Zion Station

Updated Final Safety Analysis Report

Volume 6

Commonwealth Edison Company

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15. SAFETY ANALYSES

15.0 GENERAL

This chapter presents an evaluation of the safety aspects of the plant and demonstrates that the plant can be operated safely even if highly unlikely events are postulated. It also shows that radiation exposures resulting from occurrences of these highly unlikely accidents do not exceed the guidelines of 10CFR100.

The chapter is divided into eight sections, each dealing with a different initiating event. The limiting case accident is considered to be the large break loss-of-coolant accident (LOCA) which is presented in Section 15.6.5.

15.0.1 Classification of Plant Conditions

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

1. Condition I - Normal Operation and Operational Transients;
2. Condition II - Faults of Moderate Frequency;
3. Condition III - Infrequent Faults; and
4. Condition IV - Limiting Faults

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

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle. This means that only seismic Category I, Class 1E, and IEEE qualified equipment, instrumentation, and components are used in the ultimate mitigation of the consequences of Condition II, III and IV events.

15.0.1.1 Condition I - Normal Operation and Operational Transients

Condition I events are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I events are accommodated with margin between any plant parameter and the value of that parameter which would

require either automatic or manual protective action. Inasmuch as Condition I events occur 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 described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is given below:

1. Steady state and shutdown operations:

    a. Power operation;

    b. Startup;

    c. Hot standby;

    d. Hot shutdown;

    e. Cold shutdown; and

    f. Refueling.

2. Operation with permissible deviations - Various deviations which may occur during continued operation as permitted by the Technical Specifications 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 limited clad defects;

    c. Excessive radioactivity in the reactor coolant;

    d. Operation with steam generator leaks; and

    e. Testing.

3. Operational transients:

    a. Plant heatup and cooldown;

    b. Step load changes (up to ± 10%);

    c. Ramp load changes (up to 5%/minute); and

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

15.0.1.2 Condition II - Faults of Moderate Frequency

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

1. Uncontrolled rod cluster control assembly (RCCA) withdrawal from a subcritical condition;

2. Uncontrolled RCCA withdrawal at power;

3. RCCA misalignment (dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly);

4. Chemical and Volume Control System (CVCS) malfunction;

5. Partial loss of reactor coolant flow;

6. Startup of an inactive reactor coolant loop;

7. Loss of external electrical load;

8. Loss of normal feedwater;

9. Excessive heat removal due to feedwater system malfunctions;

10. Excessive load increase;

11. Loss of all non-emergency ac power to the plant auxiliaries; and

12. Inadvertent opening of a steam generator relief or safety valve.

15.0.1.3 Condition III - Infrequent Faults

By definition, Condition III events are faults which may occur very infrequently during the life of the plant. They will be accommodated 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 event will not, by itself, generate a Condition IV event or result in a consequential loss of function of the RCS or containment barriers. The following faults are included in this category:

1. Steam system piping failure (minor);

2. Complete loss of reactor coolant flow;

3. LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break);

4. Radioactive gaseous waste system leak or failure;

5. Radioactive liquid waste system leak or failure; and

6. Postulated radioactive releases due to liquid tank failures.

Note that the complete loss of reactor coolant flow and the minor steamline ruptures are conservatively analyzed to Condition II acceptance limits.

15.0.1.4 Condition IV - Limiting Faults

Condition I events are faults which 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. They are the most drastic which must be designed against and represent limiting design cases. Condition IV events are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10CFR100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System (ECCS) and the Containment. The following faults are included in this category:

1. Steam system pipe break;

2. Reactor coolant pump shaft seizure (locked rotor);

3. Spectrum of RCCA ejection accidents;

4. Steam generator tube rupture;

5. LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break); and

6. Design basis fuel handling accidents.

Note that the steam system pipe break event is conservatively analyzed to Condition II acceptance limits.

15.0.2 Optimization of Control Systems

A control system setpoint study is performed in order to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and equipment performance.

For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for various levels of power operation.

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

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed in analyses. Two ratings are given:

1. The Nuclear Steam Supply System (NSSS) design thermal power rating of 3250 MWt.

2. The engineered safety features design rating of 3396 MWt. The NSSS-supplied engineered safety features (ESFs) are designed for thermal output higher than the NSSS design value in order not to preclude realization of future potential power capability. This higher thermal power value is designed as the ESF design rating.

Where the initial power operating conditions are assumed in the accident analyses, the NSSS thermal power output value is assumed. Where the demonstration of adequacy of an ESF (e.g., auxiliary feedwater) is concerned, the ESF design rating is assumed unless noted otherwise. Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The NSSS thermal power value used for each transient analyzed is given in Table 15.0-2B. In all cases where the 3396 MWt rating is used in an analysis, the resulting transient and results are conservative when compared to using the 3250 MWt rating.

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

15.0.3.2 Initial Conditions

For most accidents which are departure from nucleate boiling (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 DNB ratio (DNBR) as described in Reference 1. The values specific to Zion Station are documented in Reference 2. This procedure is known as the Revised Thermal Design Procedure (RTDP) and these accidents utilize the WRB-1 DNB correlations (Reference 3). RTDP allowances may be more restrictive than non-RTDP 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 RTDP transients. This flow accounts for a 3.5% allowance for calorimetric uncertainty including an allotment for feedwater venturi fouling.

For accidents which are not DNB limited, or for which the RTDP is not employed, the WRB-1 DNB correlation is used when coolant conditions are within the range of the correlation; otherwise the W-3 DNB correlation is used. The initial conditions are obtained by adding the maximum steady-state errors to rated values in such a manner to maximize the impact on the limiting parameter. Table 15.0-7 lists the conservative steady-state errors that were assumed in the analysis of these accidents.

Tables 15.0-2A and 15.0-2B summarize the principal initial conditions, computer codes used, DNB correlations, and thermal hydraulic methods. Other accident specific initial conditions are given in those sections describing the accident. A maximum level of 17% steam generator tube plugging was assumed for each transient listed in Table 15.0-2A. The 17% limit applies to each steam generator as well as the plant average. Using 17% steam generator tube plugging in the non-LOCA safety analyses is conservative compared with the plant maximum steam generator tube plugging allowance of 15%.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies, control rods, and operating instructions. Power distribution may be characterized by the radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits used in the fuel transition analyses are listed below:

F_Q (total peaking factor)	=	2.40
$F_{\Delta H}$ (enthalpy rise hot channel factor)	=	1.65

For transients which may be DNB limited the radial peaking factor is of importance. The radial peaking factor increases with decreasing power

level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the design thermal power level.

For transients which may be overpower limited, F_Q is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with or conservative with respect to reactor operation as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, for example, the CVCS malfunction event which lasts many minutes, and the excessive load increase event which may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4. For overpower transients which are fast with respect to the fuel rod thermal time constant, for example, the uncontrolled RCCA bank withdrawal from subcritical and RCCA ejection incidents which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation must be performed.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. The moderator feedback is modeled in the analyses in terms of a density coefficient. For the beginning of life (or minimum feedback) analyses, a moderator density coefficient consistent with Figure 15.0-2 is used. For the end of life (or maximum feedback) cases, a constant moderator density coefficient of 0.40 $\Delta K/g/cc$ is used.

In the analysis of certain events, conservatism requires the use of large (absolute value) reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small (absolute value) reactivity coefficient values. The values are given in Tables 15.0-2A and 15.0-2B. Figure 15.0-3 shows the upper and lower bound Doppler power coefficients as a function of power that are used in the safety analyses. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life although these combinations may not represent possible realistic situations.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position of the RCCAs versus time 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% of the rod cluster travel.

The RCCA position versus time assumed in accident analyses is shown in Figure 15.0-4. The RCCA insertion time to dashpot entry is taken as 2.4 seconds. This time is bounding for low parasitic (LOPAR), Optimized Fuel Assembly (OFA), and VANTAGE 5 fuel. Drop time testing requirements are specified in the Technical Specifications.

Figure 15.0-5 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in the safety analyses.

There is inherent conservatism in the use of Figure 15.0-5 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figures 15.0-4 and 15.0-5. A total negative reactivity insertion following a trip of 4% $\Delta K/K$ is assumed in the safety analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. For Figures 15.0-4 and 15.0-5, the RCCA drop time is normalized to 2.4 seconds.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0-6) is used in those transient analyses for which a point-kinetics core model is used. Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0-4 is used as code input.

15.0.6 Trip Points and Time Delays to Trip Assumed in the Accident Analyses

A reactor trip signal acts to open two trip breakers, connected in series, which feed power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation,

in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached until the time at which the rods are free to fall into the core. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4A.

Reference is made in that table to the overtemperature ΔT and overpower ΔT trips shown in Figure 15.0-1. This figure presents the allowable reactor coolant loop 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 trip would occur well within the area bounded by these lines. The utility of this diagram is 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 for RTDP accidents. 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 area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); and overpower ΔT and overtemperature ΔT (variable setpoints).

The limit values, which were used as the DNBR limits for all accidents analyzed with the RTDP (see Table 15.0-2A), 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. Nominal trip setpoints are specified in the Technical Specifications. The protection system channels are calibrated periodically in accordance with the Technical Specifications.

In summary, the Reactor Protection System (RPS) is designed to prevent cladding damage for Condition II events and excessive dose releases for Condition III and IV events. Coincidence of two out of three (or two out of four) signals is required where a single channel malfunction could cause a spurious trip while at power. A single component failure in the RPS itself coincident with one stuck RCCA is permissible as a contingent failure and does not result in a violation of the protection criteria.

15.0.7 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

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

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (made equal) to this measured power on a periodic basis. The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow. The nuclear instrumentation system is calibrated by comparing individual power level indications with the power obtained calorimetrically. The error assumed in the calorimetric determination of power level (for the purposes of establishing the maximum overpower trip setpoint) is given in Table 15.0-5.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

In the accident analyses, control system operation is only considered if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the effects of an accident. For some events, the computer analysis is performed both with and without control system operation to determine the worst case. If a control system does operate, no malfunctions are considered. Note that pressurizer heaters are not assumed in any of accident analyses.

The reactor trip setpoint response time delay assumptions for the VANTAGE 5 non-LOCA safety analyses and the ESF Actuation System response time assumptions for the non-LOCA, LOCA and Containment integrity analysis are given in Tables 15.0-4A and 15.0-4B, respectively.

15.0.9 Residual Decay Heat

Residual heat in a subcritical core is calculated for the LOCA per the requirements of 10CFR50.46, Appendix K (Reference 4). These requirements include the assumption of infinite irradiation time before the core goes subcritical to determine the amount of fission product decay energy.

For all other accidents (e.g., loss of normal feedwater), unless otherwise noted in the text, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10 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 (e.g., RCS pipe break), are discussed in their respective accident analysis section. The codes used in the analyses of each transient are listed in Table 15.0-2A.

15.0.10.1 FACTRAN

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

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

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

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

FACTRAN is further discussed in Reference 5.

15.0.10.2 LOFTRAN

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

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

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

LOFTRAN is further discussed in Reference 6.

15.0.10.3 TWINKLE

The TWINKLE program is a multi-dimensional 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 multi-region fuel-clad-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 are provided, e.g., channel-wise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

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

TWINKLE is further discussed in Reference 7.

15.0.10.4 THINC

The THINC Code is described in Section 4.4.4.2.

15.0.10.5 LOCA Computer Codes

The computer codes used to analyze the large and small break LOCAs are described in Section 15.6.5.

15.0.11 Limiting Single Failures

The most limiting single failure of safety-related equipment, where one exists, is identified in each analysis description and the consequences of this failure are described therein. In some instances, because of the redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed on Table 15.0-6.

15.0.12 Operator Actions

For most of the events analyzed in this chapter, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than the normal operating procedures. The exact actions taken, and the time these actions would occur, will depend on what systems are available and plans for further plant operation. Operator action is only relied upon for mitigation in those accidents that explicitly identify that the action has been credited.

15.0.13 References, Section 15.0

1. Friedland, A.J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

2. Letter ULNRC-1227, 12/13/85, to H. Denton from D. Schnell, Additional Information Addressing Plant-Specific Items in the NRC SER on WCAP-9500.

3. Motley, F.E., et al., "New Westinghouse Correlations WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A and WCAP-8763-A, July 1984.

4. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3, January 4, 1974.

5. Hargrove, H.G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December, 1989.

6. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

7. Risher, D.H., Jr. and Barry, R.F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A and WCAP-8028-A, January 1975.

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

NSSS Design Thermal Power Rating (MWt)	3250
ESF design rating (maximum calculated turbine rating) (MWt)	3396

TABLE 15.0-2A (1 of 3)

SUMMARY OF COMPUTER CODES, INITIAL CONDITIONS, DNB CORRELATIONS AND THERMAL-HYDRAULIC METHODS

FAULTS	COMPUTER CODES UTILIZED	MODERATOR DENSITY ($\Delta\rho$/g/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
1. Increase in Heat Removal by the Secondary System					
• Excessive Heat Removal Due to Feedwater System Malfunctions	LOFTRAN, THINC	Figure 15.0-2 and 0.40*	Upper Curves of Figure 15.0-3	WRB-1	Yes
• Excessive Load Increase	LOFTRAN	0.40*	Upper and Lower Curves of Figure 15.0-3	WRB-1	Yes
• Accidental Depressurization of the Main Steam System	LOFTRAN	Function of Moderator Density, See Subsection 15.1.4 (Figure 15.0-2)	See Figure 15.0-3	W-3	No
• Major Secondary System Pipe Rupture	THINC, LOFTRAN	Function of Moderator Density, See Subsection 15.1.5 (Figure 15.0-2	See Figure 15.0-3	W-3	No

TABLE 15.0-2A (2 of 3)

SUMMARY OF COMPUTER CODES, INITIAL CONDITIONS, DNB CORRELATIONS AND THERMAL-HYDRAULIC METHODS

FAULTS	COMPUTER CODES UTILIZED	MODERATOR DENSITY ($\Delta\rho$/g/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
2. Decrease in Heat Removal by the Secondary System					
• Loss of External Electrical Load	LOFTRAN	Upper and Lower Curves of Figure 15.0-3	Maximum and Minimum**	WRB-1	Yes
• Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	Figure 15.0-2*	Maximum**	NA	No
• Loss of Normal Feedwater	LOFTRAN	Figure 15.0-2*	Maximum**	NA	No
3. Decrease in Reactor Coolant System Flow Rate					
• Single and Multiple Reactor Coolant Pump Trips	LOFTRAN, THINC, FACTRAN	Figure 15.0-2*	Maximum**	WRB-1	Yes
• Reactor Coolant Pump Locked Rotor	LOFTRAN, FACTRAN, THINC	Figure 15.0-2*	Maximum**	NA	No

TABLE 15.0-2A (3 of 3)

SUMMARY OF COMPUTER CODES, INITIAL CONDITIONS, DNB CORRELATIONS AND THERMAL-HYDRAULIC METHODS

FAULTS	COMPUTER CODES UTILIZED	MODERATOR DENSITY ($\Delta\rho$/g/cc)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE
4. Reactivity and Power Distribution Anomalies					
• Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition	TWINKLE, FACTRAN, THINC	Refer to Subsection 15.4.1.2	Consistent with upper limit shown on Figure 15.0-2**	WRB-1/W-3	No
• Uncontrolled Rod Cluster Assembly Withdrawal at Power	LOFTRAN	Figure 15.0-2* and 0.40	Maximum and Minimum**	WRB-1	Yes
• Rod Cluster Control Assembly Misalignment	LOFTRAN, THINC	NA	NA	WRB-1	Yes
• Startup of an Inactive Reactor Coolant Loop	LOFTRAN, FACTRAN, THINC	0.40	Minimum**	WRB-1	No
• Chemical and Volume Control System Malfunction	NA	NA	NA	NA	NA
• Rod Cluster Control Assembly Ejection	TWINKLE, FACTRAN	Refer to Subsection 15.4.8 min., max. feedback	Refer to Subsection 15.4.8 min., max feedback	NA	NA

* The analysis includes the effects of a +7 pcm/°F positive moderator temperature coefficient (see Figure 15.0-2)

** See Figure 15.0-3, "maximum" refers to lower curve and "minimum" refers to upper curve.

TABLE 15.0-2B (1 of 3)

SUMMARY OF INITIAL CONDITIONS

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (gpm)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (psia)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (°F)
1. Increase in Heat Removal by the Secondary System						
• Excessive Heat Removal due to Feedwater System Malfunctions	0* and 3250	357,000	547 and 562.2	2250	618.8 and 802.1	32 and 428.6
• Excessive Load Increase	3250	357,000	562.2	2250	802.1	428.6
• Accidental Depressurization of the Main Steam System	0 (Subcritical)	350,000	547	2250	618.8	100
• Major Secondary System Pipe Rupture	0* (Subcritical)	350,000	547	2250	736.53	100
2. Decrease in Heat Removal by the Secondary System						
• Loss of External Electrical Load	3250	357,000	562.2	2250	938	428.6
• Loss of Offsite Power to the Station Auxiliaries	3464	350,000	569.1	2290	927	433.8
• Loss of Normal Feedwater	3464	350,000	569.1	2290	927	433.8

TABLE 15.0-2B (2 of 3)

SUMMARY OF INITIAL CONDITIONS

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (gpm)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (psia)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (°F)
3. Decrease in Reactor Coolant System Flow Rate						
• Single and Multiple Reactor Coolant Pump Trips	3250	357,000	562.2	2250	802.1	428.6
• Reactor Coolant Pump** Locked Rotor	3250/3315	357,000/ 350,000	562.2/568.0	2250/2290	802.1	428.6
4. Reactivity and Power Distribution Anomalies						
• Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition	0	161,000	547.0	2210***	NA	NA
• Uncontrolled Rod Cluster Assembly Withdrawal at Power	3250/1950/325	357,000	562.2/556.1/ 548.5	2250	802.1/728.8/ 637.1	428.6/375.0/ 262.0
• Rod Cluster Control Assembly Misalignment	3250	357,000	562.2	2270***	NA	NA
• Startup of an Inactive Reactor Coolant Loop	2437	279,000	564.2	2210	760	400

TABLE 15.0-2B (3 of 3)

SUMMARY OF INITIAL CONDITIONS

FAULTS	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (gpm)	VESSEL AVERAGE TEMPERATURE (°F)	PRESSURIZER PRESSURE (psia)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (°F)
4. Reactivity and Power Distribution Anomalies (Cont'd)						
• Chemical and Volume Control System Malfunction	0/162.5/3250	NA	140/553.26/ 567.7	14.7/2250/ 2250	NA	NA
• Rod Cluster Control Assembly Ejection	0 and 3250	161,000 and 350,000	547 and 567.7	2210	NA	NA

* Due to LOFTRAN code limitations, these transients are initialized at 1% power.
** Separate cases are examined for MDNBR and overpressurization limits.
*** Core pressure.

TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN THE ACCIDENT ANALYSES

	Non-RTDP	RTDP
NSSS Thermal Power (MWt)	3250	3250
HFP Core Inlet Temperature (°F)	530.2	530.7
HFP Vessel Average Temperature (°F)	562.2	562.2
Reactor Coolant System Pressure (psia)	2250	2250
Reactor Coolant Flow per Loop		
Thermal Design Flow (gpm)	87,500	N/A
Minimum Measured Flow (gpm)	N/A	89,250
Total Reactor Coolant Flow (10^6 lb/hr)		
Thermal Design Flow	135.0	N/A
Minimum Measured Flow	N/A	137.7
Steam Flow from NSSS (10^6 lb/hr)	13.98	13.98
Steam Pressure at Steam Generator Outlet (psia)	701	701
Maximum Steam Moisture Content (%)	0.25	0.25
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	428.6	428.6
Average Core Heat Flux (Btu/hr-ft^2)	207,410	207,410
Steam Generator Tube Plugging Level (plant average per SG) non-LOCA	17%	17%
Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)	1.65	1.59
Total Peaking Factor (F_Q)	2.40	2.40

TABLE 15.0-4A (1 of 2)

REACTOR TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN NON-LOCA ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delay (seconds)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Overtemperature ΔT	Variable, see Figure 15.0-1	6.0*
Overpower ΔT	Variable, see Figure 15.0-1	6.0*
High pressurizer pressure	2425 psia	2.0
Low pressurizer pressure	1800 psi **	2.0
High Neutron Flux, P-8 for 3-loop operation+	84% of rated power	0.5
Low reactor coolant flow (From loop flow detectors)	85% loop flow	1.0
Undervoltage trip	See note ++	1.2
Underfrequency trip	56.5 Hz	0.6
Low-low steam generator level	0% of narrow range level span	2.0
Turbine trip	Not applicable	2.5
High steam generator level trip on the feedwater pumps and closure of feedwater system valves, and turbine trip	80.0% of narrow range level span	2.5
Safety injection system reactor trip delay	Not applicable	2.0

* Total time delay (including resistance temperature detector (RTD) bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay; 2 seconds pure time delay and 4 seconds lag time constant) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

** Does not account for environmental allowance.

TABLE 15.0-4A (2 of 2)

REACTOR TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN NON-LOCA ACCIDENT ANALYSES

+ P-8 trip is not credited in any safety analysis and Zion Station is currently not licensed for 3-loop operation.

++ Analysis assumes reactor trip to initiate event; therefore, no specific setpoint value is assumed.

Note: No other reactor trips are explicitly modeled in the accident analyses.

TABLE 15.0-4B

ENGINEERED SAFEGUARDS ACTUATION SYSTEM
RESPONSE TIMES ASSUMED IN THE VANTAGE 5 NON-LOCA, LOCA, AND CONTAINMENT INTEGRITY ANALYSES

Non-LOCA Assumptions:

Time Delay Functions	Time Delay Assumptions (sec)
Main feedwater isolation valve closure time	10.0
Main steamline isolation valve closure time	7.0
Delay to get auxiliary feedwater (AFW) pumps to speed plus sensor and logic delay	60.0
Delay from safety injection (SI) actuation to pump start with offsite power	27.0
Delay from SI actuation to pump start without offsite power	47.0

LOCA Assumptions:

Time Delay Functions	Time Delay Assumptions (sec)
ECCS with loss of offsite power (Diesel start time delay = 15 seconds)	32.0[1]
AFW with loss of offsite power (From reactor trip signal to start of AFW pumps)	75.0
Reactor containment fan cooler (RCFC) actuation	38.0
Containment spray actuation	45.0

Containment Response analysis: (Appendix 15D)

Time Delay Functions	Time Delay Assumptions (sec)
RCFC actuation [2]	58.0
Containment spray actuation [3]	110.0

[1] Although the design basis time delay is 32.0 seconds, the original VANTAGE 5 LOCA analysis assumed 31.0 seconds. See Section 6.3.3.3 for discussion of this difference.

[2] Time delay following containment High pressure setpoint.

[3] Time delay following containment High-High pressure setpoint.

TABLE 15.0-5

INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS
FOR POWER RANGE NEUTRON FLUX

	Setpoint and Error Allowance (% of Rated Power)
Nominal Setpoint	109
Calorimetric Error	1.7
Water density effects in the downcomers and radial power redistribution	4.2
Rack drift and temperature effects	1.5
Calibration accuracy	1.18
Maximum overpower trip point assuming all individual errors are simultaneously applied in the most adverse direction	118

TABLE 15.0-6

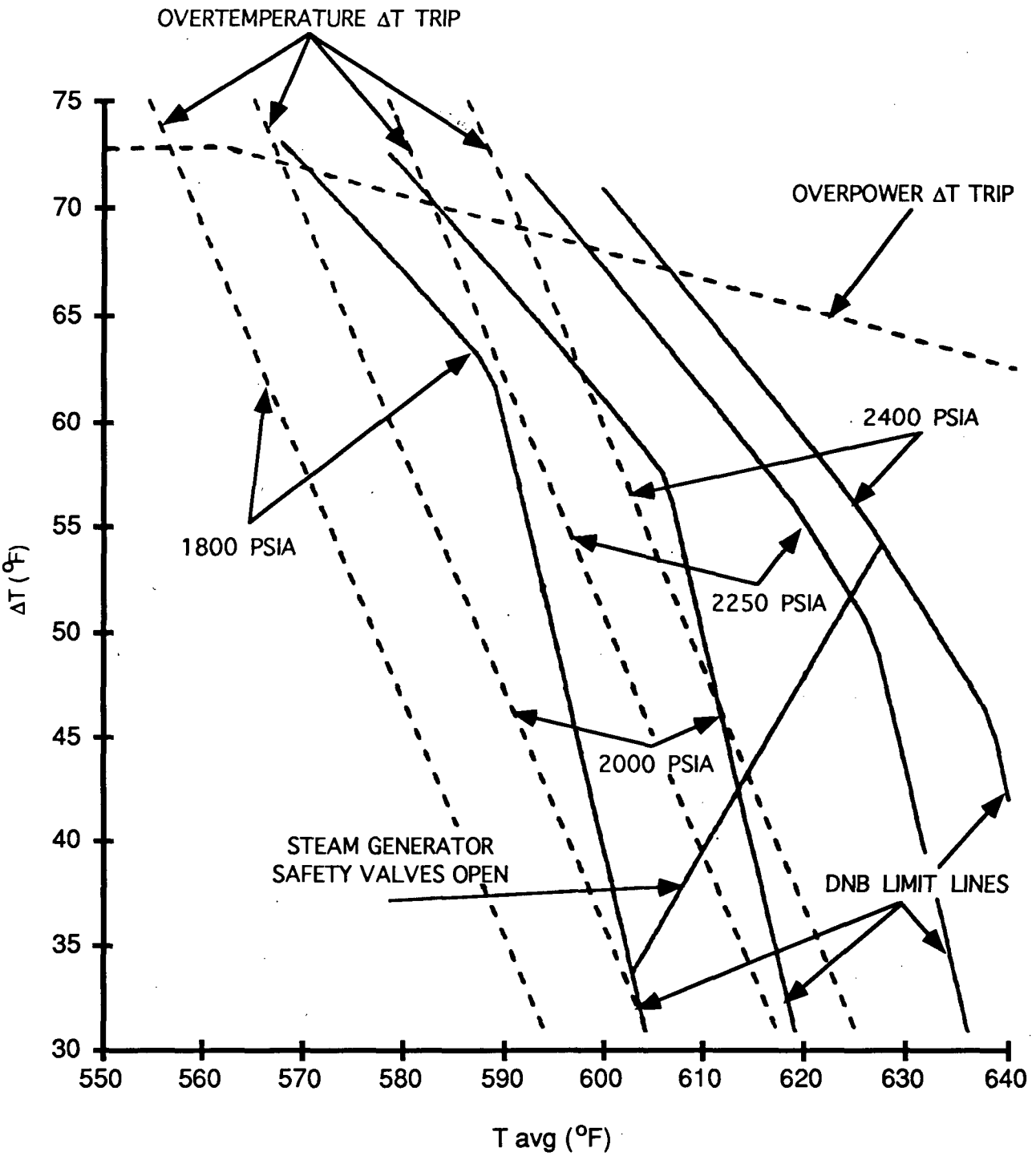
SINGLE FAILURES ASSUMED IN THE ACCIDENT ANALYSES

EVENT DESCRIPTION	WORST FAILURE ASSUMED
RCCA Bank Withdrawal from Subcritical	One Protection Train
RCCA Bank Withdrawal at Power	One Protection Train
Dropped RCCA, Dropped RCCA Bank	Nuclear Instrumentation System
CVCS Malfunction	Both centrifugal charging pumps are operating
Partial Loss of Reactor Coolant Flow	One Protection Train
Complete Loss of Reactor Coolant Flow	One Protection Train
RCP Locked Rotor	One Protection Train
Startup of an Inactive Loop	One Protection Train
Loss of External Electrical Load	One Protection Train
Loss of Normal Feedwater	Turbine-Driven AFW Pump
Feedwater System Malfunctions	One Protection Train
Excessive Load Increase	One Protection Train
Loss of Offsite Power	Turbine-Driven AFW Pump
Accidental Depressurization of the Steam System	One Safety Injection Train
Major Secondary System Pipe Rupture	One Safety Injection Train
RCCA Ejection	One Protection Train

TABLE 15.0-7

STEADY-STATE ERRORS ASSUMED IN ACCIDENT ANALYSES

	Parameter	Error Assumption
1.	Core Power	± 2% allowance for calorimetric error
2.	Average RCS Temperature	± 5.5°F allowance for controller dead band and measurement error
3.	Pressurizer Pressure	± 40 psi allowance for steady-state fluctuations and measurement error
4.	Reactor Coolant Flow	Thermal design or minimum measured flow is assumed and no steady-state errors are applied

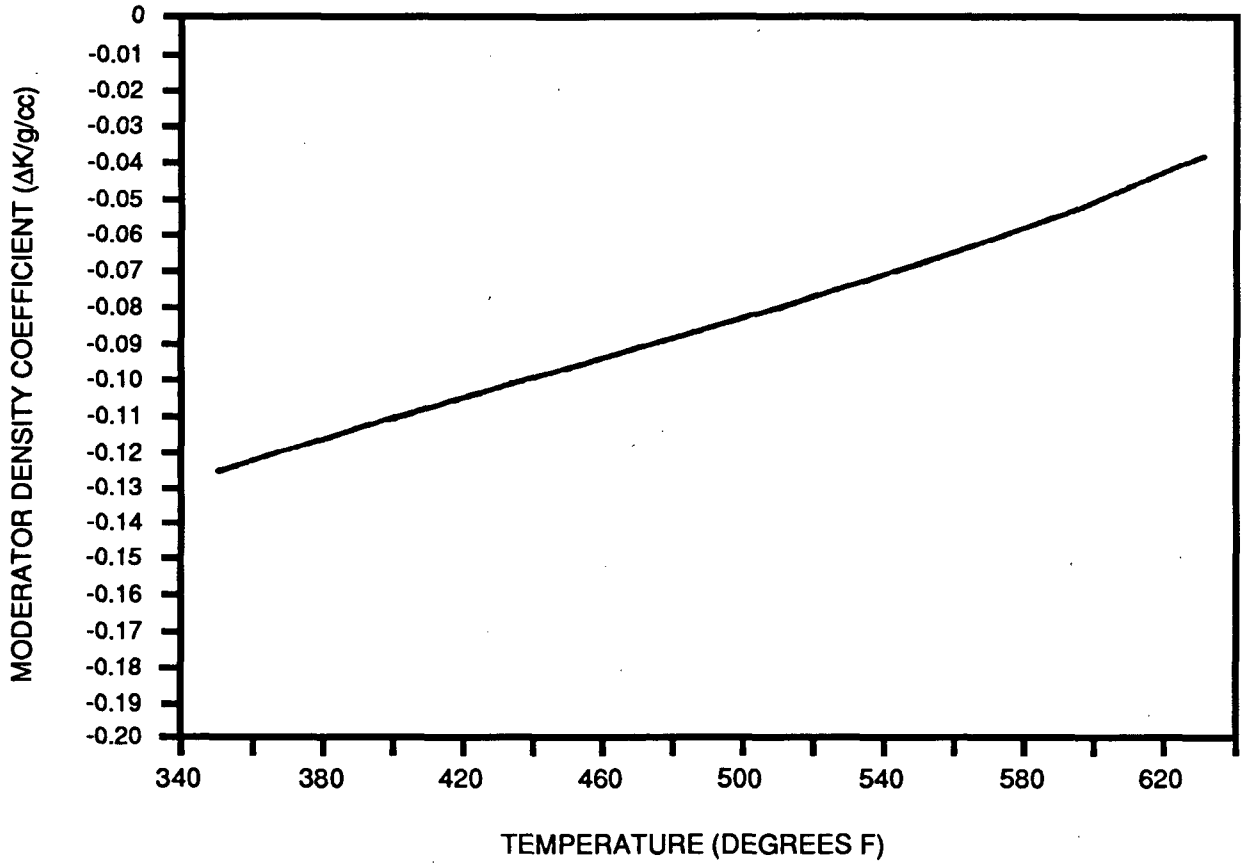


ZION STATION UFSAR

Figure 15.0-1

ILLUSTRATION OF OVERTEMPERATURE ΔT AND OVERPOWER ΔT PROTECTION

JULY 1993



ZION STATION UFSAR

Figure 15.0-2

MINIMUM MODERATOR DENSITY COEFFICIENT USED IN ANALYSIS

JULY 1993

NOTE 1: "LOWER CURVE" MOST NEGATIVE DOPPLER ONLY
POWER DEFECT = -1.94 $\Delta\rho$ (0 TO 100% POWER)
NOTE 2: "UPPER CURVE" LEAST NEGATIVE DOPPLER ONLY
POWER DEFECT = -1.0 $\Delta\rho$ (0 TO 100% POWER)

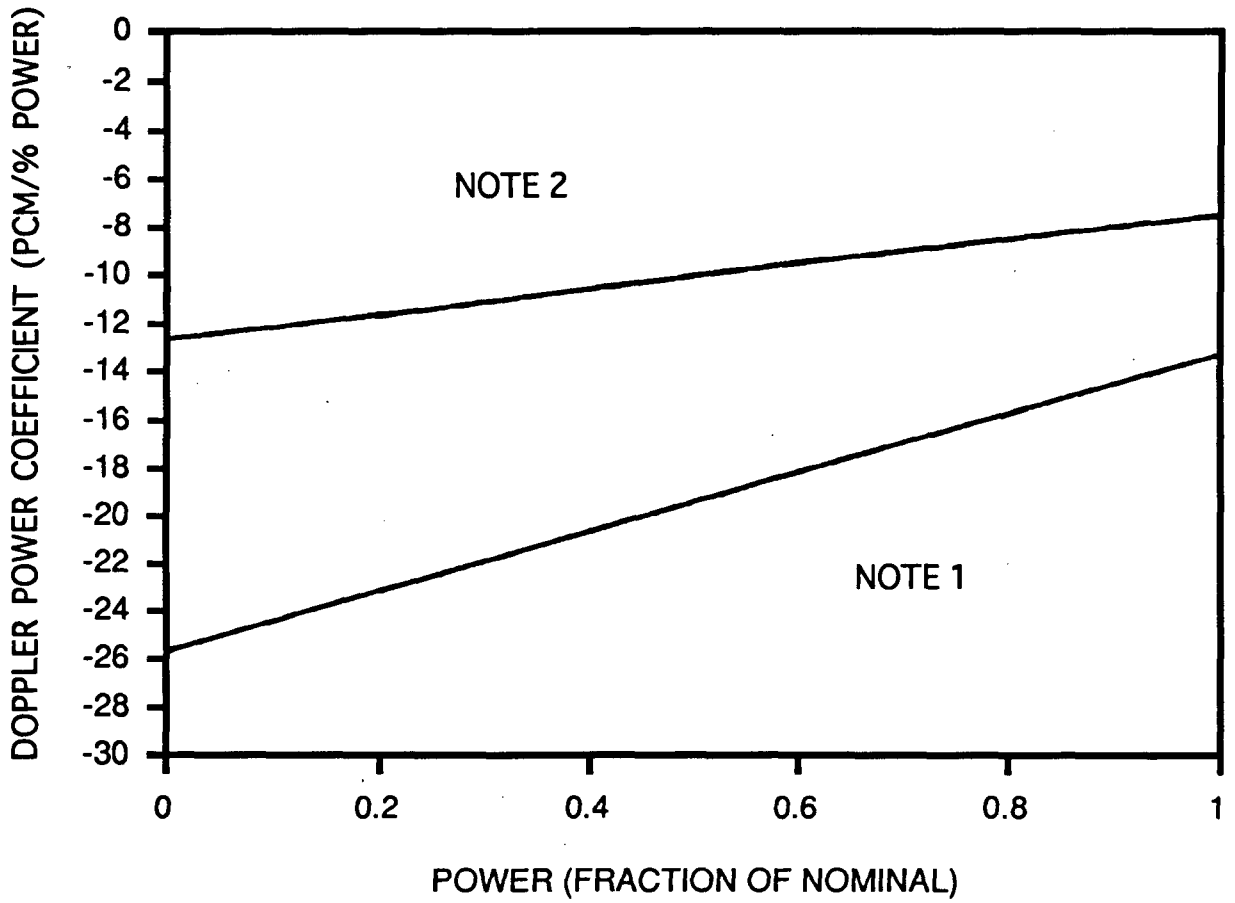


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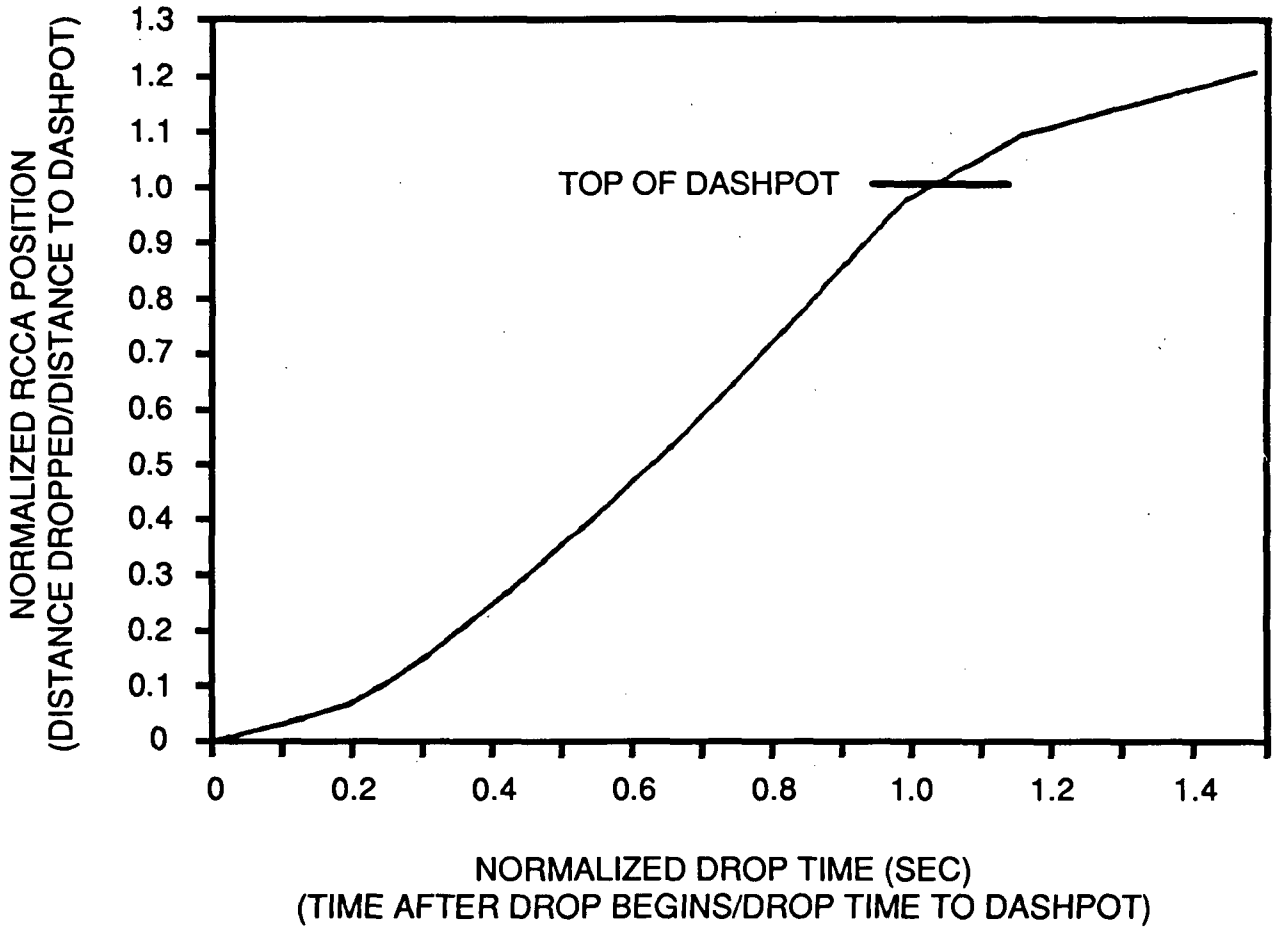
Figure 15.0-3

DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS

JULY 1993

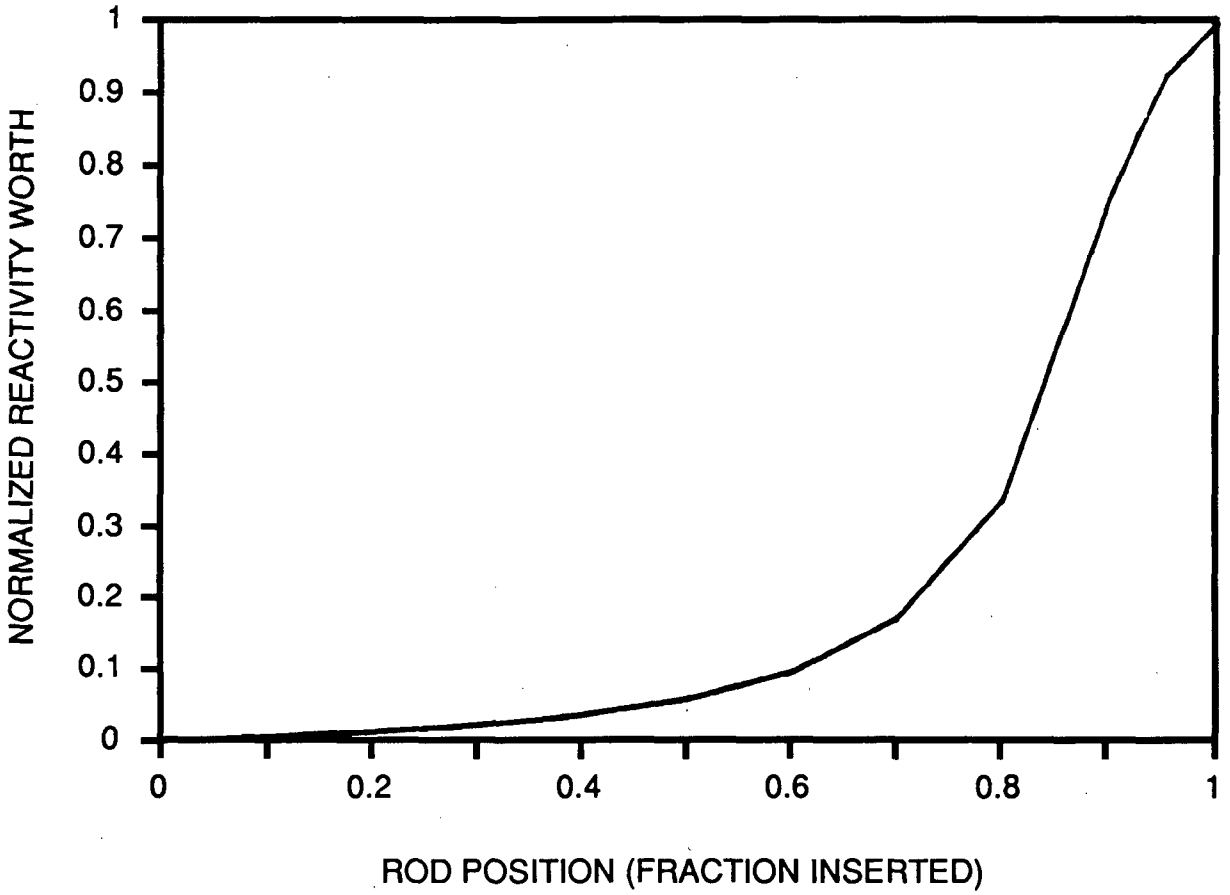


ZION STATION UFSAR

Figure 15.0-4

RCCA POSITION VERSUS TIME TO DASHPOT

JULY 1993

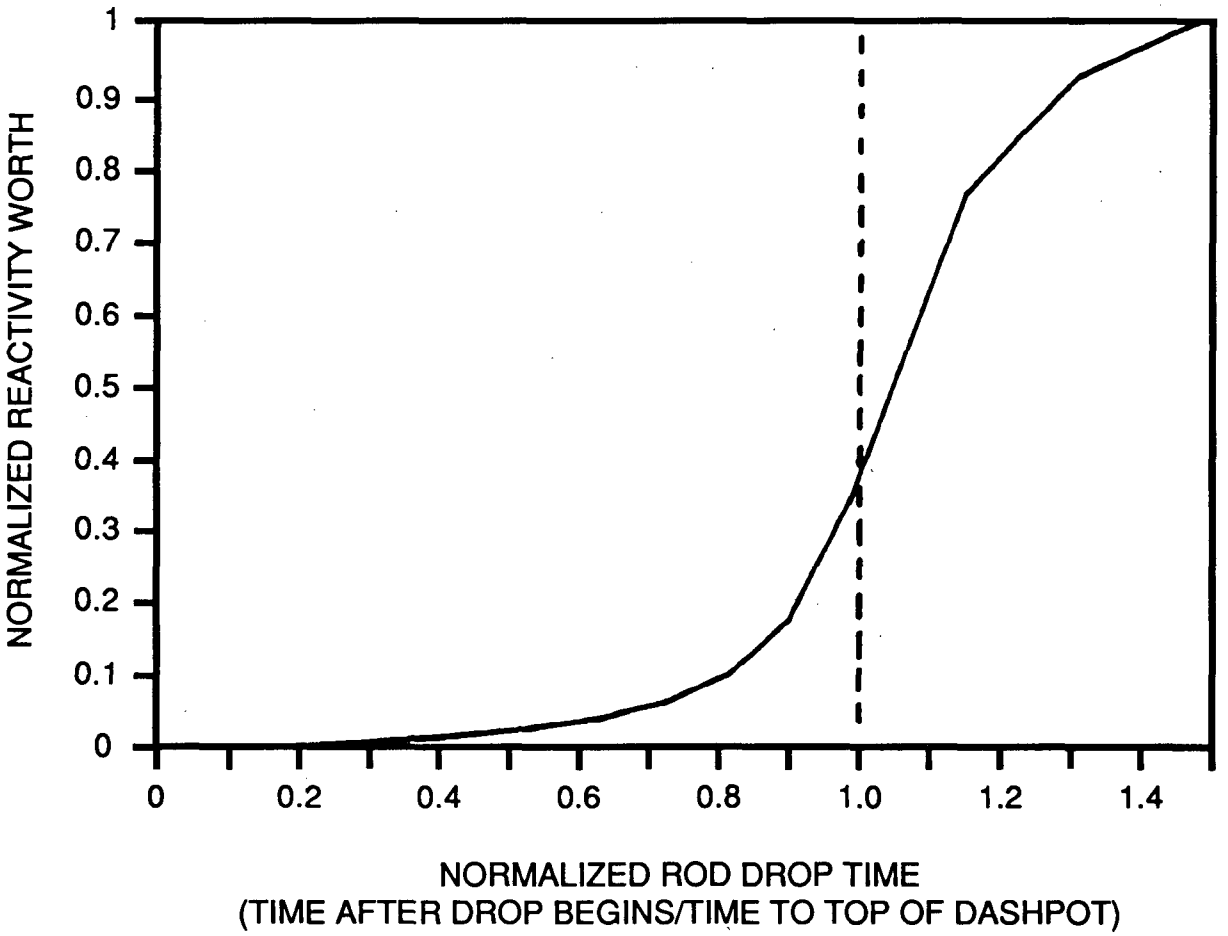


ZION STATION UFSAR

Figure 15.0-5

NORMALIZED ROD WORTH
VERSUS PERCENT INSERTED

JULY 1993



ZION STATION UFSAR

Figure 15.0-6

NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS NORMALIZED DROP TIME

JULY 1993

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.1 Excessive Heat Removal Due to Feedwater System Malfunctions

15.1.1.1 Incident Description

Reductions in feedwater temperature or excessive feedwater additions are means of increasing core power above full power due to the effects of a negative moderator temperature coefficient. Such transients are attenuated by the thermal capacity of the secondary plant and of the Reactor Coolant System (RCS). The overpower/overtemperature protection (neutron high flux, overtemperature ΔT, and overpower ΔT reactor trips) prevents any power increase that could lead to a Departure from Nucleate Boiling Ratio (DNBR) that is less than the DNBR limit.

One example of excessive feedwater flow would be the full opening of a feedwater regulating valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in 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 trip of the turbine and feedwater pumps on steam generator high-high level.

A second example of excess heat removal is the transient associated with the accidental opening of the low pressure heater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., following a large load decrease. At power, this increased subcooling will create a greater load demand on the RCS.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as discussed in Subsection 15.0.1.

15.1.1.2 Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN computer code (Reference 1). This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The feedwater system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction. Excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater

regulating valve to open fully is considered. Three cases are analyzed as follows:

1. Accidental opening of one feedwater regulating valve with the reactor just critical at zero load conditions assuming a conservatively large moderator density coefficient characteristic of end-of-life (EOL) conditions;

2. Accidental opening of one feedwater regulating valve with the reactor in automatic control at full power; and

3. Accidental opening of the low pressure heater bypass valve.

The accident is analyzed using the Revised Thermal Design Procedure (RTDP) as described in Reference 2. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

The following assumptions are used in the analysis of feedwater system malfunctions:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 2;

2. For the feedwater regulating valve accident at full power, one feedwater regulating valve is assumed to malfunction resulting in a step increase to 200% of nominal feedwater flow to one steam generator;

3. For the feedwater regulating valve accident at zero load condition, a feedwater valve malfunction occurs that results in a 5 second ramp increase in flow to one steam generator from zero to 200% of nominal full load value for one steam generator;

4. For the zero load condition, feedwater temperature is at a conservatively low valve of 32°F;

5. For the zero load condition, the initial water level in all the steam generators is at a conservatively low level;

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

7. No credit is taken for the heat capacity of the steam and water in the unaffected steam generators; and

8. The feedwater flow resulting from a fully open regulating valve is terminated by the steam generator high-high level signal that closes the affected loop feedwater regulating valve, closes the affected loop feedwater bypass valve, trips the main feedwater pumps and trips the main turbine.

Normal reactor control systems and engineered safety features (ESF) systems (e.g., safety injection) are not required to function. The Reactor Protection System (RPS) may actuate to trip the reactor due to an overpower condition. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

15.1.1.3 Results

In the case of an accidental full opening of one feedwater regulating valve with the reactor at zero power and the above mentioned assumptions, the THINC computer code (Reference 3) was used to determine the DNBR value during the limiting portion of the event. The departure from nucleate boiling (DNB) evaluation concluded that the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted for the zero power case.

The full power case (EOL maximum reactivity feedback without rod control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip is actuated when the steam generator level in the affected steam generator reaches the high-high level setpoint. Following the turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in Section 15.2.2. Assuming the reactor control to be in the automatic rod control mode results in a less limiting transient in terms of peak RCS pressure. However, the Rod Control System is not required to function for this event.

For all full power cases of excessive feedwater, continuous addition of cold feedwater is prevented by closure of the affected loop feedwater regulating valve, closure of the affected loop feedwater bypass valve, and a trip of the feedwater pumps and main turbine on steam generator high-high level. For all full power cases, a main turbine trip will result in a reactor trip. Following the reactor trip, the remaining feedwater regulating and bypass valves will close due to a reactor trip signal with low reactor coolant temperature.

The transient results seen in Figures 15.1-1 through 15.1-6 show the pressurizer pressure, core average temperature, and DNBR response to excessive feedwater addition, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater addition is terminated as a result of the high-high steam generator level signal. The DNBR does not drop below the limit safety analysis DNBR at any time.

Since the power level rises during this event, the fuel temperature will also rise until the reactor trip occurs. The core heat flux lags behind the neutron flux due to the fuel rod thermal time constant and, as a result, the peak core heat flux value does not exceed 118% of nominal. Thus, the peak fuel melting temperature will remain well below the fuel melting point.

The calculated sequence of events is shown in Table 15.1-3. The transient results show that DNB does not occur at any time during the excessive heat removal transient; thus, the ability of the primary coolant to remove heat

from the fuel rod is not reduced. Therefore, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The opening of a low pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 70°F, resulting in an increase in heat removal from the primary system. The increased thermal load, due to the open bypass valve, would result in a transient very similar (but of less magnitude) to that presented in Section 15.1.3, which evaluates the consequences of a 10% step load increase. Therefore, the transient results of this analysis are not presented.

15.1.1.4 Conclusions

The results of the analysis show that the minimum DNBR values calculated for the excessive feedwater addition at full power and zero power are at all times greater than the safety analysis limit value; hence, no fuel or clad damage is predicted. The decrease in feedwater temperature is less severe than the excessive load increase event (Section 15.1.3). Based on the results presented in that section, the applicable acceptance criteria for the decrease in feedwater temperature event have been satisfied.

15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

Refer to Section 15.1.1 for incidents and analysis.

15.1.3 Excessive Load Increase Incident

15.1.3.1 Incident 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 steam generator load demand. The Reactor Control System is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System (RPS).

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.

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

Protection against an excessive load increase accident is provided by the following RPS signals:

1. Low pressurizer pressure;
2. Overtemperature ΔT; and
3. Power range high neutron flux.

This event is classified as an ANS Condition II incident (moderate frequency) as described in Section 15.0.1.

15.1.3.2 Method of Analysis

This accident is analyzed using the LOFTRAN computer code (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generator, and steam generator 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% step load increase from rated load. These cases are as follows:

1. Reactor control in manual wiht beginning of life (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; and

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 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 has its most negative 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.

This accident is analyzed with the RTDP as described in Reference 2. Plant characteristics and initial conditions are further discussed in Section 15.0.3. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 2.

Normal reactor control systems and ESFs are not required to function for this accident. The RPS is assumed to be operable; however, reactor trip is not encountered for any of the cases analyzed due to the error allowances assumed in the setpoints. No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

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

15.1.3.3 Results

The calculated sequence of events for the excessive load increase accident is shown on Table 15.1-4. Note that a reactor trip signal was not generated for any of the four cases.

Figures 15.1-7 through 15.1-12 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 15.1-13 through 15.1-18 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.

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. Normal plant operating procedures would then be followed to reduce power.

Since the DNBR value does not fall below the safety analysis limits 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.

The time of minimum DNBR is not relevant since the conservative assumptions made in this analysis result in no reactor trip for all cases analyzed. The transient analysis results arrive at new steady-state conditions.

15.1.3.4 Conclusions

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

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Accidental Depressurization of the Main Steam System

15.1.4.1.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam (MS) System are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a MS pipe are given in Section 15.1.5.2.

The steam release, as a consequence of this accident, results in an initial increase in steam flow which 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 rod cluster control assembly (RCCA) and a single failure in the ESFs, there will be no consequential damage to the core or RCS 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 protection during an accidental depressurization of the MS System:

1. Safety Injection (SI) System actuation from any of the following:

    a. Two out of three coincident low pressurizer pressure signals; or

b. Two out of three differential pressure signals between a steamline and the remaining steamlines.

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

3. Redundant isolation of the main feedwater (MFW) lines - sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the MFW regulating valves following a reactor trip, a SI signal will close all feedwater isolation valves and trip the MFW pumps.

Accidental depressurization of the secondary system is classified as an ANS Condition II event, a fault of moderate frequency, as described in Section 15.0.1.

Plant characteristics and initial conditions are discussed in Section 15.0.3.

15.1.4.1.2 Analysis of Effects and Consequences

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN code (Reference 1) to determine RCS temperature and pressure during the transient, and the effect of SI; and

2. An analysis to determine that the DNB design basis is met.

The following conditions are assumed to exist at the time of a secondary system break accident:

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

2. A negative moderator coefficient corresponding to the EOL rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature at 1000 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-19.

3. Minimum capability for injection of boric acid solution corresponding to the most restrictive single active failure in the SI System. This corresponds to the flow delivered by one centrifugal charging pump delivering its full contents to the cold leg header (see Figure 15.1-20). No credit is taken for the low concentration boric acid that must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of RWST boric acid (2000 ppm) to the reactor coolant loops.

4. The case studies a steam flow of 223 lbs/second at 1100 psia from any steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most pessimistic initial condition.

   Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip from 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 steamline 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, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the RCS cooldown are less for steamline breaks occurring at power;

5. In computing the steam flow, the Moody Curve, (see Reference 4) for f(L/D)=0 is used; and

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

15.1.4.1.3 Results

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

Figures 15.1-22 through 15.1-25 show the transients arising as a result of a steam release having a steam flow of 223 lbs/second at 1100 psia. The assumed steam release is the maximum capacity of any single steam dump or safety valve. In this case, SI is initiated automatically by low pressurizer pressure. Operation of one high speed centrifugal charging pump is assumed. Boron solution assumed to be at 2000 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The reactivity transient for the cases shown is more severe than the case of a steam release from all four steam generators through one safety valve. The transient is quite conservative with respect to cooldown, since no credit is taken for

the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about five minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

The sequence of events is shown in Table 15.1-1.

An accidental depressurization of the MS System is analyzed in the same way as the major steam pipe rupture (MSLB) transient. Using the statepoint methodology applied to the MSLB transient, it was found that the statepoints evaluated for the DNB transient were significantly less limiting than those for the MSLB event. This is not unexpected since the maximum return to power for the steam release transient is much lower than that for the MSLB (2% nominal power vs 20% nominal power). The minimum DNBR is not a concern under these conditions.

15.1.4.1.4 Conclusions

The analysis has shown that the criteria stated earlier in this section are satisfied. For accidental depressurization of the MS System, the DNB design limits are not exceeded. This case is less limiting than the steamline rupture case described in Section 15.1.5.2.

15.1.5 Depressurization of the Main Steam System

15.1.5.1 Minor Secondary System Pipe Breaks

15.1.5.1.1 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. Minor secondary system pipe breaks are classified as an ANS Condition III incident (an infrequent incident) as discussed in Section 15.0.1.

15.1.5.1.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of the analysis presented in Section 15.1.5.2 for a major secondary system pipe rupture also meet this criterion, separate analysis for minor secondary system pipe breaks is not required. See Reference 5 for additional details.

The analysis of the more probable accidental opening of a secondary system steam dump, relief, or safety valve is presented in Section 15.1.4.1. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.1.5.1.3 Conclusions

The analysis presented in Section 15.1.5.2 demonstrates that the consequences of a minor secondary system pipe break are acceptable since a DNBR of less than the limiting value does not occur even for a more critical major secondary system pipe break.

15.1.5.2 Major Secondary System Pipe Rupture-Rupture of a Main Steamline

15.1.5.2.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a MS pipe would result in an initial increase in steam flow which 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 an insertion of positive reactivity. 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 which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the SI system.

The analysis of a MS pipe rupture is performed to demonstrate that the following criterion is satisfied:

> 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.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the analysis shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

A major steamline rupture is classified as an ANS Condition IV event as discussed in Section 15.0.1.

The major rupture of a steamline is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented in Section 15.1.5.2.2.

The following functions provide the necessary protection against a steamline rupture:

1. SI System actuation from any of the following:

    a. Two out of three differential pressure signals between a steamline and the remaining steamlines;

    b. Two out of three low pressurizer pressure signals;

c. High steamline flow in two main steamlines (one out of two per line) in coincidence with either low-low RCS average temperature or low steamline pressure in any two lines; and

d. Two out of four high containment pressure signals.

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

3. Redundant isolation of the MFW lines; sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the MFW valves, a SI signal will rapidly close all feedwater control valves and feedwater isolation valves and trip the MFW pumps.

4. Trip of the fast acting steamline stop valves (designed to close in less than five seconds after receipt of the signal) on:

   a. High steam flow in two main steamlines in coincidence with either low-low RCS average temperature or low steamline pressure in any two lines; and

   b. High-high containment pressure.

Fast-acting isolation valves with downstream check valves are provided in each steamline. These valves will fully close within 7 seconds of a large break in the steamline including the time for generation of the closure signal. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blowdown even if one of the isolation valves fails to close.

Nozzles inside the steam pipes, which are of considerably smaller diameter (16 inches) than the MS pipe, are located inside the Containment and serve to limit the maximum steam flow for any break further downstream.

15.1.5.2.2 Analysis of Effects and Consequences

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

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steamline break. The LOFTRAN code (Reference 1) has been used;

2. The radial and axial peaking factors response in the core. A detailed neutronic code was used to generate the increases in peaking factors based on statepoint information generated in item 1 above; and

3. The thermal and hydraulic behavior of the core following a steamline break. A detailed thermal and hydraulic digital-computer code, THINC (Reference 3), has been used to determine if DNB occurs for the core conditions computed in item 1 above.

The following conditions were assumed to exist at the time of a main steamline break 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 steamline break accident will not lead to a more adverse condition than the case analyzed.

2. The negative moderator coefficient corresponding 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 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-19. The effect of power generation in the core on overall reactivity is shown in Figure 15.1-21.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the statepoints for the cases analyzed. 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 nonuniform 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 found that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of boric acid (2000 ppm from the RWST) solution corresponding to the most restrictive single failure in the SI System. The Emergency Core Cooling System (ECCS) consists of three systems: 1) the passive accumulators, 2) the Residual Heat Removal System, and 3) the SI System. Only the high-head centrifugal charging pumps of the SI System and the accumulators are modeled for the steamline break accident analysis.

   The actual modeling of the SI System in LOFTRAN is described in Reference 1. The injection curve is shown in figure 15.1-20. The flow corresponds to that delivered by one charging

pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the RWST prior to the delivery of 2000 ppm boric acid solution to the Reactor Coolant loops.

For the cases where offsite power is assumed to be available, the sequence of events in the SI System is the following: after the generation of the SI signal (appropriate delays for instrumentation, logic and signal transport included), the appropriate valves begin to operate and the high head injection pump starts. In 25 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The 25 seconds envelope the sequential repositioning of the RWST and Volume Control Tank (VCT) isolation valves. The volume containing the low concentration boron is swept from the lines before the 2000 ppm boron reaches the core. This delay, as described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 20-second delay is assumed to start the diesel generators and to load the necessary SI equipment onto them.

4. The design value steam generator heat transfer coefficient is used including an allowance for the fouling factor.

5. Four combinations of break sizes and initial plant conditions have been considered in determining the core power and RCS transients:

   a. Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at no load conditions, full reactor coolant flow with offsite power available;

   b. Complete severance of pipe inside containment at the outlet of the steam generator with the plant initially at no load conditions with offsite power available;

   c. Case a above with loss-of-offsite power simultaneous with the initiation of the SI signal. Loss-of-offsite power results in coolant pump coastdown; and

   d. Case b above with the loss-of-offsite power simultaneous with the initiation of the SI signal.

6. Power peaking factors corresponding to one stuck RCCA and nonuniform 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 RCCA during the return to power phase following the steamline break. This void, in conjunction with the large negative moderator coefficient, partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

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

All the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steamline 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 steamline 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, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the RCS cooldown are less for a steamline break occurring at power.

7. In computing the steam flow during a steamline break, the Moody Curve (Reference 4) for f(L/D) = 0 is used.

8. Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitudes of the temperature decrease in the core and the pressure increase in the Containment.

15.1.5.2.3 Results

The calculated sequence of events for both cases analyzed is shown on Table 15.1-2.

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

15.1.5.2.4 Core Power and Reactor Coolant System Transient

Figures 15.1-26 through 15.1-29 show the RCS transient and core heat flux following a MS pipe rupture (complete severance of a pipe) outside the Containment, downstream of the flow measuring nozzle at initial no load condition (case a).

The break assumed is the largest break which can occur anywhere outside the Containment either upstream or downstream of the isolation valves. Offsite power is assumed available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of SI by high differential pressure between any

steamline and the remaining steamlines or by high steam flow signals in coincidence with either low-low RCS temperature or low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast action isolation valves in the steamlines by the high steam flow signals in coincidence with either low-low RCS temperature or low steamline pressure or is limited to no more than 7 seconds for the other steam generators while the faulted generator blows down. The steamline isolation valves are designed to be fully closed in less than five seconds after receipt of closure signal with no flow through them. With the high flow existing during a steamline rupture, the valves will close considerably faster.

As shown in Figures 15.1-29, 15.1-33, 15.1-37, and 15.1-41, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck assembly) before boron solution at 2000 ppm enters the RCS from the SI System. A peak core heat flux well below the nominal full-power value is attained.

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

Figures 15.1-30 through 15.1-33 show Case b, a steamline rupture at the exit of a steam generator at no load. The sequence of events is similar to that described above for the rupture outside the Containment except that criticality is attained earlier due to more rapid cooldown and a higher peak core heat flux is attained.

Figures 15.1-34 through 15.1-41 show the responses of the salient parameters for Cases c and d which correspond to the cases discussed above with additional loss-of-offsite power at the time the SI signal is generated. The SI System delay time includes 20 seconds to start the diesel and an additional 27 seconds to start the ECCS pumps and open the

appropriate valves. In 47 seconds, the diesel and pump are assumed to start and the valves are assumed to be in their final position with the pump suction transferred from the VCT to the RWST. In each case, criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. For both these cases, the peak core heat flux remains well below the nominal full power value, and is less than that predicted above with full force flow available.

Following a steamline break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss-of-offsite power, this heat is removed to the atmosphere via the steamline safety valves which have been sized to cover this condition.

15.1.5.2.5 Margin to Critical Heat Flux

A DNB analysis was performed for Case b which is most critical to DNB (i.e., highest core heat flux and the lowest RCS pressure). It was found that this case has a minimum DNBR greater than the limiting value. Thus, the DNB design basis is satisfied.

15.1.5.2.6 Conclusions

The analyses have demonstrated that the criteria stated previously in Section 15.1.5.2.1 is satisfied.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analyses, in fact, show that the DNB design basis is met.

15.1.6 References, Section 15.1

1. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2. Friedland, A.J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

3. Friedland, A.J. and S. Ray, "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-A, September, 1991.

4. Moody, F.S., "Transactions of the ASME," Journal of Heat Transfer, Figure 3, Page 134, February 1965.

5. Risher, D.H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

ACCIDENT	EVENTS	TIME (sec)
Accidental depressurization of the MS System	Inadvertent opening of one main steam safety, steam dump, or relief valve	0
	Pressurizer empties	401
	Low pressurizer pressure safety injection setpoint reached	419
	2000 ppm boron reaches core	587

TABLE 15.1-2

TIME SEQUENCE OF EVENTS FOR MAJOR SECONDARY SYSTEM PIPE RUPTURE

ACCIDENT	EVENTS	TIME (sec)
Major secondary system pipe rupture		
1. Case a	Steamline ruptures	0
	Criticality attained	14
	Pressurizer empty	21
	Boron reaches core	121
2. Case b	Steamline ruptures	0
	Criticality attained	12
	Pressurizer empty	15
	Boron reaches core	117
3. Case c	Steamline ruptures	0
	Criticality attained	18
	Pressurizer empty	24
	Boron reaches core	244
4. Case d	Steamline ruptures	0
	Criticality attained	14
	Pressurizer empty	17
	Boron reaches core	128

TABLE 15.1-3

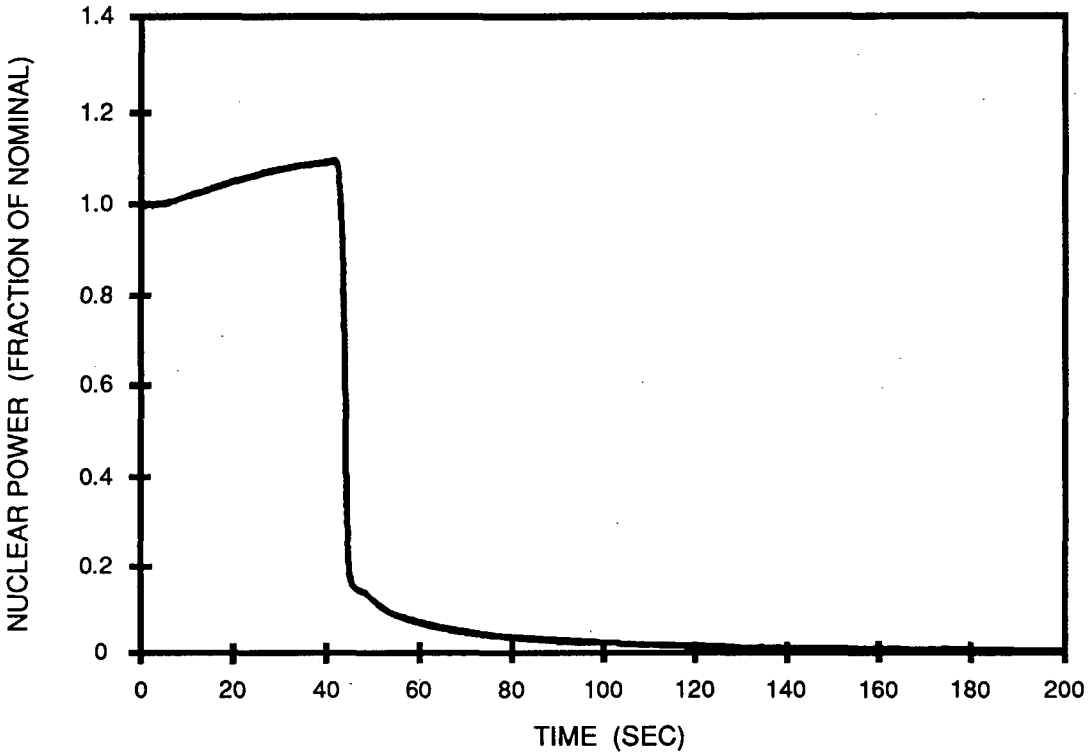
TIME SEQUENCE OF EVENTS FOR EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

ACCIDENT	EVENTS	TIME (sec)
Excessive Heat Removal Due to Feedwater System Malfunction	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level signal generated	35.7
	Turbine trip occurs due to high-high level signal	38.2
	Minimum DNBR occurs	39.5
	Reactor trip occurs due to turbine trip signal	40.7
	Feedwater regulating valves closed	45.7

TABLE 15.1-4

TIME SEQUENCE OF EVENTS FOR EXCESSIVE LOAD INCREASE

ACCIDENT	EVENTS	TIME (sec)
Excessive Load Increase		
1. Manual reactor control (BOL - minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	210
2. Manual reactor control (EOL - maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	75
3. Automatic reactor control (BOL - minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	290
4. Automatic reactor control (EOL - maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	80



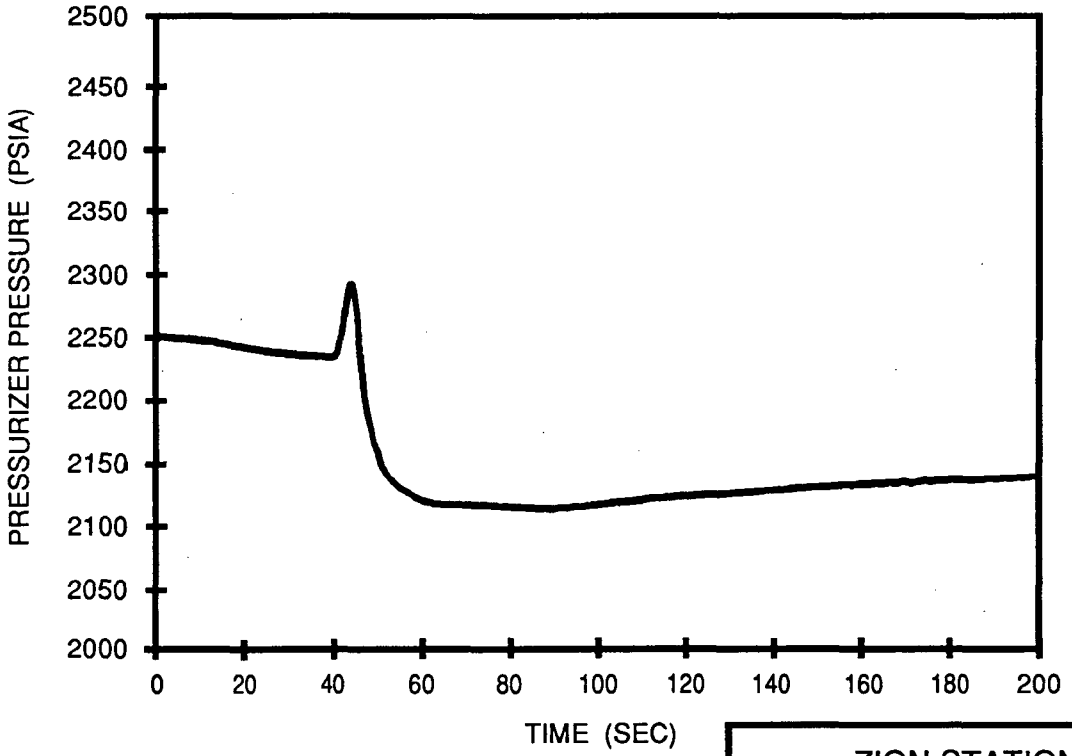


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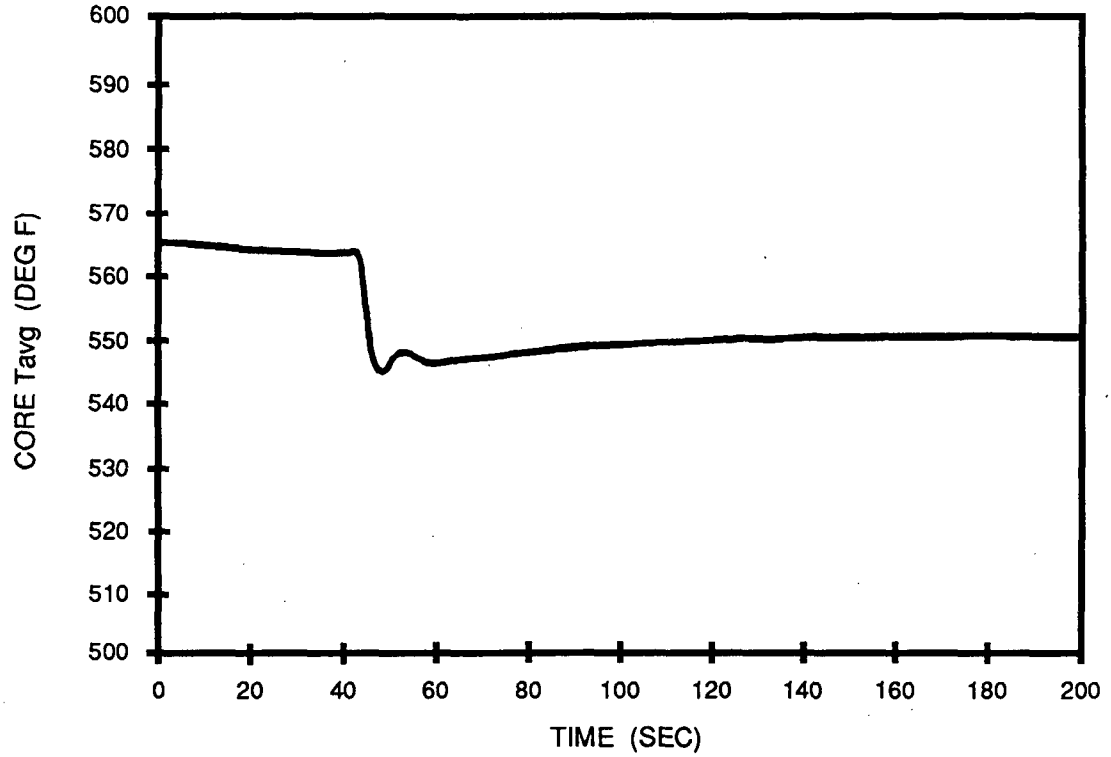
Figure 15.1-1
FEEDWATER MALFUNCTION
NUCLEAR POWER AND
PRESSURIZER PRESSURE
FULL POWER MANUAL ROD CONTROL
JULY 1993



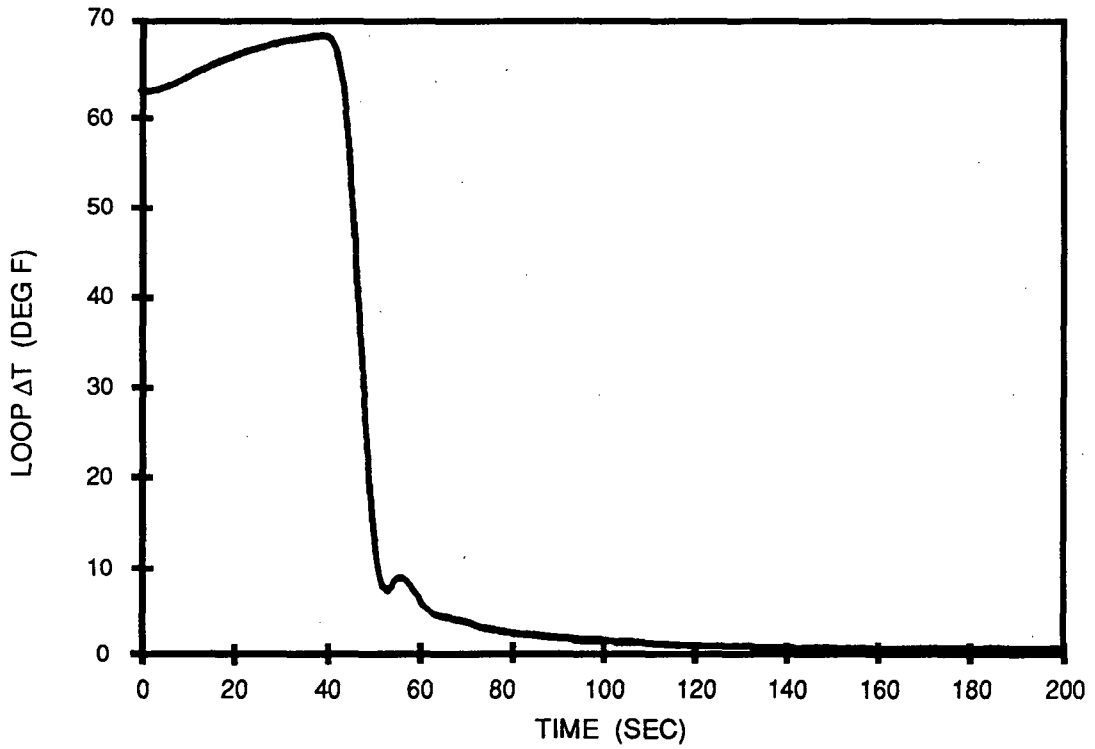


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Figure 15.1-2

FEEDWATER MALFUNCTION
CORE Tavg AND LOOP ΔT
FULL POWER MANUAL ROD CONTROL

JULY 1993

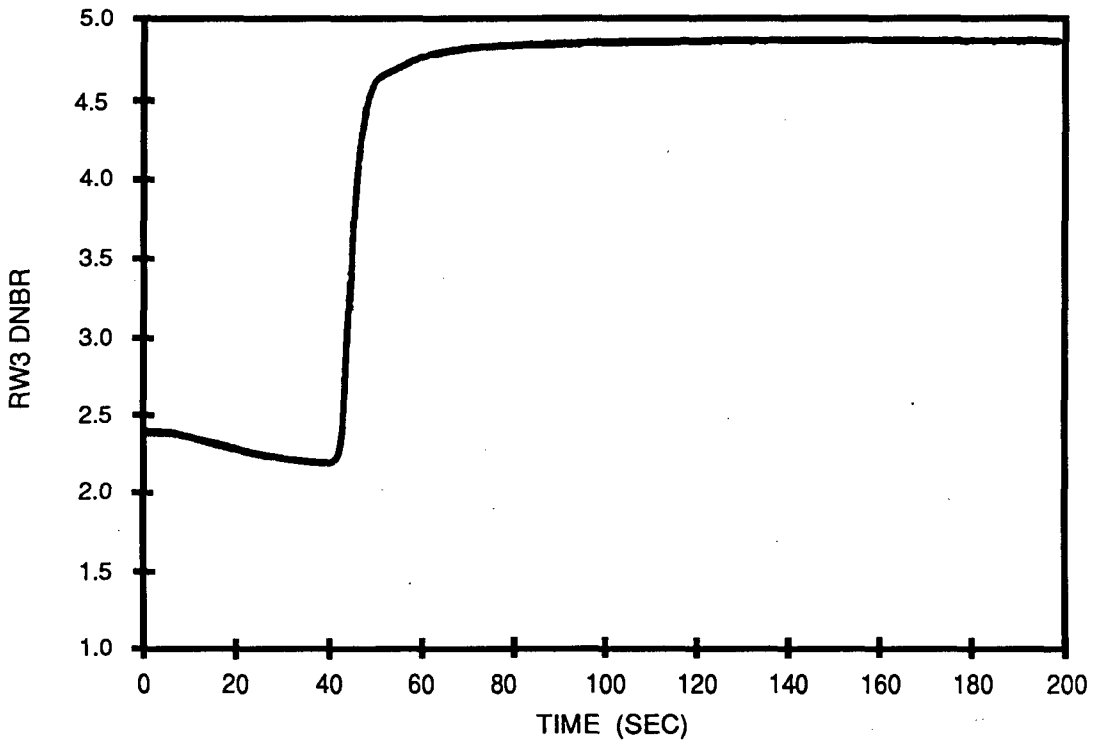

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Figure 15.1-3

FEEDWATER MALFUNCTION
DNBR VERSUS TIME
FULL POWER MANUAL ROD CONTROL

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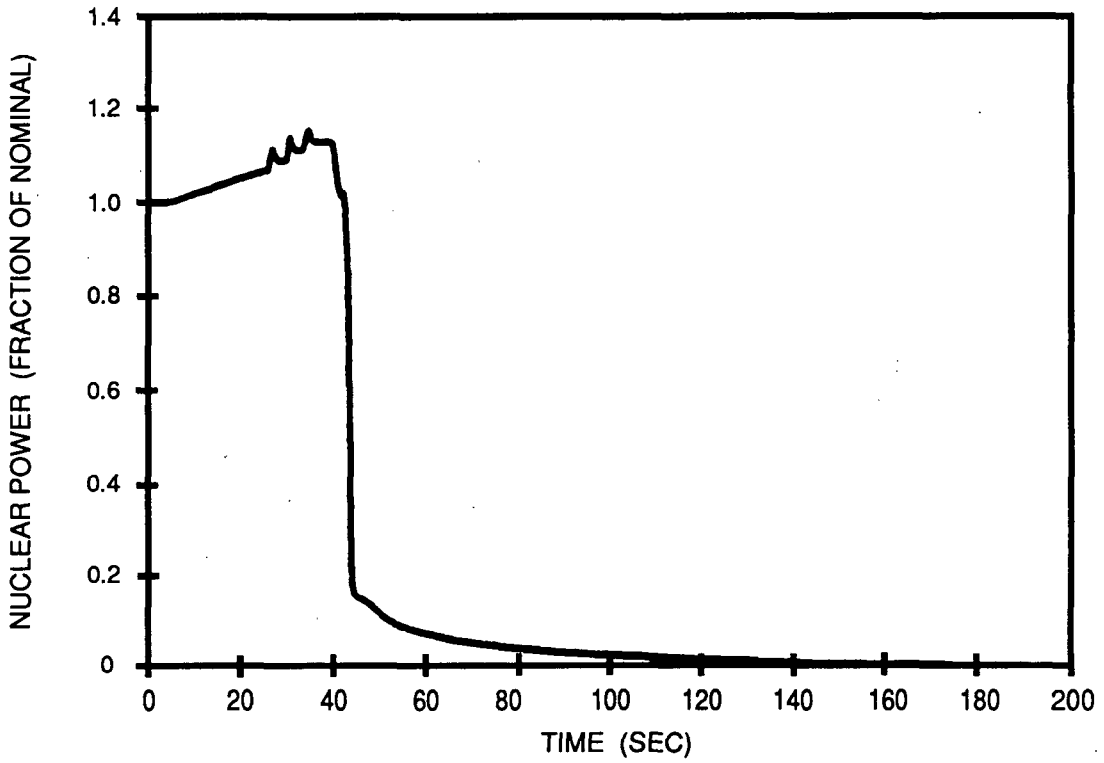

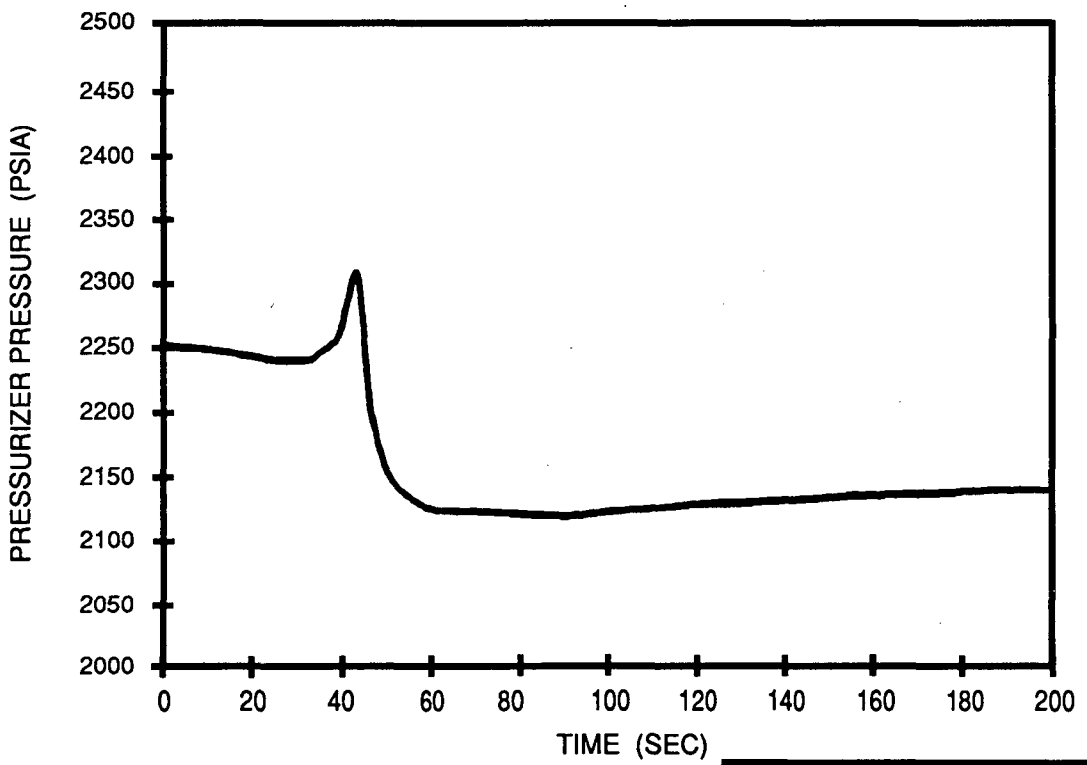

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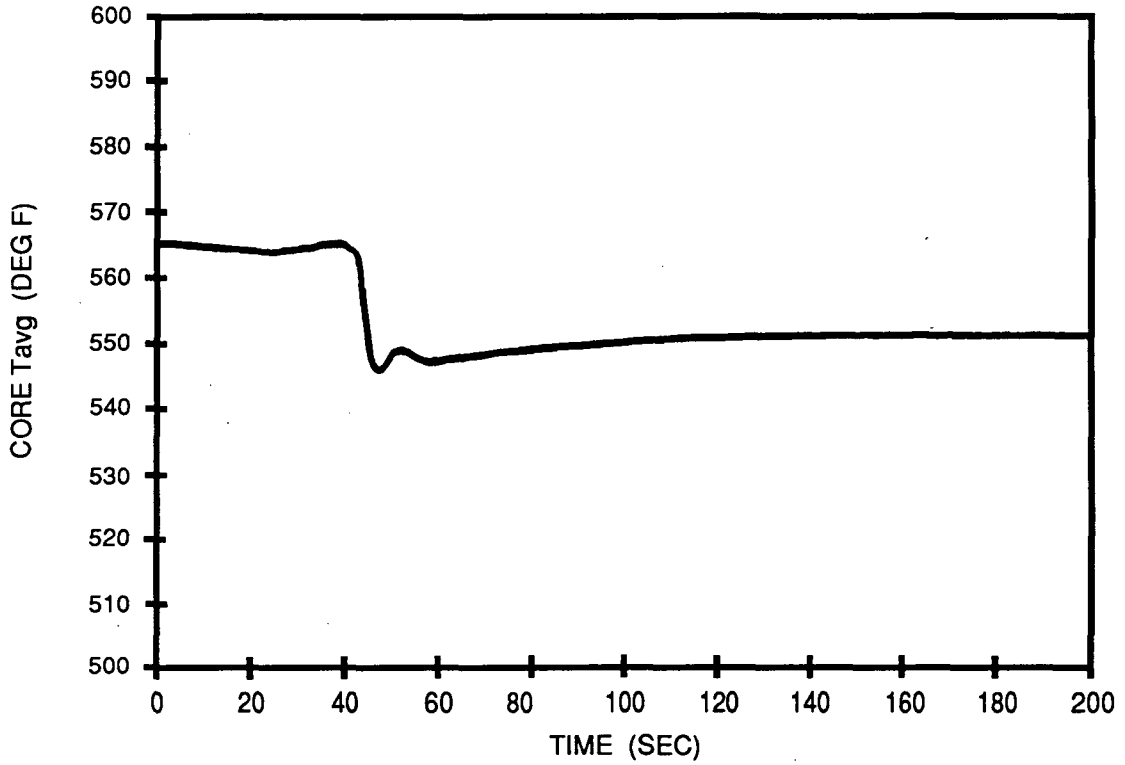
Figure 15.1-4
FEEDWATER MALFUNCTION
NUCLEAR POWER AND
PRESSURIZER PRESSURE
FULL POWER AUTO ROD CONTROL
JULY 1993



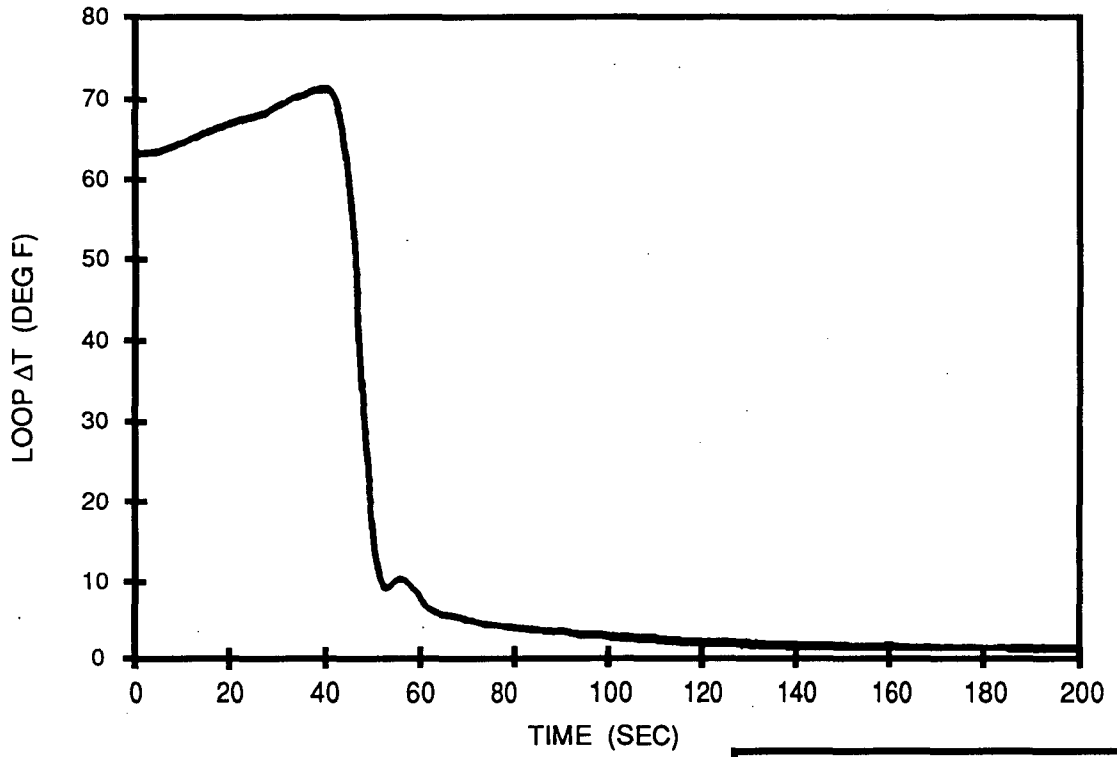


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Figure 15.1-5

FEEDWATER MALFUNCTION
CORE Tavg AND LOOP ΔT
FULL POWER AUTO ROD CONTROL

JULY 1993

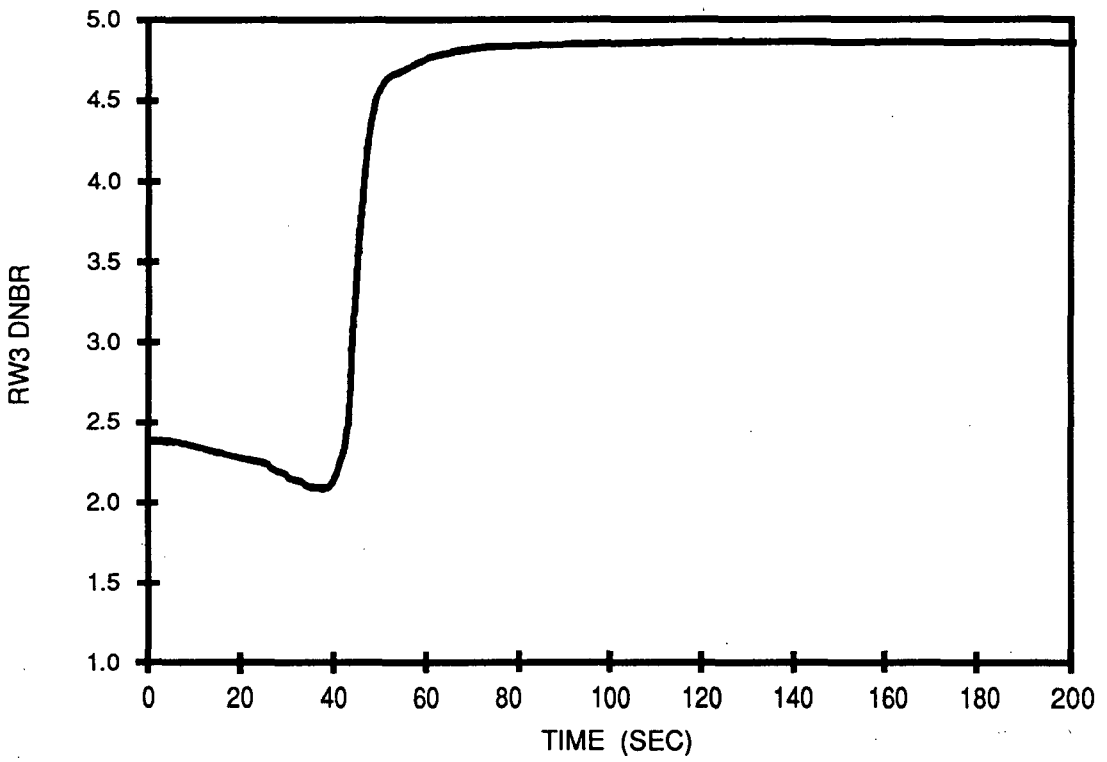

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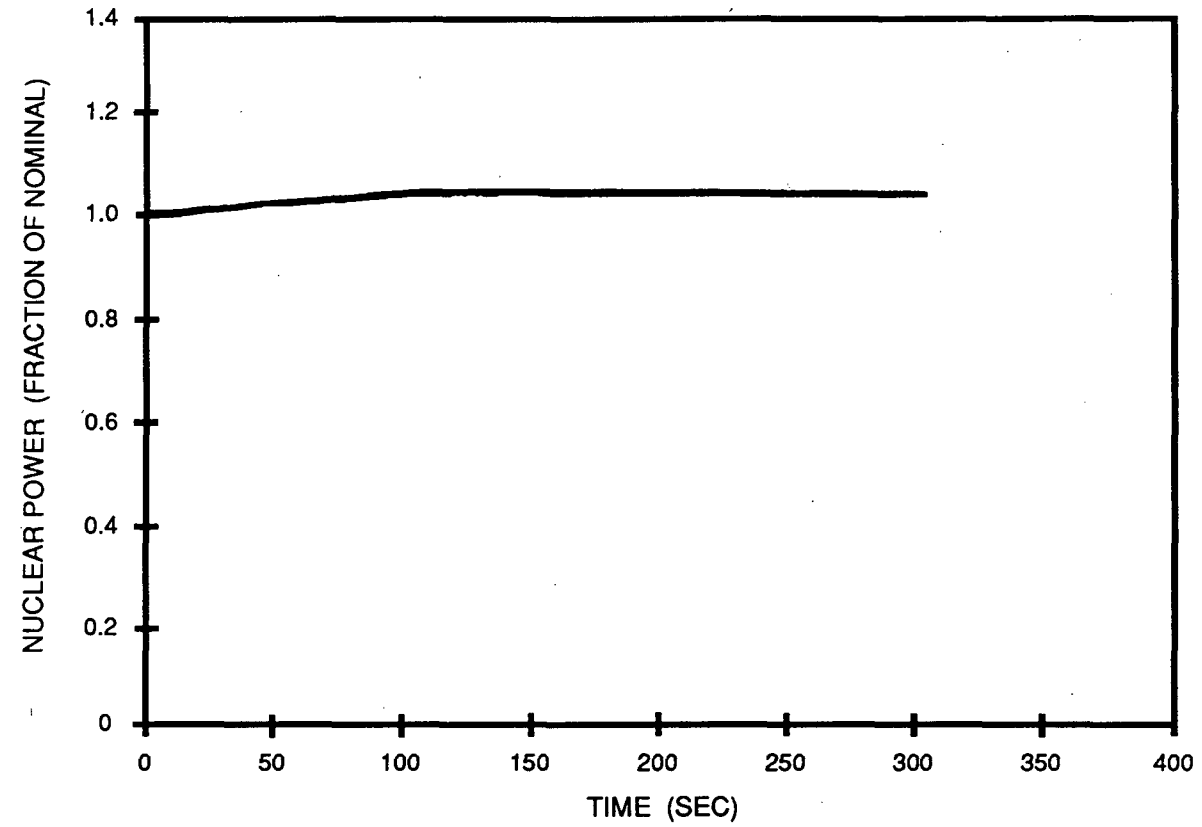
Figure 15.1-6

FEEDWATER MALFUNCTION
DNBR VERSUS TIME
FULL POWER AUTO ROD CONTROL

JULY 1993



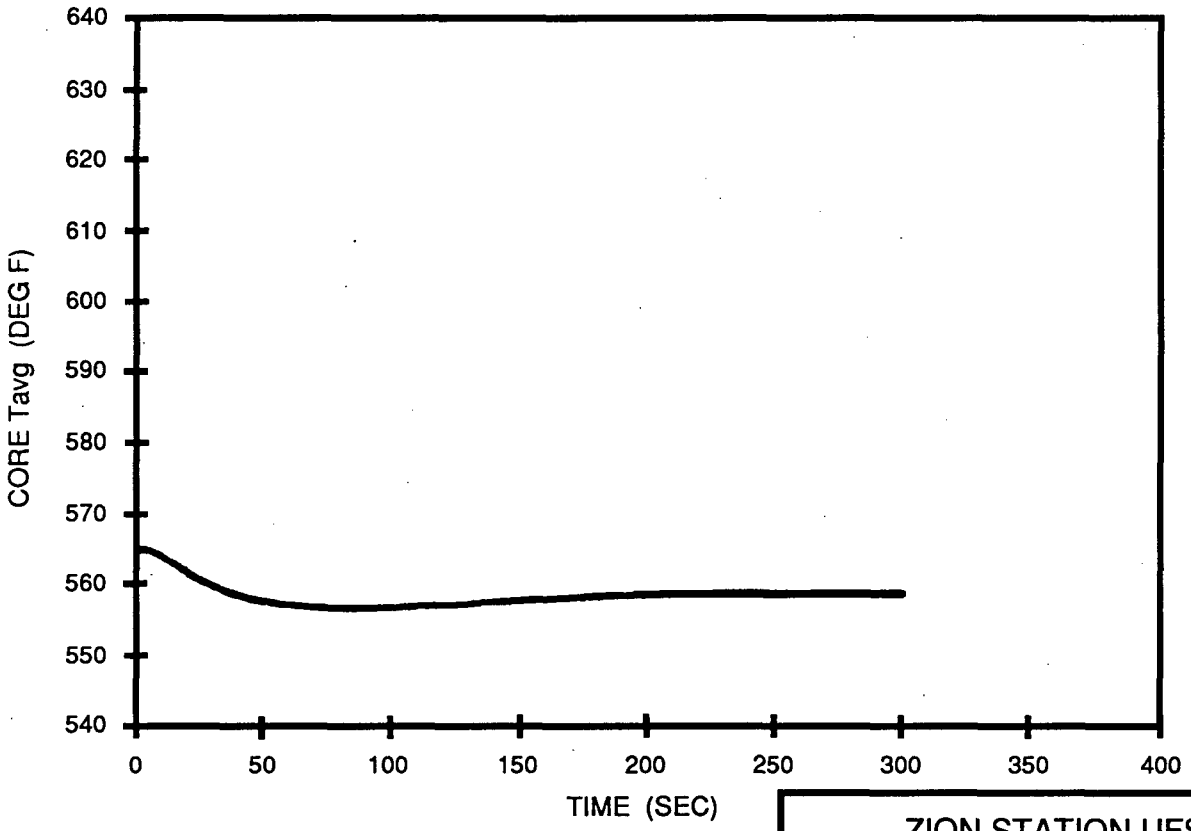


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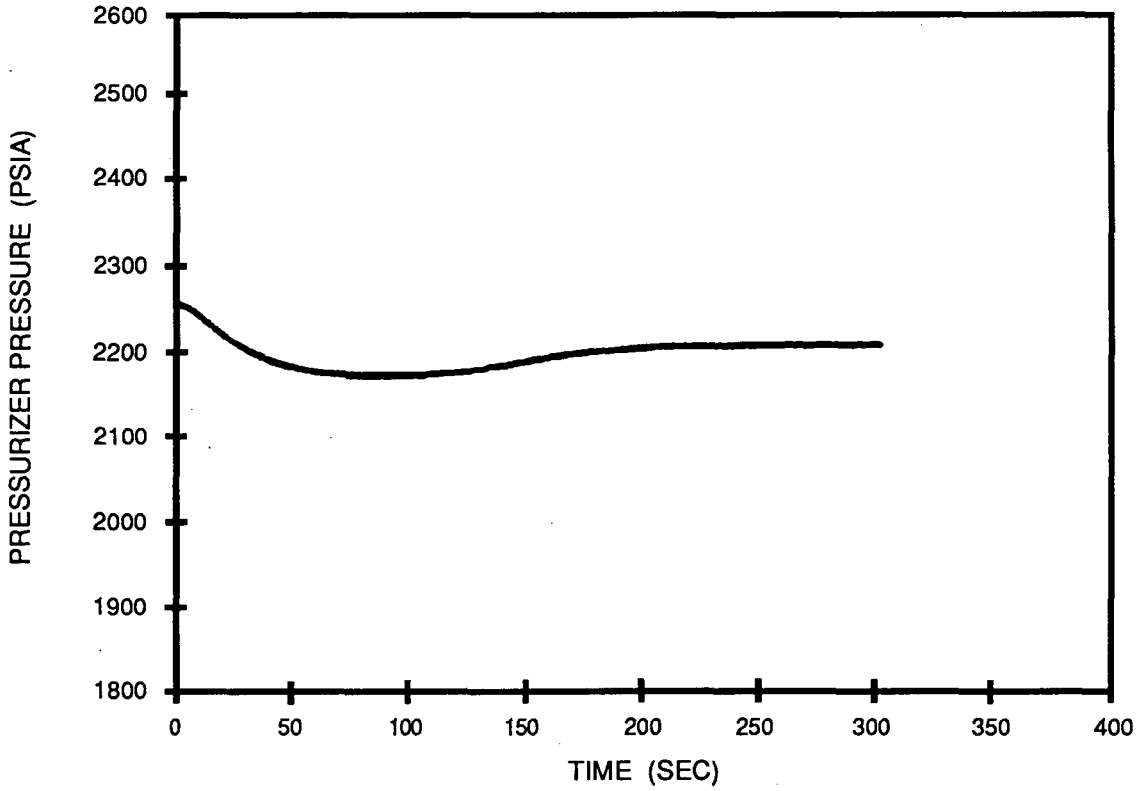
Figure 15.1-7

EXCESSIVE LOAD INCREASE
NUCLEAR POWER AND CORE Tavg
BOL MANUAL ROD CONTROL

JULY 1993



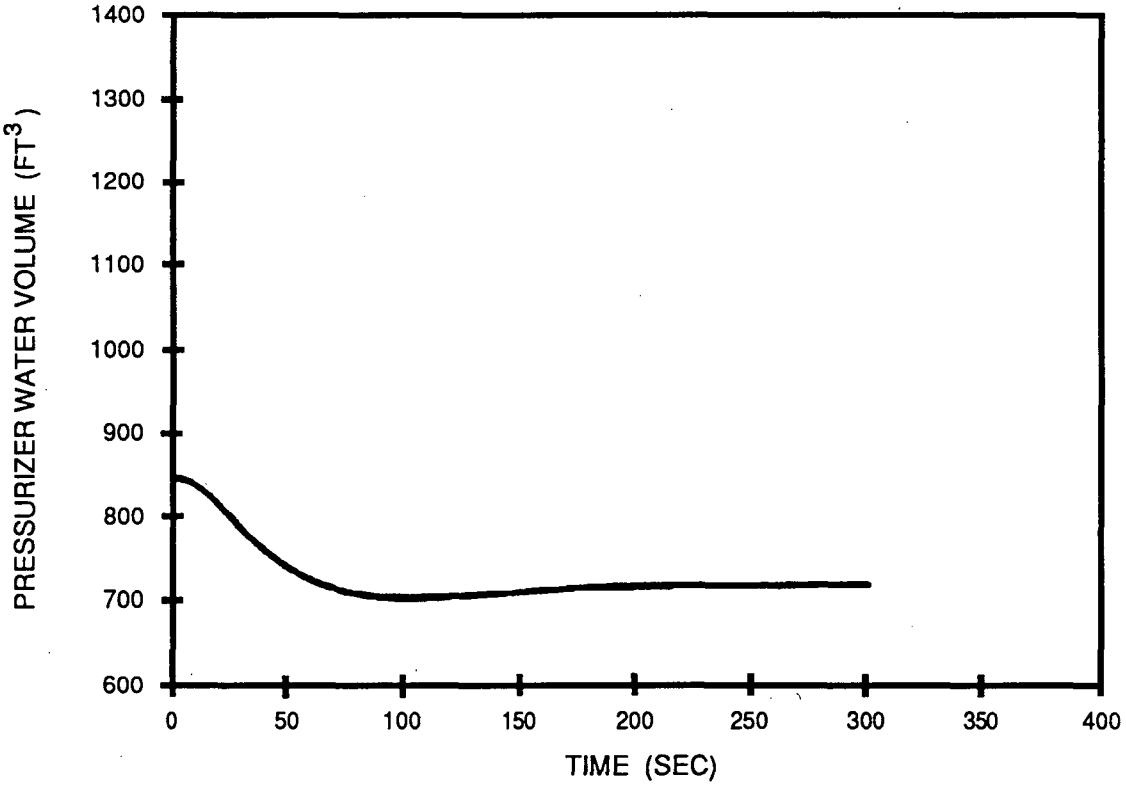


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Figure 15.1-8
EXCESSIVE LOAD INCREASE
PRESSURIZER PRESSURE
AND WATER VOLUME
BOL MANUAL ROD CONTROL

JULY 1993

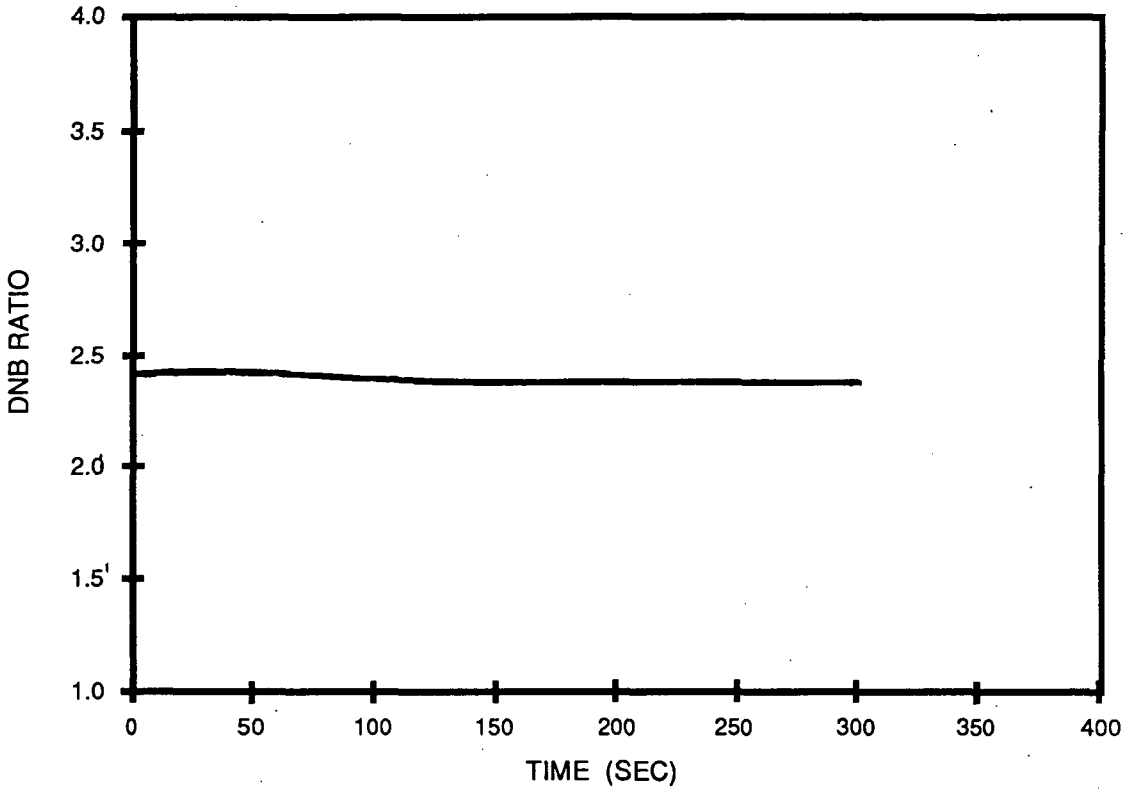

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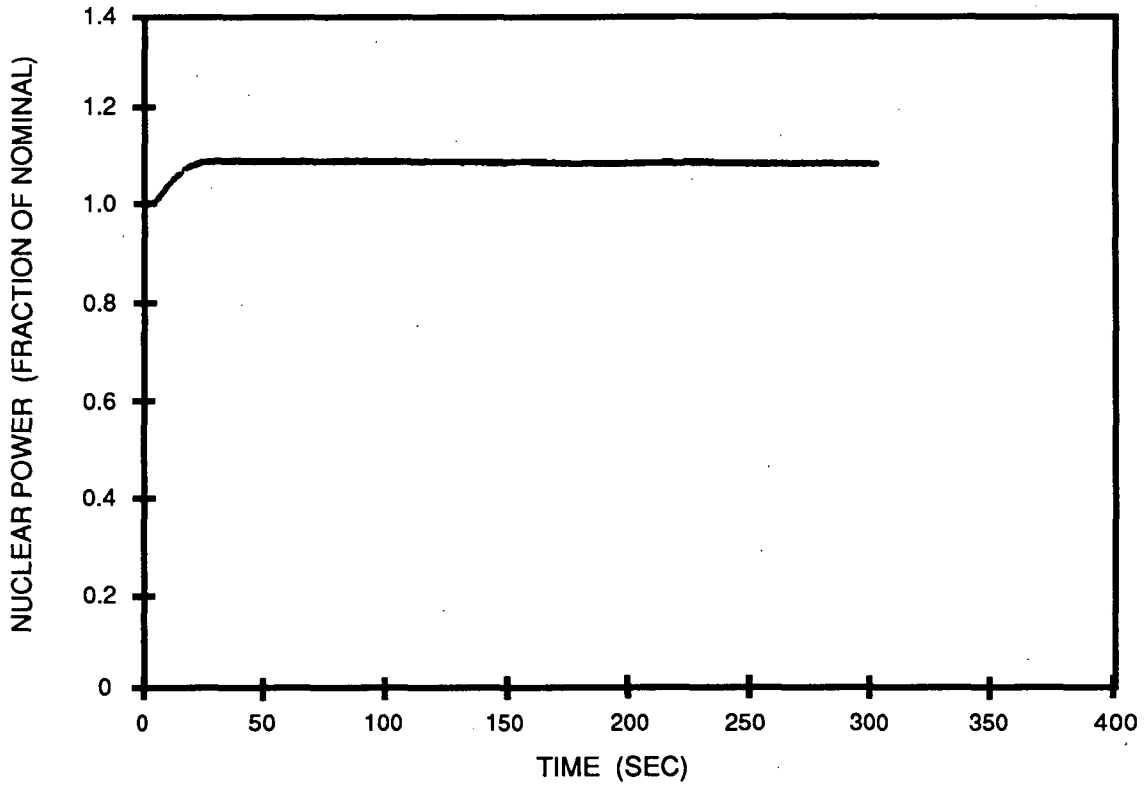
Figure 15.1-9

EXCESSIVE LOAD INCREASE
DNBR VERSUS TIME
BOL MANUAL ROD CONTROL

JULY 1993



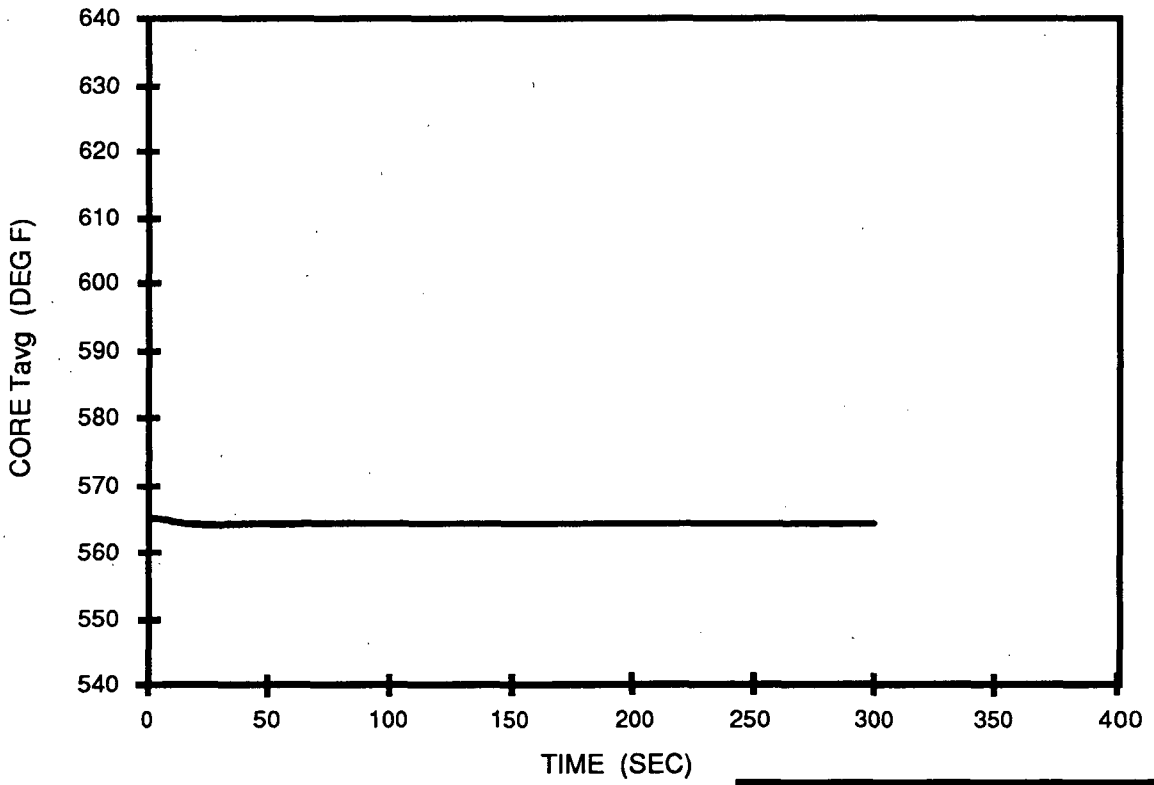


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Figure 15.1-10

EXCESSIVE LOAD INCREASE
NUCLEAR POWER AND CORE Tavg
EOL MANUAL ROD CONTROL

JULY 1993

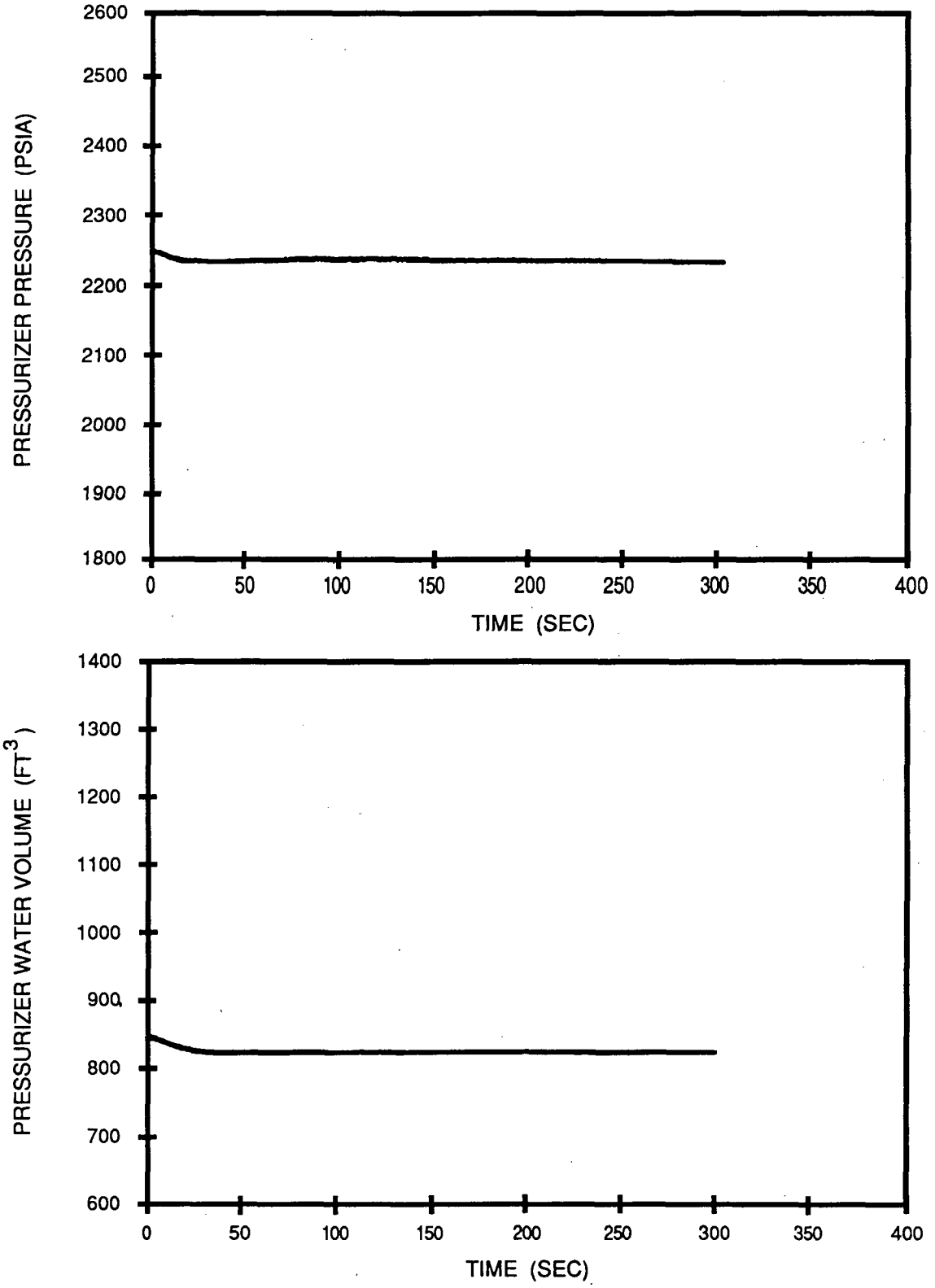

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Figure 15.1-11
EXCESSIVE LOAD INCREASE
PRESSURIZER PRESSURE
AND WATER VOLUME
EOL MANUAL ROD CONTROL

JULY 1993

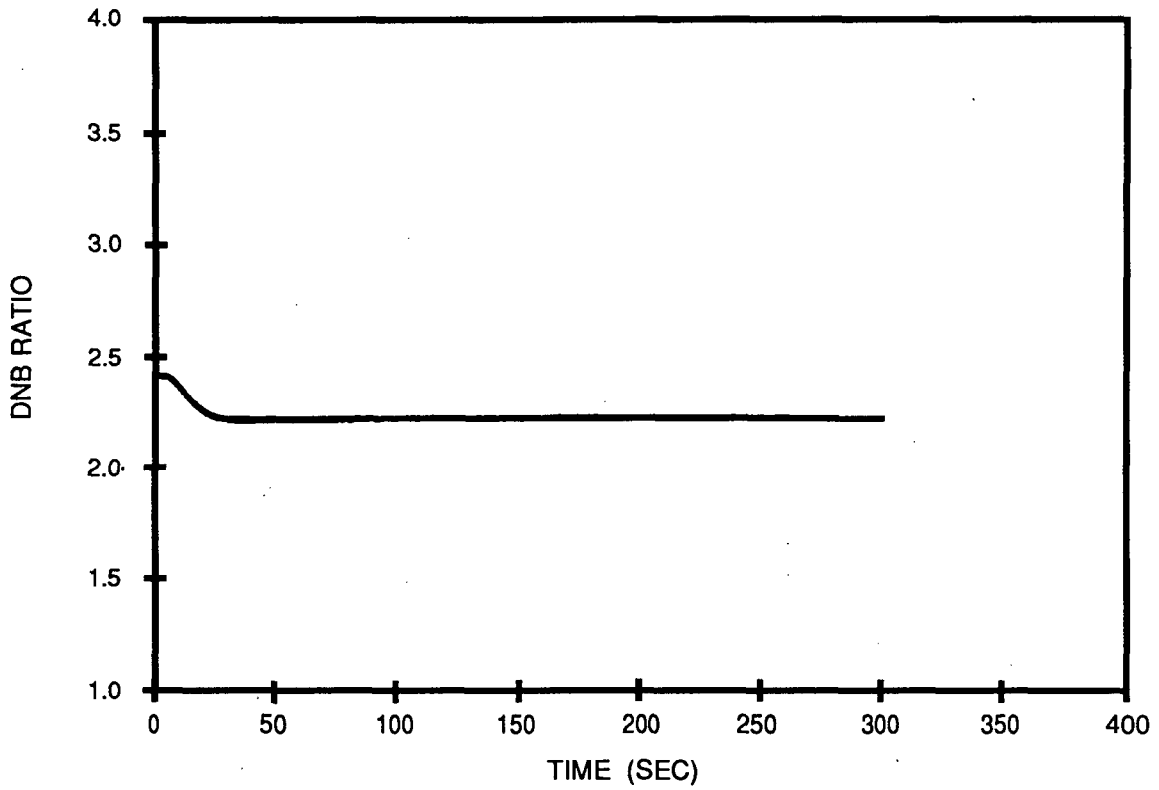

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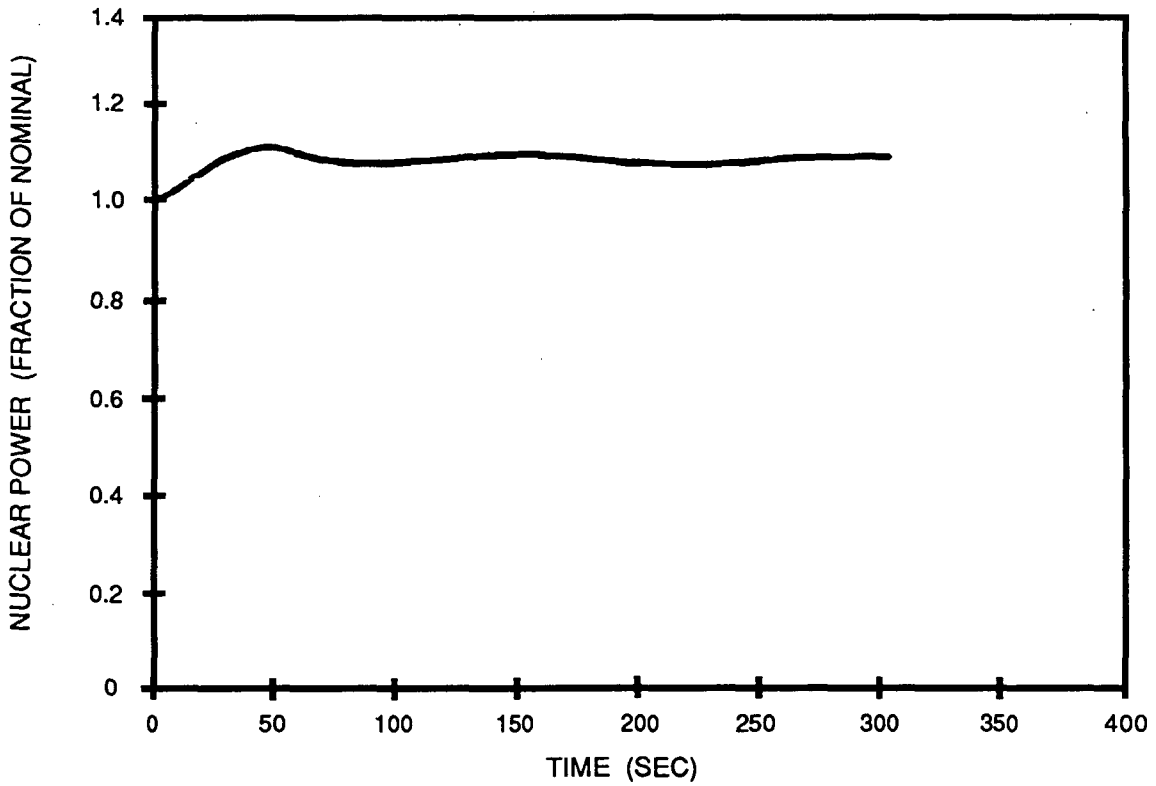
Figure 15.1-12
EXCESSIVE LOAD INCREASE
DNBR VERSUS TIME
EOL MANUAL ROD CONTROL

JULY 1993



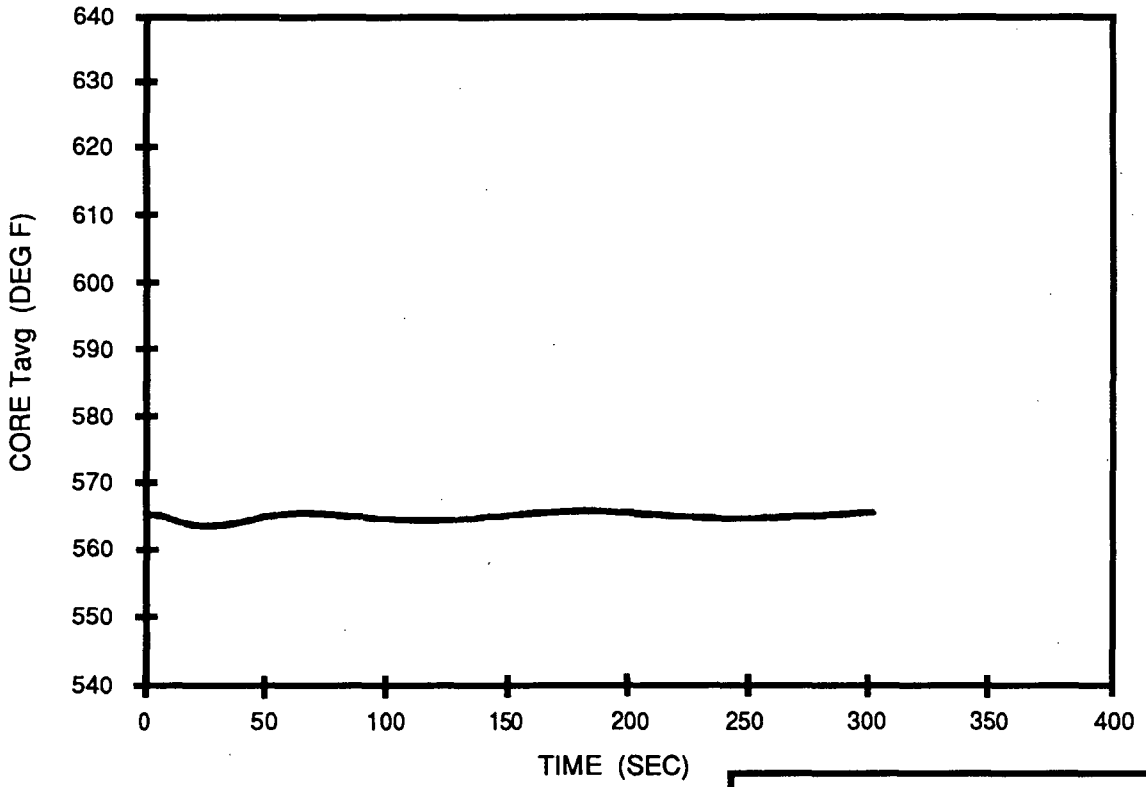


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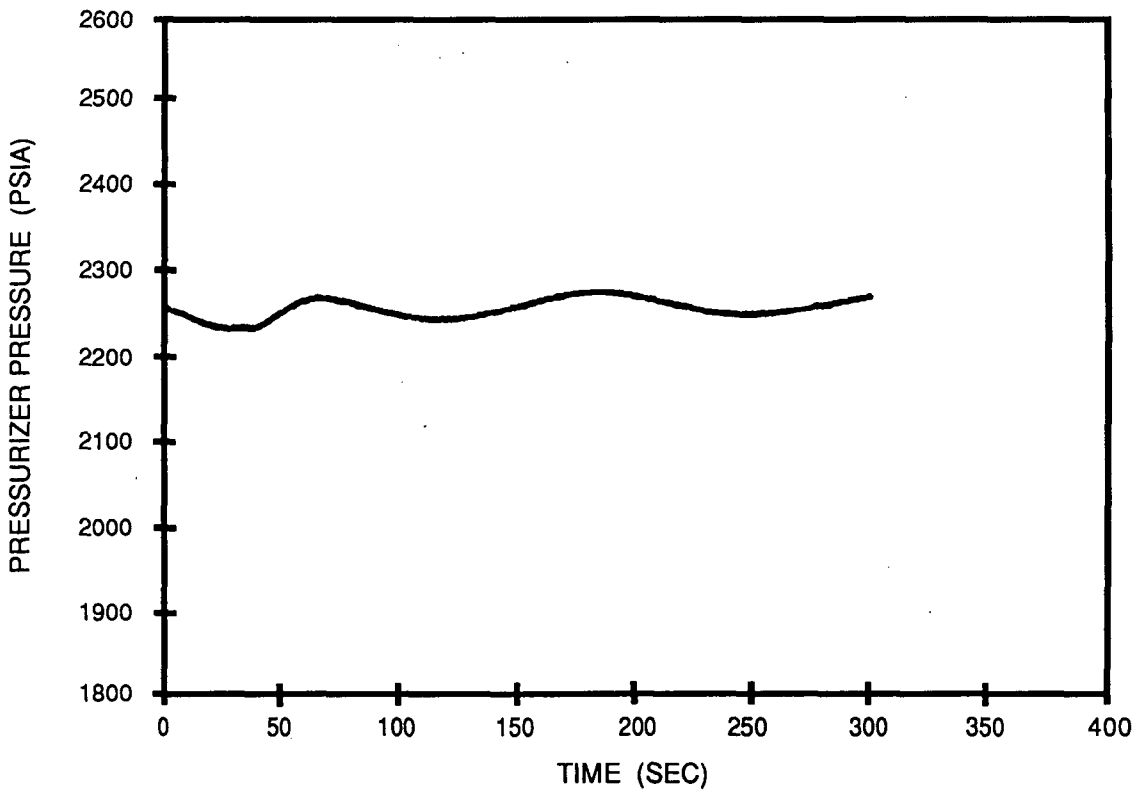
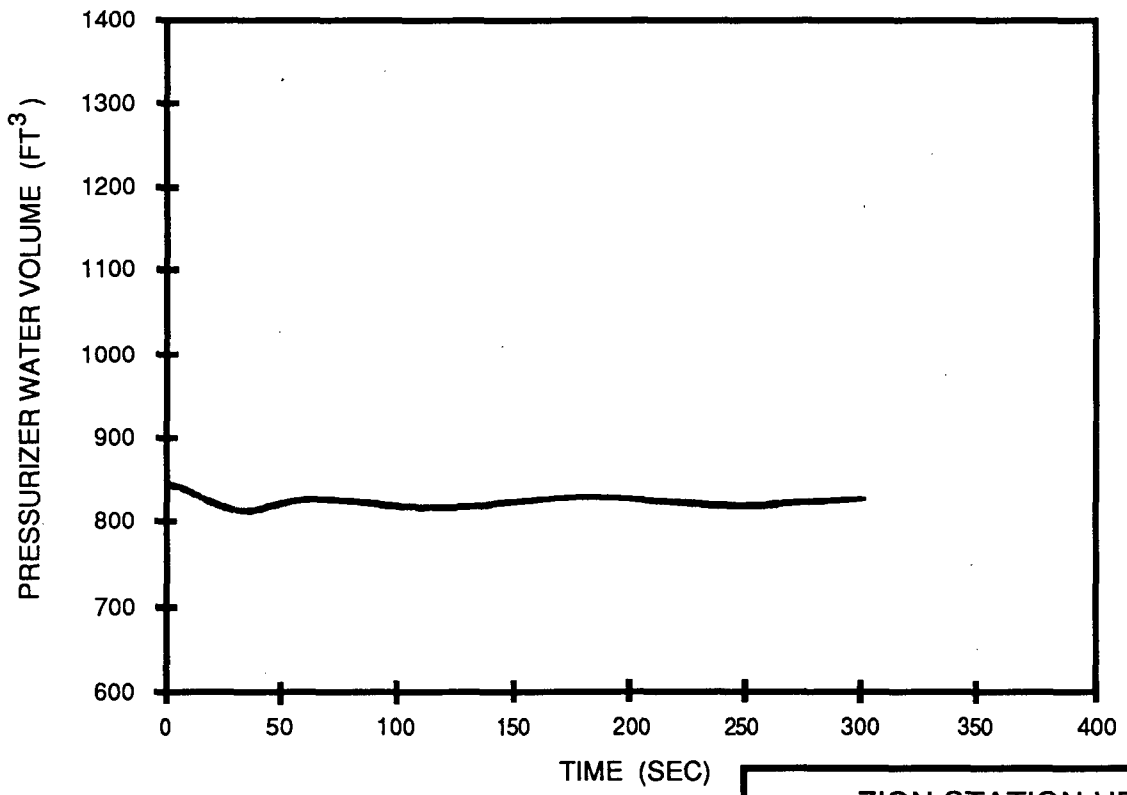
Figure 15.1-13

