Ruptured SG	2653 sec
Termination of Break Flow from Ruptured SG that flashes	3402 sec
Termination of Break Flow from Ruptured SG	5872 sec
Time of manual depressurization of the Ruptured SG	2 hours
Break Flow to Ruptured Steam Generator that flashes	Table 15.5-64C, Column "A"
Break Flow to Ruptured Steam Generator that does not flash	Table 15.5-64C, Column "B"
Tube Leakage rate to Intact Steam Generators	0.75 gpm (total for all 4 SGs; conservatively assumed for 3 intact SGs); leakage density 62.4 lbm/ft^3
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Concentration	1 μCi/gm DE I-131 (Table 15.5-78)
RCS Tech Spec Noble Gas Concentration	270 μCi/gm DE Xe-133 (Table 15.5-78)

TABLE 15.5-64A
STEAM GENERATOR TUBE RUPTURE ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
RCS Equilibrium Iodine Appearance Rates	Table 15.5-79 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentration	60 µCi/gm DE I-131 (Table 15.5-79)
Accident-Initiated Iodine Spike Appearance Rate	335 times TS equilibrium appearance rate
Duration of Accident-Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131 (Table 15.5-78)
Secondary System Release Parameters	
Initial SG liquid mass	89,707 lbm / SG
Iodine Species released to Environment	97% elemental; 3% organic
Steam flow rate to condenser from Ruptured SG before trip	63,000 lbm/min
Steam flow rate to condenser from intact SGs before trip	189,000 lbm/min
Partition Factor in Main Condenser	0.01 (elemental iodine) 1 (organic iodine and noble gases)
Steam Releases from Ruptured SG	Table 15.5-64C, Column “C”
Steam Releases from intact SG	Table 15.5-64C, Column “D”
Post-accident minimum SG liquid mass for Ruptured SG	89,707 lbm

TABLE 15.5-64A
STEAM GENERATOR TUBE RUPTURE ACCIDENT
Analysis Assumptions & Key Parameter Values

Parameter	Value
Post-accident minimum SG liquid mass for intact SGs	89,707 lbm per SG
Time period when tubes not totally submerged (intact SG)	insignificant
Fraction of Iodine Released (flashed portion)	1.0 (Released without holdup)
Fraction of Noble Gas Released from all SGs	1.0 (Released without holdup)
Iodine Partition Coefficient	100
Termination of Release from intact SG	10.73 hrs
Environmental Release Points	Plant Vent : 0 – 179 sec MSSVs/10% ADVs:179 sec – 10.73 hr
CR emergency Ventilation : Initiation Signal/Timing	
Initiation time (signal)	SIS: 219 sec Unaffected Unit inlet damper closed: 231 sec Affected Unit inlet damper closed: 257.2 sec
Control Room Atmospheric Dispersion Factors	Table 15.5-64B

TABLE 15.5-64B
STEAM GENERATOR TUBE RUPTURE ACCIDENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-179 s	179-257.2 s	257.2 s- 2 h	2-8 hr	8-10.73 hr
CR Normal Intakes					
- Plant Vent	1.29E-03	-----	-----	-----	-----
- MSSVs/10% ADVs (Note 1)	-----	8.12E-04	-----	-----	-----
CR Infiltration					
- Plant Vent	1.25E-03	-----	-----	-----	-----
- MSSVs/10% ADVs	-----	2.46E-03	2.46E-03	1.47E-03[2]	1.47E-03[2]
CR Pressurization Intake					
- MSSVs/10% ADVs	-----	-----	1.40E-05	9.40E-06[2]	9.40E-06[2]

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. Thus, the χ/Qs presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: Since the 0-2 hour activity intake following a SGTR controls the 30-day integrated dose, the SGTR dose model utilizes a simplified model with respect to selection of the χ/Q values for the 2-10.73 hr time period. Specifically, the bounding χ/Q value is selected for the release point / receptor for the 0-2 hr time period, but unlike the dose models used for the other accidents, the χ/Q values for time periods beyond t=2hr are not switched to the other unit if they display higher values.

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the Plant Vent to the CR Normal intakes (occurs prior to reactor trip) are based on Unit 1 releases to both Unit 1 and Unit 2 CR normal intakes (i.e., an average χ/Q; applied to the combined Unit 1 and Unit 2 CR normal intake flow).
- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10% ADVs releases to the Unit 2 CR intake.
- Releases from the Plant Vent to the CR Center (i.e., CR inleakage) are based on Unit 1 releases.
- Releases from the MSSVs/10% ADVs to the CR Center are based on Unit 1 10% ADV releases. (Note that the χ/Q value for the Unit 2 10% ADV to CR Center during the 2-10.73 hr period is greater than the listed value. However, the dose consequences associated with the SGTR is dominated by the 0-2 hour release, and the 0-2hr χ/Q for Unit 1 is bounding).
- Releases from the MSSVs/10% to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10% ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

TABLE 15.5-64C
STEAM GENERATOR TUBE RUPTURE ACCIDENT
Break Flows and Steam Releases

	Break Flow and Steam Release within each Time Interval			
	A	B	C	D
Time from Break (sec)	Flashed Break Flow (lbm)	Un-flashed Break Flow (lbm)	Ruptured SG Steam Releases (lbm)	Intact SGs Steam Releases (lbm)
0	167	8422	187822	563100
179	221	30003	10527	42565
853	12121	90754	113657	118
2653	135	15906	0	146
2953	779	23177	0	85467
3402	0	45026	0	97164
4324	0	16870	0	9237
4739	0	23892	0	29103
5872	0	0	0	103300
7200	0	0	27000	1,342,400
38628	0	0	0	0

Note: Data in row for T=0 is applicable to time interval between T=0 sec to T=179 sec (typ)

TABLE 15.5-77

DCPP EQUILIBRIUM CORE INVENTORY (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent, and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
AG-110	2.67E+07	IN-125	8.46E+05	**SB-125**	9.63E+05
AG-110M	6.92E+05	IN-127	1.86E+06	**SB-127**	9.14E+06
AG-111	7.09E+06	IN-129	3.55E+06	**SB-129**	3.25E+07
AG-111M	7.09E+06	IN-131	1.09E+06	**SB-130**	1.08E+07
AG-112	3.16E+06	IN-132	2.85E+05	**SB-130M**	4.38E+07
AG-115	6.21E+05	**KR-83M**	1.14E+07	**SB-131**	7.67E+07
AG-115M	2.60E+05	**KR-85**	1.11E+06	**SB-132**	4.70E+07
AG-239	4.90E-01	**KR-85M**	2.33E+07	**SB-132M**	4.37E+07
AG-241	1.32E+04	**KR-87**	4.65E+07	**SB-133**	6.32E+07
AG-242	9.40E+06	**KR-88**	6.43E+07	SB-134	1.14E+07
AG-242M	8.54E+02	**KR-89**	7.94E+07	SB-135	5.46E+06
AG-243	5.28E+03	**KR-90**	8.48E+07	SB-136	8.63E+05
AG-244	3.79E+07	KR-91	5.83E+07	**SE-83**	5.38E+06
AG-245	1.12E-03	KR-92	3.12E+07	SE-83M	5.65E+06
AS-76	3.05E+03	KR-93	1.07E+07	SE-84	2.04E+07
AS-83	7.02E+06	KR-94	5.00E+06	SE-85	9.54E+06
BA-137M	1.30E+07	**LA-140**	1.85E+08	SE-87	1.32E+07
BA-139	1.76E+08	**LA-141**	1.61E+08	SE-88	7.15E+06
BA-140	1.78E+08	**LA-142**	1.57E+08	SE-89	2.49E+06
BA-141	1.59E+08	**LA-143**	1.48E+08	**SM-153**	6.04E+07
BA-142	1.51E+08	LA-144	1.31E+08	SM-155	4.30E+06
BA-143	1.29E+08	**MO-99**	1.84E+08	SM-156	2.66E+06
BA-144	9.93E+07	**MO-101**	1.69E+08	SM-157	1.70E+06
BR-82	4.44E+05	MO-103	1.62E+08	**SN-121**	8.43E+05
BR-82M	3.88E+05	MO-104	1.33E+08	**SN-123**	6.43E+04
BR-83	1.13E+07	MO-105	9.97E+07	**SN-125**	5.25E+05
BR-84	2.10E+07	MO-106	5.86E+07	SN-125M	1.58E+06
BR-85	2.31E+07	NB-101	1.59E+08	**SN-127**	3.69E+06
BR-87	3.67E+07	NB-104	5.10E+07	SN-127M	4.95E+06
BR-88	3.52E+07	**NB-95**	1.66E+08	SN-129	1.28E+07
BR-89	2.45E+07	**NB-95M**	1.89E+06	SN-129M	1.17E+07
BR-90	1.35E+07	**NB-97**	1.59E+08	SN-130	3.28E+07
CD-115	9.17E+05	**NB-97M**	1.50E+08	SN-131	2.83E+07

TABLE 15.5-77

DCPP EQUILIBRIUM CORE INVENTORY (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent, and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
CD-115M	4.43E+04	NB-99	1.07E+08	SN-132	2.28E+07
CD-121	7.67E+05	NB-99M	7.35E+07	SN-133	6.21E+06
CE-141	1.63E+08	**ND-147**	6.59E+07	SN-134	1.07E+06
CE-143	1.50E+08	ND-149	3.90E+07	**SR-89**	9.05E+07
CE-144	1.26E+08	ND-151	2.08E+07	**SR-90**	9.67E+06
CE-147	6.18E+07	NP-238	6.20E+07	**SR-91**	1.13E+08
CF-249	2.46E-02	**NP-239**	2.16E+09	**SR-92**	1.22E+08
CM-241	3.54E+00	NP-240	6.23E+06	**SR-93**	1.39E+08
CM-242	5.88E+06	**PD-109**	4.73E+07	**SR-94**	1.39E+08
CM-244	1.31E+06	PD-109M	3.12E+05	SR-95	1.25E+08
CM-245	1.26E+02	PD-111	7.09E+06	SR-97	4.68E+07
CO-58**	0.00E+00	PD-112	3.14E+06	**TB-160**	1.87E+05
CO-60**	0.00E+00	PD-115	7.84E+05	**TC-99M**	1.63E+08
CS-132	5.75E+03	**PM-147**	1.68E+07	**TC-101**	1.69E+08
CS-134	2.41E+07	**PM-148**	1.88E+07	TC-103	1.65E+08
CS-134M	5.63E+06	**PM-148M**	2.83E+06	**TC-104**	1.40E+08
CS-136	7.01E+06	**PM-149**	6.43E+07	**TC-105**	1.18E+08
CS-137	1.37E+07	**PM-151**	2.10E+07	TC-106	8.80E+07
CS-138	1.85E+08	PM-153	9.77E+06	**TE-127**	9.03E+06
CS-139	1.72E+08	**PR-142**	9.47E+06	**TE-127M**	1.52E+06
CS-140	1.54E+08	**PR-143**	1.47E+08	**TE-129**	3.10E+07
CS-141	1.17E+08	**PR-144**	1.27E+08	**TE-129M**	6.30E+06
CS-142	6.80E+07	PR-144M	1.76E+06	**TE-131**	8.28E+07
CS-143	3.41E+07	PR-147	6.52E+07	**TE-131M**	2.04E+07
DY-166	4.91E+02	PR-149	3.57E+07	**TE-132**	1.41E+08
EU-154	9.00E+05	PR-151	1.23E+07	**TE-133**	1.09E+08
EU-155	3.83E+05	**PU-238**	5.22E+05	**TE-133M**	8.93E+07
EU-156	3.90E+07	**PU-239**	3.06E+04	**TE-134**	1.75E+08
EU-157	4.12E+06	**PU-240**	4.87E+04	TE-135	9.68E+07
EU-158	1.01E+06	**PU-241**	1.36E+07	TE-136	4.29E+07
EU-159	5.15E+05	**PU-242**	3.34E+02	TE-137	1.45E+07
GA-72	1.71E+03	PU-243	7.36E+07	TE-138	3.65E+06
GA-77	1.66E+05	RA-224	5.16E-01	**TH-228**	5.14E-01
GD-159	8.91E+05	**RB-86**	2.50E+05	U-239	2.17E+09
GE-77	6.48E+04	RB-86M	2.07E+04	**XE-131M**	1.42E+06

TABLE 15.5-77

DCPP EQUILIBRIUM CORE INVENTORY (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent, and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
GE-77M	1.70E+05	**RB-88**	6.60E+07	**XE-133**	2.01E+08
GE-83	1.24E+06	**RB-89**	8.57E+07	**XE-133M**	6.42E+06
H-3	6.10E+04	**RB-90**	7.86E+07	**XE-135**	4.92E+07
HO-166	2.58E+04	**RB-90M**	2.53E+07	**XE-135M**	4.30E+07
I-129	4.00E+00	RB-91	1.05E+08	**XE-137**	1.84E+08
I-130	3.58E+06	RB-92	9.35E+07	**XE-138**	1.70E+08
I-130M	1.92E+06	RB-93	7.89E+07	XE-139	1.25E+08
I-131	9.90E+07	RB-94	4.13E+07	XE-140	8.66E+07
I-132	1.44E+08	RB-95	2.01E+07	XE-142	1.33E+07
I-133	2.01E+08	**RH-103M**	1.66E+08	**Y-90**	1.02E+07
I-134	2.22E+08	**RH-105**	1.08E+08	Y-90M	7.71E+02
I-134M	2.07E+07	**RH-105M**	3.43E+07	**Y-91**	1.19E+08
I-135	1.92E+08	**RH-106**	7.53E+07	**Y-91M**	6.57E+07
I-136	8.73E+07	RH-109	3.65E+07	**Y-92**	1.23E+08
I-137	9.40E+07	RN-220	5.16E-01	**Y-93**	9.41E+07
I-138	4.80E+07	**RU-103**	1.66E+08	**Y-94**	1.50E+08
I-139	2.22E+07	**RU-105**	1.21E+08	**Y-95**	1.57E+08
I-140	6.06E+06	**RU-106**	6.68E+07	Y-97	1.26E+08
IN-115M	9.17E+05	RU-109	3.16E+07	ZN-72	1.71E+03
IN-121	7.55E+04	**SB-122**	1.57E+05	ZR-101	9.55E+07
IN-121M	7.82E+05	SB-122M	1.57E+04	**ZR-95**	1.65E+08
IN-123	6.87E+05	**SB-124**	1.21E+05	**ZR-97**	1.58E+08
		SB-124M	2.34E+03	ZR-99	1.66E+08

Note:

* Isotopes in **Bold** Font are dose-significant for inhalation, submersion, and direct shine. The parent, grandparent, and second parent of the isotopes in **Bold** Font are also required to address daughter product in-growth.

The group of isotopes needed to determine the "submersion and inhalation" dose in the Control Room and at the Site Boundary is typically a subset of the isotopes listed above in **bold** font, and represent a small group of reasonably long half-life isotopes with significant inhalation dose conversion factors which dominate the TEDE dose.

To determine the total effective dose equivalent (TEDE) resulting from inhalation and submersion following a LOCA, the DCPP LOCA dose consequence analysis uses the default group of 60 isotopes provided with computer code RADTRAD 3.03 plus 13 additional nuclides that were deemed to be dose significant (i.e., Br-82, Br-84, Rb-88, Rb-89, Te-133, Te-133m, Te-134, I-130, Xe-131m, Xe-133m, Xe-138, Cs-138, and Np-238).

** Co-58 / Co-60 are activation products that are developed external to the core and typically do not appear in the equilibrium core inventory

TABLE 15.5-78

PRIMARY AND SECONDARY COOLANT
Technical Specification Activity Concentrations

Nuclide	Primary Coolant (μCi/gm)	Secondary Coolant (μCi/gm)
Kr-83M	1.87E-01	-----
Kr-85M	6.60E-01	-----
Kr-85	5.60E+00	-----
Kr-87	4.41E-01	-----
Kr-88	1.22E+00	-----
Xe-131M	1.88E+00	-----
Xe-133M	1.92E+00	-----
Xe-133	1.29E+02	-----
Xe-135M	4.07E-01	-----
Xe-135	3.76E+00	-----
I-131	7.87E-01	8.06E-02
I-132	3.00E-01	1.94E-02
I-133	1.16E+00	1.08E-01
I-134	1.67E-01	4.78E-03
I-135	6.68E-01	5.09E-02

TABLE15.5-79

PRIMARY COOLANT
Pre-Accident Iodine Spike Concentrations & Equilibrium Iodine Appearance Rates

Nuclide	Pre-Accident Spike RCS Concentrations (60 µCi/gm DE I-131) (µCi/gm)	Equilibrium Iodine Activity Appearance Rates into RCS (µCi/sec)
I-131	47.2	7.18E+03
I-132	17.9	7.78E+03
I-133	69.5	1.25E+04
I-134	10.0	8.91E+03
I-135	40.1	9.91E+03

TABLE 15.5-80

ISOTOPIC GAP ACTIVITY
LOCKED ROTOR ACCIDENT / CONTROL ROD EJECTION ACCIDENT

Nuclide	Core Activity (Ci)	Fraction of Core Activity in Gap LRA	Core Gap Activity w/o Peaking Factor (Ci) LRA	Fraction of Core Activity in Gap CREA	Core Gap Activity w/o Peaking Factor (Ci) CREA
KR-85	1.11E+06	0.35	3.89E+05	0.10	1.11E+05
KR-85M	2.33E+07	0.04	9.32E+05	0.10	2.33E+06
KR-87	4.65E+07	0.04	1.86E+06	0.10	4.65E+06
KR-88	6.43E+07	0.04	2.57E+06	0.10	6.43E+06
Xe-131M	1.42E+06	0.04	5.68E+04	0.10	1.42E+05
Xe-133M	6.42E+06	0.04	2.57E+05	0.10	6.42E+05
XE-133	2.01E+08	0.04	8.04E+06	0.10	2.01E+07
XE-135	4.92E+07	0.04	1.97E+06	0.10	4.92E+06
Xe-138	1.70E+08	0.04	6.80E+06	0.10	1.70E+07
I-130	3.58E+06	0.05	1.79E+05	0.10	3.58E+05
I-131	9.90E+07	0.08	7.92E+06	0.10	9.90E+06
I-132	1.44E+08	0.23	3.31E+07	0.10	1.44E+07
I-133	2.01E+08	0.05	1.01E+07	0.10	2.01E+07
I-134	2.22E+08	0.05	1.11E+07	0.10	2.22E+07
I-135	1.92E+08	0.05	9.60E+06	0.10	1.92E+07
BR-82	4.44E+05	0.05	2.22E+04	0.10	4.44E+04
BR-84	2.10E+07	0.05	1.05E+06	0.10	2.10E+06
CS-134	2.41E+07	0.46	1.11E+07	-	-
CS-136	7.01E+06	0.46	3.22E+06	-	-
CS-137	1.37E+07	0.46	6.30E+06	-	-
CS-138	1.85E+08	0.46	8.51E+07	-	-
RB-86	2.50E+05	0.46	1.15E+05	-	-
Rb-88	6.60E+07	0.46	3.04E+07	-	-
Rb-89	8.57E+07	0.46	3.94E+07	-	-

Note: Values reported reflect the core isotopic gap activity assumed for the LRA and CREA. These values have to be adjusted for a) the failed fuel percentage (10%) and b) peaking factor (1.65), prior to assessing the associated dose consequences

For the isotopic gap activity associated with the FHA refer to Table 15.5-47C

TABLE 15.5-81

CONTROL ROOM
Analysis Assumptions & Key Parameter Values

Parameter	Value
Free Volume	170,000 ft^3
Unfiltered Normal Operation Intake	Total 4200 cfm ± 10% Unit 1: 2100 cfm ± 10% Unit 2: 2100 cfm ± 10%
Emergency Pressurization Flow Rate	650 – 900 cfm
Maximum Unfiltered Backdraft Damper Leakage during CR Pressurization Operation	100 cfm
Carbon / HEPA Filter Flow during CR Pressurization Operation	1800 – 2200 cfm
Emergency Filtered Recirculation Rate	1250 cfm (minimum)
Pressurization Intake and Recirculation Carbon/HEPA Filter Efficiency (includes filter bypass)	93% (iodine) 98% (particulates)
Unfiltered Inleakage (Normal and Pressurization Mode)	70 cfm (maximum) Includes 10 cfm egress-ingress
Occupancy Factors	0-24 hr (1.0)
	1 - 4 d (0.6)
	4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.50E-04 m^3/sec)

TABLE 15.5-82

TECHNICAL SUPPORT CENTER
Analysis Assumptions & Key Parameter Values

Parameter	Value
Free Volume	51,250 ft^3
Filtered (HEPA only) Normal Operation Intake Flow Rate	500 cfm
Normal Intake HEPA Filter Efficiency (includes filter bypass)	98% (particulates)
Filtered (Carbon / HEPA) Pressurization Flow Rate	500 cfm
Flow through Carbon / HEPA Filter during Pressurization mode	1000 cfm
Filtered Recirculation flow rate during Pressurization mode	500 cfm (minimum)
Pressurization Intake and Recirculation Carbon / HEPA Filter Efficiency (includes filter bypass)	93% (iodine) 98% (particulates)
Unfiltered Inleakage	60 cfm (maximum) Includes 10 cfm egress-ingress
Occupancy Factors	0-24 hr (1.0)
	1 - 4 d (0.6)
	4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.50E-04 m^3/sec)

TABLE 15.5-83
NON-LOCA EVENTS
Technical Support Center Limiting Atmospheric Dispersion Factors (sec/m^3)

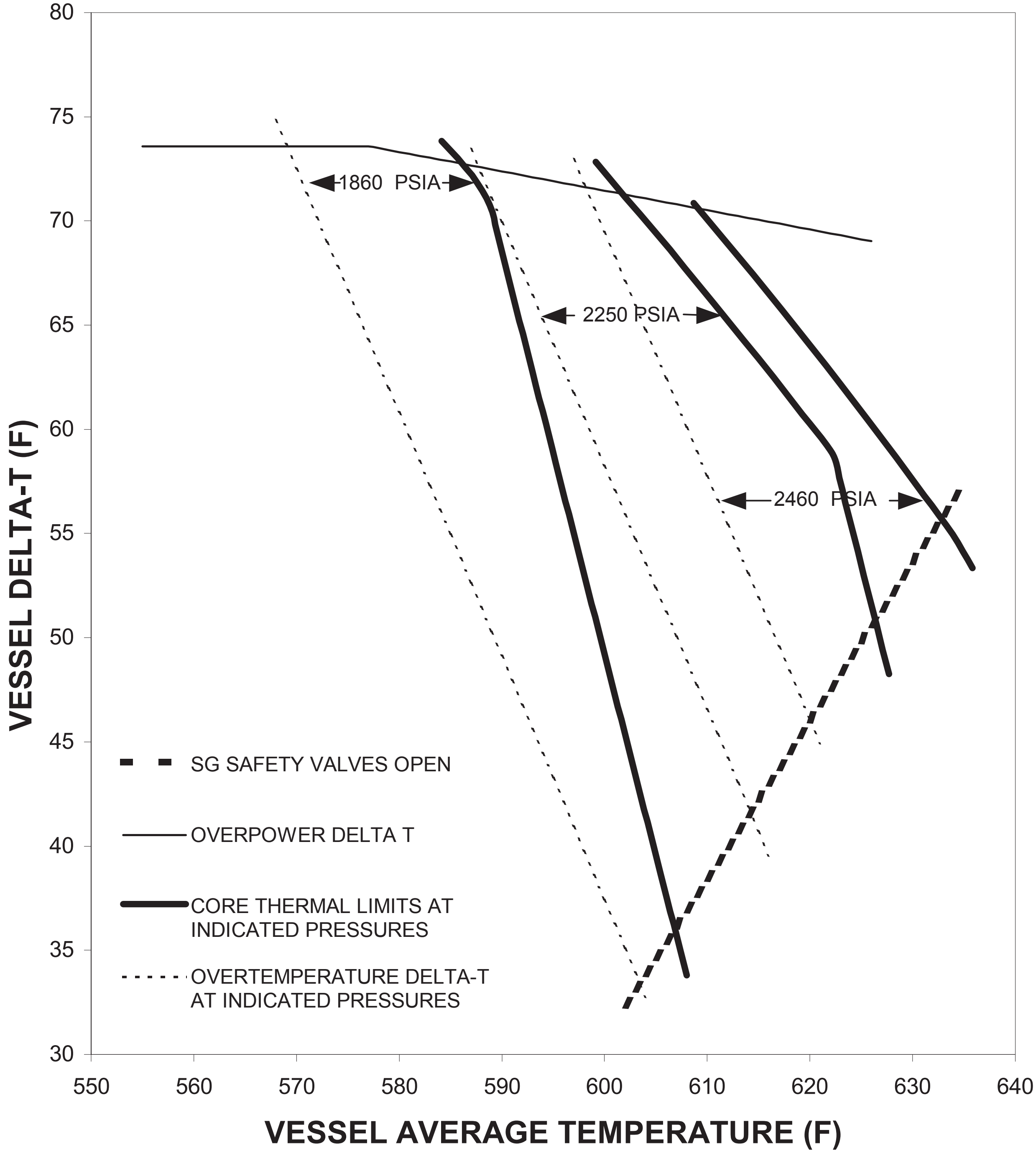
Receptor - Release Point	0-2 hr	2-8 hr	8-10.73hr	10.73-30hr	
MSLB					
TSC NOP Intake - Faulted SG (Break Location)	9.00E-04	-----	-----	-----	
TSC NOP Intake - Intact SG (MSSVs/10% ADVs)	1.80E-04	-----	-----	-----	
TSC Inleakage - Faulted SG (Break Location)	1.01E-03	4.62E-04	1.93E-04	1.93E-04	
TSC Inleakage - Intact SG (MSSVs/10% ADVs)	2.02E-04	9.24E-05	9.24E-05	-----	
CR/TSC Pressurization Intake - Faulted SG (Break Location)	-----	4.70E-05	1.85E-05	1.85E-05	
CR/TSC Pressurization Intake - Intact SG (MSSVs/10% ADVs)	-----	9.40E-06	9.40E-06	-----	
SGTR / LRA / LOL / CREA (Secondary Side Release Scenario)					
TSC Center of Roof - MSSVs/10% ADVs	2.02E-04	9.24E-05	9.24E-05	---	
FHA					
TSC Center of Roof - Equipment Hatch	7.44E-04	-----	-----	-----	
Receptor - Release Point	**0-2 hr**	**2-8 hr**	**8-24hr**	**1-4 days**	**4-30 days**
CREA (Containment Release Scenario)					
TSC NOP Intake - Containment Leakage	1.71E-03	-----	-----	-----	-----
TSC Inleakage - Containment Leakage	1.76E-03	7.16E-04	3.01E-04	2.84E-04	2.28E-04
CR/TSC Pressurization Intake - Containment Leakage	-----	3.98E-05	1.63E-05	1.37E-05	1.10E-05

Note1: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit.

- Except as noted below for the CREA Containment leakage release point to the CR/TSC Pressurization Intakes, the χ/Q values for U2 release points are bounding for all TSC receptors (i.e., the TSC NOP Intake, the TSC Center of Roof (also used for TSC Inleakage) and the CR/TSC Pressurization Intakes). Releases from the containment penetration areas are based on the Unit 2 GW/FW area release point.
- Releases from the containment penetration areas to the CR/TSC pressurization intakes are based on the Unit 2 GW/FW area releases to the Unit 1 CR/TSC intake for the 2-24 hrs time period, from the Unit 2 GE area releases to the Unit 1 CR/TSC intake for the 1-4 day time period and from the Unit 1 GW/FW area releases to the Unit 2 CR/TSC intake for the 4-30 day time period.

Note 2: The χ/Q values presented above for MSSVs / 10% ADVs reflect a factor of 5 reduction to address the high vertical velocity discharge for the first 10.73 hours of the accident.

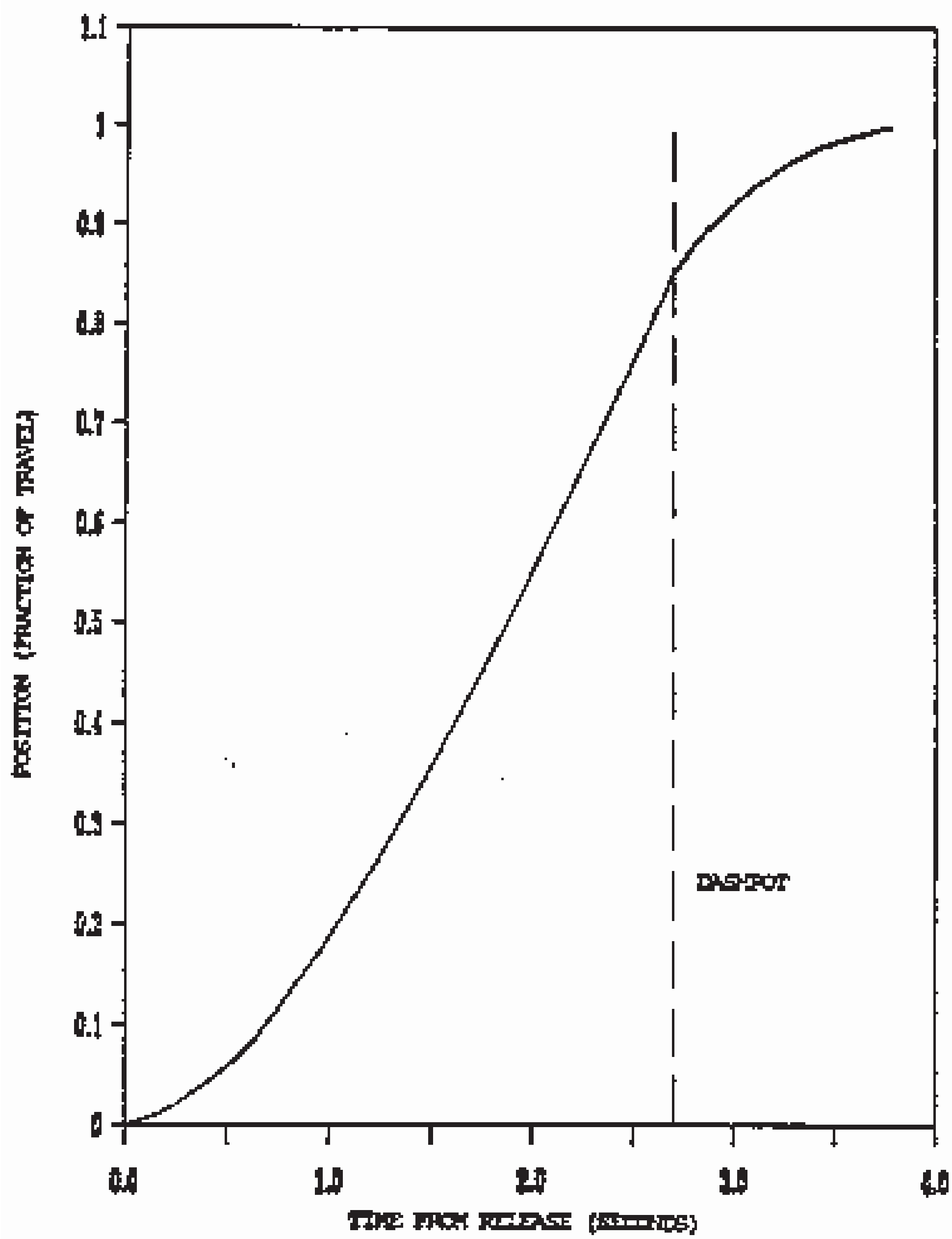
Note 3: The χ/Q values presented above for the CR/TSC pressurization intake reflect a factor of 4 reduction to address the availability of redundant safety related radiation monitors at each CR/TSC pressurization intake location, and the associated capability of initial selection of the less contaminated intake.



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.1-1
ILLUSTRATION OF OVERPOWER AND
OVERTEMPERATURE ΔT PROTECTION

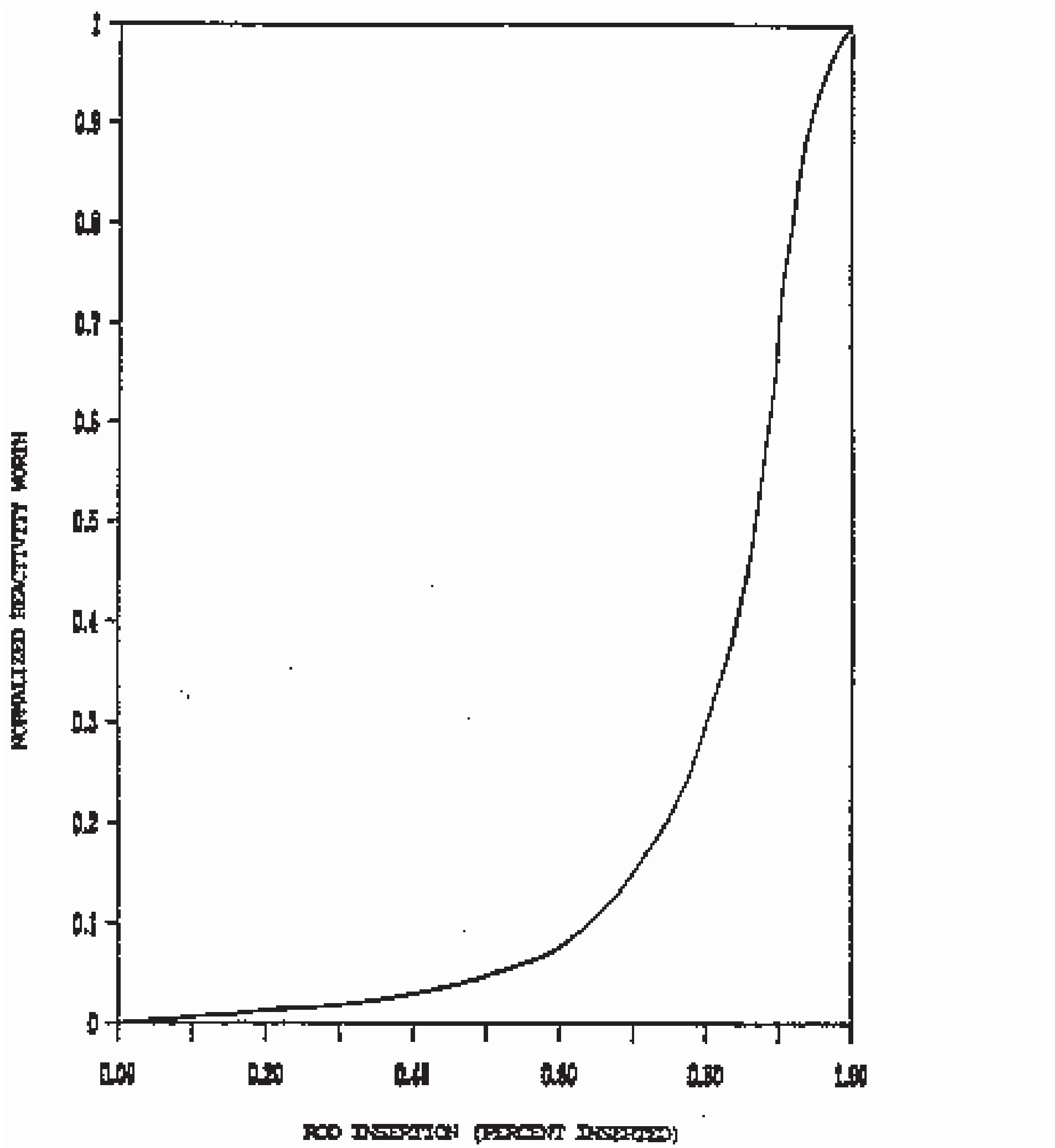


FSAR UPDATE

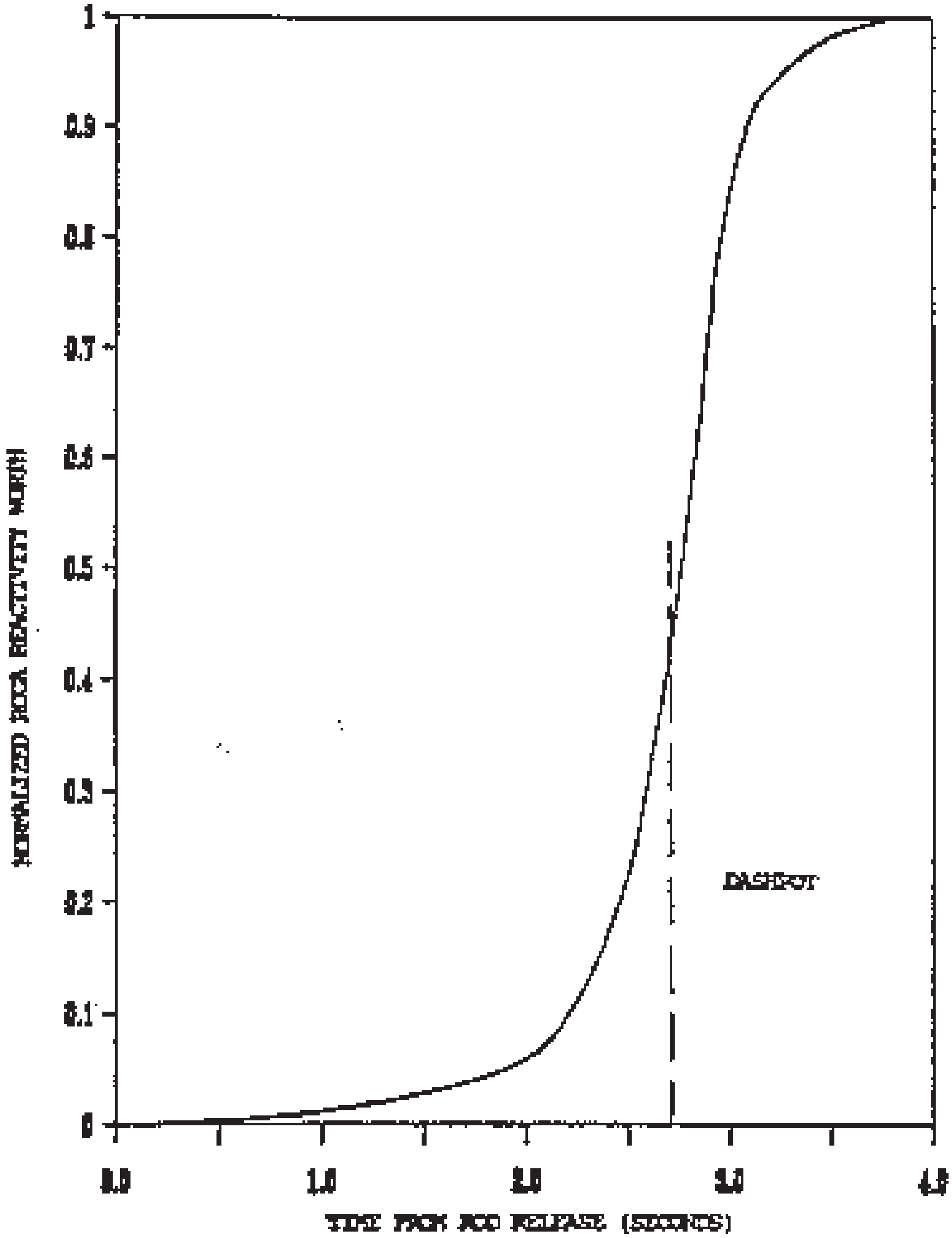
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.1-2
ROD POSITION VERSUS TIME ON REACTOR TRIP

Revision 11 November 1996



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-3 NORMALIZED RCCA REACTIVITY WORTH VERSUS PERCENT INSERTION



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.1-4
NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS TIME AFTER TRIP

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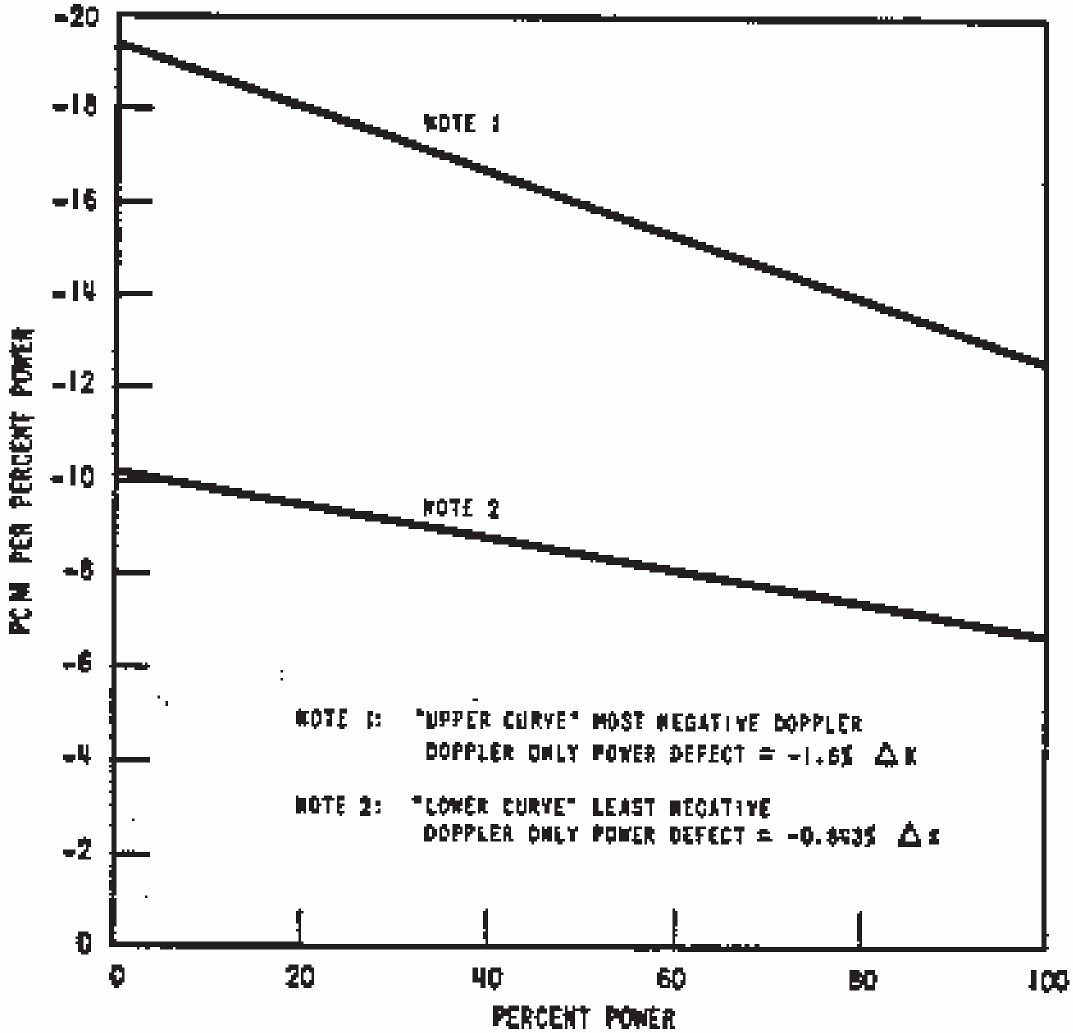

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-5 DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSIS

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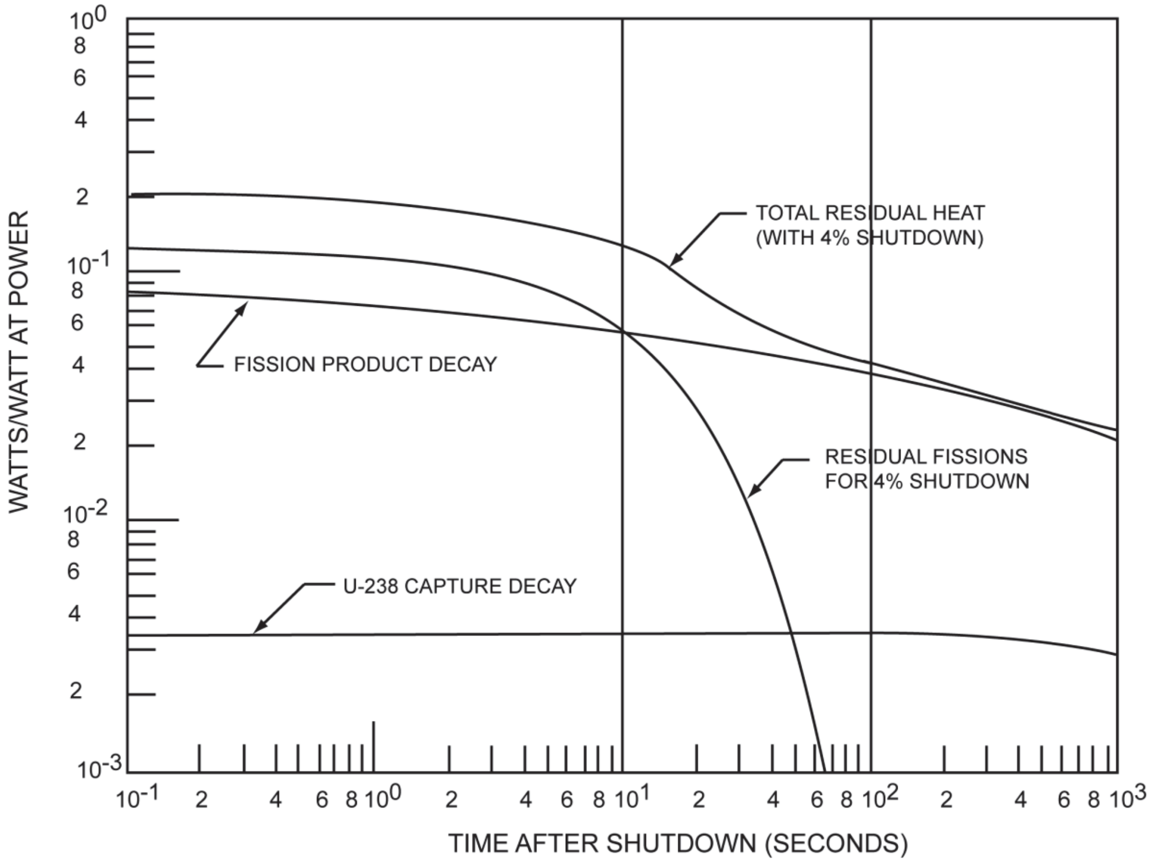
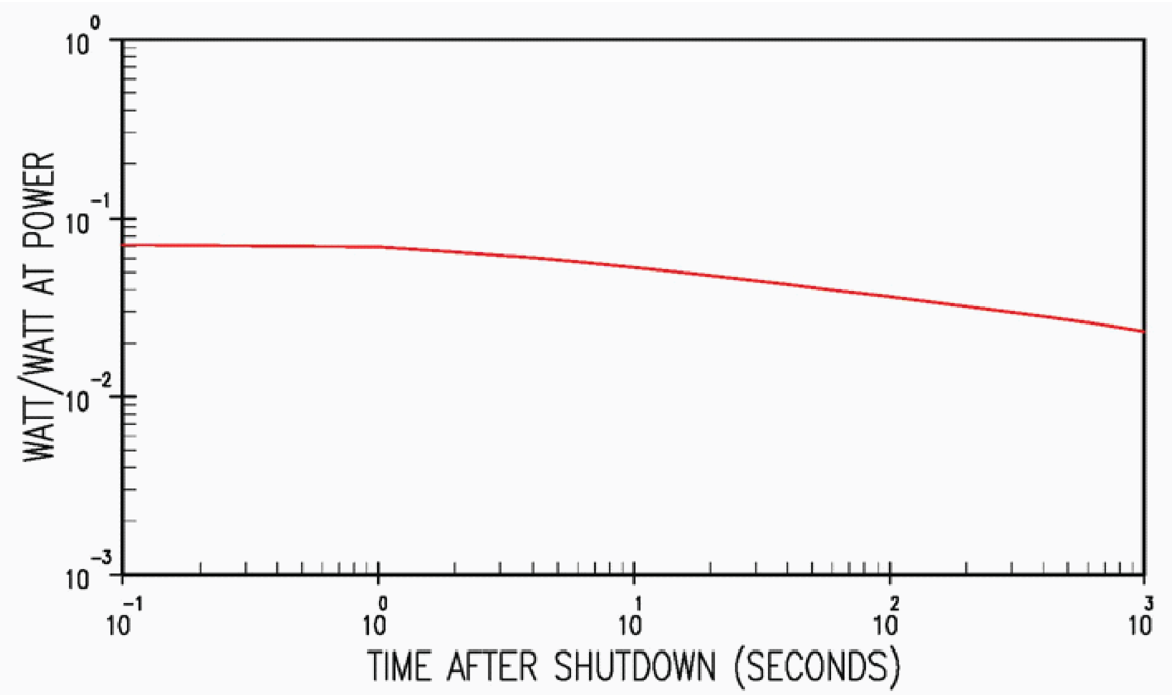
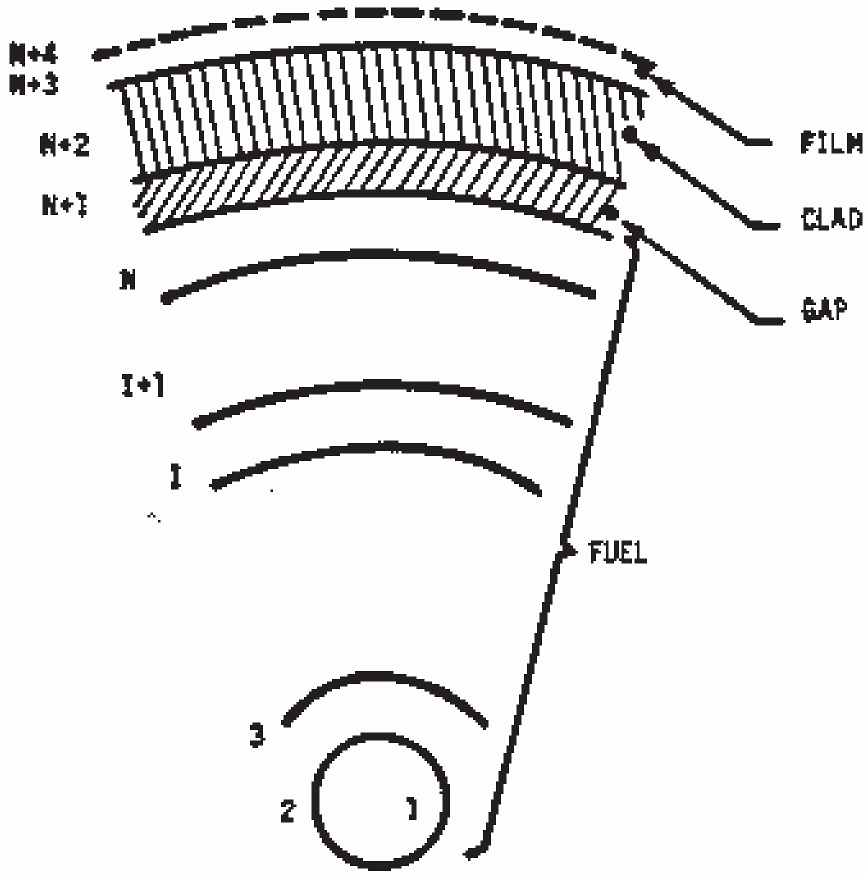


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-6 RESIDUAL DECAY HEAT (BEST ESTIMATE LBLOCA 1979 ANS DECAY HEAT)



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-7 1979 ANS DECAY HEAT CURVE (USED FOR NON-LOCA ANALYSES)



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.1-8 FUEL ROD CROSS SECTION

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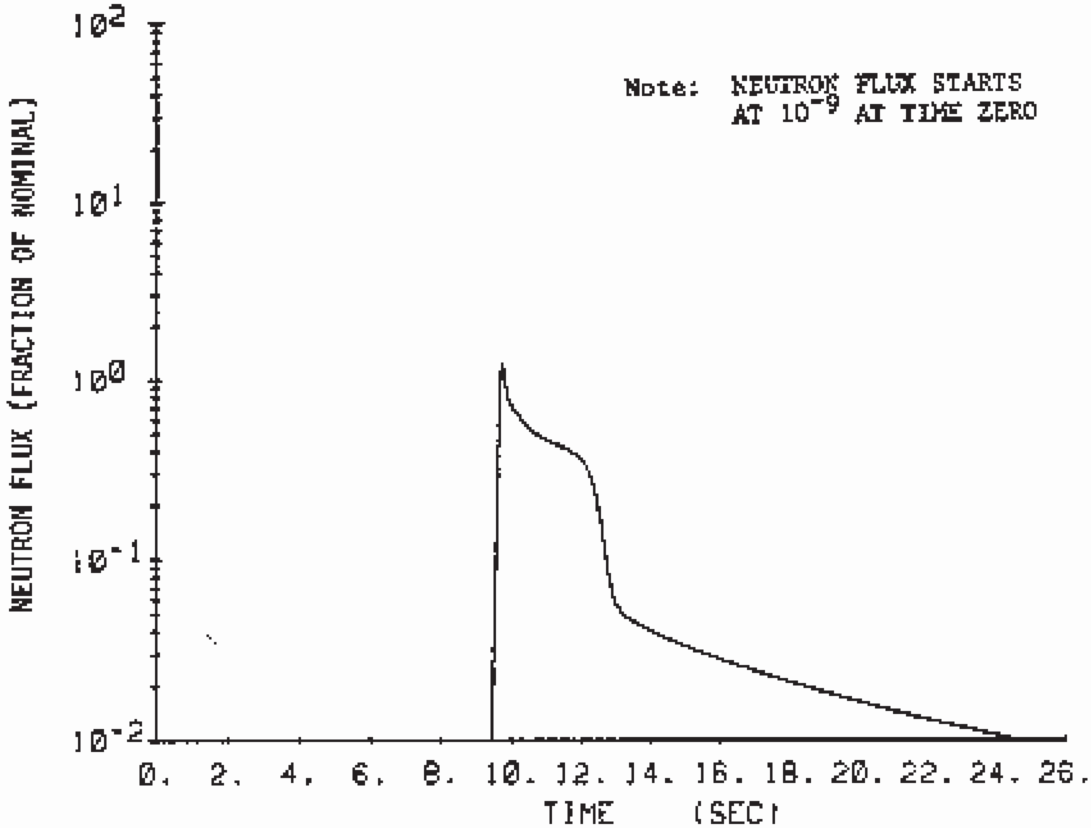
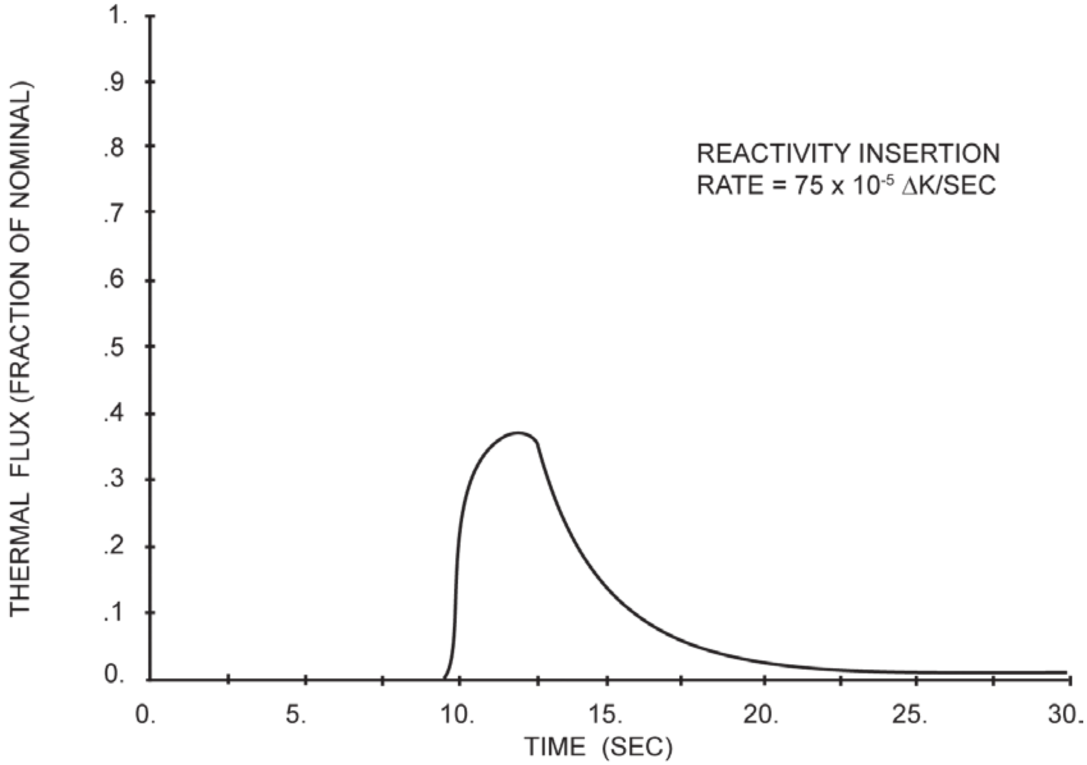


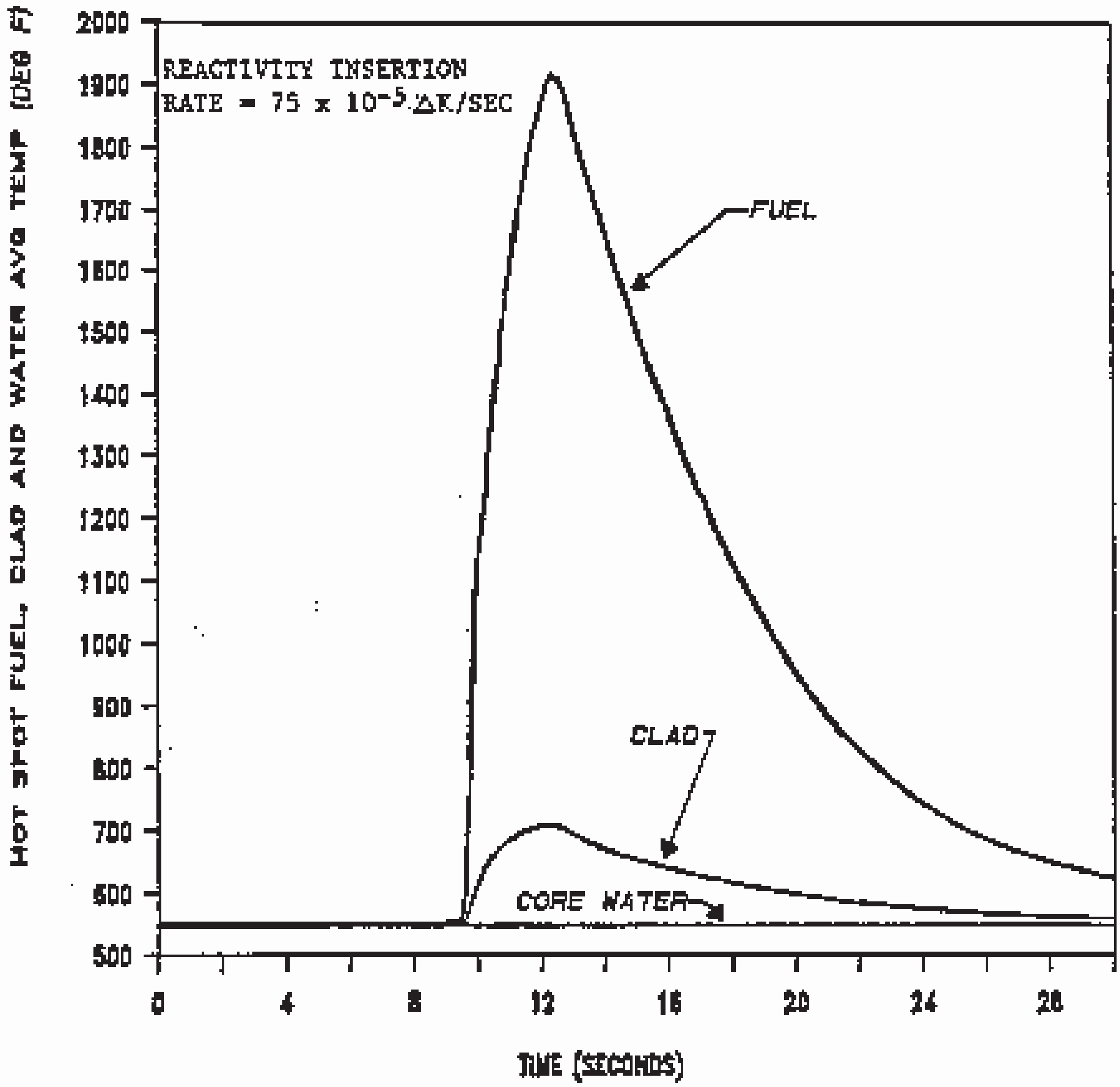
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

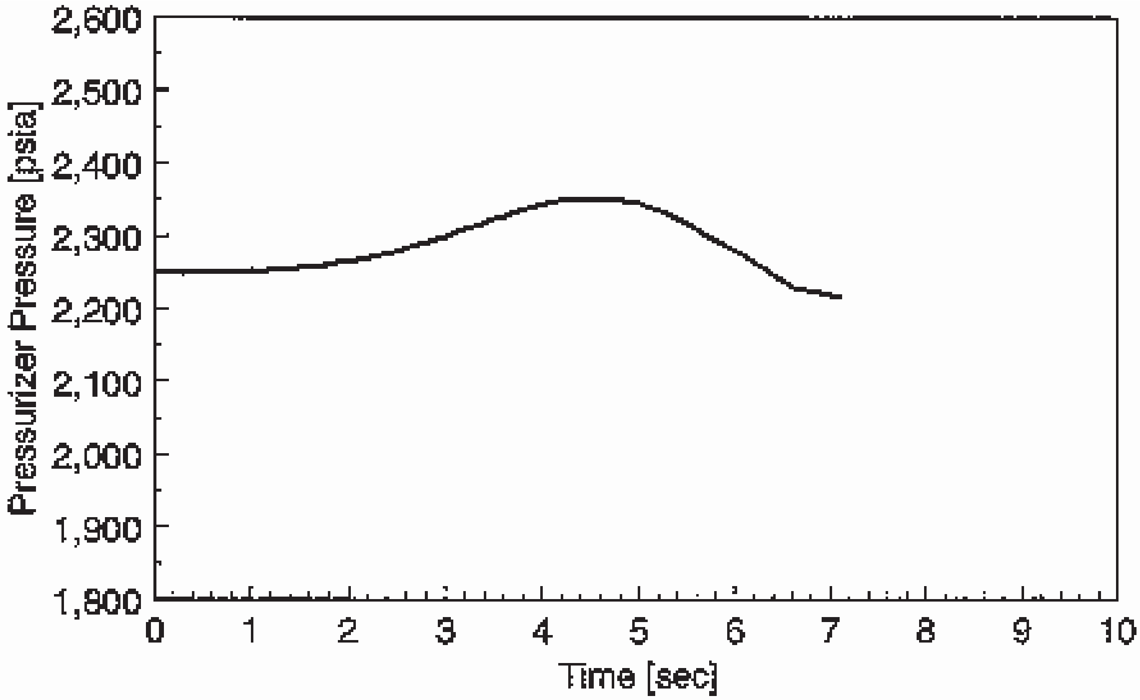
FIGURE 15.2.1-1
UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION NEUTRON FLUX VERSUS TIME

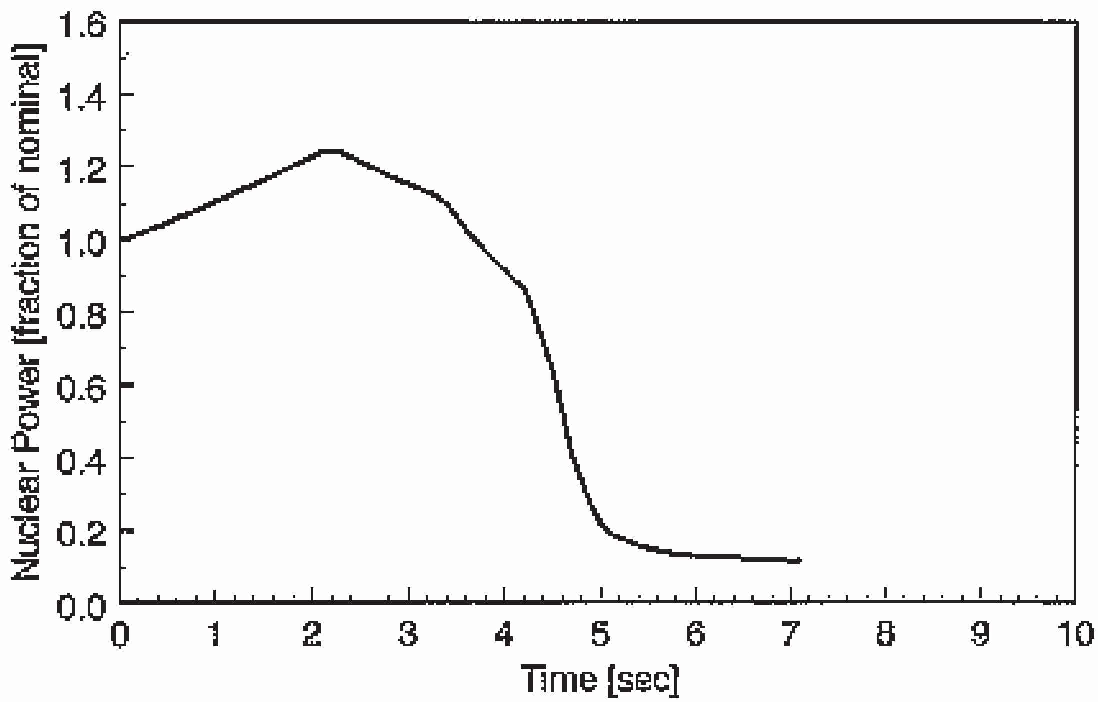


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.1-2 UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION AVERAGE CHANNEL THERMAL FLUX VERSUS TIME

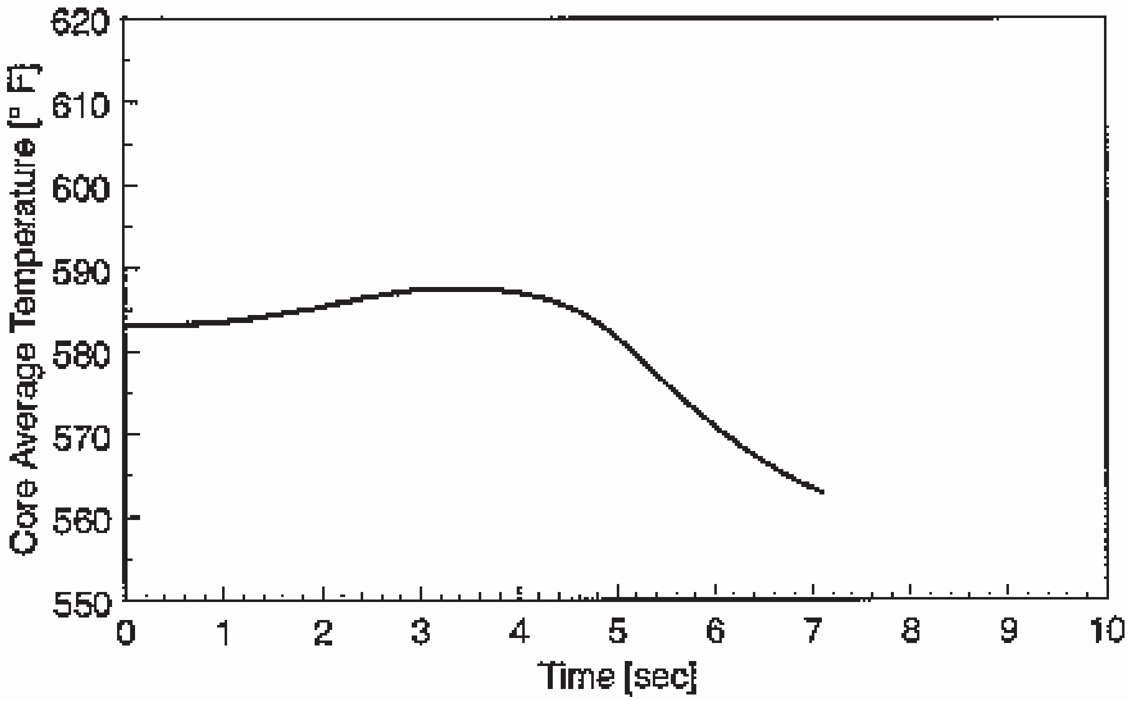


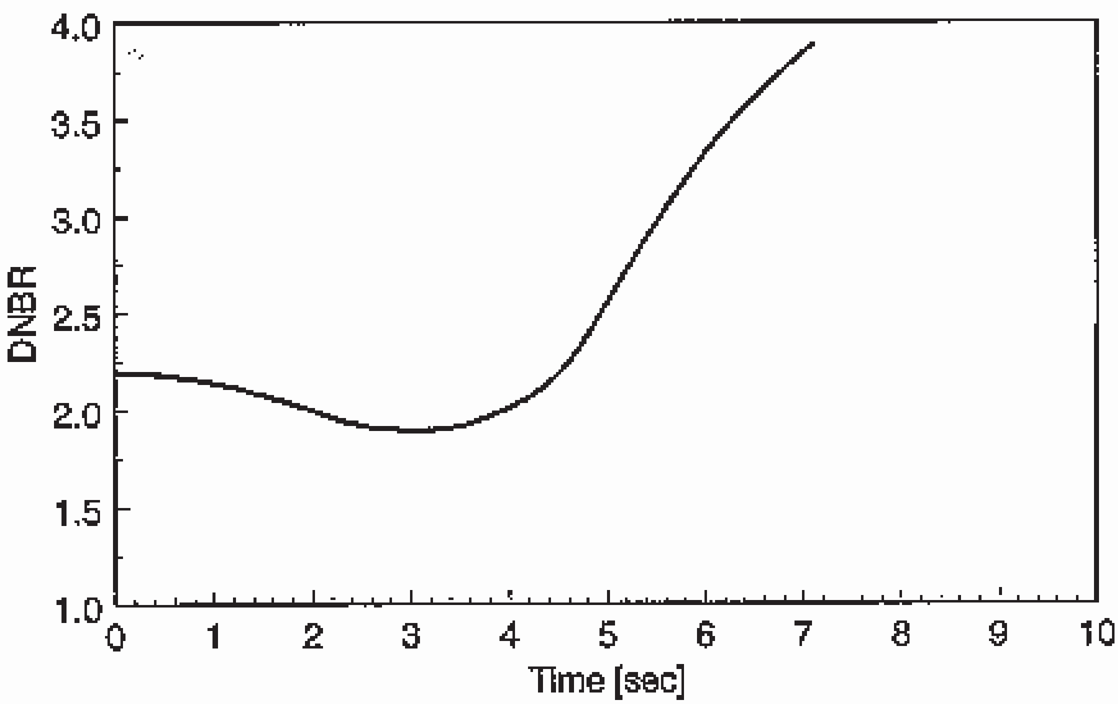
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.1-3 UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION TEMPERATURE VERSUS TIME. REACTIVITY INSERTION RATE 75 X 10^{-5} DELTA K/SEC



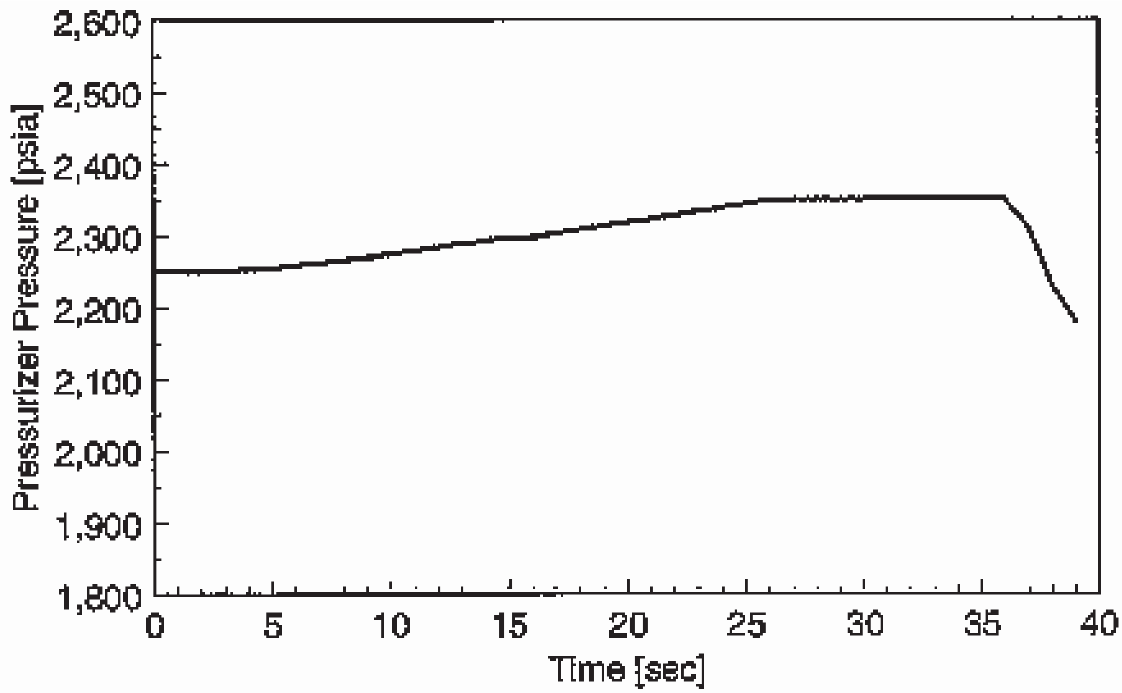


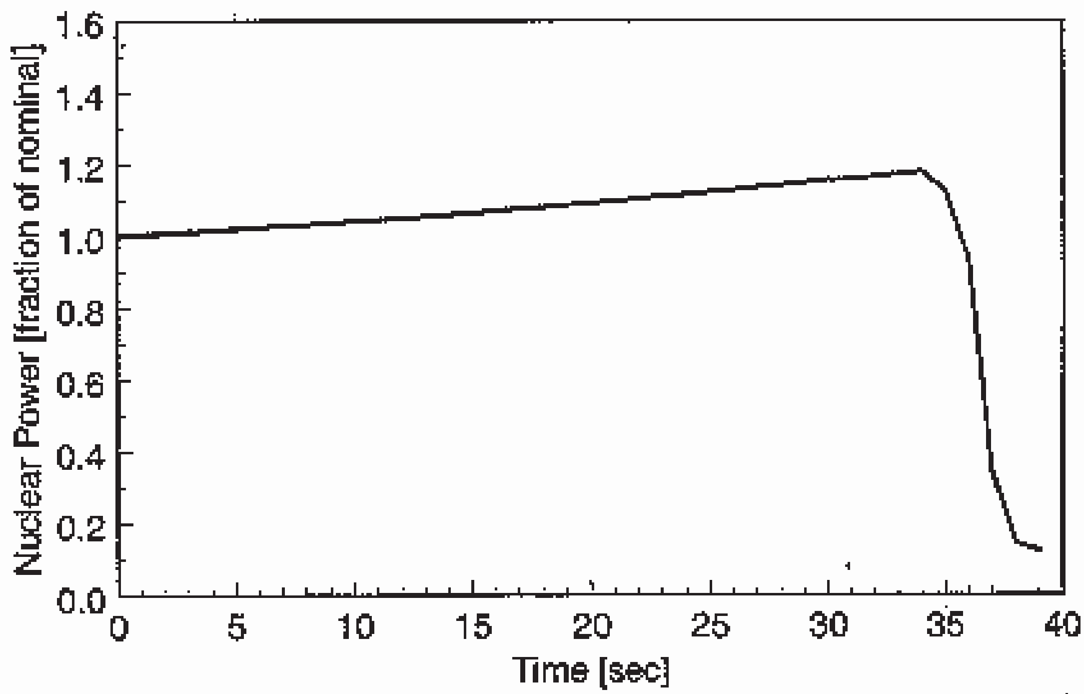
ROD WITHDRAWAL AT POWER Minimum Feedback, 75 pcm/sec Insertion Rate	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.2.2-1



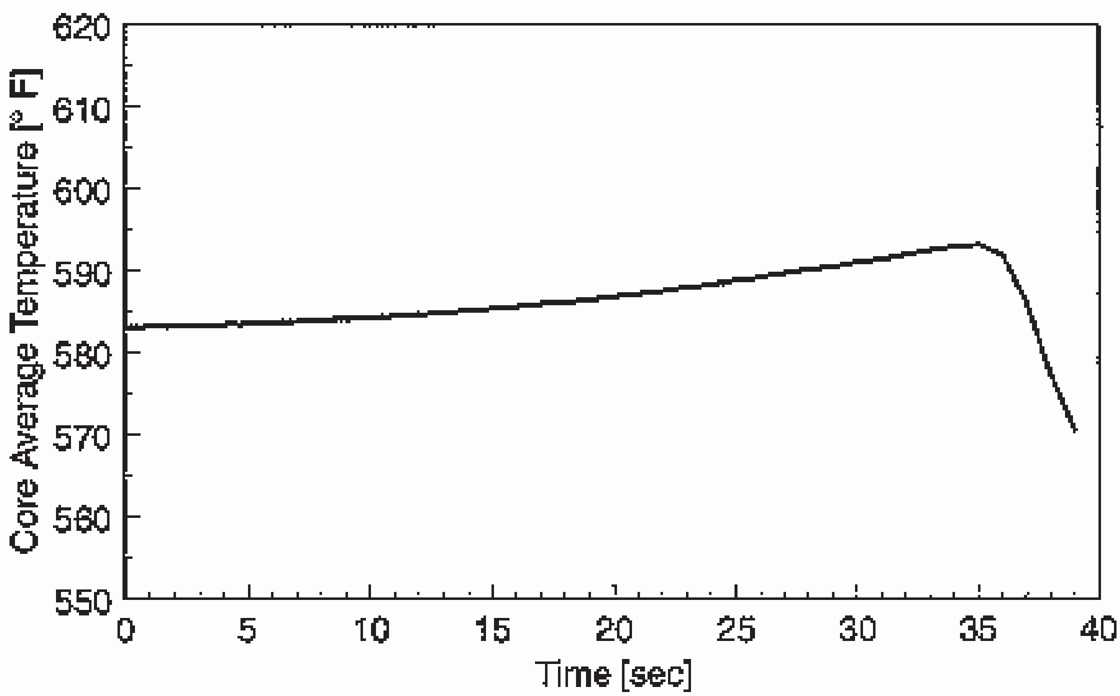


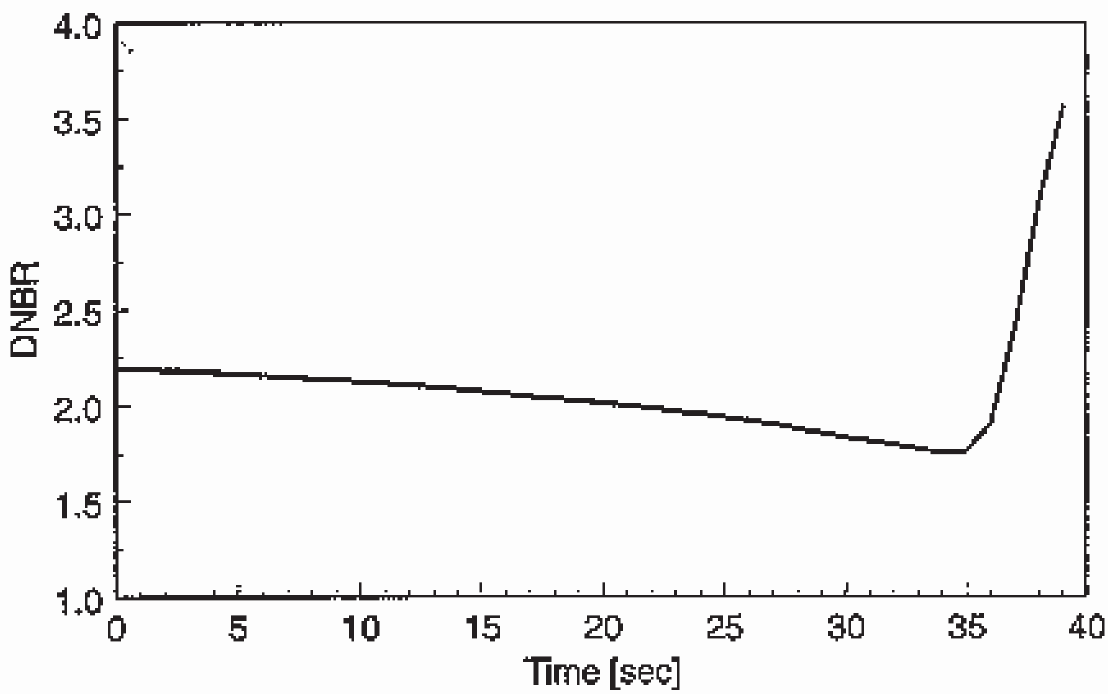
ROD WITHDRAWAL AT POWER **Minimum Feedback, 75 pcm/sec Insertion Rate**	**DIABLO CANYON UNITS 1 & 2**
	FIGURE 15.2.2-2



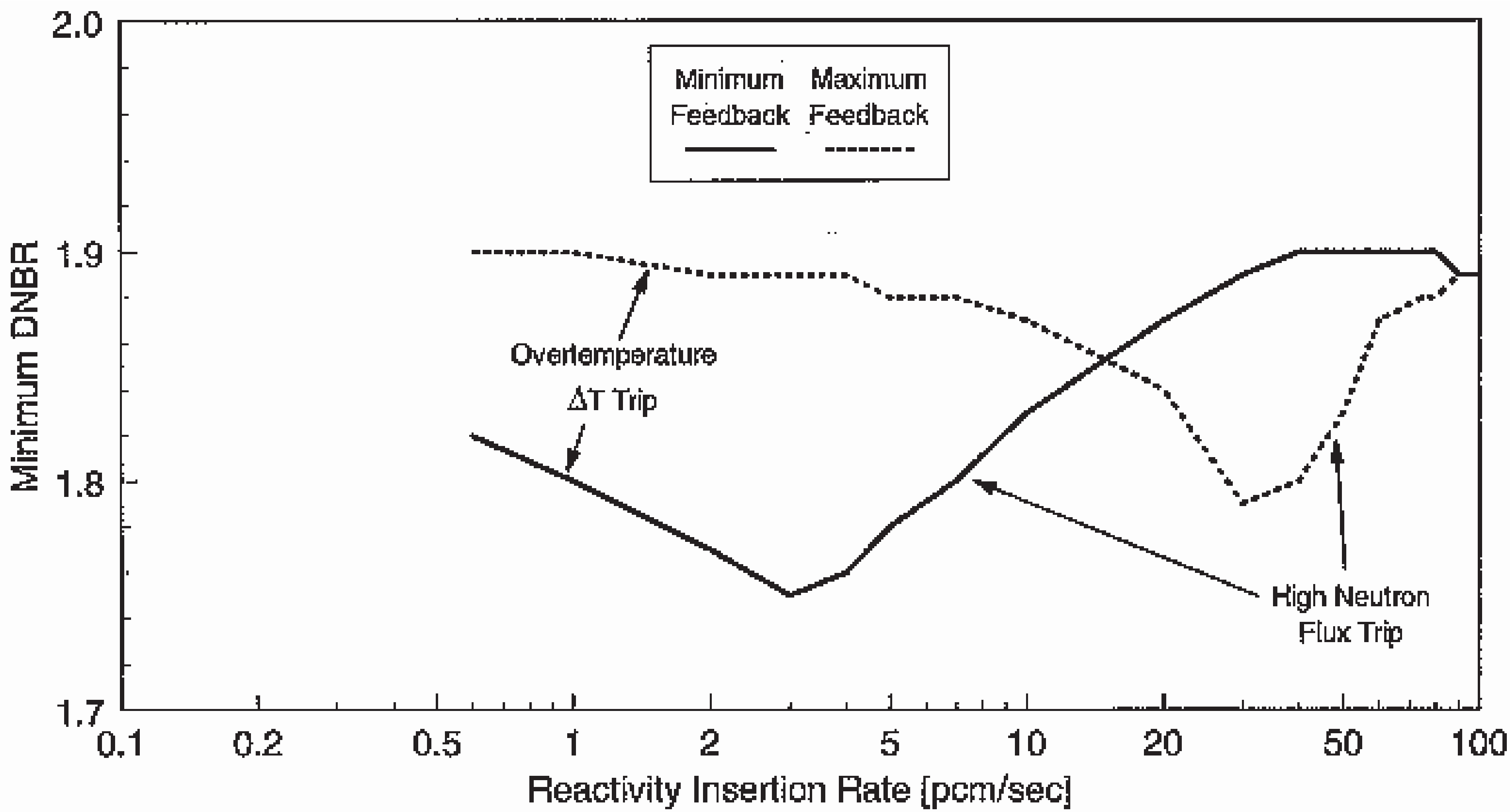


ROD WITHDRAWAL AT POWER **Minimum Feedback, 3 pcm/sec Insertion Rate**	**DIABLO CANYON UNITS 1 & 2**
	FIGURE 15.2.2-3