EXCESSIVE LOAD INCREASE
NUCLEAR POWER AND CORE Tavg
BOL AUTO ROD CONTROL

JULY 1993





ZION STATION UFSAR

Figure 15.1-14
EXCESSIVE LOAD INCREASE
PRESSURIZER PRESSURE
AND WATER VOLUME
BOL AUTO ROD CONTROL

JULY 1993

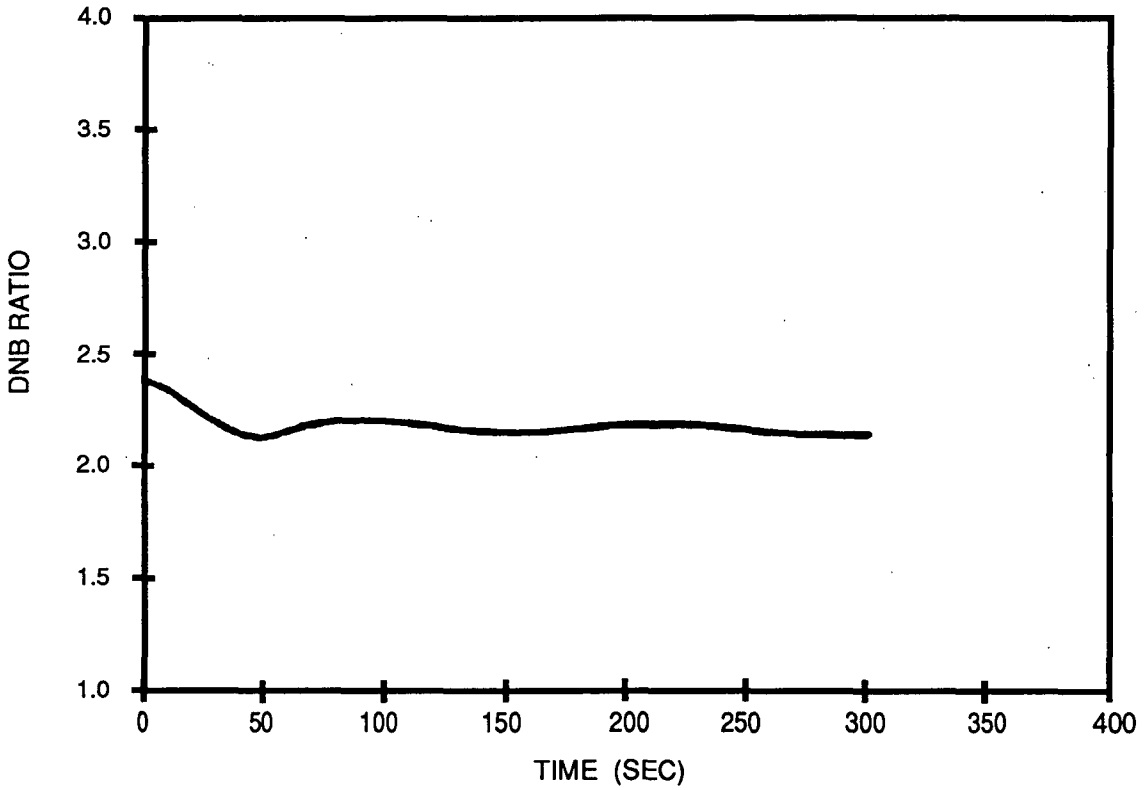


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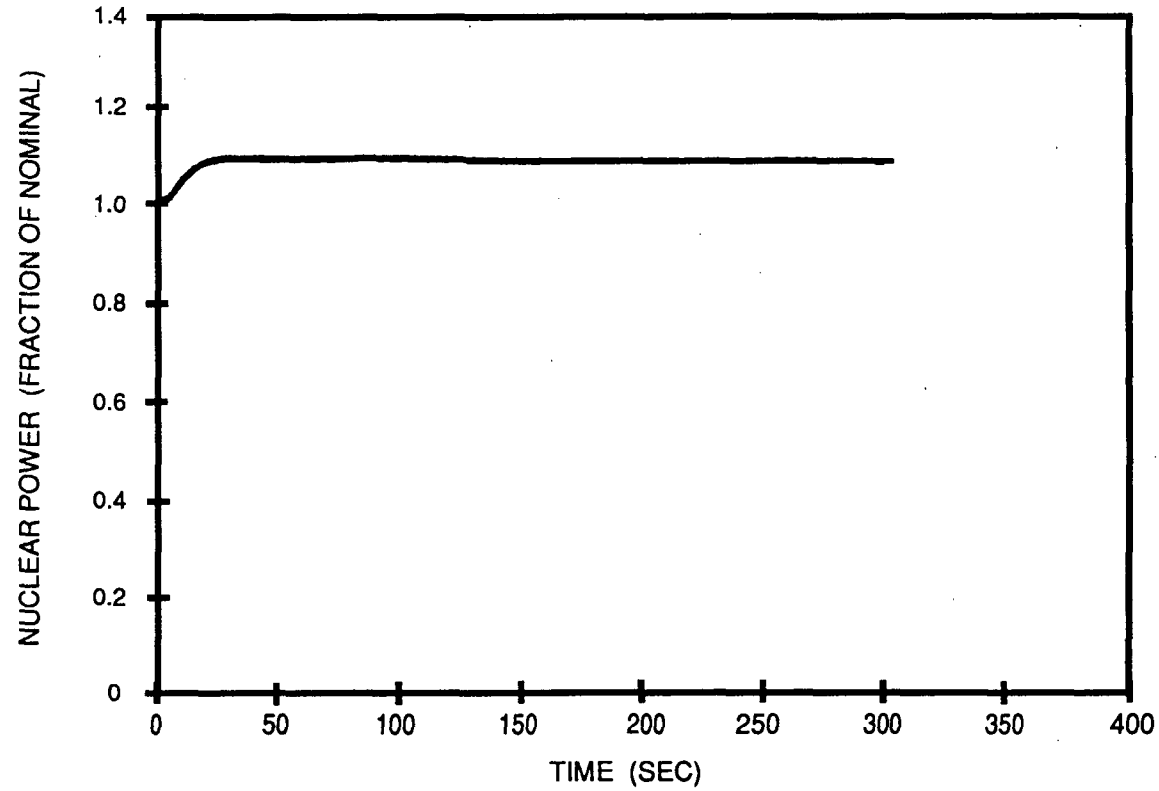
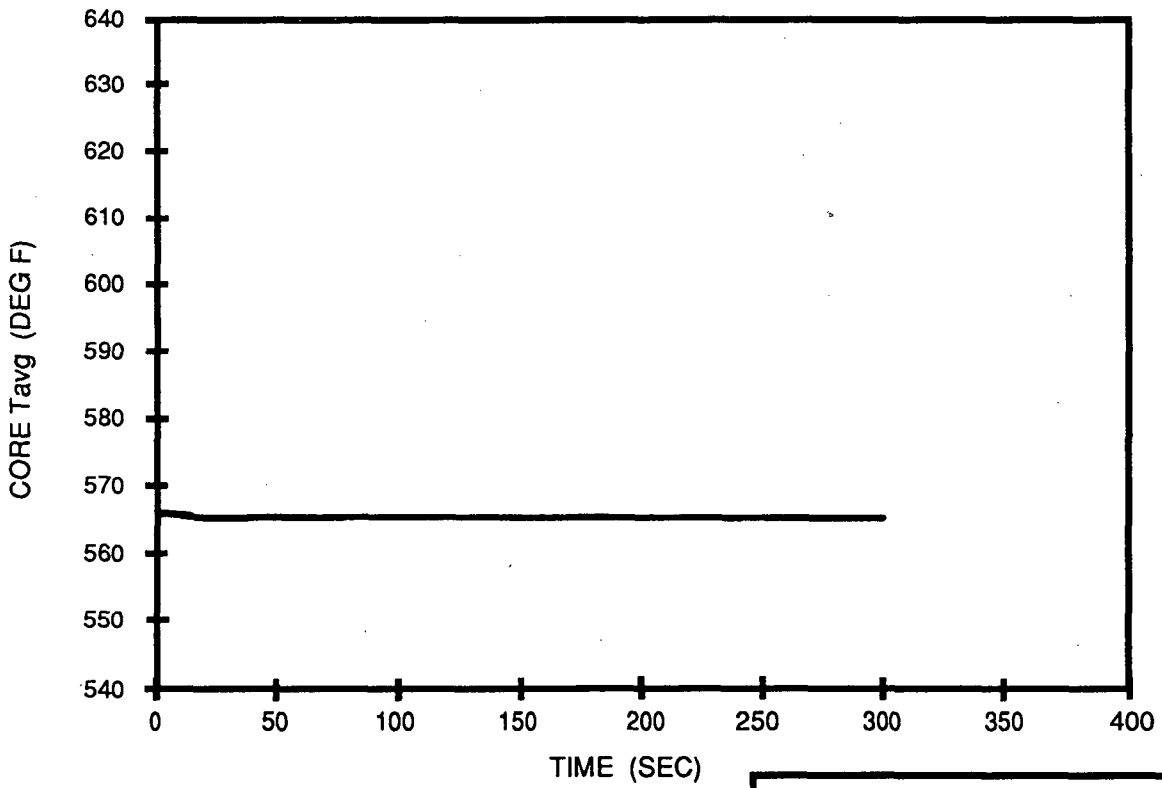
Figure 15.1-15

EXCESSIVE LOAD INCREASE
DNBR VERSUS TIME
BOL AUTO ROD CONTROL

JULY 1993



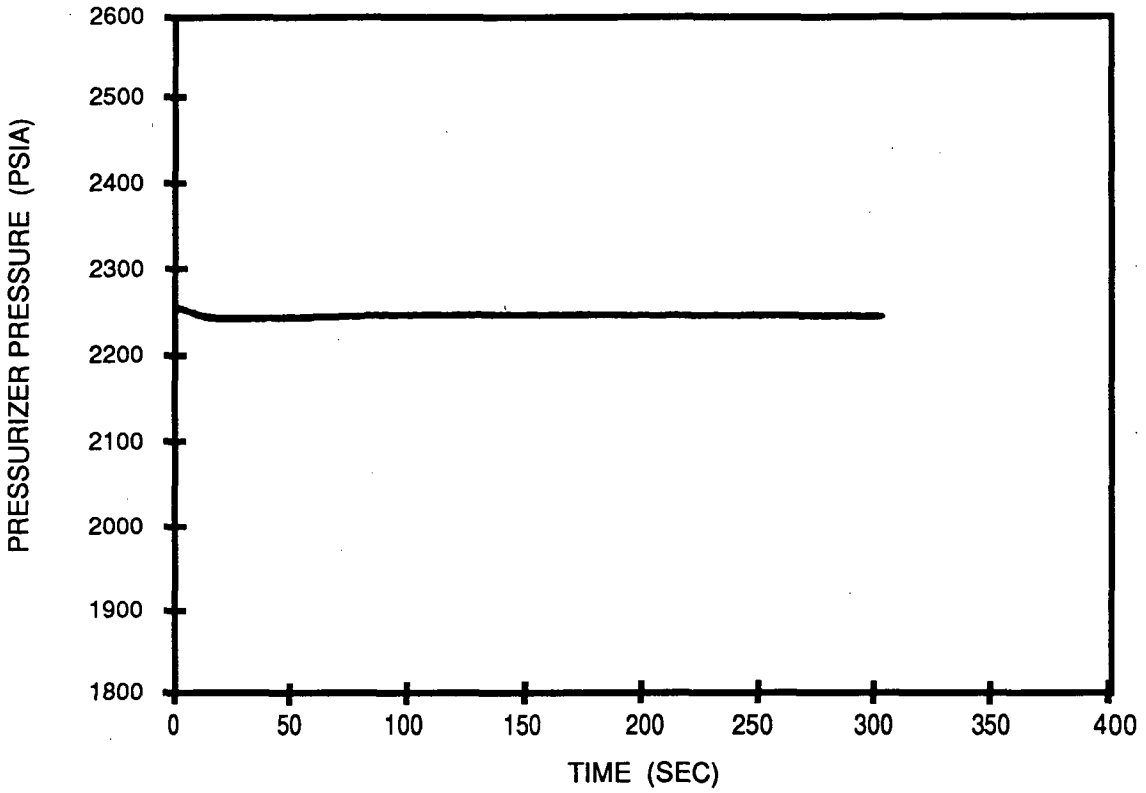


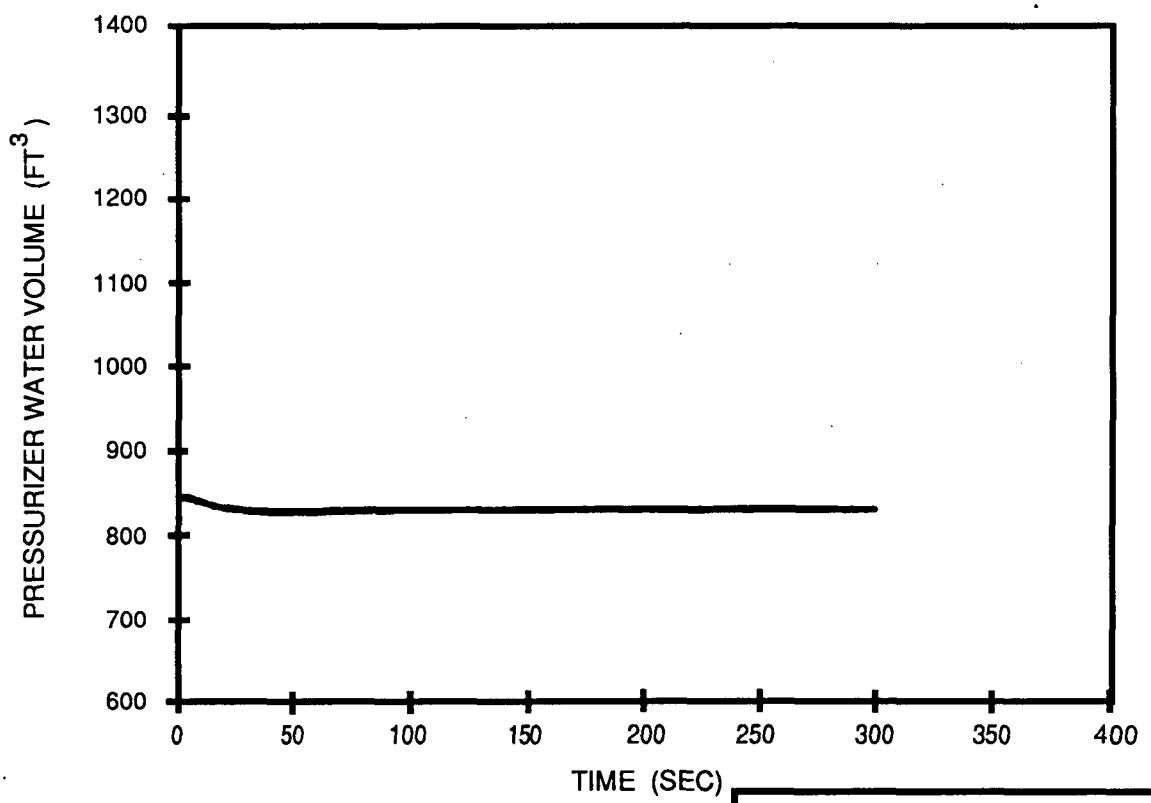
ZION STATION UFSAR

Figure 15.1-16

EXCESSIVE LOAD INCREASE
NUCLEAR POWER AND CORE Tavg
EOL AUTO ROD CONTROL

JULY 1993

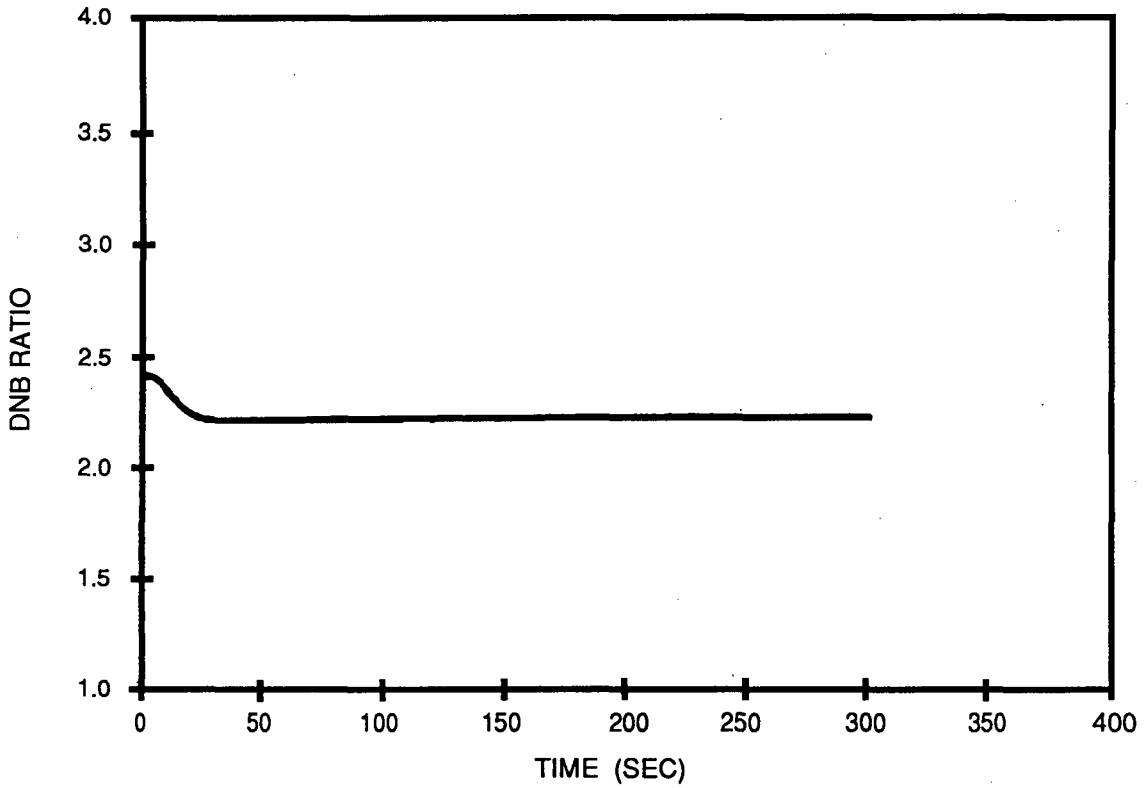




ZION STATION UFSAR

Figure 15.1-17
EXCESSIVE LOAD INCREASE
PRESSURIZER PRESSURE
AND WATER VOLUME
EOL AUTO ROD CONTROL

JULY 1993

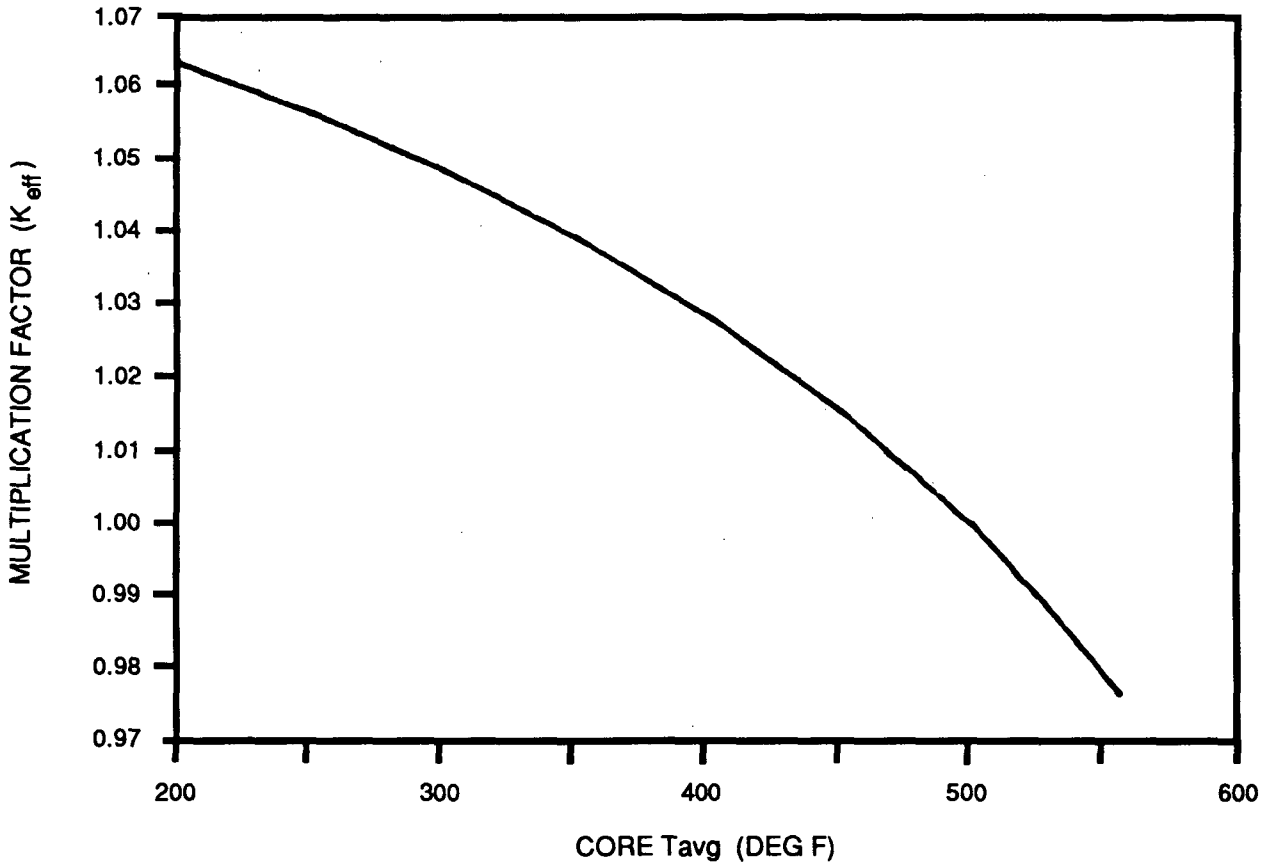


ZION STATION UFSAR

Figure 15.1-18

EXCESSIVE LOAD INCREASE
DNBR VERSUS TIME
EOL AUTO ROD CONTROL

JULY 1993

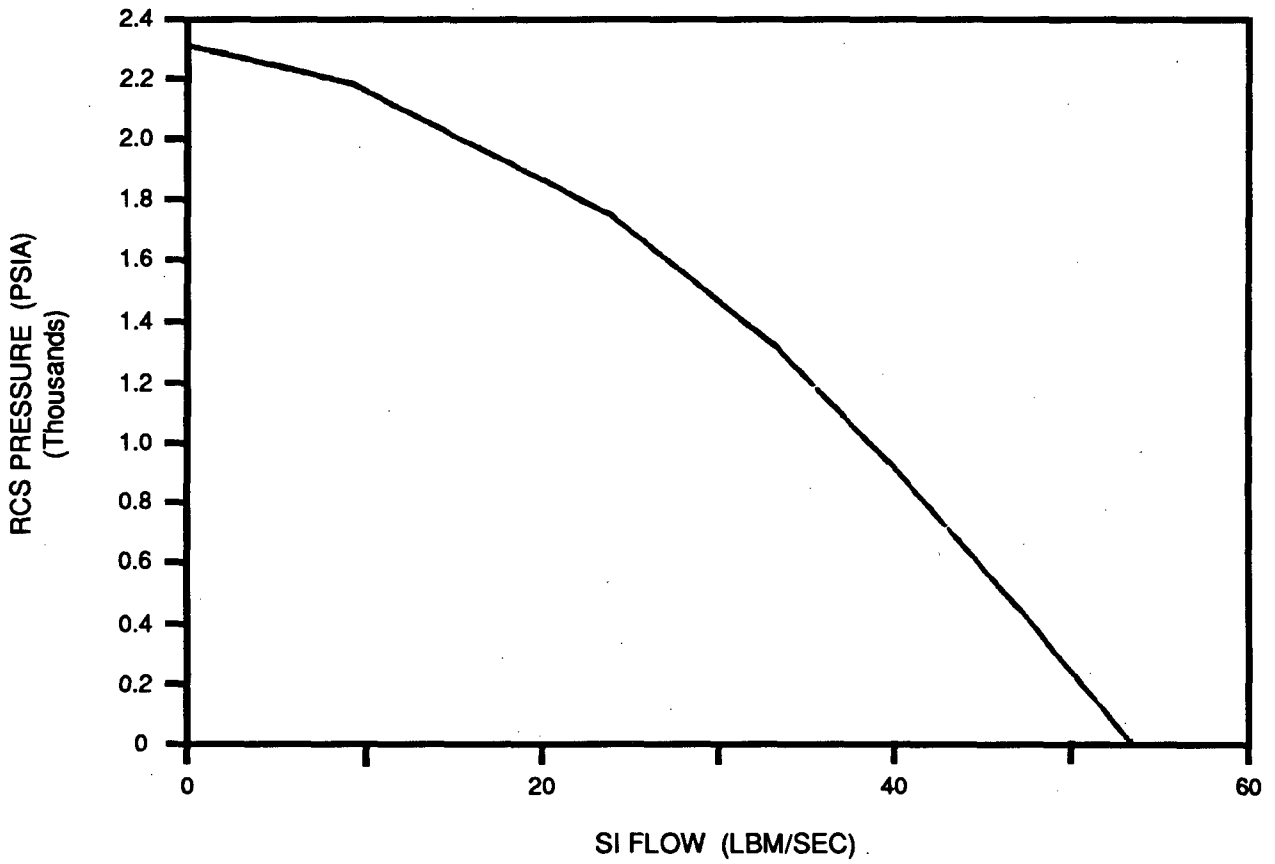


ZION STATION UFSAR

Figure 15.1-19

K_{eff} VERSUS TEMPERATURE

JULY 1993

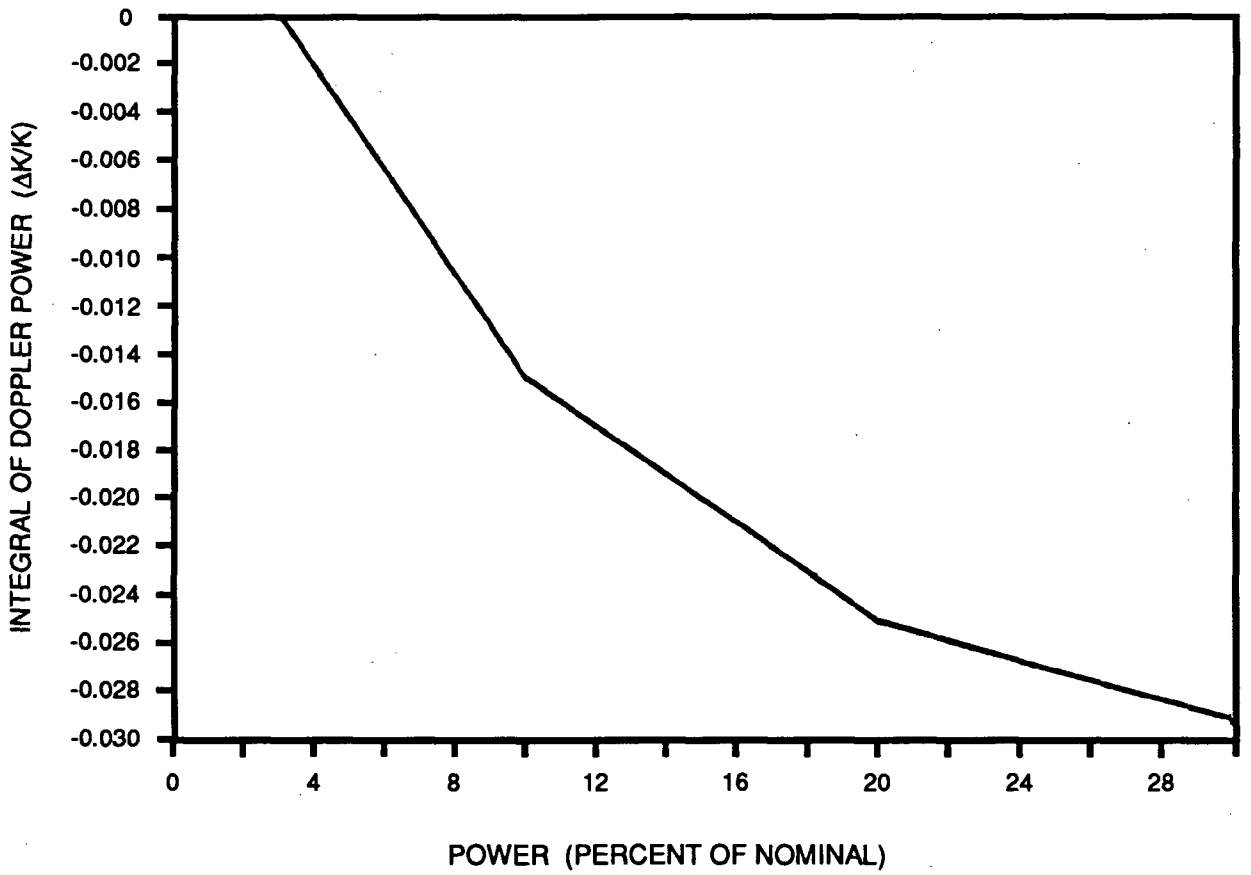


ZION STATION UFSAR

Figure 15.1-20

SAFETY INJECTION FLOW DELIVERED BY ONE CENTRIFUGAL CHARGING PUMP

JULY 1993

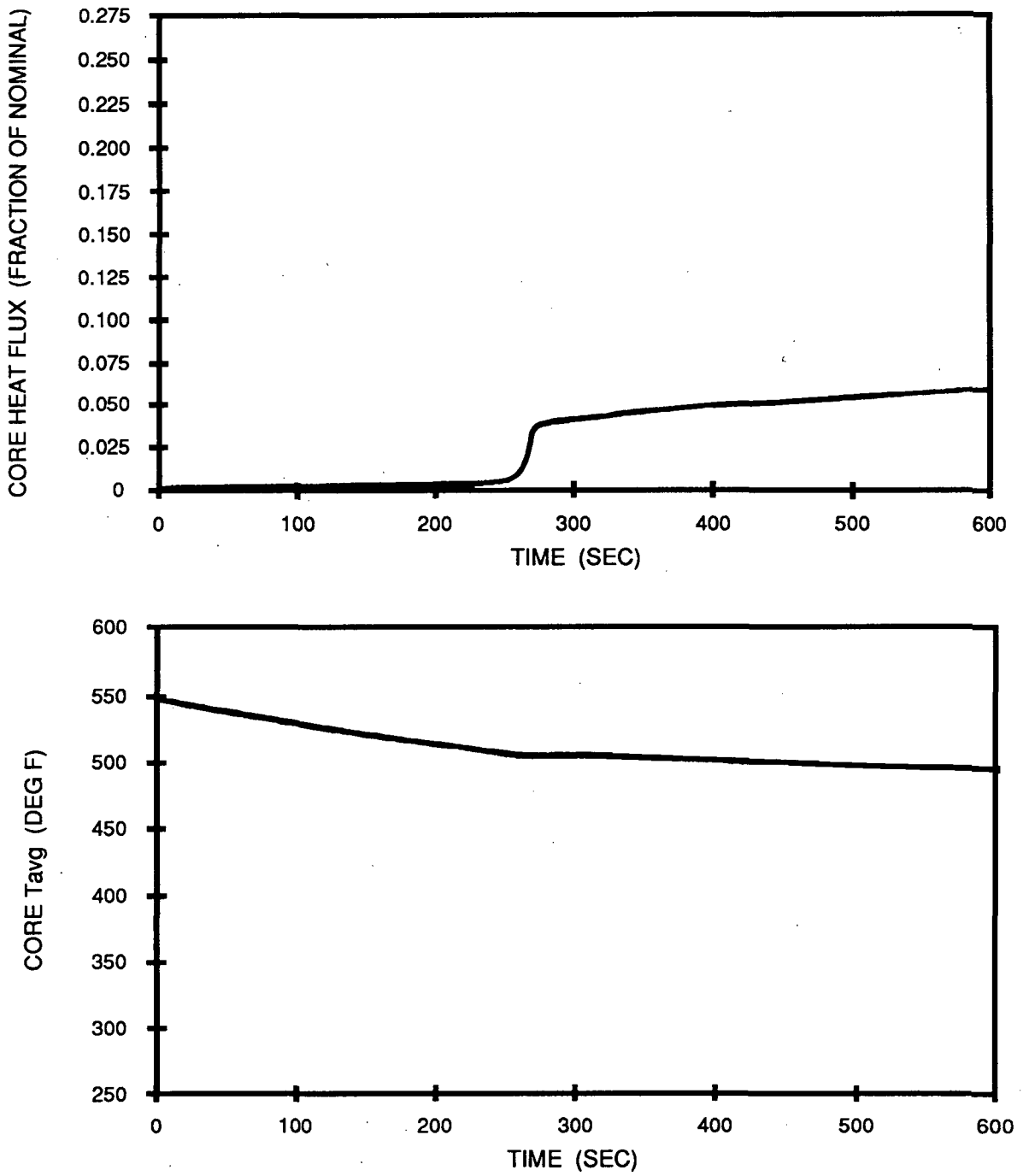


ZION STATION UFSAR

Figure 15.1-21

DOPPLER POWER FEEDBACK

JULY 1993

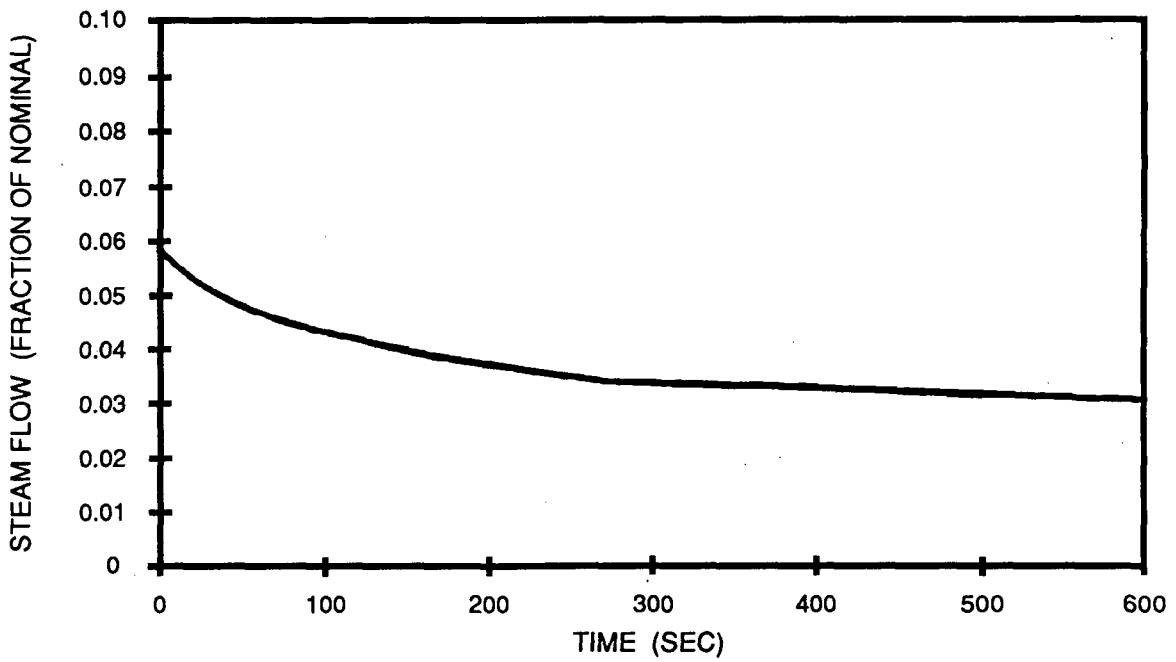


ZION STATION UFSAR

Figure 15.1-22

ACCIDENTAL DEPRESSURIZATION
OF THE STEAM SYSTEM
CORE HEAT FLUX AND Tavg

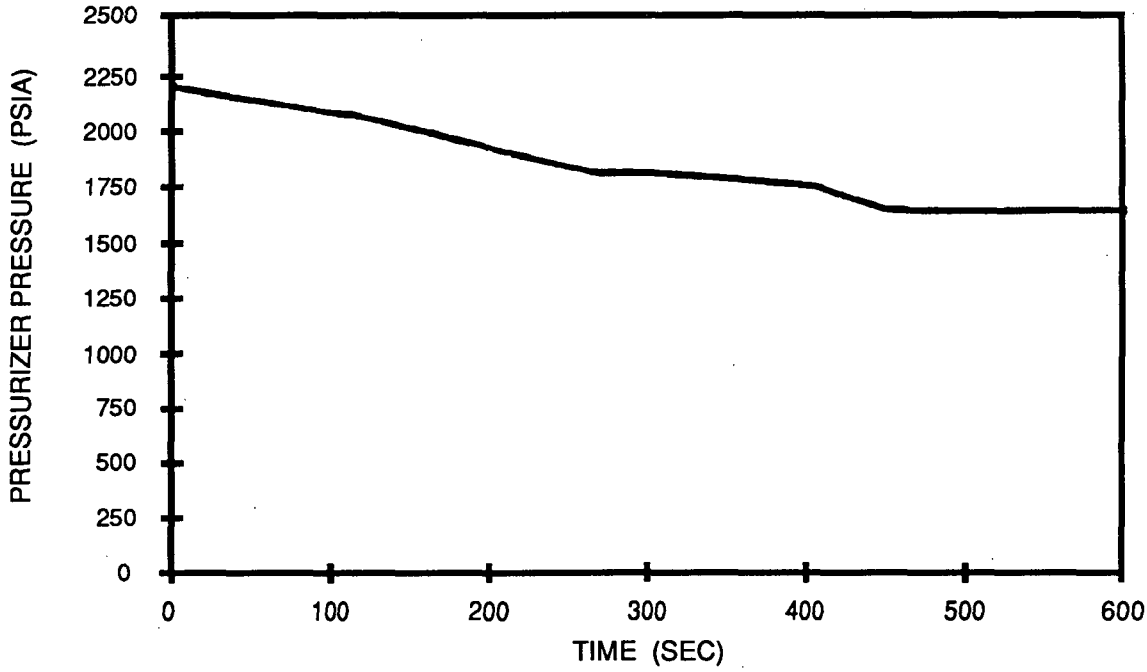
JULY 1993

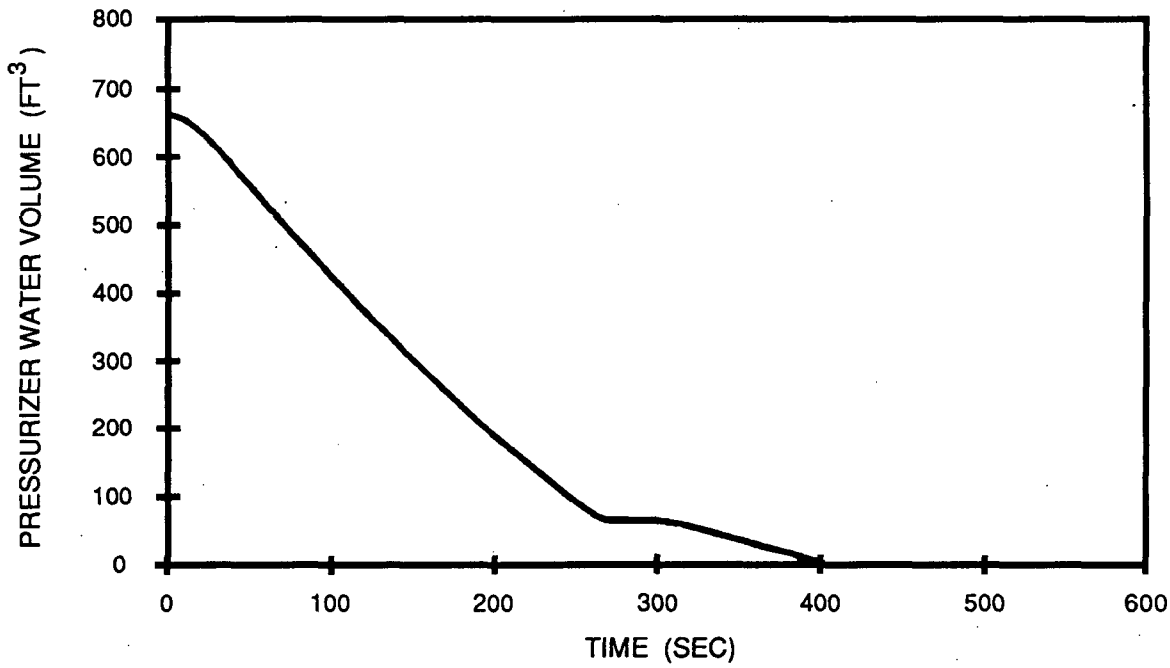


ZION STATION UFSAR

Figure 15.1-23
ACCIDENTAL DEPRESSURIZATION
OF THE STEAM SYSTEM
STEAM FLOW

JULY 1993



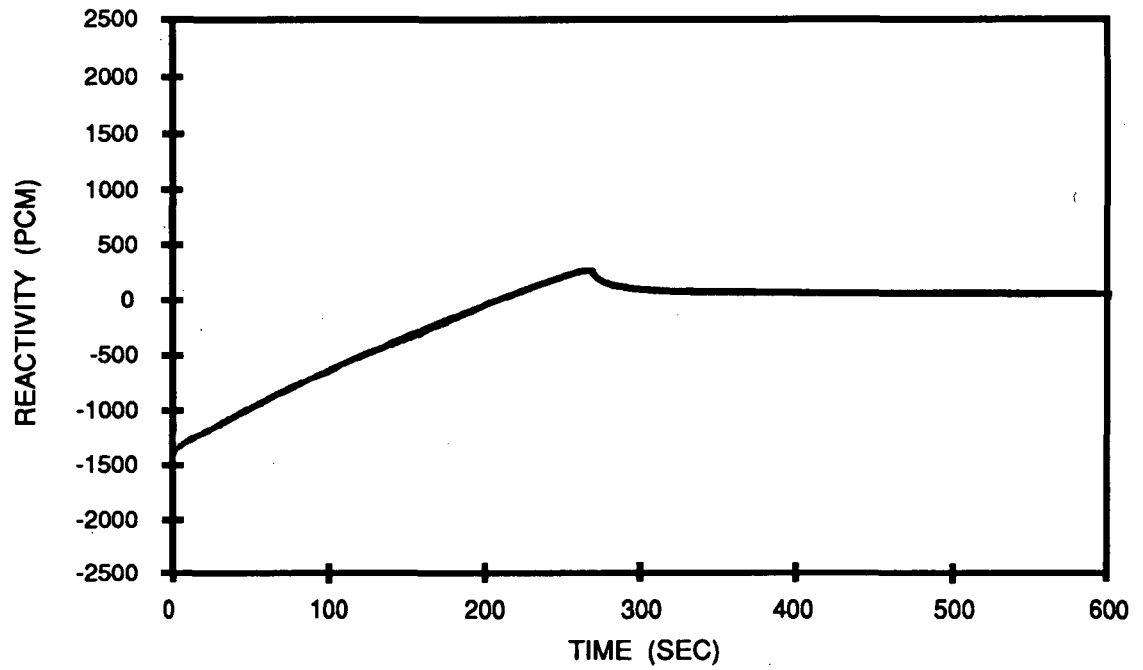


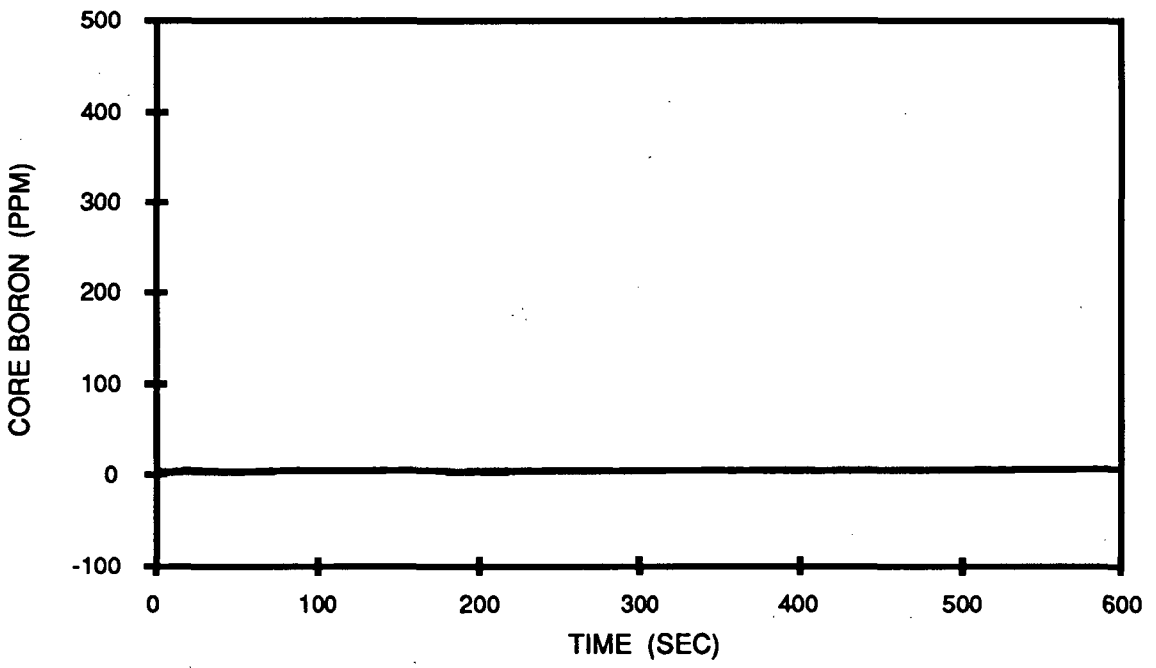
ZION STATION UFSAR

Figure 15.1-24

ACCIDENTAL DEPRESSURIZATION OF THE STEAM SYSTEM PRESSURIZER PRESSURE AND WATER VOLUME

JULY 1993

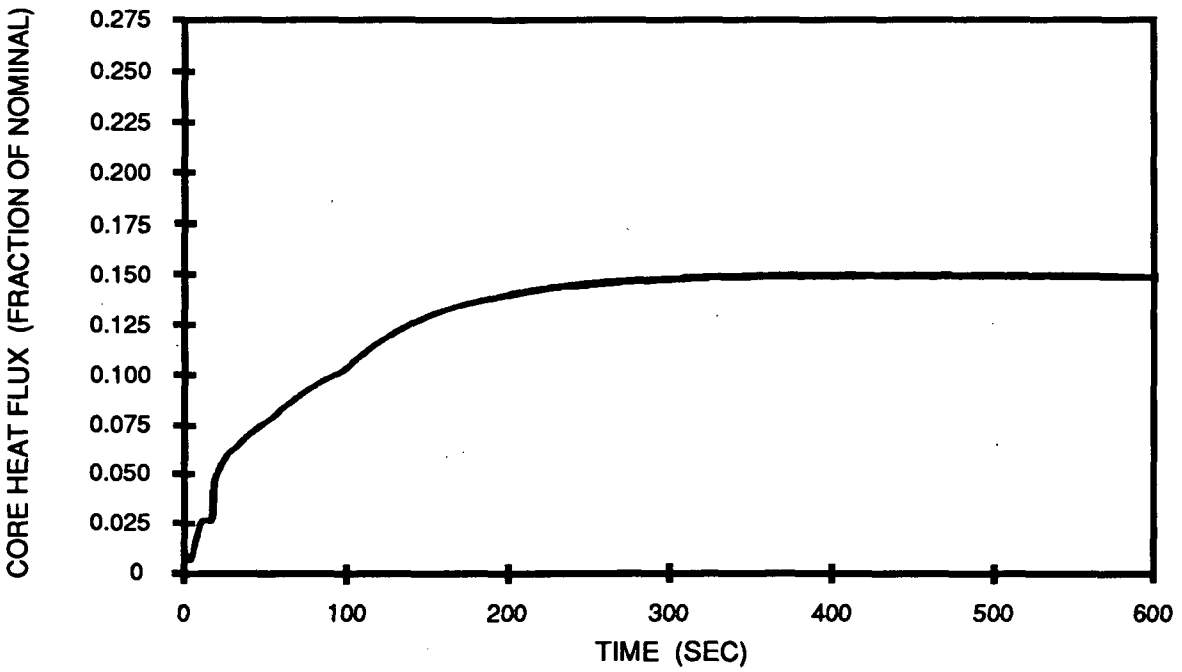


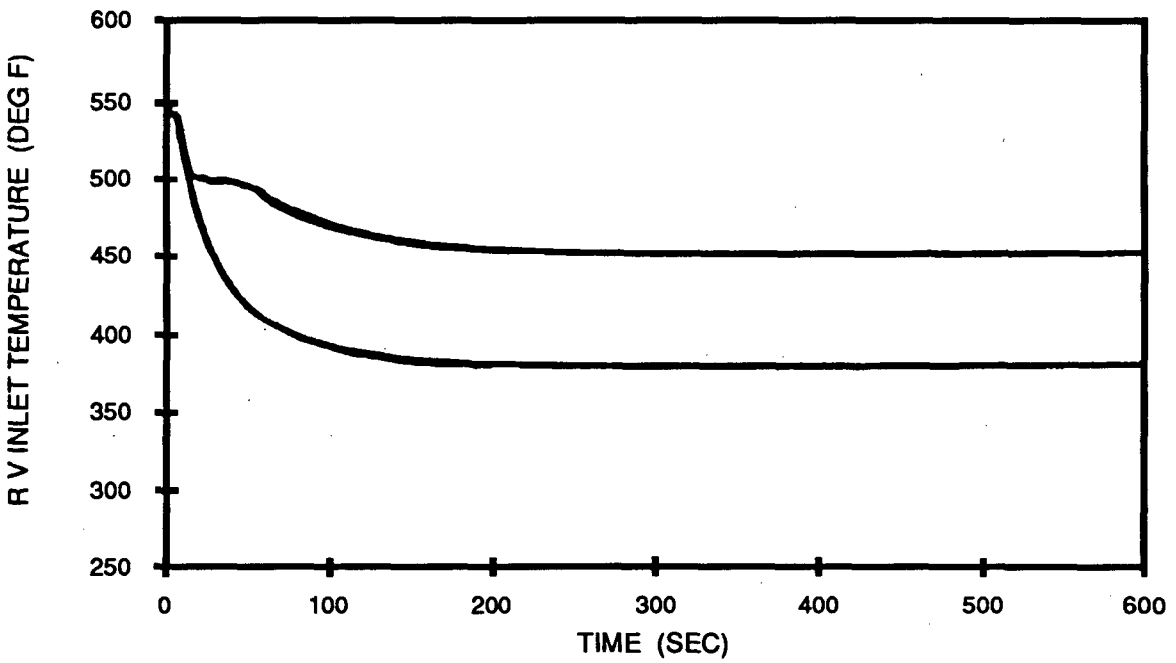


ZION STATION UFSAR

Figure 15.1-25
ACCIDENTAL DEPRESSURIZATION
OF THE STEAM SYSTEM
REACTIVITY AND
CORE BORON CONCENTRATION

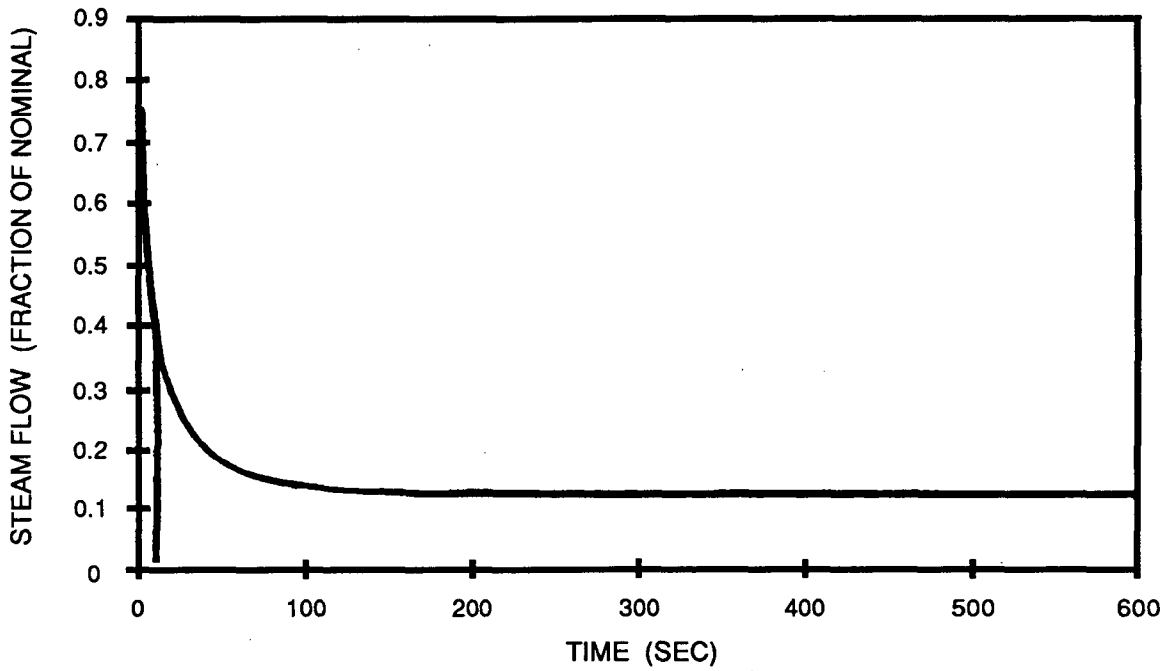
JULY 1993

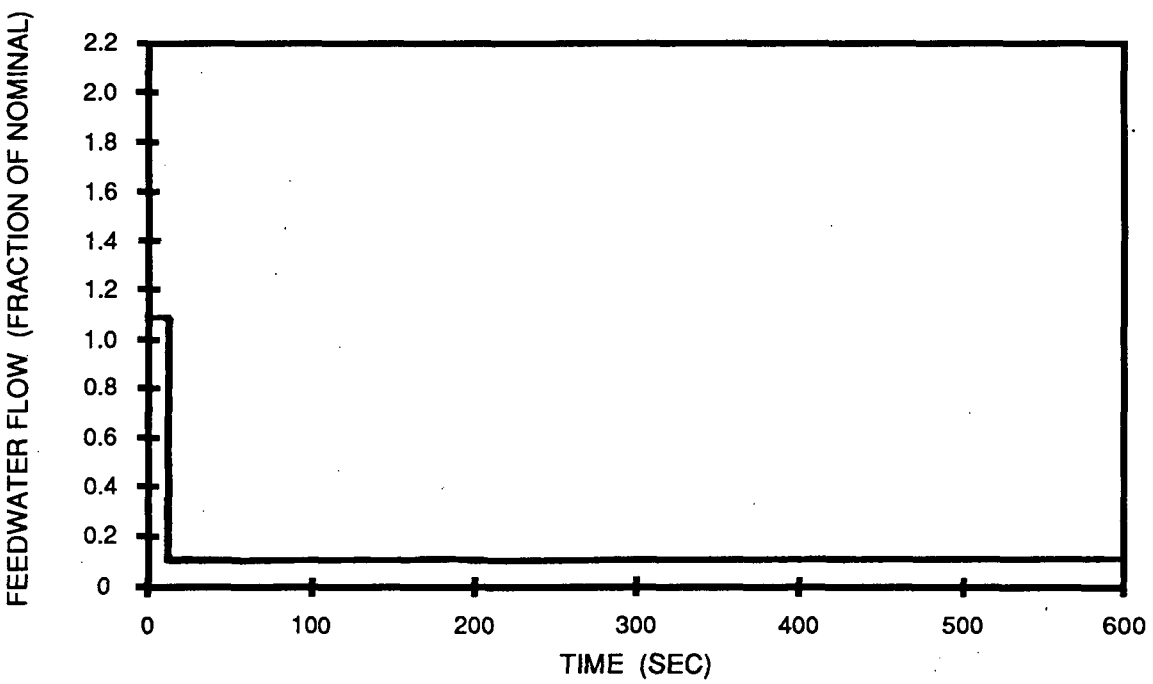




ZION STATION UFSAR

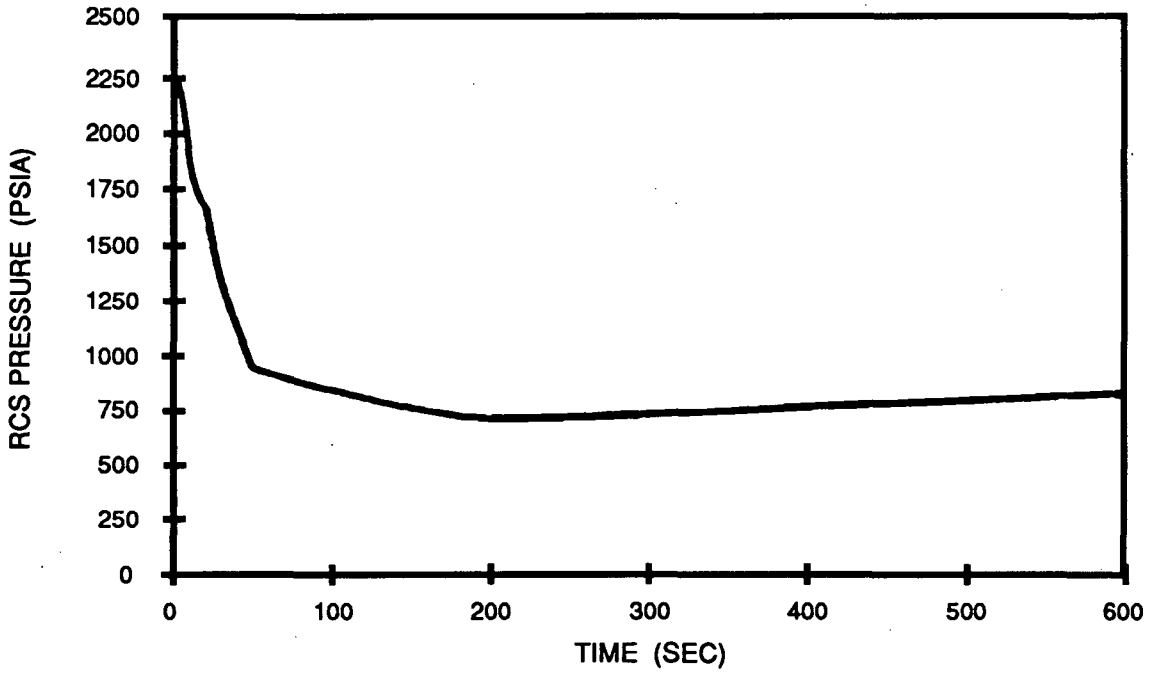
Figure 15.1-26
STEAMLINE BREAK
CORE HEAT FLUX AND
REACTOR VESSEL INLET TEMPERATURE
OUTSIDE CONTAINMENT-WITH POWER
JULY 1993

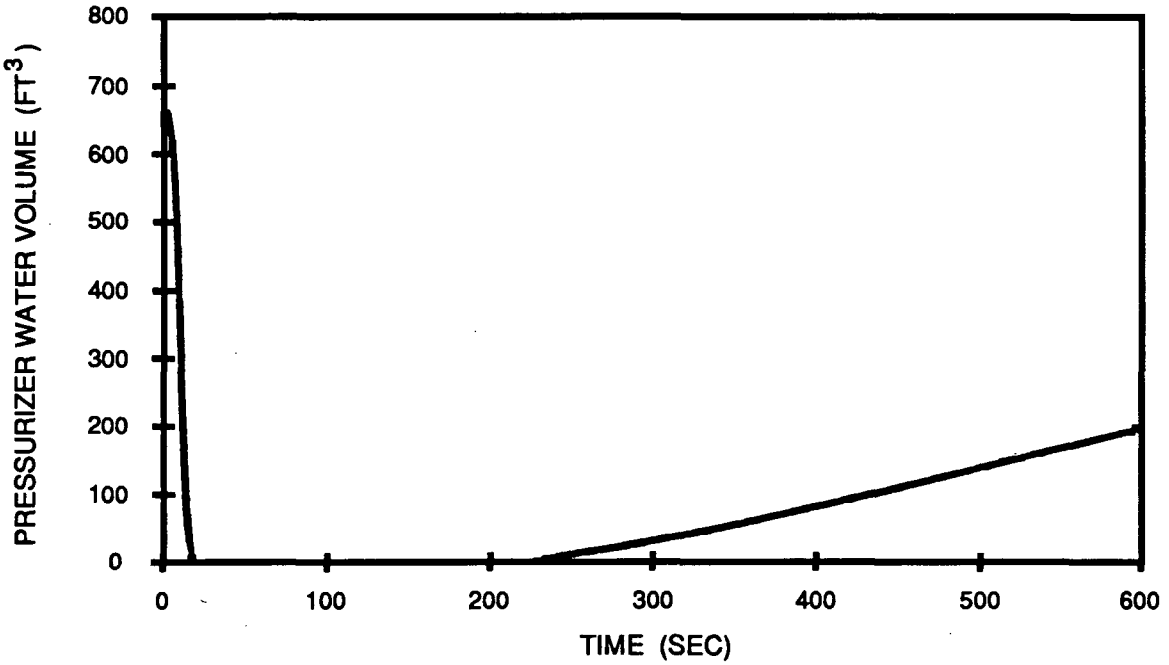




ZION STATION UFSAR

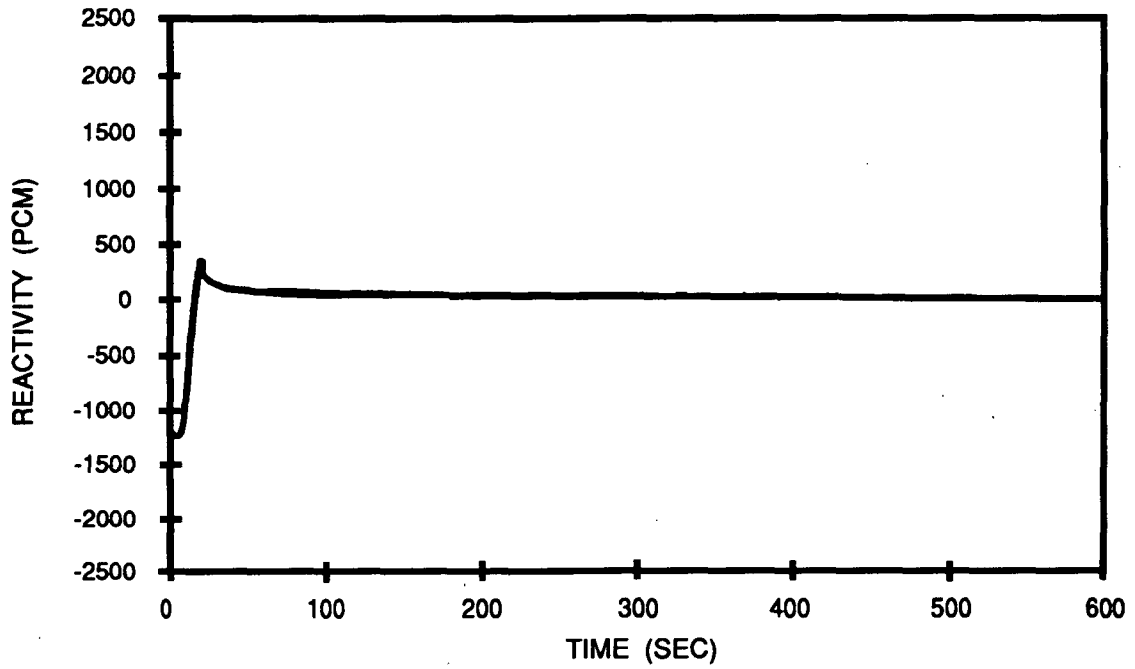
Figure 15.1-27
STEAMLINE BREAK
STEAM FLOW AND
FEEDWATER FLOW
OUTSIDE CONTAINMENT-WITH POWER
JULY 1993

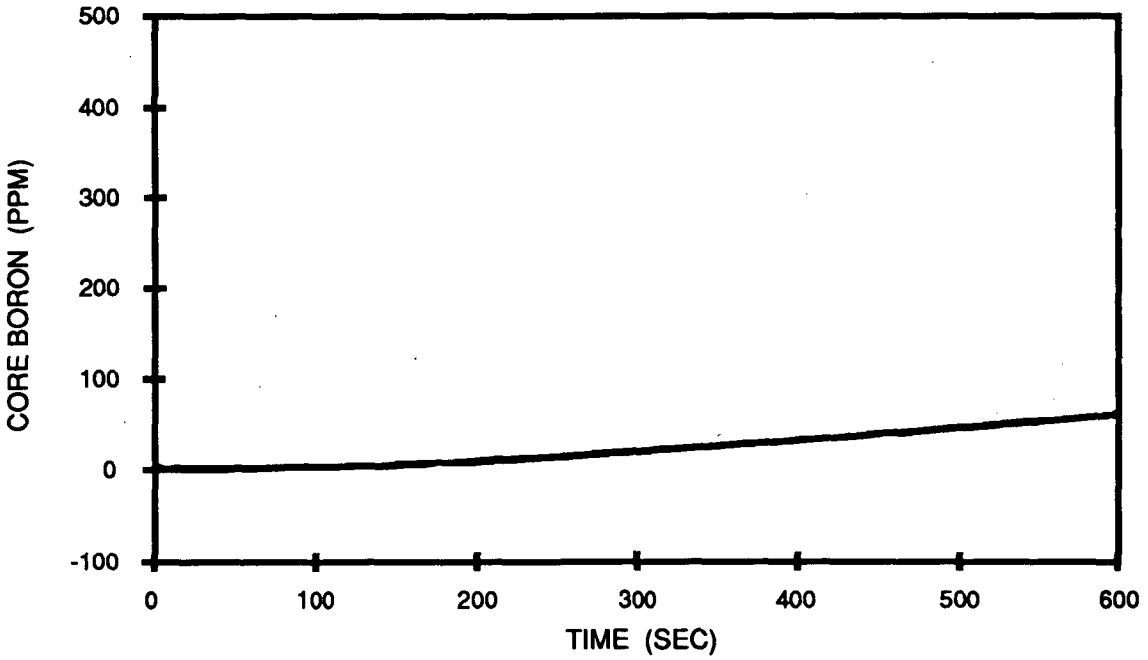




ZION STATION UFSAR

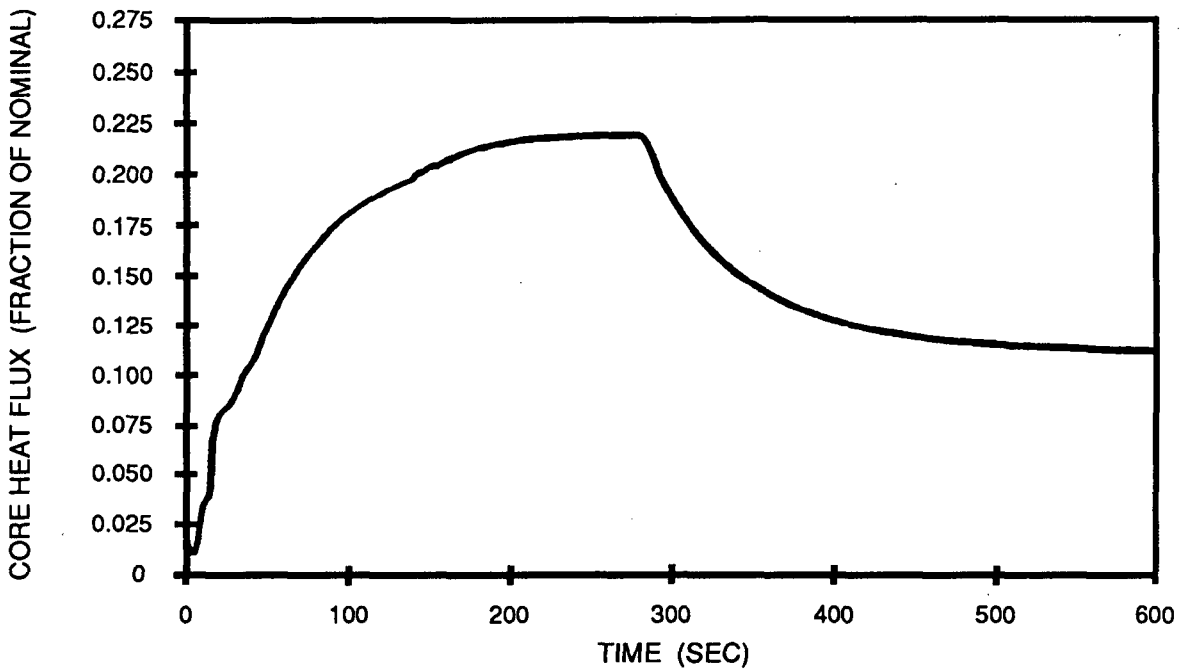
Figure 15.1-28
STEAMLINE BREAK
RCS PRESSURE AND
PRESSURIZER WATER VOLUME
OUTSIDE CONTAINMENT-WITH POWER
JULY 1993

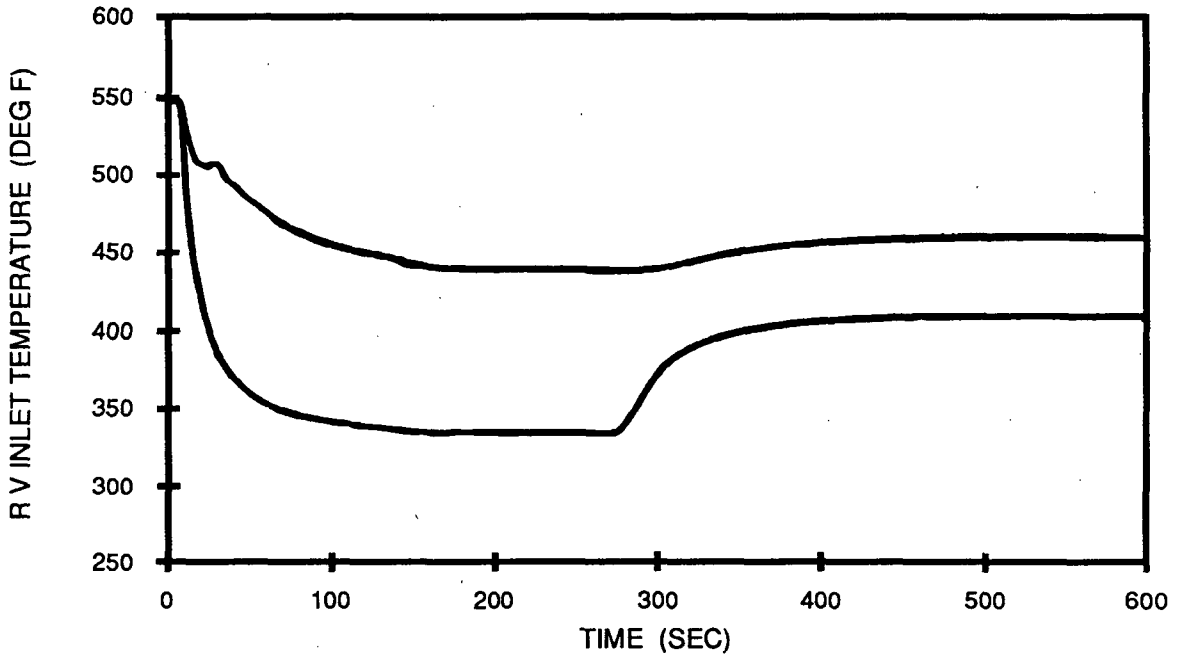




ZION STATION UFSAR

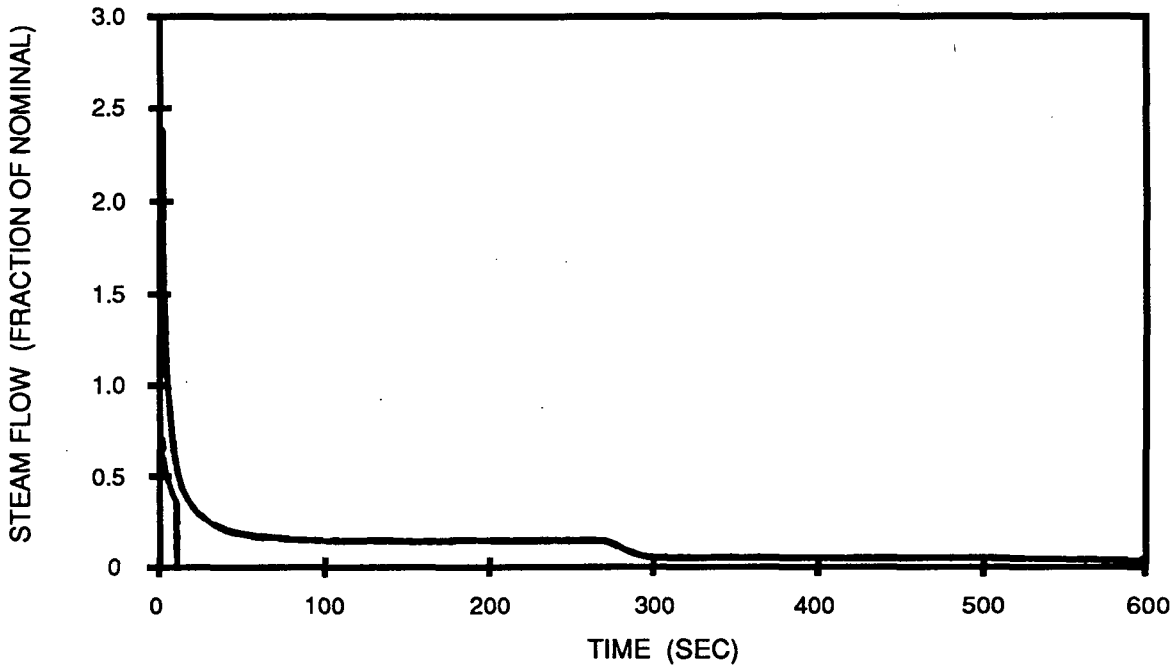
Figure 15.1-29
STEAMLINE BREAK
REACTIVITY AND
CORE BORON CONCENTRATION
OUTSIDE CONTAINMENT-WITH POWER
JULY 1993

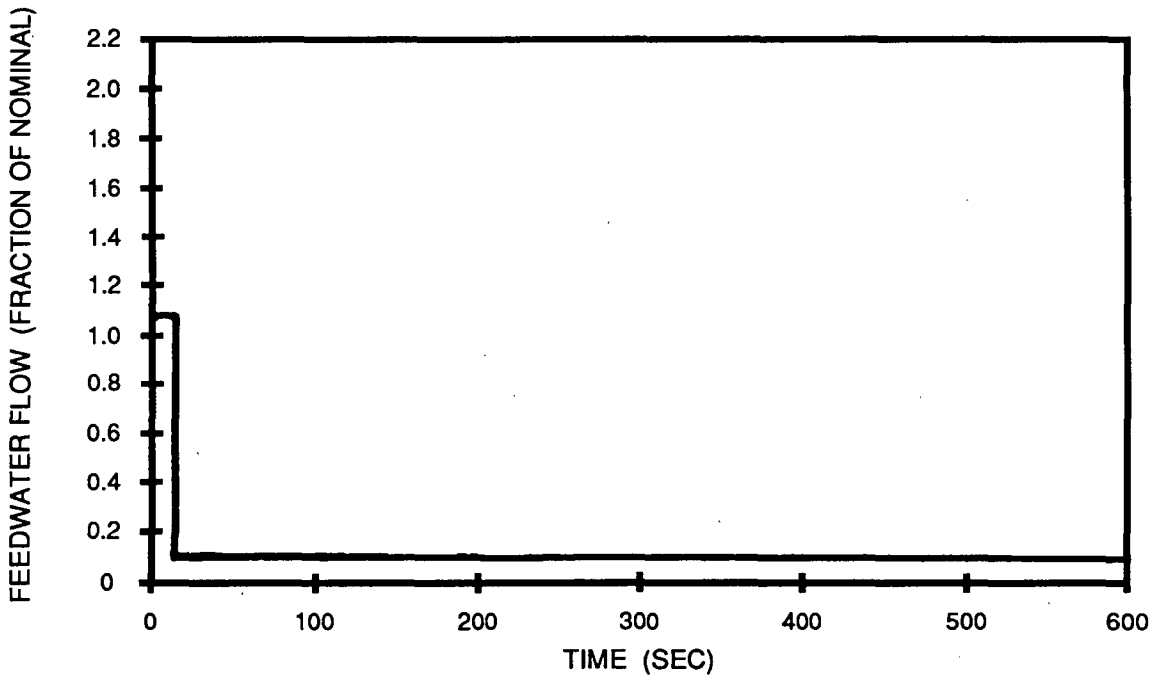




ZION STATION UFSAR

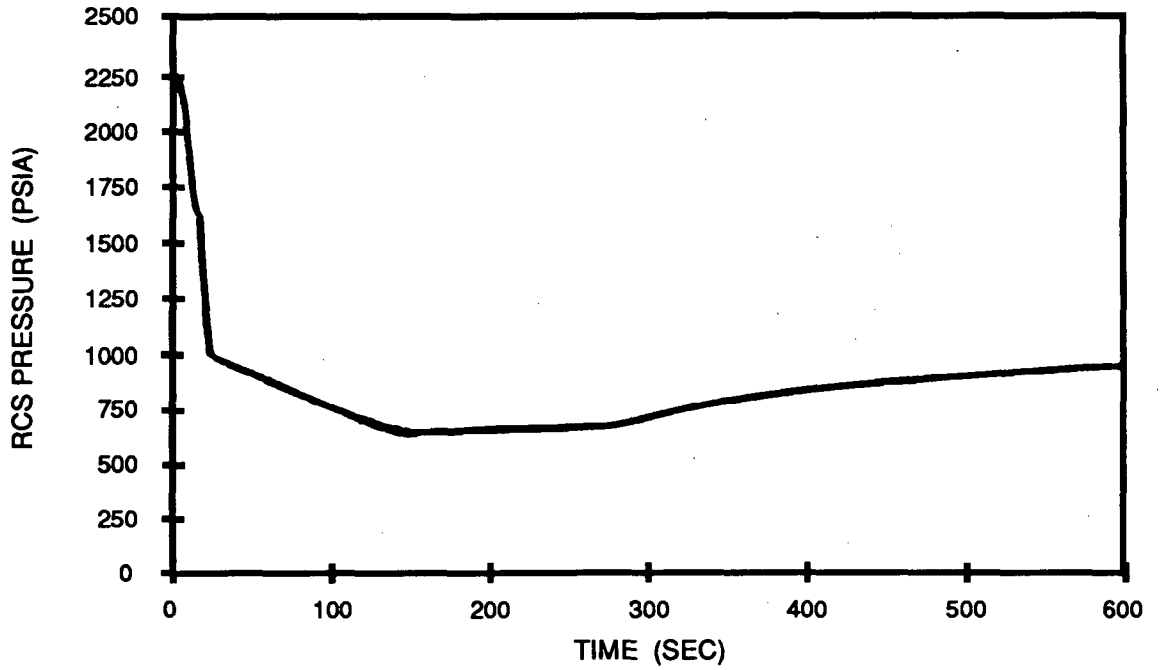
Figure 15.1-30
STEAMLINE BREAK
CORE HEAT FLUX AND
REACTOR VESSEL INLET TEMPERATURE
INSIDE CONTAINMENT-WITH POWER
JULY 1993

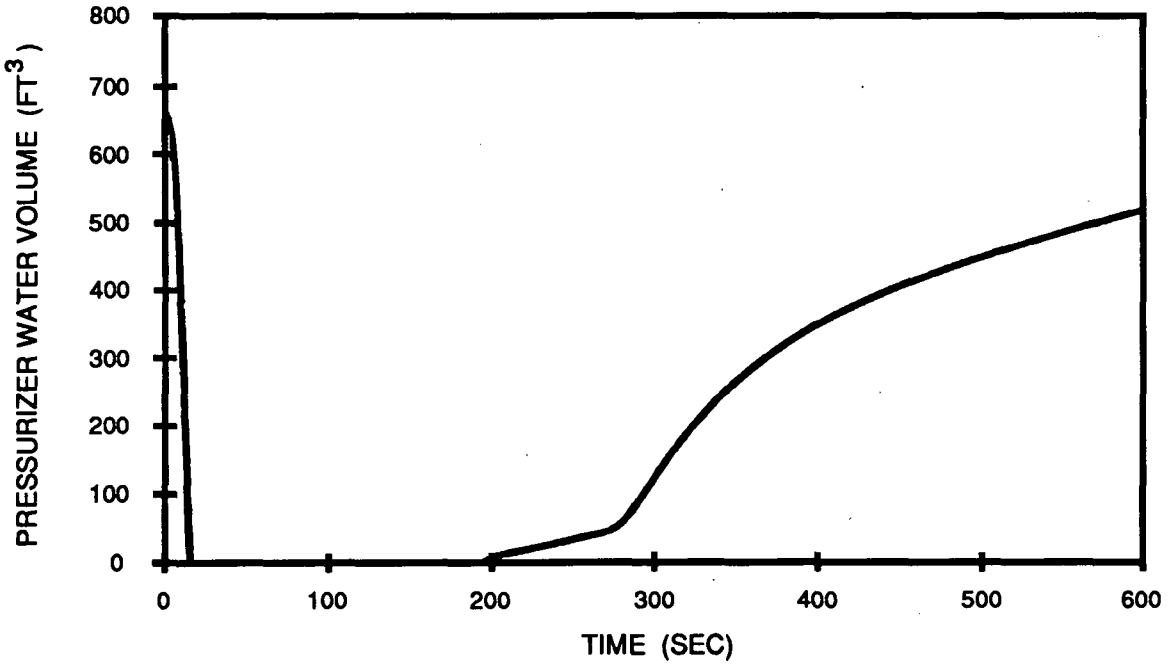




ZION STATION UFSAR

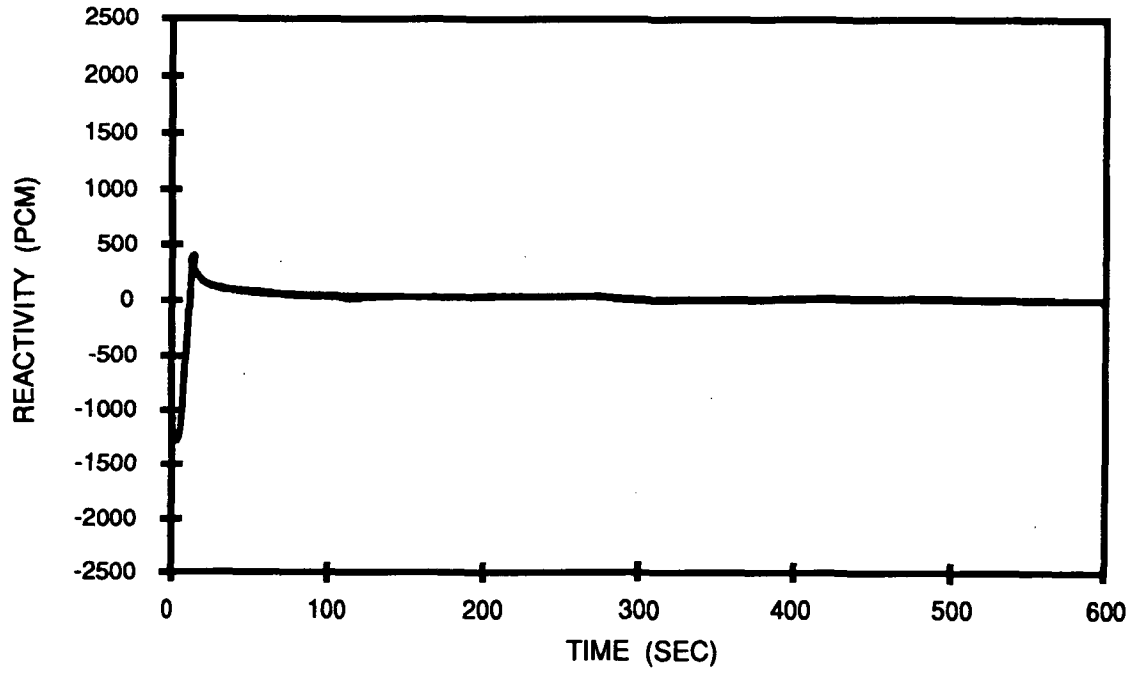
Figure 15.1-31
STEAMLINE BREAK
STEAM FLOW AND
FEEDWATER FLOW
INSIDE CONTAINMENT-WITH POWER
JULY 1993

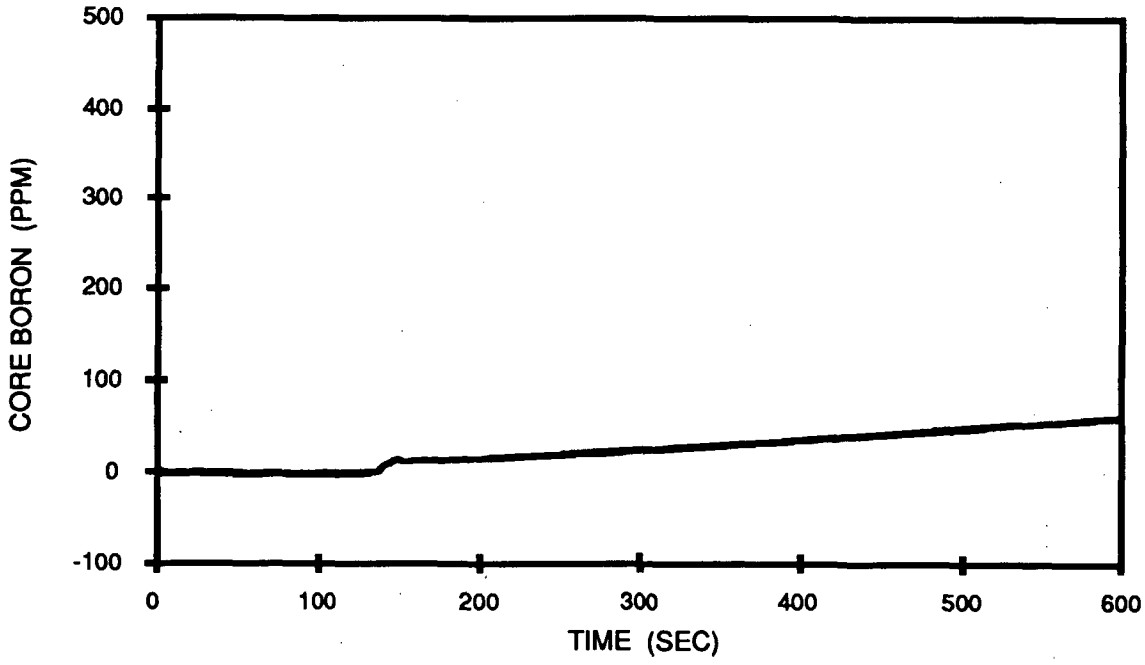




ZION STATION UFSAR

Figure 15.1-32
STEAMLINE BREAK
RCS PRESSURE AND
PRESSURIZER WATER VOLUME
INSIDE CONTAINMENT-WITH POWER
JULY 1993

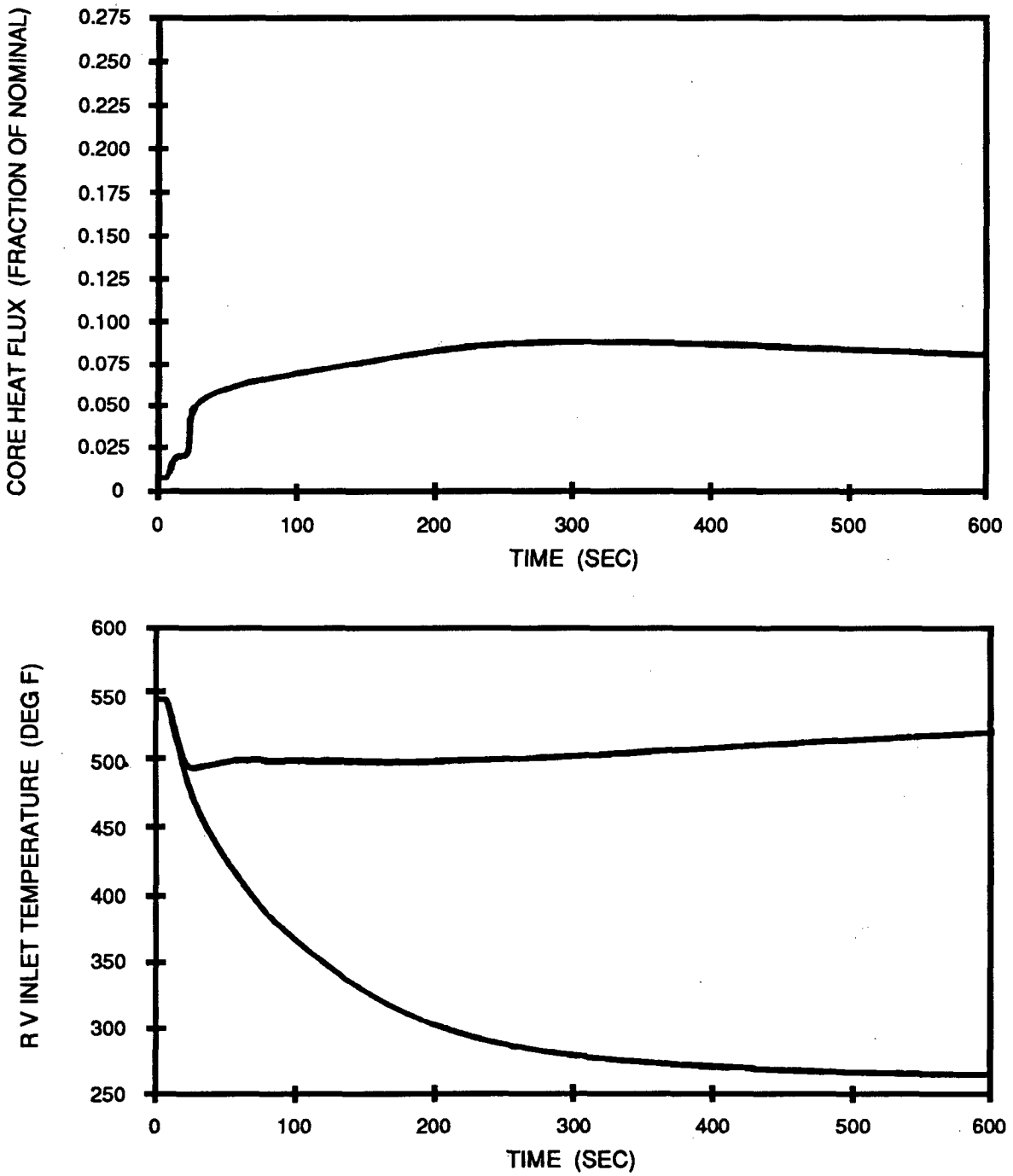




ZION STATION UFSAR

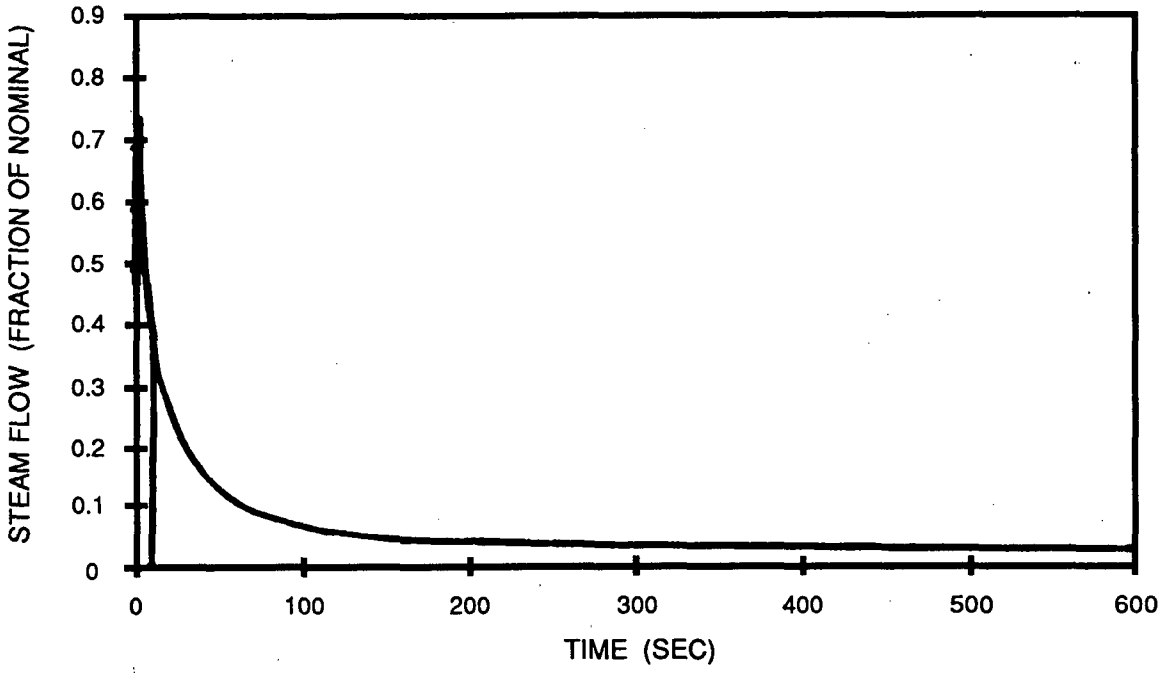
Figure 15.1-33
STEAMLINE BREAK
REACTIVITY AND
CORE BORON CONCENTRATION
INSIDE CONTAINMENT-WITH POWER

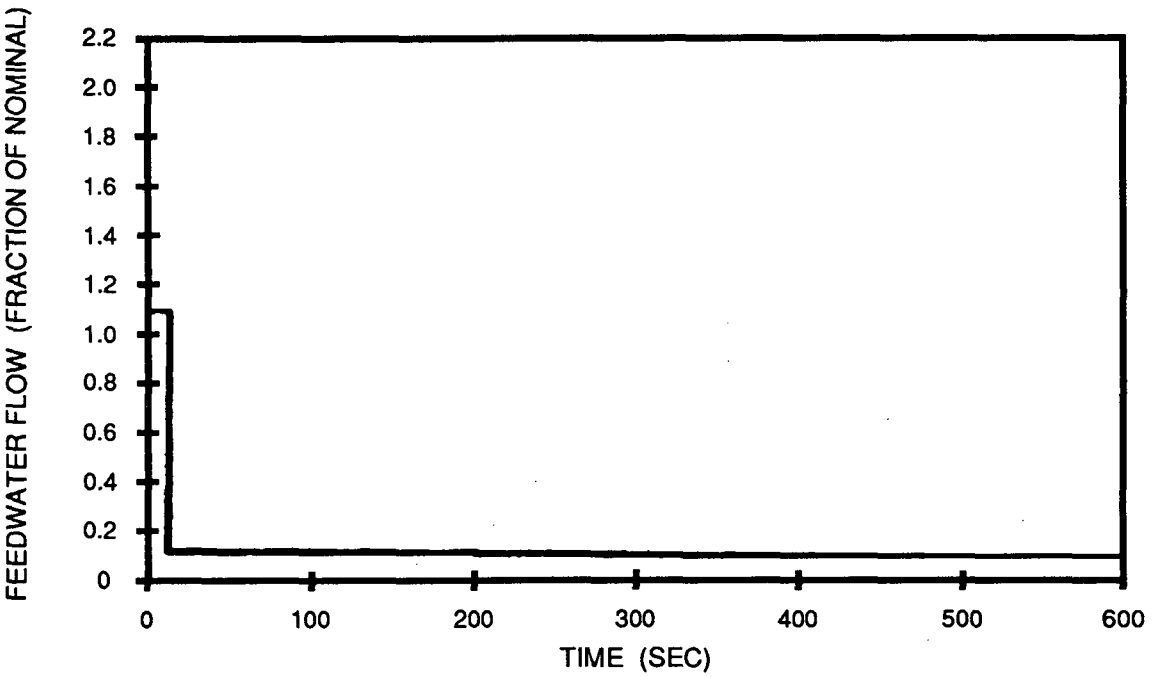
JULY 1993



ZION STATION UFSAR

Figure 15.1-34
STEAMLINE BREAK
CORE HEAT FLUX AND
REACTOR VESSEL INLET TEMPERATURE
OUTSIDE CONTAINMENT-W/O POWER
JULY 1993

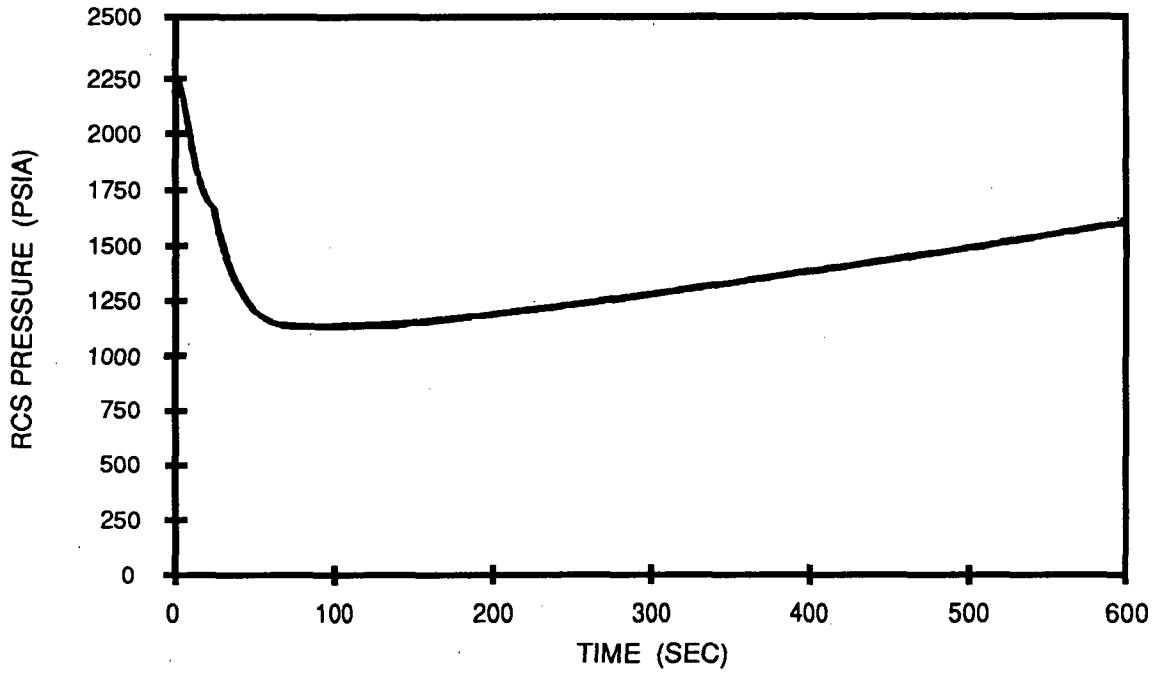


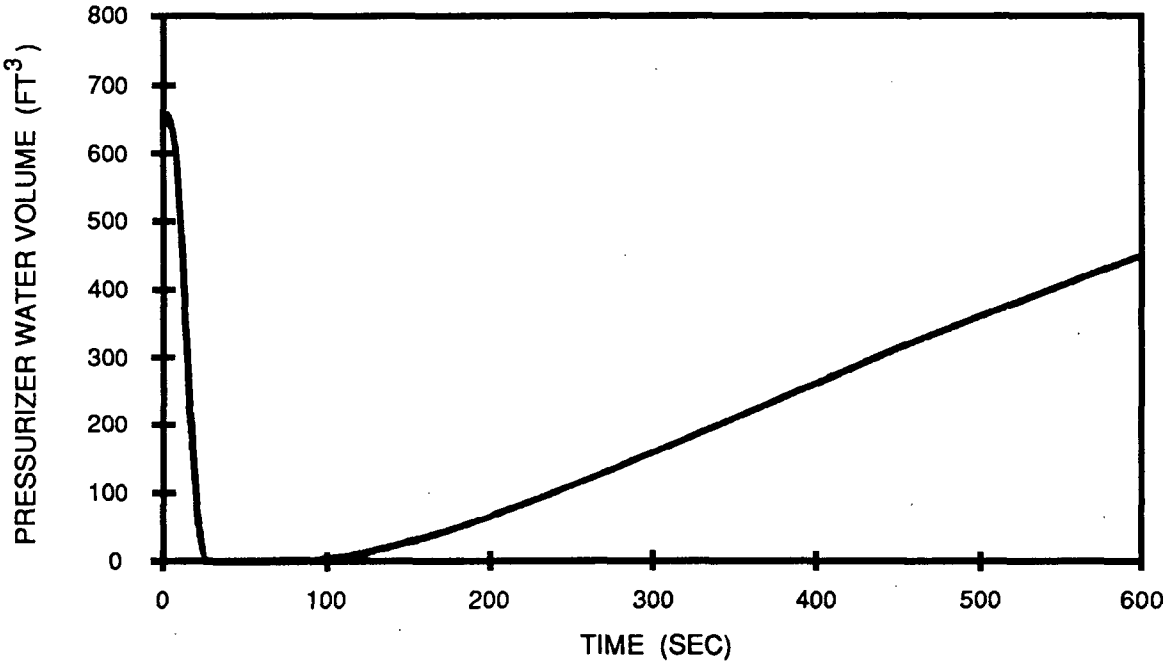


ZION STATION UFSAR

Figure 15.1-35
STEAMLINE BREAK
STEAM FLOW AND
FEEDWATER FLOW
OUTSIDE CONTAINMENT-W/O POWER

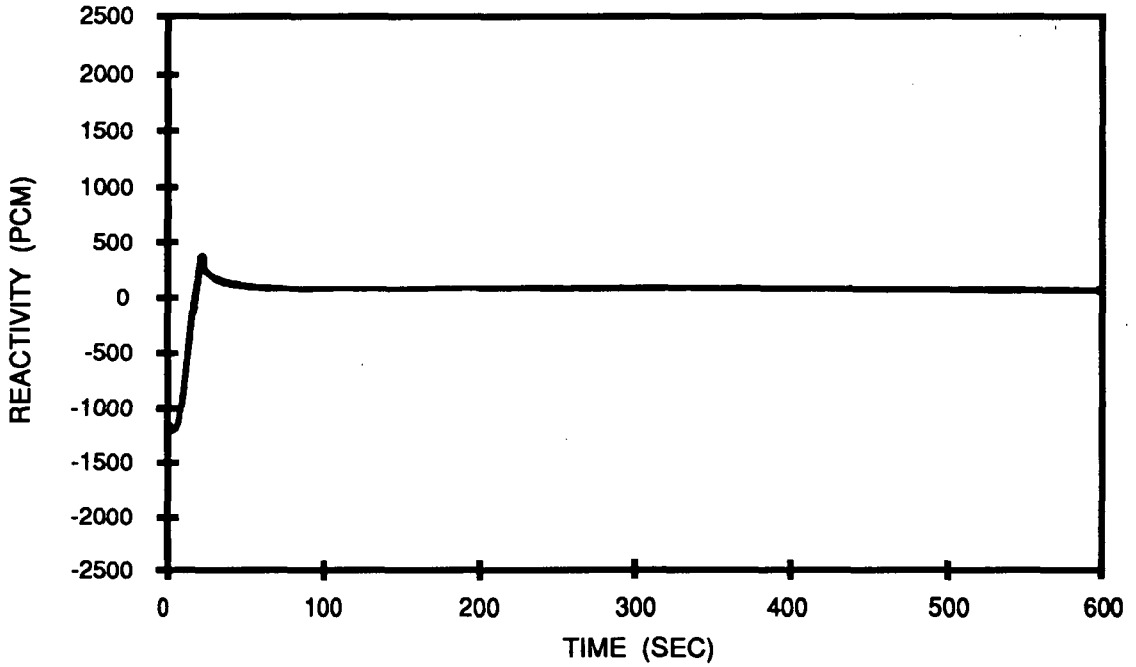
JULY 1993

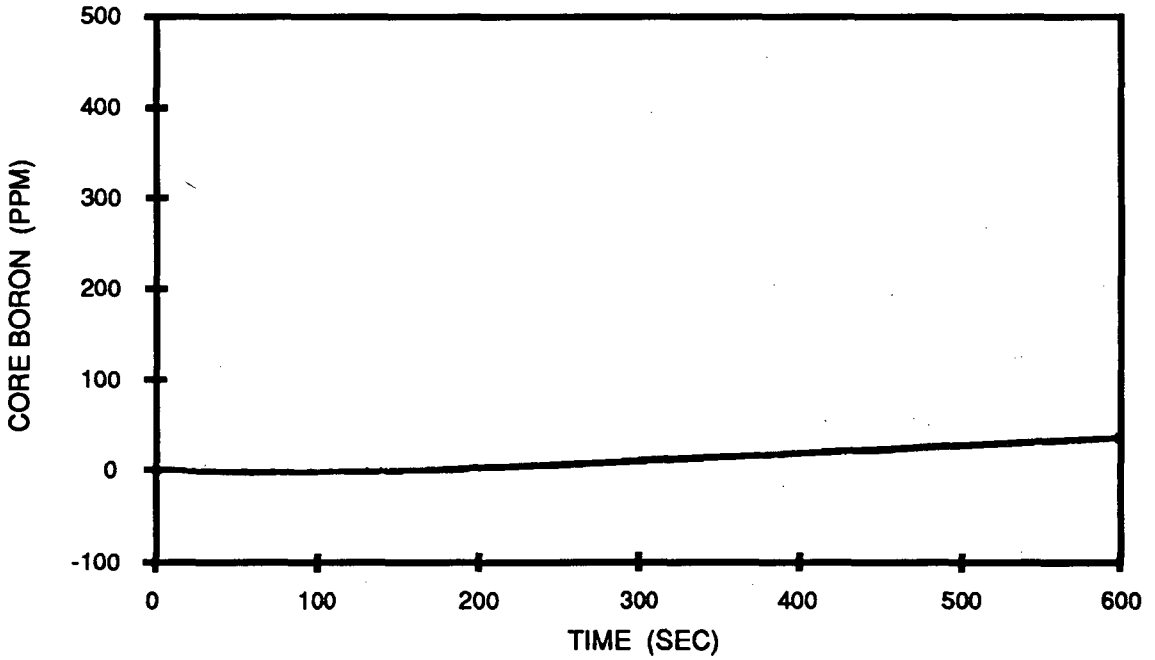




ZION STATION UFSAR

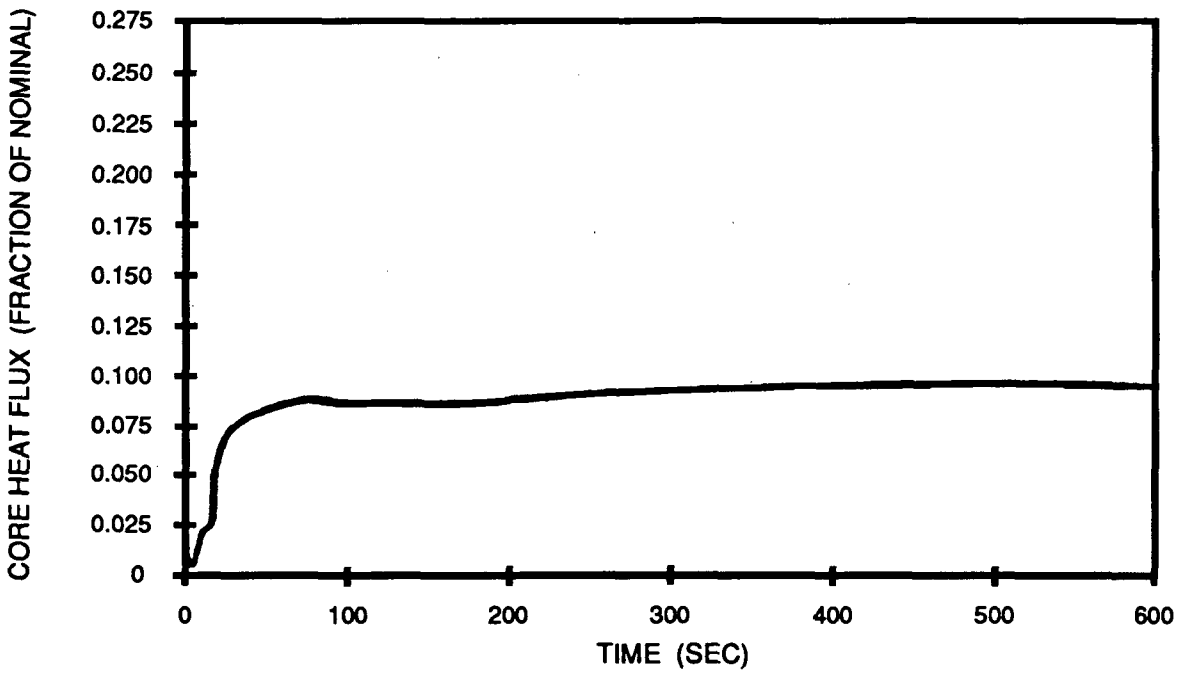
Figure 15.1-36
STEAMLINE BREAK
RCS PRESSURE AND
PRESSURIZER WATER VOLUME
OUTSIDE CONTAINMENT-W/O POWER
JULY 1993

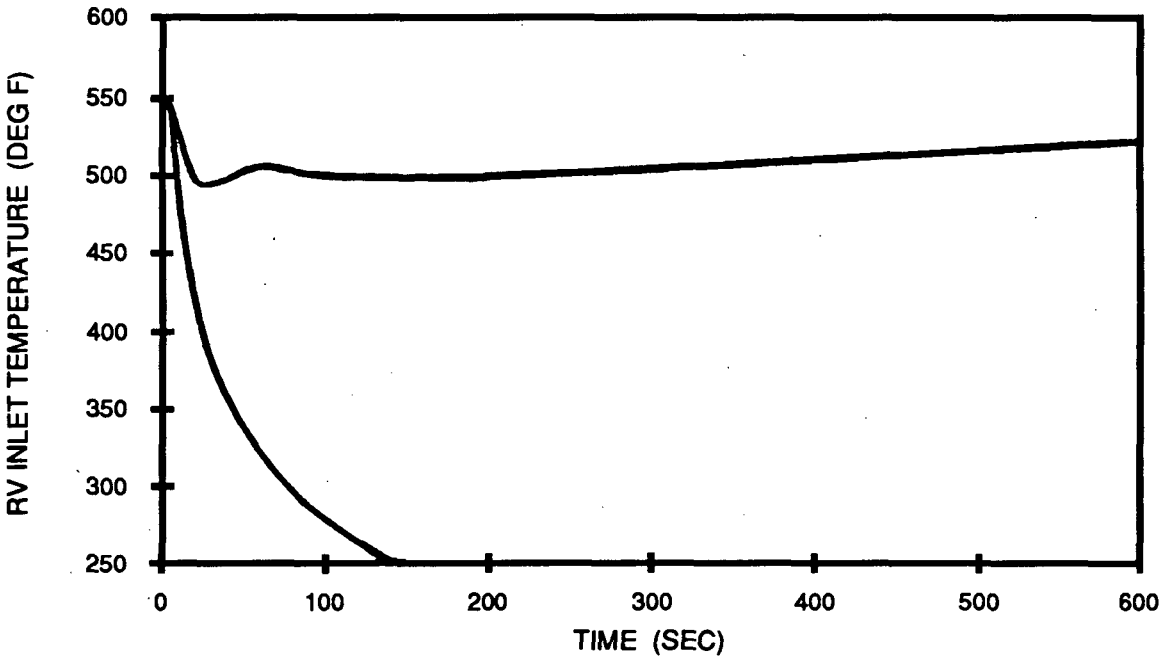




ZION STATION UFSAR

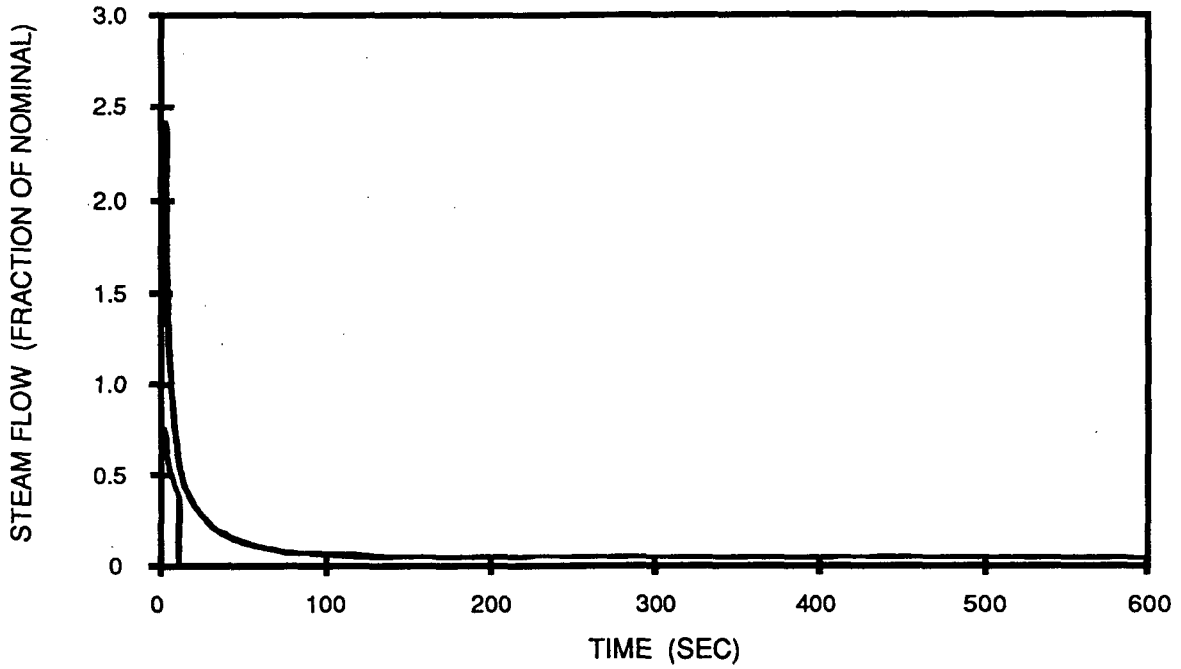
Figure 15.1-37
STEAMLINE BREAK
REACTIVITY AND
CORE BORON CONCENTRATION
OUTSIDE CONTAINMENT-W/O POWER
JULY 1993

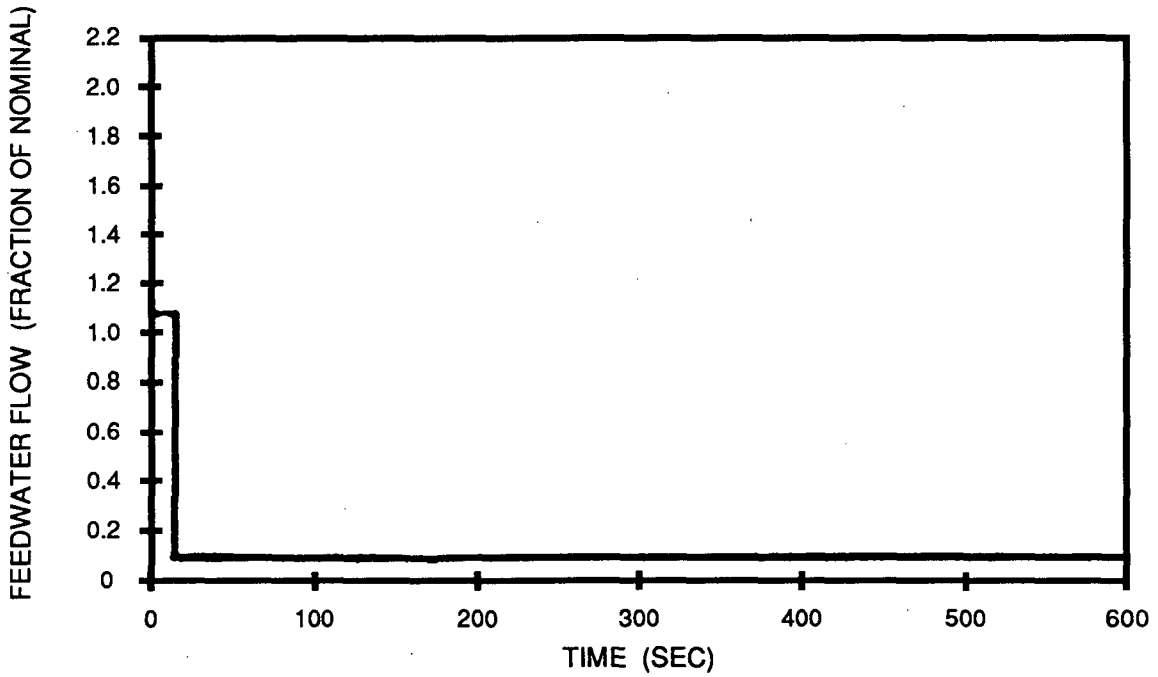




ZION STATION UFSAR

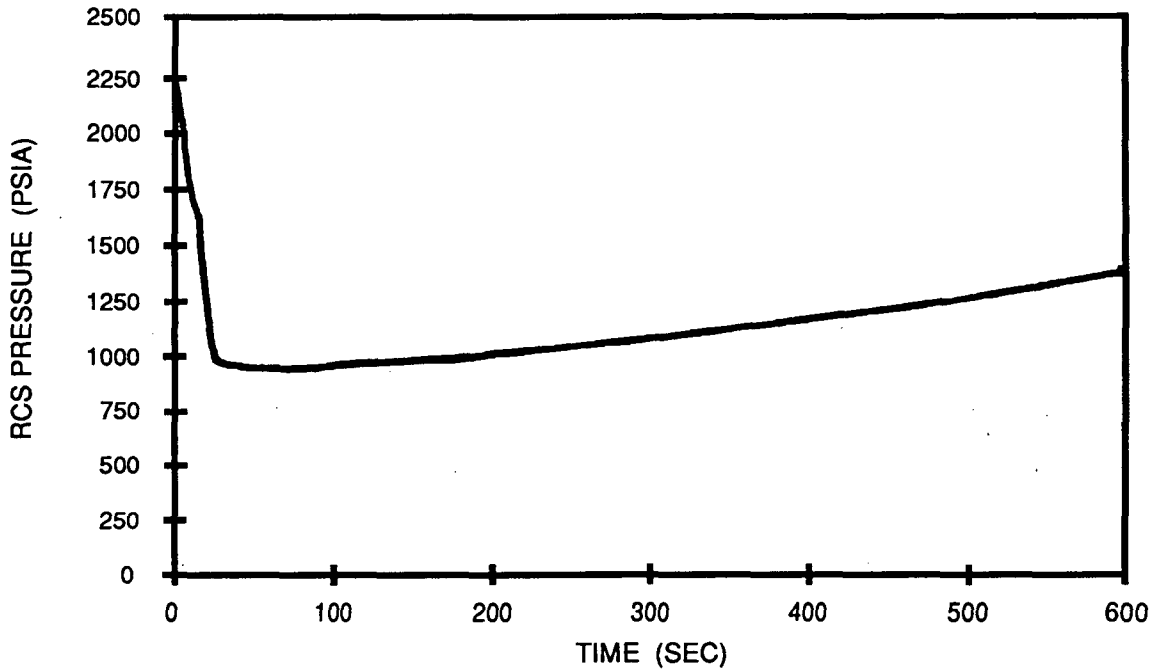
Figure 15.1-38
STEAMLINE BREAK
CORE HEAT FLUX AND
REACTOR VESSEL INLET TEMPERATURE
INSIDE CONTAINMENT-W/O POWER
JULY 1993

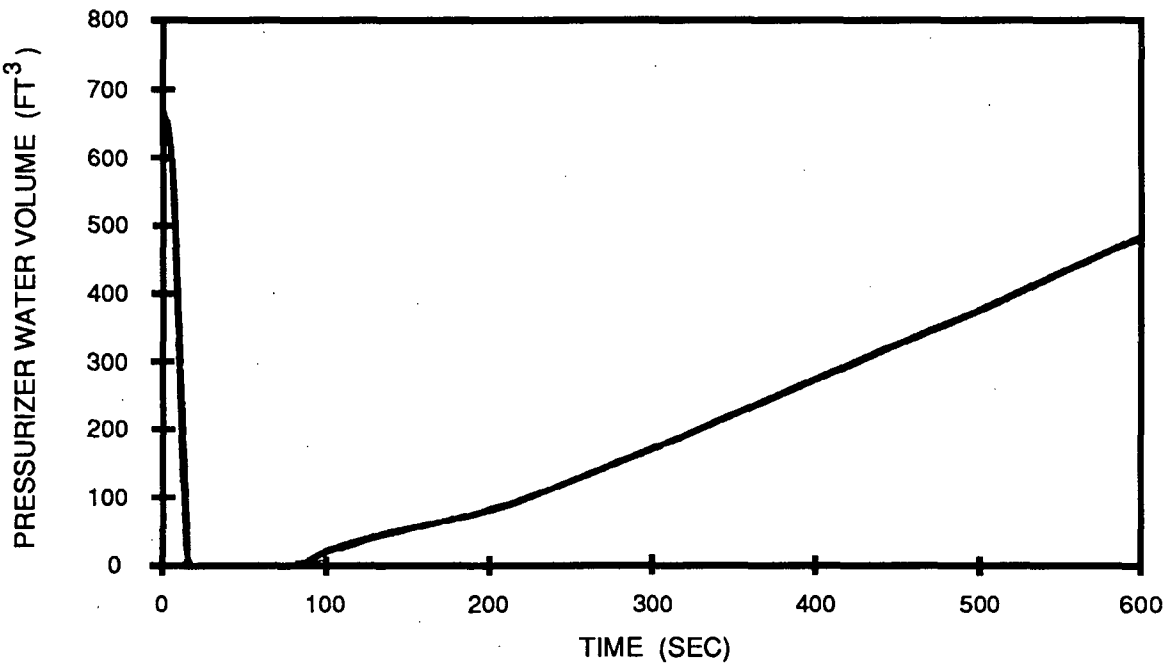




ZION STATION UFSAR

Figure 15.1-39
STEAMLINE BREAK
STEAM FLOW AND
FEEDWATER FLOW
INSIDE CONTAINMENT-W/O POWER
JULY 1993

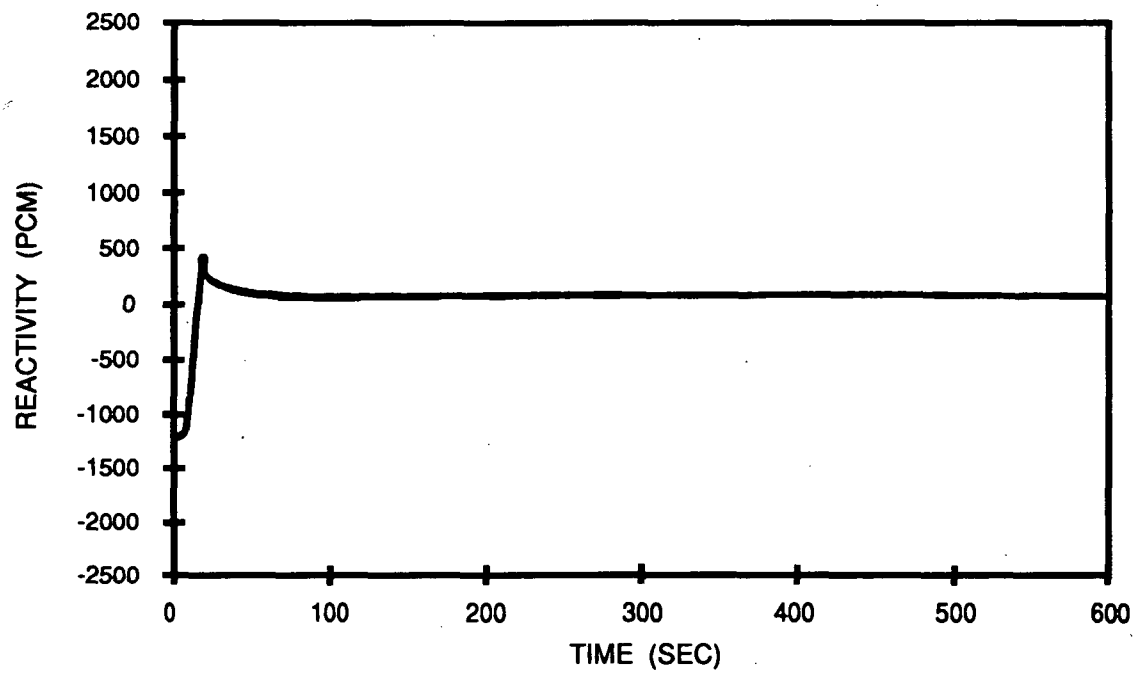


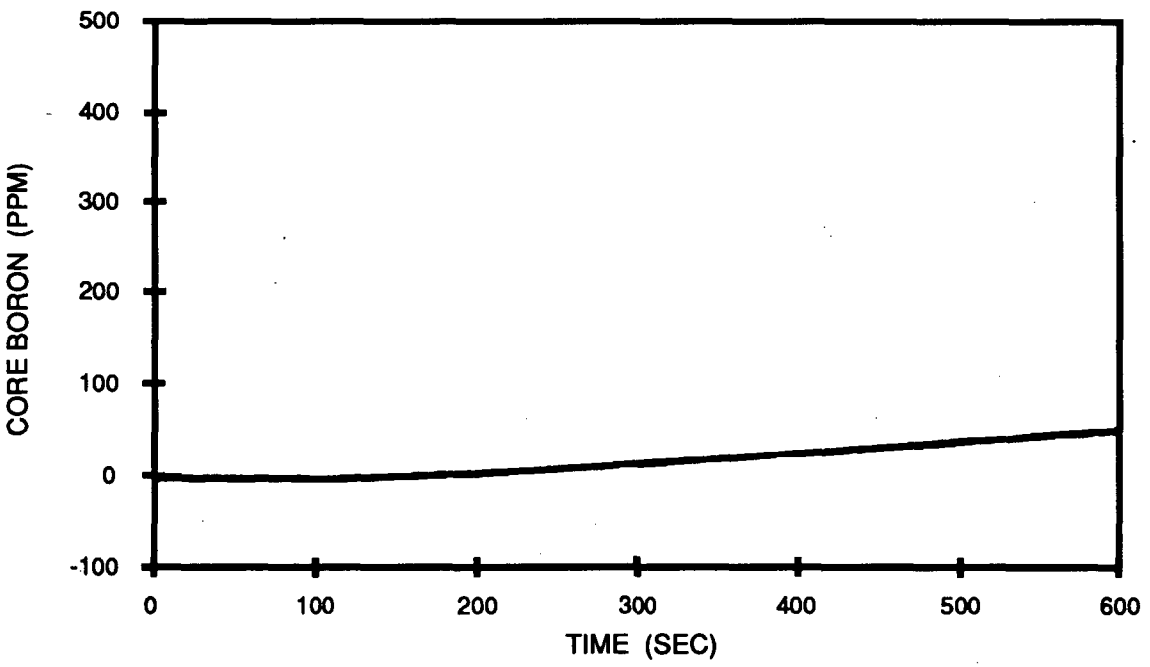


ZION STATION UFSAR

Figure 15.1-40
STEAMLINE BREAK
RCS PRESSURE AND
PRESSURIZER WATER VOLUME
INSIDE CONTAINMENT-W/O POWER

JULY 1993





ZION STATION UFSAR

Figure 15.1-41
STEAMLINE BREAK
REACTIVITY AND
CORE BORON CONCENTRATION
INSIDE CONTAINMENT-W/O POWER
JULY 1993

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

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

Discussions of the following RCS coolant heatup events are presented:

1. loss of external load;

2. loss of offsite power to the station auxiliaries; and

3. loss of normal feedwater.

The above items are considered to be ANS Condition II events. Section 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.2.2 Loss of External Electrical Load

15.2.2.1 Incident Description

The loss of external electrical load may result from an abnormal variation in network frequency, or other adverse network operating conditions. It may also result from a trip of the turbine generator or by the action of the turbine control in the unlikely event that opening of the main breaker from the generator fails to cause a turbine trip but causes a rapid large Nuclear Steam Supply System (NSSS) load reduction.

The unit is designed to accept a step loss of load from 100% to 50% load without actuating a reactor trip. The automatic steam dump system, with 40% steam dump capacity to the condenser, together with the Reactor Control System, is able to accommodate this load rejection by reducing the severity of the transient imposed upon the RCS. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the Reactor Control System. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not

lift for the 50% step loss of load with steam dump and automatic reactor control.

In the event the steam dump valves fail to open following a large load loss or in the event of a complete loss of load with steam dump operating, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the overtemperature ΔT signal, the low-low steam generator water level signal, or the high pressurizer water level signal. The steam generator shell side pressure and reactor coolant temperature will increase rapidly. The pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressure for all load losses without assuming the availability of the steam dump system. The steam dump valves will not be opened for load reductions of 10% or less. For larger load reductions, they may open dependent on the capability of the Reactor Control System.

The most likely source of a complete loss of load on the NSSS is a trip of the turbine-generator. In this case, there is a direct reactor trip signal (unless power is below approximately 10% of full power) derived from the turbine autostop oil pressure or all turbine stop valves closed. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. However, in this analysis, the behavior of the unit is evaluated for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The RCS and Main Steam (MS) System pressure relieving capacities are designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or steam dump control systems.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as discussed in Section 15.0.1. As such, the appropriate acceptance criteria are departure from nucleate boiling ratio (DNBR), peak primary pressure, and peak secondary pressure.

The turbine trip event is bounding for the loss of external load, loss of condenser vacuum, and other turbine-related events. While not explicitly required in the original plant design, the Reactor Protection System (RPS) and primary and secondary system designs preclude overpressurization, and Reference 3 provides a more complete discussion of overpressure protection.

15.2.2.2 Method of Analysis

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 assumption is made to show the adequacy of the pressure-relieving devices, to demonstrate core protection margins, and to delay 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. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The turbine trip transient is analyzed with the LOFTRAN computer code (Reference 1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level. Note that the main steam safety valves are explicitly modeled as a bank of five valves on each steam generator with staggered lift setpoints. Since higher steam pressures are conservative for this event, no blowdown or hysteresis behavior is assumed.

Major assumptions are summarized below:

1. Initial power, temperature, and pressure are assumed to be at their nominal values consistent with steady-state full power operation;

2. The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (end-of-life (EOL)) cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback (beginning of life (BOL)) cases assume a positive moderator temperature coefficient (+7 pcm/°F) and the least negative Doppler coefficient;

3. 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. No credit is taken for the operation of the steam dump system or steam generator atmospheric relief valves. This assumption maximizes secondary pressure;

5. Turbine trip is analyzed both with and without pressurizer pressure control. The pressurizer power operated relief valves (PORVs) and sprays are assumed operable for the cases with pressure control. The cases with pressure control minimize the increase in primary pressure which is conservative for the DNBR transient. The cases without pressure control maximize the pressure increase which is conservative for the RCS overpressurization criterion;

6. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur;

7. Only the overtemperature ΔT, high pressurizer pressure, and low-low steam generator water level reactor trips are assumed operable for the purposes of this analysis. No credit is taken for the direct reactor trip on turbine trip;

8. The main steam safety valves are assumed to lift 3% above their respective setpoints and are assumed to be full open 5% above the setpoints; and

9. Each main steam safety valve is assumed to relieve 100% of the flow capacity as specified by the Technical Specifications.

The accident is analyzed using the Revised Thermal Design Procedure (RTDP) as described in Reference 2. Plant characteristics and initial conditions are discussed in Section 15.0.3.

No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

15.2.2.3 Results

Four cases were analyzed: minimum feedback without pressure control; maximum feedback without pressure control; maximum feedback with pressure control; and minimum feedback with pressure control.

The calculated sequence of events for the four cases is presented in Table 15.2-1.

Figures 15.2-1 through 15.2-3 show the transient response for the turbine trip event under BOL condition without pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux increases until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Figures 15.2-4 through 15.2-6 show the transient response for the turbine trip event under EOL conditions without pressure control. The reactor is tripped on high pressurizer pressure. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Figures 15.2-7 through 15.2-9 show the transient response for the turbine trip event under EOL conditions with pressure control. The reactor is tripped on low-low steam generator water level. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer relief valves and sprays maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value.

Figures 15.2-10 through 15.2-12 show the transient response for the turbine trip event under BOL conditions with pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux increases until the reactor is tripped, and although the DNBR value decreases below the initial value, it remains well above the limit throughout the entire transient. The pressurizer relief and safety valves and sprays maintain primary pressure below 110% of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110% of the design value. This case results in the maximum amount of steam flow (66%) through the main steam safety valves.

15.2.2.4 Conclusions

The results of this analysis show that the plant design is such that a total loss of external electrical load without a direct reactor trip presents no hazard to the integrity of the RCS or the MS System. All of the applicable acceptance criteria are met. The minimum DNBR for each case is greater than the limit value. The peak primary and secondary pressures remain below 110% of design at all times.

15.2.3 Turbine Trip (Stop Valve Closure)

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.2.4 Inadvertent Closure of Main Steam Isolation Valve

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.2.5 Loss-of-Condenser Vacuum

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.2.6 Loss of Offsite Power to the Station Auxiliaries

15.2.6.1 Incident Description

In the event of a turbine trip and a complete loss of offsite power, there will be a loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The first few seconds of the transient will be identical to the "four-pump loss-of-flow" case presented in Section 15.3 (i.e. core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor). After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or core. This portion of the transient is identical to that presented in Section 15.3.

Section 15.3 demonstrates that the DNB design basis is satisfied. Therefore, the DNB aspects of this event are not explicitly evaluated in this analysis. The analysis shows that following a loss of normal feedwater, the Auxiliary Feedwater (AFW) System is capable of removing the stored and residual heat thus preventing either overpressurization of the RCS or the secondary side or uncovery of the reactor core, and returning the plant to a safe condition.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as discussed in Section 15.0.1.

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

1. Plant vital instruments are supplied by the emergency power sources;

2. The diesel generators will start on loss of voltage on the Engineered Safety Features (ESF) buses to supply plant vital loads; and

3. As the steam system pressure rises following the trip, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor. As the no-load temperature is approached, the self-actuated safety valves are used to dissipate the residual heat and maintain the plant at the hot shutdown condition.

The following provide the necessary protection against a loss of offsite power:

1. Reactor trip on low-low water level in any steam generator;

2. Reactor trip on steam flow-feedwater flow mismatch in coincidence with low steam generator water level in any loop;

3. Two motor-driven AFW pumps that are started on:

   a. Low-low level in any steam generator;

   b. Trip of all three main feedwater pumps;

   c. Any safety injection signal;

   d. Loss of offsite power (automatic transfer to diesel generators); and

   e. Manual actuation.

4. One turbine-driven AFW pump that is started on:

   a. Low-low level in any two steam generators;

   b. Undervoltage on any two reactor coolant pump buses; and

   c. Manual actuation.

The AFW System is started automatically as discussed in the loss-of-normal-feedwater analysis (Section 15.2.7). The steam-driven AFW pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank or Service Water (SW) System for delivery to the steam generators. The AFW System ensures feedwater supply of at least 308 gpm upon loss of power to the station auxiliaries.

The steam-driven and the motor-driven pumps can be tested at any time. The valves in the system can be operationally tested at any time.

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

15.2.6.2 Method of Analysis

A detailed analysis using the LOFTRAN computer code (Reference 1) is performed in order to determine the plant transient following a loss of offsite power. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the AFW System, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The major assumptions used in this analysis are identical to those used in the loss of normal feedwater analysis (Section 15.2.7) with the following exceptions:

1. No credit is taken for the immediate insertion of the control rods as a result of the loss of offsite power;

2. Power is assumed to be lost to the reactor coolant pumps (RCPs) coincident with rod motion. This assumption results in the maximum amount of stored energy in the RCS.

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation is assumed following the RCP coastdown;

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

5. The AFW System provides flow 70 seconds after AFW initiation. This delay includes the time allotted for diesel generator startup and loading.

Plant characteristics and initial conditions are further discussed in Section 15.0.3. Consistent with the loss of normal feedwater analysis, the most limiting single failure occurs in the AFW System.

15.2.6.3 Results

Figures 15.2-13 through 15.2-15 show the transient response following a loss of offsite power.

The first few seconds after the loss of power to the RCPs will closely resemble a simulation of the complete loss of flow incident; i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The

LOFTRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The capacity of the motor-driven AFW pump is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to establish enough natural circulation flow to dissipate core residual heat without water relief from the RCS relief or safety valves. From Figure 15.2-13 it can be seen that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-2. As shown in Figures 15.2-13 through 15.2-15, the plant approaches a stabilized condition following reactor trip, pump coastdown, and auxiliary feedwater initiation.

15.2.6.4 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. The AFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized and that the core is not adversely affected.

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 clad damage.

15.2.7 Loss of Normal Feedwater

15.2.7.1 Incident Description

A loss of normal feedwater results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, primary plant damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the system, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Loss of significant water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a departure from nucleate boiling (DNB) condition.

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

1. Reactor trip on low-low water level in any steam generator.

2. Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in any steam generator.

3. Turbine trip on Anticipated Transient Without Scram (ATWS) Mitigation System (AMS) signal (see Section 7.2.1.1.4.2.4).

4. Two motor-driven AFW pumps which are started on:

   a. low-low level in any steam generator,
   b. any safety injection signal,
   c. blackout signal,
   d. associated ESF bus 2nd level undervoltage signal present for greater than a timed interval,
   e. manually, and
   f. AMS signal.

5. One turbine-driven AFW pump which is started on:

   a. low-low level in any two steam generators,
   b. undervoltage on two-of-four RCP buses,
   c. any safety injection signal,
   d. manually,
   e. blackout signal,
   f. ESF bus 149(249) 2nd level undervoltage signal present for greater than a timed interval, and
   g. AMS signal.

The motor-driven AFW pumps are supplied by the diesel generators if a loss-of-offsite power occurs and the turbine-driven pump utilizes steam from the secondary system. Both types of pumps are designed to start within one minute after an initiating signal, even if a loss of all ac power occurs simultaneously. The turbine exhausts the secondary steam to the atmosphere. The AFW pumps take suction from the condensate storage tank or the SW System for delivery to the steam generators.

The above discussion shows considerable backup in equipment and protection circuit logic to ensure that reactor trip and automatic AFW flow will occur following any loss of normal feedwater, including that caused by a loss-of-offsite ac power.

The purpose of the analysis performed was to show that, following a loss of normal feedwater, the AFW System is capable of removing the stored and residual heat and, thus, preventing either overpressurization of the RCS or uncovery of the reactor core.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as discussed in Section 15.0.1.

15.2.7.2 Method of Analysis

A detailed analysis using the LOFTRAN computer code (Reference 1) is performed in order to determine the plant transient following a loss of

normal feedwater. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the AFW System, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions in this analysis are:

1. The plant is initially operating at 102% of the engineered safeguards design rating;

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

3. Core residual heat generation based on the 1979 version of ANS 5.1 (Reference 4). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed;

4. The AFW System is actuated by the low-low steam generator water level signal. A 60 second delay is assumed between signal actuation and the pumps delivering full rated flow;

5. The worst single failure in the AFW System occurs (turbine-driven pump) and one motor-driven pump is assumed to be unavailable. The AFW System is assumed to supply a total of 308 gpm to four steam generators from the available motor-driven pump;

6. The pressurizer sprays and PORVs are assumed operable. This maximizes the peak transient pressurizer water volume;

7. Secondary system steam relief is achieved through the self-actuated main steam safety valves. Note that steam relief will, in fact, be through the PORVs or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis, these have been assumed unavailable;

8. The main steam safety valves are assumed to lift 3% above their respective setpoints and are assumed to be full open 5% above the setpoints;

9. Each main steam safety valve is assumed to relieve 100% of the flow capacity as specified by the Technical Specifications;

10. The initial reactor coolant average temperature is 5.5°F higher than the nominal value which is comprised of the uncertainty on nominal temperature. The initial pressurizer pressure uncertainty is 40 psi;

11. An AFW line purge volume of 110.0 ft^3 and an initial AFW temperature of 120°F are assumed. As a result, an additional 642 seconds was assumed before the feedwater lines were purged and the relatively cold (120°F) AFW entered the steam generators; and

12. Heat addition from the RCPs is not modeled.

The assumptions detailed above are designed to minimize the heat removal capability of the secondary system and to maximize the potential for water relief from the RCS by maximizing the expansion of the primary system.

For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value. The RCPs may be manually tripped at some later time.

15.2.7.3 Results

Figures 15.2-16 through 15.2-18 show the pertinent plant parameters following a loss-of-normal feedwater accident for the assumptions listed above.

Following the reactor trip on low-low water level in any steam generator, the water level in the steam generators will fall due to the reduction of steam generator 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 level trip, the AFW pump is automatically started, reducing the rate of water level decrease. The capacity of the AFW pump provides for enough flowrate to prevent the water level in the steam generators from decreasing below a minimum level. This minimum level provides for a sufficient heat transfer area to dissipate core residual heat without water relief from the RCS relief or safety valves.

From Figure 15.2-16, it can be seen that at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of one motor-driven pump, the initial reactor power is less than 102% of 3250 MWt, or the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip, then the result will be a steam generator minimum water level higher than shown and an increased margin to the point at which reactor coolant water relief occurs.

The calculated sequence of events for this accident is listed in Table 15.2-3. As shown in Figure 15.2-16 through 15.2-18, the plant approaches a stabilized condition following reactor trip and AFW initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.4 Conclusions

The loss of normal feedwater does not adversely affect the core, RCS or steam system. The feedwater loss does not result in water relief from the pressurizer relief or safety valves.