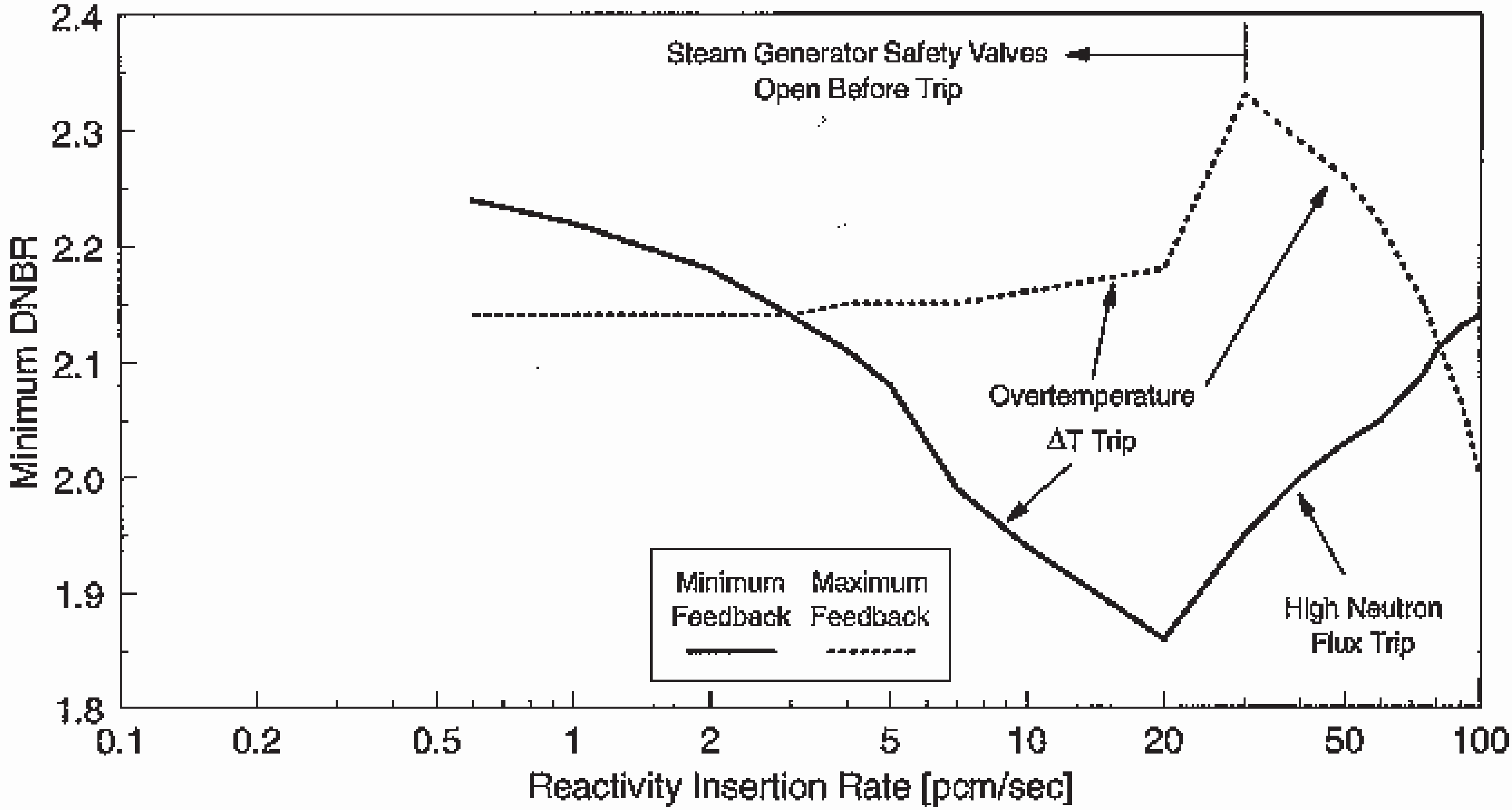




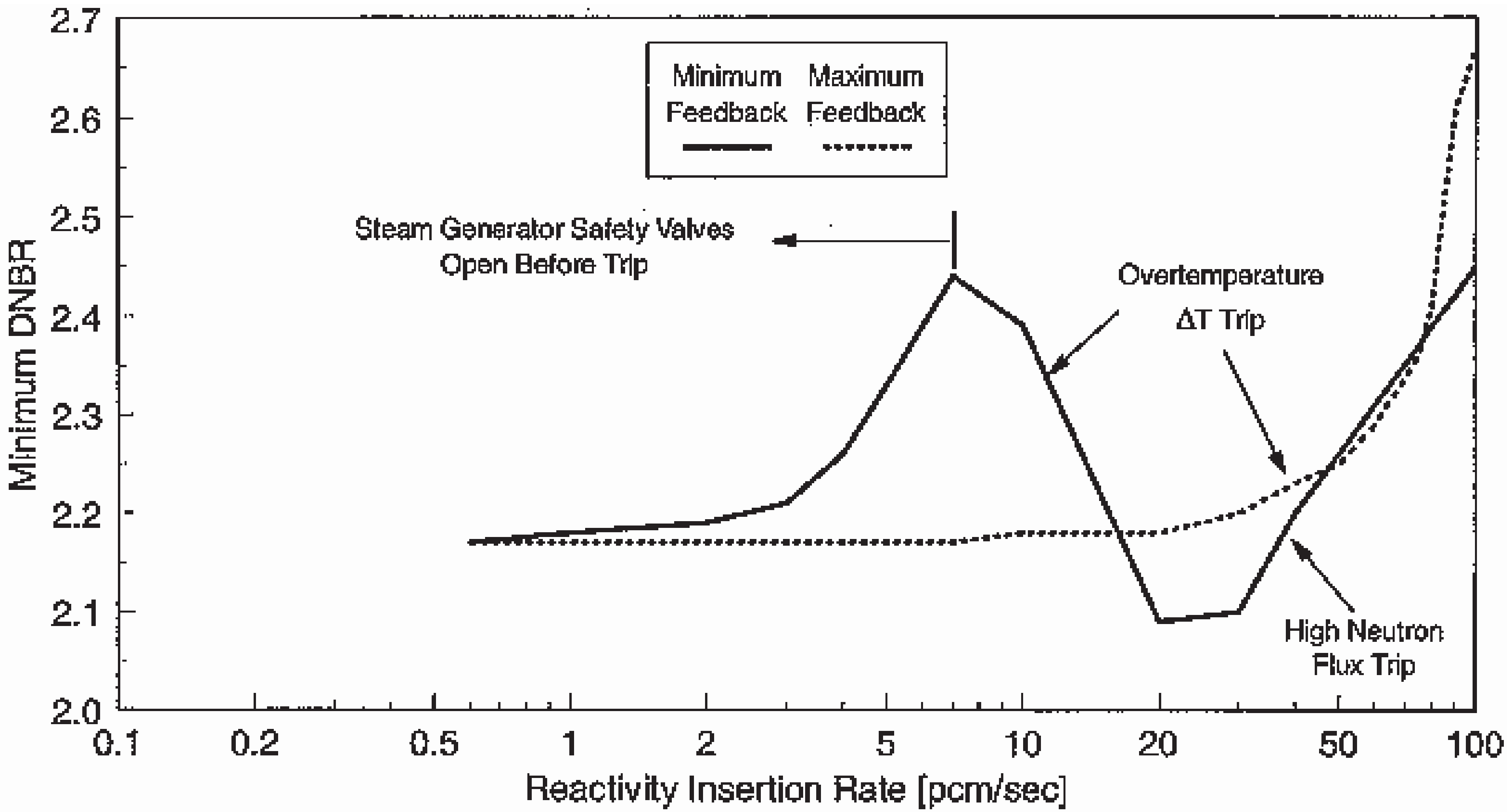
ROD WITHDRAWAL AT POWER Minimum Feedback, 3 pcm/sec Insertion Rate	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.2.2-4



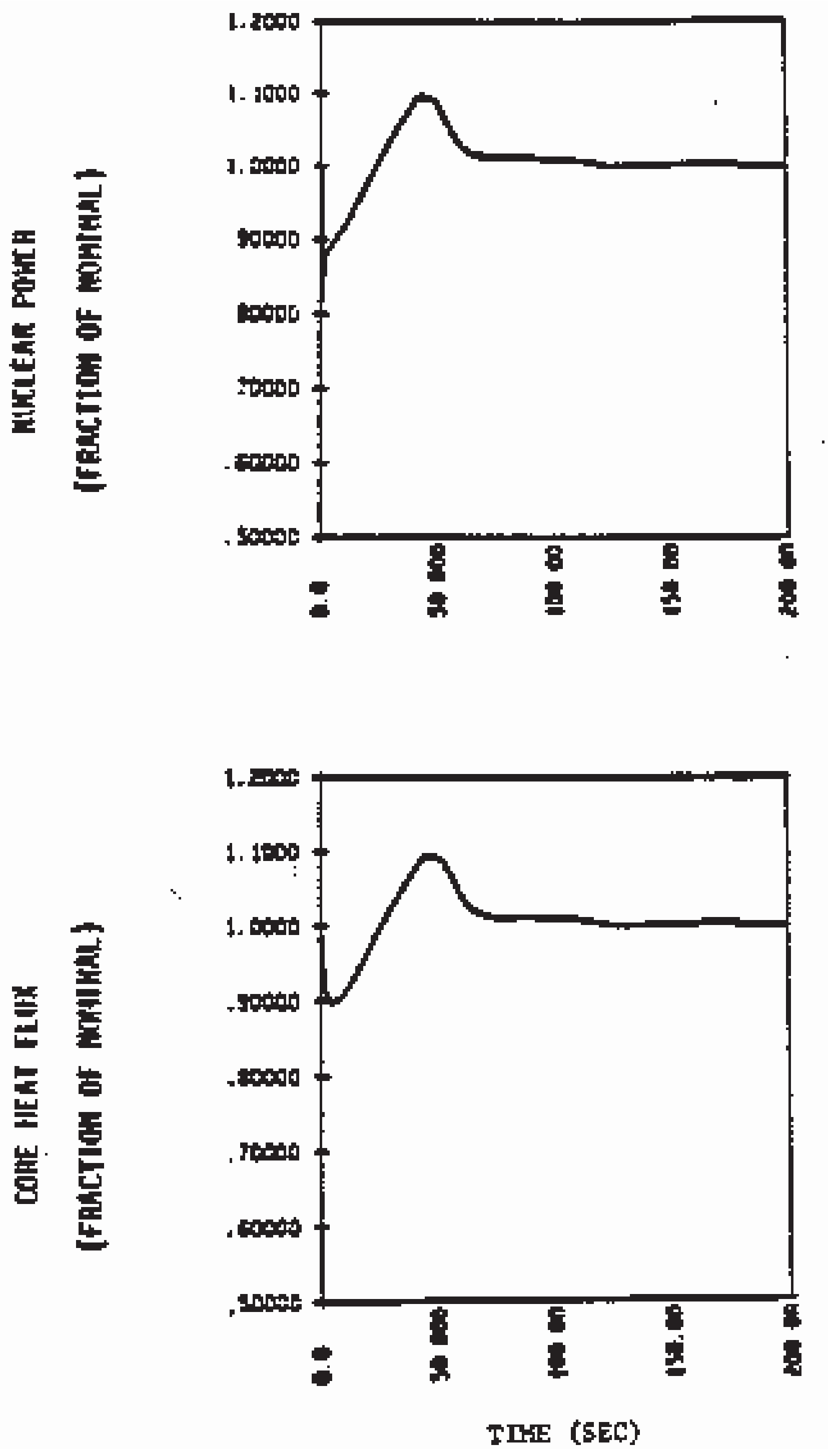
ROD WITHDRAWAL AT POWER Reactivity Insertion Rate vs. DNBR For 100% Power Cases	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.2.2-5



ROD WITHDRAWAL AT POWER **Reactivity Insertion Rate vs. DNBR** **For 60% Power Cases**	**DIABLO CANYON** **UNITS 1 & 2**
	FIGURE 15.2.2-6



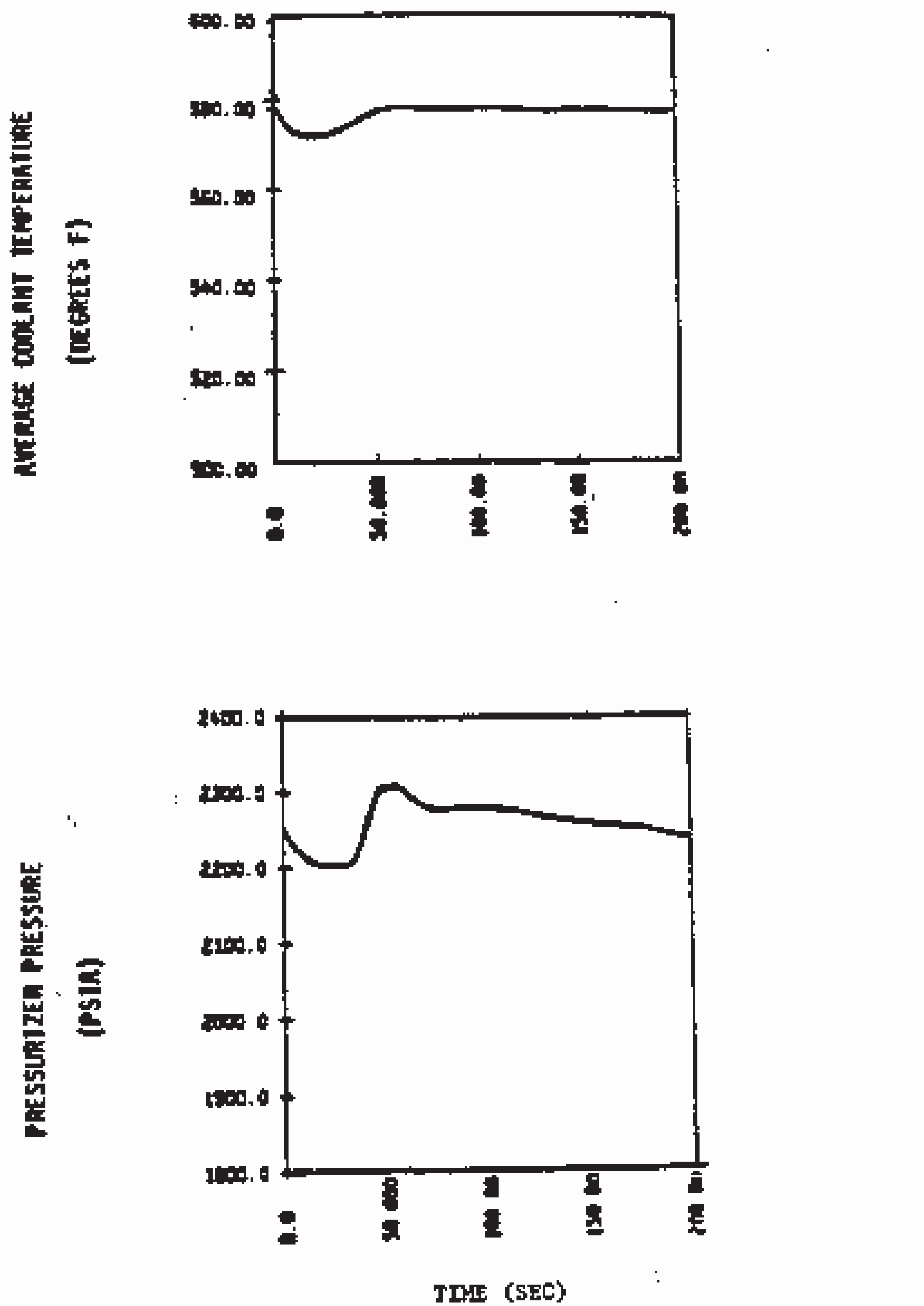
ROD WITHDRAWAL AT POWER Reactivity Insertion Rate vs. DNBR For 10% Power Cases	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.2.2-7



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

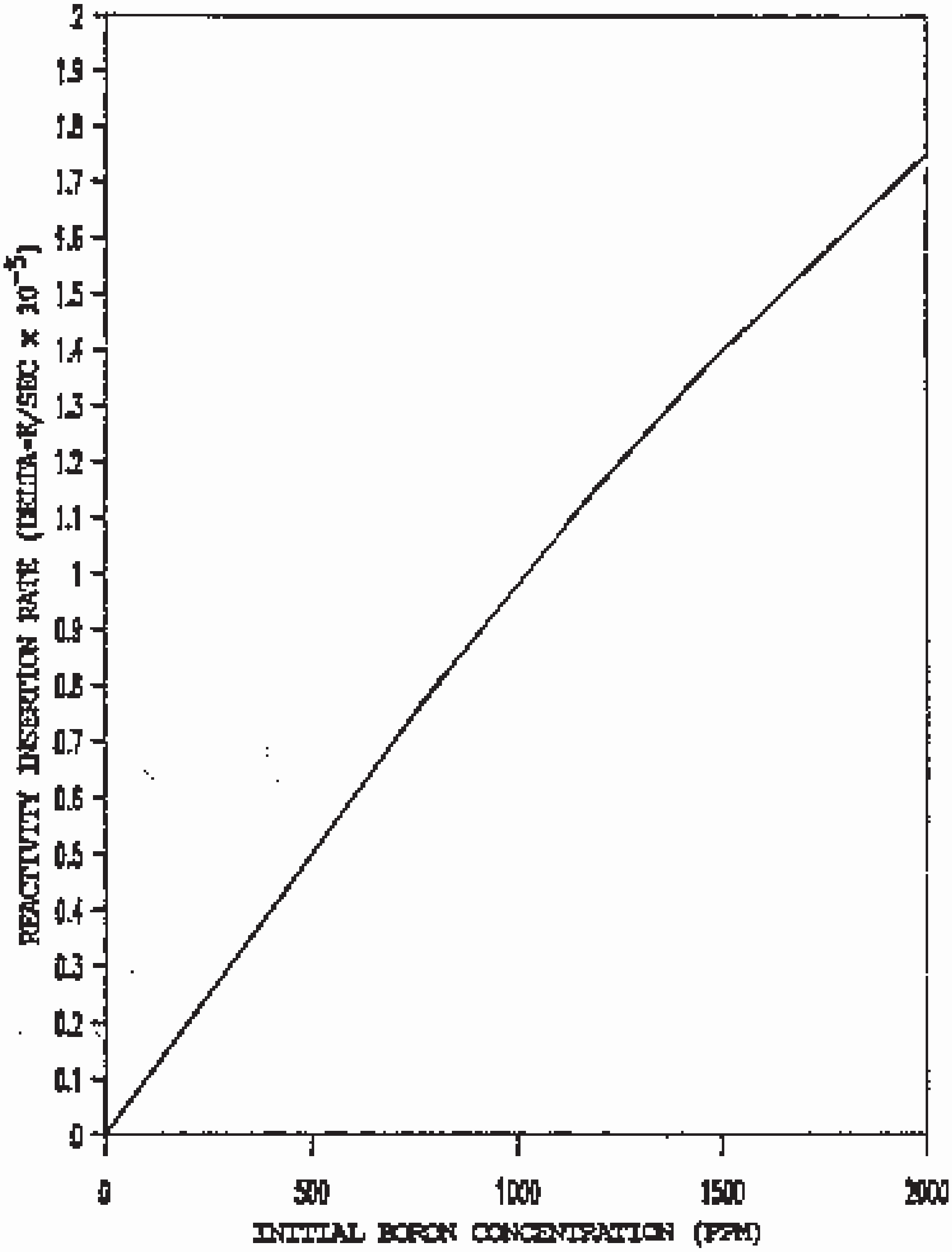
FIGURE 15.2.3-1
TRANSIENT RESPONSE TO DROPPED ROD CLUSTER CONTROL ASSEMBLY



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

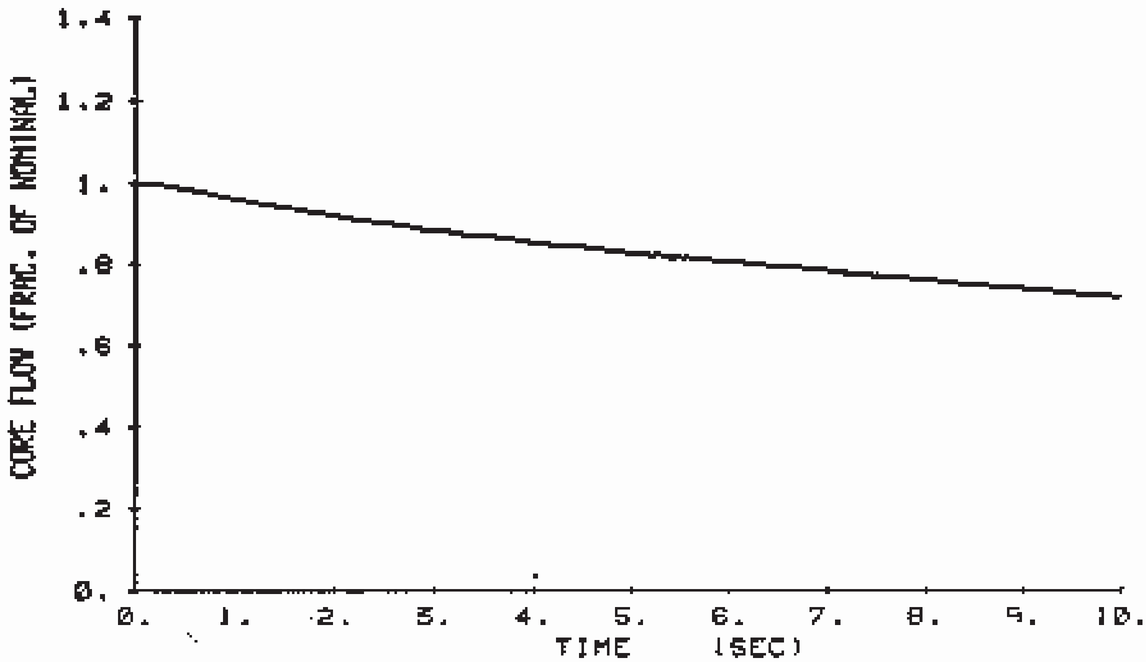
FIGURE 15.2.3-2
TRANSIENT RESPONSE TO DROPPED
ROD CLUSTER CONTROL ASSEMBLY



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.4-1
VARIATION IN REACTIVITY INSERTION RATE WITH INITIAL BORON CONCENTRATION FOR A DILUTION RATE OF 262 GPM

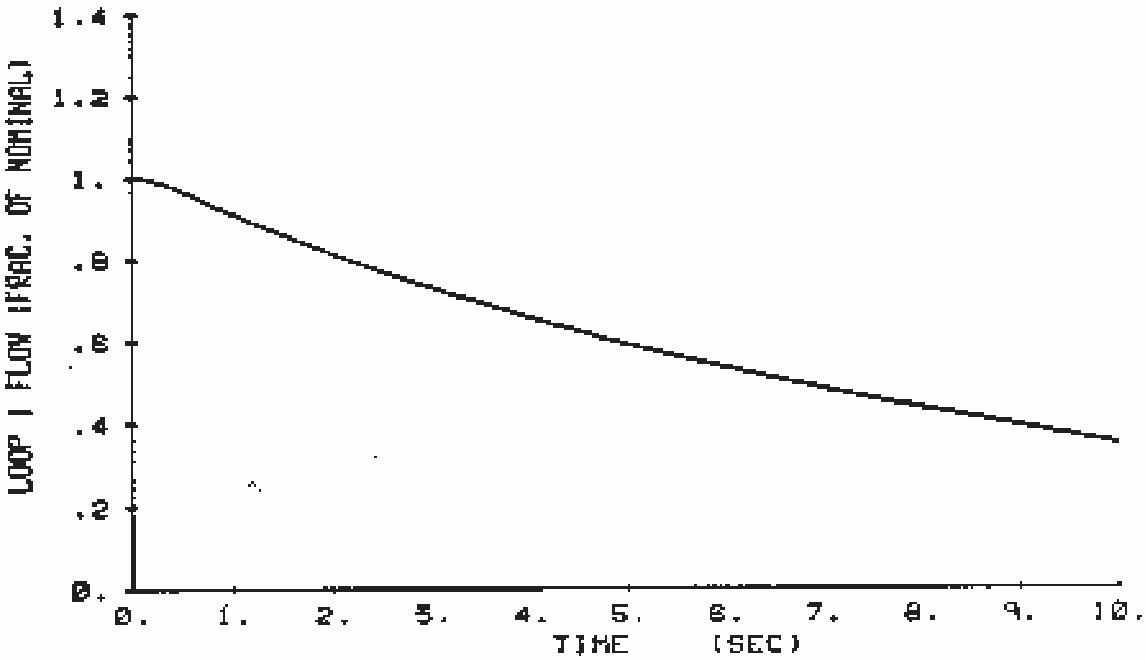


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.5-1
ALL LOOPS OPERATING
TWO LOOPS COASTING DOWN
CORE FLOW VERSUS TIME
HISTORICAL

Revision 11 November 1996

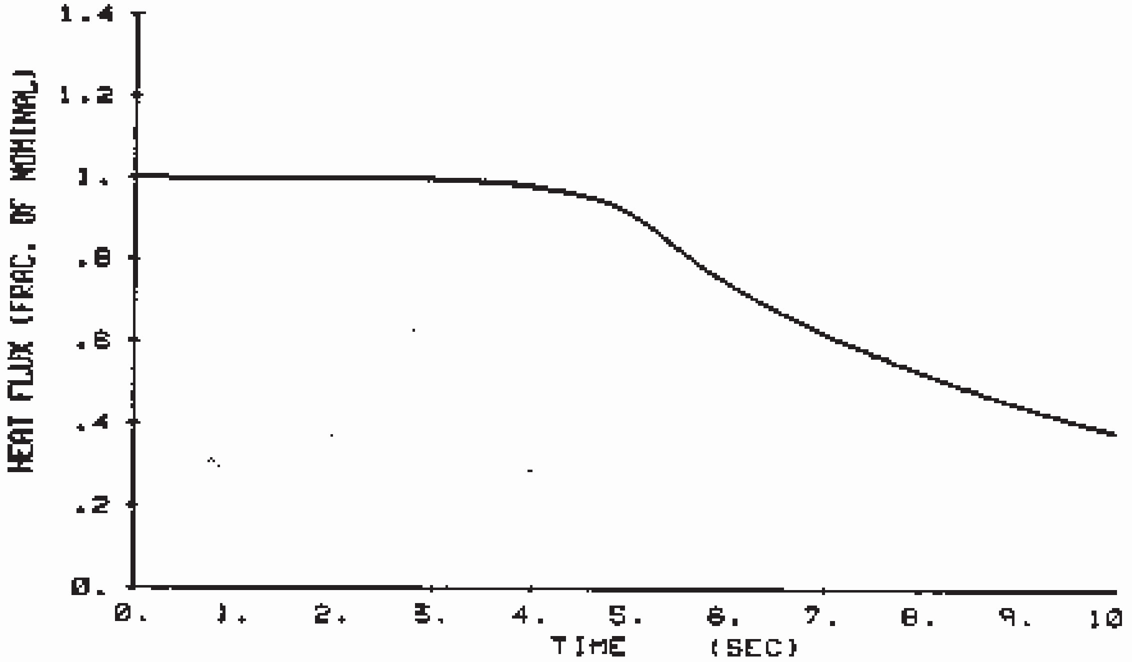


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.5-2
ALL LOOPS OPERATING
TWO LOOPS COASTING DOWN
FAILED LOOP FLOW VERSUS TIME
HISTORICAL

Revision 11 November 1996



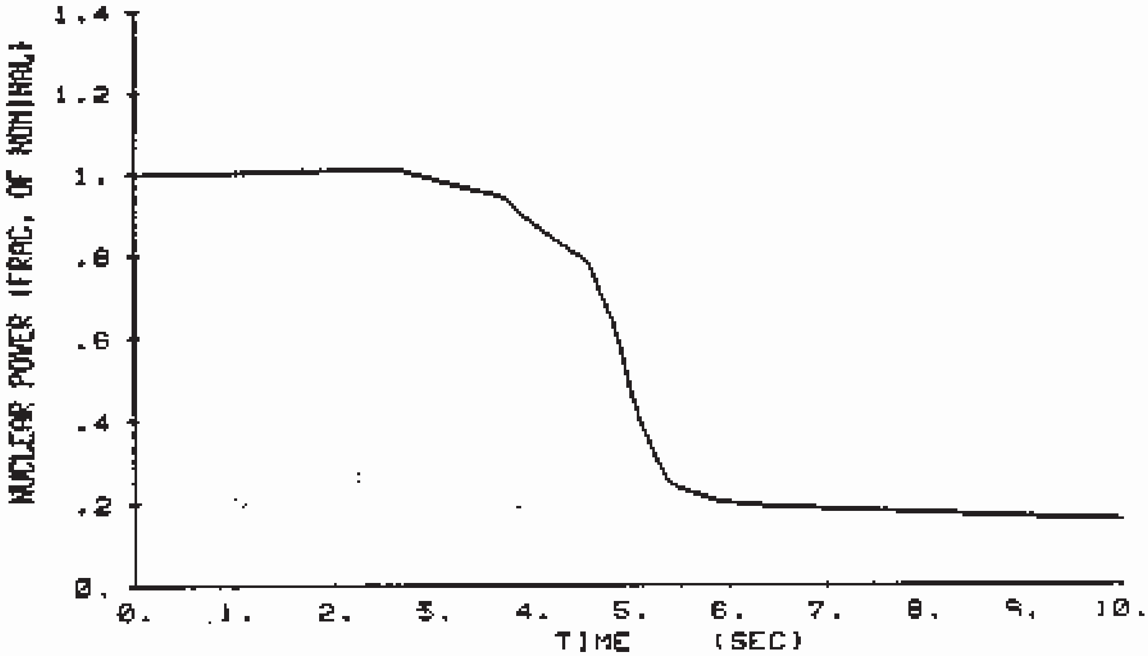
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.5-3
ALL LOOPS OPERATING
TWO LOOPS COASTING DOWN
HEAT FLUX VERSUS TIME

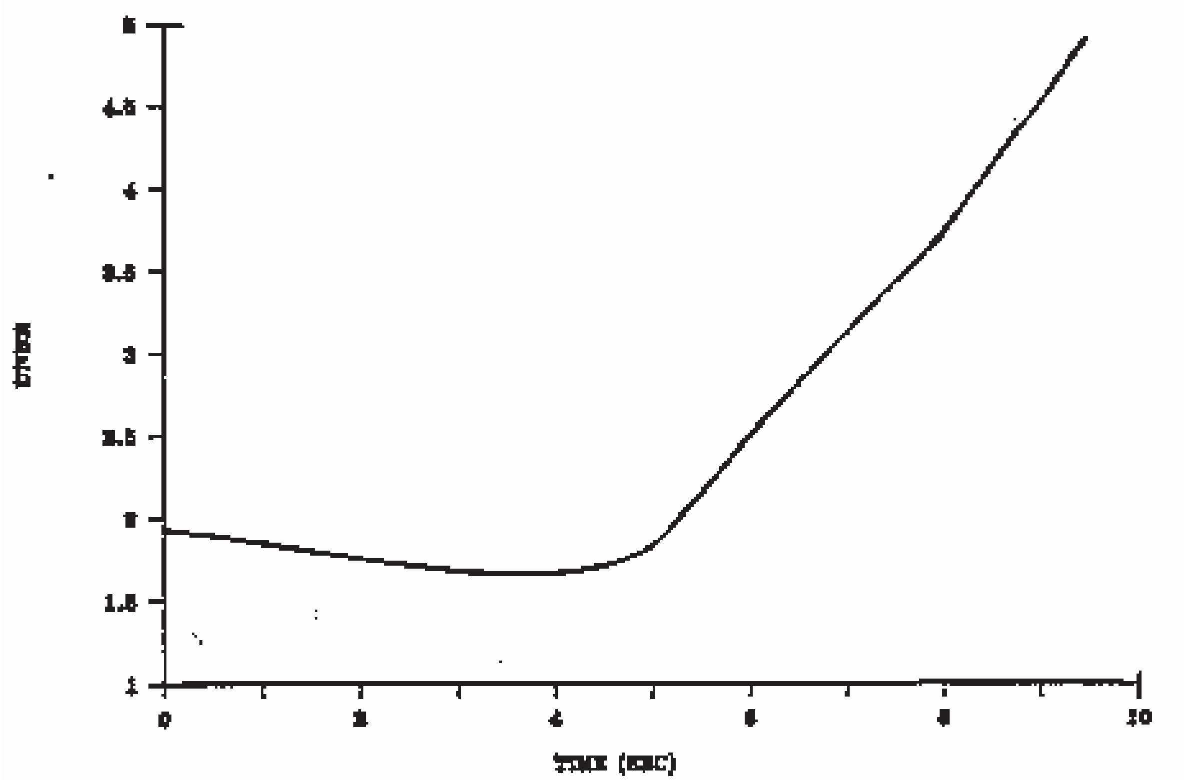
HISTORICAL

Revision 11 November 1996



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.5-4 ALL LOOPS OPERATING TWO LOOPS COASTING DOWN NUCLEAR POWER VERSUS TIME **HISTORICAL**

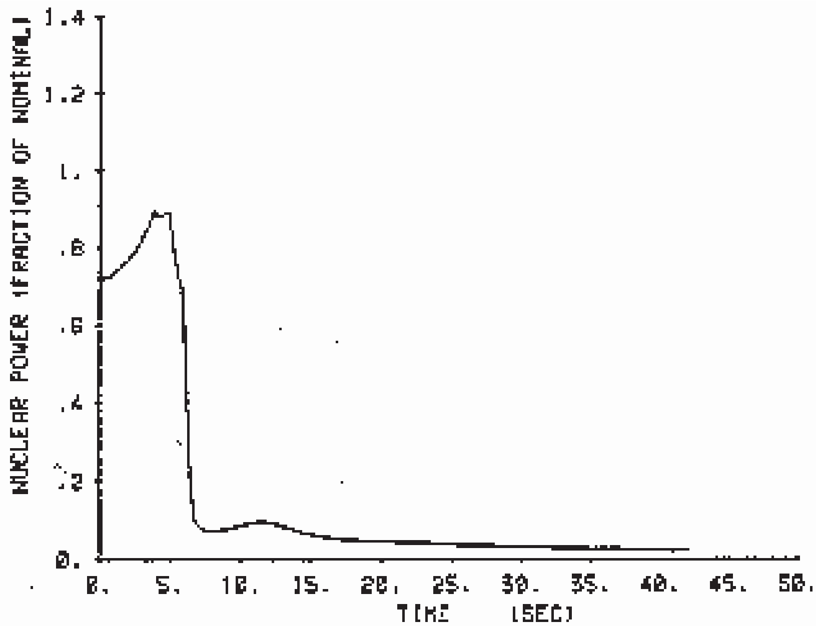
Revision 11 November 1996



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.5-5
ALL LOOPS OPERATING
TWO LOOPS COASTING DOWN
DNBR VERSUS TIME
HISTORICAL



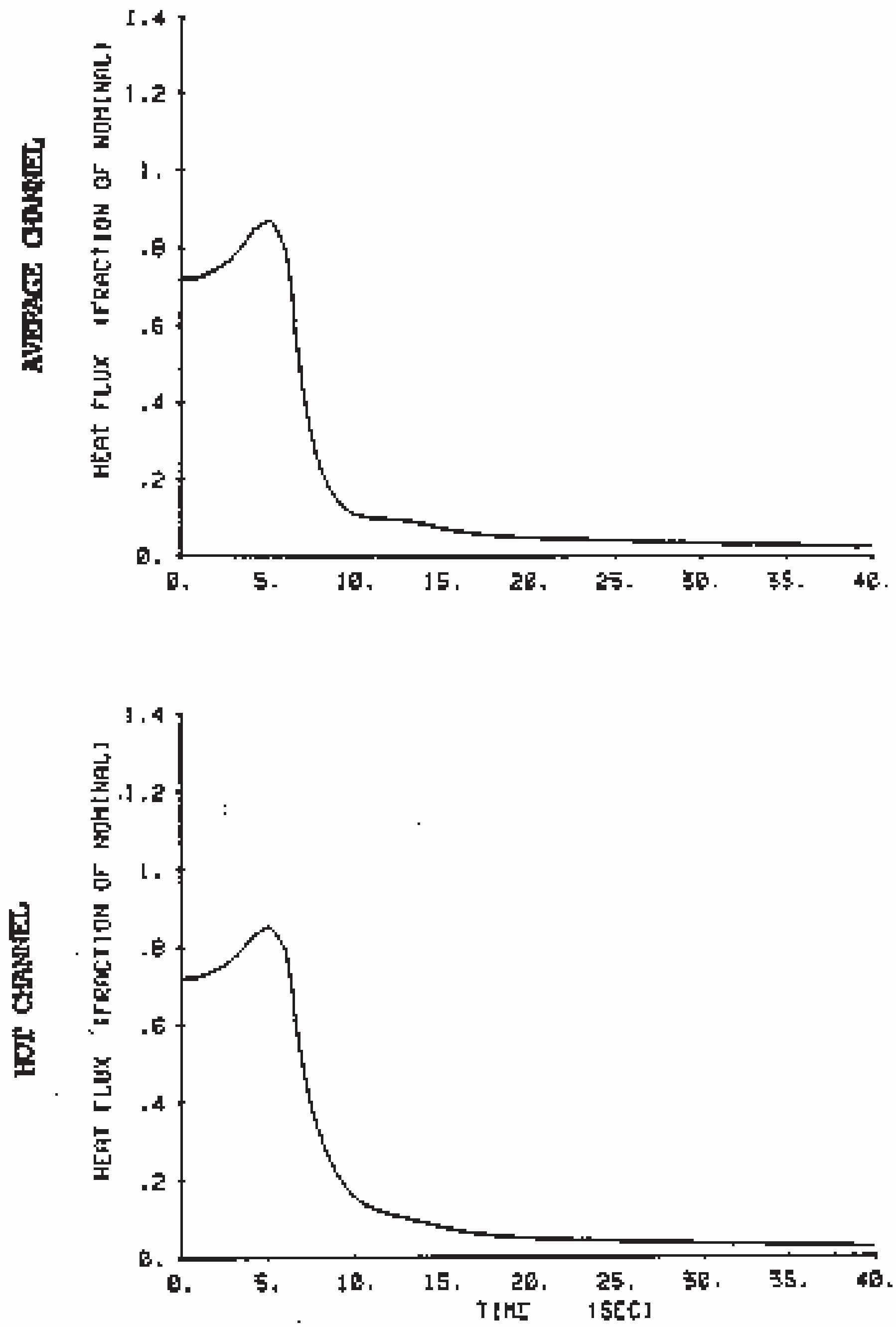
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.6-1

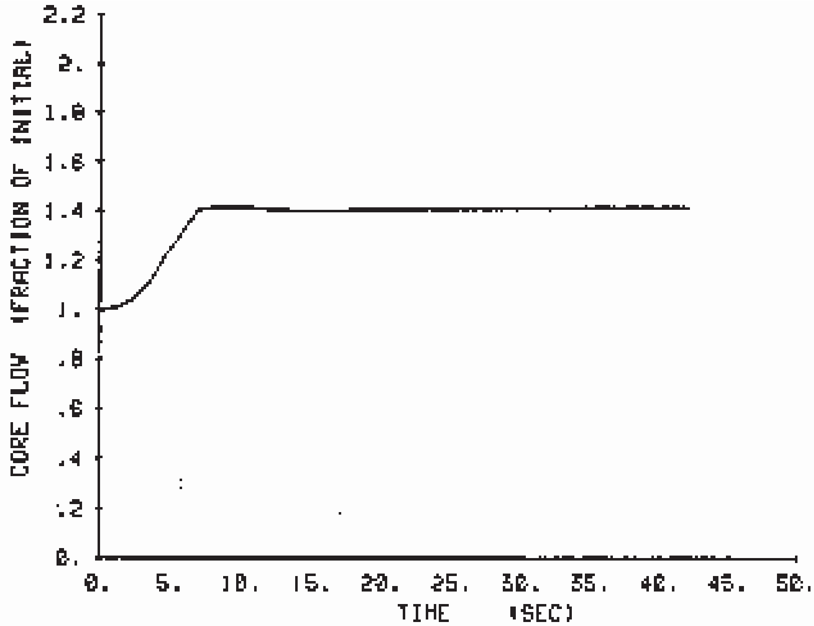
NUCLEAR POWER TRANSIENT DURING STARTUP OF AN INACTIVE LOOP

Revision 11 November 1996



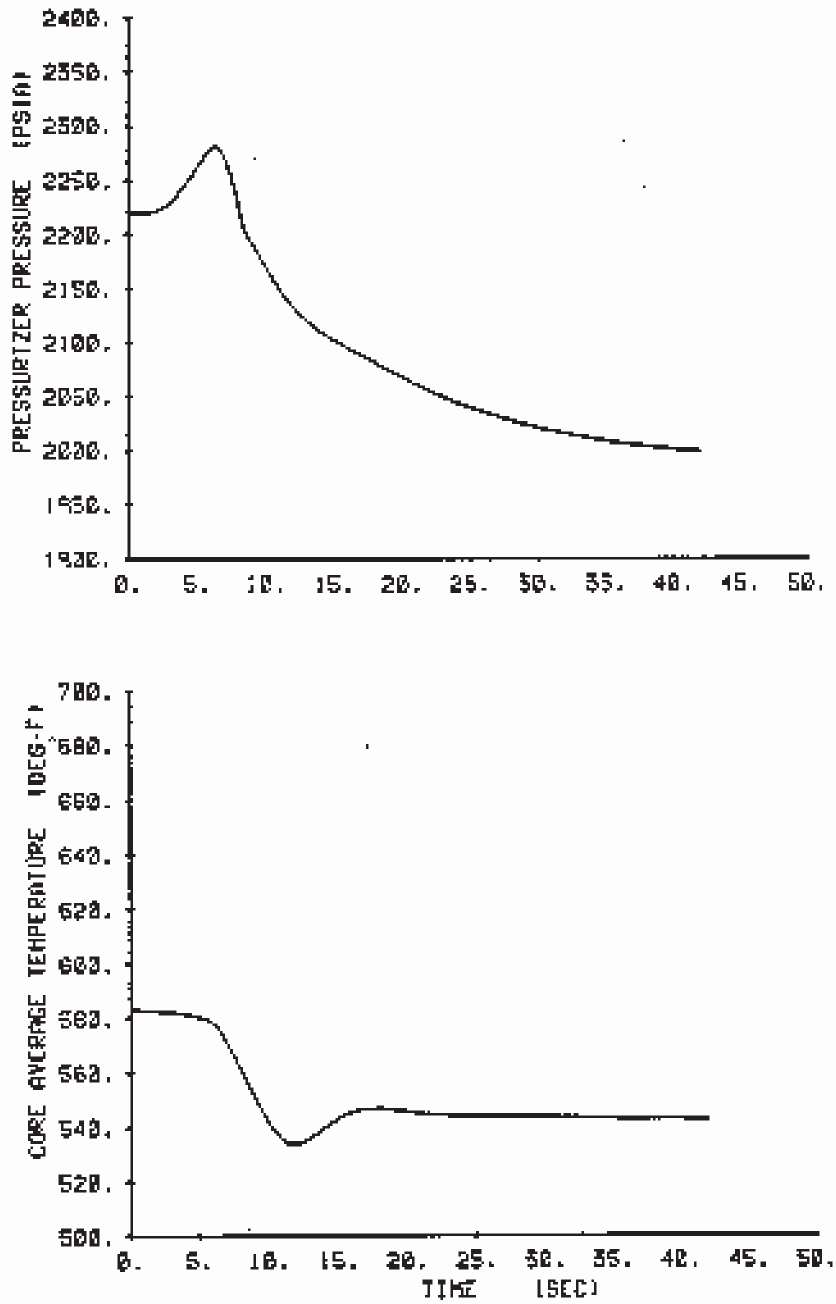
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.6-2 AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS DURING STARTUP OF AN INACTIVE LOOP

Revision 11 November 1996



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.6-3 CORE FLOW DURING STARTUP OF AN INACTIVE LOOP

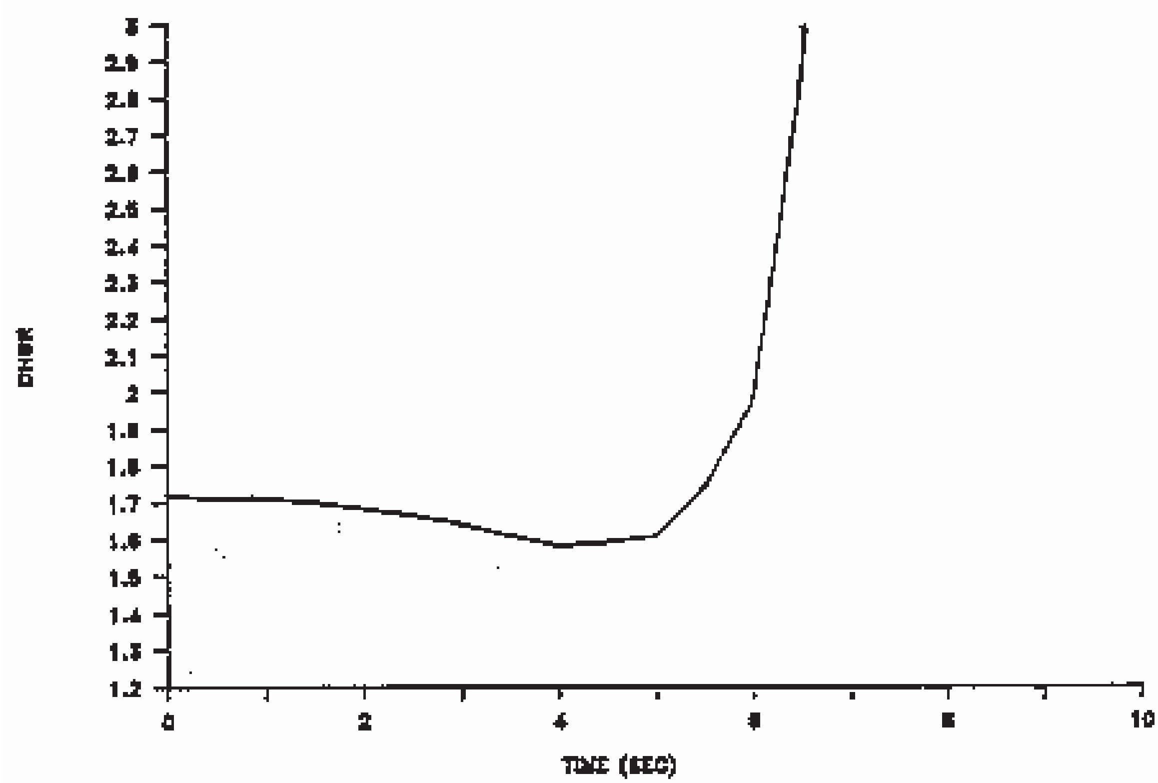
Revision 11 November 1996



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.6-4
PRESSURIZER PRESSURE TRANSIENT AND CORE AVERAGE TEMPERATURE TRANSIENT DURING STARTUP OF AN INACTIVE LOOP

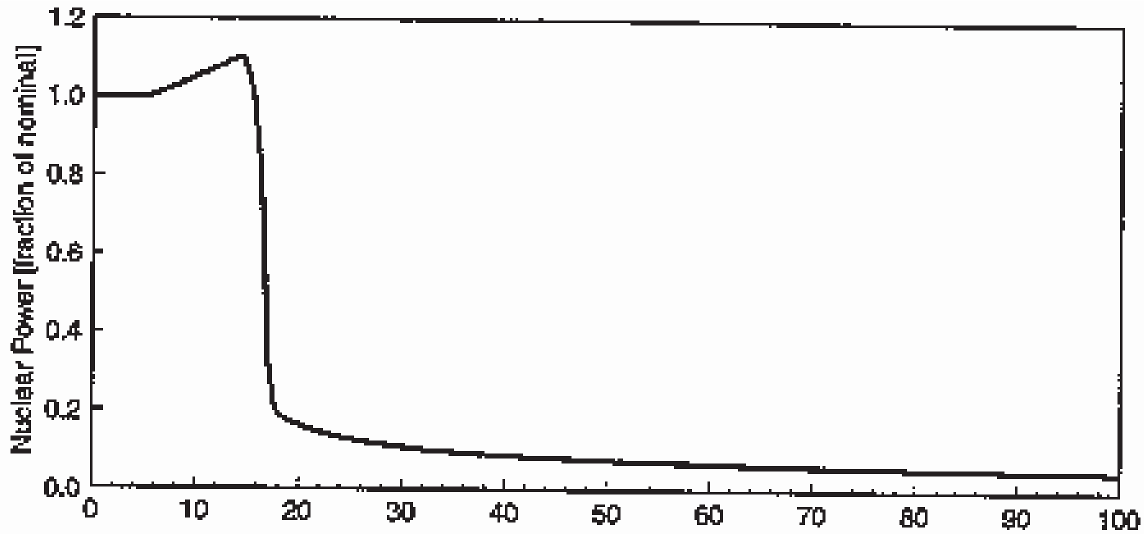


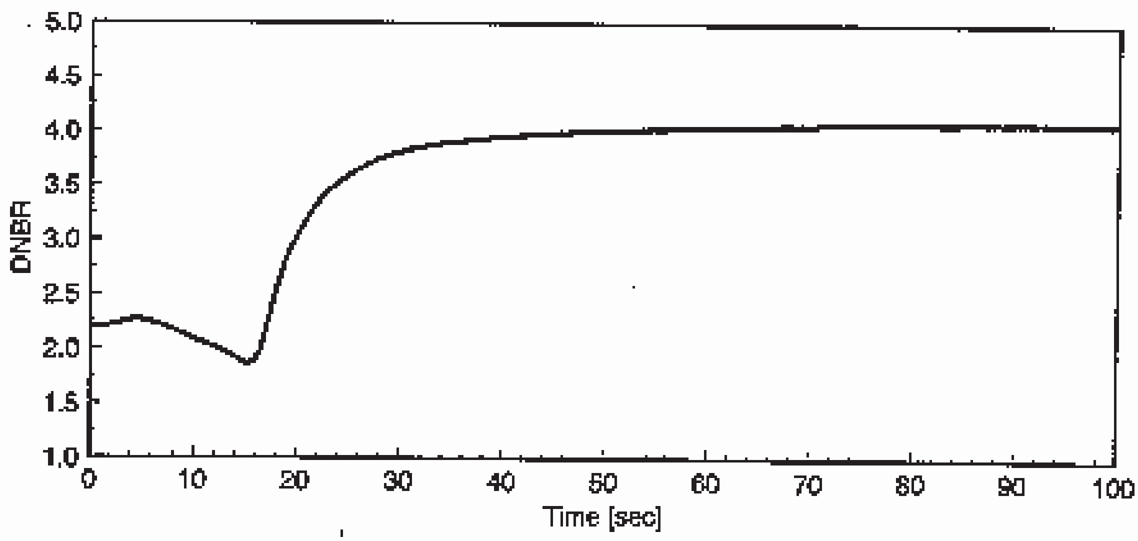
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

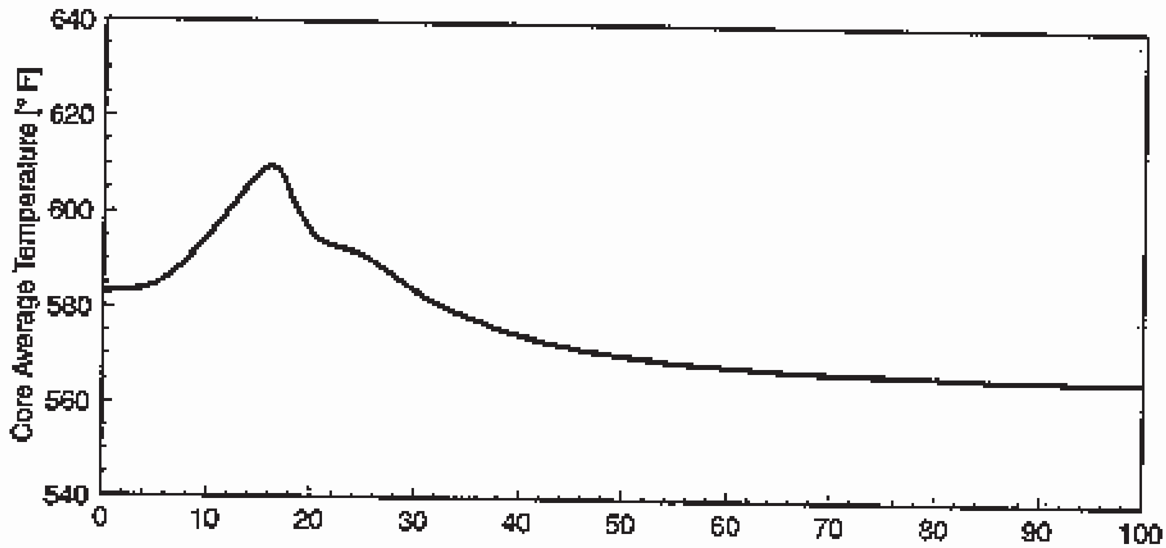
FIGURE 15.2.6-5
DNBR TRANSIENT
DURING STARTUP OF AN INACTIVE LOOP

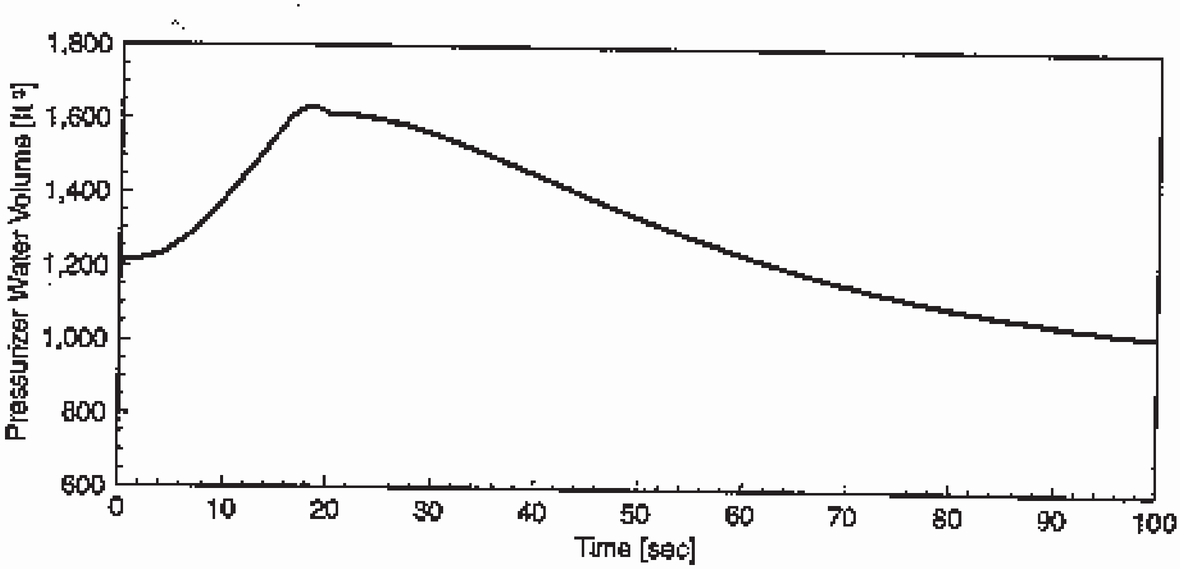
Revision 11 November 1996





LOSS OF LOAD **With Pressurizer Spray and Power Operated Relief Valve For DNB Concern at Beginning of Life**	**DIABLO CANYON UNITS 1 & 2** **FIGURE 15.2.7-1**

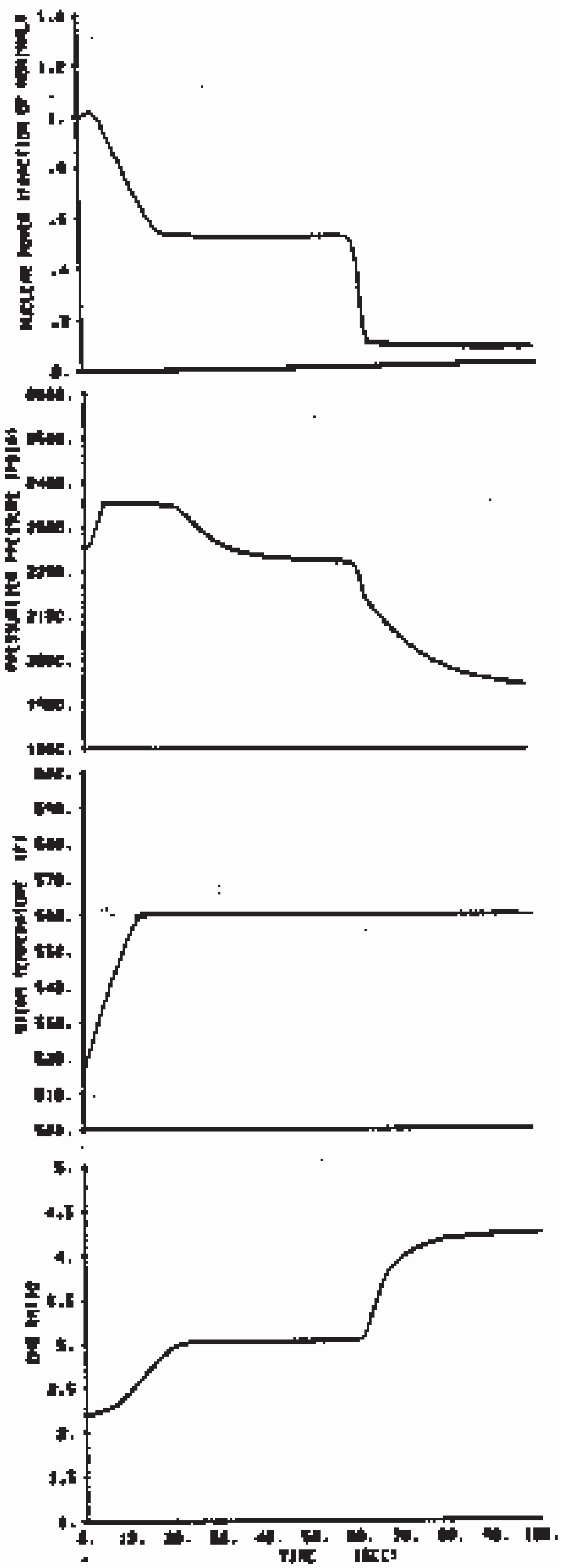




LOSS OF LOAD
With Pressurizer Spray and Power Operated Relief Valve For DNB Concern at Beginning of Life

DIABLO CANYON UNITS 1 & 2

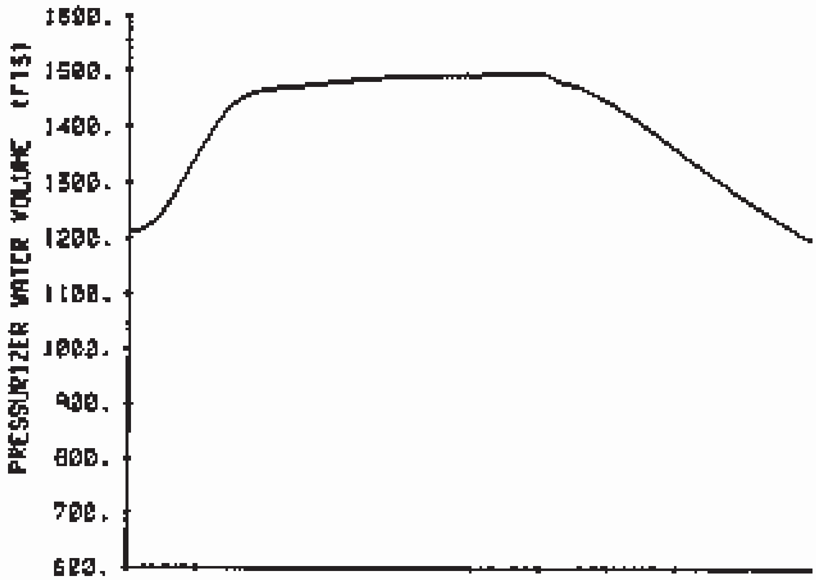
FIGURE 15.2.7-2

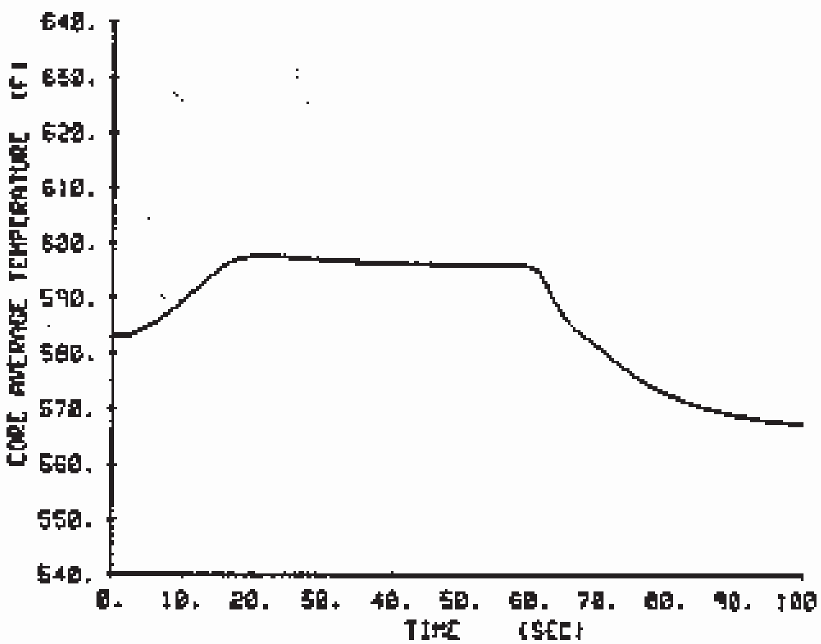


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.7-3
LOSS OF LOAD WITH PRESSURIZER SPRAY
AND POWER OPERATED RELIEF VALVE
FOR DNB CONCERN AT END OF LIFE

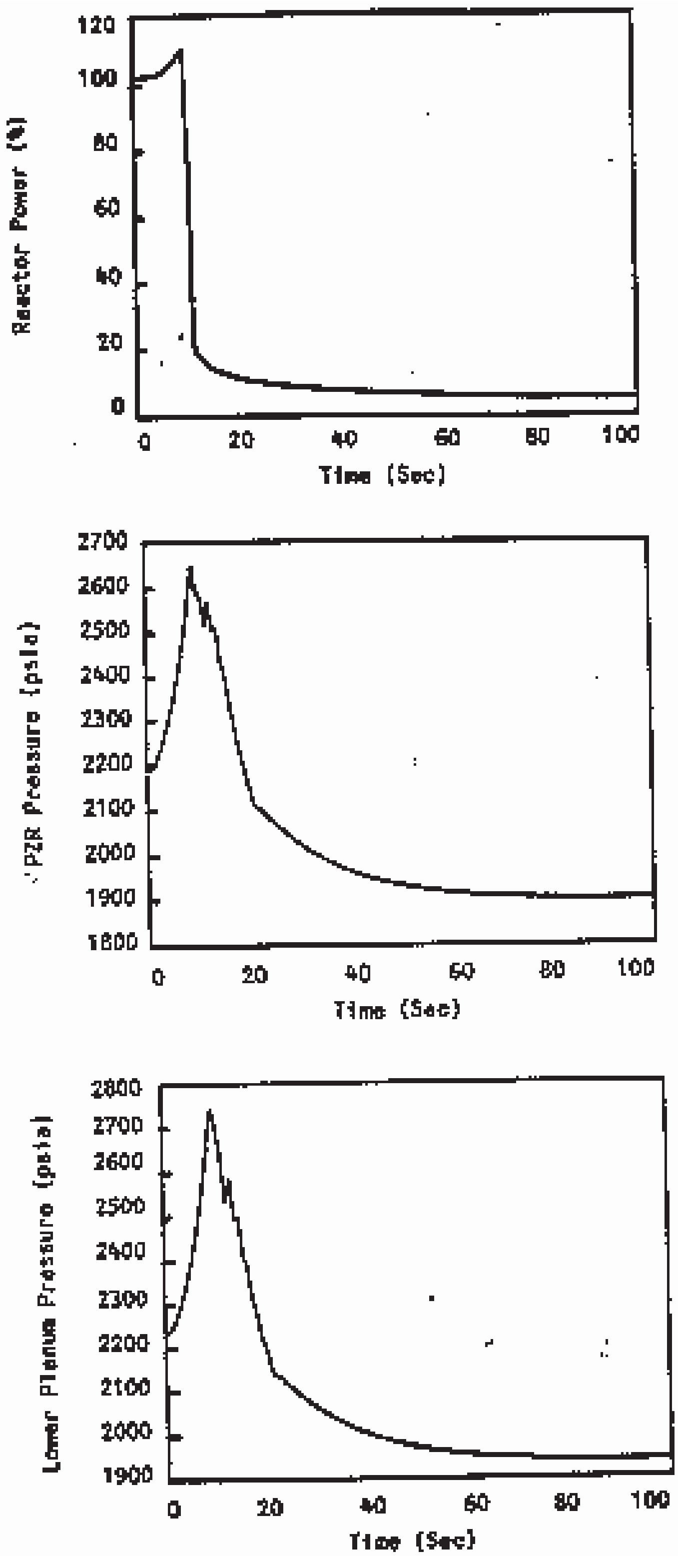




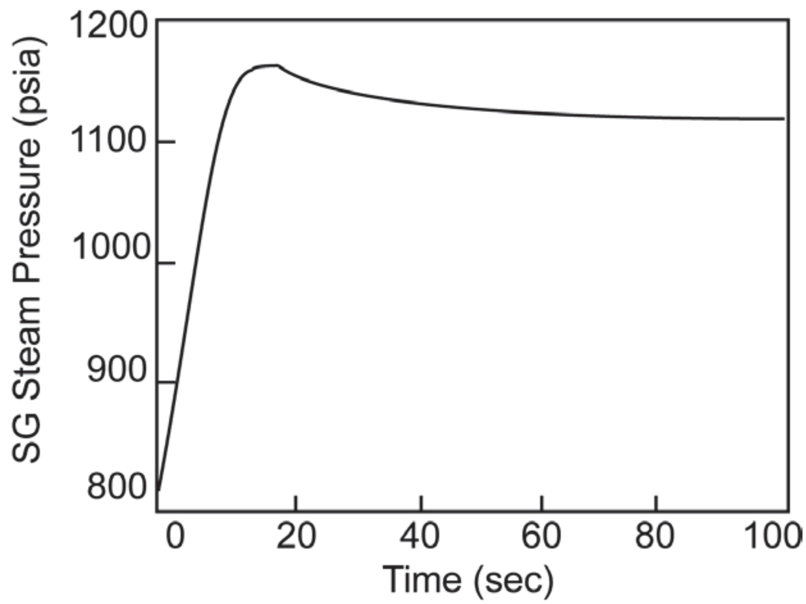
FSAR UPDATE

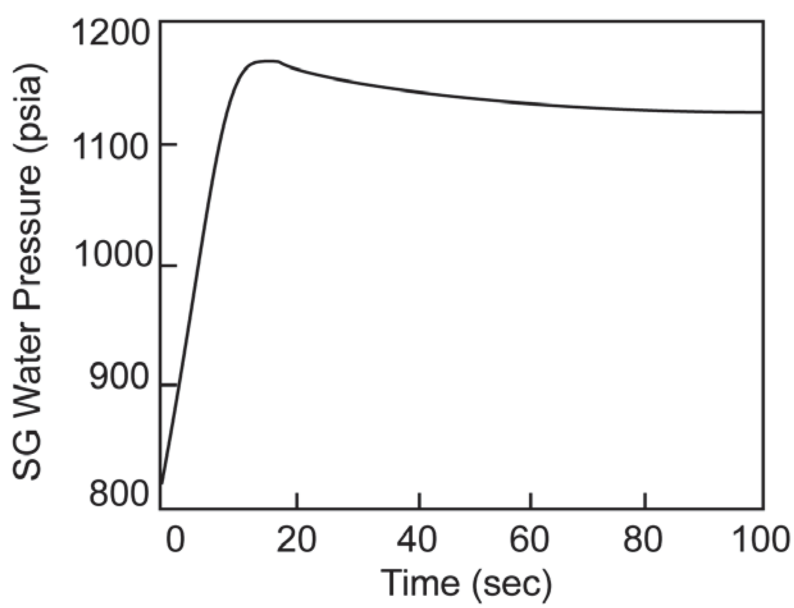
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.7-4
LOSS OF LOAD WITH PRESSURIZER SPRAY
AND POWER OPERATED RELIEF VALVE
FOR DNB CONCERN AT END OF LIFE

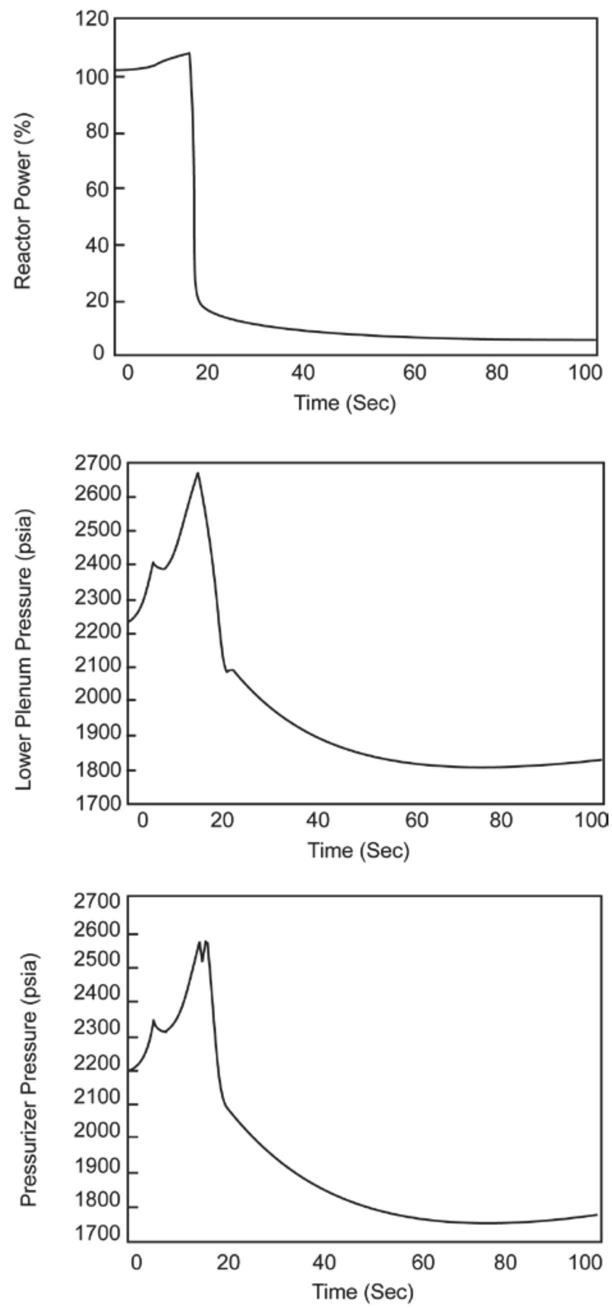


LOSS OF LOAD Without Pressurizer Spray and Power Operated Relief Valves For Overpressure Concern at Beginning of Life	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.2.7-9





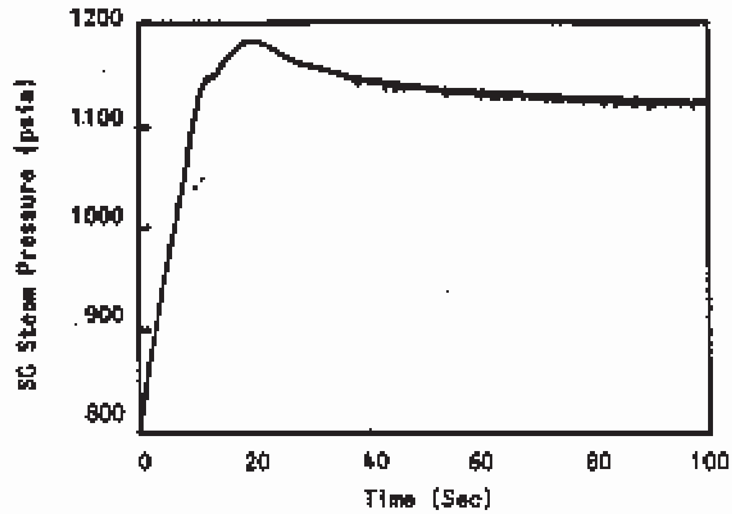
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.7-10 LOSS OF LOAD WITHOUT PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVES FOR OVERPRESSURE CONCERN AT BEGINNING OF LIFE

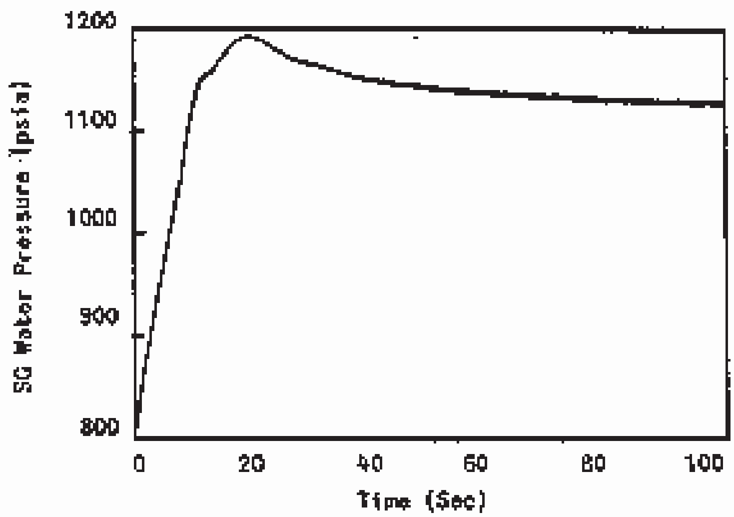


FSAR UPDATE

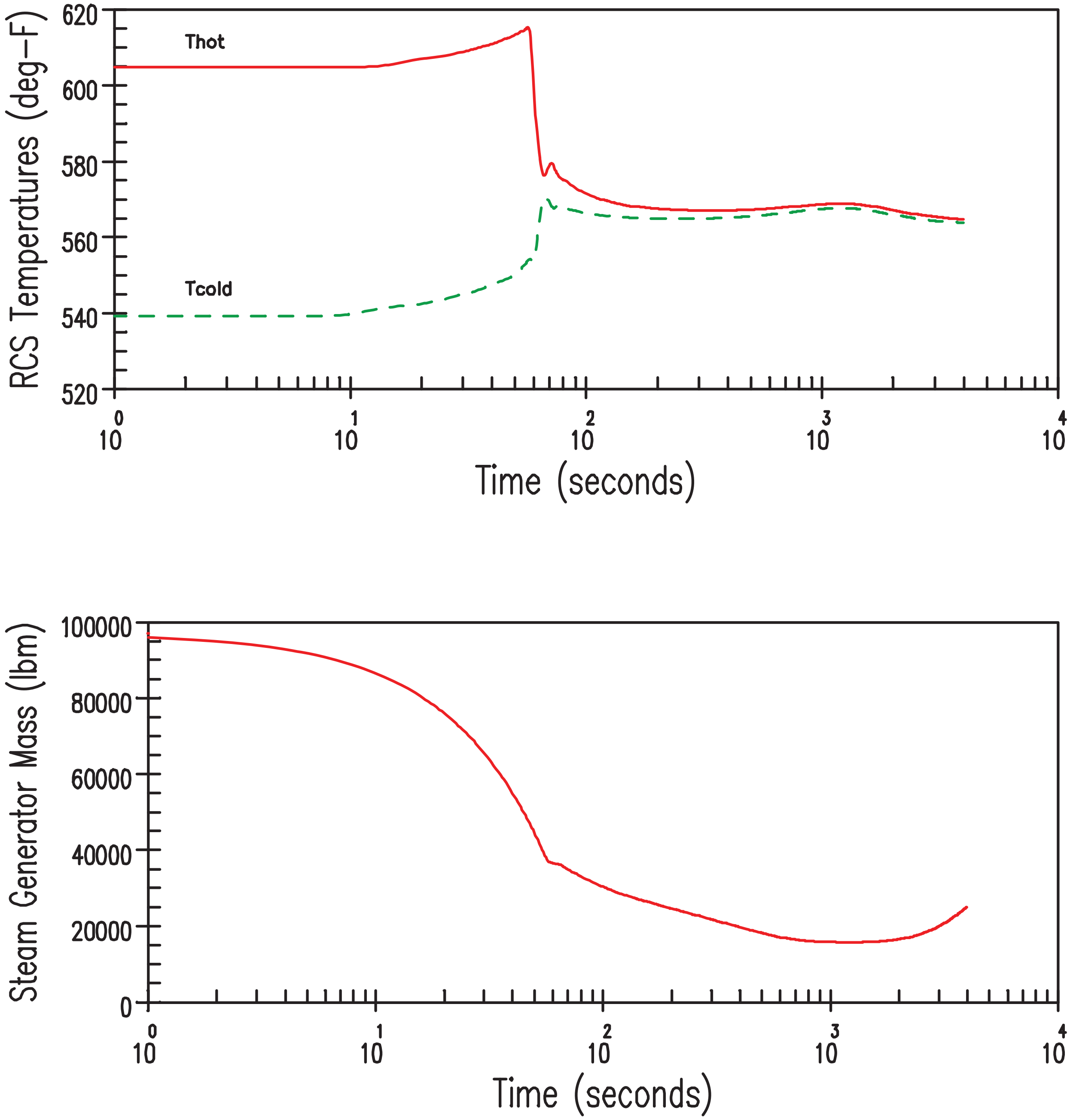
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.7-11
LOSS OF LOAD WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVES FOR OVERPRESSURE CONCERN AT BEGINNING OF LIFE

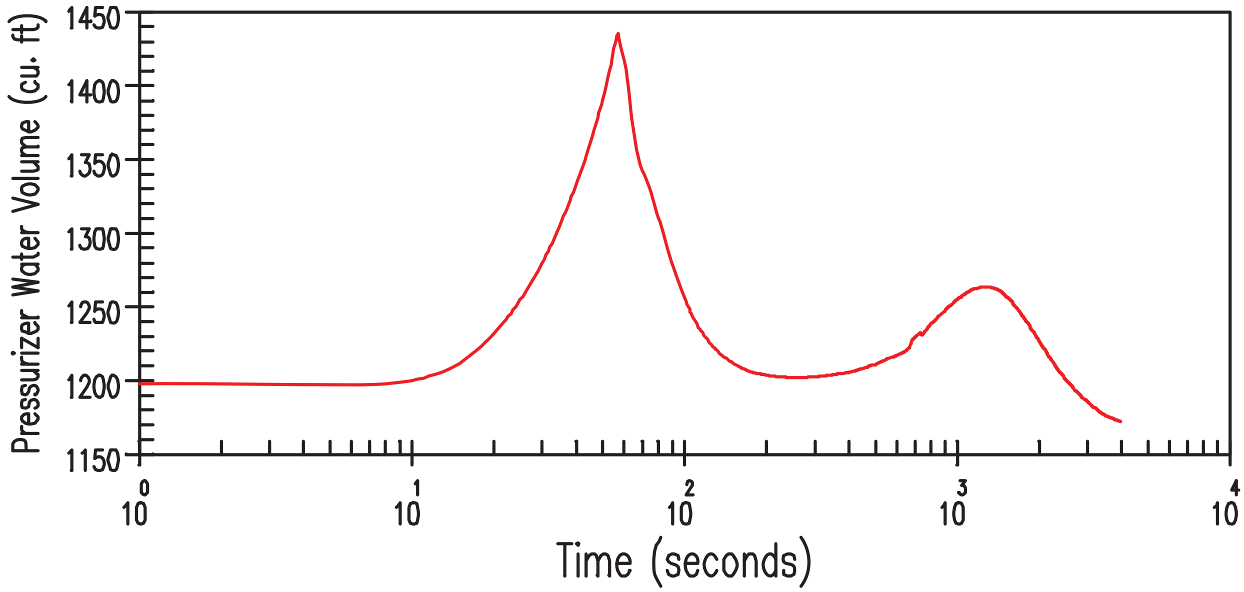


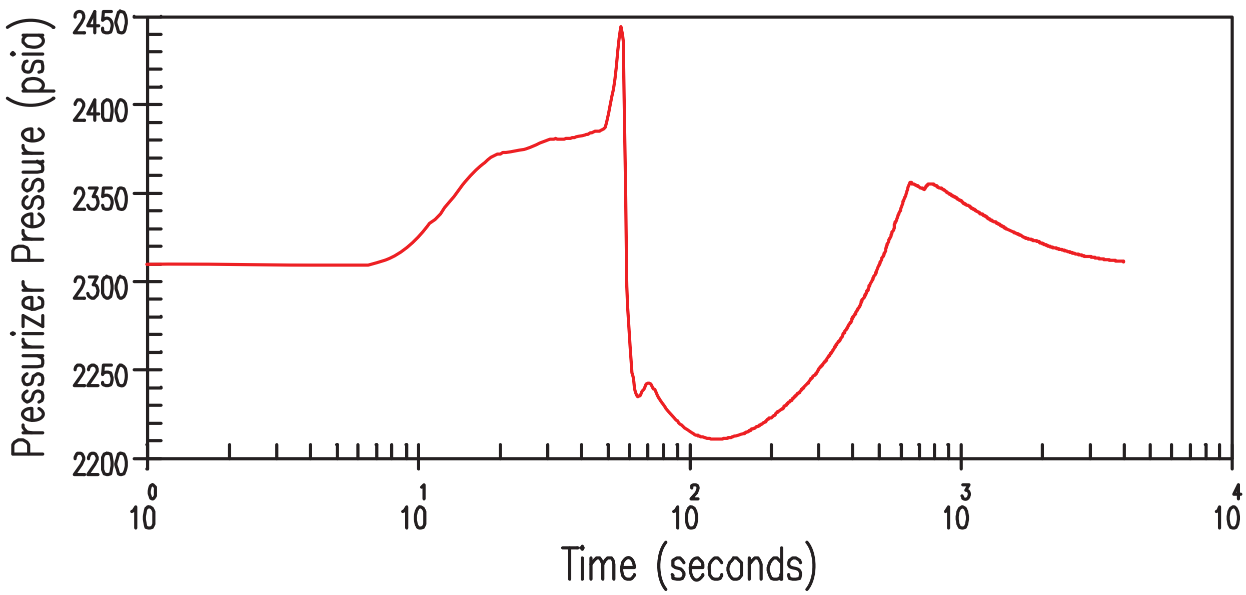


LOSS OF LOAD **With Pressurizer Spray and Power Operated Relief Valves For Overpressure Concern at Beginning of Life**	**DIABLO CANYON UNITS 1 & 2** **FIGURE 15.2.7-12**