15.2.8 Feedwater Piping Break

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.2.9 References

1. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2. Friedland, A.J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

3. Mangan, M.A., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972.

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

TABLE 15.2-1 (1 of 2)

TIME SEQUENCE OF EVENTS
LOSS OF EXTERNAL ELECTRICAL LOAD

ACCIDENT	EVENTS	TIME (sec)
Loss of External Electrical Load		
1. Without pressure control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	6.3
	Rods begin to drop	8.3
	Peak pressurizer pressure occurs	10.0
	Initiation of steam release from main steam safety valves (MSSVs)	15.5
	Minimum DNBR occurs	*
2. Without pressurizer control (EOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	6.3
	Rods begin to drop	8.3
	Peak pressurizer pressure occurs	9.5
	Initiation of steam release from MSSVs	16.5
	Minimum DNBR occurs	*
3. With pressure control (EOL)	Loss of electrical load	0.0
	Peak pressurizer pressure occurs	12.0
	Initiation of steam release from MSSVs	16.0
	Low-low steam generator water level reactor trip	78.2
	Rods begin to drop	80.2
	Minimum DNBR occurs	*

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (2 of 2)

TIME SEQUENCE OF EVENTS
LOSS OF EXTERNAL ELECTRICAL LOAD

ACCIDENT	EVENTS	TIME (sec)
4. With pressure control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip signal generated	10.5
	Rods begin to drop	12.5
	Minimum DNBR occurs	14.0
	Peak pressurizer pressure occurs	14.5
	Initiation of steam release from MSSVs	15.0

TABLE 15.2-2

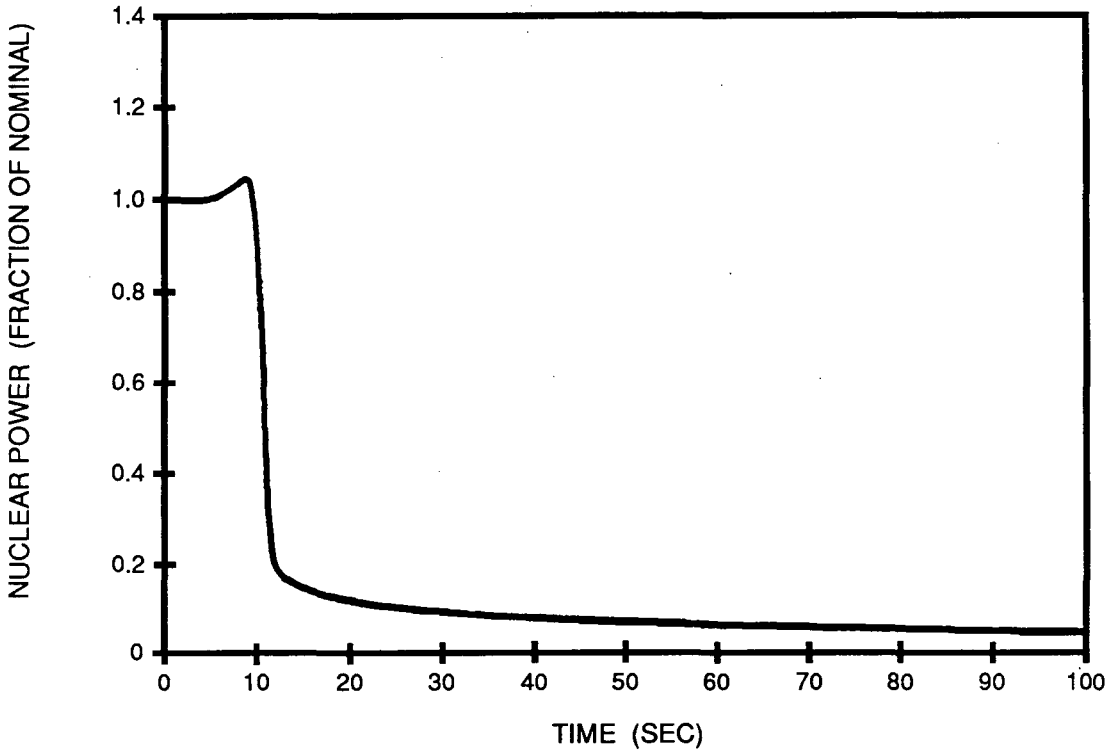
TIME SEQUENCE OF EVENTS
LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES

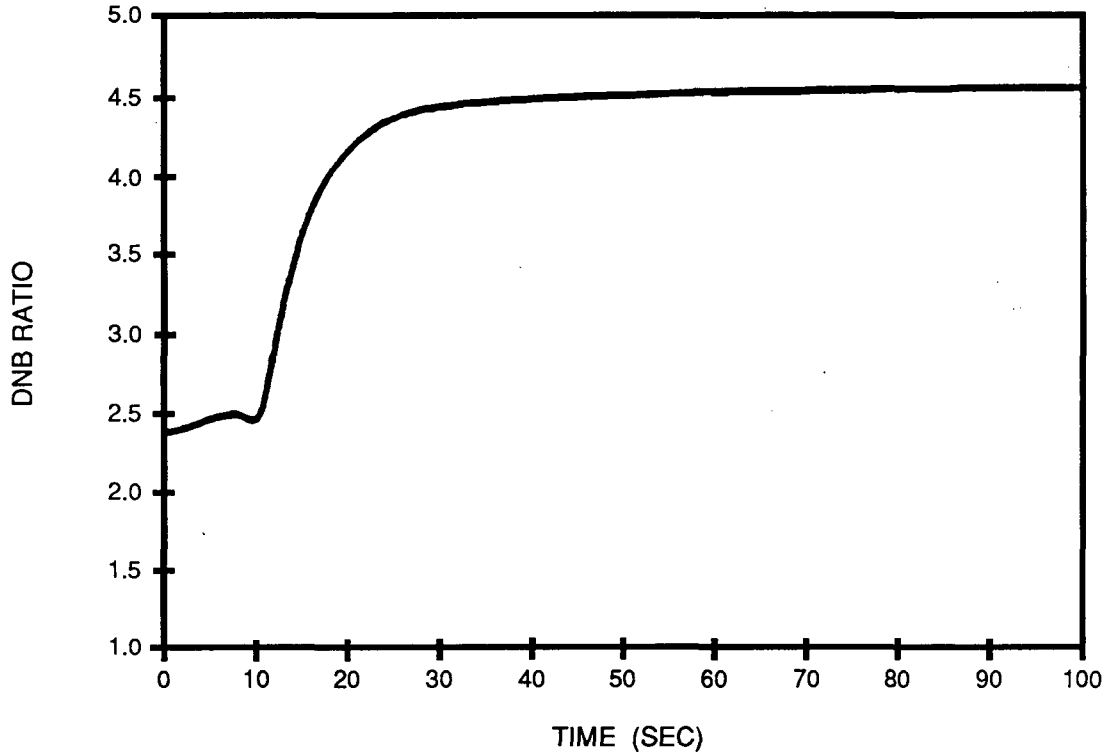
ACCIDENT	EVENTS	TIME (sec)
Loss of Offsite Power to the Station Auxiliaries	Main feedwater flow stops	10
	Low-low steam generator water level reactor trip	56
	Rods begin to drop	58
	RCPs begin to coastdown	60
	Four steam generators begin to receive auxiliary feed from one motor-driven AFW pump	126
	Cold AFW is delivered to the steam generators	768
	Peak water level in pressurizer occurs	5356
	Core decay heat decreases to the AFW heat removal capacity	~5500

TABLE 15.2-3

TIME SEQUENCE OF EVENTS
LOSS OF NORMAL FEEDWATER

ACCIDENT	EVENTS	TIME (sec)
Loss of Normal Feedwater	Main feedwater flow stops	10
	Low-low steam generator water level reactor trip	56
	Rods begin to drop	58
	Four steam generators begin to receive auxiliary feed from one motor-driven AFW pump	116
	Feedwater lines purged and relatively cold AFW is delivered to the steam generators	758
	Peak water level in pressurizer occurs	5204
	Core decay heat decreases to the AFW heat removal capacity	~5400





ZION STATION UFSAR

Figure 15.2-1
LOSS OF ELECTRICAL LOAD
NUCLEAR POWER AND DNBR
BOL WITHOUT PRESSURE CONTROL
JULY 1993

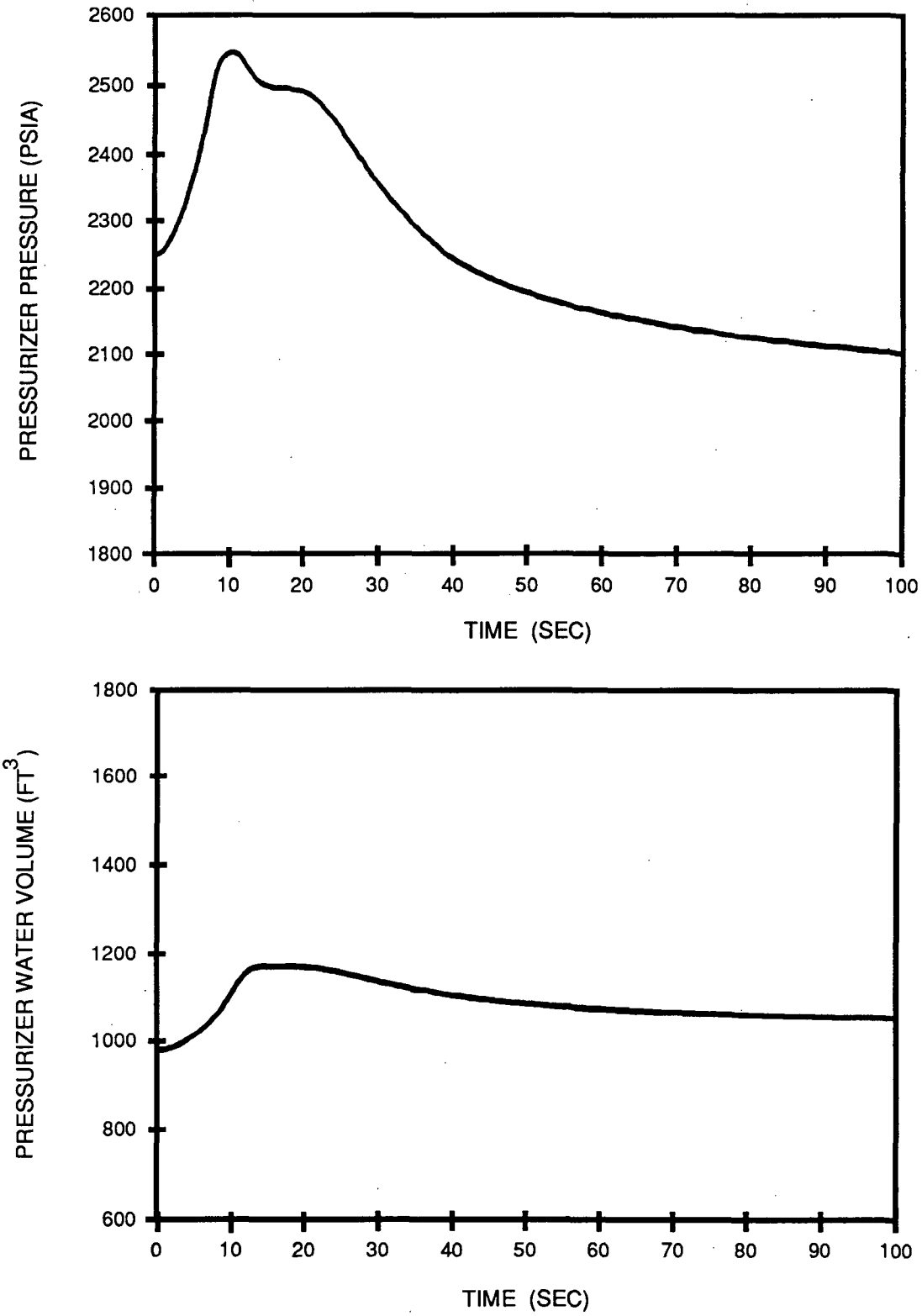


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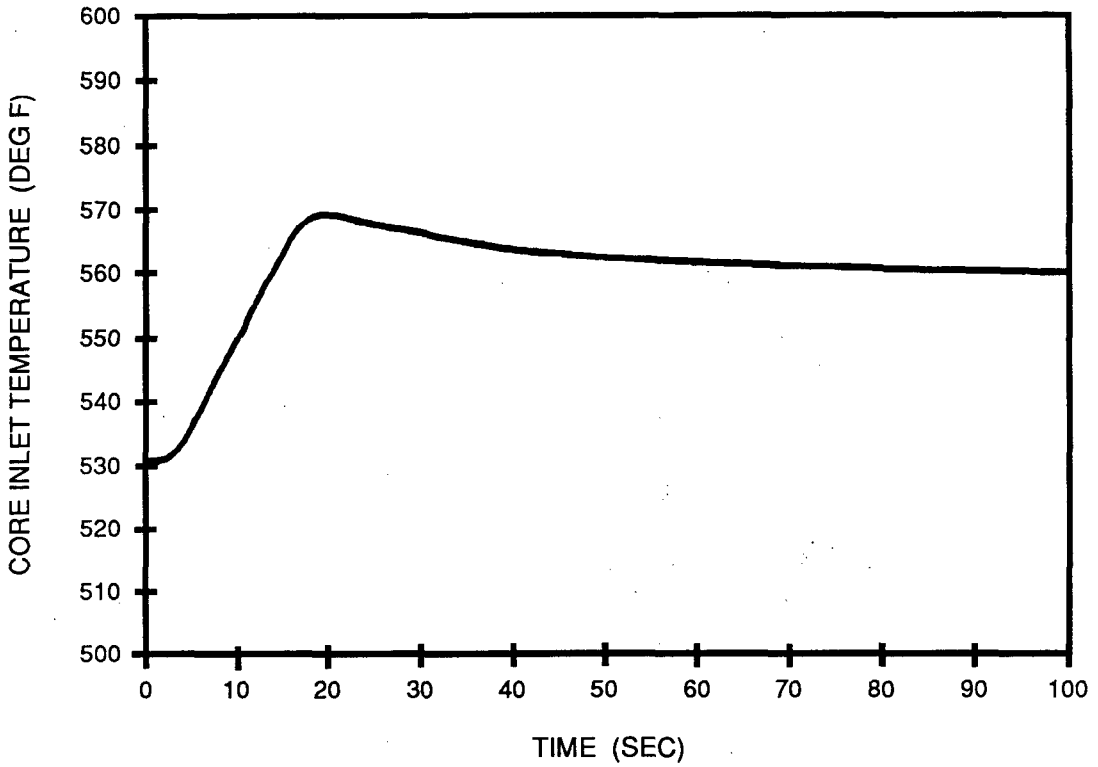
Figure 15.2-2
LOSS OF ELECTRICAL LOAD
PRESSURIZER PRESSURE
AND WATER VOLUME
BOL WITHOUT PRESSURE CONTROL
JULY 1993



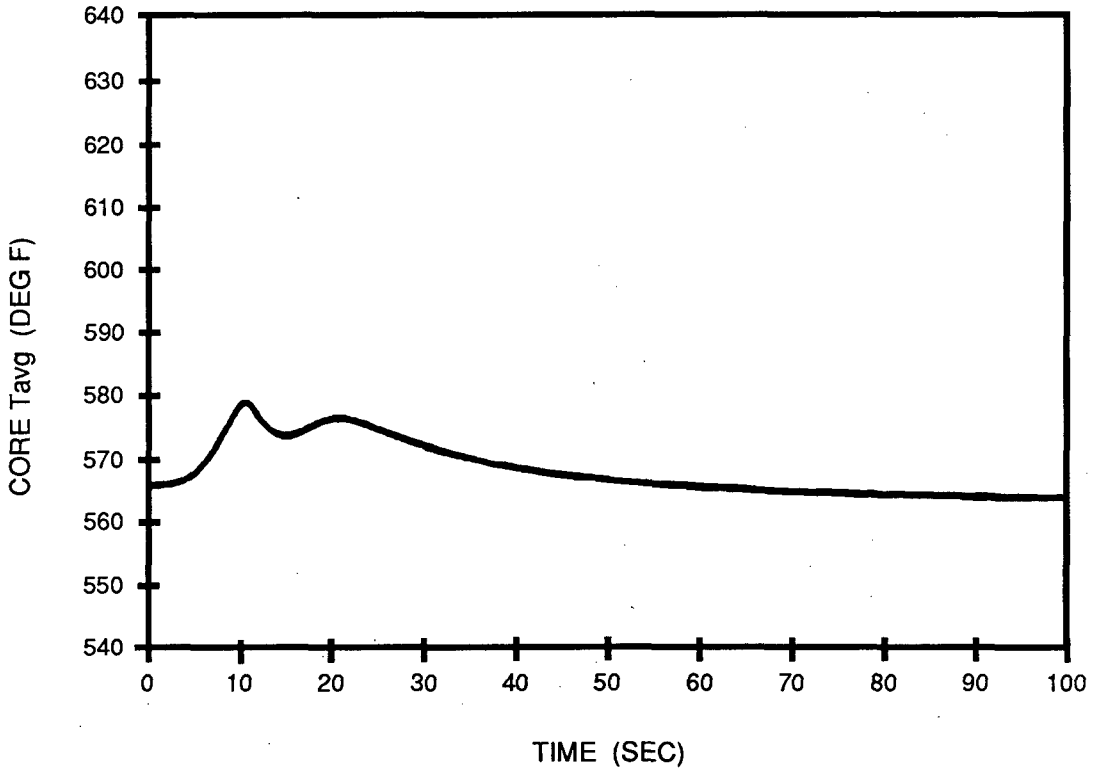


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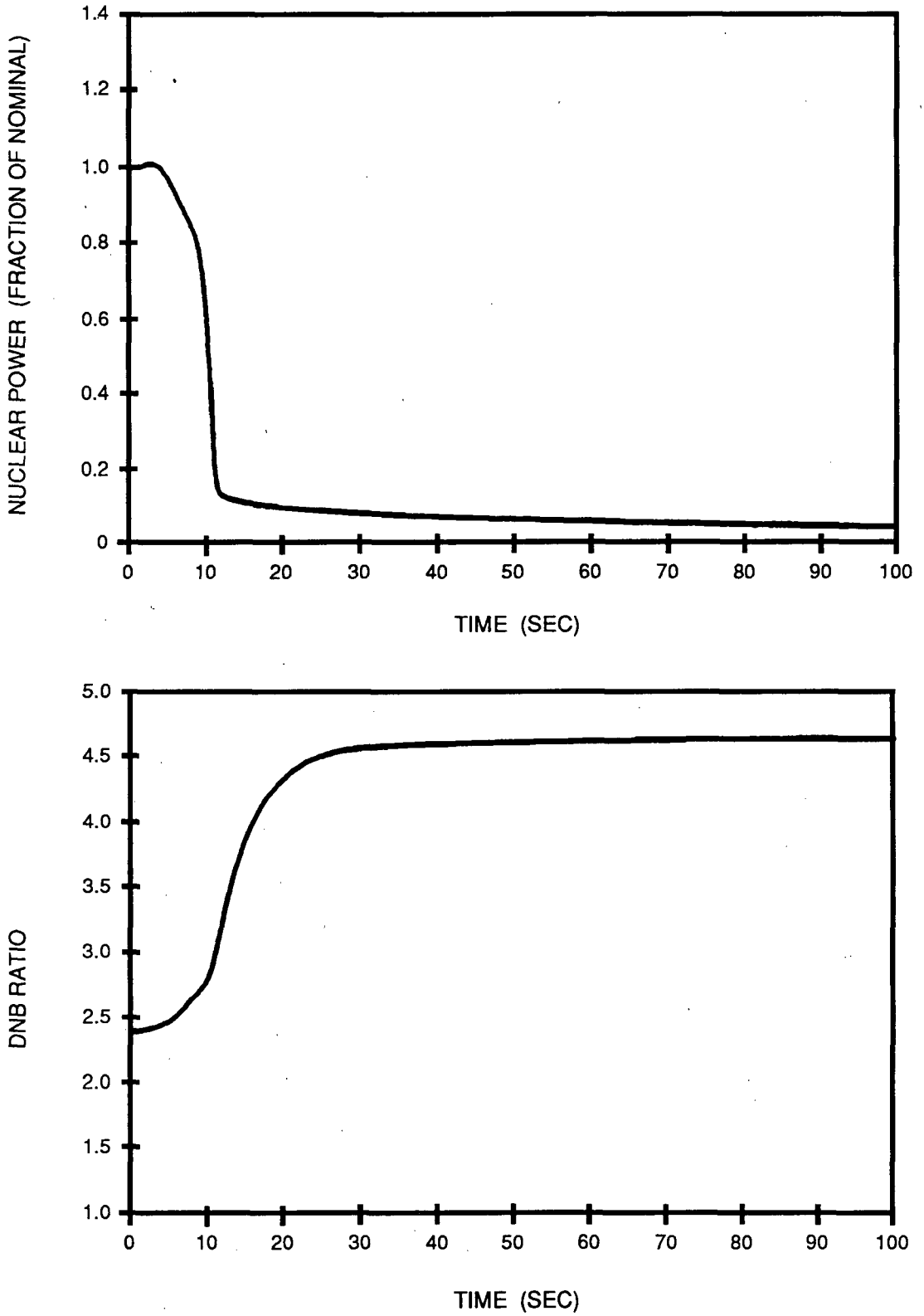
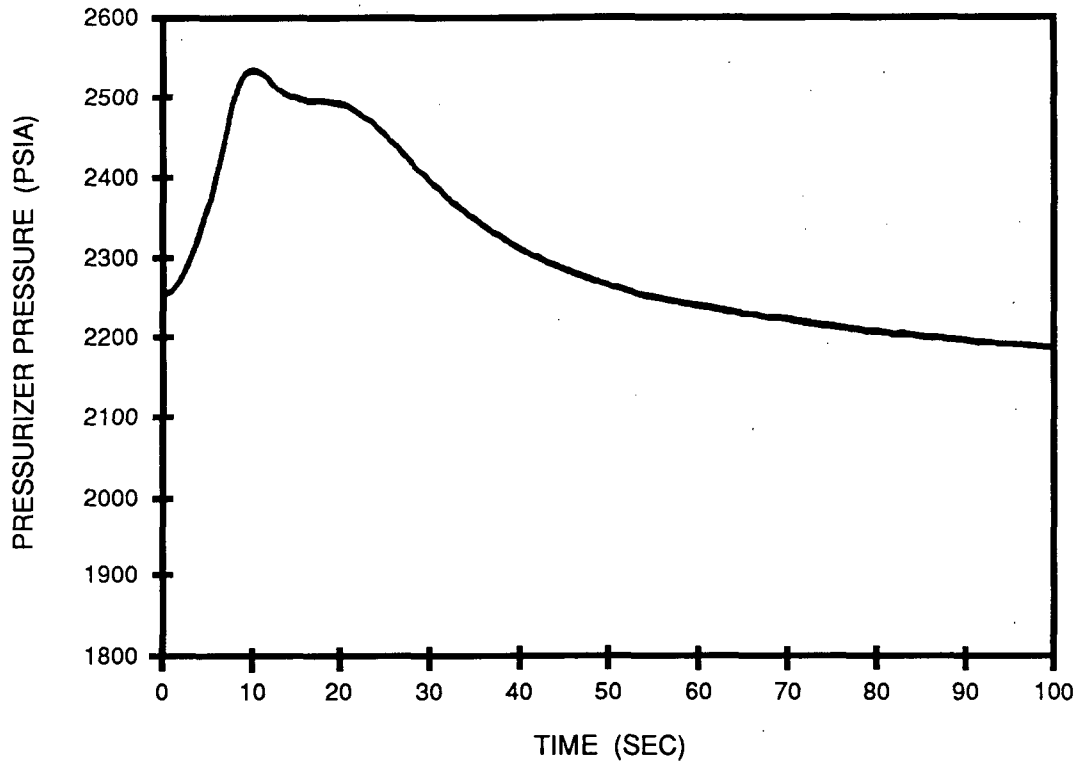
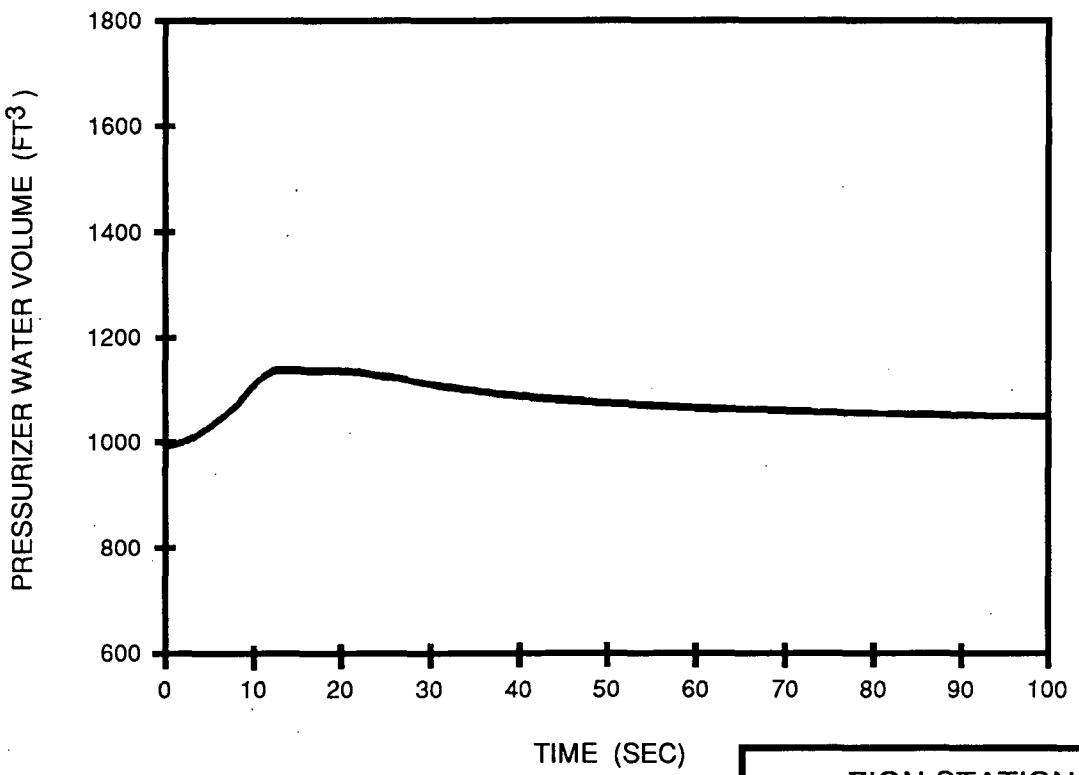
Figure 15.2-3
LOSS OF ELECTRICAL LOAD
CORE INLET TEMPERATURE AND CORE Tavg
BOL WITHOUT PRESSURE CONTROL
JULY 1993



ZION STATION UFSAR

Figure 15.2-4
LOSS OF ELECTRICAL LOAD
NUCLEAR POWER AND DNBR
EOL WITHOUT PRESSURE CONTROL
JULY 1993

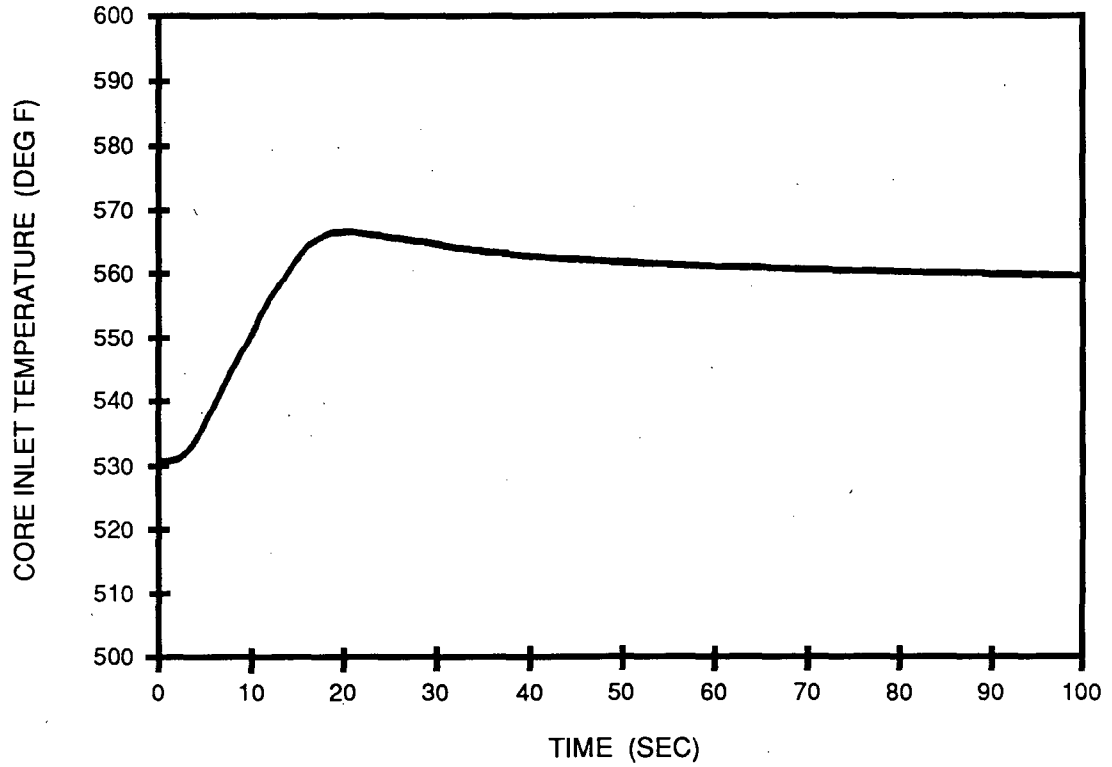


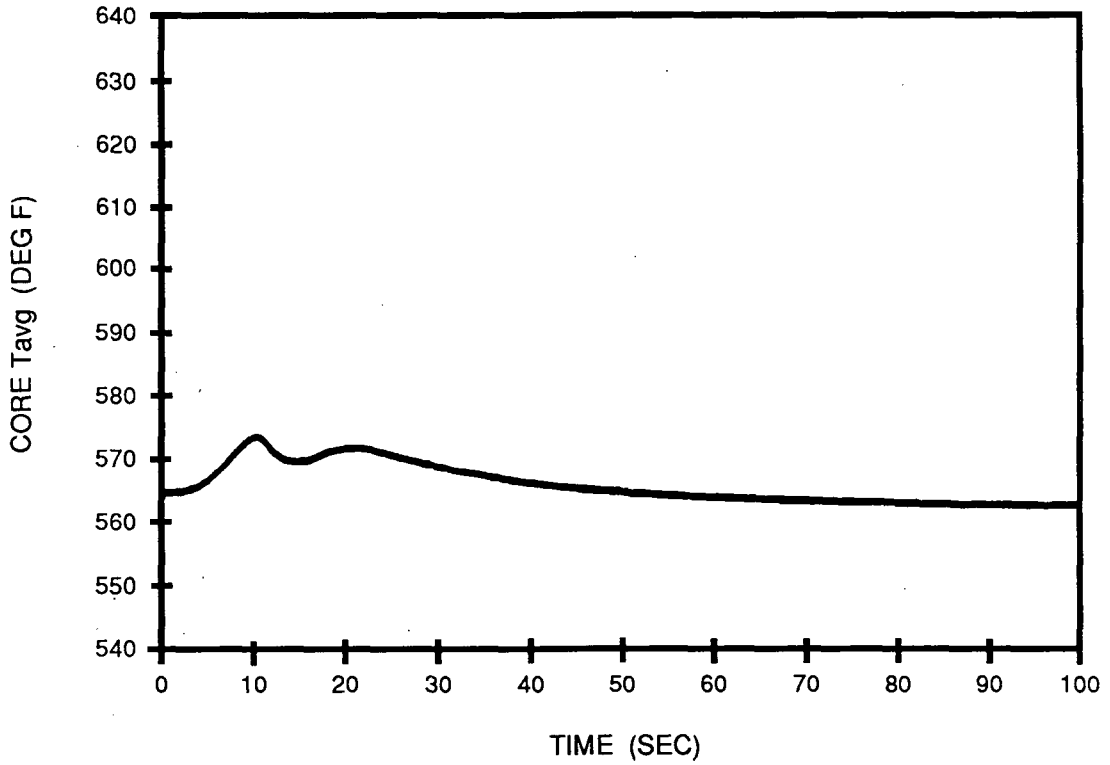


ZION STATION UFSAR

Figure 15.2-5
LOSS OF ELECTRICAL LOAD
PRESSURIZER PRESSURE
AND WATER VOLUME
EOL WITHOUT PRESSURE CONTROL

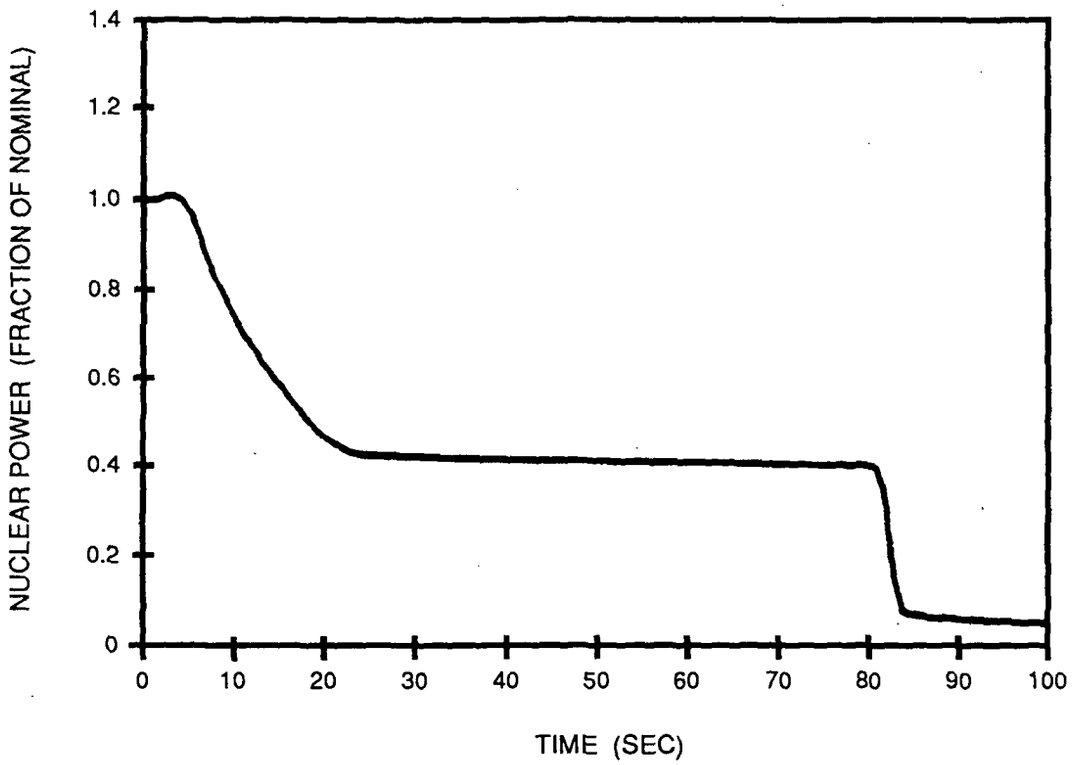
JULY 1993

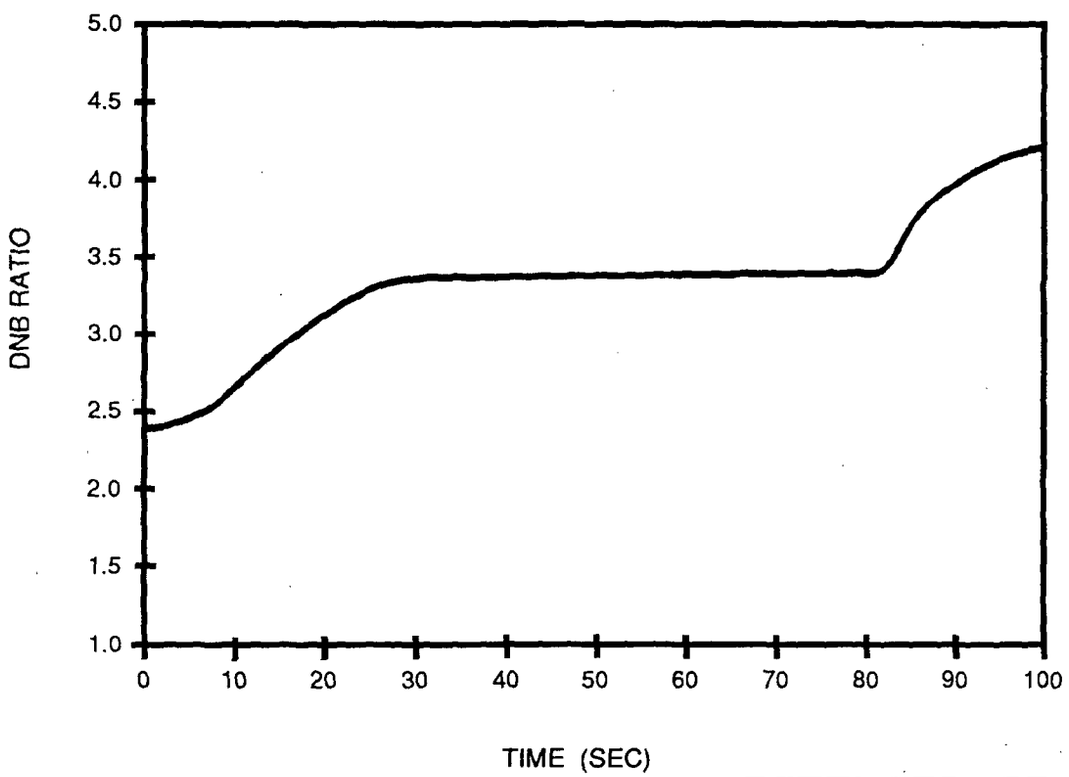




ZION STATION UFSAR

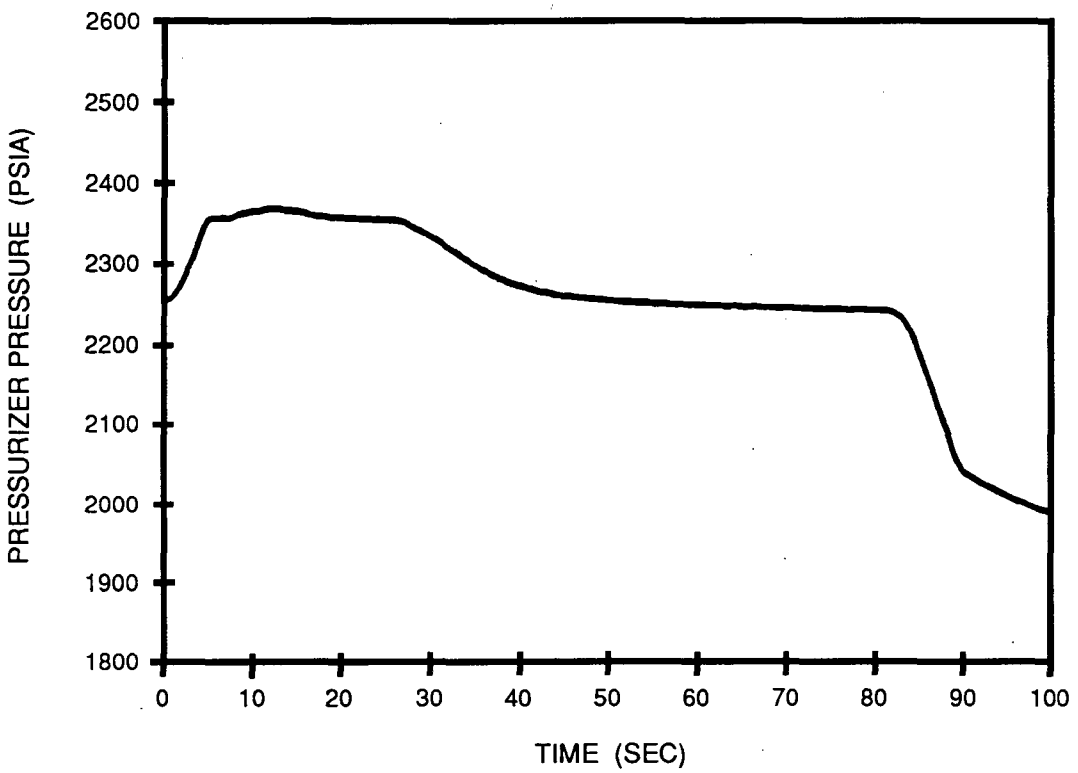
Figure 15.2-6
LOSS OF ELECTRICAL LOAD
CORE INLET TEMPERATURE AND CORE Tavg
EOL WITHOUT PRESSURE CONTROL
JULY 1993

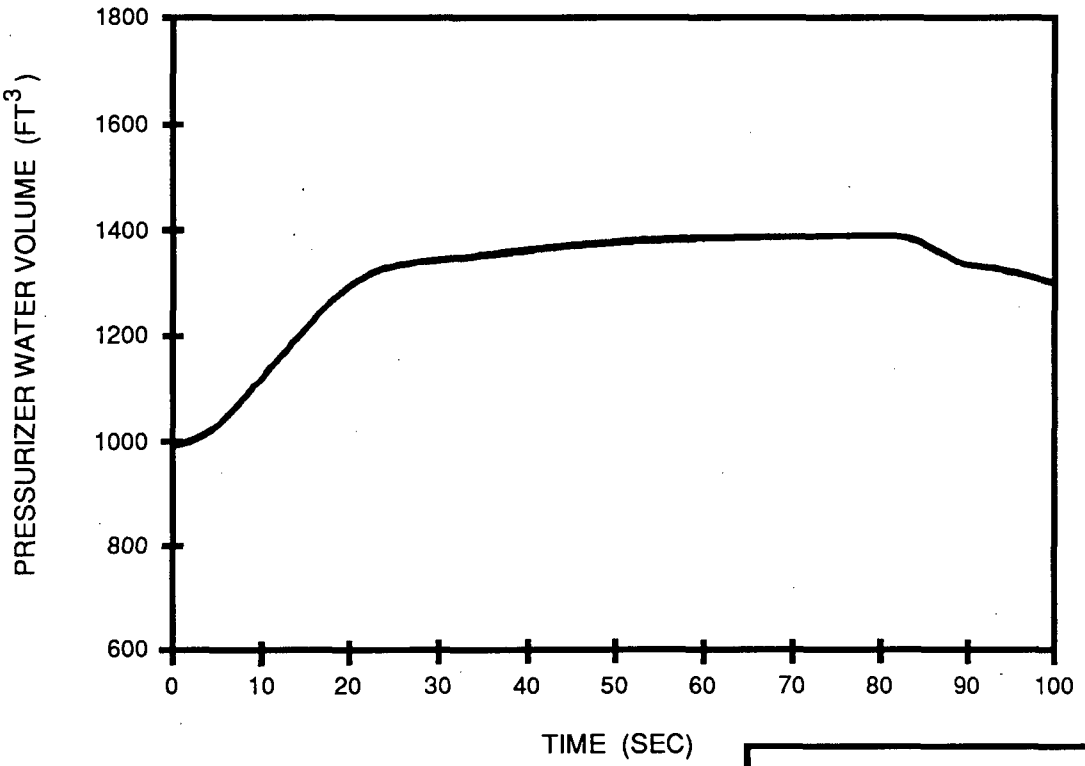




ZION STATION UFSAR

Figure 15.2-7
LOSS OF ELECTRICAL LOAD
NUCLEAR POWER AND DNBR
EOL WITH PRESSURE CONTROL
JULY 1993

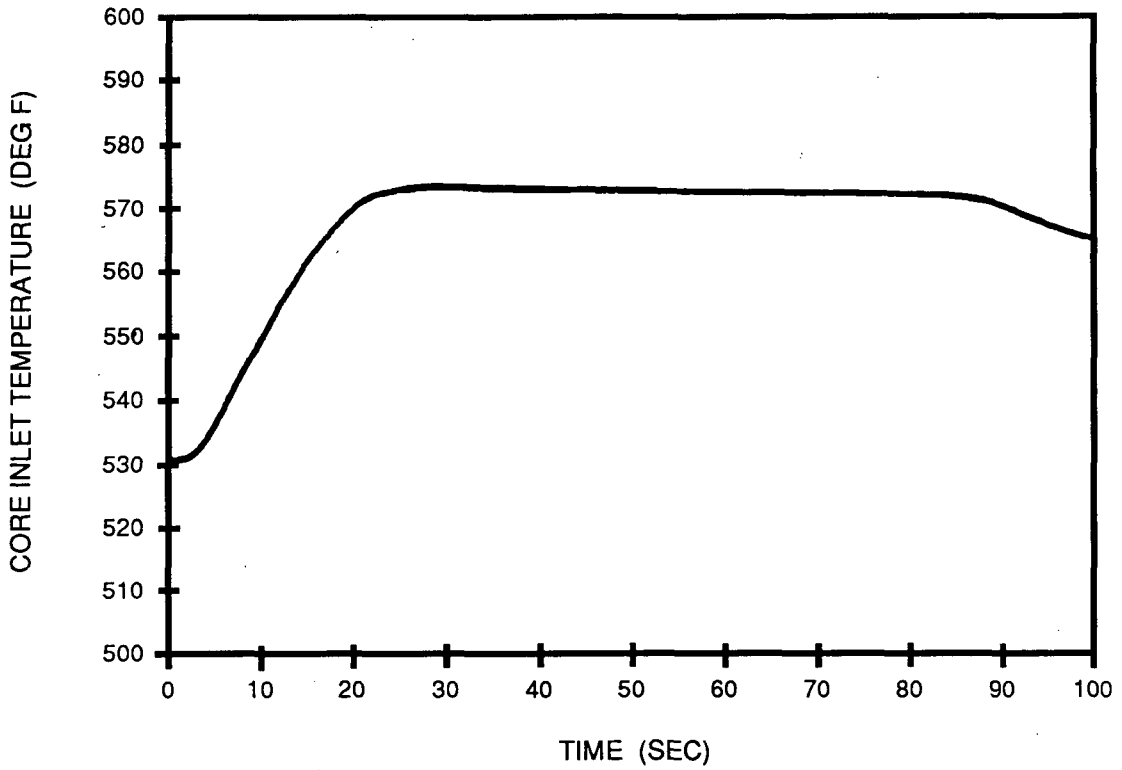


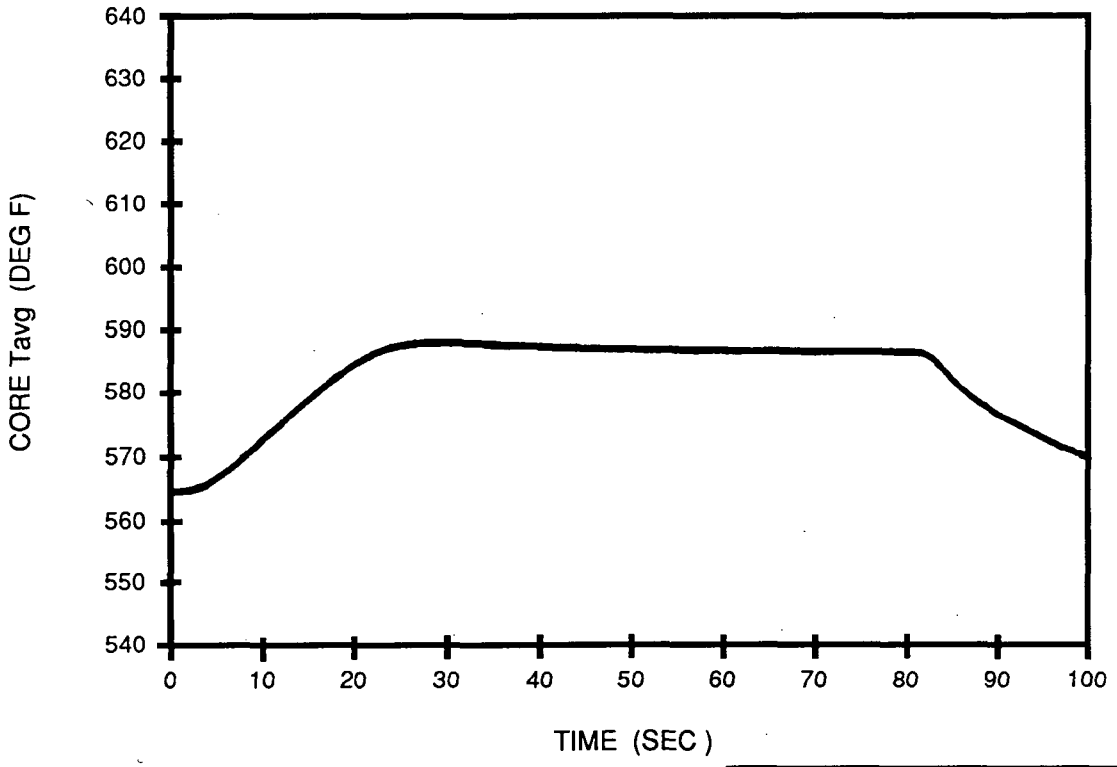


ZION STATION UFSAR

Figure 15.2-8
LOSS OF ELECTRICAL LOAD
PRESSURIZER PRESSURE
AND WATER VOLUME
EOL WITH PRESSURE CONTROL

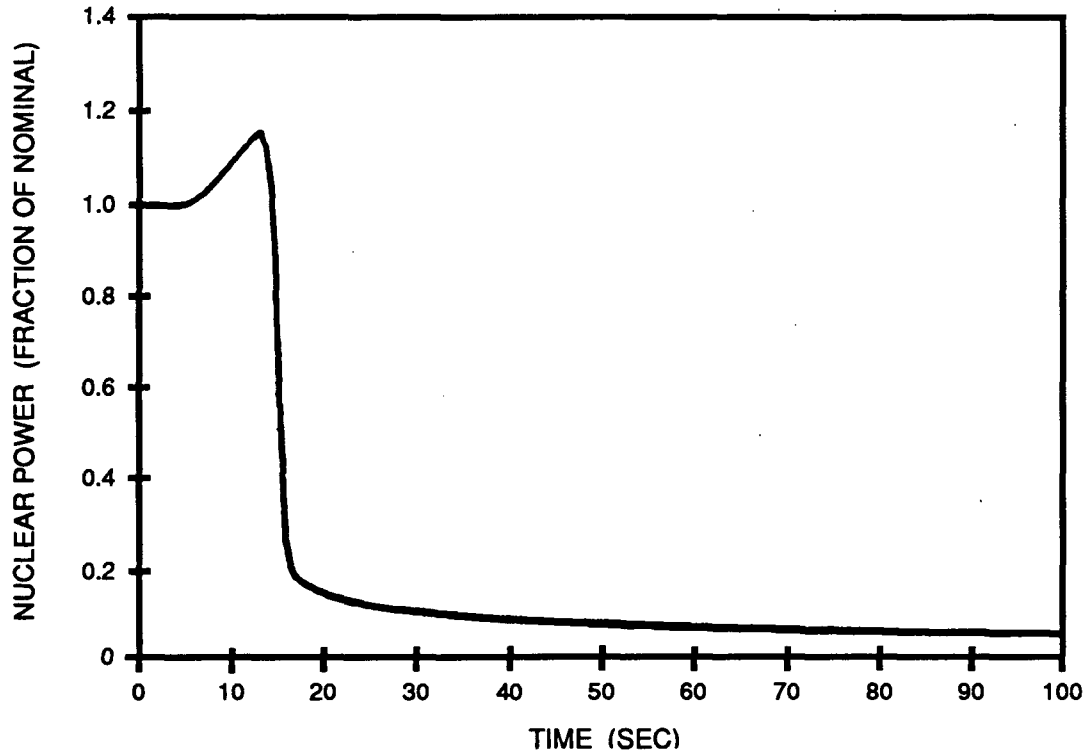
JULY 1993

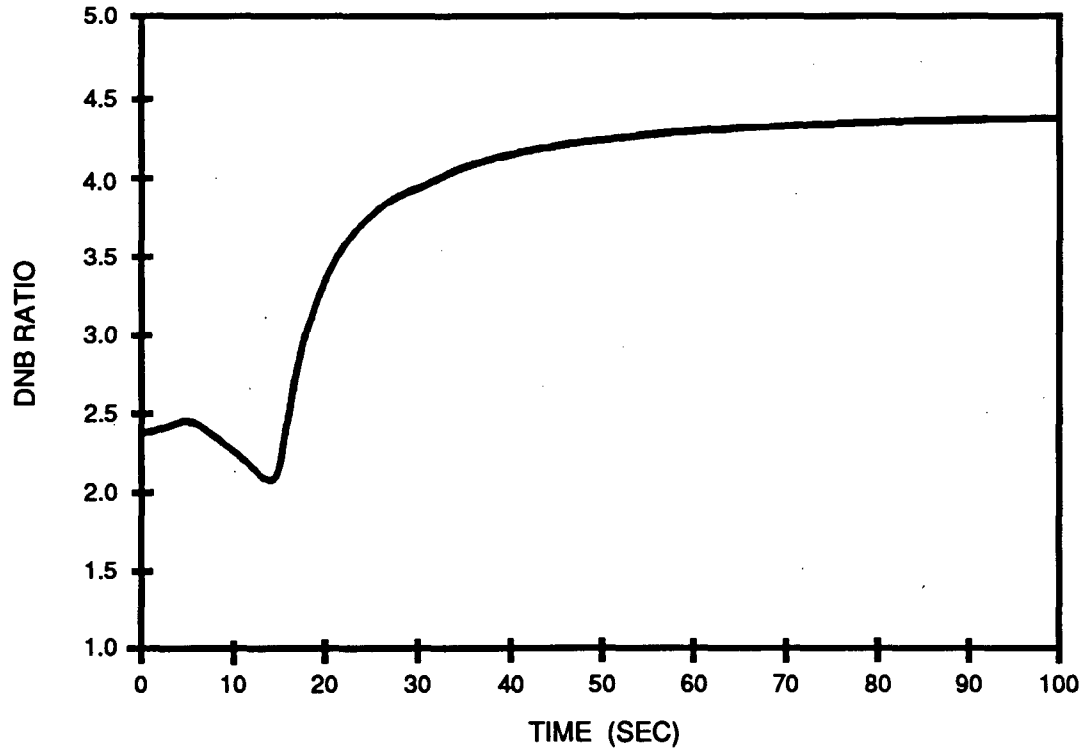




ZION STATION UFSAR

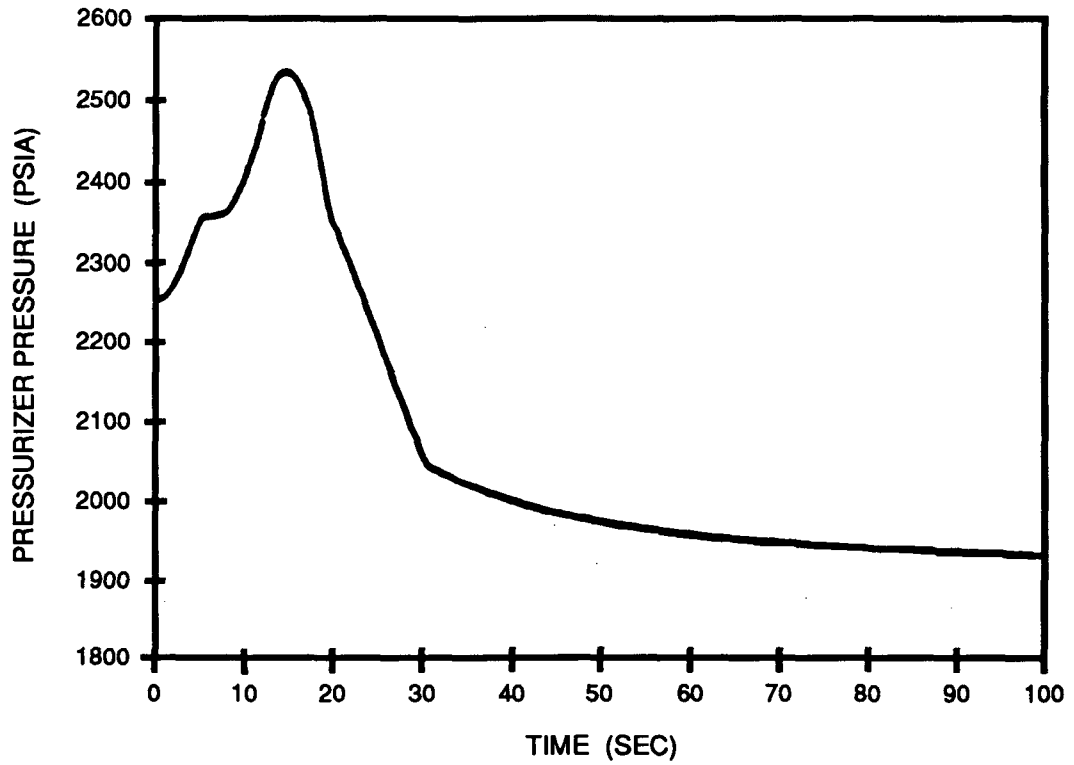
Figure 15.2-9
LOSS OF ELECTRICAL LOAD
CORE INLET TEMPERATURE AND CORE Tavg
EOL WITH PRESSURE CONTROL
JULY 1993

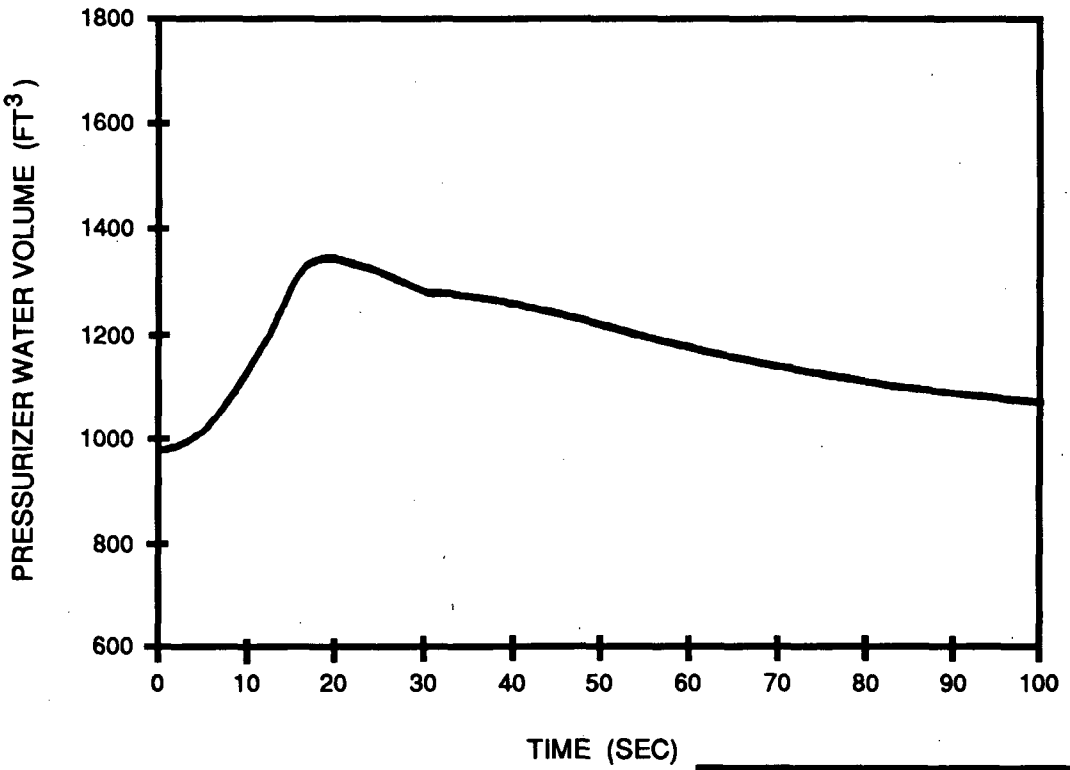




ZION STATION UFSAR

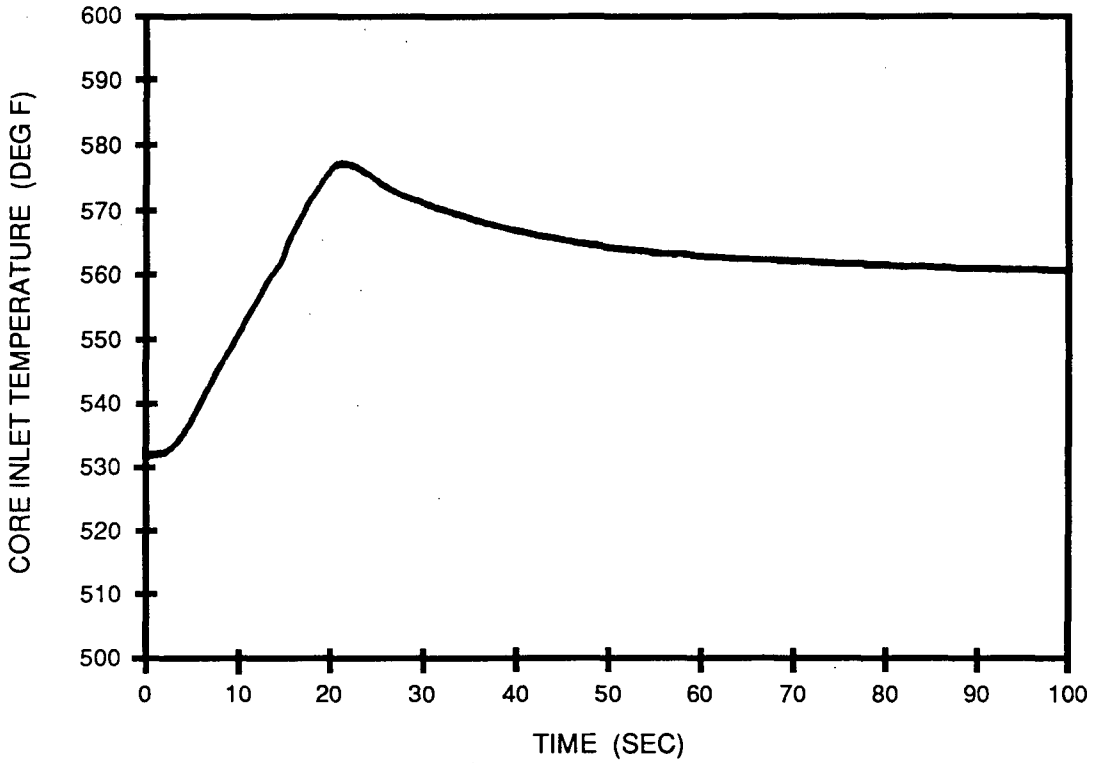
Figure 15.2-10
LOSS OF ELECTRICAL LOAD
NUCLEAR POWER AND DNBR
BOL WITH PRESSURE CONTROL
JULY 1993

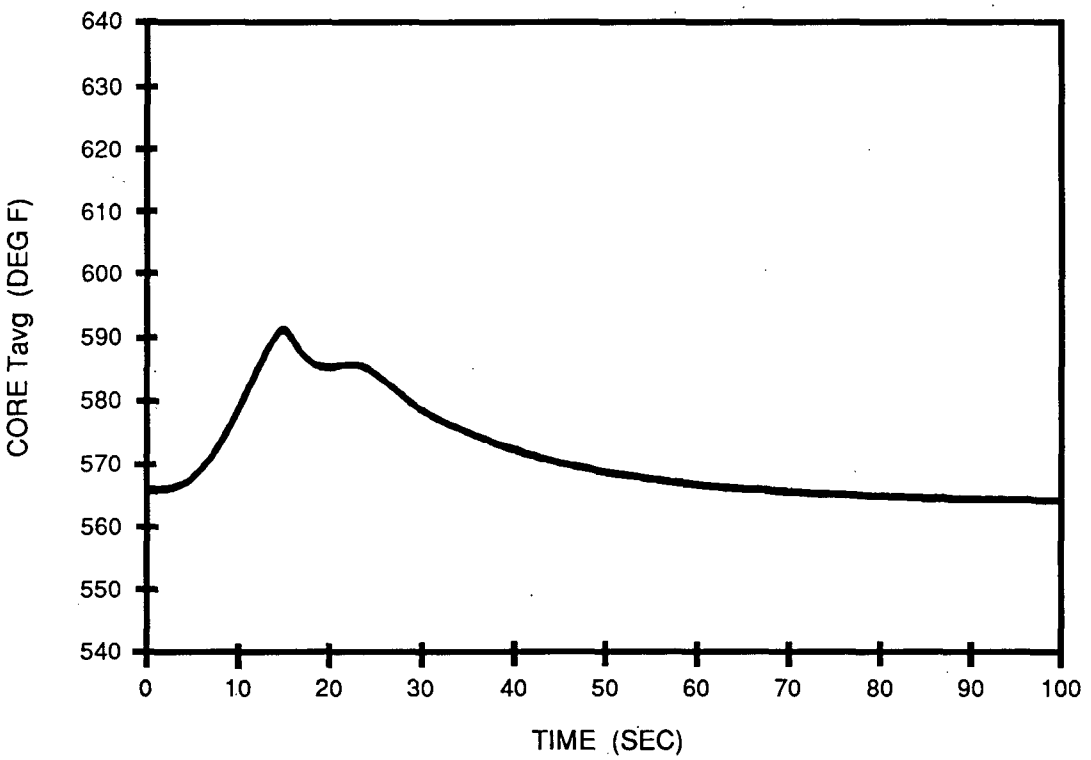




ZION STATION UFSAR

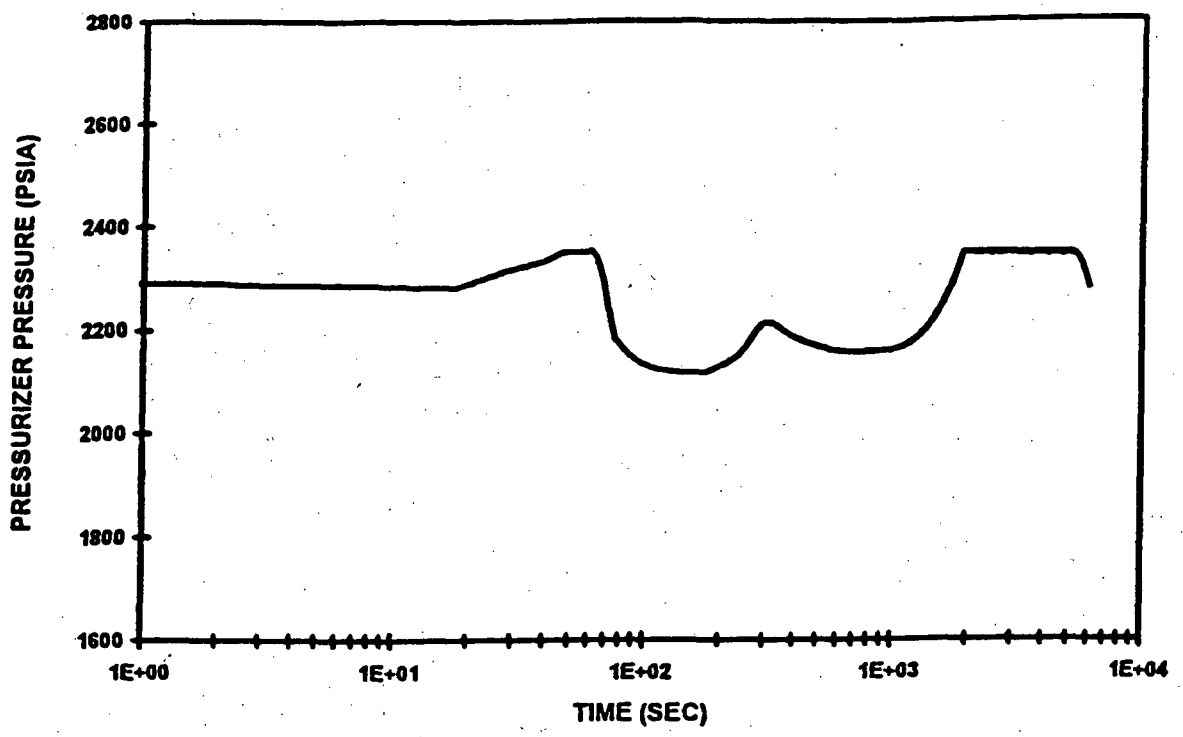
Figure 15.2-11
LOSS OF ELECTRICAL LOAD
PRESSURIZER PRESSURE
AND WATER VOLUME
BOL WITH PRESSURE CONTROL
JULY 1993

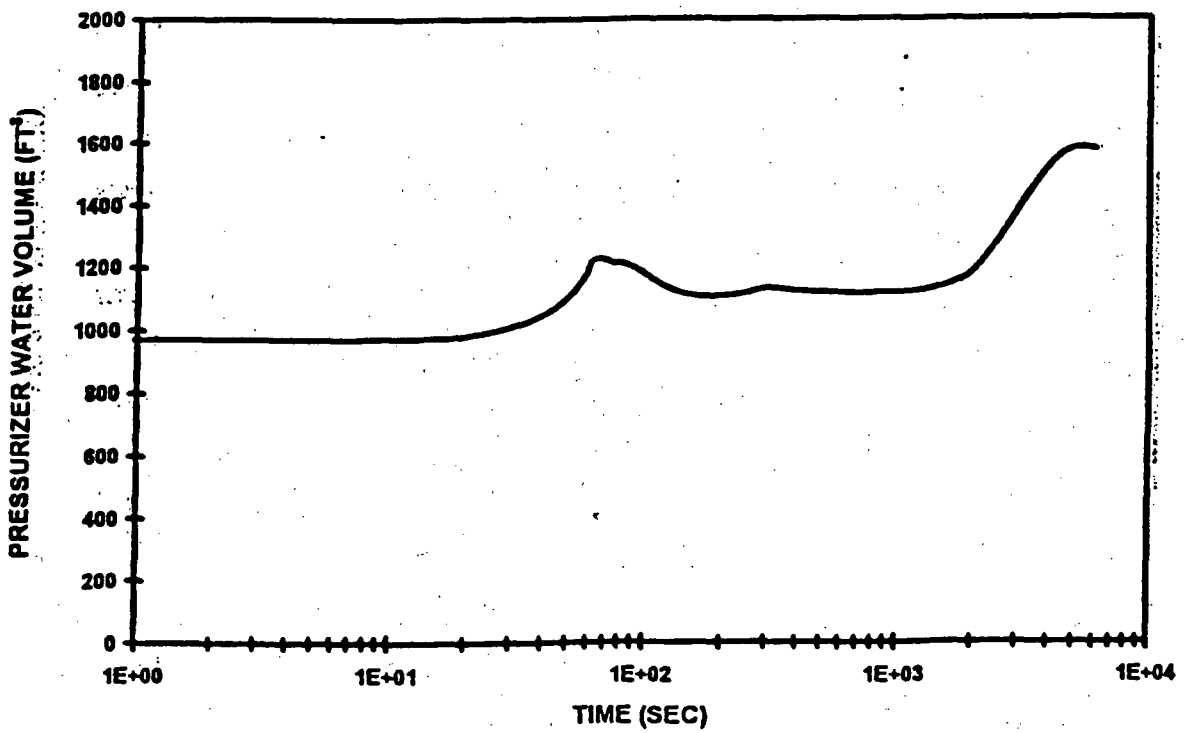




ZION STATION UFSAR

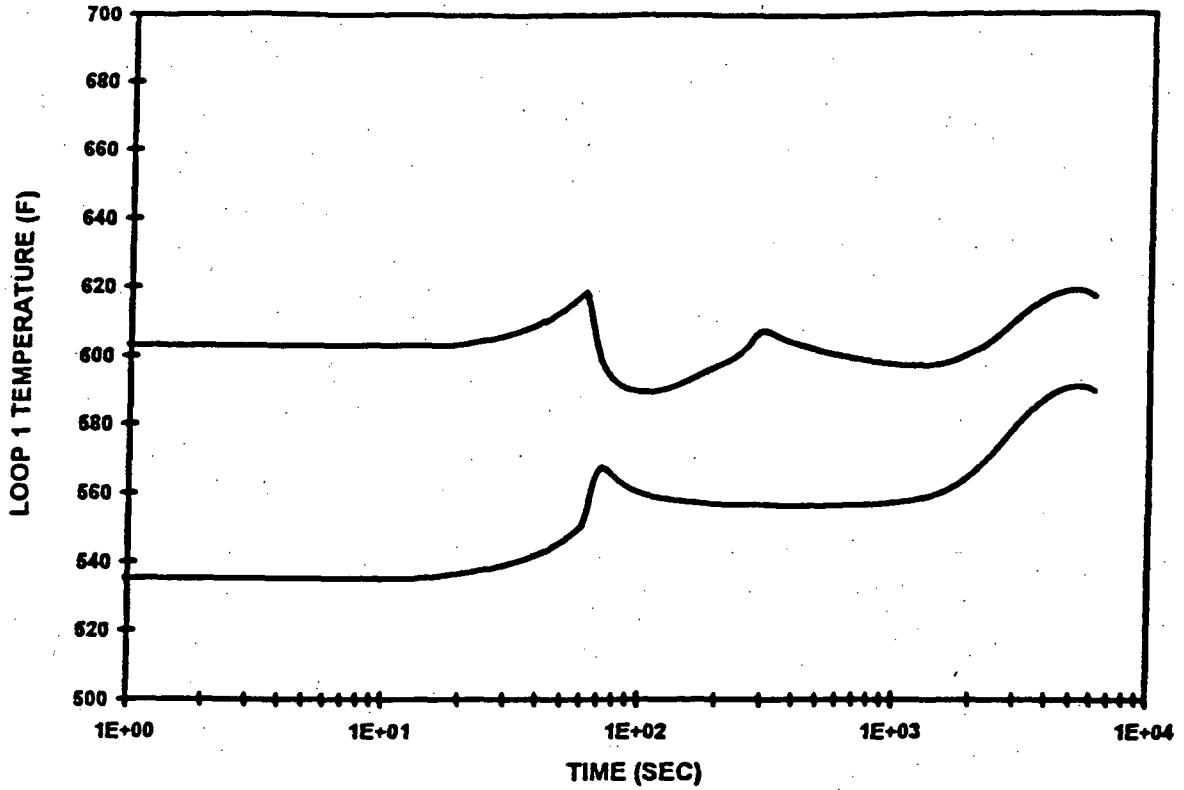
Figure 15.2-12
LOSS OF ELECTRICAL LOAD
CORE INLET TEMPERATURE AND CORE Tavg
BOL WITH PRESSURE CONTROL
JULY 1993

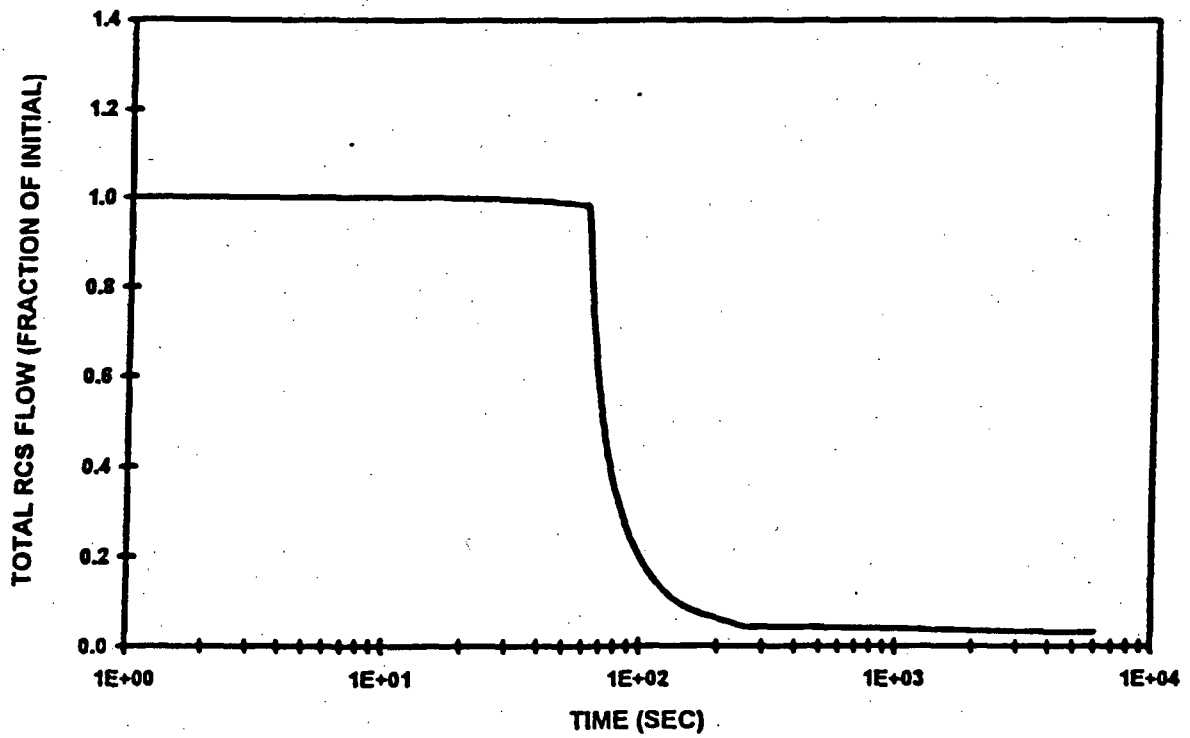




ZION STATION UFSAR

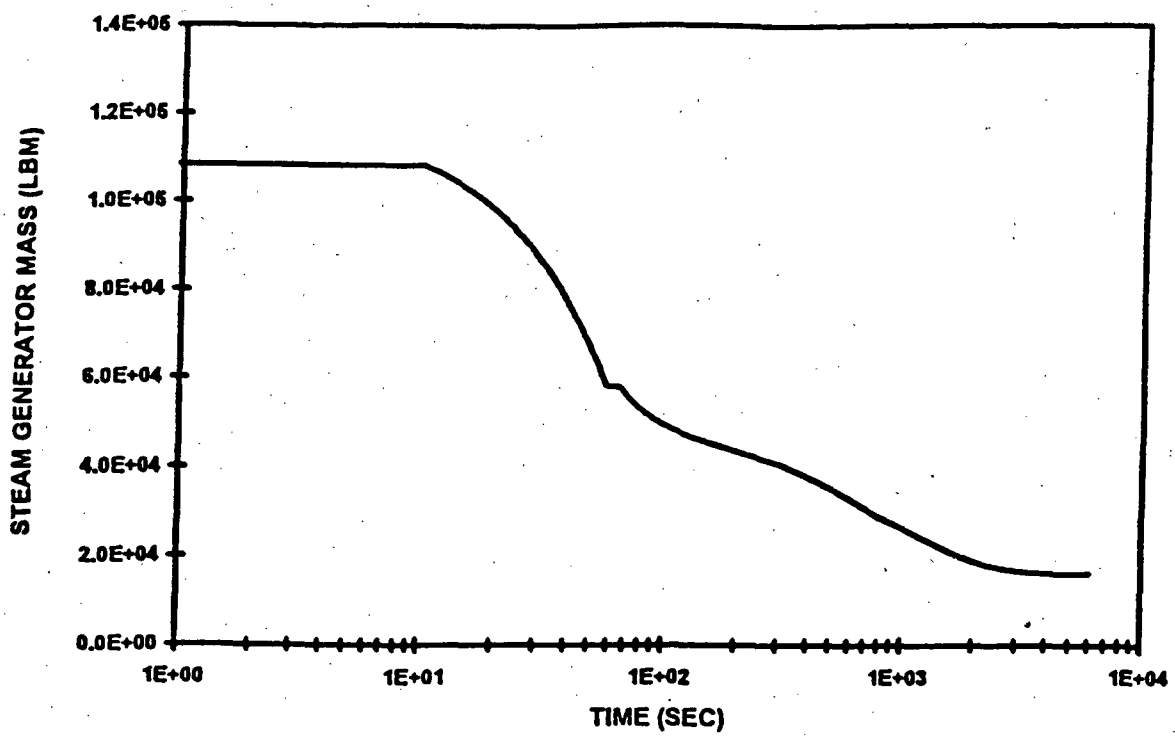
Figure 15.2-13
LOSS OF OFFSITE POWER
PRESSURIZER PRESSURE
AND WATER VOLUME
MAY 1996

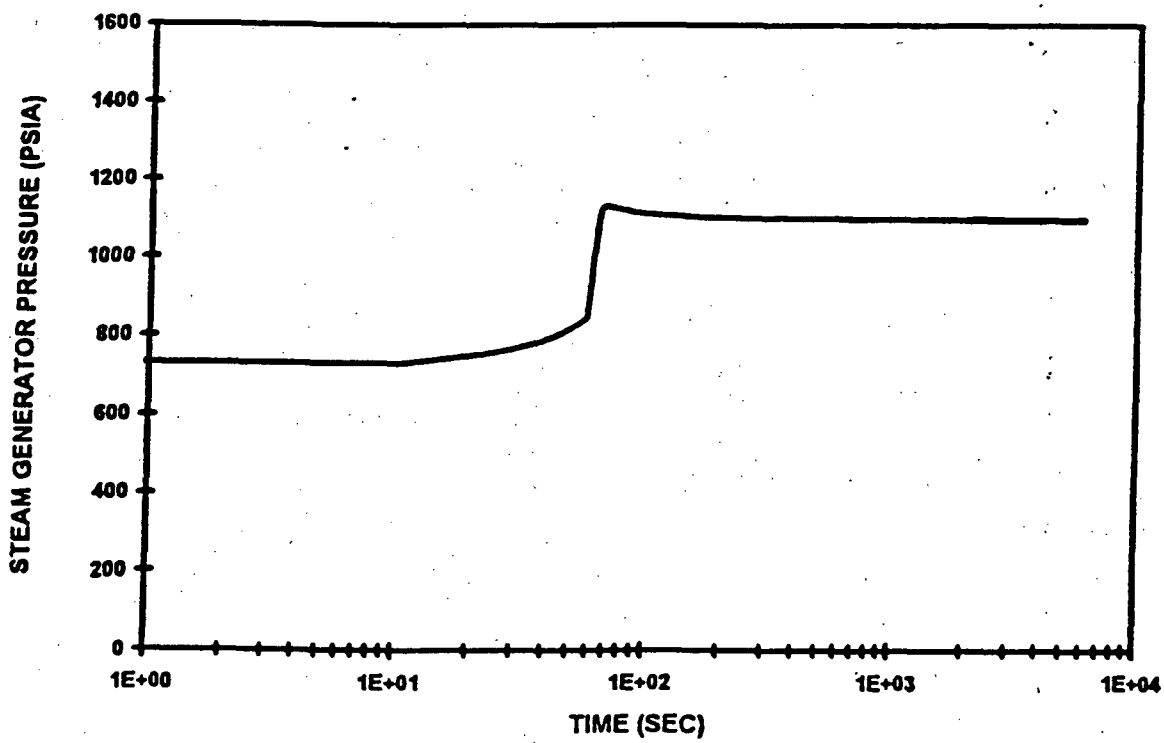




ZION STATION UFSAR

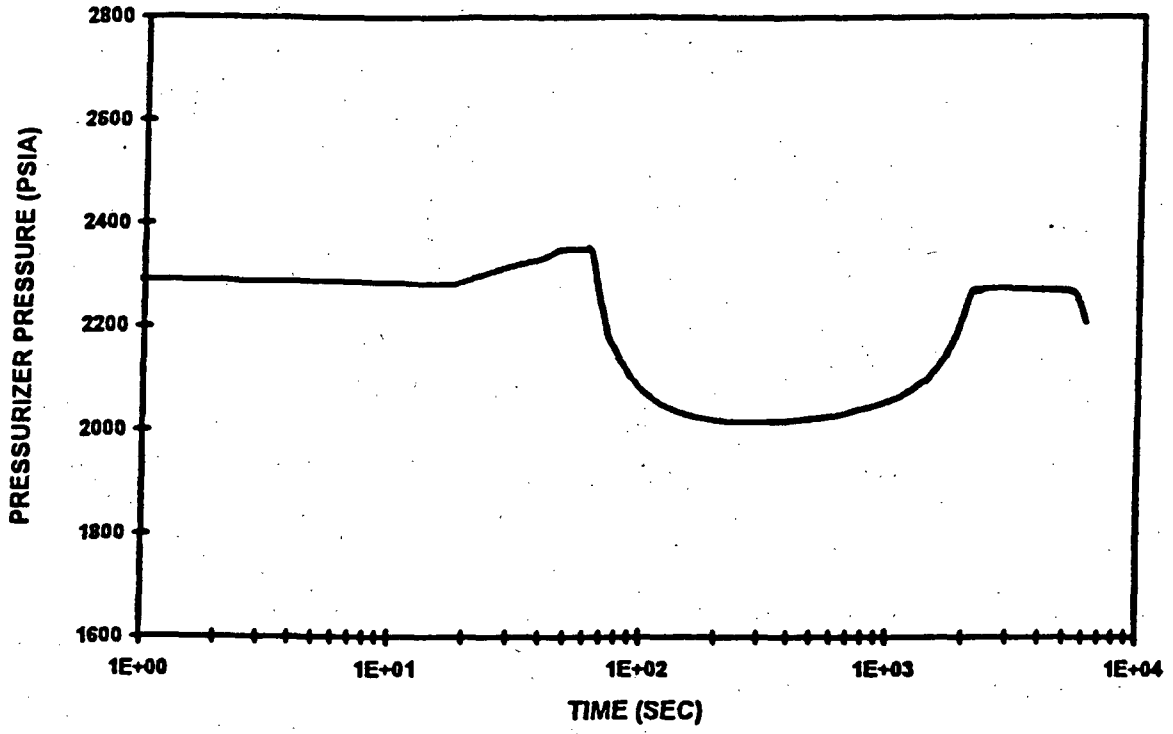
Figure 15.2-14
LOSS OF OFFSITE POWER
LOOP TEMPERATURE
AND RCS FLOW
MAY 1996

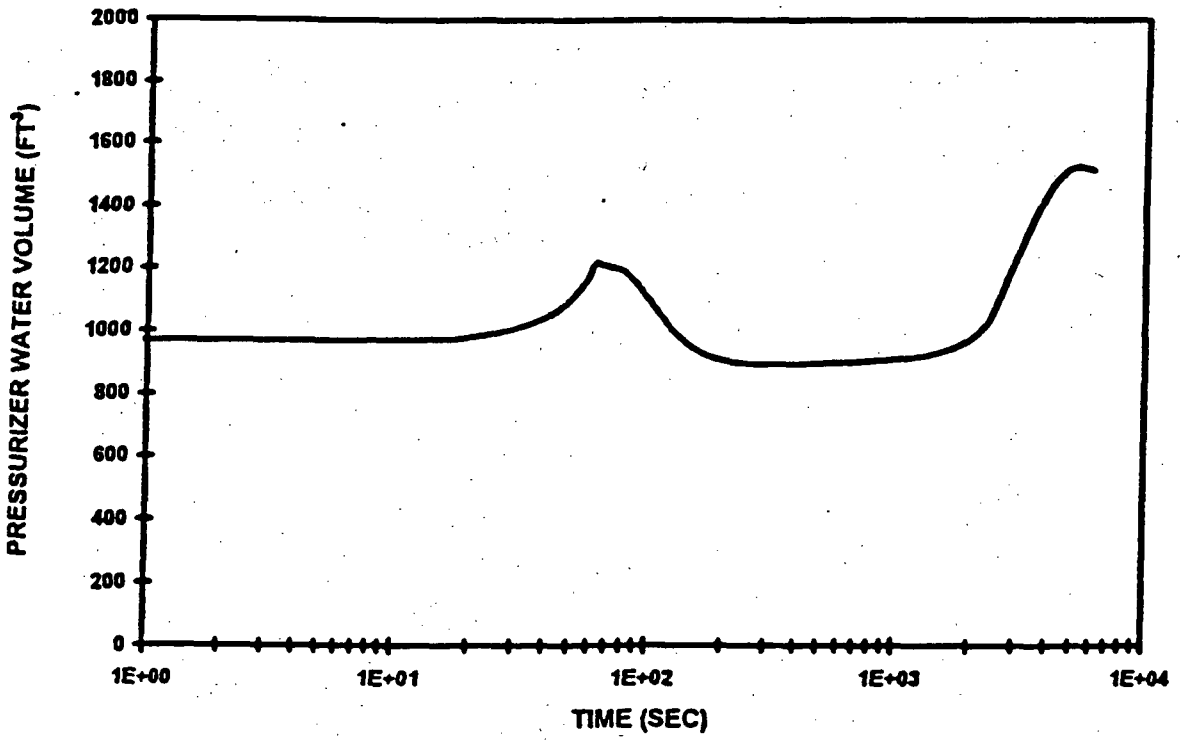




ZION STATION UFSAR

Figure 15.2-15
LOSS OF OFFSITE POWER
STEAM GENERATOR MASS
AND PRESSURE
MAY 1996

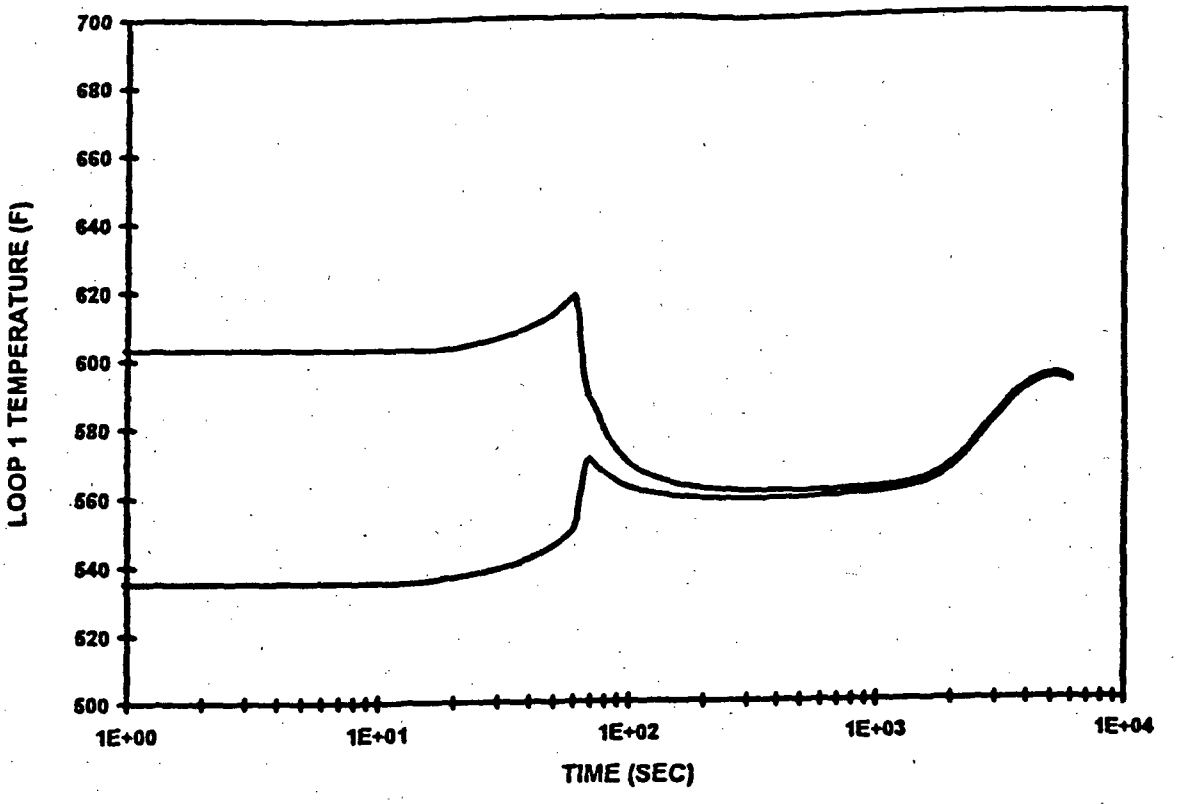


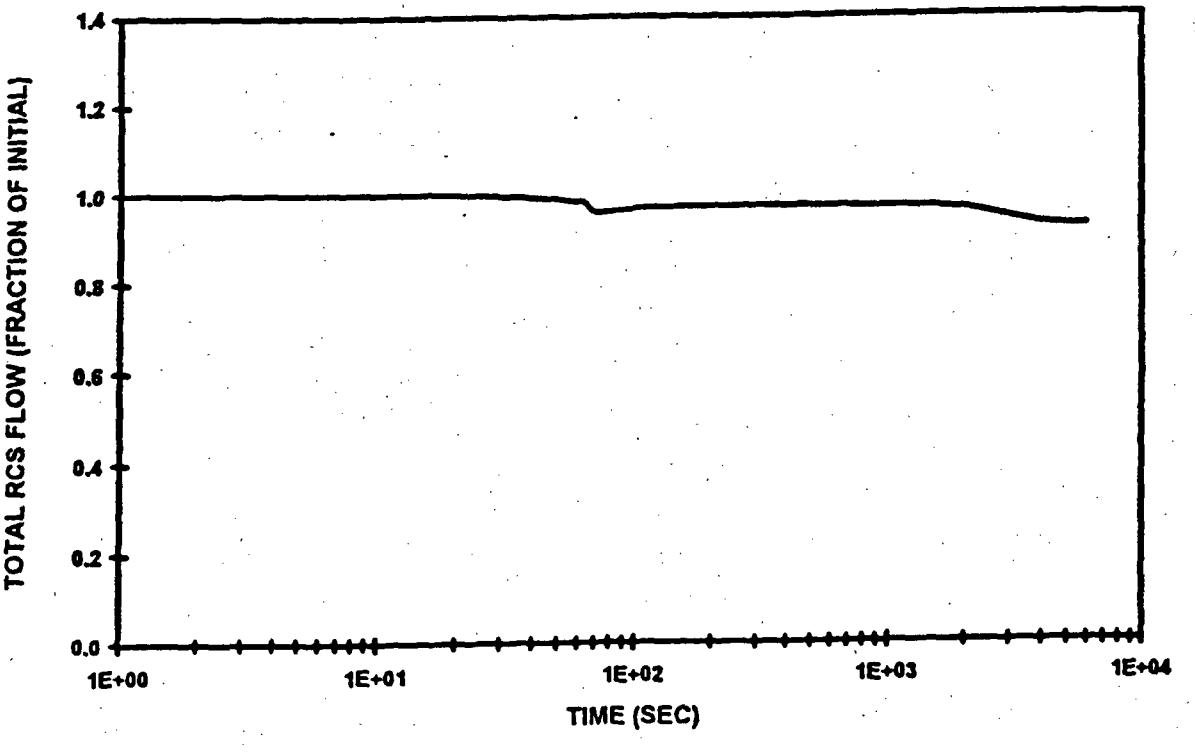


ZION STATION UFSAR

Figure 15.2-16
LOSS OF NORMAL FEEDWATER
PRESSURIZER PRESSURE
AND WATER VOLUME

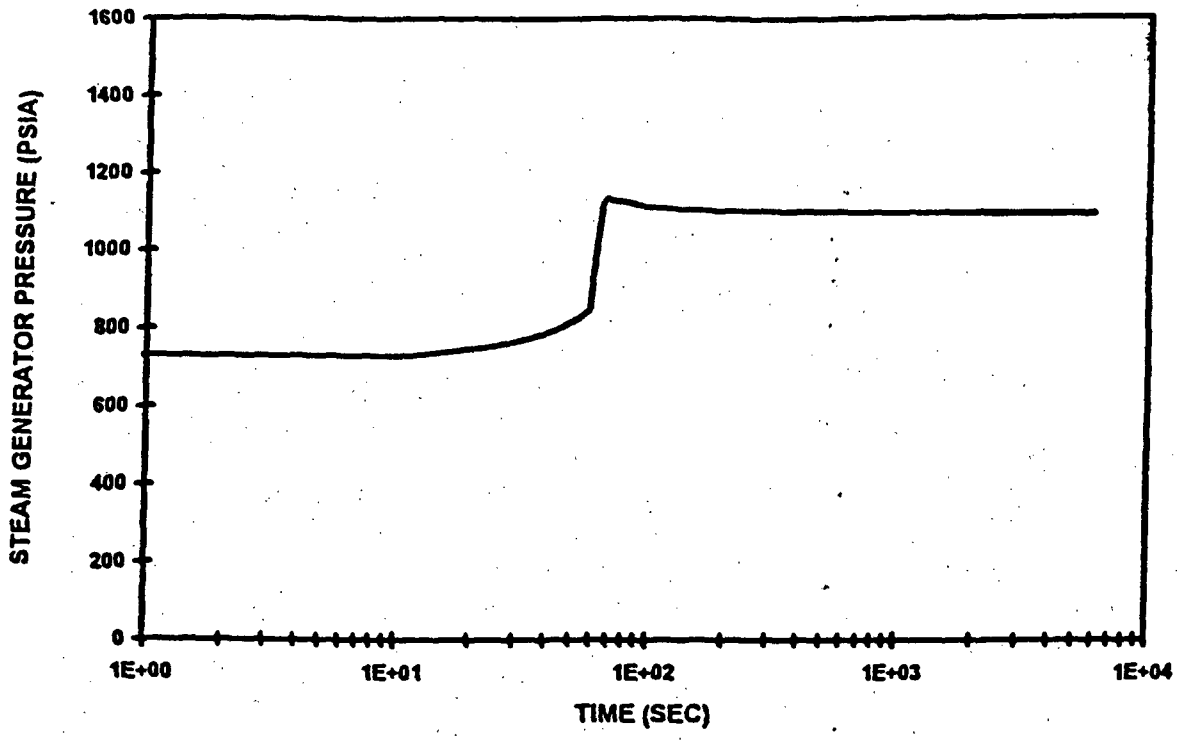
MAY 1996

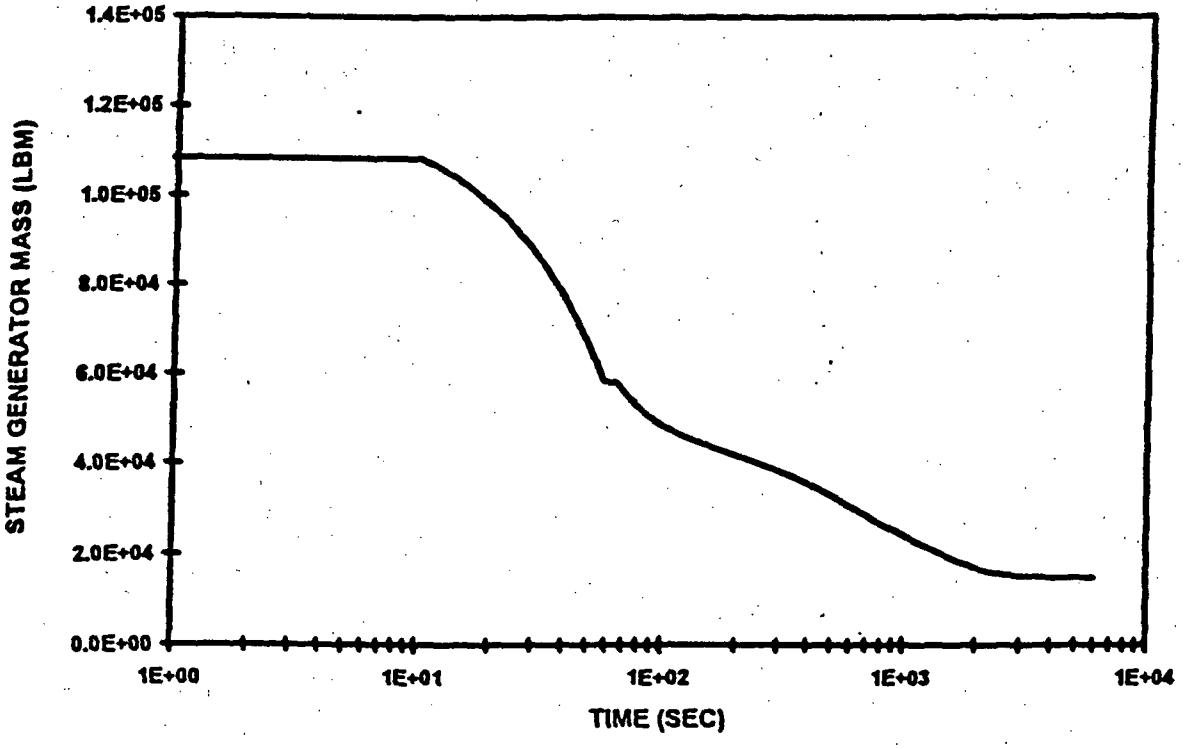




ZION STATION UFSAR

Figure 15.2-17
LOSS OF NORMAL FEEDWATER
LOOP TEMPERATURE
AND RCS FLOW
MAY 1996





ZION STATION UFSAR

Figure 15.2-18
LOSS OF NORMAL FEEDWATER
STEAM GENERATOR PRESSURE
AND MASS
MAY 1996

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Two modes of loss-of-coolant flow have been postulated to occur. The first mode involves a slow reduction in flow, where the flow through the affected loop(s) would coast down as a result of reactor coolant pump (RCP) inertia. The second mode involves a rapid reduction in flow through the affected loop, where the flow through the affected loop would be rapidly reduced as a result of the RCP being rapidly brought to a stop (locked rotor). Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented:

1. partial loss of forced reactor coolant flow;
2. complete loss of forced reactor coolant flow;
3. reactor coolant pump locked rotor; and
4. reactor coolant pump shaft break.

15.3.1 Single and Multiple Reactor Coolant Pump Trips

15.3.1.1 Incident Description

A loss-of-coolant flow incident can result from a mechanical or electrical failure in a RCP, or from a fault in the power supply for these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss-of-coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The following trip circuits provide the necessary protection against a loss-of-coolant flow incident:

1. Undervoltage or underfrequency on pump power supply buses;
2. Pump circuit breaker opening; and
3. Low reactor coolant flow.

These trip circuits and their redundancy are further described in Section 7.2.

The normal power supplies for the pumps are from four separate buses: two normally connected to the unit auxiliary transformer and two normally connected to the system auxiliary transformer. When a generator trip occurs, the pumps that are not receiving power from the system auxiliary transformer are automatically transferred to this transformer which is fed from external power lines. The pumps will continue to supply coolant flow to the core. The simultaneous loss of power to all RCPs is a highly

unlikely event. Since each pump is on a separate bus, a single bus fault would not result in the loss of more than one pump.

15.3.1.2 Method of Analysis

The following loss-of-flow cases are analyzed:

1. Complete loss of forced reactor coolant flow - Loss of four pumps at 100% nominal reactor power (3250 MWt) with four loops operating; and

2. Partial loss of forced reactor coolant flow - Loss of one pump at 100% nominal reactor power (3250 MWt) with four loops operating.

15.3.1.2.1 Complete Loss of Forced Reactor Coolant Flow

15.3.1.2.1.1 Incident 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.

Normal power for the RCPs is supplied through buses from a transformer connected to the generator auxiliary transformer and the system auxiliary transformer. When a generator trip occurs, two of the pumps continue to be supplied from the system auxiliary transformer (external power) and the pumps continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping 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.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Section 15.0.1.

The following signals provide the necessary protection against a complete loss of flow accident:

1. RCP power supply undervoltage or underfrequency,
2. Low reactor coolant loop flow.

The reactor trip on RCP undervoltage is provided to protect against conditions which can cause a loss of voltage to all RCPs, i.e., loss of ac power. This function is blocked below approximately 10% power (Permissive P-7).

The reactor trip on RCP under frequency is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. The RCP underfrequency function does not result in a direct reactor trip. When this function actuates, it opens all the RCP breakers which, in turn, generate a reactor trip on the RCP breaker position function. Reference 5 provides analyses of grid frequency disturbances and the resulting Nuclear Steam Supply System (NSSS) protection requirements which, while not explicitly required in the original design, are addressed by the Reactor Protection System (RPS).

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive P-7) and the power level corresponding to Permissive P-8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/sec, the low flow trip function will protect the core from underfrequency events. This effect is fully described in Reference 5.

15.3.1.2.1.2 Method of Analysis

One case has been analyzed: loss of all four RCPs with four loops in operation.

This transient is analyzed by three digital computer codes. First the LOFTRAN code (Reference 1) 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, the primary system pressure and temperature transients, and the nuclear power transient. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Reference 3) is used to calculate the minimum departure from nucleate boiling ratio (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.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1.2.2, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by either RCP bus undervoltage or underfrequency.

The accident is analyzed using the Revised Thermal Design Procedure (RTDP) as described in Reference 4. Plant characteristics and initial conditions are discussed in Subsection 15.0.3. Uncertainties in the initial conditions are included in the safety analysis DNBR limit as described in Reference 4.

No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

15.3.1.2.1.3 Results

Figures 15.3-1 through 15.3-4 show the transient response for the loss of power to all RCPs with four loops in operation. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-4 shows that the DNBR is always greater than the safety analysis limit value.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

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

15.3.1.2.1.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.1.2.2 Partial Loss of Forced Reactor Coolant Flow

This transient is analyzed by three digital computer codes. First, the LOFTRAN code (Reference 1) 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, the primary system pressure and temperature transients, and the nuclear power transient. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Reference 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.

The accident is analyzed using the RTDP as described in Reference 4. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

15.3.1.2.2.1 Initial Operating Conditions

Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in the initial conditions are included in the safety analysis DNBR limit as described in Reference 4. The minimum measured flow value (Table 15.0-2B) was also included.

15.3.1.2.2.2 Reactivity Coefficients

A conservatively large absolute value of the Doppler only power coefficient is used (see Figure 15.0-3). The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.0194 ΔK.

The most positive moderator temperature coefficient (+7 pcm/°F) is assumed since this results in the maximum core power and hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

For this analysis, the curve of trip reactivity versus time (Figure 15.0-6) was used.

15.3.1.2.2.3 Results

Figures 15.3-5 through 15.3-8 show the transient response for the loss of one reactor coolant pump with four loops in operation. The figures include trends of the core flow, loop flow, nuclear power, and core heat flux coastdowns. The reactor is tripped on a low loop flow signal. Figure 15.3-8 shows that the DNBR is always greater than the limit value.

For the case analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperatures do not increase far above their respective initial values.

The calculated sequence of events is shown in Table 15.3-1. The affected RCP will continue to coastdown, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.2.2.4 Conclusions

Since DNB does not occur in the partial loss-of-coolant flow incident, there is no cladding damage and no release of fission products into the reactor coolant. All applicable acceptance criteria are met. Therefore, once the fault is corrected, the plant can be returned to service in the normal manner.

15.3.2 Boiling Water Reactor Recirculation Loop Controller Malfunctions that Result in Decreasing Flow Rate

This section is not applicable to the Zion Station.

15.3.3 Reactor Coolant Pump Locked Rotor

15.3.3.1 Incident Description

A transient analysis is performed for the instantaneous seizure of an RCP rotor (locked rotor). Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator 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 the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer causes a pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves. The sequence of events initiated by the insurge depends on the rate of insurge and pressure increase. The 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 this analysis.

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

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis, which permits reverse spinning but no forward flow, has been performed and represents the most limiting condition for the locked rotor and pump shaft break accidents.

15.3.3.2 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN Code (Reference 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and the peak RCS pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN Code (Reference 2), which uses the core flow and the nuclear power values calculated by LOFTRAN. The FACTRAN Code includes a film boiling heat transfer coefficient.

One case is analyzed: one locked rotor/shaft break with four loops in operation.

15.3.3.2.1 Initial Operating Conditions

At the beginning of the postulated locked rotor accident, the plant is assumed to be operating at nominal conditions and the RTDP is used as described in Reference 4. Plant characteristics and initial conditions are discussed in Section 15.0.3.

For the peak pressure and peak clad temperature evaluations, one analysis is performed and the initial pressure is conservatively estimated as 40 psi above the nominal pressure of 2250 psia to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 15.3-10 is at the point in the RCS having the maximum pressure (i.e., the outlet of the faulted loop's RCP). The remainder of the plant is assumed to be operating under the most adverse steady-state operating condition, e.g., 102% of the NSSS design thermal power and the maximum steady-state coolant average temperature including uncertainties.

For a conservative analysis of fuel rod behavior, the hot spot evaluation assumes that DNB occurs at the initiation of the transient and continues throughout the event. This assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

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

The most positive moderator temperature coefficient (+7 pcm/°F) is assumed since this results in the maximum core power and hot spot heat flux during the initial part of the transient when the minimum DNBR is reached. A conservatively large absolute value of the Doppler only power coefficient is used (see Figure 15.0-3). The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.0194 ΔK. For this analysis, the curve of trip reactivity versus time (Figure 15.0-5) was used with a 4% ΔK trip reactivity which includes the most reactive rod cluster control assembly (RCCA) stuck out of the core.

15.3.3.2.2 (Section Deleted)

15.3.3.2.3 Evaluation of the Pressure Transient

A detailed model was used to determine the peak pressure in the RCS under the postulated accident conditions and to obtain the neutron flux response as a function of time which is used later in the analysis.

After pump seizure, neutron flux is rapidly reduced because of the control rod insertion upon plant trip. In this analysis, rod motion is assumed to begin one second after the flow in the affected loop reaches 85% of nominal flow.

No credit was 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 valves are actuated at 2500 psia and their capacity for steam relief is described in Chapter 5.

15.3.3.2.4 Evaluation of DNB 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 clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.65 times the average rod power (i.e., F_Q = 2.65) at the initial core power level.

Calculation of the extent of DNB in the core during the locked rotor event is performed with the THINC computer code (Reference 3). The THINC code uses the core heat flux calculated from FACTRAN and the core flow transient from LOFTRAN.

15.3.3.2.5 Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation (Reference 5). The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate 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 the clad temperature response. As indicated earlier, DNB was assumed to start at the beginning of the accident.

15.3.3.2.6 Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) have a pronounced influence on the

thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. For the first part of the transient, a high gap coefficient produces higher clad temperatures since the heat stored and generated in the fuel pellet tries to redistribute itself in the cooler clad. This effect of the gap coefficient is reversed when the clad temperature exceeds the pellet temperature in cases where a zirconium-steam reaction is present. The gap coefficient was taken to be the conservatively large value of 10,000 Btu/hr-ft^2-°F which is greater than the highest value calculated during core life.

15.3.3.2.7 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following correlation, which defines the rate of the zirconium-steam reaction, was introduced into the model (see Reference 4):

$$\frac{dw^2}{dt} = (33.3 \times 10^6)\ e^{-\left(\frac{45,500}{1.986T}\right)}$$

where w = amount reacted, mg/cm^2
t = time, seconds
T = temperature, °K

The reaction heat is 1510 cal/gm.

15.3.3.3 Results

The transient results without offsite power available are shown in Figures 15.3-9 through 15.3-12. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than the 2700°F transient limit for the locked rotor accident. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. The results of these calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are also summarized in Table 15.3-3.

The calculated sequence of events is shown in Table 15.3-2. Figure 15.3-9 shows that the core flow rapidly coasts down to a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.4 Conclusion

The results of the transient analysis show that 5% of the fuel rods will have calculated DNBR values below the safety analysis limit value.

Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered. Also, since the peak clad average temperature calculated for the hot spot during the worst transient remains considerably less than the 2700°F transient limit, the core will remain in place and intact with no loss of core cooling capability.

15.3.4 Reactor Coolant Pump Shaft Break

Refer to Section 15.3.3.

15.3.5 References, Section 15.3

1. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2. Hargrove, H.G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

3. Friedland, A.J. and S. Ray, "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

4. Friedland, A.J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

5. Bishop, A.A., Sanberg, R.O. and Tong, L.S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS
LOSS OF REACTOR COOLANT FLOW

ACCIDENT	EVENTS	TIME (sec)
Partial Loss of Reactor Coolant Flow	Coastdown begins	0.0
	Low loop flow reactor trip signal generated	1.47
	Rods begin to drop	2.47
	Minimum DNBR occurs	3.54
Complete Loss of Reactor Coolant Flow	All operating pumps lose power and begin coastdown	0.0
	Rods begin to drop after an undervoltage reactor trip signal is generated	1.2
	Minimum DNBR occurs	3.0

TABLE 15.3-2

TIME SEQUENCE OF EVENTS
REACTOR COOLANT PUMP LOCKED ROTOR

ACCIDENT	EVENTS	TIME (sec)
Reactor Coolant Pump Locked Rotor	Rotor locks (Pump shaft breaks)	0.0
	Low loop flow reactor trip setpoint reached	0.03
	Rods begin to drop	1.03
	Maximum RCS pressure occurs	3.4
	Maximum clad temperature occurs	3.7

TABLE 15.3-3

SUMMARY OF RESULTS FOR LOCKED ROTOR ACCIDENT

	Four Loops Operating
Maximum Reactor Coolant System Pressure (psia)	2557
Maximum Clad Average Temperature at the Core Hot Spot (°F)	1967
$Zr-H_2O$ Reaction at the Core Hot Spot (% by weight)	0.52

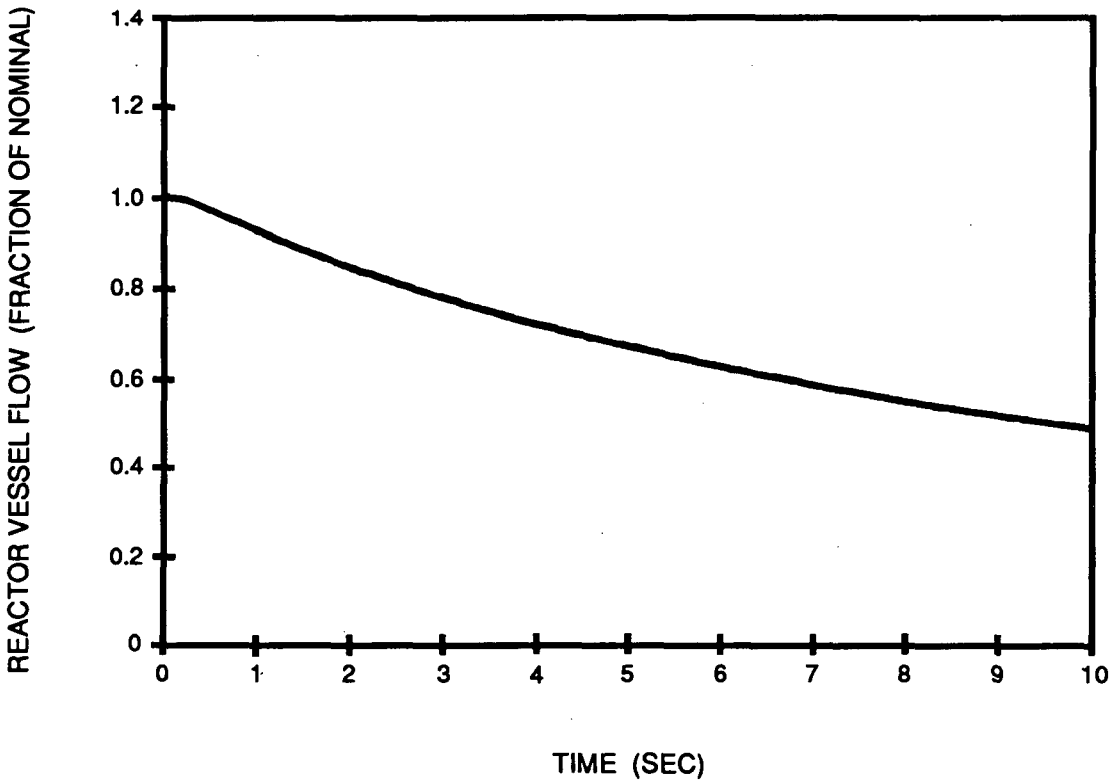


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Figure 15.3-1

COMPLETE LOSS OF FLOW
CORE FLOW VERSUS TIME

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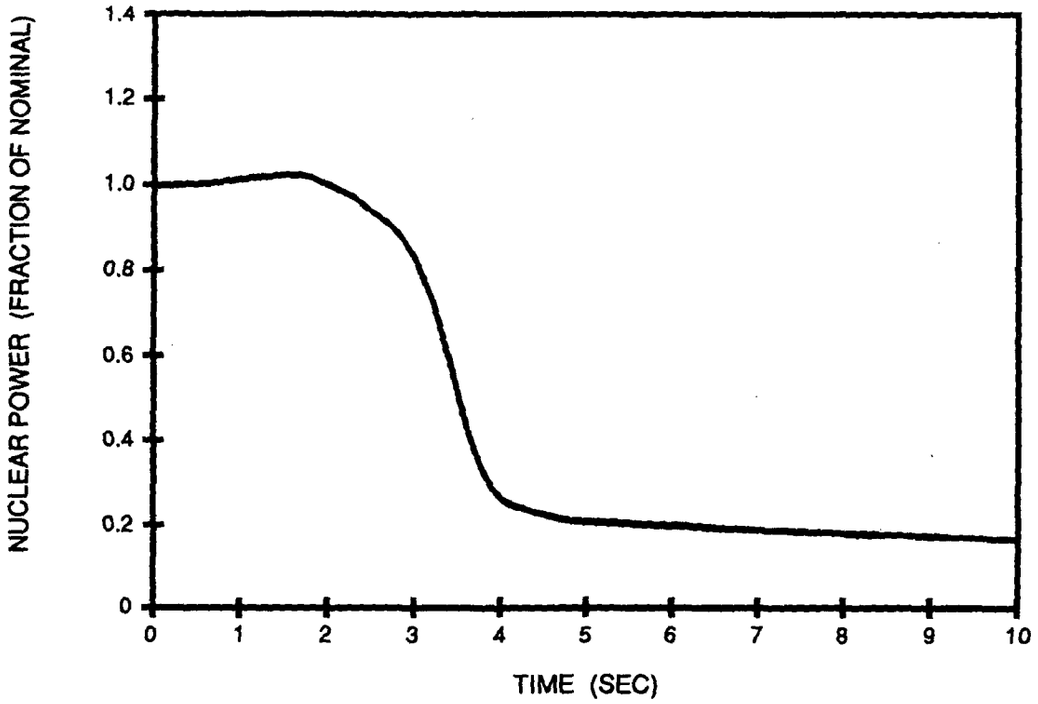

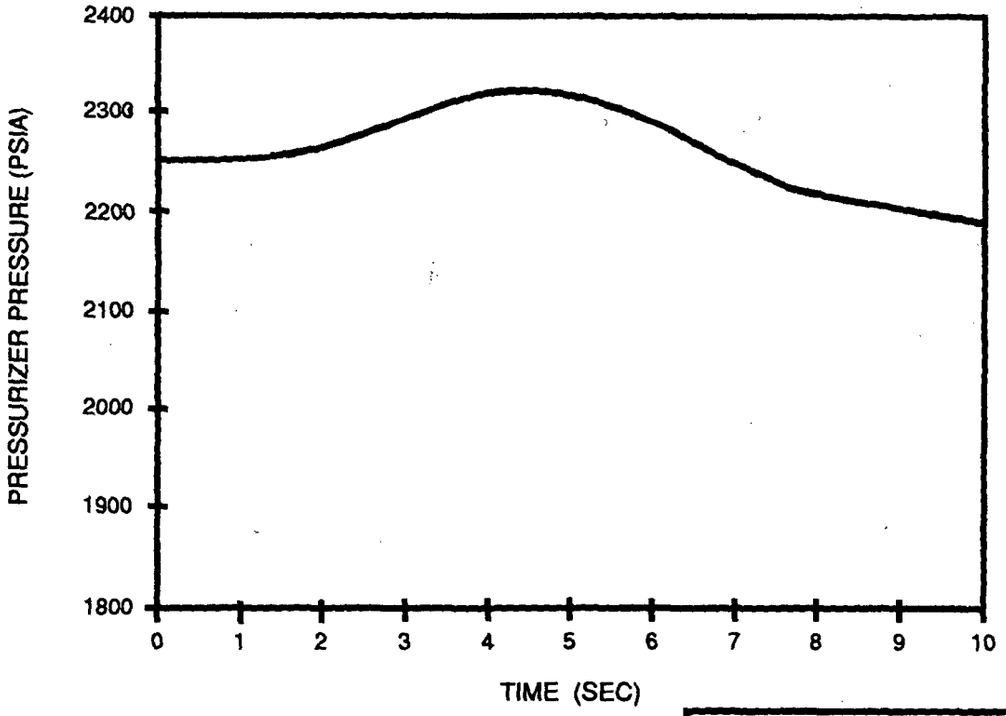

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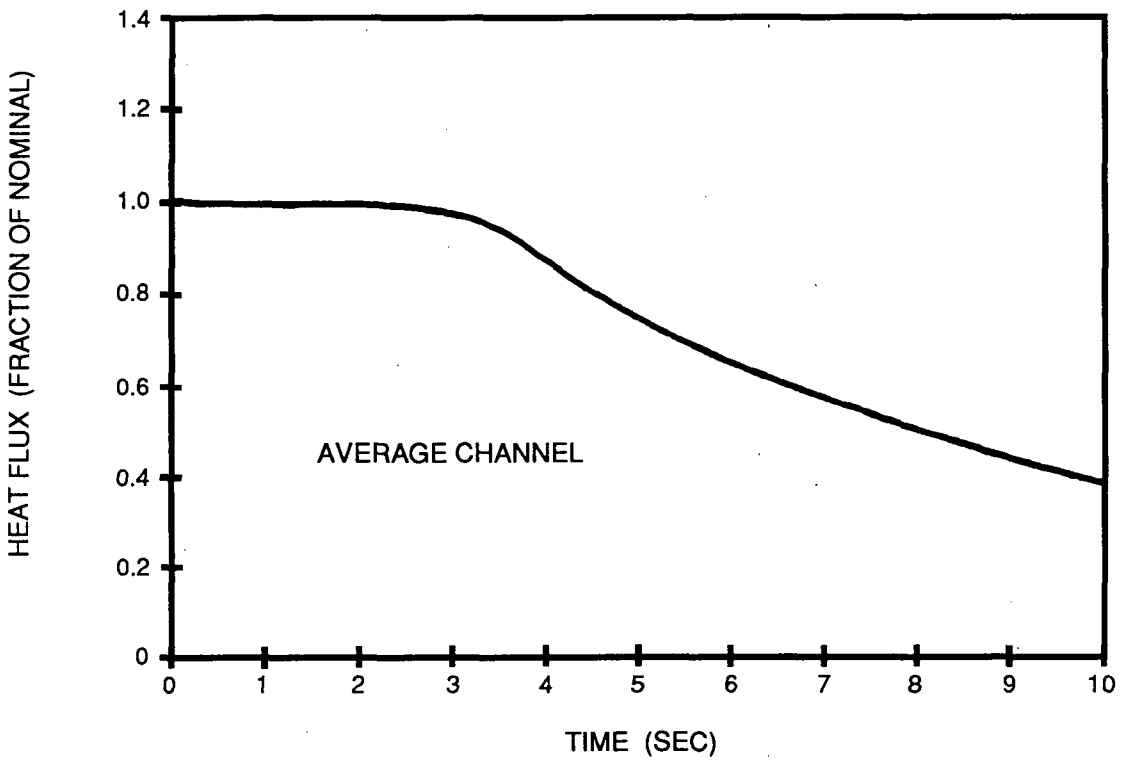
Figure 15.3-2
COMPLETE LOSS OF FLOW
NUCLEAR POWER AND
PRESSURIZER PRESSURE

JULY 1993



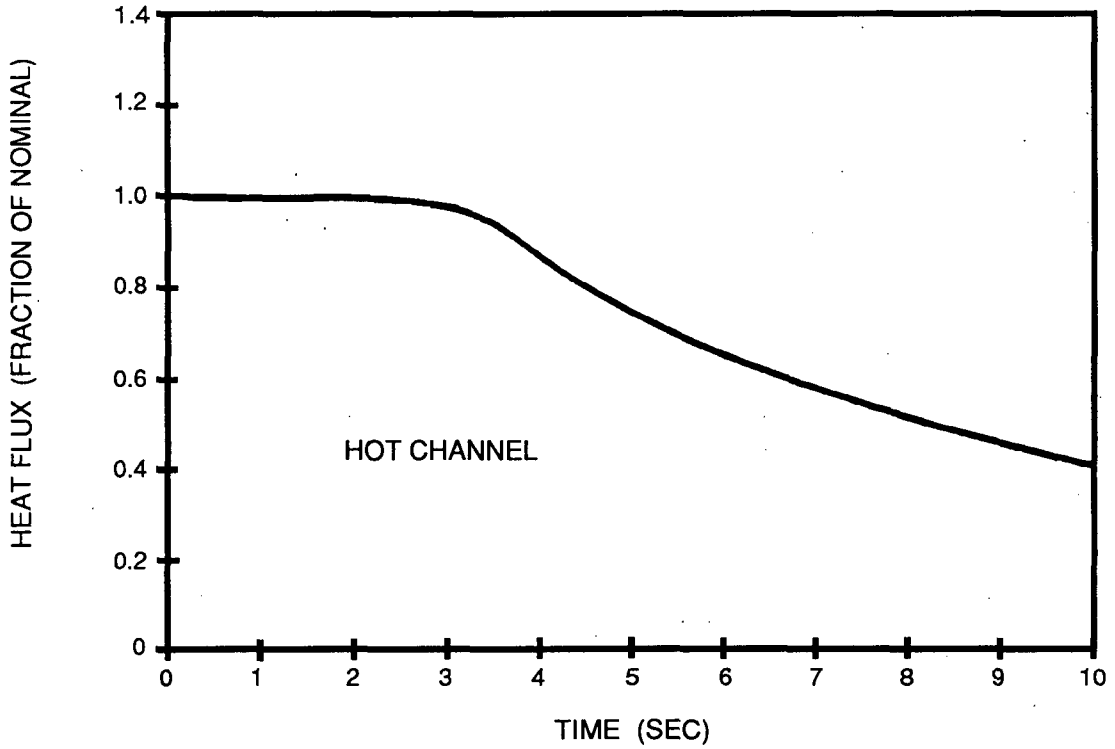


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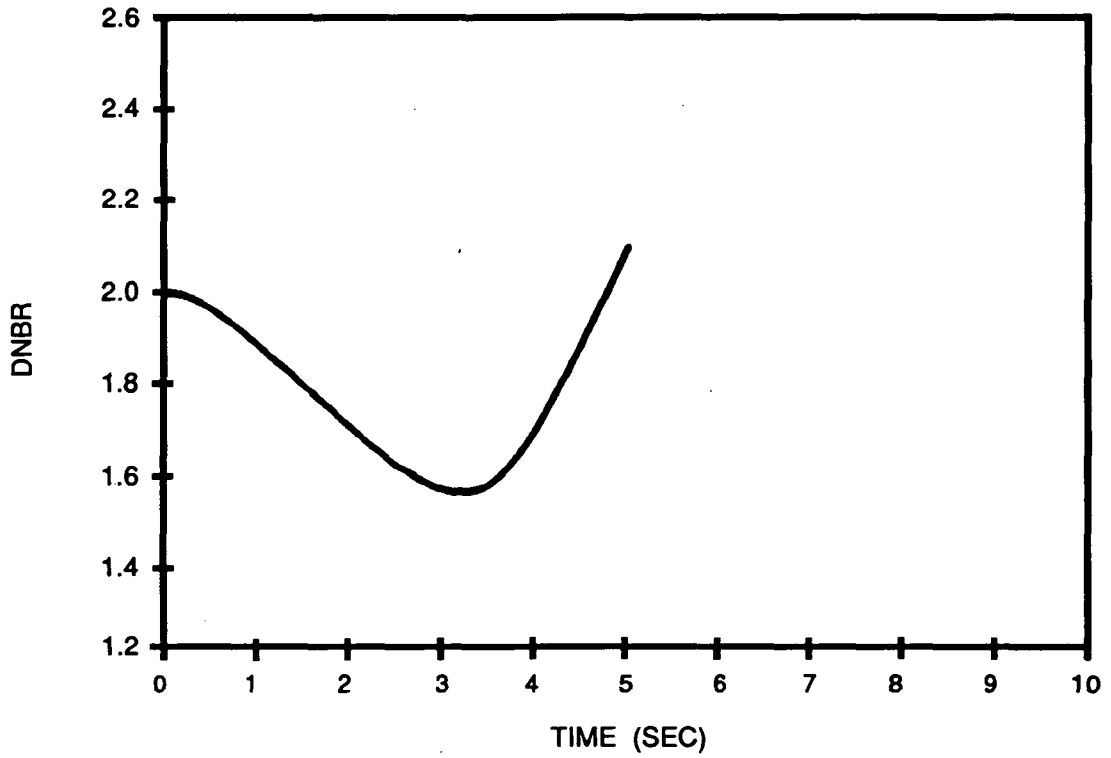
Figure 15.3-3
COMPLETE LOSS OF FLOW
AVERAGE CHANNEL HEAT FLUX AND
HOT CHANNEL HEAT FLUX

JULY 1993

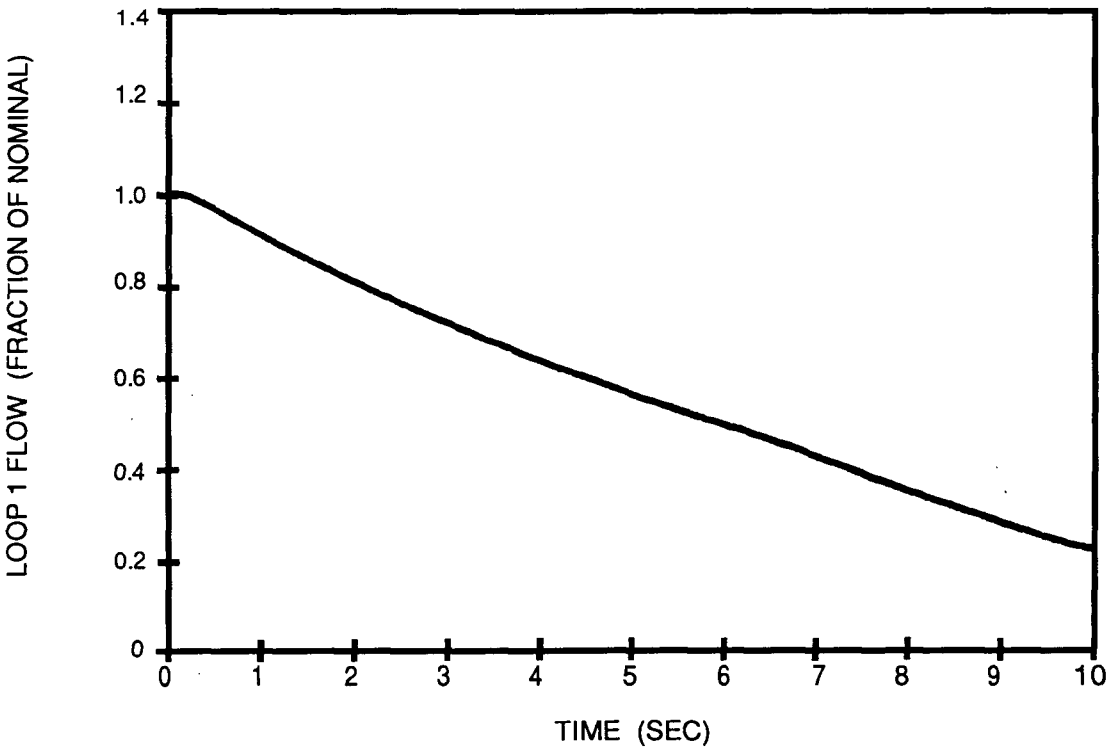


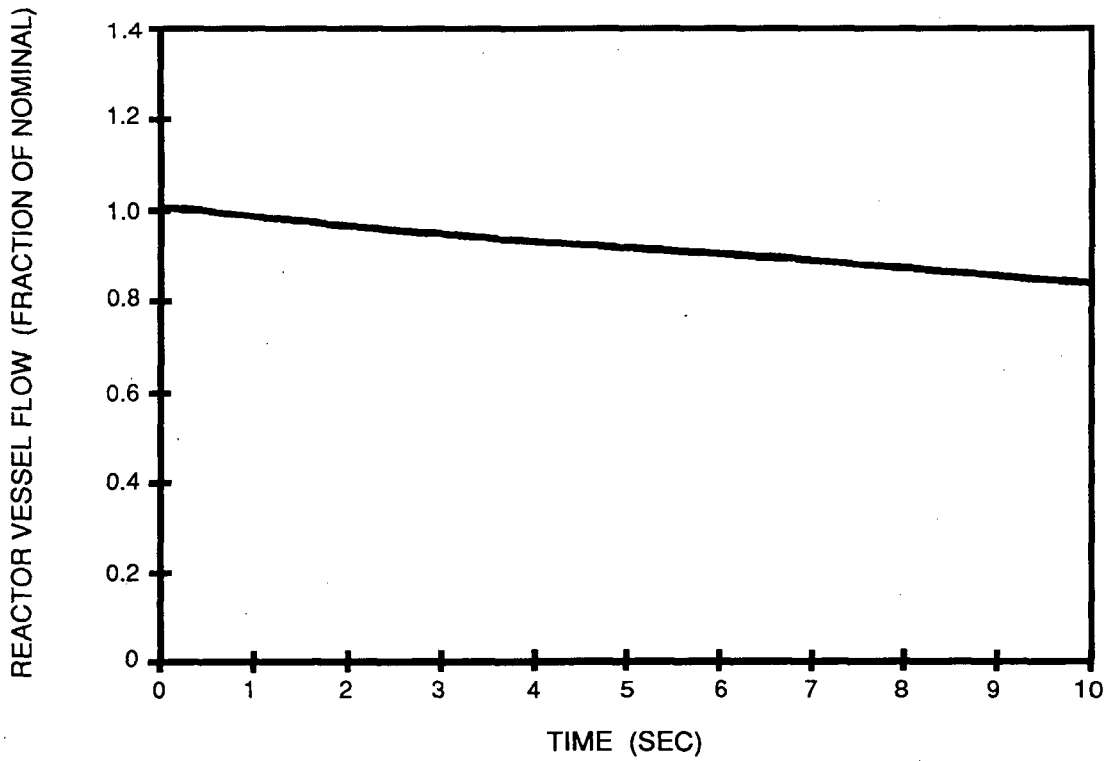
ZION STATION UFSAR

Figure 15.3-4

COMPLETE LOSS OF FLOW
DNBR VERSUS TIME

JULY 1993

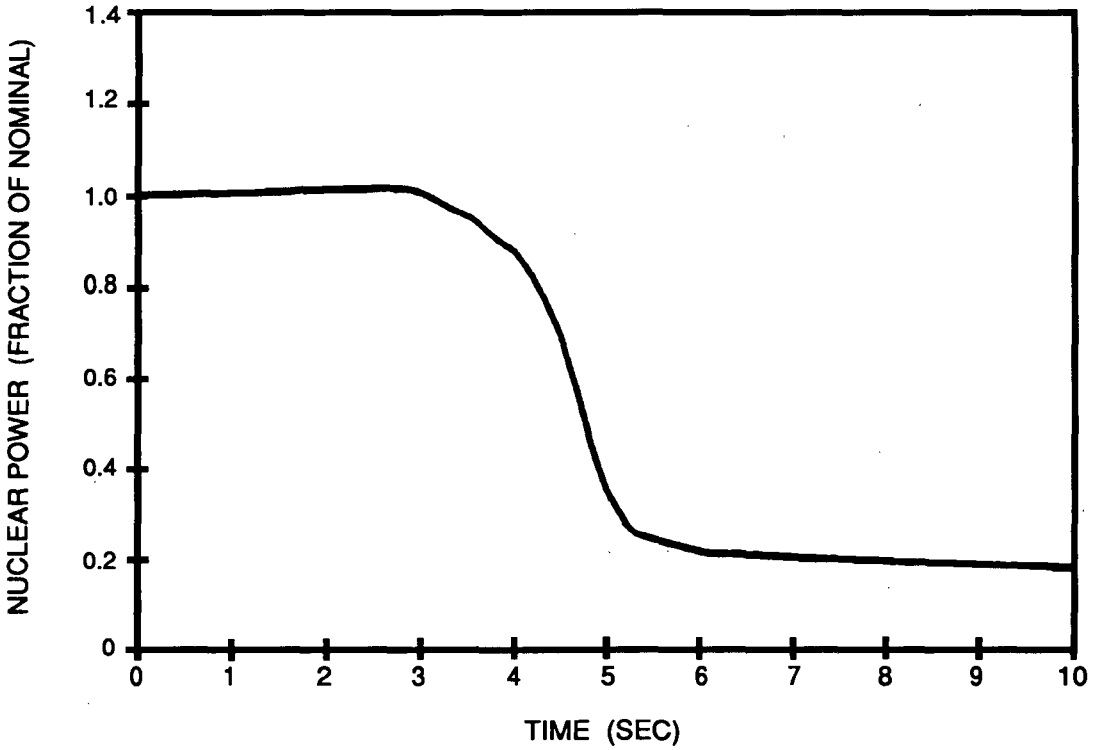


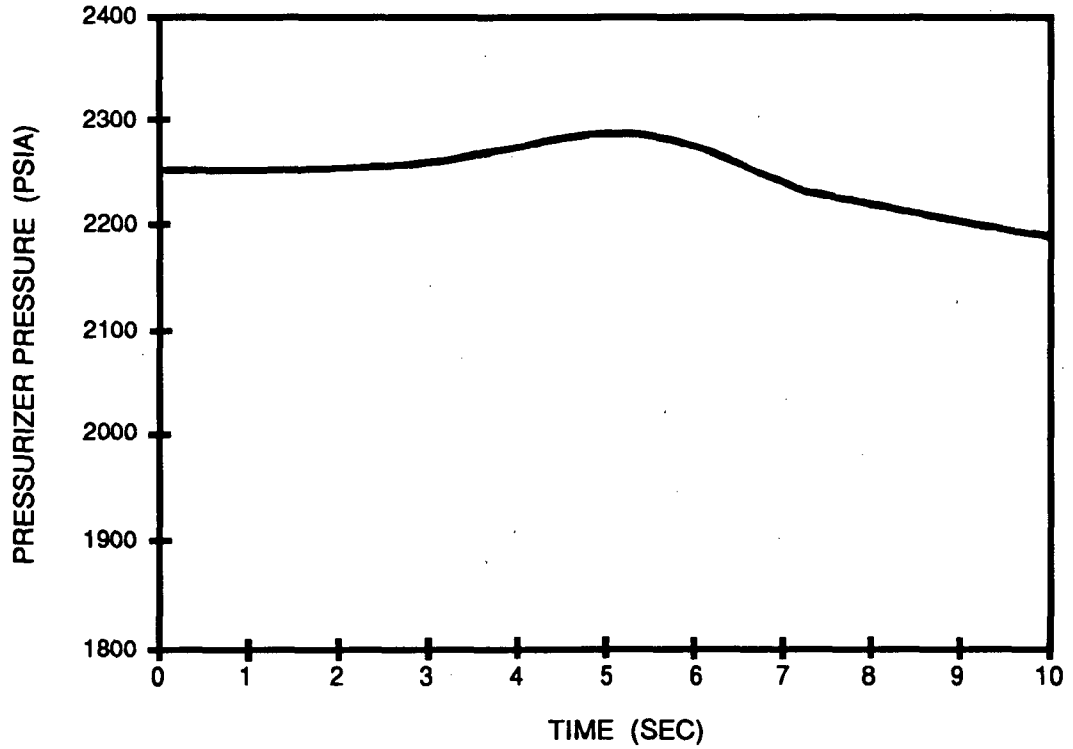


ZION STATION UFSAR

Figure 15.3-5
PARTIAL LOSS OF FLOW
CORE FLOW AND LOOP FLOW

JULY 1993

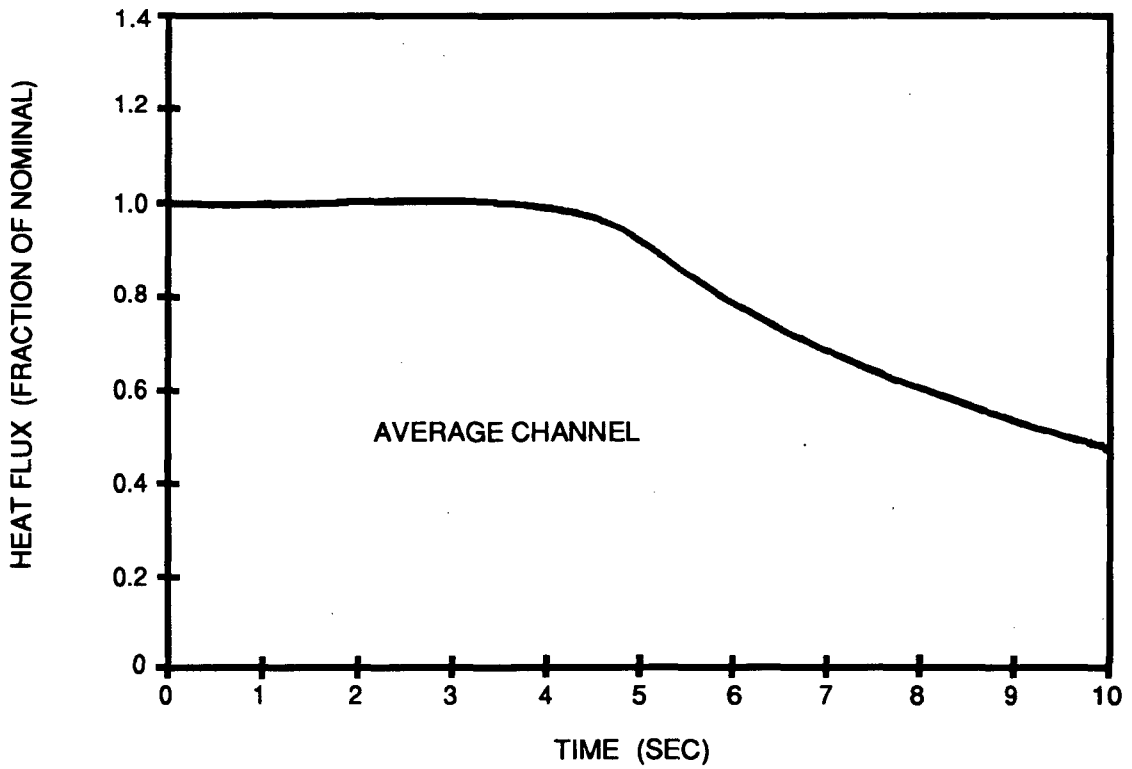




ZION STATION UFSAR

Figure 15.3-6
PARTIAL LOSS OF FLOW
NUCLEAR POWER AND
PRESSURIZER PRESSURE

JULY 1993



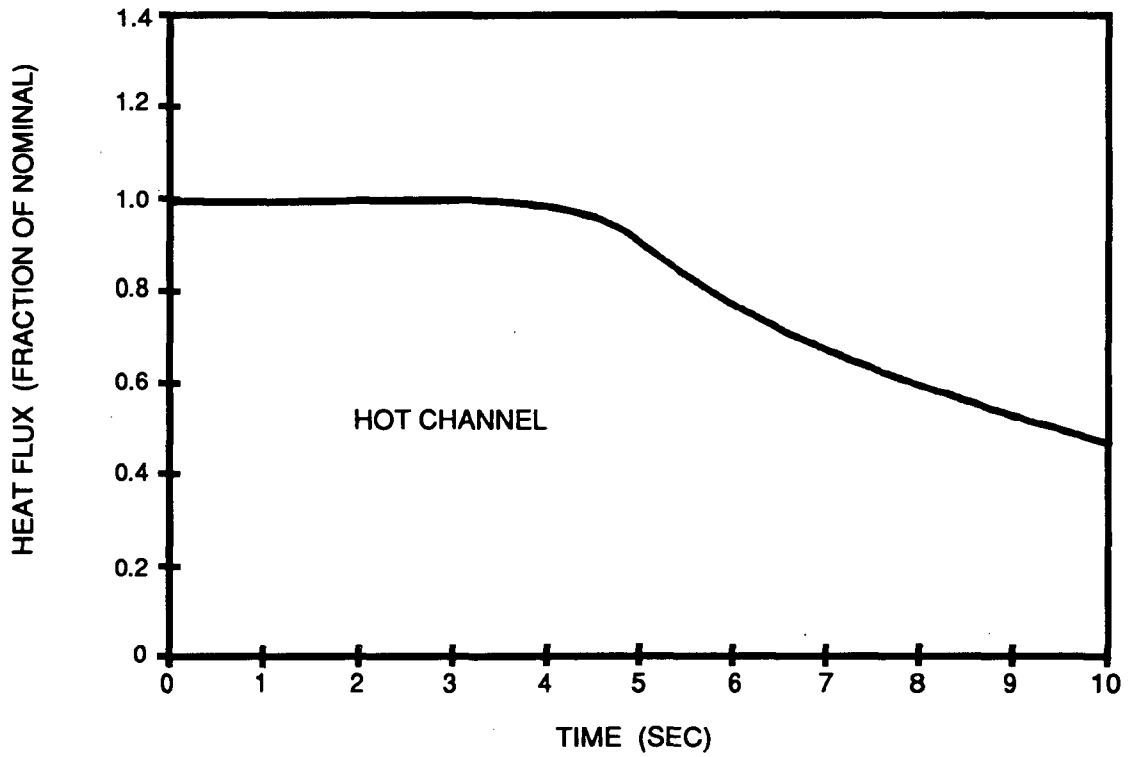


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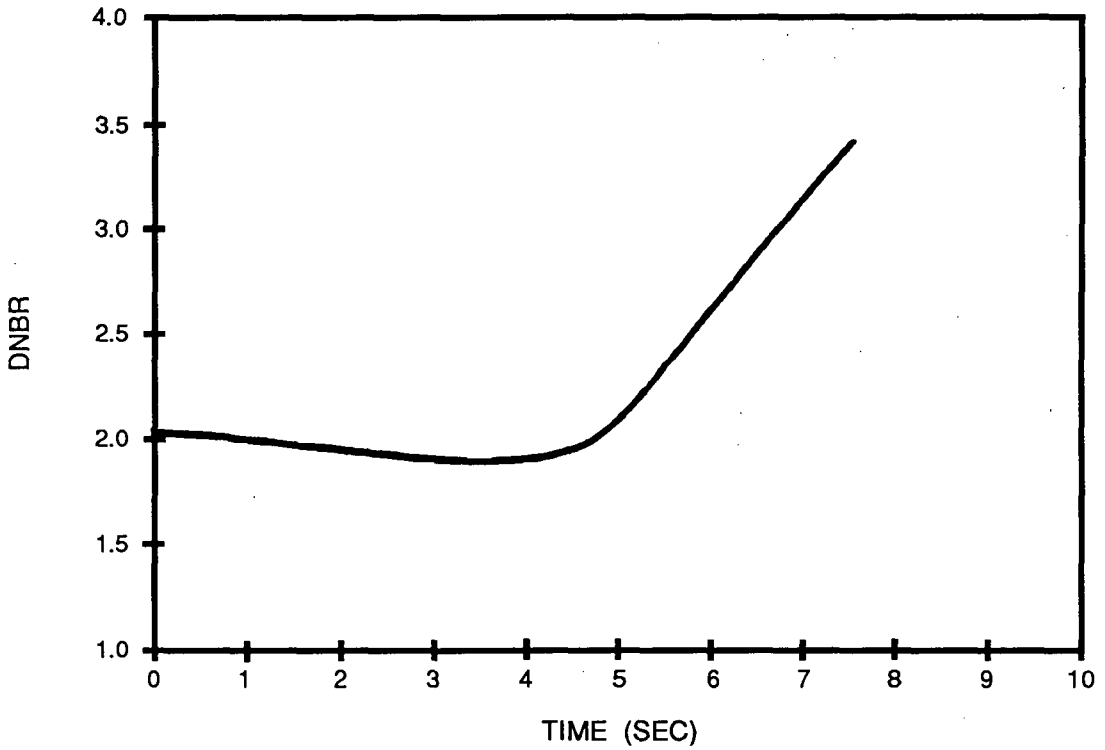
Figure 15.3-7
PARTIAL LOSS OF FLOW
AVERAGE CHANNEL HEAT FLUX AND
HOT CHANNEL HEAT FLUX