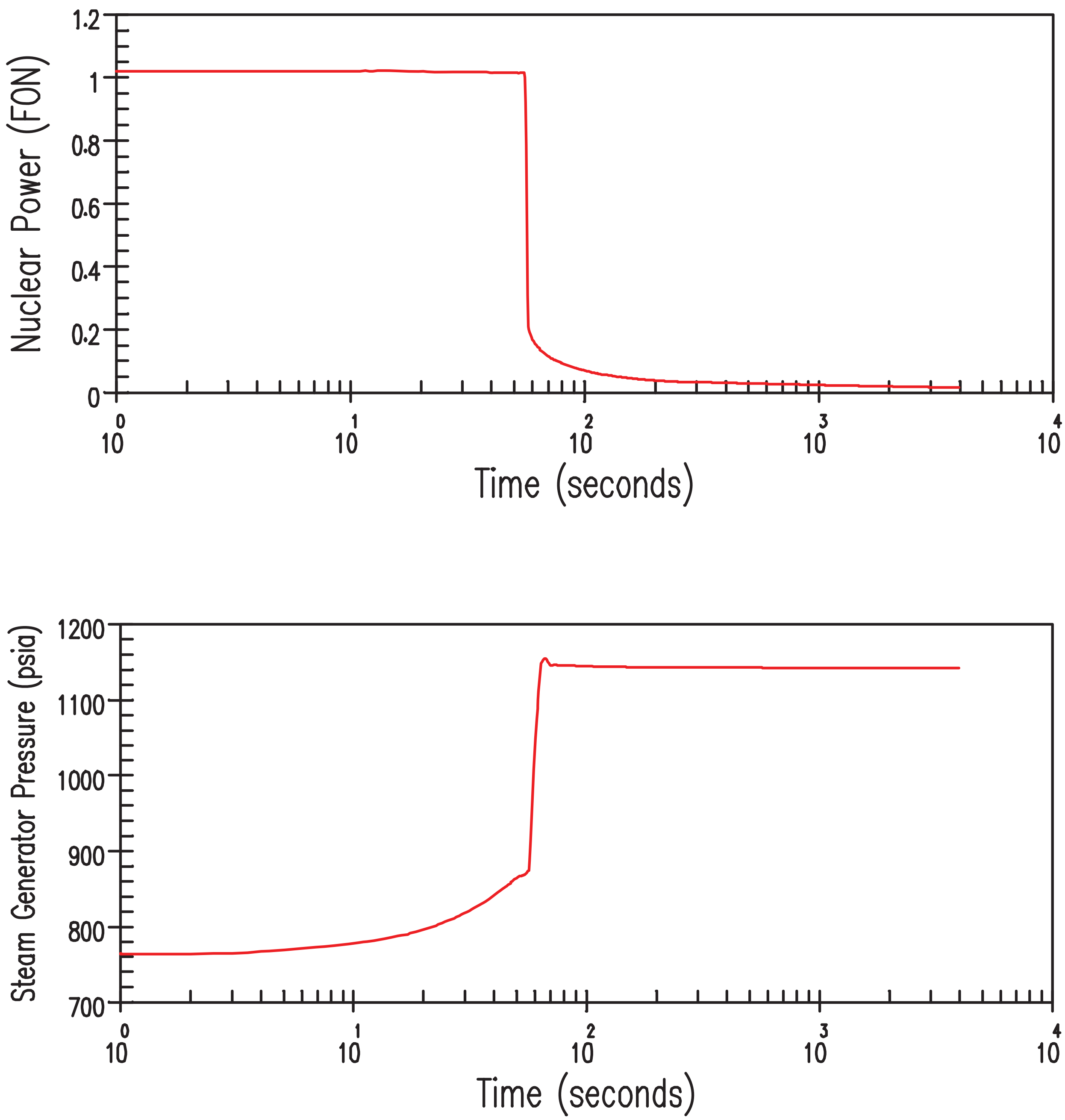


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.8-1 LOSS OF NORMAL FEEDWATER – RCS TEMPERATURES AND STEAM GENERATOR MASS TRANSIENTS

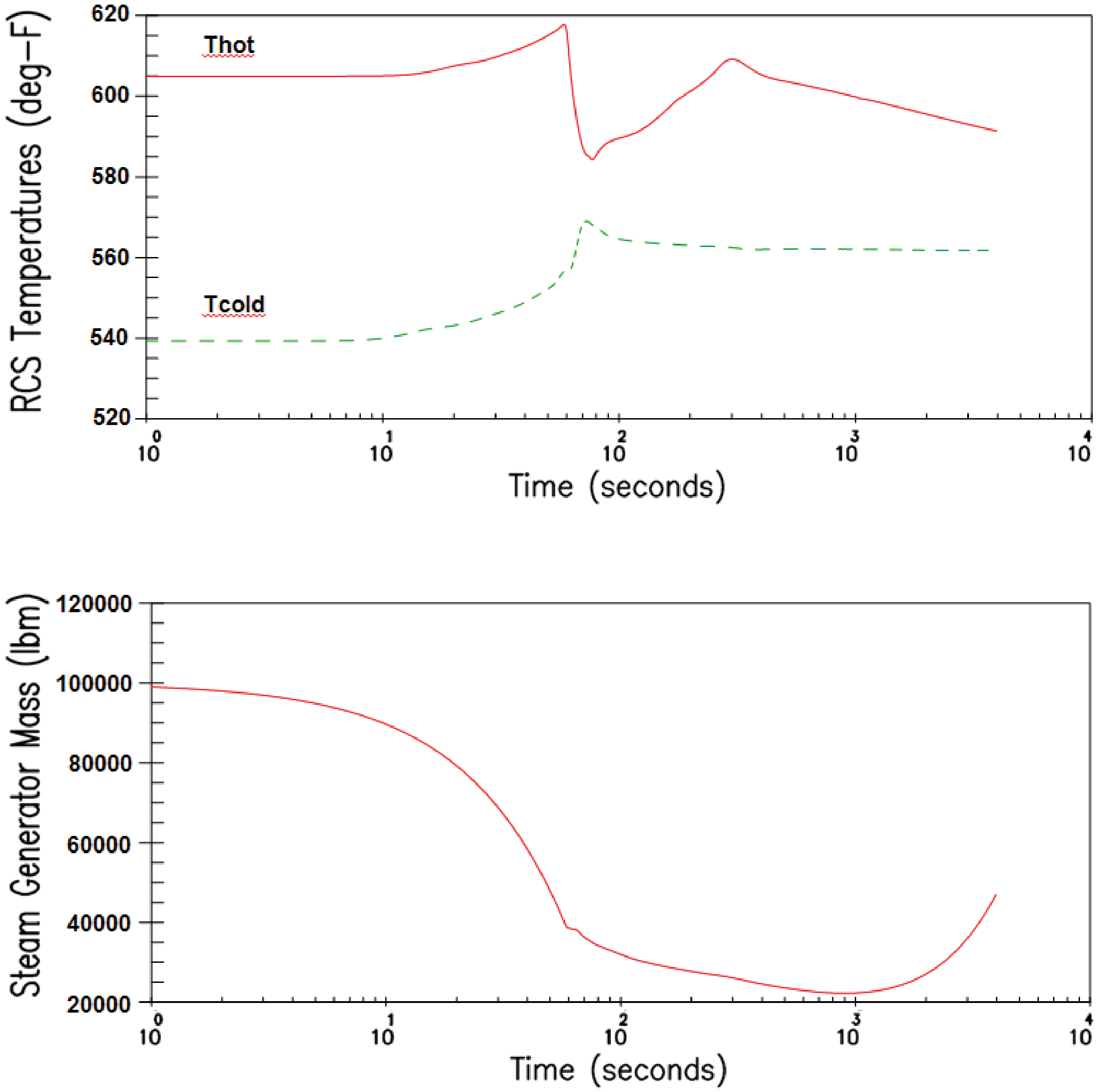




FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.8-2 LOSS OF NORMAL FEEDWATER PRESSURIZER WATER VOLUME AND PRESSURIZER PRESSURE TRANSIENTS



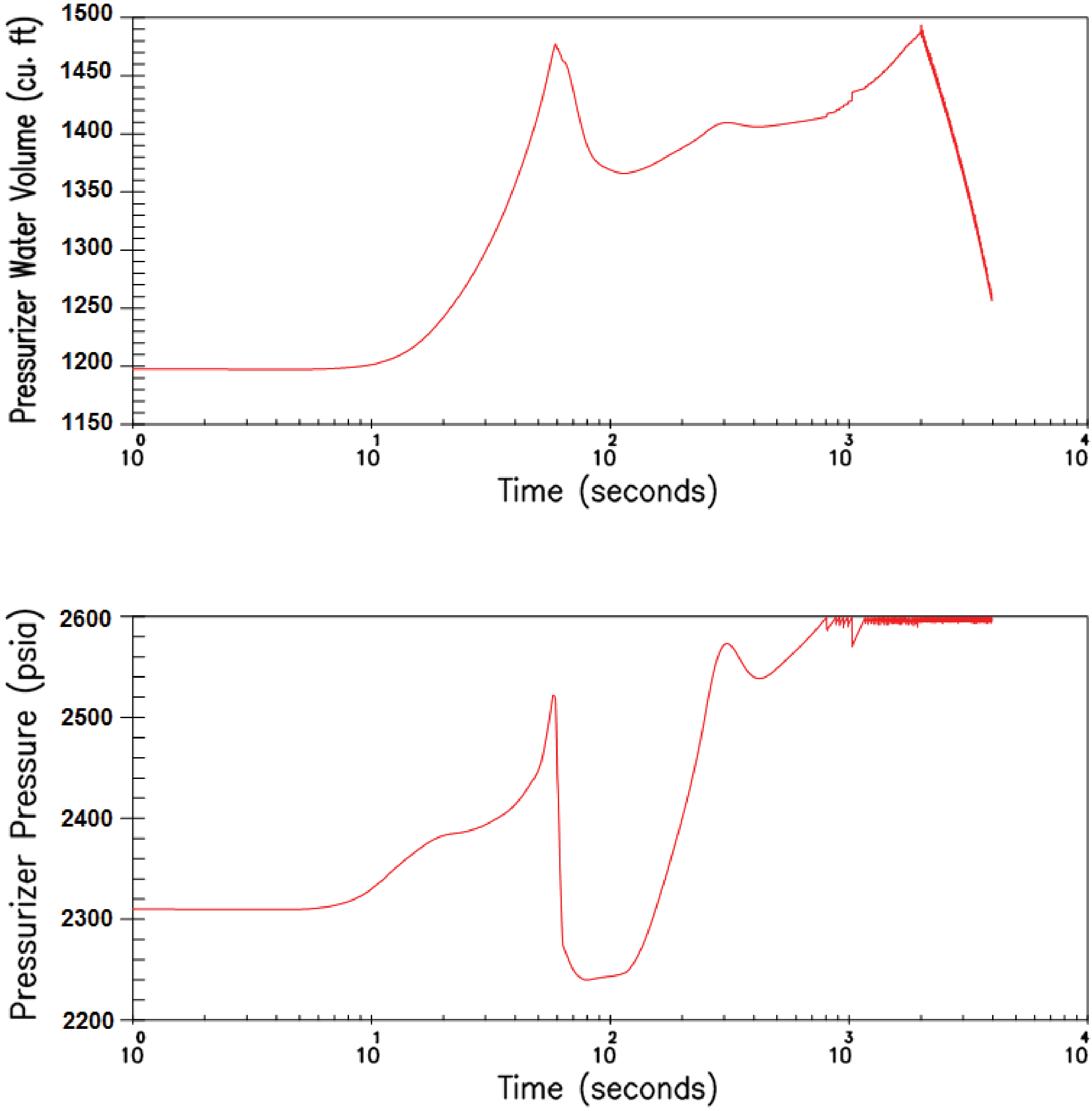
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.8-3 LOSS OF NORMAL FEEDWATER NUCLEAR POWER AND STEAM GENERATOR PRESSURE TRANSIENTS



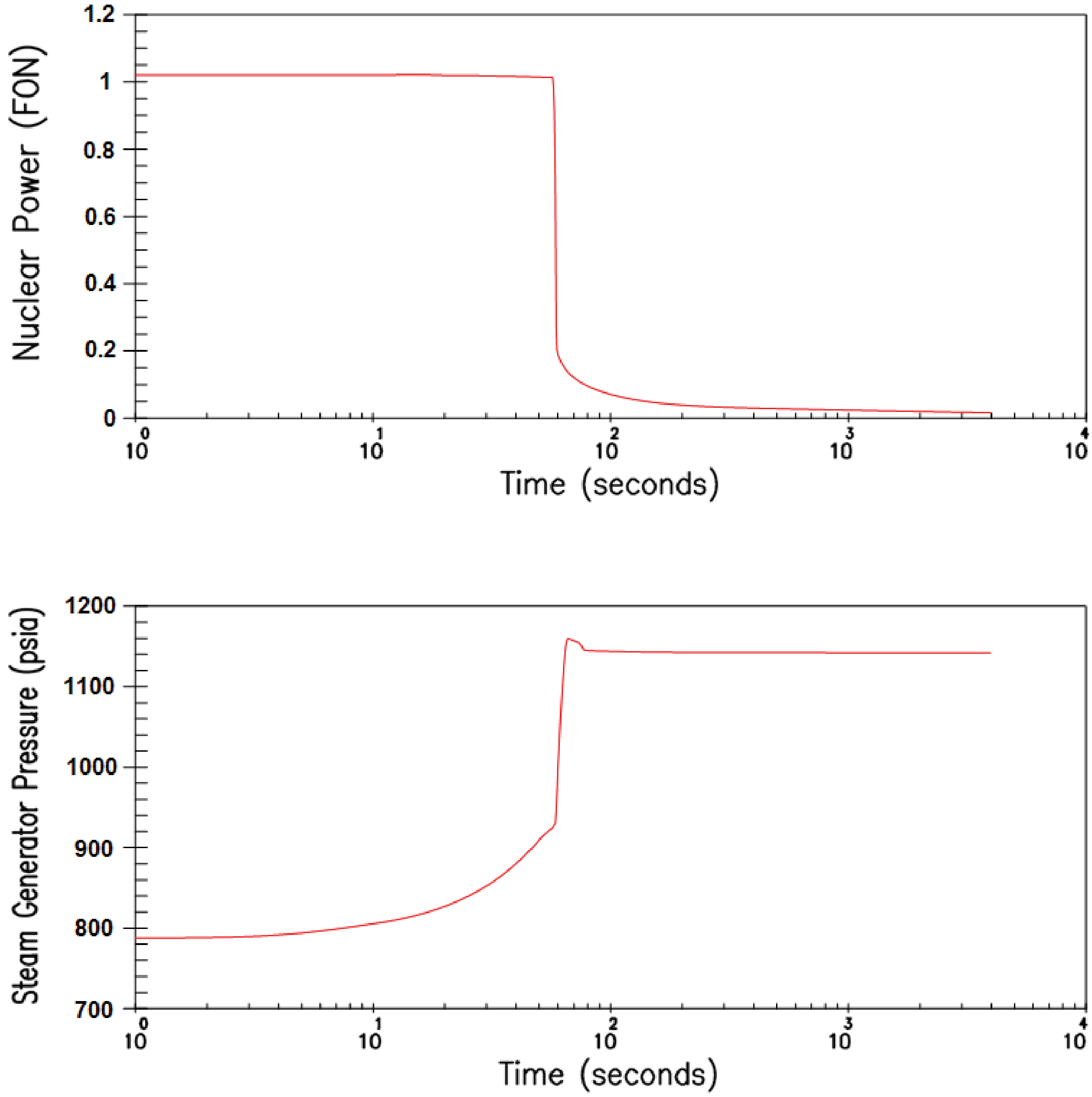
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.9-1
LOSS OF OFFSITE POWER
RCS TEMPERATURES AND STEAM
GENERATOR MASS TRANSIENTS



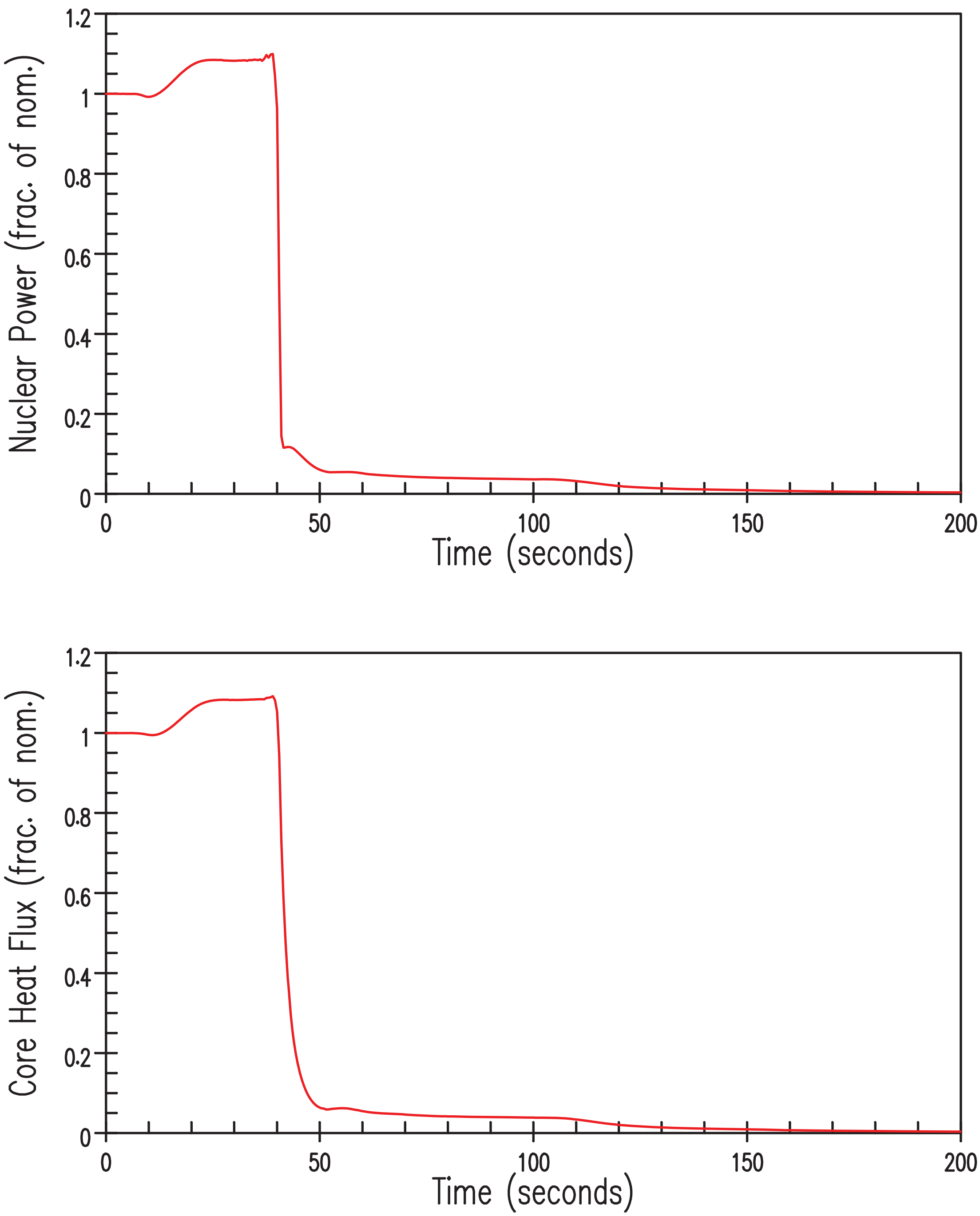
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.9-2 LOSS OF OFFSITE POWER PRESSURIZER WATER VOLUME AND PRESSURIZER PRESSURE TRANSIENTS



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.9-3
LOSS OF OFFSITE POWER
NUCLEAR POWER AND STEAM
GENERATOR PRESSURE TRANSIENTS

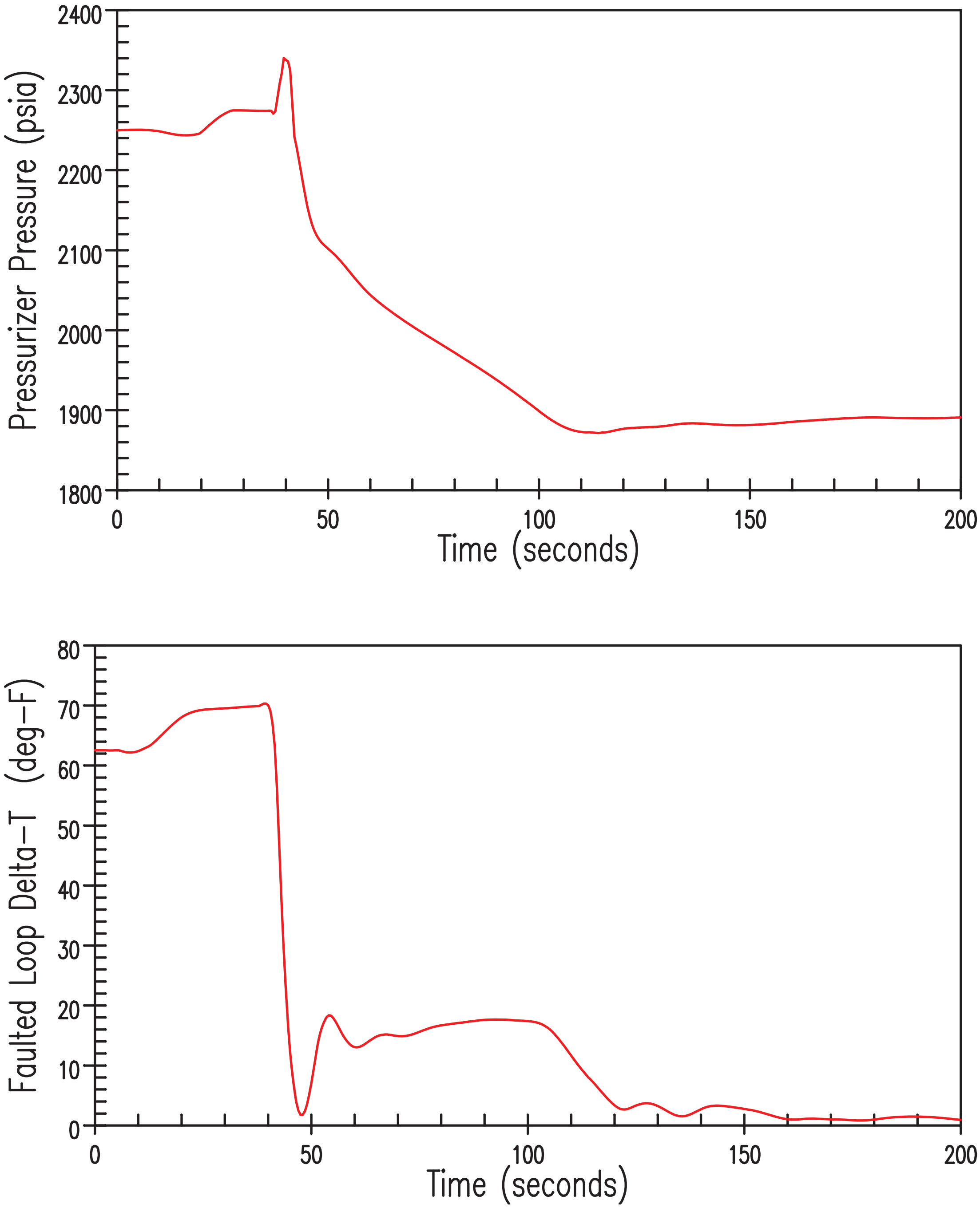


FSAR UPDATE

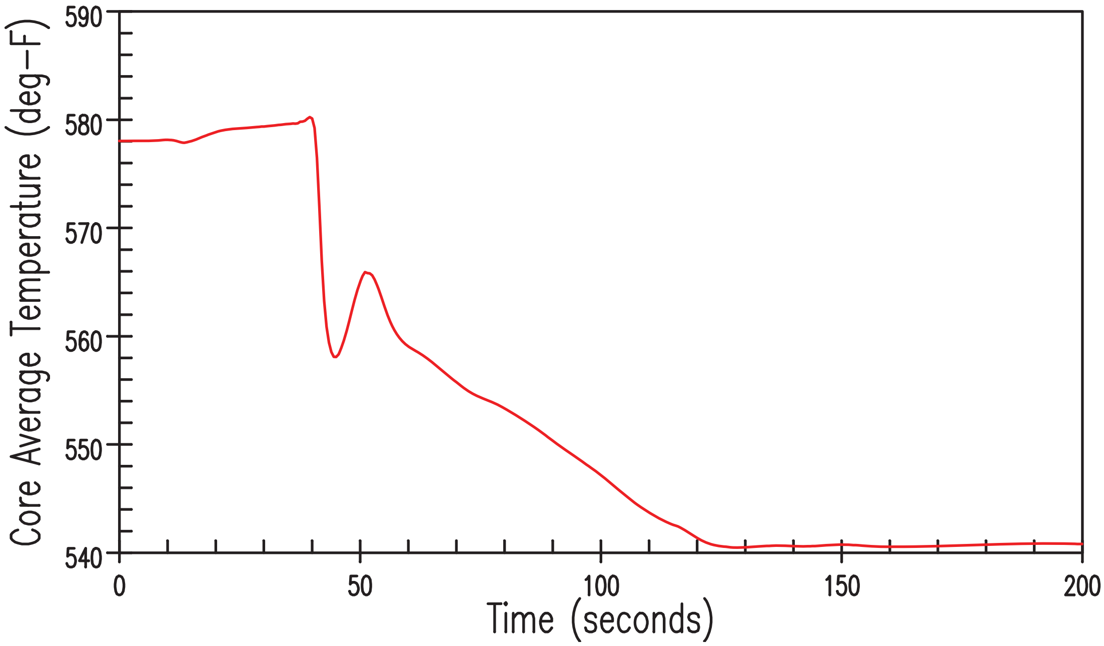
UNITS 1 AND 2
DIABLO CANYON SITE

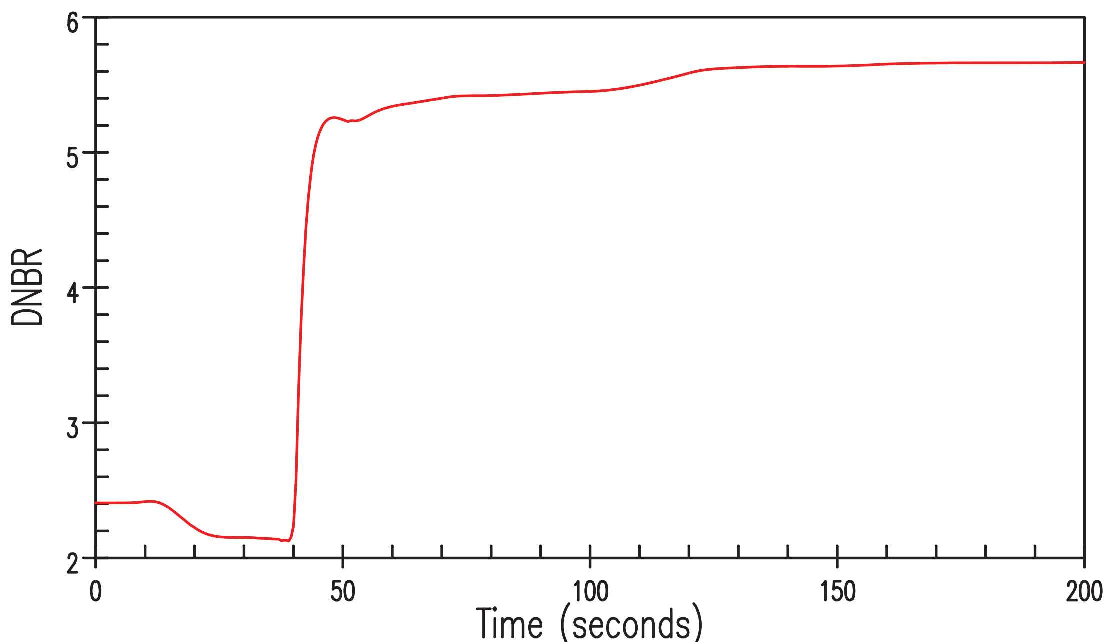
FIGURE 15.2.10-1

MAIN FEEDWATER REGULATING VALVE MALFUNCTION FULL POWER, MANUAL ROD CONTROL NUCLEAR POWER AND AVERAGE CHANNEL CORE HEAT FLUX TRANSIENTS



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.10-2 MAIN FEEDWATER REGULATING VALVE MALFUNCTION FULL POWER, MANUAL ROD CONTROL PRESSURIZER PRESSURE AND FAULTED LOOP DELTA-T TRANSIENTS

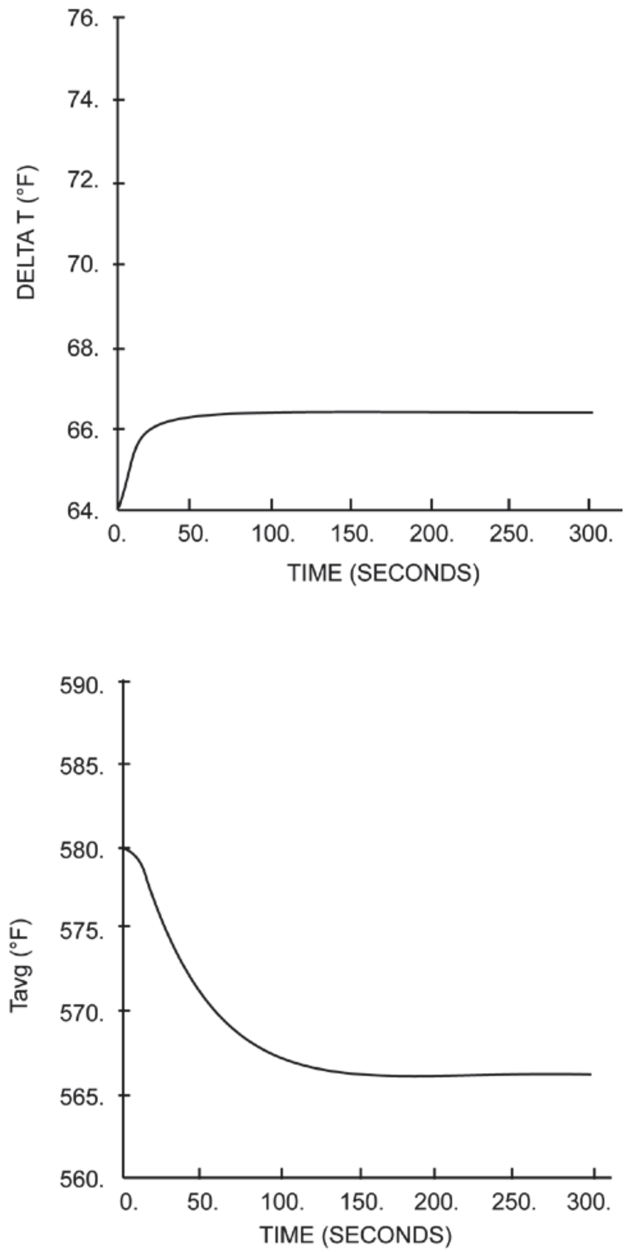




FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

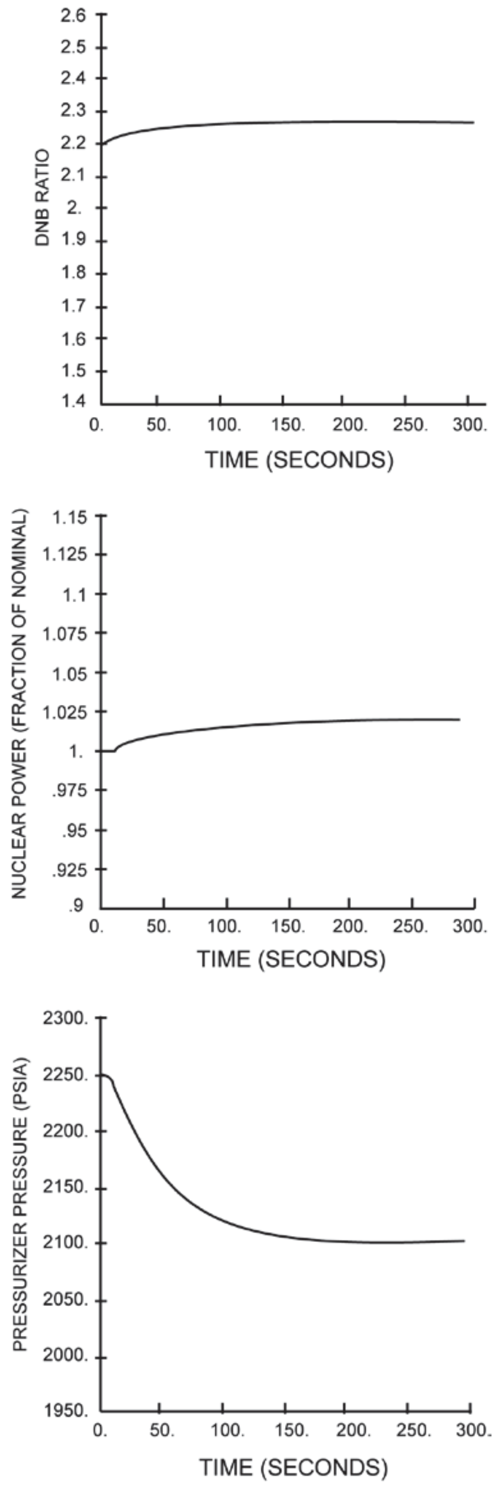
FIGURE 15.2.10-3
MAIN FEEDWATER REGULATING VALVE MALFUNCTION FULL POWER, MANUAL ROD CONTROL CORE AVERAGE TEMPERATURE AND DNBR TRANSIENTS



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

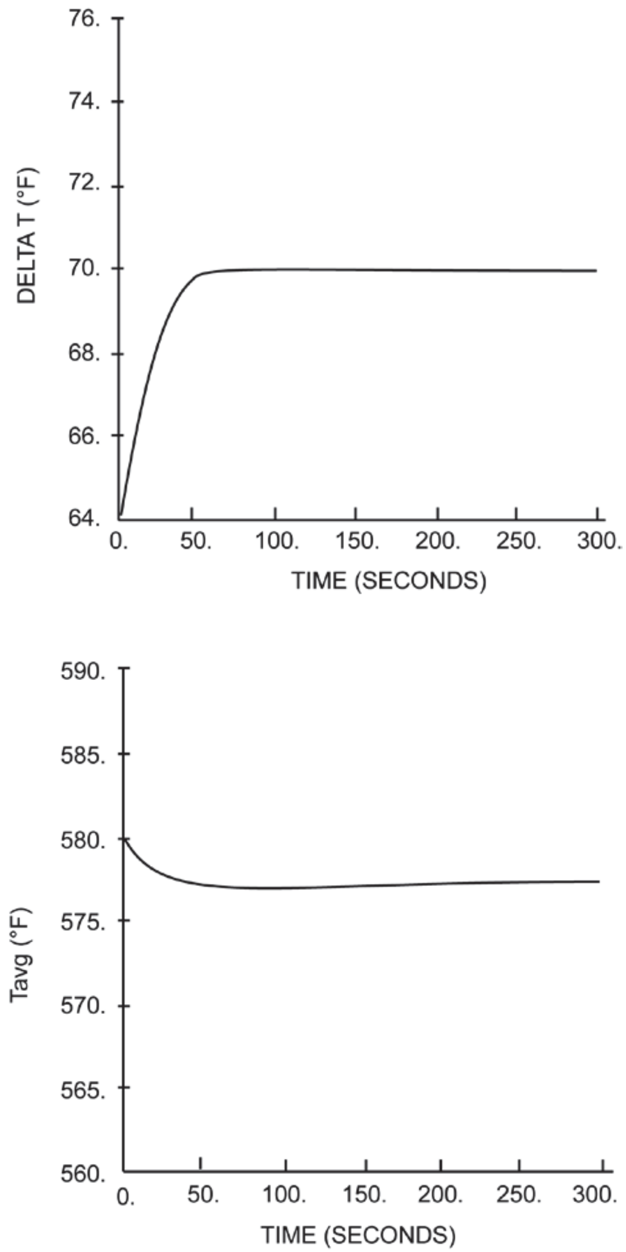
FIGURE 15.2.12-1
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

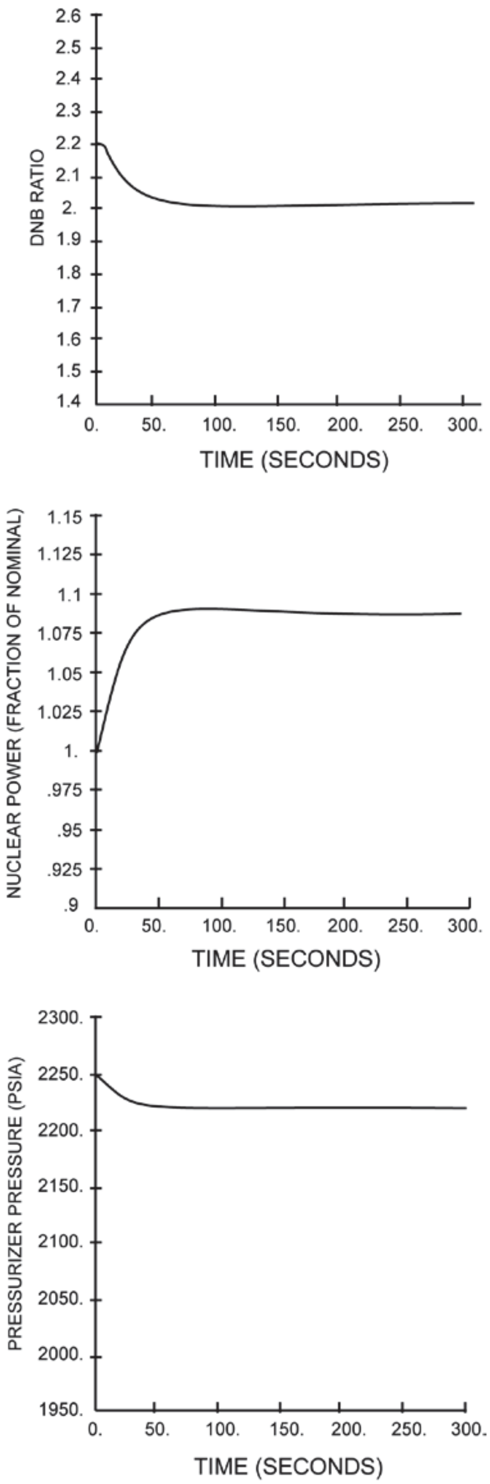
FIGURE 15.2.12-2
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME



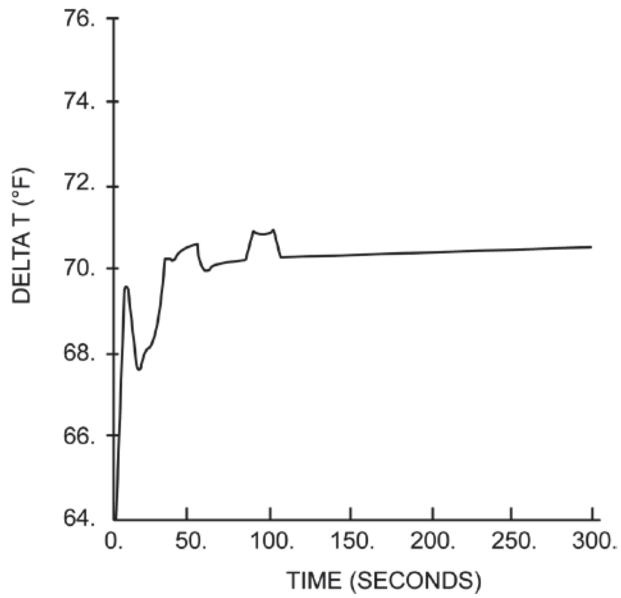
FSAR UPDATE

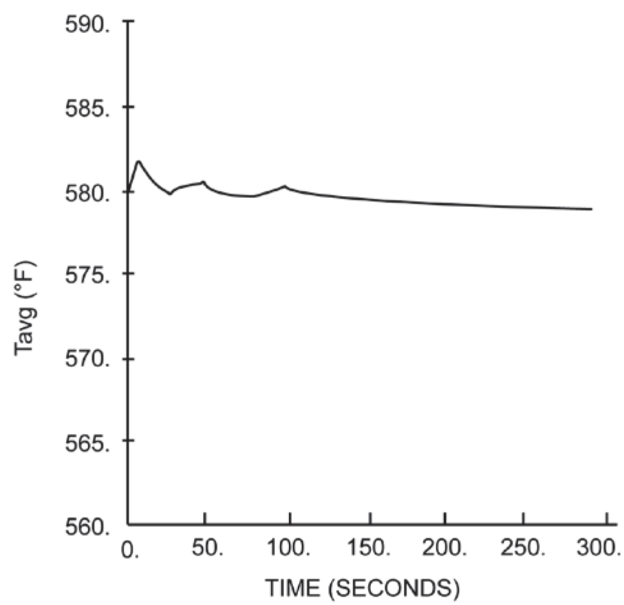
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.12-3
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME

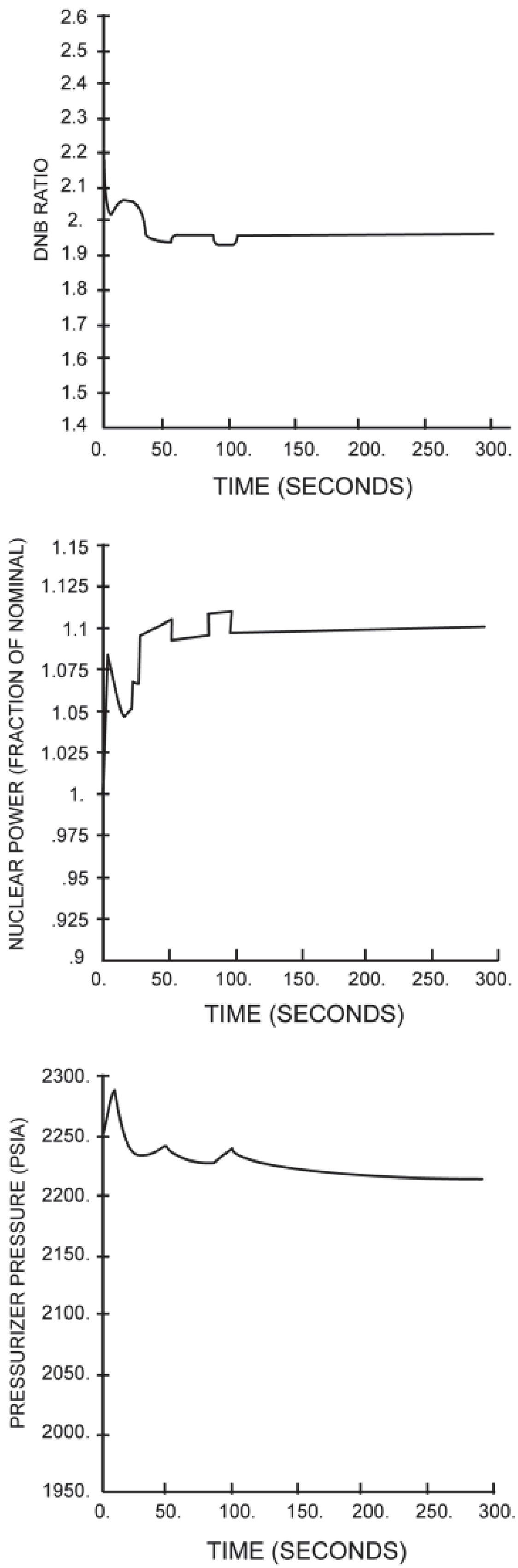


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.12-4 EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME





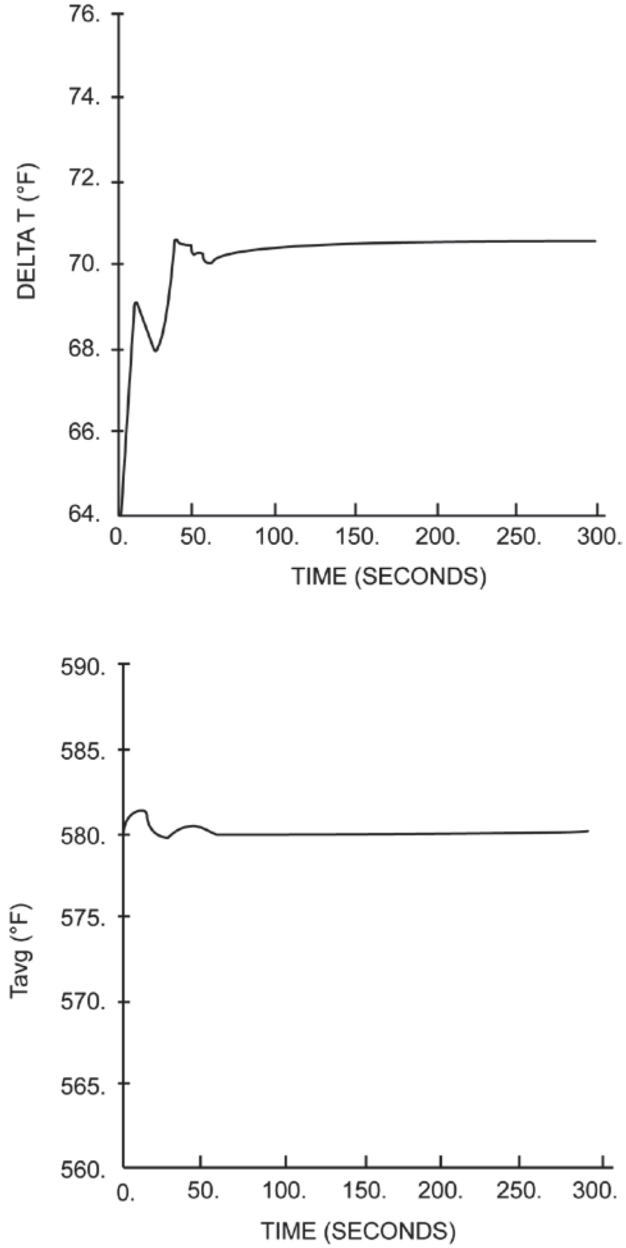
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.12-5 EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME



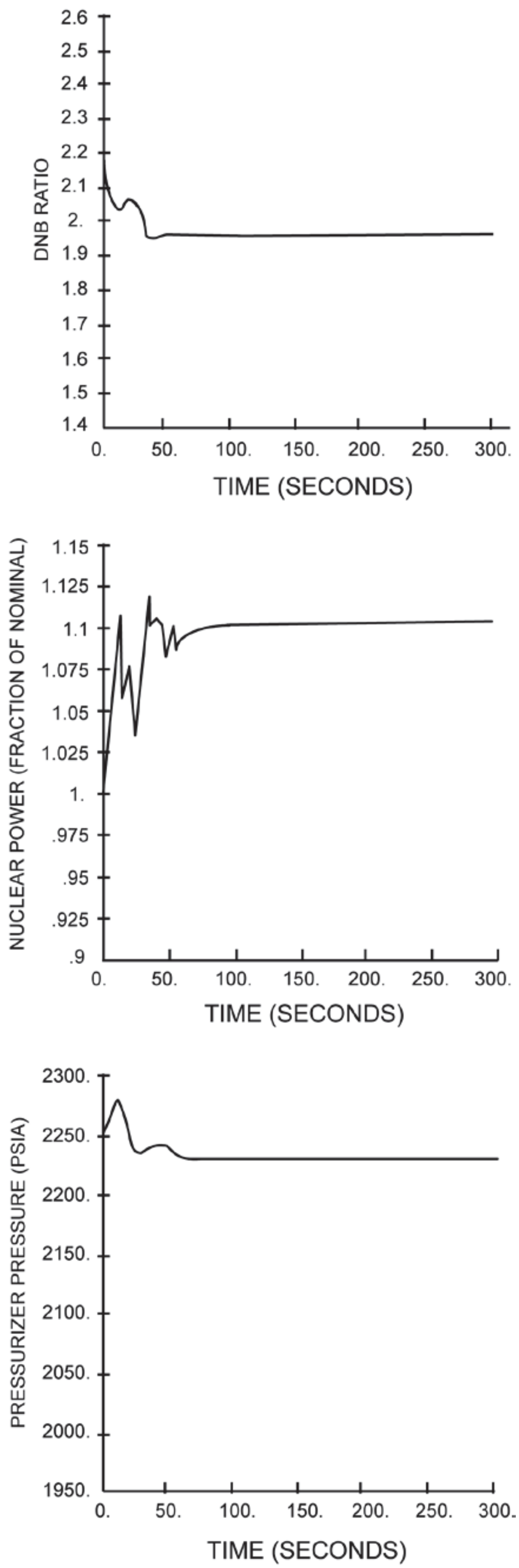
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

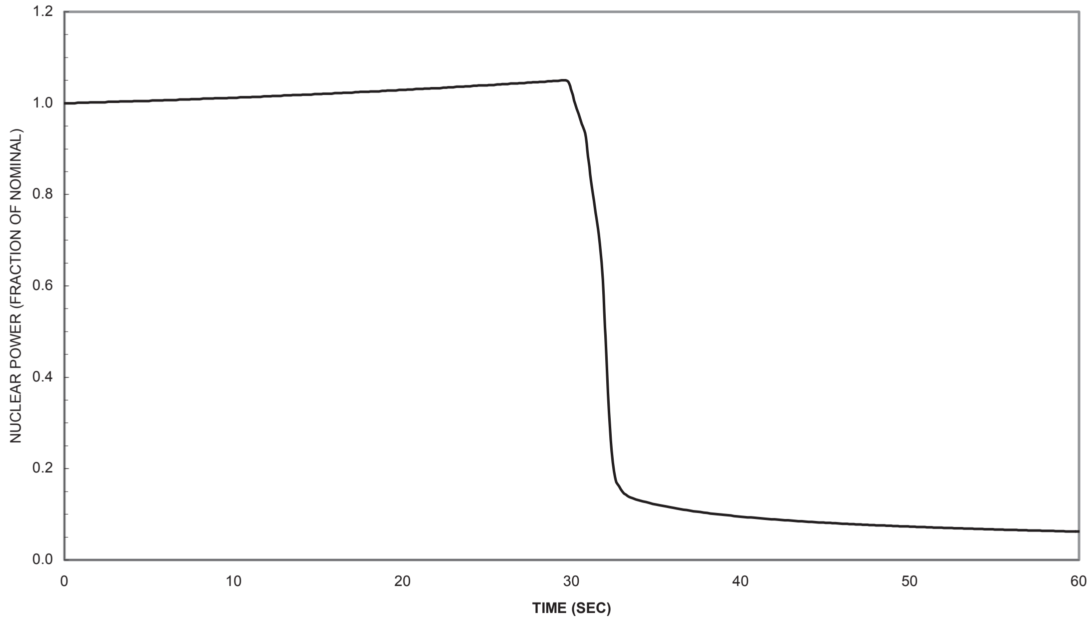
FIGURE 15.2.12-6
EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME

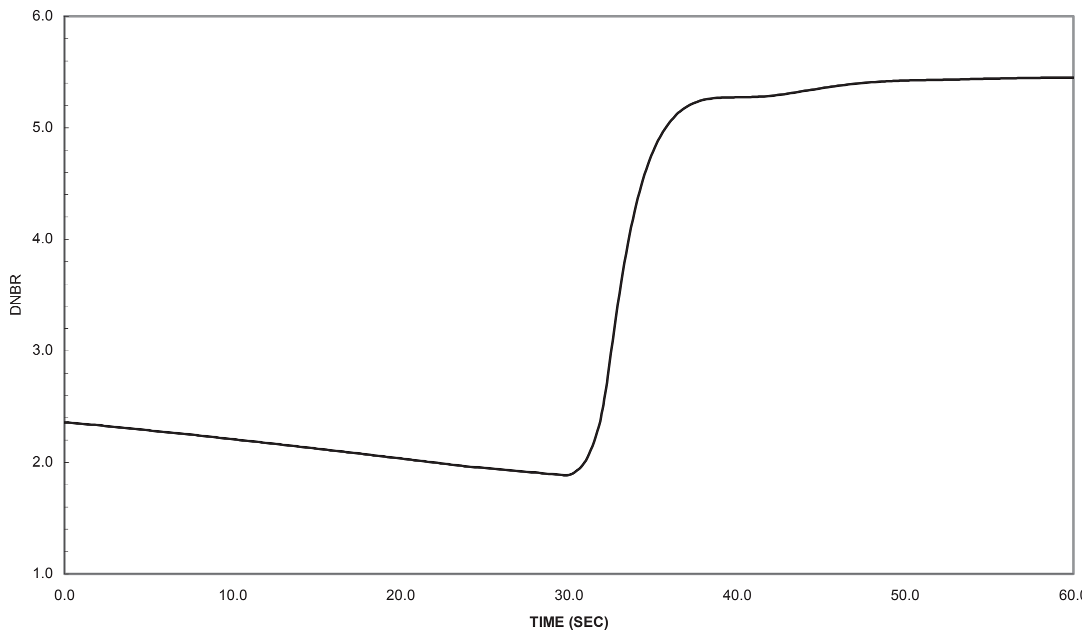


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.12-7 EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.12-8 EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME

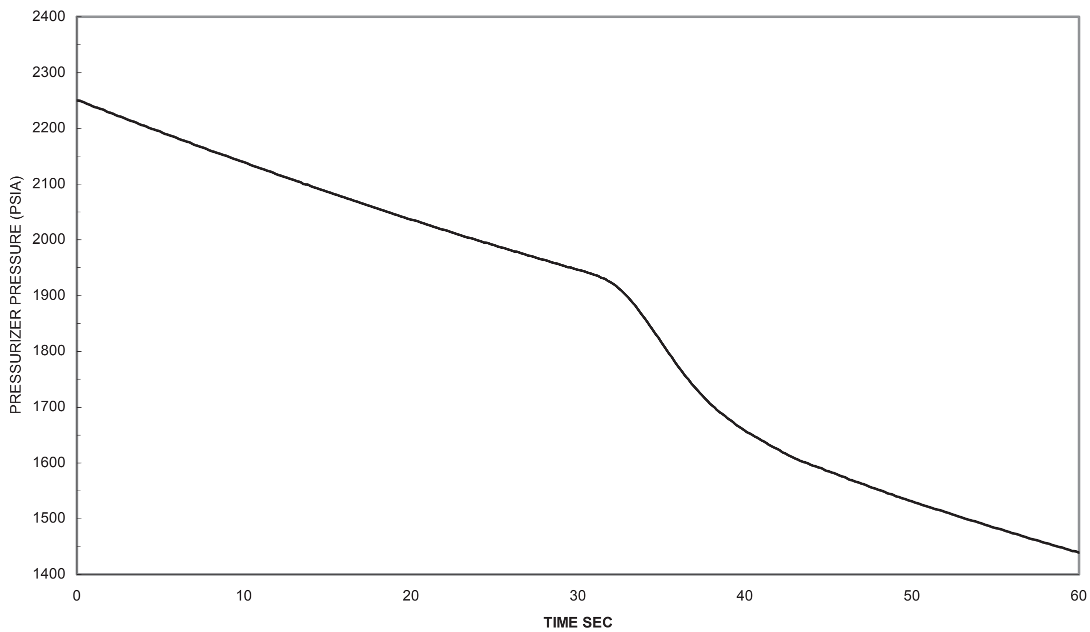


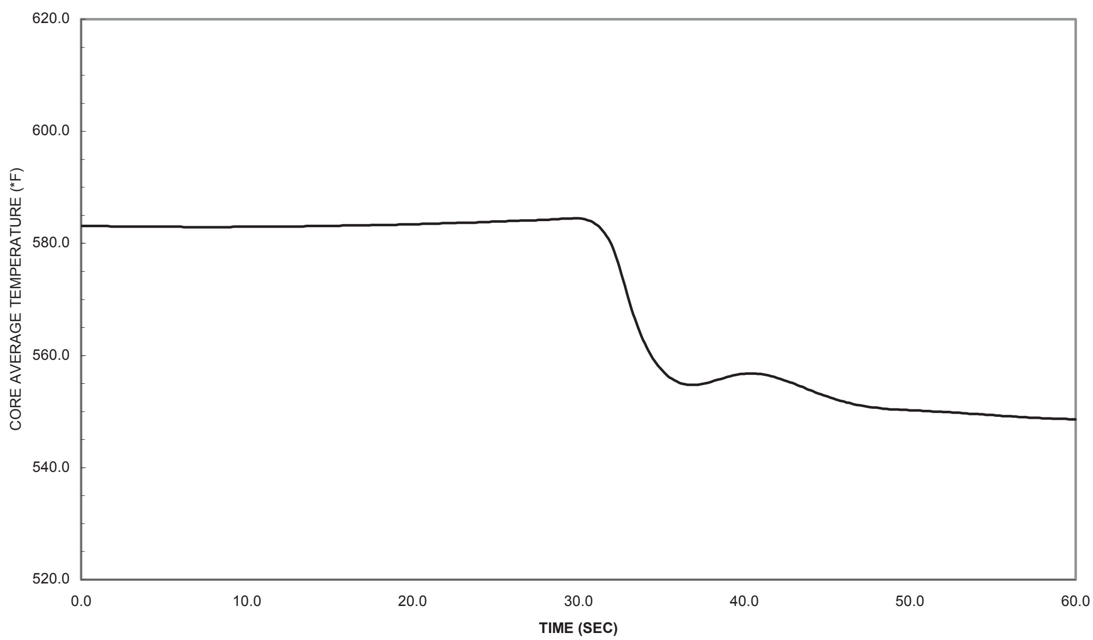


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.13-1
NUCLEAR POWER AND DNBR TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

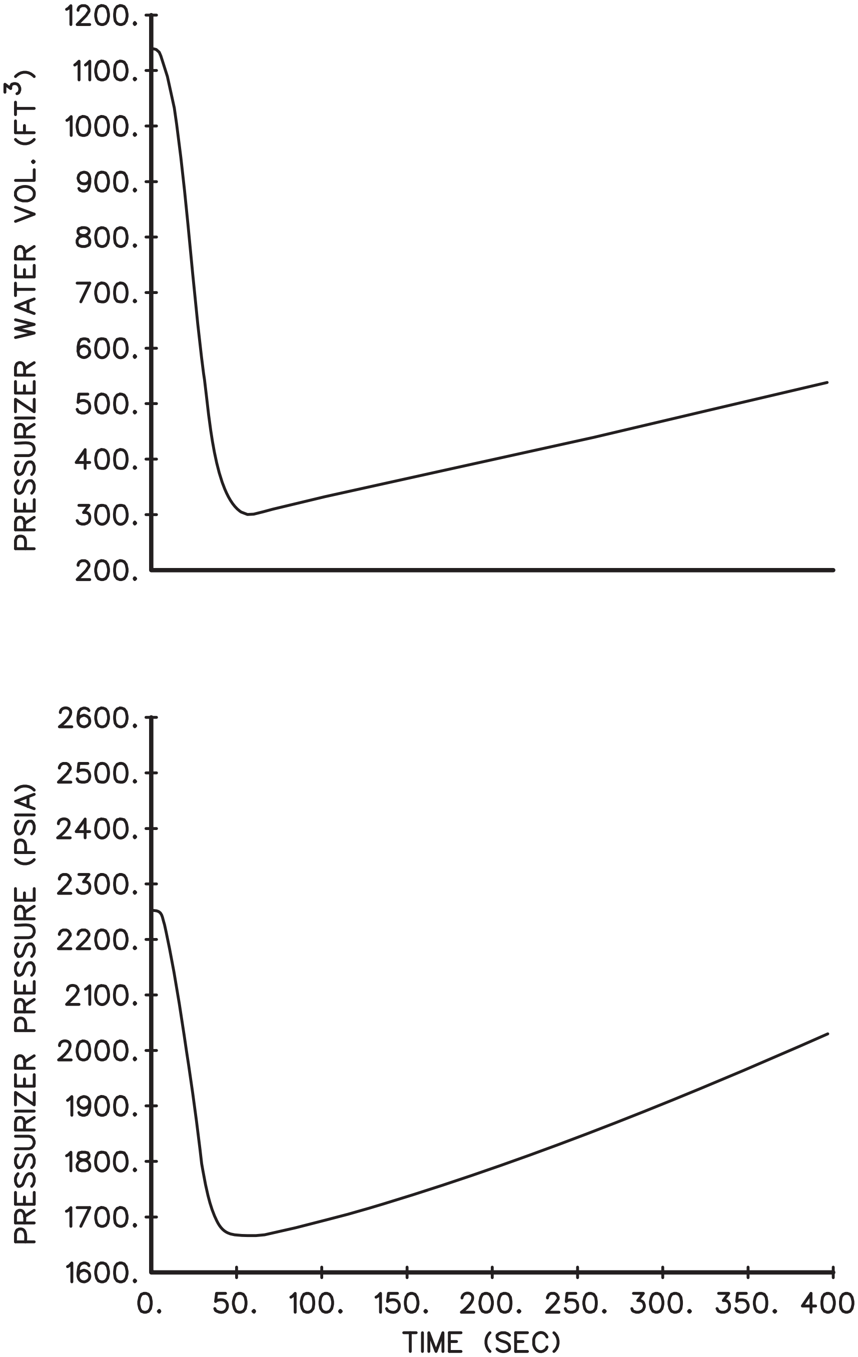




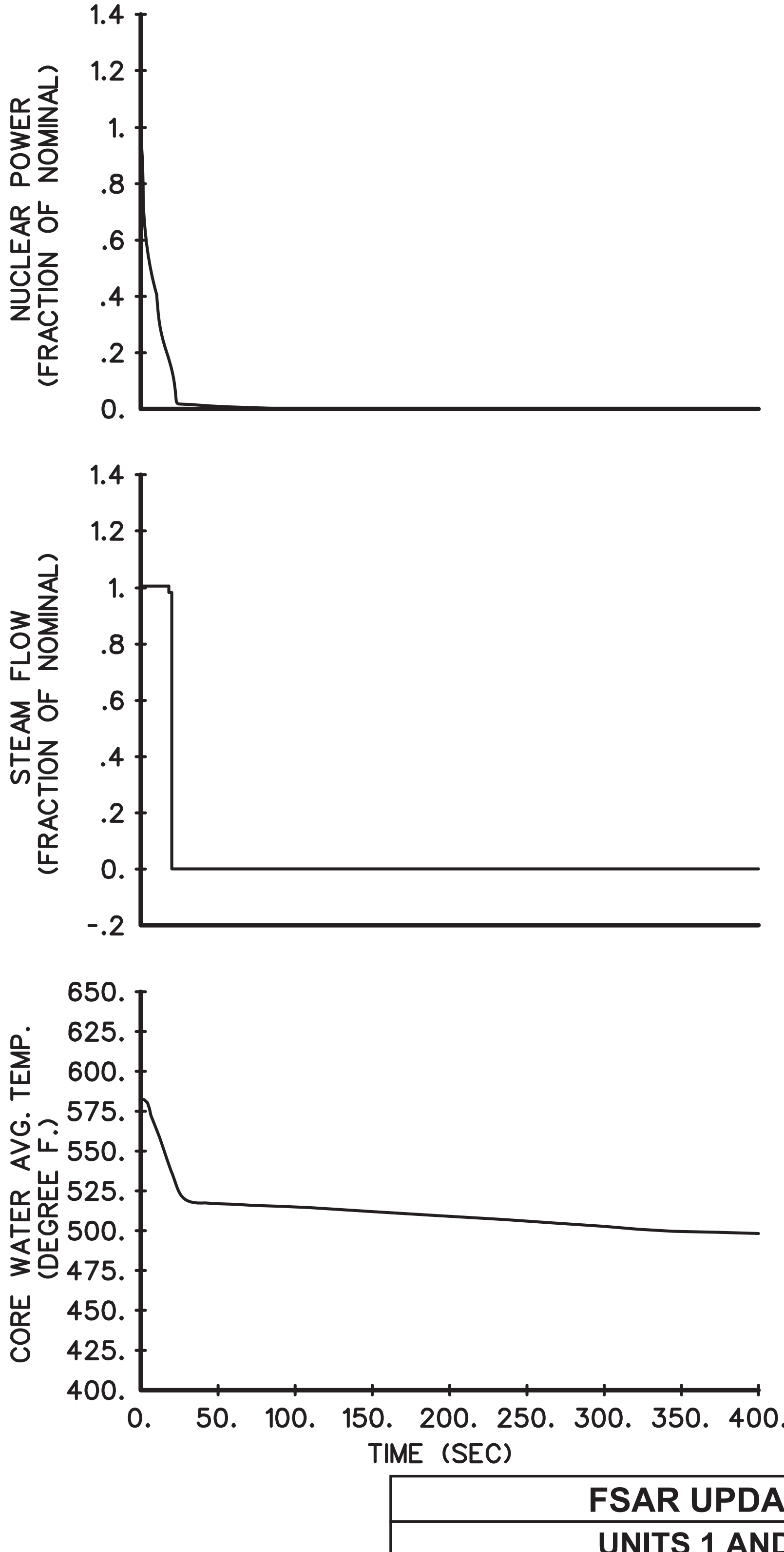
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.13-2
PRESSURIZER PRESSURE AND CORE AVERAGE TEMPERATURE TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM



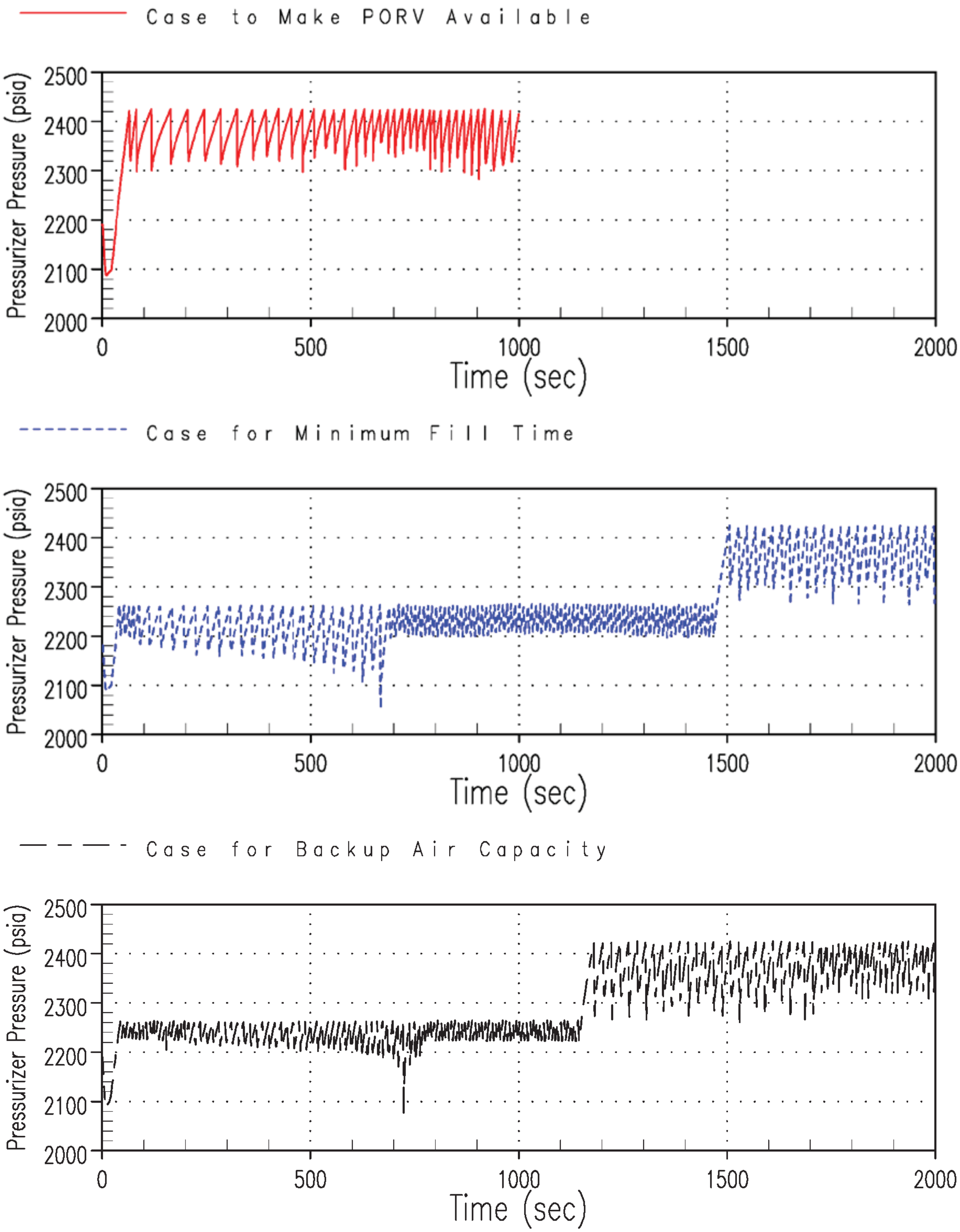
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.15-1 SPURIOUS ACTUATION OF SAFETY INJECTION SYSTEM AT POWER DNBR ANALYSIS – PRESSURIZER WATER VOLUME AND PRESSURIZER PRESSURE VERSUS TIME



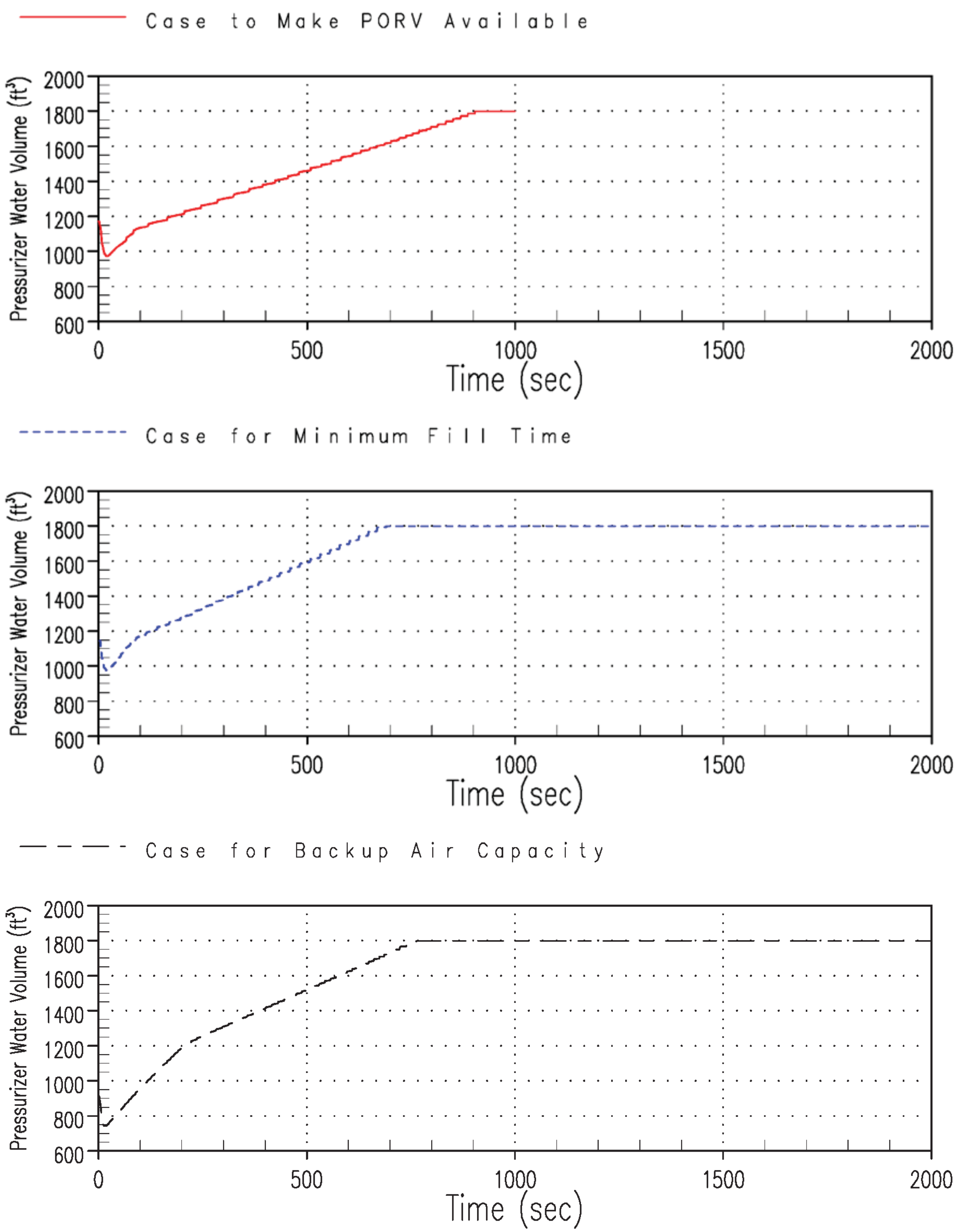
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

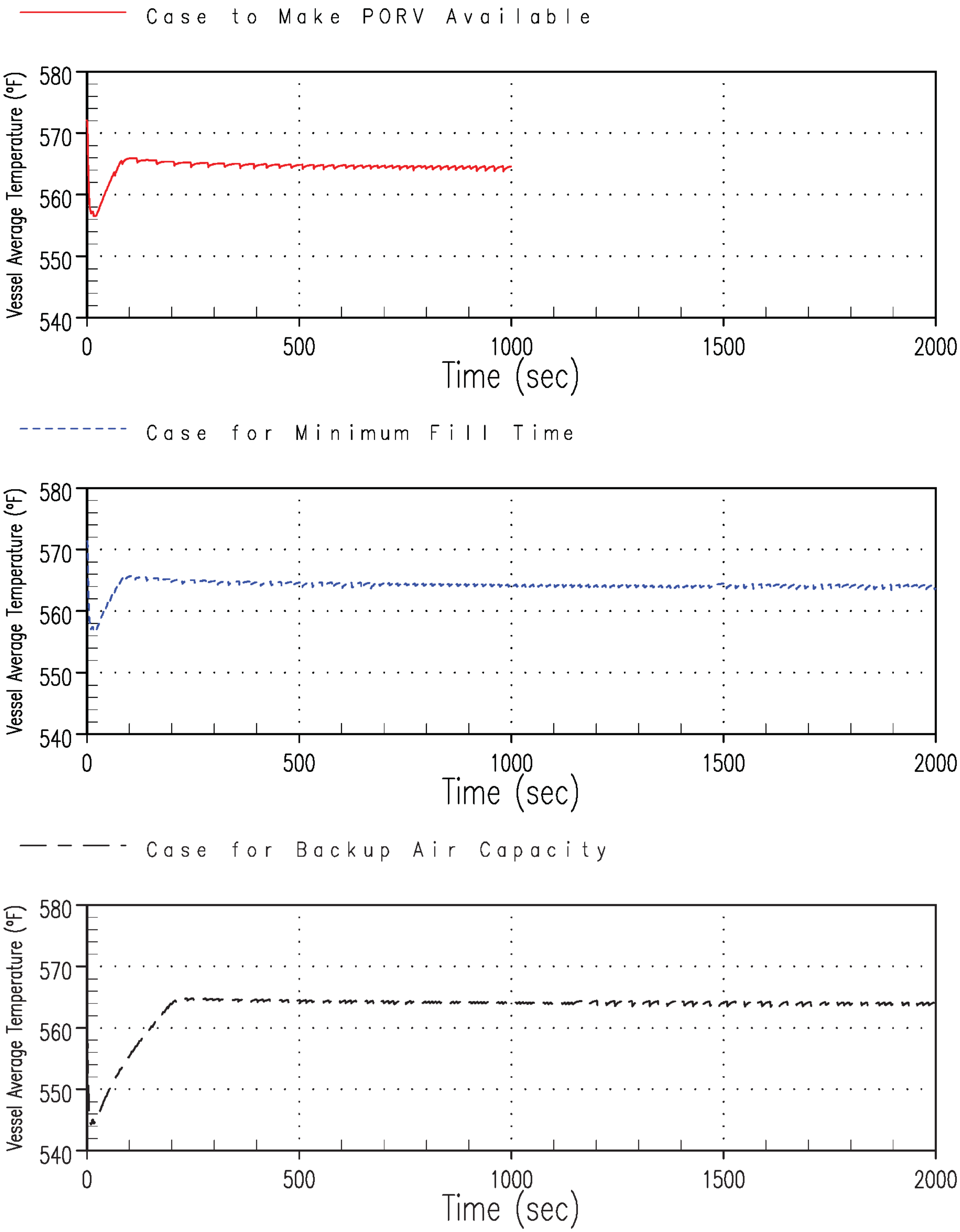
FIGURE 15.2.15-2
SPURIOUS ACTUATION OF SAFETY INJECTION SYSTEM AT POWER DNBR ANALYSIS – NUCLEAR POWER, STEAM FLOW, AND CORE WATER TEMPERATURE VERSUS TIME



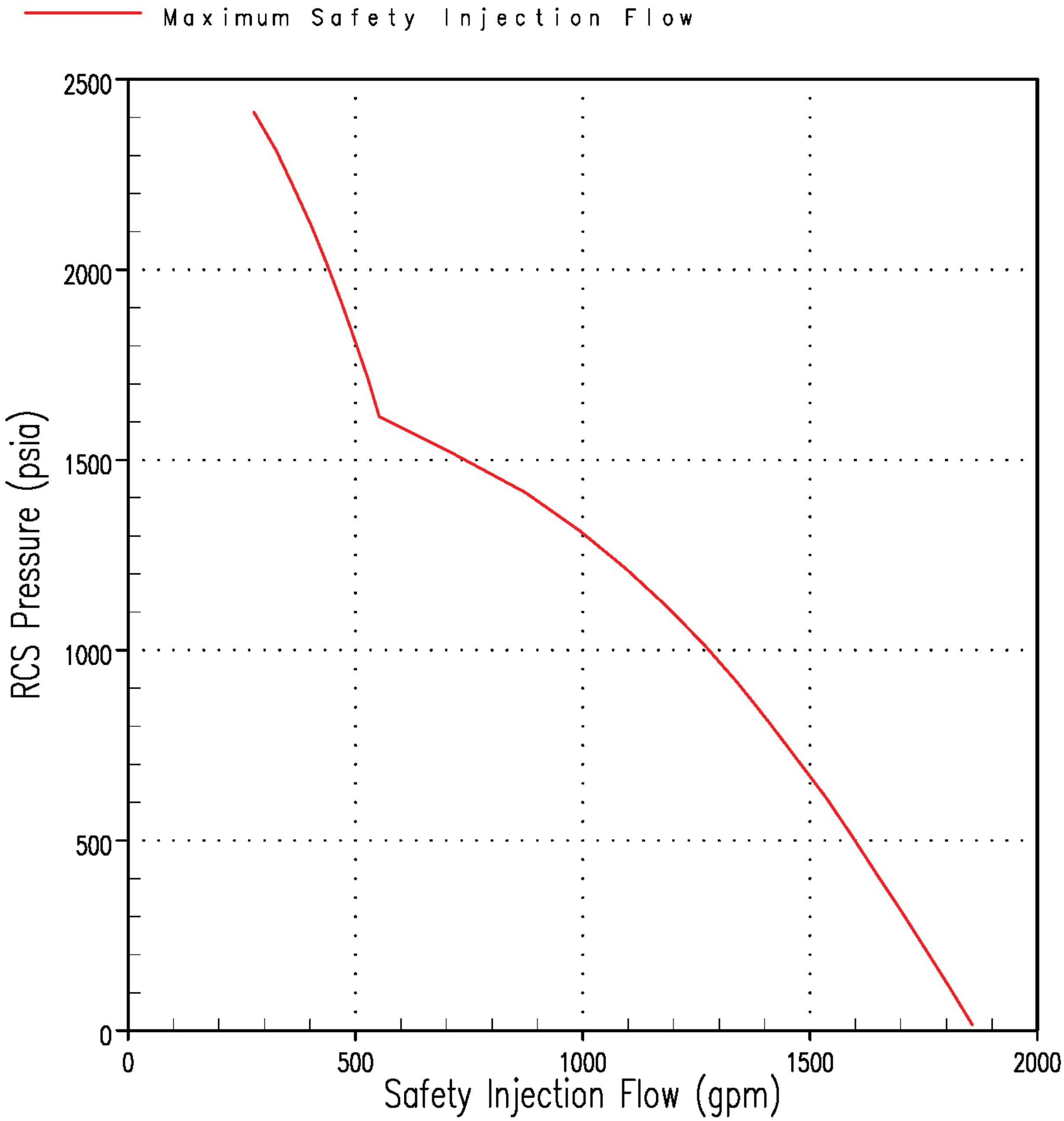
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.15-3 SSI PRESSURIZER FILLING ANALYSIS TYPICAL PRESSURIZER PRESSURE RESPONSE



FSAR UPDATE
UNITS 1 AND 2 **DIABLO CANYON SITE**
FIGURE 15.2.15-4 **SSI PRESSURIZER FILLING ANALYSIS TYPICAL PRESSURIZER LIQUID VOLUME RESPONSE**



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.2.15-5 SSI PRESSURIZER FILLING ANALYSIS TYPICAL VESSEL AVERAGE TEMPERATURE RESPONSE



FSAR UPDATE

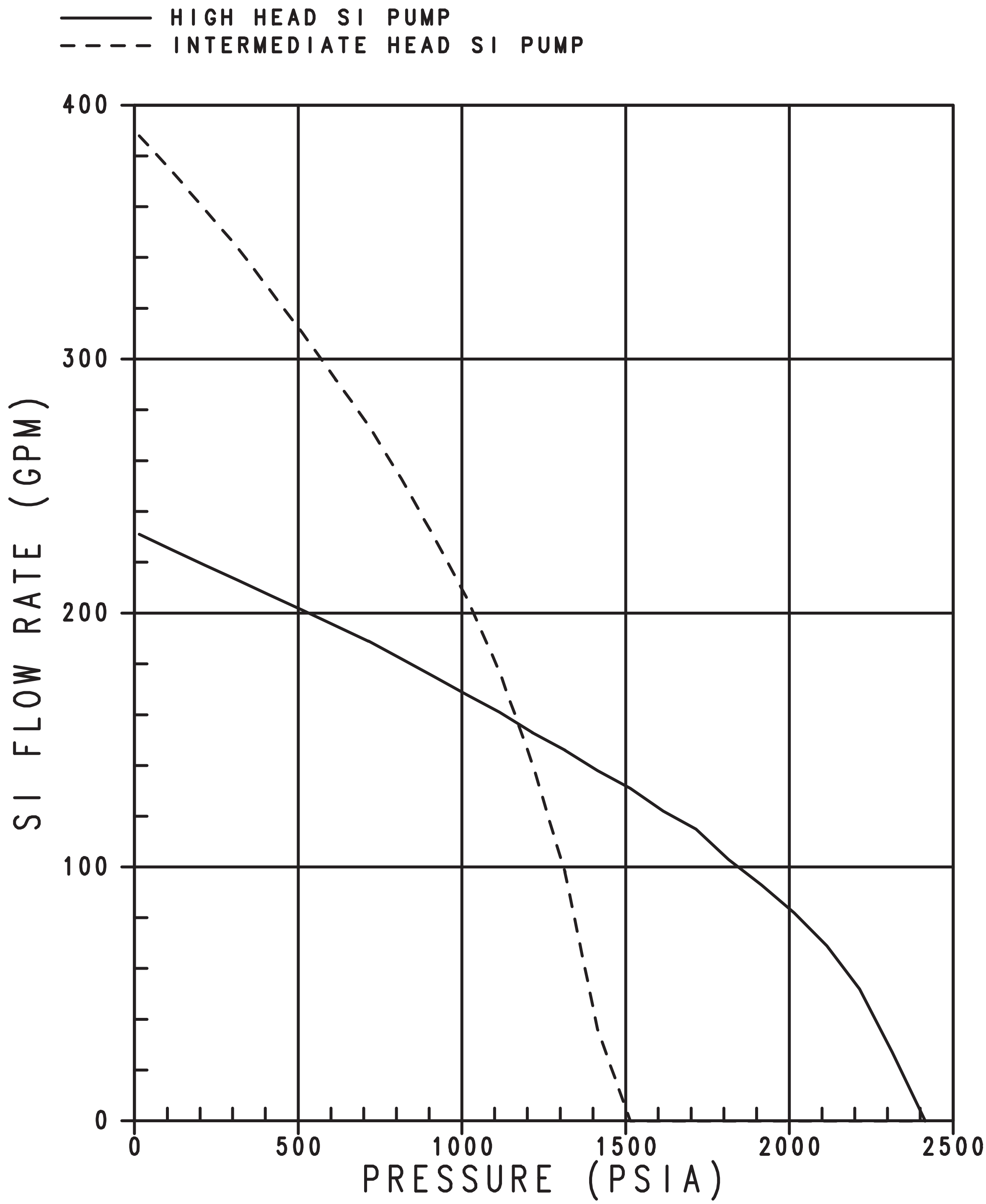
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.2.15-6

SSI PRESSURIZER FILLING
ANALYSIS MAXIMUM SAFEGUARDS

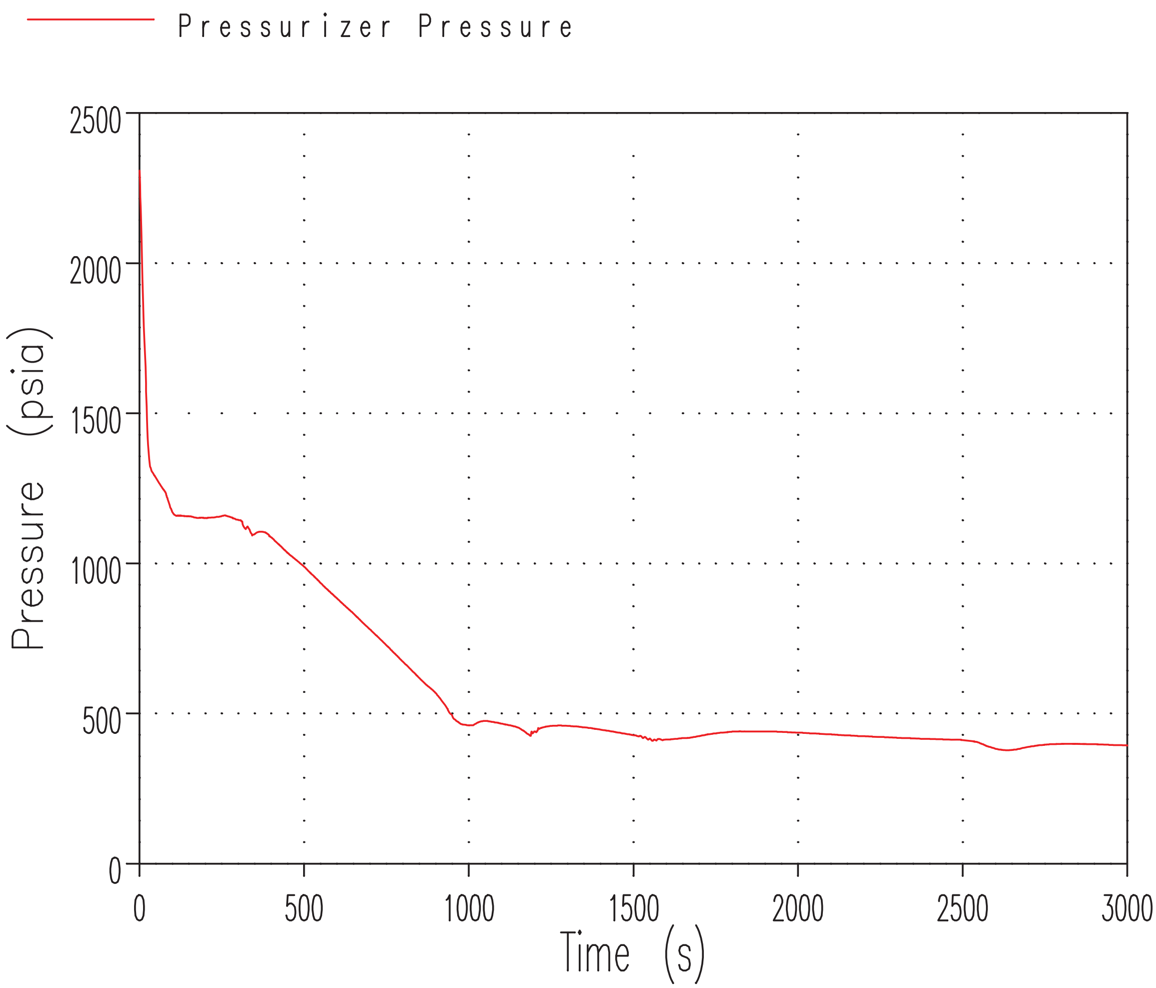
SI FLOW PROFILE

Revision 24, September 2018



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-1 SAFETY INJECTION FLOW RATE FOR SMALL BREAK LOCA

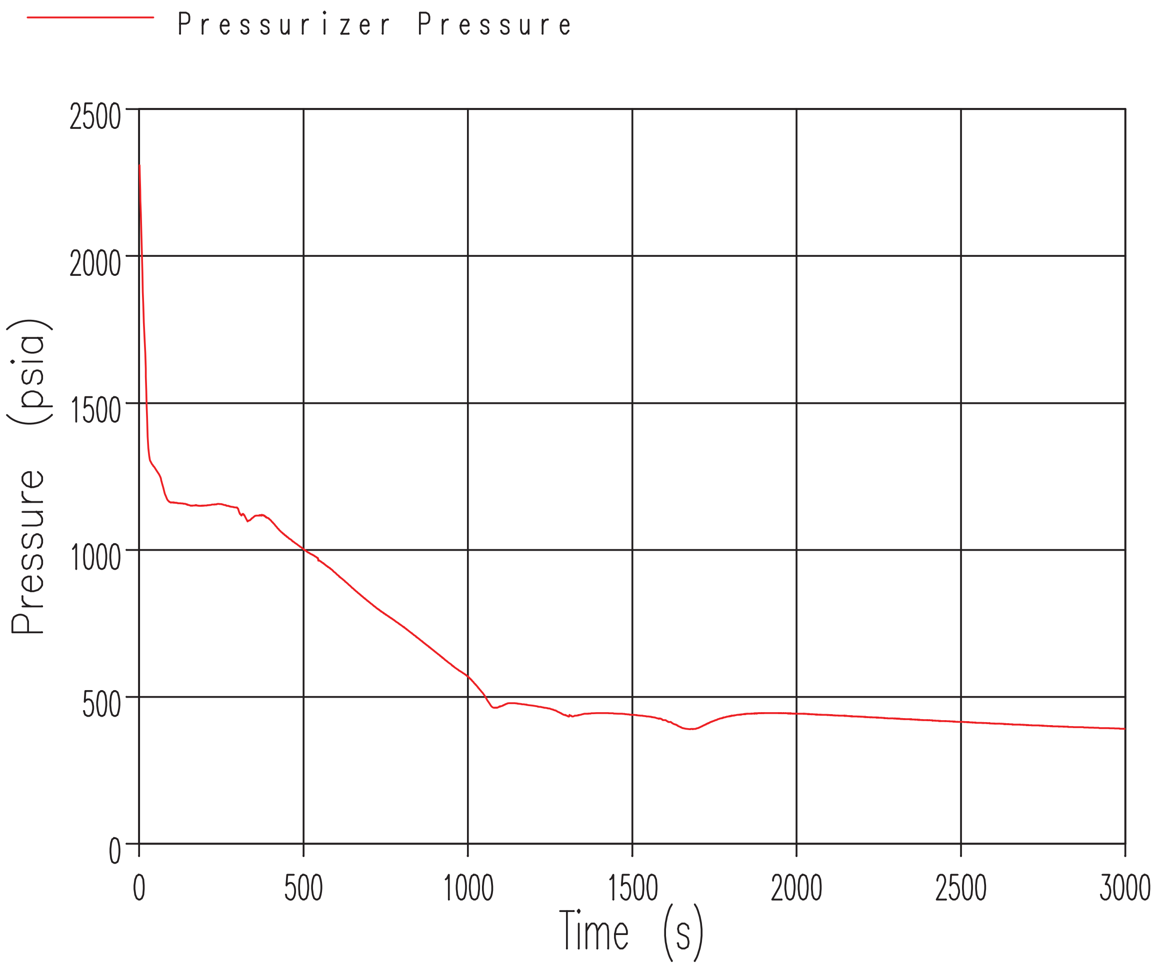
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-2 (Sheet 1 of 2) RCS DEPRESSURIZATION 4-INCH COLD LEG BREAK

Revision 21 September 2013

DCPP Unit 2



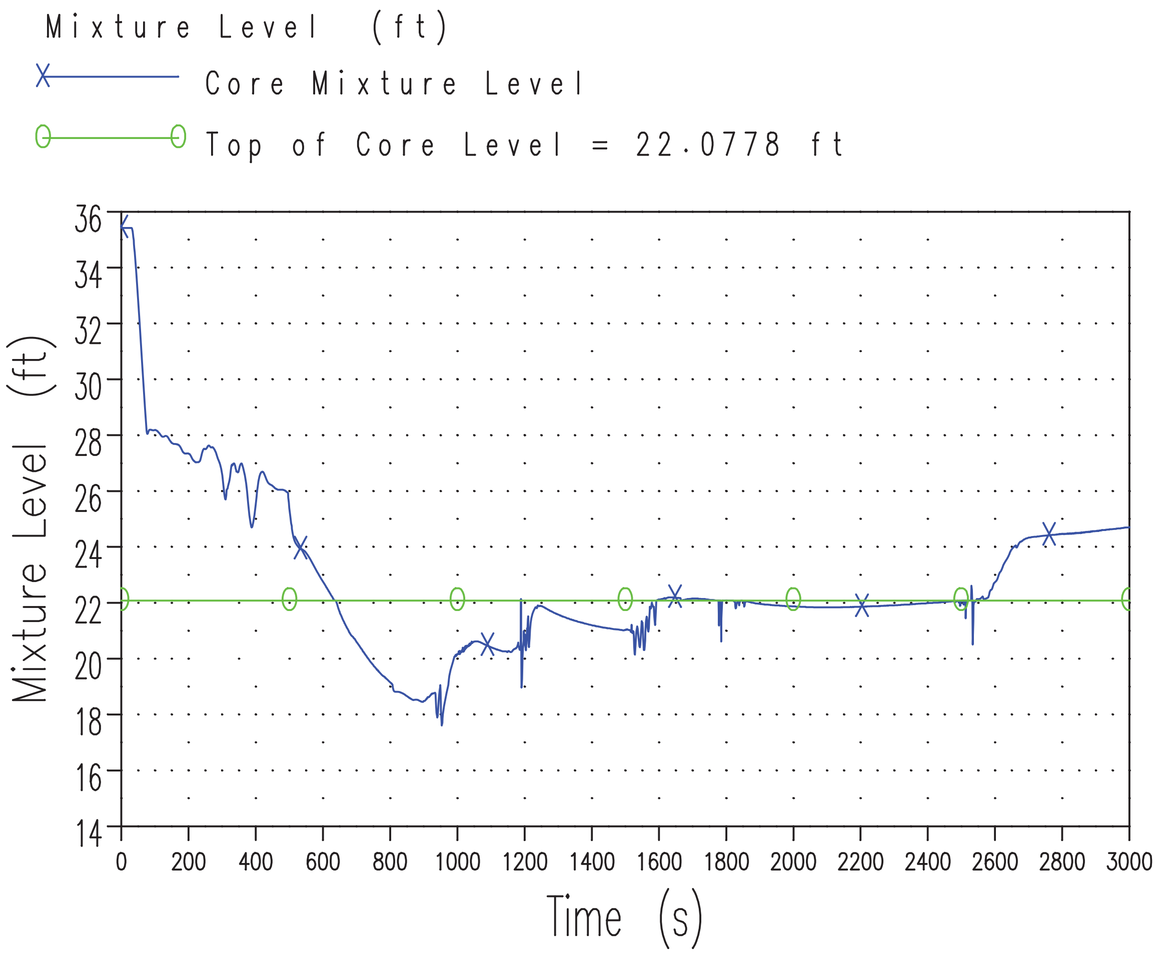
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-2 (Sheet 2 of 2)
RCS DEPRESSURIZATION
4-INCH COLD LEG BREAK

Revision 21 September 2013

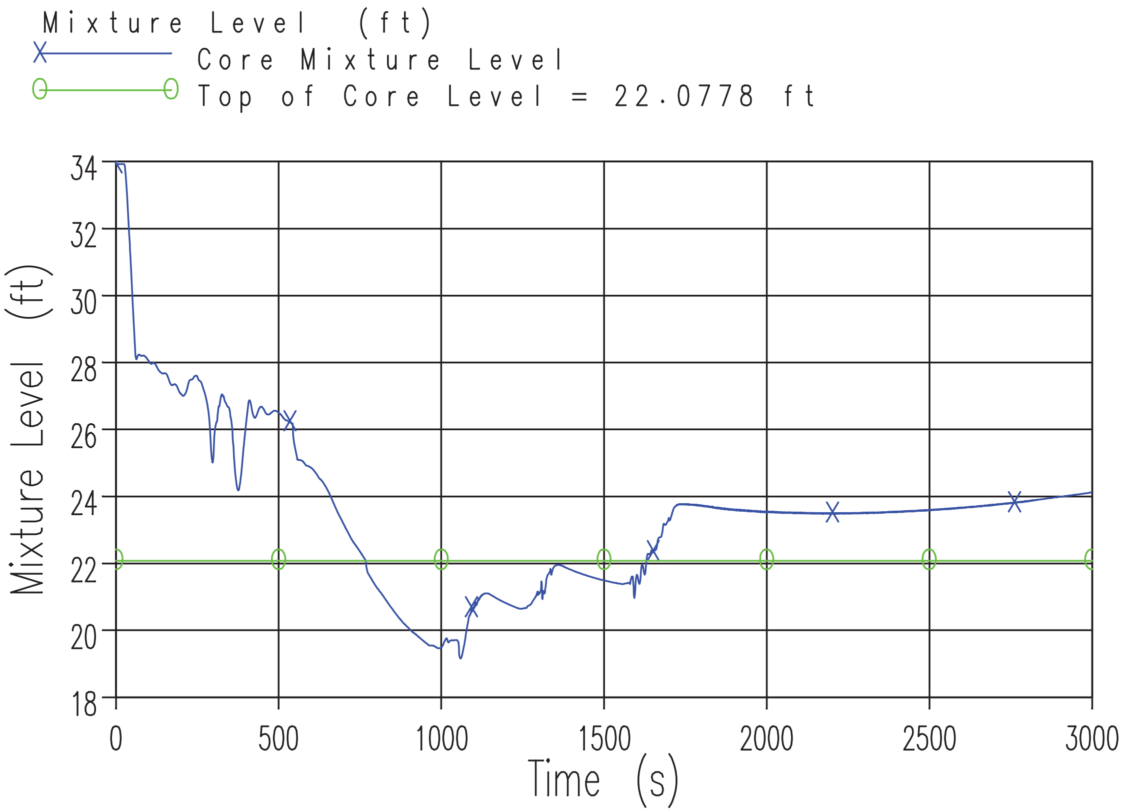
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-3 (Sheet 1 of 2) CORE MIXTURE ELEVATION 4-INCH COLD LEG BREAK

Revision 21 September 2013

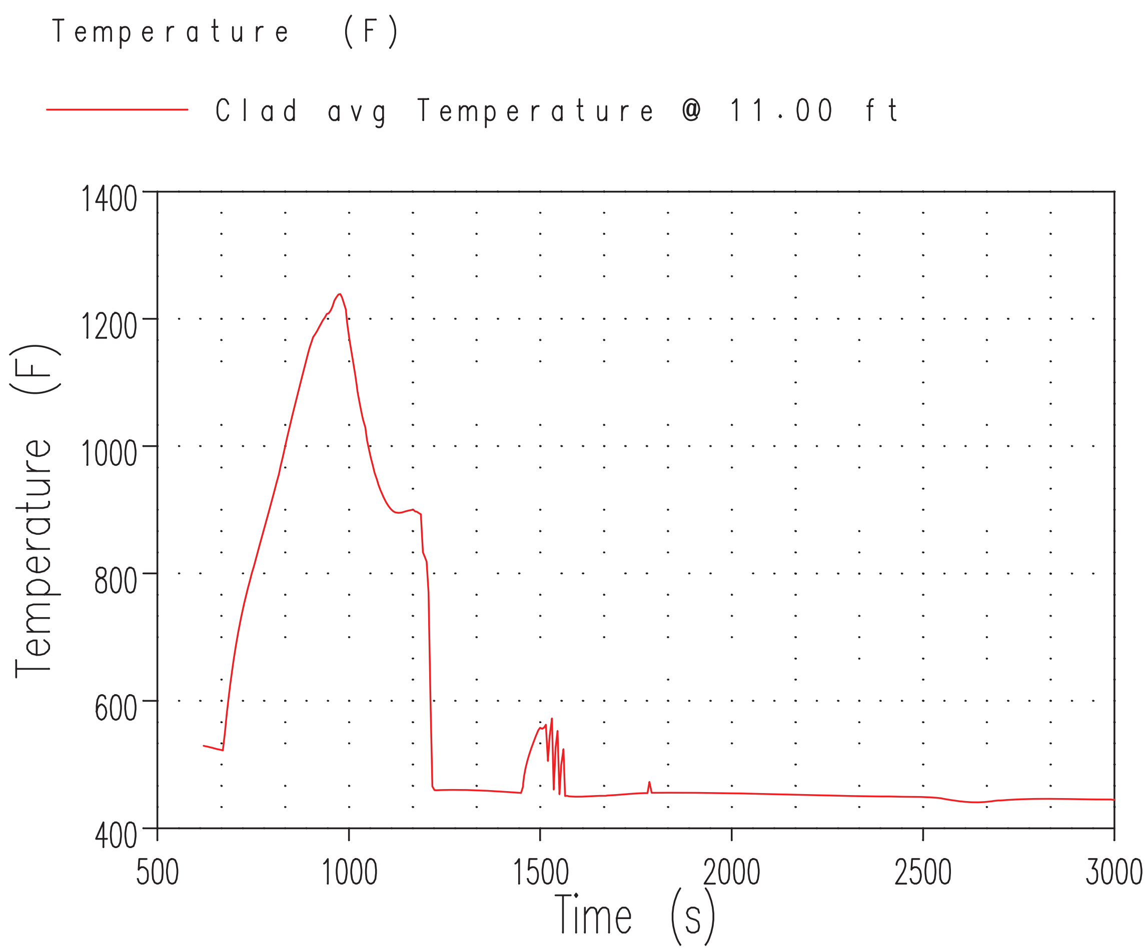
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-3 (Sheet 2 of 2) CORE MIXTURE ELEVATION 4-INCH COLD LEG BREAK

Revision 21 September 2013

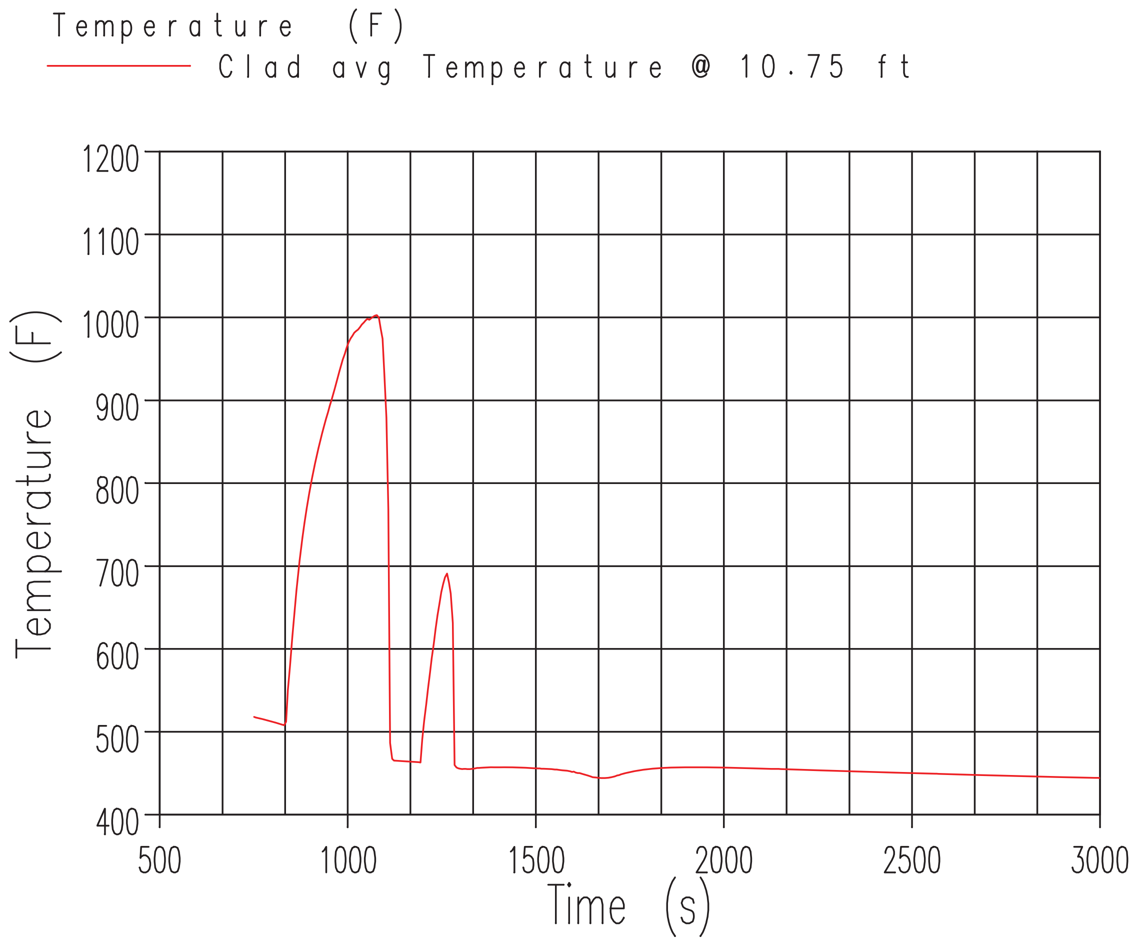
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-4 (Sheet 1 of 2) CLADDING TEMPERATURE TRANSIENT 4-INCH COLD LEG BREAK

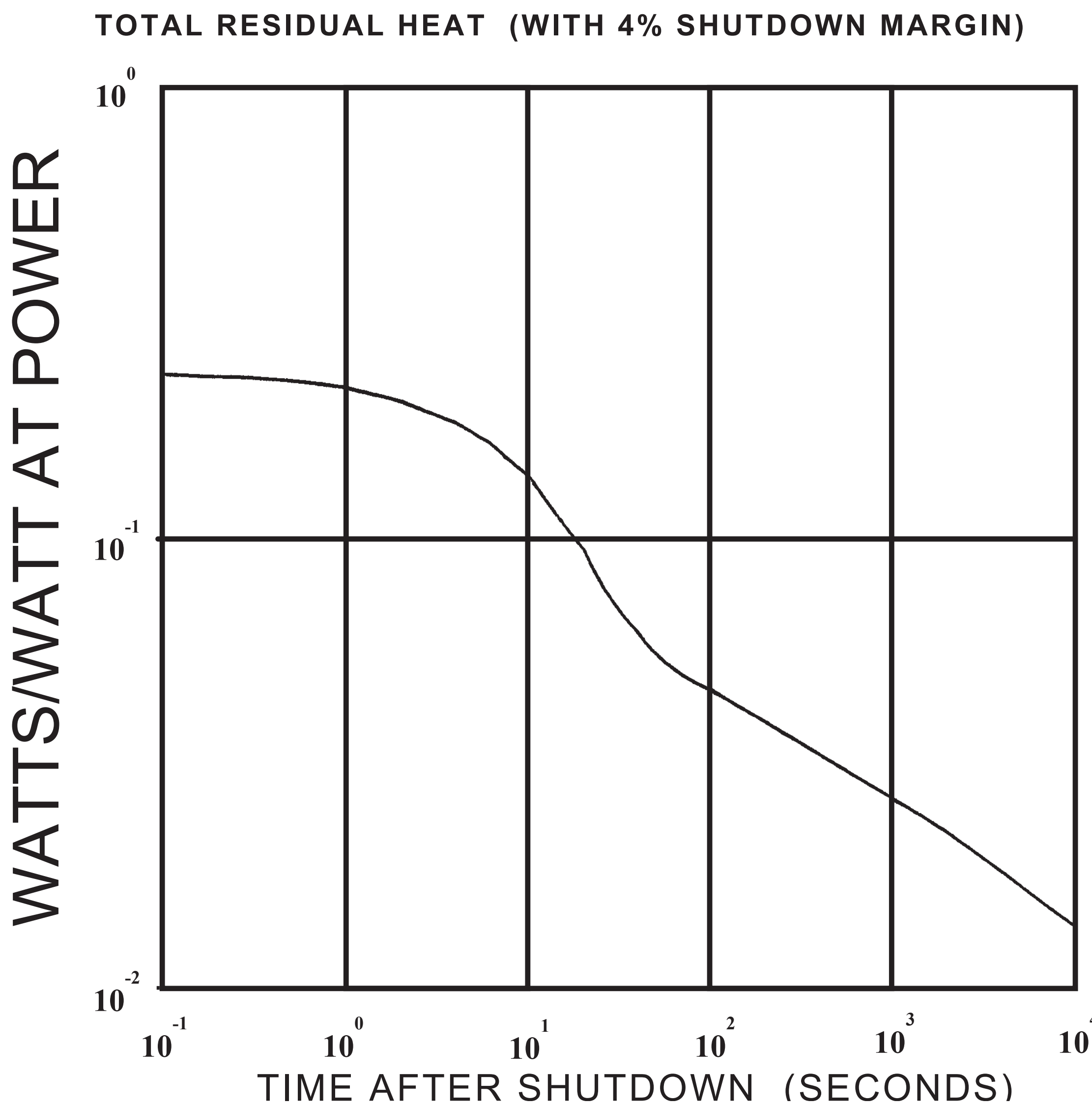
Revision 21 September 2013

DCPP Unit 2



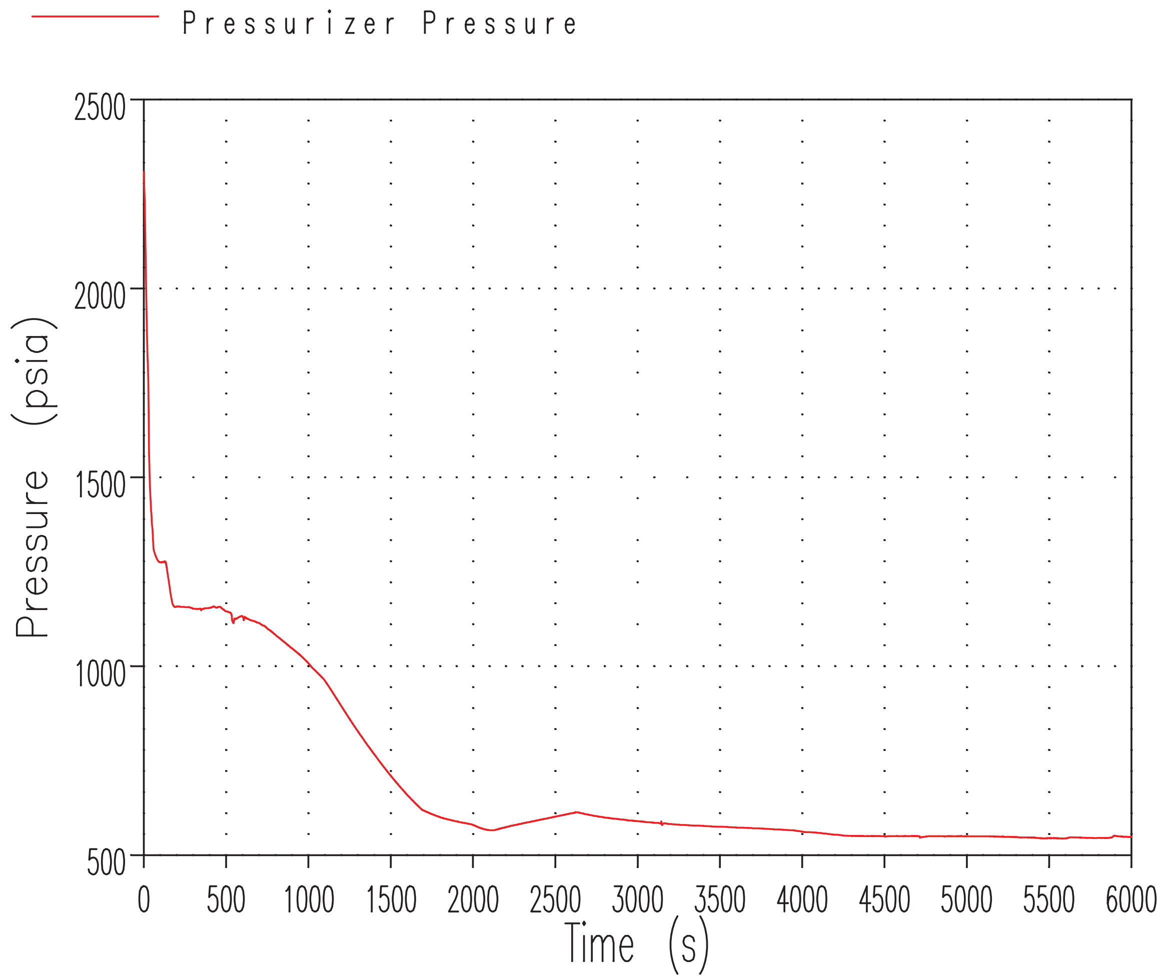
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-4 (Sheet 2 of 2) CLADDING TEMPERATURE TRANSIENT 4-INCH COLD LEG BREAK

Revision 21 September 2013



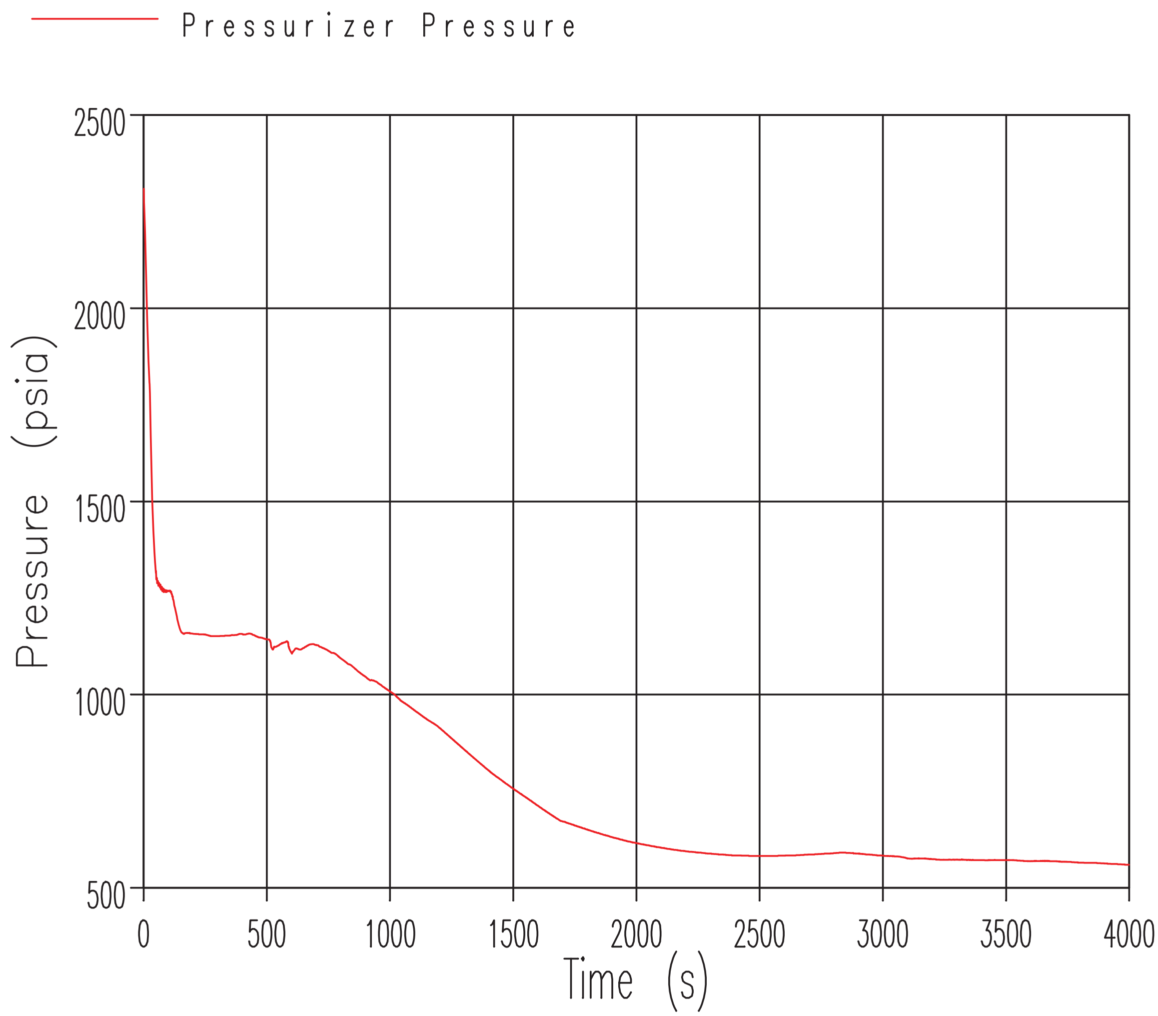
FSAR UPDATE
UNITS 1 AND 2 **DIABLO CANYON SITE**
FIGURE 15.3-8 **LOCA CORE POWER TRANSIENT**

DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-9 (Sheet 1 of 2) RCS DEPRESSURIZATION 3-INCH COLD LEG BREAK

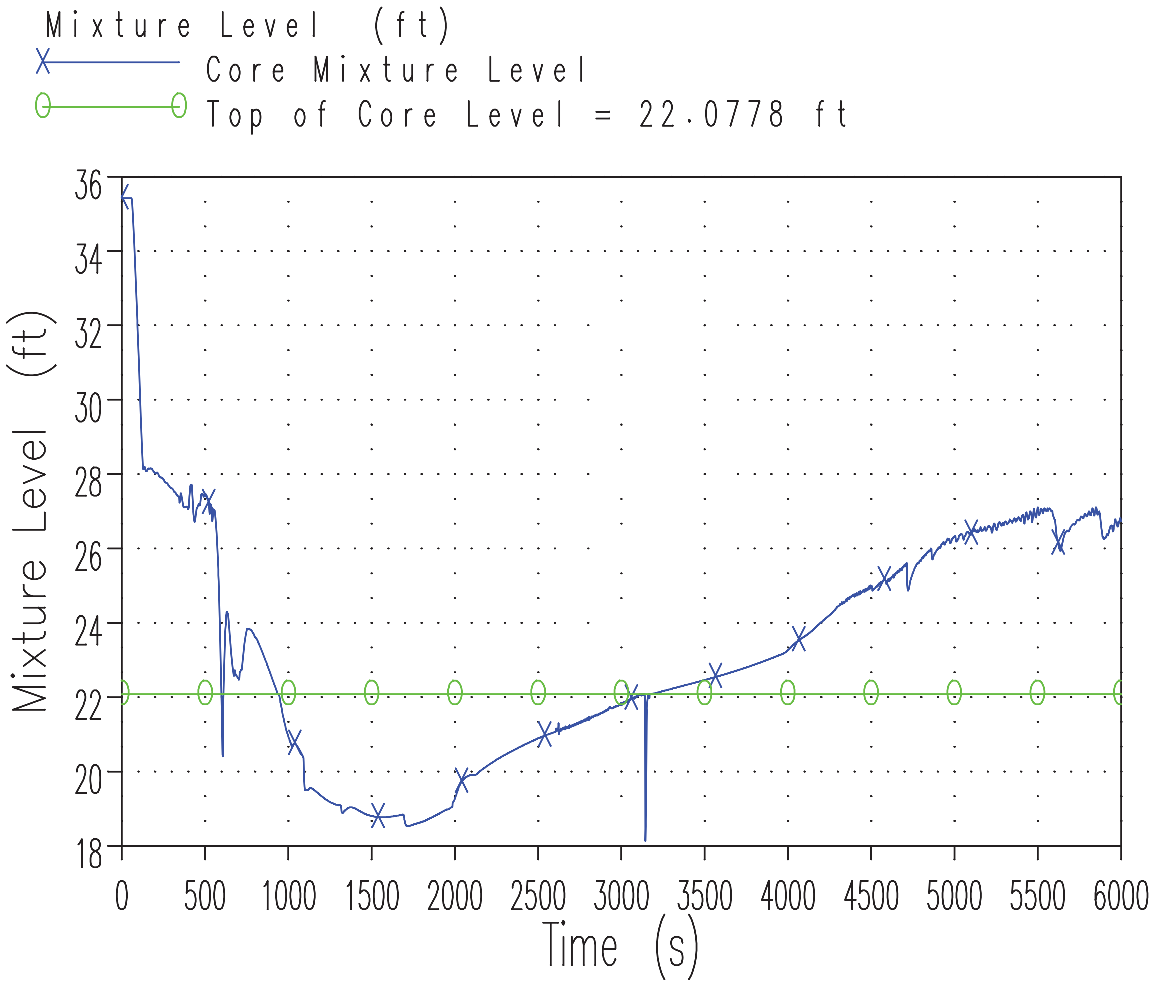
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-9 (Sheet 2 of 2) RCS DEPRESSURIZATION 3-INCH COLD LEG BREAK

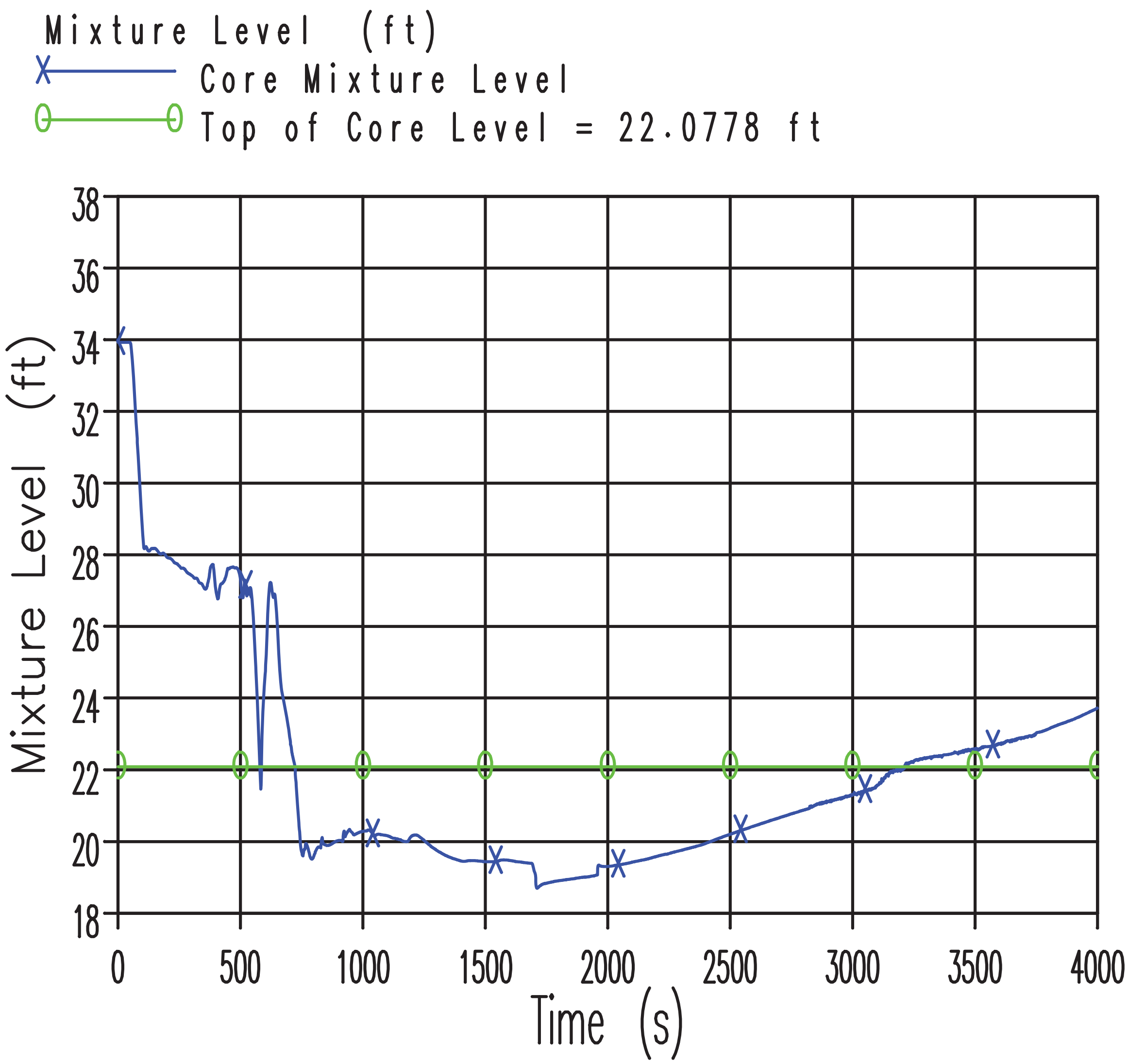
Revision 21 September 2013

DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-11 (Sheet 1 of 2) CORE MIXTURE ELEVATION 3-INCH COLD LEG BREAK

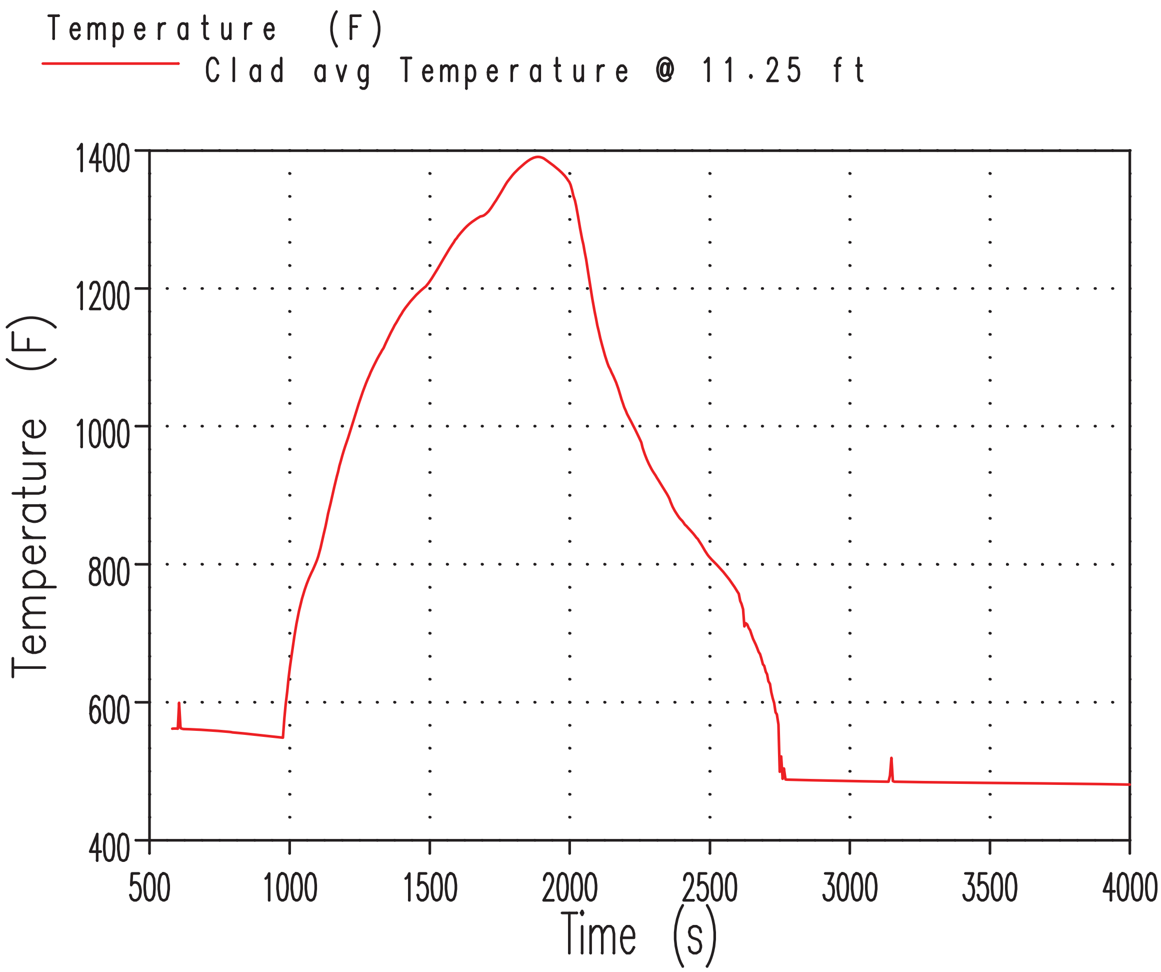
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-11 (Sheet 2 of 2) CORE MIXTURE ELEVATION 3-INCH COLD LEG BREAK

Revision 21 September 2013

DCPP Unit 1



FSAR UPDATE

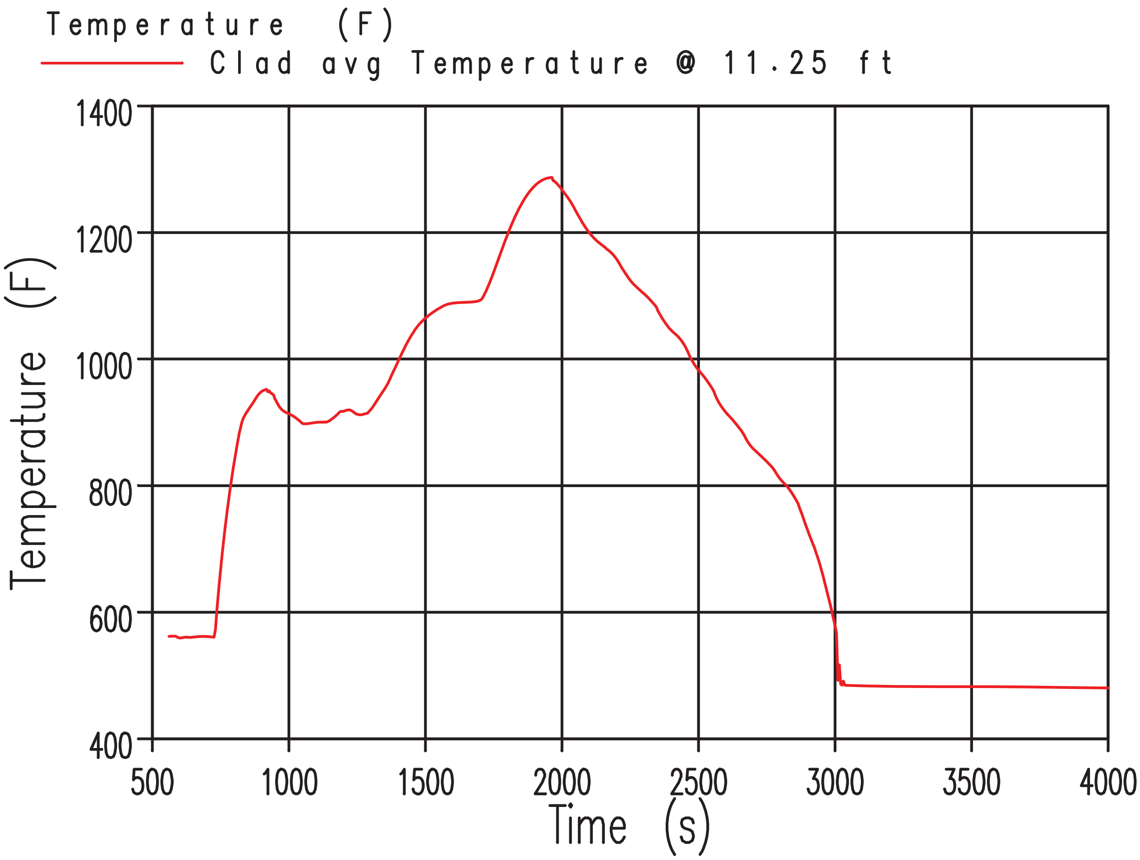
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-13 (Sheet 1 of 2)

CLAD TEMPERATURE TRANSIENT
3-INCH COLD LEG BREAK

Revision 21 September 2013

DCPP Unit 2



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-13 (Sheet 2 of 2)
CLAD TEMPERATURE TRANSIENT
3-INCH COLD LEG BREAK

Revision 21 September 2013

4026-385

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7						23.6	
14			2.5				4.7			11.1		17.6			
15					2.1			6.5							

CASE A

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-15

INTERCHANGE BETWEEN
REGION 1 AND REGION 3 ASSEMBLY

Revision 21 September 2013

4026-386

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					0.3			1.5							
2			3.2				0.8			3.2		6.0			
3			1.2		0.0			1.6						10.3	
4						0.0			2.9			6.5			
5	-2.2				-1.0			2.2			6.9				6.6
6					-1.7		0.5					8.8			
7		-3.2							5.2		16.7				5.4
8	-3.5		-3.4		-2.6		-0.7			11.4		11.3	5.8	4.4	
9				-3.6			-2.0			-2.3			2.2		
10	-3.8		-3.8			-3.6		-2.9						0.5	
11					-3.9				-4.3		-4.6		-1.5		
12		-2.8	-3.1					-4.5							
13								-4.8		-4.4		-2.6		1.4	
14			-0.4			-4.8		-4.8							
15							-4.8			-4.5					

CASE B-1

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER.

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-16

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY

4026-387

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							1.0			1.1					
2			5.1			1.0		1.0							
3								1.1		1.1		1.9		4.9	
4		1.7	1.7					1.4							
5					1.1				1.8		1.1		0.7		
6	0.0		0.2			1.8		3.9						4.0	
7				0.0			5.2			2.2			-0.3		
8	-0.7		-0.6		0.3		5.1			1.5		-0.3	-0.6	-0.7	
9		-1.0							-1.1		-0.8				-0.9
10					-1.4		-3.1					-1.3			
11	-0.9				-1.7						-1.7				-0.9
12						-2.5			-2.9			-1.1			
13			0.7		-1.9			-2.9						2.5	
14			2.3				-2.8			-2.4		-0.8			
15					-2.1			-2.8							

CASE B-2

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER.

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-17

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING TRANSFERRED TO THE REGION 1 ASSEMBLY

4026-388

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6			5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER.

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-18
ENRICHMENT ERROR: A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION

Revision 21 September 2013

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11.0			-14.0					
2			0.4			-9.2		-12.0							
3								-12.0		-14.0		-15.0		-13.0	
4		3.2	1.2					-11.0							
5					-1.5				-12.0		-15.0		-16.0		
6	9.8		7.1			-1.6		-8.0						-16.0	
7				9.2			-2.3			-12.0			-14.0		
8	20.0		17.8		10.8		0.8			-10.0		-14.0	-15.0	-16.0	
9		27.2							-5.5		-11.0				-15.0
10					20.7		5.8					-12.0			
11	42.0		X		23.6			1.9			-8.6				-13.0
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER.

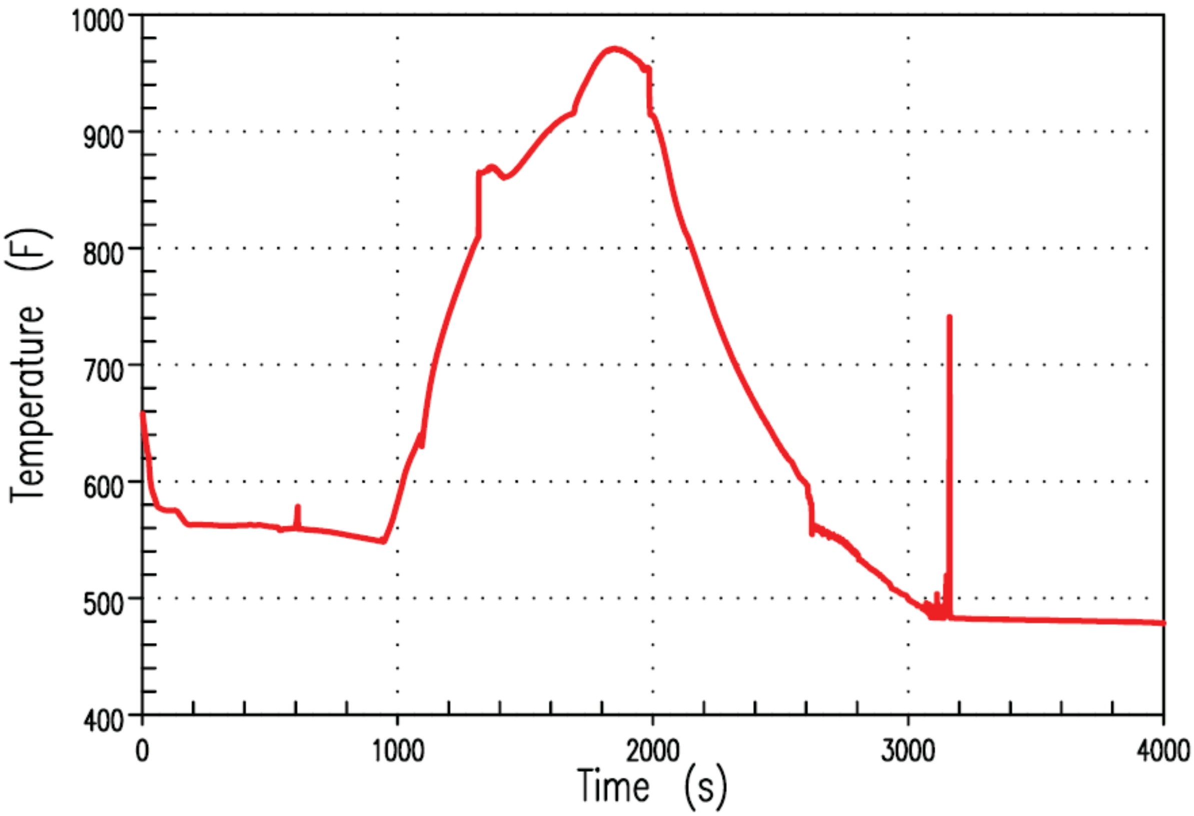
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-19

LOADING A REGION 2 ASSEMBLY INTO A REGION 1 POSITION NEAR CORE PERIPHERY

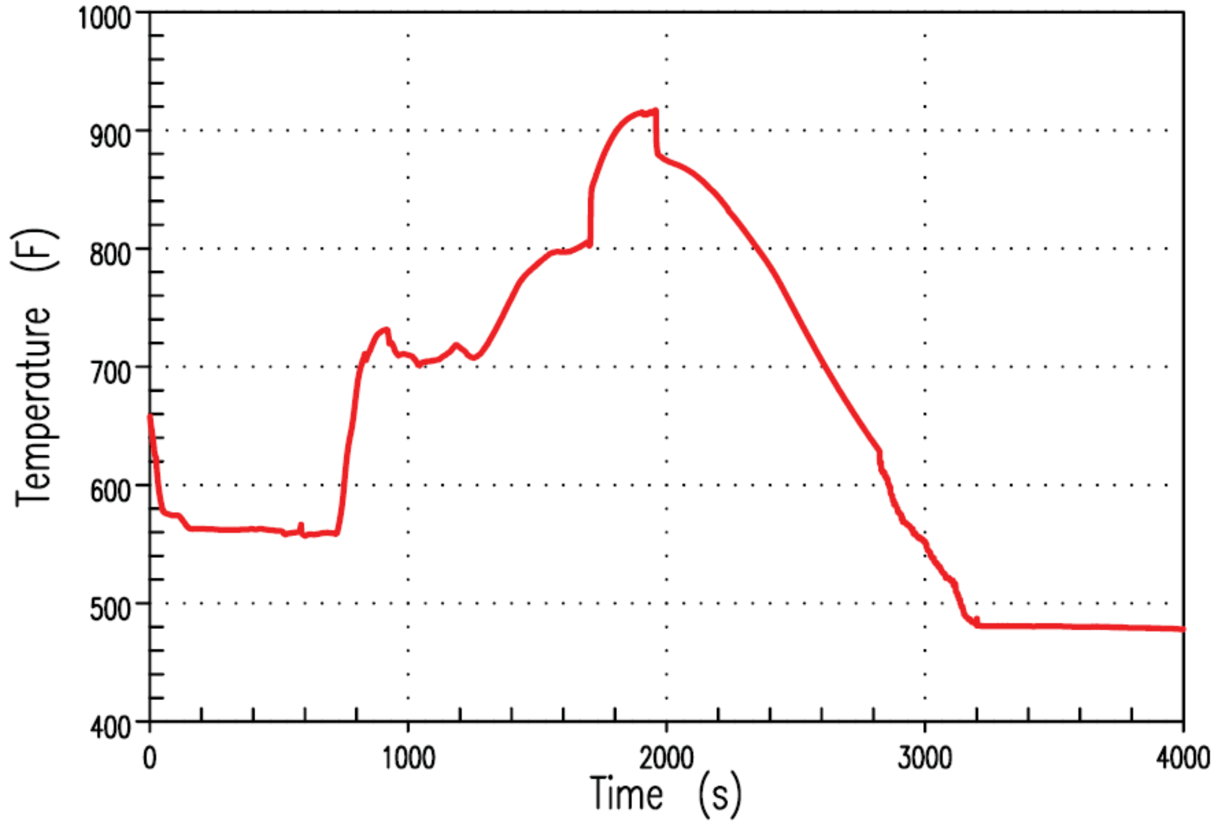
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-33 (Sheet 1 of 2) TOP CORE NODE VAPOR TEMPERATURE 3-INCH COLD LEG BREAK

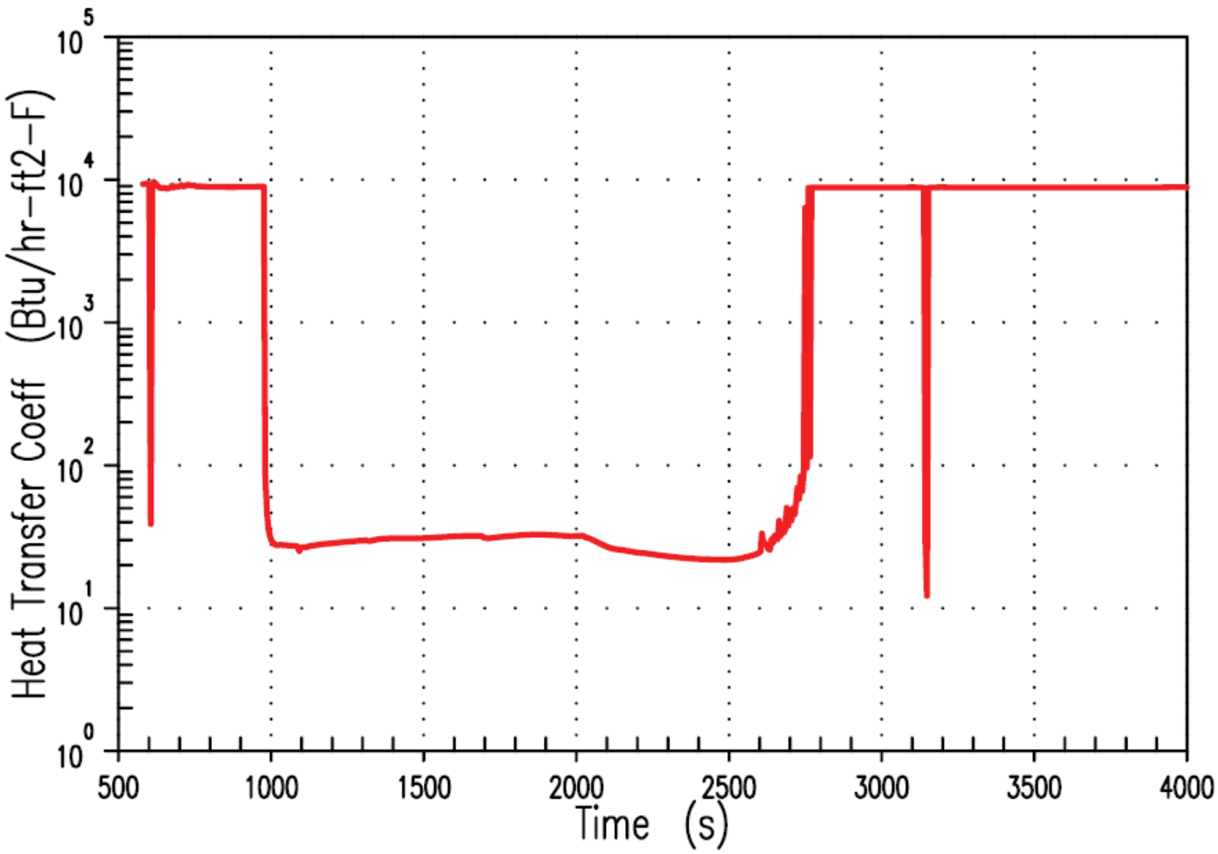
Revision 21 September 2013

DCPP Unit 2



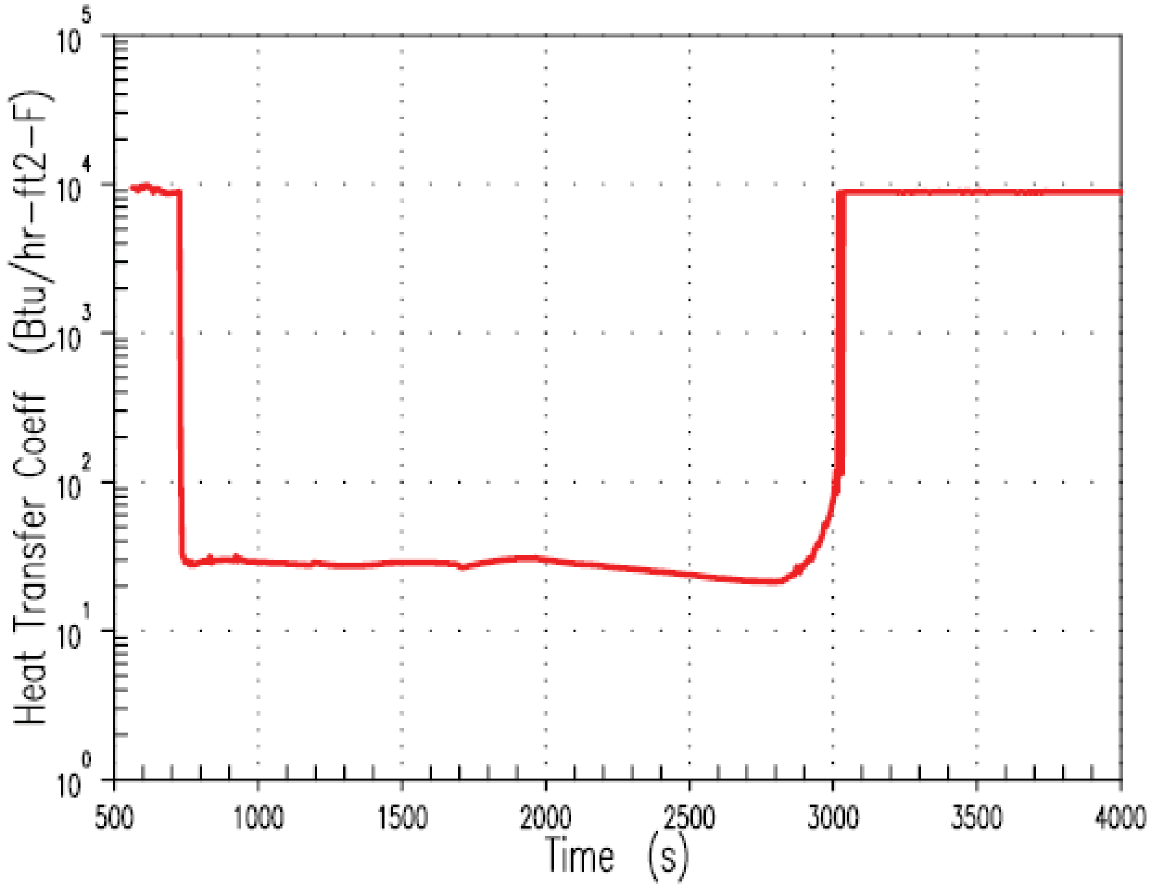
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-33 (Sheet 2 of 2) TOP CORE NODE VAPOR TEMPERATURE 3-INCH COLD LEG BREAK

DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-34 (Sheet 1 of 2) ROD FILM COEFFICIENT 3-INCH COLD LEG BREAK

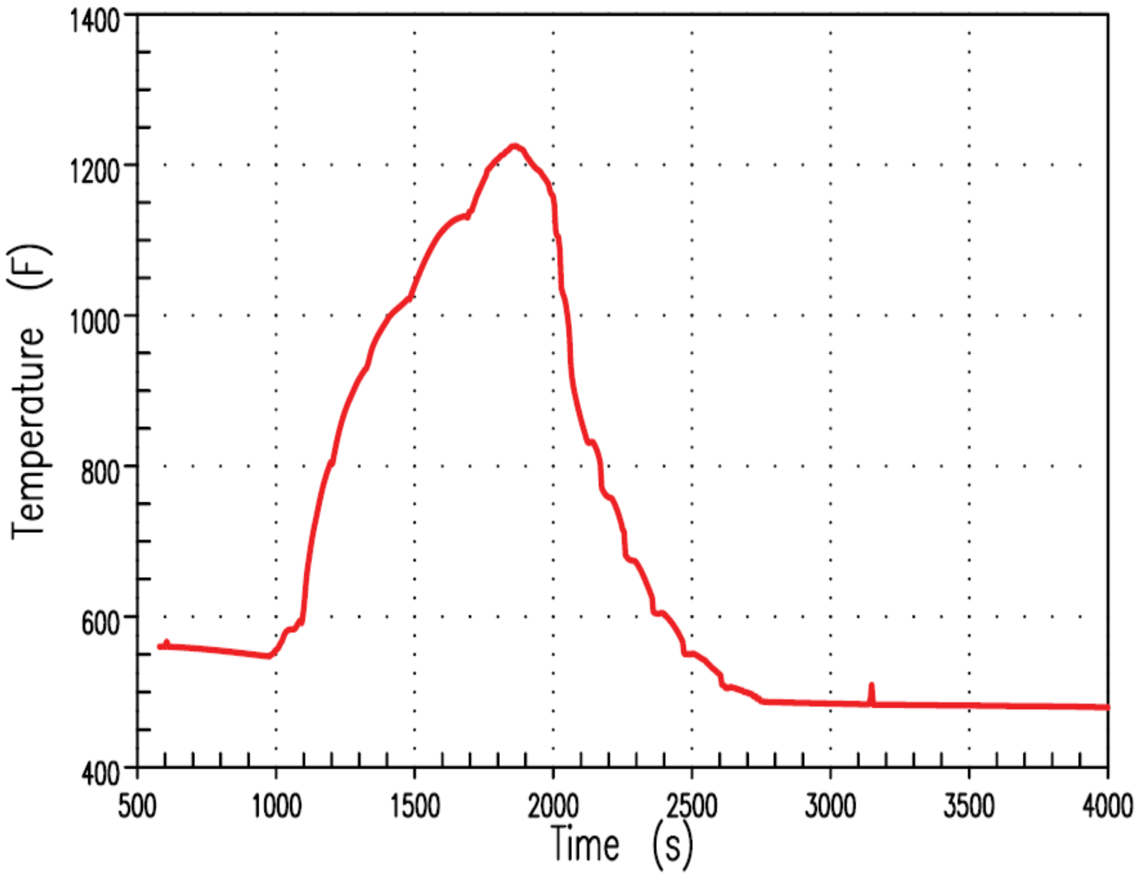
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-34 (Sheet 2 of 2) ROD FILM COEFFICIENT 3-INCH COLD LEG BREAK

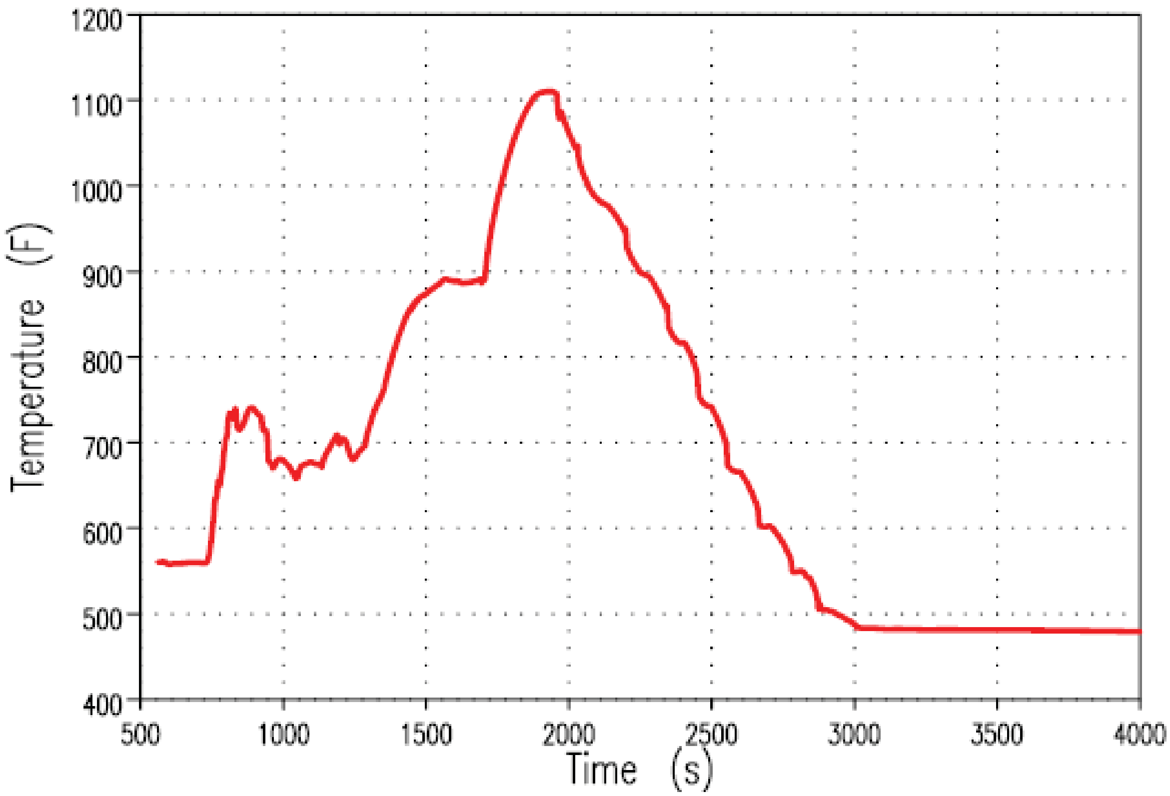
Revision 21 September 2013

DCPP Unit 1



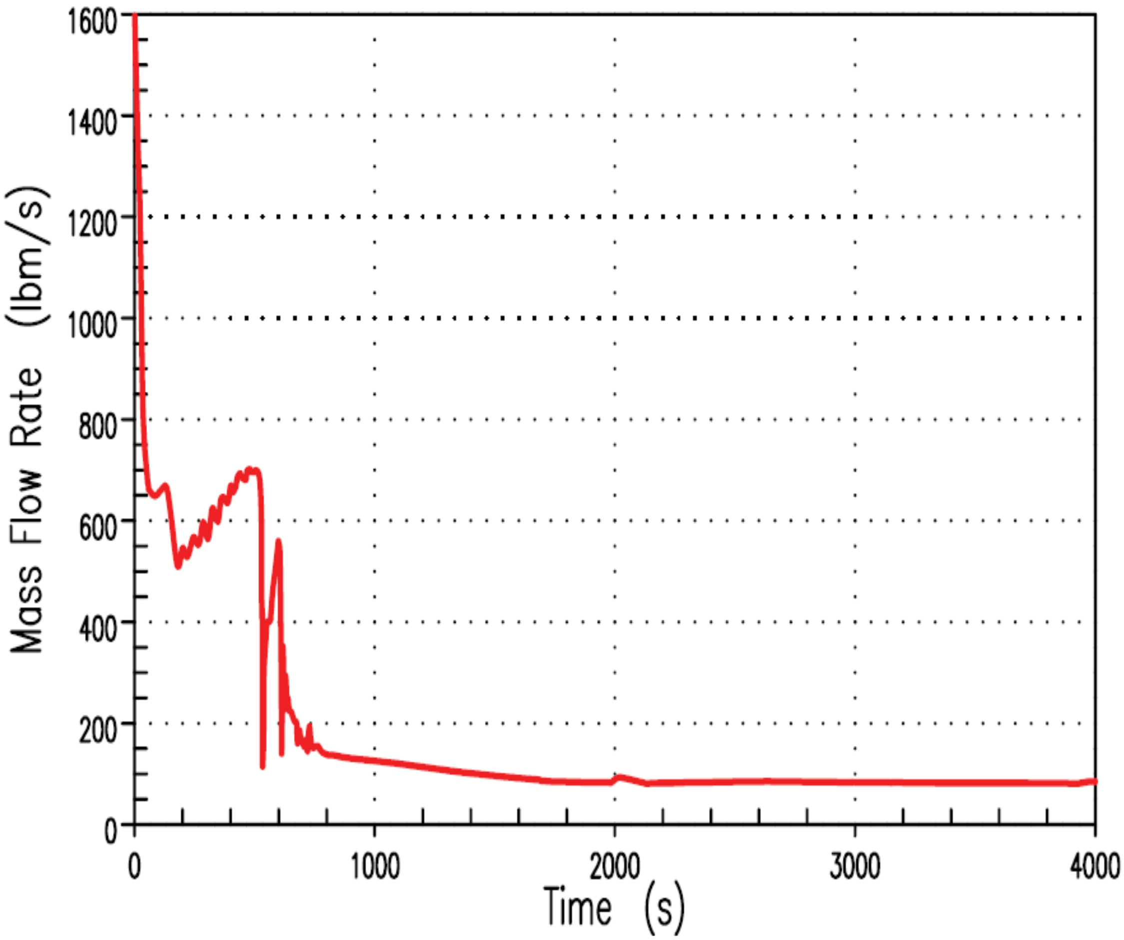
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-35 (Sheet 1 of 2) HOT SPOT FLUID TEMPERATURE 3-INCH COLD LEG BREAK

DCPP Unit 2



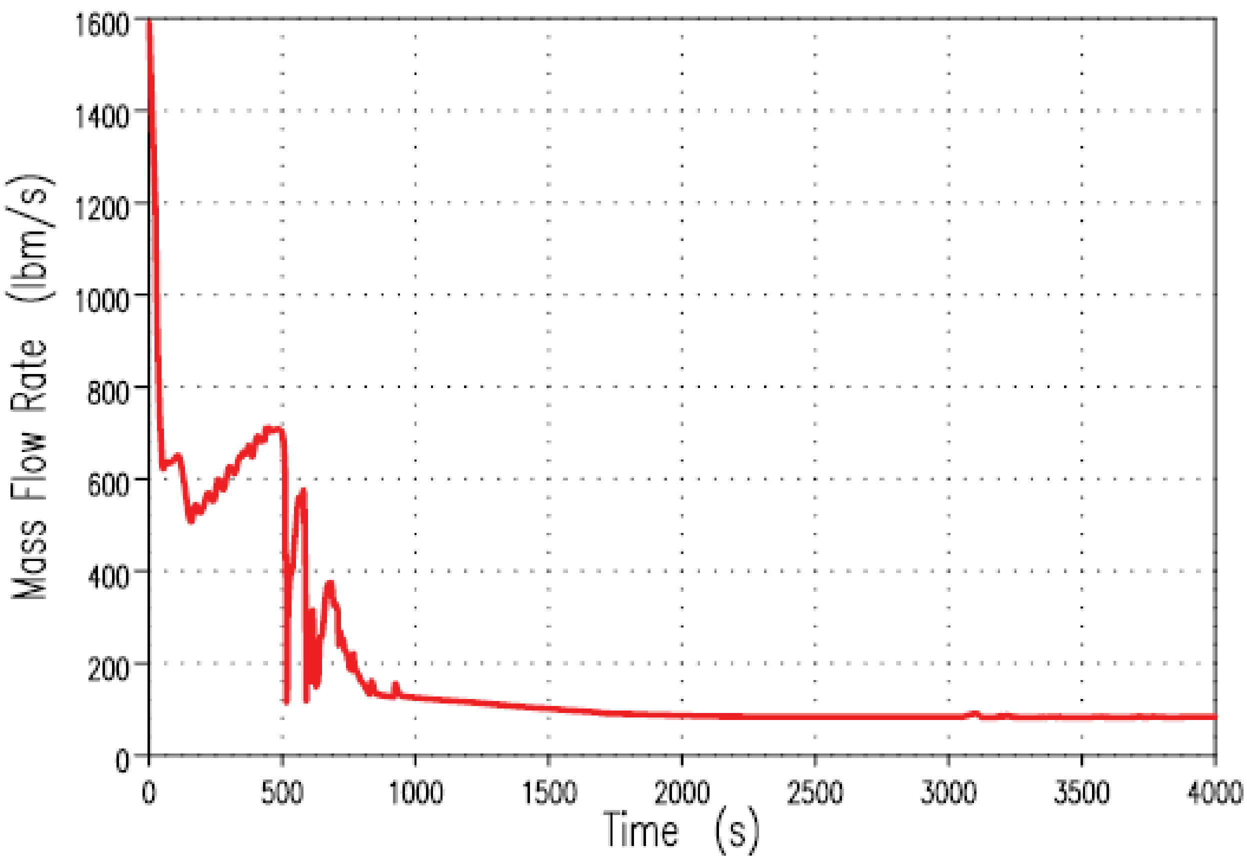
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-35 (Sheet 2 of 2) HOT SPOT FLUID TEMPERATURE 3-INCH COLD LEG BREAK

DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-36 (Sheet 1 of 2) BREAK MASS FLOW 3-INCH COLD LEG BREAK

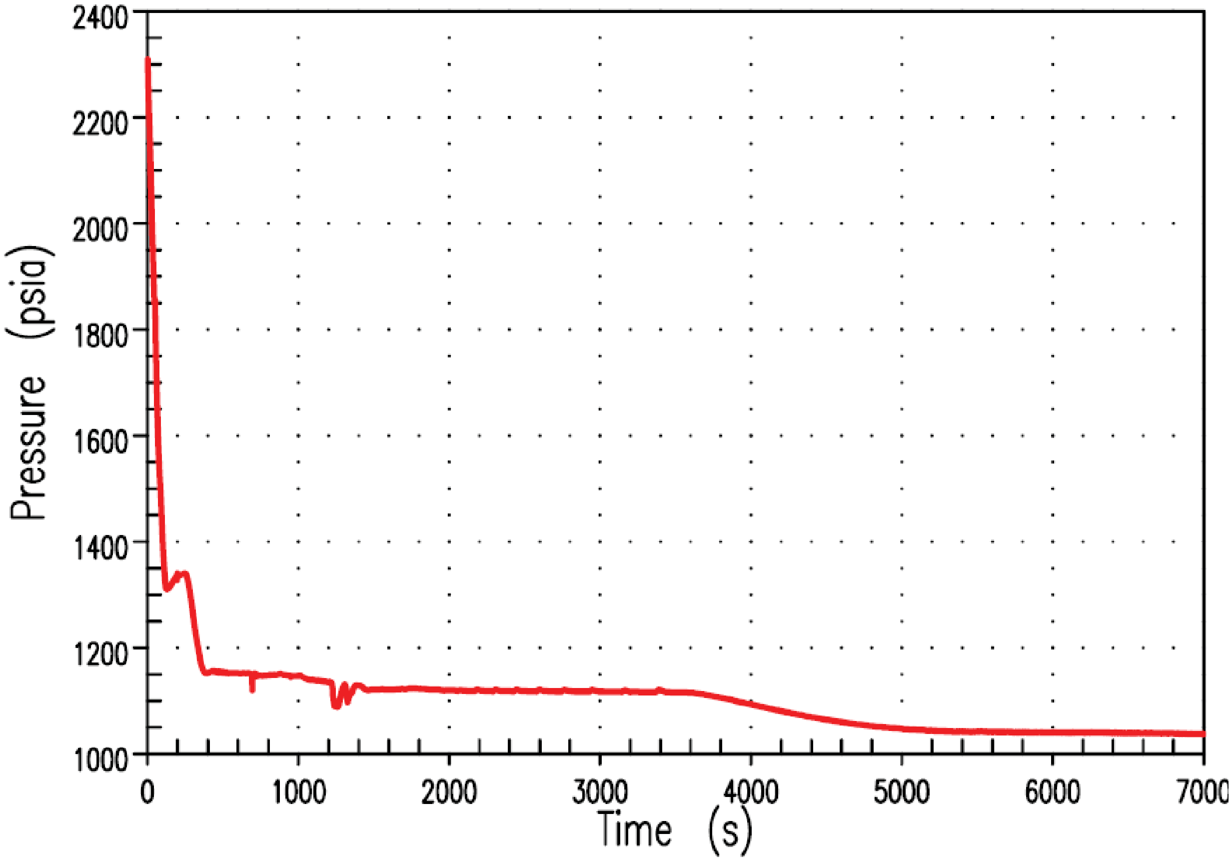
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-36 (Sheet 2 of 2) BREAK MASS FLOW 3-INCH COLD LEG BREAK