JULY 1993

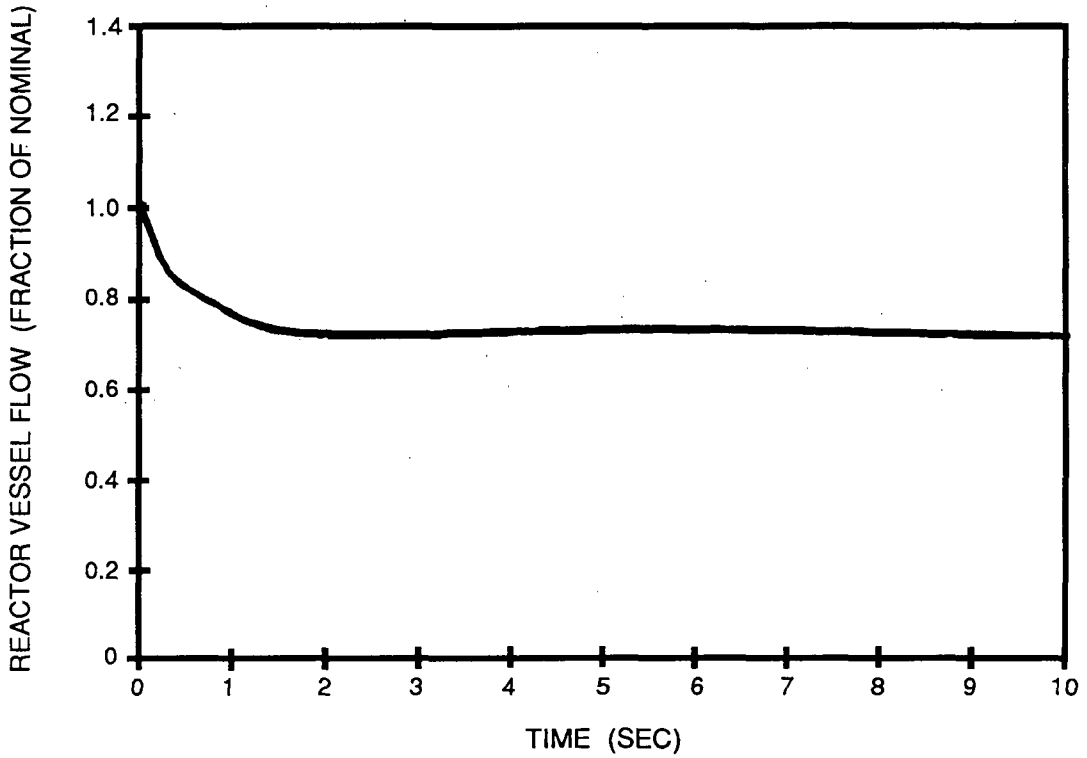


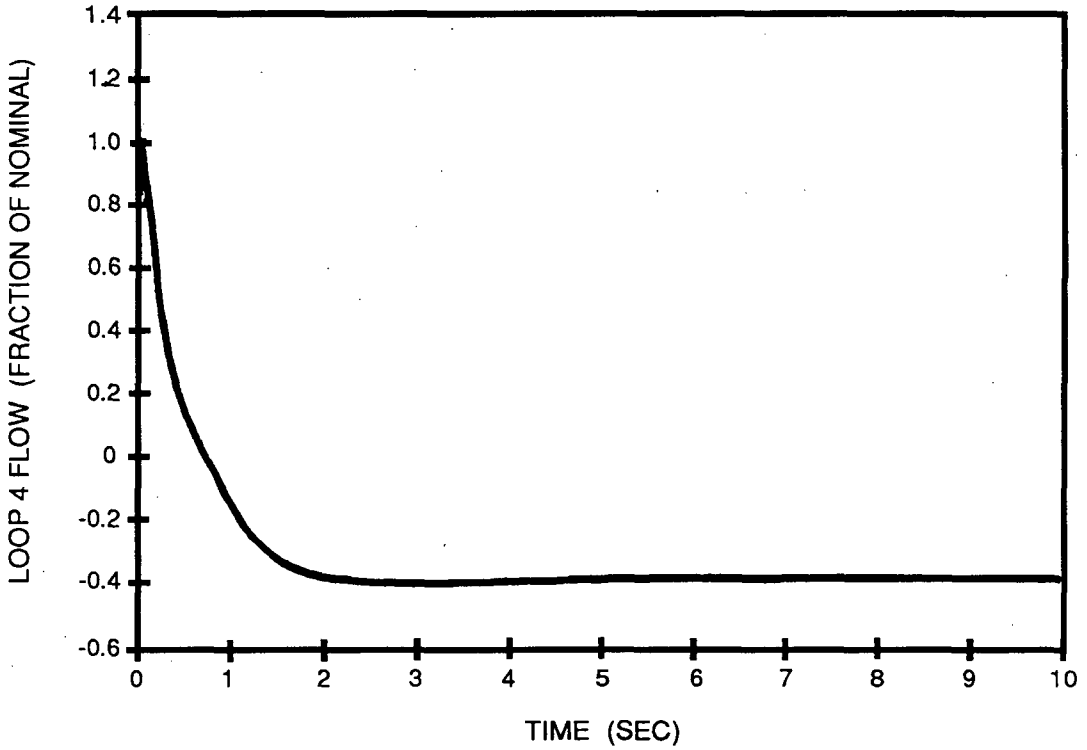
ZION STATION UFSAR

Figure 15.3-8

PARTIAL LOSS OF FLOW
DNBR VERSUS TIME

JULY 1993

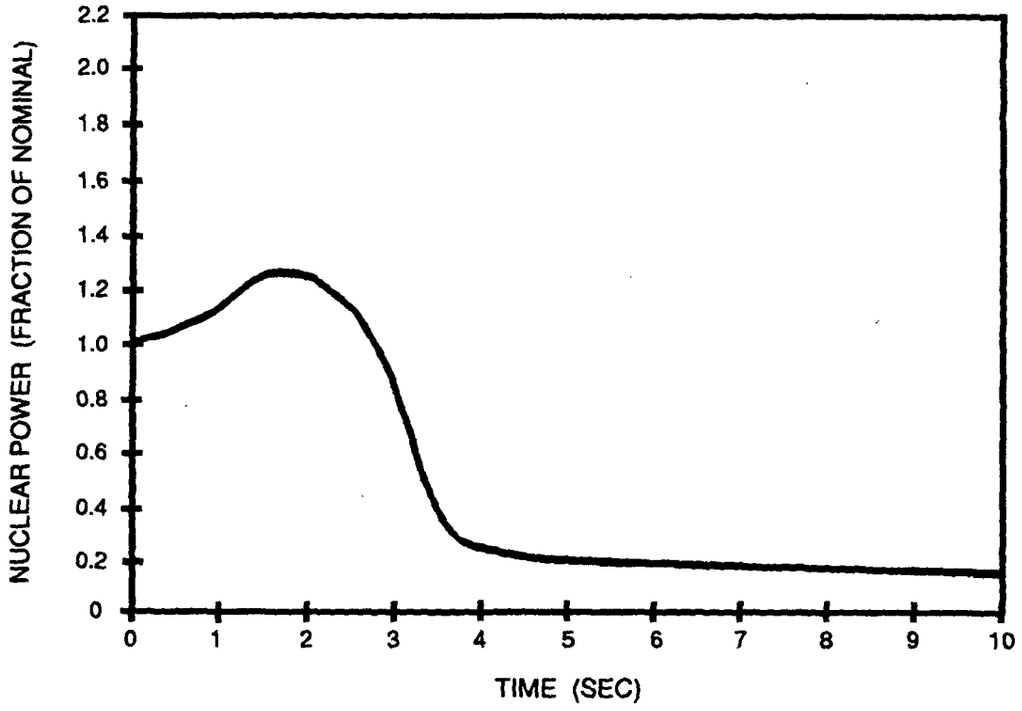


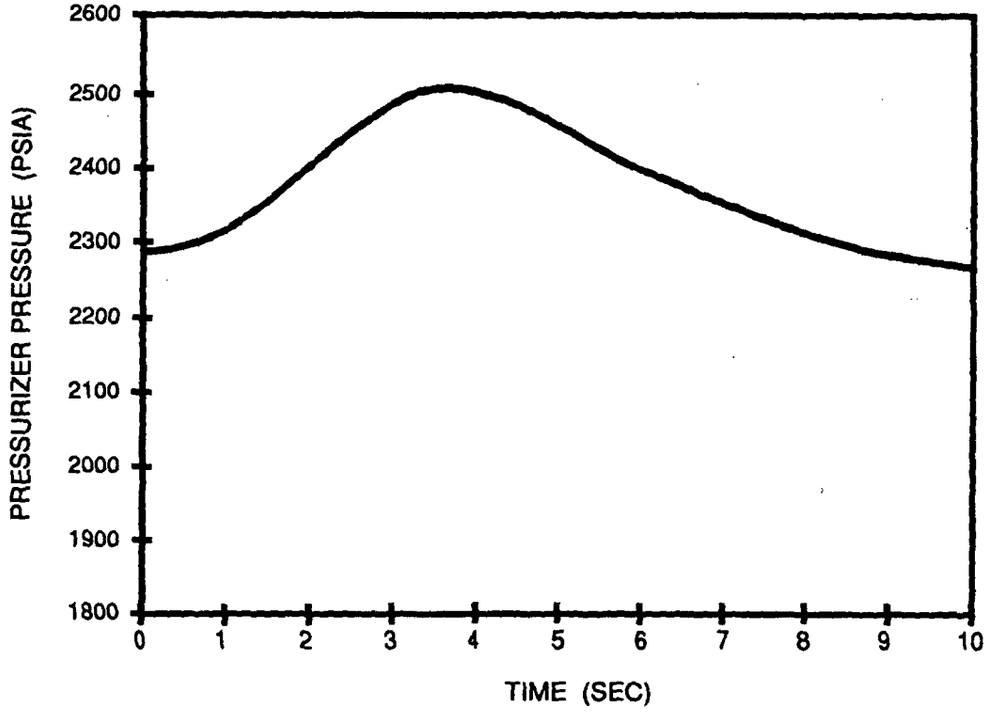


ZION STATION UFSAR

Figure 15.3-9
LOCKED ROTOR
CORE FLOW AND
FAULTED LOOP FLOW

JULY 1993

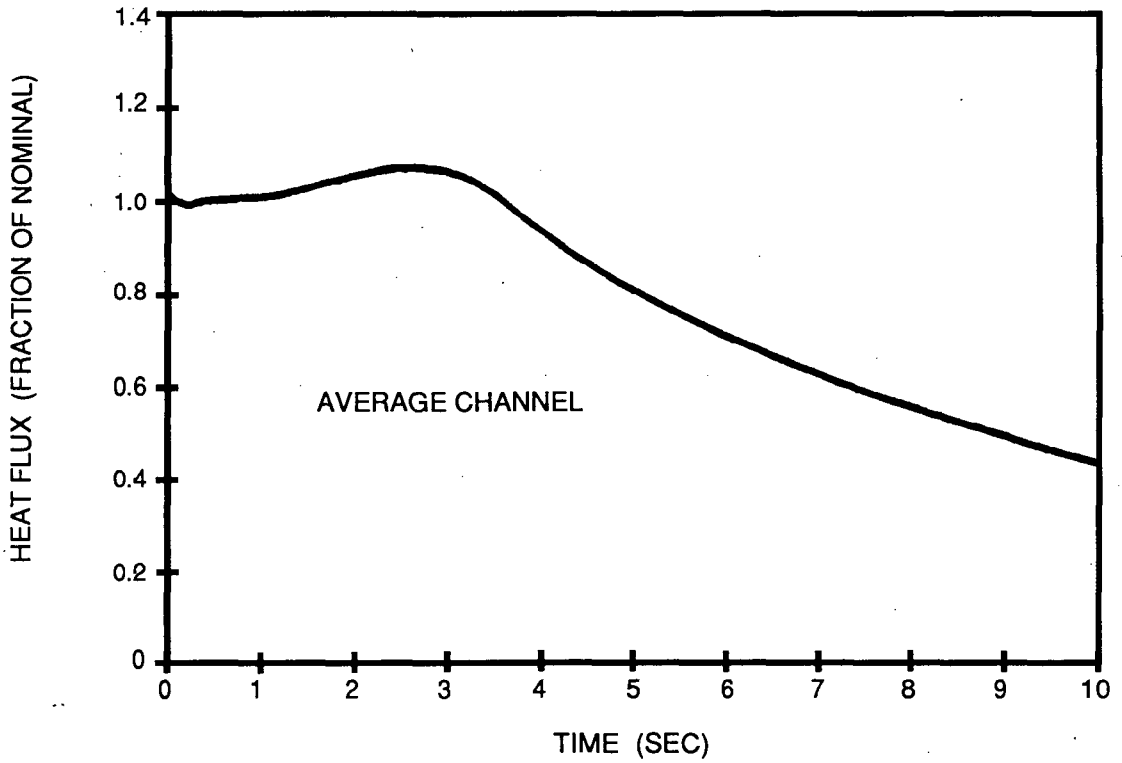


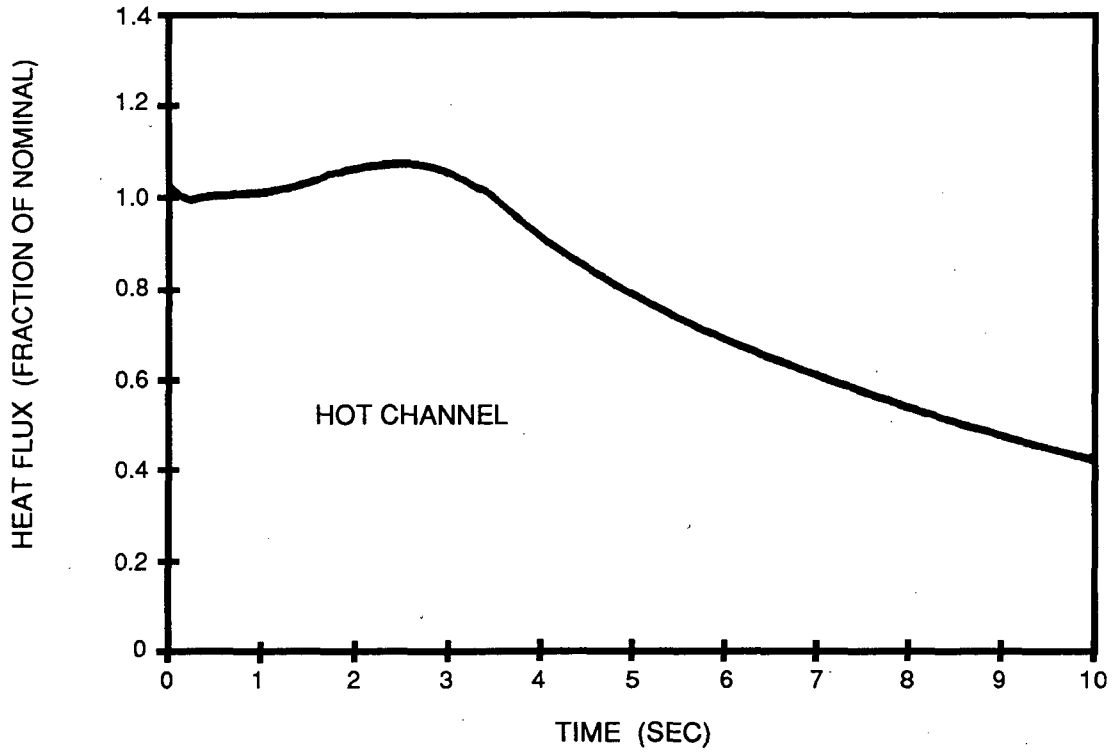


ZION STATION UFSAR

Figure 15.3-10
LOCKED ROTOR
NUCLEAR POWER AND
RCS PRESSURE

JULY 1993

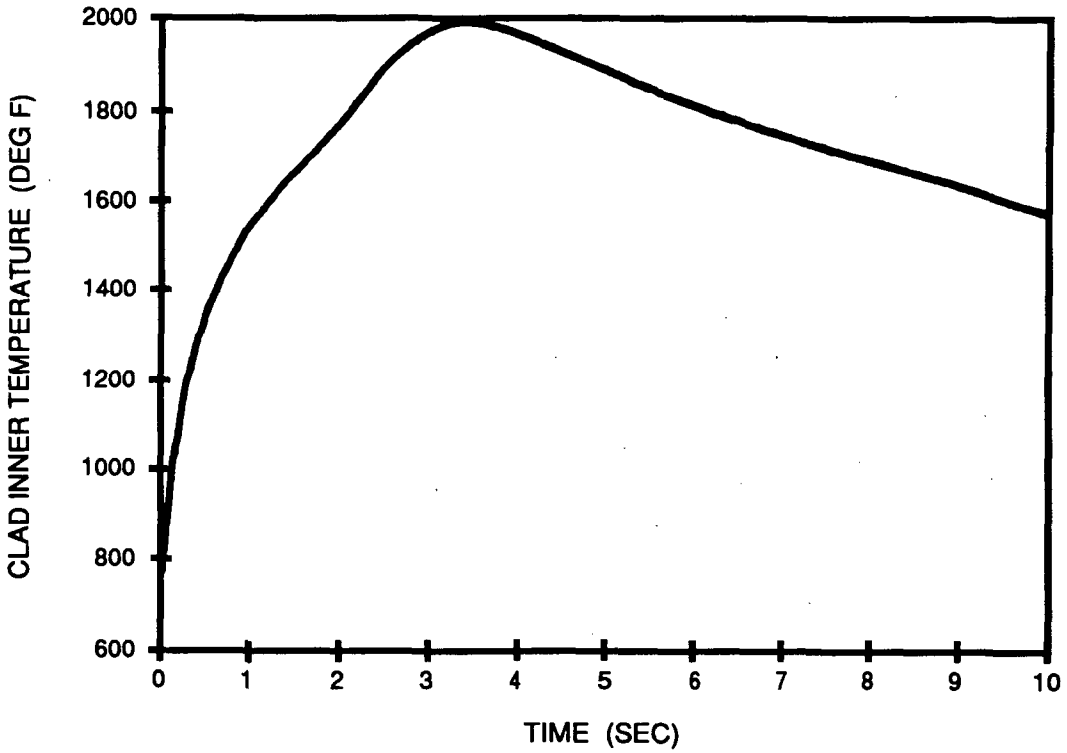




ZION STATION UFSAR

Figure 15.3-11
LOCKED ROTOR
AVERAGE CHANNEL AND
HOT CHANNEL HEAT FLUX

JULY 1993



ZION STATION UFSAR

Figure 15.3-12

LOCKED ROTOR
CLAD INNER TEMPERATURE

JULY 1993

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system (RCS). Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussion of the following incidents are presented:

1. Uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal from a subcritical condition;

2. Uncontrolled RCCA withdrawal at power;

3. RCCA misalignment;

4. Startup of an inactive reactor coolant loop;

5. Chemical and Volume Control System (CVCS) malfunction; and

6. Spectrum of RCCA ejection accidents.

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15.4.1 Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition

15.4.1.1 Incident Description

A RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of RCCAs resulting in a power excursion. This event is classified as ANS Condition II (moderate frequency) as discussed in Section 15.0.1. While the occurrence of a transient of this type is highly unlikely, such a transient could be caused by a malfunction of the Reactor Control or Control Rod Drive (CRD) Systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 15.4.2.

In bringing the reactor from a shutdown condition to a low power level during startup, reactivity is added at a prescribed and controlled rate by RCCA withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is by means of RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration. The control rod drive mechanisms (CRDMs) are wired into preselected banks, and these bank configurations are not altered during the core life. The assemblies are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The RCCA drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed by assuming the simultaneous withdrawal of the combination of the two banks of 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 fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup incident since it limits the power to a tolerable level prior to external protective action. After the initial power burst, the neutron flux is momentarily reduced and then, if the incident is not terminated by a reactor trip, the neutron flux increases again, but at a much slower rate.

Should a continuous RCCA bank withdrawal be initiated, the transient will be terminated by the following reactor trip functions:

1. Source range neutron flux level trip is actuated when either of two independent source range channels indicates a flux level above 10^5 cps. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above the source range block level. It is automatically reinstated when both intermediate range channels indicate a flux level below the source range block level;

2. Intermediate range neutron flux level trip is actuated when either of two independent intermediate range channels indicates a flux level corresponding to 25% power. This trip function may be manually bypassed when two of four power range channels are reading above 10% of full-power flux and is automatically reinstated when three of four channels indicate a flux level below this value;

3. Power range neutron flux level trip (low setting) is actuated when two out of four power range channels indicate a flux level above 25% of full-power flux. This trip function may be manually bypassed when two of four power range channels indicate a flux level above 10% of full power flux and is automatically reinstated when three of the four channels indicate a flux level below this value;

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

5. High neutron flux rate trip is actuated when the rate of change in power exceeds the positive or negative setpoint in two out of four channels. This is always active.

In addition, control rod stops on either one out of two high intermediate range flux level or one out of four high power range flux level serve to discontinue rod withdrawal and prevent the need to actuate either the intermediate range flux level trip or the power range flux level trip, respectively.

Termination of the startup incident by the protection channels identified above prevents core damage. In addition, the reactor trip from pressurizer high pressure serves as a backup to terminate the incident before an overpressure condition could occur.

15.4.1.2 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation, then an average core heat transfer calculation, and, finally, the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear power calculation is performed using the spatial neutron kinetics digital computer code TWINKLE (Reference 8). TWINKLE determines the average power generation with time including the various core feedback effects, e.g., Doppler and moderator feedback. The average core heat flux is generated by performing a fuel rod transient heat transfer calculation with FACTRAN (Reference 9). FACTRAN also calculates the fuel, cladding, and coolant temperatures. The average heat flux is then used in the THINC code (Reference 10) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Section 15.0.3. In order to give conservative results for a startup accident, the following assumptions are made:

1. Since the magnitude of the neutron flux peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservatively low value is used for the startup incident (see Section 15.0.4 and Figure 15.0-3);

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservatively high value, given in Figure 15.0-2, is used in the analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be at hot zero power (547°F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel to water heat transfer, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. The initial multiplication factor (K_o) is assumed to be closely approaching 1.0 since this results in the maximum neutron flux peak;

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrumentation and setpoint errors, as well as delays for trip signal actuation and RCCA release, are taken into account. A 10% increase has been assumed for the power

range flux trip setpoint, raising it from the nominal value of 25% to a value of 35% in addition to taking no credit for the source and intermediate range protection. The rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition to the above, the rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth control rod assembly is stuck in its fully withdrawn position;

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 in/min);

6. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux;

7. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, is assumed in the analysis; and

8. Only two reactor coolant pumps (RCPs) are assumed to be in operation. This is conservative for the DNB calculation.

No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

15.4.1.3 Results

Figures 15.4-1 and 15.4-2 show the transient behavior for the uncontrolled RCCA bank withdrawal incident with the accident terminated by reactor trip at 35% nominal power. The reactivity insertion rate used in the analysis is greater than that for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region.

The neutron flux overshoots the full power nominal value but 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 departure from nucleate boiling (DNB) considerations, is also shown in Figure 15.4-1. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value. Figure 15.4-2 shows the response of the hot spot fuel average temperature and the hot spot clad temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR remains above the limiting value at all times.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.4 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of this transient.

15.4.2 Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

15.4.2.1 Incident Description

An uncontrolled RCCA bank withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator lags behind the power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to prevent the possibility of damage to the cladding, the Reactor Protection System (RPS) is designed to terminate any such transient before the DNBR falls below the safety analysis limit value.

This event is classified as ANS Condition II (an incident of moderate frequency) as discussed in Section 15.0.1.

The automatic features of the RPS which prevent core damage in a RCCA bank withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip on high neutron flux if two out of four channels exceed an overpower setpoint;

2. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature and pressure to protect against the DNB;

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with coolant average temperature and with axial power distribution to ensure the allowable fuel power rating is not exceeded;

4. A high pressure reactor trip, actuated from any two out of four pressure channels, is set at 2385 psig. This set pressure is less than the set pressure of 2485 psig for the pressurizer safety valves; and

5. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is set at 92%.

In addition to the above listed reactor trips, RCCA withdrawal blocks are initiated on one out of four high neutron flux, two out of four overpower ΔT, or a two out of four overtemperature ΔT.

The manner in which the combination of overpower and overtemperature ΔT trips provide protection against a DNBR of less than the safety analysis limit value over the full

range of RCS conditions is illustrated in Figure 15.0-1. This figure represents the allowable conditions of reactor coolant loop average temperature and ΔT with the design power distribution and flow as a function of primary coolant pressure.

The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors, so that under nominal conditions, trips would occur well within the area bounded by these lines.

The utility of the diagram just described is that the limit imposed by any given DNBR can be represented as a line on this coordinate system. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is completely bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); and overpower and overtemperature ΔT (variable setpoints). These trips are designed to prevent the calculated DNBR from exceeding the DNBR limit for the correlation being applied. See Section 4.3.4.1.3.

The purpose of this analysis is to demonstrate the manner in which the above protective systems function for various reactivity insertion rates from different initial conditions. Reactivity insertion rates and initial conditions govern which protective function occurs first.

15.4.2.2 Method of Analysis

This transient is analyzed with the LOFTRAN (Reference 11) code. This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Revised Thermal Design Procedure (RTDP) as described in Reference 12. Plant characteristics and initial conditions are described in Section 15.0.3.

The following assumptions are made:

1. Initial reactor power, pressure and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 12;

2. Reactivity Coefficients - two cases are analyzed:

   a. Minimum Reactivity Feedback - A negative moderator density coefficient (i.e., a positive moderator temperature coefficient of +7 pcm/°F) is assumed corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed; and

   b. Maximum Reactivity Feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed consistent with end of life conditions.

3. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors. The delays for trip signal actuation are assumed to be at their maximum values;

4. The rate of negative reactivity insertion corresponding to the trip of the RCCAs is based on the assumption that the control rod assembly of highest worth is stuck in its fully withdrawn position; and

5. The maximum reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two rod control banks having the greatest combined worth at maximum speed.

No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

15.4.2.3 Results

Figures 15.4-3, 15.4-4, and 15.4-5 show the response of neutron power, heat, pressurizer pressure, water volume, average coolant temperature, and DNBR to a rapid RCCA withdrawal incident 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. A large margin to DNB is maintained.

The response of neutron power, heat flux, pressurizer pressure, water volume, average coolant temperature, and DNBR for a slow withdrawal from full power is shown in Figures 15.4-6 through 15.4-8. Reactor trip on power range high neutron flux occurs after a longer period. The rise in temperature and pressure is consequently larger than for the rapid RCCA withdrawal. The minimum DNBR is never less than the safety analysis limit value.

Figure 15.4-9 shows the minimum DNBR as a function of credible reactivity insertion rate from initial full-power operation for both the minimum and maximum reactivity feedback cases. It can be seen that two reactor trip channels provide protection over the whole range of possible 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 value.

Figures 15.4-11 and 15.4-12 show the minimum DNBR as a function of credible reactivity insertion rate for RCCA withdrawal incidents starting at 60% and 10% power, respectively. The results are similar to the 100% power case, except that as the initial power is decreased, the range over which the overtemperature ΔT trip operates is increased. In neither case does the DNBR fall below the safety analysis limit.

Since DNB does not 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 15.4-2. With the reactor tripped, the plant eventually returns to a stable condition. The plan may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.4 Conclusions

In the unlikely event of a RCCA bank withdrawal incident during power operation, the core and RCS are not adversely affected since the minimum value of the DNBR is always greater than the safety analysis limit for all possible reactivity insertion rates. Adequate protection is provided by the high neutron flux and the overtemperature ΔT trips.

15.4.3 Rod Cluster Control Assembly Misalignment

15.4.3.1 Incident Description

RCCA misalignment incidents include one or more dropped RCCAs within the same group, a dropped RCCA group, and statically misaligned RCCAs.

Each RCCA has a rod position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by rod bottom lights. Bank (demand) position is also indicated. The full-

length assemblies are always moved in preselected banks and the banks are always moved in the same preselected sequence.

Dropped RCCAs or a dropped RCCA bank are detected by:

1. sudden drop in the core power level;
2. asymmetric power distribution as seen on excore neutron detectors or core exit thermocouples;
3. rod bottom light(s); and/or
4. rod deviation alarm.

Misaligned RCCAs are detected by:

1. asymmetric power distribution as seen on excore neutron detectors or core exit thermocouples; and/or
2. rod deviation alarm.

The resolution of the rod position indicator channel is ±5% of span (the span is 12 feet). Deviation of any assembly from its bank by twice this distance (10% of span, or 14.4 inches) will not cause power distributions worse than the design limits. The rod deviation alarm alerts the operator to rod deviation greater than or equal to 12 steps (7.5 inches).

If one or more rod position indicator channels are out of service, Technical Specification surveillance requirements are followed to assure the alignment of the nonindicated assemblies. The surveillance requirements specify that for operation between 50% and 100% of rated power, the position of the control rod(s) associated with the inoperable rod position indicator channel(s) be checked indirectly by excore detectors and/or incore thermocouples and/or incore detectors every shift, or after any rod motion of the nonindicating rod(s) exceeding 24 steps, whichever occurs first.

15.4.3.2 Method of Analysis

15.4.3.2.1 One or More Dropped RCCAs Within the Same Group

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN computer code (Reference 11). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

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 analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC computer code (Reference 10). The

transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 13. Note that the analysis does not take credit for the negative flux rate reactor trip.

15.4.3.2.2. Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in Reference 13, assumptions made for the dropped RCCAs analysis provide a bounding analysis for the dropped RCCA bank.

15.4.3.2.3 Statically Misaligned RCCA

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then used as input to the THINC code to calculate the DNBR. The analysis examines the cases with the reactor initially at full power with the worst rod fully inserted and the worst rod withdrawn from Bank D which is inserted at the insertion limit. The allowance for increased peaking factors at reduced power is also analyzed but is generally less limiting. The analysis is performed at the worst time in life. Generally, this occurs early in the cycle since it results in the maximum value of moderator temperature coefficient (most positive). This assumption maximizes the power rise and minimizes the tendency of large moderator temperature coefficient (most negative) to flatten the power distribution.

15.4.3.3 Results

15.4.3.3.1 One or More Dropped RCCAs Within the Same Group

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 rapidly decreasing. Either reactivity feedback or control bank withdrawal will reestablish power.

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. Figure 15.4-13 shows a transient response to a dropped RCCA (or RCCAs) in automatic control for a typical plant.

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

In all cases, the minimum DNBR remains above the safety analysis limit value. Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 12.

15.4.3.3.2 Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of more than 800 pcm. The core is not adversely affected during this period since power is rapidly decreasing. The transient will proceed as described in Section 15.4.3.3.1; however, the return to power will be less due to the greater worth of the entire bank. The power transient will also be symmetric due to the location of the rods in the bank. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 minutes following the incident.

15.4.3.3.3 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 safety analysis limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control Bank D must then be chosen to be above that position and will usually be 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. Uncertainties in the initial conditions were included in the DNB evaluation as described in Reference 12.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for the fully withdrawn analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the Bank D case discussed above assuming insertion limits on the other banks equivalent to a Bank D full-in insertion limit.

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. Uncertainties in the initial conditions were included in the DNB evaluation as described in Reference 12.

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 Technical Specifications and operating instructions.

15.4.3.4 Conclusions

For all cases of dropped RCCAs or dropped banks, the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For all cases of any RCCA inserted, or Bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

15.4.4 Startup of an Inactive Reactor Coolant Loop

Currently, three-loop operation is not allowed in Technical Specification Mode 1 or Mode 2; however, the three-loop analysis is discussed for completeness and to show its impact on nonpower transients.

The plant can be operated with an inactive loop in either of two configurations. First, if the loop stop valves in the inactive loop are in the normal fully open position, there is reverse flow through the inactive loop. Second, with the loop stop valves of the inactive loop closed, there is no flow from the reactor vessel and active loops to the inactive loop and the plant operates much as if it were a plant with only three loops. The startup of an inactive reactor coolant loop will be considered for the case of the loop stop valves initially open and for the case of the loop stop valves initially closed, in Sections 15.4.4.1 and 15.4.4.2 respectively.

Following implementation of the revised Technical Specifications (Amd. 173/160), the loop stop isolation valve interlocks will no longer be tested. Administrative controls are used for return of an isolated reactor coolant loop to service.

15.4.4.1 Loop Stop Valves Initially Open

15.4.4.1.1 Incident Description

If the plant is operated with the pump in one loop turned off and with the stop valves open, there is reverse flow through the loop. The cold leg temperature in the inactive loop is identical to the cold leg temperatures of the active loops and to the reactor core inlet temperature. If the reactor is operated at power, there is a temperature drop across the steam generator in the inactive loop and, due to the reverse flow which exists, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures will require that the plant be brought to a load of less than 25% prior to starting a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. The starting of the idle reactor coolant pump without bringing the inactive loop hot leg temperature closer to the core inlet temperature would result in the injection of cold water into the core. The cold water would cause a rapid increase of reactivity and power.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as discussed in Section 15.0.1.

Should the startup of an inactive reactor coolant loop at an incorrect temperature occur, the transient will be terminated automatically by a reactor trip on low primary coolant 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.

15.4.4.1.2 Method of Analysis

This transient is analyzed by three digital computer codes. The LOFTRAN code (Reference 11) is used to calculate the loop and core flow, nuclear power, core pressure, and temperature transients following the startup of an idle pump. FACTRAN (Reference 9) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC code (Reference 10) is then used to calculate the DNBR during the transient based on system conditions (flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

The following assumptions are made:

1. The reactor is assumed to be initially at 77% of 3250 MWt with the secondary side of all four steam generators at the same pressure and reverse reactor coolant flow through the idle loop steam generator. 77% power is the maximum steady-state power level allowed with three loops in operation and the loop stop valves in the inactive loop open. The 77% includes 2% allowance for calibration and instrument errors.

The high initial power assumed is conservative since it 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 accelerates linearly to its nominal full-flow value;

3. A conservatively large negative moderator coefficient representative of end of life (EOL) is assumed;

4. A conservatively low Doppler power coefficient is assumed; and

5. The reactor trip is assumed to occur on low primary coolant flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84% of rated power which corresponds to the nominal setpoint plus 9% for nuclear instrumentation errors.

15.4.4.1.3 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4-14 through 15.4-16. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit 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 the increase in reactor coolant flow, and as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown in Figure 15.4-14.

The calculated sequence of events for this accident is shown in Table 15.4-3. The transient results illustrated in Figures 15.4-14 through 15.4-16 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.1.4 Conclusions

The transient results for the startup of an inactive reactor coolant loop with the loop stop valves initially open show that the DNB design basis is satisfied; thus, no fuel or clad damage is predicted.

15.4.4.2 Loop Stop Valves Initially Closed

15.4.4.2.1 Incident Description

If the stop valves in one loop are closed, the isolated section of the loop could cool down below the temperature of the active loops. Administrative

procedures require that the plant be brought to zero load, the temperature of the isolated loop brought to within 10°F of the highest temperature of the four loops, and the boron concentration of the isolated loop verified to be greater than or equal to the active loop boron concentration prior to opening the loop stop valves and returning the loop to service.

Interlocks ensure that the stop valves are not opened when an isolated loop has a lower temperature or a lower boron concentration than the core and active loops, thereby preventing an unacceptable transient. The interlocks

insure that flow from the isolated loop to the remainder of the RCS takes place through the relief line bypassing the cold leg stop valve for a period of about 105 minutes before the cold leg stop valve can be opened. The flow through the relief line is kept low (slightly over 100 gpm) so that the temperature and boron concentration in the isolated loop are brought to equilibrium with the remainder of the system at a relatively slow rate.

The Loop Stop Valve Interlock System of the isolated loop is not required to be operable when opening reactor coolant loop isolation valves, provided that: 1) the reactor is in the cold shutdown condition with sufficient boron concentration for 70°F operation; and 2) if the isolated loop has been drained and refilled, it has been borated to a boron concentration of 25% greater than that required for cold shutdown.

Interlocks are provided to:

1. Prevent opening of a hot leg loop stop valve unless the cold leg loop stop valve in the same loop is fully closed.

2. Prevent starting a reactor coolant pump unless:

   a. The cold leg loop stop valve in the same loop is fully closed and the bypass line valve is open; or
   b. Both the hot leg loop stop valve and cold leg loop stop valve are fully open.

3. Prevent opening of a cold leg loop stop valve unless:

   a. The cold leg temperature in the isolated loop is within 10°F of the highest cold leg temperature of the four loops.
   b. The hot leg temperature in the isolated loop is within 10°F of the highest hot leg temperature of the four loops.
   c. The following conditions have existed throughout the entire limiting interval (105 minutes) prior to opening of the cold leg stop valve:

      - The bypass line valve in the isolated loop is open.
      - The hot leg loop stop valve, in the isolated loop, is open.
      - A sufficiently high (100 gpm) flow rate exists in the relief line, bypassing the cold leg stop valve in the isolated loop.

The requirement that coolant temperatures in both the hot and cold legs of the isolated loop be within 10°F of the highest hot and cold leg temperatures, respectively, of any of the four loops ensures acceptable reactivity insertion rates following opening of the cold leg stop valve.

This is true for any period in core life, even assuming extremely conservative values for the moderator temperature coefficient.

The interlock on hot leg temperatures is provided as a backup for the interlock on the cold leg temperatures. Thus, the single failure criterion (as stated in IEEE 279) applies to the combination of the two interlocks and not each separately. The bypass line flow provides for temperature equalization in the isolated loop; bypass line valve opening, as stated above (and discussed below), is a necessary condition for opening of the cold leg stop valve. Therefore, temperature equalization prior to opening of the cold leg stop valve in an isolated loop is assured.

As stated above, three conditions must exist throughout the timing cycle. These are discussed, in order, below:

1. The bypass line valve must be open. Opening of the bypass line valve ensures that temperature and boron concentration equalization exists throughout the isolated loop.

   Valve position is derived from two independent limit switches, neither of which are operated from the same gear train. One limit switch is placed on the valve stem so that no single failure (including shaft or gear failure) can affect both limit switches. Both limit switches must indicate the valve is open for the "bypass line valve open" signal to exist.

2. The hot leg stop valve must be open. Opening of the hot leg stop valve, prior to opening of the cold leg valve, allows for temperature and boron concentration equalization of the isolated loop with the rest of the RCS via the flow through the relief line, as discussed below.

   In exactly the same way as for the bypass line valve, hot leg stop valve position is derived from two independent limit switches. Therefore, no single failure can negate both limit switches; both limit switches must indicate that the valve is open for a signal to exist.

3. Adequate flow must exist in the relief line. Flow through the relief line allows temperature and boron concentration equalization (along with the open hot leg stop valve as discussed above) of the isolated reactor coolant loop with the rest of the RCS.

   Relief line flow information is derived from two differential pressure switches in each line, one for each logic train. Signals confirming the three conditions listed above must be present in both logic trains before the cold leg stop valve can be opened.

It should be noted that flow through the relief line indicates:

a. The valves in the line are open; and
b. The reactor coolant pump in the isolated loop is running.

The three conditions discussed above must exist throughout the entire 105-minute timing cycle before the cold leg stop valve can be opened (if the temperature interlock requirements are satisfied). Interruption of any one of the three signals causes reset of the timer. That is, if any one of the three signals is interrupted, the entire timing cycle will begin again when all signals are present.

In addition to the above interlocks, administrative procedures prohibit return of an isolated reactor coolant loop to service unless the plant is in the hot shutdown condition with all shutdown rods withdrawn, or the plant is in the cold shutdown condition.

Administrative procedures specify the following to ensure that boron concentration is equalized:

1. Boron concentration in the isolated loop must be measured, logged, and verified as being greater than or equal to that in the rest of the RCS, on a periodic basis, during the entire time period that a loop is isolated.

2. Whenever startup of an isolated loop is initiated, the boron concentration in the isolated loop must be verified as being greater than or equal to that in the remainder of the RCS:

   a. immediately prior to opening of the hot leg stop valve; and
   b. immediately prior to opening of the cold leg stop valve.

3. During the time interval prior to opening of the cold leg stop valve, the count rate from the nuclear instrumentation must be logged every five minutes. Should the count rate increase by a factor of two over the initial count rate, the procedures require that the hot leg valve be reclosed.

Any reactivity insertion due to cold water addition from an isolated loop would be more severe at the hot shutdown condition since the value of the moderator temperature coefficient of reactivity is a decreasing function of temperature (i.e., is less negative at higher temperatures). Accordingly, the transient associated with return of an isolated loop to service has been analyzed for the plant in the hot shutdown condition.

15.4.4.2.2 Method of Analysis and Results

The startup of an inactive reactor coolant loop with the loop stop valves initially closed has been analyzed. The inadvertent starting of an idle

pump in an isolated loop results in dilution of the boron in the core if the boron concentration in the isolated loop is less than that in the core. Two dilution scenarios have been examined.

A full power situation was examined by determining the time available for operator intervention. The event was started assuming that the active portion of the RCS was at a critical boron concentration corresponding to beginning of life (BOL), hot full power, no xenon, Bank D at the insertion limits when the bypass valves were opened. The time for the RCS to change from that concentration to a boron concentration corresponding to critical at hot zero power, no xenon, N-1 rods inserted was calculated assuming a dilution flowrate from the previously isolated loop of 300 gpm. The active volume of the RCS was assumed to be at the full power nominal average temperature plus uncertainties while the inactive loop was conservatively assumed to be 20°F lower minus uncertainties.

Additionally, a hot shutdown condition scenario has been analyzed. The event was started assuming that the active portion of the RCS was at a BOL, hot zero power, no xenon, all rods out (1.3% shutdown) boron concentration when the bypass valves were opened. The time for the RCS to change from that concentration to a boron concentration corresponding to critical at hot zero power, no xenon, all rods out was calculated assuming a dilution flowrate from the previously isolated loop of 300 gpm. The active volume of the RCS was conservatively assumed to be at the no load temperature plus uncertainties while the inactive loop was assumed to be 20°F lower minus uncertainties.

For the full power condition, the initial reactivity insertion rate is calculated to be 2.07×10^{-5} Δk/sec, considerably less than the reactivity insertion rates considered in Section 15.4.2. There exists greater than 19 minutes until the shutdown margin is lost.

for the hot shutdown condition, the initial reactivity insertion rate is calculated to be 2.16×10^{-5} Δk/sec, considerably less than the reactivity insertion rate considered in Section 15.4.1. It takes more than 9 minutes after the beginning of the dilution before a critical boron concentration is reached assuming the plant is 1.3% shutdown at the beginning of the event. This is ample time for the operator to recognize a high count rate signal and terminate the dilution by turning off the RCP in the inactive loop or by borating to counteract the dilution.

15.4.4.2.3 Conclusions

With the loop stop valves initially closed, the redundant interlocks provided ensure that the temperature and boron concentration in an isolated loop are brought slowly to equilibrium with the remainder of the system. Should administrative procedures be violated and an attempt made to open stop valves when the isolated loop temperature or boron concentration is lower than that in the core, the reactivity addition rate is slow enough to allow the operator to take corrective action before shutdown margin is lost.

15.4.5 Chemical and Volume Control System Malfunction

15.4.5.1 Incident Description

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

The opening of the primary water makeup control valve provides the only supply of makeup water to the RCS which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve. In order for makeup water to be added to the RCS, at least one charging pump must also be running in addition to the primary water pumps.

The rate of addition of unborated water makeup to the RCS when it is not at pressure is limited by the capacity of the primary water supply pumps. The maximum addition rate in this case is 175 gpm with both primary water supply pumps running. The 175 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating while the other is on standby. With the RCS at pressure, the maximum delivery rate is less than the capability of the charging pumps. Assuming two centrifugal pumps are operating, the maximum delivery rate is 208 gpm. Normally, only one charging pump is in operation.

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 flow rates of boric acid and primary grade water as controlled from and indicated on the control board. In order to dilute, two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode; second, the start switch must be turned on. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the CVCS.

To cover all phases of the plant operation, boron dilution during refueling, startup, and power operation are considered in this analysis.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as discussed in Section 15.0.1.

15.4.5.2 Method of Analysis and Results

15.4.5.2.1 Dilution During Refueling

During refueling, the following conditions exist:

1. One residual heat removal pump is operating to ensure continuous mixing in the reactor vessel;

2. The valves in the seal injection water header to the reactor coolant pumps are normally closed;

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

4. The boron concentration of the refueling water is sufficient to provide a shutdown of at least 5% $\Delta k/k$ with all RCCAs in; periodic sampling ensures that this concentration is maintained; and

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

A minimum water volume in the RCS of 3535 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. A maximum dilution flow of 175 gpm, limited by the capacity of the two primary water supply pumps, and uniform mixing are also considered.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the Main Control Room. The count rate increase is a function of the subcritical effective multiplication factor.

At the beginning of the core life, equilibrium cycle core, the boron concentration must be reduced from 2560 ppm to 1400 ppm before the reactor will go critical. The acceptance criterion for this case is that the operator must have at least 30 minutes following the initiation of the dilution until the shutdown margin is lost. This time criterion provides ample time for the operator to recognize the alarm signal and isolate the reactor makeup water source by closing valves and stopping the primary water supply pumps before a complete loss of shutdown margin.

The results of this analysis concluded that the operator has at least 50 minutes before the shutdown margin is lost and the reactor goes critical. The sequence of events for all cases is presented on Table 15.4-4.

15.4.5.2.2 Dilution During Startup

Prior to startup, the RCS is filled with borated (1940 ppm) water from the refueling water storage tank. Core monitoring is by external source range detectors. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps. High source level and all reactor trip alarms are effective.

In the analysis, a maximum dilution flow of 208 gpm is considered. The volume of reactor coolant is approximately 9765 ft^3 which is the active volume of the RCS excluding the pressurizer.

With all RCCAs inserted, the reactor would go critical at a reactor coolant boron concentration of 1825 ppm. The minimum time required to reduce the boron concentration to 1825 ppm is 16 minutes. This is more than adequate time for the operator to recognize the high count rate signal and terminate the dilution flow.

15.4.5.2.3 Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the primary water pumps. However, the effective reactivity addition rate for a boron dilution was conservatively calculated for a flow of 208 gpm with an active RCS volume of 9765 ft^3. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be 1.6×10^{-5} Δk/sec which is within the range of insertion rates analyzed for the RCCA bank withdrawal at power event.

The dilution at power case is analyzed for both automatic and manual reactor control (i.e., rod control).

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. Before reaching this point, however, two alarms would be actuated to warn the operator of the 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 23.5 minutes after the low-low alarm before the total shutdown margin 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 reboration.

If the reactor is in manual control and the operator takes no action, the power and temperature will rise to the power range high neutron flux setpoint. The boron dilution accident in this case is essentially identical to a RCCA withdrawal accident. The reactivity insertion rate from the boron dilution (1.6 pcm/sec) is well within the range of reactivity insertion rates considered in Section 15.4.2, Uncontrolled Rod Cluster Control Assembly Withdrawal at Power. There is ample time (approximately 22.2) minutes) available after a reactor trip before the reactor can return to critical for the operator to determine the cause of dilution, isolate the primary grade water, and initiate boration.

15.4.5.3 Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered highly unlikely. Nevertheless, if it does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before shutdown margin is lost. These corrective actions are detailed in operating instructions and are familiar to the plant operator. It is considered incredible for the operator to ignore all the indications available.

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

Fuel and core loading errors are discussed in Section 4.3.4.2.

15.4.7 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.7.1 Accident Description

This accident is a result of an extremely unlikely mechanical failure of a control rod mechanism pressure housing such that the RCS pressure would then eject the control rod and drive shaft. The consequences of this mechanical failure, in addition to being a minor loss-of-coolant accident, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage for severe cases. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by a reactor trip actuated by high neutron flux signals.

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

15.4.7.2 Design Precautions and Protection

Certain features in Westinghouse pressurized water reactors (PWR) 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, a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

15.4.7.2.1 Mechanical Design

The mechanical design is discussed in Section 4.5. An evaluation of the mechanical design and quality control procedures indicates that a failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core should not be considered credible for the following reasons:

1. Each CRDM housing is completely assembled and shop-tested at 4100 psi;

2. The mechanism housings are individually hydrotested to 3750 psig as they are installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed RCS;

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 design earthquake can be accepted within the allowable primary working stress range specified by the ASME Boiler and Pressure Vessel Code, Section III, for Class A components; and

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.

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

15.4.7.2.2 Nuclear Design

Even if a rupture of the CRDM is postulated, the operation of a plant using chemical shim is such that the severity of an ejected rod is inherently limited. In general, the reactor is operated with control rods inserted only far enough to control axial flux. 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 ejected rod. Therefore, should a control rod be ejected from the reactor vessel during normal operation, there would probably be no reactivity excursion, since most of the control rods are fully withdrawn from the core, or a minor reactivity excursion if an inserted rod is ejected from its normal position.

However, it may occasionally be 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 control rods above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all control rods is continuously indicated in the Control Room. An alarm will occur if a bank of control rods approaches its insertion limit or if one rod deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at the low level alarm and emergency boration at the low-low alarm. The control rod position monitoring and alarm systems have been described in detail in Section 7.7.

15.4.7.2.3 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 1. The protection for this accident is provided by the high neutron flux trip (high and low settings) and the high positive rate neutron flux trip.

15.4.7.2.4 Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a control rod mechanism housing failure, investigations have shown that failure of a control rod housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings that would increase the severity of the initial accident.

15.4.7.3 Limiting Criteria

Due to the extremely low probability of a rod ejection accident, some fuel damage could be considered an acceptable consequence, provided there is no possibility of the offsite consequences exceeding the guidelines of 10CFR100. Although severe fuel damage to a portion of the core may in fact be acceptable, it is difficult to treat this type of incident on a sound theoretical basis. For this reason, criteria for the threshold of fuel failure are established and it is demonstrated that this limit will not be exceeded.

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 (see Reference 2). Extensive tests of UO_2-Zirconium clad fuel

rods representative of those in PWR type cores have demonstrated failure thresholds in the range of 240 to 257 cal/g. However, other rods of a slightly different design have exhibited failures as low as 225 cal/g. These results differ significantly from the TREAT (see Reference 3) results, which indicated a failure threshold of 280 cal/g. Limited results have indicated this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be by melting for zero burnup rods and by brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/g for unirradiated rods and 200 cal/g for irradiated rods. However, catastrophic failure (large fuel dispersal, large pressure rise) even for irradiated rods did not occur below 300 cal/g.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria (Reference 14) are:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel;

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits; and

3. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

15.4.7.4 Method of Analysis

The calculation of the rod ejection transient is performed in two stages, first an average channel core calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients 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 asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed investigation using this method, and a demonstration of the conservativeness of the calculation compared to three-dimensional spatial kinetics, is presented in Reference 4.

15.4.7.4.1 Average Core Analysis

The spatial kinetics computer code TWINKLE (Reference 8) is used for the average core transient analysis. This computer 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 multiregion, transient fuel clad to coolant heat transfer model for the calculation of pointwise Doppler and moderator feedback effects.

In this analysis, the code is used primarily 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. (This method has also been described in Reference 6.) A more detailed description of TWINKLE appears in Section 15.0.10.

15.4.7.4.2 Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors. The hot spot analysis is performed using a detailed fuel and clad transient heat transfer computer code FACTRAN (Reference 9). This computer code calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the heat flux at the surface of the rod using as input the nuclear power versus time for the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and a correlation (see Reference 7) 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 may be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. A more detailed description of FACTRAN appears in Section 15.0.10.

15.4.7.4.3 System Overpressure Analysis

Because the safety limits for fuel damage (Section 15.4.7.3) 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 steam generators, 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.

15.4.7.5 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this core. The more important parameters are discussed below. Table 15.4-5 presents the parameters used in this analysis.

15.4.7.5.1 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths are calculated using either three-dimensional static methods or a synthesis of one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level. Adverse xenon distributions are considered in the calculation.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distributions before and after ejection for a worst case can be found in Reference 15.

15.4.7.5.2 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 a region is dependent on flux, these regions have high weights. This means that the reactor feedback is larger than that indicated by a simple single channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting

factors take the form of multipliers, which when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method was employed, the axial weighting was not used. In addition, no weighting was applied to the moderator feedback. A very conservative radial weighting factor was applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time, accounting for the missing spatial dimension. These weighting factors have been shown to be conservative compared to three-dimensional analysis.

15.4.7.5.3 Moderator and Doppler Coefficients

The critical boron at the BOL and EOL were adjusted in the nuclear code in order to obtain moderator density coefficient curves which are representative of actual design conditions for this plant. As discussed above, no weighting factor was applied to these results.

The resulting moderator temperature coefficient is at least +7 pcm/°F at the appropriate zero or full power nominal average temperature, and becomes less positive for higher temperatures. This is necessary since the TWINKLE code is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler coefficient used does not directly correlate with Figure 15.0-3 because, as discussed previously, the TWINKLE code, on which the neutronic analysis is based, is a diffusion-theory code rather than a point-kinetics approximation. The Doppler defect used as an initial condition is 900 pcm for BOL conditions and 840 pcm for EOL conditions. The Doppler weighting factor will increase under accident conditions, as discussed above.

15.4.7.5.4 Delayed Neutron Fraction

Calculations of the effective delayed neutron fraction (β_{eff}) have yielded values of 0.70% at BOL and 0.50% at EOL.

The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero-power transients. Pessimistic estimates of β_{eff} of 0.53% at BOL and 0.45% at EOL were used in the analysis.

15.4.7.5.5 Trip Reactivity Insertion

The trip reactivity assumed is shown on Table 15.4-5, and includes the effect of one stuck rod. These values were 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 second after the high neutron flux trip point was reached. This delay was assumed to consist of 0.2 second for

the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second for the coil to release the rods. The analysis assumes that full insertion of the rod to the dashpot region does not occur until 2.4 seconds after the start of fall, although measurements have indicated that this value should be closer to 1.3 seconds. The choice of such a conservative insertion rate means there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full-power accidents.

15.4.7.6 Results

15.4.7.6.1 Limiting Hot Channel Factors

Cases are presented for both BOL and EOL at zero and full power.

For BOL at full power, control Bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.17% $\Delta k/k$ and 6.5, respectively. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

For BOL at zero power, control Bank D was assumed to be fully inserted and Banks B and C were at their insertion limits. The worst ejected rod is located in control Bank D and has a worth of 0.75% $\Delta k/k$ and a hot channel factor of 14.8. The peak fuel center temperature was 4129°F.

For EOL at full power, control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.19% $\Delta k/k$ and 6.2, respectively. The peak hot spot fuel temperature reached melting, conservatively assumed at 4800°F. However, melting was restricted to less than 10% of the pellet.

The ejected rod worth and hot channel factor for EOL at zero power were obtained assuming control Bank D to be fully inserted and Bank C at its insertion limit. The results were 0.90% $\Delta k/k$ and 21.5, respectively. The peak fuel center temperature was 4053°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-5. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (BOL at full power and EOL at zero power) are presented in Figures 15.4-17 through 15.4-20.

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-17 through 15.4-20, is presented in Table 15.4-6. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant-accidents (LOCAs) are discussed elsewhere in this report. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

15.4.7.6.2 (Deleted)

15.4.7.6.3 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% of the rods entered DNB based on a detailed three-dimensional THINC analysis (Reference 10).

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.

15.4.7.6.4 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar 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 (Reference 13). Since the severity of the present analysis does not exceed the worst case analysis, the accident will not result in an excessive pressure rise or further damage to the RCS.

15.4.7.6.5 Lattice Deformations

In the region of the hot spot, there will be a large temperature gradient. 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 will produce a net fluid away from that region. However, the fuel heat is released to the water relatively slowly. 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 will be pessimistically ignored in the analyses.

15.4.7.7 Offsite Exposure

Following a postulated RCCA ejection accident with an assumed 10% of the rods entering DNB, coincident with loss of offsite power and primary to secondary leakage, the resultant offsite doses would be considerably less than in the case of a double-ended main coolant pipe break (the design basis accident), and therefore within the guidelines of 10CFR100.

15.4.7.8 Conclusions

Even on the most pessimistic basis, the analyses indicated the fuel and clad limits (Section 15.4.7.3) were not exceeded. It was concluded that there was no danger of sudden fuel dispersal into the coolant. The peak pressure does not exceed that which would exceed the faulted condition stress limits, and it was concluded that there was no danger of consequential damage to the primary circuit. The amount of fission products released as a result of clad rupture during DNB is considerably less than in the case of the double-ended main coolant pipe break (the design basis accident), and therefore within the guidelines of 10CFR100.

15.4.8 References, Section 15.4

1. Burnett, T.W.T. et al., "Reactor Protection System Diversity in Westinghouse PWRs," WCAP-7306, April 1969.

2. Taxelius, T.G., ed. "Annual Report - Spert Project, October 1968 September 1969," Idaho Nuclear Corporation TID-4500, June 1970.

3. Liimatainen, R.C. and Testa, F.J., "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core simulated Fuel Elements," Argonne National Laboratory Chemical Engineering Division Semi-Annual Report, ANL-7225, January-June 1966.

4. Risher, D.H., "An Evaluation of the Rod Ejection Accident in Westinghouse PWRs Using Spatial Kinetics Methods," Revision A-1, WCAP-7588, January 1975.

5. (Deleted)

6. French, R.J. et al., "Indian Point Unit No. 2 Rod Ejection Analysis," WCAP-2940, May 1966.

7. (Deleted)

8. Risher, D.H., Jr. and Barry, R.F., "TWINKLE - A Multidimensional Neutron Kinetics Computer Code," WCAP-7979-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

9. Hargrove, H.G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

10. Friedland, A.J. and S. Ray, "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

11. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

12. Friedland, A.J. and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A and WCAP-11398-A, April 1989.

13. Haessler, R.L., et al., "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394-P-A and WCAP-11395-A, January 1990.

14. Letter, NS-NRC-89-3466, W.J. Johnson (Westinghouse) to R.C. Jones (USNRC), October 23, 1989.

15. Bishop, A.A.; Sandberg, R.O.; and Tong, L.S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

TABLE 15.4-1

TIME SEQUENCE OF EVENTS
UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL

ACCIDENT	EVENTS	TIME (sec)
Uncontrolled RCCA Bank 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	10.3
	Peak nuclear power occurs	10.5
	Rods begin to fall into core	10.8
	Minimum DNBR occurs	12.2
	Peak heat flux occurs	12.2
	Peak hot spot average clad temperature occurs	12.3
	Peak hot spot average fuel temperature occurs	12.4

TABLE 15.4-2

TIME SEQUENCE OF EVENTS
UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

ACCIDENT	EVENTS	TIME (sec)
Uncontrolled RCCA Bank Withdrawal at Power		
1. High Reactivity Insertion Rate	Initiation of uncontrolled RCCA withdrawal at full power at high reactivity insertion rate with minimum reactivity feedback (7.5×10^{-4} Δk/sec)	0.0
	Power range high neutron flux high trip setpoint reached	1.5
	Rods begin to fall into core	2.0
	Minimum DNBR occurs	3.1
2. Low Reactivity Insertion Rate	Initiation of uncontrolled RCCA withdrawal at full power at low reactivity insertion rate with minimum reactivity feedback (3.0×10^{-5} Δk/sec)	0.0
	Power range high neutron flux high trip setpoint reached	32.0
	Rods begin to fall into core	32.5
	Minimum DNBR occurs	33.2

TABLE 15.4-3

TIME SEQUENCE OF EVENTS
STARTUP OF AN INACTIVE REACTOR COOLANT LOOP
OPEN STOP VALVES

ACCIDENT	EVENTS	TIME (sec)
Startup of an Inactive Reactor Coolant Loop	Initiation of pump startup	0.0
	Power reaches high nuclear flux trip	4.7
	Rods begin to drop	5.2
	Minimum DNBR occurs	5.4

TABLE 15.4-4

TIME SEQUENCE OF EVENTS
CVCS MALFUNCTION

ACCIDENT	EVENTS	TIME (sec)
CVCS Malfunction		
1. Dilution during refueling	Dilution begins	0.0
	Shutdown margin lost (if dilution continues)	3180
2. Dilution during startup	Dilution begins	0.0
	Shutdown margin lost (if dilution continues)	2238
3. Dilution during full power operation		
a. Automatic reactor control	Dilution begins	0.0
	Shutdown margin lost	1624
b. Manual reactor control	Dilution begins	0.0
	Reactor trip on high neutron flux reached due to dilution	81
	Shutdown margin lost (if dilution continues)	1543

TABLE 15.4-5

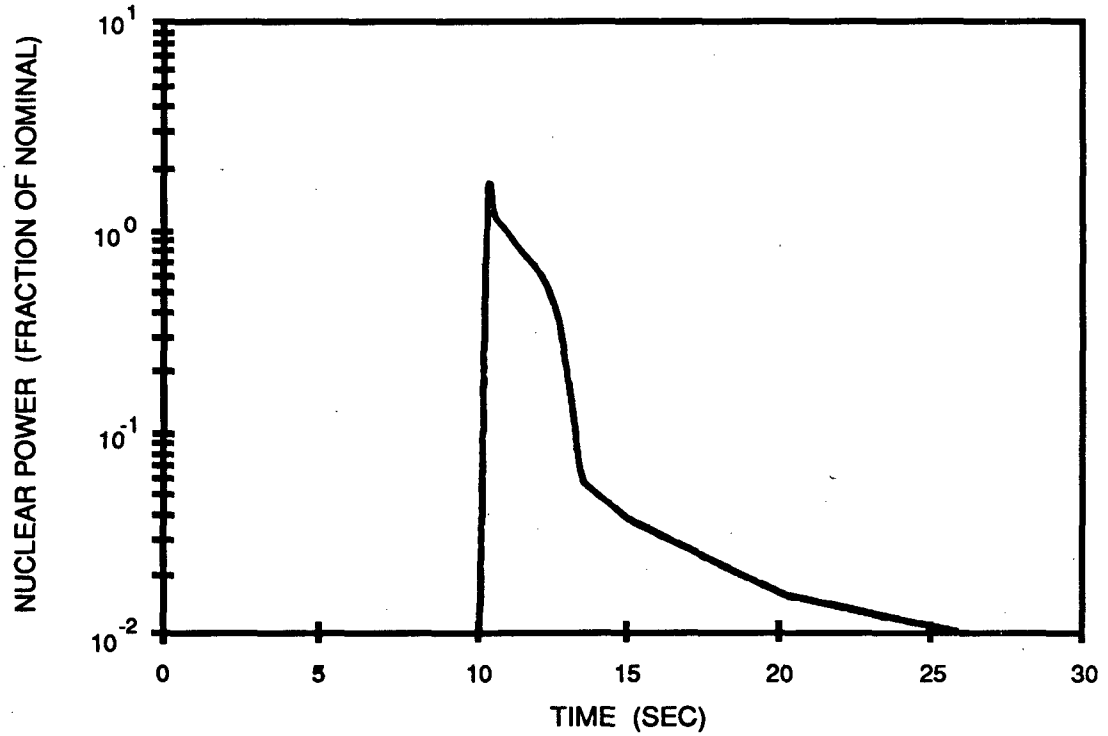
PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

TIME IN LIFE	FULL POWER BOL	ZERO POWER BOL	FULL POWER EOL	ZERO POWER EOL
Power Level, %	102	0	102	0
Initial average coolant temperature, °F	567.7	547	567.7	547
Ejected rod worth, % $\Delta k/k$	0.17	0.75	0.19	0.90
Delayed neutron fraction, %	0.53	0.53	0.45	0.45
Feedback reactivity weighting	1.30	2.398	1.30	3.19
Trip reactivity, % $\Delta k/k$	4.0	2.0	4.0	2.0
F_Q before rod ejection	2.55	-	2.55	-
F_Q after rod ejection	6.5	14.8	6.62	21.5
Number of operational pumps	4	2	4	2
Max. fuel pellet average temperature, °F	4085	3662	3813	3688
Max. fuel center temperature, °F	>4900	4129	>4800	4053
Max. fuel store energy, cal/g	179	157	165	158
% Fuel Melt	<10	0	<10	0

TABLE 15.4-6

TIME SEQUENCE OF EVENTS
RCCA EJECTION

ACCIDENT	EVENTS	TIME (sec)
Rod Cluster Control Assembly Ejection		
1. BOL Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Rods begin to fall	0.55
	Peak nuclear power occurs	0.80
	Peak fuel average temperature occurs	2.07
	Peak heat flux occurs	2.22
2. EOL Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.16
	Peak nuclear power occurs	0.19
	Rods begin to fall	0.66
	Peak heat flux occurs	1.48
	Peak fuel average temperature occurs	1.68



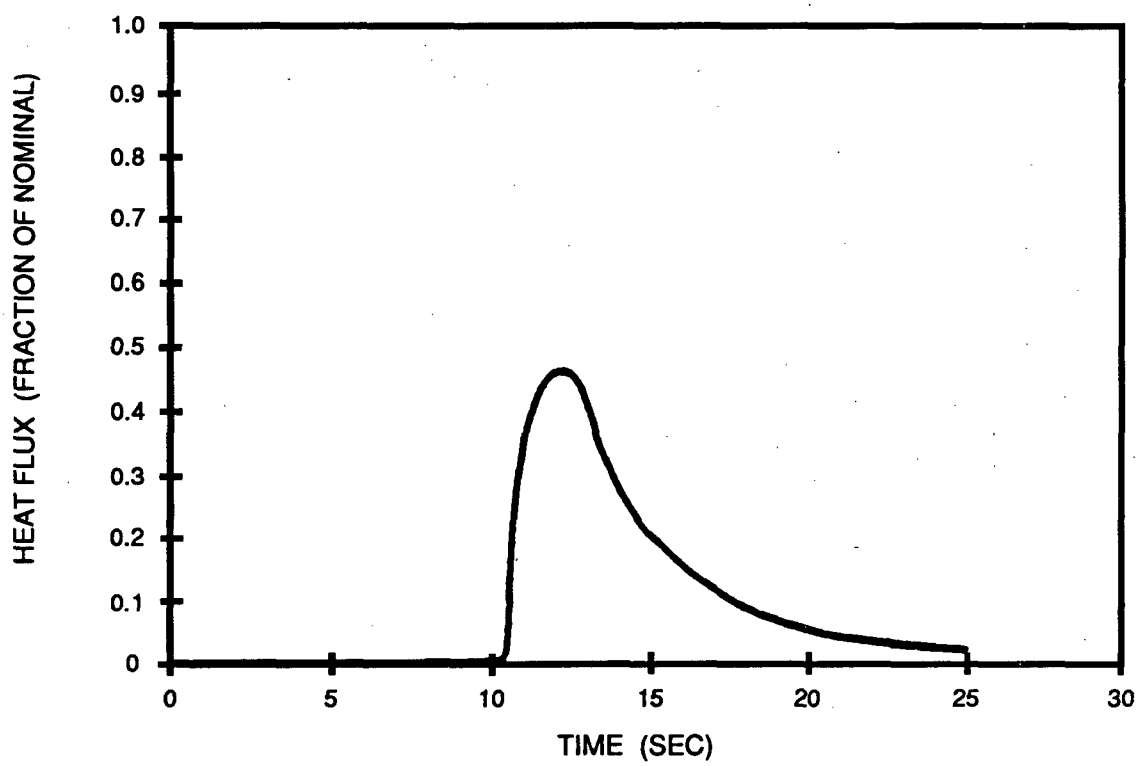


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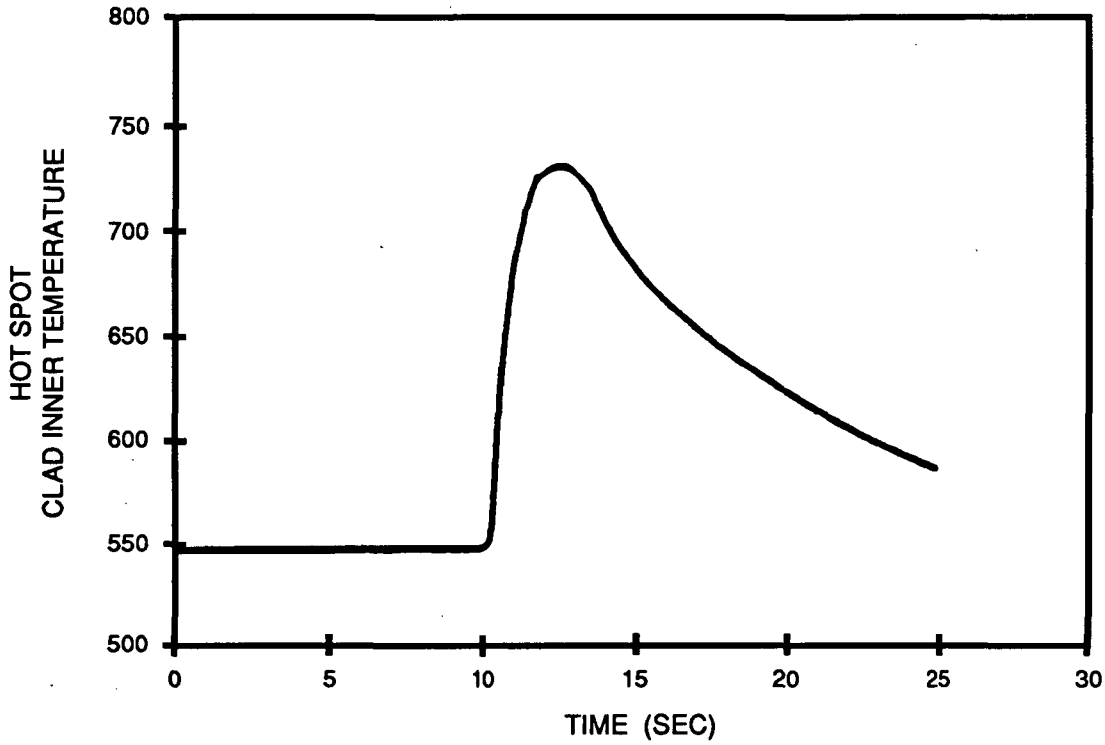
Figure 15.4-1
ROD WITHDRAWAL FROM SUBCRITICAL
NUCLEAR POWER AND CORE HEAT FLUX

JULY 1993



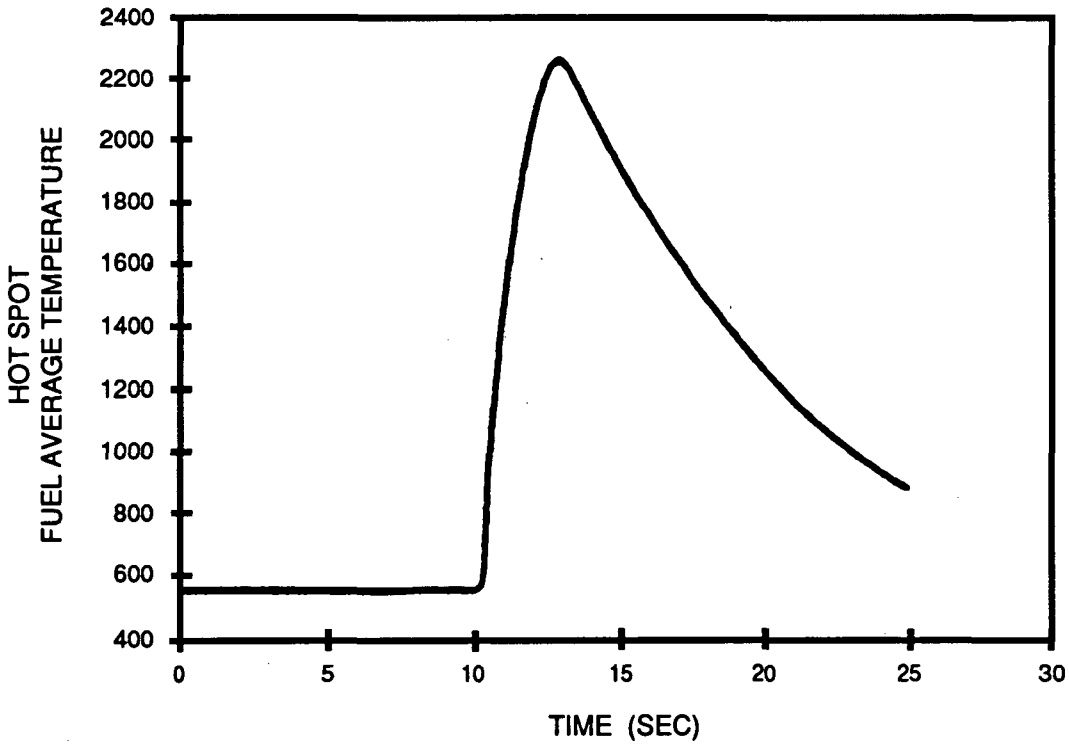


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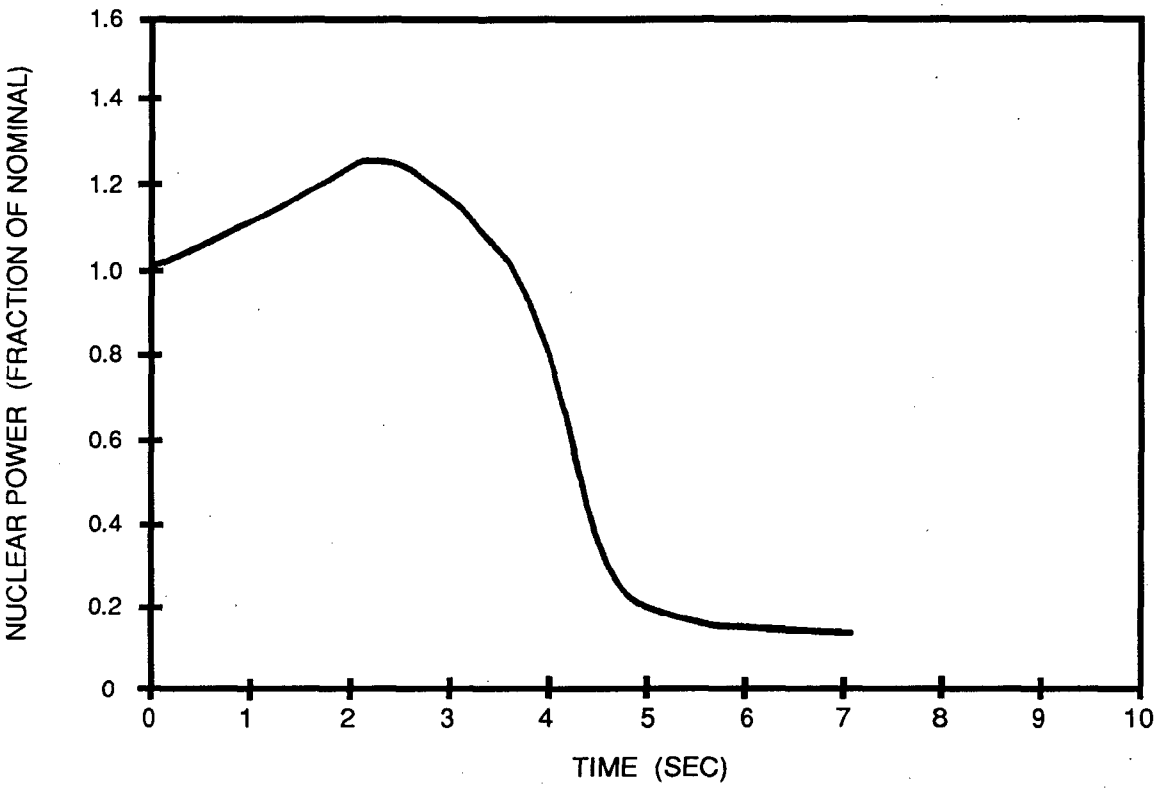
Figure 15.4-2
ROD WITHDRAWAL FROM SUBCRITICAL
CLAD INNER TEMPERATURE AND
FUEL AVERAGE TEMPERATURE

JULY 1993



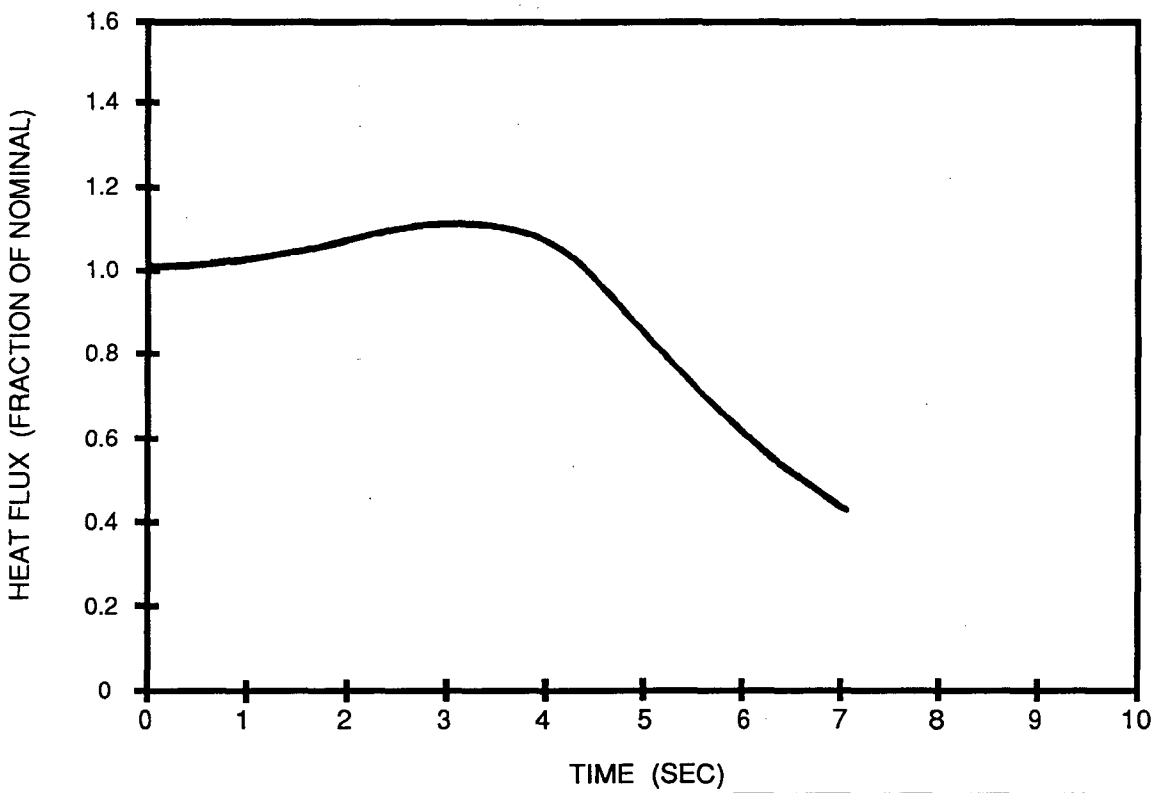


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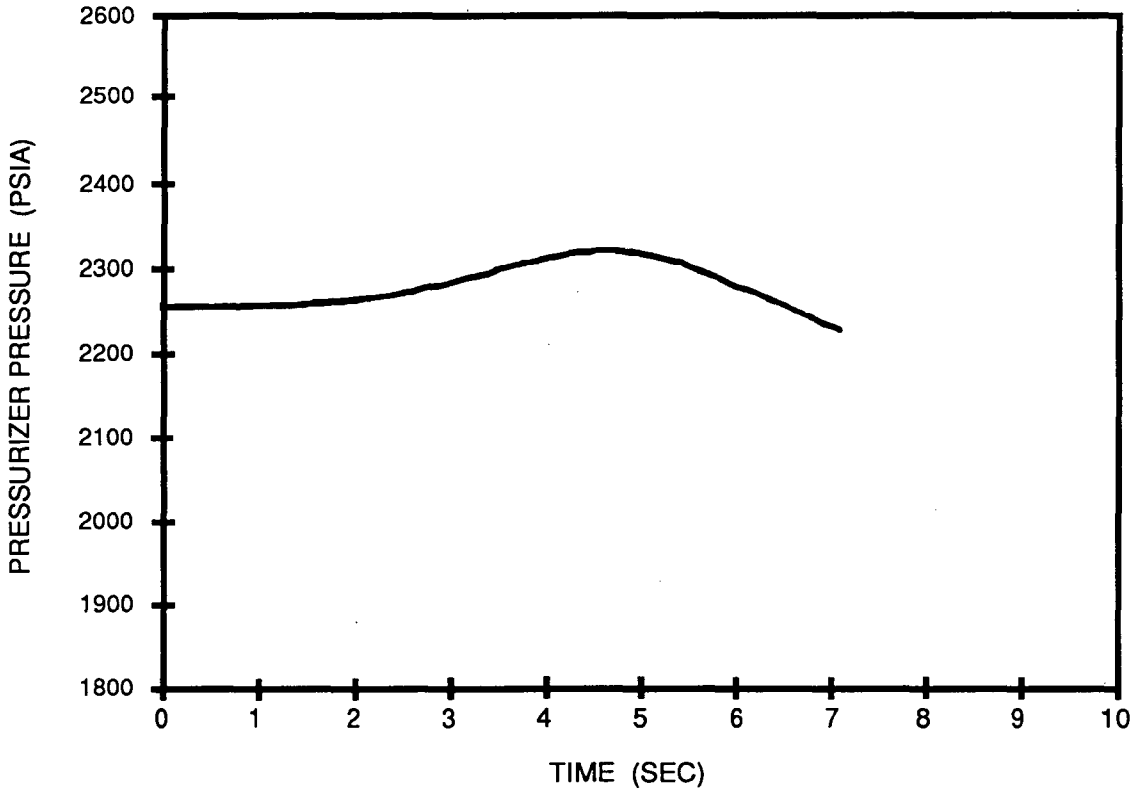
Figure 15.4-3
RCCA BANK WITHDRAWAL AT POWER
NUCLEAR POWER AND CORE HEAT FLUX
MINIMUM FEEDBACK - 75 PCM/SEC
JULY 1993



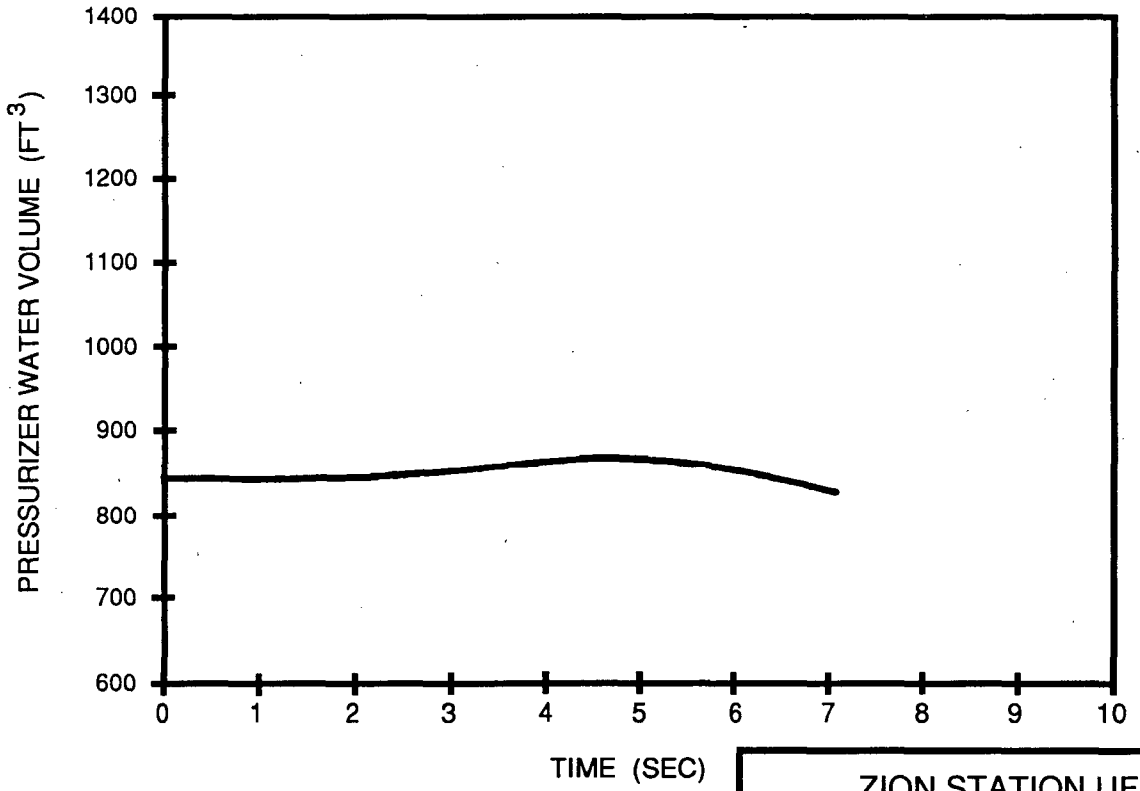


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Figure 15.4-4
RCCA BANK WITHDRAWAL AT POWER
PRESSURIZER PRESSURE
AND WATER VOLUME
MINIMUM FEEDBACK - 75 PCM/SEC

JULY 1993

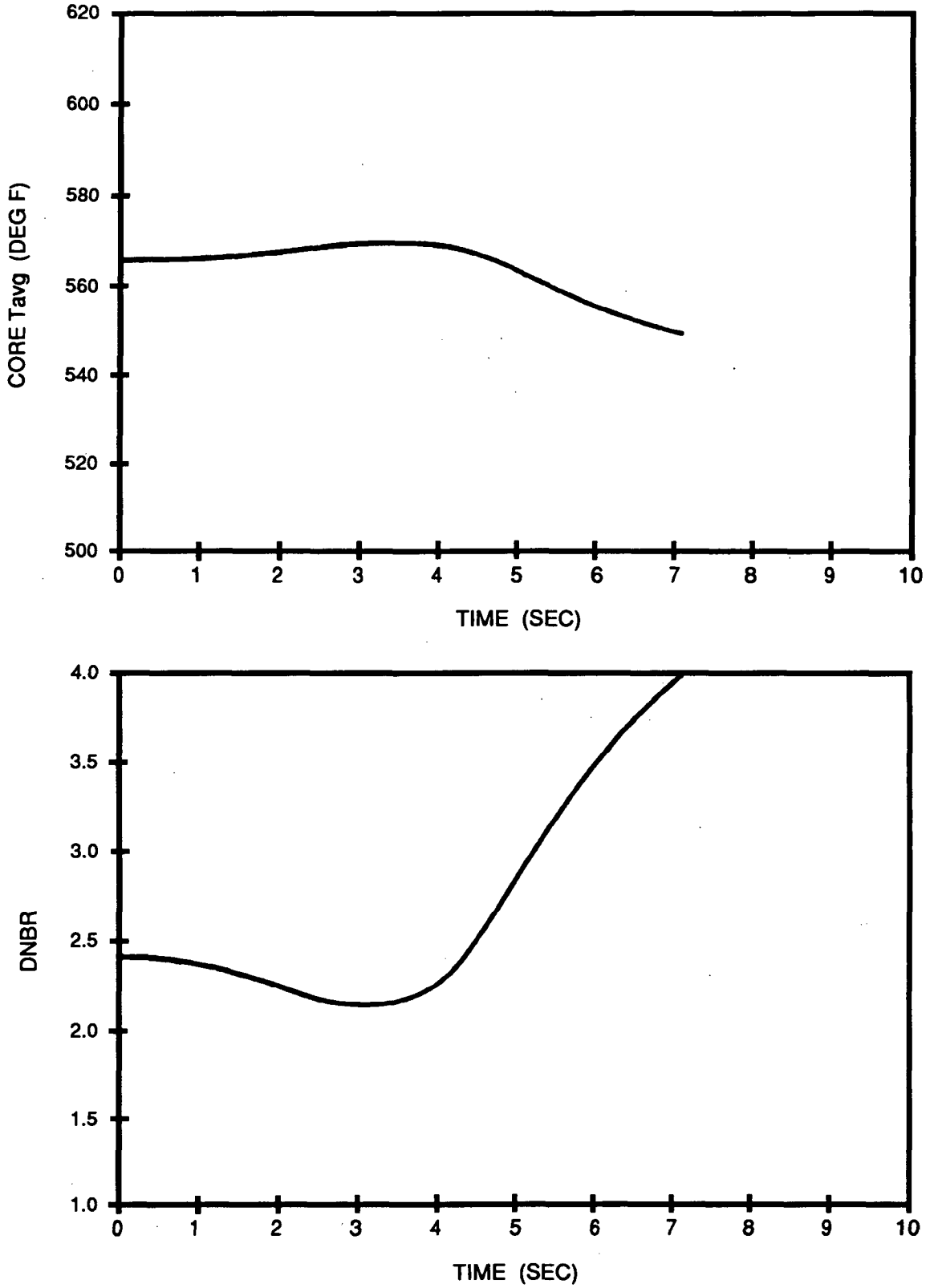

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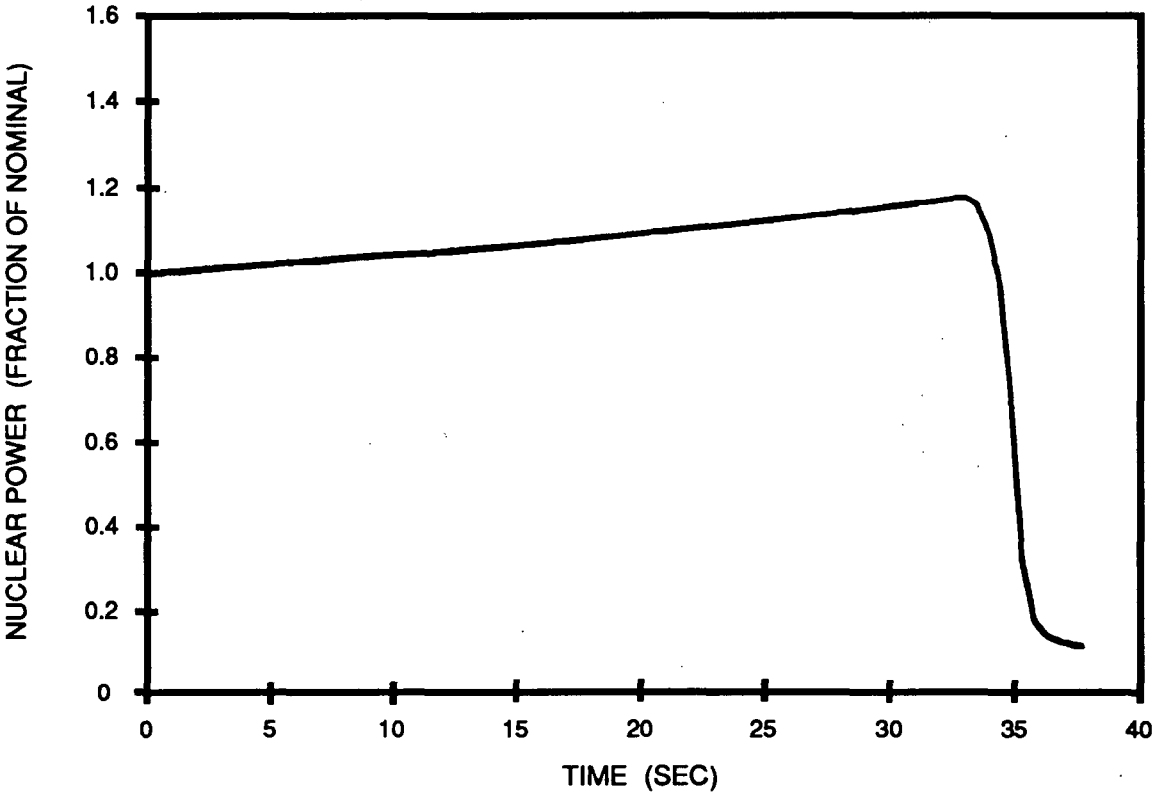
Figure 15.4-5
RCCA BANK WITHDRAWAL AT POWER
CORE Tavg AND DNBR
MINIMUM FEEDBACK - 75 PCM/SEC

JULY 1993



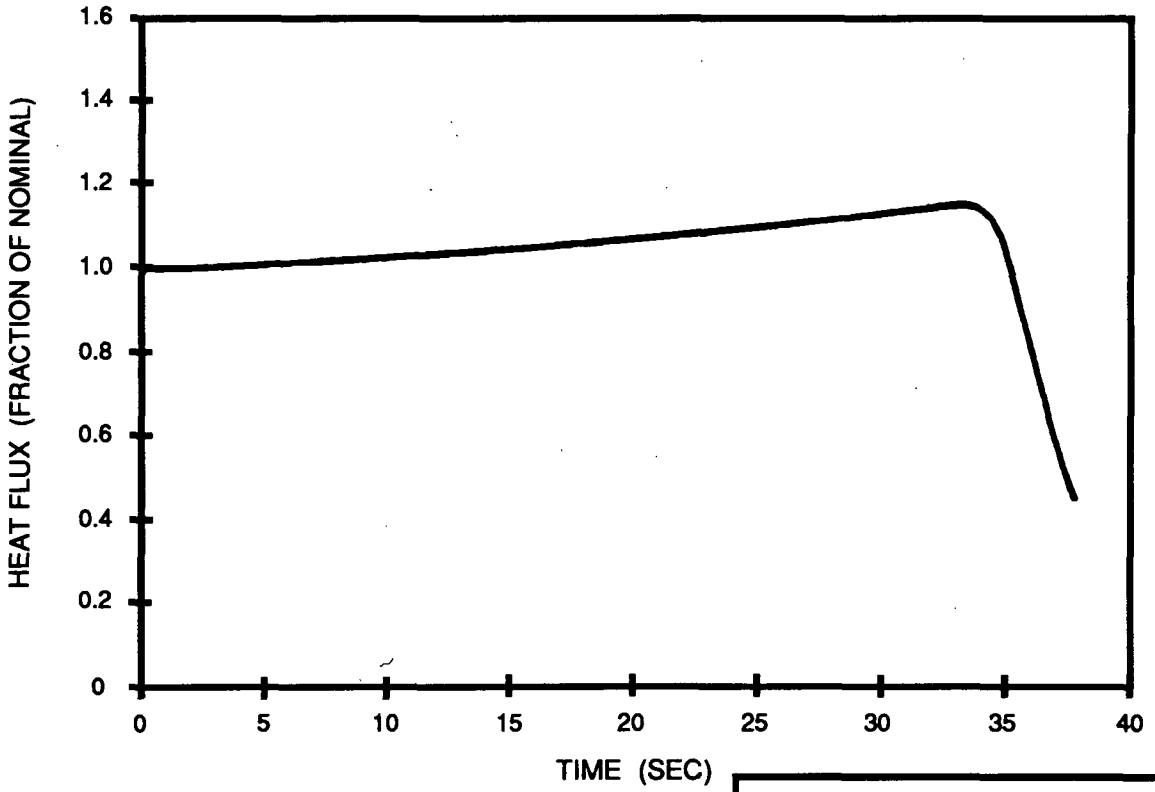


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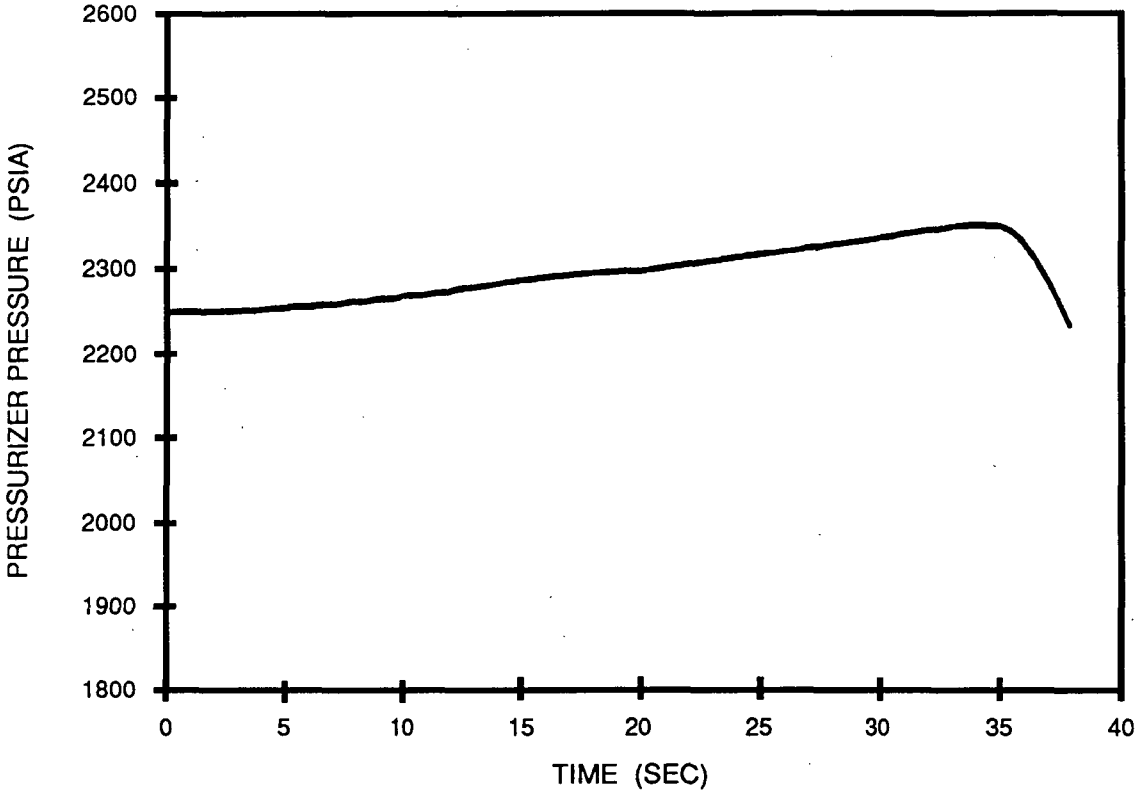
Figure 15.4-6
RCCA BANK WITHDRAWAL AT POWER
NUCLEAR POWER AND CORE HEAT FLUX
MINIMUM FEEDBACK - 3 PCM/SEC
JULY 1993



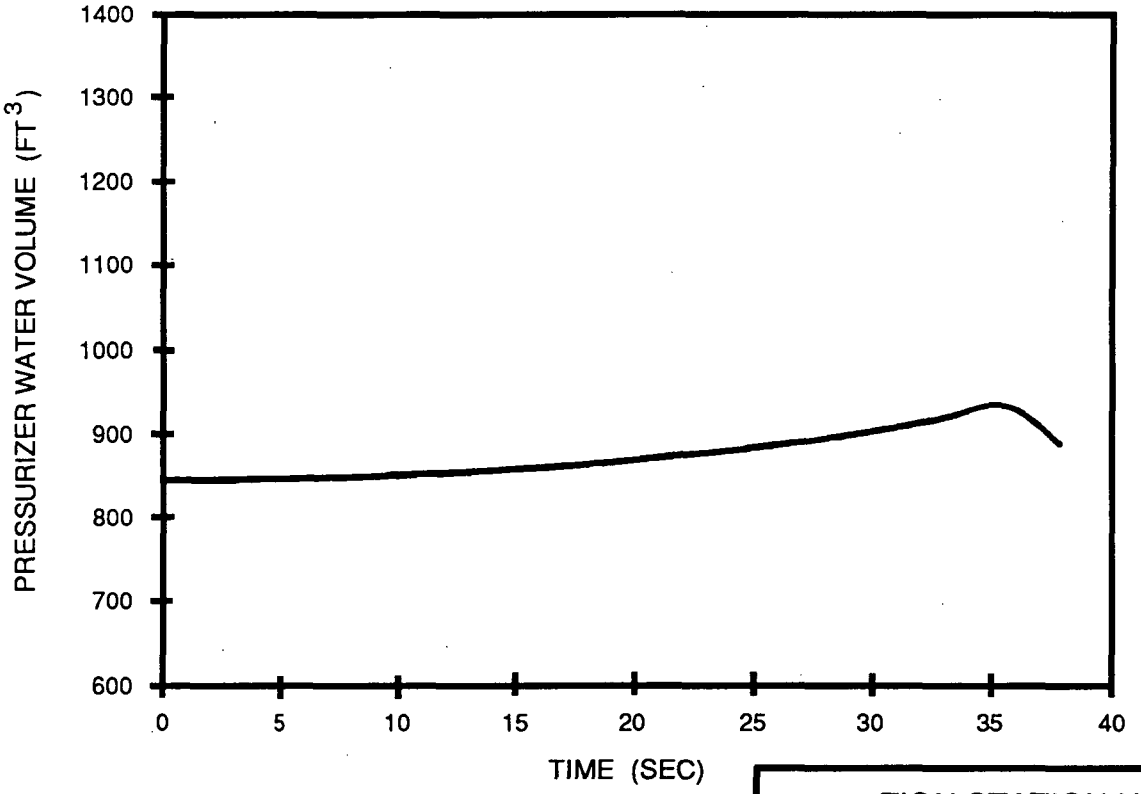


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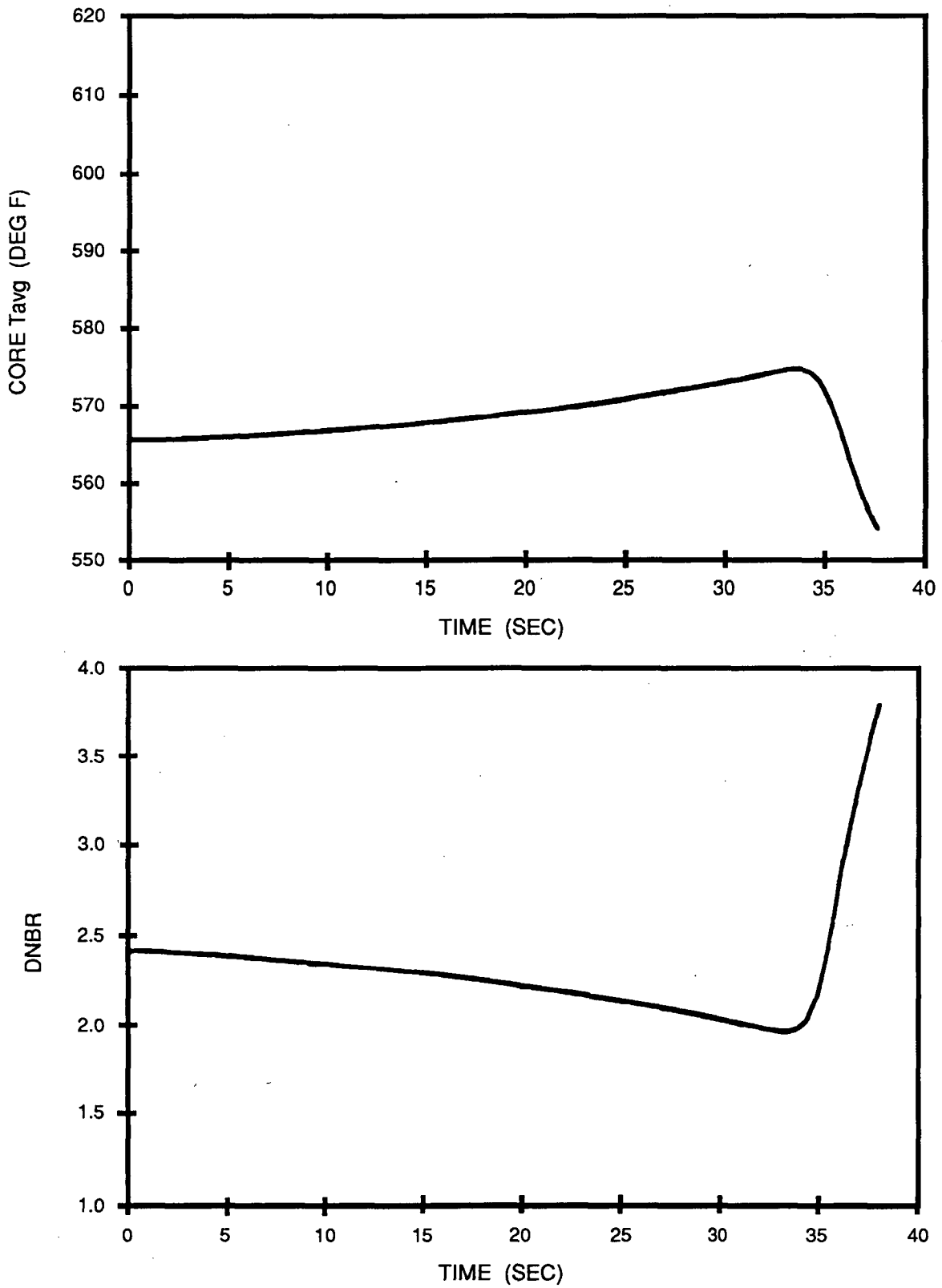
Figure 15.4-7
RCCA BANK WITHDRAWAL AT POWER
PRESSURIZER PRESSURE
AND WATER VOLUME
MINIMUM FEEDBACK - 3 PCM/SEC

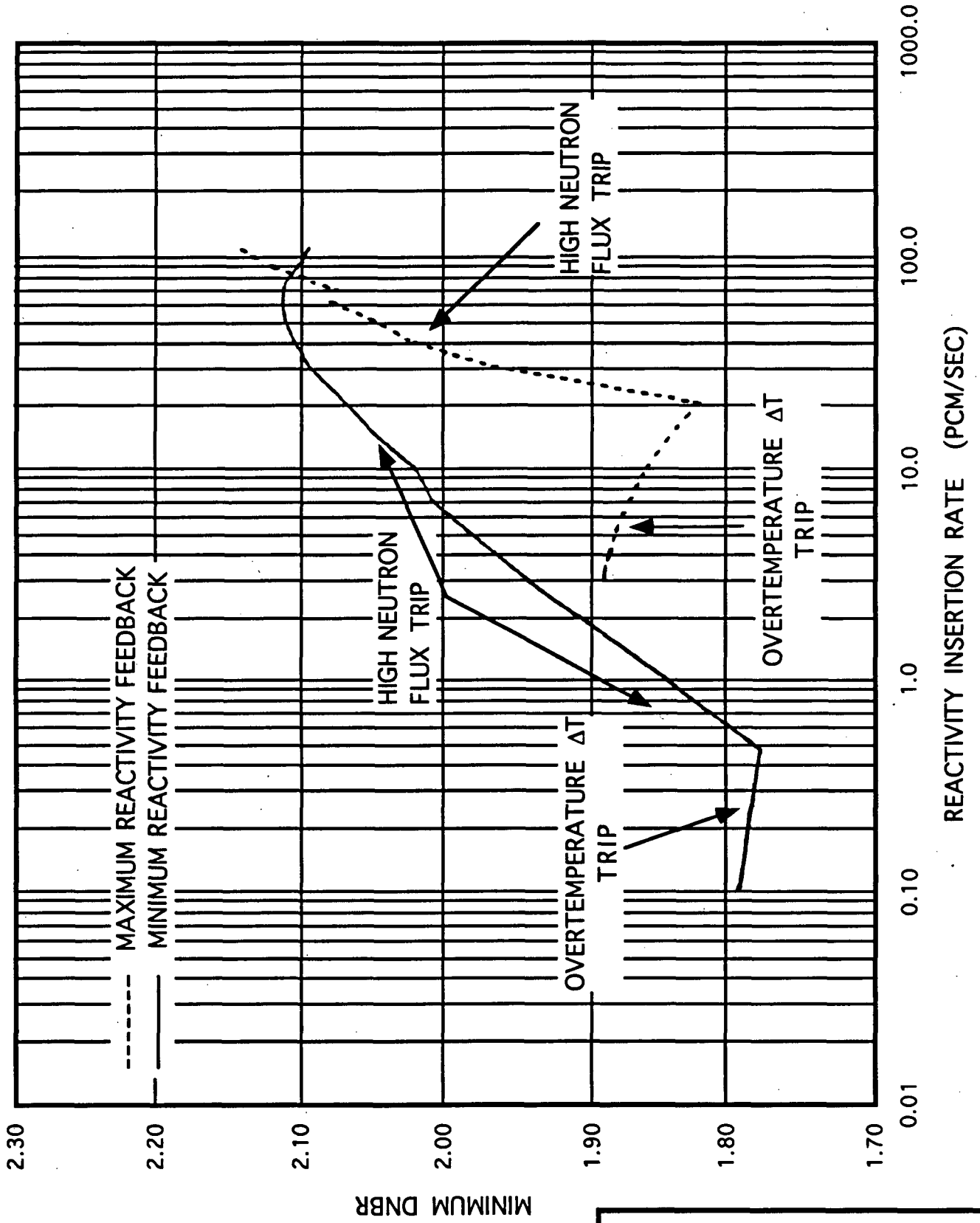
JULY 1993



ZION STATION UFSAR

Figure 15.4-8
RCCA BANK WITHDRAWAL AT POWER
CORE Tavg AND DNBR
MINIMUM FEEDBACK - 3 PCM/SEC

JULY 1993

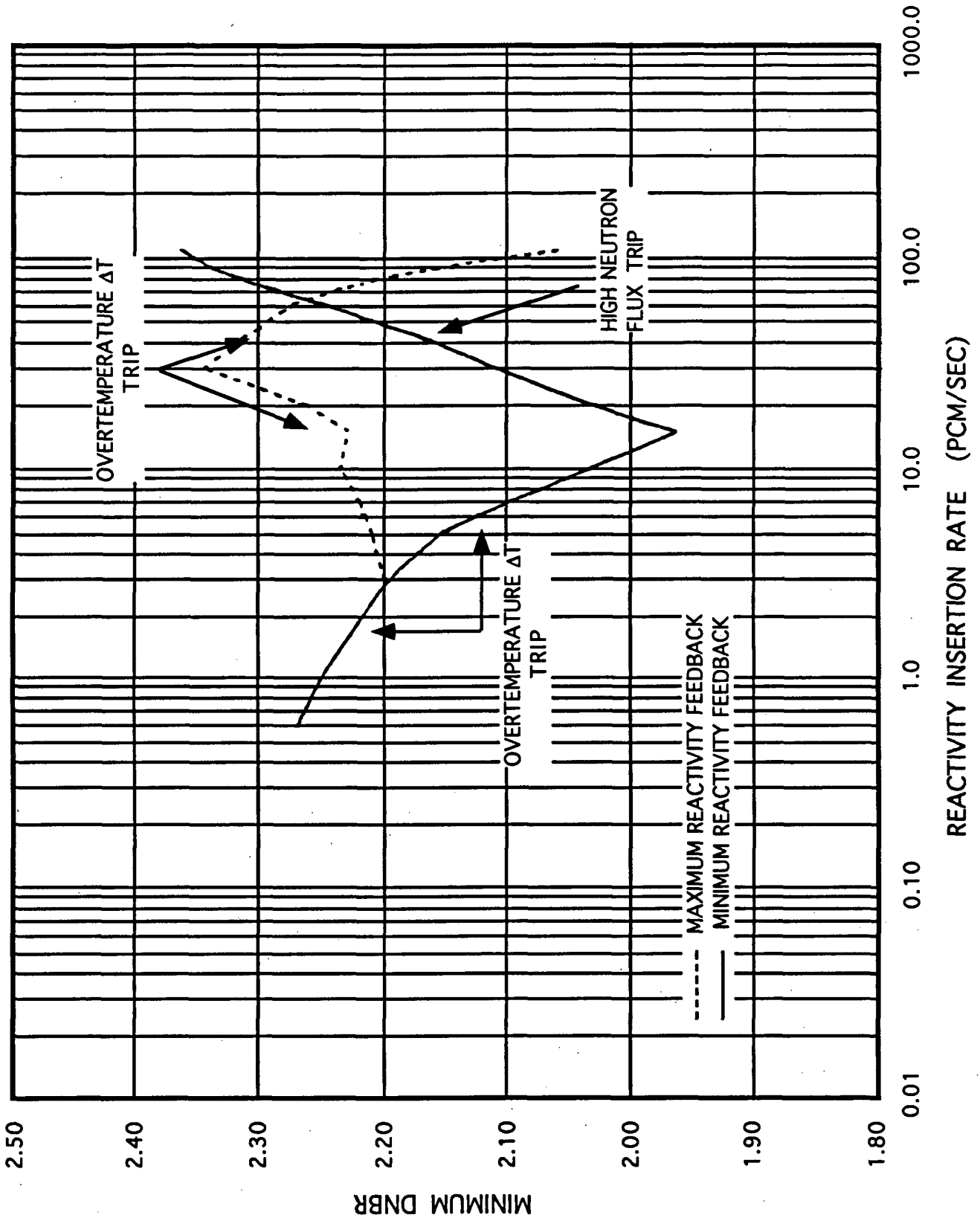


ZION STATION UFSAR

Figure 15.4-9
RCCA BANK WITHDRAWAL AT POWER
MINIMUM DNBR VERSUS INSERTION RATE
100% POWER

JULY 1993

- Figure 15.4-10 was Intentionally Deleted -

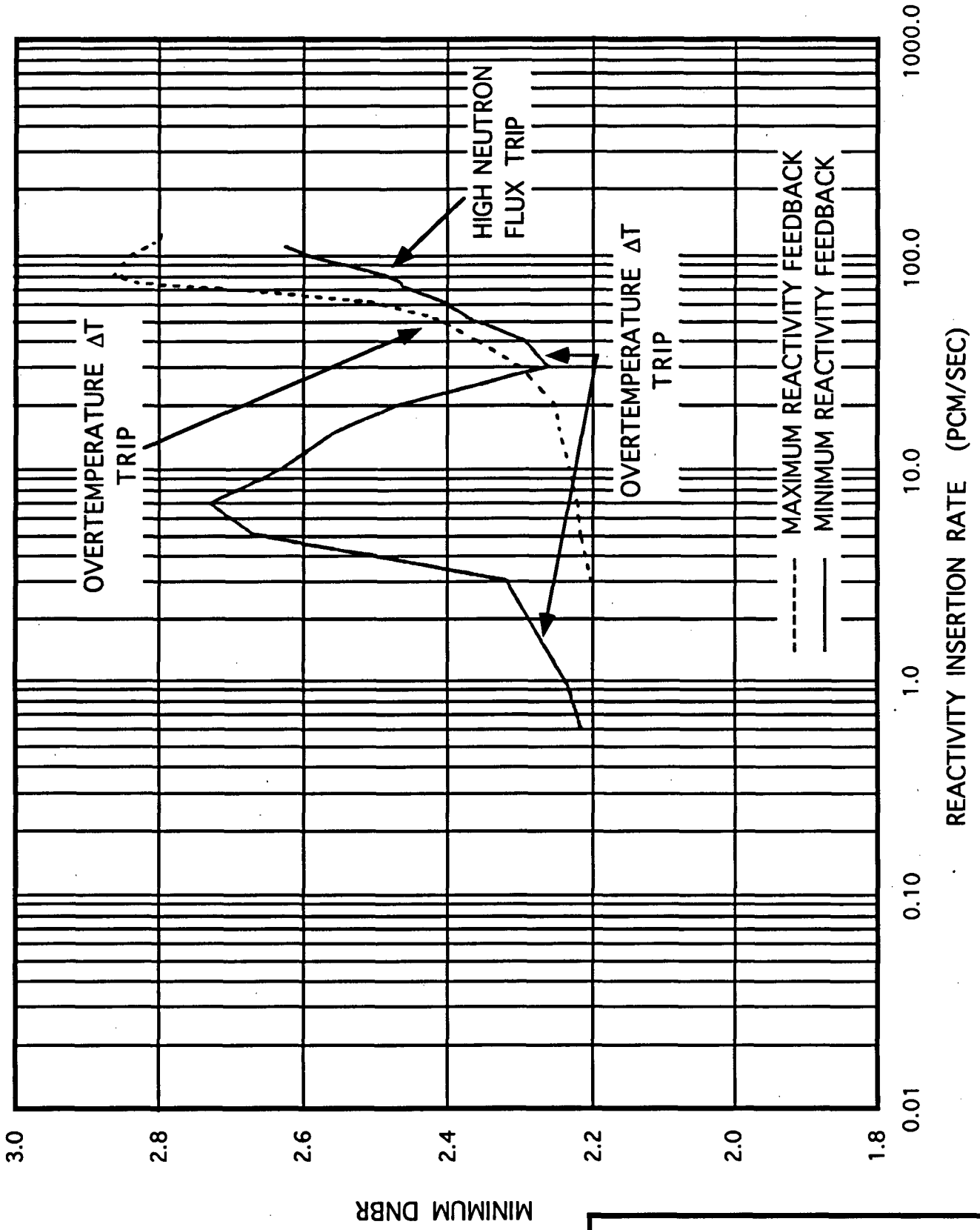


ZION STATION UFSAR

Figure 15.4-11

RCCA BANK WITHDRAWAL AT POWER
MINIMUM DNBR VERSUS INSERTION RATE
60% POWER

JULY 1993

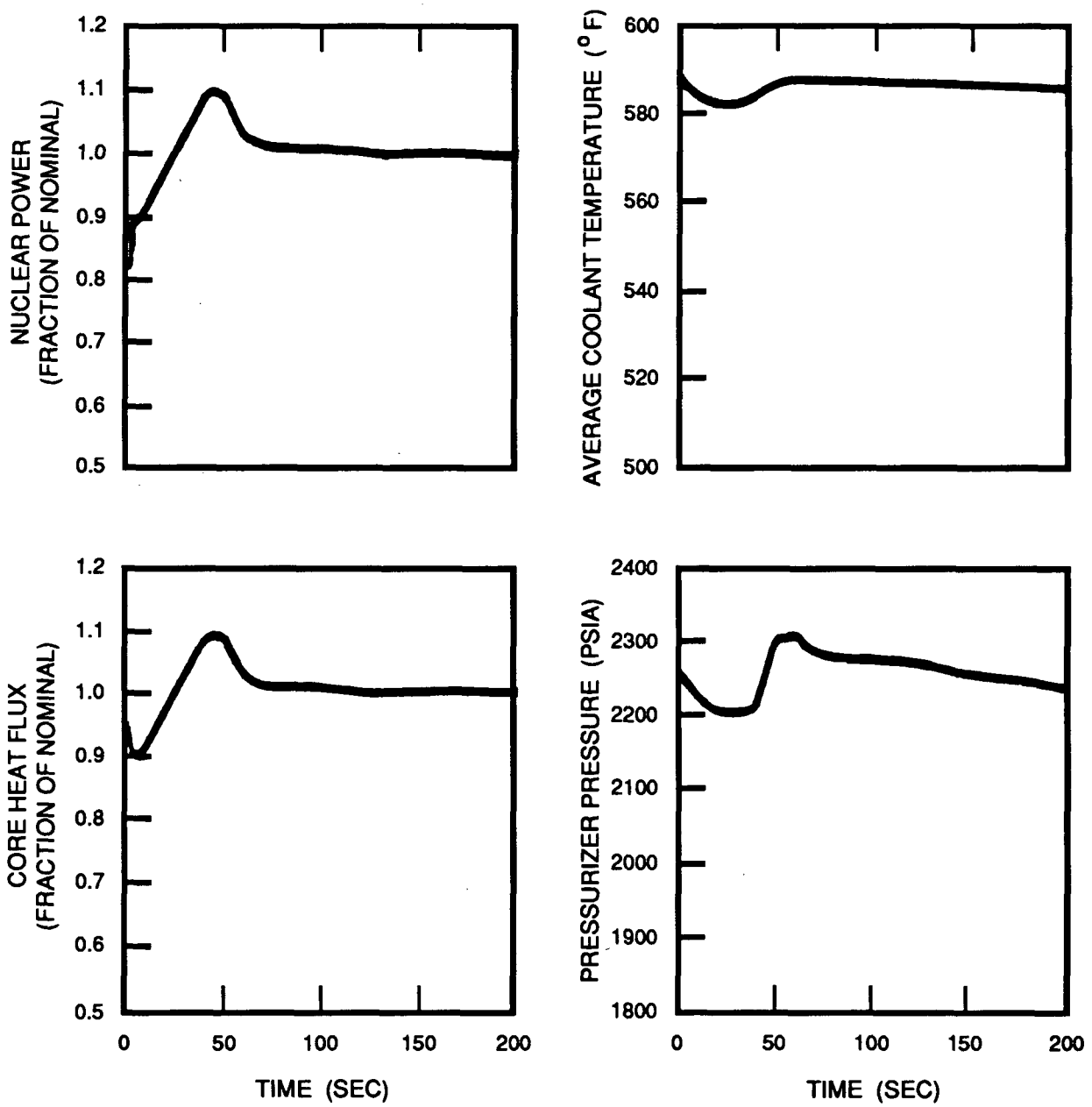


ZION STATION UFSAR

Figure 15.4-12

RCCA BANK WITHDRAWAL AT POWER
MINIMUM DNBR VERSUS INSERTION RATE
10% POWER

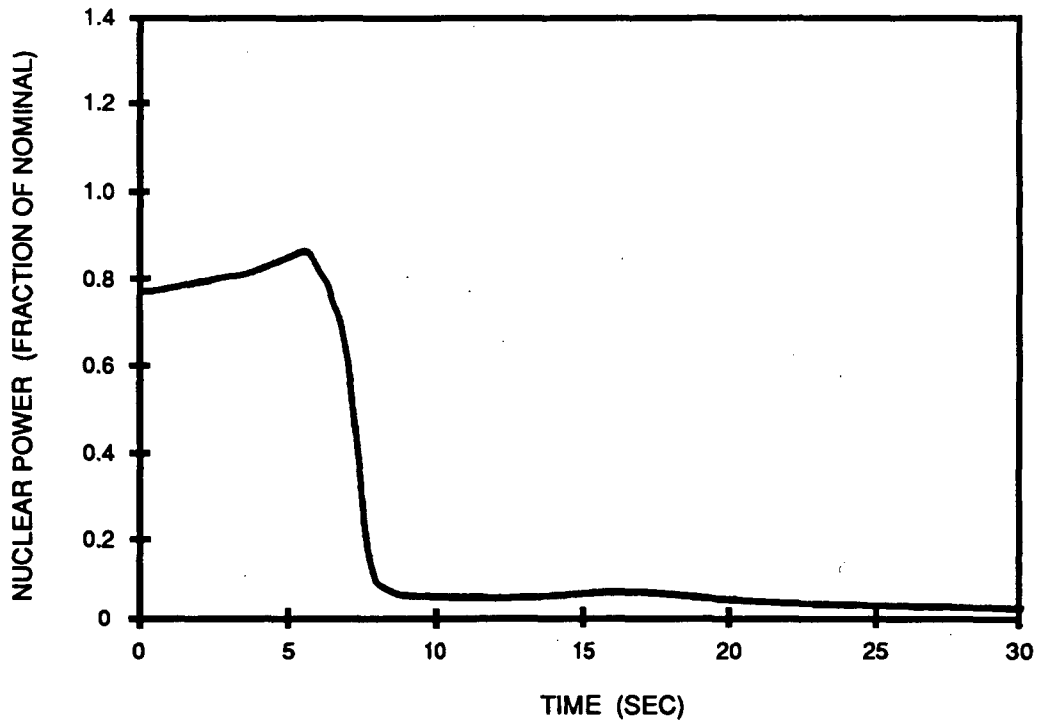
JULY 1993

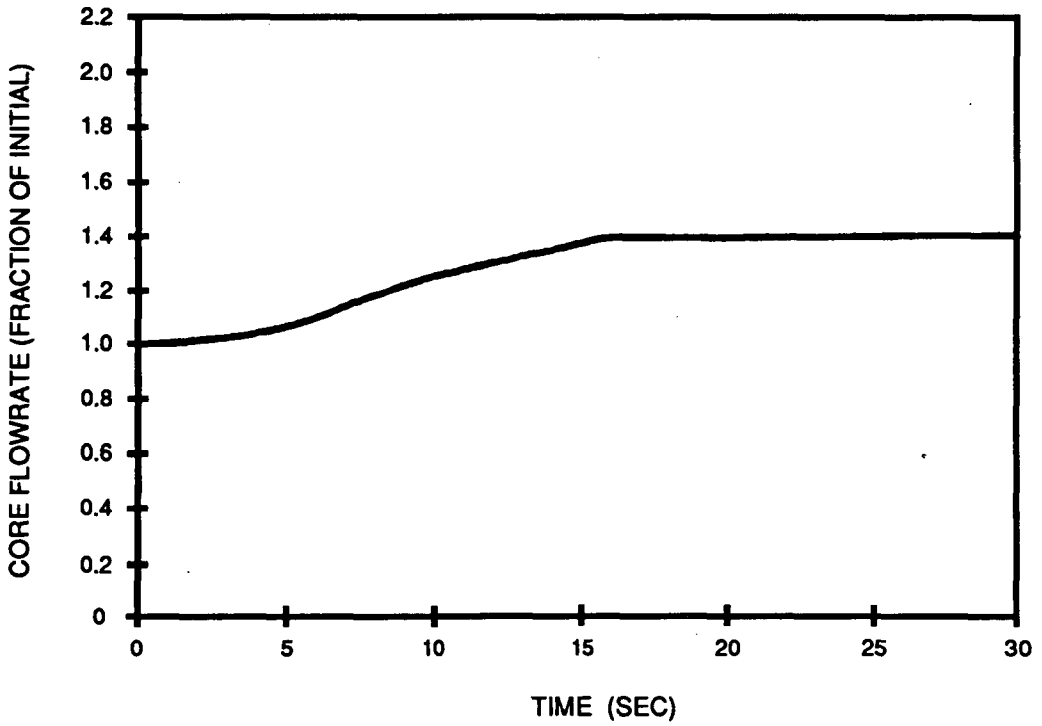


ZION STATION UFSAR

Figure 15.4-13
TRANSIENT RESPONSE TO A DROPPED ROD CLUSTER CONTROL ASSEMBLY AUTOMATIC ROD CONTROL

JULY 1993

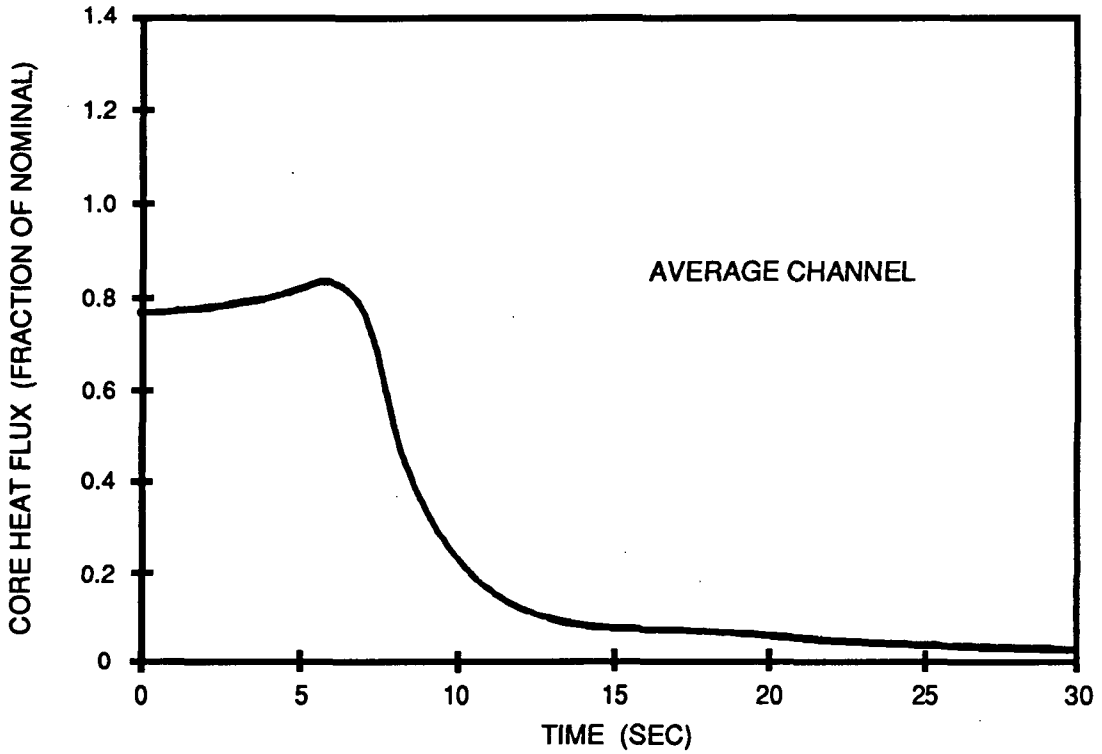


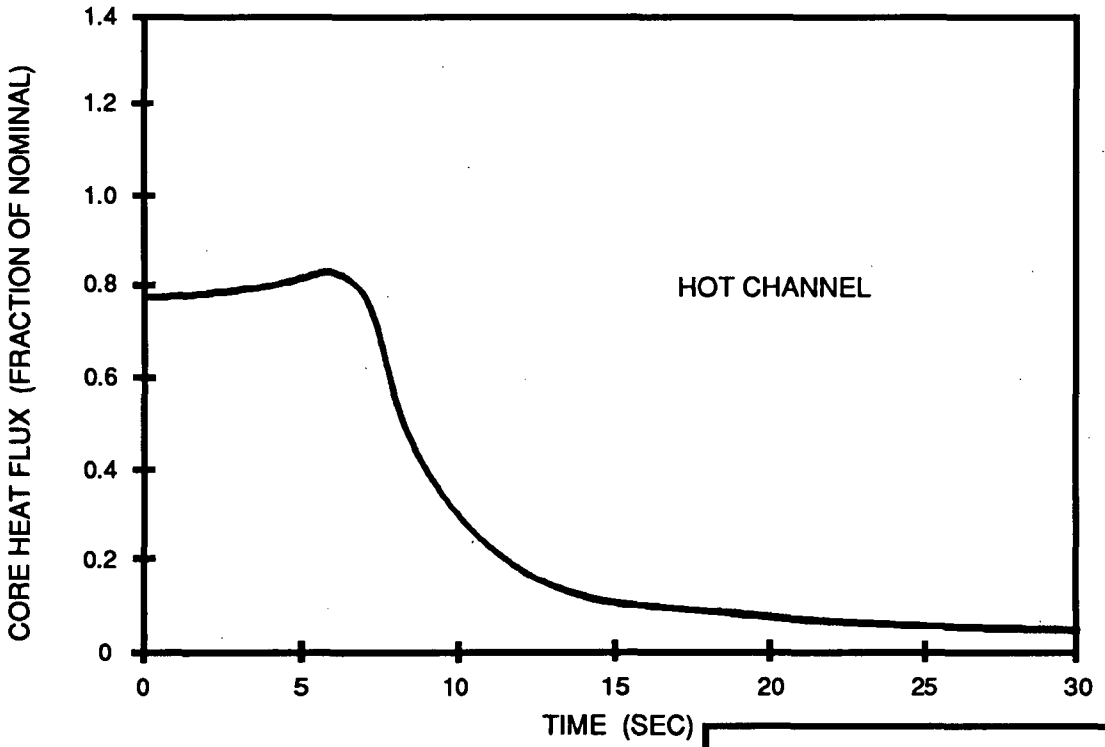


ZION STATION UFSAR

Figure 15.4-14
START UP OF AN INACTIVE LOOP
WITH LOOP STOP VALVES OPEN
NUCLEAR POWER AND CORE FLOW

JULY 1993

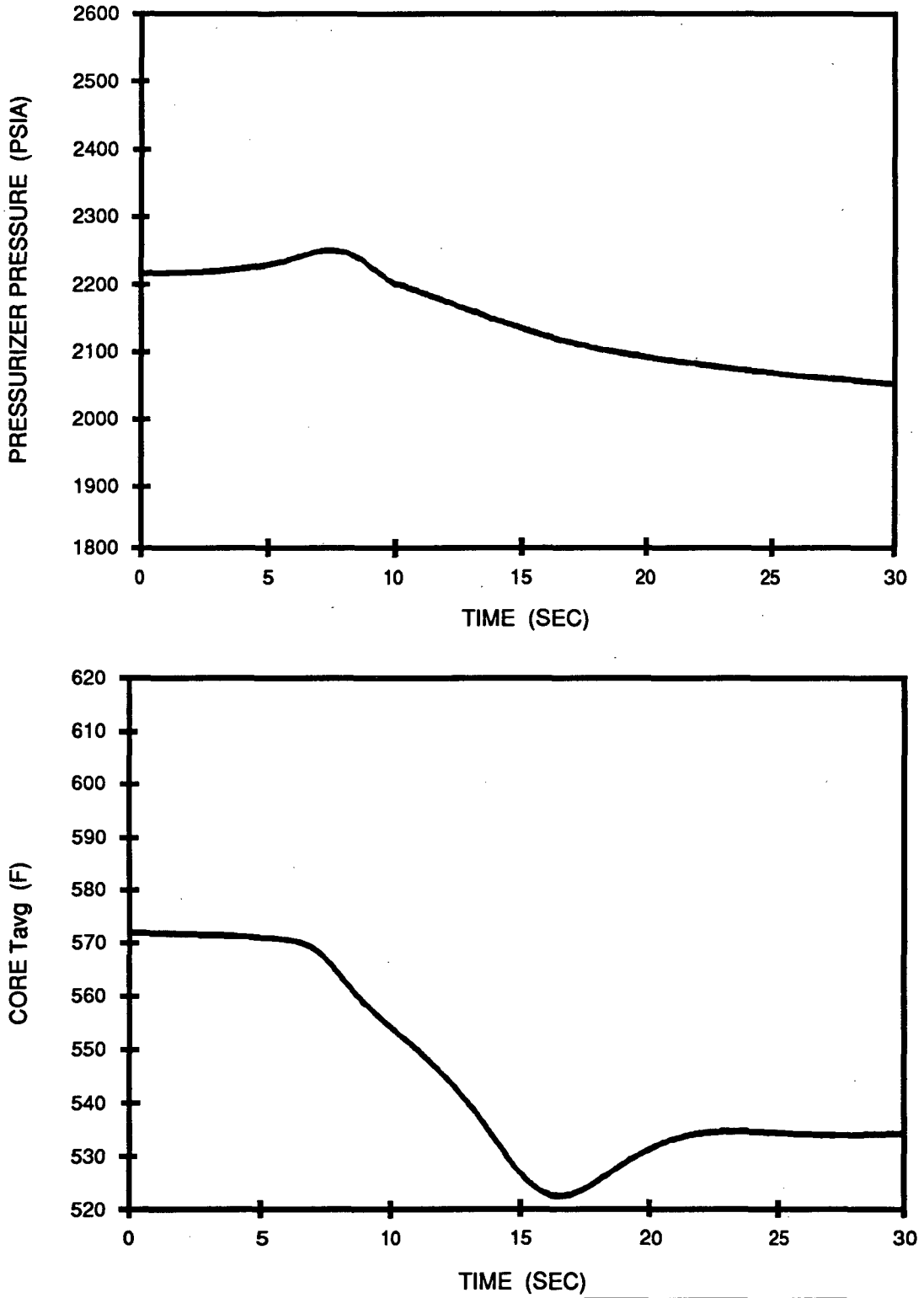




ZION STATION UFSAR

Figure 15.4-15
START UP OF AN INACTIVE LOOP
WITH LOOP STOP VALVES OPEN
AVERAGE CHANNEL HEAT FLUX AND
HOT CHANNEL HEAT FLUX

JULY 1993

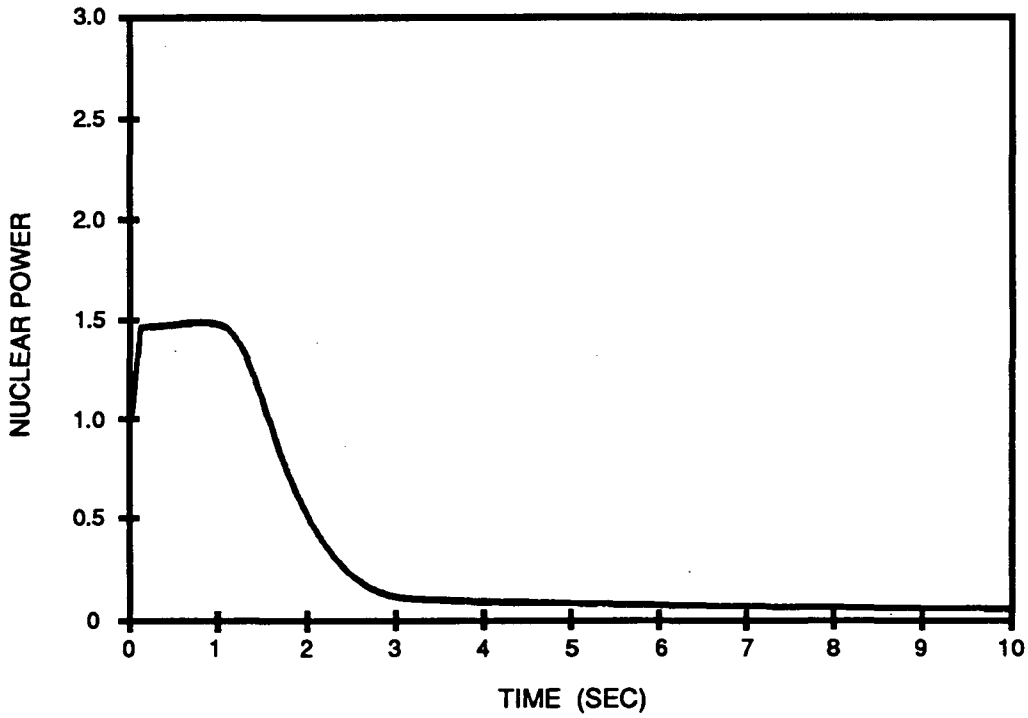


ZION STATION UFSAR

Figure 15.4-16

START UP OF AN INACTIVE LOOP WITH LOOP STOP VALVES OPEN PRESSURIZER PRESSURE AND CORE Tavg

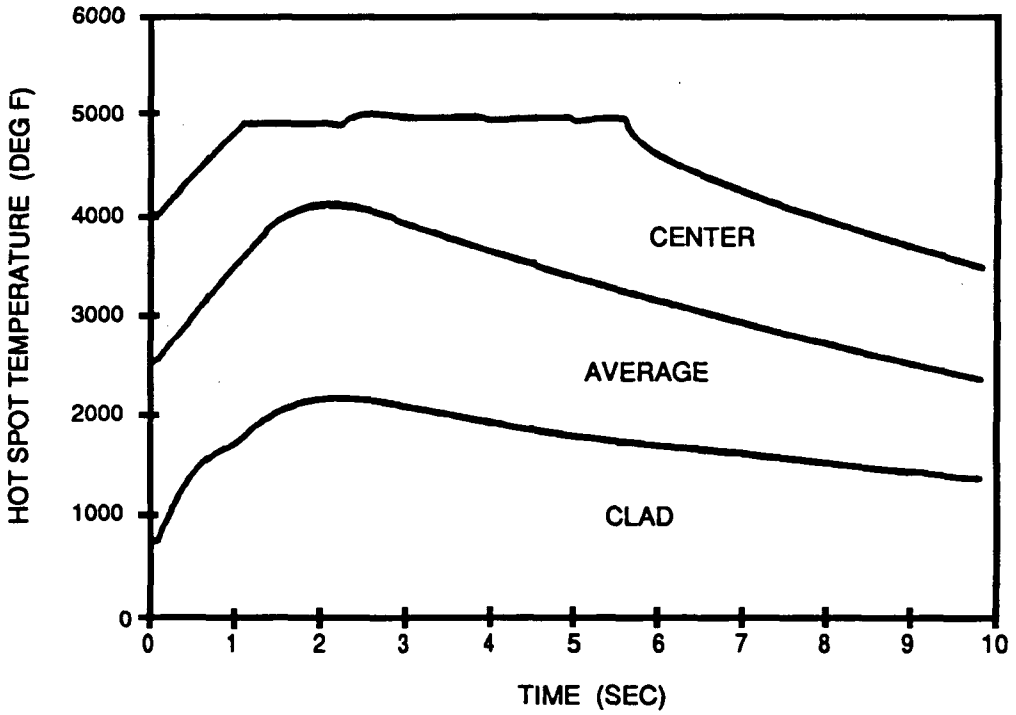
JULY 1993



ZION STATION UFSAR

Figure 15.4-17
ROD EJECTION
NUCLEAR POWER
BOL HOT FULL POWER

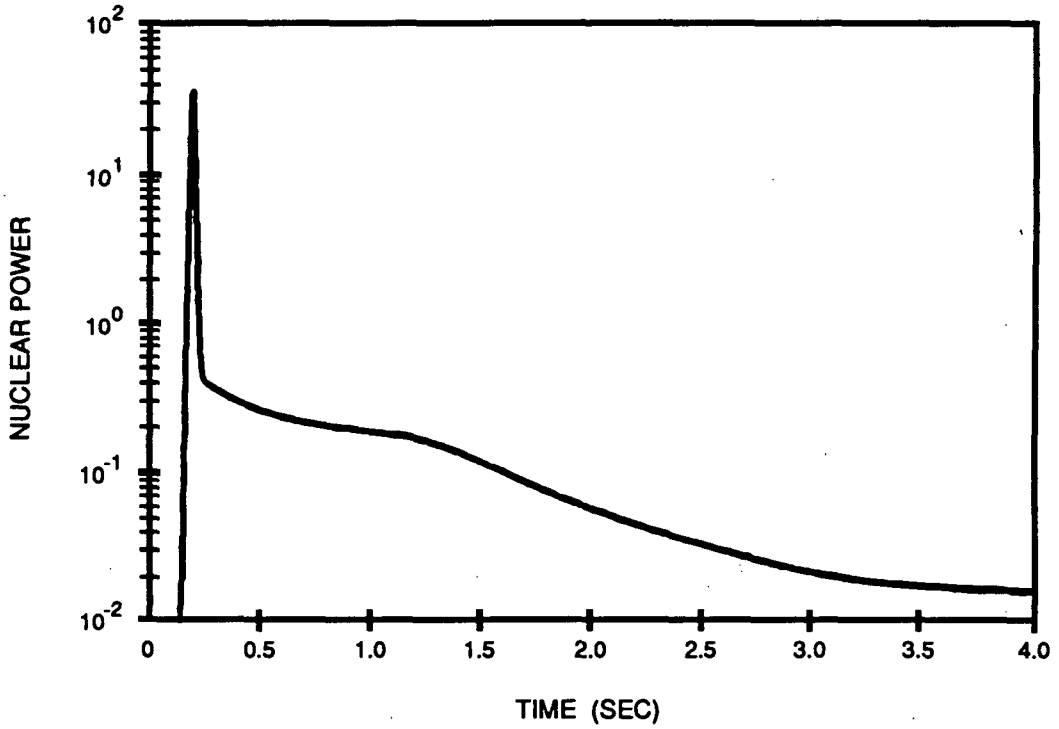
JULY 1993



ZION STATION UFSAR

Figure 15.4-18
ROD EJECTION
FUEL CENTERLINE, AVERAGE
AND CLAD TEMPERATURES
BOL HOT FULL POWER

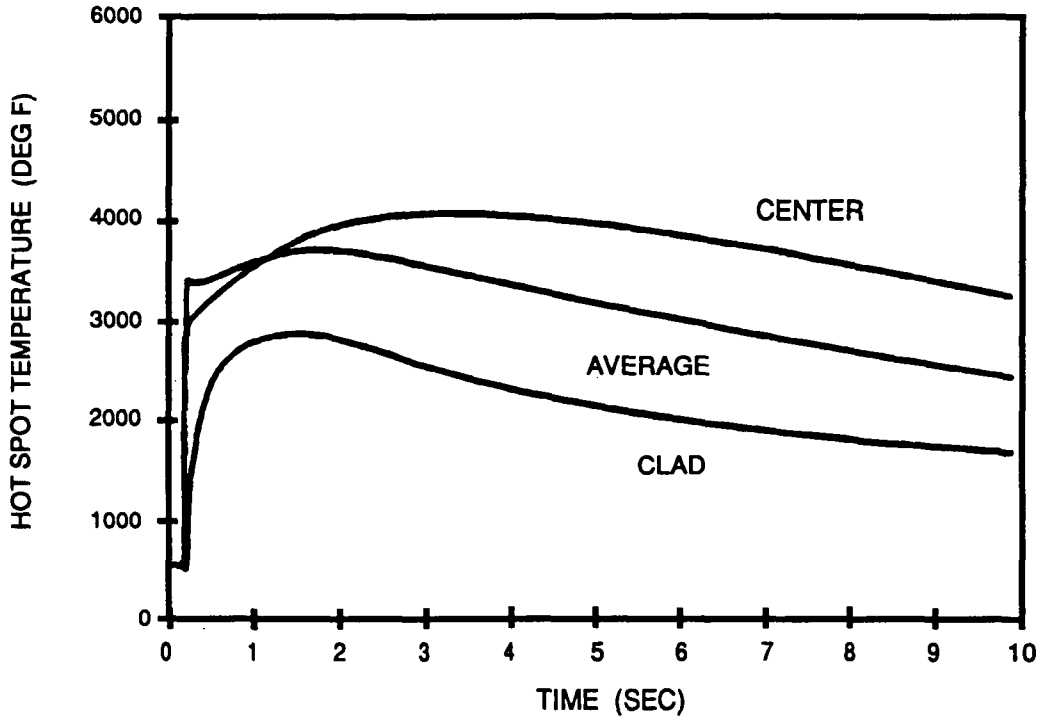
JULY 1993



ZION STATION UFSAR

Figure 15.4-19
ROD EJECTION
NUCLEAR POWER
EOL HOT ZERO POWER

JULY 1993



ZION STATION UFSAR

Figure 15.4-20
ROD EJECTION
FUEL CENTERLINE, AVERAGE
AND CLAD TEMPERATURES
EOL HOT ZERO POWER

JULY 1993

- Figure 15.4-21 was Intentionally Deleted -

- Figure 15.4-22 was Intentionally Deleted -

- Figure 15.4-23 was Intentionally Deleted -

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- Figure 15.4-27 was Intentionally Deleted -

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent Operation of Emergency Core Cooling System During Power Operation

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.5.2 Chemical and Volume Control System Malfunction (or Operator Error) That Increases Reactor Coolant Inventory

Chemical and Volume Control System Malfunctions, which result in a decrease in boron concentration in the reactor coolant, are discussed in Section 15.4.5.