Revision 21 September 2013

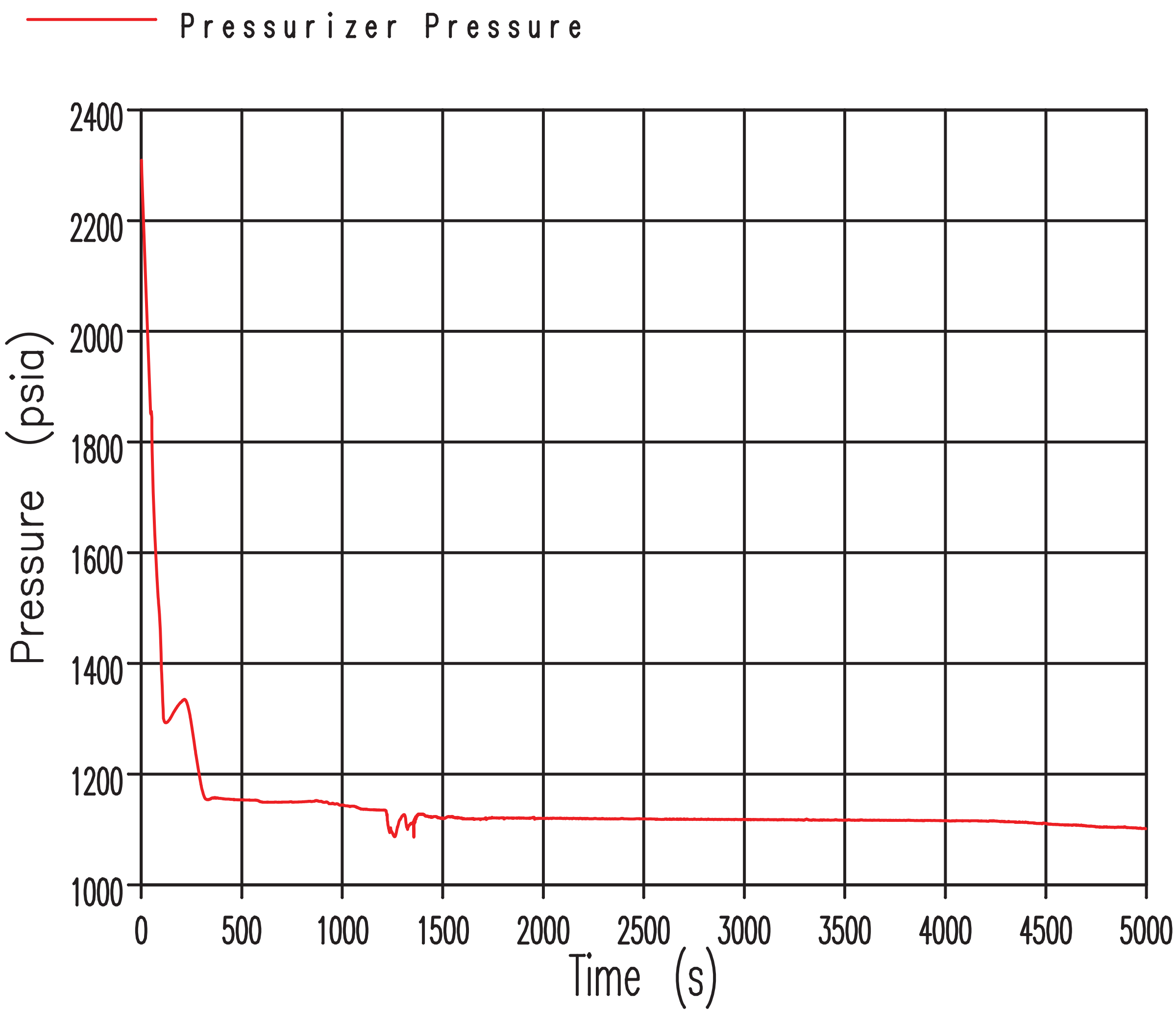
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-37 (Sheet 1 of 2) RCS DEPRESSURIZATION 2-INCH COLD LEG BREAK

Revision 21 September 2013

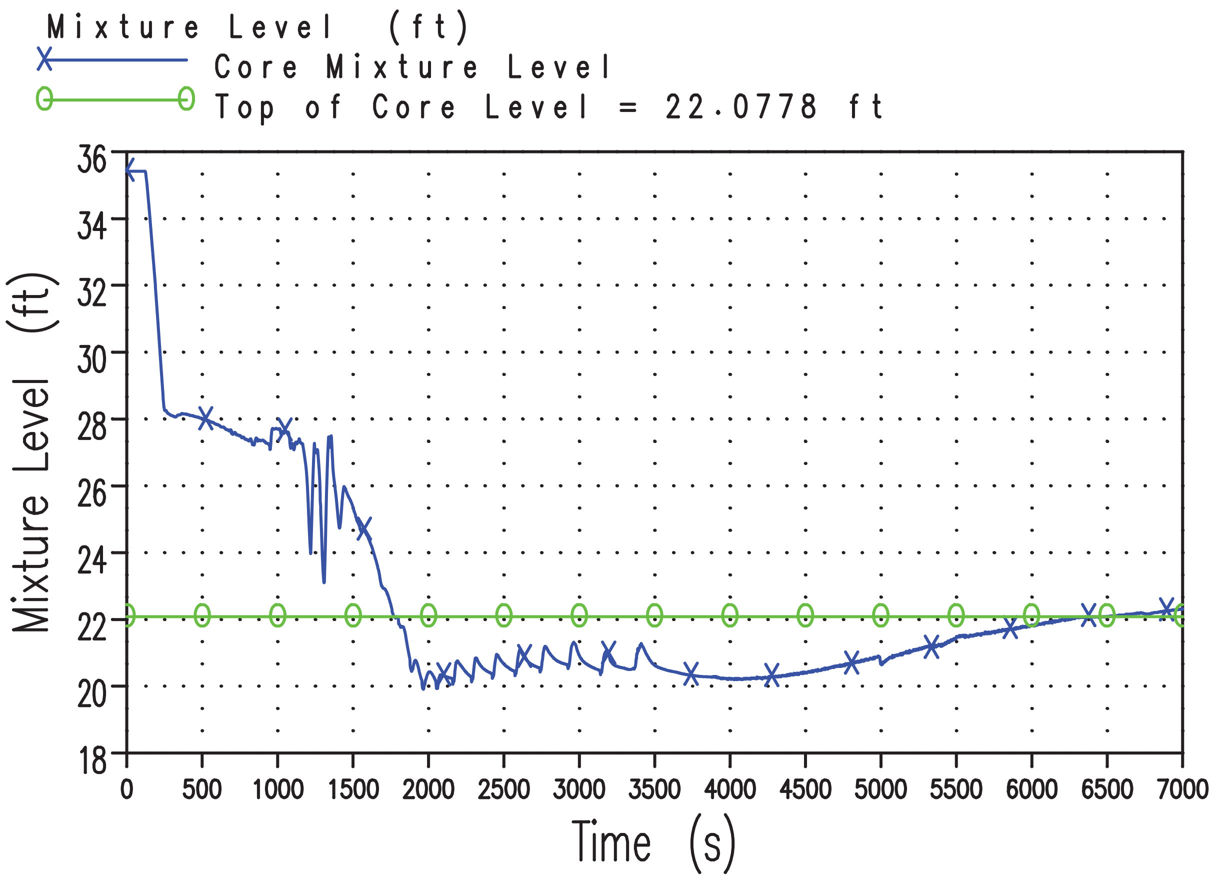
DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-37 (Sheet 2 of 2) RCS DEPRESSURIZATION 2-INCH COLD LEG BREAK

Revision 21 September 2013

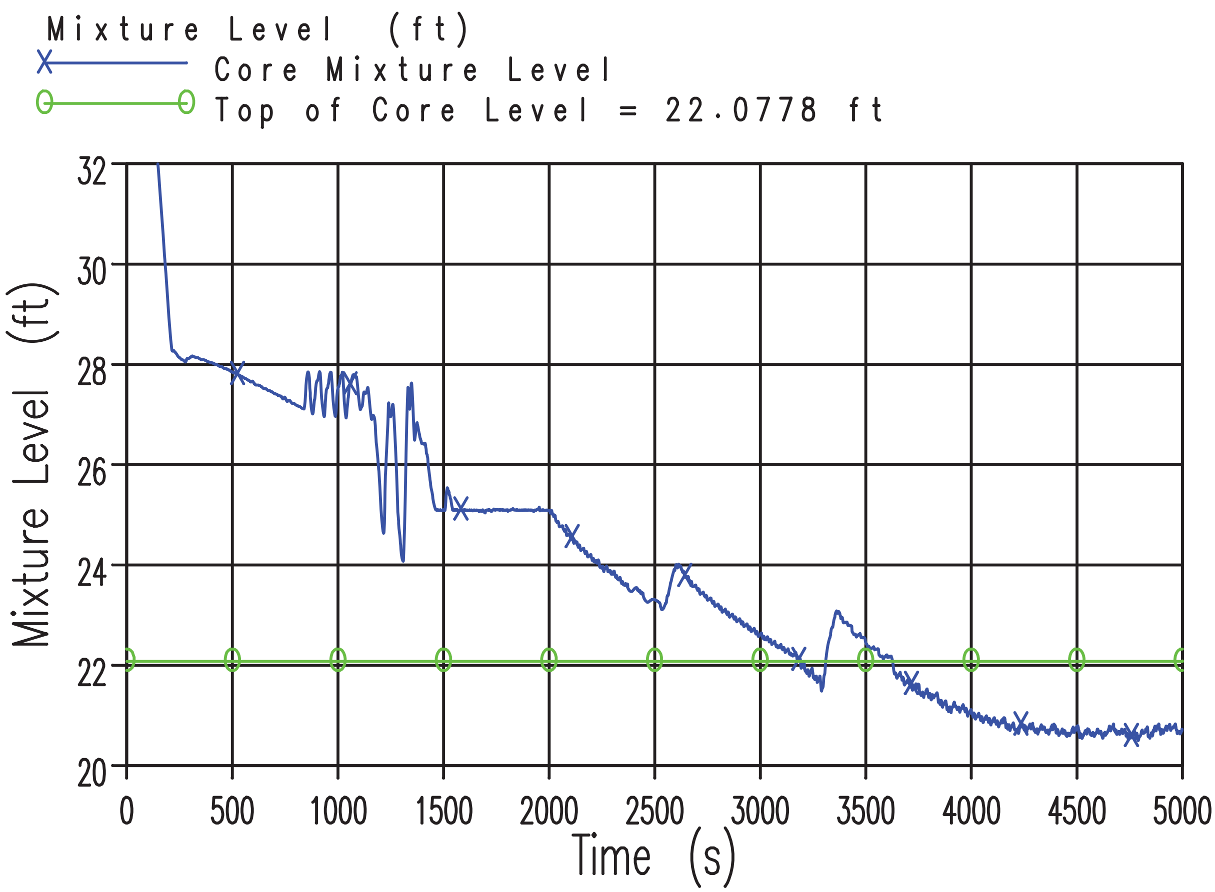
DCPP Unit 1



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-38 (Sheet 1 of 2) CORE MIXTURE ELEVATION 2-INCH COLD LEG BREAK

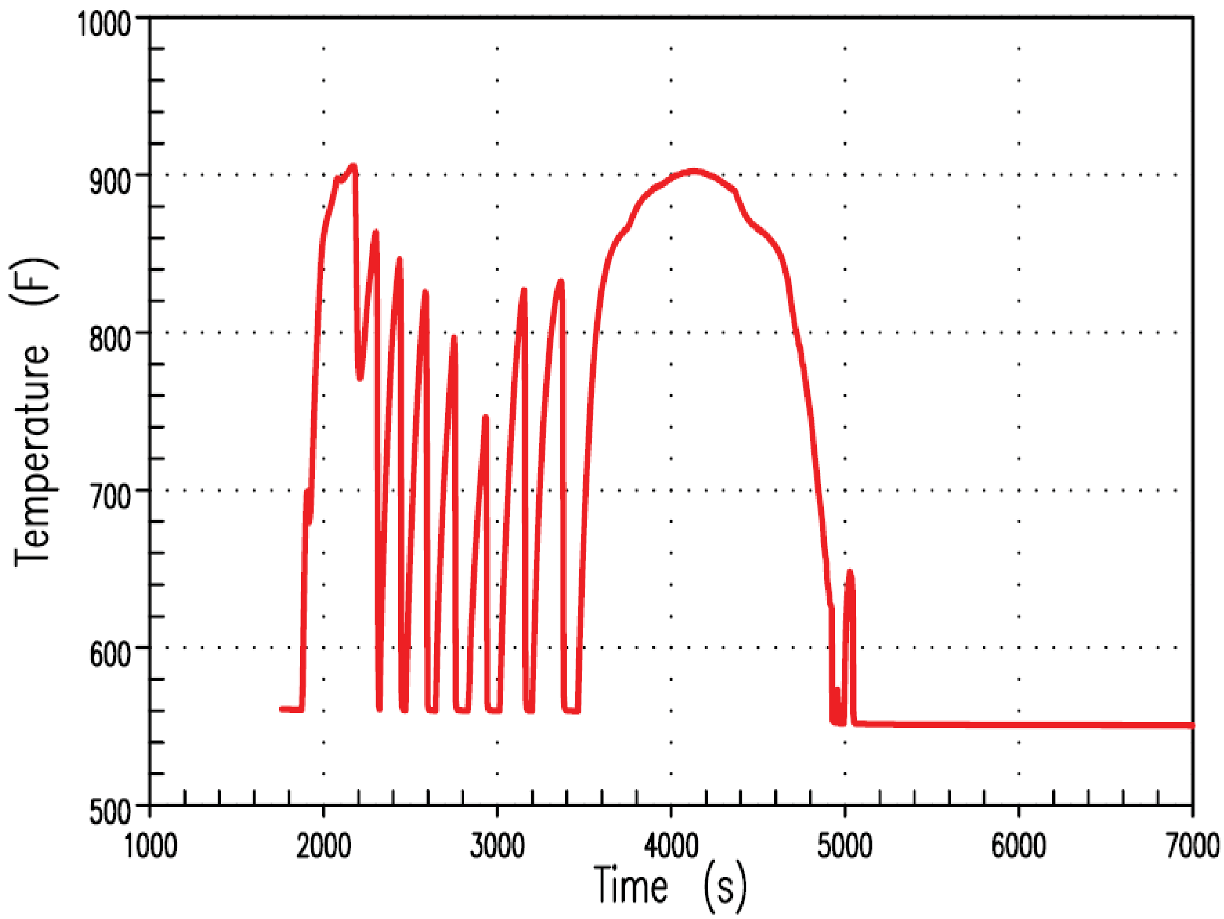
Revision 21 September 2013

DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-38 (Sheet 2 of 2) CORE MIXTURE ELEVATION 2-INCH COLD LEG BREAK

DCPP Unit 1

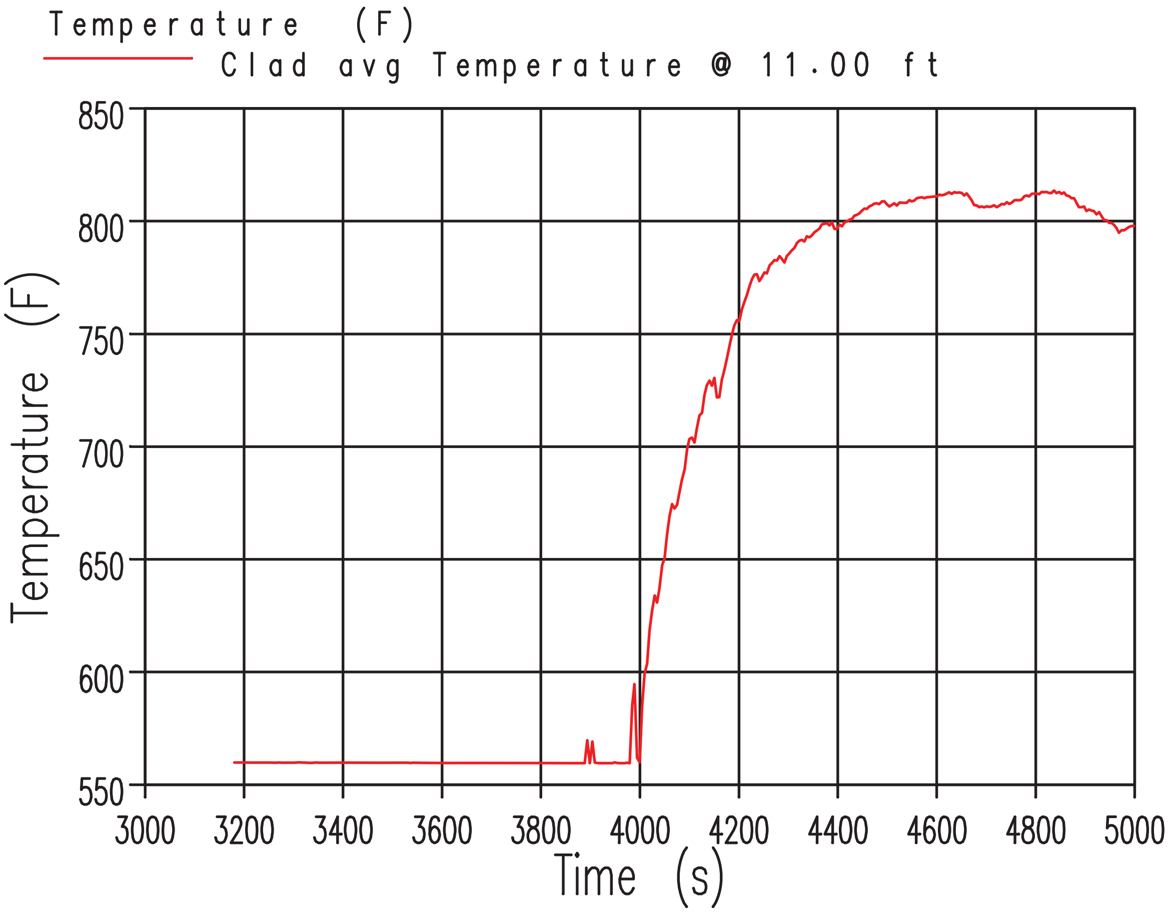


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.3-39 (Sheet 1 of 2)
CLADDING TEMPERATURE TRANSIENT
2-INCH COLD LEG BREAK

DCPP Unit 2



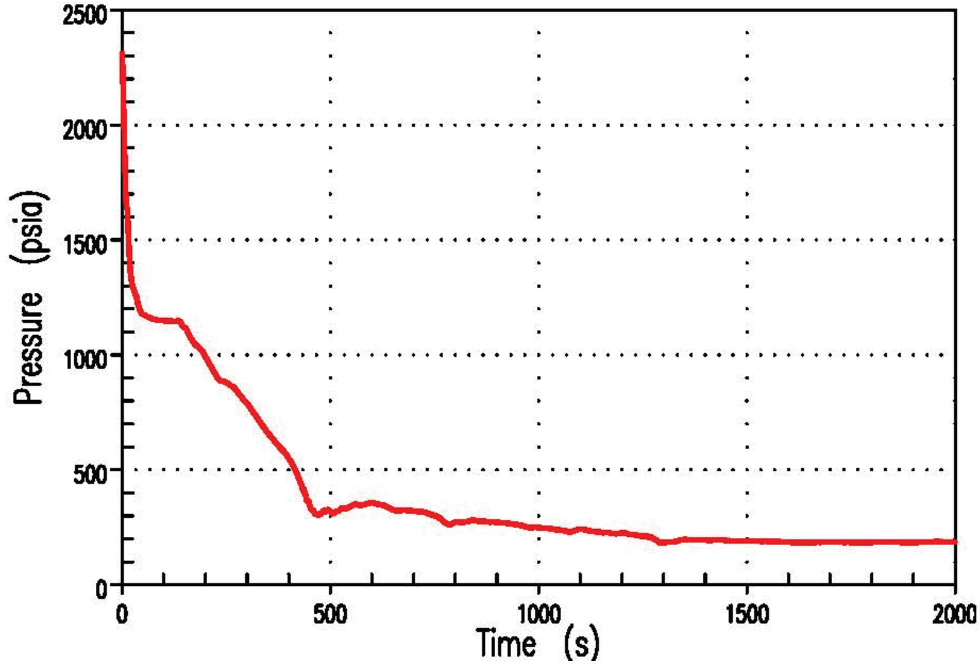
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

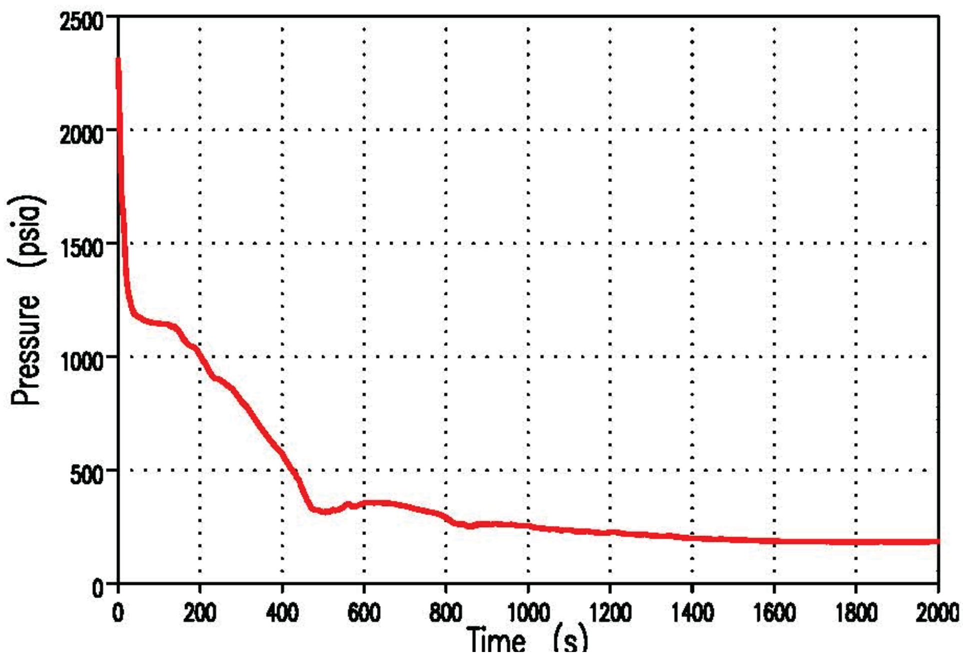
FIGURE 15.3-39 (Sheet 2 of 2)
CLADDING TEMPERATURE TRANSIENT
2-INCH COLD LEG BREAK

Revision 21 September 2013

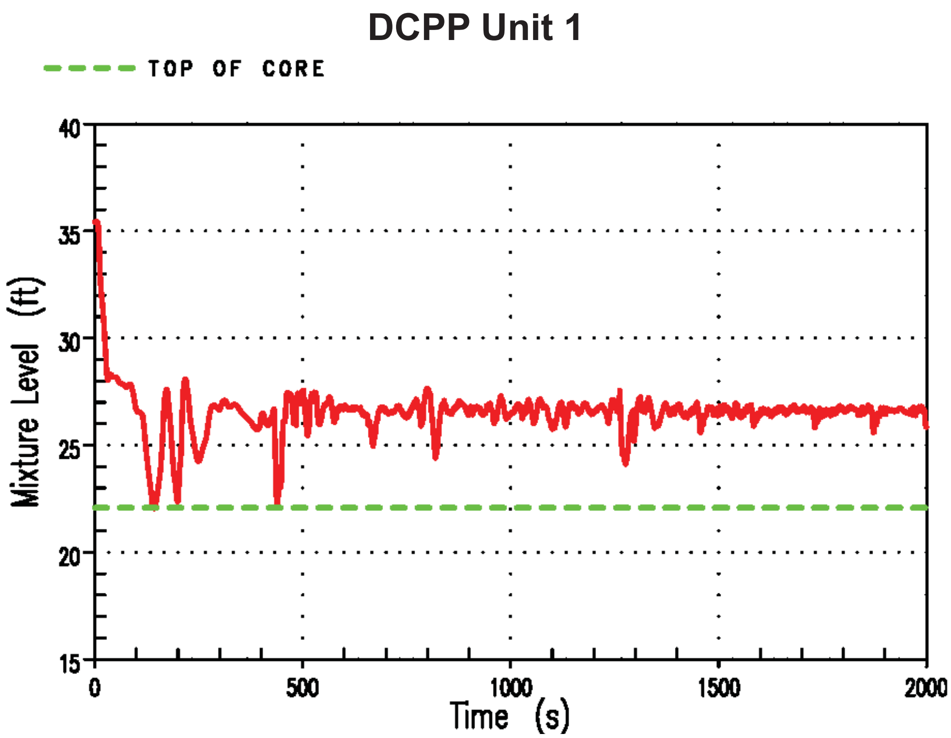
DCPP Unit 1

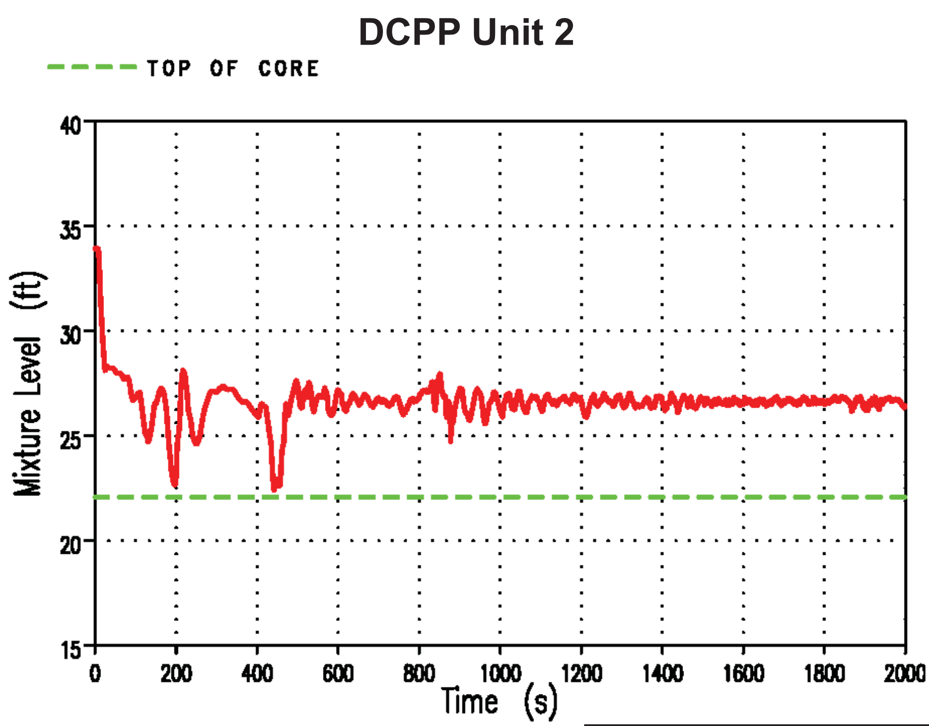


DCPP Unit 2



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3-40 RCS DEPRESSURIZATION 6-INCH COLD LEG BREAK



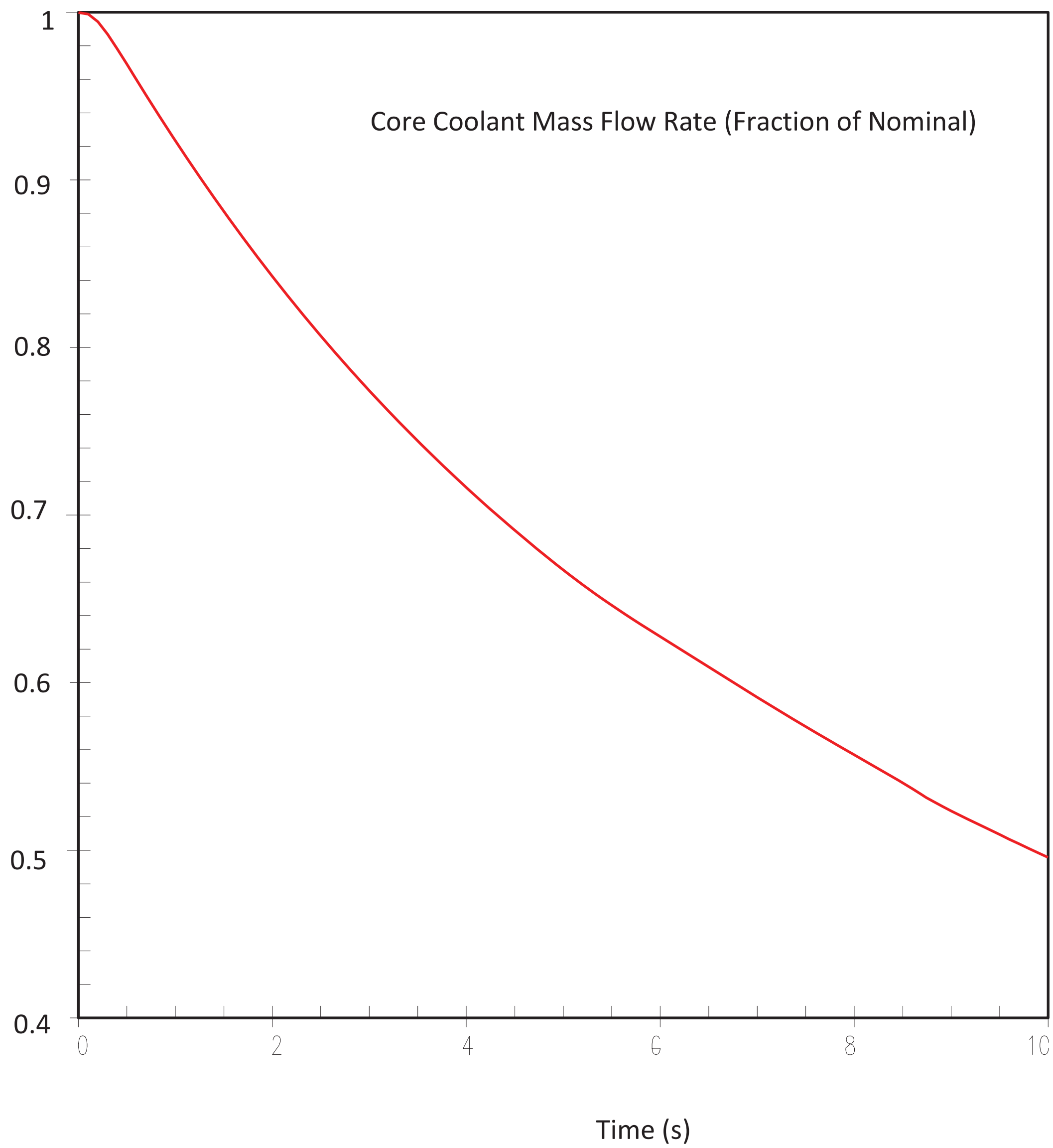


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

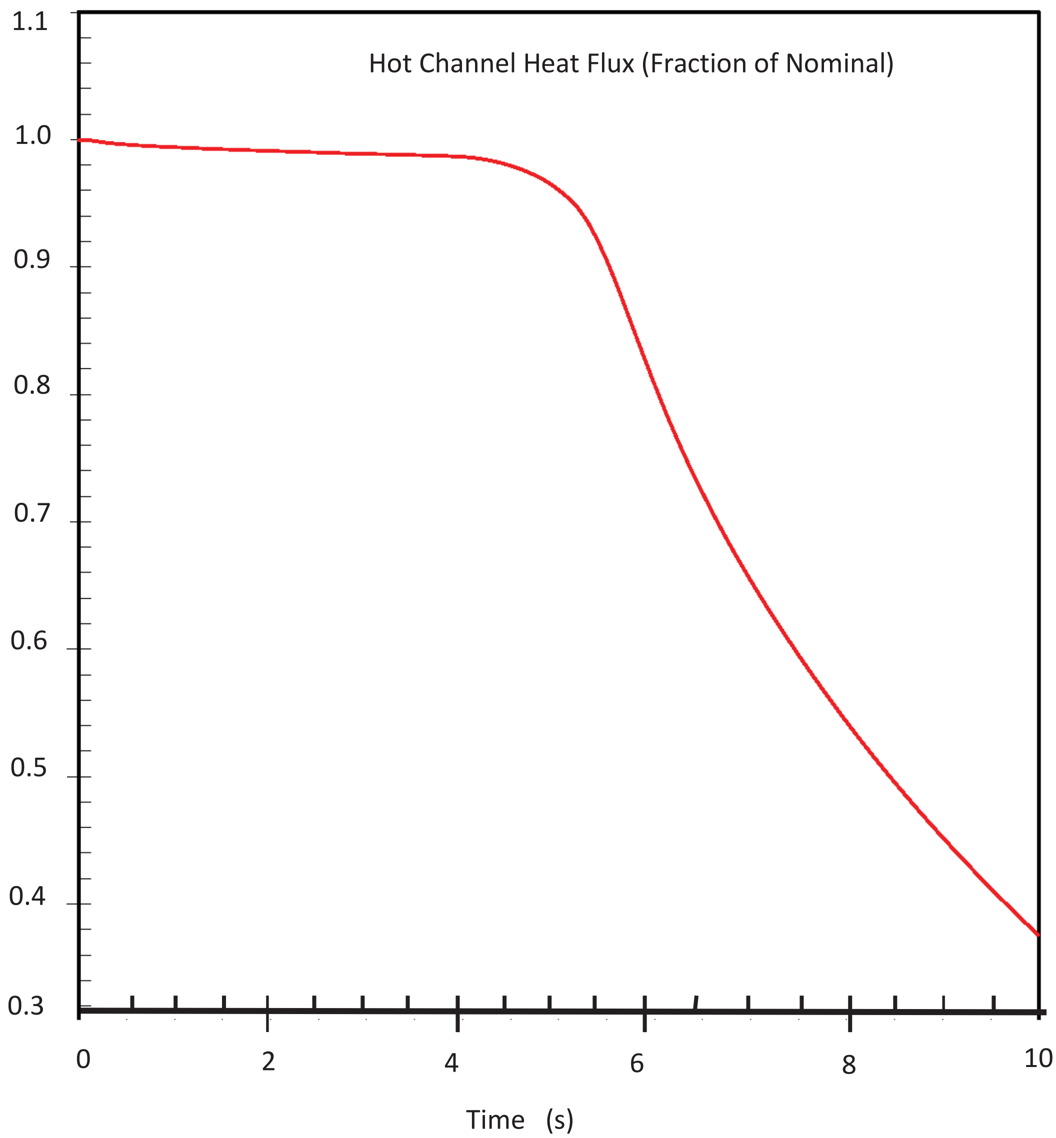
FIGURE 15.3-41

CORE MIXTURE ELEVATION
6-INCH COLD LEG BREAK



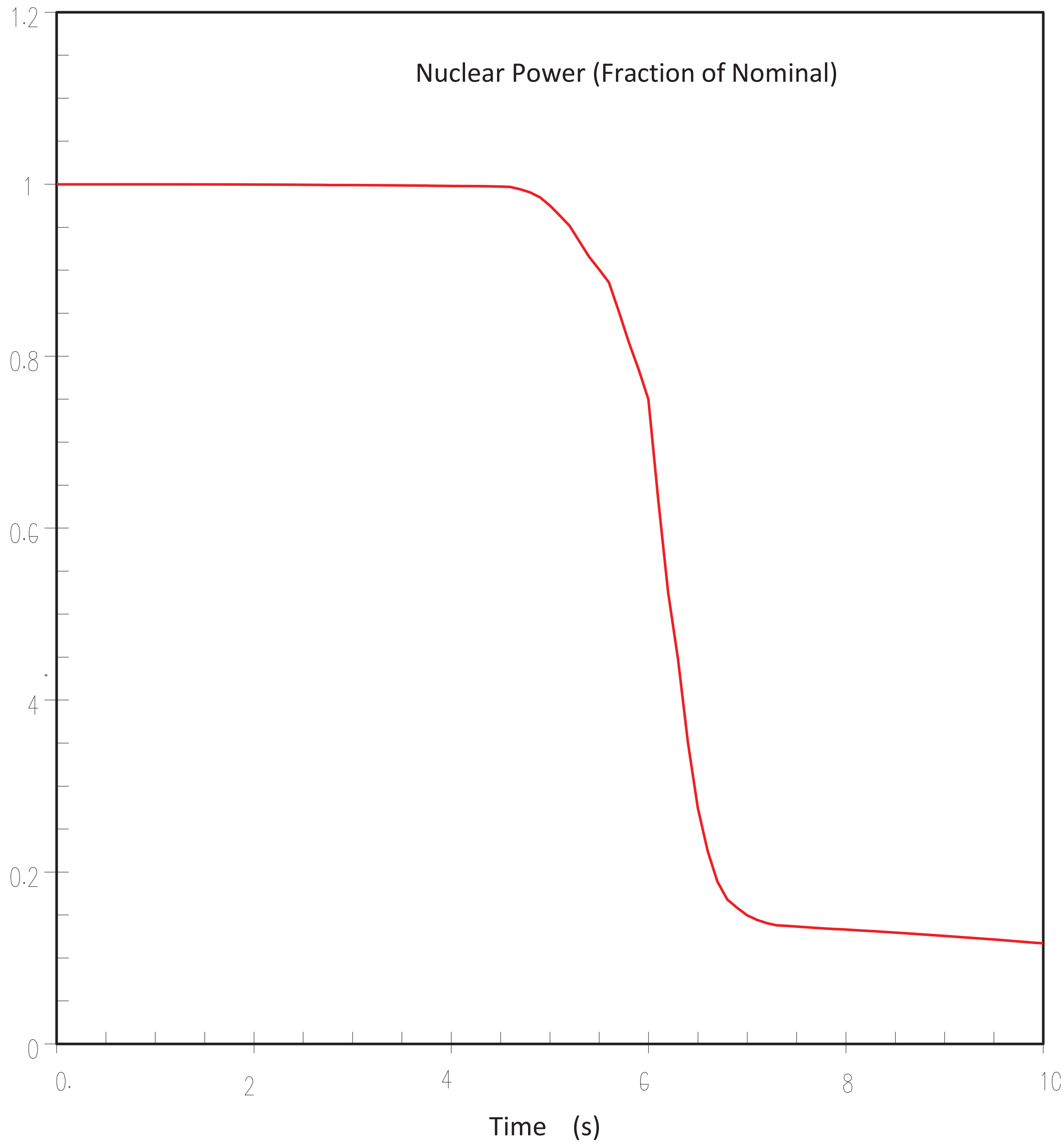
Note: The results for Unit 1 are nearly identical to those for Unit 2; therefore, the figures for Unit 1 are representative of the results for Unit 2.

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3.4-1 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW ALL LOOPS OPERATING ALL LOOPS COASTING DOWN FLOW COASTDOWN VERSUS TIME



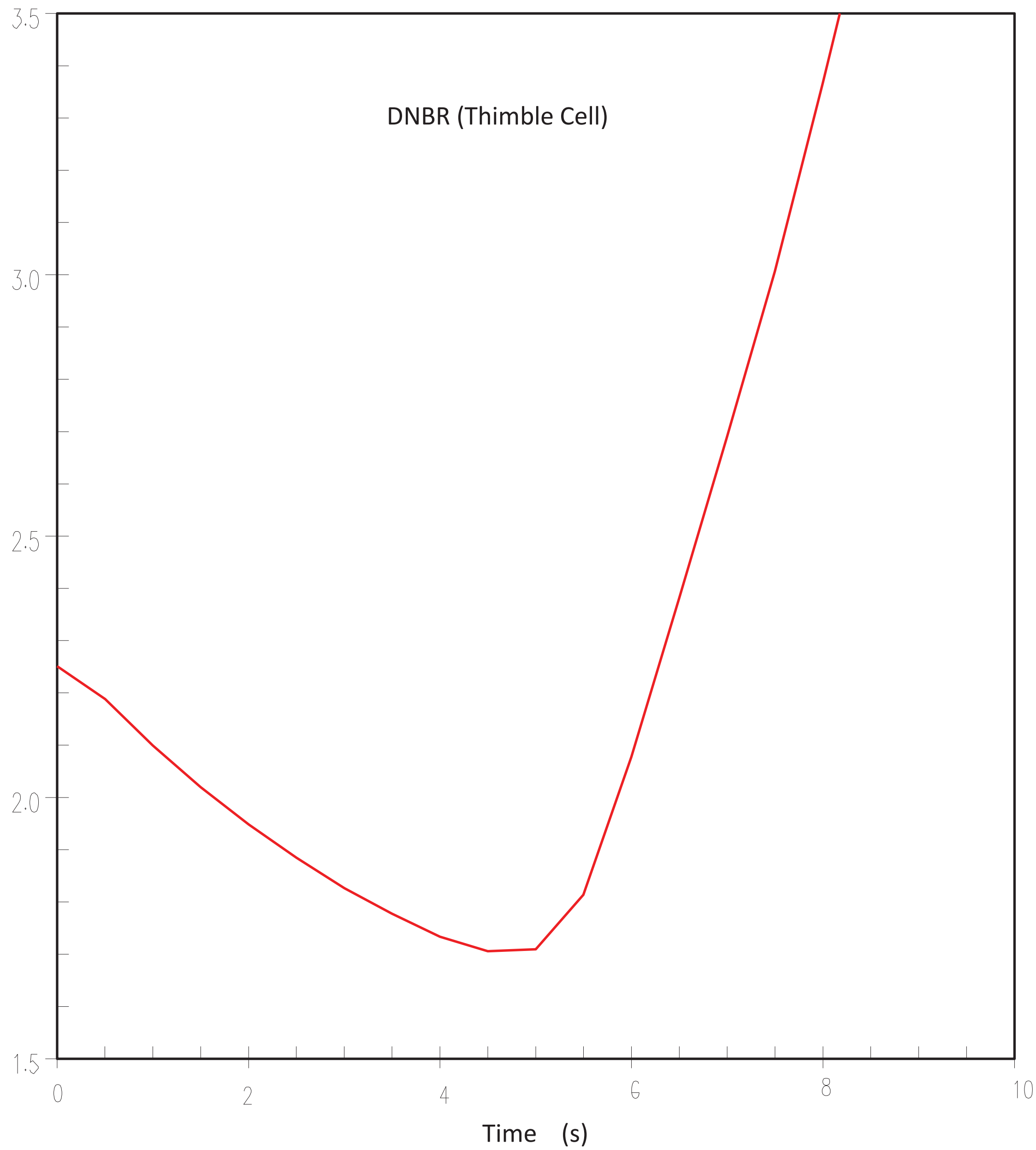
Note: The results for Unit 1 are nearly identical to those for Unit 2; therefore, the figures for Unit 1 are representative of the results for Unit 2.

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3.4-2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW ALL LOOPS OPERATING ALL LOOPS COASTING DOWN HEAT FLUX VERSUS TIME



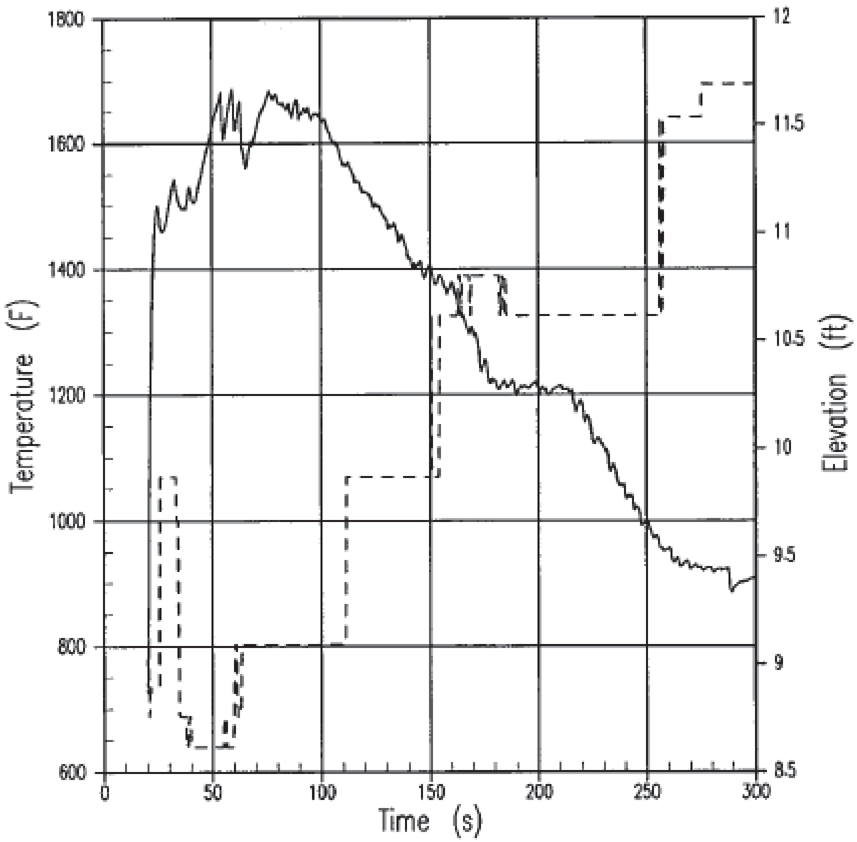
Note: The results for Unit 1 are nearly identical to those for Unit 2; therefore, the figures for Unit 1 are representative of the results for Unit 2.

FSAR UPDATE
UNITS 1 AND 2
DIABLO CANYON SITE
FIGURE 15.3.4-3
COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW
ALL LOOPS OPERATING
ALL LOOPS COASTING DOWN
NUCLEAR POWER VERSUS TIME



Note: The results for Unit 1 are nearly identical to those for Unit 2; therefore, the figures for Unit 1 are representative of the results for Unit 2.

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.3.4-4 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW ALL LOOPS OPERATING ALL LOOPS COASTING DOWN DNBR VERSUS TIME



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-1A REFERENCE TRANSIENT PCT AND PCT LOCATION

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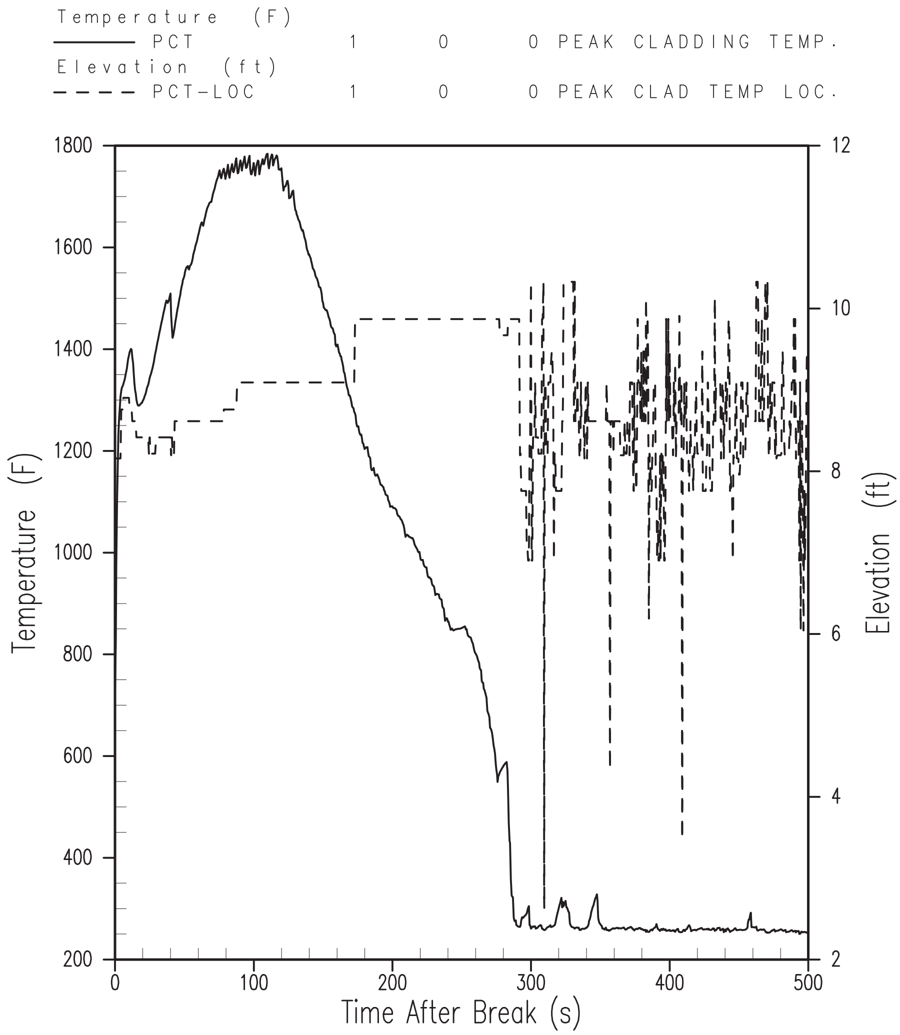


929:942:247300/18-Jul-05

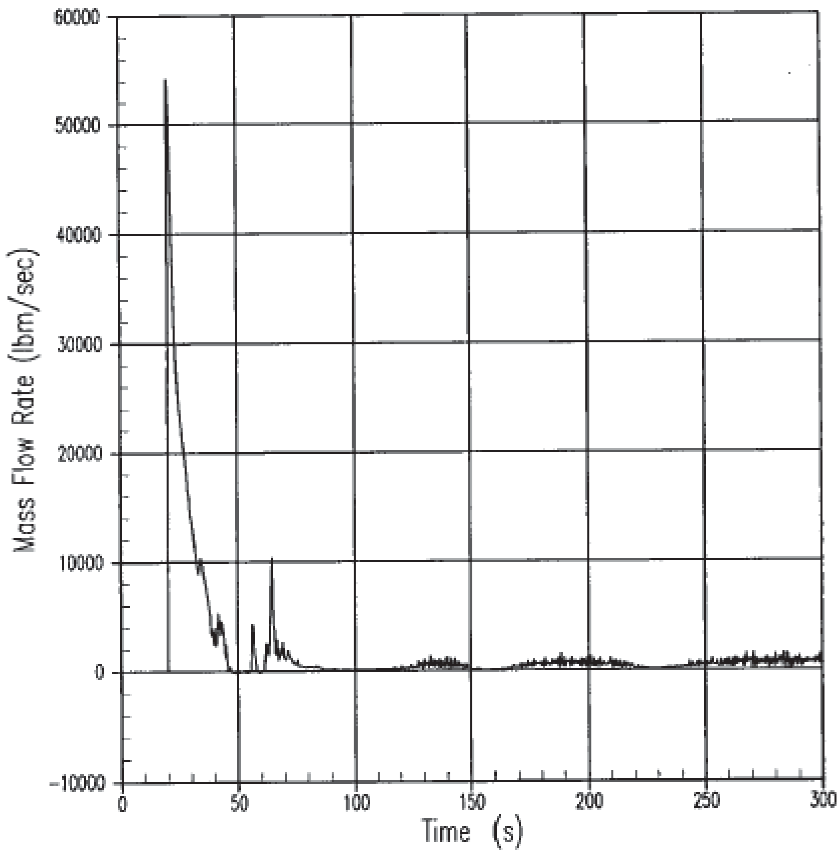
929:942:247300/18-Jul-05

FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-1B

LIMITING PCT CASE
AND PCT LOCATION



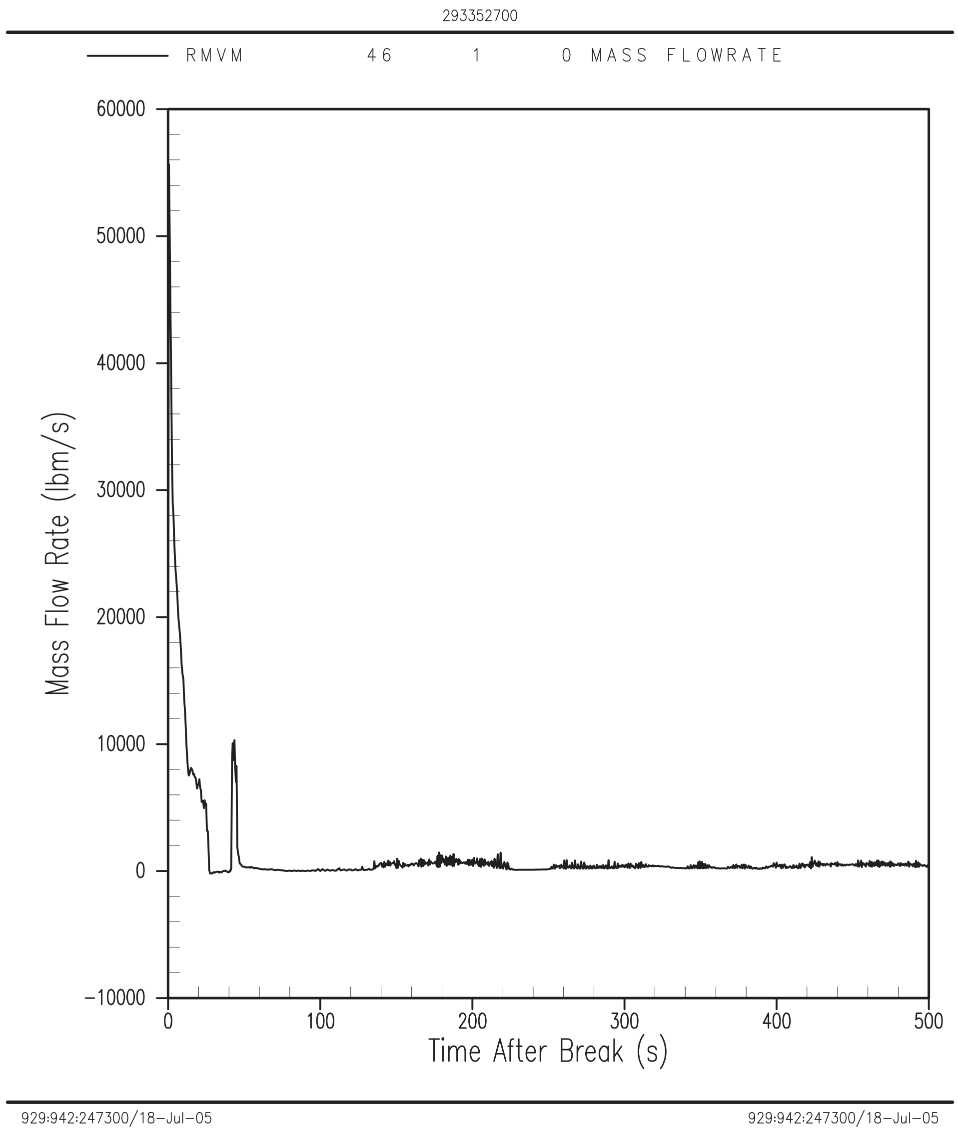
FSAR UPDATE

UNIT 1
DIABLO CANYON SITE

FIGURE 15.4.1-2A

REFERENCE TRANSIENT
VESSEL SIDE BREAK FLOW

Revision 18 October 2008

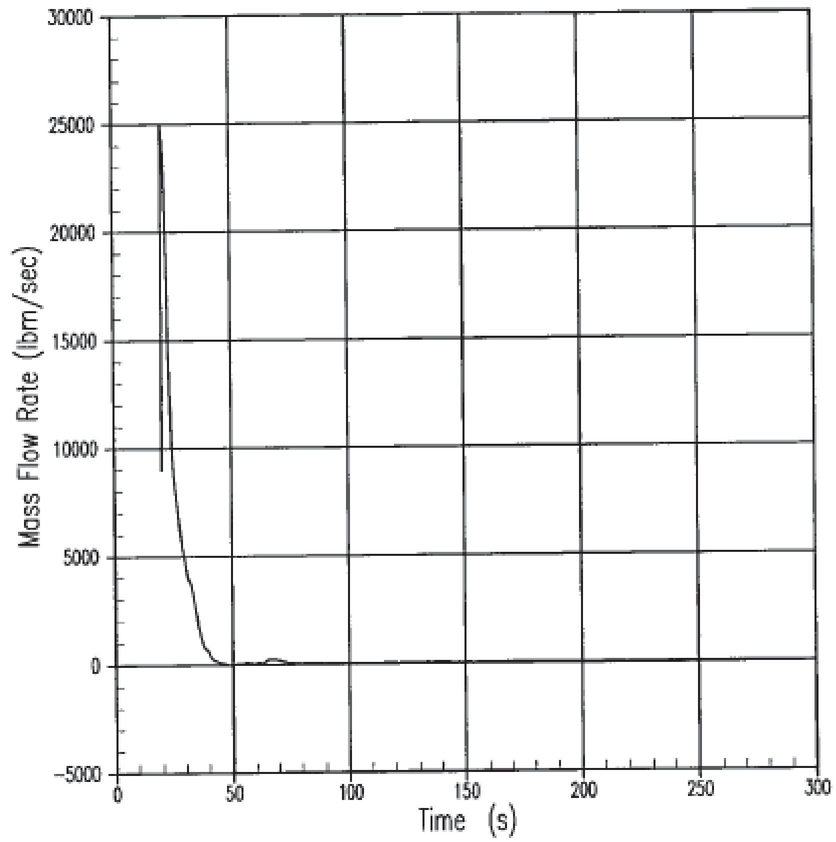


FSAR UPDATE

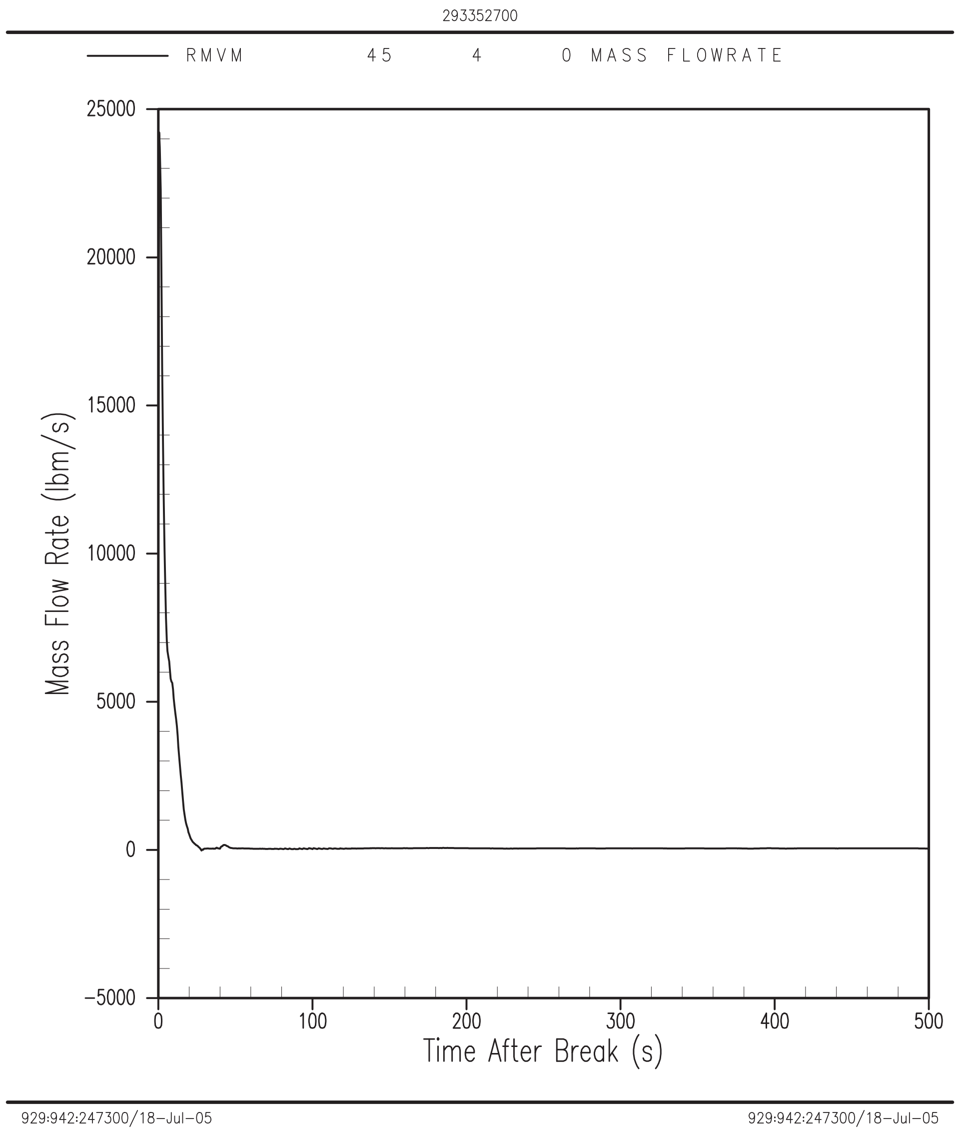
UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-2B

LIMITING PCT CASE
VESSEL SIDE BREAK FLOW



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-3A REFERENCE TRANSIENT LOOP SIDE BREAK FLOW



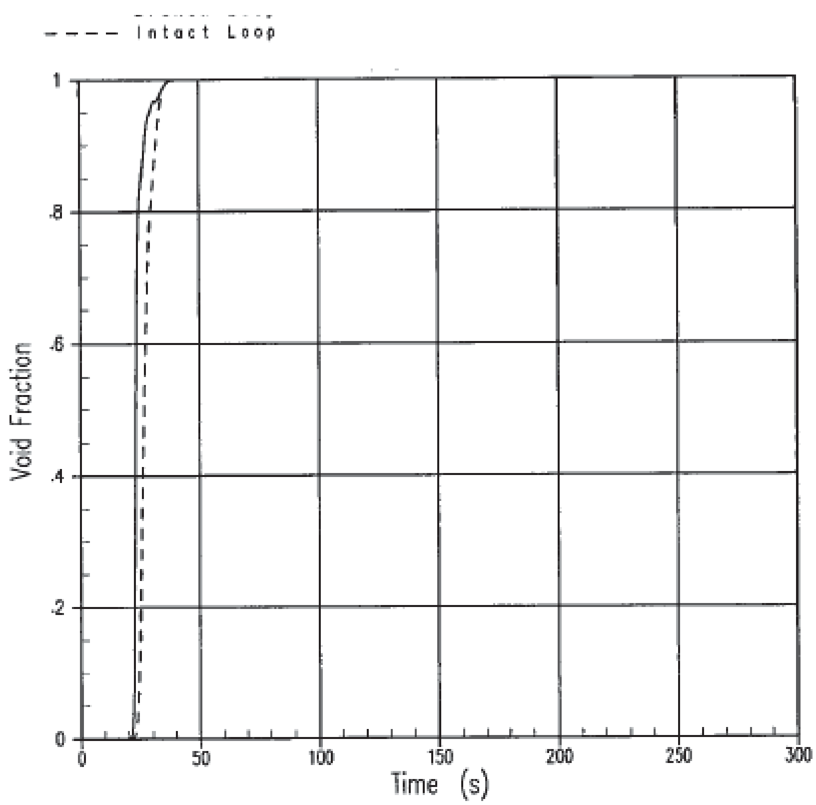
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-3B

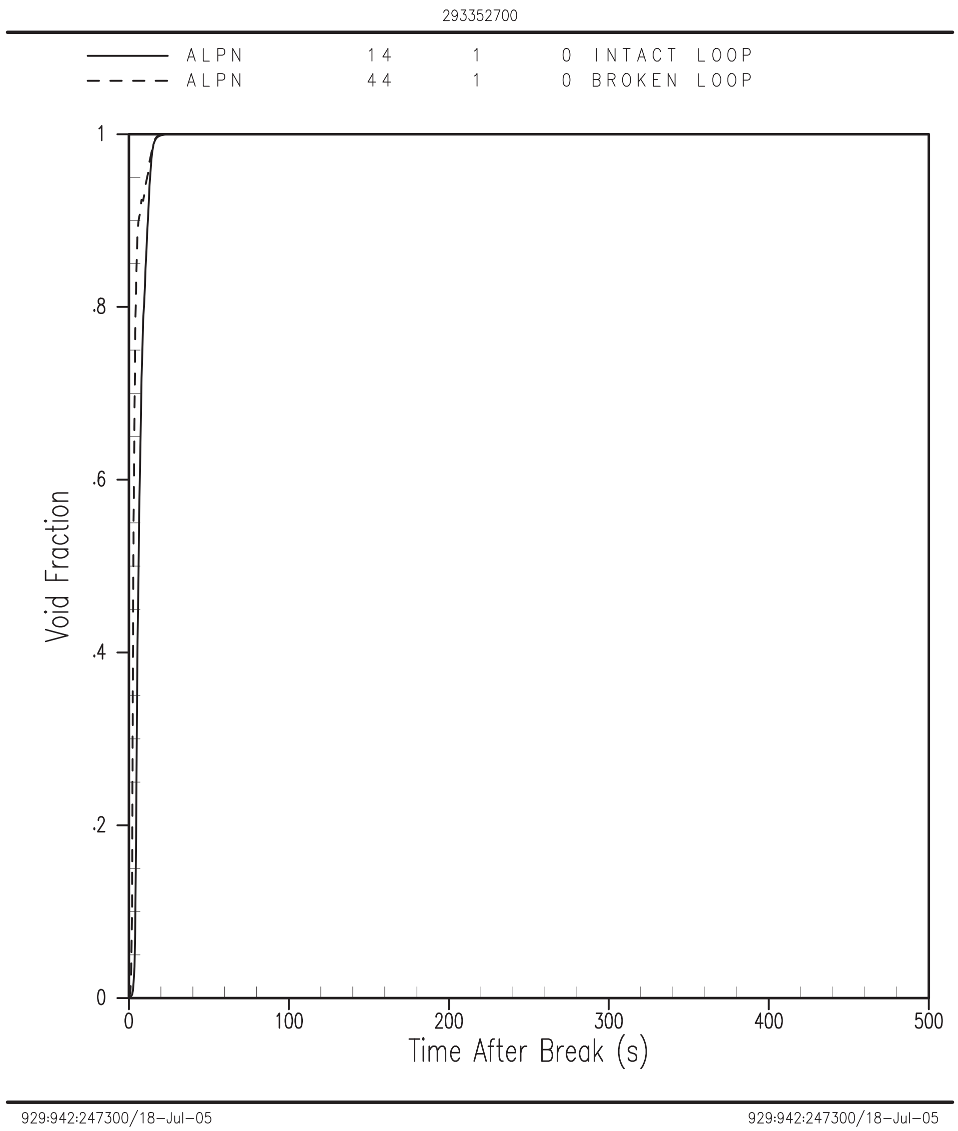
LIMITING PCT CASE LOOP SIDE BREAK FLOW

Revision 18 October 2008



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-4A REFERENCE TRANSIENT BROKEN AND INTACT LOOP PUMP VOID FRACTION

Revision 18 October 2008



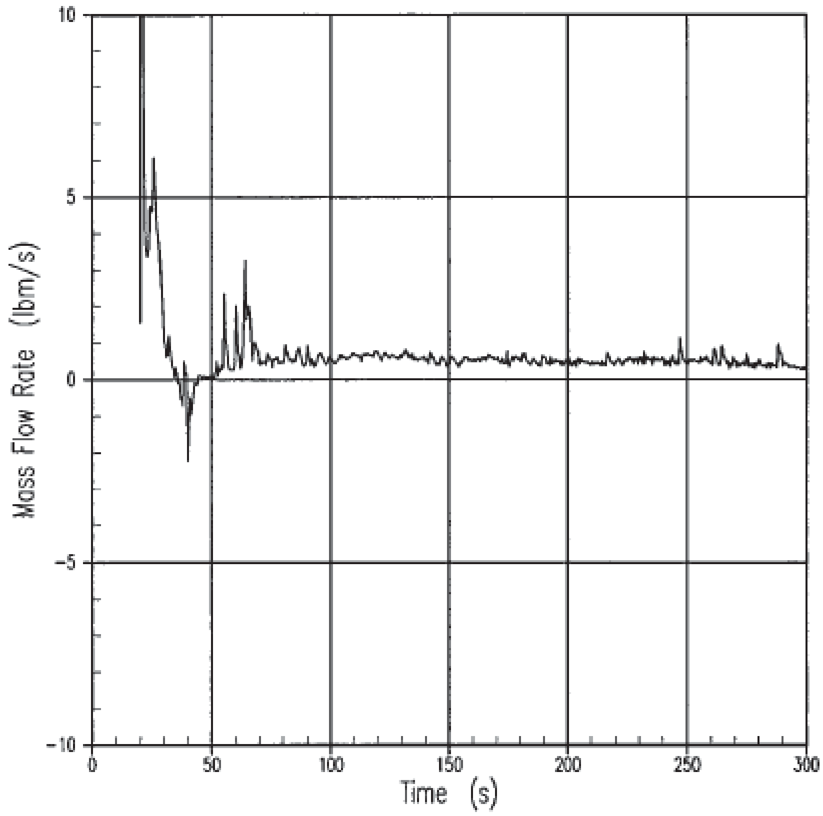
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

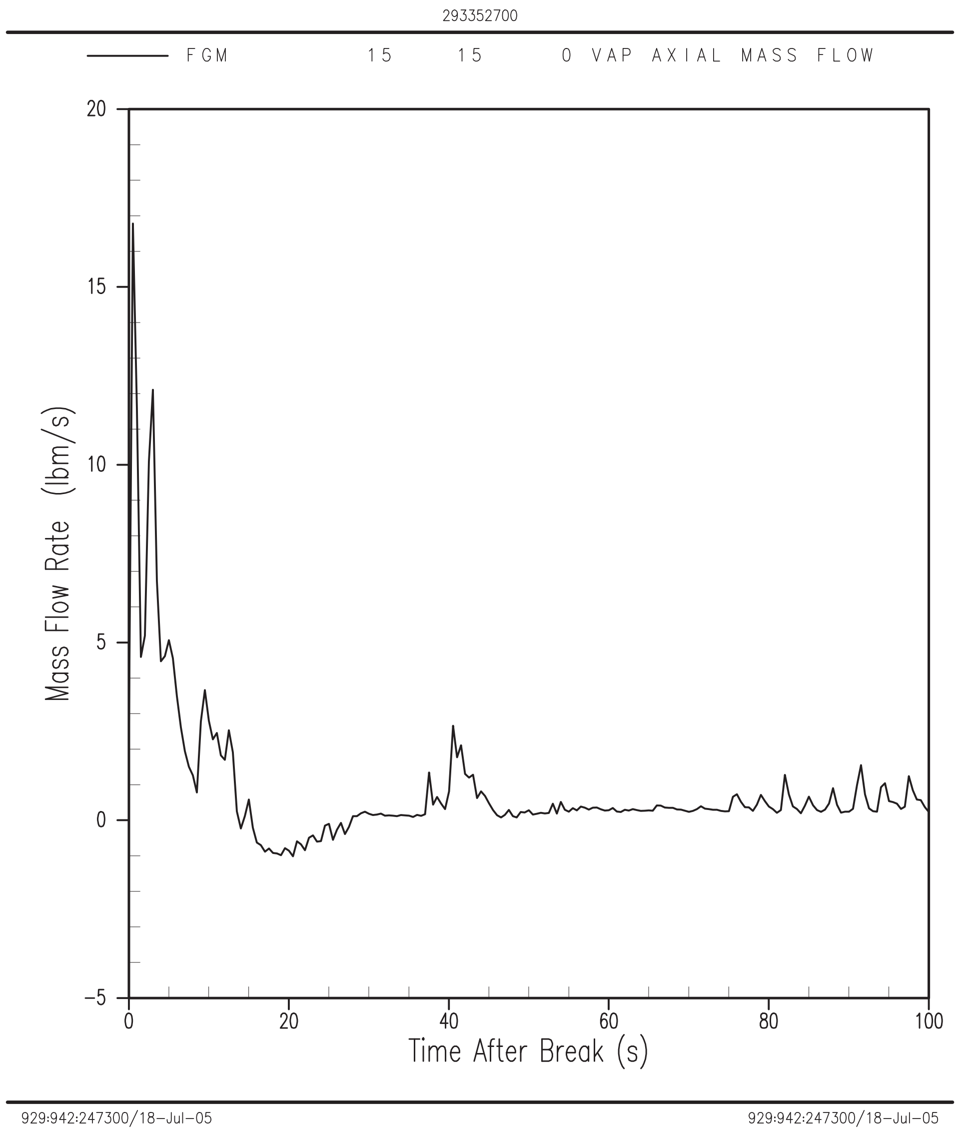
FIGURE 15.4.1-4B

LIMITING PCT CASE
BROKEN AND INTACT LOOP
PUMP VOID FRACTION

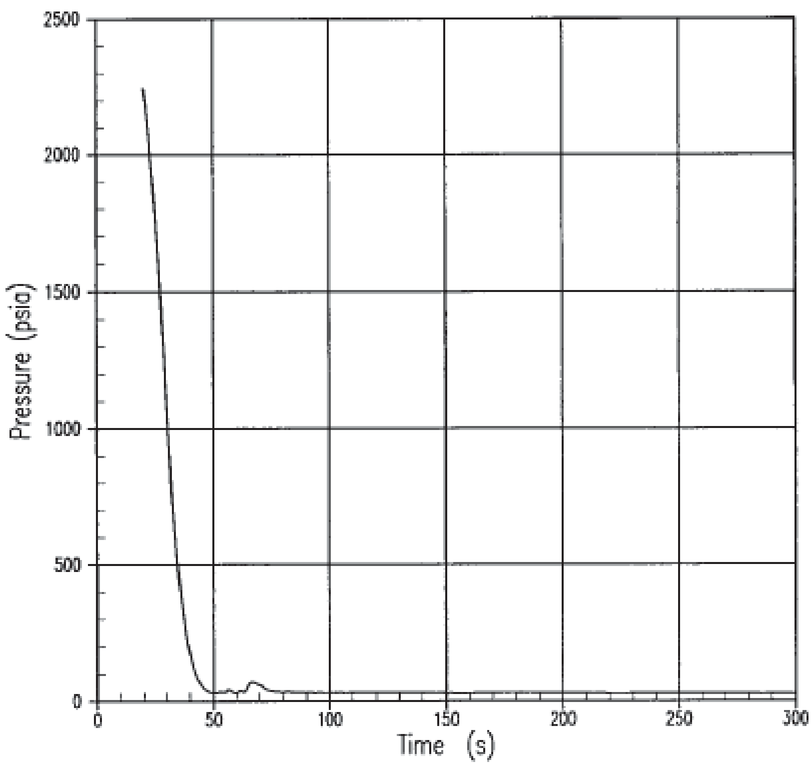
Revision 18 October 2008



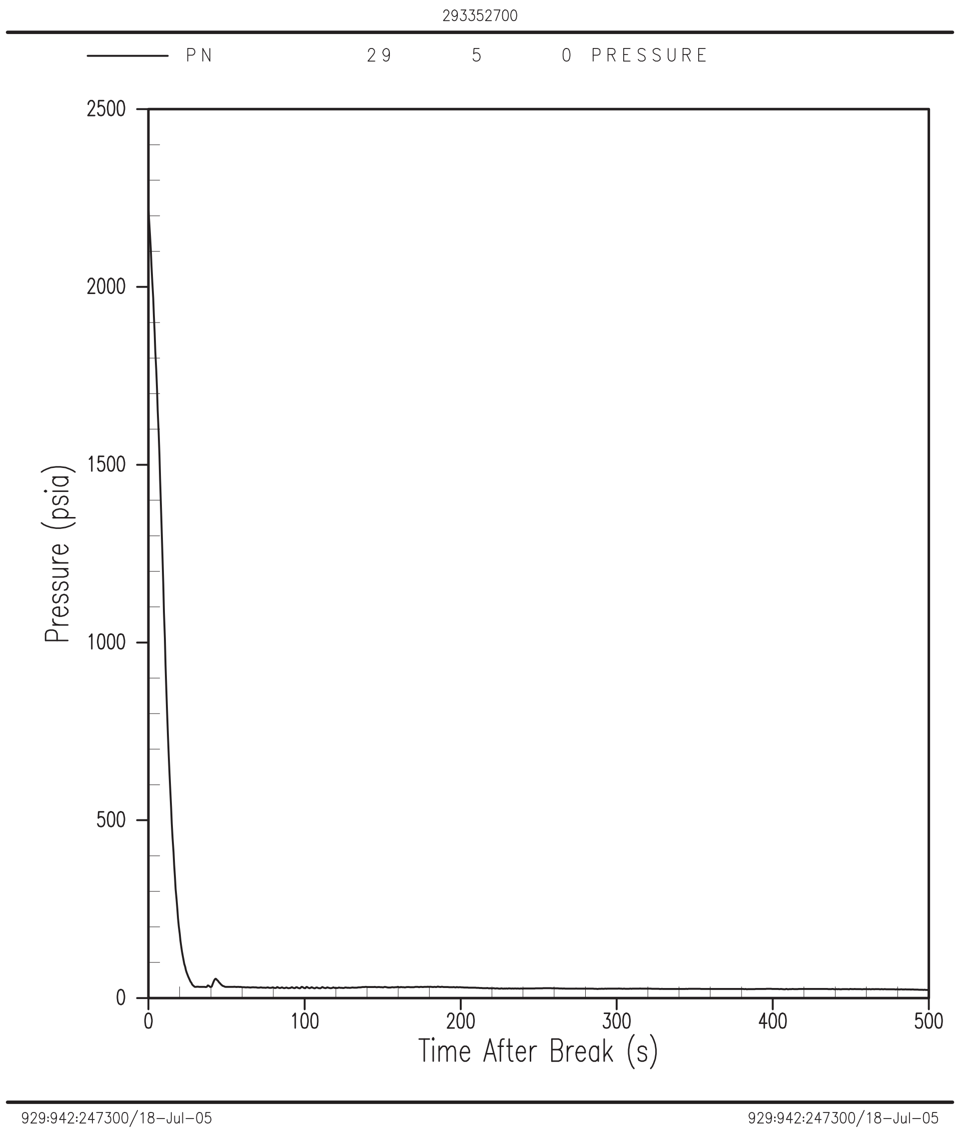
FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-5A REFERENCE TRANSIENT HOT ASSEMBLY/TOP OF CORE VAPOR FLOW



FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 15.4.1-5B LIMITING PCT CASE HOT ASSEMBLY/TOP OF CORE VAPOR FLOW



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-6A REFERENCE TRANSIENT PRESSURIZER PRESSURE

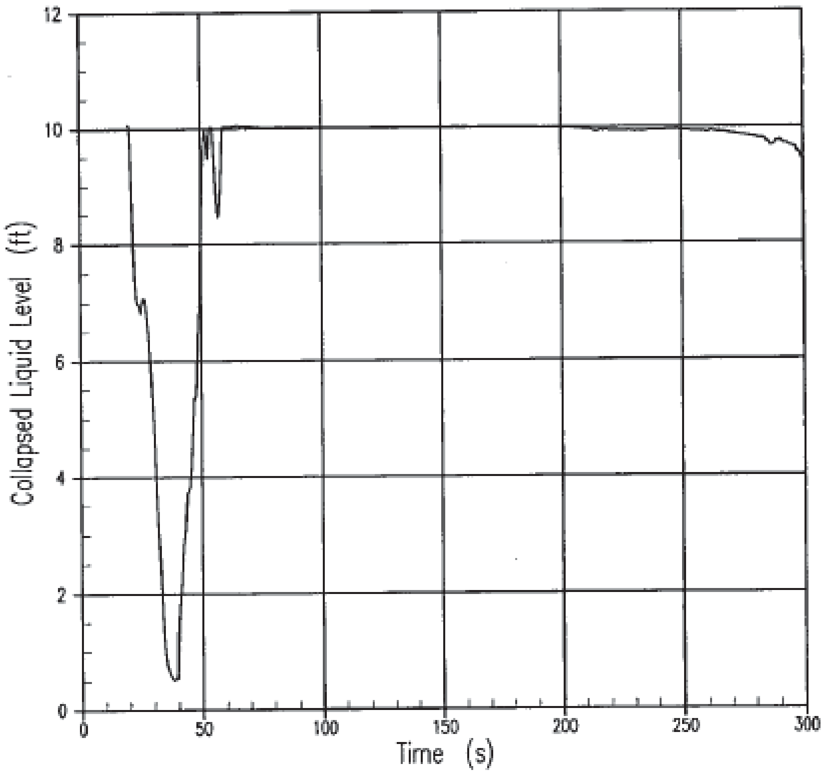


FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-6B

LIMITING PCT CASE
PRESSURIZER PRESSURE



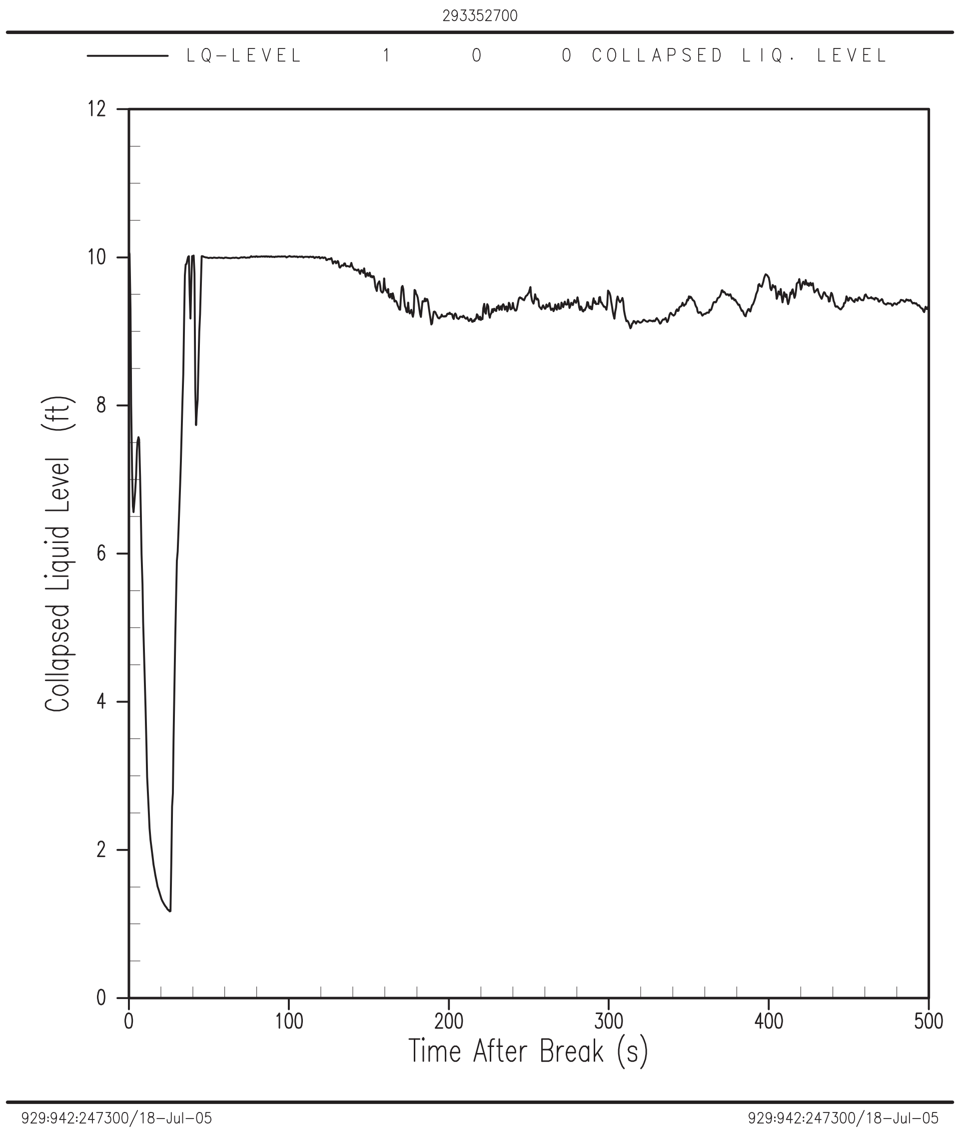
FSAR UPDATE

UNIT 1
DIABLO CANYON SITE

FIGURE 15.4.1-7A

REFERENCE TRANSIENT
LOWER PLENUM COLLAPSED
LIQUID LEVEL

Revision 18 October 2008

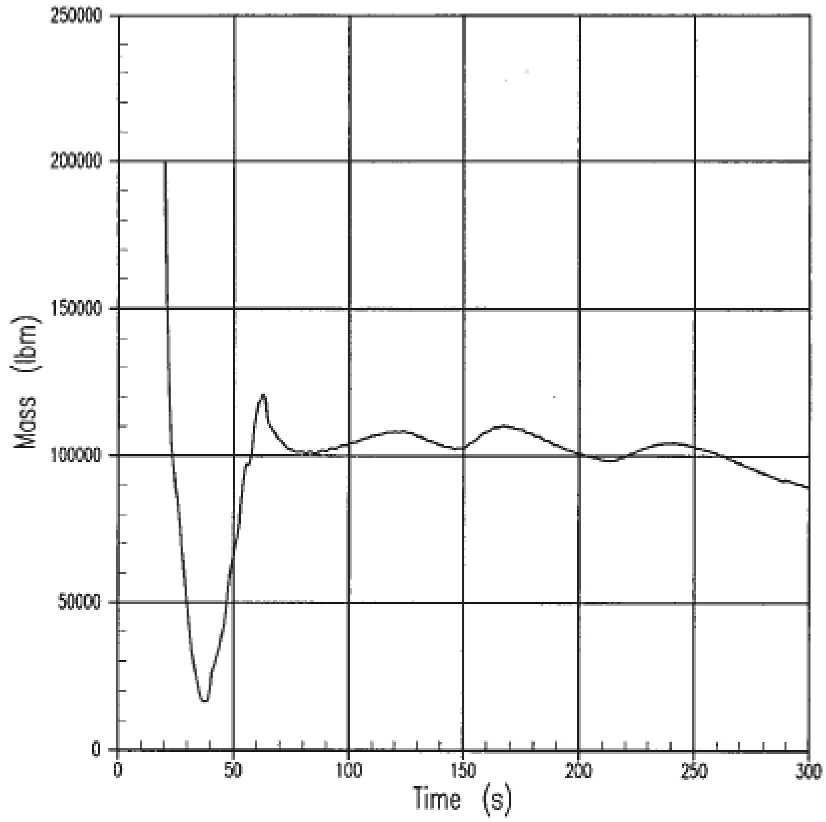


FSAR UPDATE

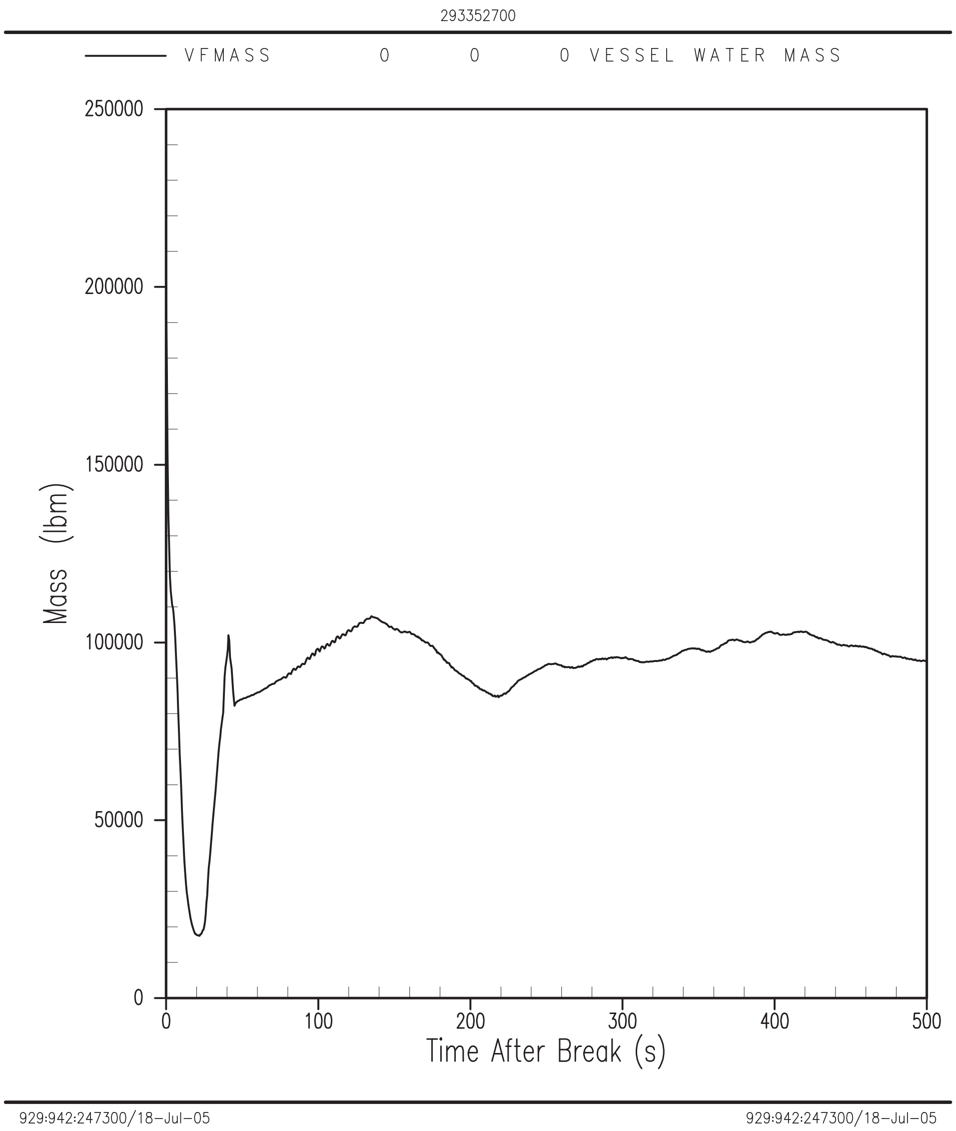
UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-7B

LIMITING PCT CASE
LOWER PLENUM COLLAPSED
LIQUID LEVEL



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-8A REFERENCE TRANSIENT VESSEL WATER MASS



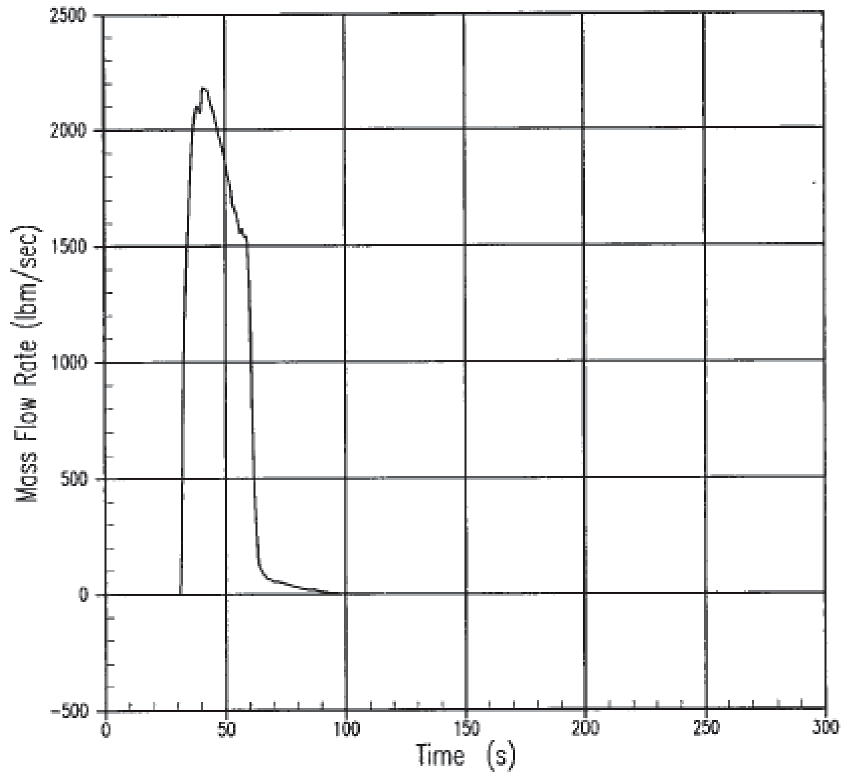
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

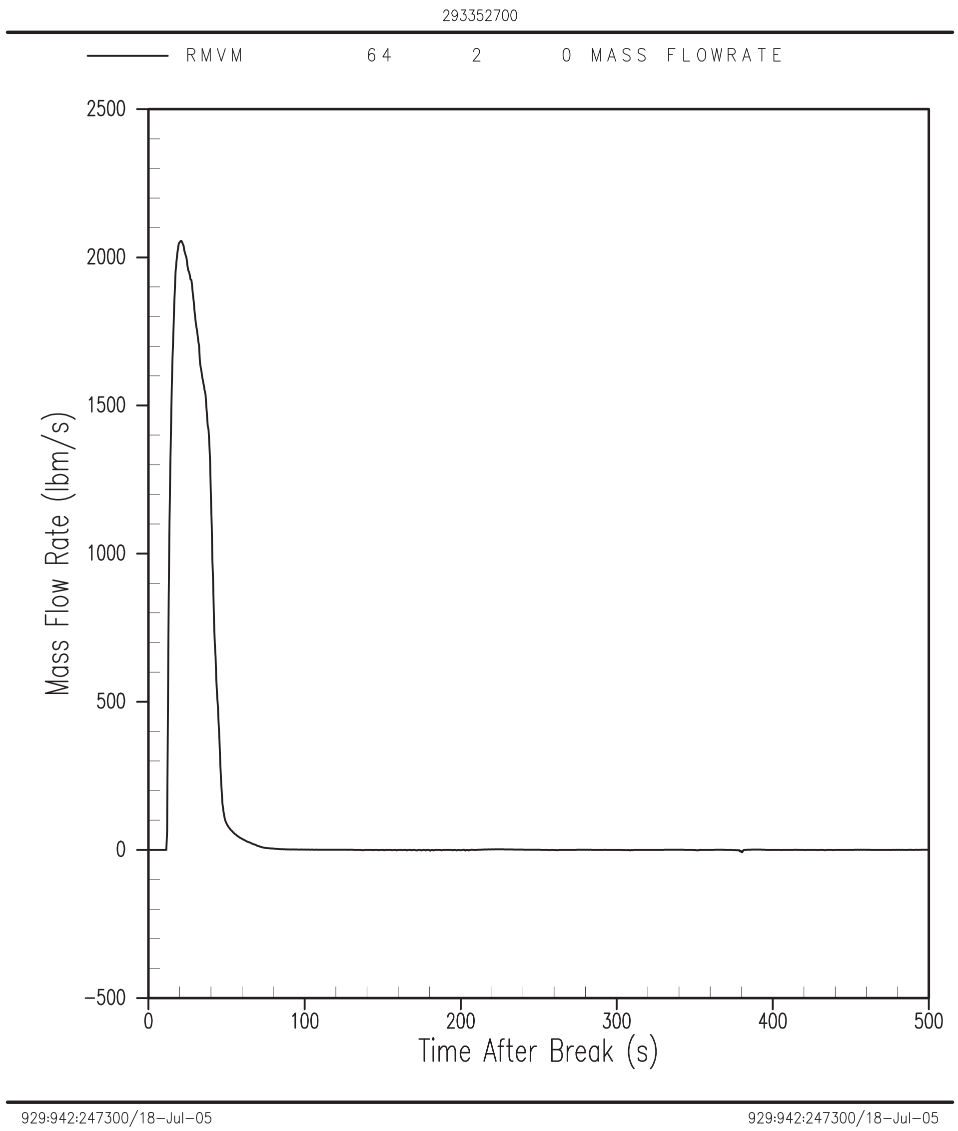
FIGURE 15.4.1-8B

LIMITING PCT CASE
VESSEL FLUID MASS

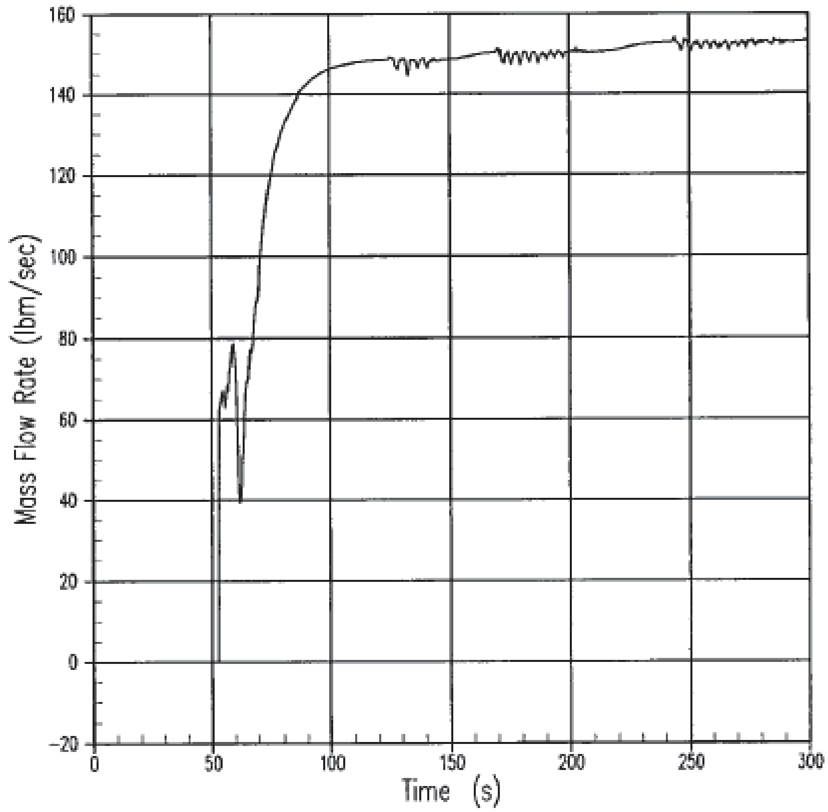
Revision 18 October 2008



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-9A REFERENCE TRANSIENT LOOP 1 ACCUMULATOR FLOW

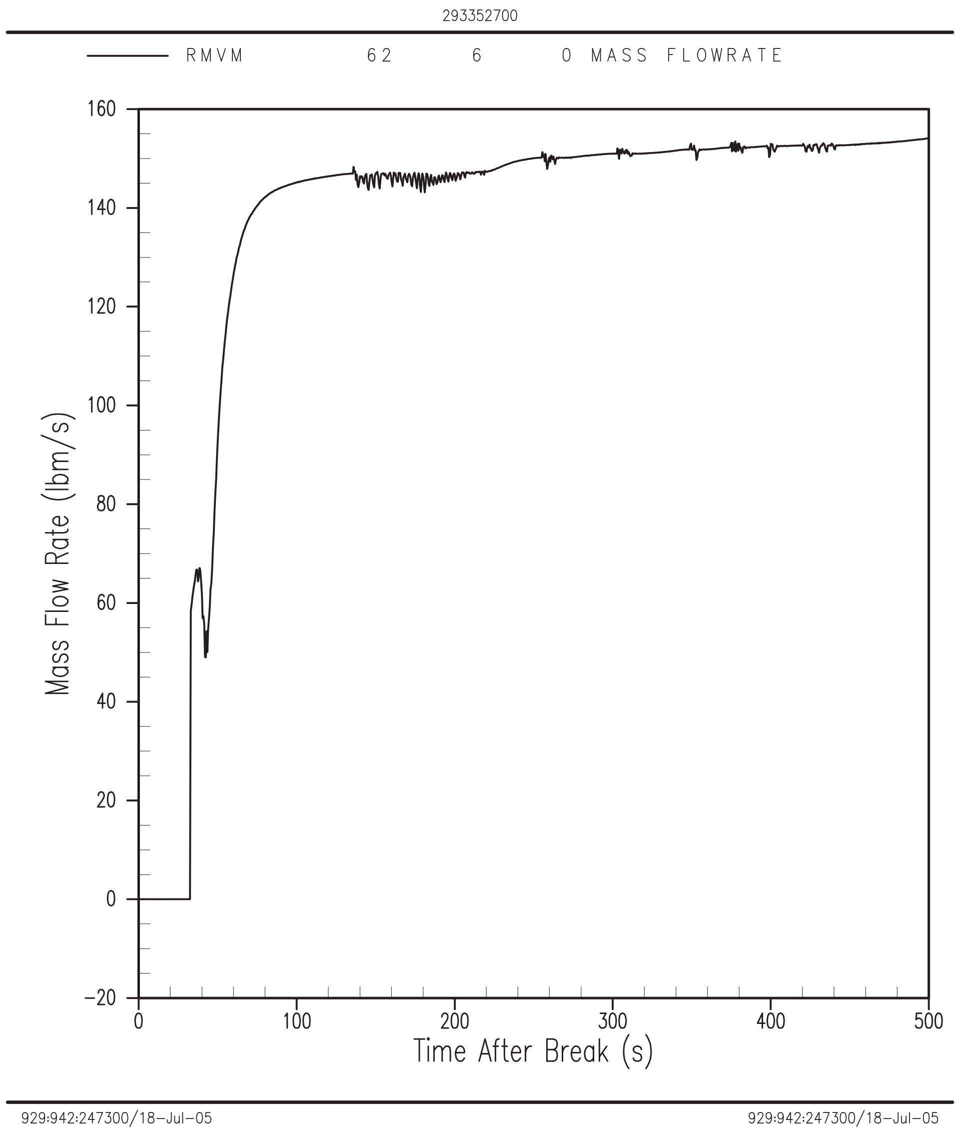


FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 15.4.1-9B LIMITING PCT CASE LOOP 1 ACCUMULATOR FLOW



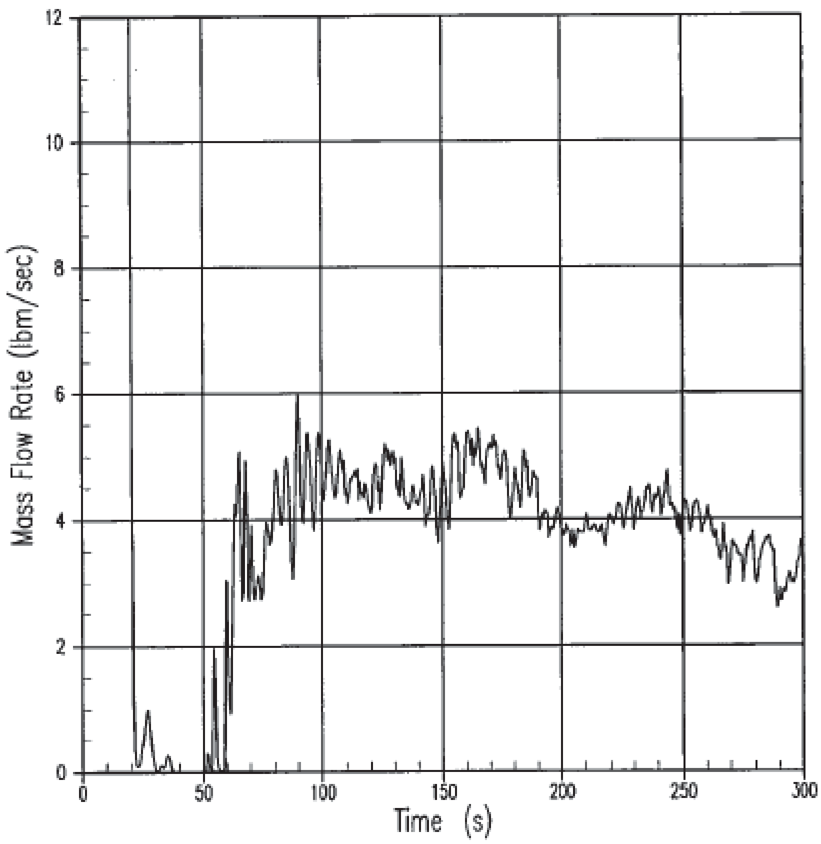
FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-10A REFERENCE TRANSIENT LOOP 1 SAFETY INJECTION FLOW

Revision 18 October 2008

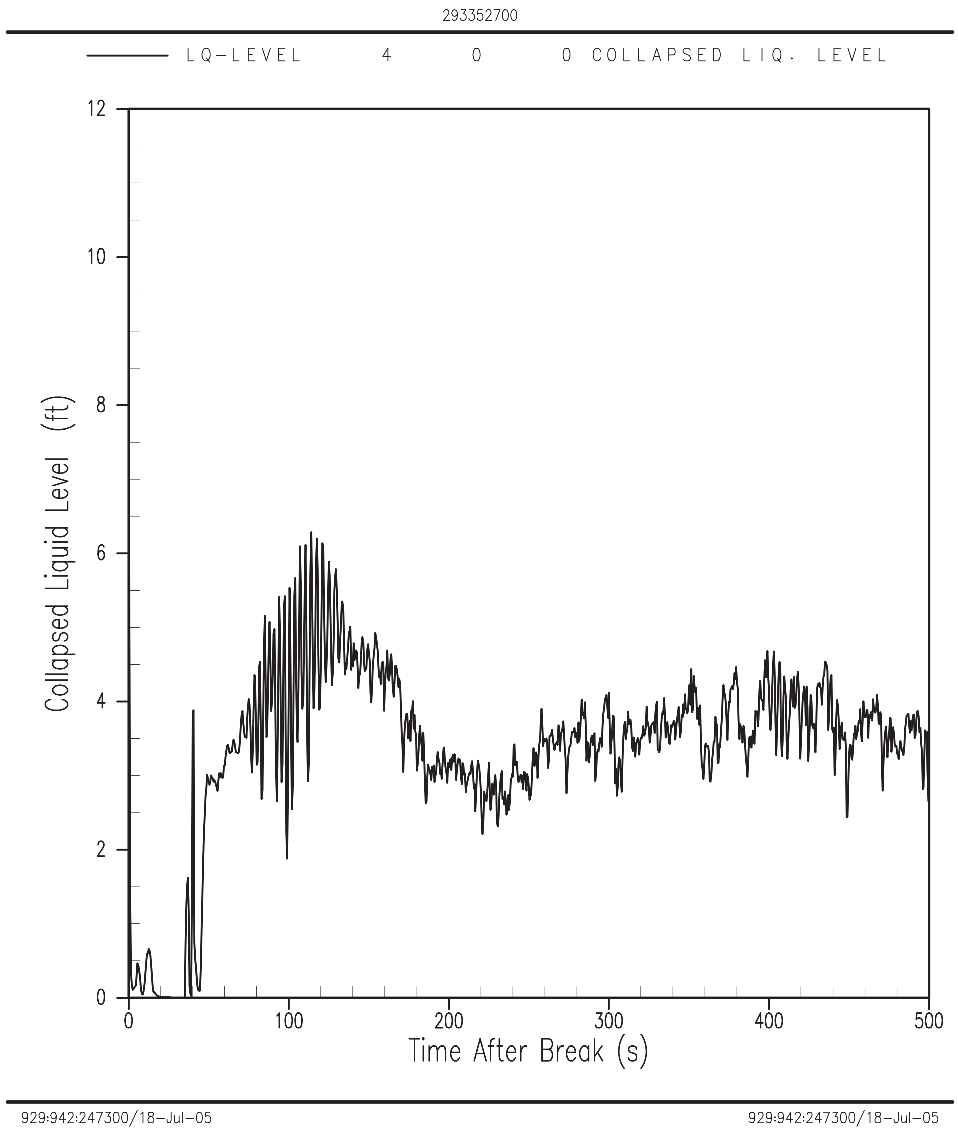


FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 15.4.1-10B LIMITING PCT CASE LOOP 1 SAFETY INJECTION FLOW

Revision 18 October 2008



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-11A REFERENCE TRANSIENT CORE AVERAGE CHANNEL COLLAPSED LIQUID LEVEL



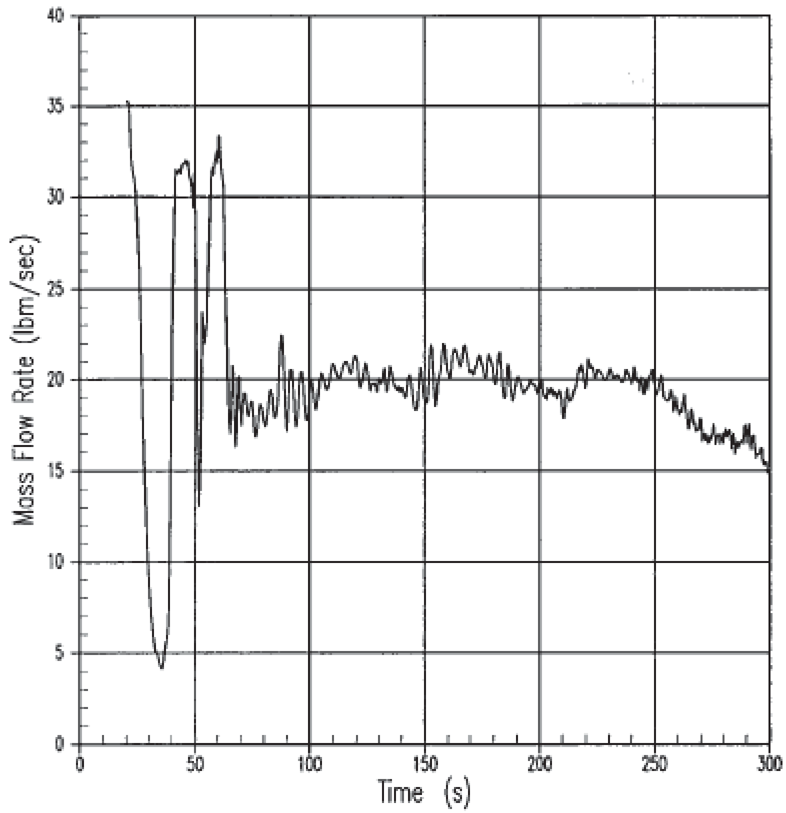
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

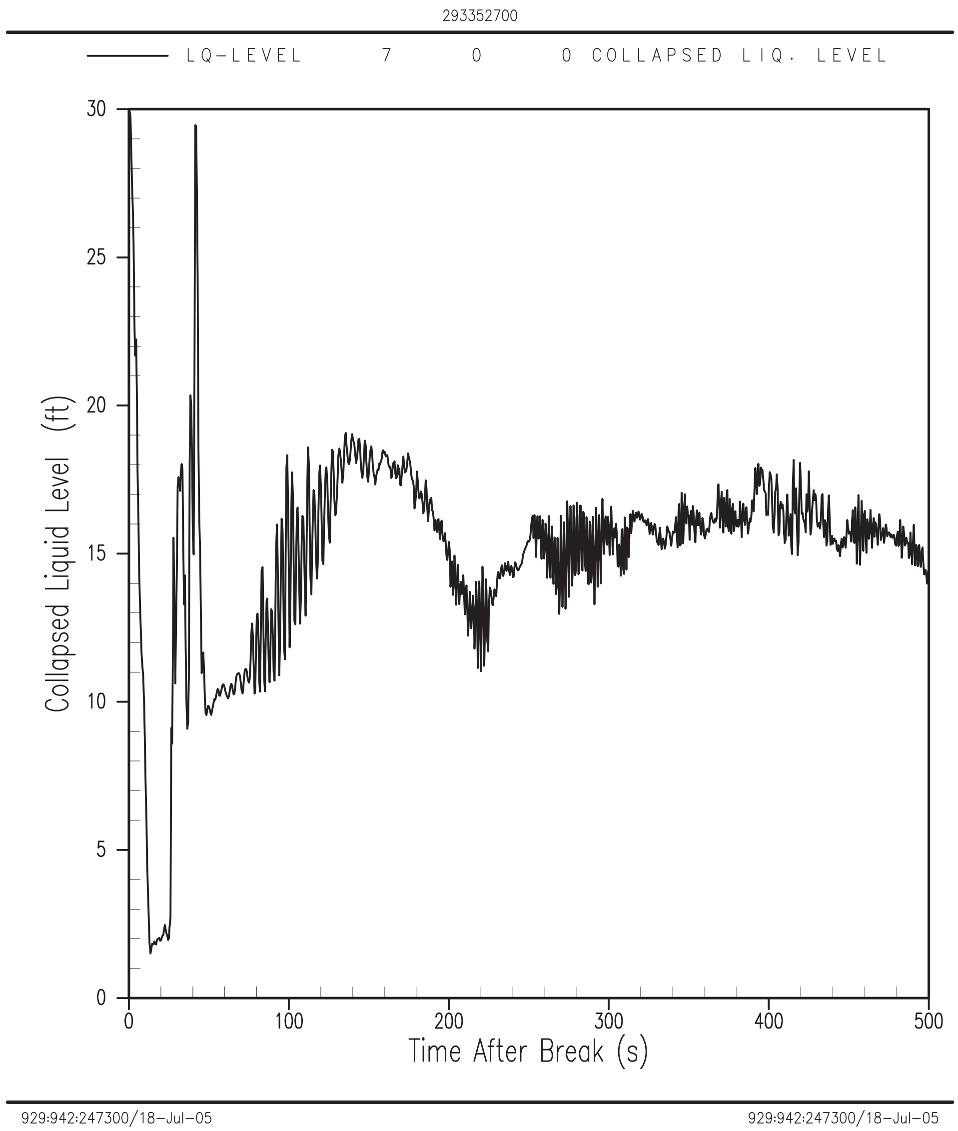
FIGURE 15.4.1-11B

LIMITING PCT CASE
CORE AVERAGE CHANNEL
COLLAPSED LIQUID LEVEL

Revision 18 October 2008

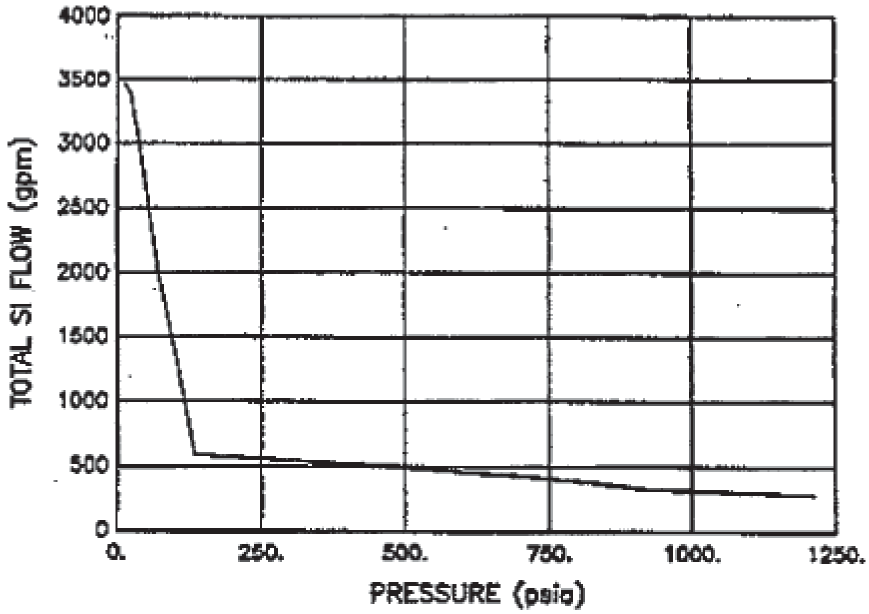


FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-12A REFERENCE TRANSIENT LOOP 1 DOWNCOMER COLLAPSED LIQUID LEVEL

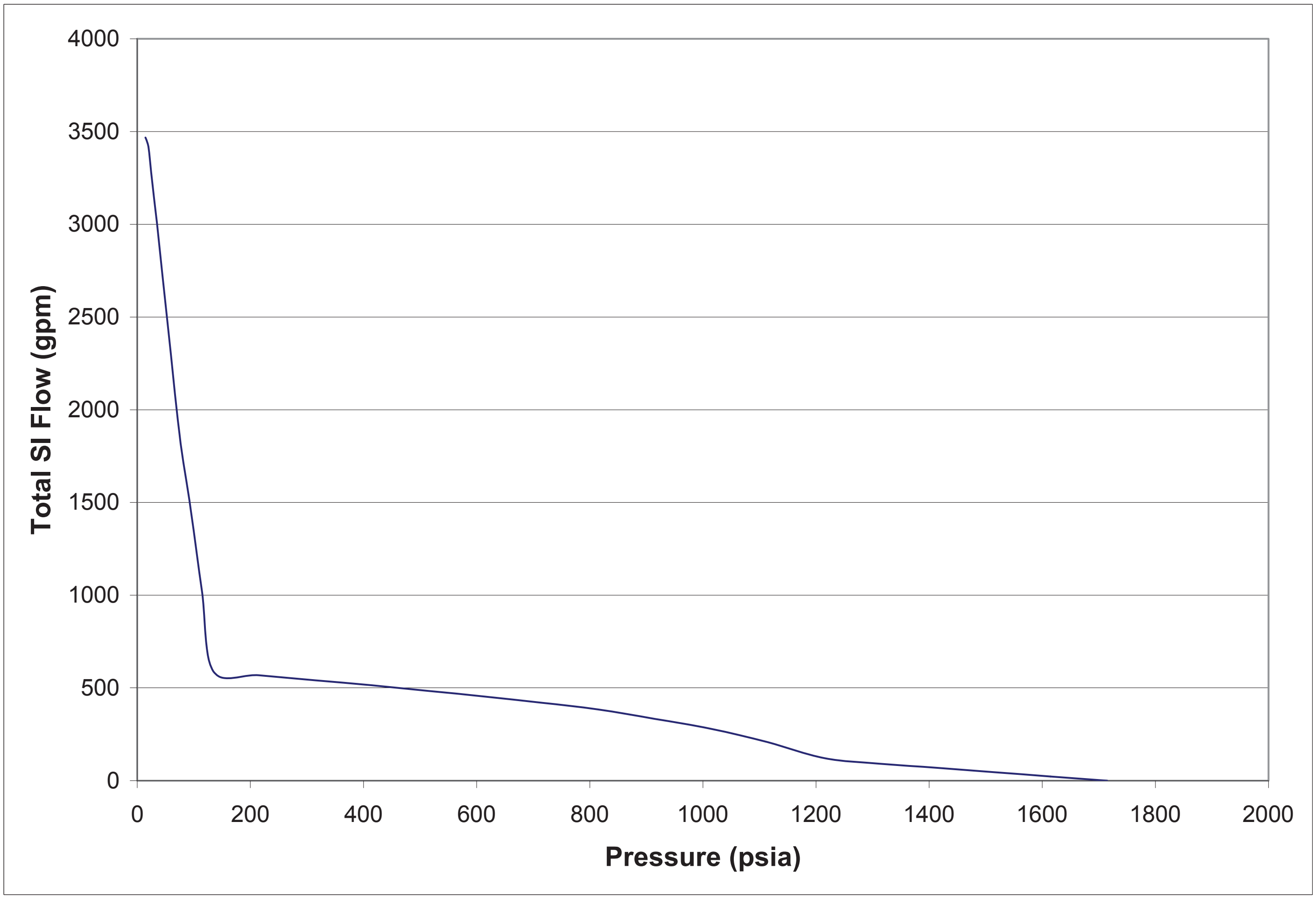


FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 15.4.1-12B LIMITING PCT CASE LOOP 1 DOWNCOMER COLLAPSED LIQUID LEVEL

Revision 18 October 2008

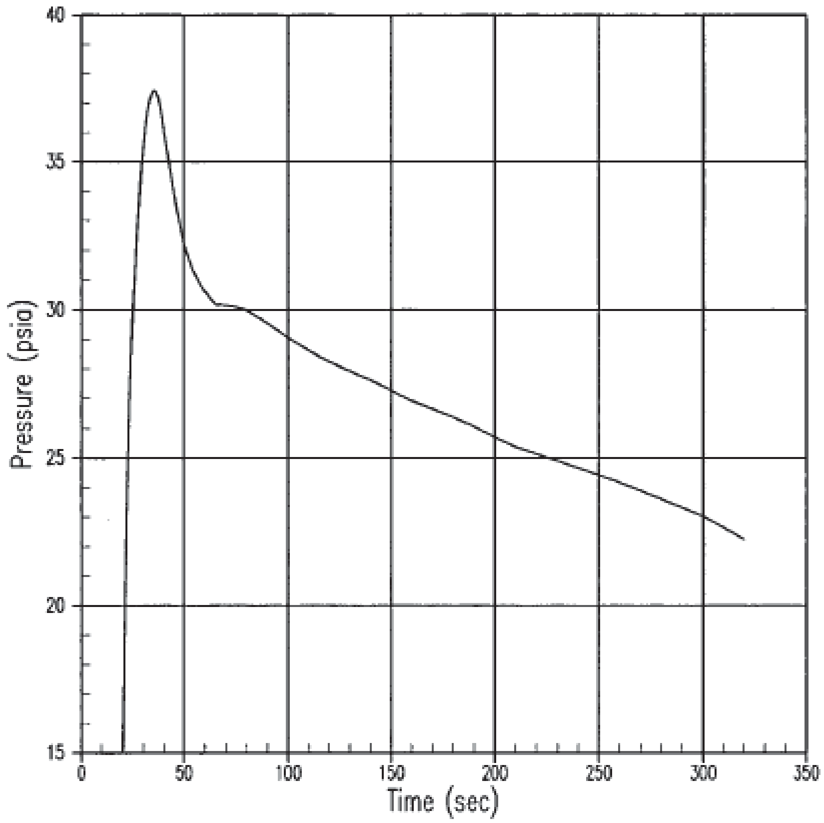


FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-13A TOTAL ECCS FLOW (3 LINES INJECTING)



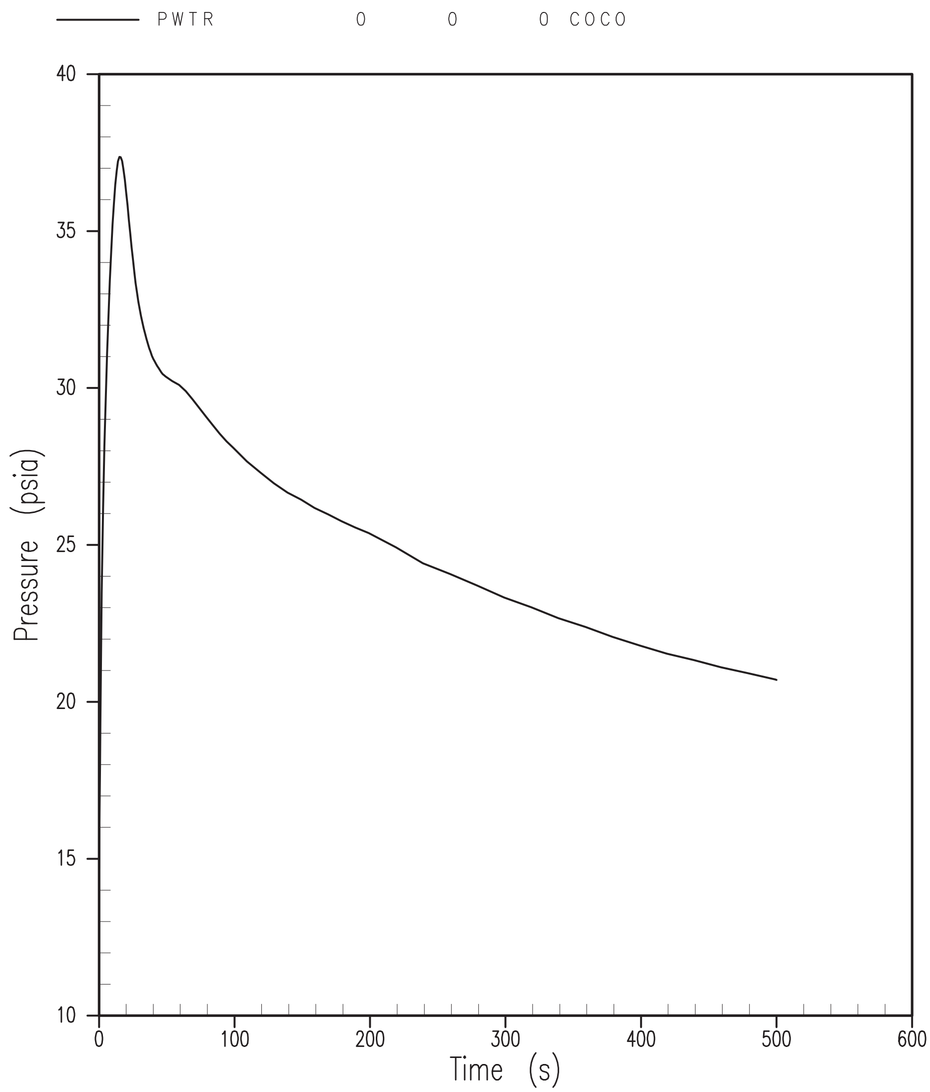
FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 15.4.1-13B TOTAL ECCS FLOW (3 LINES INJECTING)

Revision 18 October 2008



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-14A REFERENCE TRANSIENT PRESSURE TRANSIENT

334749375



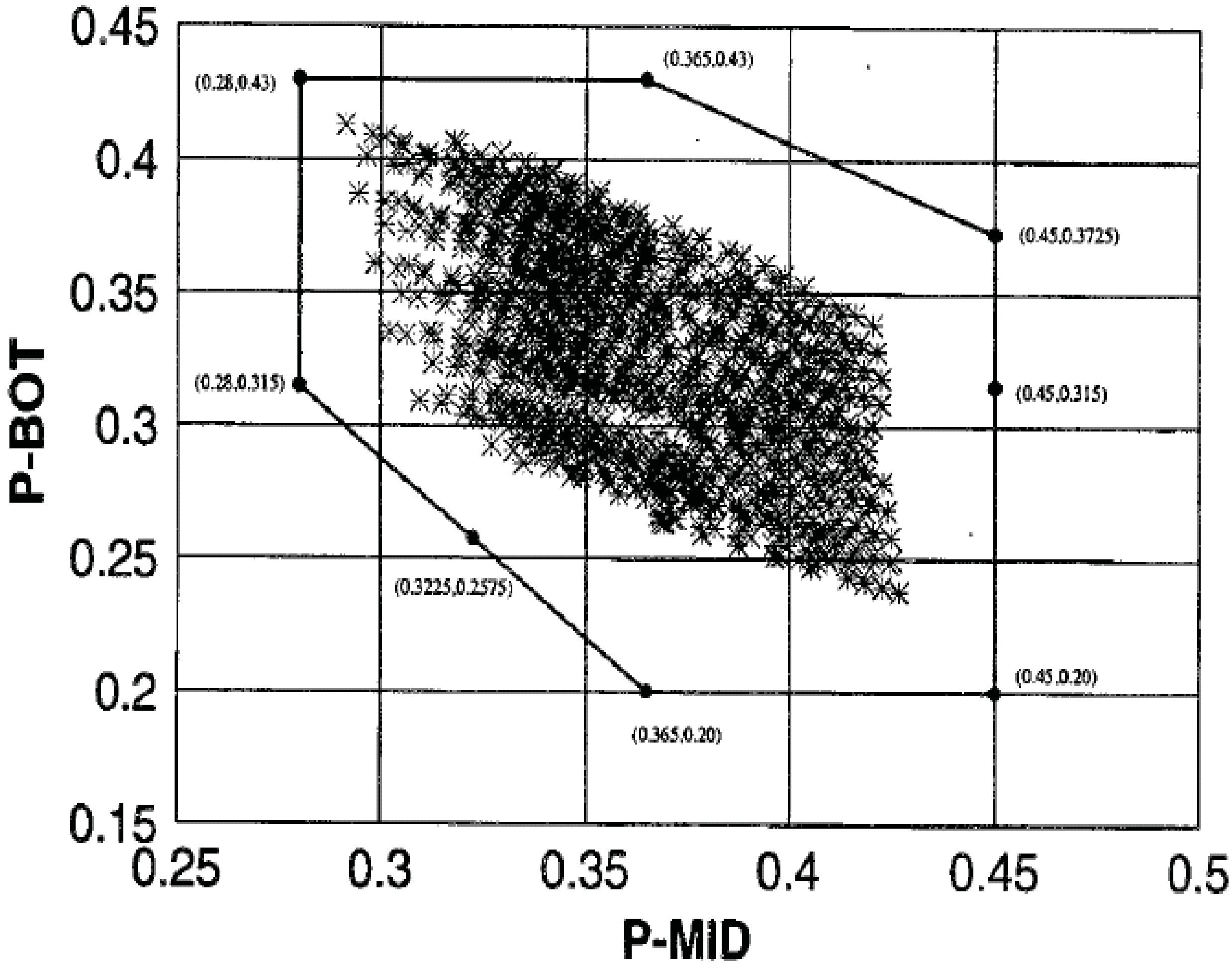
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

FIGURE 15.4.1-14B

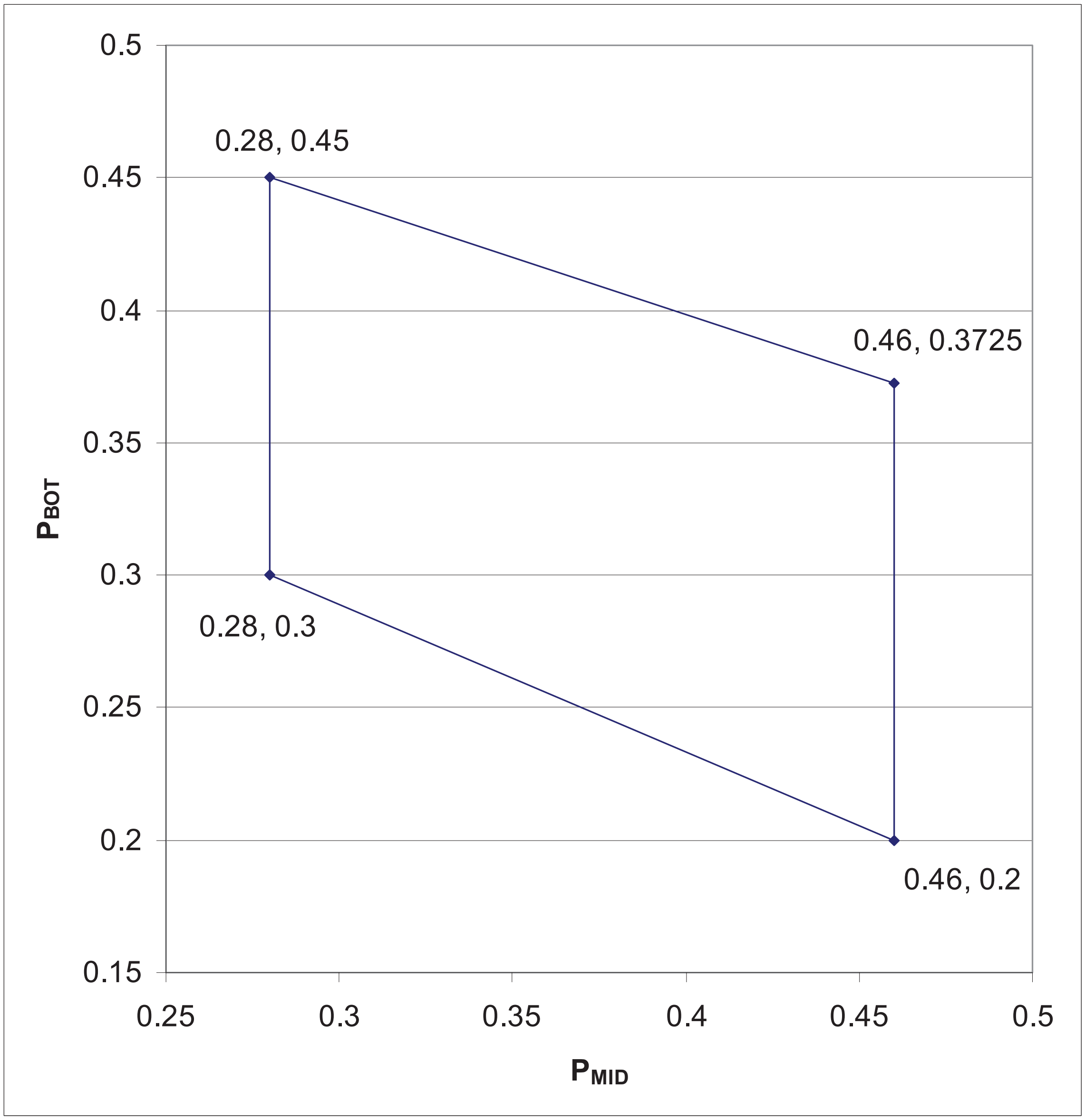
LOWER BOUND COCO CONTAINMENT PRESSURE TRANSIENT

Revision 18 October 2008

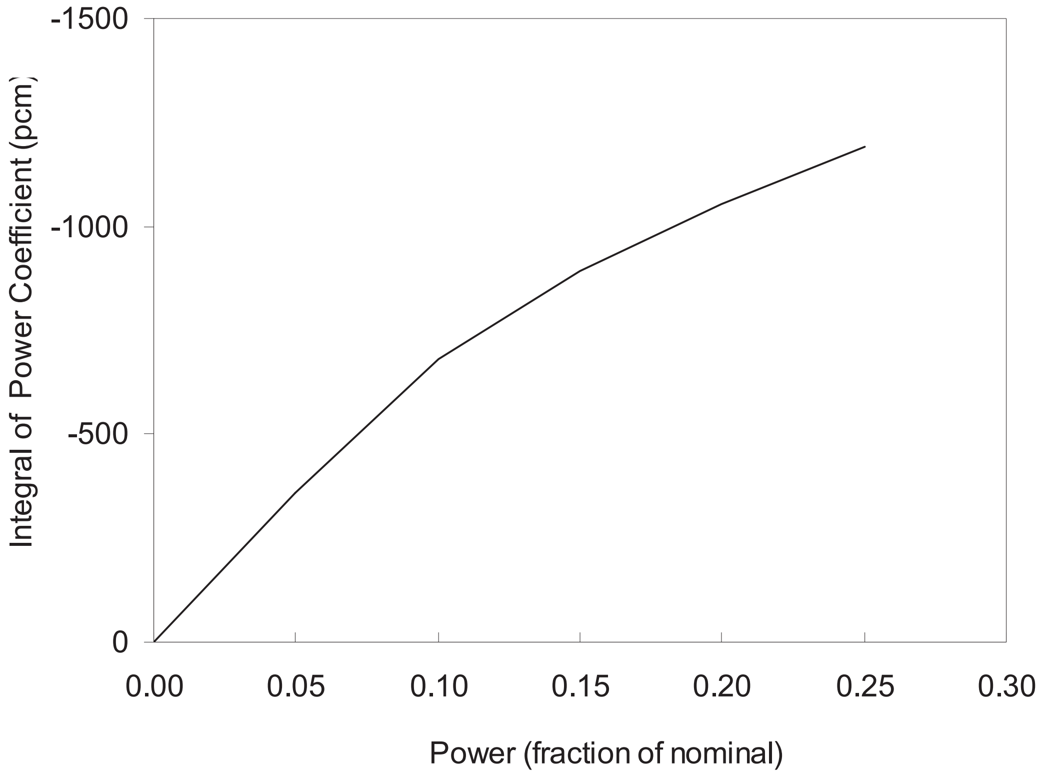


FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
FIGURE 15.4.1-15A AXIAL POWER DISTRIBUTION LIMITS

Revision 21 September 2013



FSAR UPDATE
UNIT 2 **DIABLO CANYON SITE**
FIGURE 15.4.1-15B **AXIAL POWER** **DISTRIBUTION LIMITS**



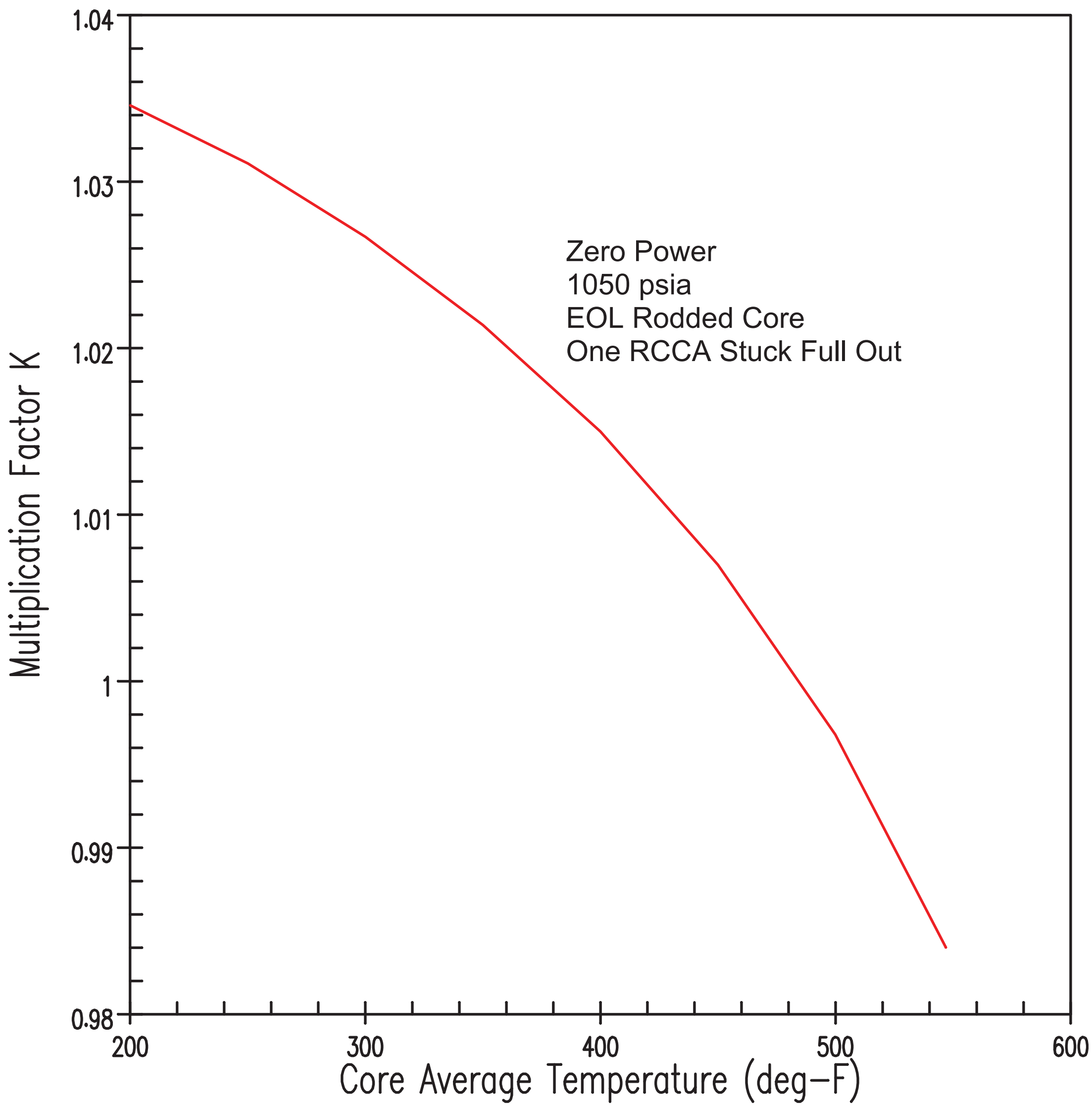
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-1

RUPTURE OF A MAIN STEAM LINE

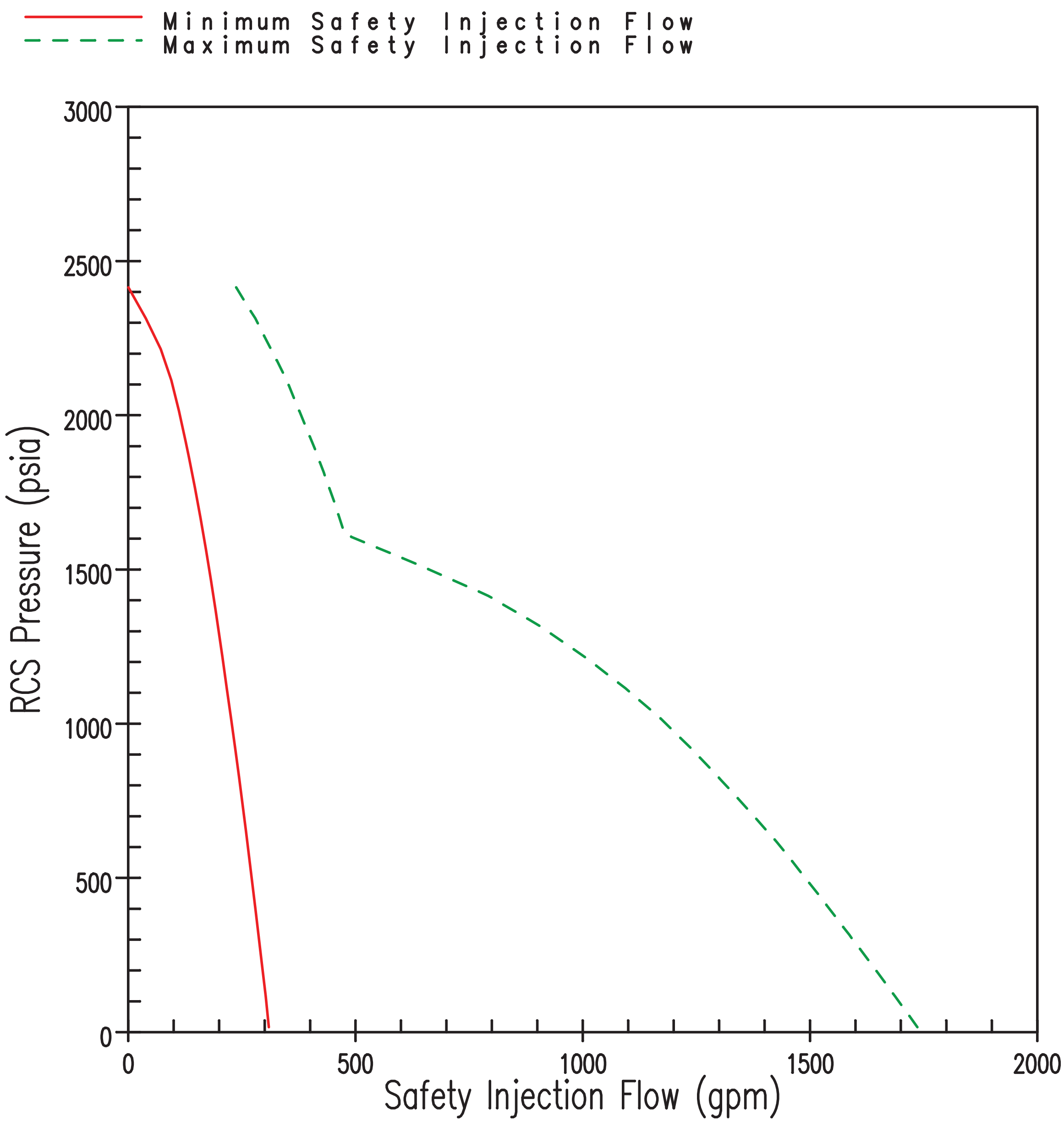
VARIATION OF REACTIVITY WITH POWER AT CONSTANT CORE AVERAGE TEMPERATURE



FSAR UPDATE

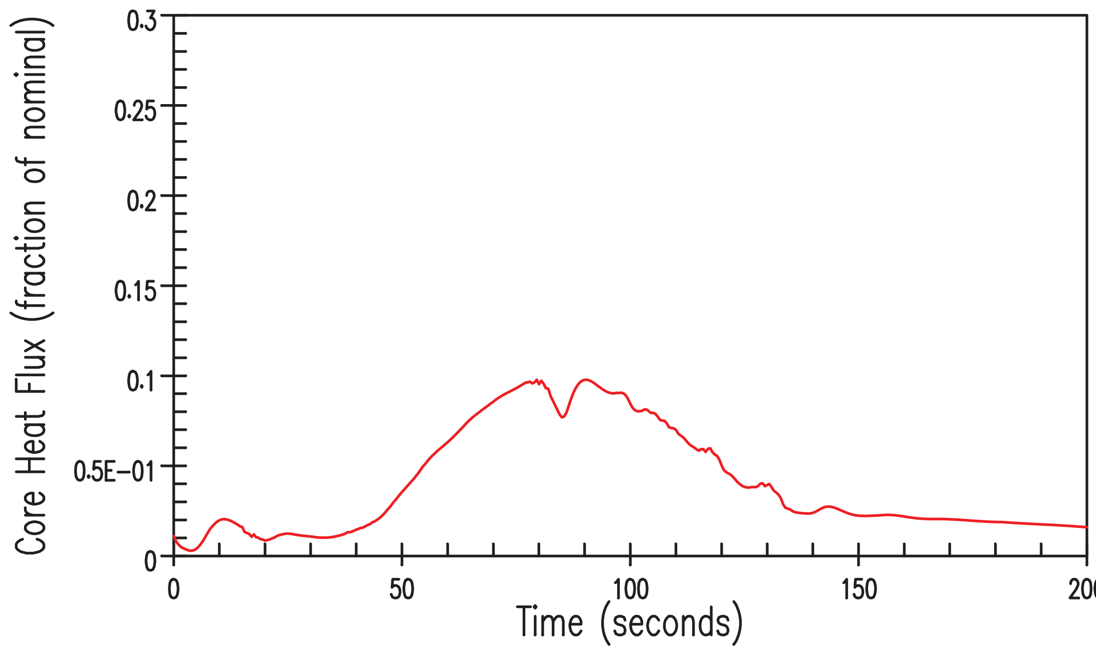
UNITS 1 AND 2
DIABLO CANYON SITE

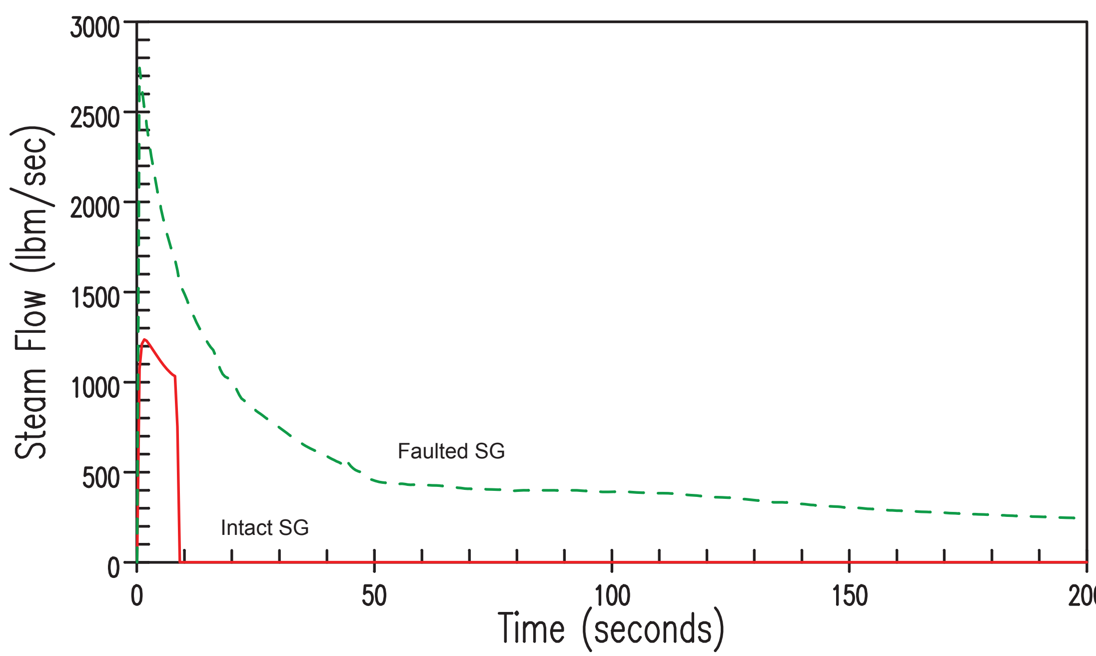
FIGURE 15.4.2-2
RUPTURE OF A MAIN STEAM LINE
VARIATION OF K_{EFF} WITH
CORE AVERAGE TEMPERATURE



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-3 RUPTURE OF A MAIN STEAM LINE SAFETY INJECTION CURVE

Revision 21 September 2013





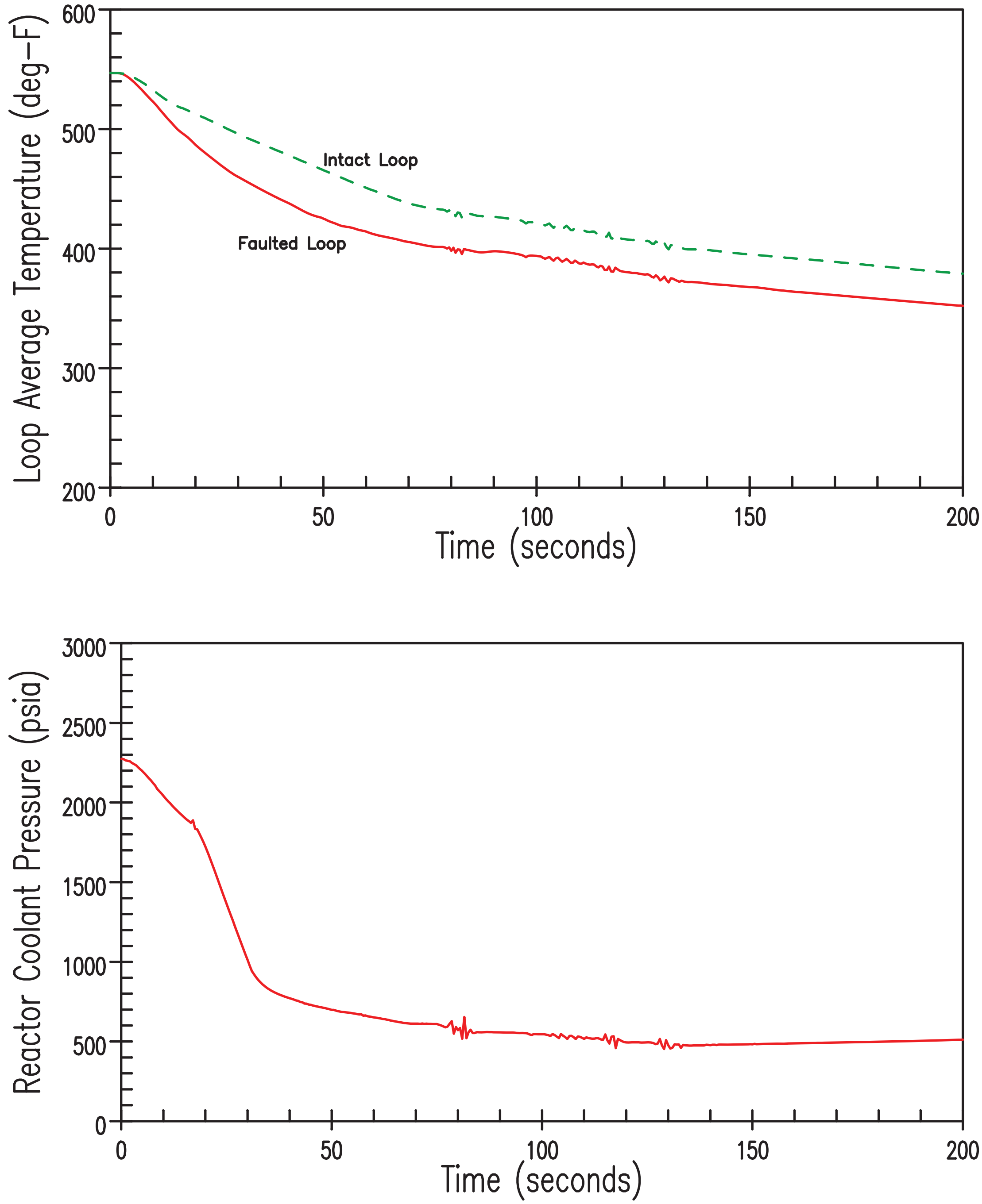
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-4

RUPTURE OF A MAIN STEAM LINE WITH OFFSITE POWER AVAILABLE

CORE HEAT FLUX AND STEAM FLOW TRANSIENTS



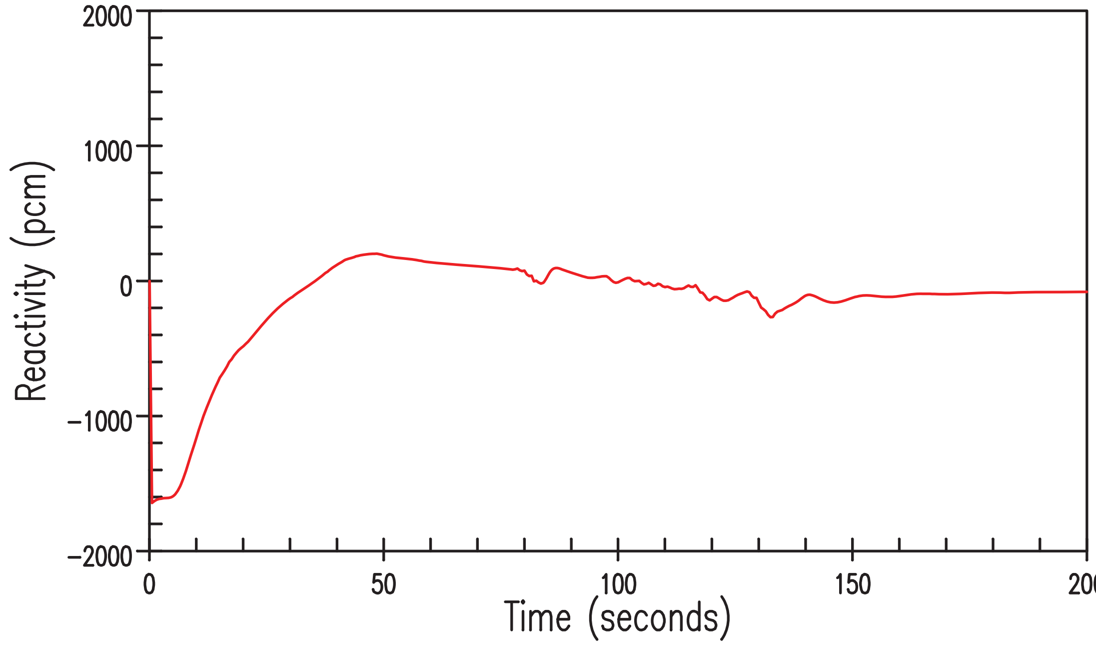
FSAR UPDATE

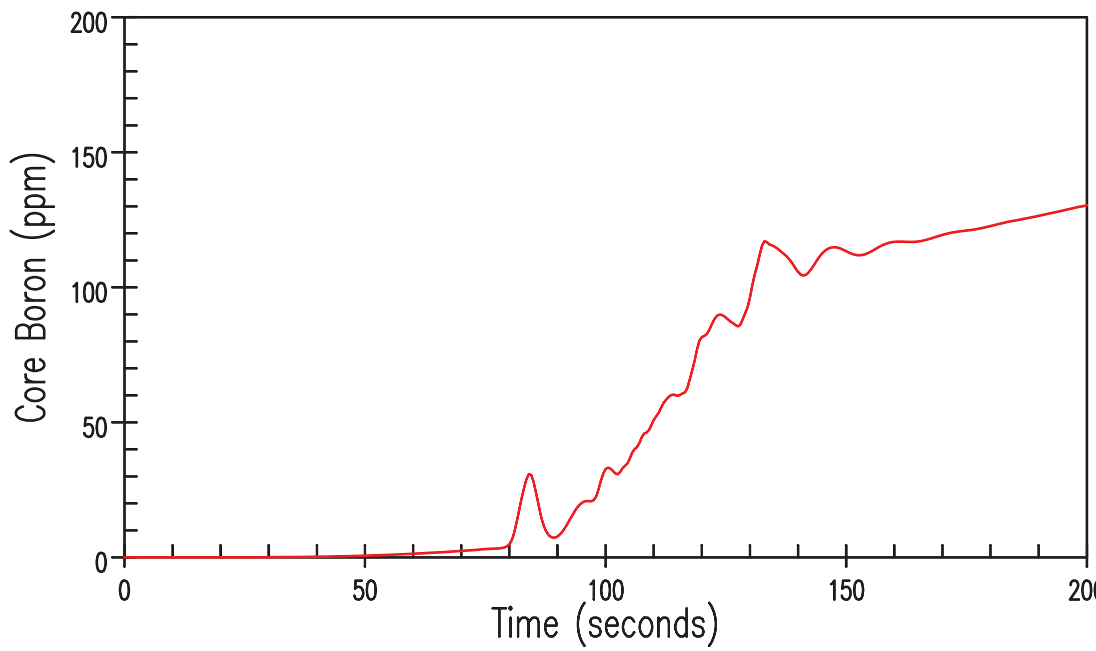
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-5

RUPTURE OF A MAIN STEAM LINE WITH OFFSITE POWER AVAILABLE

LOOP AVERAGE TEMPERATURE AND REACTOR COOLANT PRESSURE TRANSIENTS





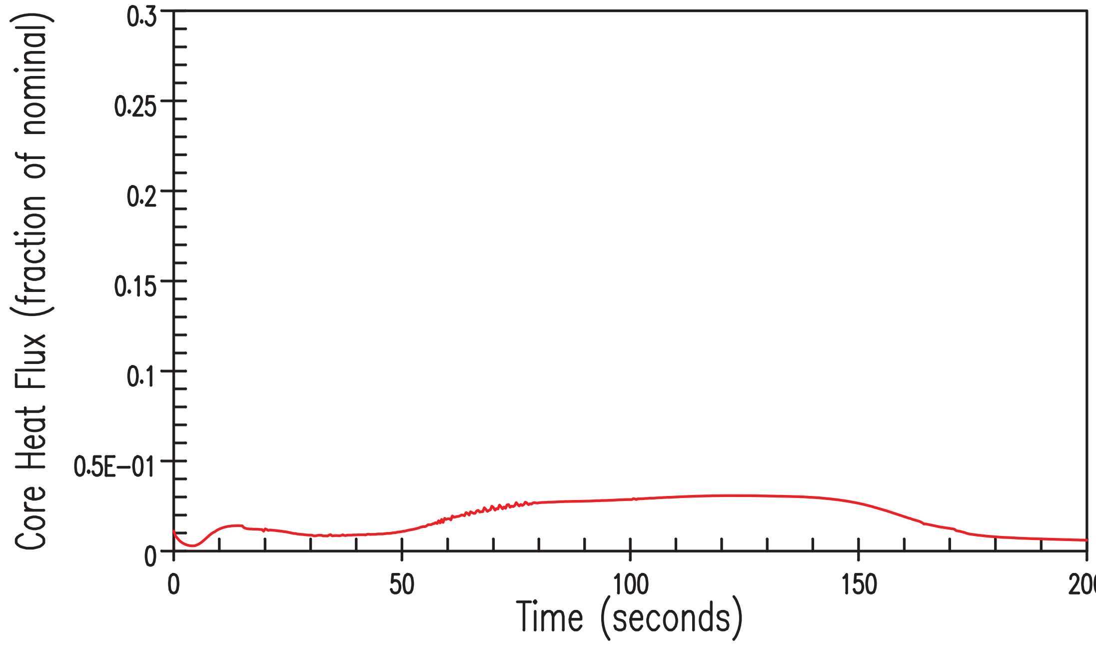
FSAR UPDATE

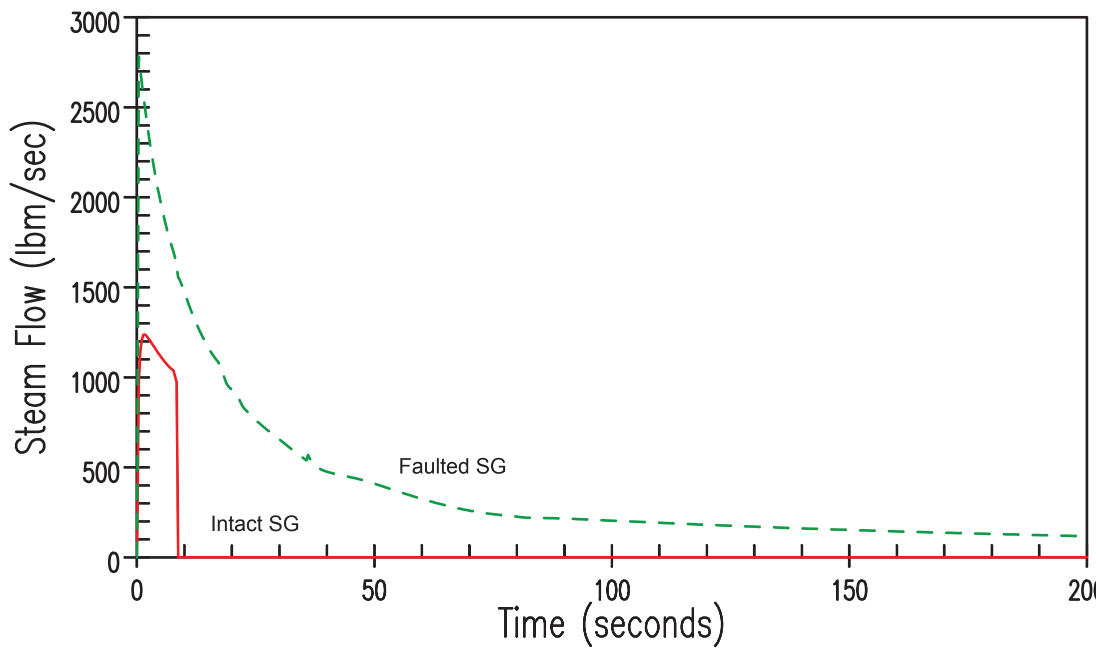
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-6

RUPTURE OF A MAIN STEAM LINE
WITH OFFSITE POWER AVAILABLE

REACTIVITY AND CORE BORON TRANSIENTS





FSAR UPDATE

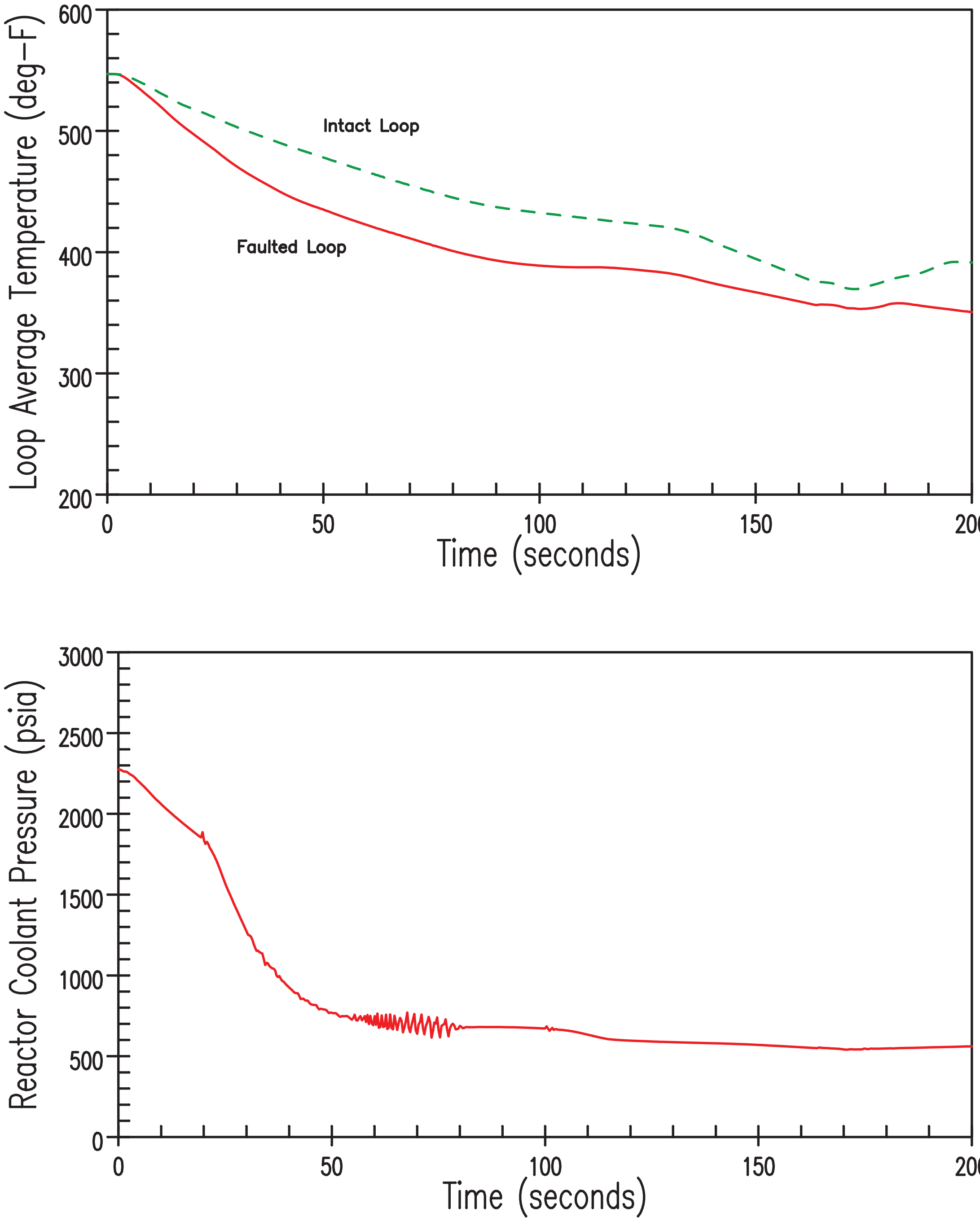
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-7

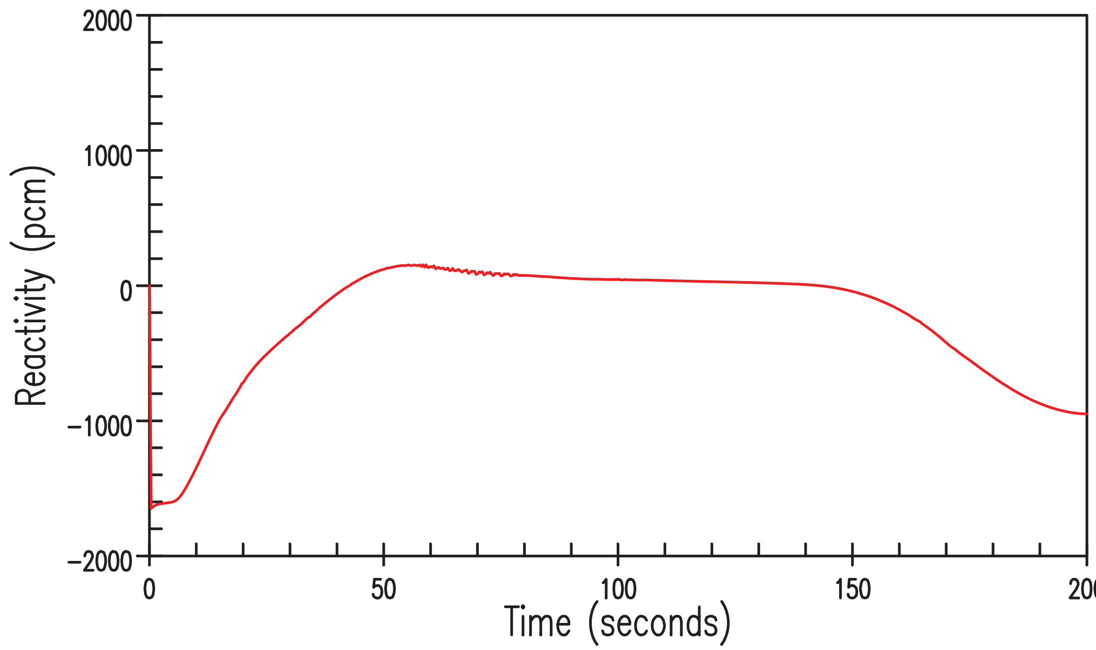
RUPTURE OF A MAIN STEAM LINE WITHOUT OFFSITE POWER AVAILABLE

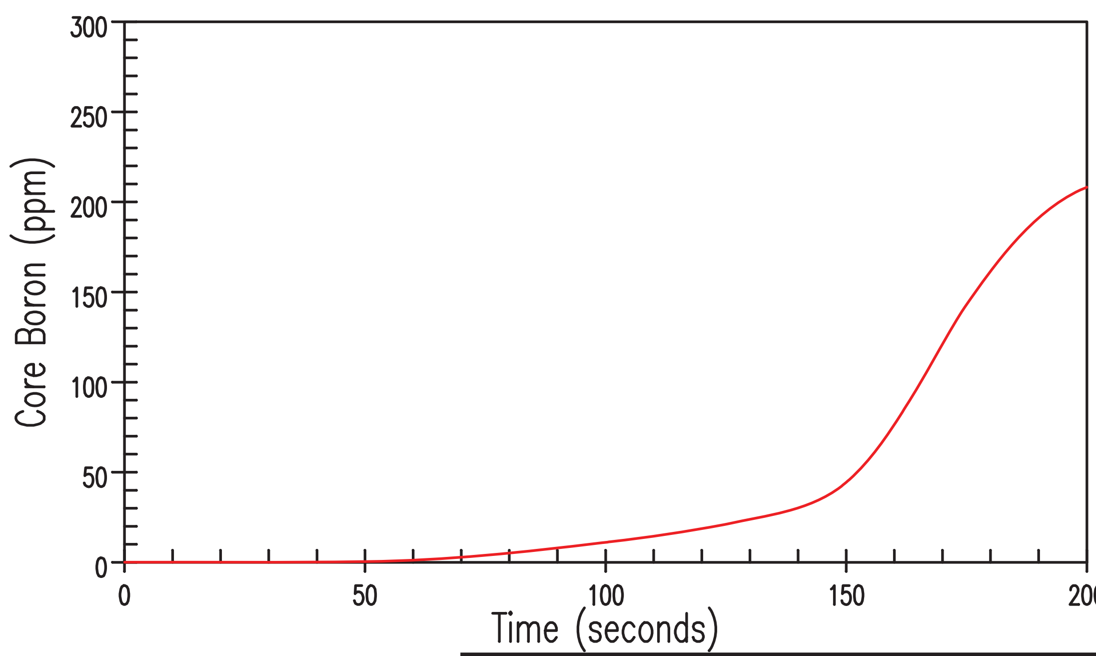
CORE HEAT FLUX AND STEAM FLOW TRANSIENTS

Revision 24 September 2018



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-8 RUPTURE OF A MAIN STEAM LINE WITHOUT OFFSITE POWER AVAILABLE LOOP AVERAGE TEMPERATURE AND REACTOR COOLANT PRESSURE TRANSIENTS

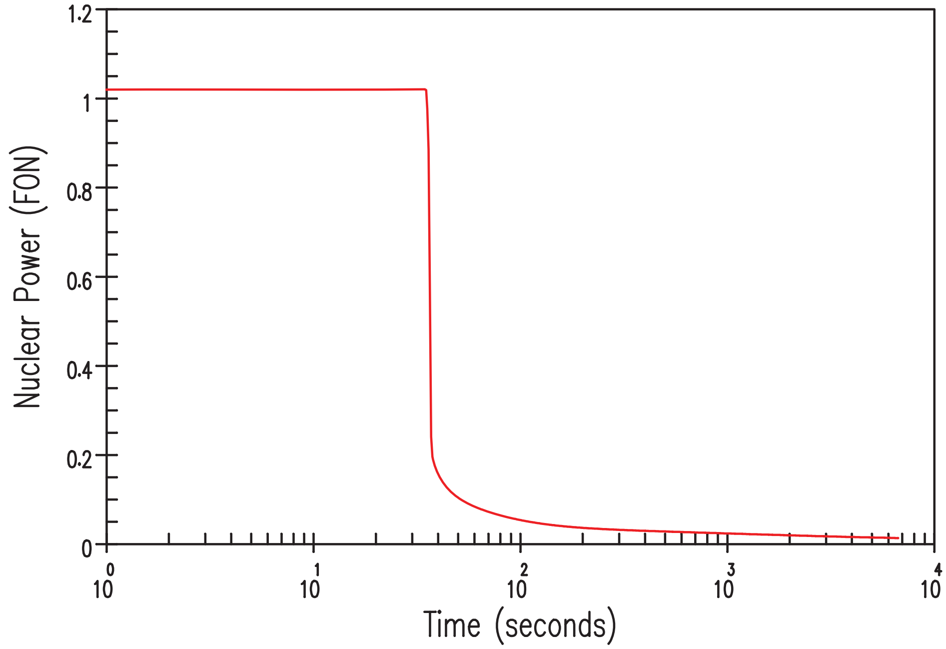


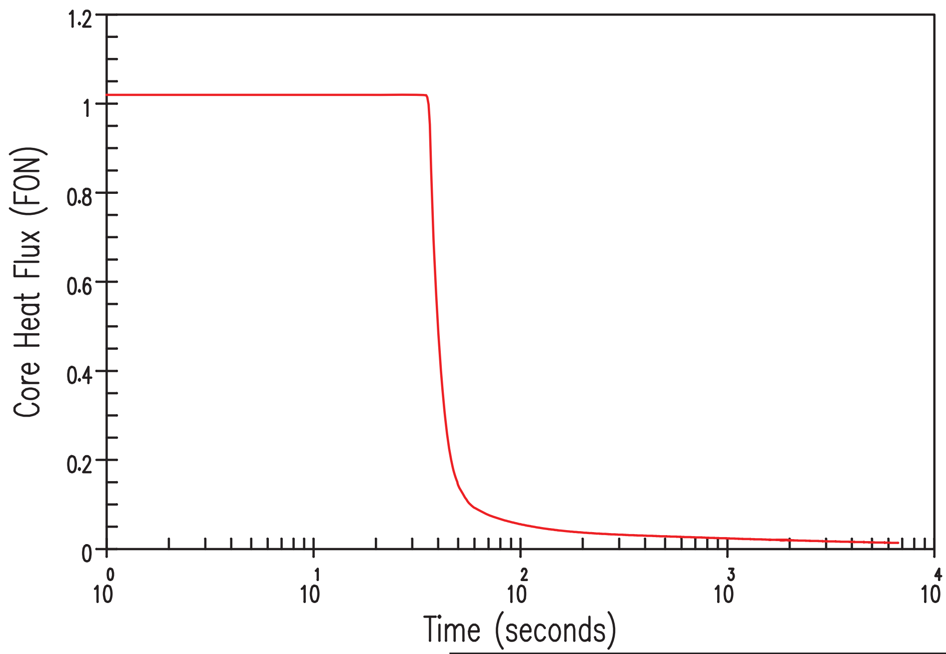


FSAR UPDATE

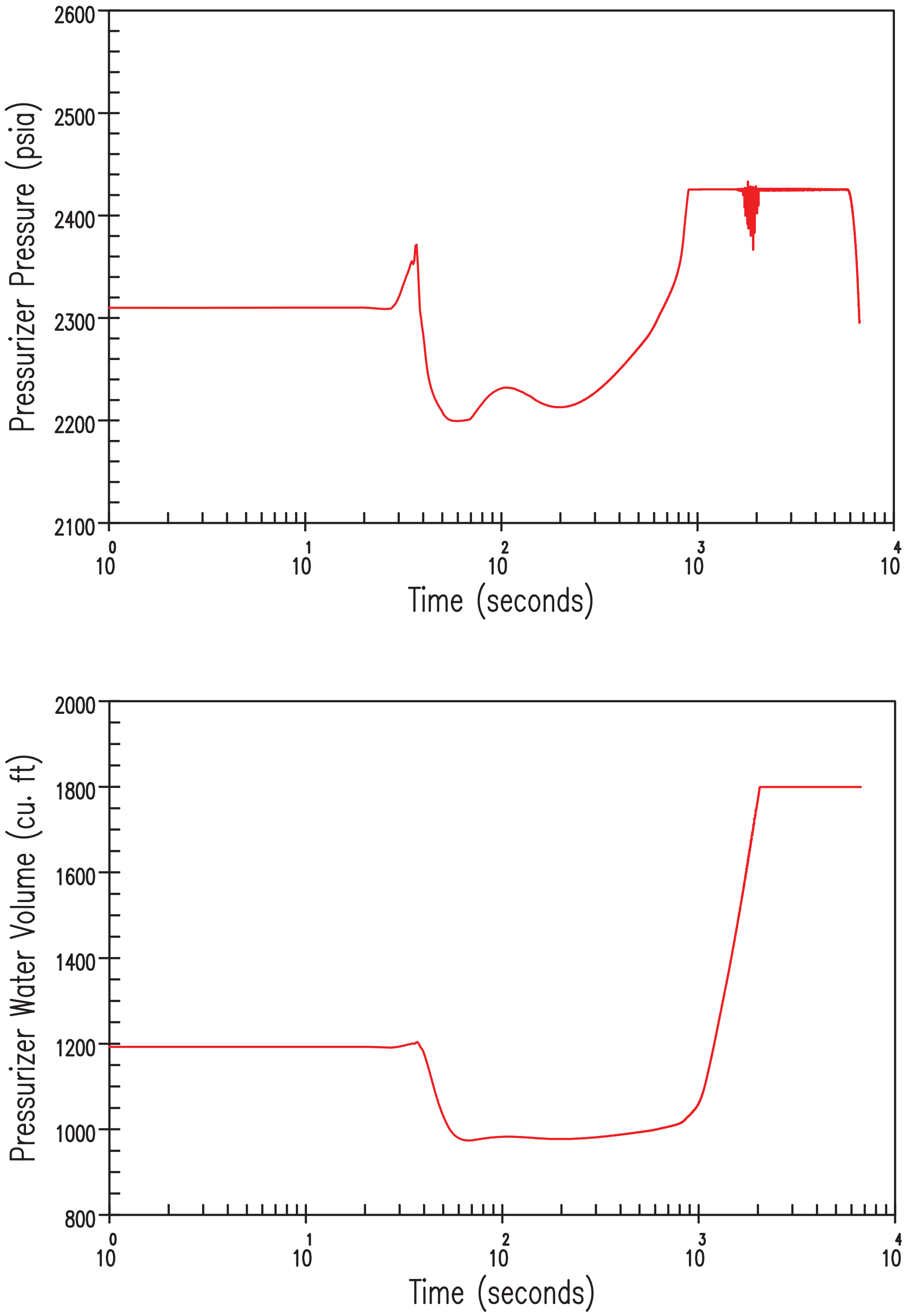
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-9
RUPTURE OF A MAIN STEAM LINE
WITHOUT OFFSITE POWER AVAILABLE
REACTIVITY AND CORE BORON TRANSIENTS





FSAR UPDATE
UNITS 1 AND 2 **DIABLO CANYON SITE**
FIGURE 15.4.2-10 **MAIN FEEDLINE RUPTURE** **WITH OFFSITE POWER AVAILABLE** **NUCLEAR POWER AND** **CORE HEAT FLUX TRANSIENTS**



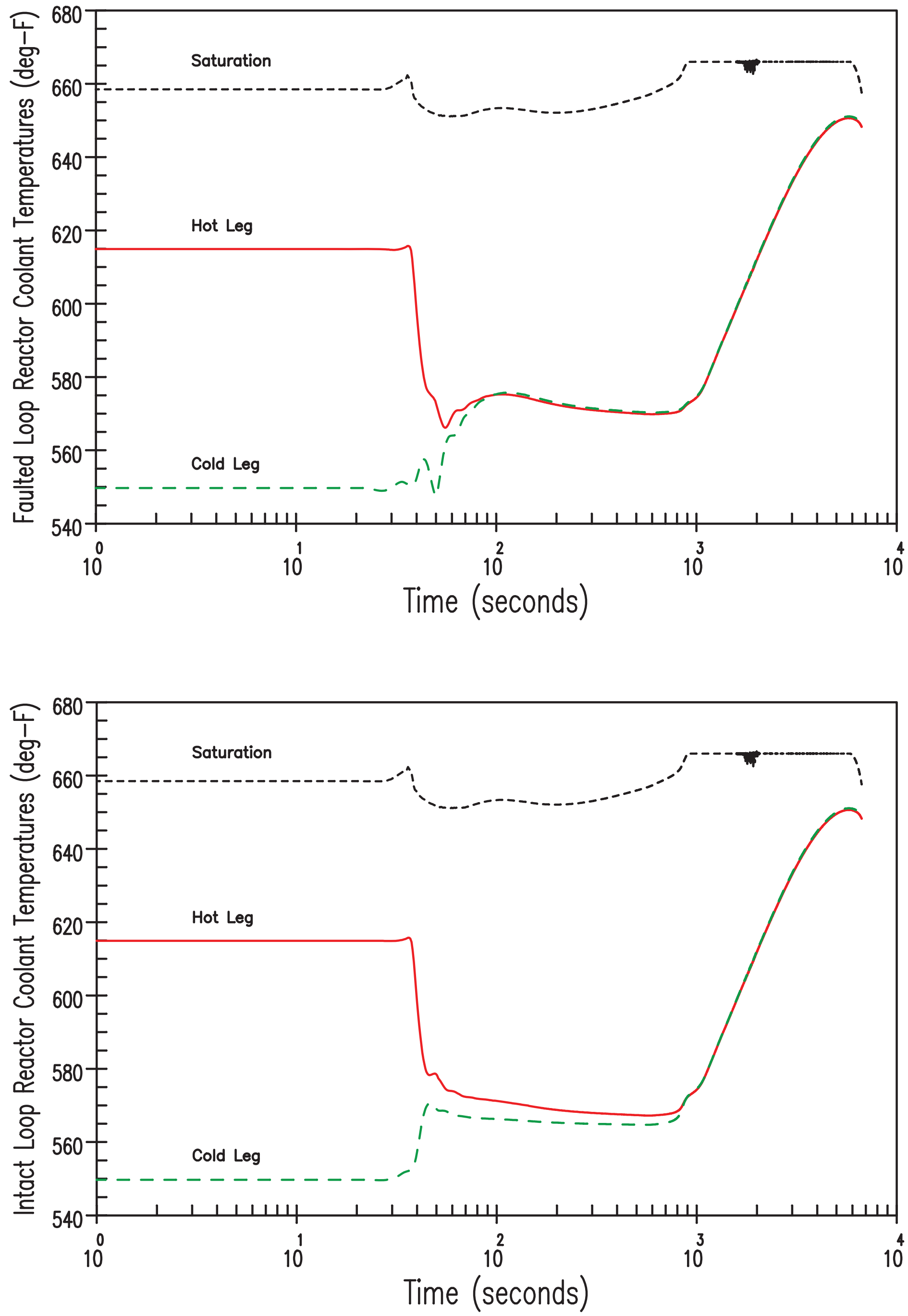
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-11

MAIN FEEDLINE RUPTURE
WITH OFFSITE POWER AVAILABLE

PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS



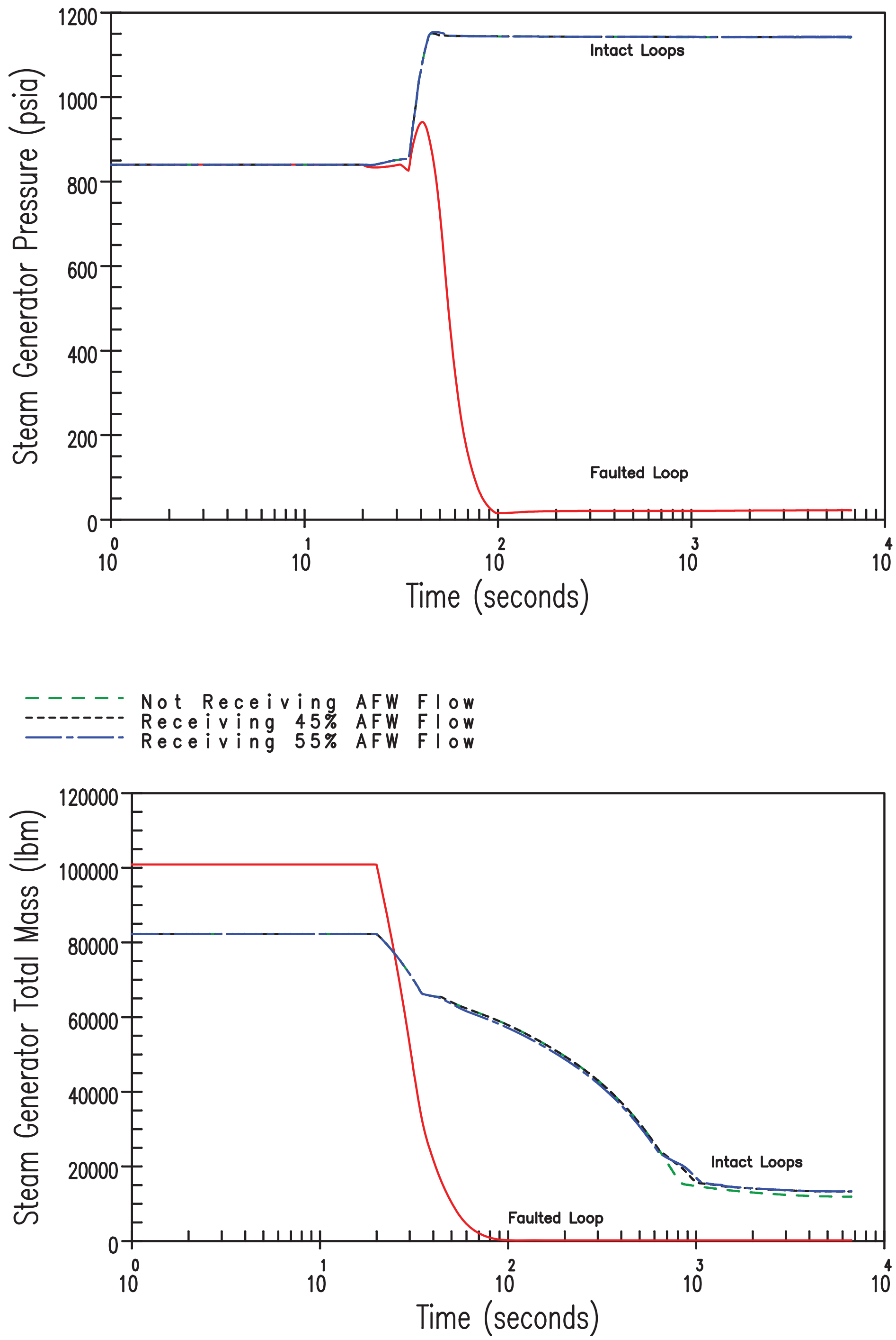
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-12

MAIN FEEDLINE RUPTURE
WITH OFFSITE POWER AVAILABLE

REACTOR COOLANT TEMPERATURE
TRANSIENTS FOR THE FAULTED
AND INTACT LOOPS

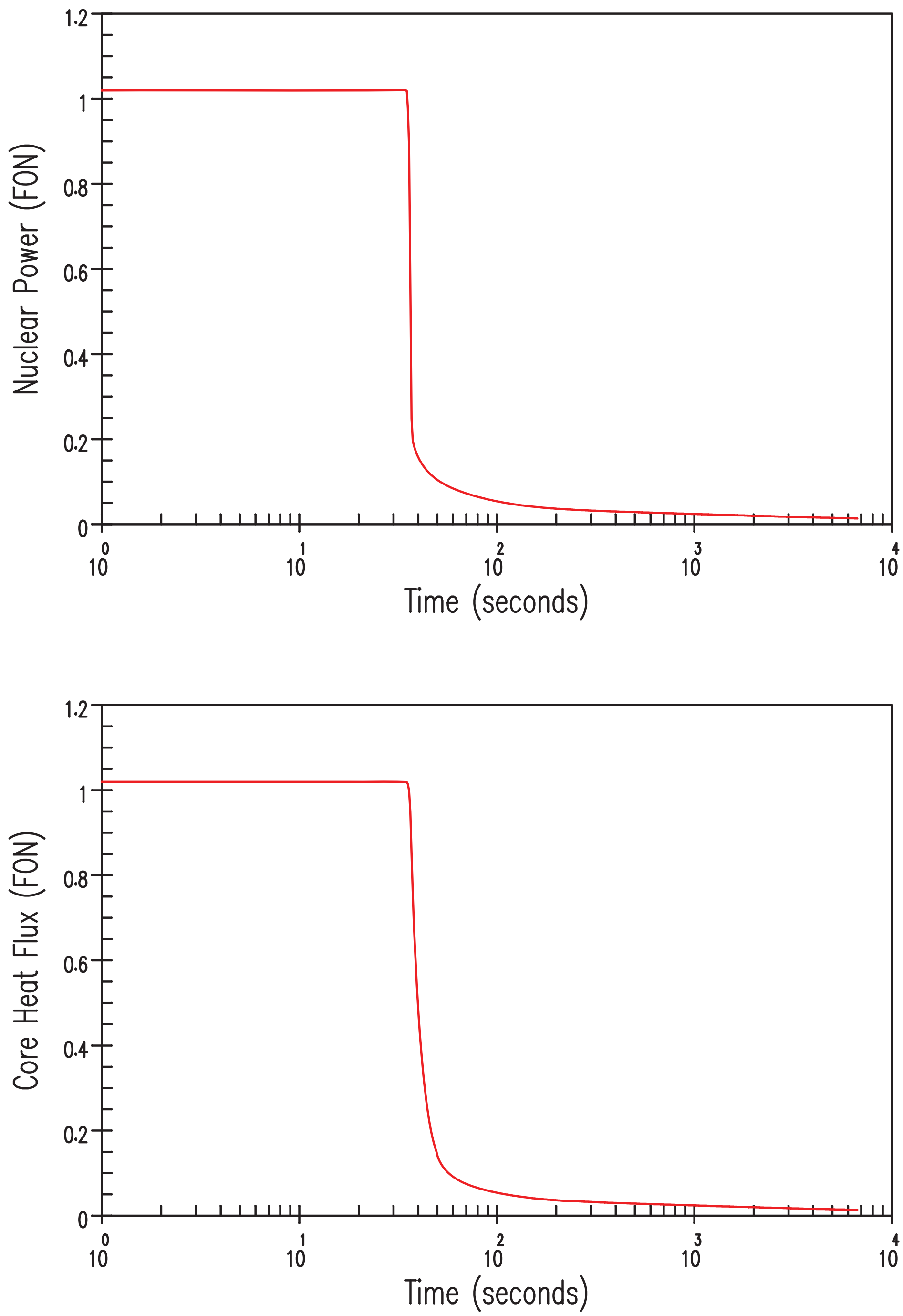


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-13

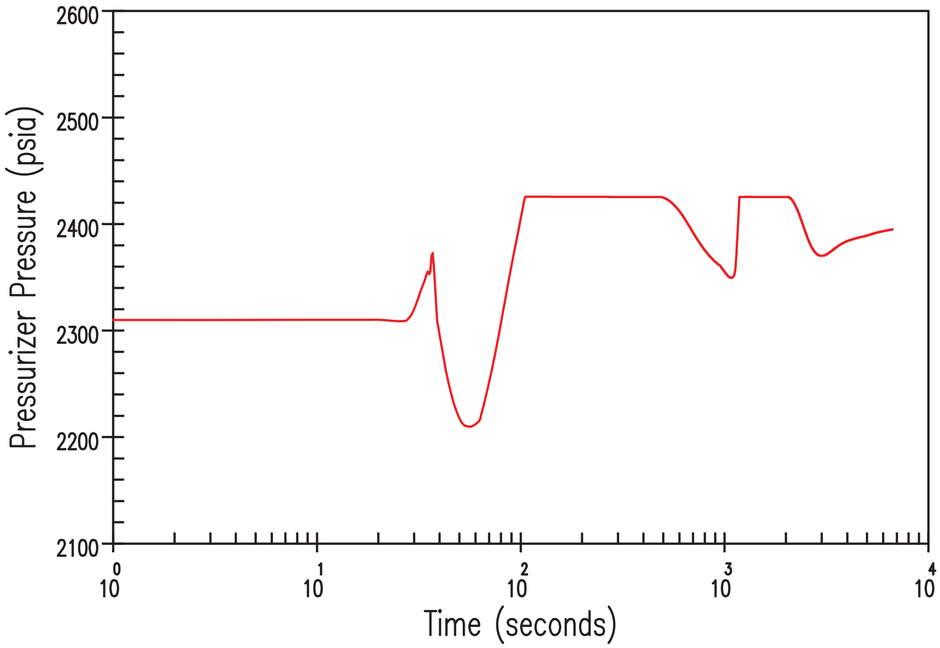
MAIN FEEDLINE RUPTURE
WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR PRESSURE AND
TOTAL MASS TRANSIENTS

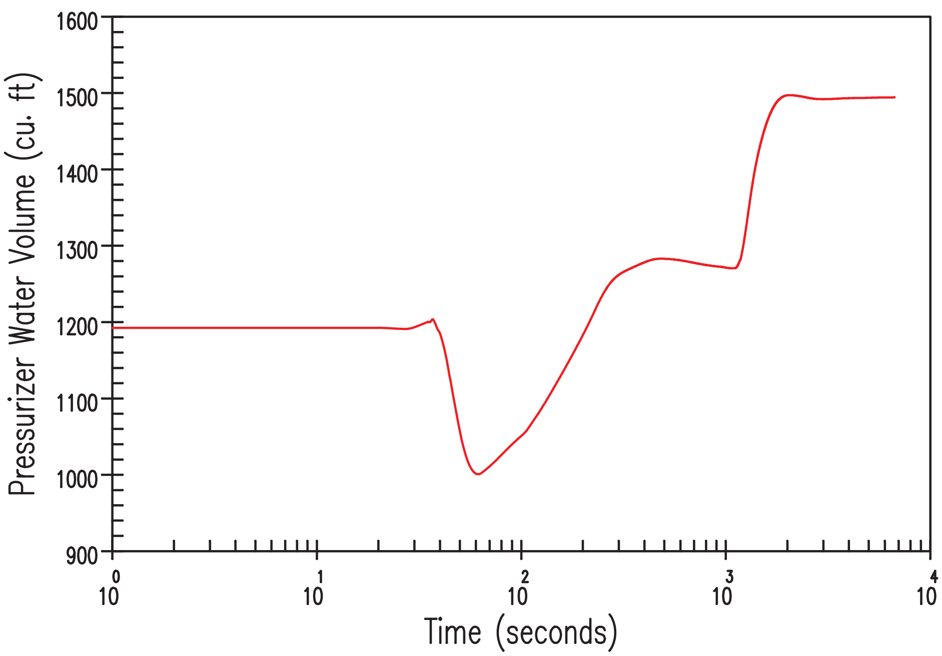


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-14
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER AVAILABLE
NUCLEAR POWER AND
CORE HEAT FLUX TRANSIENTS

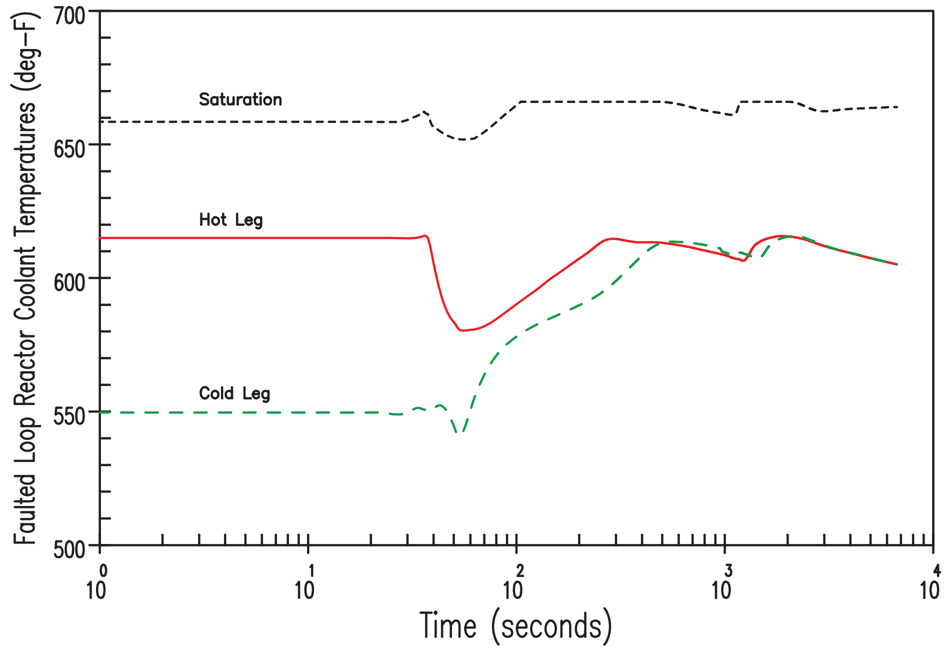


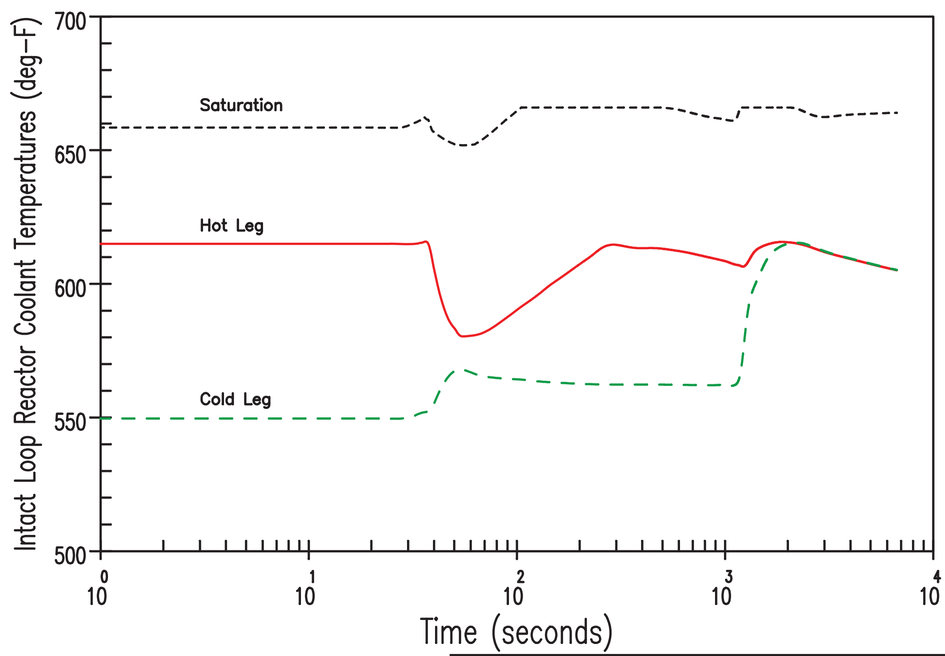


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-15
MAIN FEEDLINE RUPTURE WITHOUT OFFSITE POWER AVAILABLE
PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS

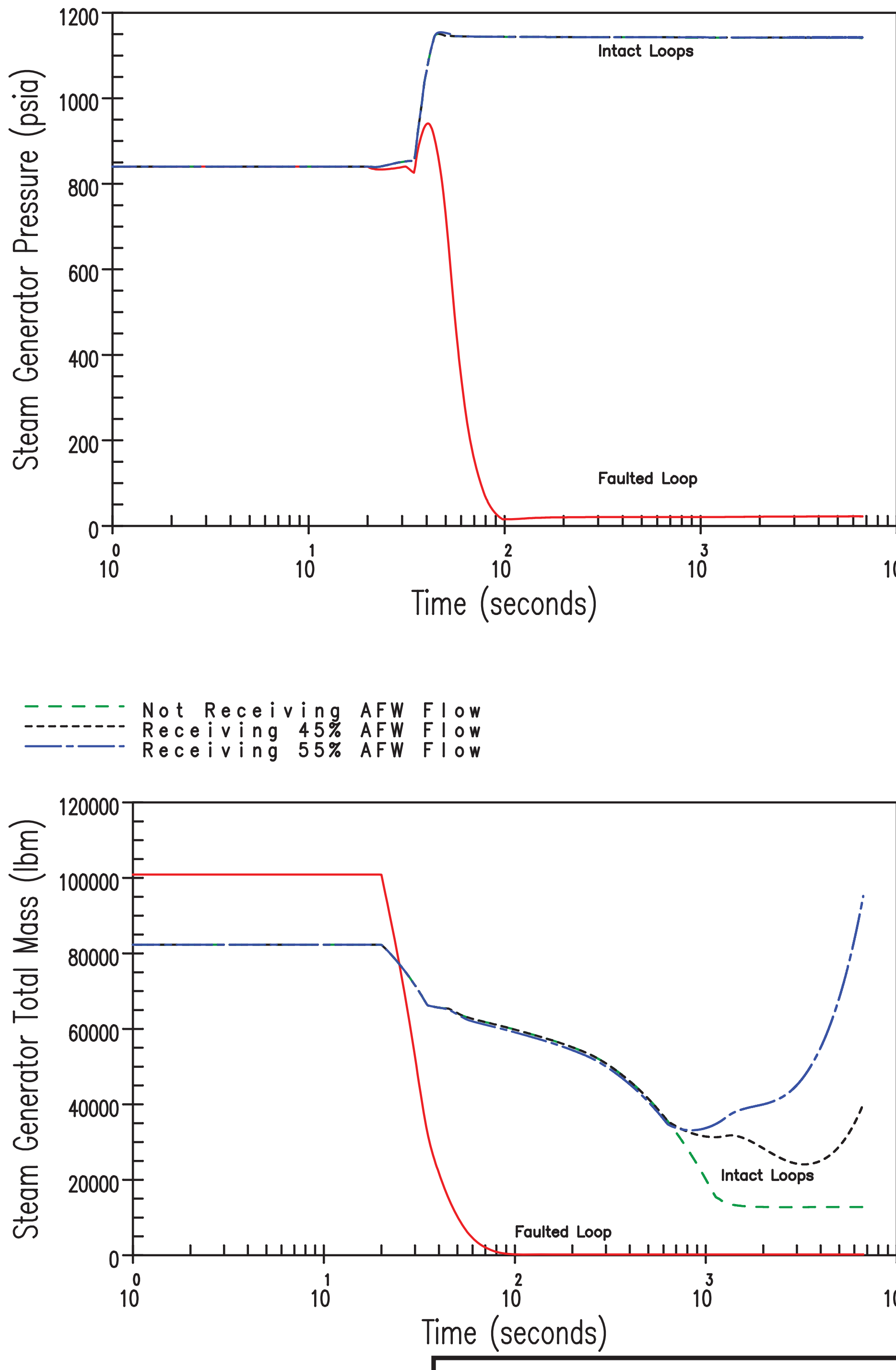




FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-16
MAIN FEEDLINE RUPTURE
WITHOUT OFFSITE POWER AVAILABLE
REACTOR COOLANT TEMPERATURE
TRANSIENTS FOR THE FAULTED
AND INTACT LOOPS



FSAR UPDATE

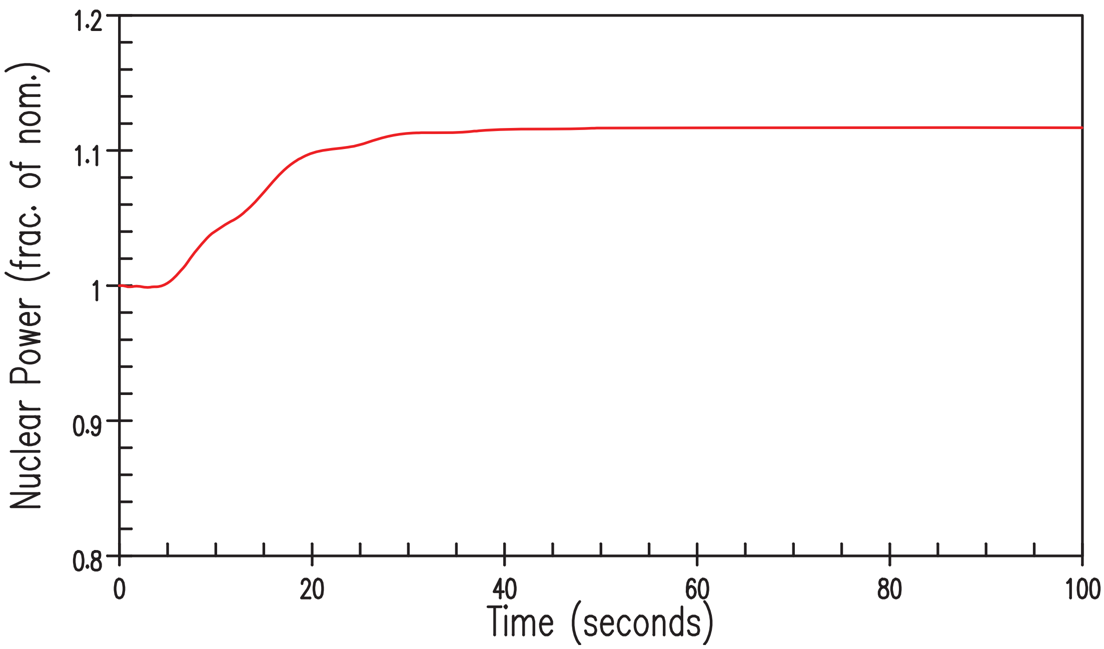
UNITS 1 AND 2
DIABLO CANYON SITE

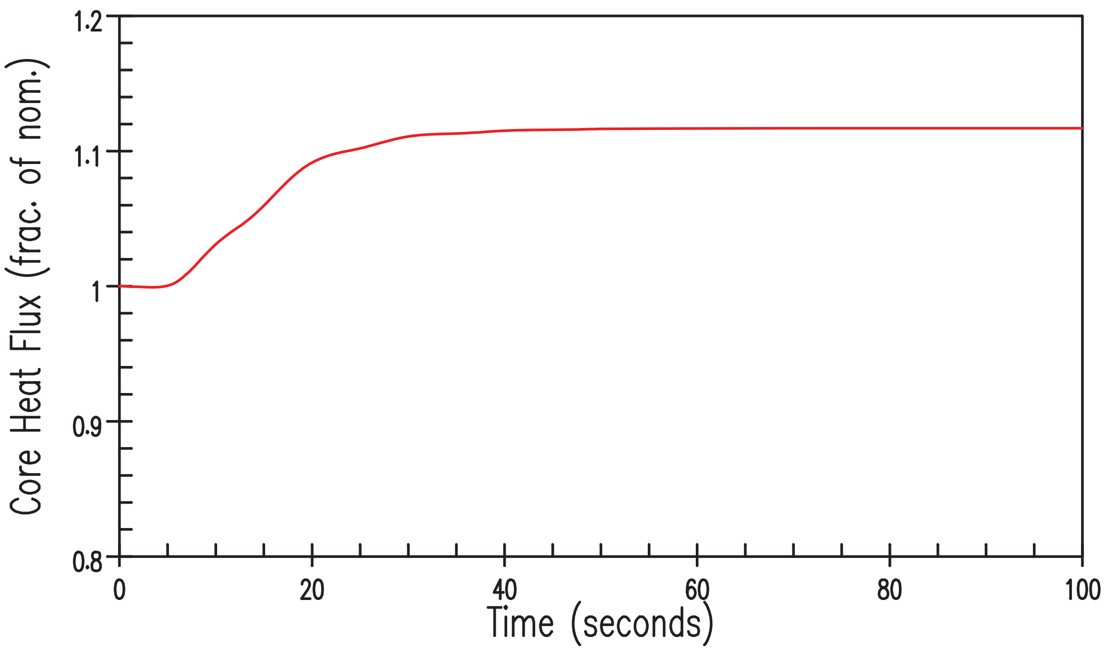
FIGURE 15.4.2-17

MAIN FEEDLINE RUPTURE WITHOUT OFFSITE POWER AVAILABLE

STEAM GENERATOR PRESSURE AND TOTAL MASS TRANSIENTS

Revision 19 May 2010





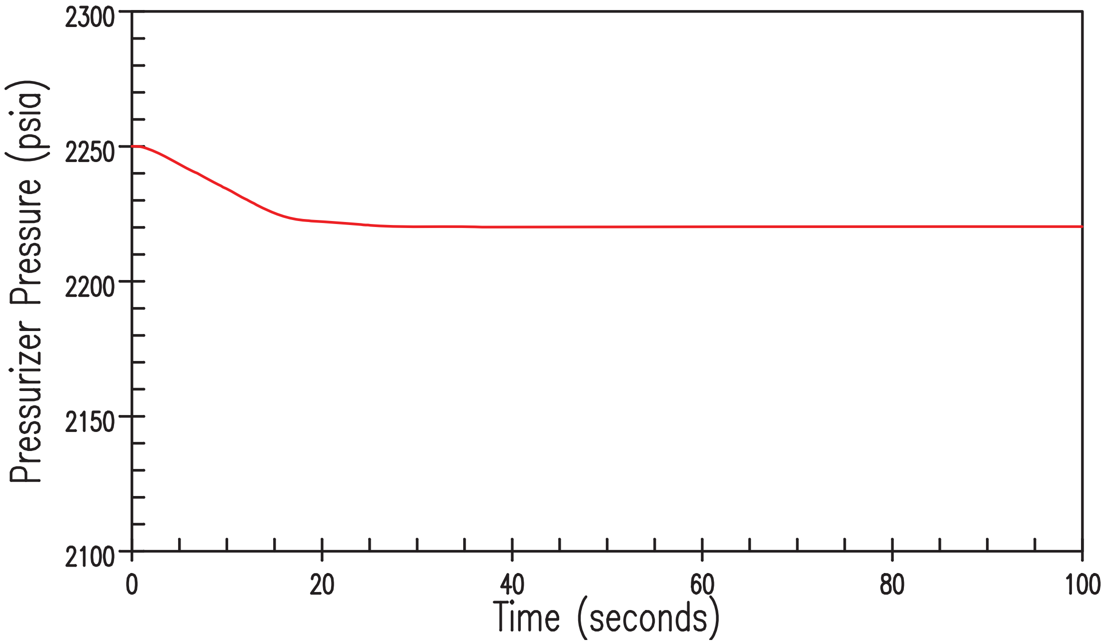
FSAR UPDATE

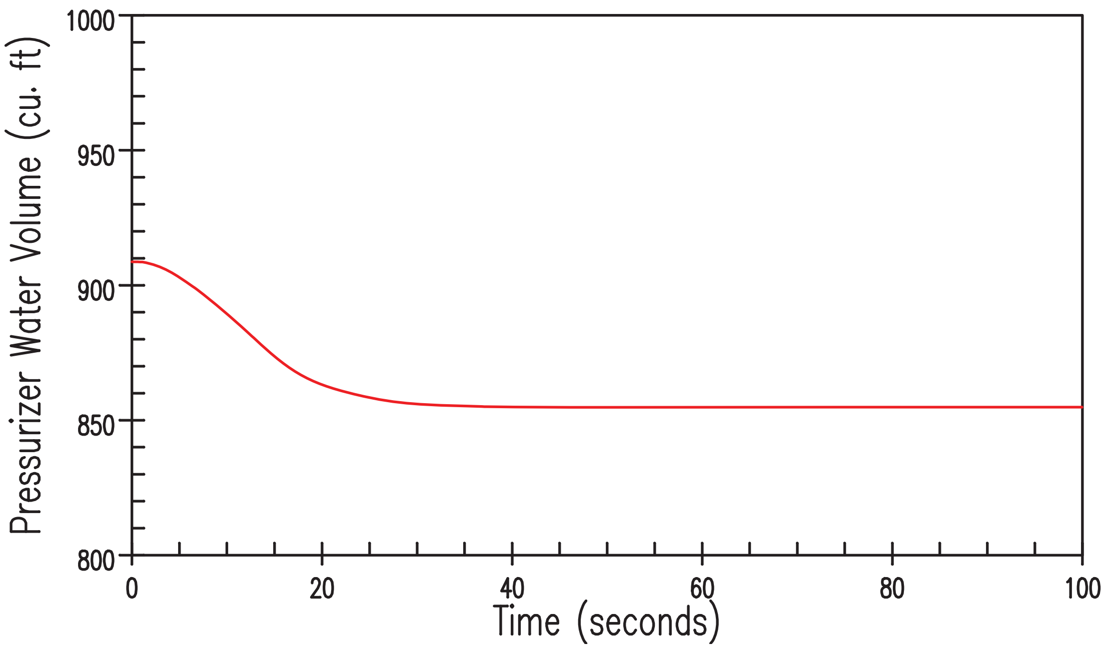
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-18

MAIN STEAM LINE RUPTURE AT FULL POWER, 0.49 ft^2 BREAK

NUCLEAR POWER AND CORE HEAT FLUX TRANSIENTS





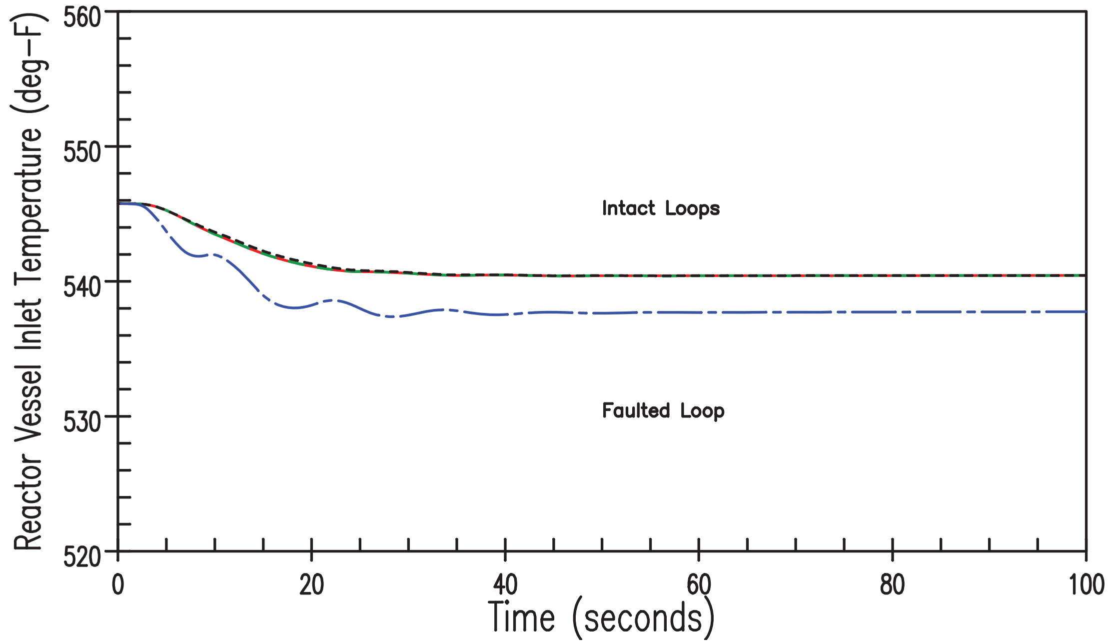
FSAR UPDATE

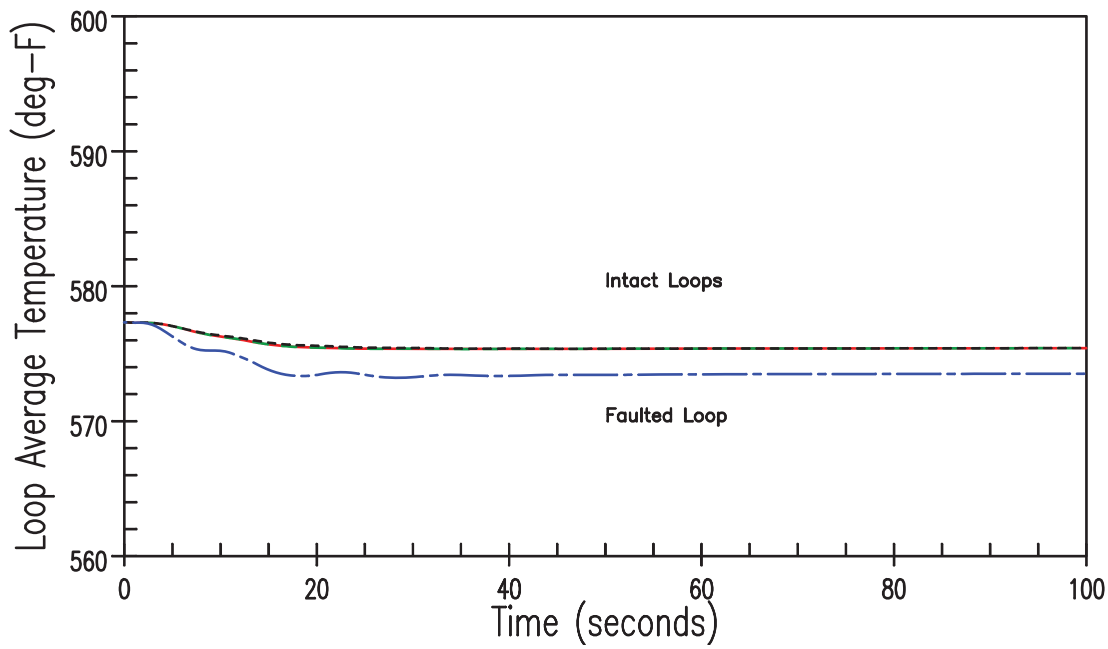
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-19

MAIN STEAM LINE RUPTURE AT FULL POWER, 0.49 ft^2 BREAK

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS





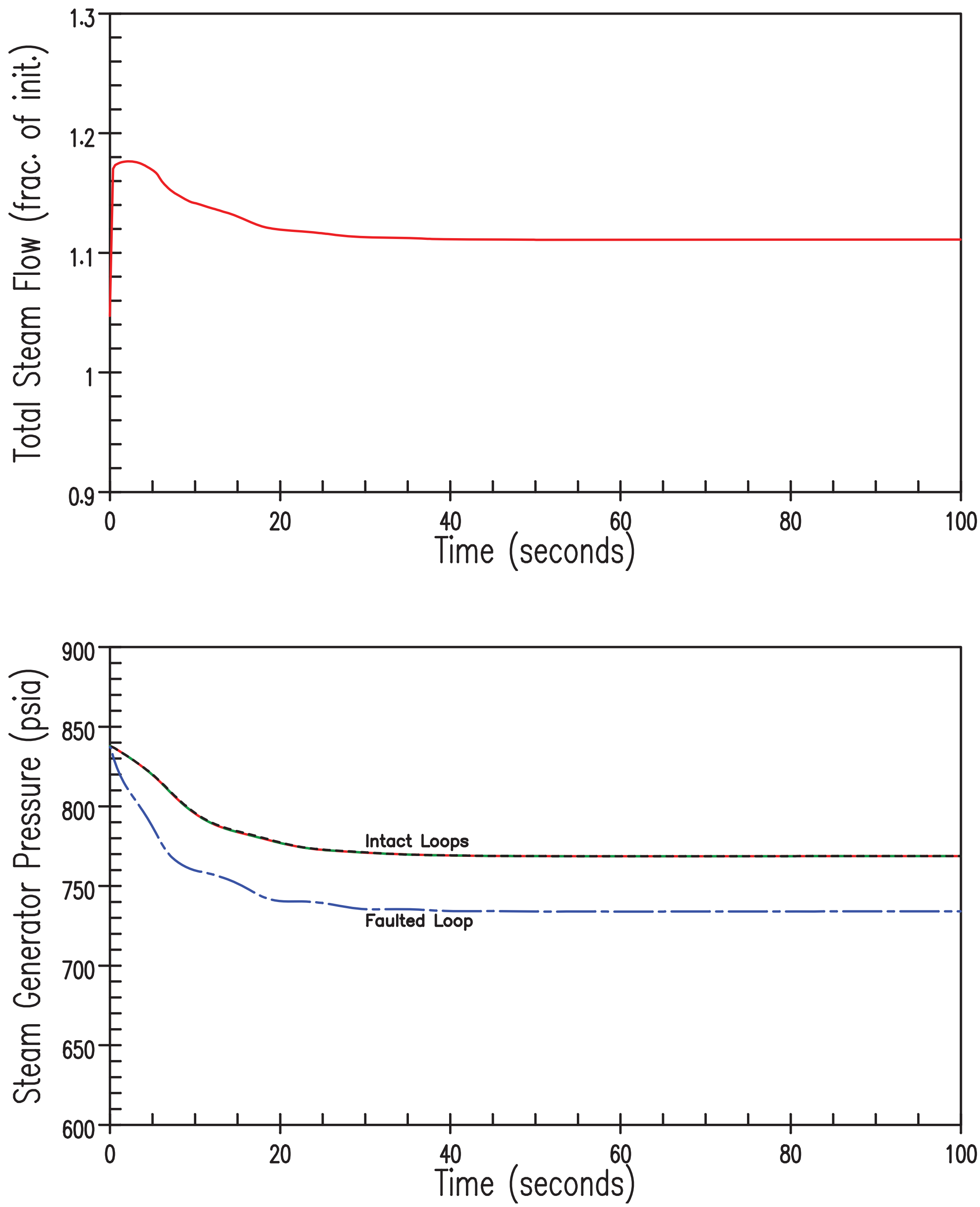
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

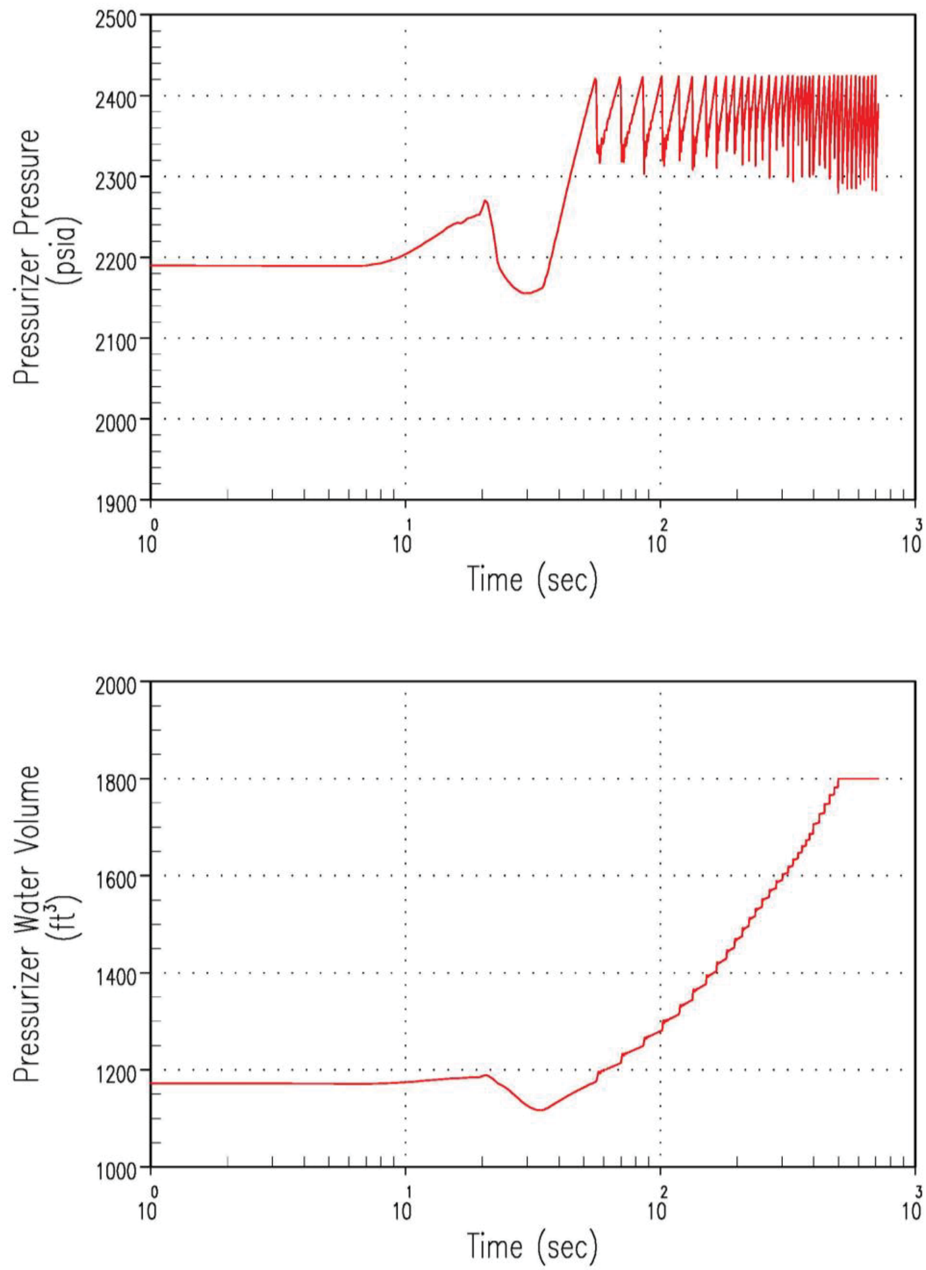
FIGURE 15.4.2-20

MAIN STEAM LINE RUPTURE AT FULL POWER, 0.49 ft^2 BREAK

REACTOR VESSEL INLET TEMPERATURE AND LOOP AVERAGE TEMPERATURE TRANSIENTS



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-21 MAIN STEAM LINE RUPTURE AT FULL POWER, 0.49 ft^2 BREAK TOTAL STEAM FLOW AND STEAM PRESSURE TRANSIENTS



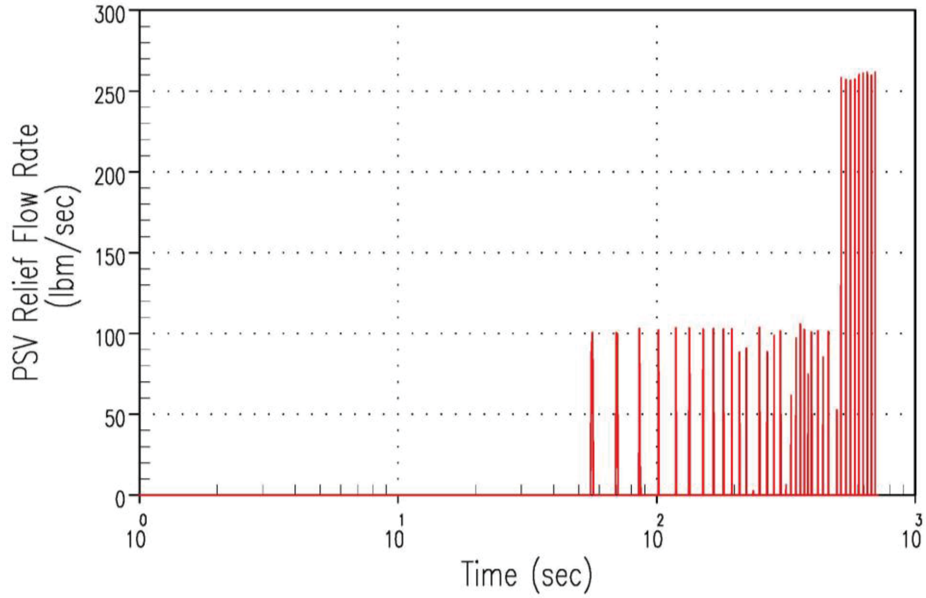
FSAR UPDATE

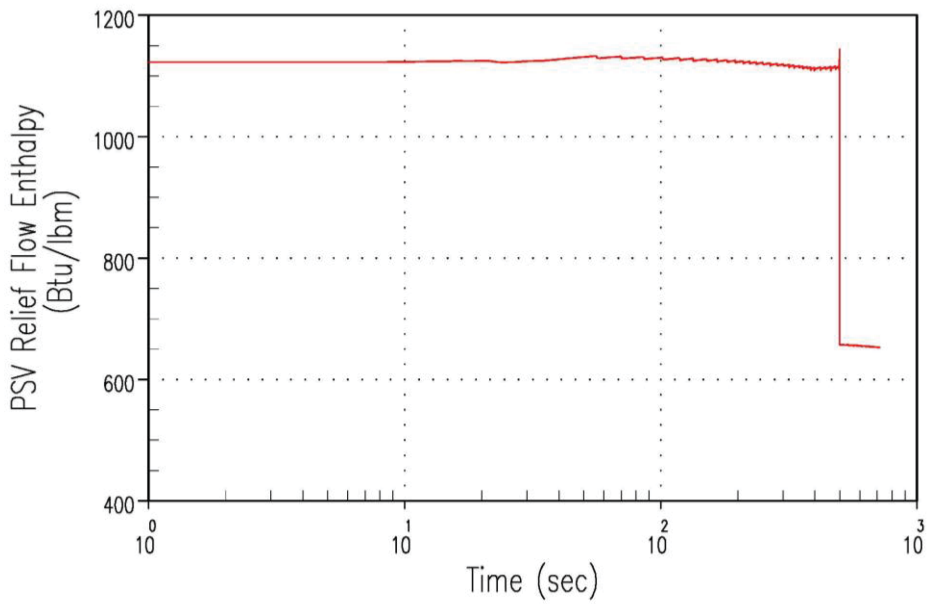
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-22

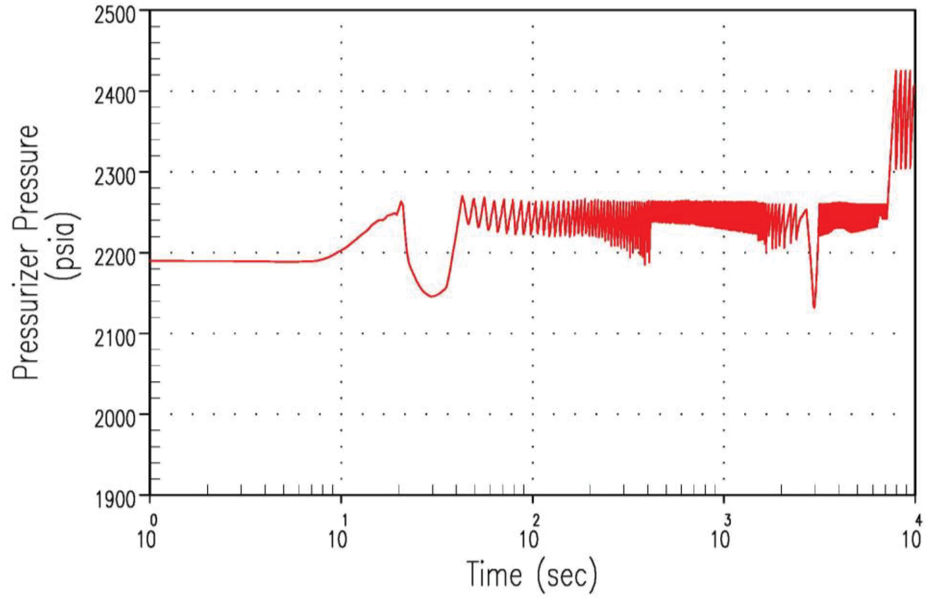
MAIN FEEDLINE RUPTURE
FOR PRESSURIZER FILLING
(UNBLOCK PRESSURIZER PORV)

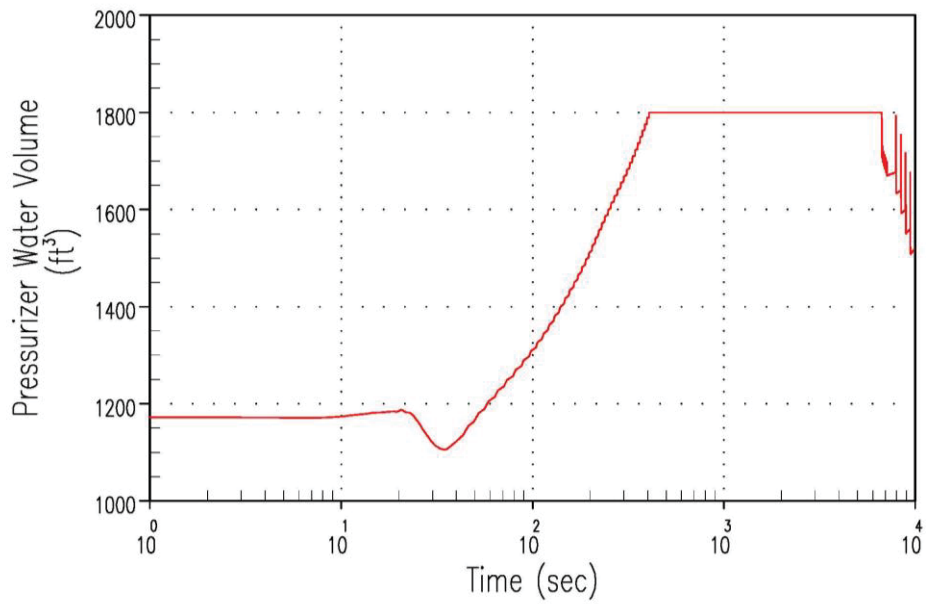
PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS



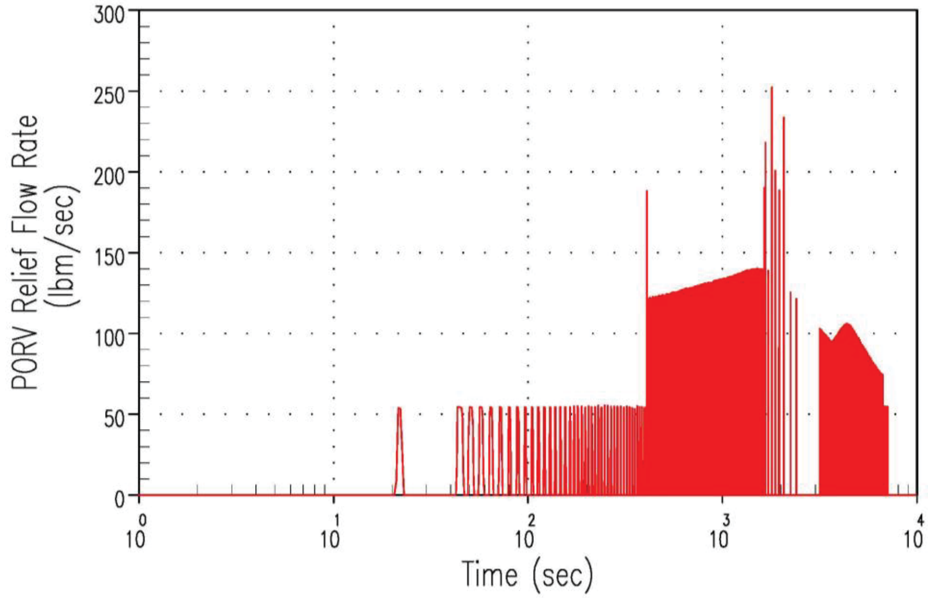


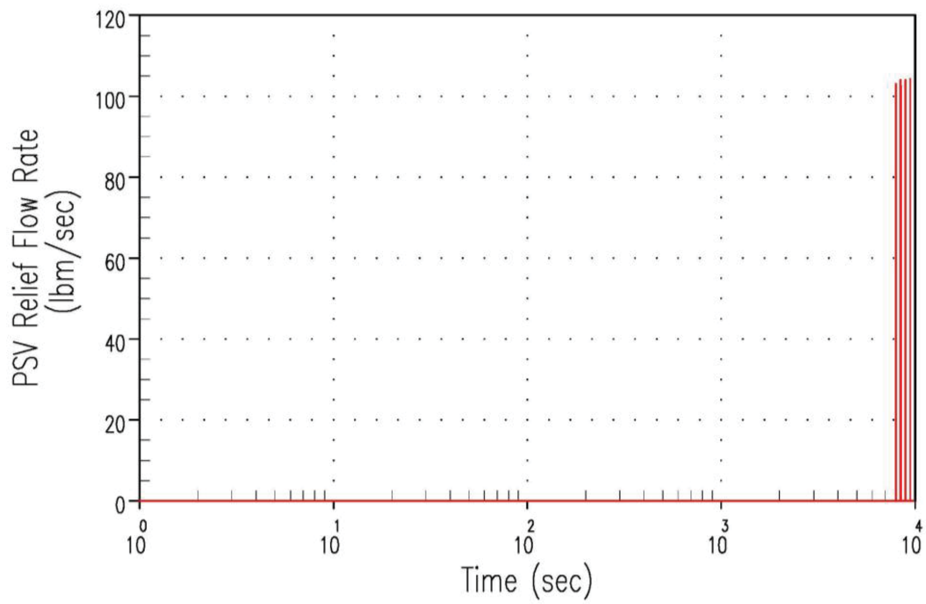
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-23 MAIN FEEDLINE RUPTURE FOR PRESSURIZER FILLING (UNBLOCK PRESSURIZER PORV) PSV RELIEF FLOW RATE AND ENTHALPY TRANSIENTS



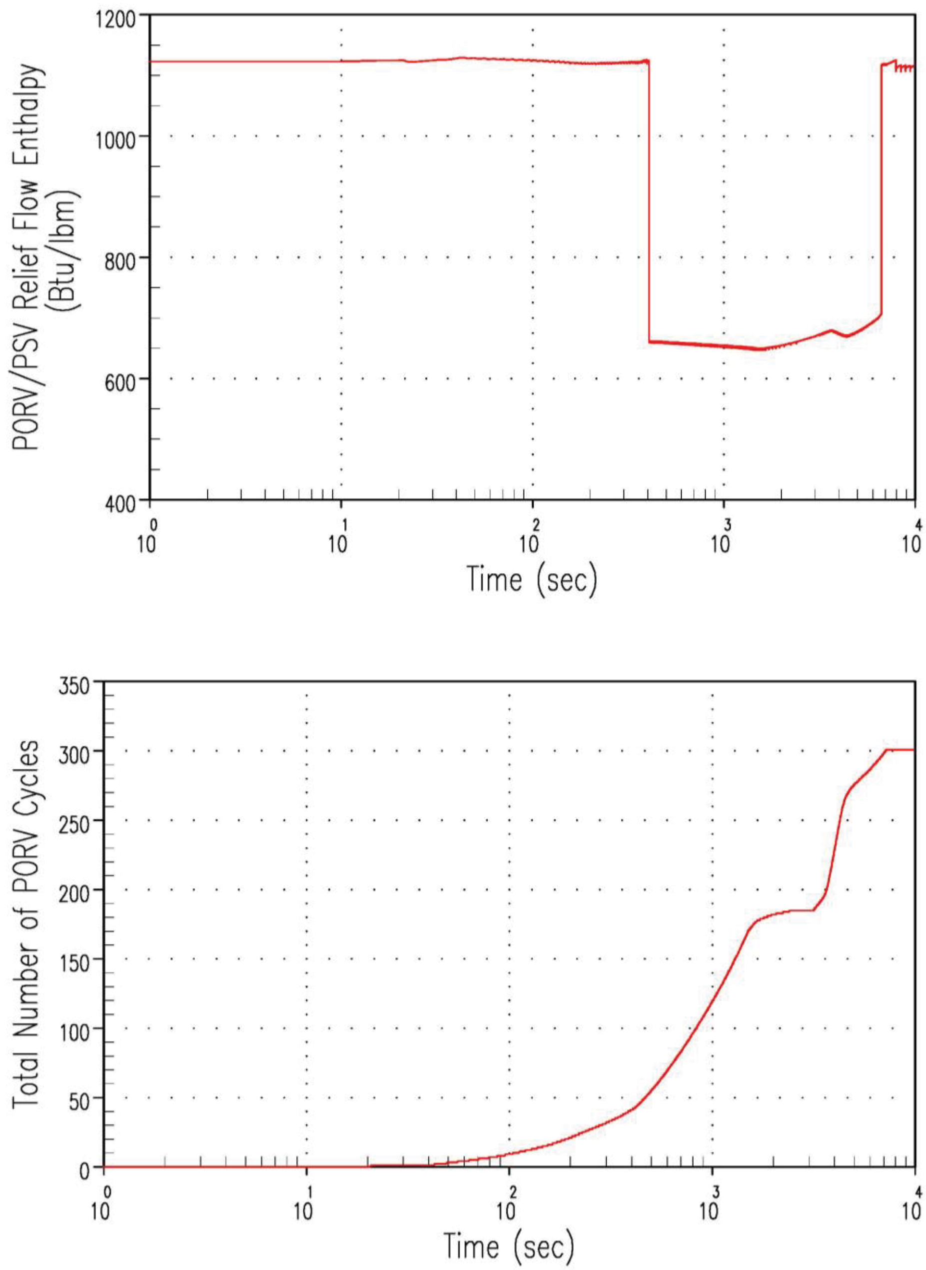


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-24 MAIN FEEDLINE RUPTURE FOR PRESSURIZER FILLING (ISOLATE CHARGING AND STOP RCP SEAL INJECTION FLOW) PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS





FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.2-25 MAIN FEEDLINE RUPTURE FOR PRESSURIZER FILLING (ISOLATE CHARGING AND STOP RCP SEAL INJECTION FLOW) PORV AND PSV RELIEF FLOW RATE TRANSIENTS



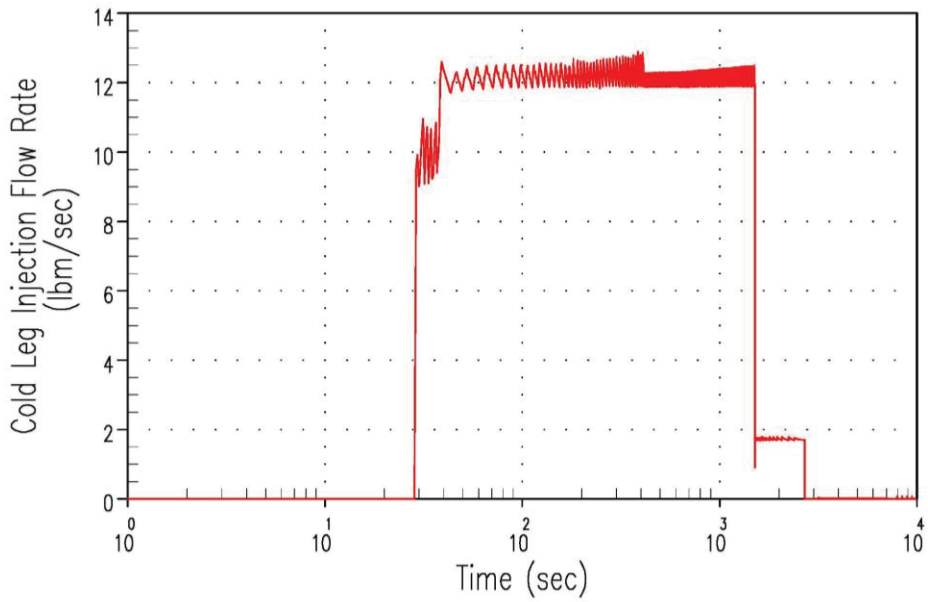
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-26

MAIN FEEDLINE RUPTURE
FOR PRESSURIZER FILLING
(ISOLATE CHARGING AND
STOP RCP SEAL INJECTION FLOW)

PORV/PSV RELIEF FLOW ENTHALPY
AND TOTAL NUMBER OF PORV
CYCLES TRANSIENTS



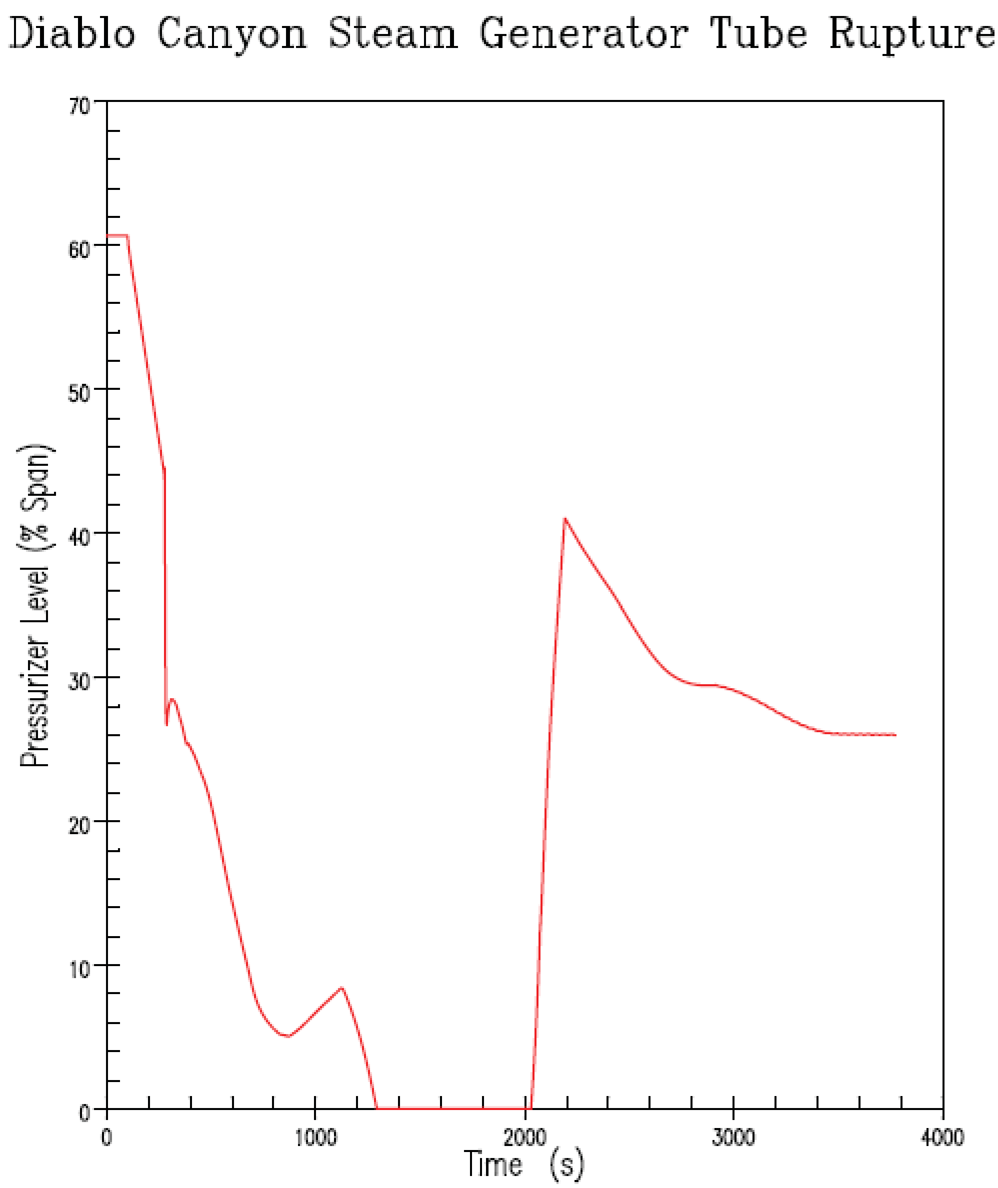
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.2-27

MAIN FEEDLINE RUPTURE
FOR PRESSURIZER FILLING
(ISOLATE CHARGING AND
STOP RCP SEAL INJECTION FLOW)