15.5.3 Boiling Water Reactor Transients

This section is not applicable to the Zion Station.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

15.6.2 Break in Instrument Lines or Other Lines From Reactor Coolant Pressure Boundary That Penetrate Containment

See Appendix 3B for details.

15.6.3 Steam Generator Tube Rupture

15.6.3.1 General

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with 1% of fuel rods defective. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the Reactor Coolant System (RCS). In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated relief valves (PORVs).

The steam generator tube material is Inconel 600, and, as the material is highly ductile, it is considered that the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin; this has been verified by operational experience.

The main objective of the operator is to determine that a steam generator tube rupture has occurred and to identify and isolate the faulty steam generator 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 faulty unit. The recovery procedure can be carried out in a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily. Consideration of the indications provided at the control board, together

with the magnitude of the break flow, leads to the conclusion that the isolation procedure can be completed within 30 minutes of accident initiation.

15.6.3.2 Accident Description

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, prior to plant trip, 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 steam generator is reduced due to the additional break flow which is now being supplied to that unit.

2. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer until a reactor trip signal is generated by low pressurizer pressure. The safety injection (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.

3. The secondary side radiation monitors will alarm, indicating a sharp increase in radioactivity in the secondary system.

4. The plant trip will automatically shut off the steam supply to the turbine. If offsite power is available, the condenser bypass valves open, permitting steam dump to the condenser. In the event of a coincident station blackout, the condenser bypass valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator safety valves and/or PORVs.

5. Following the plant trip, the continued action of AFW supply and borated SI flow (supplied from the refueling water storage tank (RWST)) provides a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser (or in the case of loss of offsite power, steam relief to atmosphere) is attenuated during the 30 minutes in which the recovery procedure leading to isolation is being carried out.

6. SI flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

15.6.3.3 Recovery Procedure

The immediately apparent symptoms of a tube rupture accident, such as falling pressurizer pressure and level and increased charging pump flow, are also symptoms of small steam breaks and loss-of-coolant accidents (LOCAs). It is therefore important for the operator to determine that the accident is the rupture of a steam generator tube in order to carry out the correct recovery procedure. The accident under discussion is uniquely identified by a condenser air ejector radiation alarm, a steam generator blowdown radiation alarm, and/or main steamline radiation alarm. In the event of a relatively large rupture, it will be clear soon after a trip that the level in one steam generator is rising more rapidly than in the others. This, too, is a unique indication of a tube rupture accident.

The operator carries out the following procedures which lead to isolation of the faulty steam generator and, subsequently, to unit cooldown.

1. Before the faulty steam generator is identified, feedwater flow is regulated to all steam generators to maintain a minimum onscale water level.

2. The operator verifies that condenser steam dump maintains the no-load T_{avg} and transfers steam dump to the steam pressure mode.

3. The faulty steam generator is identified by rising water level, radiation levels at the individual main steamlines, or individual steam generator blowdown samples. The main steam isolation valve (MSIV), MSIV bypass valve, and the blowdown valves associated with the faulty steam generator are closed (if the faulty steam generator is "A" or "D", the steam supply valve to the turbine-driven AFW pump is also closed). All feedwater flow to the faulty steam generator is then terminated. This completes isolation of the faulty steam generator.

4. After the faulted steam generator has been identified and isolated, cooldown of the RCS is initiated at the maximum rate achievable to a temperature equal to the saturation temperature corresponding to the faulted steam generator pressure minus 50°F. One of the following methods of cooldown is used:

   a. If offsite power and the condenser are available, steam is dumped to the condenser from the intact steam generators by manual control of the steam dump in the steam pressure mode; or

   b. If offsite power or the condenser is not available, steam is vented from the intact steam generators through the steam generator atmospheric relief valves.

5. After the reactor coolant temperature has been reduced to the temperature mentioned in Step 4 above, the RCS is depressurized to a value equal to the faulted steam generator pressure by one of the following methods:

   a. If the reactor coolant pumps (RCPs) are operating, normal pressurizer spray is used;

   b. If the RCPs are not operating, a pressurizer PORV is briefly opened; or

   c. If neither the normal pressurizer spray flow or the pressurizer PORVs are available, auxiliary spray flow from the charging flow header is used.

6. As water level returns in the pressurizer, the RCS subcooling has been verified to be greater than 15°F, RCS pressure is stable or increasing (after verified closure of the spray valves and PORVs), and secondary heat sink availability is assured, then all Emergency Core Cooling System (ECCS) pumps not needed for normal charging and RCP seal injection are placed in standby.

7. When the above actions have been carried out as necessary, the reactor systems necessary for a normal shutdown are established and plant shutdown to the cold condition is begun.

8. After the Residual Heat Removal (RHR) System is in operation, the condensate accumulated in the secondary system can be examined and processed through the Waste Disposal System (WDS).

15.6.3.4 Results

In determining the mass transfer from the RCS through the broken tube, the following assumptions were made:

1. Plant trip occurs automatically as a result of low pressurizer pressure.

2. Following the initiation of the SI signal, both centrifugal charging pumps are started and continue to deliver flow for at least 30 minutes. The emergency instructions for a tube rupture accident indicate that the operator should switch off one pump when the accident has been identified, level returns in the pressurizer, and RCS subcooling is being maintained.

3. After plant trip, the break flow reaches equilibrium at the point where incoming SI flow is balanced by outgoing break flow as shown in

Figure 15.6-1. The resultant break flow persists from plant trip until 30 minutes after the accident.

4. The steam generators are controlled at the safety valve setting rather than the PORV setting.

5. The operator identifies the accident type and terminates break flow to the faulty steam generator within 30 minutes of accident initiation.

The above assumptions lead to a conservative upper bound value of 125,000 pounds for the total amount of reactor coolant transferred to the faulty steam generator.

Because the iodine is soluble in water, considerable separation will occur. Assuming equilibrium is reached between the liquid and vapor iodine concentrations, the effective partition factor is 0.1. This factor is defined as the ratio of the amount of iodine per unit mass of steam to the amount of iodine per unit mass of liquid (see Reference 1).

Therefore, in the case of operator action at 30 minutes, leakage of 125,000 pounds of reactor coolant results in 172 curies of equivalent iodine being released to the steam generator. With an effective partition factor of 0.1 and a value of χ/Q of 8.9×10^{-4}, the activity released to the atmosphere results in a thyroid dose at the site boundary of less than 23 rem. Thus, the dose is well under the limits of 10CFR100 even should the operator delay in taking action when warned by alarms and instruments.

Commonwealth Edison Company has analyzed the steam generator tube rupture incident without a coincident loss of outside power on the basis of 10CFR20 limits and criteria. This analysis is summarized below.

The whole body site boundary dose is limiting. The 10CFR20 limit is 0.5 rem. The calculation is summarized below:

$$WBD = (1/2)(\overline{E})(A)(V)(\chi/Q)(3.7 \times 10^{10})(1.33 \times 10^{-11})$$

where:

WBD = whole body dose
$\overline{E}$ = average energy per disintegration
A = coolant activity for 1% failed fuel
V = volume of carryover
χ/Q = defined for 10CFR20 analysis as the annual average site boundary value

The results of this analysis show the site boundary whole body dose to be approximately 3.5 mrem.

The two hour thyroid dose calculation is summarized below:

$$D_{inh} = (Q_R)\ (B)\ (DCF)\ (\chi/Q)\ (SGPF)\ (CPF)\ (700)$$

where:

Q_R	=	total iodine release in curies to the secondary side of the steam generator
B	=	breathing rate
DCF	=	weighted dose conversion factor for the mix of iodine isotopes
χ/Q	=	defined for 10CFR20 analysis as the annual average site boundary value
SGPF	=	iodine release fraction for the steam generator
CPF	=	iodine release fraction for the condenser
700	=	reconcentration factor for iodine

The results of the above analysis for 1% failed fuel show the thyroid dose to be 0.292 mrem.

It can be seen that the doses resulting from a steam generator tube rupture added to a previously occurring steam generator design basis leak are well within 10CFR20 limits when analyzed using 10CFR20 criteria. Even variations in χ/Q of a factor of 10 do not change this picture. The thyroid dose calculations are even more conservative in that reconcentration is assumed from a cow at the site boundary. In view of the land usage, this is highly unlikely.

15.6.3.5 Conclusions

There is ample time available to carry out the recovery procedure presented in Section 15.6.3.3 such that isolation of the affected steam generator is established before water level rises into the main steam pipes. The available time scale is improved by the termination of feedwater flow to the faulty steam generator and the regulation of pressurizer water level with only the required ECCS pumps. Normal operator vigilance therefore assures that excessive water level will not be attained. Offsite doses resulting from this incident have been shown to be well within 10CFR100 limits.

Based on a review of the historical data on steam generator leak propagation, the probability of a tube rupture is so low as to make the use of 10CFR20 limits unnecessarily restrictive. The postulated combined tube rupture and blackout is not credible in view of the above analysis. Therefore, there appears to be no need for limits on primary or secondary activity in terms of concern for the effects of steam generator tube

ruptures in the light of 10CFR20 limits. The various doses resulting from design basis operation with a steam generator leak are presented in Section 15.7.1.4.

15.6.4 Spectrum of Boiling Water Reactor Steam Piping Failures Outside of Containment

This text is not applicable to the Zion Station.

15.6.5 Reactor Coolant System Pipe Rupture (Loss-of-Coolant Accident)

A comprehensive safety analysis of postulated pipe ruptures within the RCS boundary has been performed. This analysis has included cases of the LOCA resulting from a broad spectrum of small and large pipe ruptures including the design basis accident (DBA) case of the double-ended break of the largest RCS pipe.

The objective of the analysis has been to determine the conditions of the RCS, core, and Containment in the event of a postulated LOCA, and to determine that the various ECCSs have the capability to control each LOCA, including the DBA.

15.6.5.1 General

15.6.5.1.1 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 shutdown 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 requirements of 10CFR50 (see Reference 2). The requirements are:

1. Peak clad temperatures do not exceed 2200°F;

2. The amount of cladding that chemically reacts with the coolant does not exceed 1% of the total Zircaloy cladding surrounding the fuel, excluding the clad surrounding the plenum volume;

3. Oxidation of the cladding does not exceed 17% of the original cladding thickness, which precludes embrittlement problems;

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling; and

5. After initial operation of the ECCS, core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived isotopes remaining in the core.

15.6.5.1.2 Accident Description

A LOCA would result from a rupture of the RCS or of any line connected to that system up to the first closed valve. The charging pumps have the capability to make up for leakage resulting from ruptures of a small cross section, thus permitting an orderly shutdown. The coolant released would remain in the Containment.

For a postulated large break, a reactor trip is initiated when the pressurizer low pressure setpoint is reached while the SI signal is actuated by pressurizer low pressure. The reactor trip and SI 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 supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to the delayed fission product decay; and

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

Before the reactor trip occurs, the reactor is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat, heat from hot internals and the vessel is transferred to the RCS fluid and then to the secondary system. The secondary 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 AFW flow by starting the AFW pumps. If offsite power is available, the steam is dumped to the condenser; if not, the steam is dumped to the atmosphere. The secondary flow aids in the reduction of RCS pressure. When the RCS pressure falls below 600 psia, the accumulators begin to inject borated water.

The RCPs are assumed to be tripped at the initiation of the accident due to loss of offsite power and the effects of pump coastdown are included in the blowdown analyses.

15.6.5.2 Thermal Analysis

The analysis specified by 10CFR50.46 is presented in this section. The time sequence of events for the LOCA is provided in Table 15.6-40. The results of the LOCA analysis are shown in Table 15.6-1 and show compliance

with the acceptance criteria. The highest peak clad temperature was calculated for a double-ended cold leg guillotine (DECLG) break with a Moody discharge coefficient (C_D) of 0.4.

The analysis is based on reactor conditions shown in Table 15.6-1 and is applicable to both Unit 1 and 2. The analytical techniques used are in compliance with Appendix K of 10CFR50 and are described in the topical report (see Reference 3). The detailed description of the LOCA analysis is given in References 6, 7, 9, 10, 11 and 37. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The BASH (Reference 37), BART-A1 (Reference 6), LOCTA-IV (Reference 7), COCO (Reference 9), SATAN-VI (Reference 10), and WREFLOOD (Reference 11) codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The conditions of Table 15.6-1 reflect four-loop operation. Containment parameters used in the analysis are given in Table 15.6-2.

The method of analysis to determine peak clad temperature is divided into two types of analysis: 1) large break LOCA, and 2) small break LOCA. The method of analysis for large and small break LOCA is described below and results are given.

15.6.5.2.1 Large Break Analysis

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. The SI Signal is actuated when the appropriate setpoint is reached. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling (DNB) is calculated, consistent with Appendix K of 10CFR50 (see Reference 2). Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

When the RCS pressure falls below 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. The termination of bypass is defined as the commencement of a continuous flow of water down the downcomer into the lower plenum.

For the large break sensitivity studies (see Reference 4), a break spectrum was performed utilizing a DECLG with various values of discharge coefficient (C_D) and a range of split-type break sizes ranging from a 1.0 ft^2 area to a full double-ended area of the cold leg. This study determined that the DECLG type break was both the most limiting type and location; therefore, only the spectrum of DECLG cases were analyzed for Zion Station.

15.6.5.2.1.1 Method of Analysis, Large Break

The calculation of peak clad temperature is performed by modeling the hottest fuel assembly (from the reactor core) in the LOCBART code.

The Improved Fuel Performance model, described in Reference 38, generates the initial fuel rod conditions input to LOCBART.

The hot fuel assembly is subdivided into three regions:

1. The hottest rod,

2. Adjacent rod to the hottest rod, and

3. The average fuel channel in the hot assembly.

The peak clad temperature occurs on the hottest rod. The LOCBART code is used in conjunction with other computer codes which determines necessary thermal-hydraulic boundary conditions for the LOCBART fuel rod heatup analysis.

The large break LOCA transient can be conveniently divided into three periods: blowdown, refill, and reflood. Also, three physical parts of the transient are analyzed for each period: the thermal-hydraulic transient in the RCS, the containment pressure and temperature, and the fuel and clad temperatures of the hottest rod. These considerations lead to the use of a system of computer codes designed to model the LOCA transient.

The LOCBART code is used throughout the transient to compute fuel and clad temperatures in the hottest rod. The COCO code is used to model the containment pressure transient. The SATAN-VI code evaluates the thermal-hydraulic transient during blowdown while the WREFLOOD code computes, using output from the SATAN-VI code, the time to bottom of core recovery (BOC), RCS conditions at BOC and mass and energy release from the break during the reflood phase of the LOCA. Since the mass flow rate to the Containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. The BOC conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the BASH code. Data from both the SATAN-VI code and the WREFLOOD code out to BOC are input to the LOCBART code which calculates core average conditions at BOC for use by the BASH code.

BASH provides a more realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by WREFLOOD. Figure 15.6-2A illustrates how BASH has been substituted for WREFLOOD in

calculating transient values of core inlet flow, enthalpy, and pressure for the detailed fuel rod model, LOCBART. The BASH code provides a much more sophisticated treatment of steam/water flow phenomena in the RCS during core reflood. A more dynamic interaction between core thermal-hydraulics and system behavior is expected, and experiments have shown this behavior. The BART code has been coupled with a loop model to form the BASH code and BART provides the entrainment rate for a given flooding rate. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated inlet flow is used by BART to calculate a new entrainment rate to be fed back to the loop code. This process of transferring data between BART, the loop code, and back to BART, forms the calculational process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a more dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

In the 1981 + BASH ECCS model, the cladding heat-up transient is calculated by LOCBART which is a combination of the LOCTA-IV code with BART. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART, the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

Figure 15.6-2A shows the interaction of the 1981 + BASH large break model and the relationship of the computer codes to the LOCA sequence of events.

A detailed description of the various aspects of the LOCA analysis is given in WCAP-8339 (see Reference 3). This document describes the major phenomena modeled, the interfaces among the computer codes and features of the codes which maintain compliance with the acceptance criteria. The individual codes are described in detail in separate reports (see References 7, 9, 10, and 11).

Code modifications are specified in Reference 12.

15.6.5.2.2 Small Break Loss-of-Coolant Accident Reanalysis

15.6.5.2.2.1 Method of Analysis

The Zion Station small break LOCA analysis was performed using the Westinghouse ECCS Small Break Evaluation model (Reference 39) which utilizes the NOTRUMP (References 40 and 41) and LOCTA-IV computer codes. The small break was analyzed assuming a full core of VANTAGE 5 with Intermediate Flow Mixers (IFMs) fuel to determine the calculated peak cladding temperature. This is consistent with the methodology of Reference 43. VANTAGE 5 with IFMs fuel, as analyzed for Zion Station, has all of the features presented in Reference 43 and also utilizes a low pressure drop structural grid for the mid-grids. Figure 15.6-2B shows the interaction of the computer codes used to evaluate the small break cases.

The time sequence of events for the small break LOCA analysis is shown in Table 15.6-41.

The core power level utilized in the small break LOCA analysis was 102% of 3250 MWt, the core licensed power. The peak linear power, core power and peaking factor used in the analyses are also given in Table 15.6-43. Additional assumptions for the small break LOCA analysis were:

1. operation at a loop flow rate of 86,800 gpm;

2. 20% steam generator tube plugging (SGTP) uniform among the four steam generators per plant (the 20% SGTP value is composed of a 15% plugging limit with the additional 5% allocated to offset the concern for steam generator tube collapse under LOCA and seismic loads);

3. 40 second delay in delivery of pumped ECCS assuming loss of offsite power coincident with reactor trip;

4. main steam safety valves set 3% above the Technical Specification setpoint value and requiring an additional 3% accumulation before being assumed fully open; and

5. the flow of the main steam safety valves was set to 50% of the rated capacity.

Figure 15.6-3 presents the hot rod power shape utilized to perform the small break analysis. This actual power shape was chosen because it provides that distribution of power versus core height which will maximize peak cladding temperature given an upper limit of +13% for the Technical Specification axial offset.

A spectrum of cold leg breaks covering the range of 2, 3, and 4-inch breaks was analyzed in order to determine the most limiting break size.

The NOTRUMP and LOCTA-IV computer codes are used to perform the analysis of LOCAs due to small breaks in the RCS. The NOTRUMP computer code, approved for this use by the Nuclear Regulatory Commission (NRC), is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the flow through the reactor core and break. This code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these new features are the utilization of nonequilibrium thermal calculation in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stack fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA ECCS evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611 (Reference 47).

In NOTRUMP, the RCS is subdivided into fluid filled control volumes (fluid nodes) and metal nodes interconnected by flowpaths and heat transfer links. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied to these nodes. The broken loop is modeled explicitly, and the intact loops are lumped into a second loop.

In the NOTRUMP model, the reactor core is represented as a vertical stack of heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables the 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 LOCA.

Clad thermal analysis is performed with the LOCTA-IV computer code which uses as input the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input. For all computations, the NOTRUMP and LOCTA-IV calculations were terminated slightly after the time the core mixture level returned to the top of the core following core uncovery.

A schematic representation of the computer code interfaces is given in Figure 15.6-2B.

15.6.5.2.3 Results

15.6.5.2.3.1 Results of Large Break Analysis

The large break LOCA analysis assumed fuel having all the features presented in Reference 43, applied to a 15x15 fuel lattice, and utilizing a low pressure drop structural grid for the mid-grids. Fuel having all of these features is referred to as VANTAGE 5 with IFMs. Additionally, the results of fuel hydraulic testing reported in Reference 44 were

incorporated into the analysis. The time sequence of events for the breaks analyzed is given in Table 15.6-40. Table 15.6-1 presents the peak clad temperature and hot spot metal reaction for a range of Moody discharge coefficients for DECLG break. The DECLG break location was determined to be the limiting location for peak clad temperature from sensitivity studies reported in Reference 4. The analysis of the LOCA is performed at 102% of design power. The peak linear power, core power and peaking factor used in the analyses are also given in Table 15.6-1. Additionally, the analysis assumed that:

1. the RCS was operating at a loop flow rate of 86,800 gpm;

2. 20% SGTP uniform among the four steam generators (composed of a 15% SGTP allowance and 5% to address the concern for steam generator tube collapse under seismic and LOCA loads);

3. a 31 second delay in delivery of pumped ECCS assuming loss of offsite power coincident with the break;

4. pumped ECCS flow for the RHR, Safety Injection, and Centrifugal Charging pumps was modeled to inject into the accumulator lines (Reference 45); and

5. a 5% reduction in pumped ECCS flow rates was assumed.

A specific requirement of LOCA analysis (10CFR50, Appendix K) is that a range of power distributions shall be studied and the distribution resulting in the most severe calculated consequences shall be used in the analysis. The Westinghouse design method of using a chopped cosine power shape was followed in performing the large break LOCA analysis.

Figures 15.6-4 through 15.6-26 and Figures 15.6-62 through 15.6-93 present the transients for the principal parameters for the break case analyzed. The following items are noted:

Figures 15.6-4 - 6;	Quality, mass velocity and clad heat transfer coefficient for the hot spot/burst locations. Figures 15.6-4A - 6A are for the maximum ECCS case.
Figures 15.6-7 - 9;	Core pressure, break flow, and core pressure drop. The break flow is the sum of the flow rates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet.
Figures 15.6-10 - 12;	Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot/burst location. Figures 15.6-10A and 15.6-11A are for the maximum ECCS case.
Figures 15.6-13 - 14;	Downcomer and core water level during reflood and core exit velocity. Figures 15.6-13A and 15.6-14A are for the maximum ECCS case.
Figures 15.6-15 - 17;	ECCS flow rates, for both accumulator and pumped safety injection. Figure 15.6-17A is the pumped safety injection for the maximum ECCS case.
Figures 15.6-18 - 19;	Containment pressure and core power transient. Figure 15.6-18A is the containment pressure for the maximum ECCS case.
Figures 15.6-20 - 24;	Quality, mass velocity, cladding heat transfer coefficient, cladding temperature, and fluid temperature for the hot spot for VANTAGE 5 without IFMs.
Figures 15.6-25 - 26;	Break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break.

The calculated peak clad temperature was 2066.43°F for the VANTAGE 5 without IFMs design and 2065.12°F for the VANTAGE 5 with IFMs design. The hot spot metal-water reaction reached was 7.39% for the VANTAGE 5 without IFMs and 7.45% for VANTAGE 5 with IFMs, both of which are well below the embrittlement limit of 17%, as required by 10CFR50.46. The total core metal-water reaction is less than 1.0% for all breaks as compared with the 1% criterion of 10CFR50.46. Based on these results, a core coolable geometry was maintained for the hypothetical large break LOCA and therefore, long-term core cooling is assured by continued operation of the ECCS. Additionally, the analysis for the VANTAGE 5 with IFMs fuel was repeated assuming no single failure within the ECCS or ESF (Maximum ECCS) and the results were slightly less limiting than the results for the case assuming a single failure in the ECCS.

Several sensitivity cases are presented in Table 15.6-1 which were performed to determine the most limiting fuel type. Additionally, Westinghouse performed sensitivity studies to determine the transition core penalty. These studies determined that the 50°F penalty licensed in Reference 43 would continue to apply to transition cores utilizing VANTAGE 5 with IFMs fuel. The last planned use of fresh OFA fuel was Cycle 12. The Cycle 13 fuel will contain several VANTAGE 5 fuel features but without the low pressure drop structural grids or the IFM grids. The Cycle 13 fuel has been treated as OFA fuel in performing LOCA analyses and therefore VANTAGE 5 without IFMs is the limiting fuel type for the Cycle 13 design for both Units 1 and 2. VANTAGE 5 with IFMs fuel, having both the low pressure drop structural grids and IFMs, will be implemented for Cycle 14; therefore, the Cycle 14 core design will contain both VANTAGE 5 without IFMs fuel from Cycle 13 and fresh VANTAGE 5 with IFMs. A comparison of the fresh VANTAGE 5 with IFMs LOCA results to the VANTAGE 5 without IFMs fuel shows that starting with Cycle 14, VANTAGE 5 with IFMs will lead the core resulting in the highest calculated peak cladding temperature of 2065.12°F plus a 50.0°F transition core penalty for a net peak cladding temperature of 2115.12°F. This result leaves margin to the 10CFR50.46 limit of 2200°F. Once all fuel with the OFA structural grids has been removed and as long as fuel with the OFA structural grids is not re-introduced, the 50°F transition penalty can be removed.

15.6.5.2.3.2 Results of Small Break Analysis

The small break LOCA analyses were performed with the assumptions appearing in Table 15.6-43. Charging/SI flows including flow imbalance between the branch lines, and the required flow to the Reactor Coolant Pump seals were used in the small break LOCA analyses. Additionally, the results of fuel hydraulic testing reported in Reference 44 were incorporated into the analysis. The ECCS modeling concern reported in Reference 45 was not incorporated into the small break analysis since these breaks do not result in large amounts of accumulator injection and when accumulator injection does occur the associated flow rates are small. Thus, when accumulator injection does occur, the injection section pressure does not significantly increase above the RCS pressure.

The time sequence of events and parameters of interest for the 2, 3, and 4-inch breaks analyzed are shown in Table 15.6-41 and 15.6-42, respectively. Peak clad temperature for the limiting break (3-inch) was 1962.6°F occurring at an assembly average burnup of 500 MWD/MTU. The maximum local zirconium oxidation was 5.52% and the core wide oxidation was less than the 1% criterion. These results indicate that a coolable geometry was maintained for small break LOCAs and therefore, long-term core cooling is assured by continued operation of the ECCS. Analyses were performed to determine the most limiting time in life for the 3-inch break transient. These analyses confirmed that the 500 MWD/MTU burnup was the most limiting time in life. Figure 15.6-3 shows hot rod axial power shape. Figures 15.6-94 through 15.6-111 show RCS pressure, core mixture height, peak clad temperature, steam flow rate, rod film coefficient, and hot spot fluid temperatures versus time for each break.

15.6.5.2.4 Conclusions - Thermal Analysis

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will meet the acceptance criteria as presented in Section 15.6.5.1.1 for a total core peaking factor of 2.40 and an $F_{\Delta H}^{N}$ of 1.65. The small break LOCA analyses were performed at more limiting peaking factors of F_{Q}^{T} = 2.5 and $F_{\Delta H}^{N}$ of 1.7.

15.6.5.3 Core and Internals Integrity Analysis

The response of the reactor core and vessel internals under excitation produced by a simultaneous complete severance of a reactor coolant pipe and seismic excitation for typical four-loop plant internals has been determined. Reference 46 assessed the impact of the transition to VANTAGE 5 fuel on the analyses. It was concluded that the application of VANTAGE 5 fuel, with or without fuel assembly thimble plugging devices, will be compatible with the existing reactor internals and analyses. Furthermore, these proposed changes will not adversely impact the performance or integrity of this system and its components. A detailed description of the analysis, applicable to the Zion design, is given in Reference 15.

15.6.5.3.1 Reactor Internals Response Under Blowdown and Seismic Excitation

A LOCA may result from a rupture of reactor coolant piping. During the blowdown of the coolant, critical components of the core are subjected to vertical and horizontal excitation as a result of rarefaction waves propagating inside the reactor vessel.

For these large breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. (The subsequent refilling of the core by the ECCS uses borated water to maintain the core in a subcritical state.) Therefore, the main requirement is to assure effectiveness of the ECCS. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic

instability (buckling) or large deflections of the upper core barrel or both is the possible response of the barrel during hot leg blowdown. In addition to the above effects, the hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefaction wave propagates along a reactor inlet pipe arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a nonaxisymmetric

expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady-state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the Design Basis Earthquake (DBE) is postulated with the LOCA, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

15.6.5.3.2 Acceptance Criteria for Results of Analyses

The criteria for acceptability in regard to mechanical integrity analysis is that adequate core cooling and core shutdown must be assured. This implies the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts in addition to a stress criterion to assure integrity of the components.

15.6.5.3.2.1 Allowable Deflection and Stability Criteria

15.6.5.3.2.1.1 Upper Barrel

The upper barrel deformation has the following limits:

1. To ensure a shutdown and cooldown of the core during blowdown, the basic requirement is a limitation on the outward deflection of the barrel at the locations of the inlet nozzles connected to the unbroken lines. A large outward deflection of the barrel in front of the inlet nozzles, accompanied with permanent strains, could close the inlet area and stop the cooling water coming from the accumulators. (The remaining distance between the barrel and the vessel inlet nozzle after the accident must be such that the inlet flow area be approximately the same as that of the accumulator pipes.) Consequently, a permanent barrel deflection in front of the unbroken inlet nozzles larger than a certain limit, called the "no-loss of function" limit, could impair the efficiency of the ECCS; and

2. To ensure rod insertion and to avoid disturbing the rod control cluster guide structure, the barrel should not interfere with the guide tubes. This condition also requires a stability check to assure the barrel will not buckle under the accident loads.

15.6.5.3.2.1.2 Rod Control Cluster Guide Tubes

The guide tubes in the upper core support package house the control rods. The deflection limits were established from tests.

The deflection limitations within the fuel assembly are related to the stability of the thimbles in the upper end. The upper end of the thimbles must not experience stresses above the allowable dynamic compressive stresses. Any buckling of the upper end of the thimbles due to axial compression could distort the guide line and thereby affect the free fall of the control rod.

The deflection limitations in the upper package are related to the local vertical deformation of the upper core plate, where a guide tube is located. This deformation shall be below 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 inch. This limit will prevent the guide tubes from undergoing compression. For a plate local deformation of 0.150 inch, the guide tube will be compressed and deformed transversely to the upper limit previously established; consequently, the value of 0.150 inch is adopted as the no loss of function local deformation, with an allowable limit of 0.100 inch.

15.6.5.3.2.2 Allowable Stress Criteria

For this faulted condition, the allowable stress criteria is given by Table 15.6-6. This table defines various criteria based upon their corresponding method of analysis. To account for multi-axial stresses, the von Mises theory is also considered. Note that irradiation is not considered to effect the allowable stresses. Figure 15.6-30 graphically shows the allowable stress criteria.

15.6.5.3.3 Method of Analysis

15.6.5.3.3.1 Blowdown Model

BLODWN-2 is a digital computer program developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a LOCA (see Reference 16). This program applies to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs, such as WHAM (see Reference 17), which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. BLODWN-2 is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time. Although spatially one-dimensional conservation laws are employed, the code can be applied to describe

three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps, and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc. are considered.

BLODWN-2 predictions have been compared with numerous test data as reported in WCAP-7401 (see Reference 18). It is shown that the BLODWN-2 digital computer program gives good agreement in both the subcooled and the saturated blowdown regimes.

15.6.5.3.3.2 FORCE Model for Blowdown

BLODWN-2 evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. These pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE which utilizes a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which FORCE calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element;

2. Flow stagnation on and unrecovered orifice losses across the element; and

3. Friction losses along the element.

Input to the code, in addition to the BLODWN-2 pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The mechanical analysis has been performed using conservative assumptions in order to obtain results with extra margin. Some of the most significant are:

1. The mechanical and hydraulic analysis has been performed separately without including the effect of the water-solid interaction. Peak pressures obtained from the hydraulic analysis will be attenuated by the deformation of the structures;

2. When applying the hydraulic forces, no credit is taken for the stiffening effect of the fluid environment which will reduce the deflections and stresses in the structure; and

3. The multimass model described below is considered to have a sufficient number of degrees-of-freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

15.6.5.3.3.3 Vertical Excitation Model for Blowdown

For the vertical excitation, the reactor internals are represented by a multimass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also incorporated in the multimass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb type friction is assumed in the event that sliding between the rods and the grid fingers occurs. Figure 15.6-31 shows the spring-mass system used to represent the internals. In order to obtain an accurate simulation of the reactor internals response, the effects of internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rods motion relative to the grids, and preloads in hold down springs have been incorporated in the analytical model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs. Table 15.6-7 lists nomenclature of the various masses, springs, etc.

The appropriate dynamic differential equations for the multimass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program which computes the response of the multimass model when excited by a set of time dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements, and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe ruptures is analyzed. The forcing functions used in the study are obtained from hydraulic analyses of the pressure and flow distribution around the entire RCS as caused by double ended severance of a RCS pipe.

15.6.5.3.3.4 Vertical Excitation Model for Earthquake

As shown in Reference 15, the reactor internals are modeled as a single-degree-of-freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support is increased by an amplification due to the building-soil interaction.

15.6.5.3.3.5 Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

15.6.5.3.3.5.1 Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse. The barrel is then analyzed for dynamic buckling using these conditions and the following conservative assumptions: (a) the effect of the fluid environment is neglected (water stiffening is not considered); and (b) the shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a nonaxisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1. The upper core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower core barrel weldment without taking credit for the supports at the barrel midspan offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel;

2. The upper core barrel is analyzed as a shell with four variable sections to model the support flange, upper barrel, reduced weld section, and a portion of the lower core barrel; and

3. The barrel with the core and thermal shield, is analyzed as a beam fixed at the top and elastically supported at the lower radial support and the dynamic response is obtained.

15.6.5.3.3.5.2 Guide Tubes

The dynamic loads on rod cluster control assembly (RCCA) guide tubes are more severe for a LOCA caused by hot leg rupture than for an accident by cold leg rupture since the cold leg break leads to much smaller changes in the transverse coolant flow over the RCCA guide tubes. Thus, the analysis is performed only for a hot leg blowdown.

The guide tubes in closest proximity to the ruptured outlet nozzle are the most severely loaded. The transverse guide tube forces during the hot leg blowdown decrease with increasing distance from the ruptured nozzle location.

A detailed structural analysis of the RCCA guide tubes was performed to establish the equivalent cross-section properties and elastic end support conditions. An analytical model was verified both dynamically and statically by subjecting the RCCA guide tube to a concentrated force applied at the transition plate. In addition, the guide tube was loaded experimentally using a triangular distribution to conservatively approximate the hydraulic loading. The experimental results consisted of a load deflection curve for the RCCA guide tube plus verification of the deflection criteria to assure RCCA insertion.

The response of the guide tubes to the transient loading due to blowdown may be found by utilizing the equivalent single-degree-of-freedom system for the guide tube using experimental results for equivalent stiffness and natural frequency.

The time dependence of the hydraulic transient loading has the form of a step function with constant slope front with a rise time to peak force of the same order of the guide tube fundamental period in water. The dynamic amplification factor in determining the response is a function of the ramp impulse rise time divided by the period of the structure.

15.6.5.3.3.5.3 Upper Support Columns

Upper support columns located close to the broken nozzle during hot leg break will be subjected to transverse loads due to cross flow.

The loads applied to the columns were computed with a similar method to the one used for the guide tubes; i.e., taking into consideration the increase in flow across the column during the accident. The columns were studied as beams with variable section and the resulting stresses were obtained using the reduced section modulus at the slotted portions.

15.6.5.3.3.6 Transverse Excitation Model for Earthquake

The reactor building with the reactor vessel support, the reactor vessel, and the reactor internals are included in this analysis. The mathematical model of the building, attached to ground, is similar to that used to evaluate the building structure. The reactor internals are mathematically modeled by beams, concentrated masses, and linear springs.

All masses, water, and metal are included in the mathematical model. All beam elements have the component weight or mass distribution uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components are attached uniformly, their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is also included as a distributed mass. Horizontal components are considered as a concentrated mass acting on the barrel. This concentrated mass also includes components attached to the horizontal members, since these are the media through which the

reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points.

The concentrated masses attached to the barrel represent the following:

1. the upper core support structure, including the upper vessel head and one-half of the upper internals;

2. the upper core plate, including one-half the thermal shield and the other half of the upper internals;

3. the lower core plate, including one-half of the lower core support columns;

4. the lower one-half of the thermal shield; and

5. the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional area is selected along with a value for Poisson's ratio. The fuel assembly moment of inertia is derived from experimental results by static and dynamic tests performed on fuel assembly modes. These tests provide stiffness values for use in this analysis.

The fuel assemblies are assumed to act together and are represented by a single beam. The following assumptions are made in regard to connection restraints. The vessel is pinned to the vessel support and part of the Containment Building. The barrel is clamped to the vessel at the barrel flange, spring connected to the vessel at the barrel flange, and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

The response spectrum method has been used in the calculation. After computing the transverse natural frequency and obtaining the normal modes of the complete structure, the maximum response is obtained from the superposition of the usual mode responses with the conservative assumptions that all the modes are in phase and that all the peaks occur simultaneously.

15.6.5.3.4 Conclusions - Mechanical Analysis

The results of the analysis, applicable to the Zion design, are presented in Table 15.6-8 and Table 15.6-9. These tables summarize the maximum

deflections and stresses for blowdown, seismic, and blowdown plus seismic loadings.

The stresses due to the DBE (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

These results indicate the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion with the exceptions shown in Table 15.6-8. It can be seen in the table that 54 of the 61 guide tubes are below the "no-loss of function" limit. For those guide tubes deflected above the "no-loss of function" limit, it must be assumed that the rods will not drop. However, the conclusion reached is that the core will shut down in an orderly fashion due to the formation of voids, and this orderly shutdown will be aided by the great majority of rods that do drop.

15.6.5.4 Containment Integrity Evaluation

15.6.5.4.1 Method of Analysis

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer codes, COCO or CONTEMPT4/MOD5 (Reference 8). The analytical model is restricted to the containment volume and structure. Transient phenomena within the RCS affect containment conditions by means of convective mass and energy transport through the pipe break.

The model may be divided into three parts: Blowdown, when the system pressure drops from 2250 psia to containment pressure; Refill, when the vessel inventory is increased to the bottom of the core; and Reflood, where the water level moves into the core. A brief description of the calculation model used for blowdown (SATAN code) is included in Section 15.6.5.4.2. Calculations for the refill period have been minimized by making the conservative assumption that the bottom of core recovery occurs immediately at the end of blowdown.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into two systems. The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of

conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the codes.

The following are the major assumptions made in the analysis:

1. Discharge mass and energy flow rates through the RCS break are established from the coolant blowdown and core thermal transient analysis (described in the preceding paragraphs);

2. For COCO, the discharge flow separates into steam and water phases at the break point. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the Containment. In the Appendix 15D CONTEMPT4 model, the entering blowdown is instantaneously mixed with the vapor region. The amount of liquid in the atmosphere is then determined based on the total compartment conditions. Then the liquid in the atmosphere is transferred to the compartment pool region;

3. Homogeneous mixing is assumed. The steam-air mixture and the water phase have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and the water phase; and

4. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

During the transient, there is energy transfer from the steam-air and water systems to the internal structures and equipment within the shell.

Provision is made in the computer analysis for the effects of several engineered safety features, including internal spray, fan coolers, and recirculation of sump water. The heat removal from containment steam-air phase by internal spray is determined by allowing the spray water temperature to rise to the steam-air temperature.

15.6.5.4.2 Available Energy Sources

The amount of mass and energy carried into the Containment during blowdown is calculated by the SATAN computer code as adjusted to reflect conservatively high core heat transfer coefficients. The SATAN code is used with a break in the appropriate piping. All accumulators inject except for the cold leg break, where one accumulator is spilled to the containment floor. The following is a summary of all the energy sources potentially available for transfer to the Containment for a LOCA:

1. Reactor coolant energy;
2. Accumulator energy (mixes with RCS);
3. Initial core stored energy;
4. Core internals metal energy;

5. Reactor vessel metal (below vessel nozzles);
6. Core power generation (shutdown energy and decay heat); and
7. Zr-H_2O reaction.

All the initial core stored energy and the power generated by the core during blowdown is available for transfer to the coolant, and thence to the Containment. The core stored energy has been re-evaluated using results from a detailed temperature analysis of the pellet, clad and gap. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad, gap and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, have been included in the thermal analysis methods to enable the determination of their net effects on temperature profiles. The temperature predictions of this detailed temperature analysis have been compared to inpile fuel temperature measurements and pellet melt radius data with good results.

A conservative value for core stored energy of 6.59 x 1.2 or 7.9 full-power seconds, which includes fuel densification and additional margin, was used in the analysis presented in Section 15.6.5.4.5. Moreover, core stored energy was based on a conservative value of 102% of the engineered safety features designed rated power level which is 3391 MWt. The margins cited above clearly indicate that the values employed in this analysis represent a conservative upper bound of the core stored energy. The initial metal sensible energy is transferred to the coolant by a time dependent temperature difference calculation. It should be emphasized that the energy transferred from the core to the coolant for the containment evaluation far exceeds that transferred from the core thermal evaluation. That is to say a conservatively high core heat transfer coefficient is used for the containment evaluation, while a conservatively low coefficient is used during core thermal evaluation.

The amount of heat released from the core over blowdown had been studied and an upper bound had been determined by a suitably conservative analysis. Specifically, an average channel heat release analysis was performed using the LOCTA code. The transition boiling correlation and DNB time were

modified to obtain a conservatively high release rate. The resulting upper bound value is used in the analysis presented in Section 15.6.5.4.5.

Any energy addition resulting from a Zr-H_2O reaction is also considered. The reaction energy reaches the Containment by transfer to coolant, while the recombination energy of the H_2 generated in the reaction is added directly to the steam-air mixture in the Containment. The hydrogen is assumed to burn as it is produced. This is not modeled for the cases discussed in Appendix 15D.

Finally, hot metal surfaces not cooled by SI water (reactor vessel above nozzles and steam generator tubes) are simulated as hot walls in contact with the containment steam-air mixture. A small heat transfer coefficient is employed to reflect actual conditions since these surfaces are covered by stagnant steam inside the RCS. This is not modeled in CONTEMPT4 since the hot metal energy is accounted for in the mass and energy release for the cases analyzed as discussed below.

The core reflood subsequent to blowdown has been analyzed considering the steam generators as an active heat source. The analysis' results were performed for several break locations; the one which yields the highest energy flow rate during the post blowdown period is the pump suction break. This is because of the following: for the cold leg break, all of the fluid leaving the top of the core passes through the steam generators and may become superheated. However, the flooding rate is limited to a relatively low value by the resistance of the pump in the broken loop. For a hot leg break, the flooding rate is not so restricted, but the majority of the fluid leaving the top of the core bypasses the steam generators and is not superheated. Thus, the steam generators add much less energy. The pump suction break, on the other hand, has the relatively high flooding rate combined with all of the fluid passing through the primary side of the steam generators. A discussion of these break analyses is presented in Section 15.6.5.4.5.

The following are some additional conservative assumptions used in the analysis:

1. The reactor power is based on operation at the maximum calculated power of 3391 MWt (which is 4.3% greater than the application at 3250 MWt) plus an allowance for calorimetric error;
2. The decay heat is based on power operation for an infinite time;
3. Coolant temperatures are the maximum levels attained in steady-state operation, including 4°F allowance for instrument error and deadband;
4. Gross system volumes are calculated from component dimensions, plus an allowance for margin on fluid volume; and
5. Pressurizer liquid inventory at the nominal full-power level plus an appropriate margin for instrument error and deadband.

15.6.5.4.3 Available Energy Sinks

15.6.5.4.3.1 Containment Structures

Provision is made in the containment pressure transient analysis for heat storage in both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite difference form accounts for transient conduction into and out of the node and temperature rise of the node. Table 15.6-11 is a summary of the containment structural heat sinks used in the analysis. The information used to compile this data was provided by Sargent & Lundy. All Sargent & Lundy input to the Westinghouse pressure study was physical information based upon the actual plant design. This information included such items as areas of exposed steel and concrete, density of steel and concrete and similar physical factors. None of this information requires a basis, since all information was of a factual nature.

The heat transfer coefficient to the containment surface is calculated by the code based primarily on the work of Tagami (see Reference 19). From this work it was determined that the value of the heat transfer coefficient can be modeled to increase parabolically to the peak value at the end of blowdown and then decrease exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air weight ratio.

It should be noted that this method is different than that presented in the Preliminary Facility Description and Safety Analysis Report. In that report, the heat transfer coefficients have been based on the work of Kolflat (see Reference 20). The revised method of calculation results in decreased heat transfer to the Containment structure during blowdown.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as:

$$\frac{\text{total coolant energy transferred into Containment}}{\text{(containment vessel volume) (time interval to peak pressure)}}$$

15.6.5.4.3.1.1 Heat Transfer Coefficient Calculations in COCO

From the above, steel's maximum value of h is calculated in COCO as:

Equation (1)

$$h_{max} = 75 \left(\frac{E}{t_p V}\right)^{0.60}$$

where:

h_{max} = maximum value of h (Btu/hr-ft^2-^{o}F)
t_p = time from start of accident to end of blowdown

V = containment volume (ft^3)
E = coolant energy discharged (Btu)

The parabolic increase to the peak value is given by:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{1/2} \qquad 0 \le t \le t_p \qquad \text{Equation (2)}$$

where:

h_s = heat transfer coefficient for steel (Btu/hr-ft^2-°F)
t = time from start of accident (sec)

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\ e^{-.05(t-t_p)} \qquad t > t_p \qquad \text{Equation (3)}$$

where:

h_{stag} = $2 + 50\,\chi \qquad 0 \le \chi \le 1.4$
h_{stag} = h for stagnant conditions (Btu/hr-ft^2-°F)
χ = steam-to-air weight ratio in Containment

For concrete, the heat transfer coefficient is taken as 40% of the value calculated for steel in both the COCO and CONTEMPT4 models.

15.6.5.4.3.1.2. Heat Transfer Coefficient Calculations in CONTEMPT4

For the CONTEMPT4/MOD5 code, the Tagami heat transfer correlation is represented by:

Equation (1a)

$$h_{max} = c\left(\frac{E}{t_p V}\right)^{0.62}$$

where: C=a constant equal to 0.607 for SI units

The Tagami heat transfer coefficient is linearly increased to its maximum value depending on the actual problem time, t, using:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)$$

The heat transfer coefficient following the peak Tagami value is abruptly changed to a value based on the Uchida correlation which is also a function of the containment air-to-steam mass ratio (Ref. 8) similar to the COCO model.

15.6.5.4.3.2 Containment Fan Coolers

The ability of the containment fan coolers to function properly in the accident environment was originally demonstrated by the Westinghouse computer code HECO. The most recent RCFC performance information is given in Reference 14; however, the calculations are similar to those that follow. The code determines the plate-fin cooling coil heat removal rate when operating in a saturated steam-air mixture.

In the code, a mass flowrate of cooling water is first established. This determines the tube inside film coefficient. Next, the resistance to heat transfer between the cooling water and the outside of the fin collars is

computed; including inside film coefficient, fouling factor[1], tube radial conduction, fin-collar interface resistance, and conduction across the fin collars. The analysis now becomes iterative. One now assumes an overall heat transfer rate Q_{tot} and the temperature at the outside of the fin collars is determined from Q_{tot} and the sum of the resistances cited above.

A second iterative procedure is now established. The variable whose value is assumed is the effective film coefficient between the fins and the gas stream, which involves the effect of convective heat transfer and mass transfer. With this value of $h_{effective}$, one can determine fin efficiency and the fin temperature distribution.

It is assumed that a condensate film exists on the vertical fins. An analysis is performed which relates this film thickness to the rate of removal due to gravity and shear, and the rate of addition of condensate by mass transfer from the bulk gas. In the process, from an energy balance, one determines the temperature of the interface between the bulk gas and the condensate; this is necessary for determining the mass transfer rate from the gas. Now that the thickness of the condensate film is known, the value of the assumed $h_{effective}$ is checked from the relation $h_{eff} = K$ water/δ_{film}. If the assumed and computed values are not the same, a new guess is made and calculations repeated until the assumed and computed values are equal.

When this occurs, the heat transfer rate from the fins and fin collar is computed, using the standard equations for fin and fin collar heat transfer and the values of $h_{effective}$ and film-bulk gas interface temperature. If this value is not the same as Q_{tot}, initially assumed in order to determine fin-collar temperature, the whole analysis is repeated with a new estimate of Q_{tot}. When, finally, the heat transfer rate to the cooling water from the fin collar equals the resulting computed rate to the fin collar and fins from the gas, the effect of this heat transfer rate on the cooling water is computed. The water exit temperature is established and this value is used as the inlet temperature for the next heat exchanger pass. Also, the effect of convective heat transfer and condensate mass transfer are determined relative to the gas composition and thermodynamic state. The updated gas state is used as inlet conditions for the next pass. The

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[1] A fouling factor of 0.001 hr-ft^2-°F/Btu, under both normal and DBA conditions, was assumed for cooling coil design purposes by Westinghouse. This value is conventionally used in sizing heat exchangers cooled by river water at 125° or less and with tube water velocity greater than 3 ft/sec (Reference 21), and is considered sufficiently conservative for this application. Computer analysis of the original Westinghouse coils selected shows that the required postaccident heat removal rate can be achieved even with a slight increase in fouling.

process is now repeated for the second, third, etc., passes until the gas exits the heat exchanger.

The mass transfer coefficients used in the HECO code were derived from analyses and reports of experimental data contained in References 21, 22, and 23.[1] From Reference 22, the mass flow rate of condensate is defined by

Equation (1)

$$\dot{m} = \overline{h}_D \, (\rho_{sg} - \rho_{sw})$$

From Reference 22, pp. 471-473, experimental data for mass and heat transfer correlate well by the expression

$$\left(\frac{\overline{h}_D}{u_s}\right) (Sc)^{-2/3} = \overline{St} \, (Pr)^{-2/3}$$

as shown in Figure 16-10 of Reference 22. Thus

Equation (2)

$$\overline{h}_D = u_s \cdot St \left(\frac{Sc}{Pr}\right)^{2/3}$$

$$\overline{h}_D = \frac{u_s \cdot h}{\rho C \, u_s} \left(\frac{Sc}{Pr}\right)^{2/3}$$

As Reference 22 points out, for large partial pressures of the condensing components, Equation (2) must be corrected by a factor P_t/P_{am}. Thus h_D is defined by

Equation (3)

$$\overline{h}_D = \frac{h}{\rho C} \, \frac{P_t}{P_{am}} \left(\frac{Sc}{Pr}\right)^{2/3}$$

This is essentially the same result as reported by Reference 23, pg. 343, and Reference 24.

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[1] Nomenclature used is given at the end of this discussion.

Reference 22 states that experiments show Equation (1) to be valid when the Schmidt number does not differ greatly from 1.0. Equations (1) and (3) are combined to give the mass transfer rate, which is

$$\dot{m} = \frac{h}{\rho C} \frac{P_t}{P_{am}} \left(\frac{Sc}{Pr}\right)^{2/3} (\rho_{sg} - \rho_{sw})$$

An approximation was made in assuming that $\left(\frac{Sc}{Pr}\right)^{2/3} \cong 1.0$; thus, the local mass transfer rate was computed from

$$\dot{m} = \frac{h}{\rho C} \frac{P_t}{P_{am}} (\rho_{sg} - \rho_{sw})$$

The heat transfer rate due to condensation is computed from

$$q_1 = \frac{\dot{m}\lambda h P_t}{\rho C P_{am}} (\rho_{sg} - \rho_{sw})$$

The heat transfer coefficient (h) was determined from experiments on Westinghouse plate-fin coils which are the same geometry as would be used in this application.

The heat transfer rate, locally, is computed from

$$q_2 = h\ (T_g - T_i)$$

The basis for selecting these values is that the authorities cited as references have shown, through analyses and through cited experiments, that the methods used are accurate.

The air side pressure drop across the original Westinghouse cooling coils under DBA conditions was estimated to be approximately 2.1 inches H_2O, or 0.076 psi. This will have negligible effect on the heat removal capability of the cooling coils.

The pressure of noncondensable gases are taken into consideration by virtue of the fact that the theory behind the analysis assumed that the condensable vapor must diffuse through a noncondensable gas.

Application of this method resulted in the fan cooler heat removal rate per fan as presented in Figure 15.6-32 for the original Westinghouse coils.

A testing program has been completed to confirm the validity of the above design procedures and the performance of the Westinghouse cooling coils under postaccident conditions. Details of the testing program are described in WCAP 7336-L (Westinghouse Proprietary) (see Reference 25).

Nomenclature

$\dot{m}$ = mass flow rate of condensate, lbm/hr-ft^2
h_D = mass transfer coefficient, ft/hr
ρ_{sg} = density of saturated steam at local bulk gas temp., lbm/ft^3
ρ_{sw} = density of saturated steam at local condensate-gas interface temp., lbm/ft^3
u_s = free steam gas velocity, ft/min
Sc = Schmidt number, $\mu/\rho D$, dimensionless
μ = viscosity of bulk gas, lbm/ft-hr
ρ = bulk gas density, lbm/ft^3
D = Gas-air diffusion coefficient, ft^2/hr
St = Stanton number, $h/\rho C u_s$, dimensionless
h = convective heat transfer coefficient, Btu/hr-ft^2-°F
C = Specific heat of bulk gas, Btu/lbm-°F
Pr = Prandtl number, $\mu C/k$, dimensionless
k = thermal conductivity of bulk gas, Btu/hr-ft-°F
P_t = total gas pressure, lbf/ft^2

P_{am} = air log-mean pressure $\dfrac{P_{aw} - P_{ag}}{\ln \dfrac{P_{aw}}{P_{ag}}}$, lbf/ft^2

P_{aw} = partial pressure of air at the local gas-condensate interface, lbf/ft^2
P_{ag} = Partial pressure of air at the local bulk gas temperature, lbf/ft^2
λ = Latent heat of vaporization (or condensation) at the local gas condensate interface temperature, Btu/lbm
q_1 = local heat transfer rate due to condensation, Btu/hr-ft^2
q_2 = local heat transfer rate due to convection, Btu/hr-ft^2
T_g = local bulk gas temperature, °F
T_i = local gas-condensate interface temperature, °F

The original Westinghouse cooling coils were replaced in the 1986 timeframe. The heat removal capability of the replacement coils, supplied by Marlo Coil Nuclear Cooling Inc., is described in Reference 14. The method to determine the heat removal capability is similar to that described previously. The analysis described in Appendix 15D uses RCFC heat removal based on the Marlo Coils of Reference 14 and a service water flow rate that bounds the minimum required flow rate in the Technical Specifications.

15.6.5.4.3.3 Containment Spray

When a spray drop enters the hot saturated steam-air environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the

atmosphere and the drop will cause a heat flow to the drop. Both of these mechanisms will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will be essentially equal to the temperature of the steam-air mixture.

The terminal velocity of the drop can be calculated using the formula given by Weinberg (see Reference 26) where the drag coefficient C_D is a function of the Reynolds number:[1]

Equation (1)

$$V^2 = \frac{4Dg\ (\rho - \rho_m)}{3C_D\ \rho_m}$$

For the 700 micron drop size expected from the nozzles, the terminal velocity is less than 7 ft/sec. For a 1000 micron drop, the velocity would be less than 10 ft/sec. The Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu′ (Sherwood Number), can be calculated from the empirical relations given by Ranz and Marshall (see Reference 27).

Equation (2)

$$\underline{Nu} = 2 + 0.6\ (\underline{Re})^{1/2}\ (\underline{Pr})^{1/3}$$

Equation (3)

$$\underline{Nu'} = 2 + 0.6\ (\underline{Re})^{1/2}\ (\underline{Sc})^{1/3}$$

The Prandtl number and the Schmidt number for the conditions assumed are approximately 0.7 and 0.6, respectively. Both of these are sufficiently independent of pressure, temperature and composition to be assumed constant under containment conditions (see References 28 and 29). The coefficients of heat transfer (h_c) and mass transfer (k_G) are calculated from Nu and Nu′, respectively. The equations describing the temperature rise of a falling drop are:

Equation (4)

$$\frac{d}{dt}(Mu) = mh_g + q$$

[1] Nomenclature used is given at the end of this discussion.

Equation (5)

$$\frac{d}{dt}(M) = m$$

where:

Equation (6)

$$q = h_c A \ (T_s - T)$$

Equation (7)

$$m = k_G A \ (P_s - P_v)$$

These equations can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the liquid drop temperature rises to the steam-air mixture temperature in less than 0.5 second, which occurs before the drop has fallen five feet. These results demonstrate that the spray will be 100% effective in removing heat from the atmosphere.

Nomenclature

A	=	area
C_D	=	drag coefficient
D	=	droplet diameter
g	=	acceleration of gravity
h_c	=	coefficient of heat transfer
h_s	=	steam enthalpy
k_G	=	coefficient of mass transfer
M	=	droplet mass
m	=	diffusion rate
Nu	=	Nusselt number for heat transfer
Nu'	=	Nusselt number for mass transfer
P_s	=	steam partial pressure
P_v	=	droplet vapor pressure
Pr	=	Prandtl number
q	=	heat flow rate
Re	=	Reynolds number
Sc	=	Schmidt number
T	=	droplet temperature
T_s	=	steam temperature
t	=	time
u	=	droplet internal energy
V	=	velocity

ρ = droplet density
ρ_m = steam-air mixture density

15.6.5.4.4 Containment Pressure Analysis

Appendix 15D, Containment Pressure Analysis provides the containment pressure analysis based on VANTAGE 5 fuel, increased Containment Spray delay time and decreased Service Water flow to the Reactor Containment Fan Coolers.

15.6.5.4.4.1 Assumptions

The containment pressure was calculated for a range of large area ruptures of the RCS. The rupture sizes considered were:

1. Double-ended rupture,
2. 6 ft^2 break,
3. 4.5 ft^2 break, and
4. 3 ft^2 break.

Figure 15.6-33 presents a generalized result of the original transients. For all cases, a pressure peak of less than 47 psig was calculated (the design pressure for Zion Station).

In these transients, two spray pumps and three fans starting at 45 seconds were assumed. (The results of an analysis using a containment spray (CS) initiation time of 92.5 seconds are given in the notes to Table 15.6-12.) These acted to quickly reduce the pressure after the peak pressures were reached. Each of the spray pumps provides 2615 gpm of spray flow at Containment design pressure. After the RWST is exhausted, containment spray is continued via the RHR system. One of two RHR loops is utilized to recirculate and cool water from the containment sump, providing a source of cooled water for the containment spray and ECCS functions. Table 15.6-12 contains a chronology for the event actions for the calculation of the original design containment pressure transients.

Sections 15.6.5.4.4.2 and 15.6.5.4.4.3 provide a summary of the energy sources and sinks used in the calculations.

15.6.5.4.4.2 Energy Sources

The energy sources presented in Table 15.6-13 are potentially available to be transferred to the Containment during the blowdown time.

In the above energy summation, all sensible energy sources are referenced to the datum of saturated water at containment design pressure, which is the maximum amount of energy that can be transferred from the metal to the coolant.

The integrated energy balance at the end of blowdown is presented in Table 15.6-14. The values were determined by the SATAN code, as modified by conservative core heat transfer assumptions.

In this calculation, the sensible energy sources are transferred to the coolant as a function of time and, for longer blowdown times, more sensible energy is absorbed. For the very large breaks, very little energy is transferred to the steam generators, because of the rapid uncovering of the tubes; while for smaller breaks, the tubes do not uncover as rapidly and significant heat transfer results.

A negligible amount of energy is transferred from the reactor vessel during the relatively fast blowdown.

15.6.5.4.4.3 Energy Sink

Figure 15.6-34 presents the energy absorption capability within the 2.715×10^6 ft^3 free volume of the Containment. Thus, the internal energy of the steam-air mixture must be increased to 311.4×10^6 Btu for the containment pressure to reach the design pressure of 47 psig.

The integrated containment energy balance at the end of blowdown is given by:

$$U_f = U_i + \Sigma (mh)_{in} + \Sigma Q_{in} - \Sigma Q_{out}$$

Where:

U_f	=	Final internal energy in the Containment
U_i	=	Initial external energy in the Containment
$\Sigma(mh)_{in}$	=	Enthalpy added by blowdown sources and spray water
ΣQ_{in}	=	Energy added directly to containment atmosphere by hydrogen-oxygen recombination
ΣQ_{out}	=	Heat removal by containment structure and cooling system.

The internal energy is made up of three sources: air, steam, and sump water. Only the air-steam mixture with their respective partial pressures contribute to the containment total pressure. The internal energy for the initial assumed containment conditions, 120°F and 15 psia, is as follows:

Steam (m) (u)	= (2360) (1051)	= 2.48×10^6 Btu
Air (m) (C_v) (T)	= (186,088) (0.172) (120)	= 3.84×10^6 Btu
Sump (m) (u)	= (12,343) (88.0)	= 1.09×10^6 Btu
		7.41×10^6 Btu

The internal energy balance at the end of blowdown is given in Table 15.6-15.

The difference between the internal energies given by the energy balance equation and by the COCO program represents an error of less than plus or minus 1% in the calculation.

Figure 15.6-35 shows the heat transfer coefficient calculated for the various break sizes.

15.6.5.4.4.4 Containment Margin Evaluation

Evaluation of the capability of the Reactor Containment and containment cooling systems to absorb energy additions without exceeding the containment design pressure requires consideration of two periods of time following a postulated large area rupture of the RCS.

The first period is the blowdown phase. Since blowdown occurs too rapidly for the containment cooling systems to be activated, there must be sufficient energy absorption capability in the free volume of the Containment (with due credit for energy absorption in the Containment structures) to limit the resulting pressure below design.

The second period is the postblowdown period where the containment cooling systems must be able to absorb any postulated postblowdown energy additions and continue to limit the containment pressure below design.

15.6.5.4.4.5 Margin-Blowdown Peak to Design Pressure

Point A in Figure 15.6-36 corresponds to the internal energy at the end of the double-ended blowdown, 261 x 10^6 Btu. In order for the pressure to increase to design pressure (47 psig), the internal energy must be increased to 311 x 10^6 Btu (Point B). The allowed energy addition is therefore 50 x 10^6 Btu. Since energy transferred to the Containment from the core is in the form of steam, the total transferred core energy corresponding to allowed energy addition is as follows:

$$Q_{core} = \frac{h_{fg}}{h_g} Q_{Allowed} = \frac{914.1}{1178.2} x\ 50\ x\ 10^6 = 38.8\ x\ 10^6\ Btu$$

This allowable value of energy which could be transferred from the core to the Containment without increasing the transient containment pressure to design pressure can be compared to the energy stored in the reactor vessel and transferred to the steam generator during blowdown for the double-ended break. The thick metal of the reactor vessel was not considered since a negligible amount of this energy can be transferred in the short blowdown time.

Stored in the core	16.0×10^6 Btu
Core internals metal	4.9×10^6 Btu
Transferred to steam generators	-1.8×10^6 Btu
	19.1×10^6 Btu

Thus, the Containment has the capability to limit containment pressure below design even if all of the available energy sources were transferred to the Containment at the end of blowdown. This would also include no credit for energy absorption in the steam generator. For this to occur, an extremely high core to coolant heat transfer coefficient is necessary. This would result in the core and internals being completely subcooled and limit the potential for release of fission products.

15.6.5.4.4.6 Additional Energy Added As Superheat

Line A to C on Figure 15.6-36 represents a constant mass line extended into the superheated region. Comparison of the energy addition allowable for the superheated case relative to the saturated case shows a lesser ability of the Containment to absorb an equivalent amount of energy as superheat. An addition of 12×10^6 Btu of energy after blowdown would cause the containment pressure to increase to design. The recombination of hydrogen and oxygen from a 12.9% Zr-H_2O reaction completed before the end of blowdown would be required to generate 12×10^6 Btu of energy. For the case analyzed, the core was assumed to be in a subcooled state and no Zr-H_2O reaction would be possible. In order for Zr-H_2O reaction to occur before the end of blowdown, all of the stored initial energy must remain in the core. If this occurred, a blowdown peak containment pressure of only 34.0 psig would be reached instead of 39.4 psig in the case analyzed. Lines D and E on Figure 15.6-36 represent the superheat energy addition required to increase the pressure to the design pressure and this corresponds to the hydrogen-oxygen recombination energy from a 20.8% Zr-H_2O reaction.

It is, therefore, concluded that the Containment has the capability to absorb the maximum energy addition from any LOCA without reliance on the containment cooling system. In addition, the Containment has the capability, with substantial margin, to absorb any possible energy additions from arbitrary energy sources.

15.6.5.4.4.7 Margin-Post Blowdown Energy Additions

The ECCS is designed to rapidly subcool the core and stop the addition of mass and energy to the Containment. Thus, it is expected that there will not be any significant energy addition to the Containment following blowdown. However, the following cases are presented to demonstrate the capability of the Containment to withstand postaccident energy additions without credit for core cooling.

Case 1 - Blowdown from a large area rupture with continued addition of the core residual energy and hot metal energy to the Containment as steam.

Case 2 - Same as Case 1 but with the energy addition from a maximum Zirconium-water reaction.

Figure 15.6-37 presents the containment pressure transient for Case 1. For this case, the decay heat generated for a 3391 MWt core operated for an infinite time is conservatively assumed. This decay heat is added to the Containment in the form of steam by boiling off water in the reactor vessel. For this case, injection water merely serves as a mechanism to transfer the residual energy to the Containment as it is produced. Injection water is in effect throttled at the required rate.

In addition, all the stored energy in the core and internals, which is calculated to remain at the end of blowdown, is added in the same way during the time interval between 22 and 48 seconds (corresponds to accumulator injection time). Also, all the sensible heat of the reactor vessel is added as steam exponentially over a 2000 second time interval.

The containment cooling system capability assumed in the analysis was two-of-three available CS pumps and three-of-five available containment fan coolers. This is the minimum equipment available considering the single failure criterion in the emergency power system, the spray system, and the fan cooler system.

The containment heat removal capability, started at 45 seconds, does not exceed the energy addition rate until 50 seconds. Thus the containment pressure exceeds the initial blowdown value, reaching a peak of 42.6 psig. An extended depressurization time results due to the increased heat load on the containment coolers.

It should be emphasized that this situation is highly unrealistic in that continued addition of steam to the Containment at these high rates after blowdown could not occur. The accumulator and ECCS act to rapidly reflood and subcool the core.

Figure 15.6-38 presents the containment pressure transient for Case 2. To realistically account for the energy necessary to cause a metal-water reaction, sufficient energy must be stored in the core. Storing the energy in the core rather than transferring it to the coolant causes a decrease in the blowdown peak.

The reaction was calculated using the parabolic rate equation developed by Baker and assuming that the clad continues to react until zirconium oxide melting temperature of 4800°F is reached. An additional 10% reaction of the unreacted clad is assumed when the oxide melting temperature is

reached. A total reaction of 32.3% has occurred by 1000 seconds. Previous analysis has shown that steam limited reactions could result in a higher total reaction, but at a much later time. The reaction provided by the parabolic rate equation, therefore, imposes the greatest load on the containment cooling system.

As in Case 1, the residual heat and sensible heat are added to the Containment as steam. The energy from the Zr-H_2O reaction is added to the Containment as it is produced. The hydrogen was assumed to burn as it entered the Containment from the break.

The blowdown peak was reduced to 34.0 psig and a peak pressure of 42.2 psig was reached at 400 seconds. At this time, the heat removal capability of the containment cooling system assumed to be operating, two CS pumps and three fan coolers, exceeded the energy addition from all sources.

For comparison, the containment pressure transients for Cases 1, 2, and the double-ended blowdown are replotted in Figure 15.6-39. It is concluded that operation of the minimum containment cooling system equipment provides the capability of limiting the containment pressure below its design pressure with the addition of all available energy sources and without credit for the cooling effect from the ECCS.

15.6.5.4.4.7.1 Discussion of Additional Energy Sources Used in Cases 1 and 2

The following is a summary of the energy sources and the containment heat removal capacities used in the containment capability study. Figure 15.6-40 presents the rate of energy addition from core decay heat, Zr-H_2O reaction energy, and the hydrogen-oxygen recombination energy. The heat removal capability for the partial containment cooling (one spray pump and three Westinghouse fan coolers) is also presented. (As discussed in section 15.6.5.4.3.2, Westinghouse cooling coils are no longer used.) These heat removal values are for operation with the Containment at design pressure.

The integrated heat additions and heat removals for Cases 1 and 2 are plotted in Figures 15.6-41 and 15.6-42, respectively. These curves are presented in a manner that demonstrates the capability of the Containment and the cooling systems to absorb energy. The integrated heat removal capacity is started at the internal energy corresponding to design pressure, while the integrated heat additions begin from the internal energy calculated at the end of blowdown for each case. The upper line on each curve is the Containment structure's and containment cooling system's capability to absorb energy additions without exceeding design pressure. The lower curve for each is the energy addition curve. Since these energy additions are the maximum possible with no credit for core cooling, there is more than adequate capability to absorb arbitrary additions.

The curves in Figures 15.6-43 and 15.6-44 present the individual contribution of the heat removal and heat addition source, respectively.

15.6.5.4.5 Containment Pressure Analysis Including Reflood Effects

15.6.5.4.5.1 Analysis Discussion

Subsequent to the pressure transients described previously in this section, an additional pressure transient analysis was performed to evaluate the additional effects of reflood and to evaluate the following postulated break cases:

1. Double Ended Pump Suction (DEPS),
2. Double Ended Hot Leg (DEHL),
3. Double Ended Cold Leg (DECL), and
4. 0.6 - Double Ended Pump Suction.

The most current analysis, is discussed in Appendix 15D.

Mass and energy release rates for blowdown and reflood are given in Figures 15.6-45 and 15.6-46, respectively. The core inlet velocity as a function of time is given in Figure 15.6-47.

The basis for the safeguards performance in this analysis is the loss of one fan cooler and one spray pump due to the failure of one diesel generator. This failure is more severe than the usual minimum safeguards case, since the usual case would limit the available water flow to the core and thus limit the mass and energy flow from the RCS to the Containment during the reflood period. To illustrate this conservatism, the double ended pump suction break with the minimum safeguards operation of one ECCS train, two spray pumps, and three fan coolers has been analyzed.

To demonstrate containment margin, the nonmechanistic case which assumes the operation of two ECCS trains, two spray pumps, and three fan coolers has been analyzed. The results of these cases are tabulated in Table 15.6-16. The sensitivity to different safeguards assumptions is shown in Figure 15.6-48, the sensitivity to break location is shown in Figure 15.6-49, and the sensitivity to break size is shown in Figure 15.6-50.

In addition to the above, the following analyses have been performed to further verify that the peak calculated pressure following a LOCA will not exceed the design pressure.

Case 1 - The double-ended pump suction break has been analyzed including the following assumptions:

1. Entrainment during reflood continues until the quench front reaches the 8-foot level;

2. Water in the lower plenum is initially saturated with temperature decreasing during the reflooding transient to 230°F;

3. The core flooding rate has been analyzed using a more detailed pressure drop calculation and local densities throughout the loop. This calculation is described in Section 15.6.5.4.5.2; and

4. The additional structural heat sinks given in Table 15.6-17 have been used.

This case resulted in a peak containment pressure of 42.2 psig.

Case 2 - The double-ended pump suction break has been further analyzed showing a peak pressure of 43.8 psig as shown in Figure 15.6-51. The assumptions in this case were the same as those in Case 1 above except that the entrainment during reflood was arbitrarily extended until the quench front reached the 10-foot level to show additional margin. The mass and energy releases for this case are given in Tables 15.6-18 and 15.6-19.

The core inlet velocity is given in Figure 15.6-52. The heat transfer coefficient to the structures is given in Table 15.6-20 for these two cases. Figure 15.6-53 gives the schematic loop representations for the reflood calculations. Table 15.6-21 gives a summary of areas, K factors, lengths, and diameters used, and Table 15.6-22 gives the local density distribution at 80 seconds after blowdown.

An overall inventory and energy balance for the above cases are given in Tables 15.6-23 and 15.6-24.

Table 15.6-11 lists the structural heat sinks available within each Containment. The last entry allots 50,000 square feet to miscellaneous steel structures without any identification as to the equipment and components considered. Table 15.6-25 lists each of these steel structures and its exposed area. Table 15.6-17 lists additional steel heat sinks totaling 118,932 square feet available within each Containment that were not accounted for previously. These additional steel heat sinks were determined by examination of "as-built" drawings of equipment and components installed within the Containment structure. Additional investigation would undoubtedly reveal even more steel heat sinks.

15.6.5.4.5.2 <u>REFLOOD Code Description</u>

The REFLOOD code consists of a fixed vessel model, two variable-geometry loops, and models for accumulators and pumped injection. In the vessel

model, water levels in both the downcomer and core are calculated from the mass balance and momentum equations and the appropriate correlation for liquid carryover from the core. REFLOOD includes the effect of inertia in the core-downcomer liquid and the pressure drop due to the elevation head of two-phase liquid above the core water front.

The model used for each of the coolant loops (broken and lumped unbroken loops) is very general. Each of the loops may have a maximum of 29 series resistance elements. A typical schematic is shown in Figure 15.6-53. Provision is made for pressure drops within each element due to friction $(f \cdot L)/D$, form-factors (commonly called K-factors), and the dynamic pressure drop due to density change. The dynamic pressure drop due to area change is included at the interface between loop elements (and at the interface between the first element of each loop and the core). In the REFLOOD code, the density of fluid flowing in each resistance element is determined from the local pressure and enthalpy. The loops are assumed to be quasistatic - there is no provision for mass buildup in any loop element.

The REFLOOD code currently provides the following models and features:

1. The pressure at the top of the downcomer can be specified as the pressure of any element in either loop, or as containment back pressure;

2. In each loop, any element can be specified as the steam generator element. (The local enthalpy changes to that of superheated steam at the steam generator secondary side temperature at the inlet of the steam generator element);

3. Pumped injection may be specified as a tabular head-flow curve, with delivery pressure specified as the pressure in any loop flow element, or containment back pressure; and

4. Accumulator injection may be specified as a linear ramp in time.

15.6.5.4.5.3 Verification of REFLOOD Code

Since the REFLOOD code is basically an extension of the previous analysis, a comparison to the previous analysis can provide a good base point for REFLOOD code verification. The major verification process was accomplished by first eliminating all discernable differences in REFLOOD. Both analyses were then run with identical input to establish a base point comparison for the equivalent codes. From that point, individual differences were re-established one at a time in REFLOOD and new runs were made. In this way, the REFLOOD code solution method could be verified against the previous analysis and the effects of individual differences assessed. It should be noted at this point that the verification process was done only for a model which considered the break located in the pump suction piping

(the break location of primary interest in containment energy release determinations).

The comparison was done by removing the momentum pressure drop terms and the pressure drop due to the head of two-phase mixture above the core water front in the vessel. Densities were calculated internally based on containment pressure instead of local pressure. It is noted that the standard design procedure for the previous analysis is to use core pressure vis-a-vis containment pressure. The loop model consisted of seven elements (three in the broken loop, four in the unbroken loop) arranged as in the previous analysis. Pressure drop data was input as form factors (Ks), no equivalent lengths were used.

Case 1 was duplicated using the previous analysis. A comparison of the two Case 1 runs is indicated in Figure 15.6-54.

15.6.5.4.6 Evaluation of Containment Internal Structures

15.6.5.4.6.1 General

The containment internal structures such as the reactor coolant loop compartments and the reactor shield wall are designed for the pressure buildup that could occur following a LOCA. If a LOCA were to occur in these relatively small volumes, the pressure would buildup at a rate faster than the overall compartments.

The compartment bounded by the crane wall and the reactor shield wall and by the base slab and operating floor slab at El 617′ was designed for a differential transient pressure of 14 psi for the design basis LOCA. The concrete systems were designed for this pressure differential at working stress allowable by American Concrete Institute (ACI)-318 63. The major RCS equipment compartments are interconnected to form essentially one large volume. The differential pressure is based on the energy release from a LOCA and the available venting areas for energy release to other containment internal areas.

A digital computer code, COMCO, was developed to analyze the pressure buildup in the reactor coolant loop compartments. The COMCO code is largely an extension of the COCO code in that a separation of the two-phase blowdown into steam and water is calculated and the pressure buildup of the steam-air mixture in the compartment is determined. Each compartment has a vent opening to the free volume of the Containment.

The main calculation performed is a mass energy balance within the control volume of a compartment and is based on such factors as energy released, structural heat sinks available, compartment volumes and vent areas. The pressure builds up in the compartment until a mass and energy relief through the vent exceeds the mass and energy entering the compartment from the break. The reactor coolant loop compartments are designed for the

maximum calculated differential pressure resulting from an instantaneous double-ended rupture of the reactor coolant pipe.

Using this calculation, the peak differential pressure across the reactor loop compartment wall was less than 10 psi. The walls are designed for a differential pressure of 14 psi.

During construction, a control limit was placed on the vent area to ensure that the design pressure would not be exceeded in the final calculation. This was done by developing a curve of pressure versus vent area and then controlling the design to always ensure adequate vent area. When this curve was developed, a pressure of 14 psi was selected as the design basis based upon a review of the preliminary design and the fact that the final design would conservatively allow the final vent area to be greater than that corresponding to a pressure peak of 14 psi.

The compartments evaluated for pressure buildup within the Containment following a pipe rupture within that compartment are as follows:

1. Reactor vessel capacity,
2. Reactor coolant loops,
3. Main steam and feedwater, and
4. Pressurizer.

15.6.5.4.6.2 Reactor Vessel Cavity

Although the current basis for the reactor cavity analysis is a single-ended rupture of a reactor coolant pipe inside the biological shielding, the analysis was based on a longitudinal split of the reactor vessel. (See Section 3.5) Since the reactor cavity design is sufficient to ensure containment integrity and protect against possible missile generation in the unlikely event of a reactor pressure vessel split, it will certainly be able to withstand the pressures resulting from a single-ended rupture of a reactor coolant pipe.

The reactor vessel cavity volumes are given in Table 15.6-26.

15.6.5.4.6.3 Reactor Coolant Loops Compartments

The loop compartment analysis determines the peak pressure in the regions between the crane wall and the biological shield following a double-ended rupture of a reactor coolant pipe. The tool used in this analysis is COMCO, a Westinghouse computer code which calculates the pressure in a compartment with a given volume and vent area.

The COMCO code is used to calculate the pressure in a single compartment. This code uses a technique similar to that of the COCO code with regard to methods of mass and energy balance in the steam-air mixture and the separation of steam and water phases. This method of solution involves the

manipulation of algebraic volume and state equations to arrive at a set of explicit differential equations. All variables are updated simultaneously by a one-step integration technique. The solution of the simultaneous equation is based on a Gauss-Johnson reduction procedure applied to a matrix of the variables.

The model allows up to two vents in the compartment. Flows through the vent are calculated using relations for isentropic flow through a short tube with a vena contracta correction. The model is completed with ideal gas critical flow relations to handle flows in this range. These flows are based upon the pressure differentials across the vents, which are calculated using an input backpressure. The time steps used in calculating pressure differentials are computed by the code. The basis of the time step calculation is the rate of change of the compartment steam pressure. If the rate of change of the steam pressure is found to be changing rapidly when compared to 1% of the rate of change in the previous time step, the time step is shortened. If the pressure is found to be rising or falling at a steady rate or staying constant, the time step is increased. Time steps as small as 0.00005 second are commonly used at the beginning of a transient while at the end of a transient with a nearly steady-state conditions established, the time steps can be increased to the order of 0.01 to 0.05 second.

For the actual analysis of the loop compartments, a double-ended severance of a cold leg reactor coolant pipe was assumed. The time dependent mass and energy flow corresponding to this break size was used as input to the analysis. With the loop compartment volume (see Table 15.6-27) and the breakflow rate known, a sensitivity study of the effect of vent area upon compartmental pressure was undertaken. The study calculated compartmental pressures for vent areas varying between 500 ft^2 and 2000 ft^2. This information was correlated into a curve of compartment pressure versus vent area shown in Figure 15.6-55.

The analysis was then extended to study the effects of different backpressures inhibiting flow from the compartment. A series of COMCO runs with input backpressures indicated that the pressure increase in the loop compartment due to this effect would be between one-half and one psi. For a loop compartment with a volume of 87,000 ft^3 and a vent area of 1275 ft^2, this analysis showed that, even allowing for the effect of containment backpressure, the maximum differential across the wall would be 7 psi. Since the design pressure of these compartments is 14 psig, there is a 100% margin. While the maximum absolute pressure experienced by the walls is the peak containment pressure for the LOCA, at largest absolute pressure at the time of the maximum differential is 8.9 psig.

An analysis of the pressure in the reactor cavity was also performed. This analysis resulted in a peak pressure of 600 psig. It should be noted that

this analysis was conducted using a design basis much more severe than the current design basis.

As part of the loop compartment analysis, a sensitivity study concerning the compartment volume was performed. This study involved the calculation of compartment pressure transients for loop compartments of 50,000 ft^3 and 100,000 ft^3. A comparison of the peak differential pressures for the cases showed that, for this range of volume, peak differential pressure was not sensitive to volume.

The important sensitivity is to vent flow since the differential pressure will continue to rise until the flow out of the compartment matches the blowdown flow. The compartment volume can only affect the time required for the peak differential pressure to be achieved. For the range of volumes of concern, this effect is small and the peak differential insensitive.

The loop compartment analysis was performed both with and without backpressure. The cases with the maximum differential were used.

15.6.5.4.6.4 Main Steam and Feedwater Compartments

The main steam and feedwater piping compartments are all very nearly equal in volume and vent area. The volume and vent area of each of the compartments is 4520 ft^3 and 178 ft^3 respectively. A rupture of a main steamline is the limiting break for these compartments and, for this case, the peak compartment pressure is below the design pressure of the compartments.

15.6.5.4.6.5 Pressurizer Compartment

The pressurizer compartment was evaluated for a rupture of the surge line at the nozzle connection to the pressurizer and rupture of a relief valve line at the top of the pressurizer compartment. Since this compartment is completely open at the bottom where the surge line connects and is partially open at the top, no significant pressure buildup is experienced in this compartment.

Assuming a differential pressure of 10 psi exists in all the compartments evaluated, the pressurizer compartment is the most limiting from a stress standpoint.

15.6.5.5 Environmental Consequences of a Loss-of-Coolant Accident

15.6.5.5.1 General

UFSAR Chapters 3 and 6 describe the protective systems and features which are specifically designed to limit the consequences of a major LOCA. The

capability of the SI System for preventing melting of the fuel clad and the ability of the Containment and containment cooling systems to absorb the blowdown resulting from a major loss of coolant have been discussed in Sections 15.6.5.2 and 15.6.5.4, respectively. The capability of the safeguards to meet the dose limits set forth in 10CFR100 is demonstrated in this section. The exposures given in this analysis are based on calculated techniques used at the time the original FSAR was written. Although present day analyses use different assumptions and models, it is believed that exposures would still be less than 10CFR100 limits and that the values given here are still conservative.

Because of the design conservatism and care taken during fabrication and installation of the RCS, a break of the system integrity of any size is considered highly unlikely. For break diameters up to four inches, clad damage is not expected and hence, activity release to the Containment would be limited to that contained in the coolant. The corrosion product activity and fission product activity has diffused to the coolant through assumed fuel defects. For larger break sizes up to and including the hypothetical double-ended rupture of a coolant loop, some clad rupture would occur and a portion of the activity contained in the fuel pellet-clad gap would be released to the Containment and would be available for leakage.

For the purpose of evaluating radiation exposure, a double-ended rupture of a reactor coolant loop is considered with partial safeguards operating from the diesel generator power system. As shown in Section 15.6.5.2, the ECCS, with power from two of the three diesel generators, will maintain clad temperature well below the melting point of Zircaloy-4 and will limit zirconium-water reaction to an insignificant amount. However, as a result of the cladding temperature increase and the rapid system depressurization, cladding failure may result in the hotter regions of the core. Release of the inventory of the volatile fission products in the pellet-cladding gap might follow.

It is assumed that all of the gaseous activity present in the pellet-cladding gap of all the fuel rods is released. The ability of the safeguards to limit environmental activity release and hence whole body and thyroid dose is analyzed.

15.6.5.5.2 Sources

15.6.5.5.2.1 Initial Release Fractions

The offsite doses have been analyzed for two cases. In the first case, the DBA, it has been assumed that the entire inventory of volatile fission products contained in the pellet-cladding gap is released during the time the core is being flooded by the ECCS. Of this gap inventory, 50% of the halogens and 100% of the noble gases are considered to be released to the containment vessel atmosphere. It has also been assumed that 2.5% of the

halogens originally present in the gaps are available for leakage from the Containment in organic forms and are not subject to plateout. The remaining 47.5% of the gap inventory is considered to be in elemental form, and is assumed to be subject to condensation and plateout on the containment surfaces of the lower compartment at the rate of 0.865 hr^{-1}. After two hours, no further decrease is assumed to occur due to condensation or plateout. The basic inventories used in the analysis of this case are listed in Table 15B.2-1 of Appendix 15B.

In the second case, the offsite doses resulting from a hypothetical accident assuming larger activity releases have been analyzed. Activity releases of these magnitudes have a considerably lower probability than those associated with the DBA. For the analysis of the hypothetical case, it has been assumed that 50% of the core inventory of halogens and 100% of the core inventory of noble gases are released to the containment atmosphere. It has also been assumed that 50% of this containment vessel halogen inventory platesout immediately on the interior surfaces of the Containment. Of the remaining 25% of the core inventory of halogens, 10% is assumed to be in organic form. The basic inventories used in the analysis of this case are also listed in Table 15B.2-1 of Appendix 15B.

15.6.5.5.2.2 Organic Iodine Inventory

It is not expected that significant organic iodine would be liberated from the fuel as a result of a LOCA. This conclusion is based on the absence of indications of such release from the inpile fuel meltdown experiments conducted by Oak Ridge National Laboratory (ORNL). The fraction expected would be on the order of 0.2% or less on the basis that, in the region of the fuel rod where conditions would be most favorable for the existence of organic iodine, the rates of thermal radiolytic decomposition would exceed the rate of replenishment.

The more plausible mechanism for organic iodine formation is by reaction of elemental iodine in an absorbed state on organic-contaminated surfaces. Whether limited by diffusion to the surface or by the reaction rate of absorbed iodine, the resulting fractional conversion of airborne iodine per unit time is proportional to the surface to volume ratio of the enclosure. Therefore, observed yields of organic iodine as a function of aging time in various test enclosures were extrapolated to the Containments in proportion to the surface/volume ratio. These results, in no case exceeded a calculated conversion rate of 0.0035% of the atmospheric iodine per hour.

At this rate, the formation of organic iodine has a negligible effect on the consequences of containment leakage. In short, the mechanisms which are believed to have produced significant amounts of organic iodine in test facilities would be so diminished in effect by the vastly reduced relative surface to volume in the plant Containment, that the organic iodine component will be of minor importance.

Recent experiments have shown that the formation of organic forms of iodines is largely dependent on specific conditions of the test such as activity concentration, pressure, temperature, humidity, radiation field level presence of impurities, etc.

On the basis of the data available at the time of license, the amount of iodine in organic form which could possibly exist in the Reactor Containment after the LOCA will not exceed 5% of the total airborne fraction.

However, for this analysis, a conservative value of 10% of the total airborne iodine inventory was assumed to be in organic form.

15.6.5.5.3 Method of Analysis

To evaluate the ability to meet the suggested 10CFR100 guideline, the thyroid dose and the whole body dose are calculated as a function of distance from the reactor. Results are presented for offsite exposure at the site boundary and low population zone distance.

15.6.5.5.3.1 Basic Activity Transport Model

The quantities of activity of elemental iodine and noble gases released from the Containment were calculated with the PREL digital computer code, which calculates the time dependent concentrations of a material in up to eight volumes simultaneously. Eight processes of removal (absorption) are allowed in each volume, and the isotope may be transferred from volume to volume as time passes. The integrated amounts absorbed by each process over the period of the run are also determined.

The basic set of equations which represent the time dependent variations of the amounts of material in N volumes are:

$$\frac{dC(J)}{dt} = -\lambda(J)\ C(J) + \sum_{I=1}^{N} \lambda_T(I,J)\ C(I)$$

There is one of these equations for each volume J. The first term on the right of the equation is just the total removal due to decay, cleanup leakage, etc. The second term is production in volume J due to transfer in from other volumes. Specifically, the terms used in this and following equations are defined below.

t	=	time, hours
C(J)	=	amount of species in volume J, curies
λ(J)	=	total removal rate of species from volume J, curies removed/hr/curie present

$\lambda_A(J,K)$	=	absorption rate of species by a specific process K from volume J, curies absorbed/hr/curie present
$\lambda_T(I,J)$	=	transfer rate from volume I to volume J, curies trans./hr/curie present
$\lambda_A(J)$	=	total absorption of species in volume J due to all processes curies absorbed/hr/curie present
$\lambda_I(J)$	=	total removal of species from volume J due to transfer other volumes, curies trans./hr/curie present
J	=	index referring to different volumes
K	=	index referring to different processes of "absorption"
I	=	index referring to volume from which transfer occurs
T	=	time period over which problem is run, hours
N	=	number of volumes in the problem
M	=	number of processes of absorption

The total rate of absorption in volume J as a result of M processes is given by:

$$\lambda_A(J) = \sum_{K=1}^{M} \lambda_A(J,K)$$

The total rate of removal from volume J due to transfer of species to the N-1 other volumes, with $\lambda_T(J,J) = 0$, is given by:

$$\lambda_T(J) = \sum_{I=1}^{N} \lambda_T(J,I)$$

The total rate of removal from volume J, including both absorption and transfer out is given by:

$$\lambda(J) = \lambda_A(J) + \lambda_T(J)$$

After the problem has run for the time period specified, the total activity of each isotope released from the Containment for each time period is computed.

15.6.5.5.3.2 Model for Offsite Doses

The activity releases computed by PREL are used in the WEDOSE digital code to calculate the offsite inhalation and whole body doses. The WEDOSE code uses the following standard relationship for the inhalation dose from each isotope for each time period:

$$D(I,T) = Q(I,T) \cdot DCF(I) \cdot B(T) \cdot \frac{\chi}{Q}(x,T)$$

where:

D(I,T)	=	inhalation dose from isotope I during period T, rems
Q(I,T)	=	activity of isotope I released in time (t_2 - t_1), curies
DCF(I)	=	inhalation dose conversion factor for isotope I, rem/curie
B(T)	=	breathing rate m^3/sec
$\frac{\chi}{Q}(x,T)$	=	atmospheric dispersion factor, sec/m^3

For the computation of the whole body doses from cloud immersion, the equation for the semi-infinite spherical model was used, as follows:

$$D(I,T) = 0.246\ Q(I,T) \cdot E(I) \cdot \frac{\chi}{Q}(x,T)$$

The values of average energy per disintegration, E(I), the decay constants, and the dose conversion factors used are listed in Table 15.6-28. Values of the atmospheric dilution factor for the site boundary and low population zone distances are listed in Table 15.6-29. The breathing rates are given in Table 15.6-28.

For the direct containment dose, the sources were assumed to be homogeneously distributed within the free volume of the Reactor Containment. The source intensity as a function of time was determined by considering natural decay only. The doses were based on a point kernel attenuation model, with the source region divided into a number of incremental source volumes, and the associated attenuation and gamma ray buildup computed between each source point and the dose point.

15.6.5.5.4 Effectiveness of Safeguards for Limiting Activity Release

15.6.5.5.4.1 Effectiveness of Containment and Isolation Features in Terminating Activity Release

The Reactor Containment serves as an activity leakage limiting boundary. The Containment is steel lined and designed to withstand internal pressure in excess of that resulting from the design basis LOCA (Section 3.8.1). All weld seams and penetrations are designed with a double barrier to inhibit leakage. In addition, the Weld Channel and Penetration Pressurization (PP) Systems supply a pressurized air or nitrogen seal, at a pressure above the containment design pressure, between the double barriers

so that if leakage occurred it would be into the Containment (Section 6.2.4.5). The Containment Isolation System (Section 6.2.4) provides a minimum of two barriers in piping penetrating the containment. The Isolation Valve Seal Water (IVSW) System (Section 6.2.4.4) provides a water seal at a pressure above containment design in the piping lines that could be a source of leakage. The Containment is designed to leak at a rate of less than 0.1% per day at design pressure without including the benefit of either the IVSW System or the PP System. The weld seams and penetrations are pressurized continuously during reactor operation causing zero outleakage through these paths. The IVSW System is actuated on the containment isolation signal and prevents leakage through the pipelines which could be a leak source. This system would be actuated within one minute to terminate containment leakage.

15.6.5.5.4.2 Effectiveness of Spray System for Iodine Removal

The effectiveness of the spray system for removal of inorganic iodine from the containment atmosphere is evaluated in detail in Appendix 6B.

If there is a large excess of chemical reagent to react with the iodine and convert it to a nonvolatile form with little or no tendency to return to the gas phase, the elemental iodine removal rate by spray can be expressed by:

Equation (1)

$$\frac{dA}{dt} = -\lambda_s A$$

where:

A = inventory of elemental iodine which is available for leakage at any time, t

λ_s = elemental iodine removal coefficient

Integration of equation (1) gives:

Equation (2)

$$A = A_o e^{-\lambda_s t}$$

As discussed in Appendix 6B, an elemental iodine removal coefficient of 54 hr^{-1} is expected for two of three spray pumps operating. It is also assumed that the sprays are no longer effective after the inorganic iodine inventory in the Containment is reduced by a factor of 100.

15.6.5.5.5 Dispersion

The meteorological dispersion of the leakage from the Containment has been calculated using the Pasquill dispersion model and dispersion parameters presented by Hilsmeier and Gifford (Reference 30). The model has been modified to account for additional dispersion of the leakage due to turbulence in the wake of the Containment Building. Conservative dispersion characteristics applicable to four time periods were selected based on the WINDIF results described in Chapter 2 and the doses calculated for each period.

The Pasquill equation for the dispersion of a point source at ground level modified for building turbulence gives the ground level plume concentration as a function of distance:

$$\chi = \frac{Q}{(\pi\sigma_y\sigma_z + cA)\,u}\; e^{\left(-\frac{y^2}{2\sigma_y^2}\right)}$$

where:

σ_y	=	lateral dispersion parameters, m
σ_z	=	vertical dispersion parameter, m
u	=	mean wind speed, m/sec.
c	=	factor, ranging between 0.5 and 2
A	=	projected horizontal area of containment structure, m^2
y	=	lateral distance from the plume centerline, m
Q	=	point source release term

The first and second periods of the dose calculation used this dispersion relationship, a value for c of 0.5 and a buildup area of 1650 m^2. The meteorological parameters used are summarized below:

Period	Time Interval	Stability	Wind Speed, m/sec
1	0 - 2 hours	Pasquill "F"	1
2	2 - 12 hours	Pasquill "F"	2

These choices were based on an examination of five years of record each for Chicago-O'Hare and Milwaukee-General Mitchell Field. In no case did a Class F condition exist for longer than eight hours with invariant wind direction. The mean wind speed for Class F conditions is, at both stations, about 4.2 knots, or 2.2 m/sec.

The third period extends from 12 to 96 hours after the accident and was based on the frequency distribution of persistent wind directions, as shown

in Figure 2.3-6. Milwaukee data shows the probability of winds persisting in a single sector for 38 hours or more to be 10^{-4}; for the same probability, Chicago data indicates a persistence duration of 80 hours. Thus, the assumption of 96 hours for invariant wind direction is conservative. For this period, the stability was assumed to be equally distributed between neutral (Pasquill "D") and stable (Pasquill "F") conditions with corresponding mean wind speeds of 5 and 2 m/sec, respectively. For this period of four days (less the first 12 hours), the average concentration was determined by integrating the Gaussian equation from $y = -\infty$ to $y = +\infty$ and averaging over the sector are:

$$\chi_{av} = \frac{(2)^{1/2}\, f\, F_i\, Q}{(\pi)^{1/2}\, \sigma_{zi}\, \overline{u}_i\, x}$$

where:

- f = fraction of time that the wind direction is in the sector of interest (fraction per radian),
- F_i = fraction of time that stability class i occurs
- x = downwind distance, m
- Q, σ_z and u are as defined previously

For the third period, f was assumed to be 1 and F_i was set equal to 0.5 for the "D" and "F" conditions.

For the fourth period, extending from 4 to 30 days after the accident, a distribution of stability conditions and direction frequencies was selected on the basis of the WINDIF output, as shown below:

Pasquill Class	F_i	f	$\overline{u}_i$, m/sec
C	0.10	0.2	3
D	0.65	0.3	5
F	0.25	0.4	2

This results in a wind direction frequency of over 30% in the sector of interest. The most probable direction frequency during the worst season at either Chicago or Milwaukee was less than 15%; thus, the assumptions herein are conservative.

The average (χ/Q) values were calculated using the relationship given above, and the parameters appropriate to the extended period noted in the

listing. These values, and those calculated for the first five periods are listed in Table 15.6-29.

15.6.5.5.6 Cases Analyzed

15.6.5.5.6.1 One Minute Isolation of Containment

With the IVSW System and the PP System provided to block potential leak paths, the leakage would terminate within one minute, the approximate actuation time for the above systems. For this case it is assumed that the Containment leaks at its design rate for one minute, at which time leakage terminates. This case was analyzed for the case of a TID-14844 (Reference 35) release and for the gap release case. The results are given in Tables 15.6-30 and 15.6-31.

15.6.5.5.6.2 Containment Leaks at Design Rate

The capability of the safeguards systems has also been evaluated without taking credit for the leakage reduction afforded by the IVSW and PP Systems. These results are also presented in Tables 15.6-30 and 15.6-31. For these cases, the integration has been performed over five time intervals with B(t), χ/Q, (x,t) and L(t) constant over each interval. The intervals are:

0	to	2 hrs
2 hrs	to	12 hrs
12 hrs	to	24 hrs
1 day	to	4 days
4 days	to	30 days

It is assumed that iodine release terminates after 30 days because of pressure suppression and removal of airborne iodine activity inside the Containment by the engineered safety features. The Containment is assumed to have a leak rate of 0.1% per day for the first 24 hours and 0.045% per day for the remainder of the 30-day period. The breathing rates used are given in Table 15.6-28.

The values of λ_j and DCF_j are also given in Table 15.6-28. The initial inventories available for release to the Containment are the same as those for the previous case. The sprays are assumed to have an iodine removal coefficient (λ_s) of 54 hr^{-1}.

Figure 15.6-56 shows the thyroid dose at the site boundary and low population zone as a function of spray effectiveness. Figures 15.6-57 through 15.6-60 show the 2-hour and 30-day whole body dose from the leakage cloud taking no credit for leakage termination at one minute.

15.6.5.5.6.3 Recirculation Leakage

Subsequent to the emptying of the RWST during the initial phase of SI and CS, water from the containment sump is recirculated by the RHR pumps and cooled via the RHR heat exchangers and then returned to the RCS. Because the LOCA may cause the sump water to contain radioactivity, the potential offsite exposures due to operation of these external recirculation paths have been evaluated.

As shown in Section 6.3, the maximum estimated leakage to the Auxiliary Building from the components and joints of the SI System components during recirculation is approximately 1380 cubic centimeters per hour.

During the recirculation phase of postaccident cooling, the sump water temperature is calculated to be below saturation at the initiation of recirculation so that essentially no leakage will flash to vapor. For conservatism in the analysis, it is assumed that approximately 10% of leakage or 1380 cc/hr will vaporize and carry the entrained iodine to the atmosphere of the Auxiliary Building for a period of one hour following initiation of recirculation. The Auxiliary Building Ventilation System will discharge the vapor to the atmosphere.

It was also assumed that all of the released iodine activity is in the water in the sump which has a total volume of about 1.3×10^9 cc, including reactor coolant and injection water. The combined leakage from the recirculation system results in a dose of about 0.1 rem to the thyroid in the one hour at the site boundary. The actual dose would be negligible as the temperature of the recirculated water will be substantially reduced so that little or no vaporization should occur. Also the iodine in the sump will be combined with the solution from CS and will remain trapped.

The assumptions used for the releases as assumed for the LOCA are conservatively high enough so that recirculation leak doses would be included.

15.6.5.5.7 Results

The effectiveness of the various safeguards in limiting thyroid dose are evaluated in terms of the reduction in offsite exposure which is achieved, relative to the exposure which would result if the safeguards were not provided. The reference case where no safeguards are employed is evaluated using the fission product release described in TID-14844 (Reference 35).

The ECCS, by preventing core meltdown and limiting the iodine activity released to the 3% of the core inventory contained in the coolant and pellet-cladding gaps, achieves a reduction of 25/3.0 in the iodine source available for leakage. Since the ECCS limits the source, its dose

reduction factor of 8.3 is independent of the other safeguards operating and of the location or duration of exposure.

The dose reduction effect of the IVSW System in terminating leakage in one minute would be a factor of 120 in two hours if it were the only safeguard system provided for reducing iodine leakage. The CS System will reduce the iodine leakage by a factor of seven (based on two of the three spray pumps) in two hours with no termination of leakage. The dose reduction factor for both the isolation system and the spray system are functions of exposure duration.

The dose reduction factor required to meet 10CFR100 limits are 4.8 at the site boundary in two hours and 1.66 at the low population zone boundary in 30 days. The reduction in dose afforded by either the spray system or the isolation system is seen to easily meet 10CFR100 requirements even if no credit were to be taken for the reduction in leakage source obtained in preventing TID-14844 meltdown.

The closest approach to the plant site boundary is 415 meters from the surface of the Reactor Containment. The two-hour thyroid dose at this exclusion distance is 196 rem without termination of containment leakage and with only two of the three spray pumps functioning. For the same condition, the 30-day thyroid dose at the 1610-meter low population zone is 167 rem.

In addition to the thyroid inhalation exposure, the whole body exposures were also evaluated, both due to direct radiation from the Containment and from exposure due to immersion in the leakage plume.

The whole body exposure (beta plus gamma) due to immersion in the plume of leaking gaseous fission products released from the core is about 7.6 rem at the site boundary in two hours with no isolation and about 0.18 rem when containment leakage is terminated in one minute. With no isolation, the corresponding 30-day dose at the low population zone is 4.7 rem.

For the direct dose, the sources are assumed to be homogeneously distributed within the free volume of the Reactor Containment. The source intensity as a function of time after the accident is determined by considering natural decay only, with no credit taken for removal by sprays or washdown. The dose is based on a point kernel attenuation model, with the source region divided into a number of incremental source volumes, and the associated attenuation and gamma ray buildup computed between each source point and the dose point. The combined direct and scattered radiation dose from the confined activity is less than 0.03 mrem for two hours and less than 0.2 mrem for the duration of the accidents at distances beyond the site boundary.

15.6.5.5.8 Summary and Conclusions

Offsite exposure is summarized in Tables 15.6-30 and 15.6-31. It is concluded that even with very pessimistic assumptions that do not take full credit for the safeguards systems provided, both the whole body and thyroid dose after a LOCA would be within the 10CFR100 suggested guidelines.

15.6.5.5.9 Radiological Evaluation Including Life Core Fission Products

The radiological evaluation of a postulated LOCA as described in the above sections was based in TID-14844 fission product inventories. The fission product inventories presented in TID-14844 do not reflect an end-of-life core fission product inventory, which includes contributions from plutonium. The fission product inventories were re-evaluated, taking into account the fission product yield of both uranium and plutonium that is representative of an end-of-life core. The core fission product inventories for TID-14844, and an equilibrium core (beginning, middle, and end-of-life which included plutonium yields) are presented in Table 15.6-32. The assumptions used to calibrate these inventories are given in Table 15.6-33.

The radiological consequences of a postulated LOCA were evaluated using the equilibrium end-of-life core inventories presented in Table 15.6-32. The remainder of the assumptions used in the postulated LOCA analysis are identical to those used in the above sections.

The resulting doses from this analysis are given in Tables 15.6-34 and 15.6-35. The doses are well within the guidelines of 10CFR100.

15.6.5.5.10 Control Room Radiological Impact

A re-evaluation of the Control Room Ventilation System in terms of post-LOCA thyroid doses was performed based on the following set of assumptions:

1. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full-power operation of the core is immediately available for leakage from the primary Reactor Containment, per Atomic Energy Commission (AEC) Safety Guide 4, paragraph C.1.d.

2. Initial gross iodine and tellurium inventories used are slightly higher than given by the procedure outlined in TID 14844 due to an allowance for higher yields from plutonium fissions in ripe fuel (8.2×10^8 curies of iodine isotopes, 2.8×10^8 curies of tellurium isotopes).

3. One percent of the initial tellurium inventory is immediately available for leakage from the primary Reactor Containment. Residual

tellurium in the fuel is allowed to decay into iodine, one fourth of which then becomes available for leakage.

4. Leakage from the primary Reactor Containment is assumed to average 0.1% for the first day and 0.045% per day thereafter.

5. The plume of escaping isotopes is assumed to travel in the worst possible direction; directly into the Turbine Building from which fresh makeup air is drawn.

6. Circulation in the Turbine Building through convective ventilation out of the upper windows is assumed to continue after the LOCA at 2.5 x 10^6 cfm.

7. Free volume of Turbine Building is taken to be 12,000,000 ft^3.

8. Free volume of Control Room - 132,500 ft^3.

9. Tellurium and iodine decay is allowed to proceed during residence in the Turbine Building and Control Room.

10. The eight-hour breathing rate from AEC Safety Guide 4, paragraph 2.c (10 $meters^3$/8 hours) is used throughout.

11. Iodine dose conversion factors as given by ICRP Pub. 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959, are used (see also Table 15.6-28).

12. The containment spray removes 90% of the iodine present as elemental and particulate iodine. No credit is taken for removal of any of the organic iodides.

13. Ingress and egress from the Control Room is assumed to take six minutes per shift (2 times 400 meters to site boundary at 4 miles per hour). It is conservatively assumed that the individual breathes air with the same iodine content as that at the control room filter intake.

Results of the recalculation of control room doses based on the above assumptions are presented in Table 15.6-36.

The results are very conservative since they are based on the 60-day persistence of the worst possible meteorological condition and include no allowances for plateout, fallout, washout, or similar loss factors. In addition, no iodine intake has been allowed for the activated charcoal filters used for smoke and odor removal in the recirculation part of the Control Room Air Conditioning System. Since the tandem HEPA charcoal filters contain a total of approximately 420 pounds of activated charcoal,

the iodine removed from the air supply to the Control Room represents an extremely light loading. No decrease in breathing rate after eight hours has been assumed and no use of available respiratory equipment during ingress and egress has been assumed.

15.6.5.6 Reduction of Hydrogen In Containment After a Loss-of-Coolant Accident

15.6.5.6.1 Introduction and Summary

The Hydrogen Recombiner System will remove hydrogen from the containment after a LOCA. The recombiners are manually aligned by operators to reduce the hydrogen concentration below 0.5%. The recombiners can be operated when the hydrogen concentration in the containment is less than 4% and containment pressure is less than 10 psig. For the design basis LOCA, the containment pressure reduces to less than 10 psig after approximately 10 hours; the postulated hydrogen concentration inside containment at this time is well below 4%. The hydrogen concentration does not reach the flammable limit until after 33 days. This provides for ample time for alignment and operation of the recombiners.

15.6.5.6.2 Hydrogen Production and Accumulation

Hydrogen accumulation in the containment atmosphere following the DBA is the result of production from several sources. Based on Safety Guide 7, the potential sources of

hydrogen that have been identified are the zirconium-water reaction, corrosion of materials of construction, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis has been appraised from two aspects - core solution radiolysis and sump solution radiolysis. Collectively, this analysis of the various hydrogen sources provides a conservative prediction of hydrogen production following the DBA. Table 15.6-37 provides the total volume of hydrogen generated from each of these sources.

15.6.5.6.2.1 Zirconium-Water Reaction

The zirconium water reaction is described by the chemical equation:

$$Zr + 2\ H_2O \rightarrow ZrO_2 + 2\ H_2 + heat$$

The quantity of zirconium which reacts with the core cooling solution will depend on the functioning of the ECCS. System analysis has shown that core cooling initiation is sufficiently rapid to limit the zirconium-water reaction to a maximum 0.1%. For this case, however, 5% (a factor of 50 greater than anticipated) of the fuel cladding has been conservatively assumed to react with the core cooling solution. On this basis, the 41,994 pounds of zirconium cladding in the core will produce approximately 16,600 scf of hydrogen which has been assumed to be released immediately to the containment atmosphere. This is equivalent to approximately 17% of the 30-day inventory of hydrogen.

15.6.5.6.2.2 Corrosion of Plant Metals

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in Containment in the emergency core cooling solution at DBA conditions (see Reference 31). Metals tested include Zircalloy, Inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel and copper. The results of these corrosion tests are reported in detail in Reference 31. Tests conducted at ORNL (see References 32 and 33) also have verified the compatibility of the various materials (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates in the plant, the results of the corrosion tests have shown that only aluminum will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2\ Al + 3\ H_2O \rightarrow Al_2O_3 + 3\ H_2$$

Therefore, three moles of hydrogen are produced for every two moles of aluminum that is oxidized. (Approximately 20 standard cubic feet of hydrogen for each pound of aluminum corroded.)

Following the first 50 minutes of the accident, a steady state corrosion rate of 200 mils/yr is assumed. This steady state corrosion rate corresponds to a constant hydrogen production rate of 0.0326 scfm. Because of the extremely large surface area of paint, the aluminum in the paint is assumed to immediately produce hydrogen at the conversion given above (20 scf for each pound of aluminum). Also, during the first 50 minutes of the accident, it is assumed that the aluminum is exposed to the high pH containment spray solution and elevated containment temperatures. This has the effect of accelerating the steady state corrosion rate. For these assumptions, the corrosion of aluminum within the Containment yields 3100 cubic feet of hydrogen after one day. The 30-day hydrogen production from aluminum corrosion corresponds to approximately 7400 scf which is equivalent to approximately 7.4% of the total 30-day inventory.

The corrodible aluminum materials considered as sources of hydrogen are listed in Table 15.6-38.

Galvanized carbon steel was not considered to be an important source of hydrogen. This is because of low production rate of hydrogen from galvanized carbon steel, in addition to the fact that galvanized coatings are extremely thin.

15.6.5.6.2.3 Emergency Core Cooling Solution Radiolysis

Water radiolysis is a complex process involving the reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$H_2O \rightarrow H_2 + (1/2)O_2$$

The quantitative definition of the rates and extent of radiolytic hydrogen production following the DBA is of primary concern for this analysis.

An extensive program has been conducted by Westinghouse Electric Corporation and ORNL to investigate the radiolytic decomposition of the core cooling solution following the DBA. It has been apparent from this program that two separate radiolytic environments exist in the Containment at LOCA conditions. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In

the other case, the decay of dissolved fission products, which have escaped from the core, results in the radiolysis of the sump solution.

15.6.5.6.2.3.1 Core Solution Radiolysis

As the emergency core cooling solution flows (both injection and recirculation phase) through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy absorption results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from a knowledge of the fission products contained in the core and a detailed analysis of the dissipation of the decay energy between core materials and the solution. Such a detailed analysis has been presented in Reference 34. The results of this study show essentially all of the beta energy will be absorbed within the fuel and cladding and that this represents 50% of the total beta-gamma decay energy. The study further shows that, of the gamma energy, only 7.4% maximum will be absorbed by the solution incore.

For this analysis, it is assumed that all of the beta energy from fission products within the fuel is absorbed within the fuel and cladding. Therefore, 0% is absorbed by the coolant in the core region. It is also assumed that 10% of the gamma energy from fission products within the fuel is absorbed by the coolant in the core region. Thus, an overall absorption factor of 5% of the total core decay energy ($B + \gamma$) has been used to compute solution radiation dose rates and the time-integrated dose. The contained decay energy incore is the same as the assumed TID release of 50% halogens and 1% other fission products. The noble gases are assumed by the TID model and Safety Guide 7 to escape to the containment vapor space, where little or no water radiolysis would result from decay of these nuclides.

The radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse and ORNL. The Westinghouse results have been reported, in part, in Reference 31, a proprietary Westinghouse topical report, and give some guidance to the selection of hydrogen yields or $G(H_2)$ values. These results illustrate, from static capsule tests, that hydrogen yields are much lower than the maximum of 0.44 molecules per 100 ev that can be expected where the gas to liquid volume ratio approaches zero, as would be the case in the core (see Reference 31). Indeed, with no gas space for hydrogen formed in solution to escape to, the rapid back reactions of molecular radiolytic products in solution to reform water is sufficient to result in very low net hydrogen yields.

However, the differences between the static capsule tests and the dynamic condition incore, where core cooling fluid is continuously flowing have been recognized. Such flow, was reasoned to disturb the steady-state conditions which are observed in static capsule tests, and while the occurrence of back reactions would still be significant, the overall, net yield of hydrogen would be somewhat higher in the flowing system. Thus, calculations of hydrogen yield from core radiolysis have been done assuming

a conservative value of 0.5 molecules/100 ev. The results show that approximately 38,000 scf of hydrogen is produced by core solution radiolysis in 30 days. This is equivalent to approximately 38% of the 30-day hydrogen inventory.

15.6.5.6.2.3.2 Sump Solution Radiolysis

Another important source of hydrogen assumed for the postaccident periods arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by dissolved fission products. In this consideration, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis.

The energy deposited in solution was computed using the following assumptions:

1. A TID release model (see Reference 35) was assumed wherein 50% of the total core halogens and 1% of all other fission products, excluding noble gases, are released to the core cooling fluid (sump solution);

2. The quantity of fission product release was postulated to be equal to that from a reactor operating at full power for 830 days prior to the accident; and

3. The total decay energy from the released fission products, both beta and gamma, was postulated to be fully absorbed in the solution.

The first assumption is grossly conservative in that only gap activity is expected to be released from the reactor core, which would represent less than 3% of the contained halogens, not 50%. In this assumption alone, there is conservatism to the extent of a factor greater than 15.

Within the assessment of energy release by dissolved fission products, account was made of the decay of halogens and a separate accounting for the slower decay of the 1% other fission products. To arrive at the energy deposition rate and time-integrated energy deposited, the contribution from each individual fission product was computed.

The yield of hydrogen from sump solution radiolysis is most nearly represented by the static capsule tests performed by Westinghouse and ORNL with the alkaline sodium borate solution. The differences between these tests and the actual conditions for the sump solution are important and render the capsule tests conservative in their predictions of radiolytic hydrogen yields.

First of all, the sump solution will have considerable depth, which prevents the ready diffusion of hydrogen from solution, which was not the case with shallow-depth capsule tests. This retention of hydrogen in solution would have a significant effect in reducing the hydrogen yields to the containment atmosphere. The buildup of hydrogen concentration in solution would enhance the back reaction reformation of water to lower the

net hydrogen yield, in the same manner as a reduction in gas-to-liquid volume ratio would reduce the yield.

Secondly, the solution temperature for the capsule tests was approximately 72°F, considerably lower than the 140°F to 160°F sump temperature anticipated for the postaccident condition. The effect of higher temperature is to reduce the net hydrogen yield because of the increased rate of temperature dependent chemical back reactions to reform water. With these considerations taken into account, a reduced hydrogen yield is a reasonable assumption to make for the case of sump radiolysis. While the yield will be on the order of 0.1 or less, a conservative value of 0.5 molecules/100 ev has been used in this assessment.

The results of this analysis show that approximately 39,000 scf of hydrogen is generated by sump solution radiolysis in the 30 days following the LOCA. This is equivalent to approximately 39% of the 30-day hydrogen inventory.

15.6.5.6.3 Hydrogen Production from Sources not Assumed in Safety Guide 7

Zion Station's coating system meets all criteria of ANSI Standard N5.9 - 1956 and also the proposed ANSI Standard N101.2 including performance under DBA conditions. ANSI Standard N5.9-1056 does not include any reference to hydrogen evolution. Proposed ANSI Standard N101.2 also does not include any reference to hydrogen evolution, but this question was considered during the preparation of N101.2. It was the industry consensus at that time that hydrogen evolution from inorganic zinc coating systems would not be significant and therefore further consideration was dropped from the initial issue of N101.2. Finally, Safety Guide 7 does not contain any reference to the production of hydrogen from zinc-based containment coating systems. For these reasons, the hydrogen generated from the containment coating system is not included in the Safety Guide 7 results given in Table 15.6-37. The discussion of hydrogen produced from coating systems is discussed below for completeness.

Studies performed by ORNL (ORNL-TM-3212 and 3342), based on experimental work by H.E. Zitel of ORNL, indicate that hydrogen can be released from zinc-bearing paint in a high temperature environment.

Evolution of hydrogen from the containment coating system could contribute as much as 7250 cubic feet of hydrogen in 26 hours, after which it is anticipated there would be no further hydrogen yield of any significance from the coating. The H_2 yield value used to calculate the hydrogen generated from the coating system was 1.9 cm^3/cm^2. Table 3.1 of ORNL-TM-3342 also reports a yield value of 1.9 cm^3/cm^2 for sample B-3-1 under the DBA test conditions of five minutes at 240°F, 45 minutes at 220°F, and 22 hours 15 minutes at 160°F. Sample B-3-1 consisted of a 2.0 mil DFT inorganic zinc primer coat and a 3.5 mil DFT organic top coat, whereas the coating used at Zion Station consisted of a 2.5 to 3.5 mil DFT inorganic zinc primer coat and a 3.5 mil DFT organic topcoat. The difference in hydrogen yield between that reported for sample B-3-1 and that which would be obtained from a test of the subject coating system would unquestionably be very small.

Using a revised hydrogen yield, H_y, value of 1.9 cm^3/cm^2 (= 0.0625 ft^3/ft^2), the calculation for total hydrogen yield is as follows:

Total approximate surface area, A_c, of interior face of containment liner, including the dome = 116,000 ft^2

Total evolved hydrogen = $A_c \times H_y$ = 116,000 x 0.0625 = 7,250 ft^3/26 hours.

Numerous DBA tests have been conducted by the manufacturer of the subject coating system which clearly demonstrate acceptable levels of coating stability and surface adherence following a DBA. In addition, numerous tests of various coating systems, including the subject system, have been conducted by ORNL, and these demonstrate the same stability and adhesive qualities following a DBA; one such series of tests are reported in ORNL-TM-2412, Part V, October 1970, by Griess-Row-Watson and West.

15.6.5.6.4 Summary of Hydrogen Sources Considered

The total hydrogen production from the zirconium-water reaction, corrosion of aluminum, core solution radiolysis, and sump solution radiolysis corresponds to the assumptions given in Safety Guide 7 and is shown in Figure 15.6-61. Figure 15.6-61 shows the total hydrogen production rate as a function of time following the DBA. A tabulation of hydrogen generated within the Containment during the first 100 days post-LOCA based upon Safety Guide No. 7 assumptions is given in Table 15.6-37.

The hydrogen generation resulting from the zirconium-water reaction, as described previously, was considered to be complete during the first day. Also, as previously discussed, the hydrogen generated from the 140 lbs. of aluminum within the paint is assumed to be an instantaneous release based on 20 scf for each pound of aluminum. Hydrogen contributions from the other sources, including the remaining aluminum, are reflected in the overall time function. The hydrogen generation from zinc-based containment coating systems is not considered in Safety Guide 7, therefore it is not included in the results given in Figure 15.6-61 or Table 15.6-37.

15.6.5.6.5 Hydrogen Mixing

Adequate incontainment mixing of hydrogen is assured by continued operation of the containment fan cooler units. The inlet (or suction) ducts for these units extend to, roughly, the height of the containment polar crane. The discharge from the fan cooler units is directed into the steam generator/RCP compartment located between the biological shield and the missile wall (polar crane wall). The air flow direction through the fan coolers is from the upper level of the Containment and from the area above the operating floor to the lower level of the Containment in the steam generator/RCP compartment. Evidence to date would indicate that hydrogen pocketing, if it occurs, would occur at or below the operating floor. Therefore, this air flow arrangement ensures mixing by forcing hydrogen to disperse into the main fan cooler flow. In addition, three fan coolers have sufficient capacity to process 3.5 containment air volumes per hour which further ensures total air mixing.

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15.6.6 Boiling Water Reactor Transients

This section is not applicable to Zion Station.

15.6.7 References, Section 15.6

1. Transfer of Iodine from Aqueous Solutions to Saturated Vapor (UCD-621.039.562.5) M.A. Styrikovich, et al.

2. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Vol. 39, Number 3, January 4, 1974.

3. Bordelon, F.M., Massie, H.W., and Zordan, T.A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.

4. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974.

5. (Deleted)

6. Young, M.Y., et al., "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561, January 1980.

7. Bordelon, F.M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301, June 1974.

8. C.C. Lin, "CONTEMPT4/MOD5: An Improvement to CONTEMPT4/MOD4 Multicompartment Containment System Analysis Program for Ice Containment Analysis," NUREG/CR-4001, BNL-NUREG-51824, September 1984.

9. Bordelon, F.M. and Murphy, E.T., "Containment Pressure Analysis Code COCO," WCAP-8720, June 1974.

10. Bordelon, F.M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974.

11. Kelly, R.D., et al., "Calculations Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170, June 1974.

12. Bordelon F.M., et al., "The Westinghouse ECCS Evaluation Model and Supplementary Information," WCAP-8471 (Proprietary), WCAP-8472 (Non-Proprietary), January 1975.

13. (Deleted)

14. S. Bhat, "RCFC Cooling Coils for ComEd Zion 1 & 2 Stations," Marlo Coil Nuclear Cooling Inc. Engineering Report MR-01-84-6486, January 16, 1985.

15. WCAP-7822, December 1971, Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation, G.J. Bohm.

16. S. Fabic, "BLOWDWN-2: Digital Computer Program for Calculation of Hydraulic Transients During a Loss-of-Coolant Accident," Transactions Am. Nucl. Soc. p. 358, 1969 Annual Meeting, Seattle, June 15-19.

17. S. Fabic, "Computer Program WHAM for Calculation of Pressure Velocity, and Force Transients in Liquid Filled Piping Networks," Kaiser Engineers Report No. 67-49-R (November 1967).

18. S. Fabic, "Loss of Coolant Analysis: Comparison Between BLOWDN-2 Code Results and Test Data," WCAP-7401, November 1969.

19. Tagami, Takaski, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)."

20. Kolflat, A., and Chittenden, W.A., "A New Approach to the Design of Containment Shells for Atomic Power Plants." Proc. of Amer. Power Conf., 1957 p. 651-9.

21. McAdams, W.H., Heat Transmission, 3rd Edition, McGraw-Hill Book Co., Inc., New York (1954).

22. Eckert, E.R.G., & Drake, P.M.J., Heat and Mass Transfer, McGraw-Hill Book Co., Inc., New York (1959).

23. Kern, D.Q., Process Heat Transfer, McGraw-Hill Book Co., Inc., New York (1950).

24. Chilton, T.H., and Colburn, A.P., "Mass Transfer (Absorption) Coefficients Prediction from Data on Heat Transfer and Fluid Friction," Imd. Eng. Chem., 26, (1934), pp. 1183-87.

25. WCAP 7336-L, Topical Report - Reactor Containment Fan Cooler Cooling Test Coil, W. L. Boettinger, July 1969.

26. S. Weinberg, Proc. Inst. Mech. Engr., 164, pp. 240-258, 1952.

27. W. Ranz and W. Marshall, Chem. Engr., Prog. 48, 3, pp. 141-146 and 48, 4, pp. 173-180, 1952.

28. J. Perry, "Chemical Engineers Handbook," 3rd Ed. McGraw-Hill, 1950.

29. E. Eckert and J. Gross, "Introduction to Heat and Mass Transfer," McGraw-Hill, 1963.

30. Hilsmeier, W. F., and Grifford, F. A., Graphs for Estimating Atmospheric Dispersion, ORO 545, 1962.

31. Bell, M.J., Bulkowski, J.E., and Picone, L.F., "Investigation of Chemical Additives for Reactor Containment Sprays," WCAP-7153, March 1968, Westinghouse Proprietary.

32. ORNL Nuclear Safety Research and Development Program Bi-Monthly Report for July-August 1968, ORNL-TM-2368, p. 78.

33. ORNL Nuclear Safety Research and Development Program Bi-Monthly Report for September-October 1968, ORNL-TM-2425, p. 53.

34. Sejvar, J., "Distribution of Fission Product Decay Energy in PWR Cores," WCAP-7816, December 1971.

35. DiNunno, J.J., et al, Calculation of Distance Factors for Power and Test Reactor Sites, AEC Report Number TID-14844, March 23, 1962.

36. Slade, D.H., Meteorology and Atomic Energy 1968, AEC Report Number TID-24190, July 1968.

37. Kabadi, J.N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Revision 2 with Addenda, (Proprietary), December 15, 1987.

38. Weiner, R.A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

39. Lee, N., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985.

40. Meyer, P.E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-P-A (Non-Proprietary), October 1986.

41. Rupprecht, S.D., et al., "Westinghouse Small Break LOCA Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary) and WCAP-11372-A (Non-Proprietary), October 1986.

42. Report on Small Break Accidents for Westinghouse NSSS, WCAP 9600, June 1979.

43. Davidson, S.L. and Kramer, W.R. (eds.), "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

44. Letter from S.F. Stimac, CECo, to T.E. Murley, NRC, Supplemental information in support of requested Technical Specification changes for Zion Station in support of VANTAGE 5 fuel, December 30, 1991.

45. Letter from T.M. Anderson (W) (NS-TMA-2354) to Dr. D.F. Ross, Jr. (NRC), "Report under the Requirements of Section II, 1.6 of 10CFR50 Appendix K, Accumulator Safety Injection Interaction due to Injection into the Accumulator Piping," December 22, 1980.

46. Fakhri, A.I., et al., "Reactor Pressure Vessel and Internals System Evaluations for the Zion Units 1 and 2 VANTAGE 5 Fuel Upgrade,"WCAP-12757, originally issued April 1991 (Revised March 1992).

47. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

48. Zion Calculation No. 22S-B-018M-015, "Revision of Table 15.6-37 and Figure 15.6-61 of Zion Station's UFSAR (1993 Revision)," performed by ABB Impell Corp. (later became VECTRA).

TABLE 15.6-1 (1 of 3)

LARGE BREAK RESULTS

I. VANTAGE 5 WITH IFMs FUEL - 1981 EM AND BASH

Results	DECLG C_D = 0.4	DECLG C_D = 0.6	DECLG C_D = 0.8	Maximum ECCS DECLG C_D = 0.4
Peak Clad Temperature (°F)	2065	1994	1933	2041
Peak Clad Temp. Elevation (ft)	6.25	6.0	6.25	6.25
Peak Clad Temp. Time (sec.)	65.7	53.4	57.0	65.8
Max Local Zr/H_2O Reaction (%)	7.45	5.08	4.48	6.42
Max Local Zr/H_2O Rxn Elevation (ft)	6.25	6.0	6.25	6.25
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Elevation (ft)	6.25	6.0	6.25	6.25
Hot Rod Burst Time (sec)	42.0	34.0	37.0	42.0

Inputs

NSSS Power - 102% of 3250 MWt

Peak Linear Power - 102% of 16.434 kW/ft

Local Peaking Factor (at licensed rating) = 2.40

Accumulator Water Volume[1] = 888 ft^3/tank @ 90°F

Steam Generator Tube Plugging Level[2] = 20% (uniform)

TABLE 15.6-1 (2 of 3)

LARGE BREAK RESULTS

II. VANTAGE 5 WITHOUT IFMs FUEL* - 1981 EM AND BASH

Results	DECLG C_D = 0.4	DECLG C_D = 0.6	DECLG C_D = 0.8
Peak Clad Temperature (°F)	2066.43	2023.27	1924.36
Peak Clad Temp. Elevation (ft)	6.25	6.0	6.25
Peak Clad Temp. Time (sec.)	65.64	53.50	57.06
Max Local Zr/H_2O Reaction (%)	7.39	5.627	4.50
Max Local Zr/H_2O Rxn Elevation (ft)	6.25	6.0	6.25
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Hot Rod Burst Elevation (ft)	6.25	6.0	6.25
Hot Rod Burst Time (sec)	42.38	34.49	38.01

Inputs

NSSS Power - 102% of 3250 MWt

Peak Linear Power - 102% of 16.434 kW/ft

Local Peaking Factor (at licensed rating) = 2.40

Accumulator Water Volume[1] = 888 ft^3/tank @ 90°F

Steam Generator Tube Plugging Level[2] = 20% (uniform)

*Analyzed as 15X15 OFA fuel.

TABLE 15.6-1 (3 of 3)

LARGE BREAK RESULTS

Note(s) 1. The accumulator water volume is the maximum allowable Technical Specification value. Maximum water volume in the accumulators is the limiting condition when peak cladding temperature is calculated to occur at the "burst" elevation and shortly after bottom of core recovery. The Technical Specification minimum volume for Zion Station will be 818 ft^3/tank.

2. The 20% steam generator plugging value is composed of a 15% plugging limit with the additional 5% allocated to offset the concern for steam generator tube collapse under LOCA and seismic loads.

TABLE 15.6-2 (1 of 2)

CONTAINMENT DATA FOR ECCS PERFORMANCE

NET FREE VOLUME	2.736×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	62°F
Service Water Temperature	33°F
Outside Temperature	-10°F
SPRAY SYSTEM	
Number of Pumps Operating	3
Runout Flow Rate	3600 gpm/each
Actuation Time	45.0 sec
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers	5
Fastest Postaccident Initiation of Fan Coolers	38 sec
STRUCTURAL HEAT SINKS	

Structure Thickness (in.)	Area (ft^2)
Painted Containment Cylinder .25 steel, 12 concrete; .004 paint	54,447
Painted Containment Dome .25 steel, 12 concrete; .004 paint	15,026
Containment Floor 18 concrete	15,500
Reactor Cavity .25 steel, 12 concrete	2000
Crane Wall, Operating Deck 12 concrete	36,000
Shield Walls 9 concrete	7000

TABLE 15.6-2 (2 of 2)

CONTAINMENT DATA FOR ECCS PERFORMANCE

	Thickness (in.)		Area (Ft^2)
Refueling Canal	.25	steel, 12 concrete	16,000
Misc. Steel Structures	.25	steel	54,860
Misc. Painted Steel	.375	steel; .004 paint	89,300
Misc. Stiffeners	.6249	steel	1060
	5.25	steel, 12 concrete	1147
	.64	steel, 12 concrete	1400
	10.51	steel, 12 concrete	186
	24.51	steel, 12 concrete	54
	.75	steel, 12 concrete	440
	7.287	steel, 12 concrete	603.94
	12.0308	steel, 12 concrete	180.93
Unpainted Containment Cylinder	.25	steel, 12 concrete	14,862
Unpainted Containment Dome	.25	steel, 12 concrete	3712
Unpainted Misc. Steel	.375	steel	32,000

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- Table 15.6-4 Intentionally Deleted -

- Table 15.6-5 Intentionally Deleted -

TABLE 15.6-6

REACTOR INTERNALS ALLOWABLE STRESS CRITERIA PRIMARY STRESSES
(See Notes 1, 2, 9 and 10)

	P_m	$P_m + P_B$
Elastic Analysis	2.45 S_m and 0.75 S_u	3.0 S_m
Plastic Analysis (see notes 5 and 6)	0.67 S_u	2.45 S_m and
Limit Analysis (see note 4)	1.33 L_L	1.33 L_L
Test Results (see note 7)	0.8 L_T	0.8 L_T
Stress Ratio Analysis (see note 8)	S_F	kS_F

P_m = Membrane stress
P_B = Binding stress
S_m = Design stress intensity value at temperature, given by the ASME Section III Code.

NOTES FOR TABLE 15.6-6

Note 1. The symbols P_M, P_B do not represent single quantities but rather sets of six quantities representing the six quantities representing the six stress components σ_t, σ_1, Σ_r, τ_{1r}, τ_{rt} and τ_{t1}.

Note 2. When loads are dynamically applied, consideration should be given to the use of dynamic load amplification and possible changes in modulus of elasticity.

Note 3. For configurations where compressive stresses occur, the stress limits shall be revised to take into account critical buckling stresses. The permissible equivalent static external pressure shall be taken as 2.5 times that given by the rules of Section III Pressure Vessel ASME Code. Where dynamic pressures are involved, the permissible external pressure shall be limited to 75% of the dynamic instability pressures.

Note 4. L_L = Lower bound limit load with an assumed yield point equal to 1.5 S_m. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (nonstrain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

Note 5. S_u = Ultimate strength at temperature. Multiaxiality effects on uniform strength shall be considered.

Note 6. Elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual stress strain curve at the temperature of loading or any approximation to the actual stress-strain curve which everywhere has a lower stress for the same strain as the actual curve may be used. Either the shear or strain energy of distortion flow rule shall be used to account for multiaxial effects.

Note 7. The stress limits given in this criterion need not be satisfied if it can be shown from the test of a prototype or model that the specified loads do not exceed 80% of L_T, for Faulted Conditions. L_T is the ultimate load or load combination used in the test. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for Faulted Conditions.

NOTES FOR TABLE 15.6-6
(continued)

Note 8. Stress ratio is a method of plastic analysis which uses the stress ratio combinations (combination of stresses that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. For Faulted Condition, use $S_F < 2.4S_m$ and $0.75\ S_u$, whichever is smaller.

Note 9. Where deformation is of concern in a component, the deformation shall be limited to 80% of the value given in the Design Specification (no loss of function) for Faulted Conditions.

Note 10. No limit requirements are established for fatigue and secondary stresses, however evaluation of these conditions are recommended when safety requirements are involved.

TABLE 15.6-7

COMPONENTS NOMENCLATURE

Component	Element
Vessel Supports	Number 1 and 49
Barrel Flange and Hold-Down Spring	Numbers 2 through 6
Barrel	Numbers 7 through 10
Lower Core Supports	Numbers 11 through 15
Major Fuel Assemblies	Even Numbers 16 through 38
Minor Fuel Assemblies	Odd Numbers 17 through 39
Upper Internals	Numbers 40 through 48

TABLE 15.6-8

MAXIMUM DEFLECTIONS UNDER BLOWDOWN
(1-MILLISECOND DOUBLE-ENDED BREAK)

Component		Hot Leg	Cold Leg	Seismic	Direction	Maximum Total	Allowable	No Loss of Function
Upper barrel								
- radial inward		0.057	0.0	0.002	Horizontal	0.059	5	10
- radial outward		0.029	0.431	0.002	Horizontal	0.460	4.125	8.25
Upper core plate		0.015	0.016	0	Vertical	0.016	0.100*	0.150
RCC Guide Tubes (deflection as a beam)	(54)	<Allowable		0.010	Horizontal	<Allowable	1.0	1.60 to 1.75
	(2)	<N.L.F		0.010	Horizontal	<N.L.F.	1.0	1.60 to 1.75
		>Allowable			Horizontal	<Allowable		
	(5)	>N.L.F.		0.010	Horizontal	>N.L.F.	1.0	1.60 to 1.75
Fuel Assembly Thimbles (cross section distortion)		~0	~0	~0	Horizontal	~0	0.036	0.072

* Only to assure that the plate will not touch a guide tube.

NOTE: All deflections are in inches.

TABLE 15.6-9

SUMMARY OF MAXIMUM STRESS INTENSITIES
(1-MILLISECOND PIPE BREAK AND SEISMICS)

Component	Hot Leg Break		Cold Leg Break		Maximum Total Seismic		Maximum Total Blowdown Plus Seismic
	Maximum Membrane	Maximum Total	Maximum Membrane	Maximum Total	Vertical	Horizontal	
Barrel (Girth weld)	21,440	31,340	38,900	46,200	100	400	46,700
Barrel-Flange (weld)	19,820	29,720	18,430	47,700	410	600	48,710
Fuel Assembly Top Nozzle Plate	------	28,700	------	8000	0	0	28,700
Fuel Assembly Bottom Nozzle Plate	------	38,700	------	40,800	400	---	41,200
Fuel Assembly Thimbles	6600	6600	2300	2300	---	---	6600
Upper Support Columns (1)	6200	39200	-----	-----	---	---	39,200
(55)	6200	<20,000	-----	-----	---	---	<20,300

Allowable stress:

Maximum Membrane $P_m = 2.4\ S_m = 39{,}800$ psi

Maximum Total $P_m + P_b = 3.0\ S_m = 48{,}800$ psi

S_m (evaluated at 588°F) = 16,600 psi (per winter Addenda ASME Section III Code)

NOTE: All measurements are in psi.

- Table 15.6-10 Intentionally Deleted -

TABLE 15.6-11

STRUCTURAL HEAT SINKS FOR CONTAINMENT RESPONSE ANALYSIS

Heat Sink	Material	Area ft^2	Thickness in.	Density lb/ft^3	Heat Capacity BTU/lb°F	Conductivity BTU/HR ft°F
Containment Cylinder	Steel lined concrete	77,500	1/4	511	0.11	26
Containment Dome	Steel lined concrete	19,500	1/4	511	0.11	26
Containment Floor	Unlined concrete	15,500	12	146	.24	1.6
Reactor Cavity	Steel lined concrete	2000	1/4	511	0.11	26
Crane Wall	Unlined concrete	31,000	12	150	0.186	0.8
Operating Deck	Unlined concrete	5000	12	150	0.186	0.8
Shield Walls	Unlined concrete	7000	9	150	0.186	0.8
Refueling Canal	Steel lined concrete	16,000	1/4	511	0.11	9.4
Misc. Steel Structure	Steel	50,000	1/4	511	0.11	26

TABLE 15.6-12 (1 of 2)

LOCA EVENT CHRONOLOGY FOR ORIGINAL CONTAINMENT RESPONSE ANALYSIS

Break	Double Ended-Cold Leg	6 ft^2	4.5 ft^2	3 ft^2
Break Occurs	0.0	0.0	0.0	0.0
Containment Reaches Peak Pressure (sec)	21.0	24.0	26.0	45.0
Blowdown Over, sec	21.5	24.8	26.0	50.0
ECCS starts, sec[1,4]	25	25	25	25
Containment Spray starts, sec[3,5]	45	45	45	45
Containment Fan Coolers Start, sec[2]	45	45	45	45
Accumulators start to inject water, sec	10.9	12.6	14.0	20.2
Accumulators empty, sec	48	50	50	60
Refueling water storage tank empties, sec	2013	2013	2013	2013
Recirculation and containment spray via RHR starts (Operator action required to initiate)	2013	2013	2013	2013

1 The COCO case assumes 25 seconds for all cases. The SI signal is generated as follows: a) DE-CL - .39 sec, b) 6ft^2 - .84 sec, c) 4.5 ft^2 - .85 sec, and d) 3 ft^2 - .87 sec.

The design maximum delay time between SI signal and the time that the safety injection system is ready to deliver water is 22 seconds.

2 The COCO code assumes 45 seconds for all cases. The containment fan coolers are initiated by the SI signal from pressurizer level and pressure which is generated as indicated above. The design maximum delay time between SI signal and the time that the containment fan coolers are operating at full speed is 43 seconds.

TABLE 15.6-12 (2 of 2)

LOCA EVENT CHRONOLOGY FOR ORIGINAL CONTAINMENT RESPONSE ANALYSIS

3 The COCO code assumes 45 seconds for all cases. The containment spray system is actuated by hi-hi containment pressure signal (23 psig). The delay in receiving the hi-hi containment signal is as follows: a) DE - 5 sec, b) 6 ft^2 - 6 sec, c) 4.5 ft^2 - 7 sec, and d) 3 ft^2 - 9 sec. The design maximum delay time between the hi-hi containment pressure signal and the time that the containment spray system is delivering water at full capacity is 41 seconds.

4 The containment high pressure signal is a backup for generation of the SI signal. The delay in receiving the SI signal from the containment pressure is as follows: a) DE - .75 sec, b) 6 ft^2 - .75 to 1.0 sec, c) 4.5 ft^2 - .75 to 1.0 sec, and d) 3 ft^2 - 1.0 to 2.0 sec.

5 Taking into consideration an additional 47.5 second delay in the initiation of containment spray (i.e. spray flow begins out of nozzles at 92.5 seconds), peak containment pressure will be increased by 1.04 psi.

TABLE 15.6-13

ENERGY SOURCES

1.	Reactor Coolant System Internal Energy	316.3×10^6 Btu
2.	Accumulator Internal Energy (Three)	13.9×10^6
3.	Initial Core Stored Energy	38.1×10^6
4.	Core Internals Metal Energy	12.1×10^6
5.	Reactor Vessel Metal (below vessel nozzle)	15.0×10^6
	Sub Total	395.4×10^6

6.	Core Power Generation During Blowdown		
	a. Double ended	(21.5 sec)	8.6×10^6
	b. 6 ft^2	(24.8 sec)	9.3×10^6
	c. 4.5 ft^2	(26.0 sec)	9.6×10^6
	d. 3 ft^2	(50 sec)	13.6×10^6
7.	$Zr\text{-}H_2O$ reaction		~0.0

	a. Double ended	404.1×10^6
	b. 6 ft^2	404.8×10^6
TOTALS	c. 4.5 ft^2	405.0×10^6
	d. 3 ft^2	409.1×10^6

TABLE 15.6-14

INTEGRATED ENERGY BALANCE

Outside Reactor Coolant System Control Volume	DE	6 ft^2	4.5ft^2	3ft^2
1. Blowdown Enthalpy	344.5	345.4	338.4	350.6
2. Transferred to Steam Generator	-1.8	-.3	5.0	3.0
	342.7	345.1	343.4	353.6
Inside Reactor Coolant System Control Volume				
1. Reactor Coolant Internal Energy remaining in vessel (plus accumulator addition)	16.1	16.0	19.0	26.0
2. Stored in Core	16.0	15.5	14.7	9.1
3. Core Internal Metal	4.9	4.3	3.6	1.2
4. Reactor Vessel Metal	15.0	15.0	15.0	15.0
5. Internal Energy of Water Remaining in Accumulator (Injection not complete)	9.5	8.9	9.4	4.3
	61.5	59.7	61.7	55.66
	404.2	404.8	405.1	409.2

TABLE 15.6-15

INTERNAL ENERGY BALANCE
(all entries are in millions of BTU's)

		Double Ended	6 ft^2	4.5 ft^2	3 ft^2
U_i		7.4	7.4	7.4	7.4
$\Sigma (mh)_{in}$	a/Blowdown	344.5	345.4	338.4	350.6
	b/Sprays	0	0	0	0.3
ΣQ_{in}		~ 0	~ 0	~ 0	~ 0
ΣQ_{out}	a/Structure	16.2	16.7	16.9	20.6
	b/Fans	0	0	0	.3
	Total U_f	335.7	336.1	328.9	337.4

From COCO the final conditions are:

Steam	235.0	252.3	245.2	252.8
Air	8.3	8.3	8.3	8.3
Sump	73.6	74.9	74.9	77.6
	334.9	335.6	328.4	338.7

TABLE 15.6-16

PEAK CONTAINMENT PRESSURES

Break Type	Safeguards Operating	Peak Pressure (psig)	
			+.8
Double Ended Pump Suction(DEPS)	4 fan coolers, 2 spray pumps	45.8	46.6
Double Ended Hot Leg (DEHL)	4 fan coolers, 2 spray pumps	39.8	
Double Ended Cold Leg (DECL)	4 fan coolers, 2 spray pumps	38.4	
0.6 DEPS	4 fan coolers, 2 spray pumps	45.4	
3 ft^2 PS	4 fan coolers, 2 spray pumps	43.6	
DEPS (Flow from one ECCS train)	3 fan coolers, 2 spray pumps	44.7	45.5
DEPS	3 fan coolers, 2 spray pumps	45.9	46.7

TABLE 15.6-17

ADDITIONAL STRUCTURAL HEAT SINKS FOR CONTAINMENT RESPONSE ANALYSIS

	Exposed Surface (ft^2)	Thickness (inches)
Accumulator Tanks	576 404 260	$2\ ^3/_4$ $^3/_8$ $1\ ^{13}/_{32}$
Ducts	7300	$^1/_4$
Fan Coolers	11,760	$^3/_8$
Polar Crane	40,000	$^3/_8$*
Duct	2050	$^1/_4$
CRDM Missile Shield Support Beams	1913	$^3/_8$*
Cable Pan Structural Framing	2279	$^3/_8$*
Steam Generator and Reactor Coolant Pump Supports	14,794	$^3/_8$*
Galleries	37,596	$^3/_8$*

* Both sides exposed

TABLE 15.6-18

MASS AND ENERGY RELEASE OVER BLOWDOWN

Time (Sec)	Mass Rate (lbm/sec)	Energy Rate (BTU/sec)
1.00000E-08	6.61248E+04	3.43247E+07
2.50105E-02	6.61248E+04	3.43247E+07
2.50186E-01	7.11571E+04	3.72120E+07
6.75203E-01	6.13477E+04	3.30338E+07
1.30007E+00	5.43747E+04	3.02425E+07
2.20017E+00	4.32824E+04	2.48544E+07
3.20029E+00	3.59018E+04	2.11422E+07
4.25020E+00	3.16976E+04	1.89517E+07
5.40007E+00	2.86282E+04	1.72621E+07
6.55016E+00	2.66701E+04	1.61317E+07
7.70038E+00	2.55990E+04	1.54883E+07
8.80053E+00	2.41229E+04	1.46062E+07
9.90039E+00	2.27579E+04	1.37243E+07
1.10007E+01	2.18515E+04	1.32595E+07
1.20007E+01	1.85517E+04	1.20816E+07
1.30504E+01	1.80259E+04	1.12783E+07
1.42008E+01	1.70114E+04	1.07334E+07
1.55009E+01	1.51140E+04	9.78390E+06
1.70504E+01	1.32596E+04	8.79639E+06
1.85002E+01	1.19109E+04	7.98460E+06
1.96504E+01	1.16897E+04	7.36935E+06
2.07003E+01	1.11030E+04	6.51444E+06
2.16501E+01	9.49383E+03	5.39679E+06
2.25501E+01	8.54853E+03	4.66998E+06
2.34000E+01	7.72378E+03	4.09195E+06
2.42501E+01	5.98365E+03	3.22452E+06
2.57253E+01	2.84574E+03	1.54730E+06
2.77229E+01	1.61236E+02	2.08499E+05
2.86954E+01	0.	0.
1.00000E+05	0.	0.

TABLE 15.6-19

MASS AND ENERGY RELEASE FOR REFLOOD

Time (Sec)	Mass Rate (lbm/sec)	Energy Rate (BTU/sec)
2.8490000E+01	0.	0.
2.8500000E+01	0.	0.
3.0500000E+01	0.	0.
3.0600000E+01	0.	0.
3.5500000E+01	7.8501751E+02	1.0059575E+06
3.8500000E+01	8.4178010E+02	1.0767557E+06
4.8500000E+01	8.0044957E+02	1.0182715E+06
5.0000000E+01	7.9464092E+02	1.0100316E+06
5.8500000E+01	7.6499884E+02	9.6819819E+05
6.8500000E+01	7.2489503E+02	9.1287659E+05
7.8500000E+01	6.8800906E+02	8.6229601E+05
8.8500000E+01	6.6203819E+02	8.2584164E+05
1.0000000E+02	6.3500856E+02	7.8830868E+05
1.0850000E+02	6.1456972E+02	7.6027595E+05
1.2335000E+02	6.1183357E+02	7.5262950E+05
1.4850000E+02	5.0650133E+02	6.1740934E+05
1.6177900E+02	4.9870118E+02	6.0552863E+05
1.8178100E+02	1.4368489E+02	1.7440187E+05
2.0000000E+02	1.3340997E+02	1.6191978E+05
3.0000000E+02	9.7286706E+01	1.1802778E+05
1.0000000E+03	7.3584759E+01	8.9229170E+04
2.0000000E+03	5.6246115E+01	6.8156188E+04
5.0000000E+03	4.1849646E+01	5.0638420E+04
1.0000000E+04	3.3249525E+01	4.0168814E+04

Entrainment Ends at 123.4 Seconds for Design Case - Mass Release becomes 155 lbm/sec, Energy Release becomes 1.92 x 105 BTU/sec and resumes with this Table at 200 seconds.

Entrainment Ends at 161.78 Seconds for 10' Case

TABLE 15.6-20 (1 of 2)

STRUCTURAL HEAT TRANSFER COEFFICIENTS FOR ORIGINAL CONTAINMENT RESPONSE ANALYSIS

Time (Seconds)	Heat Transfer Coefficient (BTU/hr-°F-Ft^2) Case #1	Heat Transfer Coefficient (BTU/hr-°F-Ft^2) Case #2
.25	16.7	16.7
1.0	33.5	33.5
2.0	47.3	47.3
3.0	58.0	58.0
5.0	74.8	74.8
7.0	88.5	88.5
10.0	105.8	105.8
12.0	115.9	115.9
15.0	129.6	129.6
17.0	138.0	138.0
20.0	149.6	149.6
21.0	153.3	153.3
22.0	156.9	156.9
23.0	160.5	160.5
24.0	163.9	163.9
25.0	167.3	167.3
26.0	170.6	170.6
27.0	173.9	173.9
28.0	177.1	177.1
29.0	177.5	177.5
30.0	171.8	171.8
32.0	161.2	161.2
35.0	147.2	147.2
37.0	139.0	139.0
40.0	128.2	128.2
42.0	122.0	122.0
45.0	113.7	113.7
47.0	108.9	108.9
50.0	102.5	102.5
61.0	86.3	86.3
71.0	78.1	78.1
81.0	73.5	73.4
91.0	71.0	71.0
101.0	69.9	69.9
131.0	69.5	69.9
141.0	68.7	70.3
151.0	68.0	70.7
171.0	66.9	71.1
181.0	66.4	70.5

TABLE 15.6-20 (2 of 2)

STRUCTURAL HEAT TRANSFER COEFFICIENTS FOR ORIGINAL CONTAINMENT RESPONSE ANALYSIS

Time (Seconds)	Heat Transfer Coefficient (BTU/hr-°F-Ft^2) Case #1	Heat Transfer Coefficient (BTU/hr-°F-Ft^2) Case #2
191.0	66.0	70.0
201.0	65.6	69.5
401.0	58.8	62.0
601.0	52.7	55.4
801.0	47.0	49.3
1001.0	41.7	43.6

TABLE 15.6-21

SUMMARY OF AREAS, K FACTORS, LENGTHS AND DIAMETERS USED IN 19 ELEMENT REFLOOD MODEL

#	Element Description	Area (ft^2)	K	L(ft)	D(ft)
1	Hot Leg Nozzle	4.59	.237	-----	-----
2	Hot Leg Piping	4.59	.383	-----	-----
3	Steam Generator Inlet Plenum	4.59	.455	-----	-----
4	S.G. Tubes	11.1	3.712	66.4	0.065
5	S.G. Outlet Plenum	5.24	.305	-----	-----
6	Crossover Piping	5.24	.486	-----	-----
1′	H.L. Nozzle	3 x 4.59	.237	-----	-----
2′	H.L. Piping	3 x 4.59	.383	-----	-----
3′	S.G. Inlet Plenum	3 x 4.59	.455	-----	-----
4′	S.G. Tubes	3 x 11.1	3.712	66.4	0.065
5′	S.G. Outlet Plenum	3 x 5.24	.305	-----	-----
6′	Crossover Piping	3 x 5.24	.486	-----	-----
7′	Pump (Intact Loop)	3 x 4.5	10.4	-----	-----
8′	Cold Leg Piping	3 x 4.12	.278	-----	-----
9′	C.L. Inlet Nozzle	3 x 4.12	.558	-----	-----
10′	Around Downcomer	20.0	.01	-----	-----
11′	C.L. Inlet Nozzle	4.12	.558	-----	-----
12′	C.L. Piping	4.12	.278	-----	-----
13′	Pump (Broken Loop)	4.5	16.8	-----	-----

TABLE 15.6-22

LOCAL DENSITIES FROM REFLOOD 80 SECONDS AFTER BLOWDOWN

Element #	Average Density (lbm/ft^3)
1	0.578
2	0.577
3	0.577
4	0.145
5	0.121
6	0.112
1′	0.579
2′	0.579
3′	0.579
4′	0.167
5′	0.167
6′	0.166
7′	0.165
8′	0.164
9′	0.163
10′	0.163
11′	0.161
12′	0.161
13′	0.159

TABLE 15.6-23

ENERGY INVENTORY IN LOSS-OF-COOLANT ACCIDENTS FOR CONTAINMENT ANALYSIS
(Blowdown Run - Pump Suction DE (G))

	ENERGY ($x10^6$ BTU)
I. SOURCES	
Reactor coolant	314.1
Accumulator (4 of 4)	18.5
Accumulator (Spilling)	
Initial Core Stored	31.2
Initial Thin Metal (Internals)	12.1
Initial Thick Metal (Vessel)	14.9
Core Power Generation (To End of Blowdown)	8.2
TOTAL SOURCES	399.0
II. BLOWDOWN INVENTORY	
A. Available	
Reactor Coolant	314.1
Transferred into Coolant	
Accumulator (4 of 4)	4.9
Core Heat	28.6
Thin Metal Stored Energy	4.5
Thick Metal Stored Energy	0.0
TOTAL AVAILABLE	352.1
B. Transferred Out	
Blowdown	332.9
To Steam Generator	6.9
C. Remaining in the System	12.3
TOTAL ACCOUNTED FOR	352.1
III. POST BLOWDOWN INVENTORY	
Remaining Accumulator	13.6
Spilling Accumulator	
Remaining Core Stored Energy	10.8
Remaining Thin Metal Stored Energy	7.6
Remaining Thick Metal Stored Energy	14.9
TOTAL REMAINING	46.9
IV. BALANCE	
Total Available	352.1
Total Accounted For	352.1

TABLE 15.6-24

REFLOOD ENERGY RELEASE DEPSG BREAK

	W MODEL	AEC MODEL
Safeguards Assumptions	Maximum	Maximum
Height in Core Entrainment Stop	8′	10′
Total Reflood Energy to Containment at End of Entrainment (10^6 BTU)	80.8	105.6
Energy Types:		
1. Core Stored	10.85	10.85
2. Decay Heat	14.76	19.76
3. Thin Metal (Internals)	7.64	7.64
4. Thick Metal (Reactor Vessel Metal)	3.91	5.18
5. Steam Generator Energy	39.7	57.4

TABLE 15.6-25

EXPOSED AREA OF THE ITEMS COMPRISING THE MISCELLANEOUS STEEL STRUCTURES IN TABLE 15.6-11

Item	Exposed Area (ft^2)
Lifting Rigs - $^1/_4$" thick	2840
CRD Vent Fans - $^1/_4$" thick	4710
Hangers, Snubbers - $^1/_4$" thick	8000
Electrical Cable Pans	25,071
Exposed Steel	6435
Manipulator Crane - $^1/_4$" thick	2352
Total	49,408

TABLE 15.6-26

REACTOR VESSEL CAVITY VOLUMES AND VENT AREAS

	With Insulation	Without Insulation
Below Reactor Vessel Nozzles to 541'-6"	4408 ft^3	4781 ft^3
Around Reactor Vessel Nozzles	69 ft^3	210 ft^3
Above Reactor Vessel Nozzles to 617'	39,579 ft^3	39,579 ft^3
Hot Leg Penetrations	1.39 ft^2	4.25 ft^2
Cold Leg Penetrations	1.34 ft^2	4.09 ft^2
Below Reactor Vessel Nozzles	15.38 ft^2	28.24 ft^2
Above Reactor Vessel Nozzles	16.58 ft^2	37.23 ft^2

TABLE 15.6-27

REACTOR COOLANT LOOP COMPARTMENT VOLUMES AND VENT AREAS

The reactor coolant loop compartment free volumes are:

Loop 1	48,900 ft^3
Loop 2	52,723 ft^3
Loop 3	54,073 ft^3
Loop 4	54,540 ft^3

The reactor coolant loop compartment vent areas are:

Compartment vent areas (through 617′)

Loop 1	201.6 ft^2
Loop 2	250.3 ft^2
Loop 3	297.2 ft^2
Loop 4	182.2 ft^2

Vent areas between compartments

Loop 1-4	141.2 ft^2
Loop 2-3	787 ft^2
Loop 1-2	1019 ft^2
Loop 3-4	1219 ft^2

Total compartment vent areas:

Loop 1	1362 ft^2
Loop 2	2056 ft^2
Loop 3	2303 ft^2
Loop 4	1542 ft^2

TABLE 15.6-28

PHYSICAL DATA FOR ISOTOPES

Isotope	Decay Constant λ_i* (Hr^{-1})	Beta Energy** (Mev/Dis.)	Gamma Energy** (Mev/Dis.)	Dose Conversion Factor, DCF_i* (Rem/Curie)
I-131	0.358×10^{-2}	1.83×10^{-1}	3.92×10^{-1}	1.48×10^{6}
I-132	0.296×10^{0}	4.85×10^{-1}	2.13×10^{0}	5.35×10^{4}
I-133	0.331×10^{-1}	4.93×10^{-1}	5.65×10^{-1}	4.00×10^{5}
I-134	0.792×10^{0}	9.41×10^{-1}	1.02×10^{0}	2.50×10^{4}
I-135	0.102×10^{0}	3.61×10^{-1}	1.68×10^{0}	1.24×10^{5}
Xe-133	5.47×10^{-3}	1.55×10^{-1}	2.70×10^{-2}	
Xe-133m	1.26×10^{-2}	2.07×10^{-1}	2.60×10^{-2}	
Xe-135	7.60×10^{-1}	3.04×10^{-1}	2.61×10^{-1}	
Xe-135m	1.03×10^{-1}	1.04×10^{-1}	4.16×10^{-1}	
Kr-85	7.95×10^{-6}	2.21×10^{-1}	4.00×10^{-3}	
Kr-85m	1.59×10^{-1}	2.52×10^{-1}	1.57×10^{-1}	
Kr-87	5.33×10^{-1}	1.34×10^{0}	1.59×10^{0}	
Kr-88	2.50×10^{-1}	3.72×10^{-1}	1.92×10^{0}	

BREATHING RATES

Time Period (Hours)	Breathing Rates (M^3/sec)
0 - 2	3.47×10^{-4}
2 - 24	2.21×10^{-4}
24 - 7720	2.32×10^{-4}

* Reference 34
** Reference 35

TABLE 15.6-29

SITE DISPERSION FACTORS
(χ/Q, sec/m^3)

Distance (meters)	0 - 2 hr	2 - 12 hr	12 - 96 hr	4 - 30 days
300	1.05×10^{-3}	5.3×10^{-4}	9.9×10^{-5}	2.6×10^{-5}
415	9.2×10^{-4}	4.6×10^{-4}	7.8×10^{-5}	1.9×10^{-5}
600	6.8×10^{-4}	3.4×10^{-4}	5.7×10^{-5}	1.4×10^{-5}
1000	4.8×10^{-4}	2.4×10^{-4}	3.5×10^{-5}	8.0×10^{-6}
1610	2.7×10^{-4}	1.4×10^{-4}	1.9×10^{-5}	4.2×10^{-6}
3000	1.1×10^{-4}	5.5×10^{-5}	6.8×10^{-6}	1.6×10^{-6}
6000	4.7×10^{-5}	2.4×10^{-5}	2.6×10^{-6}	5.8×10^{-7}
10,000	2.2×10^{-5}	1.1×10^{-5}	1.1×10^{-6}	2.5×10^{-7}
30,000	6.7×10^{-6}	3.4×10^{-6}	2.8×10^{-7}	6.0×10^{-8}

TABLE 15.6-30

INHALATION DOSES TO THYROID
FROM A LOSS-OF-COOLANT ACCIDENT

	Site Boundary (415 m) 0 - 2 Hours	Low Population Zone (1610 m) 0 - 30 Days
10CFR100	300 rem	300 rem
Containment leakage terminated in one minute by IVSWS, two of three spray pumps operating and PPS operating		
TID 14844 Release	21.0 rem	6.2 rem
Gap Release	0.32 rem	0.10 rem
Continuous leakage, two of three spray pumps operating; no credit for IVSWS and PPS		
TID 14844 Release	196 rem	167 rem
Gap Release	3.5 rem	3.1 rem

TABLE 15.6-31

WHOLE BODY DOSES FROM A LOSS-OF-COOLANT ACCIDENT

	Site Boundary (415 m) 0 - 2 Hours	Low Population Zone (1610 m) 0 - 30 Days
10CFR100	25 rem	25 rem
Containment leakage terminated in one minute by IVSWS; PPS operating		
TID 14844 Release	0.18 rem	0.06 rem
Gap Release	<1.0 mR	<mR
Continuous leakage-No credit for IVSWS or PPS		
TID 14844 Release	7.60 Rem	4.70 Rem
Gap Release	0.04 Rem	0.04 Rem

TABLE 15.6-32

LIFE CORE FISSION PRODUCT INVENTORY
URANIUM-PLUTONIUM FUEL
(3391 Mwt)

Isotope	TID (Curies)	Equil. Cycle BOL (Curies)	Equil. Cycle MOL (Curies)	Equil. Cycle EOL (Curies)
Kr-85M	3.76 + 7	3.14 + 7	2.89 + 7	2.70 + 7
Kr-85	1.23 + 6	1.07 + 6	9.97 + 5	9.40 + 5
Kr-87	7.23 + 7	6.01 + 7	5.50 + 7	5.08 + 7
Kr-88	1.03 + 8	8.54 + 7	7.87 + 7	7.70 + 7
Xe-133M	4.92 + 6	4.98 + 6	4.93 + 6	4.94 + 6
Xe-133	1.94 + 9	1.92 + 8	1.88 + 8	1.88 + 8
Xe-135M	5.19 + 7	5.25 + 7	5.19 + 7	5.20 + 7
Xe-135	5.24 + 7	5.43 + 7	5.43 + 7	5.42 + 7
I-131	8.53 + 7	9.13 + 7	9.22 + 7	9.34 + 7
I-132	1.29 + 8	1.36 + 8	1.37 + 8	1.38 + 8
I-133	1.91 + 8	1.83 + 8	1.79 + 8	1.76 + 8
I-134	2.22 + 8	2.22 + 8	2.18 + 8	2.19 + 8
I-135	1.72 + 8	1.78 + 8	1.76 + 8	1.78 + 8

TABLE 15.6-33

ASSUMPTIONS USED IN REANALYSIS INCORPORATING LIFE CORE FISSION PRODUCTS POWER SOURCES (Mw)

Isotope	Equilibrium Beginning of Life	Equilibrium Middle of Life	Equilibrium End of Life
U-235	2380	2035	1761
U-238	238	238	238
Pu-239	710	981	1154
Pu-241	103	137	238

FISSION PRODUCT YIELDS (%)

	U-235	U-238	Pu-239	Pu-241
Kr-85m	1.3	0.73	0.53	0.30
Kr-85	0.29	0.15	0.11	0.063
Kr-87	2.5	1.7	0.91	0.50
Kr-88	3.0	2.1	1.4	0.7
Xe-133m	0.17	0.17	0.17	0.17
Xe-133	6.7	5.5	6.5	6.5
Xe-135m	1.8	1.8	1.8	1.8
Xe-135	6.3	6.0	7.2	7.3
I-131	2.9	3.2	3.7	3.0
I-132	4.3	4.7	5.4	4.5
I-133	6.7	5.5	5.3	6.5
I-134	7.8	6.6	7.4	7.8
I-135	6.2	5.9	5.7	7.2

TABLE 15.6-34

INHALATION DOSES TO THYROID FROM A POSTULATED LOSS-OF-COOLANT ACCIDENT INCLUDING LIFE CORE FISSION PRODUCTS

	Site Boundary (415 m) 0 - 2 Hours	Low Population Zone (1610 m) 0 - 30 Days
10CFR100	300 rem	300 rem
Containment leakage terminated in one minute by IVSWS, two of three spray pumps operating and PPS operating		
TID 14844 Release	21.7 rem	6.4 rem
Gap Release	0.34 rem	0.11 rem
Continuous leakage, two of three spray pumps operating; no credit for IVSWS and PPS		
TID 14844 Release	203 rem	175 rem
Gap Release	3.7 rem	3.4 rem

TABLE 15.6-35

WHOLE BODY DOSES FROM A POSTULATED LOSS-OF-COOLANT ACCIDENT INCLUDING LIFE CORE FISSION PRODUCTS

	Site Boundary (415 m) 0 - 2 Hours	Low Population Zone (1610 m) 0 - 30 Days
10CFR100	25 rem	25 rem
Containment leakage terminated in one minute by IVSWS; PPS operating		
TID 14844 Release	0.13 rem	0.05 rem
Gap Release	<1.0 mR	<0.5mR
Continuous leakage-No credit for IVSWS or PPS		
TID 14844 Release	5.67 rem	3.64 rem
Gap Release	0.04 rem	0.07 rem

TABLE 15.6-36

CONTROL ROOM DOSE RATES AFTER A LOSS-OF-COOLANT ACCIDENT

Elapsed time after LOCA (hours)	Accumulated Thyroid Dose (Continuous Occupancy) (Rem)	Accumulated Thyroid Dose (1/3 Occupancy*) (Rem)	Iodine Trapped on Filter (milligrams)
36	.165	.125	3
92	.257	.194	6
140	.318	.238	9
308	.462	.347	18
812	.607	.456	44
1484	.632	.474	81
8760	.664	.498	85

* Including allowance for ingress and egress this dose approximates individual exposures for one eight hour shift per day.

TABLE 15.6-37 (1 of 5)

TOTAL HYDROGEN GENERATION

TOTAL HYDROGEN GENERATED BY ZIRC-WATER REACTION = 0.166E+05 SCF

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		TOTAL RECOMBINED		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	VOL. PCT.
1	.34E+01	.73E+04	.15E+01	.31E+04	.976E-01	.33E+04	0.	0.	.51E+01	.31E+05	1.23%
2	.25E+01	.12E+05	.13E+01	.52E+04	.976E-01	.34E+04	0.	0.	.40E+01	.37E+05	1.48%
3	.20E+01	.15E+05	.12E+01	.70E+04	.976E-01	.36E+04	0.	0.	.33E+01	.42E+05	1.69%
4	.16E+01	.17E+05	.11E+01	.87E+04	.976E-01	.37E+04	0.	0.	.29E+01	.46E+05	1.86%
5	.14E+01	.19E+05	.11E+01	.10E+05	.976E-01	.39E+04	0.	0.	.26E+01	.50E+05	2.02%
6	.12E+01	.21E+05	.10E+01	.12E+05	.976E-01	.40E+04	0.	0.	.23E+01	.54E+05	2.15%
7	.10E+01	.23E+05	.99E+00	.13E+05	.976E-01	.41E+04	0.	0.	.21E+01	.57E+05	2.28%
8	.91E+00	.24E+05	.96E+00	.15E+05	.976E-01	.43E+04	0.	0.	.20E+01	.60E+05	2.39%
9	.81E+00	.26E+05	.93E+00	.16E+05	.976E-01	.44E+04	0.	0.	.19E+01	.62E+05	2.50%
10	.73E+00	.27E+05	.90E+00	.17E+05	.976E-01	.46E+04	0.	0.	.18E+01	.65E+05	2.59%
11	.67E+00	.28E+05	.87E+00	.19E+05	.976E-01	.47E+04	0.	0.	.17E+01	.68E+05	2.68%
12	.61E+00	.29E+05	.85E+00	.20E+05	.976E-01	.49E+04	0.	0.	.16E+01	.70E+05	2.77%
13	.57E+00	.29E+05	.83E+00	.21E+05	.976E-01	.50E+04	0.	0.	.15E+01	.72E+05	2.85%
14	.53E+00	.30E+05	.81E+00	.22E+05	.976E-01	.51E+04	0.	0.	.15E+01	.74E+05	2.94%
15	.50E+00	.31E+05	.78E+00	.23E+05	.976E-01	.53E+04	0.	0.	.14E+01	.76E+05	3.01%
16	.47E+00	.32E+05	.77E+00	.24E+05	.976E-01	.54E+04	0.	0.	.14E+01	.78E+05	3.09%
17	.45E+00	.32E+05	.75E+00	.26E+05	.976E-01	.56E+04	0.	0.	.13E+01	.80E+05	3.16%
18	.43E+00	.33E+05	.73E+00	.27E+05	.976E-01	.57E+04	0.	0.	.13E+01	.81E+05	3.22%
19	.41E+00	.34E+05	.71E+00	.28E+05	.976E-01	.58E+04	0.	0.	.12E+01	.84E+05	3.30%
20	.40E+00	.34E+05	.70E+00	.29E+05	.976E-01	.60E+04	0.	0.	.12E+01	.86E+05	3.36%

TABLE 15.6-37 (2 of 5)

TOTAL HYDROGEN GENERATION

TOTAL HYDROGEN GENERATED BY ZIRC-WATER REACTION = 0.166E+05 SCF

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		TOTAL RECOMBINED		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	VOL. PCT.
21	.38E+00	.35E+05	.68E+00	.30E+05	.976E-01	.61E+04	0.	0.	.12E+01	.87E+05	3.43%
22	.37E+00	.35E+05	.67E+00	.31E+05	.976E-01	.63E+04	0.	0.	.12E+01	.89E+05	3.48%
23	.36E+00	.36E+05	.66E+00	.32E+05	.976E-01	.64E+04	0.	0.	.11E+01	.90E+05	3.55%
24	.35E+00	.36E+05	.65E+00	.33E+05	.976E-01	.65E+04	0.	0.	.11E+01	.92E+05	3.61%
25	.34E+00	.37E+05	.63E+00	.33E+05	.976E-01	.67E+04	0.	0.	.11E+01	.93E+05	3.67%
26	.34E+00	.37E+05	.62E+00	.34E+05	.976E-01	.68E+04	0.	0.	.11E+01	.95E+05	3.72%
27	.33E+00	.38E+05	.61E+00	.35E+05	.976E-01	.70E+04	0.	0.	.10E+01	.96E+05	3.78%
28	.32E+00	.38E+05	.60E+00	.36E+05	.976E-01	.71E+04	0.	0.	.10E+01	.98E+05	3.83%
29	.31E+00	.39E+05	.59E+00	.37E+05	.976E-01	.72E+04	0.	0.	.10E+01	.10E+06	3.88%
30	.31E+00	.39E+05	.58E+00	.38E+05	.976E-01	.74E+04	0.	0.	.99E+00	.10E+06	3.94%
31	.30E+00	.40E+05	.57E+00	.39E+05	.976E-01	.75E+04	0.	0.	.97E+00	.10E+06	3.99%
32	.30E+00	.40E+05	.56E+00	.39E+05	.976E-01	.77E+04	0.	0.	.96E+00	.10E+06	4.04%
33	.29E+00	.40E+05	.55E+00	.40E+05	.976E-01	.78E+04	0.	0.	.94E+00	.10E+06	4.09%
34	.29E+00	.41E+05	.55E+00	.41E+05	.976E-01	.80E+04	0.	0.	.93E+00	.10E+06	4.15%
35	.28E+00	.41E+05	.54E+00	.42E+05	.976E-01	.81E+04	0.	0.	.92E+00	.10E+06	4.19%
36	.28E+00	.42E+05	.53E+00	.43E+05	.976E-01	.82E+04	0.	0.	.91E+00	.10E+06	4.25%
37	.27E+00	.42E+05	.52E+00	.43E+05	.976E-01	.84E+04	0.	0.	.89E+00	.11E+06	4.28%
38	.27E+00	.42E+05	.52E+00	.44E+05	.976E-01	.85E+04	0.	0.	.88E+00	.11E+06	4.33%
39	.26E+00	.43E+05	.51E+00	.45E+05	.976E-01	.87E+04	0.	0.	.87E+00	.11E+06	4.37%
40	.26E+00	.43E+05	.50E+00	.46E+05	.976E-01	.88E+04	0.	0.	.86E+00	.11E+06	4.42%

TABLE 15.6-37 (3 of 5)

TOTAL HYDROGEN GENERATION

TOTAL HYDROGEN GENERATED BY ZIRC-WATER REACTION = 0.166E+05 SCF

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		TOTAL RECOMBINED		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	VOL. PCT.
41	.25E+00	.44E+05	.50E+00	.46E+05	.976E-01	.89E+04	0.	0.	.85E+00	.12E+06	4.48%
42	.25E+00	.44E+05	.49E+00	.47E+05	.976E-01	.91E+04	0.	0.	.84E+00	.12E+06	4.51%
43	.25E+00	.44E+05	.48E+00	.48E+05	.976E-01	.92E+04	0.	0.	.83E+00	.12E+06	4.56%
44	.24E+00	.45E+05	.48E+00	.48E+05	.976E-01	.94E+04	0.	0.	.82E+00	.12E+06	4.60%
45	.24E+00	.45E+05	.47E+00	.49E+05	.976E-01	.95E+04	0.	0.	.81E+00	.12E+06	4.65%
46	.24E+00	.45E+05	.47E+00	.50E+05	.976E-01	.96E+04	0.	0.	.80E+00	.13E+06	4.68%
47	.23E+00	.46E+05	.46E+00	.50E+05	.976E-01	.98E+04	0.	0.	.79E+00	.13E+06	4.72%
48	.23E+00	.46E+05	.46E+00	.51E+05	.976E-01	.99E+04	0.	0.	.78E+00	.13E+06	4.76%
49	.23E+00	.46E+05	.45E+00	.52E+05	.976E-01	.10E+05	0.	0.	.77E+00	.13E+06	4.80%
50	.22E+00	.47E+05	.45E+00	.52E+05	.976E-01	.10E+05	0.	0.	.77E+00	.13E+06	4.84%
51	.22E+00	.47E+05	.44E+00	.53E+05	.976E-01	.10E+05	0.	0.	.76E+00	.13E+06	4.89%
52	.22E+00	.47E+05	.44E+00	.54E+05	.976E-01	.10E+05	0.	0.	.75E+00	.13E+06	4.93%
53	.21E+00	.48E+05	.43E+00	.54E+05	.976E-01	.11E+05	0.	0.	.74E+00	.13E+06	4.97%
54	.21E+00	.48E+05	.43E+00	.55E+05	.976E-01	.11E+05	0.	0.	.74E+00	.13E+06	5.01%
55	.21E+00	.48E+05	.42E+00	.56E+05	.976E-01	.11E+05	0.	0.	.73E+00	.13E+06	5.05%
56	.20E+00	.48E+05	.42E+00	.56E+05	.976E-01	.11E+05	0.	0.	.72E+00	.14E+06	5.07%
57	.20E+00	.49E+05	.41E+00	.57E+05	.976E-01	.11E+05	0.	0.	.71E+00	.14E+06	5.10%
58	.20E+00	.49E+05	.41E+00	.57E+05	.976E-01	.11E+05	0.	0.	.71E+00	.14E+06	5.14%
59	.20E+00	.49E+05	.41E+00	.58E+05	.976E-01	.11E+05	0.	0.	.70E+00	.14E+06	5.19%
60	.19E+00	.50E+05	.40E+00	.58E+05	.976E-01	.12E+05	0.	0.	.69E+00	.14E+06	5.22%

TABLE 15.6-37 (4 of 5)

TOTAL HYDROGEN GENERATION

TOTAL HYDROGEN GENERATED BY ZIRC-WATER REACTION = 0.166E+05 SCF

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		TOTAL RECOMBINED		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	VOL. PCT.
61	.19E+00	.50E+05	.40E+00	.59E+05	.976E-01	.12E+05	0.	0.	.69E+00	.14E+06	5.26%
62	.19E+00	.50E+05	.39E+00	.60E+05	.976E-01	.12E+05	0.	0.	.68E+00	.14E+06	5.30%
63	.19E+00	.50E+05	.39E+00	.60E+05	.976E-01	.12E+05	0.	0.	.68E+00	.14E+06	5.33%
64	.18E+00	.51E+05	.39E+00	.61E+05	.976E-01	.12E+05	0.	0.	.67E+00	.14E+06	5.37%
65	.18E+00	.51E+05	.38E+00	.61E+05	.976E-01	.12E+05	0.	0.	.66E+00	.14E+06	5.40%
66	.18E+00	.51E+05	.38E+00	.62E+05	.976E-01	.12E+05	0.	0.	.66E+00	.14E+06	5.44%
67	.18E+00	.51E+05	.38E+00	.62E+05	.976E-01	.13E+05	0.	0.	.65E+00	.15E+06	5.46%
68	.17E+00	.52E+05	.37E+00	.63E+05	.976E-01	.13E+05	0.	0.	.65E+00	.15E+06	5.49%
69	.17E+00	.52E+05	.37E+00	.63E+05	.976E-01	.13E+05	0.	0.	.64E+00	.15E+06	5.53%
70	.17E+00	.52E+05	.37E+00	.64E+05	.976E-01	.13E+05	0.	0.	.63E+00	.15E+06	5.56%
71	.17E+00	.52E+05	.36E+00	.65E+05	.976E-01	.13E+05	0.	0.	.63E+00	.15E+06	5.59%
72	.16E+00	.53E+05	.36E+00	.65E+05	.976E-01	.13E+05	0.	0.	.62E+00	.15E+06	5.62%
73	.16E+00	.53E+05	.36E+00	.66E+05	.976E-01	.13E+05	0.	0.	.62E+00	.15E+06	5.66%
74	.16E+00	.53E+05	.35E+00	.66E+05	.976E-01	.14E+05	0.	0.	.61E+00	.15E+06	5.70%
75	.16E+00	.53E+05	.35E+00	.67E+05	.976E-01	.14E+05	0.	0.	.61E+00	.15E+06	5.72%
76	.16E+00	.54E+05	.35E+00	.67E+05	.976E-01	.14E+05	0.	0.	.60E+00	.15E+06	5.76%
77	.15E+00	.54E+05	.35E+00	.68E+05	.976E-01	.14E+05	0.	0.	.60E+00	.15E+06	5.78%
78	.15E+00	.54E+05	.34E+00	.68E+05	.976E-01	.14E+05	0.	0.	.59E+00	.15E+06	5.82%
79	.15E+00	.54E+05	.34E+00	.69E+05	.976E-01	.14E+05	0.	0.	.59E+00	.15E+06	5.85%
80	.15E+00	.54E+05	.34E+00	.69E+05	.976E-01	.14E+05	0.	0.	.58E+00	.16E+06	5.86%

TABLE 15.6-37 (5 of 5)

TOTAL HYDROGEN GENERATION

TOTAL HYDROGEN GENERATED BY ZIRC-WATER REACTION = 0.166E+05 SCF

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		TOTAL RECOMBINED		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	VOL. PCT.
81	.15E+00	.55E+05	.33E+00	.70E+05	.976E-01	.15E+05	0.	0.	.58E+00	.16E+06	5.89%
82	.14E+00	.55E+05	.33E+00	.70E+05	.976E-01	.15E+05	0.	0.	.58E+00	.16E+06	5.92%
83	.14E+00	.55E+05	.33E+00	.71E+05	.976E-01	.15E+05	0.	0.	.57E+00	.16E+06	5.95%
84	.14E+00	.55E+05	.33E+00	.71E+05	.976E-01	.15E+05	0.	0.	.57E+00	.16E+06	5.98%
85	.14E+00	.55E+05	.32E+00	.71E+05	.976E-01	.15E+05	0.	0.	.56E+00	.16E+06	6.02%
86	.14E+00	.56E+05	.32E+00	.72E+05	.976E-01	.15E+05	0.	0.	.56E+00	.16E+06	6.04%
87	.14E+00	.56E+05	.32E+00	.72E+05	.976E-01	.15E+05	0.	0.	.55E+00	.16E+06	6.07%
88	.13E+00	.56E+05	.32E+00	.73E+05	.976E-01	.16E+05	0.	0.	.55E+00	.16E+06	6.10%
89	.13E+00	.56E+05	.31E+00	.73E+05	.976E-01	.16E+05	0.	0.	.55E+00	.16E+06	6.13%
90	.13E+00	.56E+05	.31E+00	.74E+05	.976E-01	.16E+05	0.	0.	.54E+00	.16E+06	6.15%
91	.13E+00	.57E+05	.31E+00	.74E+05	.976E-01	.16E+05	0.	0.	.54E+00	.16E+06	6.19%
92	.13E+00	.57E+05	.31E+00	.75E+05	.976E-01	.16E+05	0.	0.	.53E+00	.16E+06	6.22%
93	.13E+00	.57E+05	.30E+00	.75E+05	.976E-01	.16E+05	0.	0.	.53E+00	.16E+06	6.24%
94	.13E+00	.57E+05	.30E+00	.76E+05	.976E-01	.16E+05	0.	0.	.53E+00	.16E+06	6.27%
95	.12E+00	.57E+05	.30E+00	.76E+05	.976E-01	.17E+05	0.	0.	.52E+00	.17E+06	6.28%
96	.12E+00	.58E+05	.30E+00	.76E+05	.976E-01	.17E+05	0.	0.	.52E+00	.17E+06	6.31%
97	.12E+00	.58E+05	.29E+00	.77E+05	.976E-01	.17E+05	0.	0.	.51E+00	.17E+06	6.33%
98	.12E+00	.58E+05	.29E+00	.77E+05	.976E-01	.17E+05	0.	0.	.51E+00	.17E+06	6.36%
99	.12E+00	.58E+05	.29E+00	.78E+05	.976E-01	.17E+05	0.	0.	.51E+00	.17E+06	6.38%
100	.12E+00	.58E+05	.29E+00	.78E+05	.976E-01	.17E+05	0.	0.	.50E+00	.17E+06	6.41%

TABLE 15.6-38

CORRODIBLE ALUMINUM MATERIALS WITHIN CONTAINMENT CONSIDERED AS SOURCES OF HYDROGEN

	WEIGHT	AREA
Nuclear Instrumentation Detectors	162 lbs	≤40 ft^2
Power Range Nuclear Instrumentation Moderator Jacket	230 lbs	130 ft^2
Miscellaneous Instrumentation and Control Equipment	20 lbs	8 ft^2
Control Rod Drive Mechanism Connectors	122 lbs	42 ft^2
Paint	140 lbs	***
RCP Vibration Monitoring Instrumentation	30 lbs	51 ft^2
FOP Ladder (One in each Containment)	<50 lbs	13.1 ft^2
Reactor Coolant Pumps (Total for 4 RCPs)	524 lbs	51.2 ft^2
Control Rod Drive Fan Wheels (Total for 2 CRD Fans)	640 lbs	75.8 ft^2
Total Al Present	1918	411.1 ft^2
Total Allowable	N/A	915 ft^2(Rev. 11)
Margin Available (Contingency)	N/A	503.9 ft^2

*** Assumed that 100% of Al in paint reacts within first 24 hours.

NOTE

With the exception of paint (see *) the total Al surface area present is assumed constant (i.e., infinite mass of Al with constant surface area is assumed), therefore the parameter of concern is area, not weight/mass.**

- Table 15.6-39 was Intentionally Deleted -

TABLE 15.6-40

LARGE BREAK - VANTAGE 5 WITH IFMs FUEL - SEQUENCE OF EVENTS

	DECLG C_D=0.4	DECLG C_D=0.6	DECLG C_D=0.8	MAXIMUM ECCS DECLG C_D=0.4
Start	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.967	0.946	0.934	0.967
SI Signal	2.88	2.35	2.17	2.88
Accumulator Injection	19.0	14.0	11.9	19.0
Pump Injection Begins	33.88	33.35	33.17	33.88
End of Bypass	38.729	30.407	27.54	38.729
End of Blowdown	38.729	30.407	27.54	38.729
Bottom of Core Recovery	55.430	45.17	42.042	54.251
Accumulators Empty	67.923	61.27	58.052	69.10

Note: All times are in seconds

TABLE 15.6-41

SMALL BREAK LOCA ANALYSIS

TIME SEQUENCE OF EVENTS

	COLD LEG BREAK 2-INCH DIAMETER	COLD LEG BREAK 3-INCH DIAMETER	COLD LEG BREAK 4-INCH DIAMETER
Start	0.00 sec	0.00 sec	0.00 sec
Reactor Trip Signal	56.52 sec	23.16 sec	13.48 sec
SI Signal	56.52 sec	23.16 sec	13.48 sec
Pumped ECCS Flow Begins	109.06 sec	63.92 sec	54.17 sec
Top of Core Uncovered	2026.98 sec	906.75 sec	602.04 sec
Accumulator Injection Begins	NA sec	1523.87 sec	838.05 sec
Peak Cladding Temperature Occurs	3487.33 sec	1583.78 sec	905.48 sec
Top of Core Recovered	5083.87 sec	3449.06 sec	1507.80 sec

TABLE 15.6-42

SMALL BREAK RESULTS - VANTAGE 5 WITH IFMs FUEL - NOTRUMP MODEL

RESULTS	BOL (500 MWD/MTU) COLD LEG BREAK 2-INCH DIAMETER	BOL (500 MWD/MTU) COLD LEG BREAK 3-INCH DIAMETER	BOL (500 MWD/MTU) COLD LEG BREAK 4-INCH DIAMETER
Peak Clad Temperature (°F)	1637.88	1962.62	1349.12
Peak Clad Temp. Elevation (ft.)	11.75	11.75	11.0
Peak Clad Temperature time (sec)	3487.33	1583.78	905.48
Max Local Zr/H_2O Reaction (%)	2.007	5.519	0.121
Max Local Zr/H_2O Rxn Elev (ft.)	11.75	11.75	11.00
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec.)	NO BURST	1581.73	NO BURST
Hot Rod Burst Elevation (ft.)	NA	11.75	NA

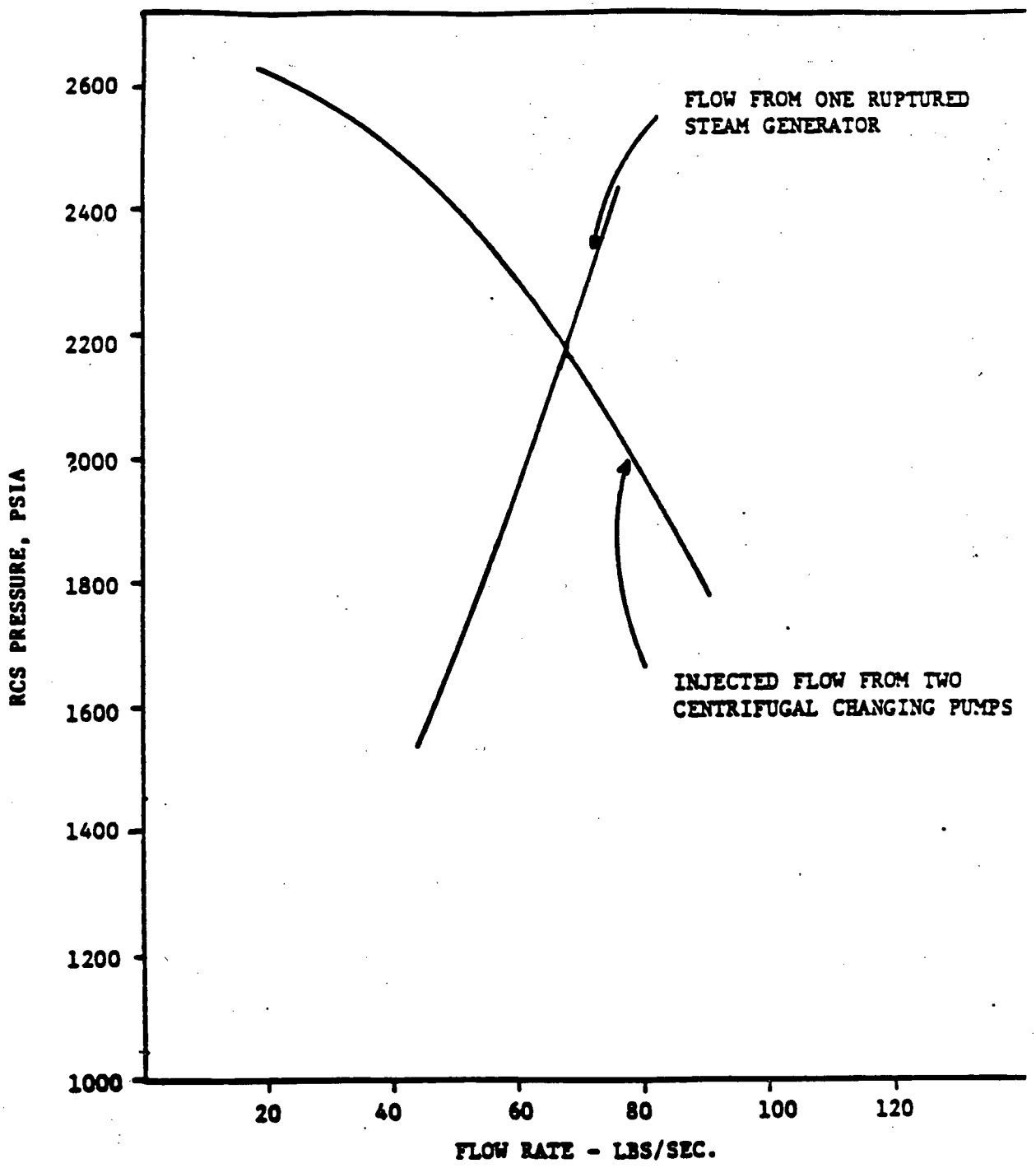
TABLE 15.6-43

SMALL BREAK LOCA ANALYSIS INPUT ASSUMPTIONS

Reactor Trip Signal	= 1700 psia, pressurizer pressure LOW
SI signal	= 1700 psia, pressurizer pressure LOW-LOW
Accumulator Water Volume[1]	= 853 ft^3/tank
Steam Generator Tube Plugging Level[2]	= 20% (Uniform)
Auxiliary Feedwater Flow Rate	= 259 gpm @ 1125 psia (Total flow to all four SGs)
MSSV Setpoint Tolerance	= Technical Specification setpoint pressure $\pm$ 3%
NSSS Power - 102% of	3250 MWt
Peak Linear Power - 102% of	See Figure 15.6-3
Maximum Allowable Peaking Factor	2.50 (0.0 to 6.0 ft), 2.31 at 12.0 ft
Enthalpy Rise Peaking Factor ($F^N_{\Delta H}$)	1.70

1 - The accumulator water volume of 853 ft^3 represents the nominal water volume between the Technical Specification allowable minimum and maximum volume. Since small break LOCA is less sensitive than large break to accumulator water volume, small break methods utilize the nominal water volume in the analysis.

2 - The 20% steam generator plugging value is composed of a 15% plugging limit with the additional 5% allocated to offset the concern for steam generator tube collapse under LOCA and seismic loads.



ZION STATION UFSAR

Figure 15.6-1

BREAK AND INJECTED MASS FLOWS

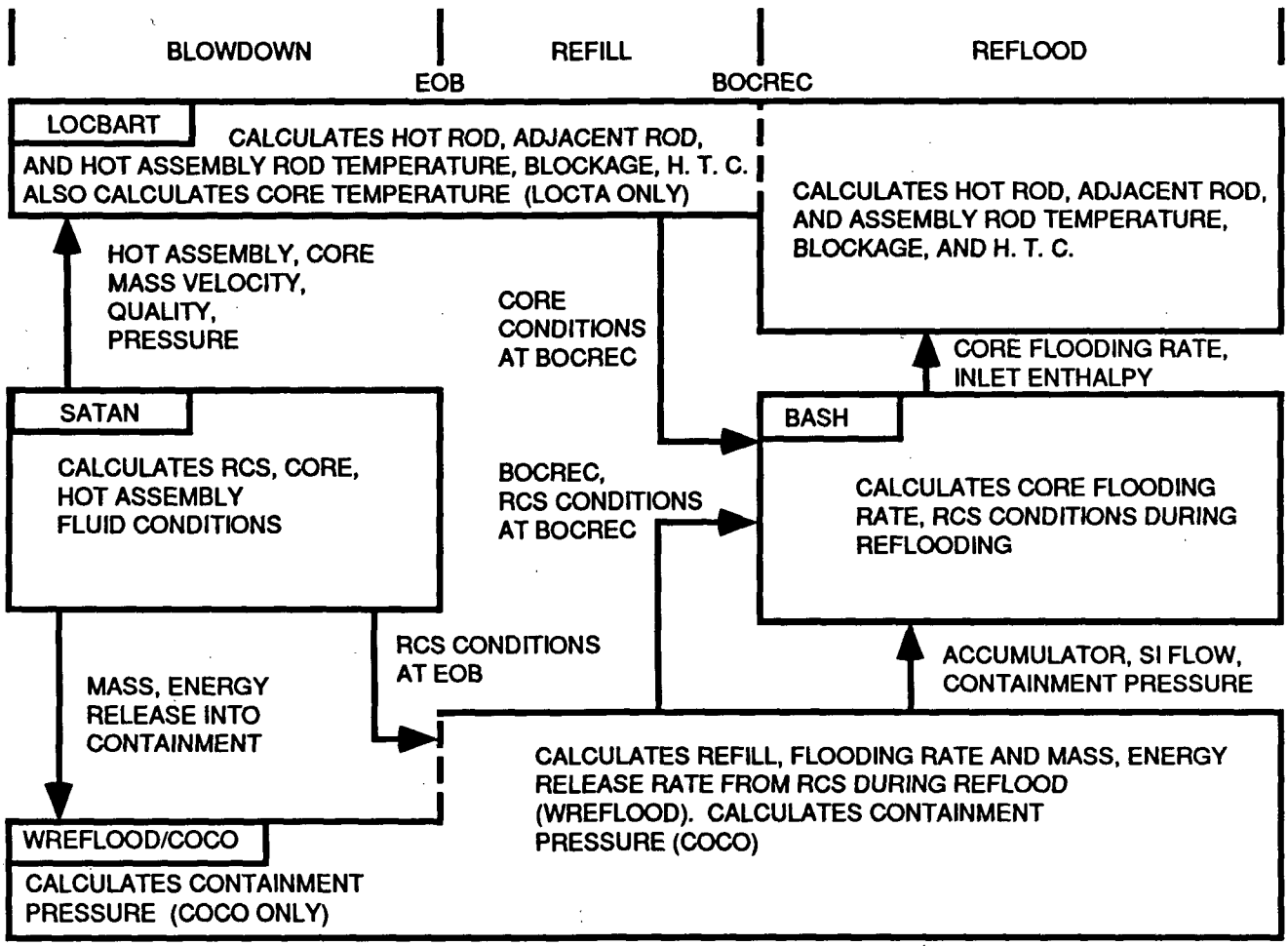


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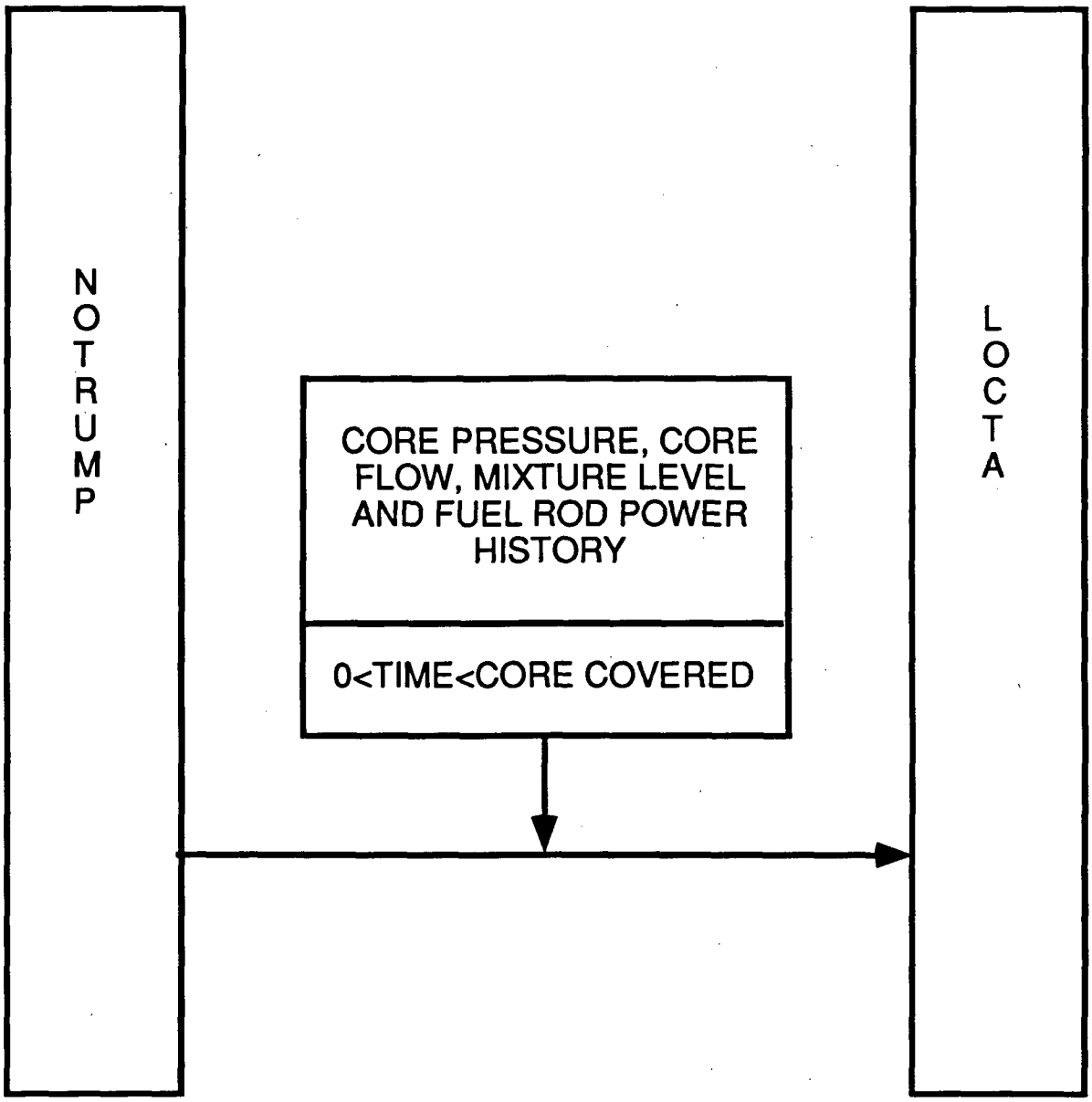
Figure 15.6-2A

CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL

JULY 1993

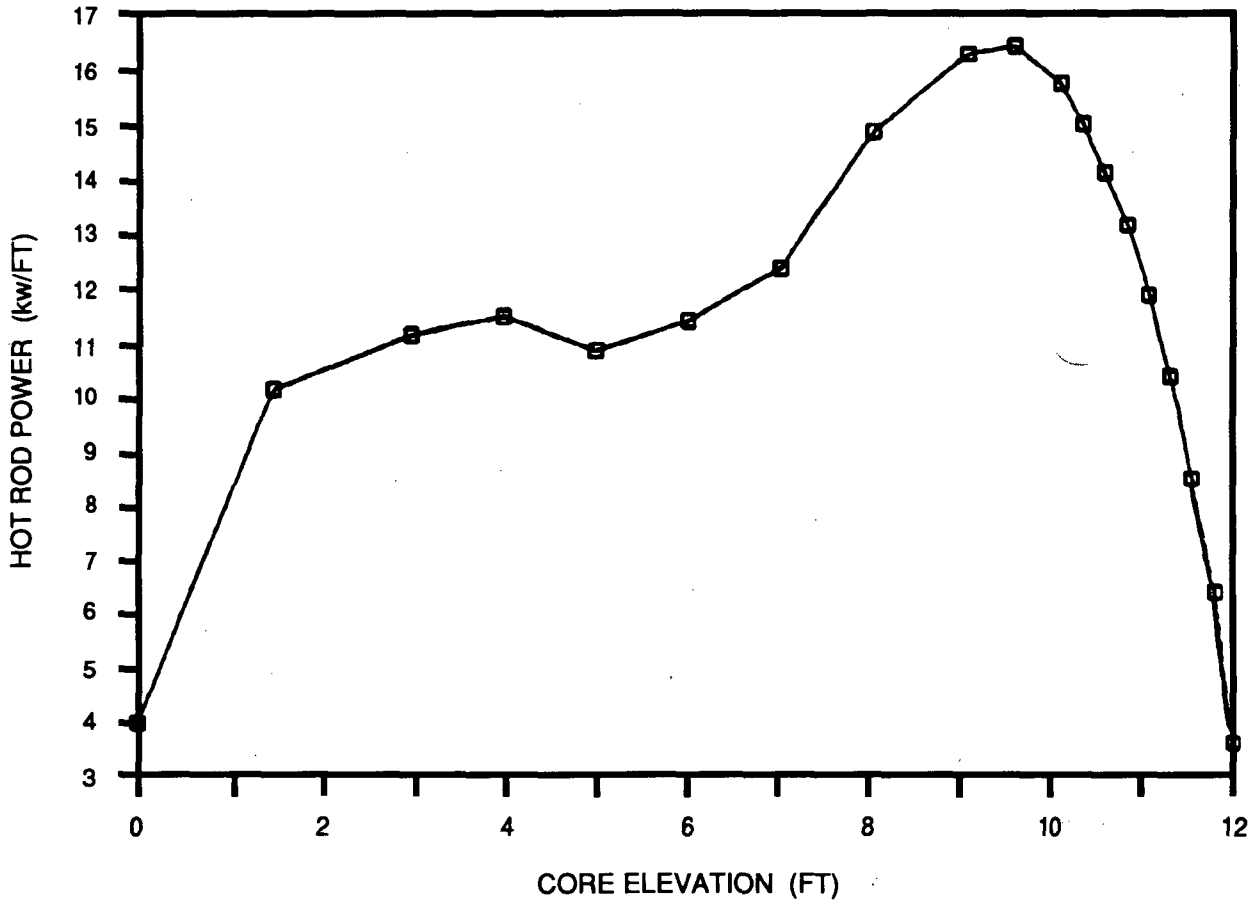


ZION STATION UFSAR

Figure 15.6-2B

CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

JULY 1993

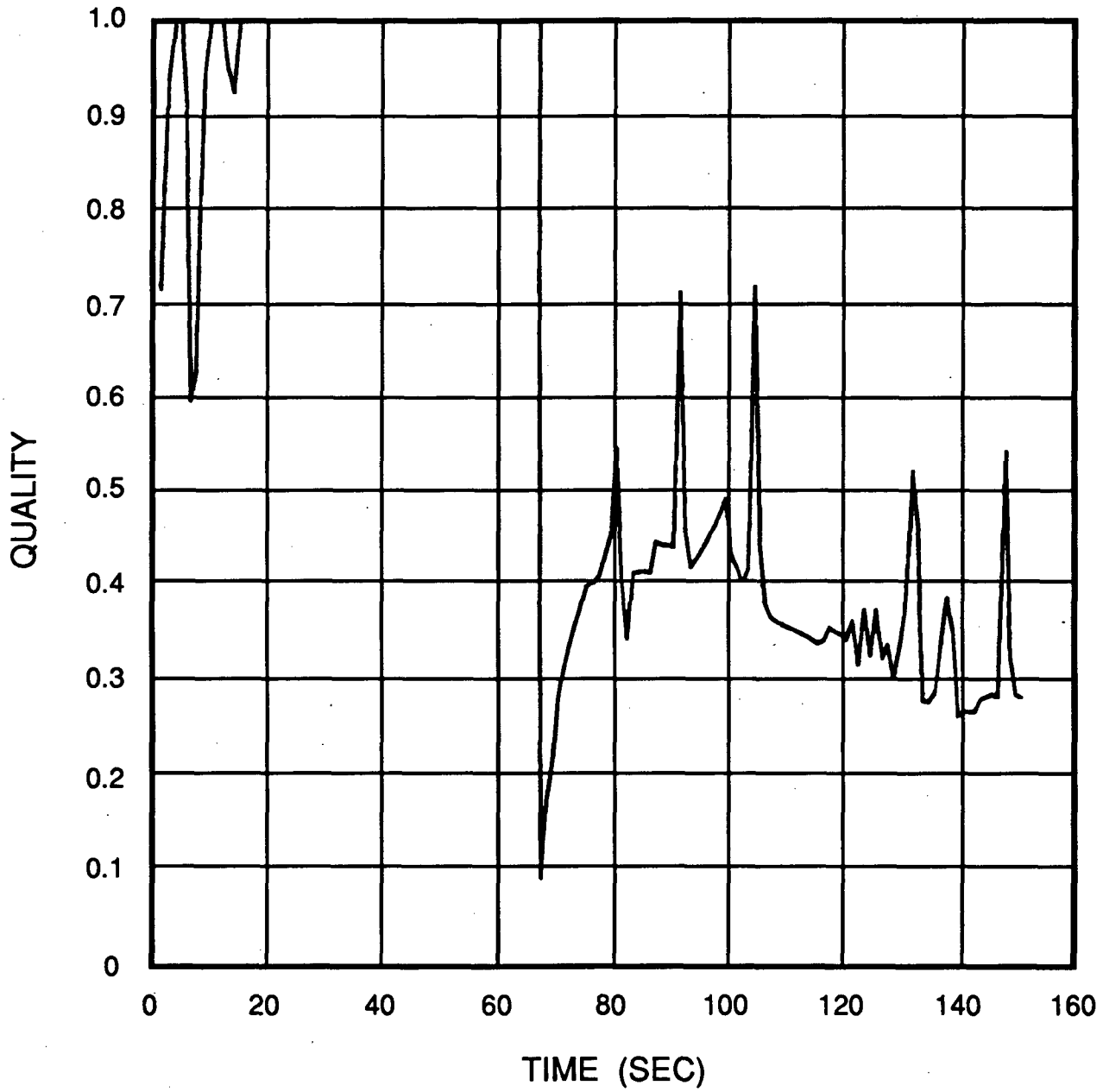


ZION STATION UFSAR

Figure 15.6-3

SMALL BREAK LOCA POWER SHAPE
HOT ROD POWER (Kw/Ft)

JULY 1993

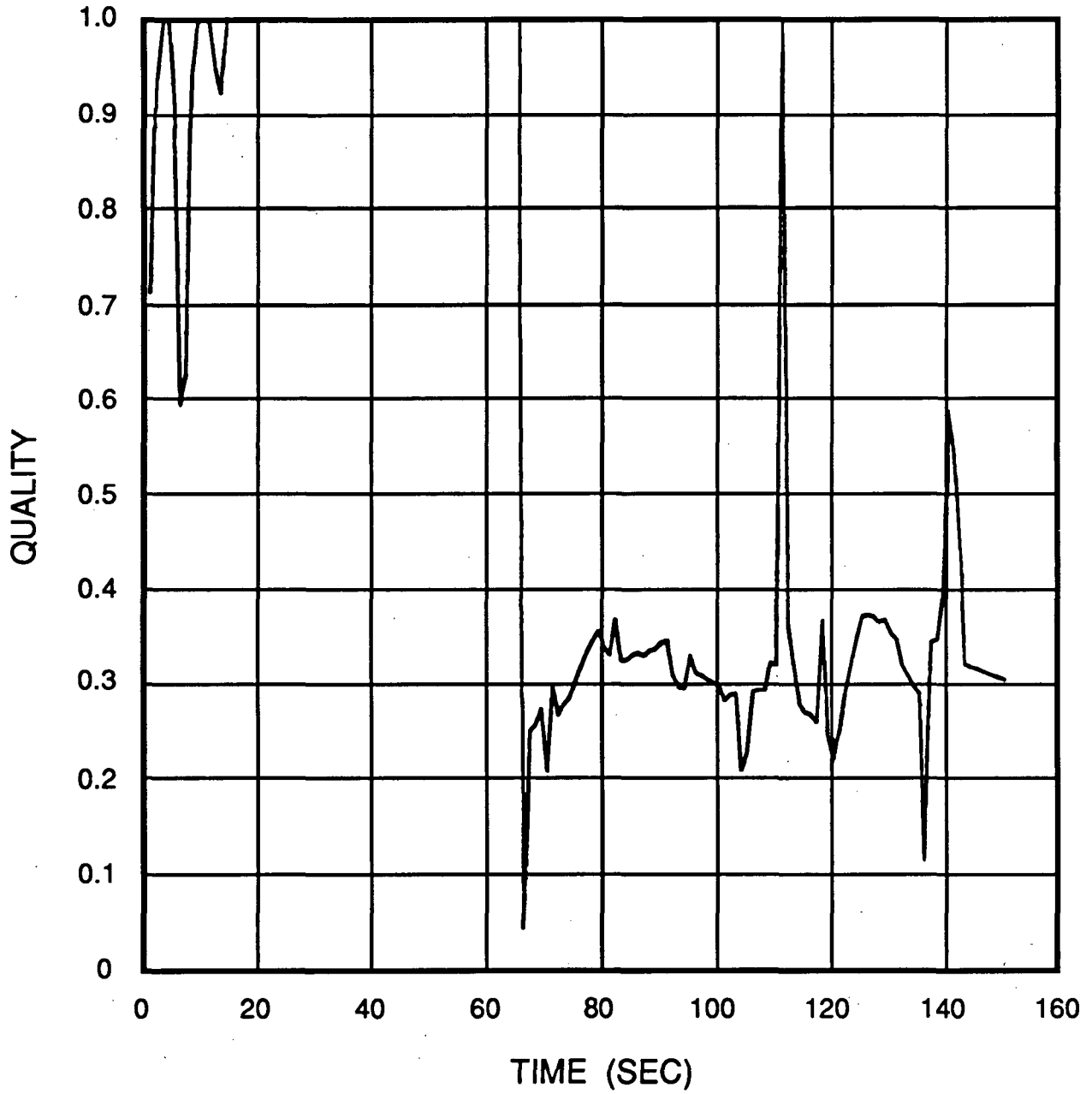


ZION STATION UFSAR

Figure 15.6-4

FLUID QUALITY
DECLG (CD=0.4)

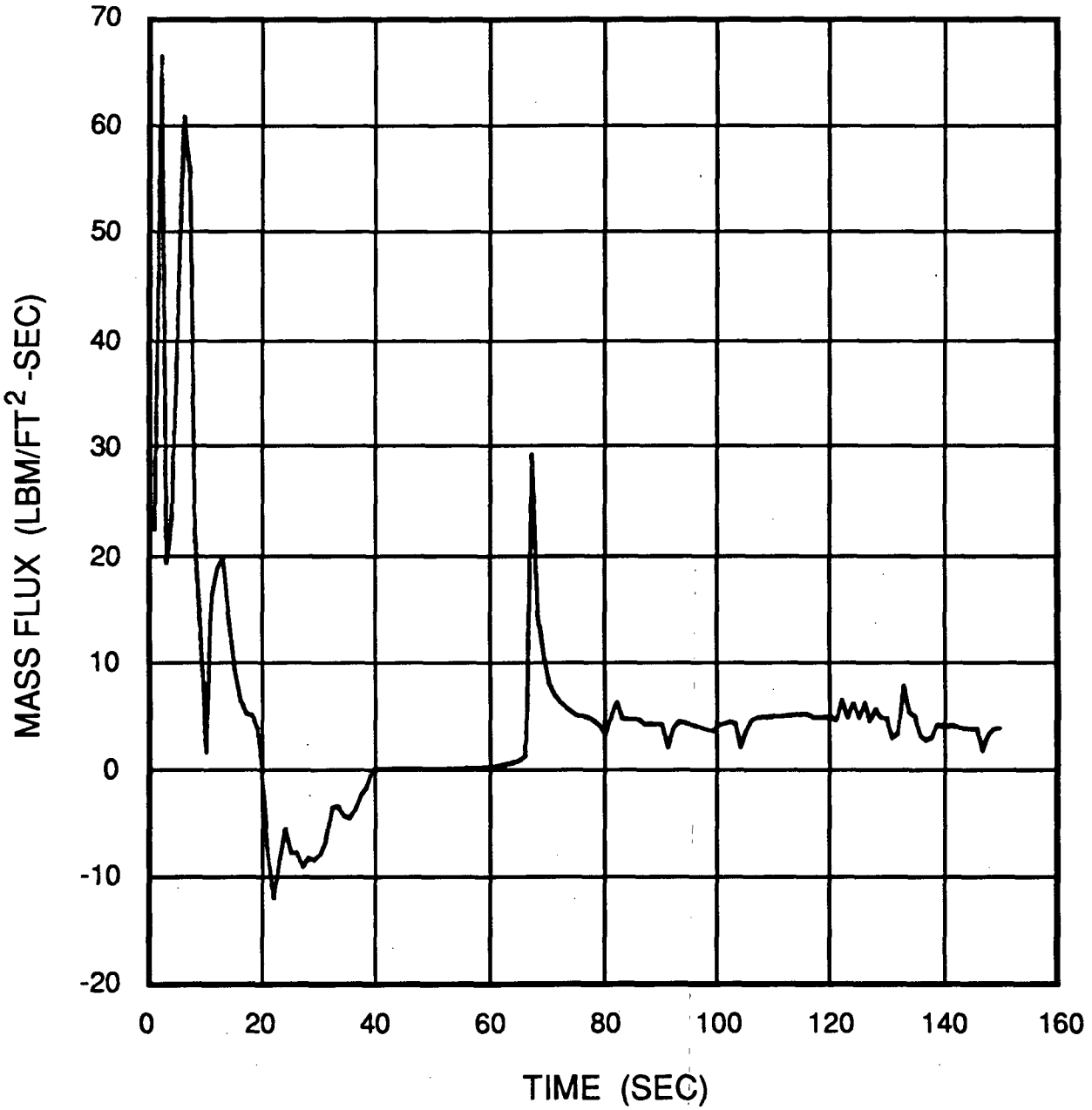
JULY 1993



ZION STATION UFSAR

Figure 15.6-4A
FLUID QUALITY
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

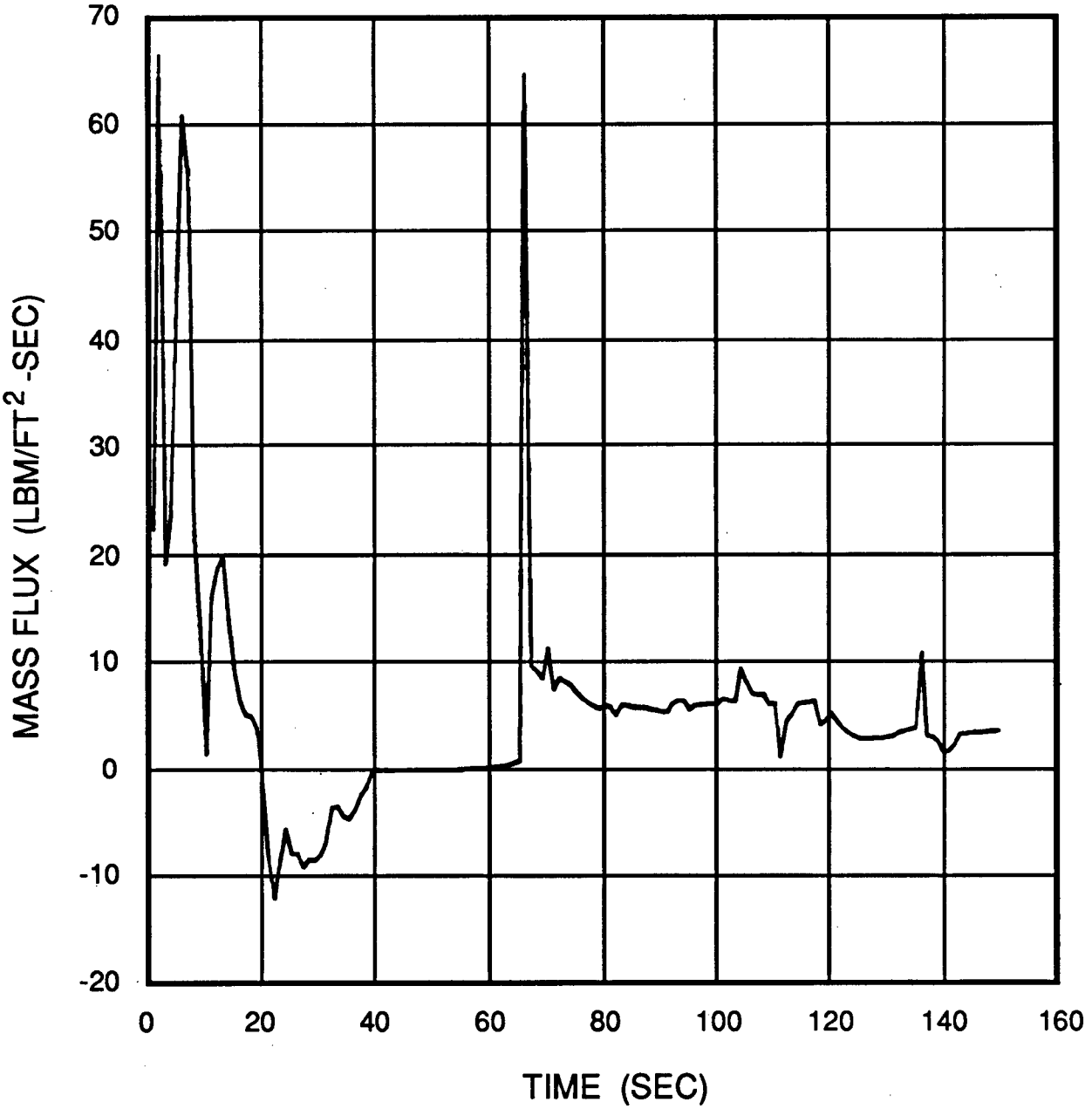


ZION STATION UFSAR

Figure 15.6-5

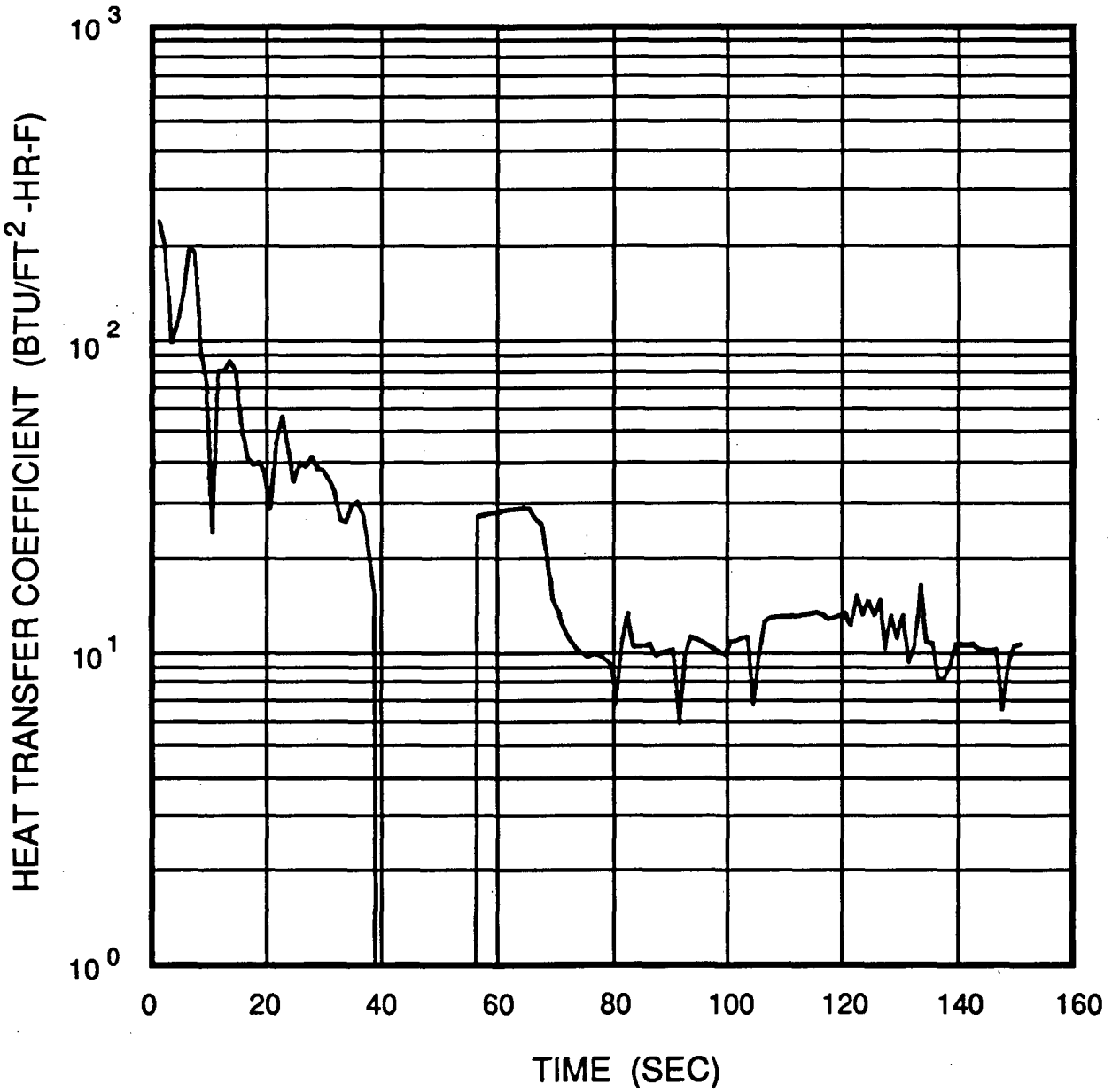
MASS VELOCITY
DECLG (CD=0.4)

JULY 1993



ZION STATION UFSAR

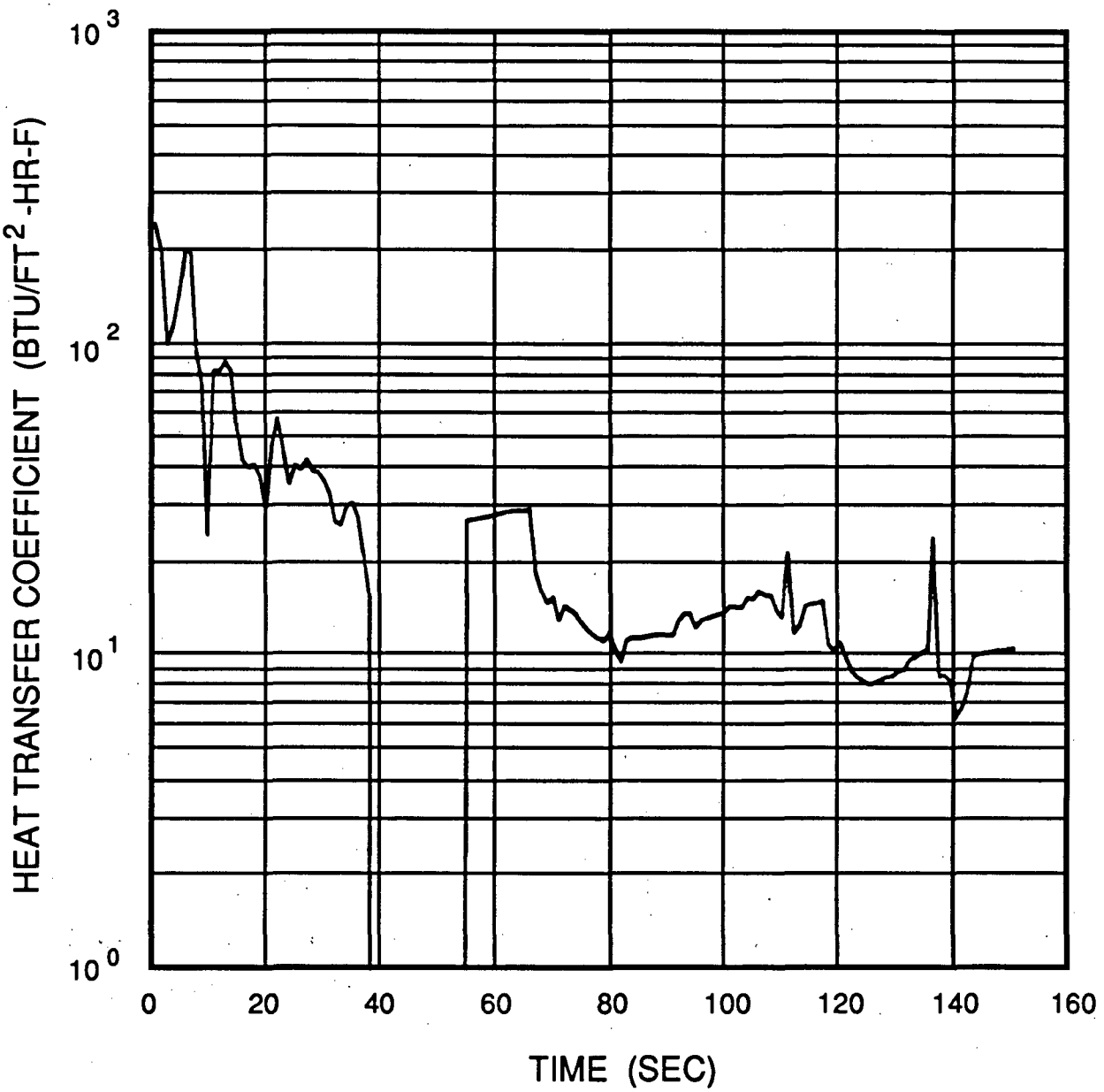
Figure 15.6-5A
MASS VELOCITY
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS
JULY 1993



ZION STATION UFSAR

Figure 15.6-6
CLAD HEAT TRANSFER COEFFICIENT
DECLG (CD=0.4)

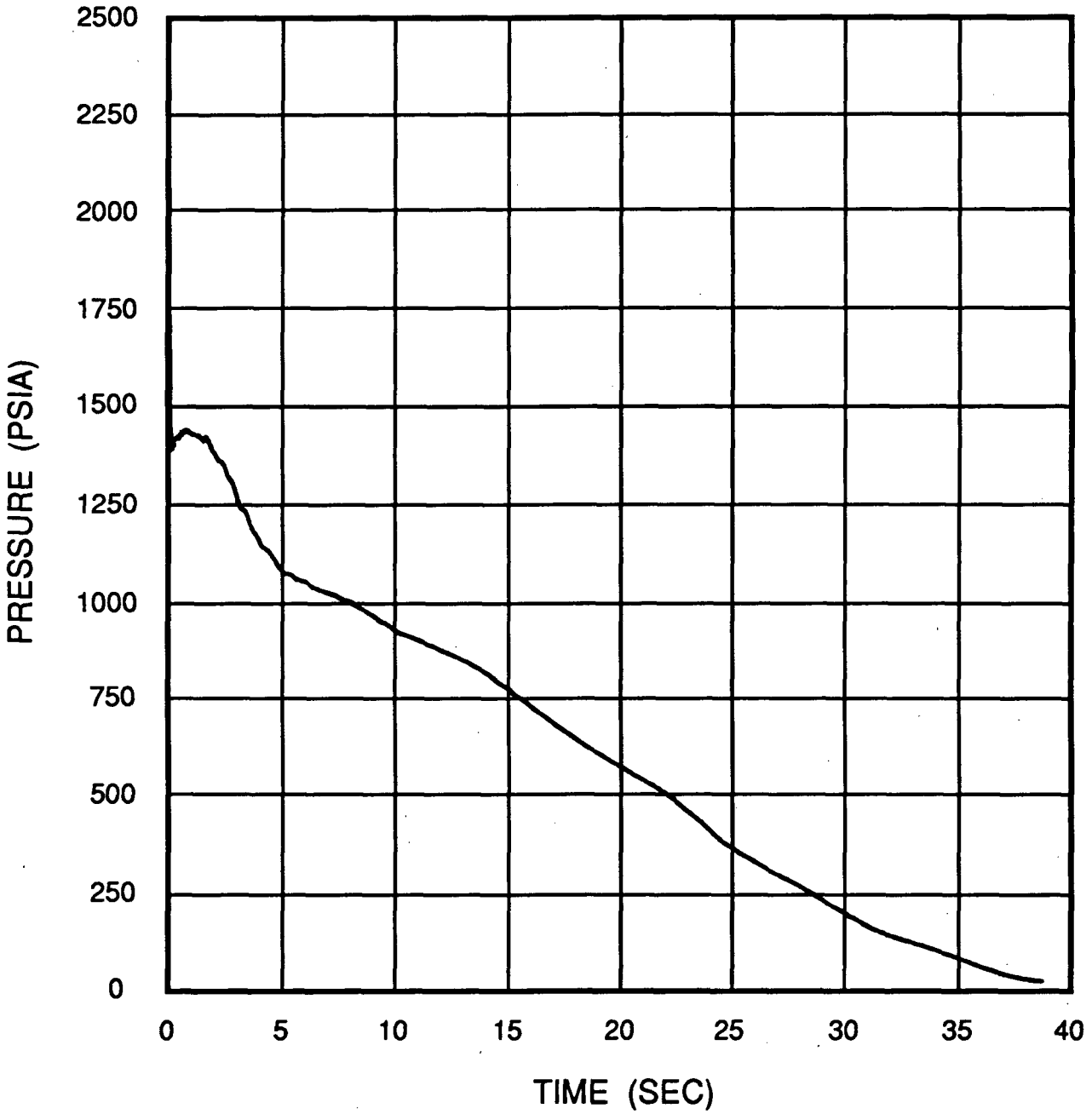
JULY 1993



ZION STATION UFSAR

Figure 15.6-6A
CLAD HEAT TRANSFER COEFFICIENT
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

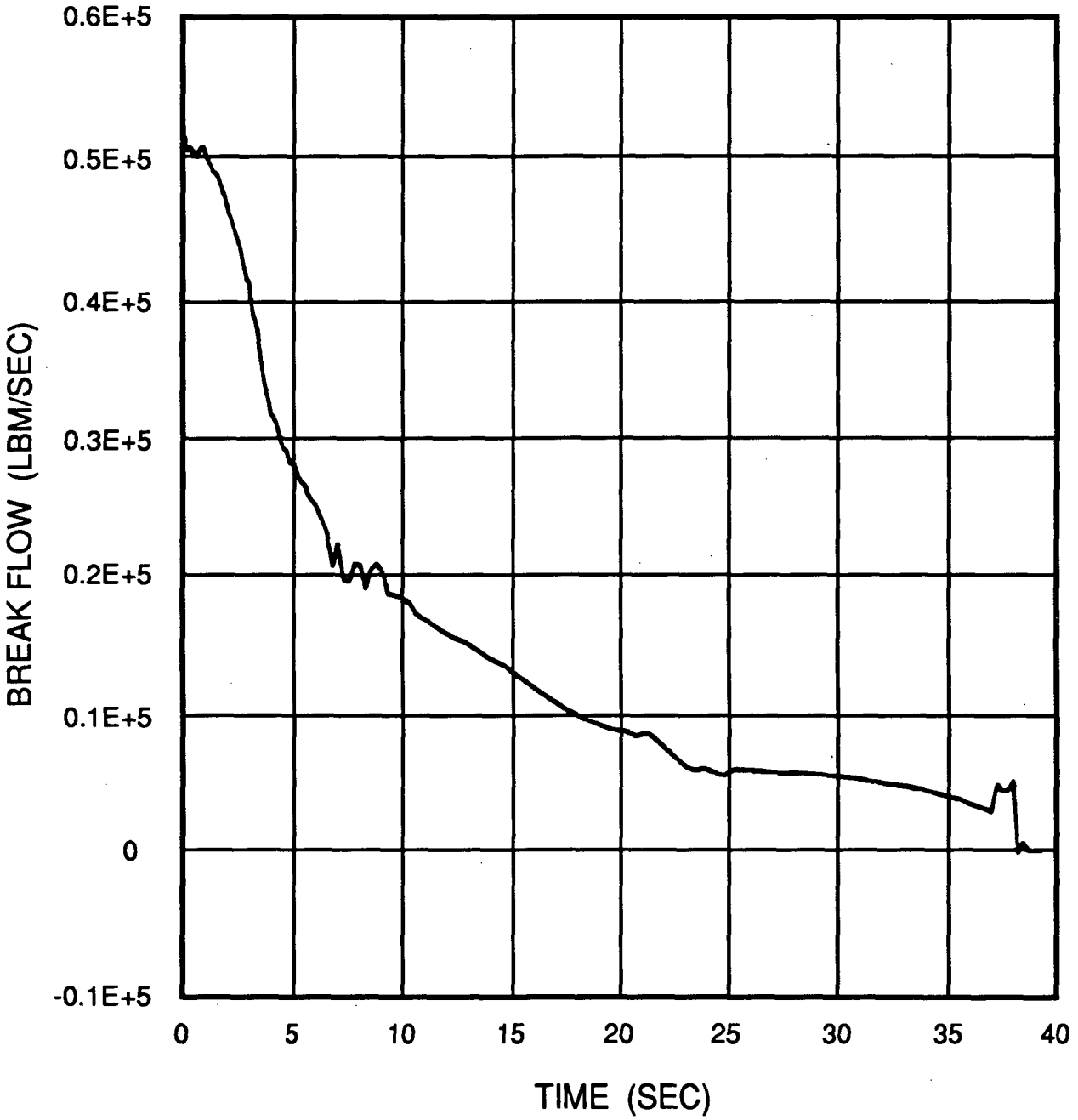


ZION STATION UFSAR

Figure 15.6-7

CORE PRESSURE
DECLG (CD=0.4)

JULY 1993

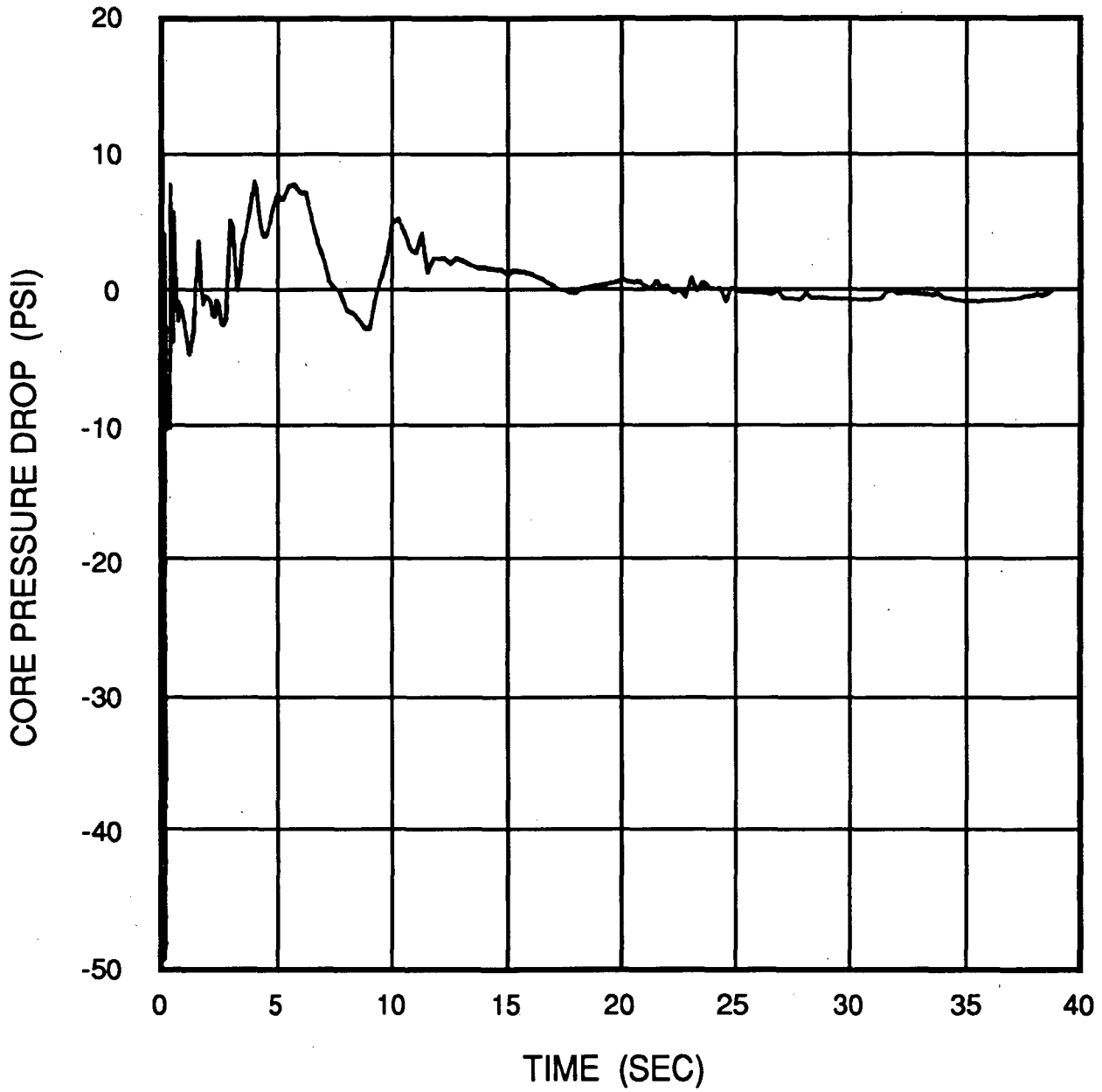


ZION STATION UFSAR

Figure 15.6-8

BREAK FLOW RATE
DECLG (CD=0.4)

JULY 1993

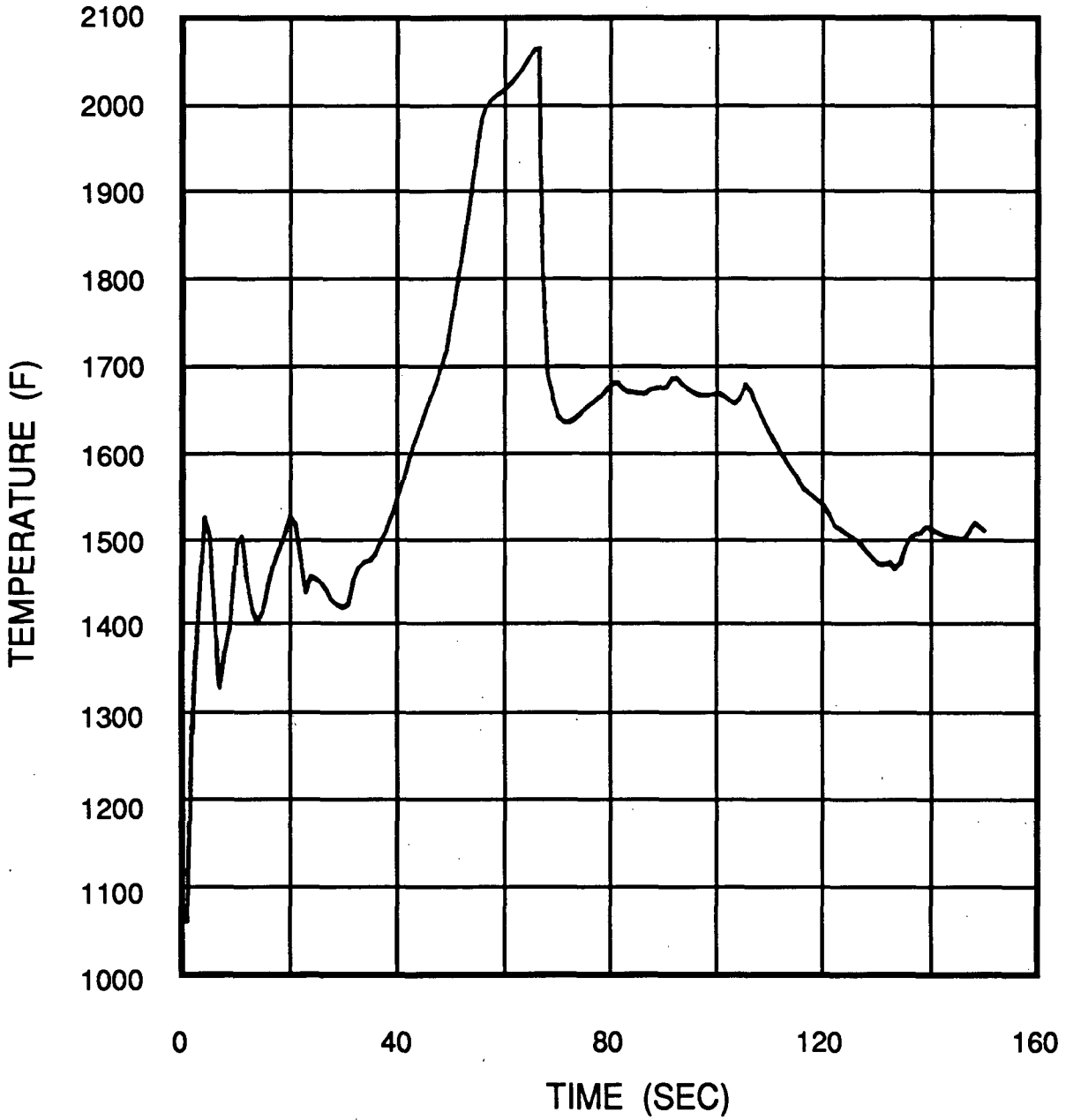


ZION STATION UFSAR

Figure 15.6-9

CORE PRESSURE DROP
DECLG (CD=0.4)

JULY 1993

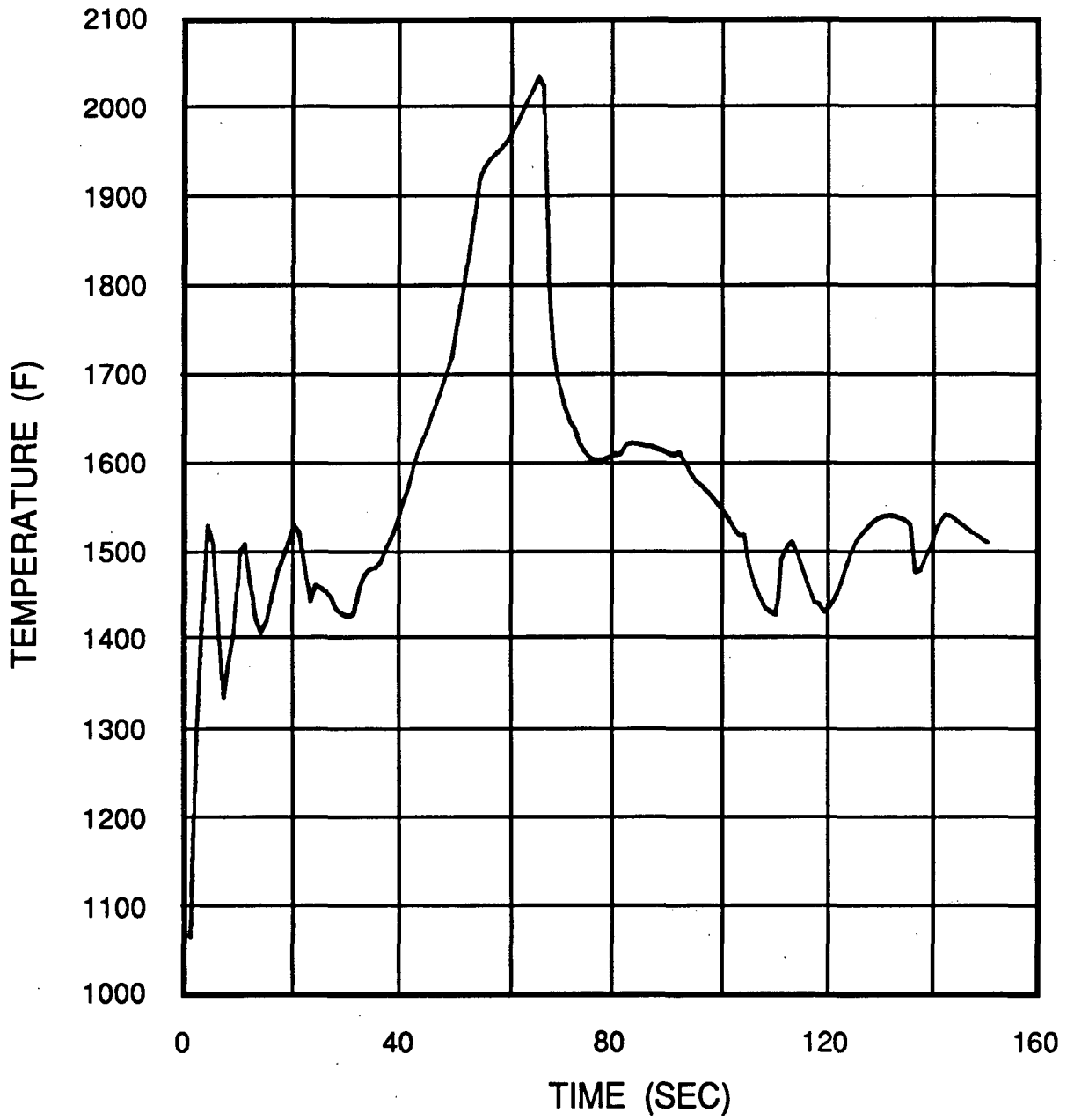


ZION STATION UFSAR

Figure 15.6-10

PEAK CLADDING TEMPERATURE
DECLG (CD=0.4)

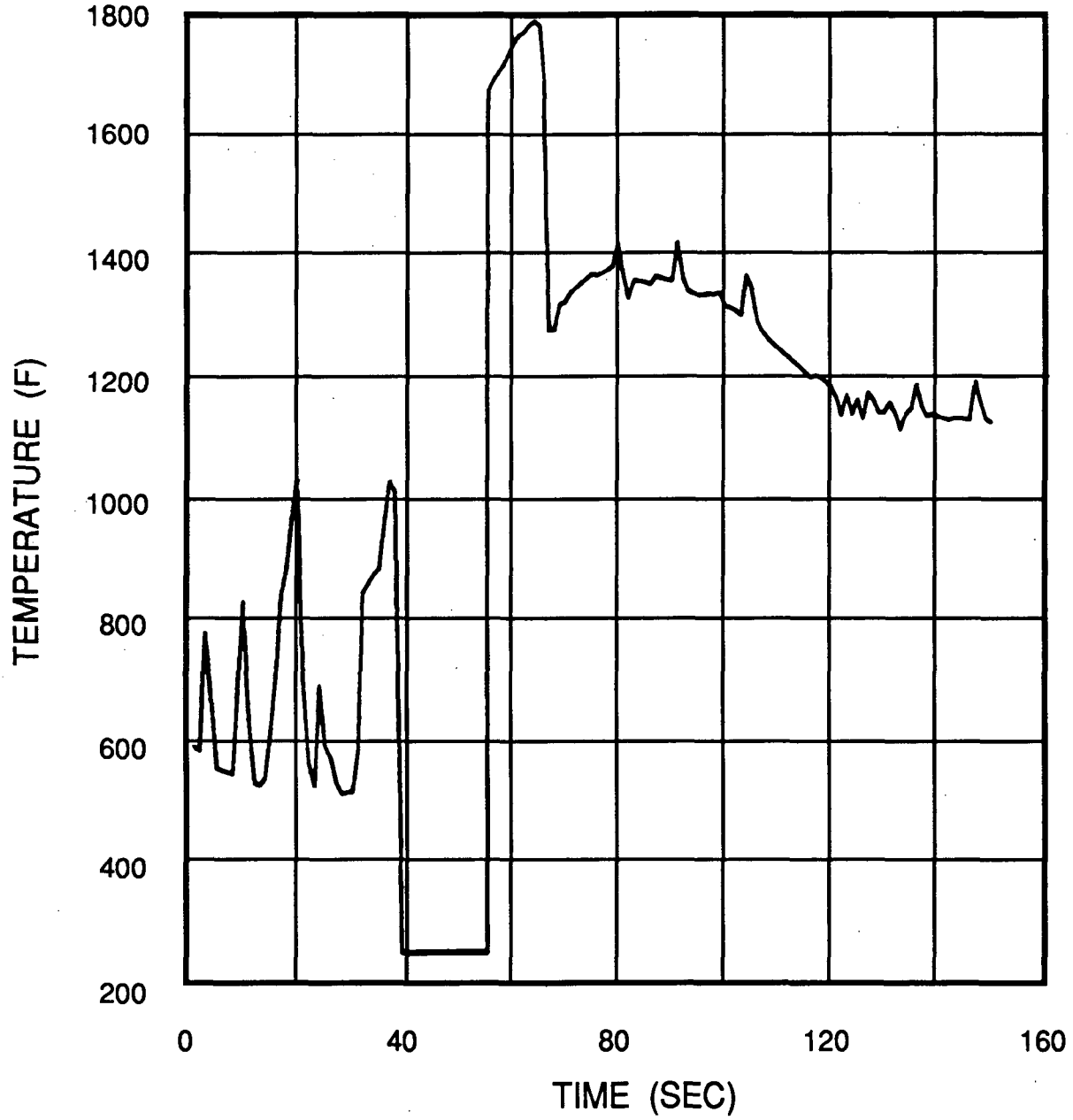
JULY 1993



ZION STATION UFSAR

Figure 15.6-10A
PEAK CLADDING TEMPERATURE
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

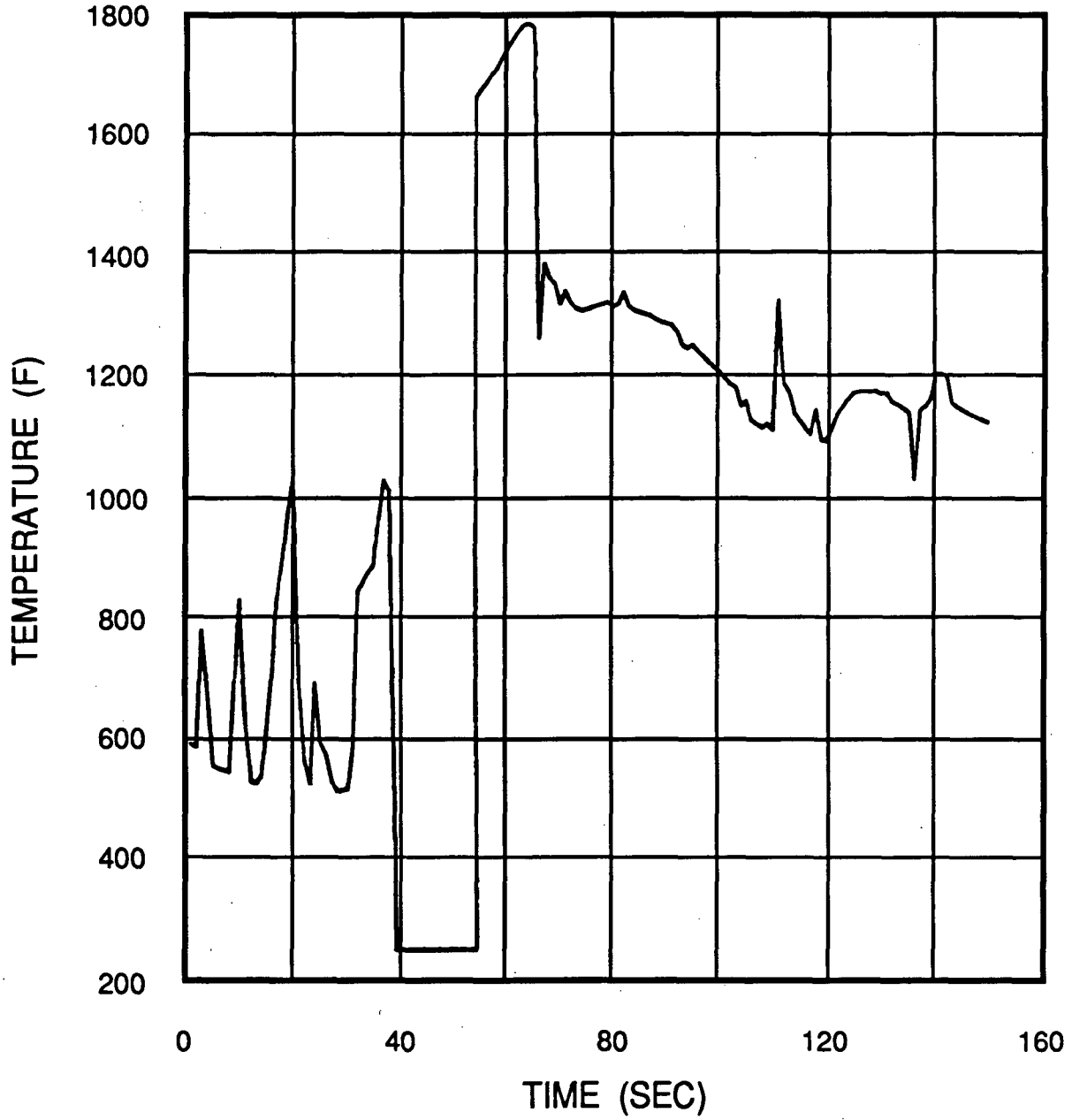


ZION STATION UFSAR

Figure 15.6-11

FLUID TEMPERATURE
DECLG (CD=0.4)

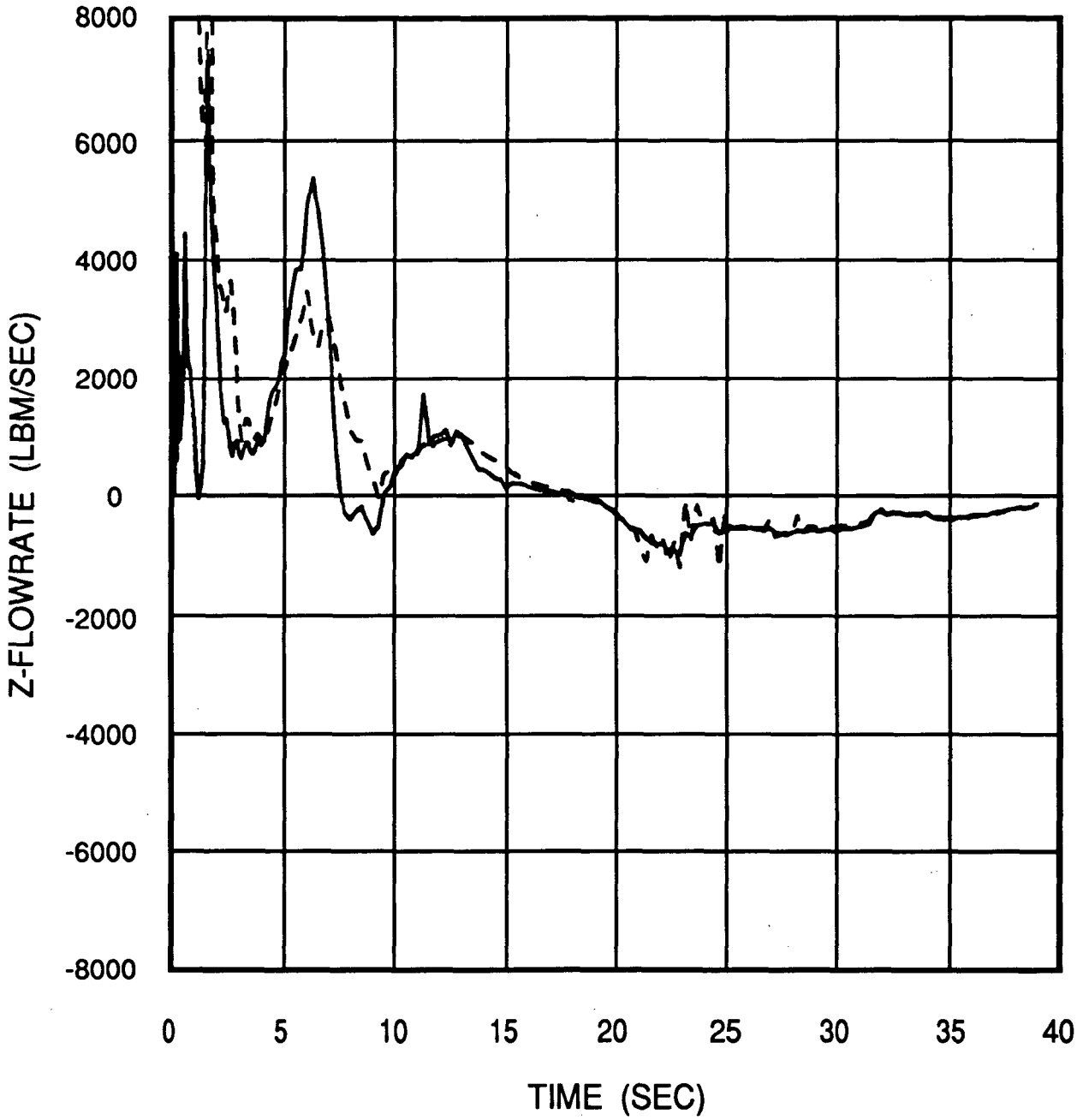
JULY 1993



ZION STATION UFSAR

Figure 15.6-11A
FLUID TEMPERATURE
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

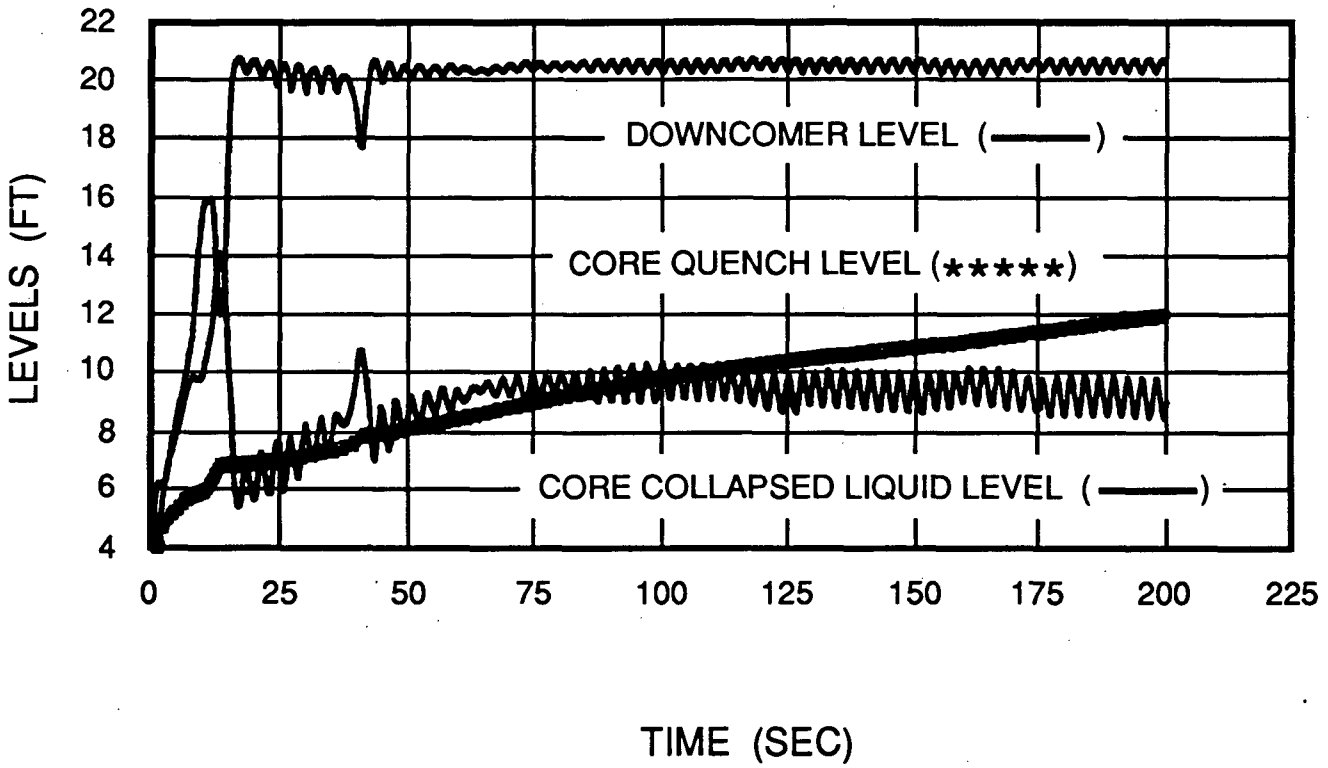


ZION STATION UFSAR

Figure 15.6-12

CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.4)