COLD LEG INJECTION FLOW RATE
TRANSIENT

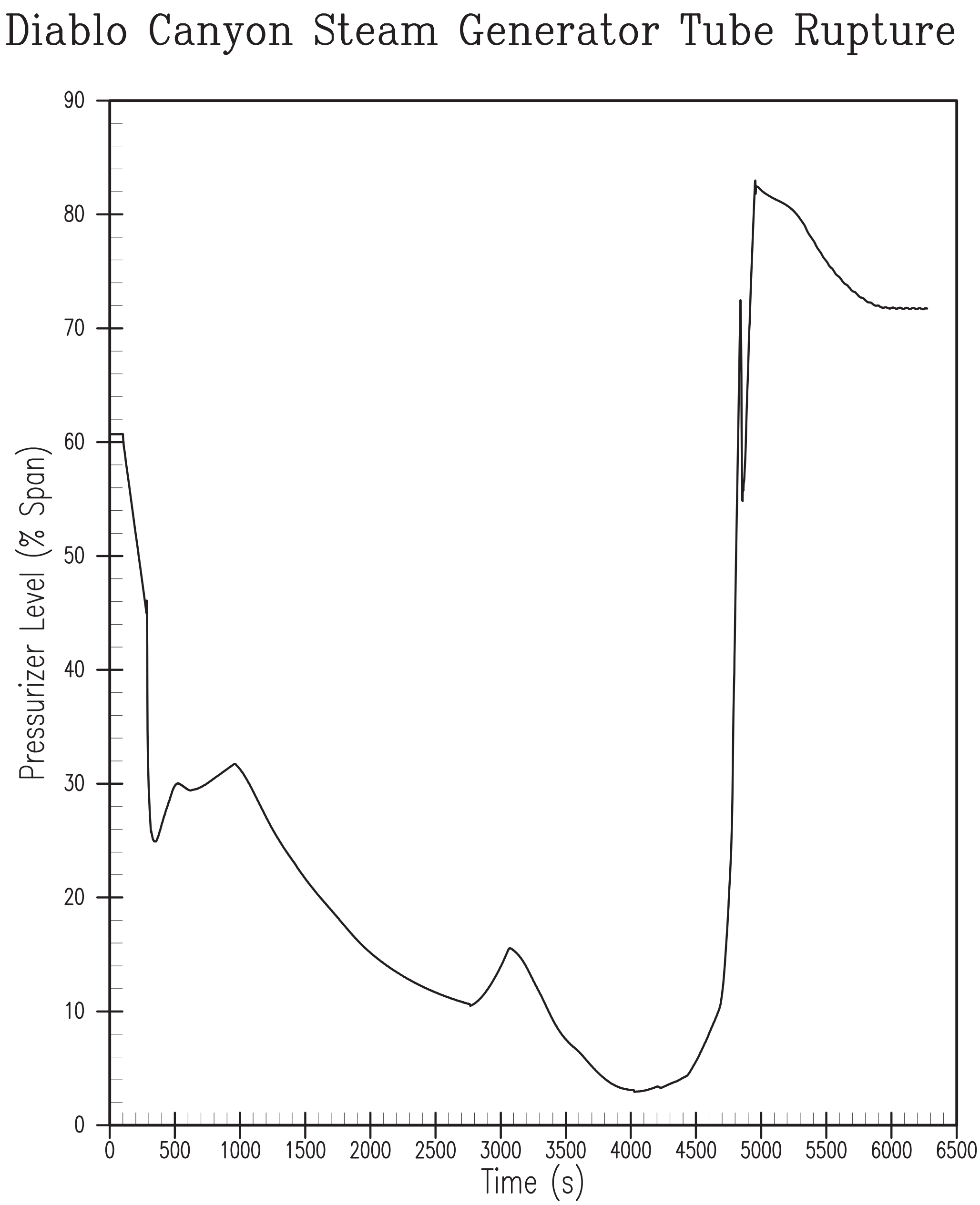


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-1A

PRESSURIZER LEVEL
SGTR MTO ANALYSIS



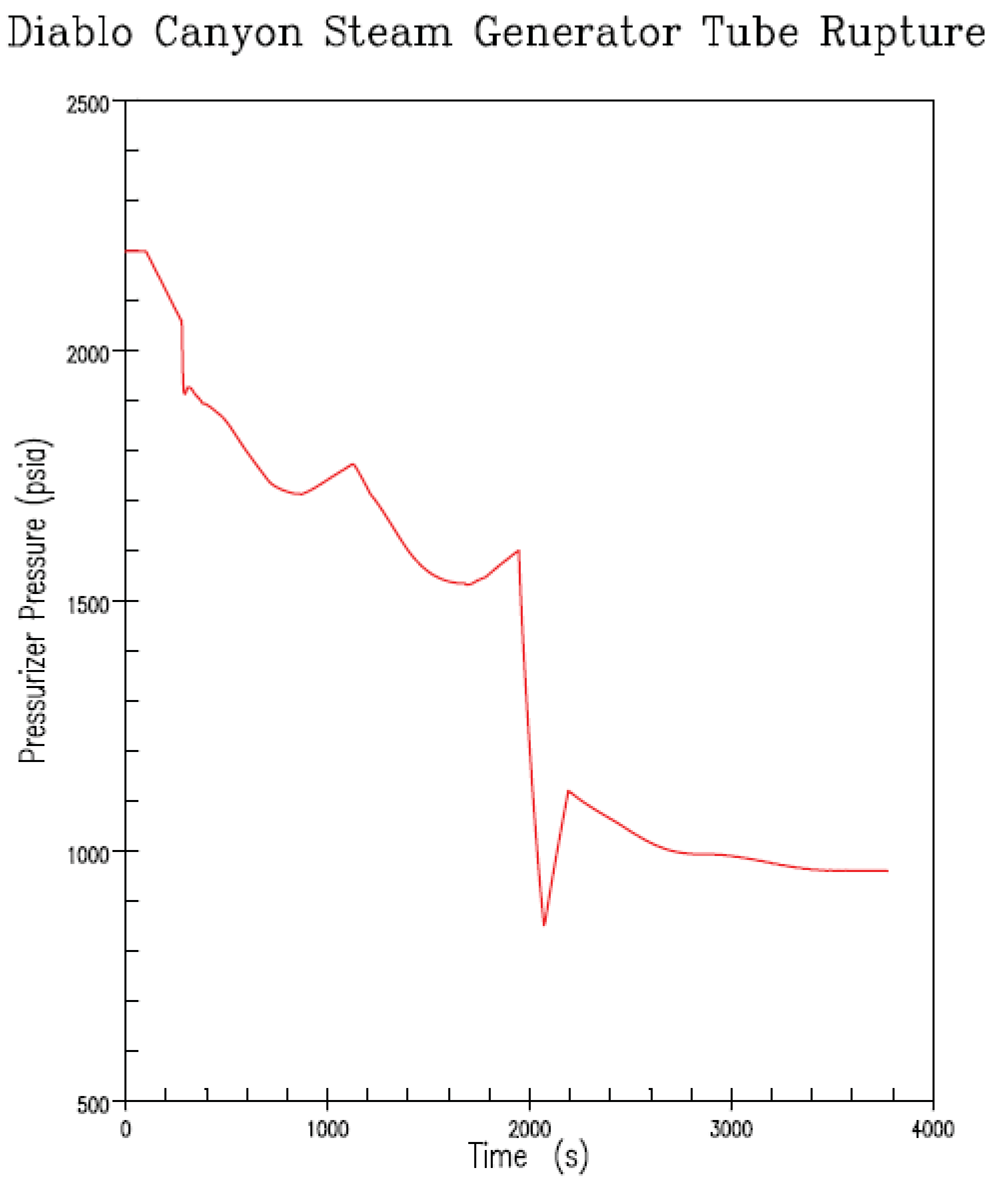
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-1B

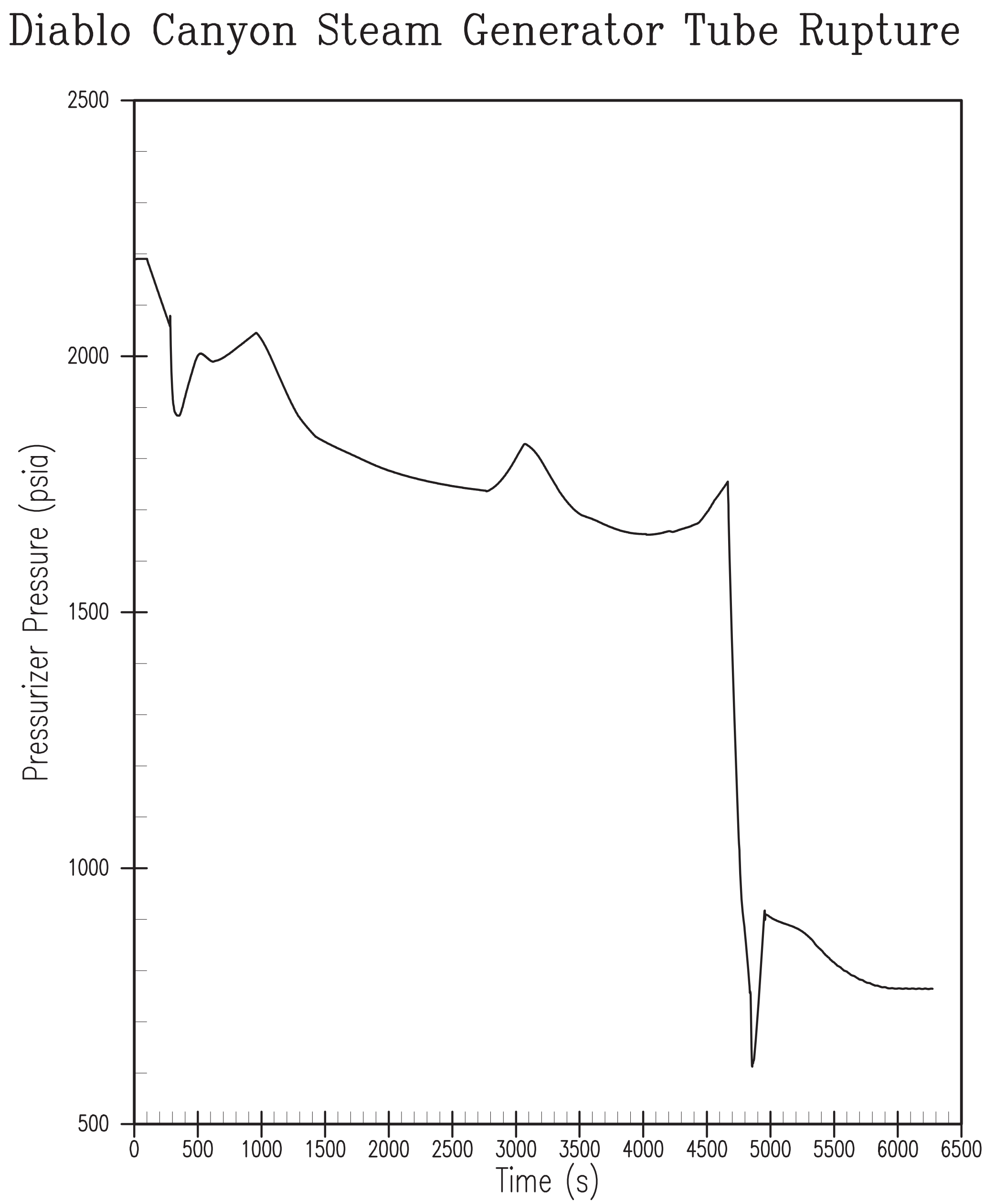
PRESSURIZER LEVEL
SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-2A PRESSURIZER PRESSURE SGTR MTO ANALYSIS

Revision 20 November 2011

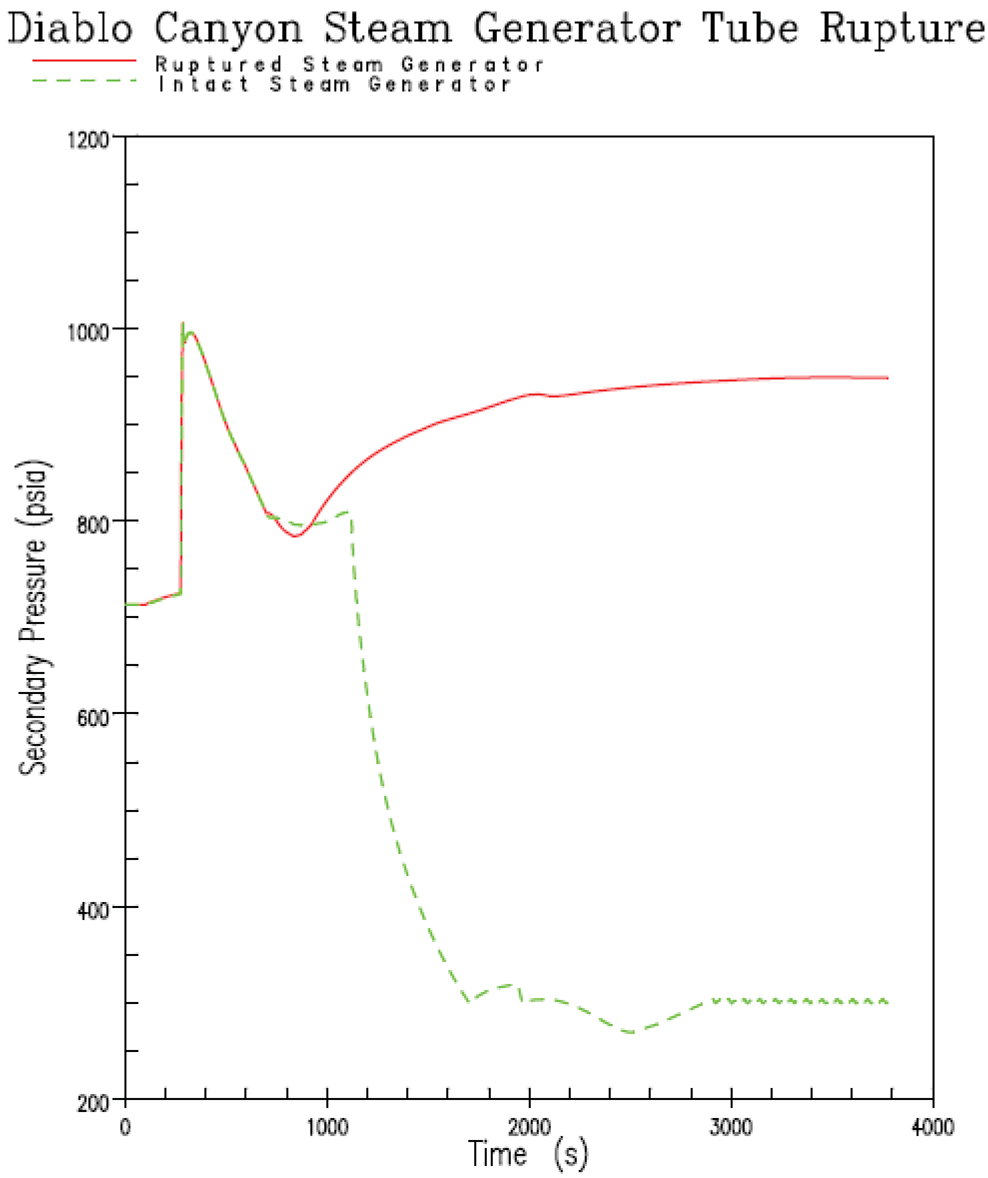


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

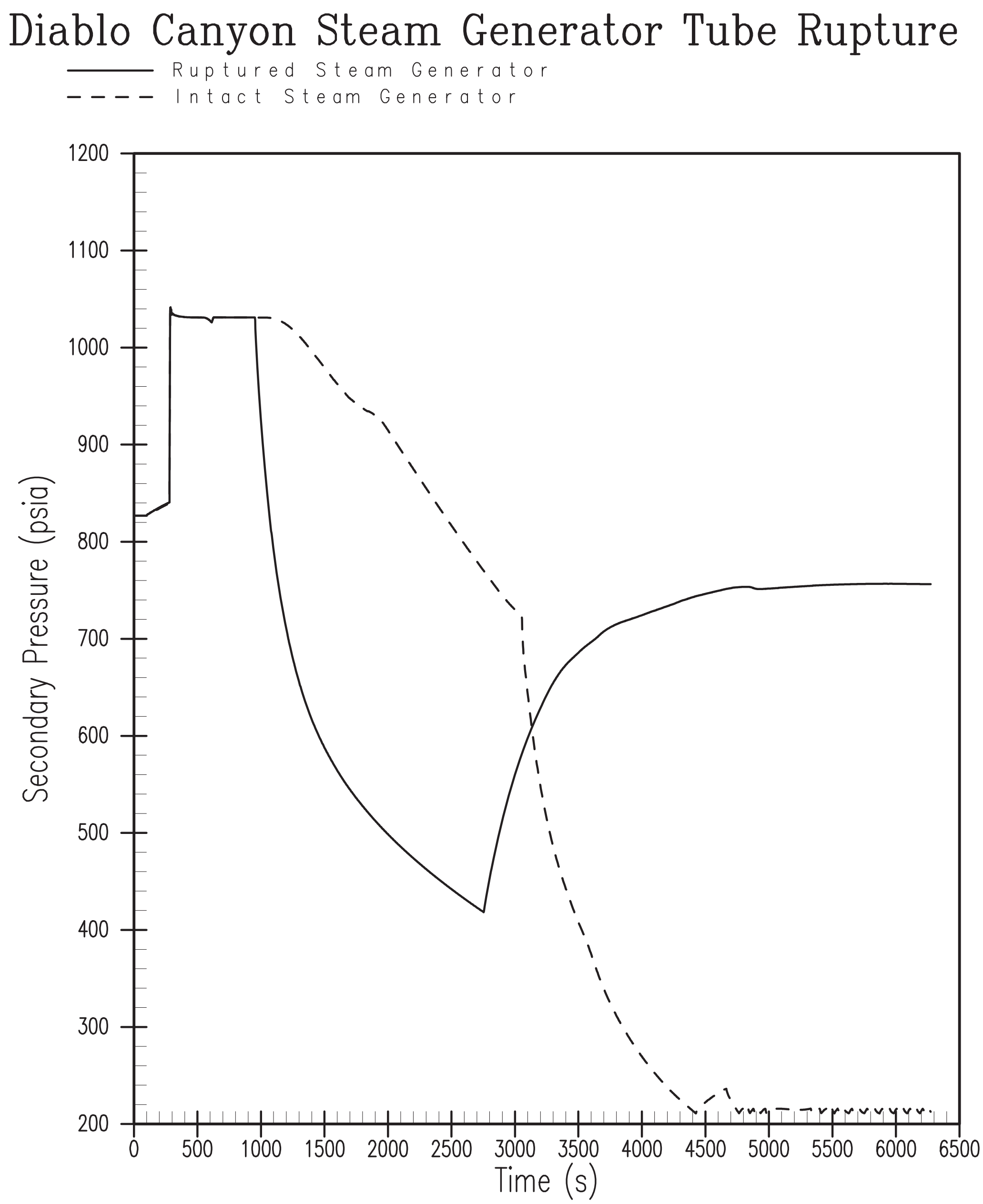
FIGURE 15.4.3-2B
PRESSURIZER PRESSURE
SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-3A SECONDARY PRESSURE SGTR MTO ANALYSIS

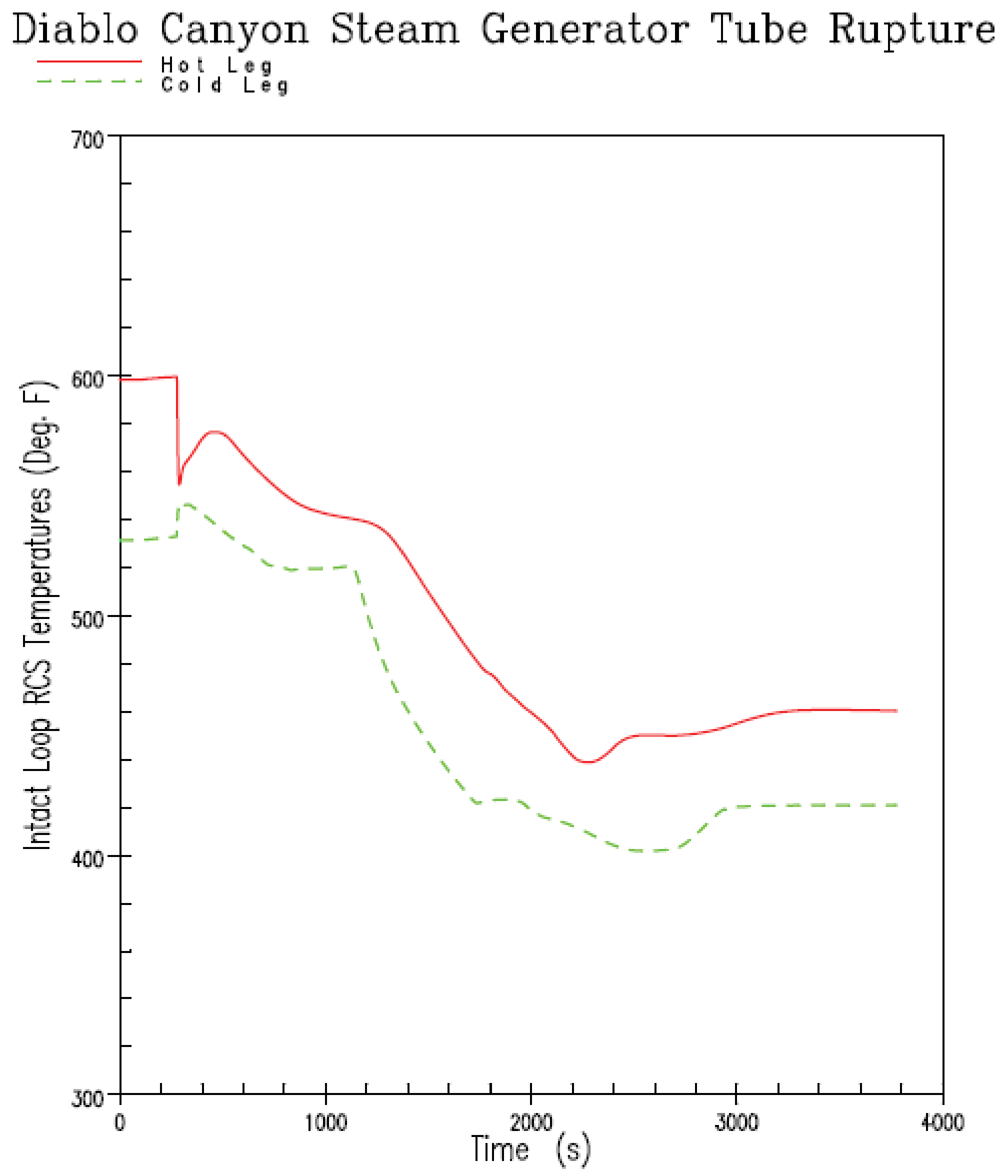
Revision 20 November 2011



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-3B
SECONDARY PRESSURE
SGTR DOSE INPUT ANALYSIS

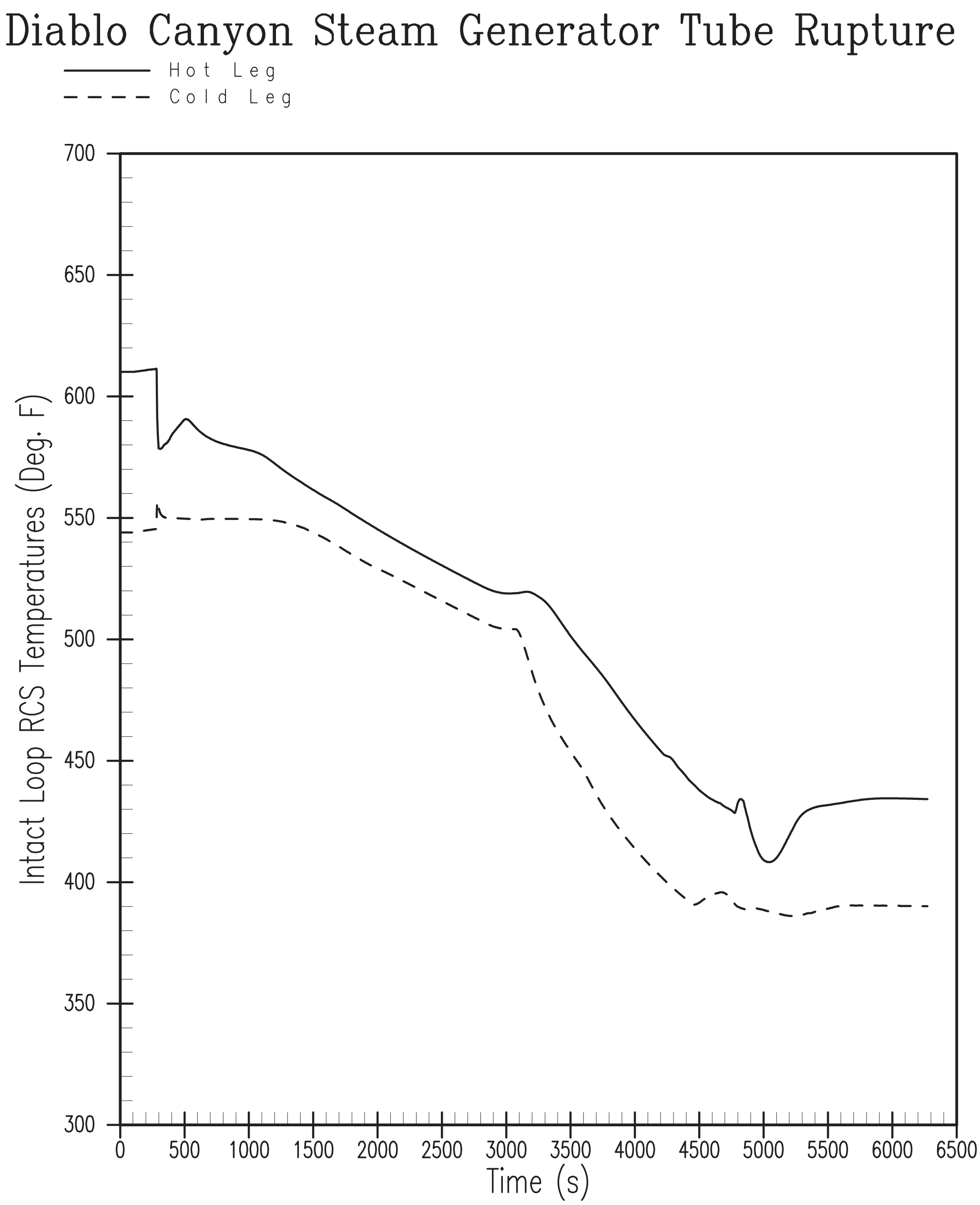


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-4A

INTACT LOOP HOT AND COLD LEG RCS TEMPERATURES SGTR MTO ANALYSIS



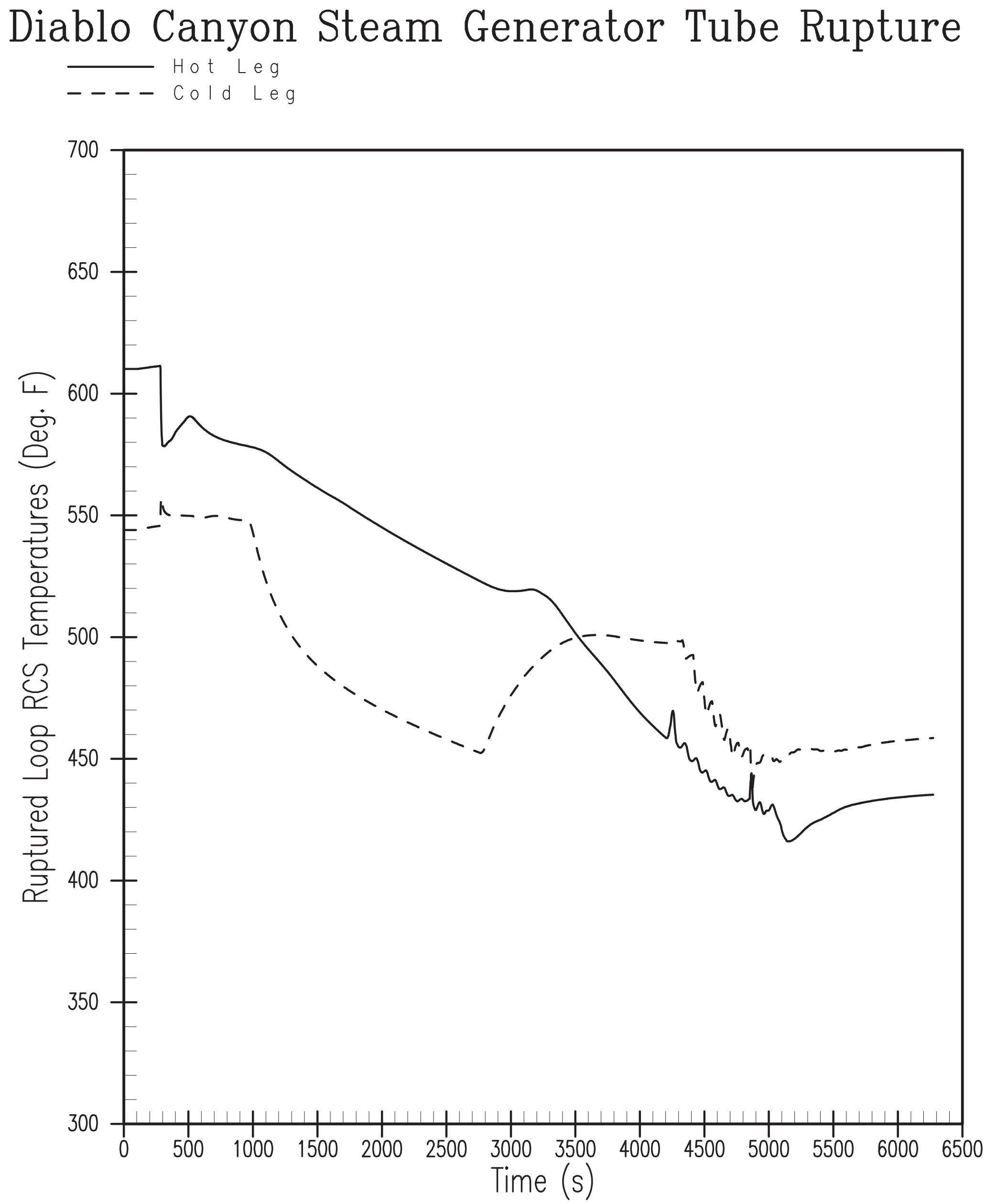
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-4B

INTACT LOOP HOT AND COLD LEG RCS TEMPERATURES SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013

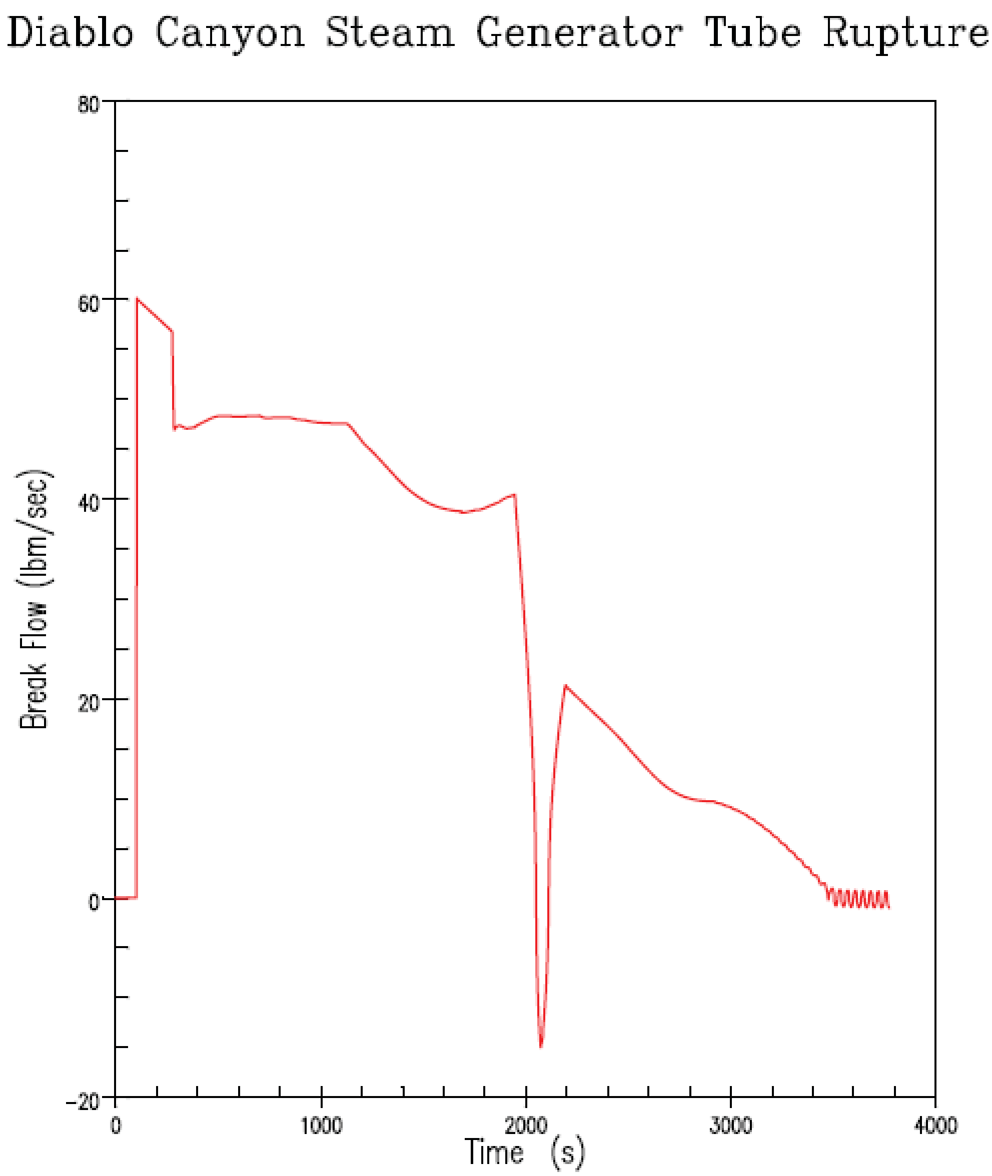


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-5B
RUPTURED LOOP HOT AND COLD LEG RCS TEMPERATURES SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013



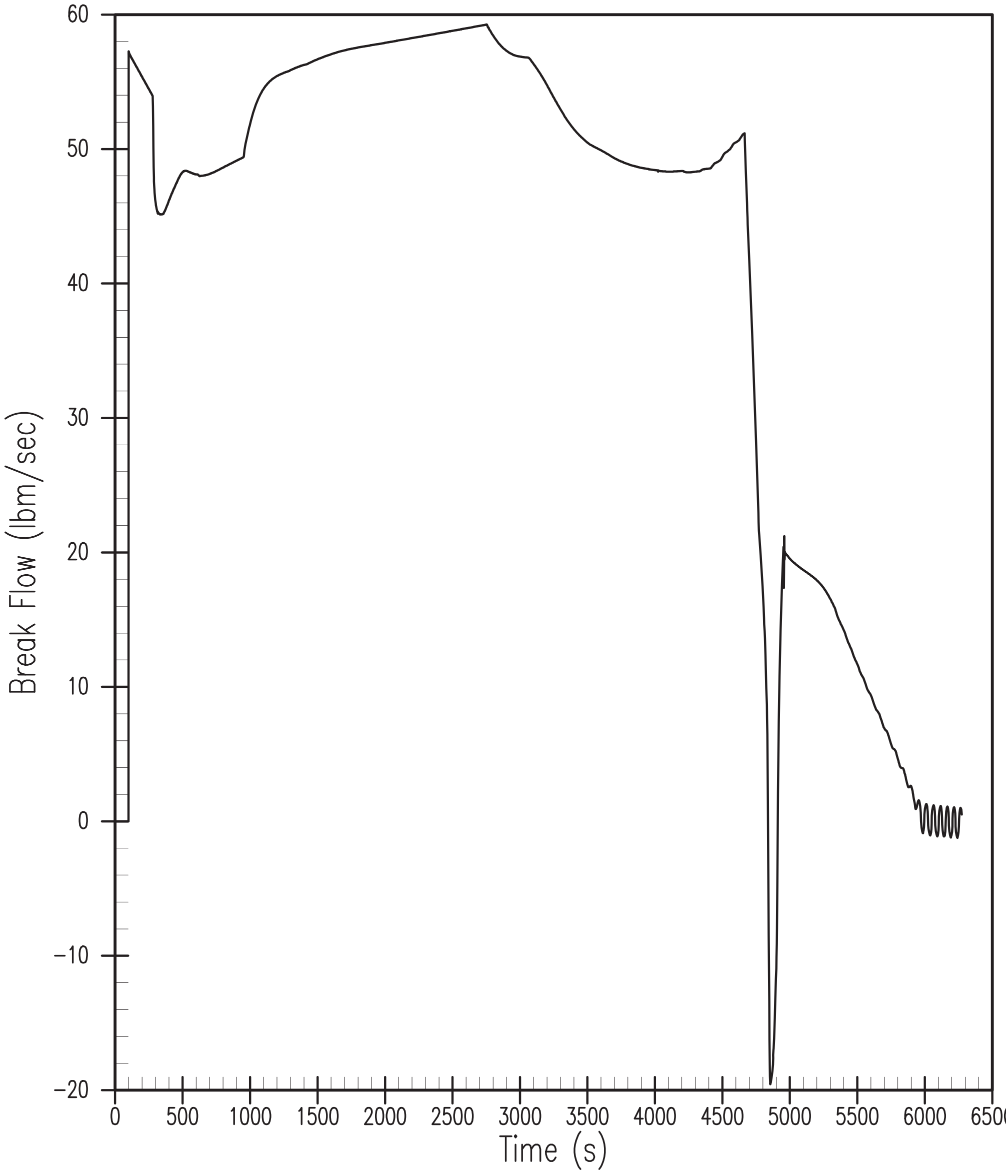
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-6A

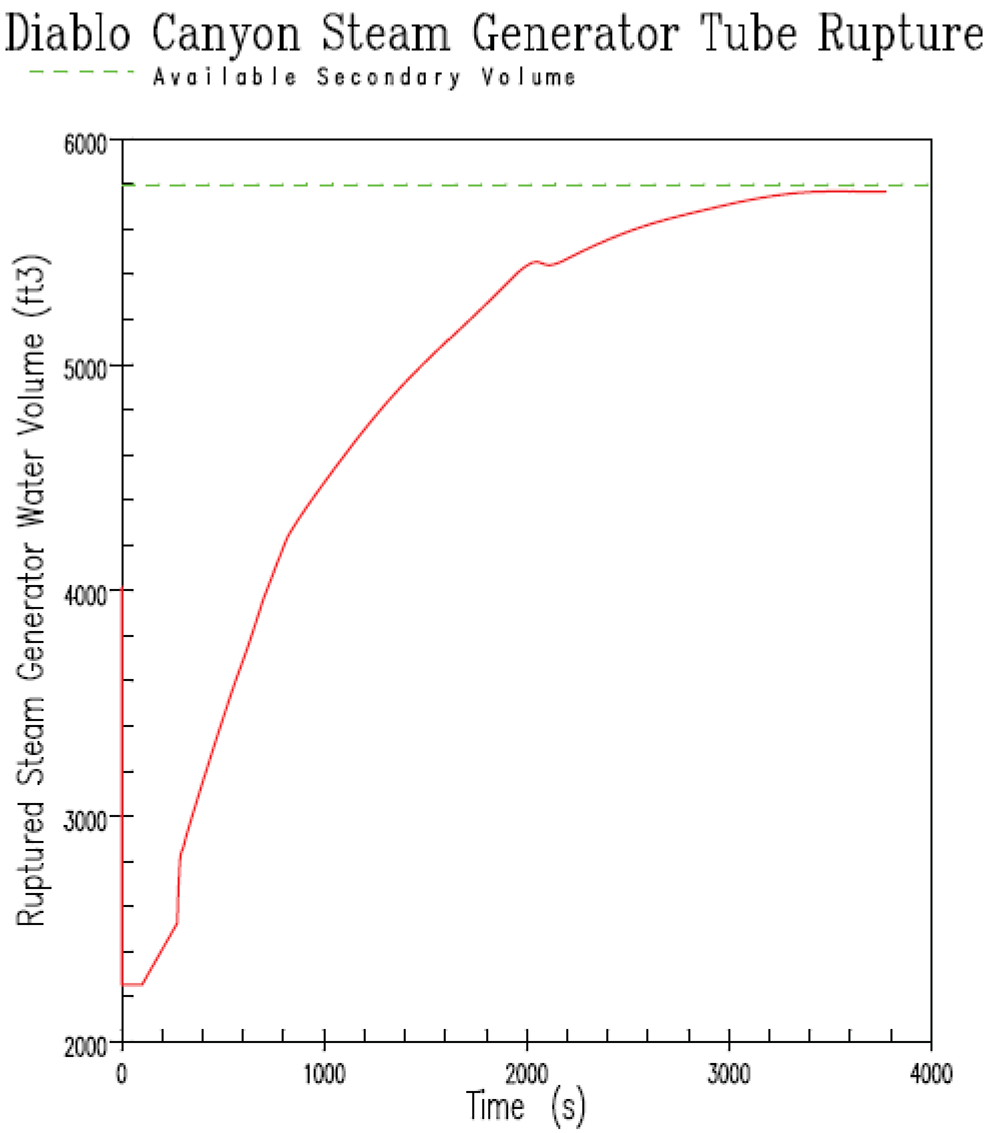
PRIMARY TO SECONDARY
BREAK FLOW RATE
SGTR MTO ANALYSIS

Diablo Canyon Steam Generator Tube Rupture



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-6B PRIMARY TO SECONDARY BREAK FLOW RATE SGTR DOSE INPUT ANALYSIS

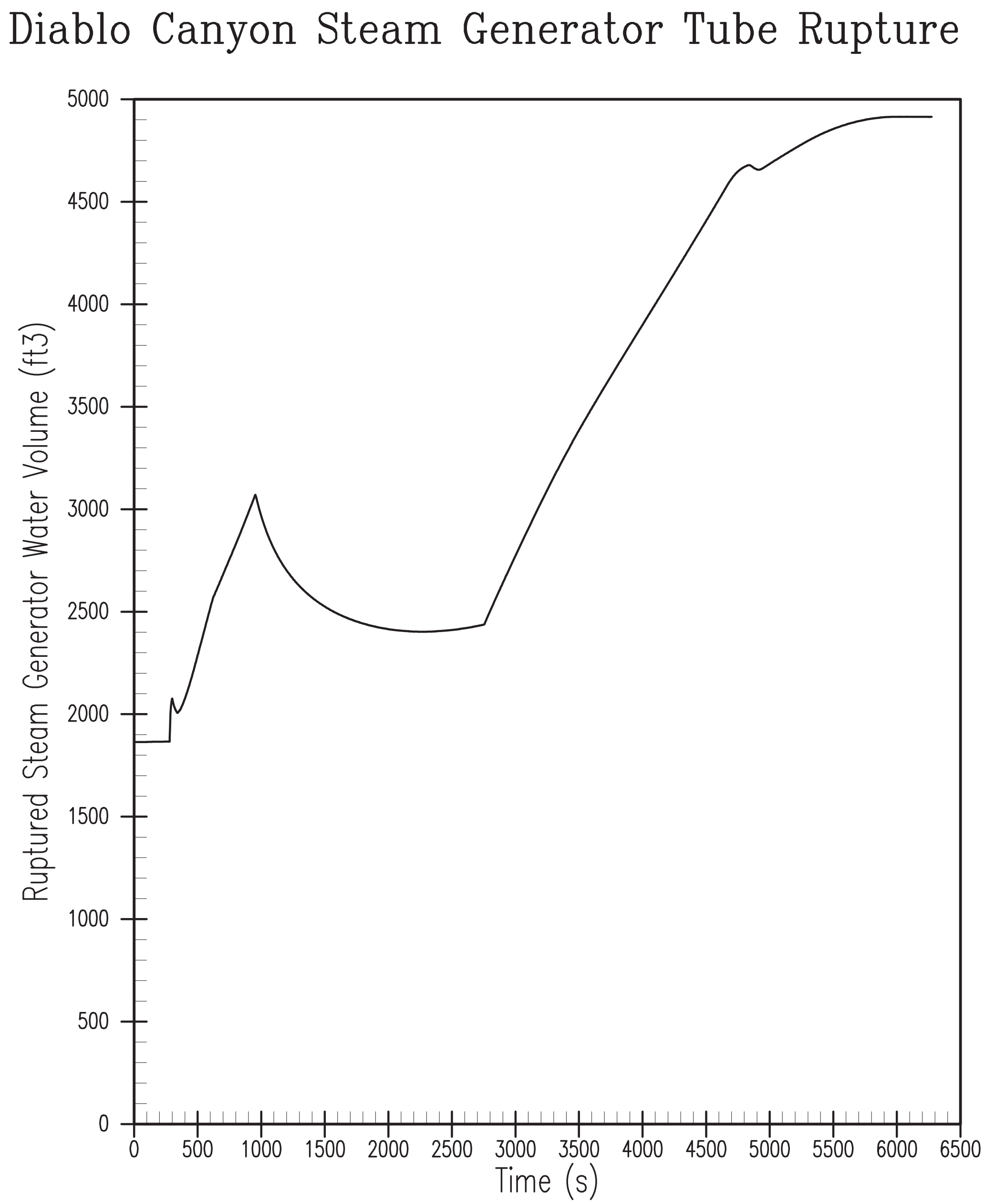
Revision 21 September 2013



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-7A
RUPTURED STEAM GENERATOR WATER VOLUME SGTR MTO ANALYSIS



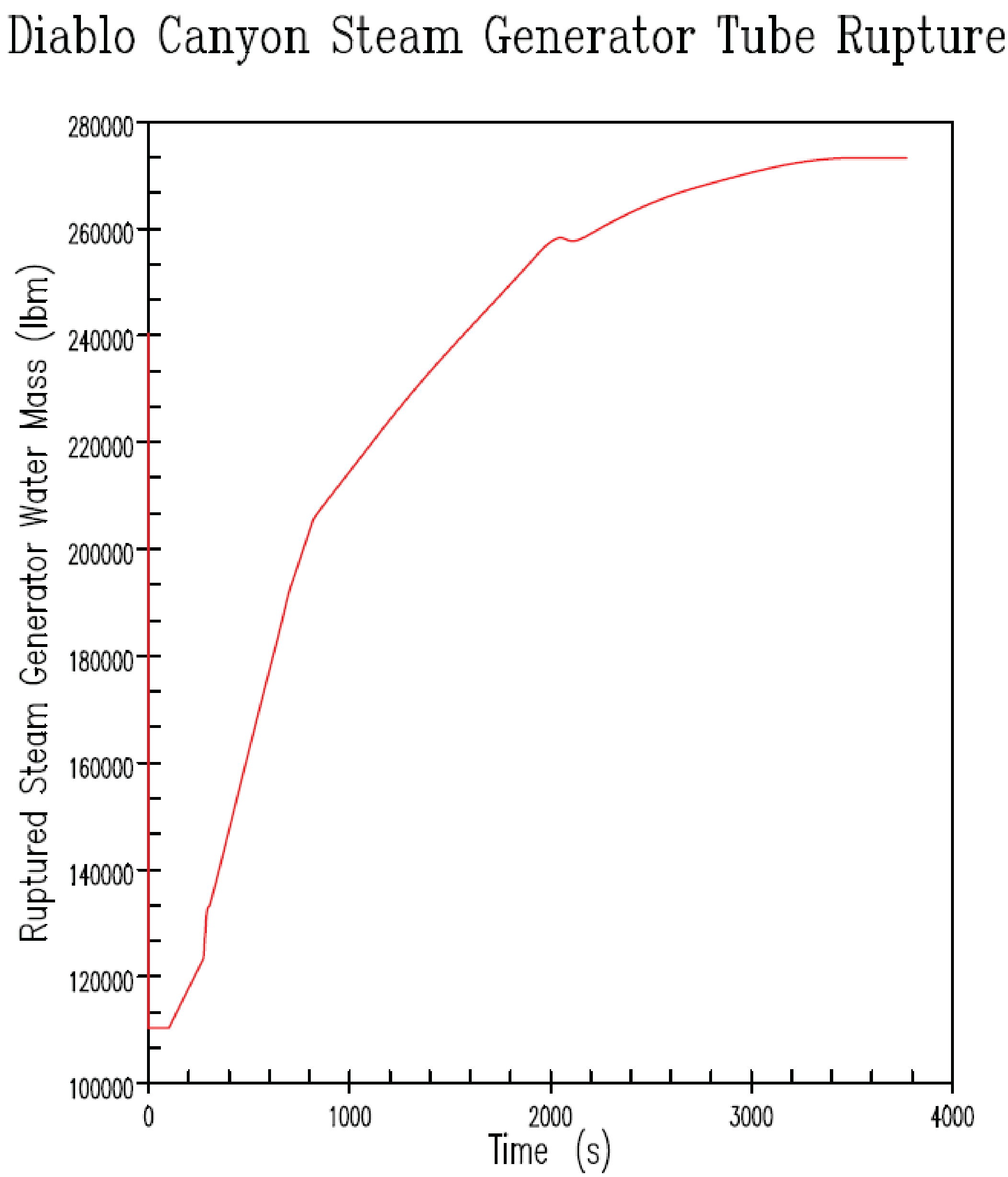
FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

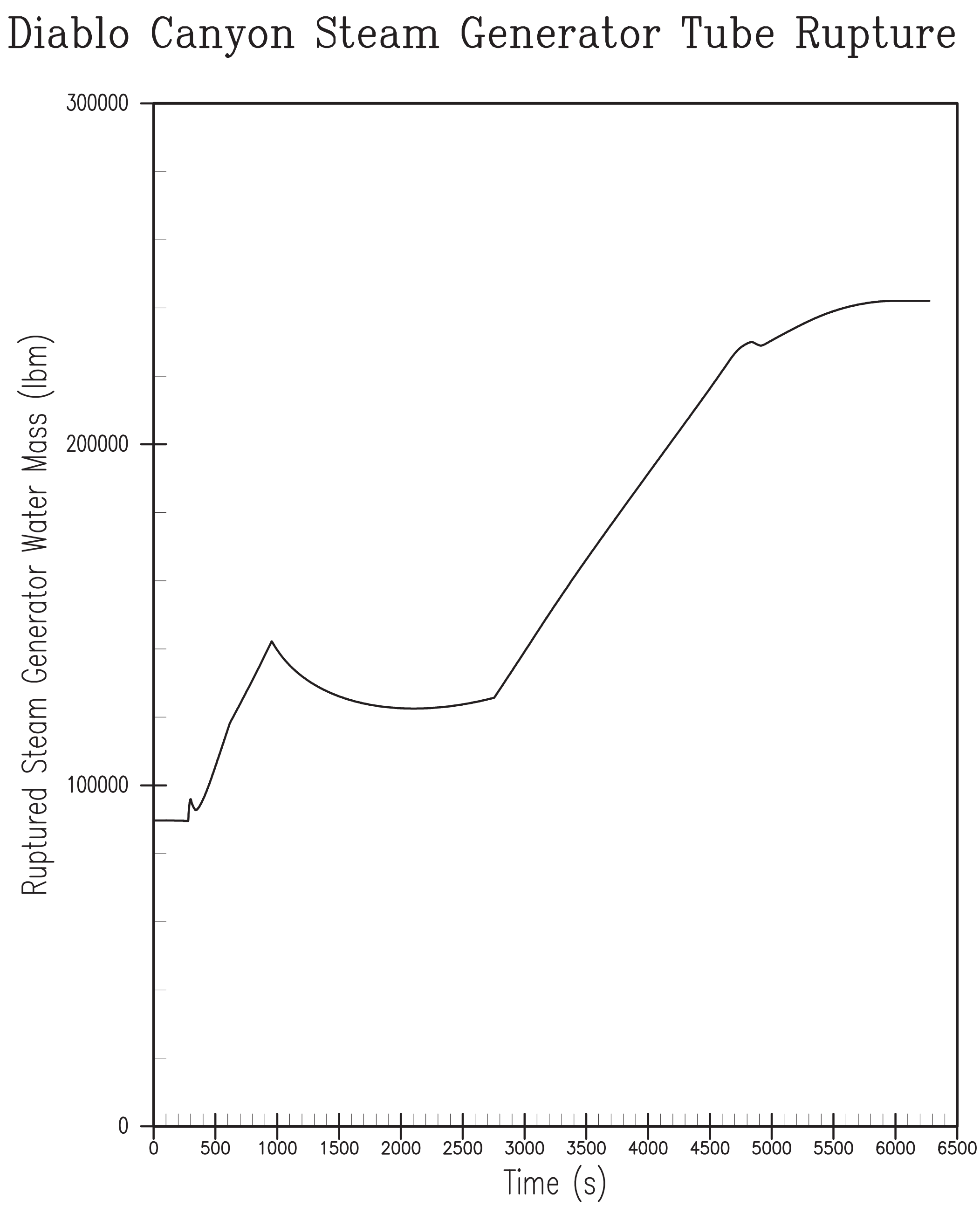
FIGURE 15.4.3-7B

RUPTURED STEAM GENERATOR WATER VOLUME SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013



FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-8A RUPTURED STEAM GENERATOR WATER MASS SGTR MTO ANALYSIS



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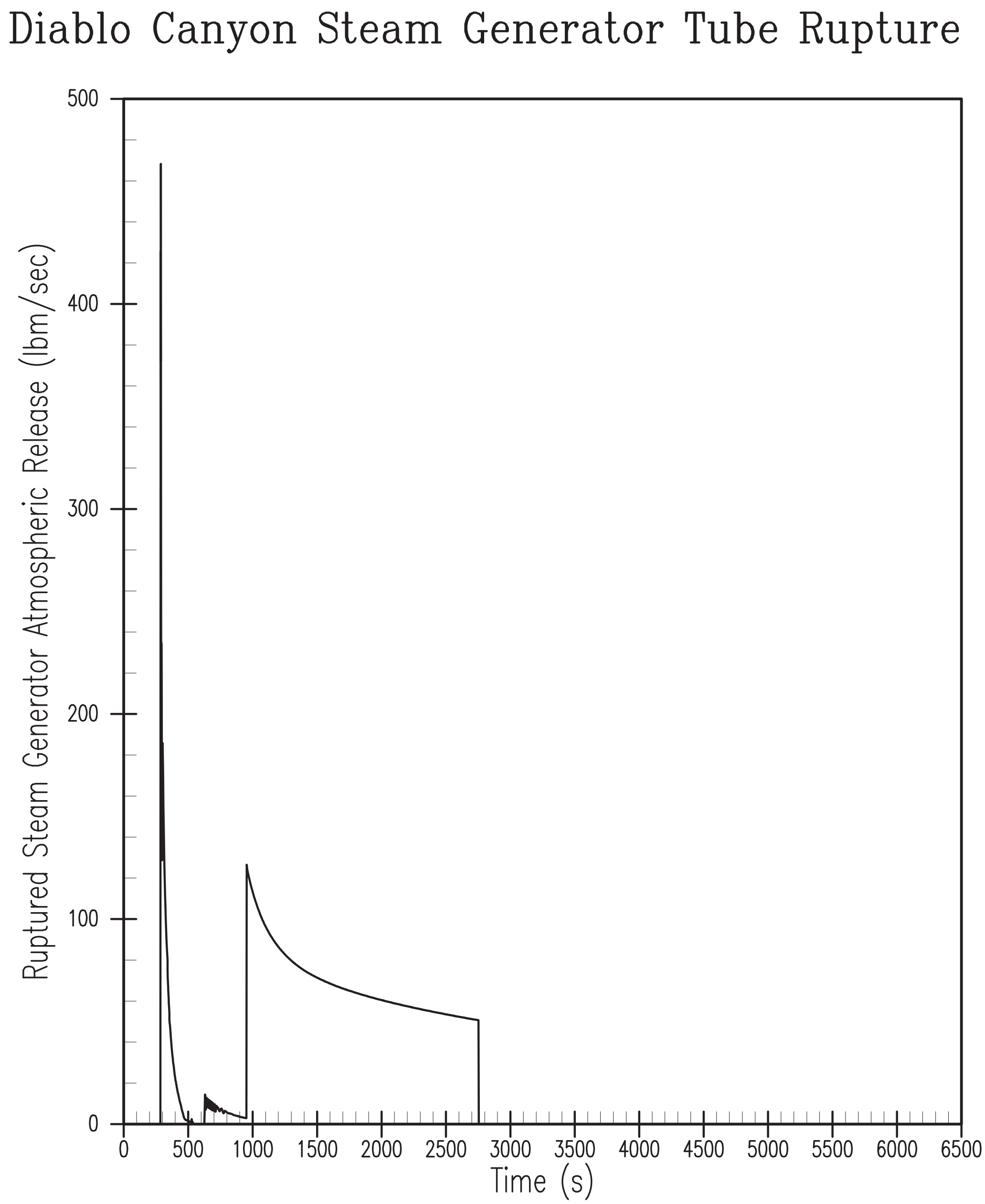
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-8B
RUPTURED STEAM GENERATOR WATER MASS
SGTR DOSE INPUT ANALYSIS

Revision 21 September 2013



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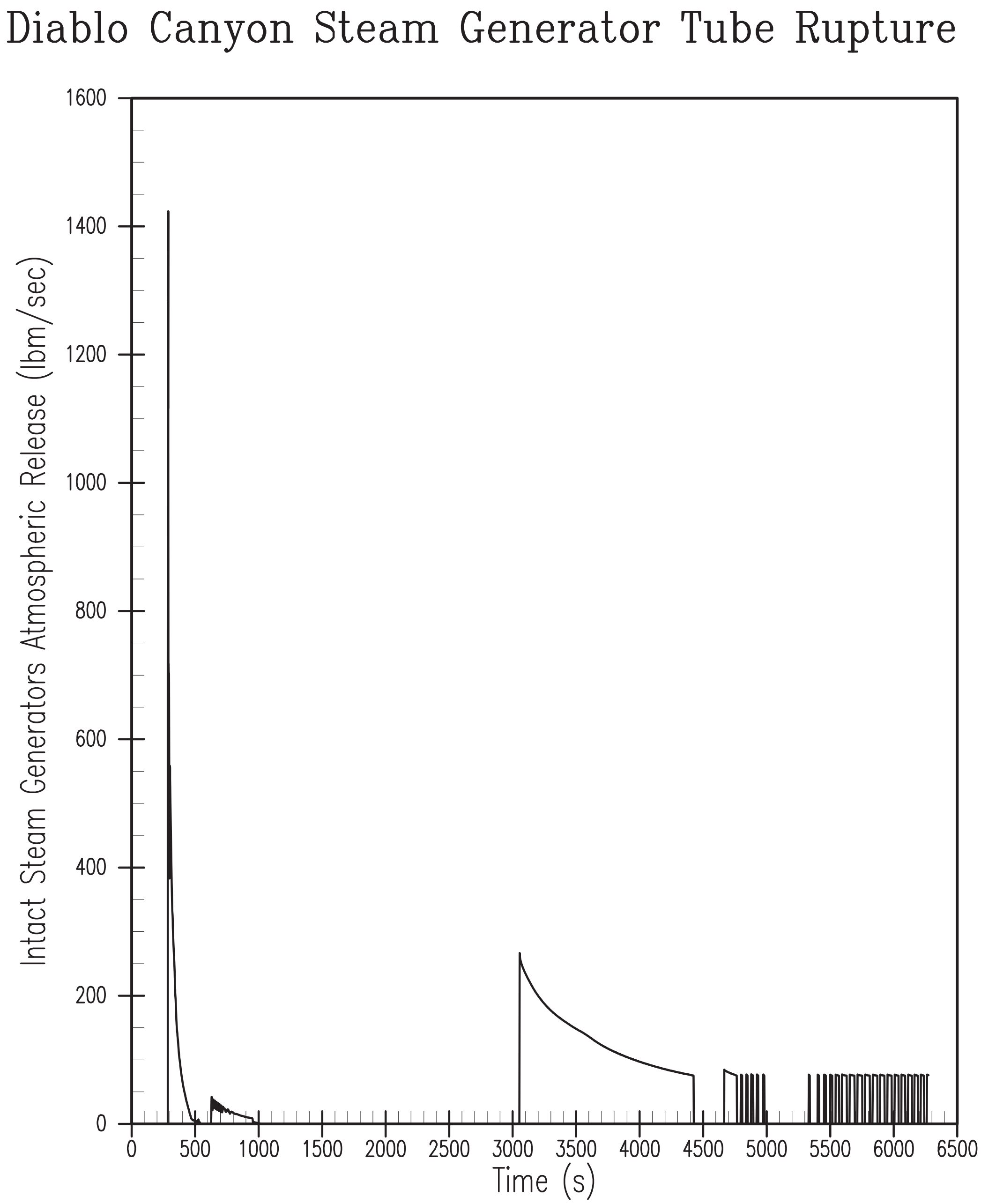
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-9 RUPTURED SG MASS RELEASE RATE TO THE ATMOSPHERE SGTR DOSE INPUT ANALYSIS



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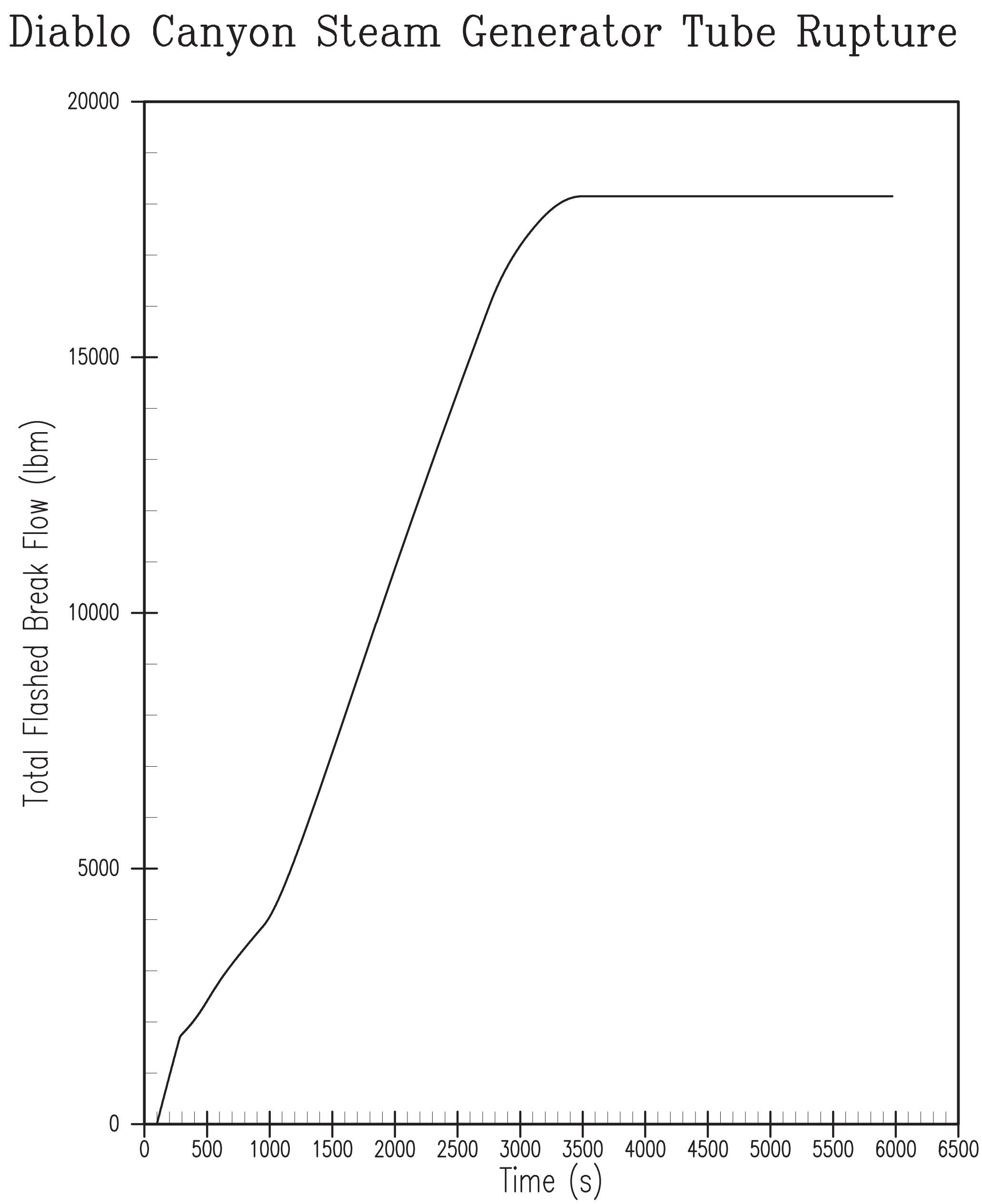
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.3-10
INTACT SGs MASS RELEASE RATE
TO THE ATMOSPHERE
SGTR DOSE INPUT ANALYSIS



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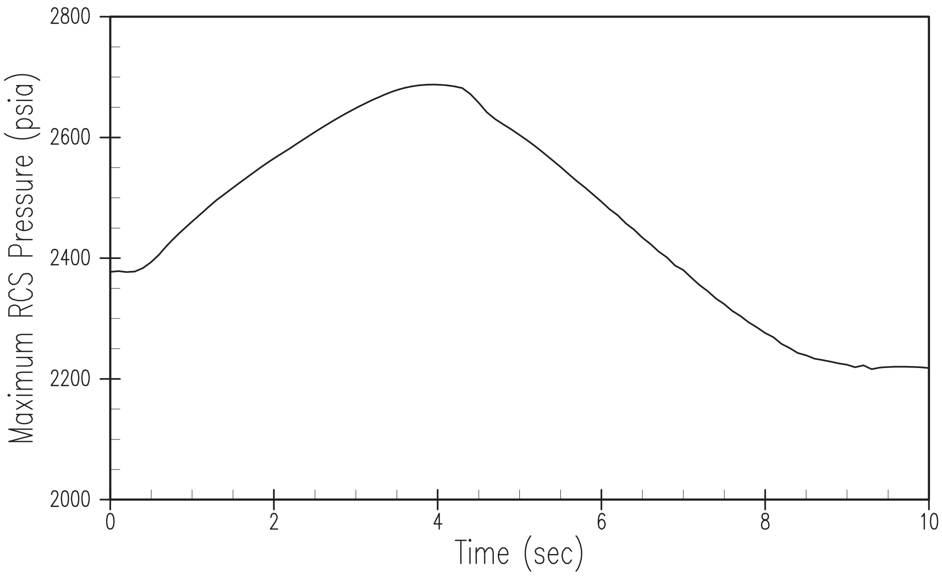
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.3-11 TOTAL FLASHED BREAK FLOW SGTR DOSE INPUT ANALYSIS



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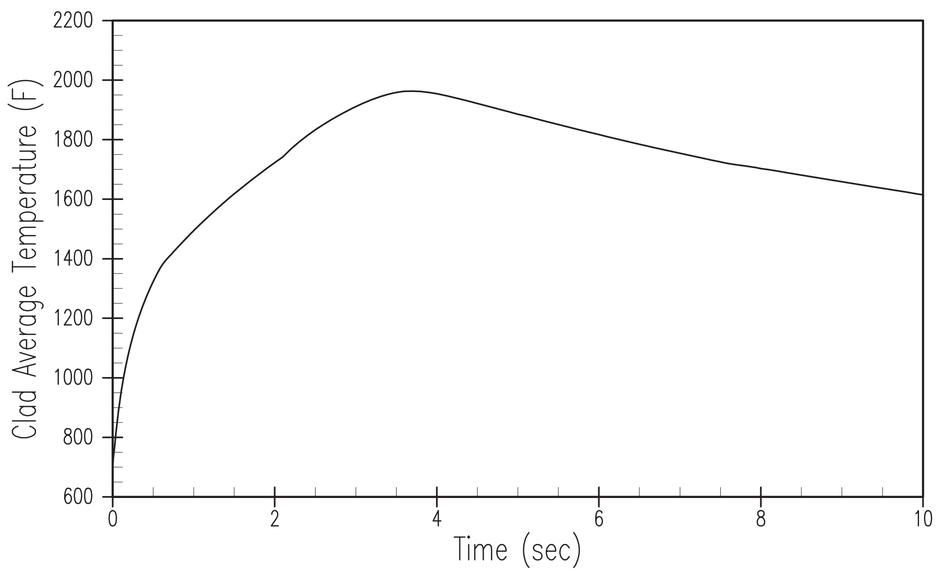
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.4-1 SINGLE REACTOR COOLANT PUMP LOCKED ROTOR MAXIMUM RCS PRESSURE VS. TIME



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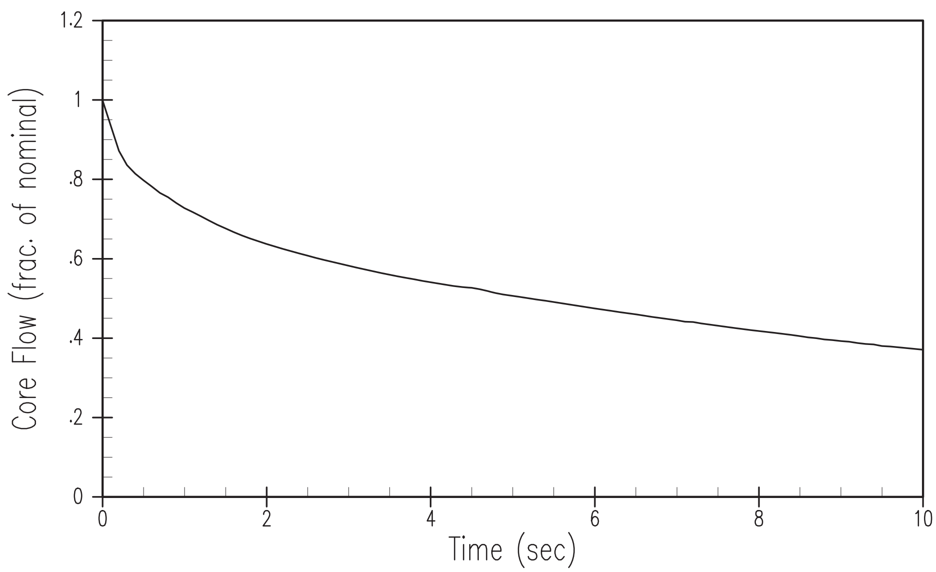
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.4-2
SINGLE REACTOR COOLANT PUMP
LOCKED ROTOR

CLAD AVERAGE TEMPERATURE VS. TIME



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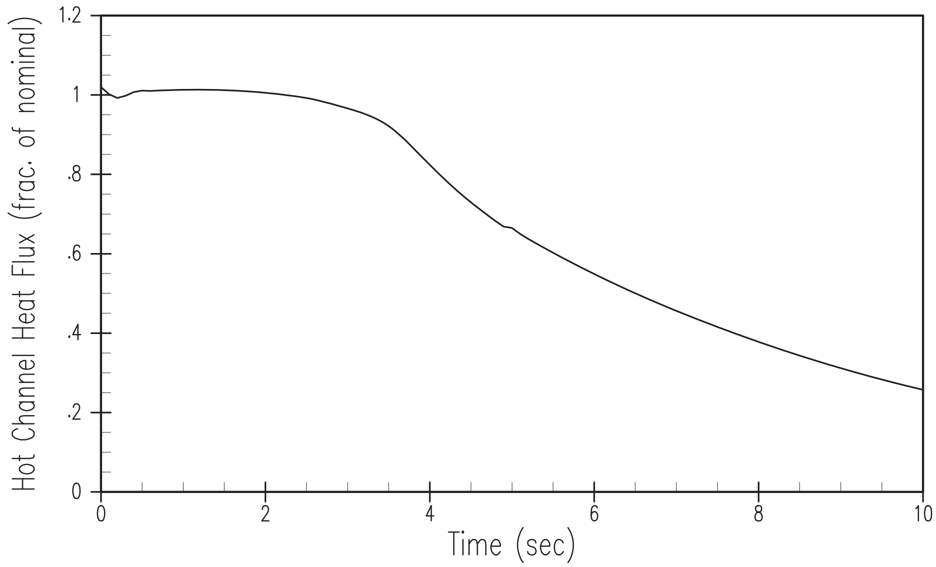
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.4-3
SINGLE REACTOR COOLANT PUMP
LOCKED ROTOR

CORE FLOW VS. TIME



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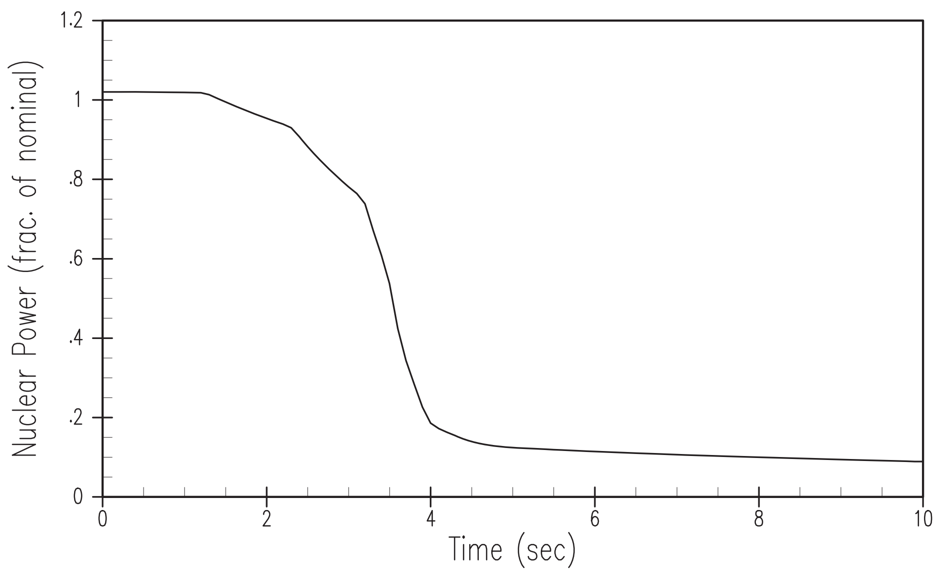
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.4-4 SINGLE REACTOR COOLANT PUMP LOCKED ROTOR HOT CHANNEL HEAT FLUX VS. TIME



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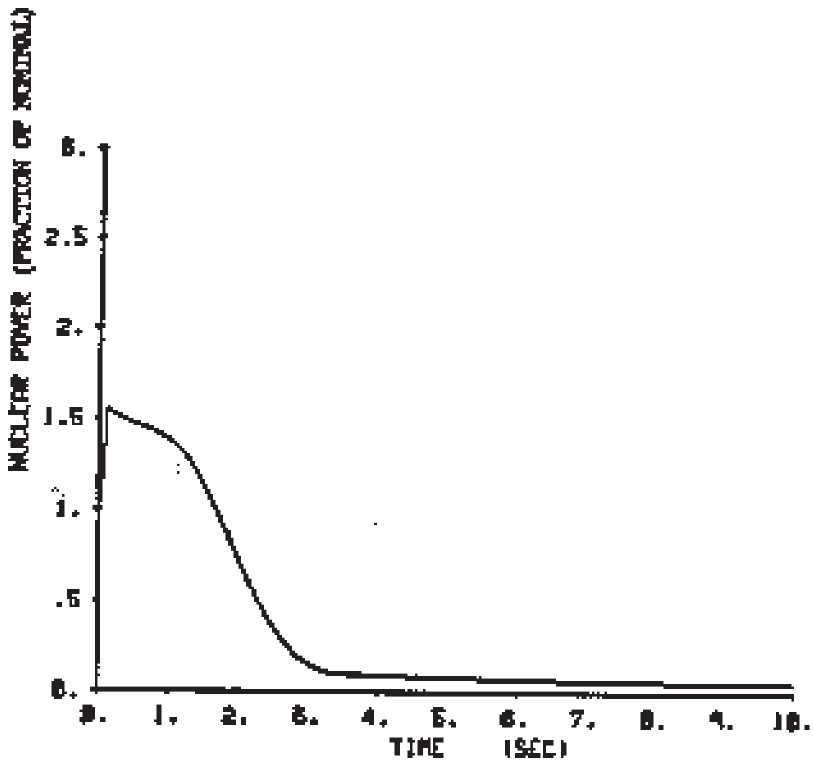
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.4-5
SINGLE REACTOR COOLANT PUMP
LOCKED ROTOR

NUCLEAR POWER VS. TIME



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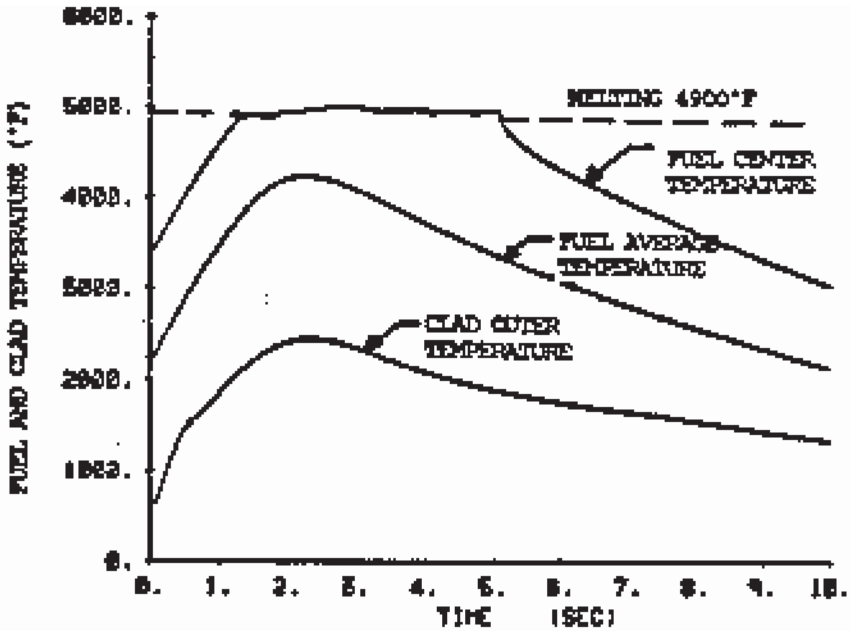
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 15.4.6-1 NUCLEAR POWER TRANSIENT, BOL, HZP, ROD EJECTION ACCIDENT



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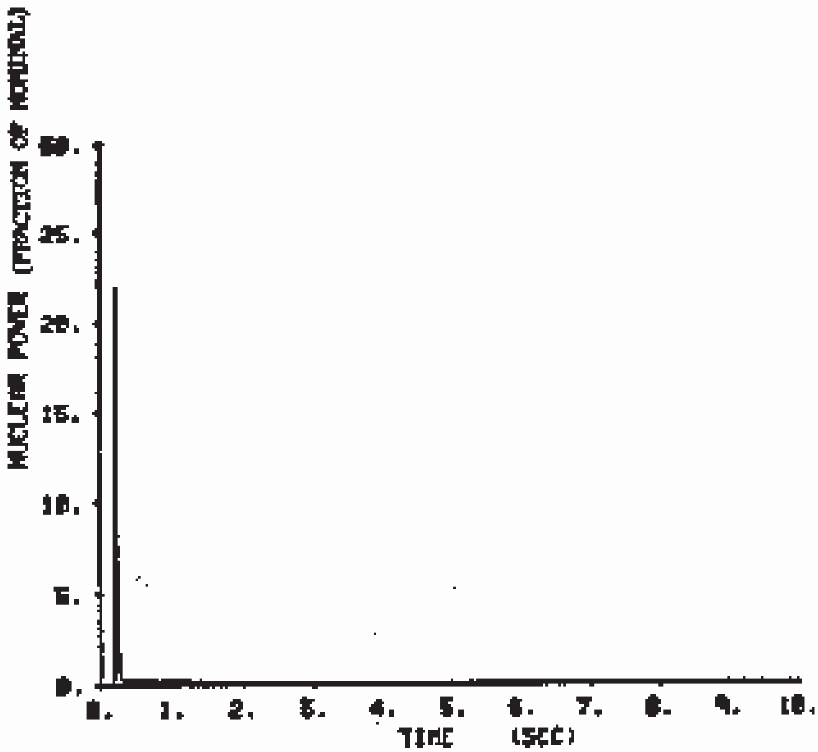
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.6-2
HOT SPOT FUEL AND CLAD TEMPERATURES VERSUS TIME, BOL, HZP, ROD EJECTION ACCIDENT



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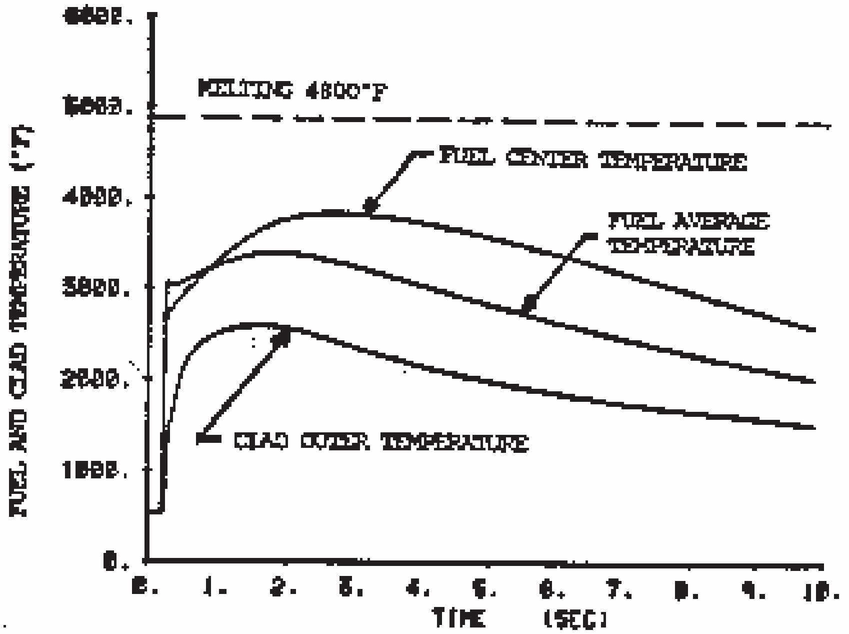
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.6-3
NUCLEAR POWER TRANSIENT, EOL, HFP, ROD EJECTION ACCIDENT



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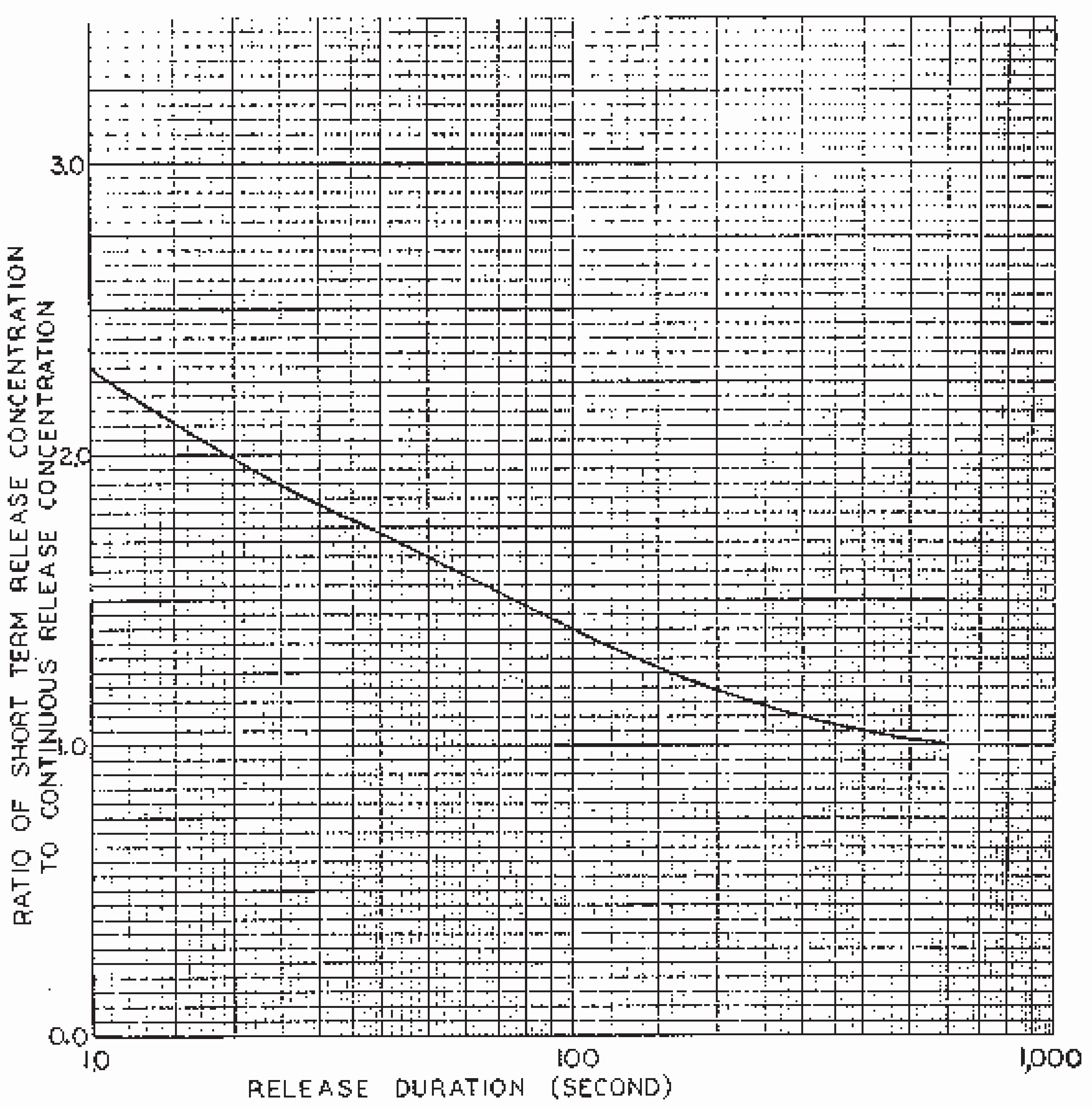
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 15.4.6-4
HOT SPOT FUEL AND CLAD TEMPERATURES VERSUS TIME, EOL, HZP, ROD EJECTION ACCIDENT



Ratio of Short-term Release Concentration to Continuous Release Concentration vs. Release Duration	DIABLO CANYON UNITS 1 & 2
	FIGURE 15.5-1

Revision 11 November 1996