JULY 1993



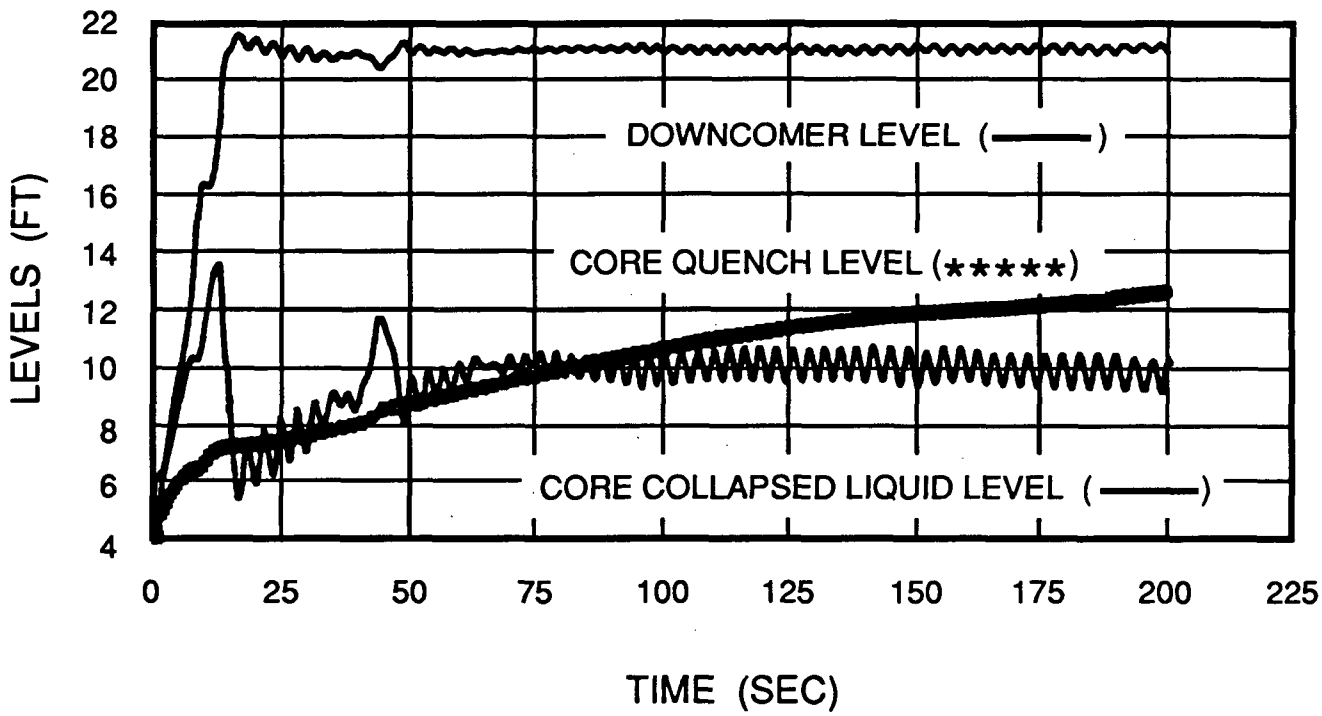
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 55.430 SECONDS FOR CD = 0.4

ZION STATION UFSAR

Figure 15.6-13
REFLOOD TRANSIENT
CORE AND DOWNCOMER LEVELS
DECLG (CD=0.4)

JULY 1993



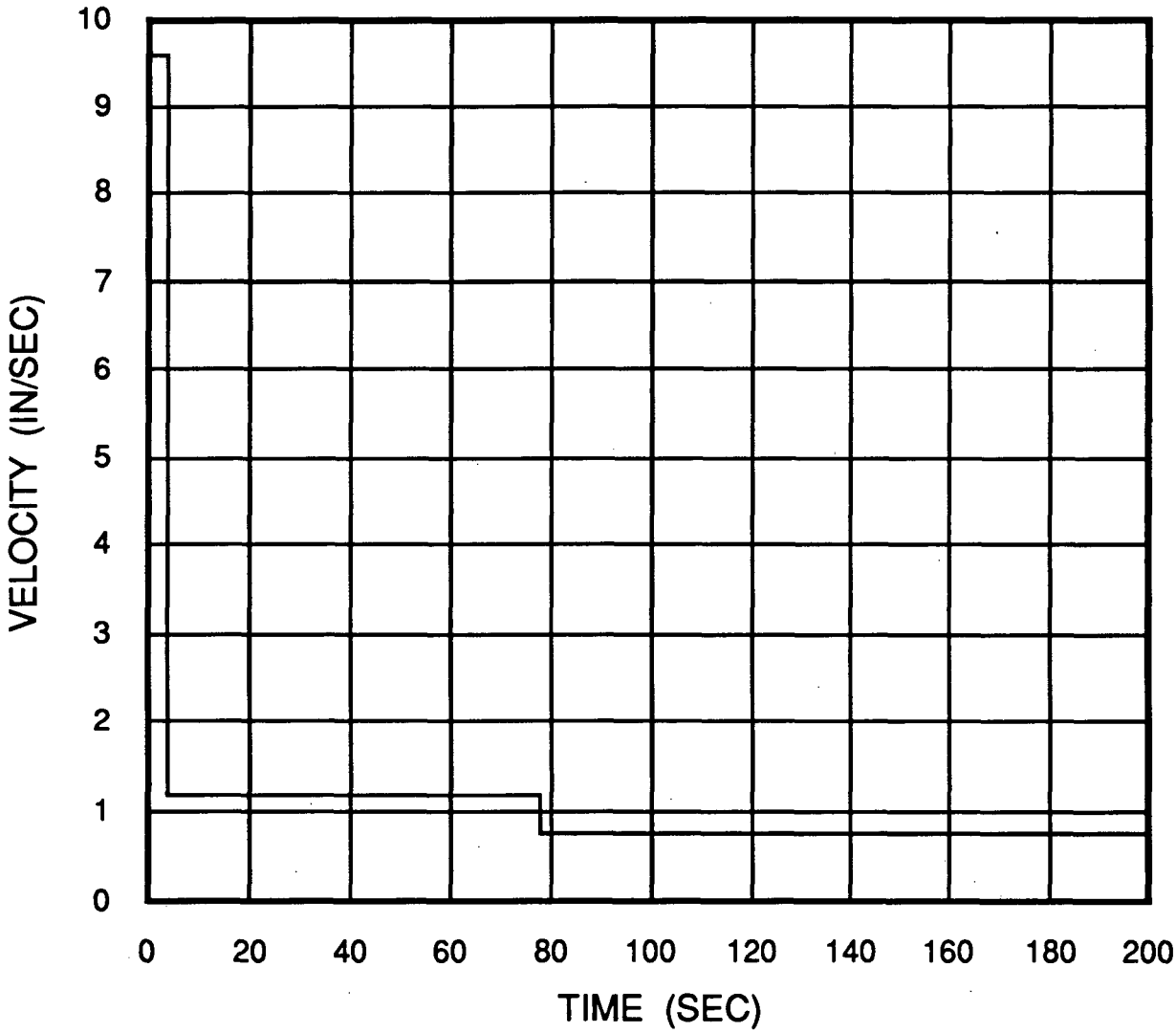
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 54.251 SECONDS FOR CD = 0.4, MAXSI

ZION STATION UFSAR

Figure 15.6-13A
REFLOOD TRANSIENT
CORE AND DOWNCOMER LEVELS
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993



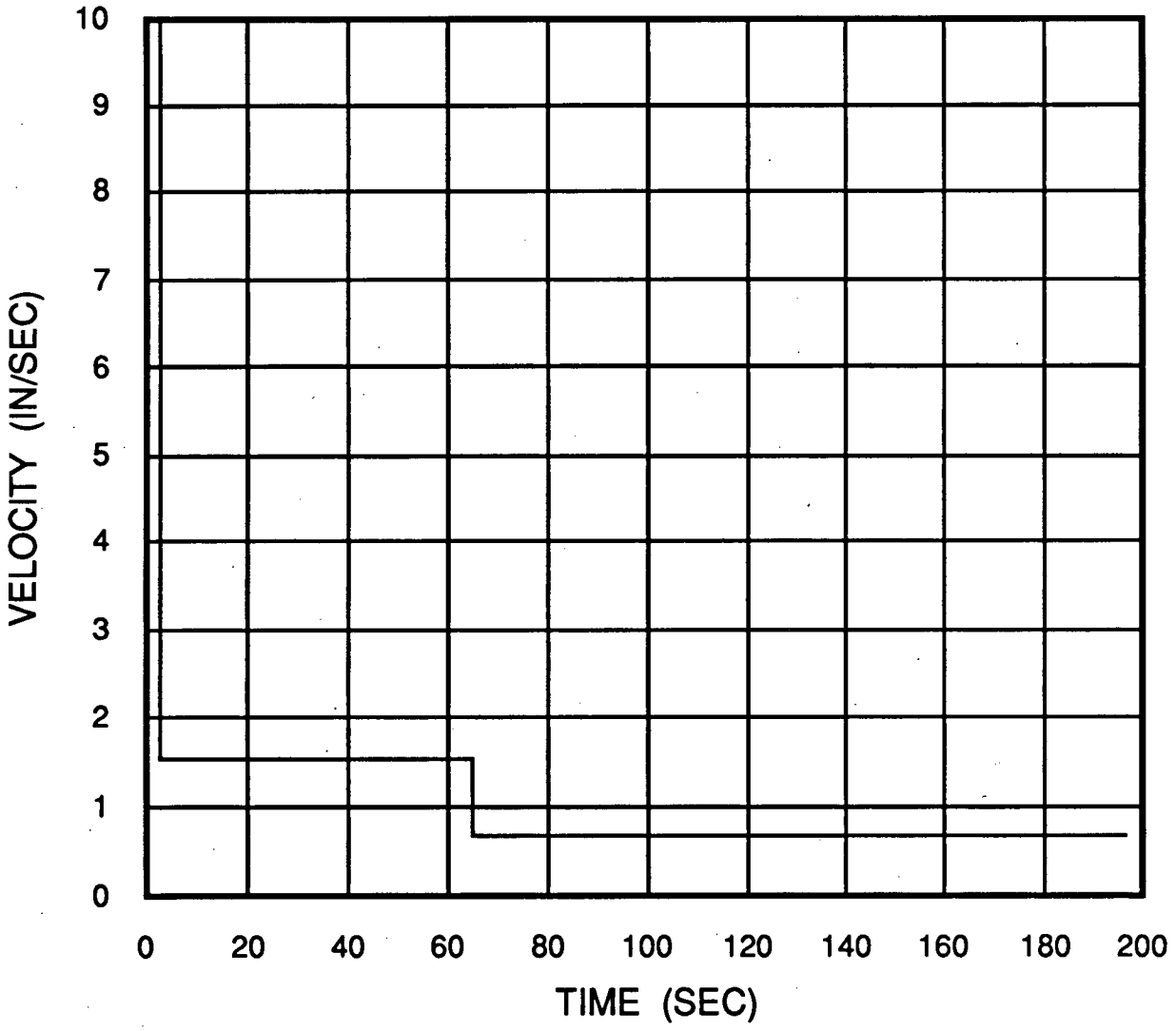
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 55.430 SECONDS FOR CD = 0.4

ZION STATION UFSAR

Figure 15.6-14
REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD=0.4)

JULY 1993

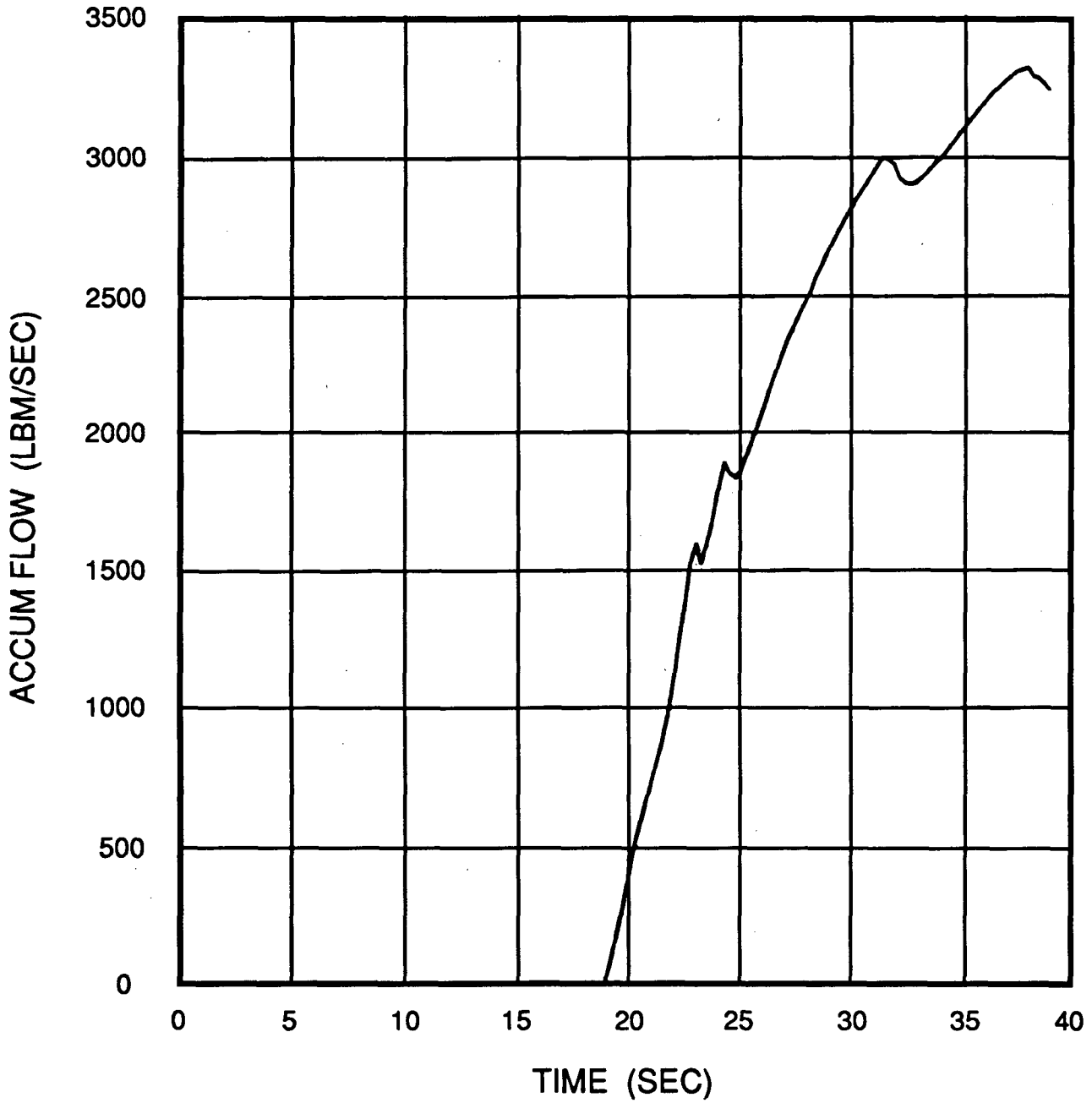


TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 54.251 SECONDS FOR CD = 0.4, MAXSI

ZION STATION UFSAR

Figure 15.6-14A
REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS
JULY 1993

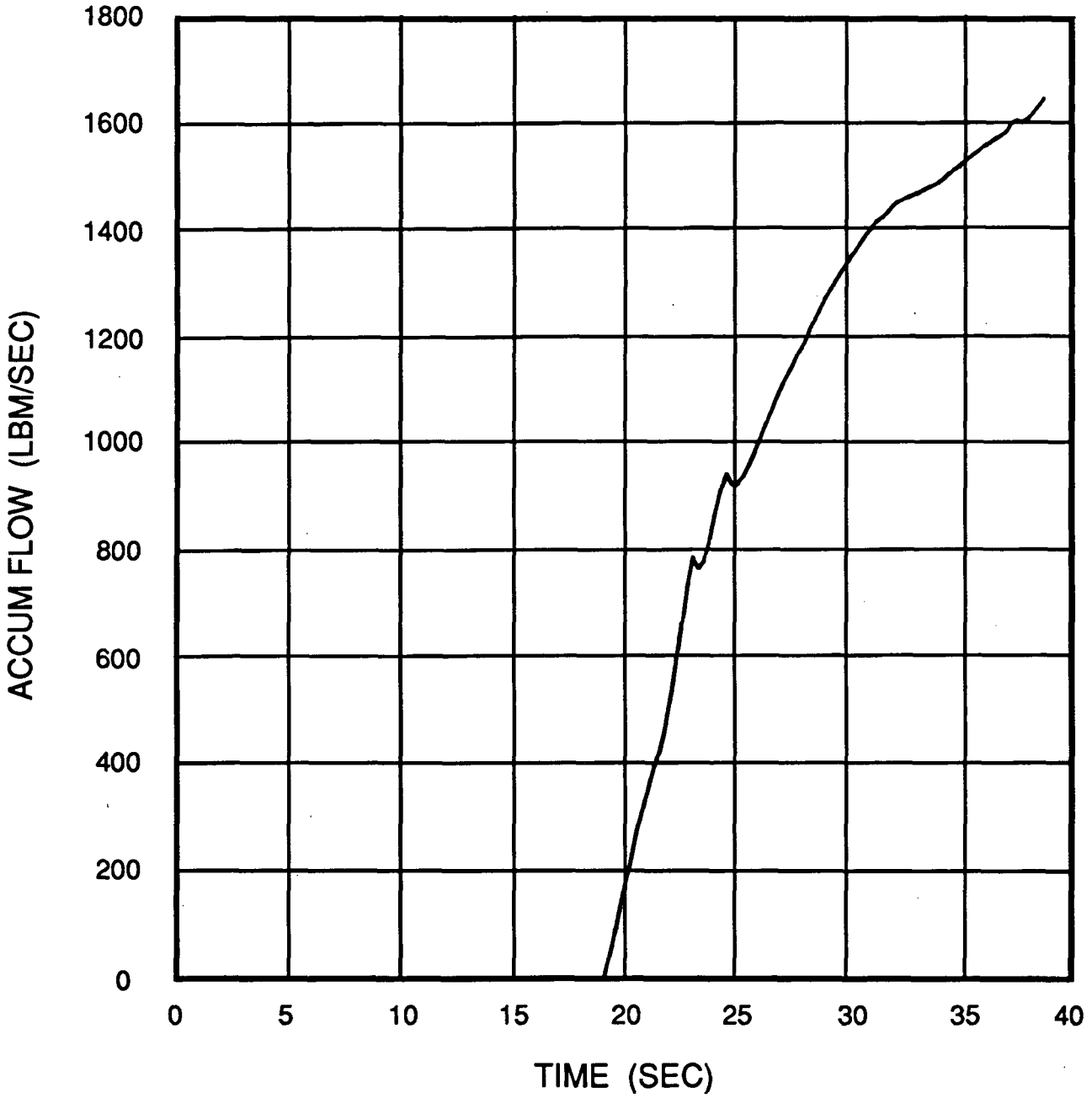


FLOW FROM TWO INTACT LOOP ACCUMULATORS

ZION STATION UFSAR

Figure 15.6-15
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.4)

JULY 1993

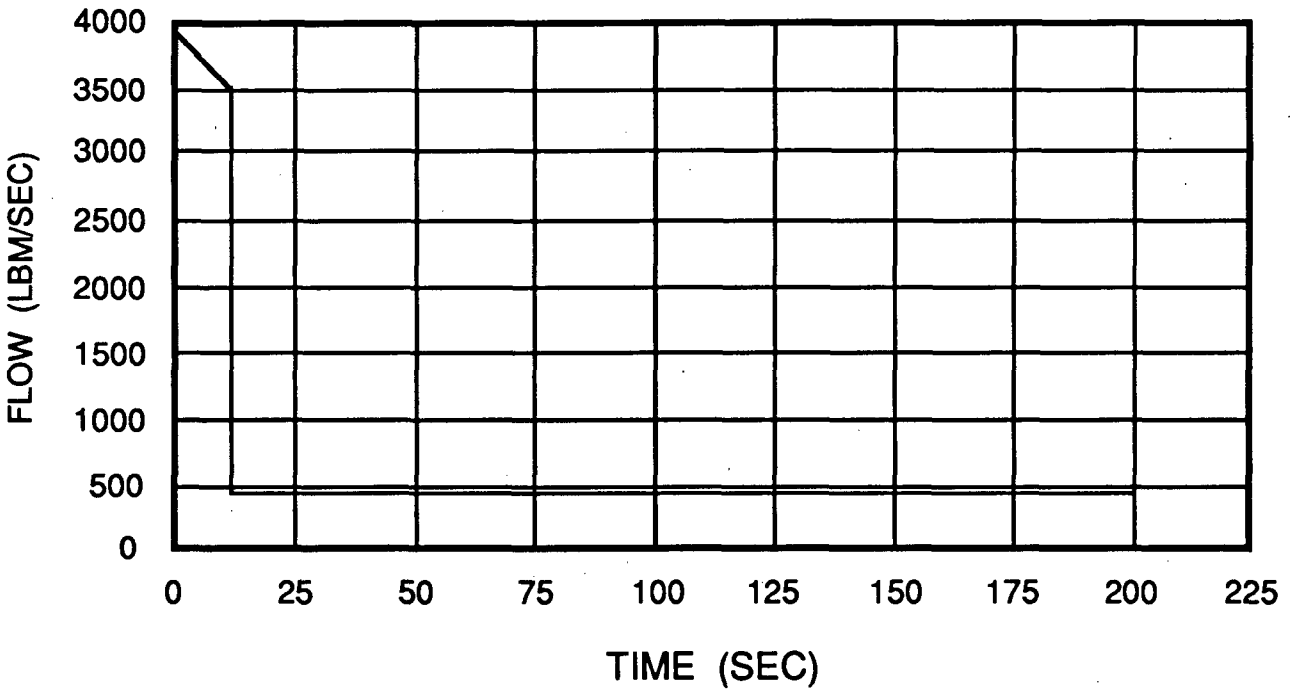


FLOW FROM ONE INTACT LOOP ACCUMULATOR

ZION STATION UFSAR

Figure 15.6-16
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.4)

JULY 1993



TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

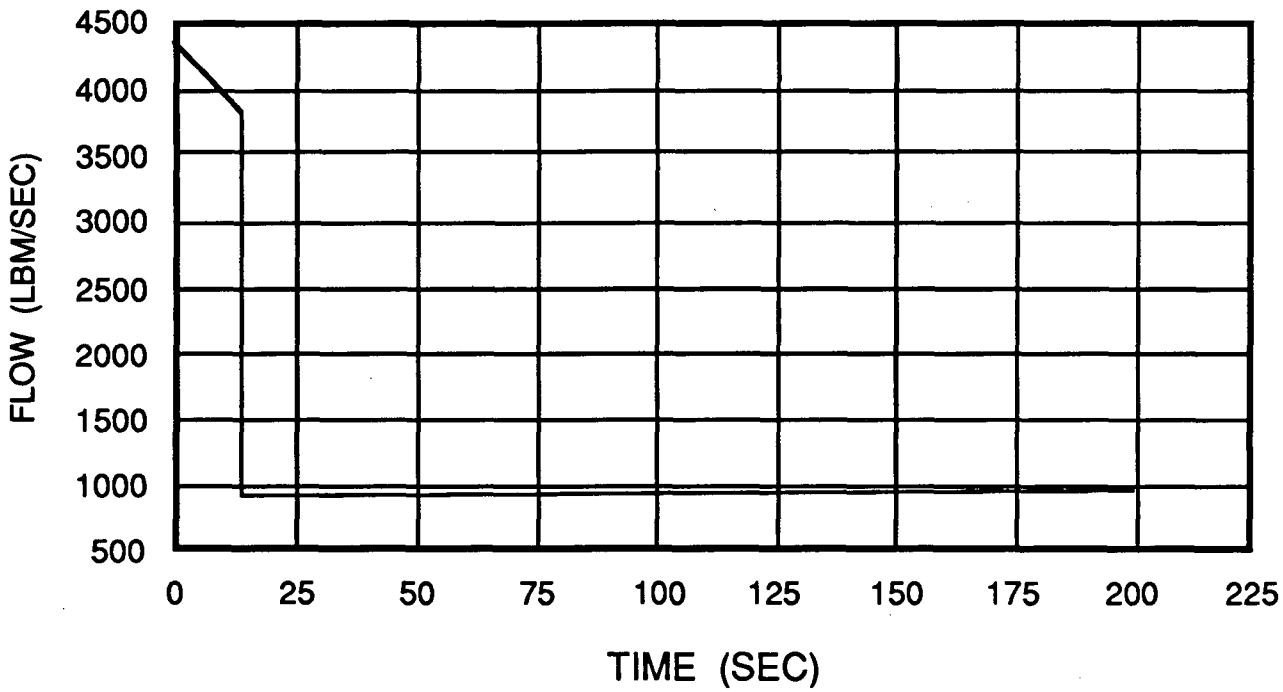
BOC = 55.430 SECONDS FOR CD = 0.4

ZION STATION UFSAR

Figure 15.6-17

PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.4)

JULY 1993



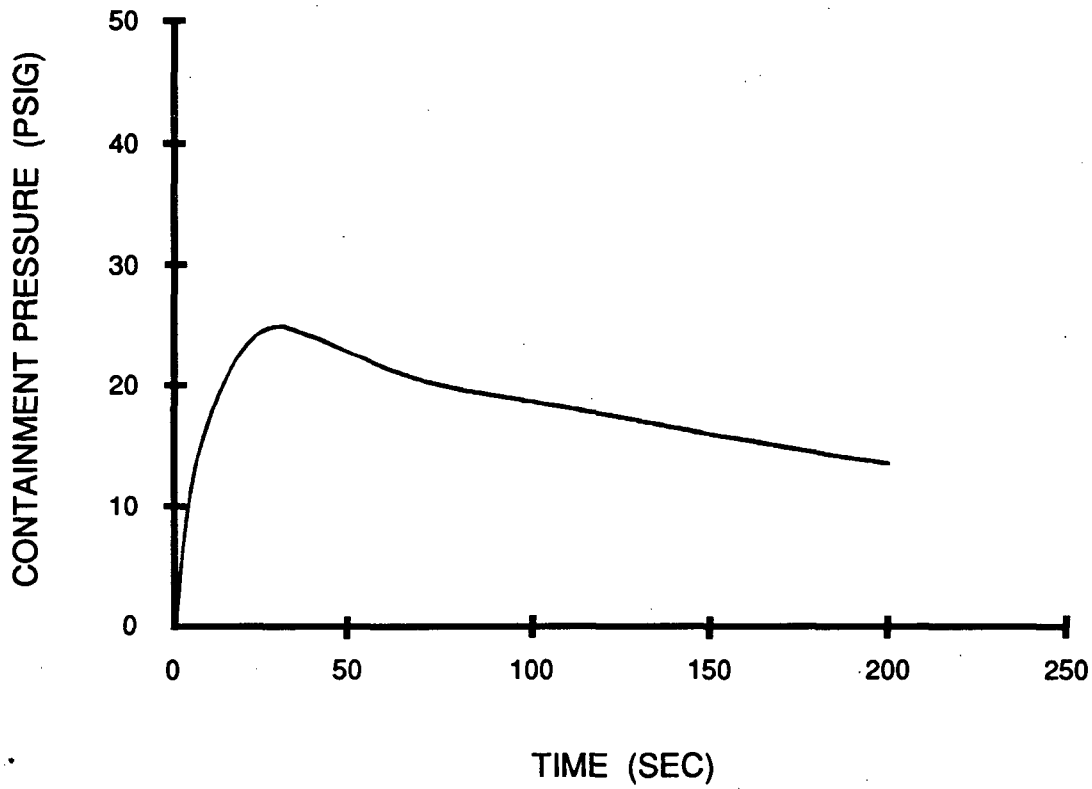
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 54.251 SECONDS FOR CD = 0.4, MAXSI

ZION STATION UFSAR

Figure 15.6-17A
PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

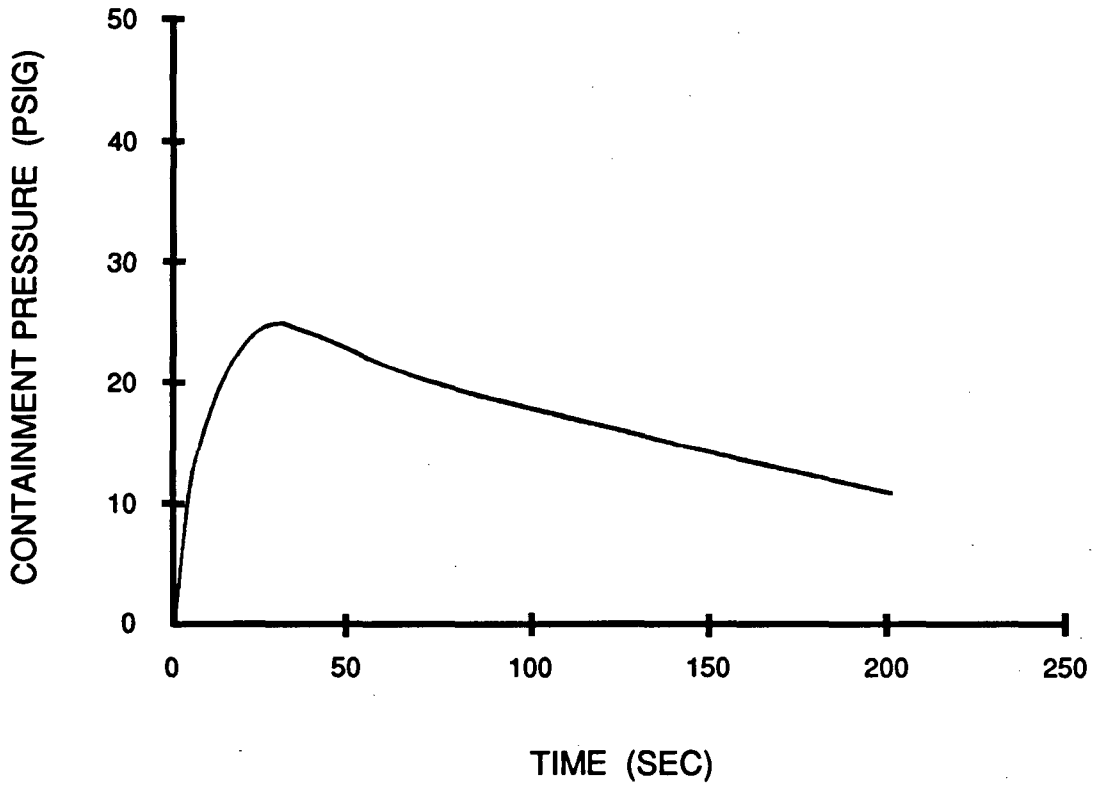


ZION STATION UFSAR

Figure 15.6-18

CONTAINMENT PRESSURE
DECLG (CD=0.4)

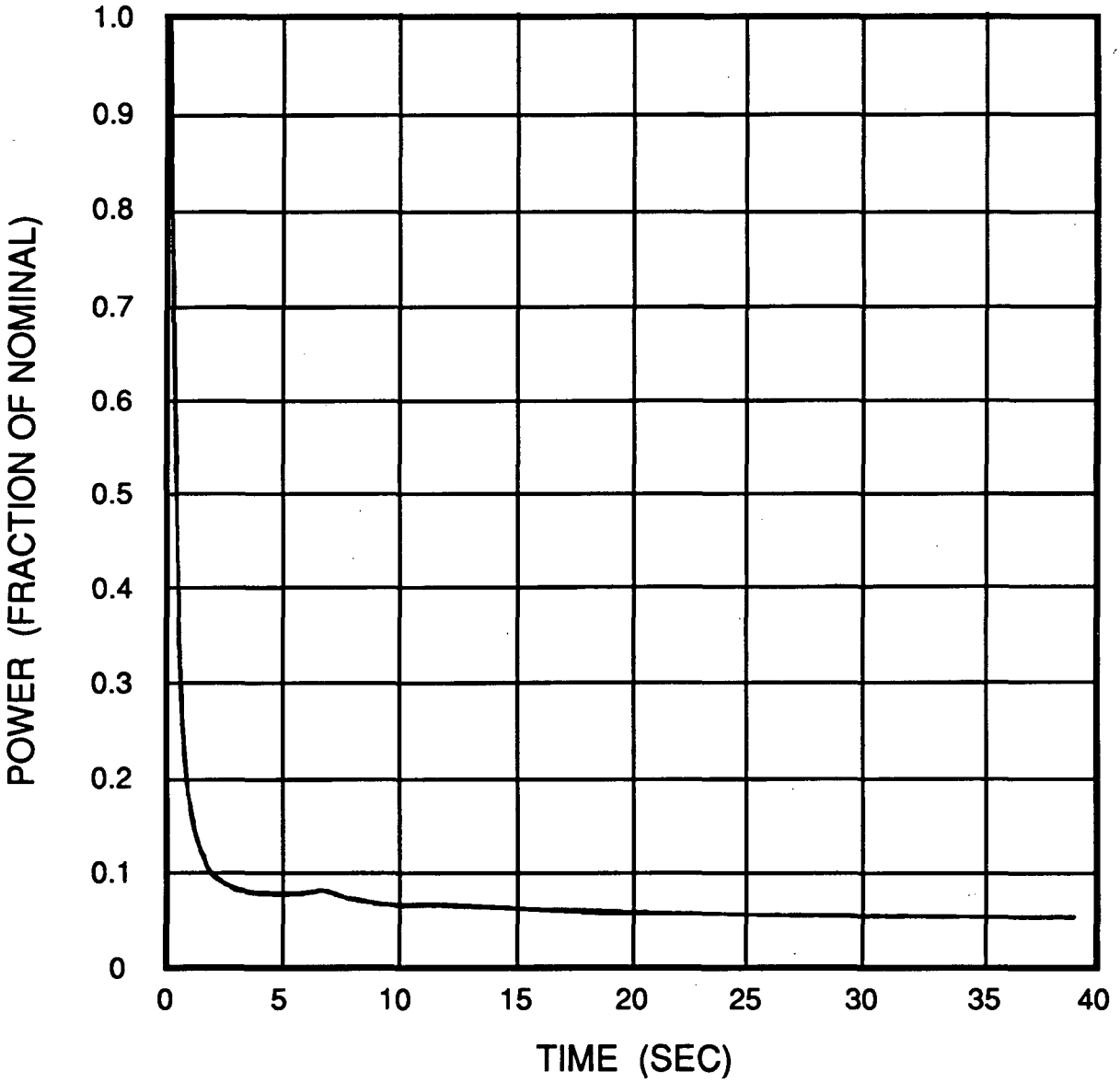
JULY 1993



ZION STATION UFSAR

Figure 15.6-18A
CONTAINMENT PRESSURE
DECLG (CD=0.4)
MAXIMUM ECCS FLOWS

JULY 1993

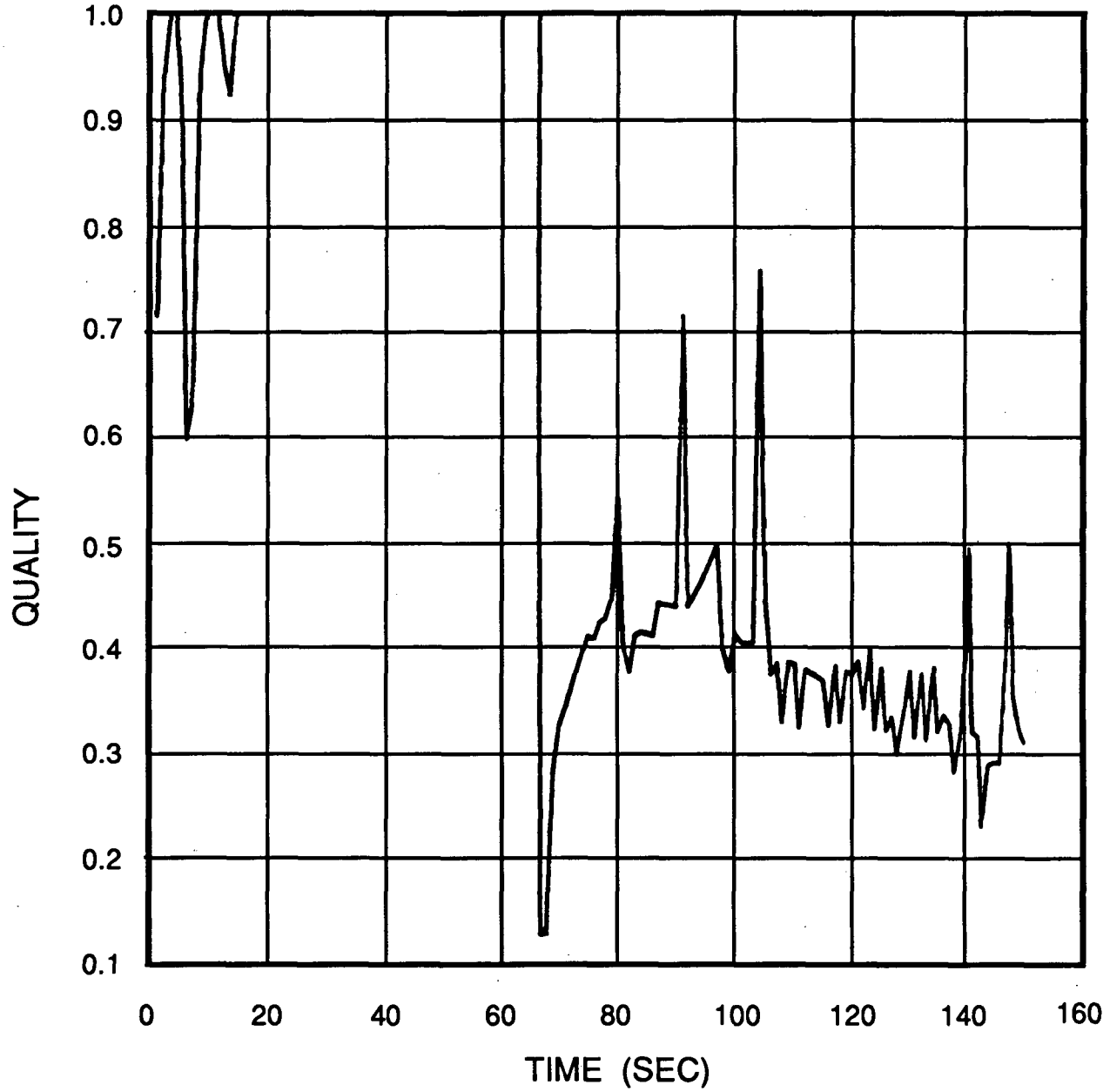


ZION STATION UFSAR

Figure 15.6-19

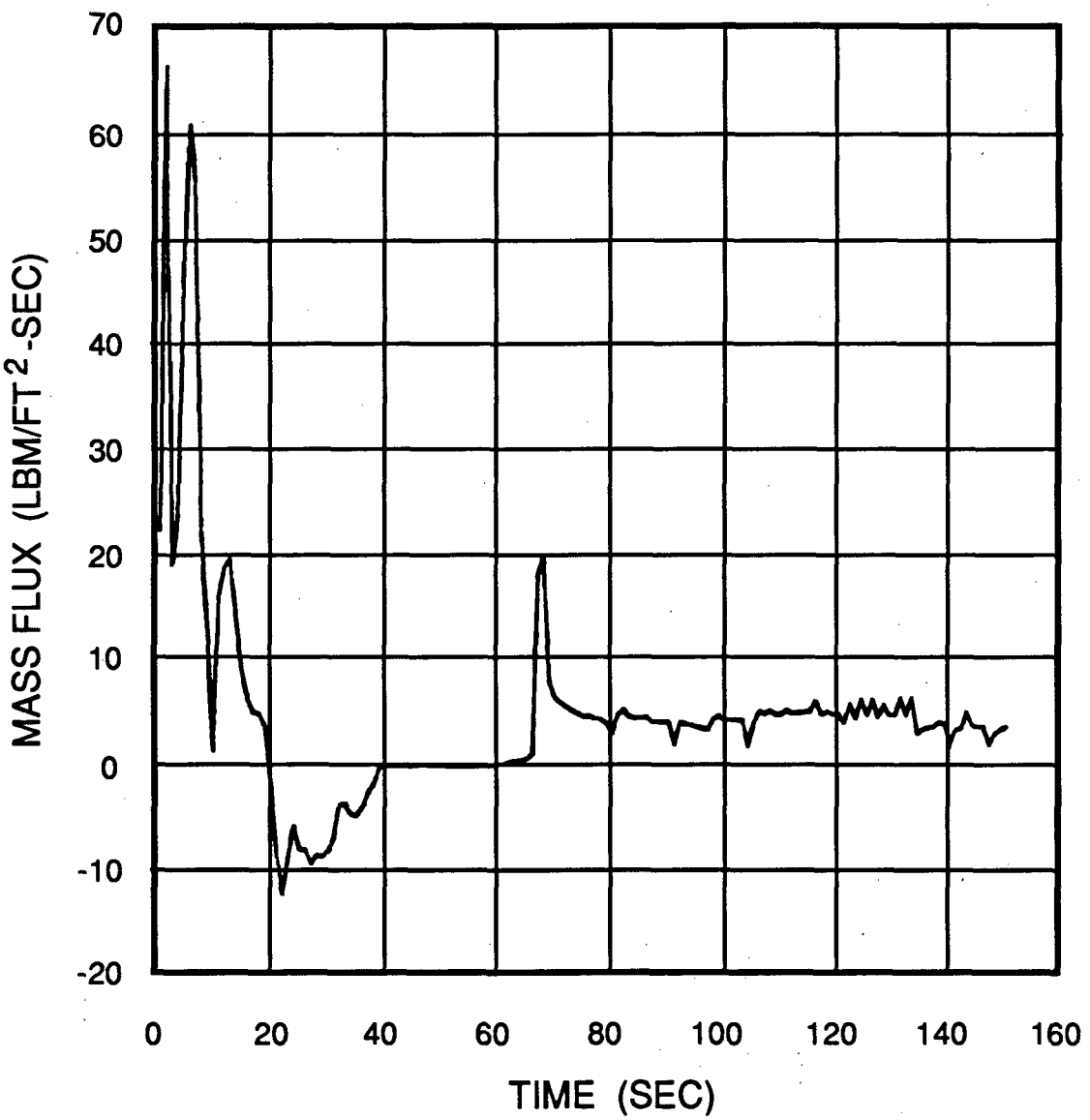
CORE POWER TRANSIENT
DECLG (CD=0.4)

JULY 1993



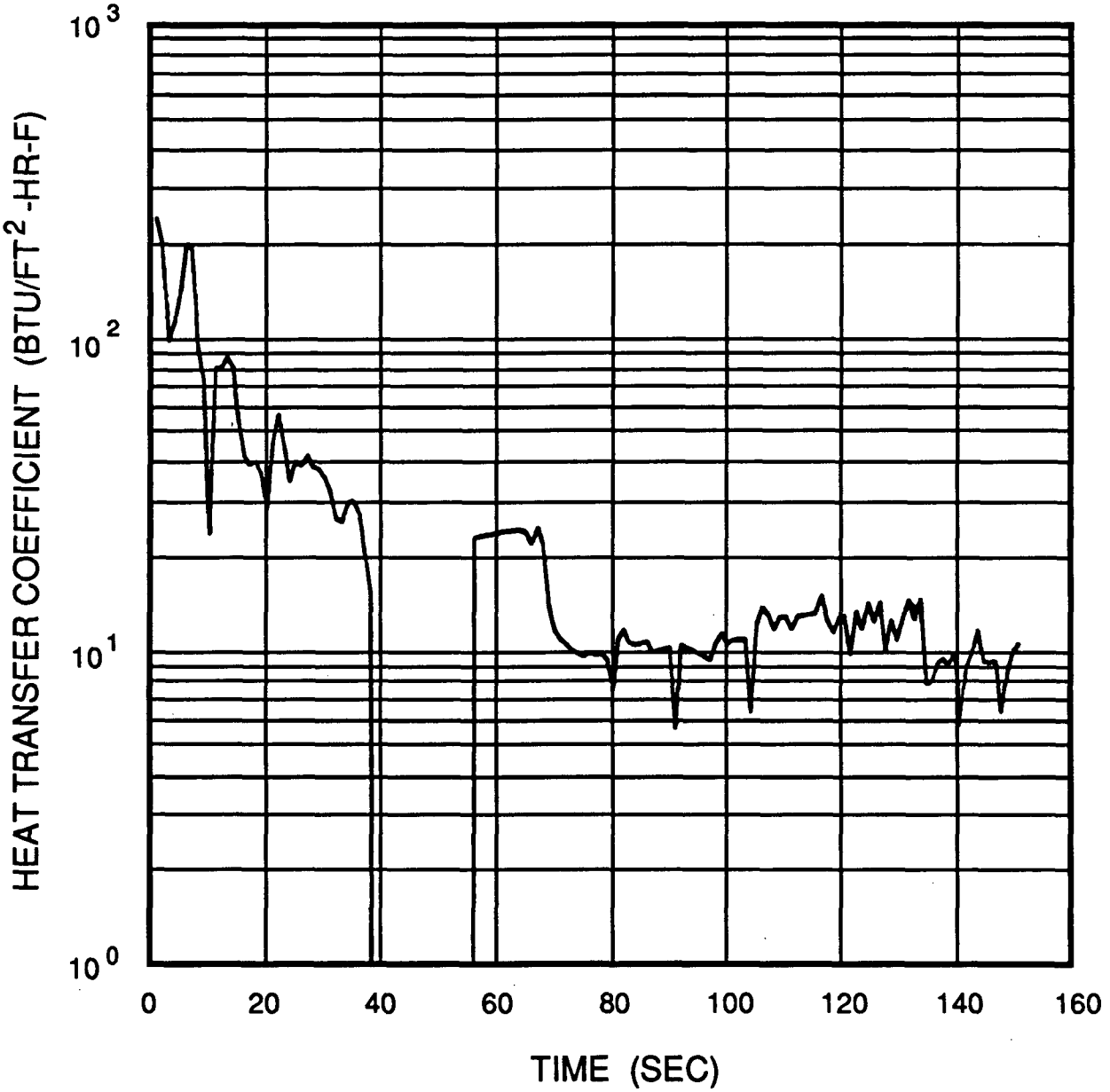
ZION STATION UFSAR

Figure 15.6-20
FLUID QUALITY
DECLG (CD=0.4)
VANTAGE 5 WITHOUT IFMs
JULY 1993



ZION STATION UFSAR

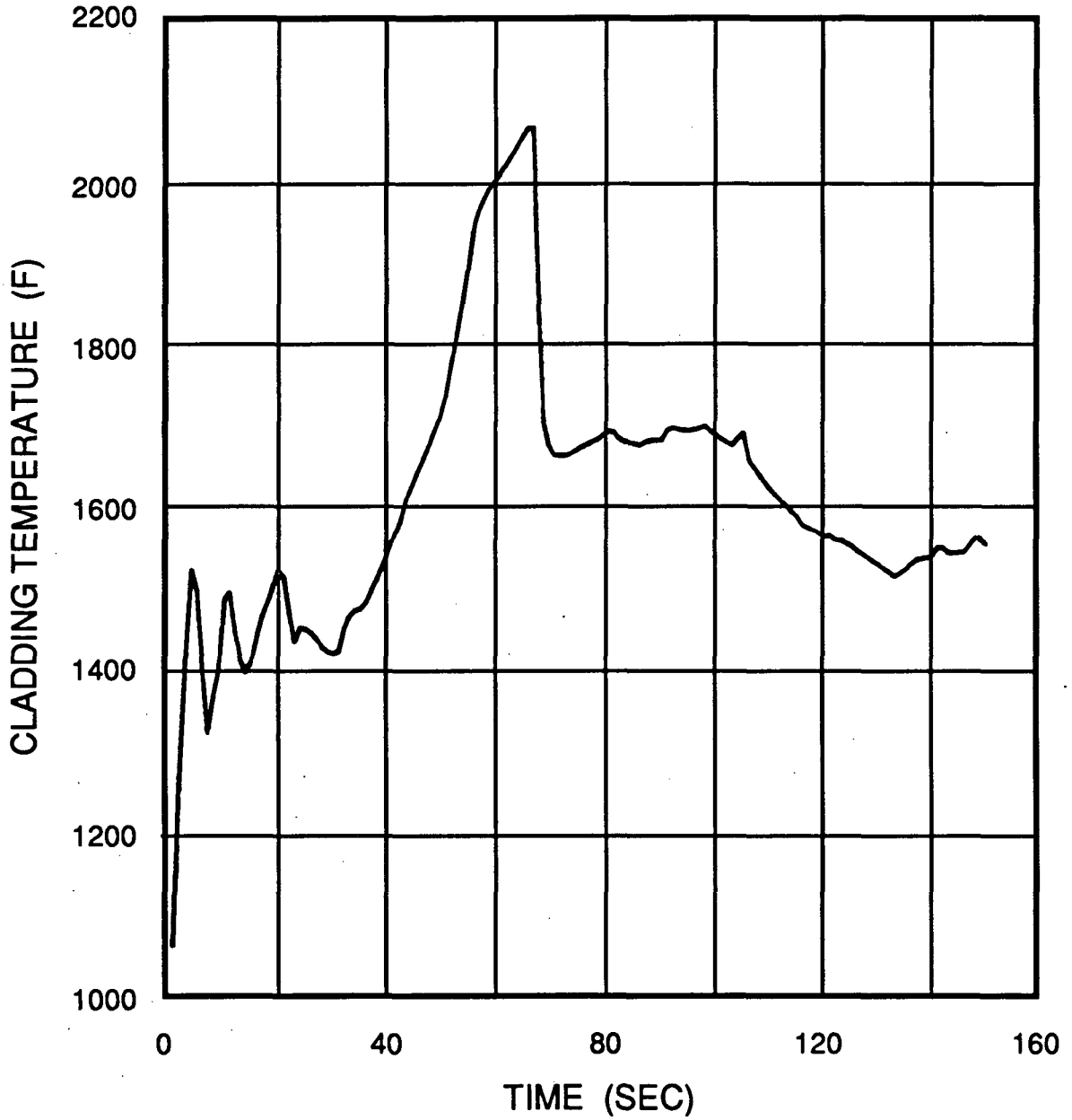
Figure 15.6-21
MASS VELOCITY
DECLG (CD=0.4)
VANTAGE 5 WITHOUT IFMs
JULY 1993



ZION STATION UFSAR

Figure 15.6-22
CLADDING HEAT TRANSFER
COEFFICIENT DECLG (CD=0.4)
VANTAGE 5 WITHOUT IFMs

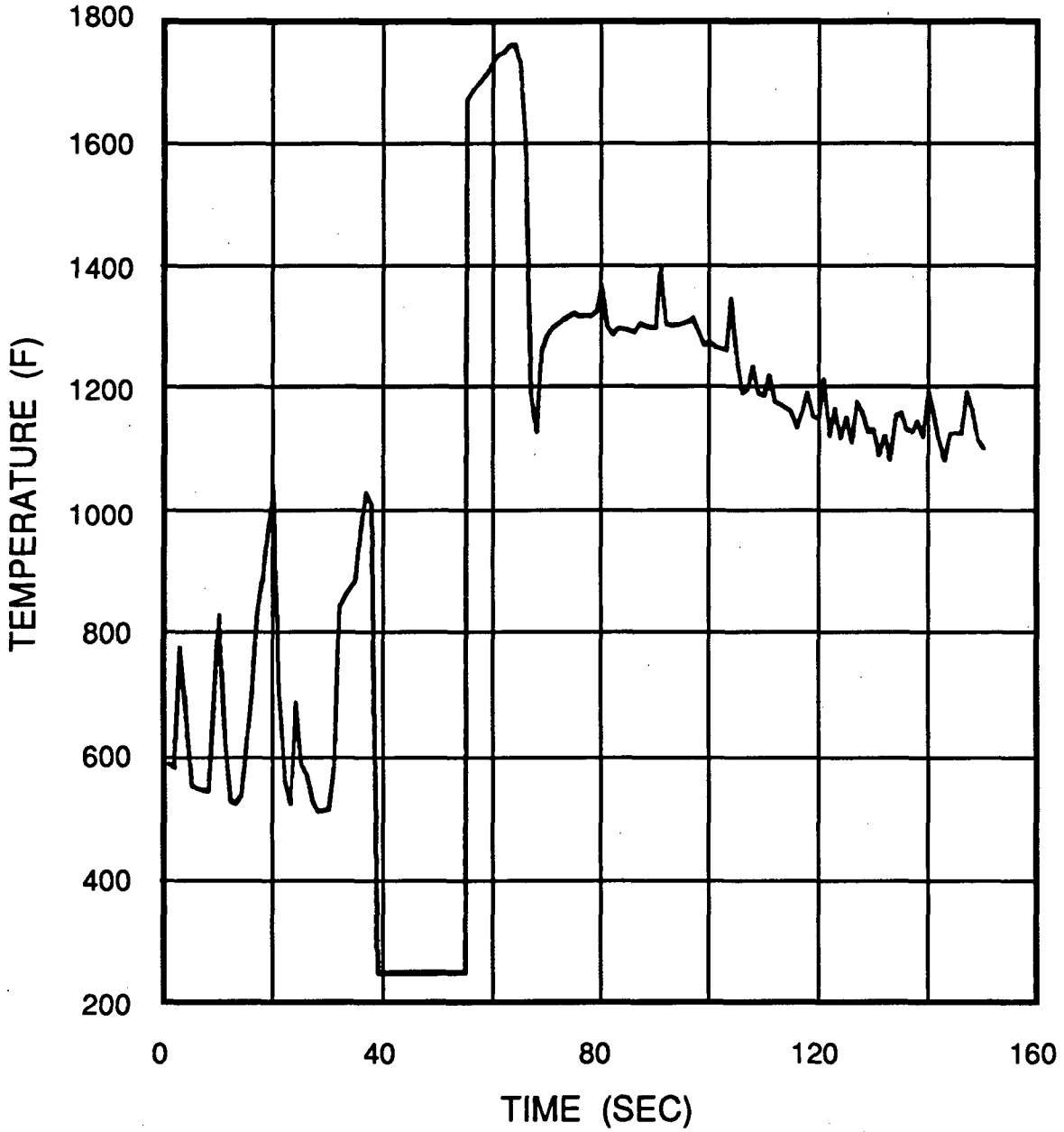
JULY 1993



ZION STATION UFSAR

Figure 15.6-23
CLADDING TEMPERATURE
DECLG (CD=0.4)
VANTAGE 5 WITHOUT IFMs

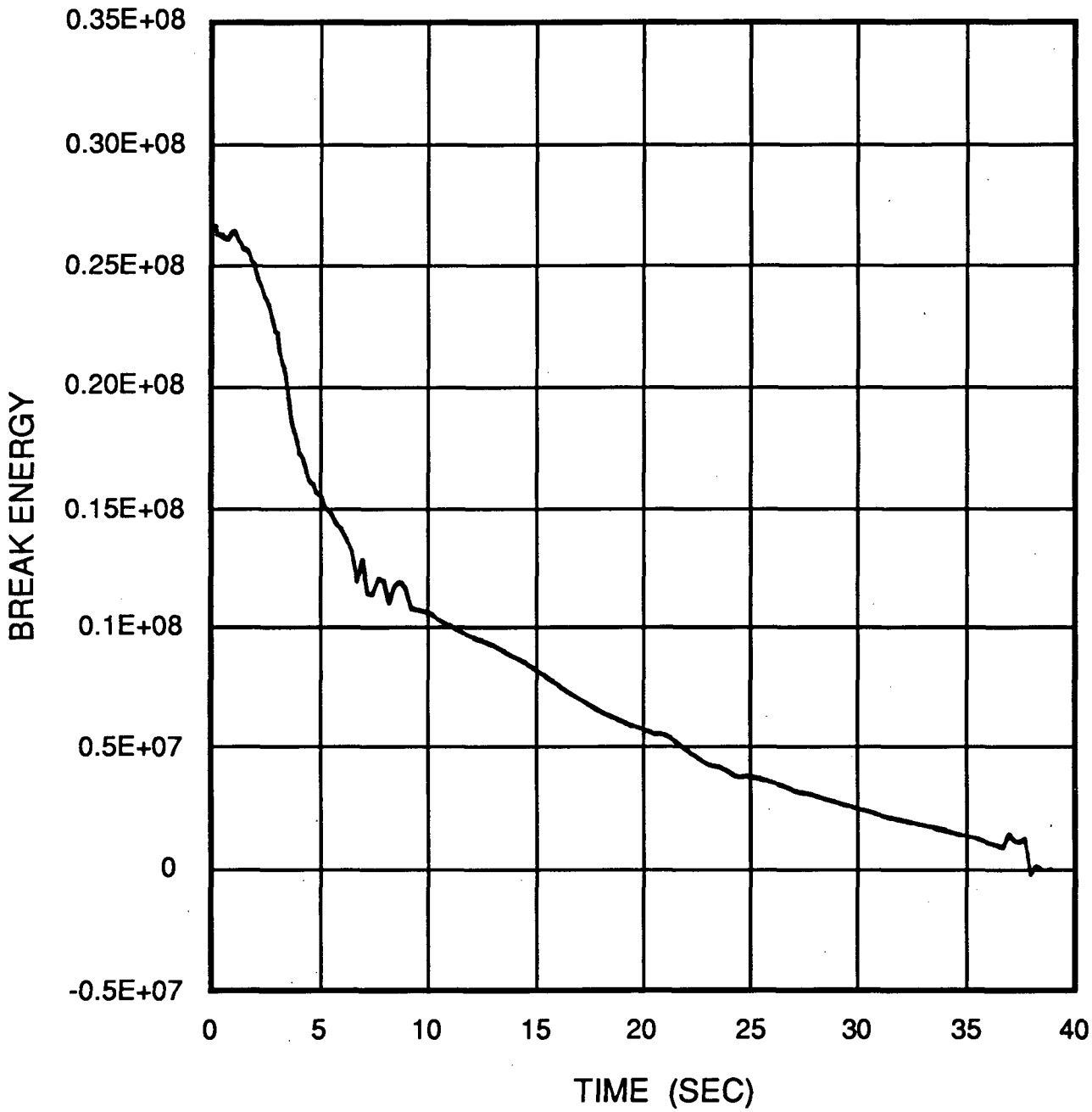
JULY 1993



ZION STATION UFSAR

Figure 15.6-24
HOT SPOT FLUID TEMPERATURE
DECLG (CD=0.4)
VANTAGE 5 WITHOUT IFMs

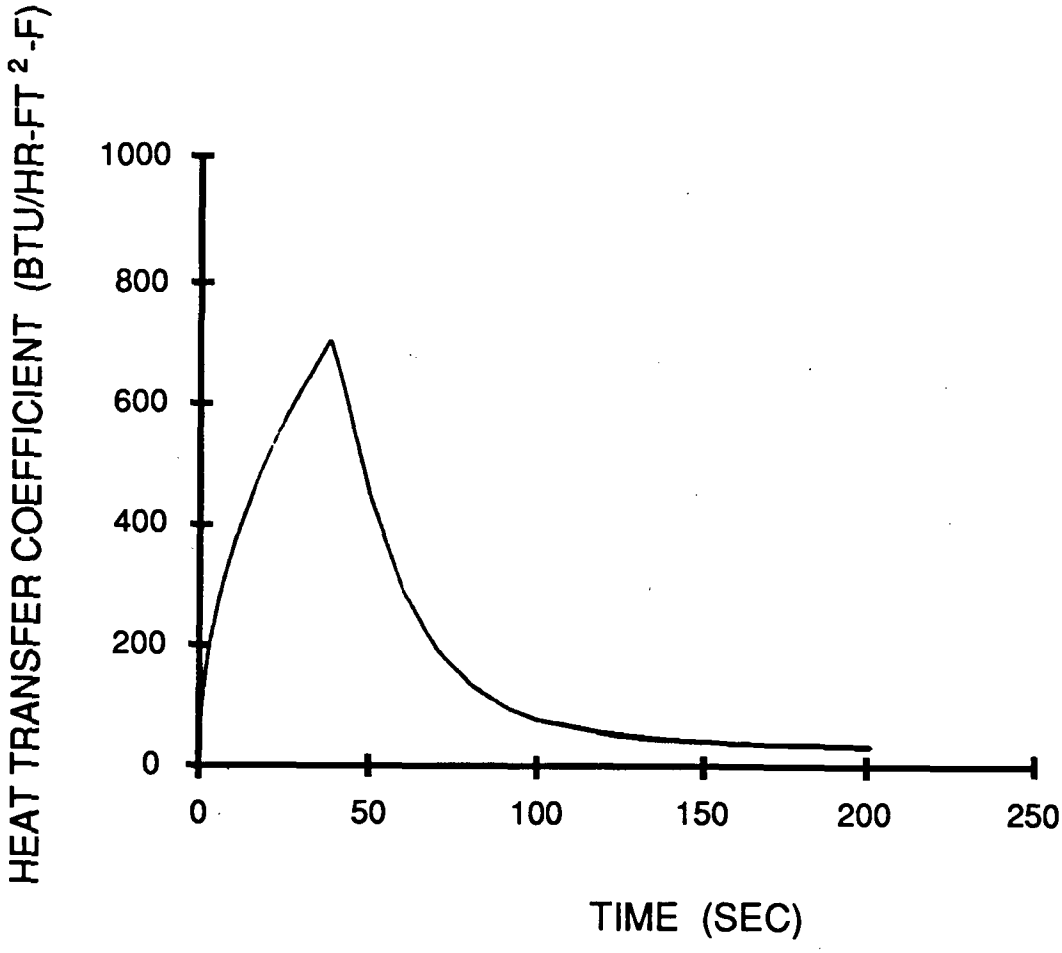
JULY 1993



ZION STATION UFSAR

Figure 15.6-25
BREAK ENERGY RELEASE
(BLOWDOWN)
DECLG (CD=0.4)

JULY 1993



ZION STATION UFSAR

Figure 15.6-26
CONTAINMENT WALL CONDENSING
HEAT TRANSFER COEFFICIENT
DECLG (CD=0.4)

JULY 1993

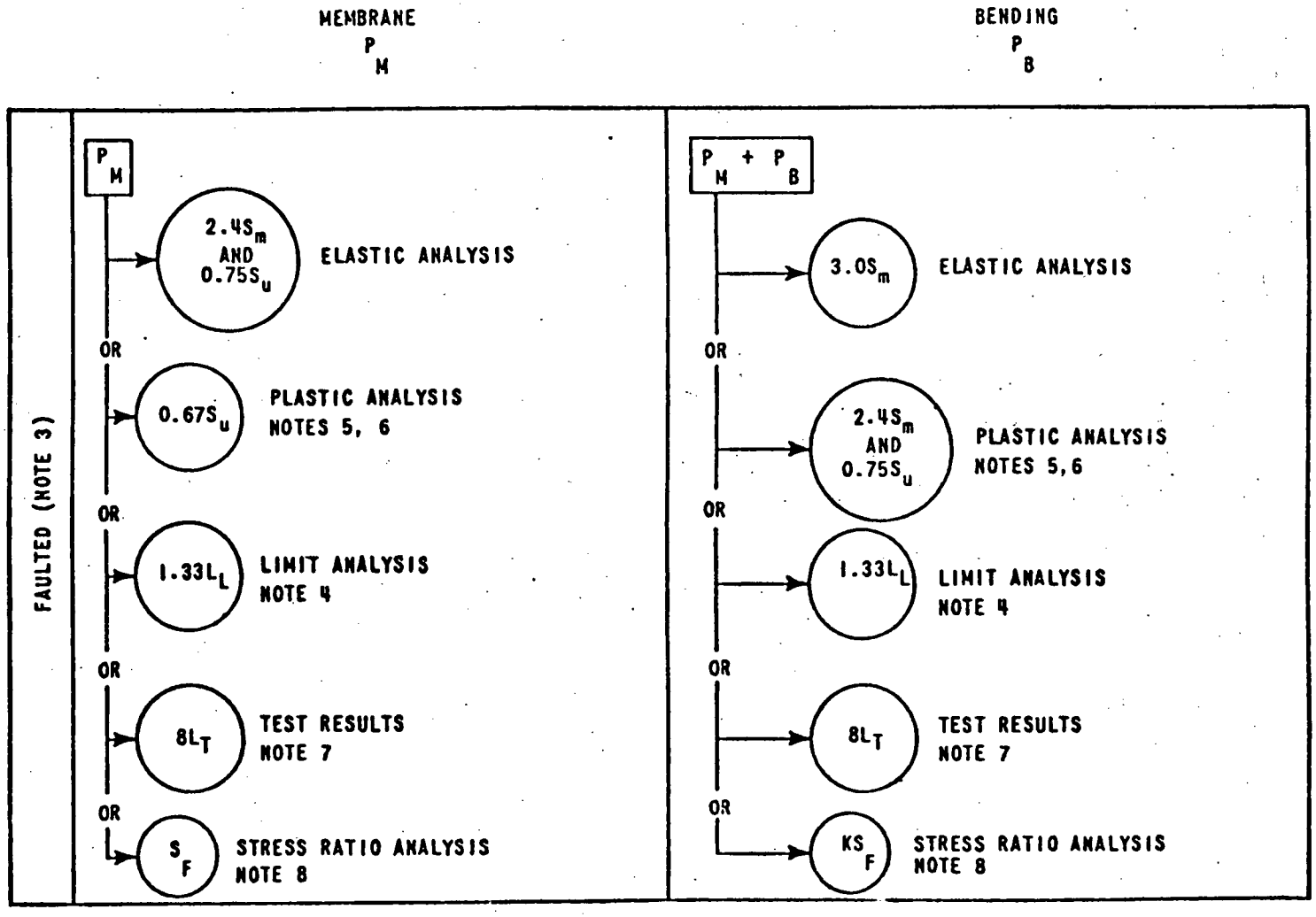
- Figure 15.6-27 was Intentionally Deleted -

- Figure 15.6-28 was Intentionally Deleted -

- Figure 15.6-29 was Intentionally Deleted -

S_m = DESIGN STRESS INTENSITY VALUE AT TEMPERATURE, GIVEN BY THE ASME SECTION III CODE

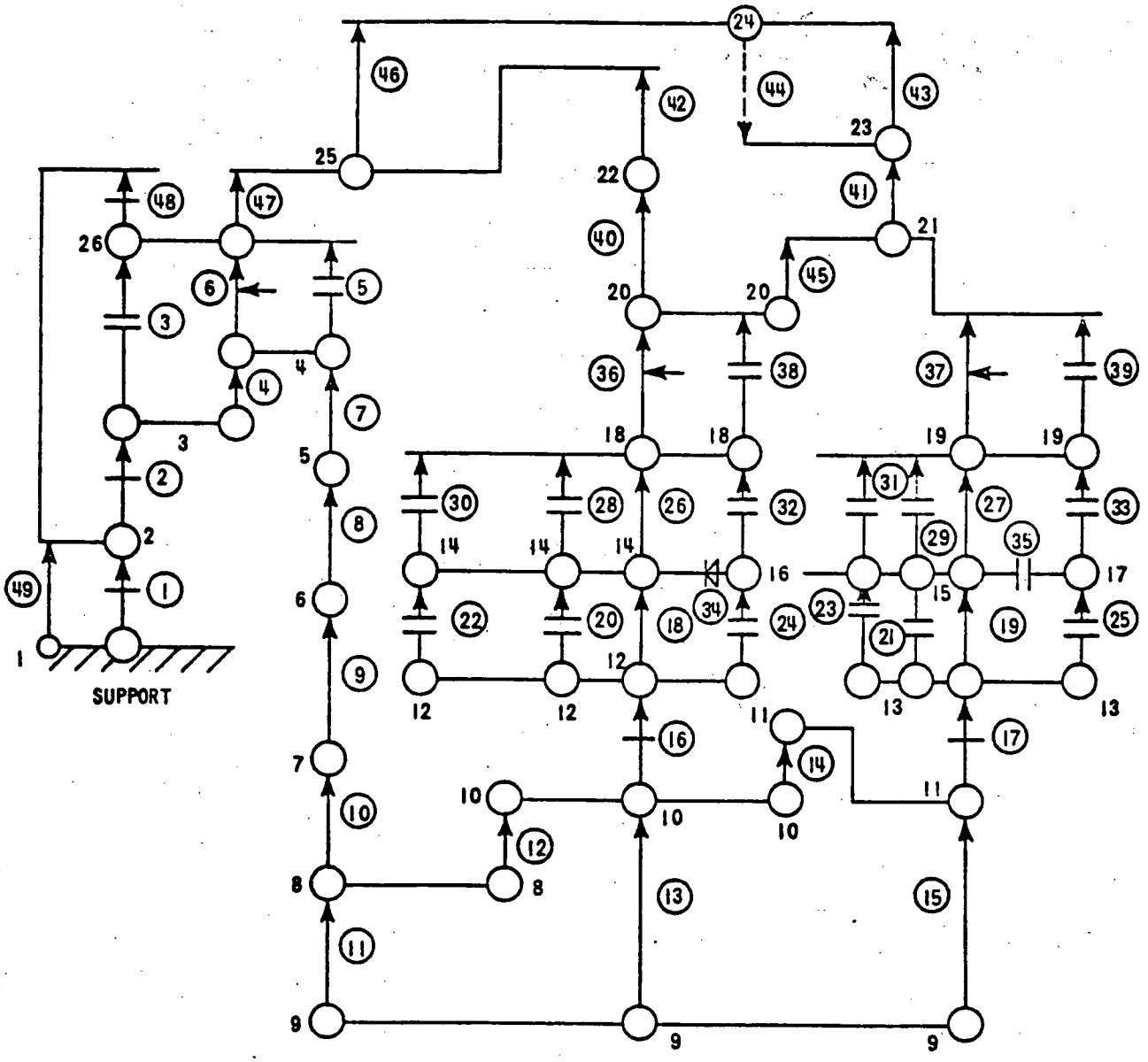
PRIMARY STRESSES (NOTES 1, 2, 9 AND 10)



PRIMARY STRESSES (NOTES 1, 2, 9 AND 10)

ZION STATION UFSAR

Figure 15.6-30
REACTOR INTERNALS ALLOWABLE STRESS CRITERIA



TWO-WAY SPRING-DAMPER

ONE-WAY SPRING-SNUBBER

PRELOADED ONE-WAY SPRING

ONE-WAY SPRING-SNUBBER WITH INITIAL GAP

SLIDING ELEMENT n

ELEMENT NUMBERED n

NODE m WITH MASS REPRESENTING INTERNAL COMPONENT

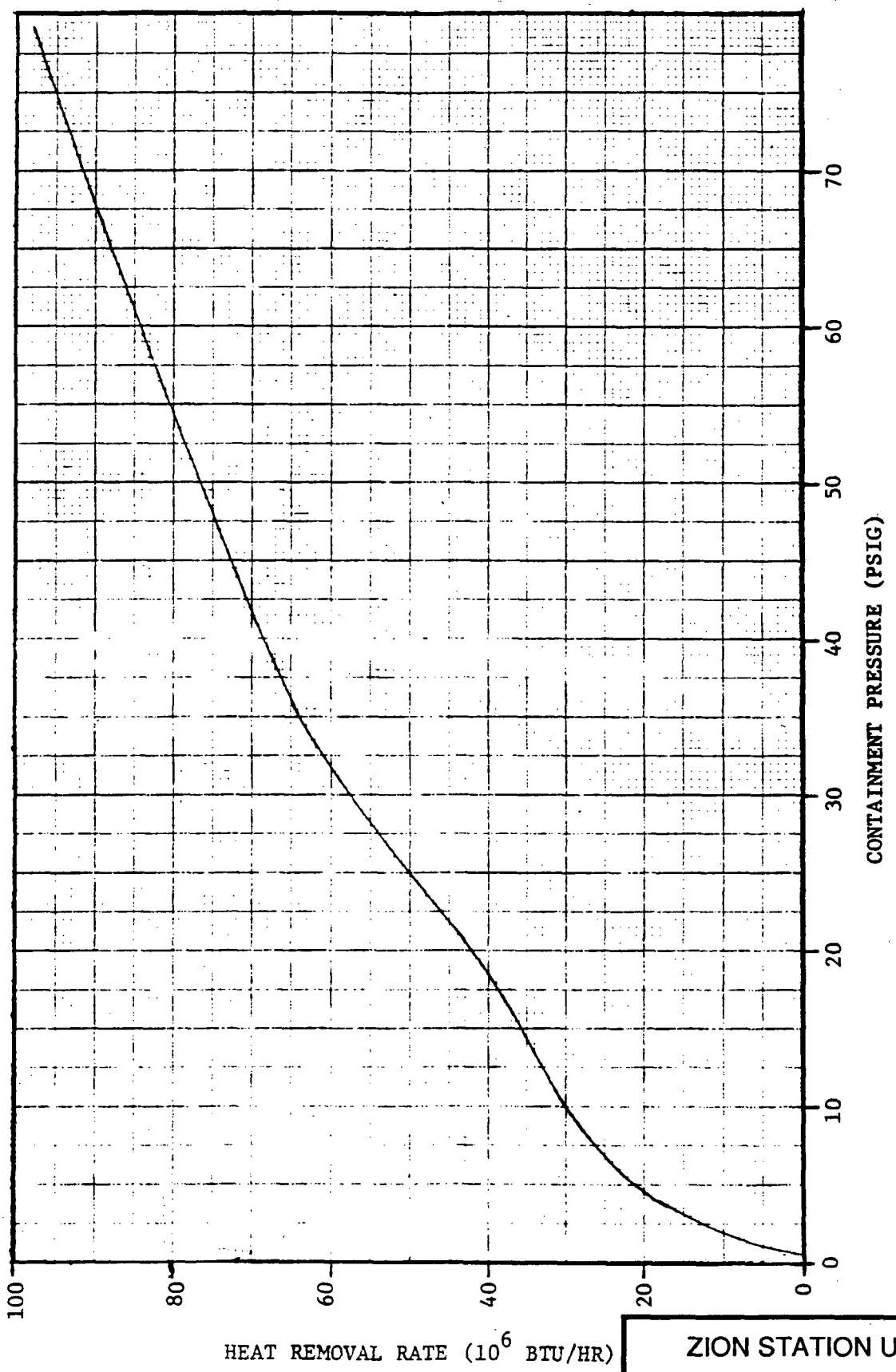
NODE m WITHOUT MASS

ELEMENT WITHOUT STIFFNESS

ZION STATION UFSAR

Figure 15.6-31

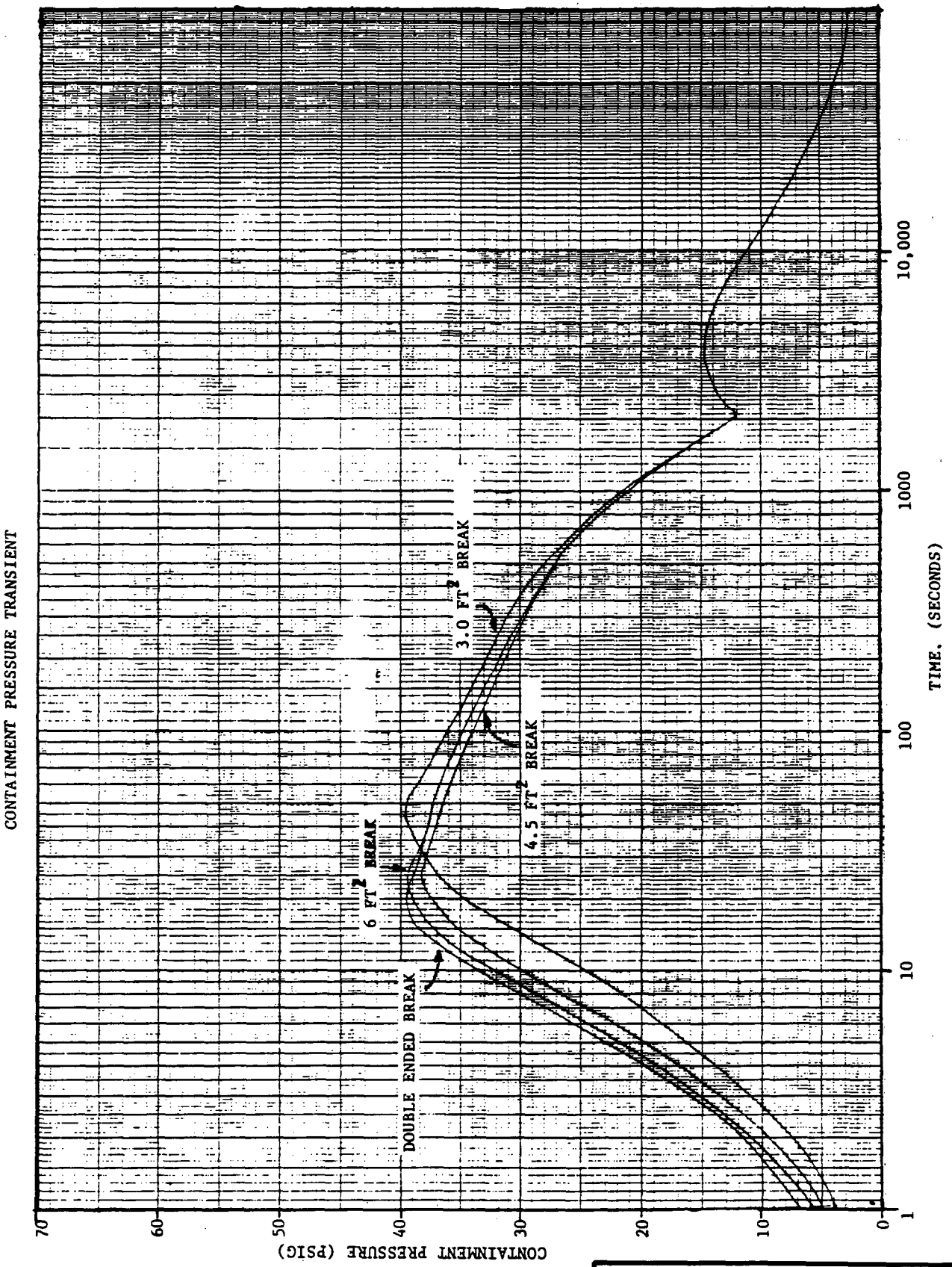
LOOP REACTOR MATHEMATICAL MODEL FOR VERTICAL RESPONSE



ZION STATION UFSAR

Figure 15.6-32

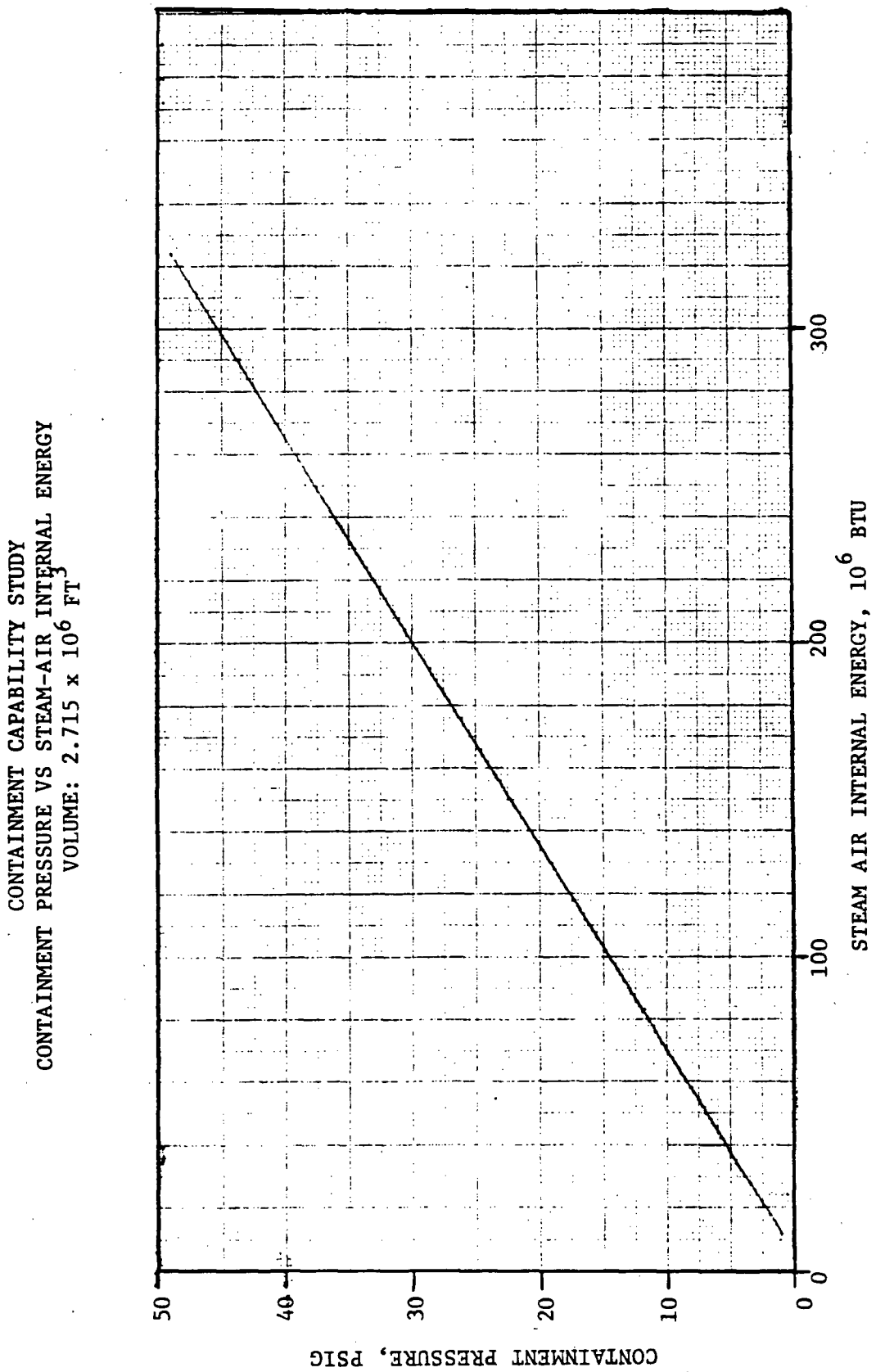
FAN COOLER HEAT REMOVAL RATE VS. CONTAINMENT PRESSURE



ZION STATION UFSAR

Figure 15.6-33

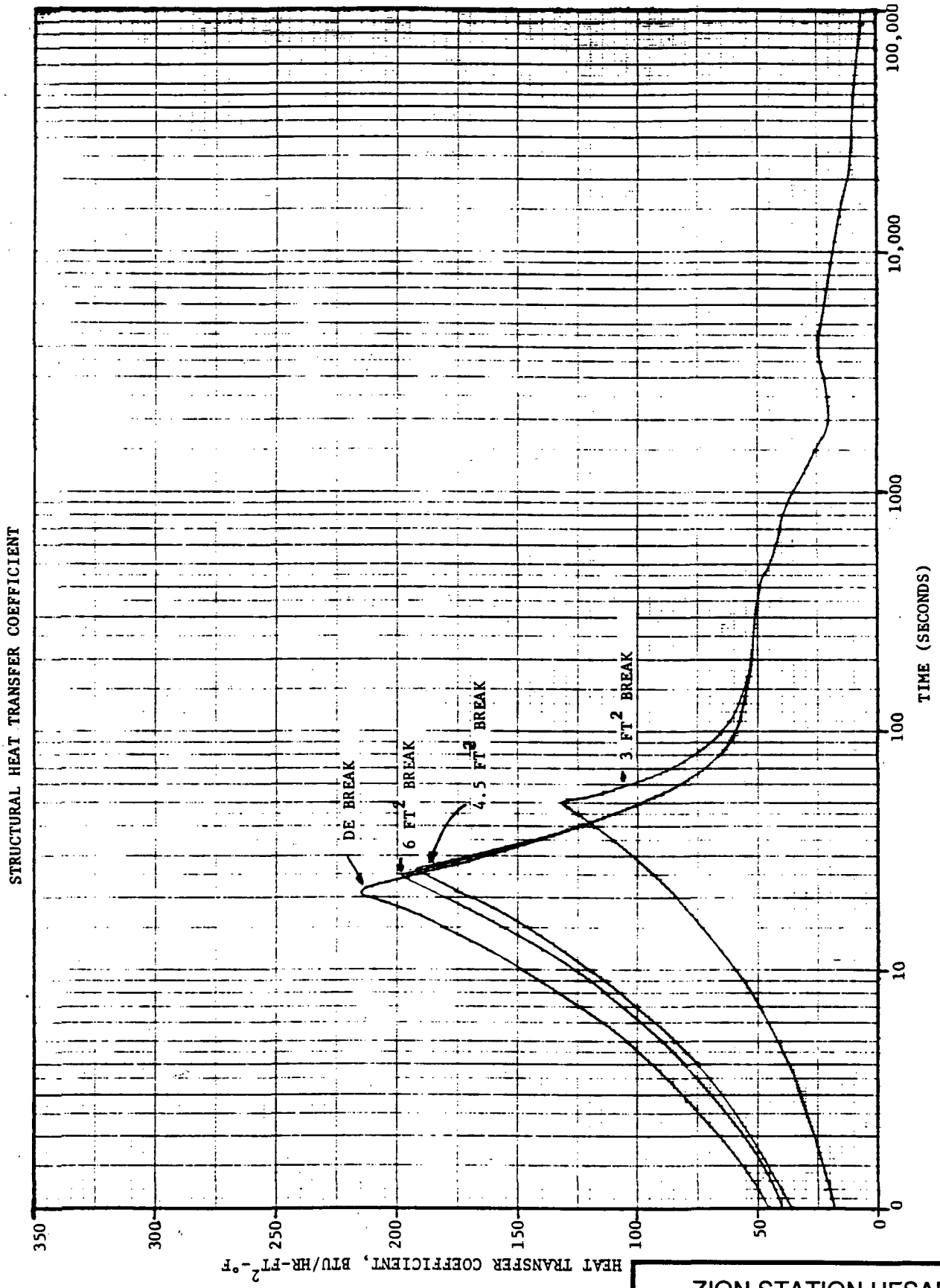
CONTAINMENT PRESSURE TRANSIENT



ZION STATION UFSAR

Figure 15.6-34

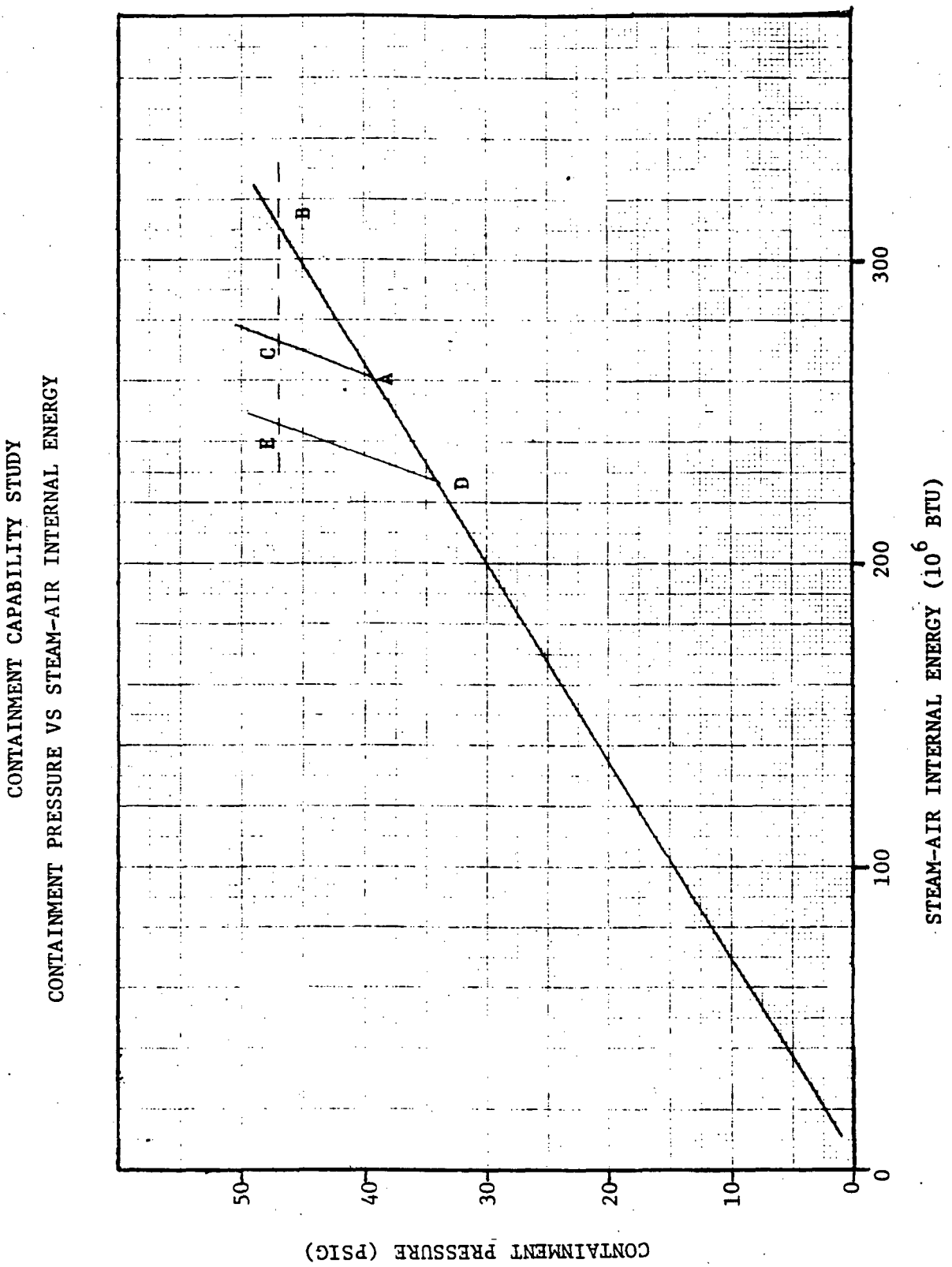
CONTAINMENT PRESSURE VS. STEAM-AIR INTERNAL ENERGY



ZION STATION UFSAR

Figure 15.6-35

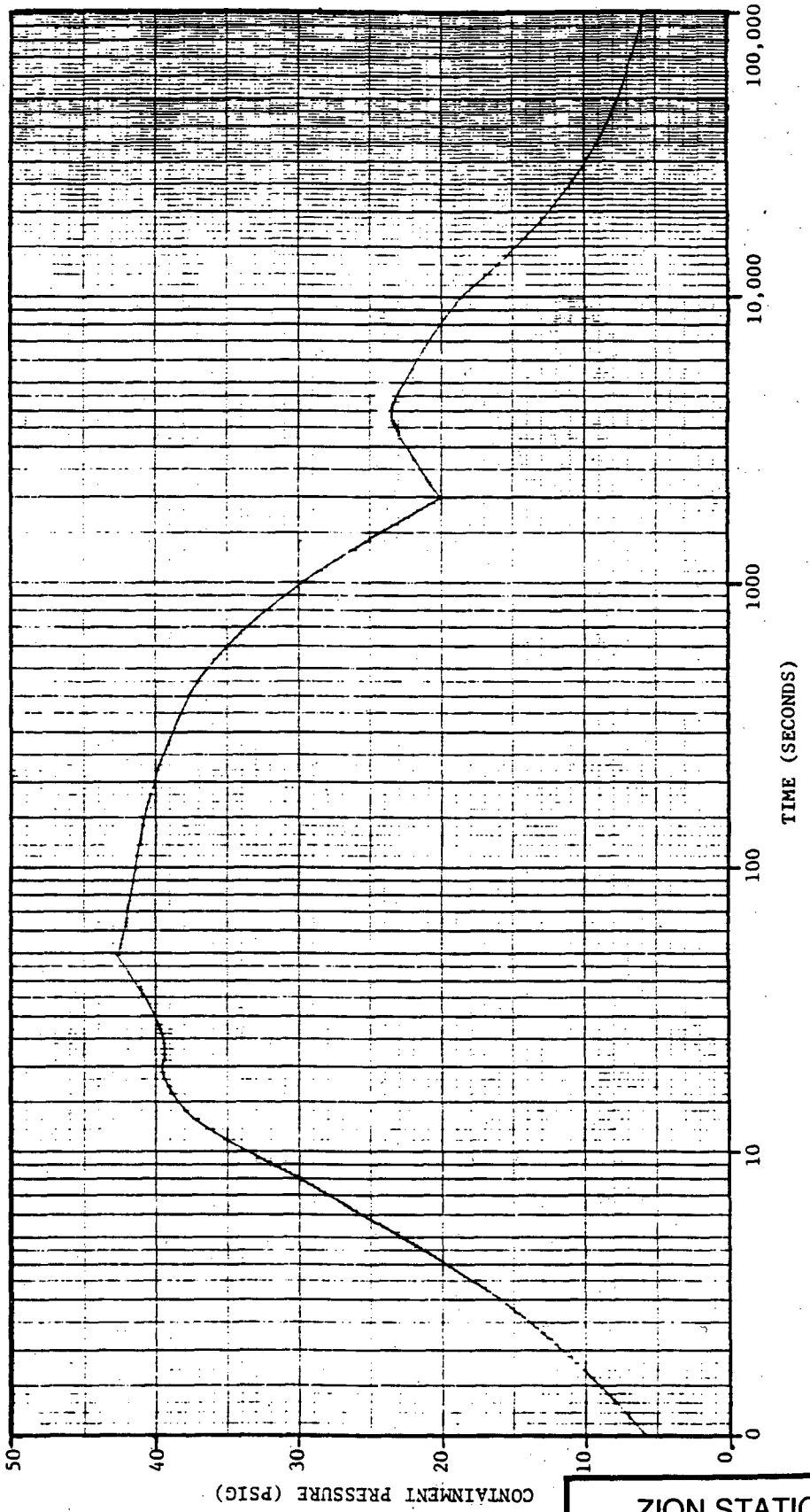
STRUCTURAL HEAT TRANSFER COEFFICIENT



ZION STATION UFSAR

Figure 15.6-36

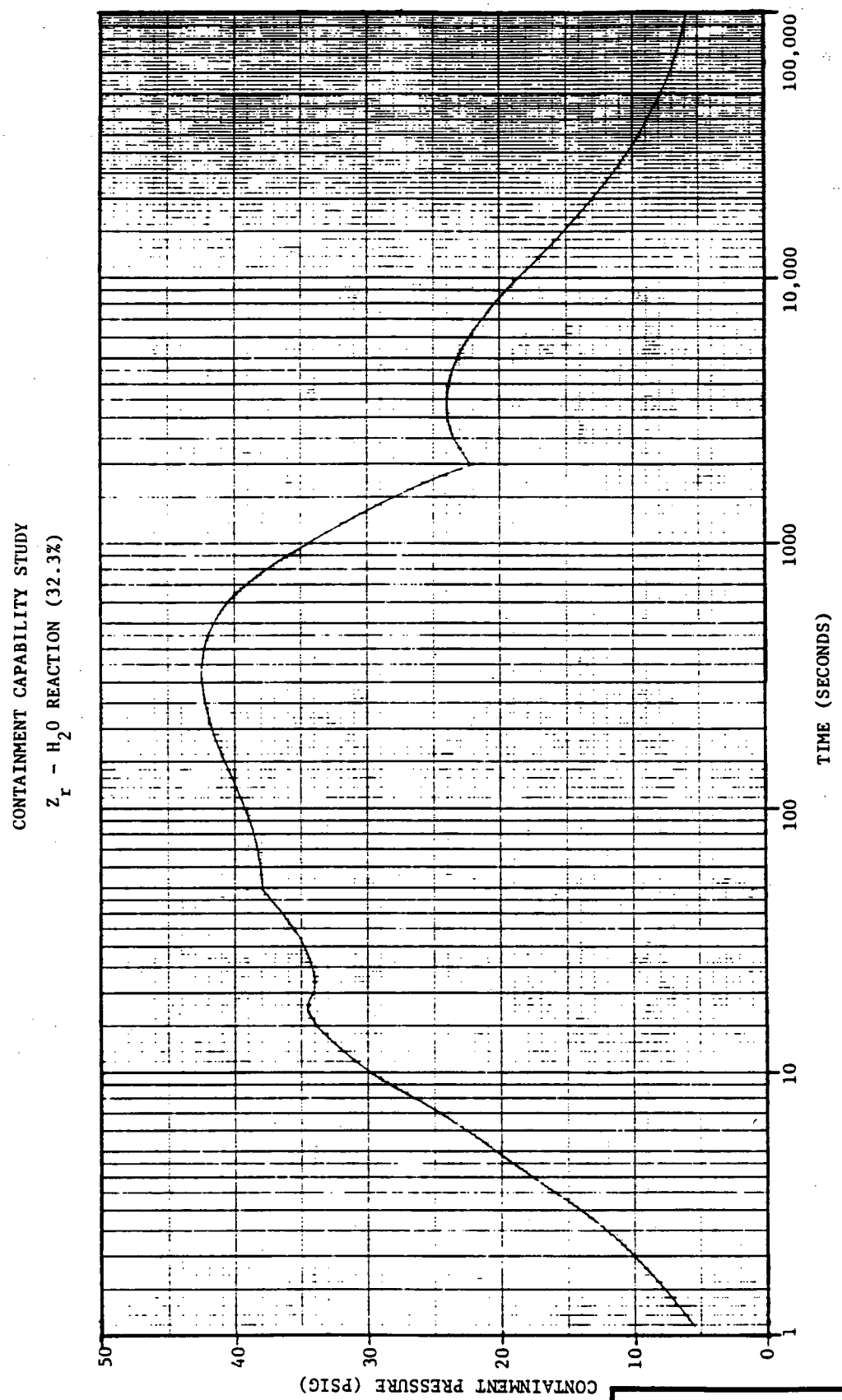
CONTAINMENT CAPABILITY STUDY, CONTAINMENT PRESSURE VS. STEAM-AIR INTERNAL ENERGY



ZION STATION UFSAR

Figure 15.6-37

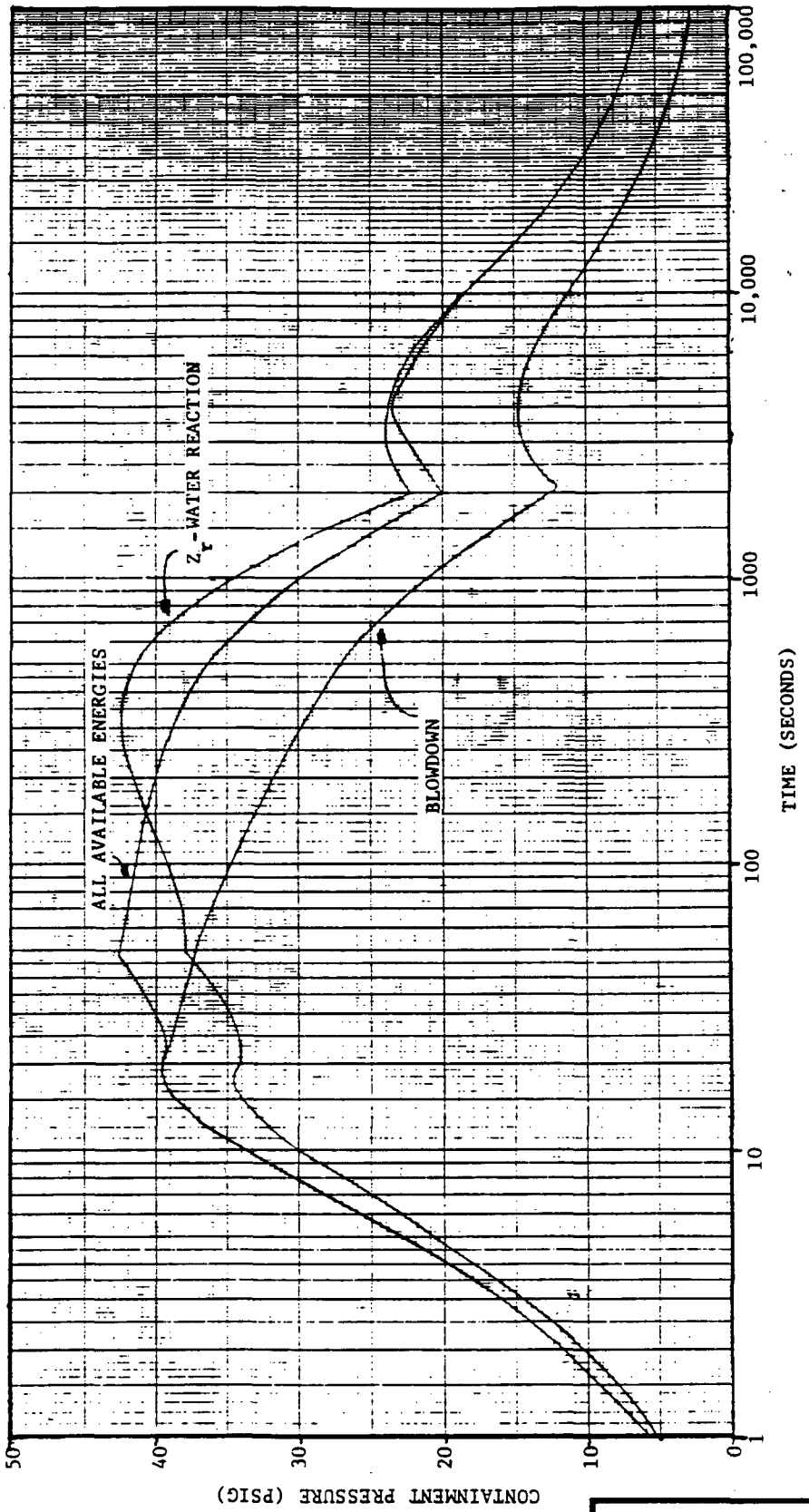
CONTAINMENT CAPABILITY STUDY, ALL AVAILABLE ENERGIES



ZION STATION UFSAR

Figure 15.6-38

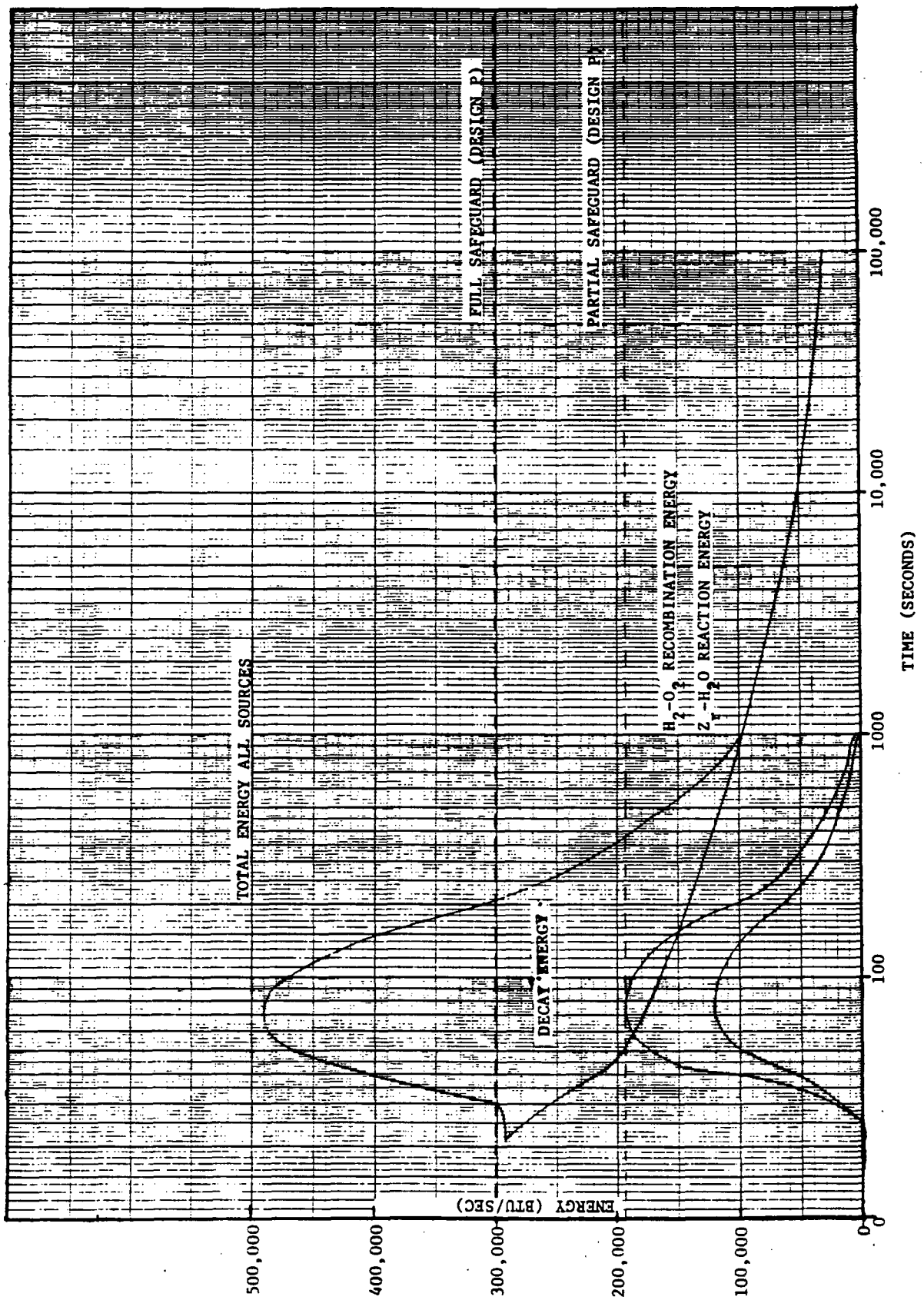
CONTAINMENT CAPABILITY STUDY, ZR - H_2O REACTION



ZION STATION UFSAR

Figure 15.6-39

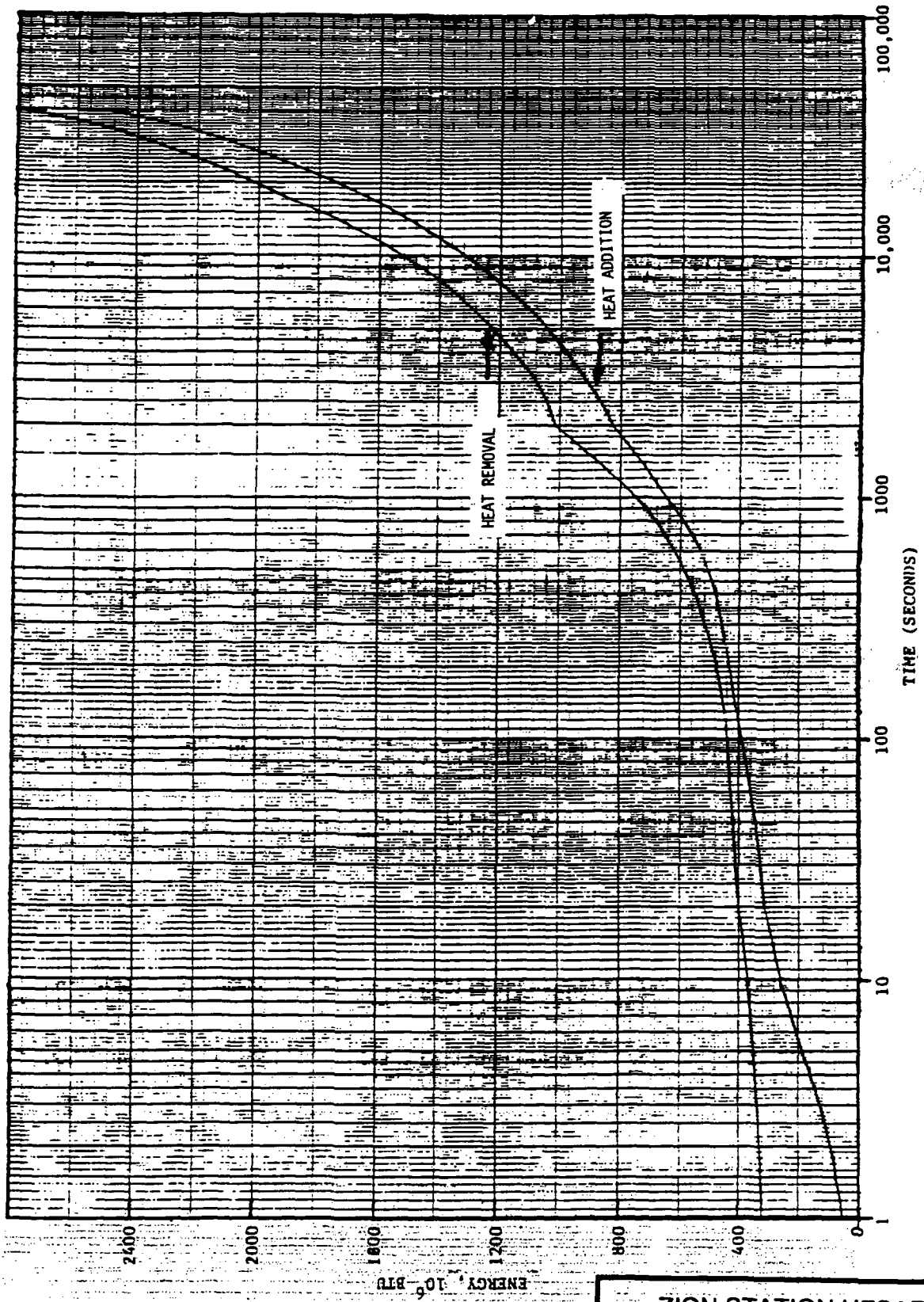
CONTAINMENT PRESSURE TRANSIENTS



ZION STATION UFSAR

Figure 15.6-40

CONTAINMENT CAPABILITY STUDY, RATE OF ENERGY ADDITION

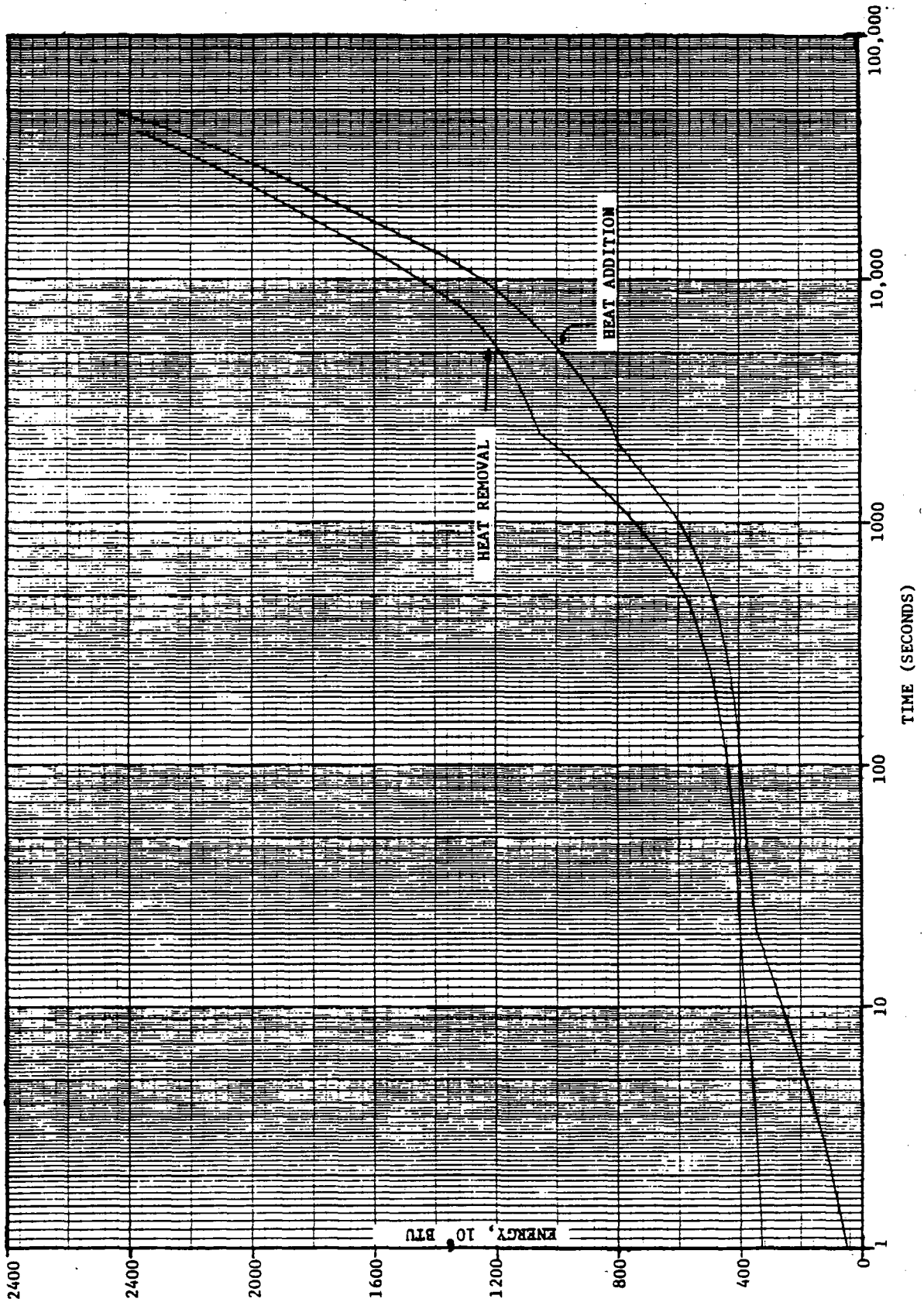


ZION STATION UFSAR

Figure 15.6-41

CONTAINMENT CAPABILITY, CASE 1

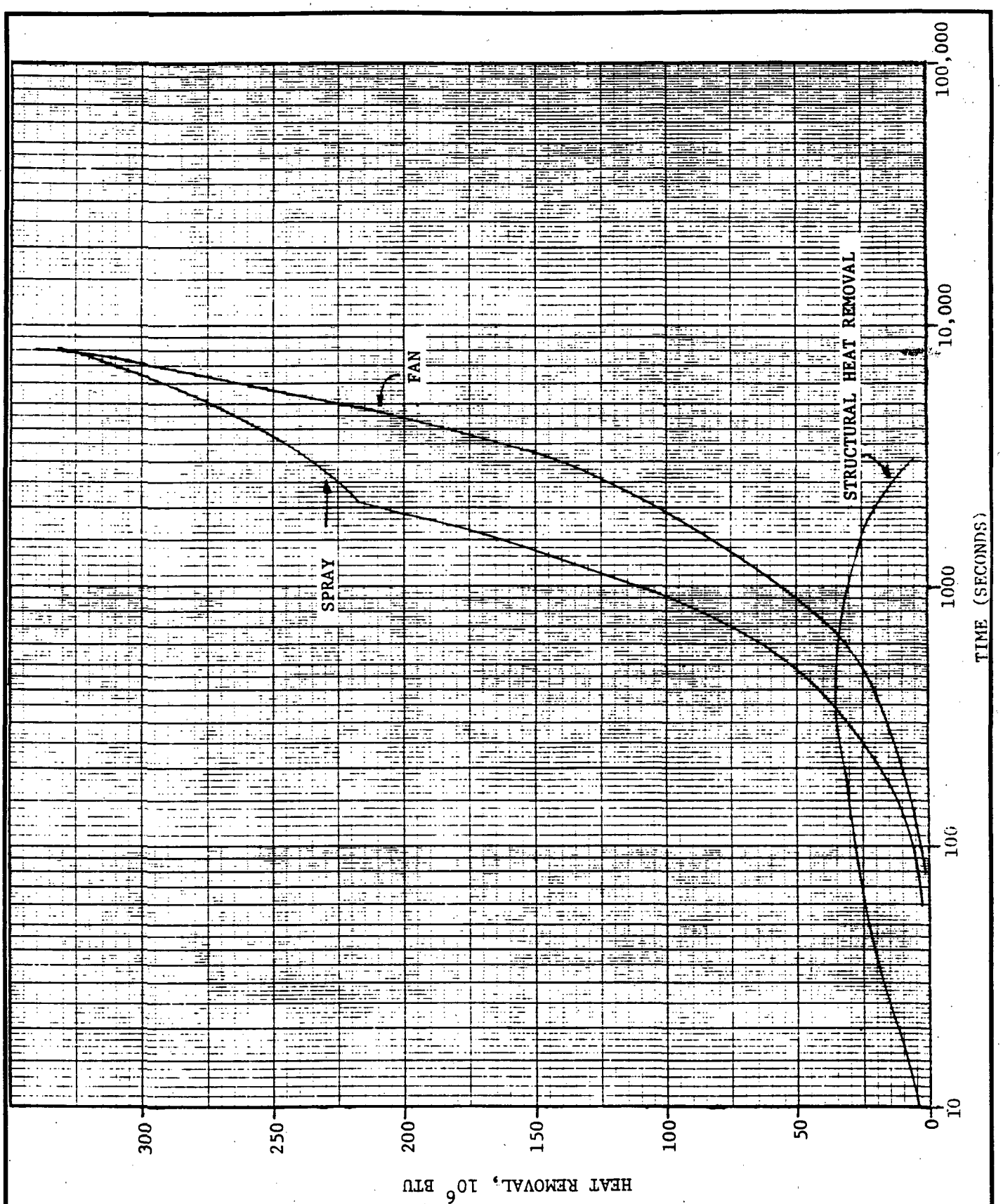
JULY 1995



ZION STATION UFSAR

Figure 15.6-42

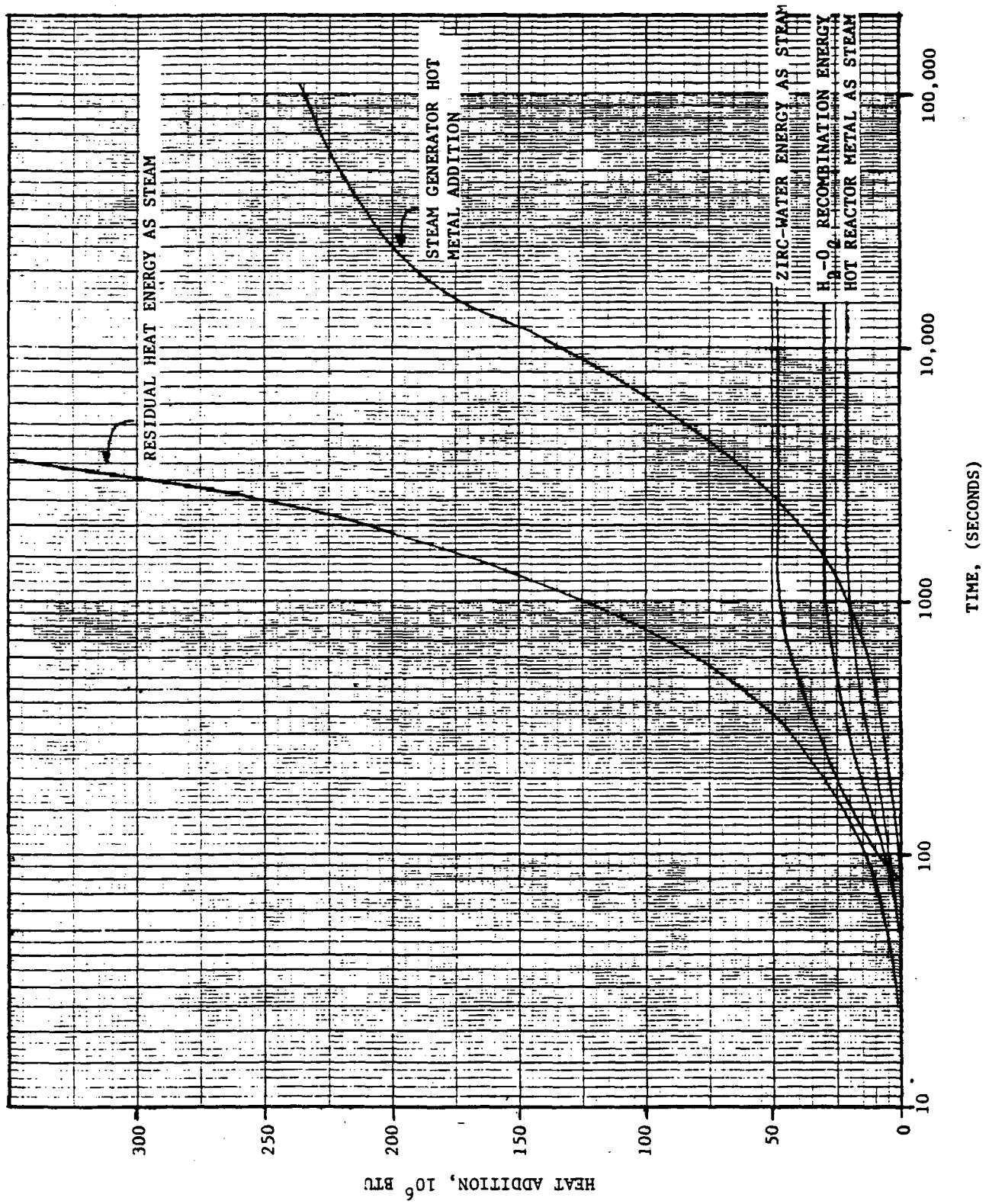
CONTAINMENT CAPABILITY, CASE 2



ZION STATION UFSAR

Figure 15.6-43

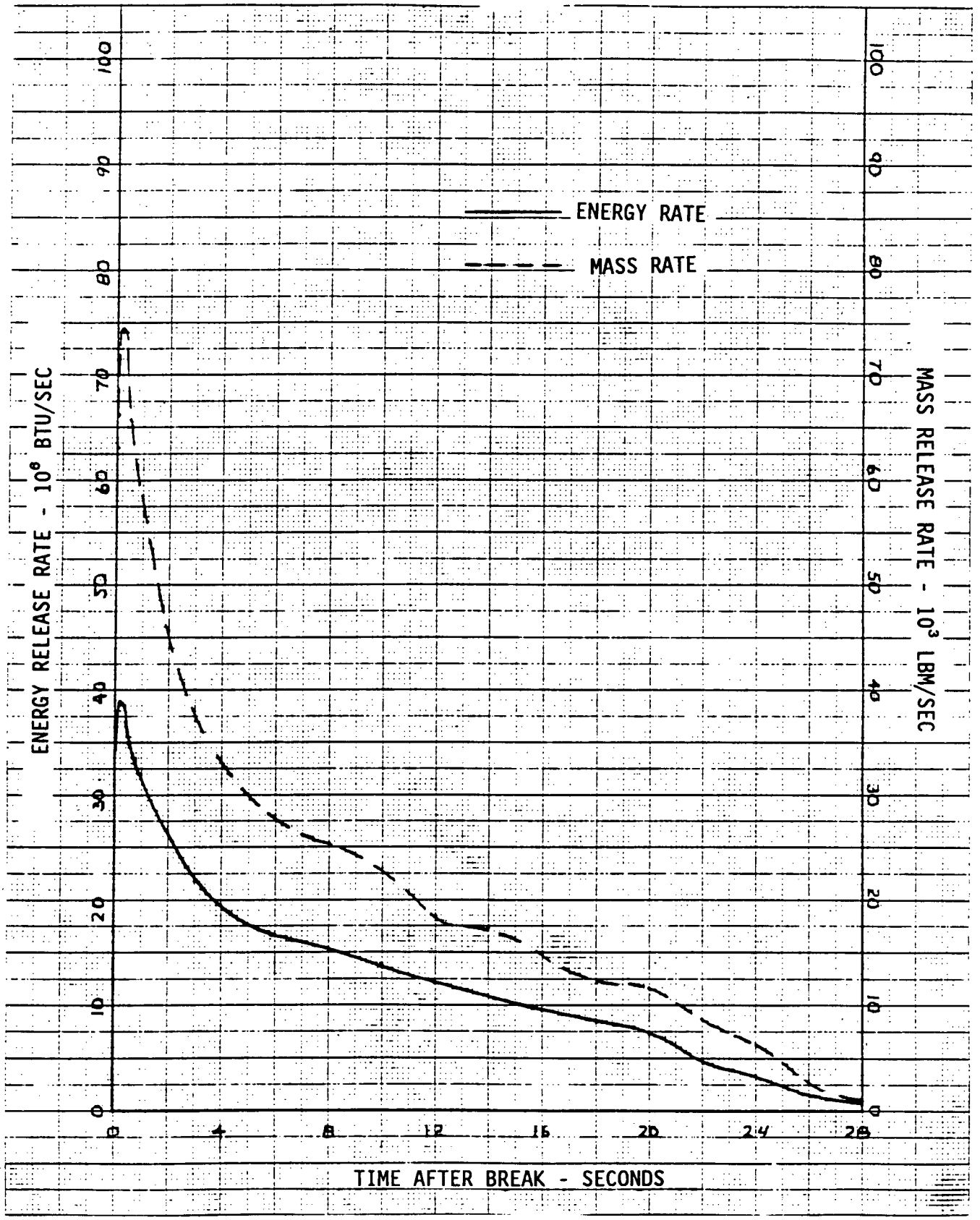
INDIVIDUAL CONTRIBUTIONS TO HEAT REMOVAL



ZION STATION UFSAR

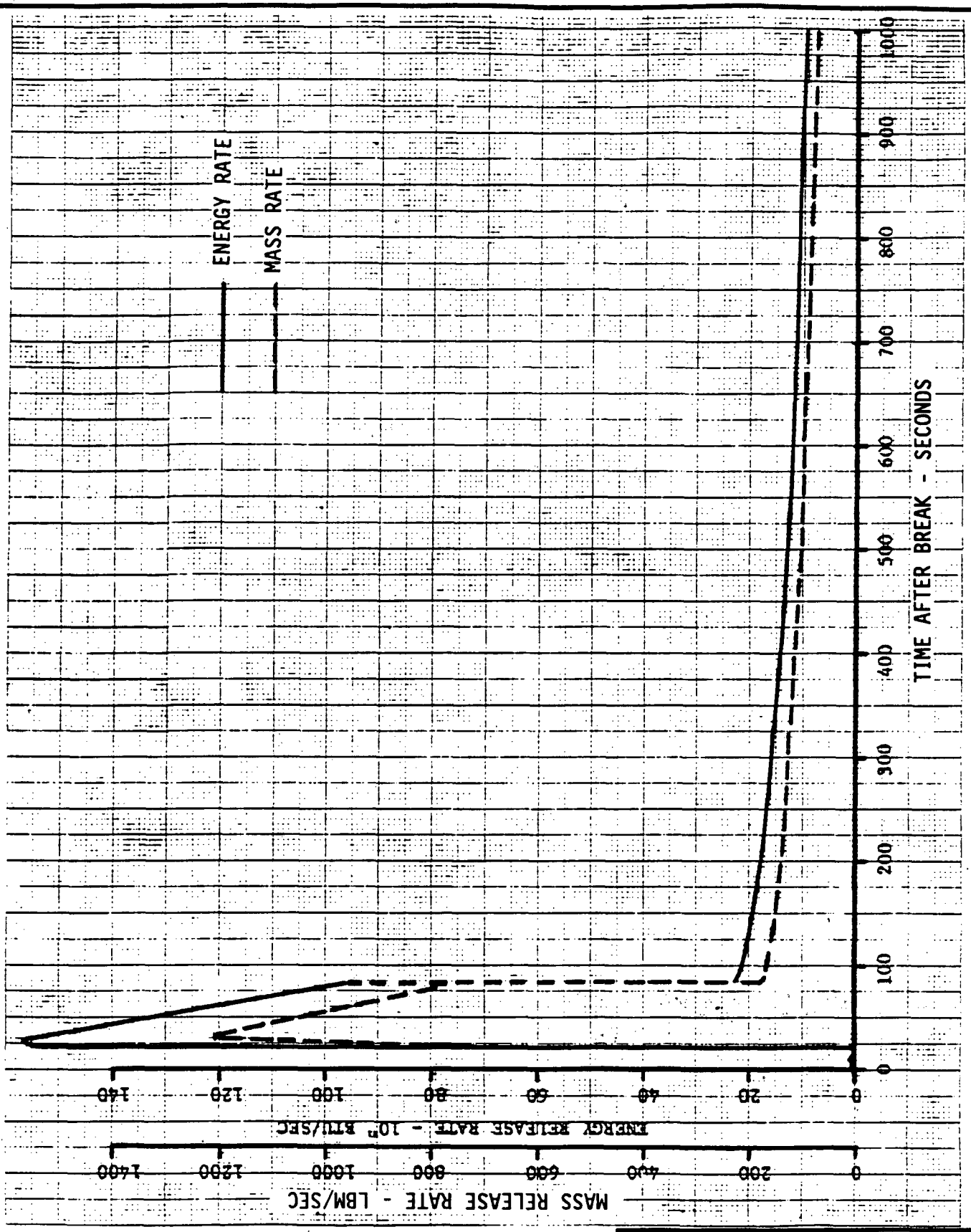
Figure 15.6-44

CONTAINMENT CAPABILITY STUDY, HEAT SOURCE



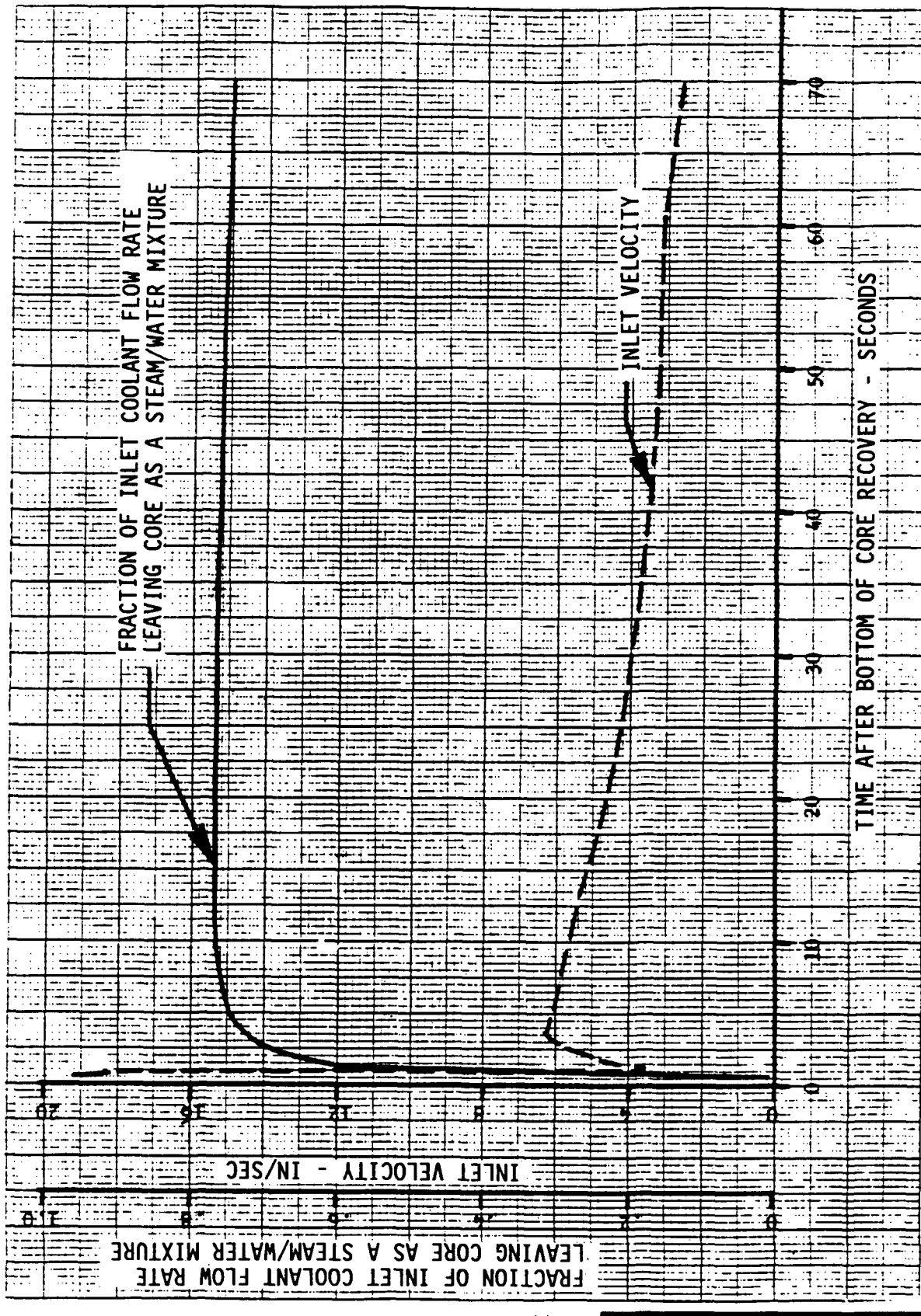
ZION STATION UFSAR

Figure 15.6-45
MASS AND ENERGY RELEASE RATES INTO CONTAINMENT DURING BLOWDOWN



ZION STATION UFSAR

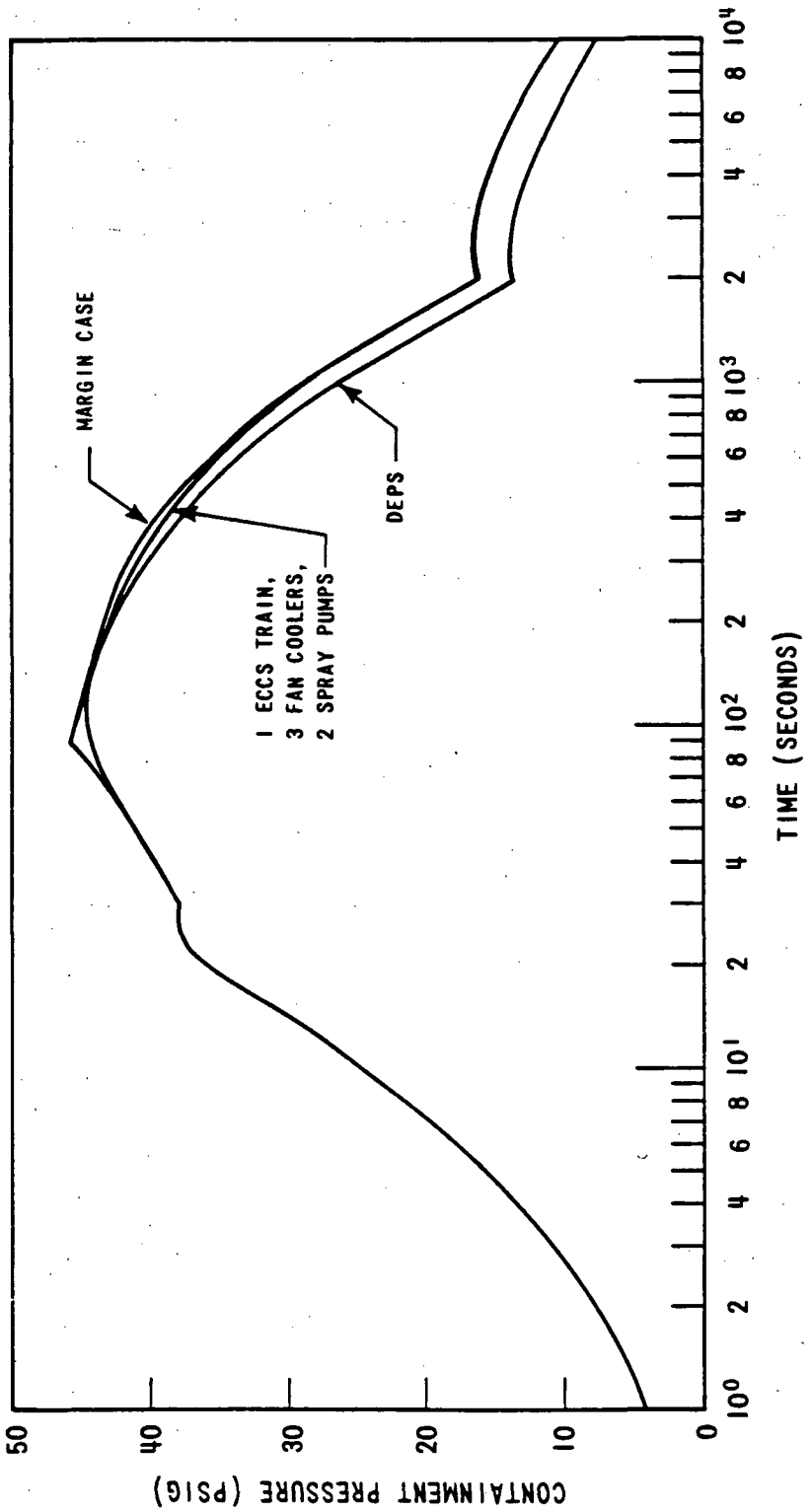
Figure 15.6-46
MASS AND ENERGY RELEASE RATES INTO CONTAINMENT DURING REFLOOD



ZION STATION UFSAR

Figure 15.6-47

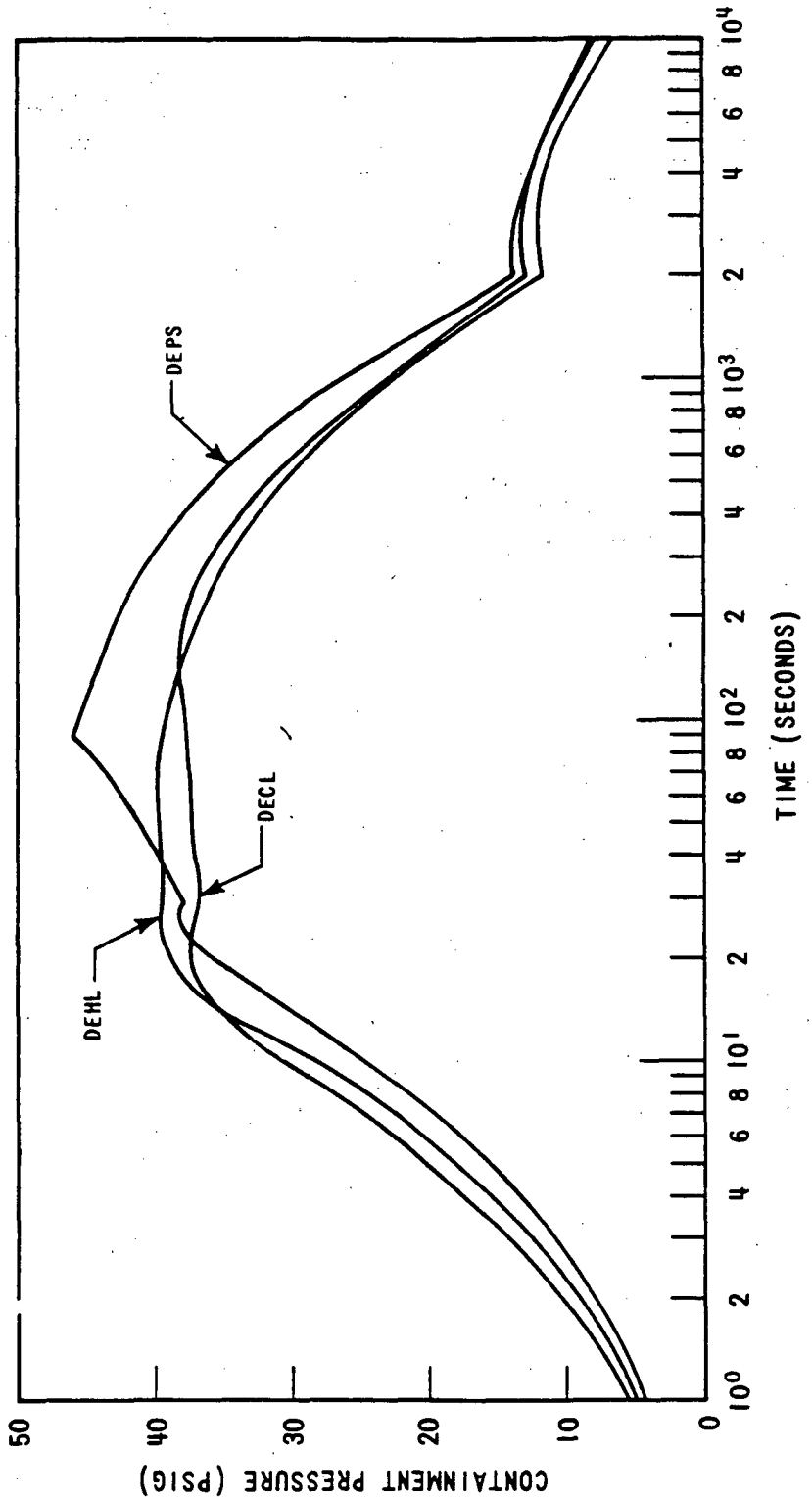
CORE INLET VELOCITY, DOUBLE ENDED PUMP SUCTION BREAK



ZION STATION UFSAR

Figure 15.6-48

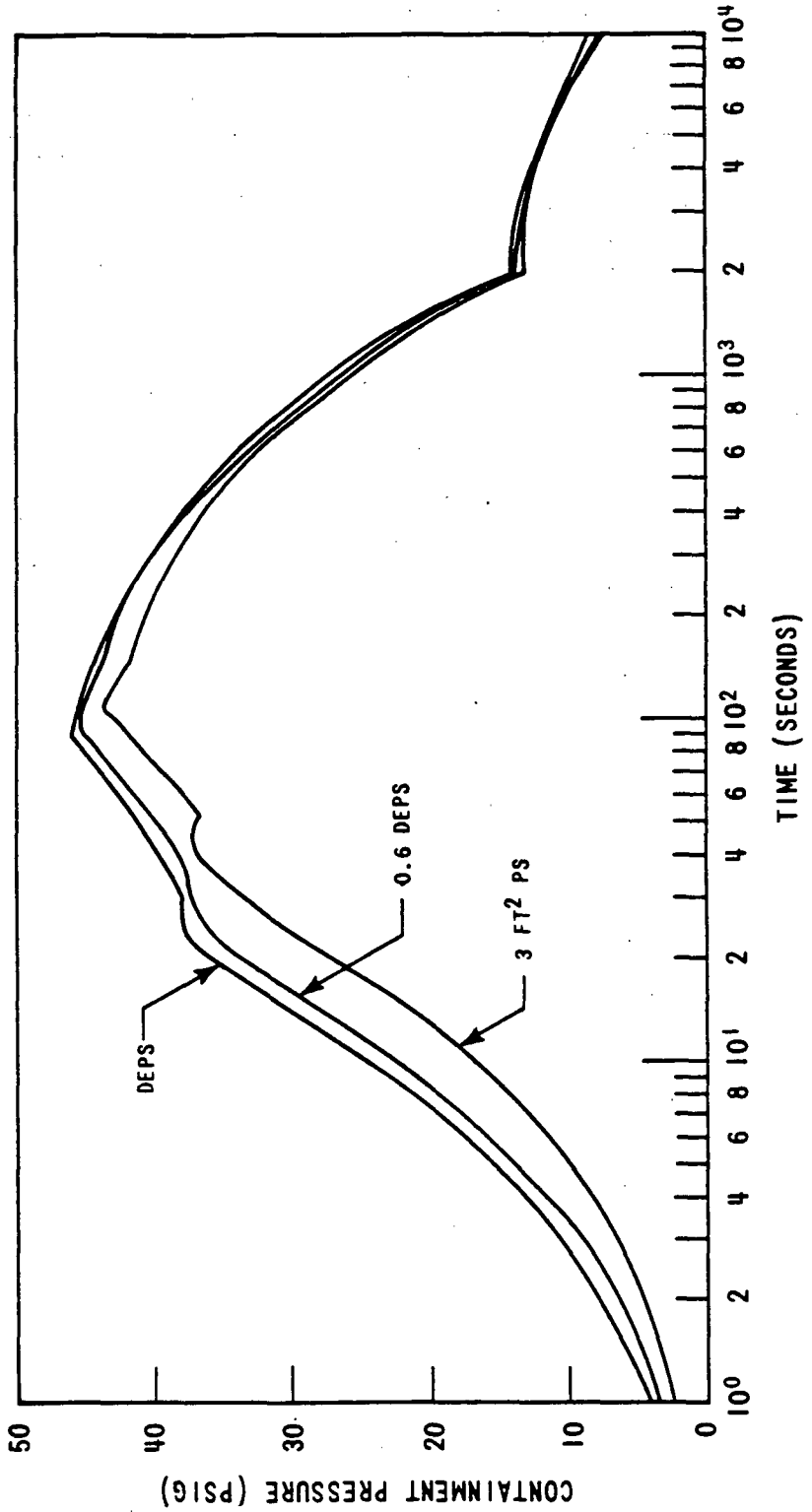
SENSITIVITY TO SAFEGUARDS OPERATION, CONTAINMENT PRESSURE



ZION STATION UFSAR

Figure 15.6-49

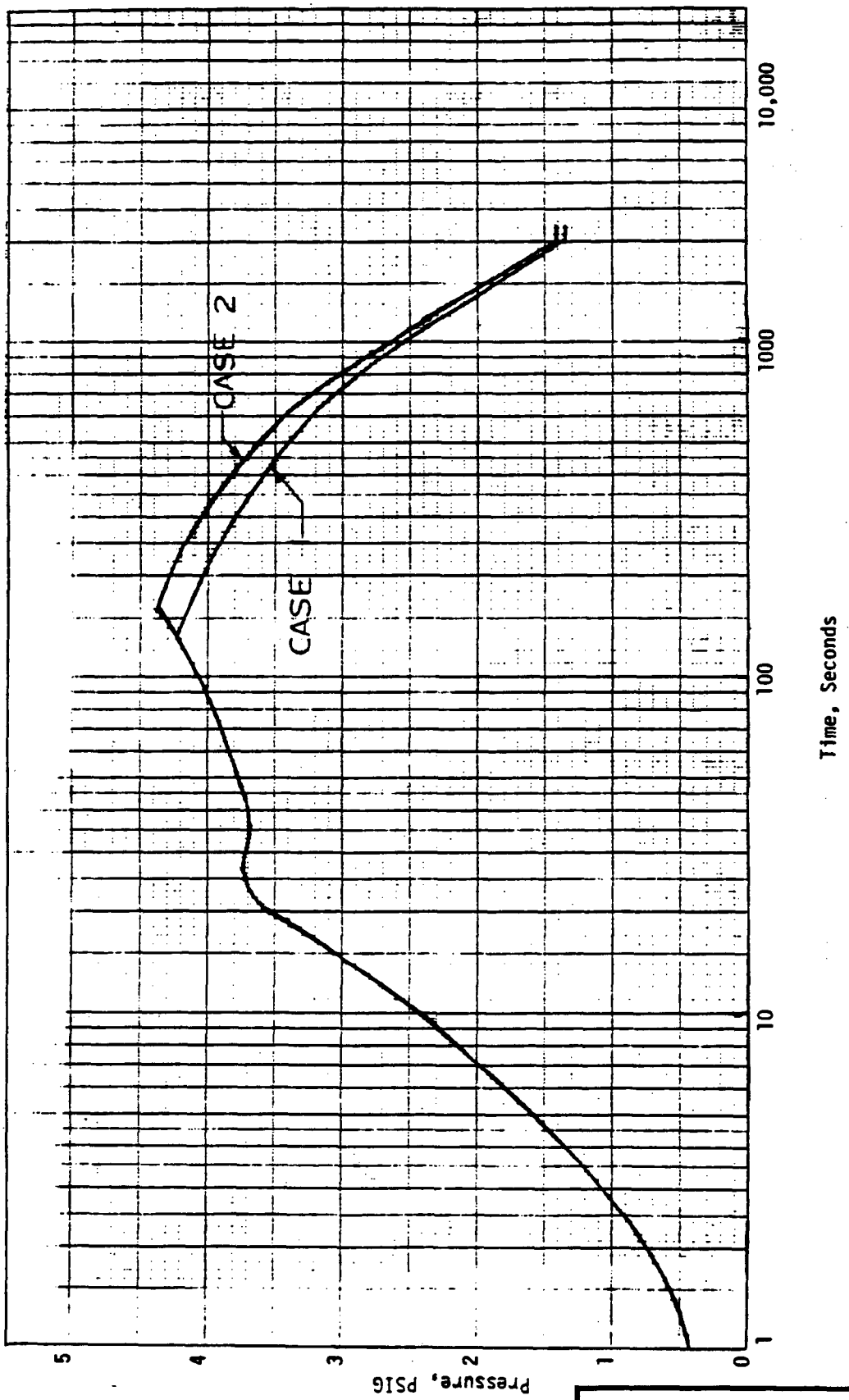
BREAK LOCATION SENSITIVITY, CONTAINMENT PRESSURE



ZION STATION UFSAR

Figure 15.6-50

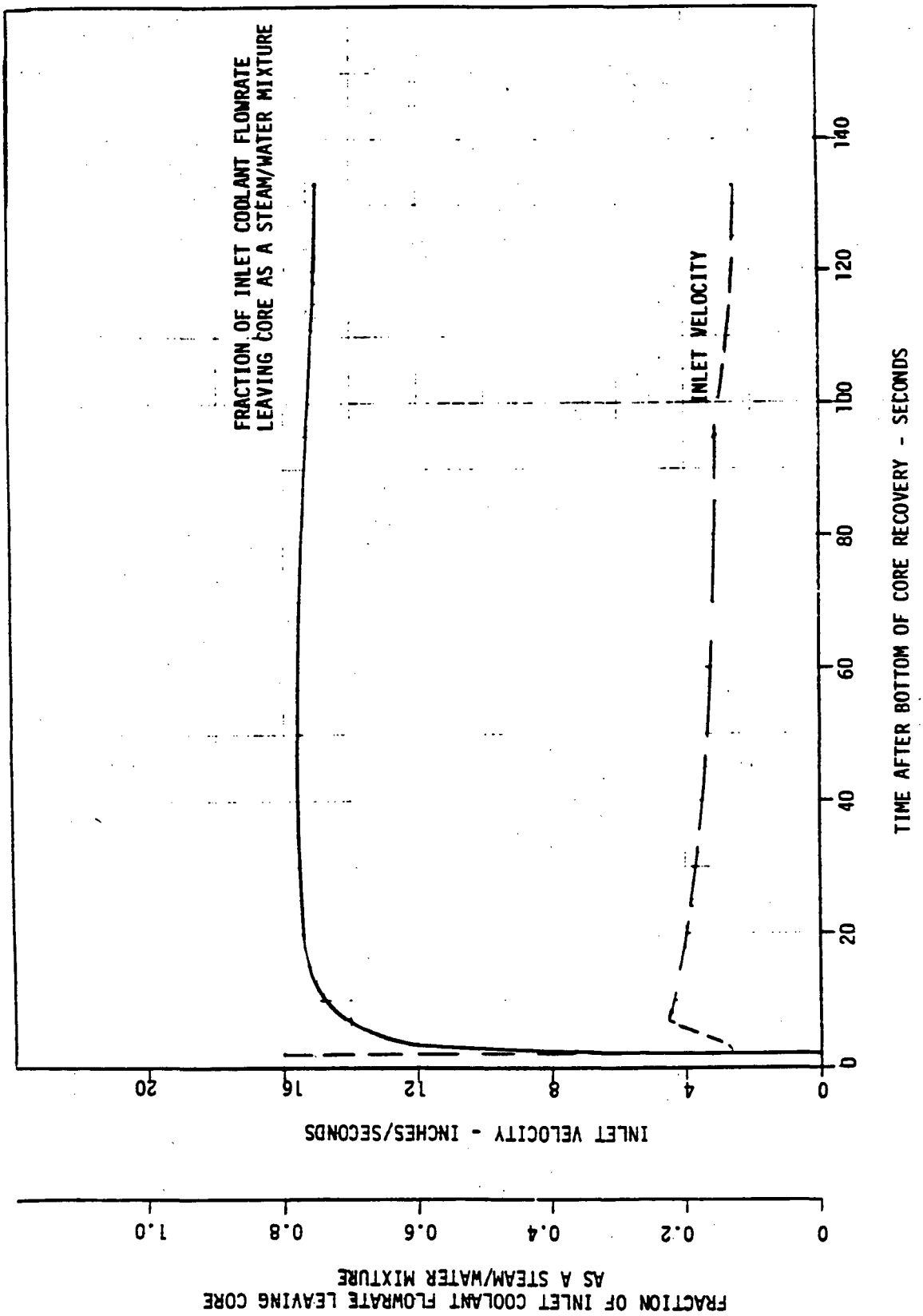
BREAK SIZE SENSITIVITY, CONTAINMENT PRESSURE



ZION STATION UFSAR

Figure 15.6-51

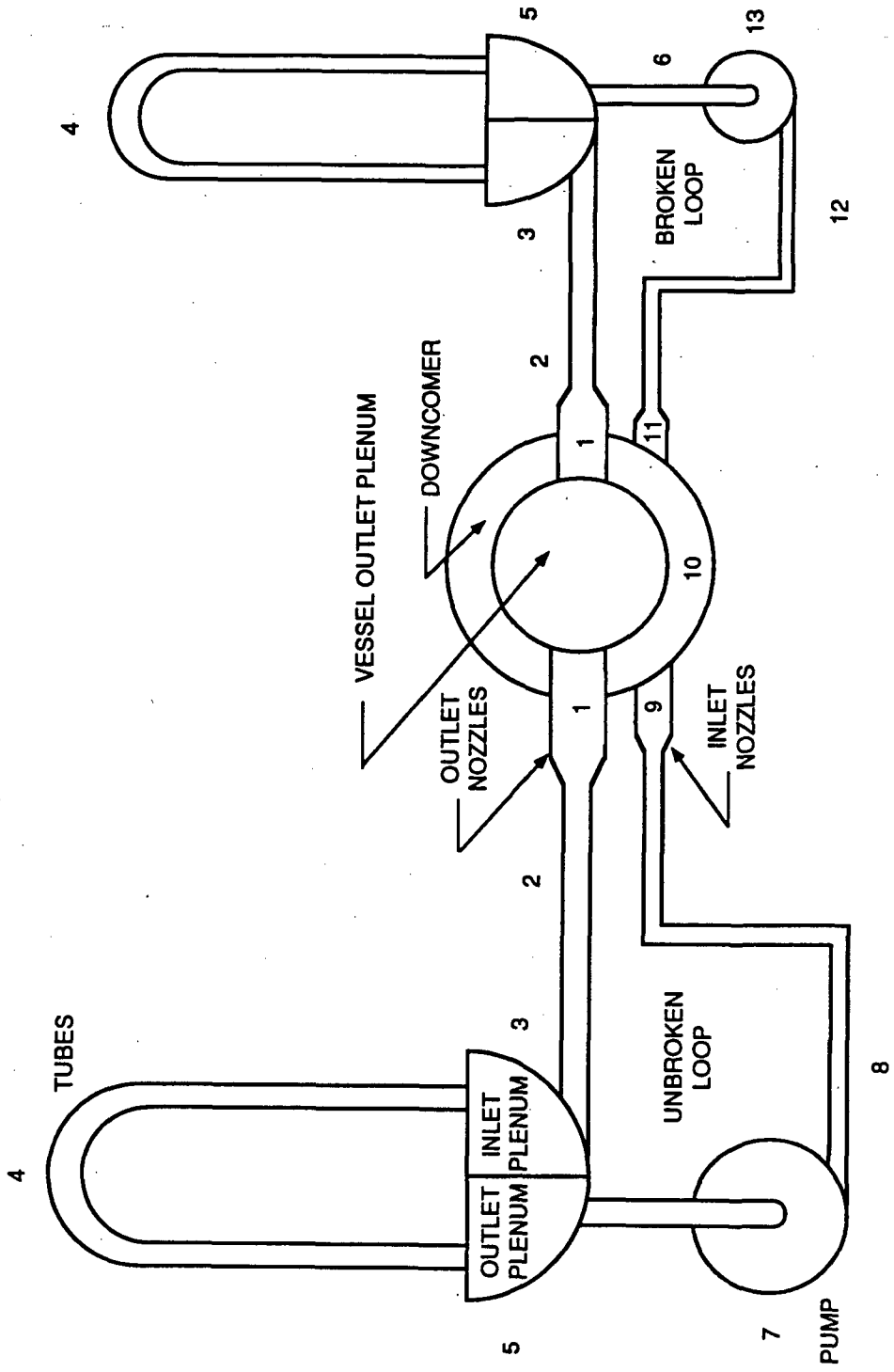
CONTAINMENT PRESSURE TRANSIENT
DOUBLE ENDED PUMP SUCTION BREAK



ZION STATION UFSAR

Figure 15.6-52

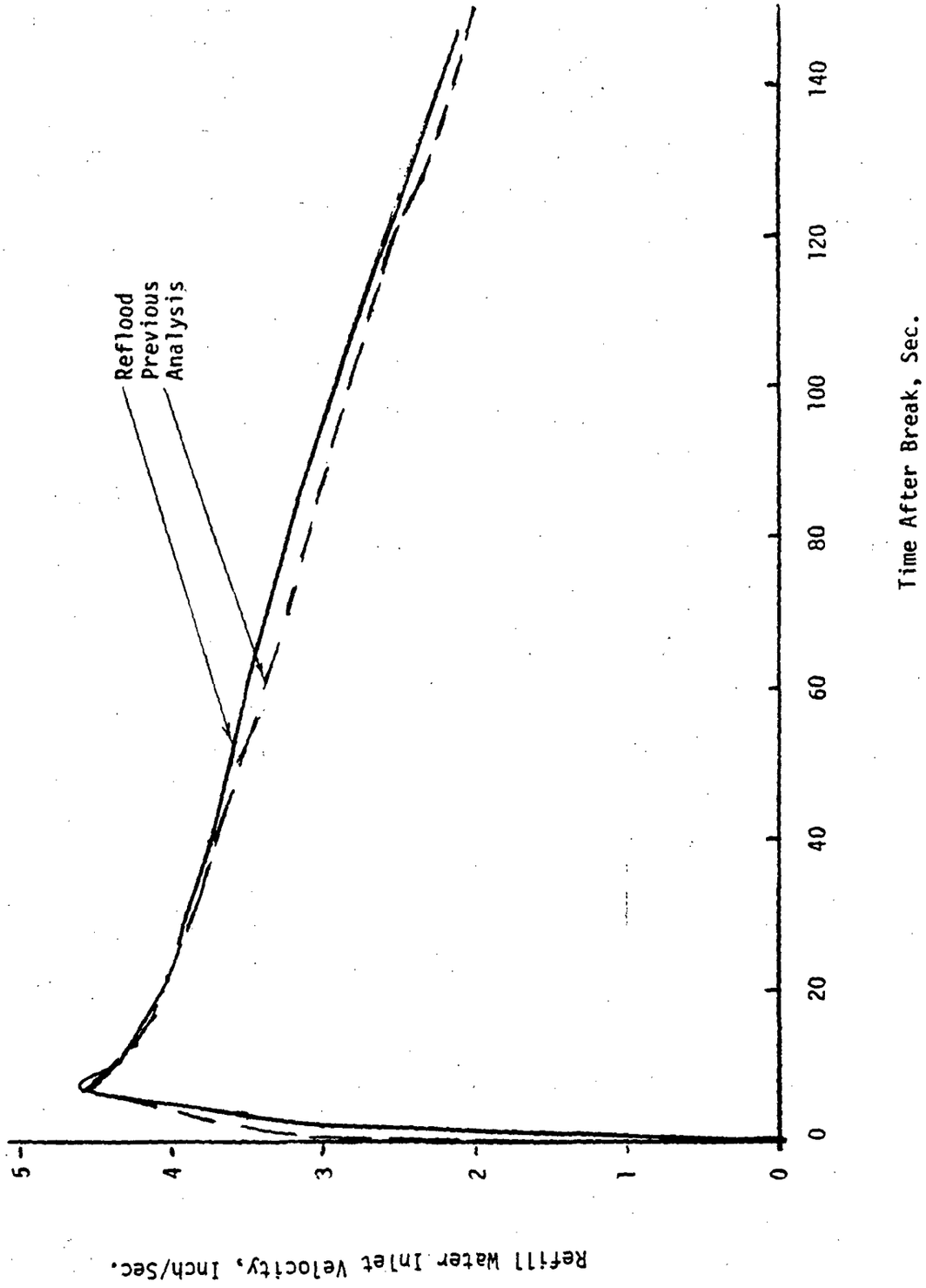
CORE INLET VELOCITY, DOUBLE ENDED PUMP SUCTION BREAK



ZION STATION UFSAR

Figure 15.6-53

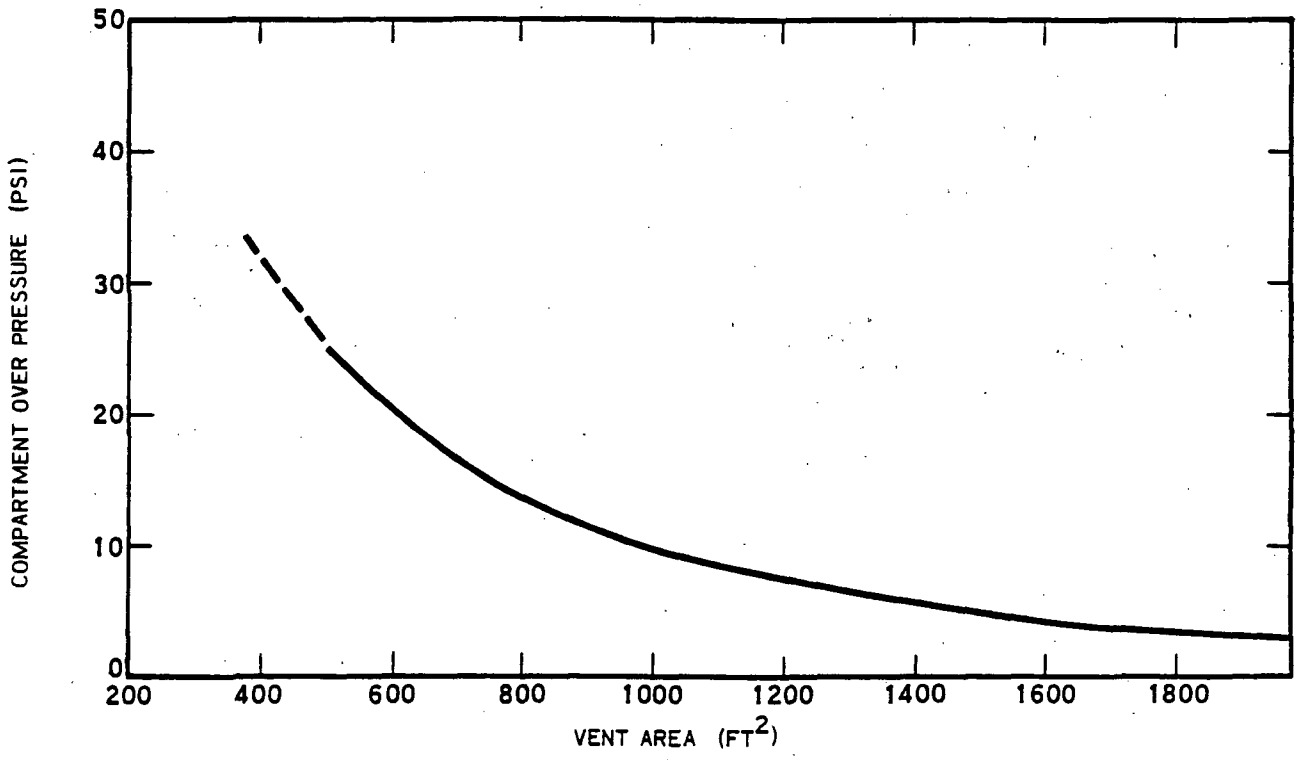
SCHEMATIC OF REFLOOD CODE
19 ELEMENT LOOP MODEL



ZION STATION UFSAR

Figure 15.6-54

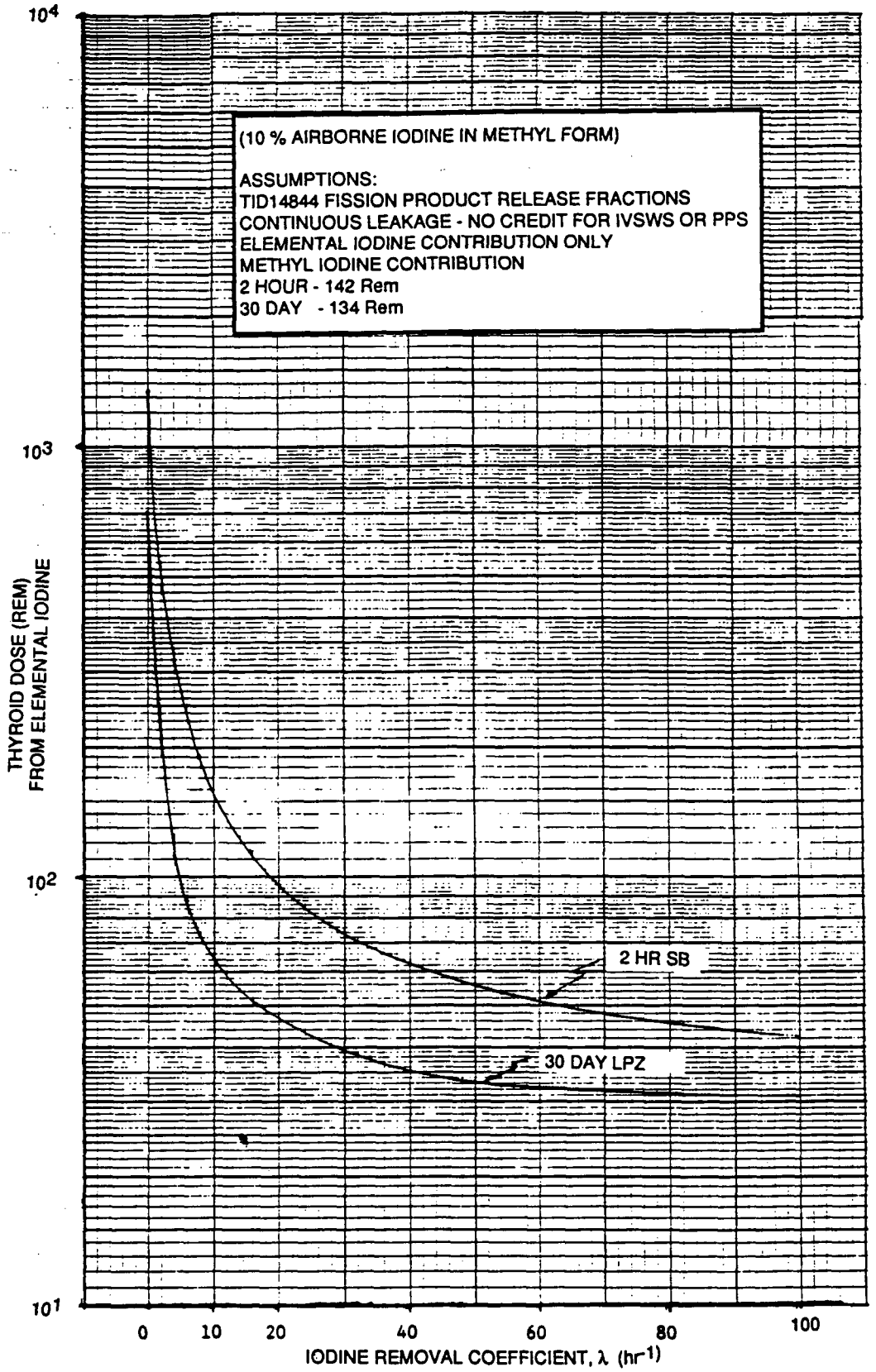
COMPARISON OF PREVIOUS ANALYSIS WITH EQUIVALENT REFLOOD



ZION STATION UFSAR

Figure 15.6-55

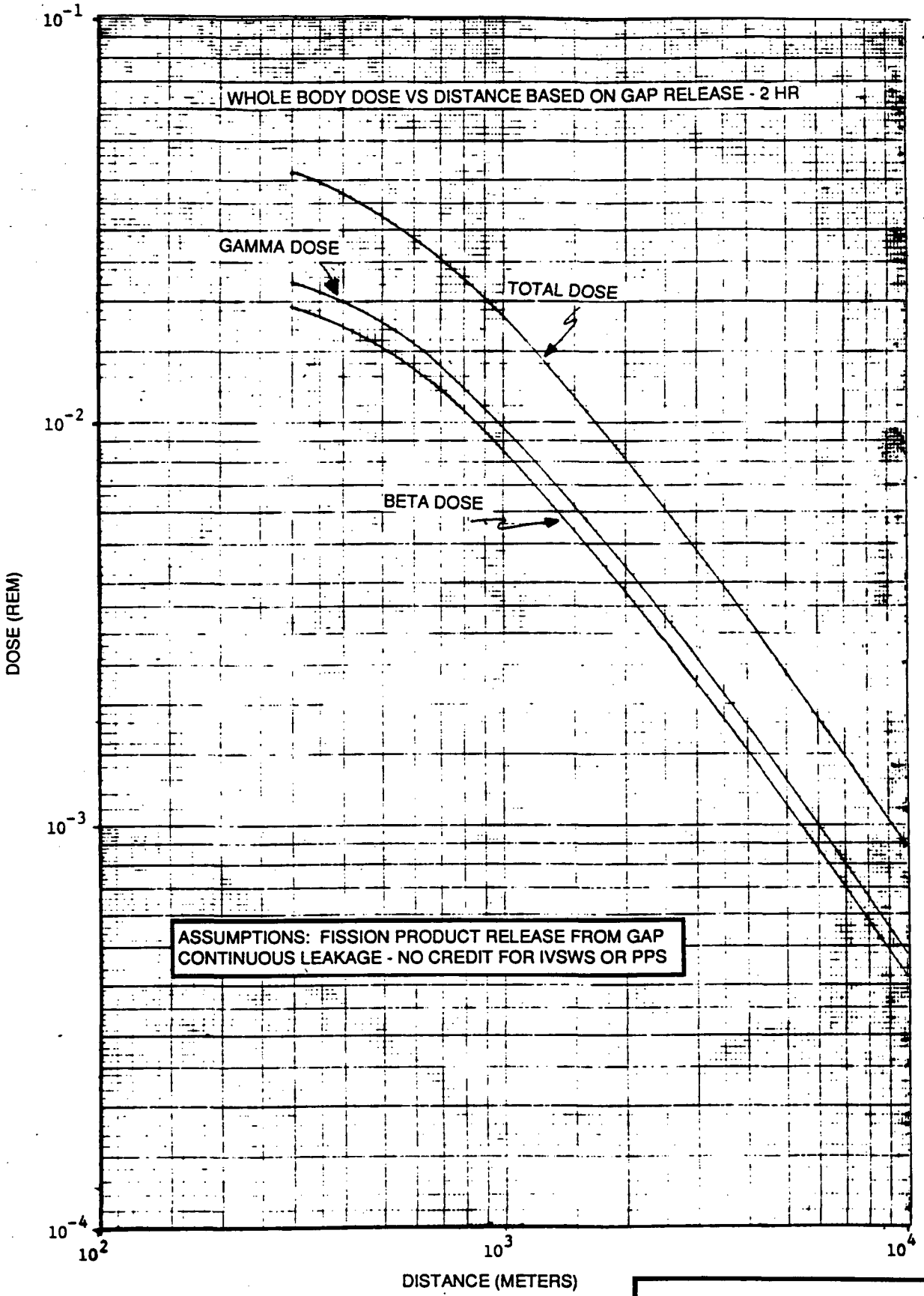
REACTOR LOOP COMPARTMENT PRESSURE VERSUS VENT AREA



ZION STATION UFSAR

Figure 15.6-56

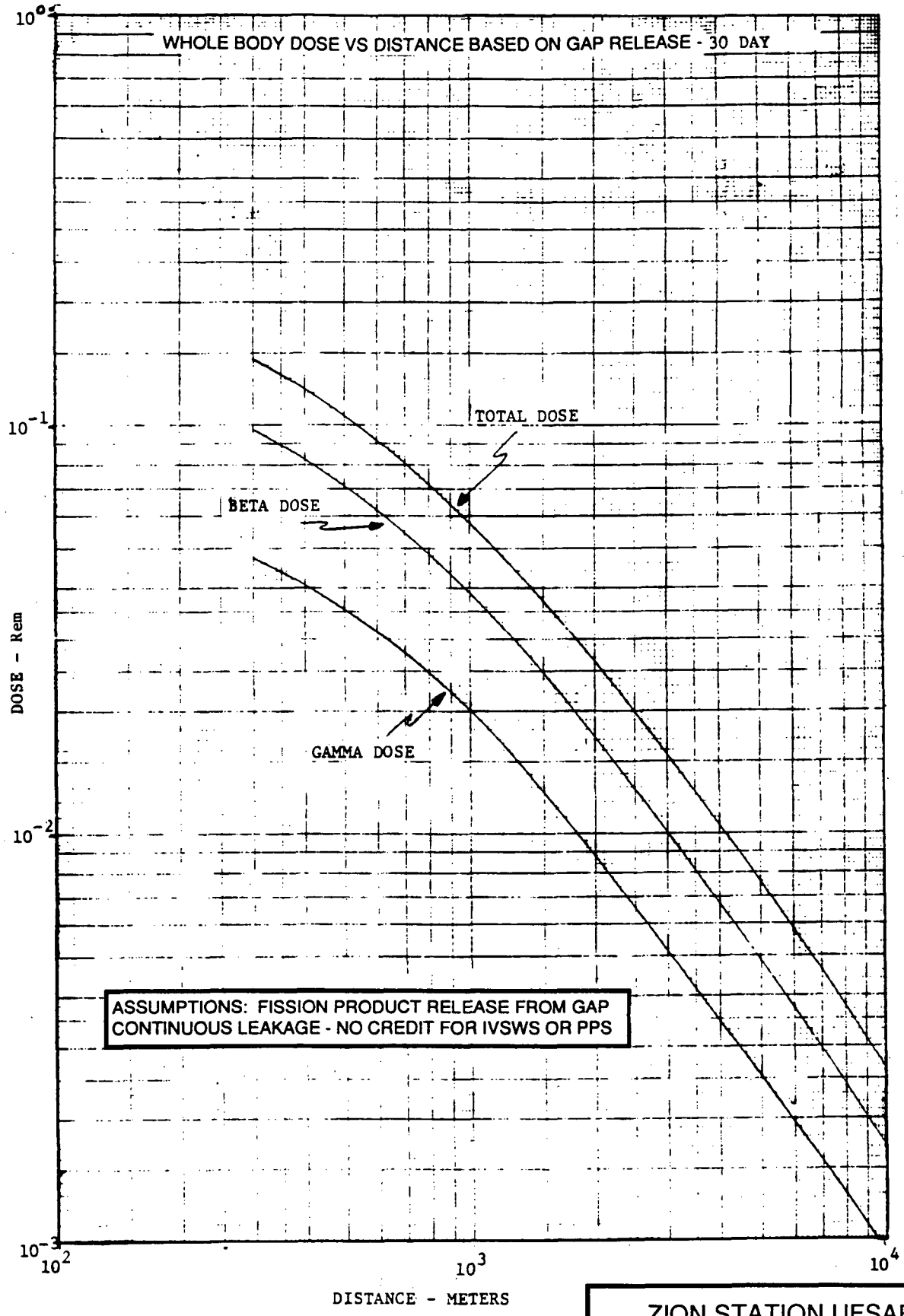
LOSS OF COOLANT ACCIDENT, 2 HOUR SITE BOUNDARY AND 30 DAY LOW POPULATION ZONE DISTANCE THYROID DOSE VS. IODINE REMOVAL COEFFICIENT



ZION STATION UFSAR

Figure 15.6-57

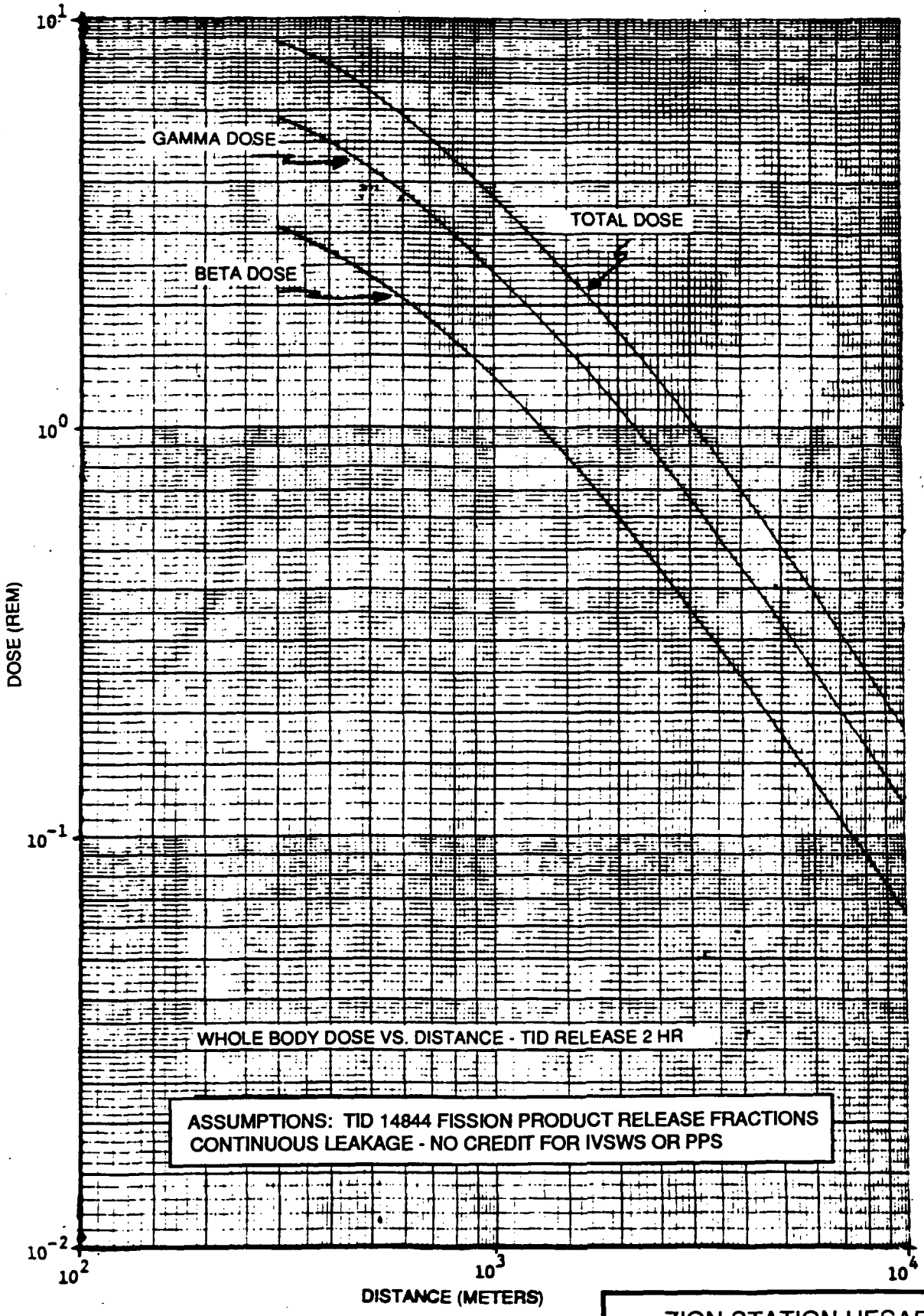
WHOLE BODY DOSE VS. DISTANCE BASED ON GAP RELEASE - 2 HR



ZION STATION UFSAR

Figure 15.6-58

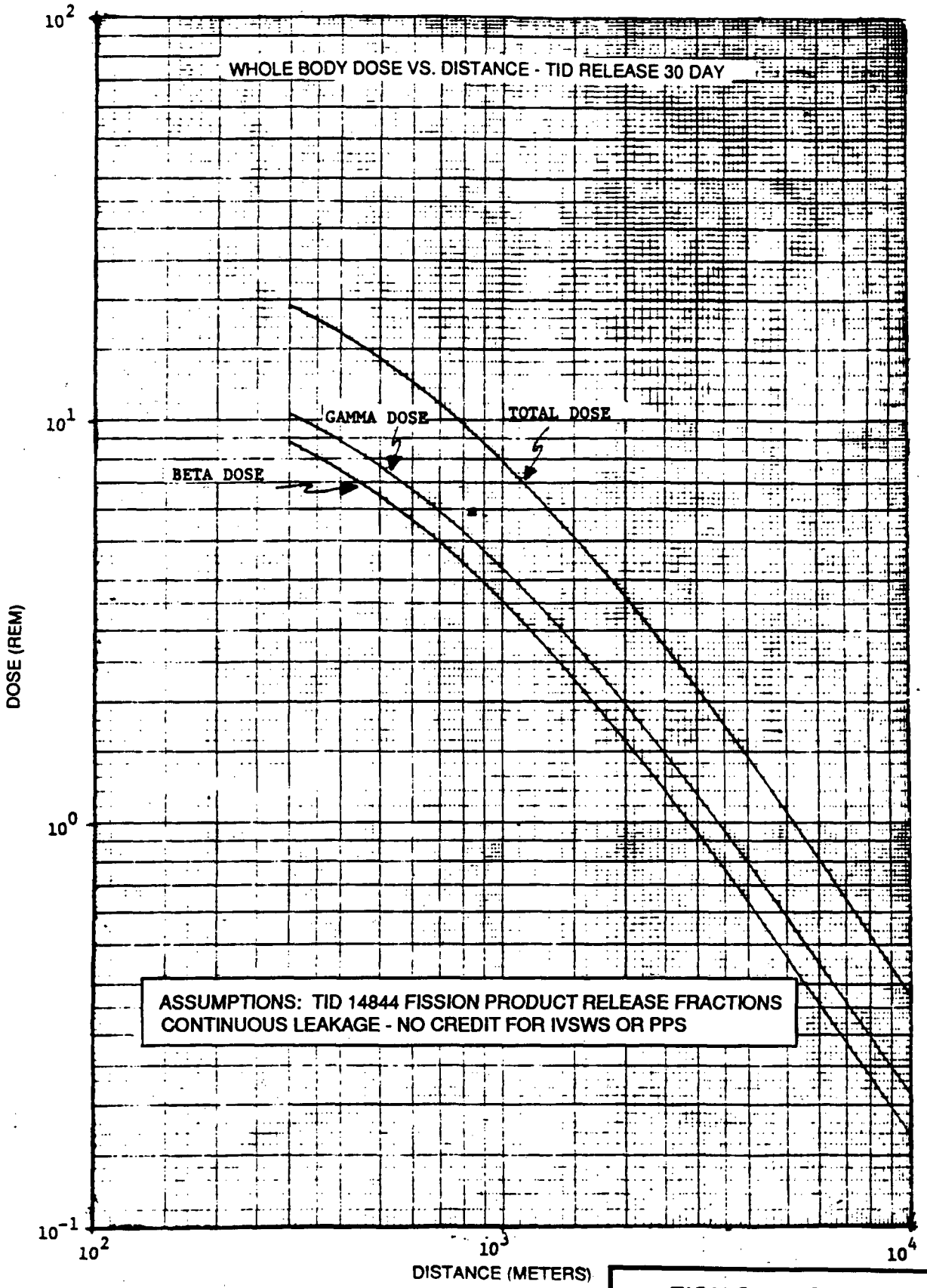
WHOLE BODY DOSE VS. DISTANCE BASED ON GAP RELEASE - 30 DAY



ZION STATION UFSAR

Figure 15.6-59

WHOLE BODY DOSE VS. DISTANCE
TID RELEASE - 2 HR

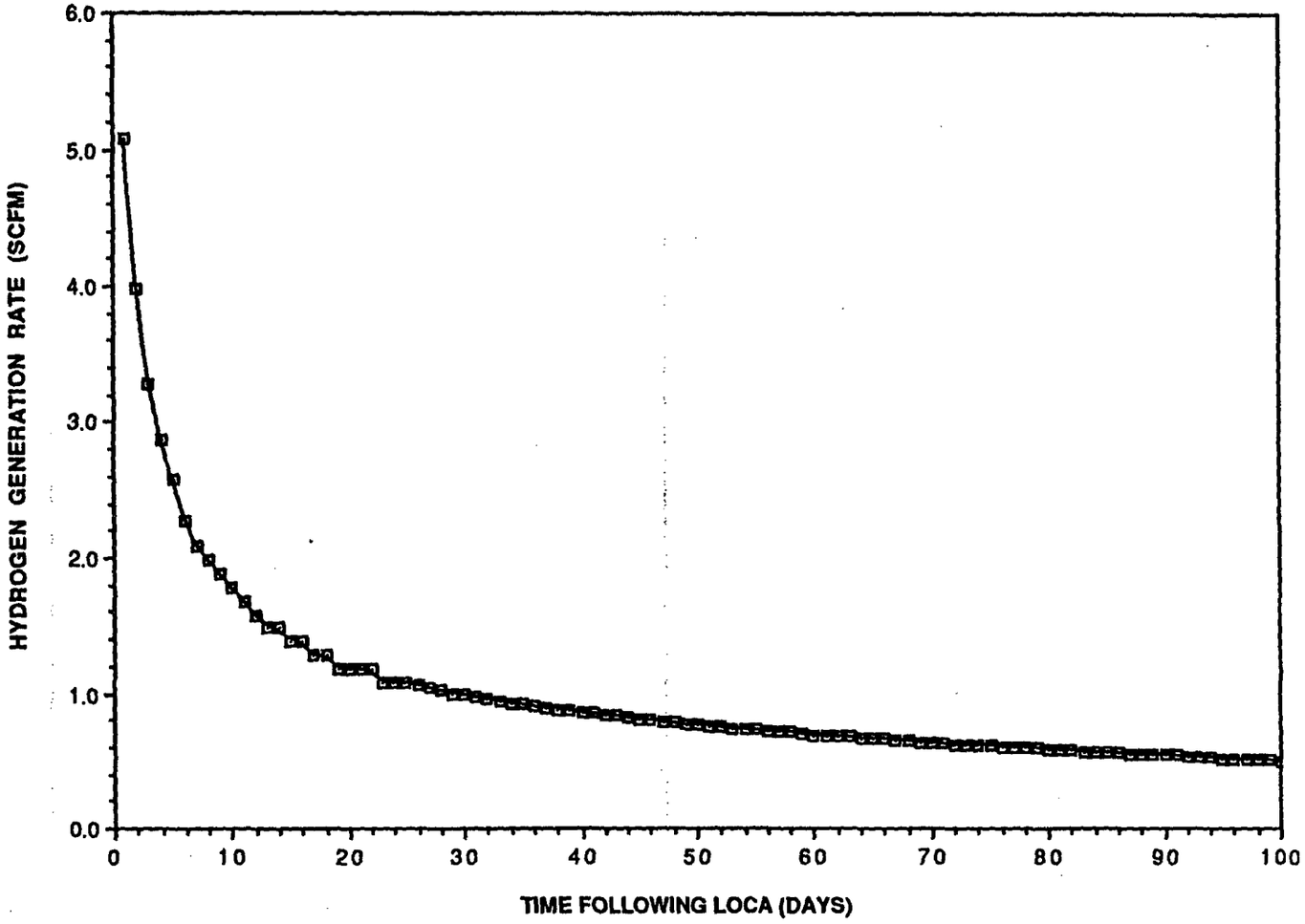


ZION STATION UFSAR

Figure 15.6-60

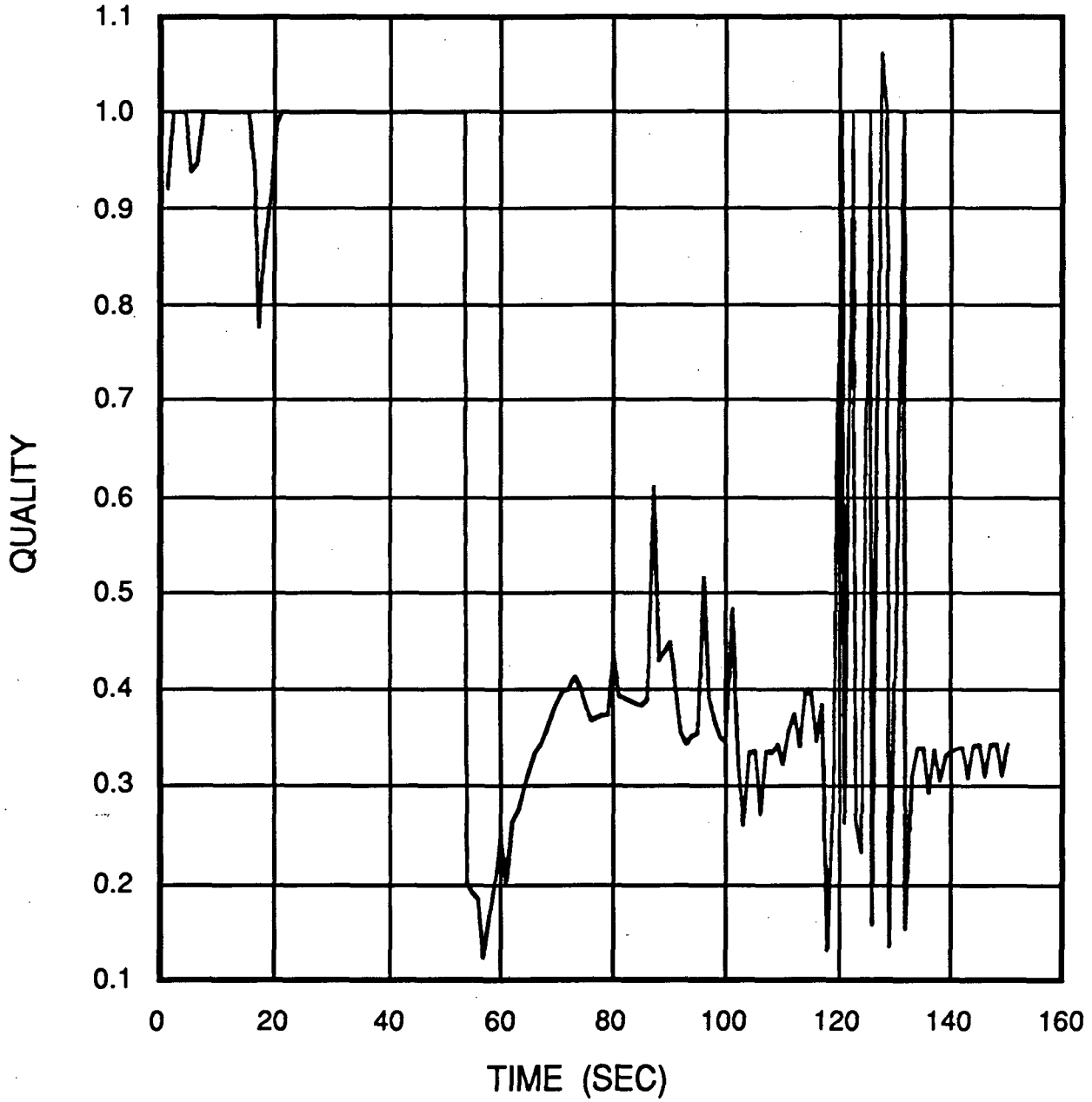
WHOLE BODY DOSE VS. DISTANCE
TID RELEASE - 30 DAY

HYDROGEN GENERATION RATE IN CONTAINMENT FOLLOWING A LOCA



REVISED FIGURE 15.6-61

JULY 1995

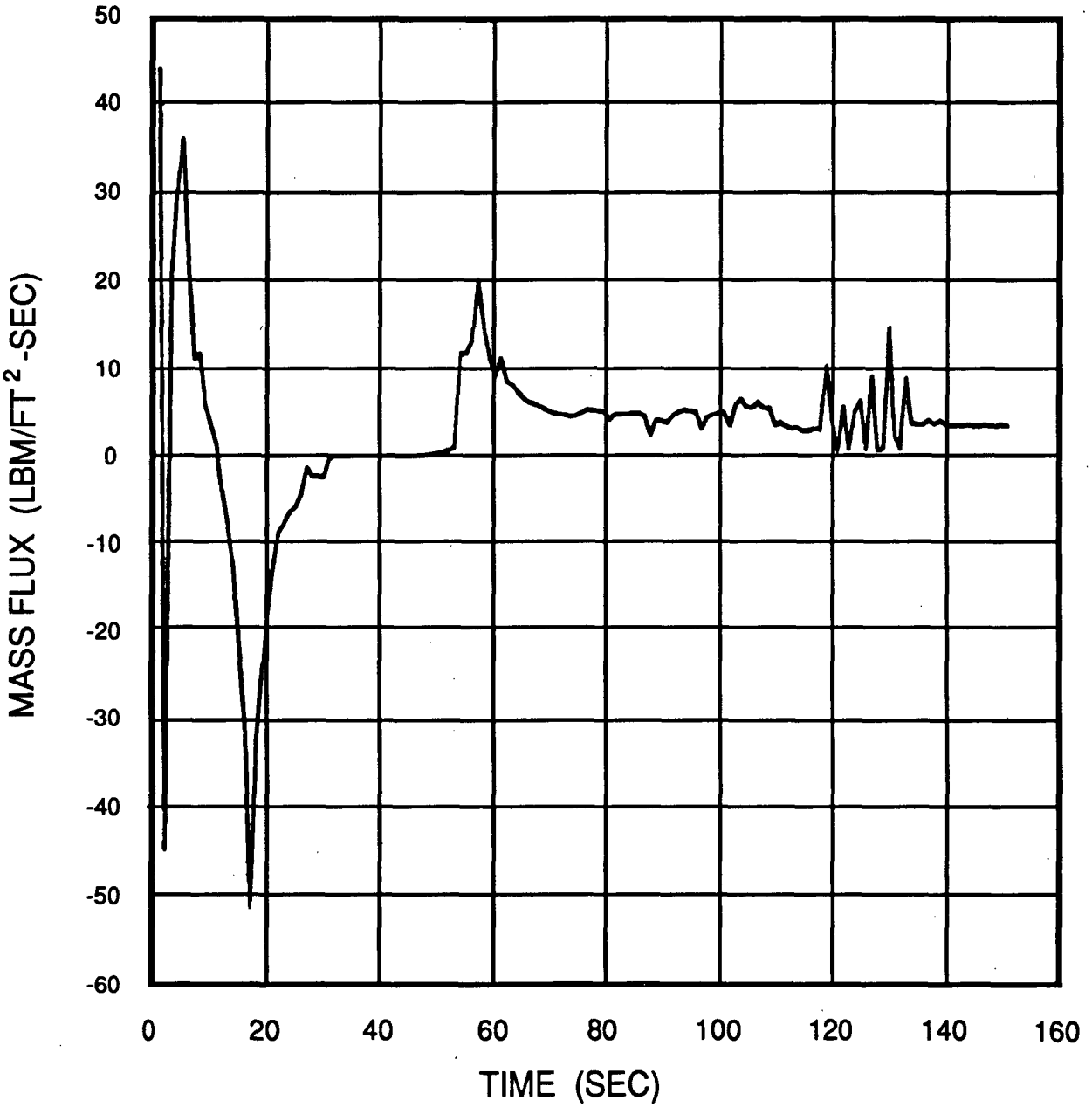


ZION STATION UFSAR

Figure 15.6-62

FLUID QUALITY
DECLG (CD=0.6)

JULY 1993

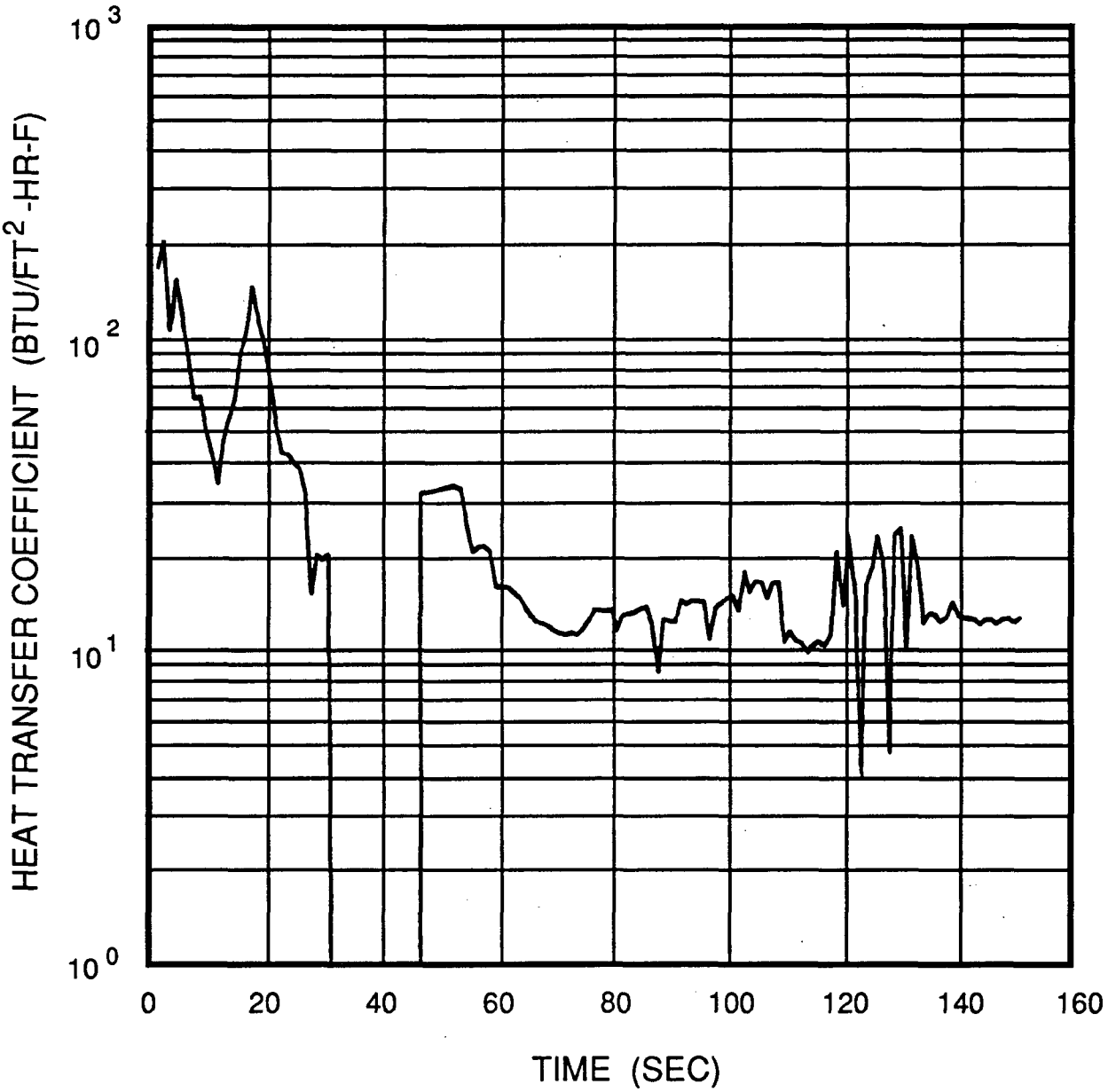


ZION STATION UFSAR

Figure 15.6-63

MASS VELOCITY
DECLG (CD=0.6)

JULY 1993

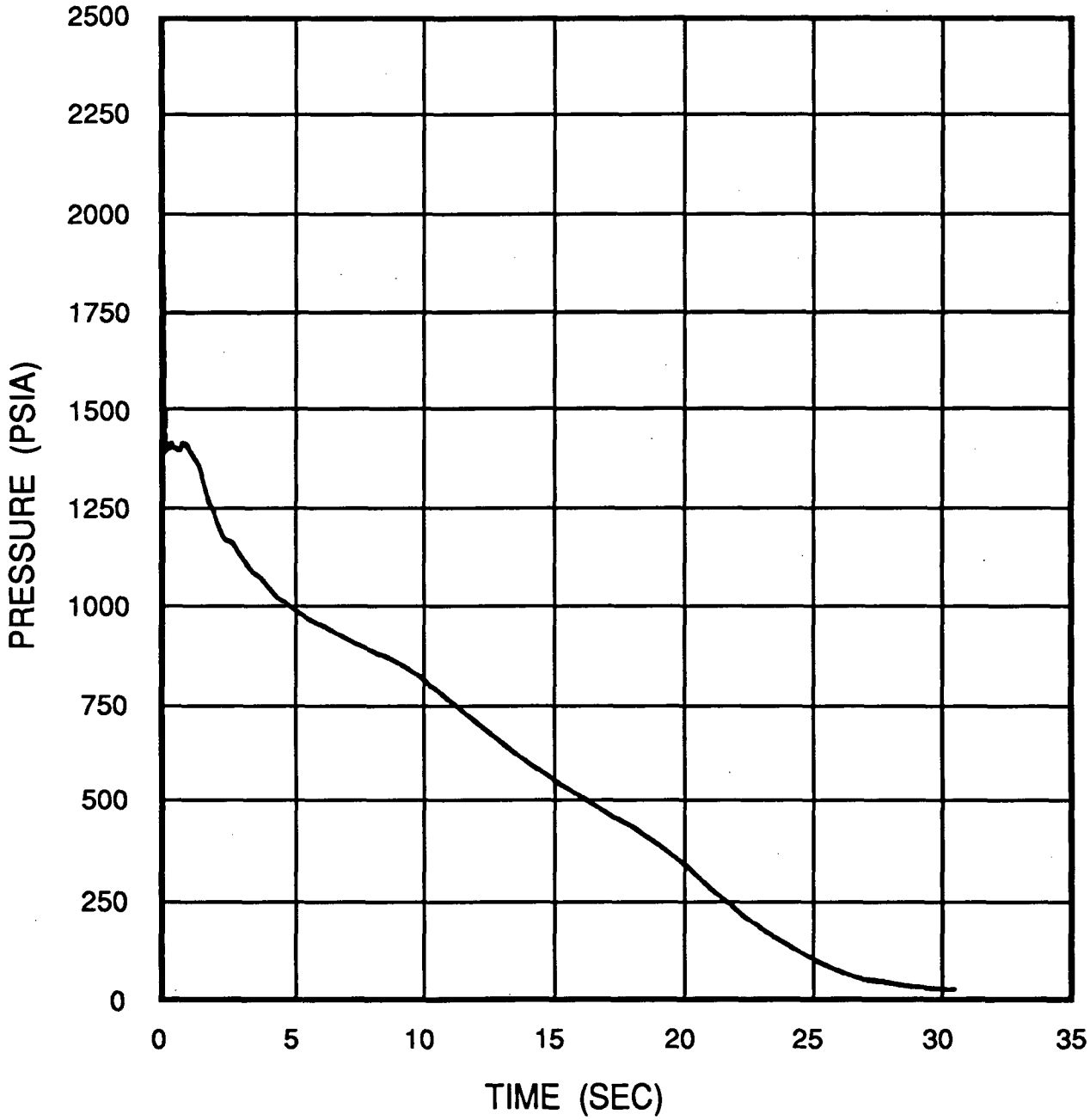


ZION STATION UFSAR

Figure 15.6-64

HEAT TRANSFER COEFFICIENT
DECLG (CD=0.6)

JULY 1993

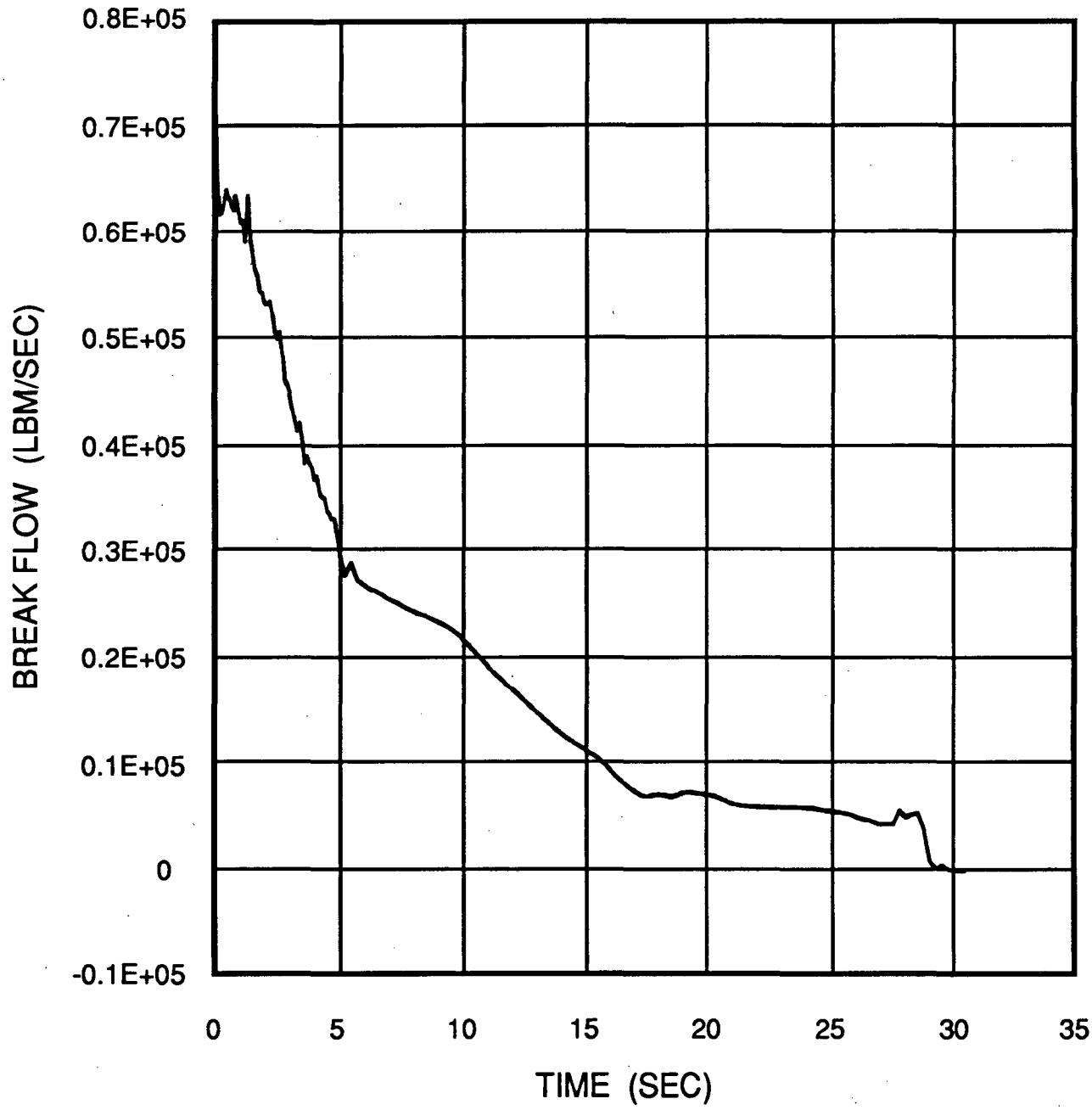


ZION STATION UFSAR

Figure 15.6-65

CORE PRESSURE
DECLG (CD=0.6)

JULY 1993

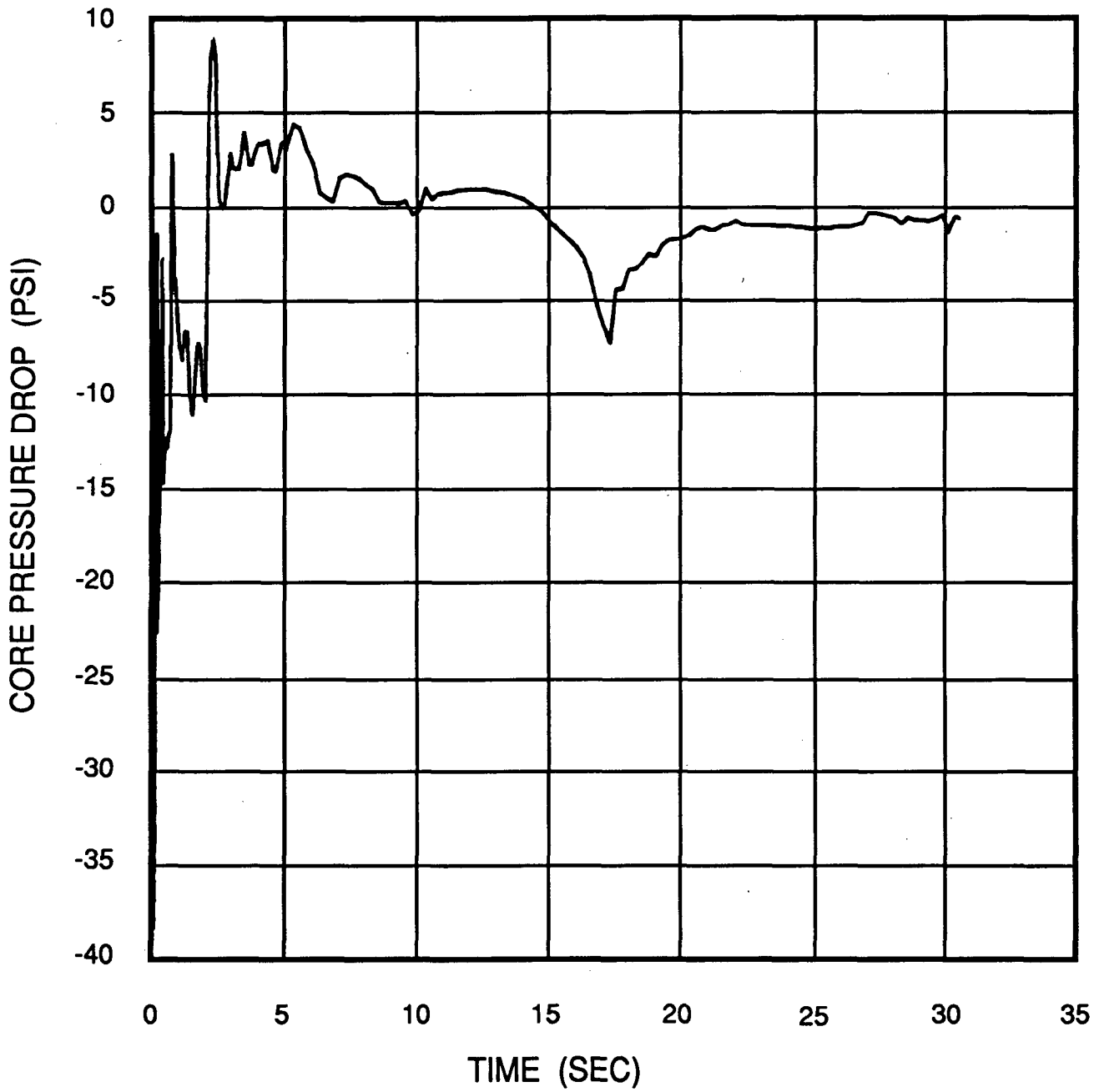


ZION STATION UFSAR

Figure 15.6-66

BREAK FLOW RATE
DECLG (CD=0.6)

JULY 1993

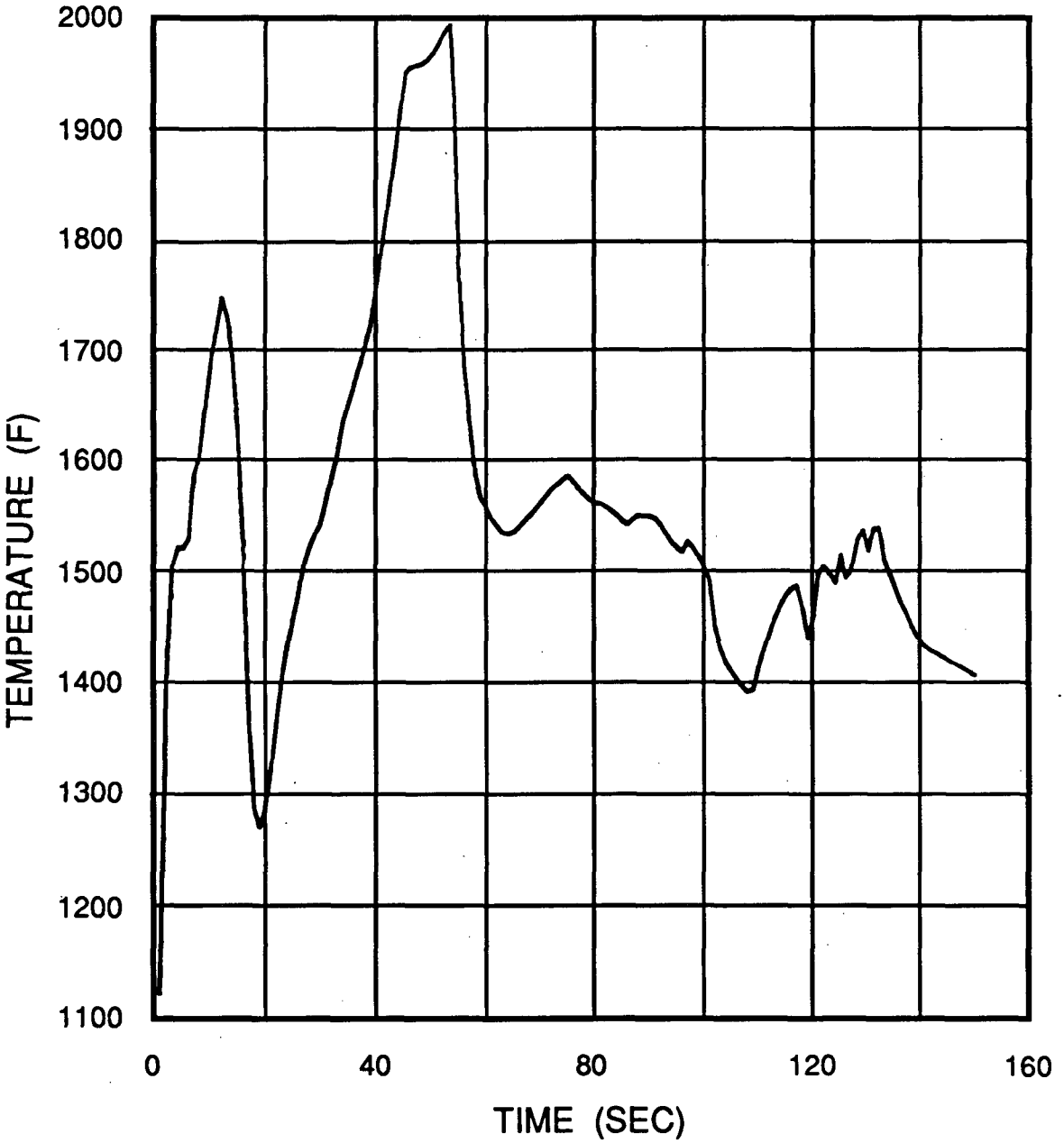


ZION STATION UFSAR

Figure 15.6-67

CORE PRESSURE DROP
DECLG (CD=0.6)

JULY 1993

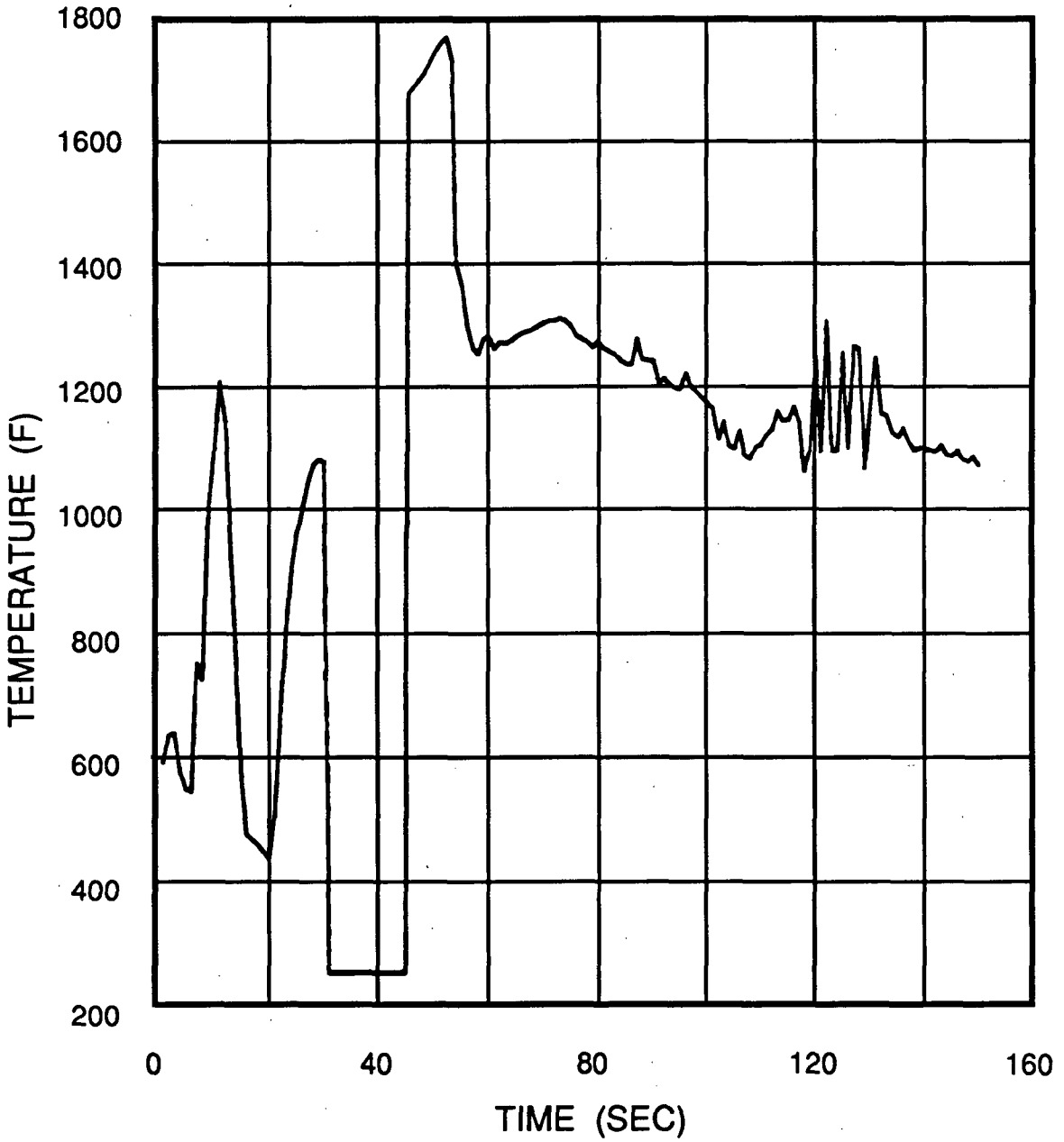


ZION STATION UFSAR

Figure 15.6-68

PEAK CLADDING TEMPERATURE
DECLG (CD=0.6)

JULY 1993

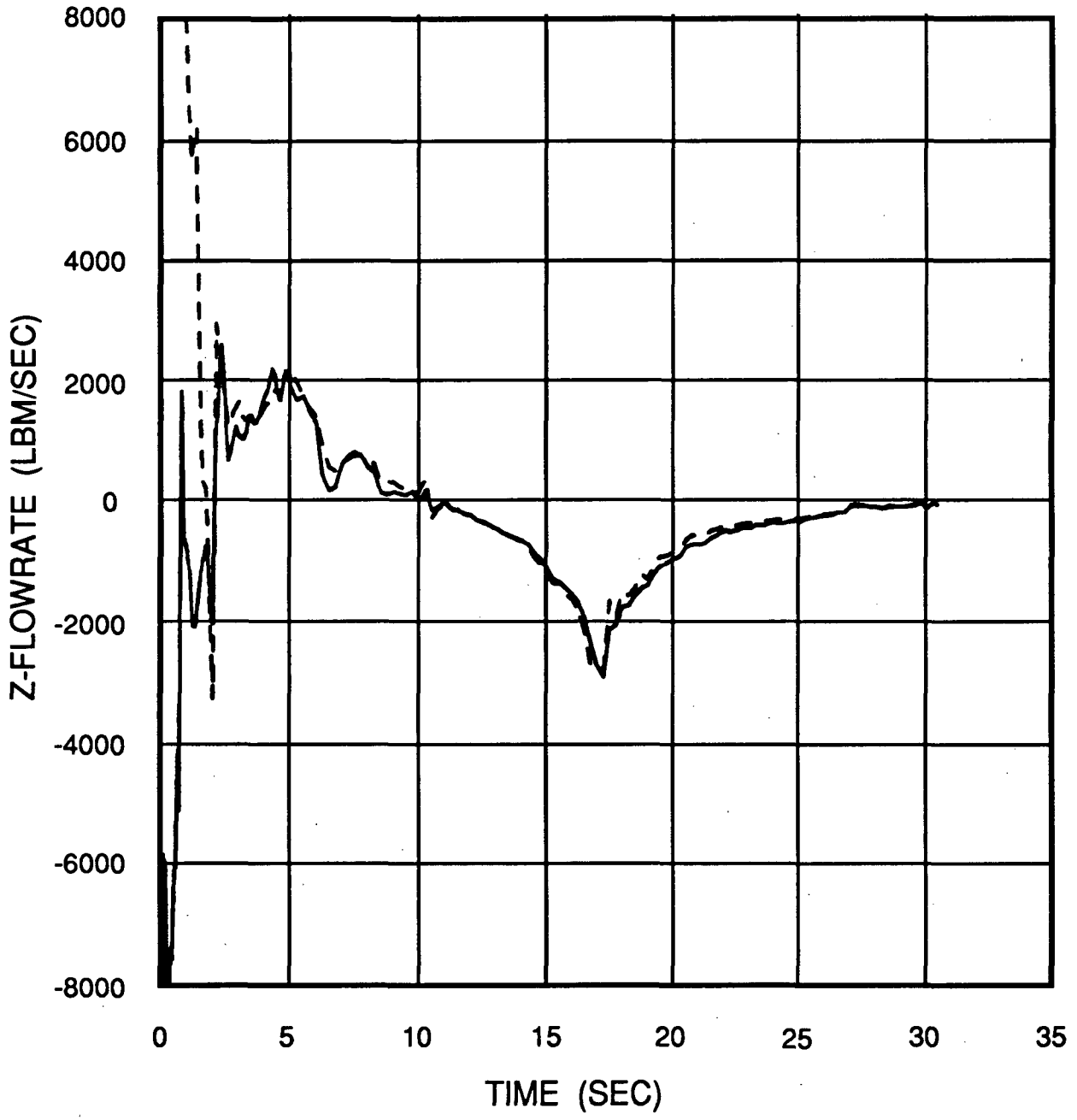


ZION STATION UFSAR

Figure 15.6-69

FLUID TEMPERATURE
DECLG (CD=0.6)

JULY 1993

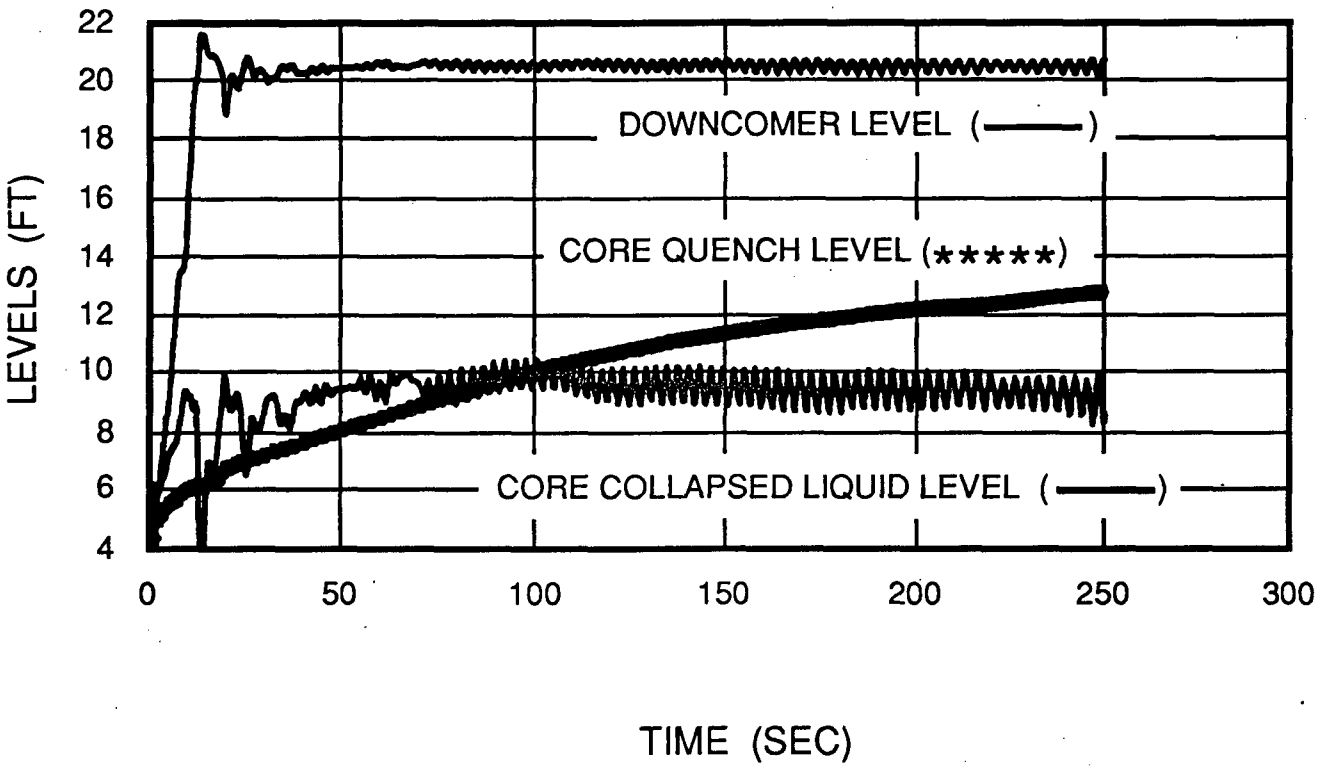


ZION STATION UFSAR

Figure 15.6-70

CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.6)

JULY 1993



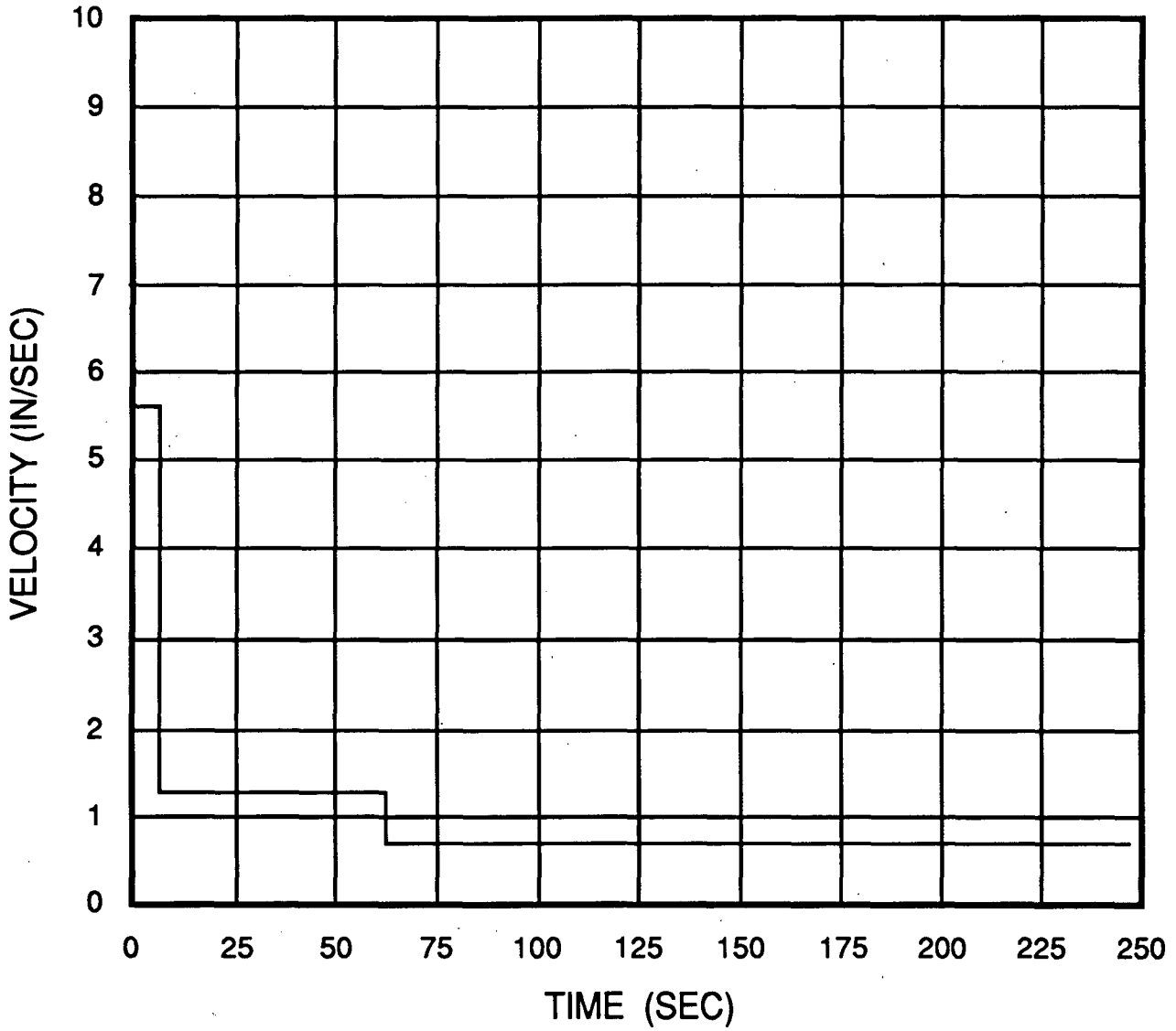
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 45.170 SECONDS FOR CD = 0.6

ZION STATION UFSAR

Figure 15.6-71
REFLOOD TRANSIENT
CORE AND DOWNCOMER LEVELS
DECLG (CD=0.6)

JULY 1993



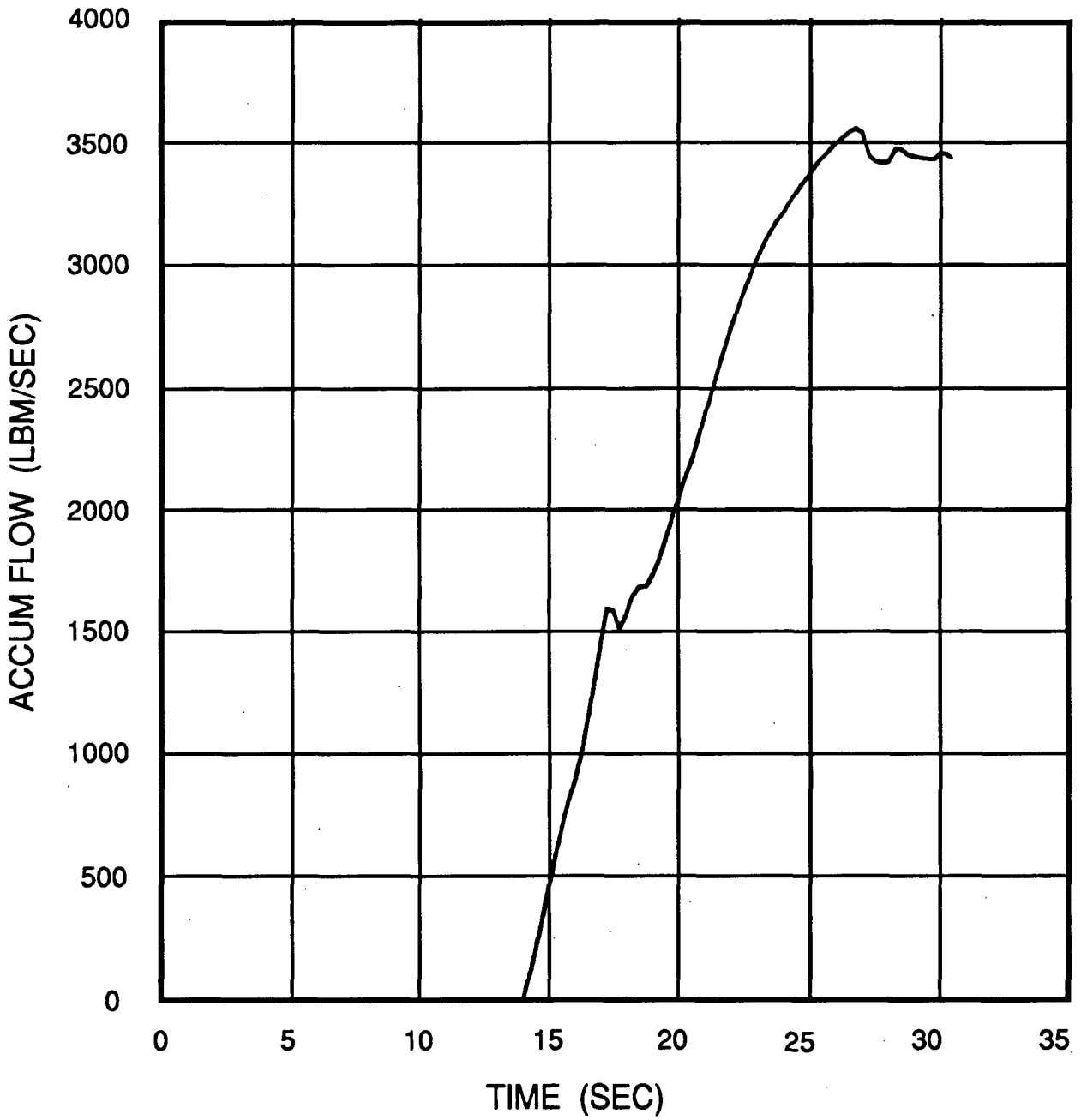
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 45.170 SECONDS FOR CD = 0.6

ZION STATION UFSAR

Figure 15.6-72
REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD=0.6)

JULY 1993

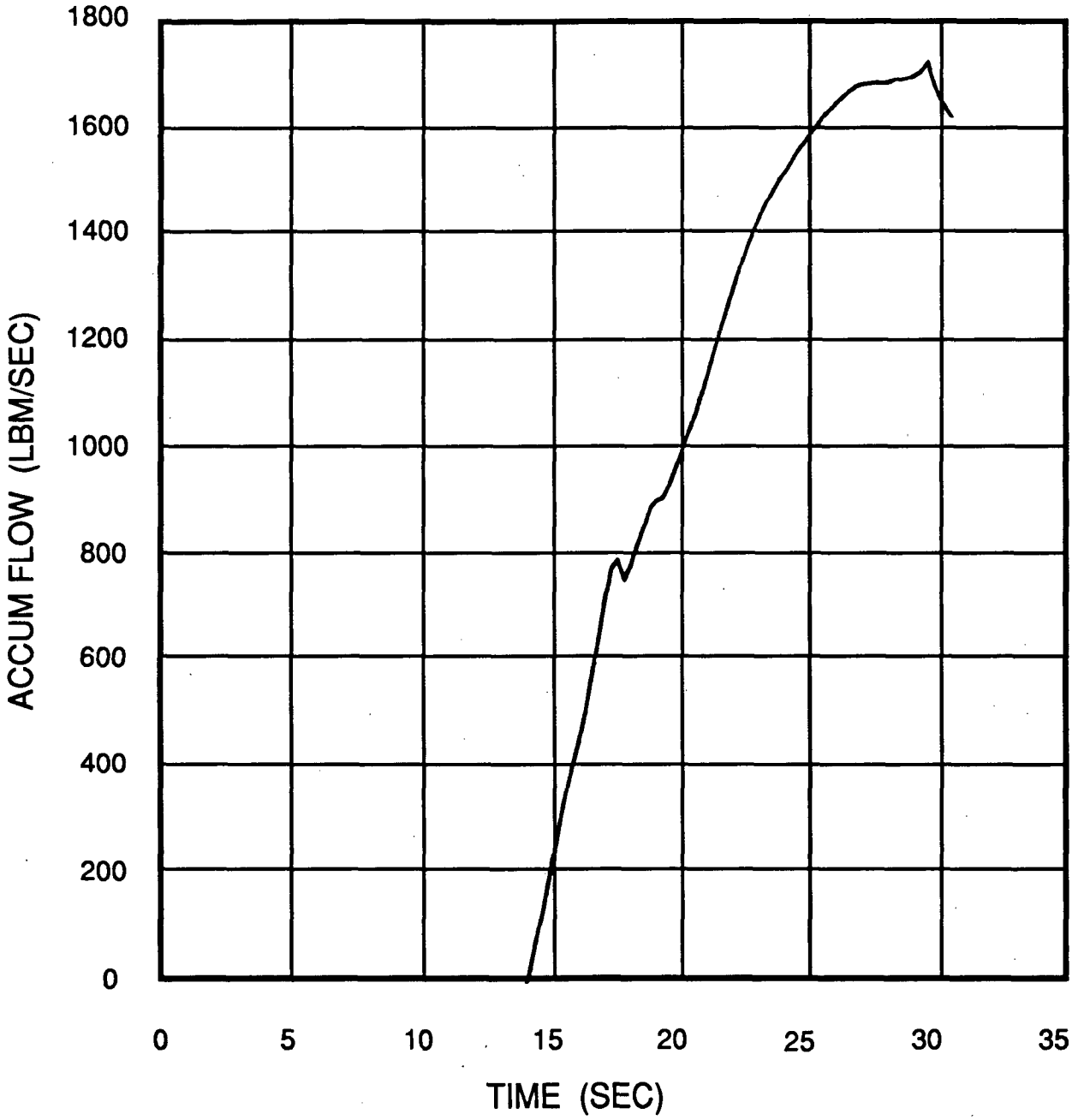


FLOW FROM TWO INTACT LOOP ACCUMULATORS

ZION STATION UFSAR

Figure 15.6-73
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.6)

JULY 1993

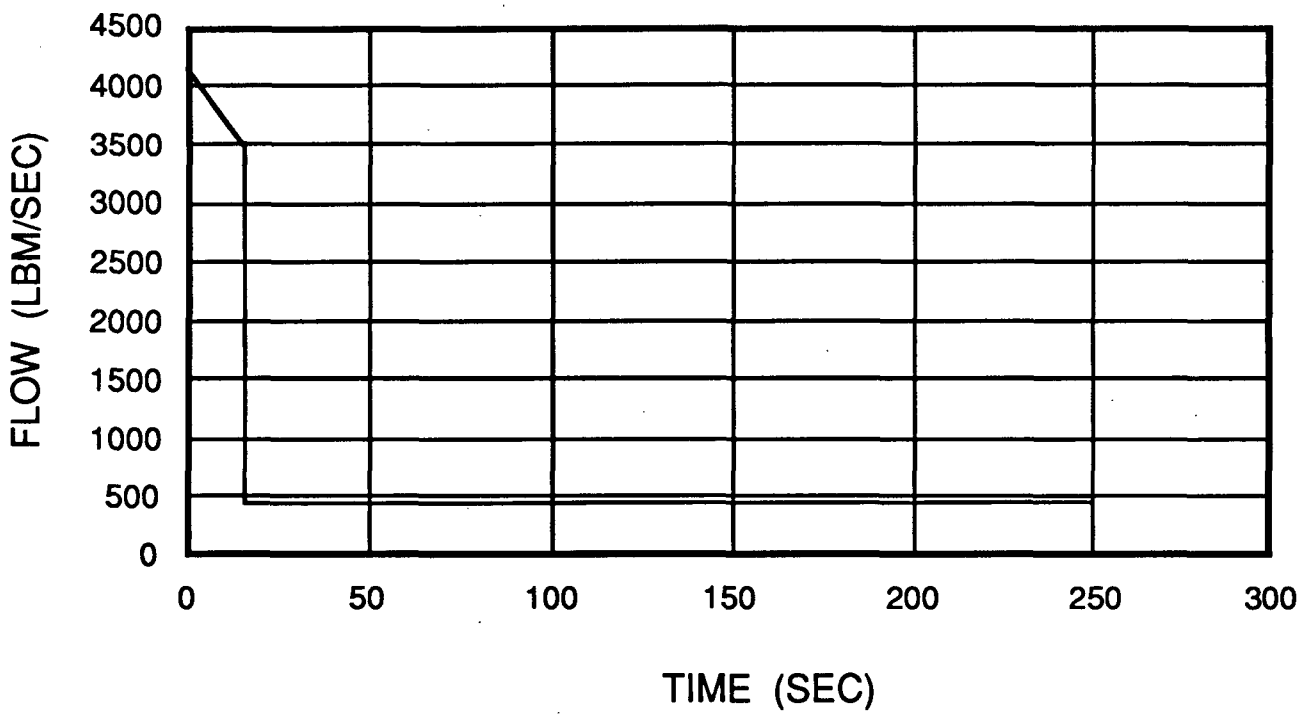


FLOW FROM ONE INTACT LOOP ACCUMULATOR

ZION STATION UFSAR

Figure 15.6-74
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.6)

JULY 1993



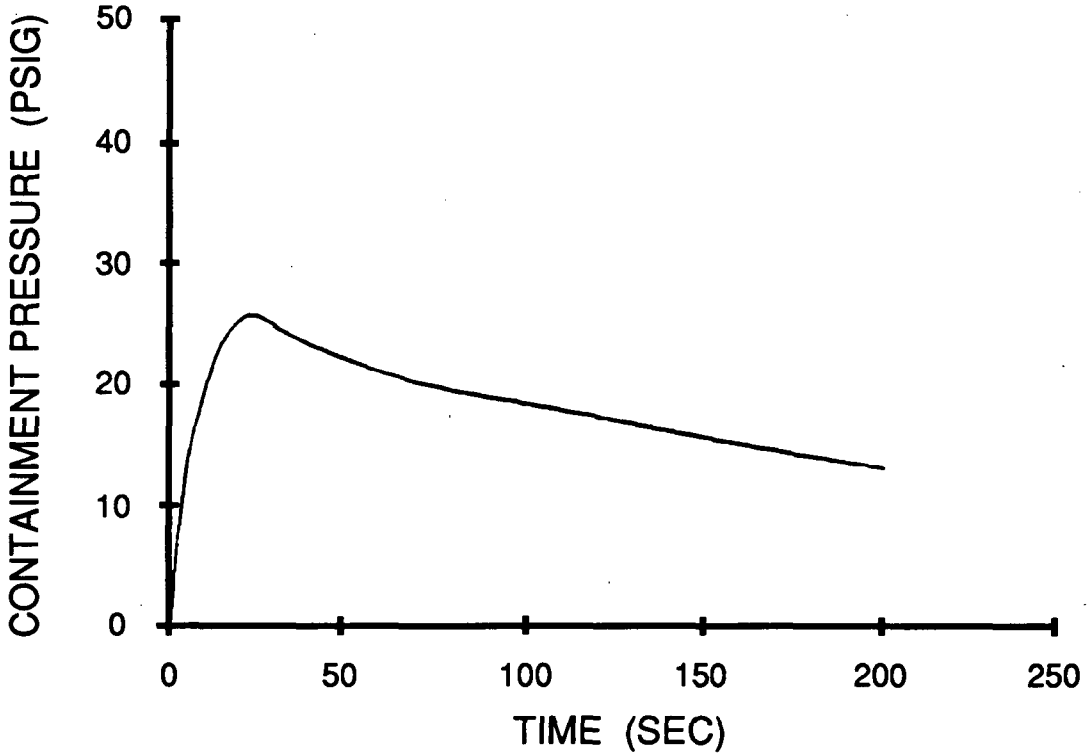
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 45.170 SECONDS FOR CD = 0.6

ZION STATION UFSAR

Figure 15.6-75
PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.6)

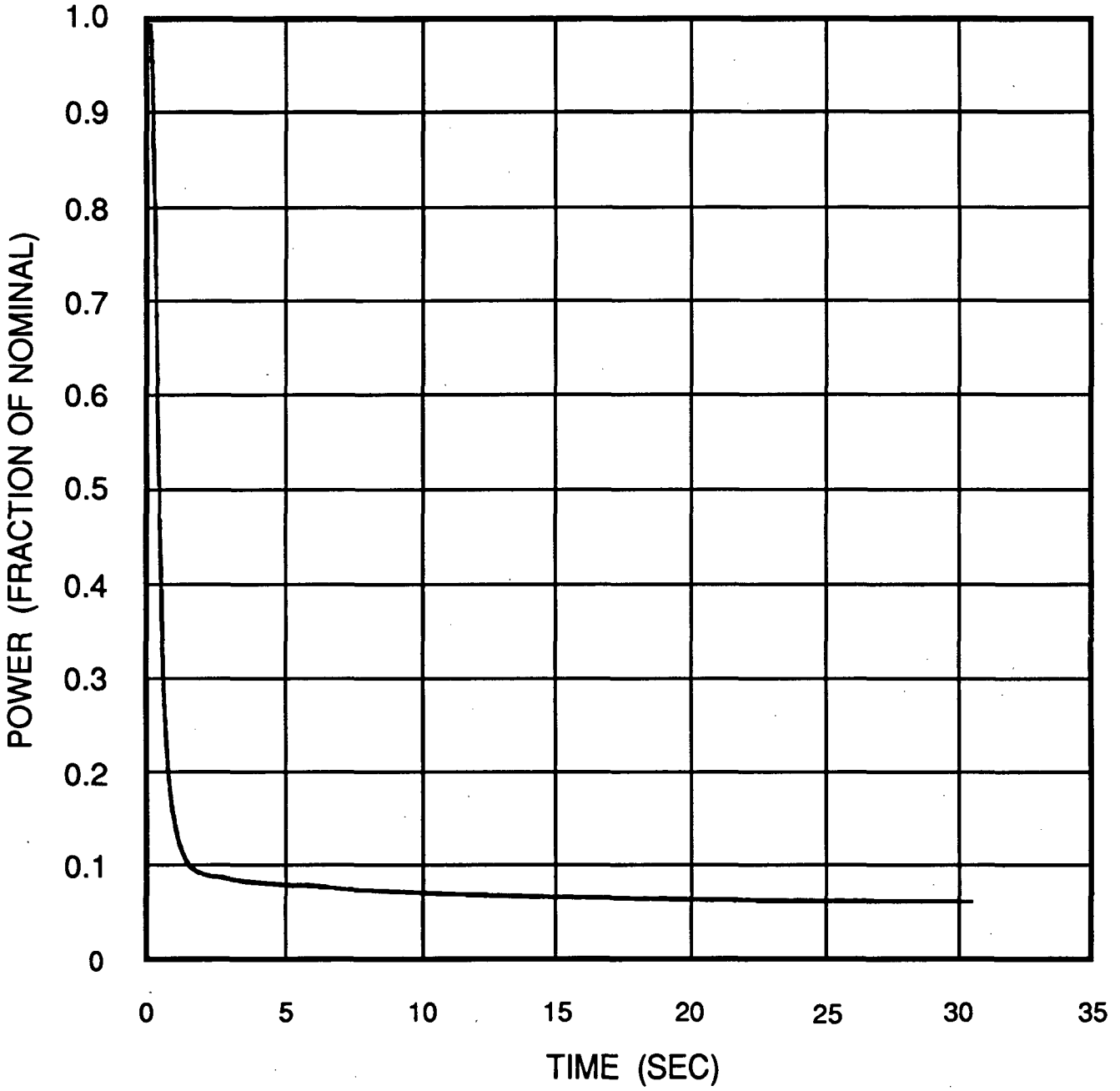
JULY 1993



ZION STATION UFSAR

Figure 15.6-76
CONTAINMENT PRESSURE
DECLG (CD=0.6)

JULY 1993

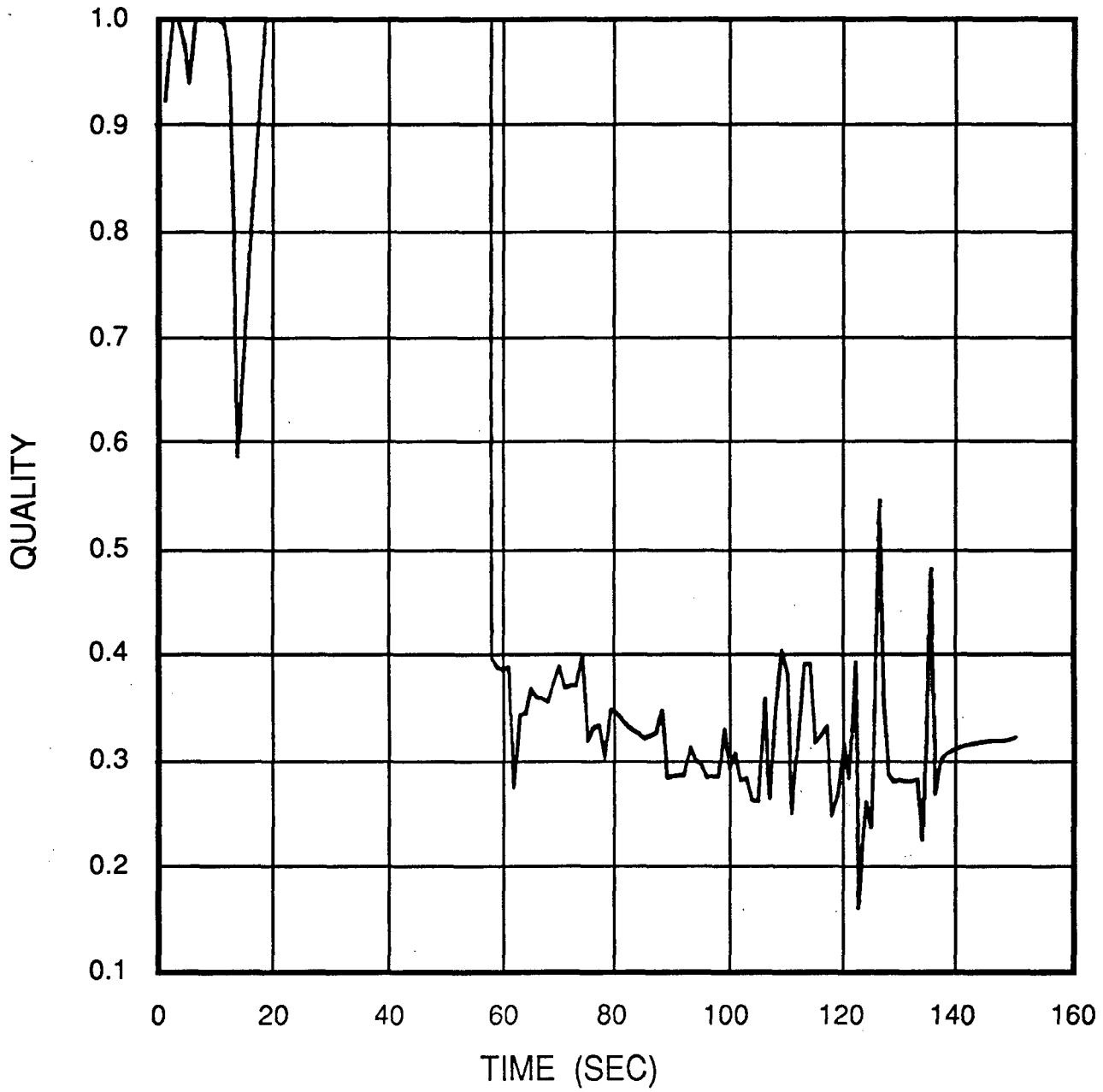


ZION STATION UFSAR

Figure 15.6-77

CORE POWER TRANSIENT
DECLG (CD=0.6)

JULY 1993

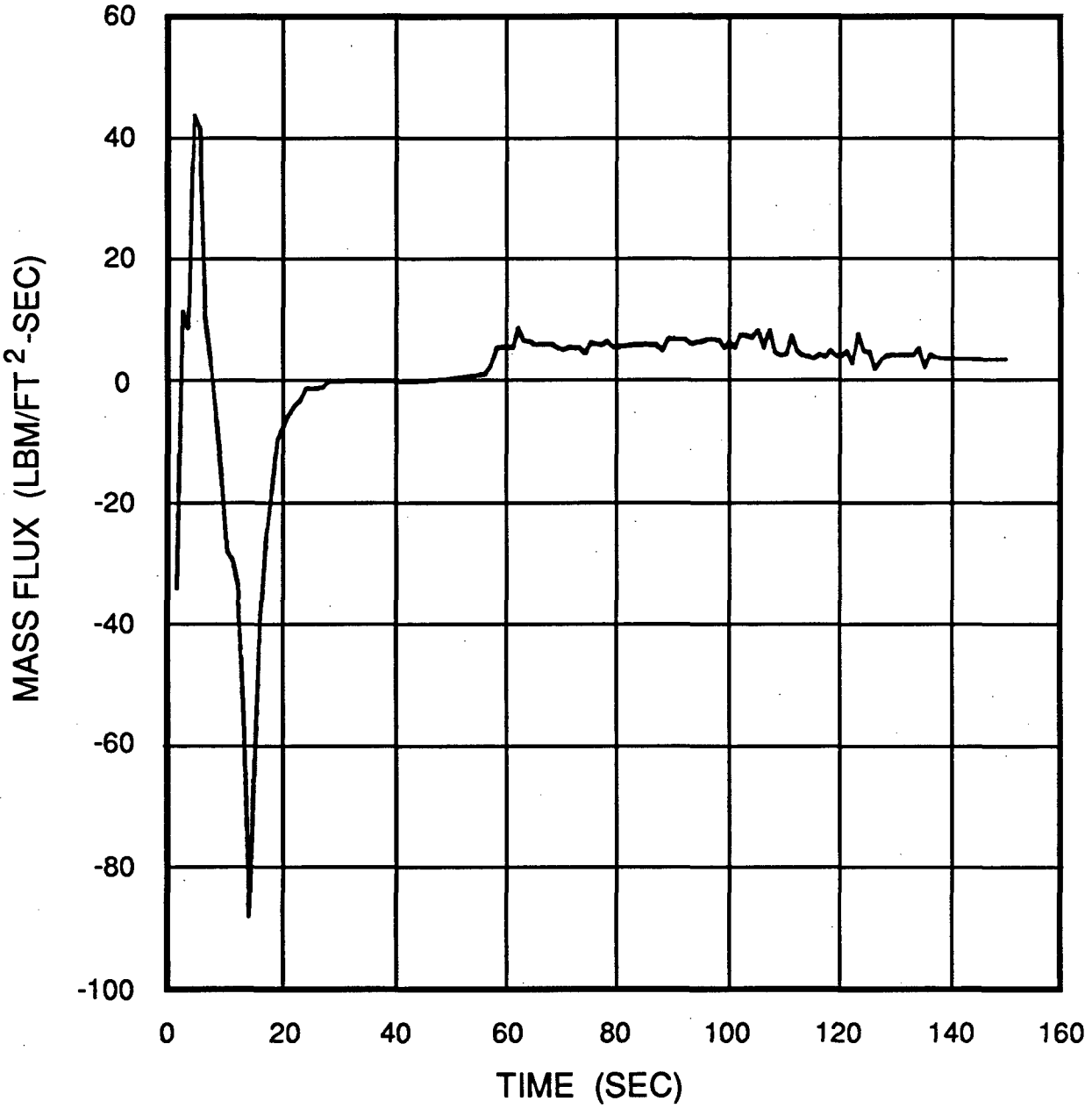


ZION STATION UFSAR

Figure 15.6-78

FLUID QUALITY
DECLG (CD=0.8)

JULY 1993

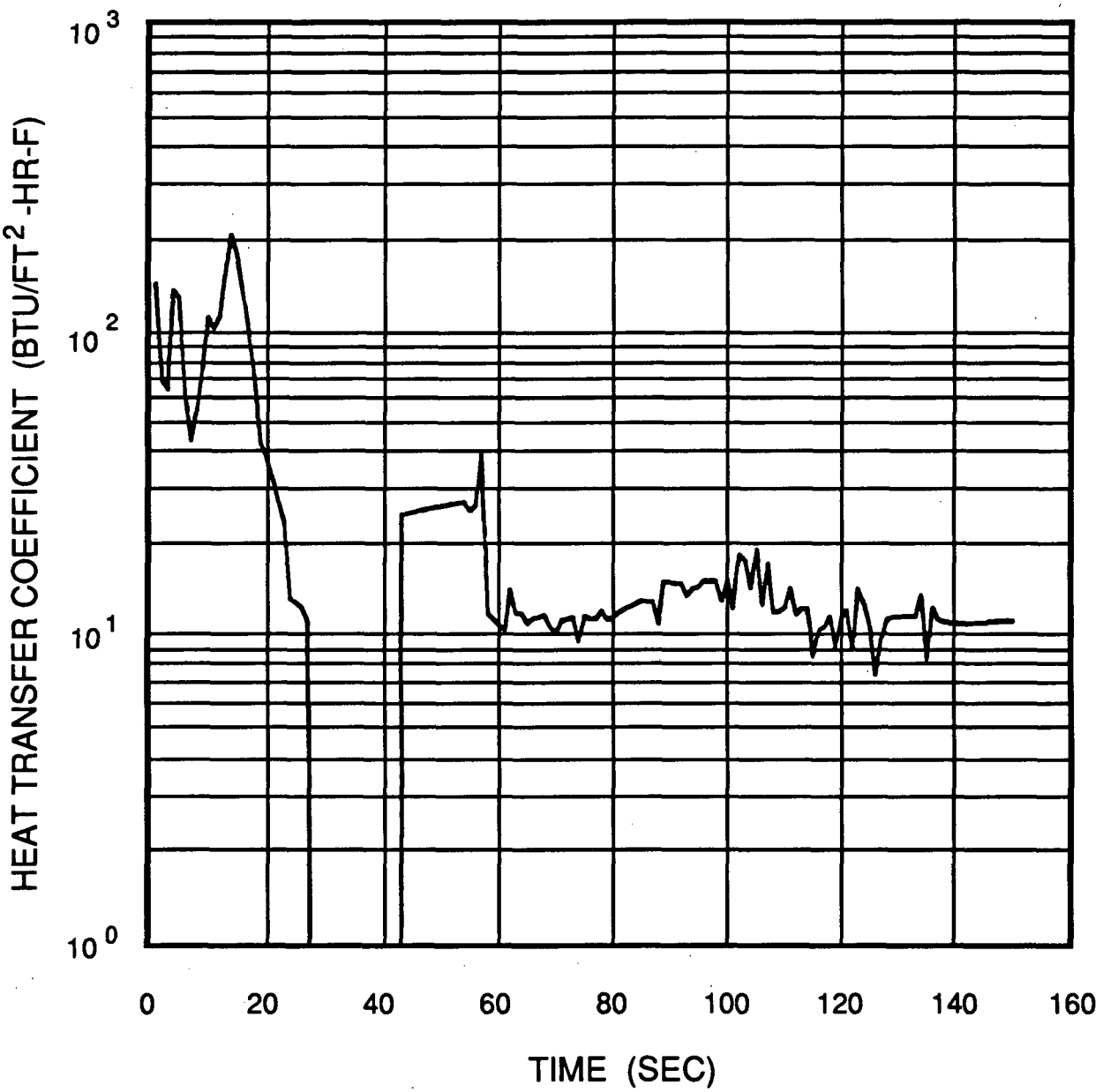


ZION STATION UFSAR

Figure 15.6-79

MASS VELOCITY
DECLG (CD=0.8)

JULY 1993

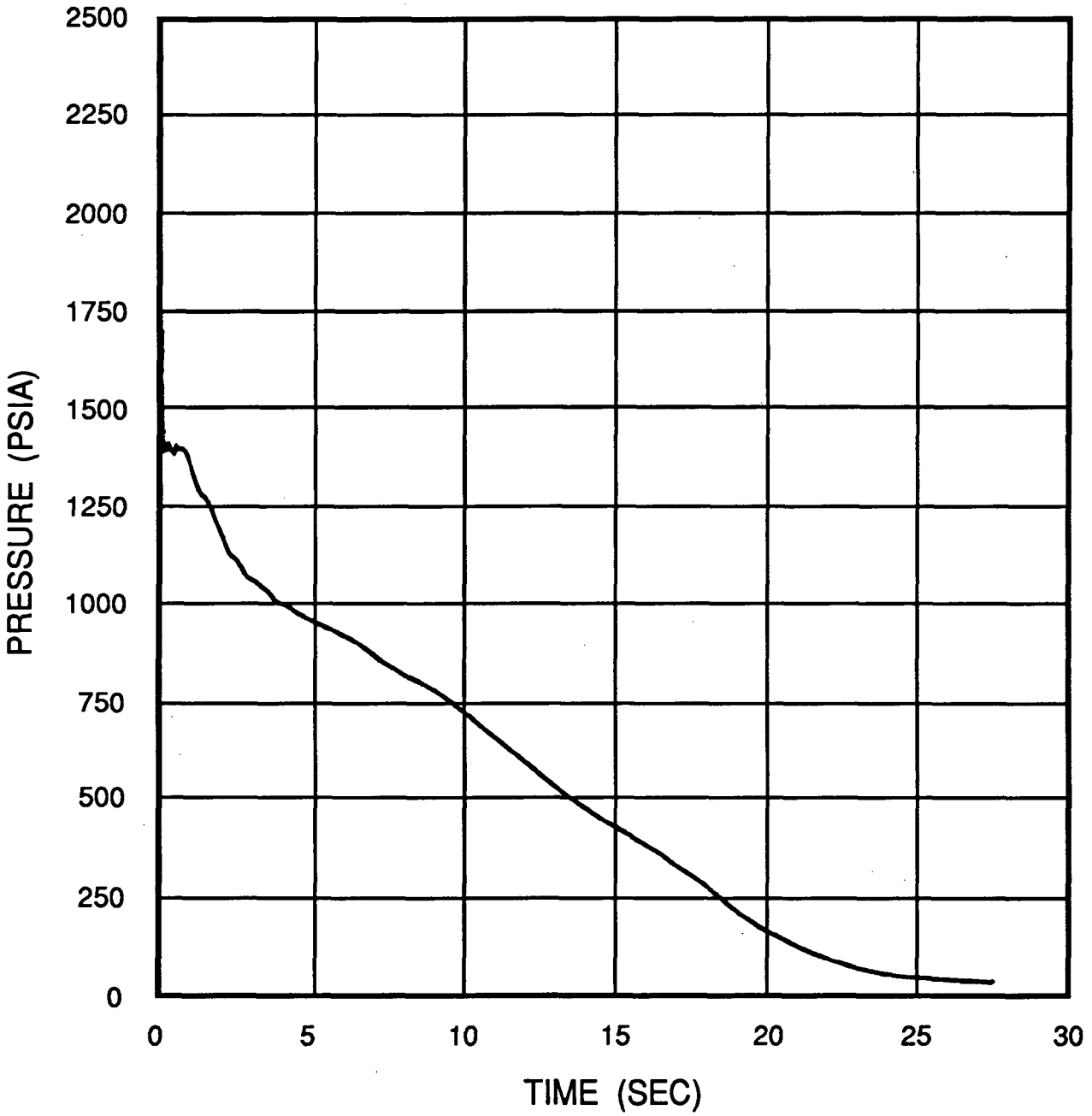


ZION STATION UFSAR

Figure 15.6-80

HEAT TRANSFER COEFFICIENT
DECLG (CD=0.8)

JULY 1993

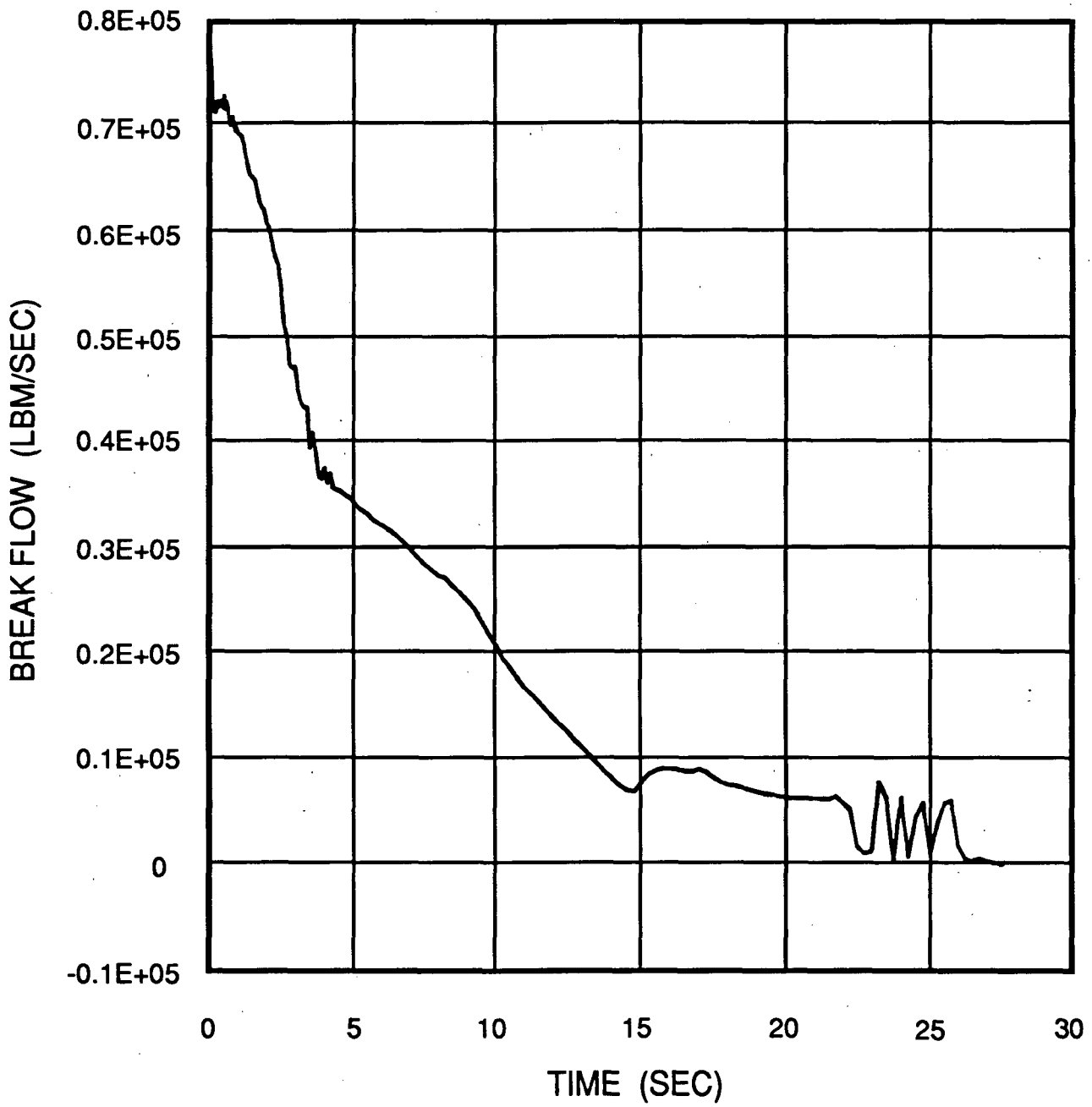


ZION STATION UFSAR

Figure 15.6-81

CORE PRESSURE
DECLG (CD=0.8)

JULY 1993

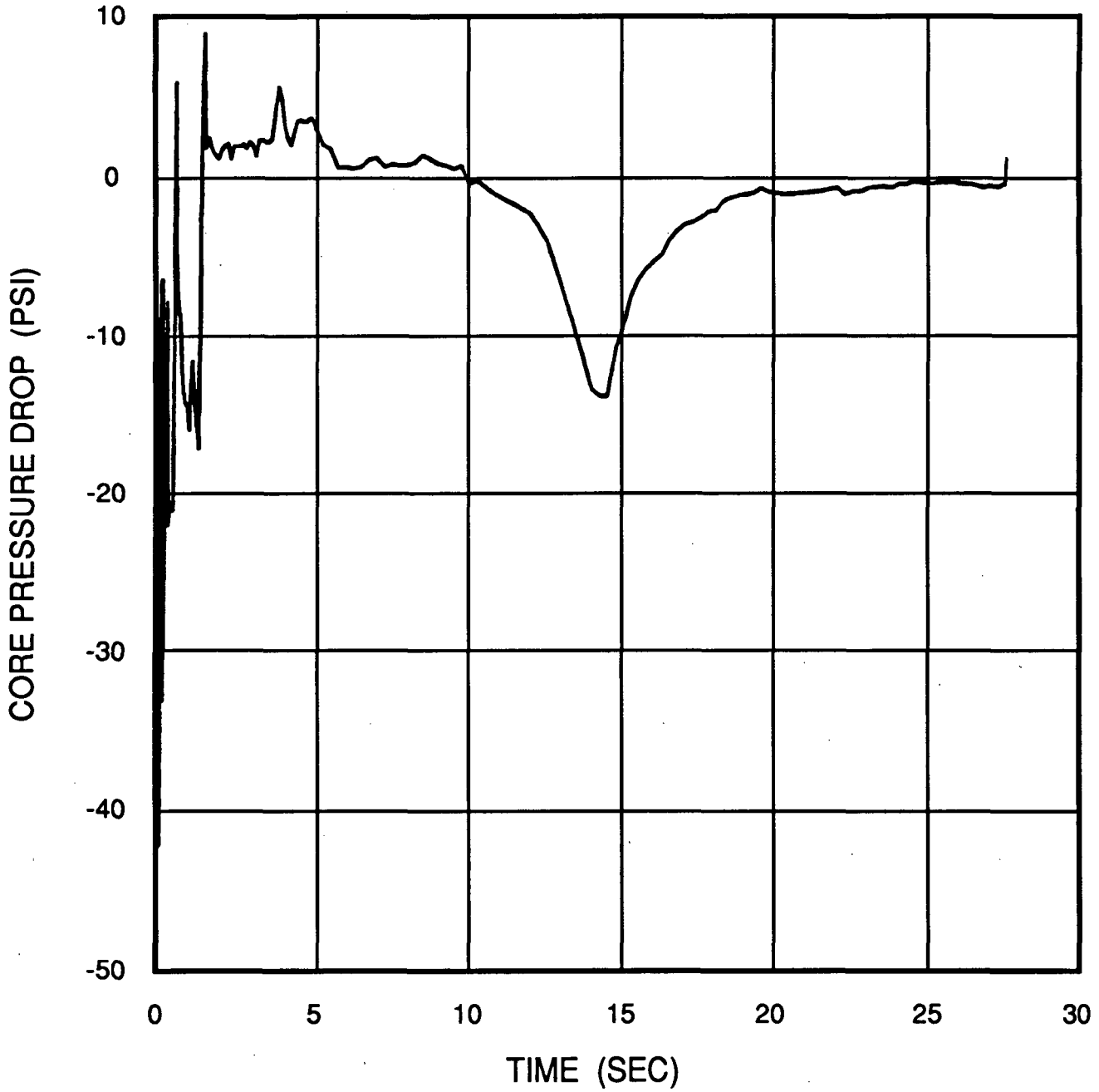


ZION STATION UFSAR

Figure 15.6-82

BREAK FLOW RATE
DECLG (CD=0.8)

JULY 1993

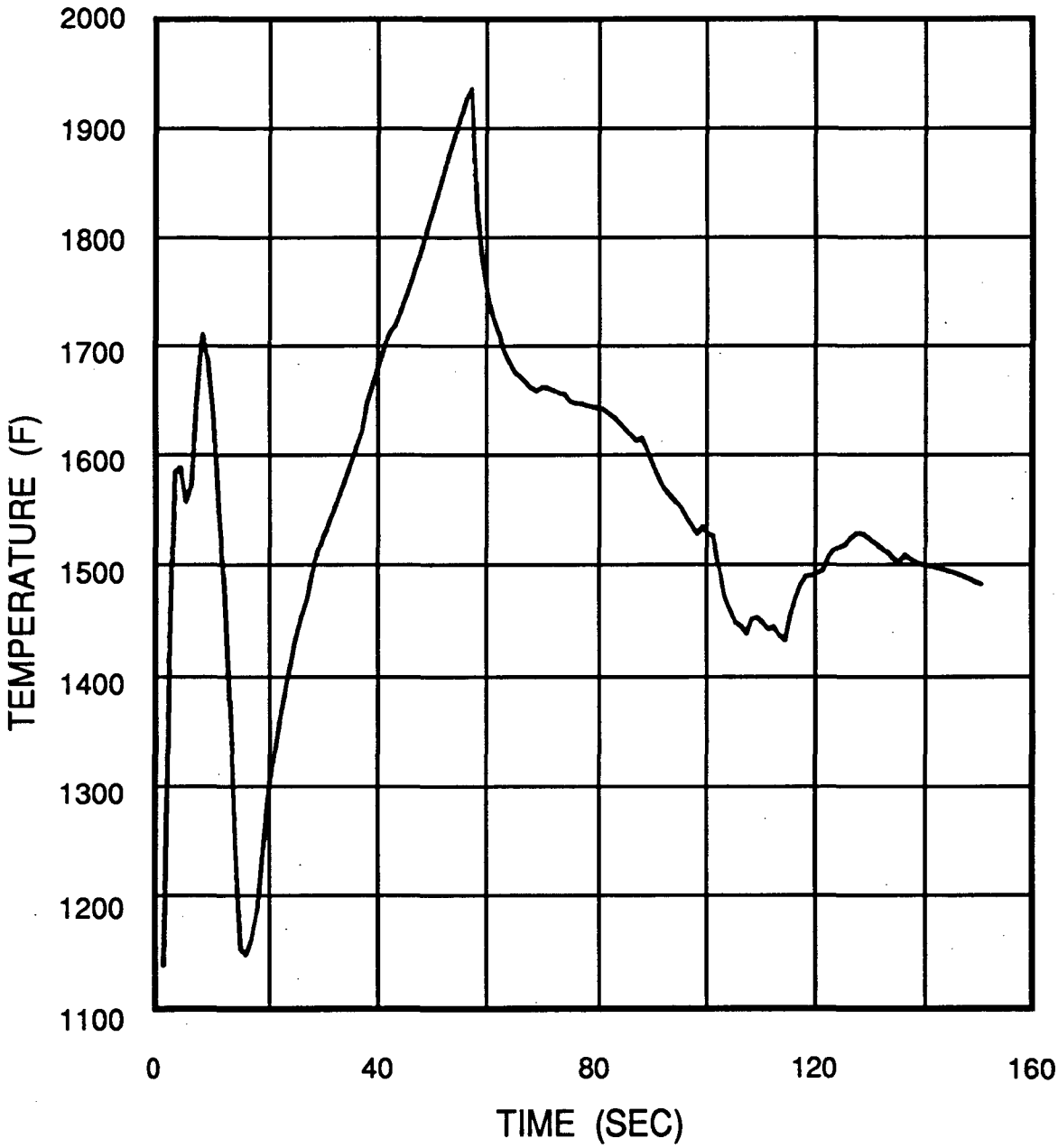


ZION STATION UFSAR

Figure 15.6-83

CORE PRESSURE DROP
DECLG (CD=0.8)

JULY 1993

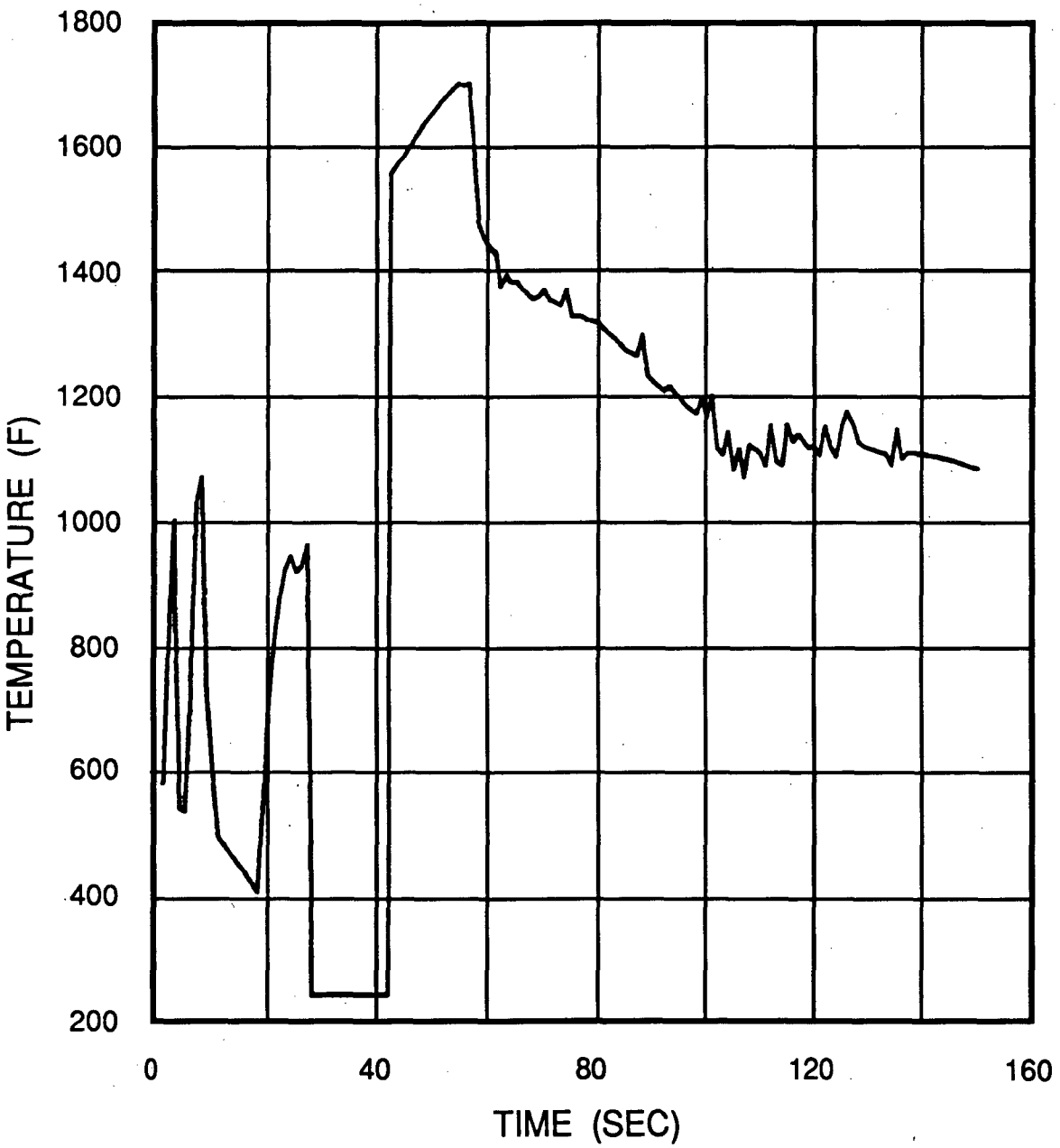


ZION STATION UFSAR

Figure 15.6-84

PEAK CLADDING TEMPERATURE
DECLG (CD=0.8)

JULY 1993

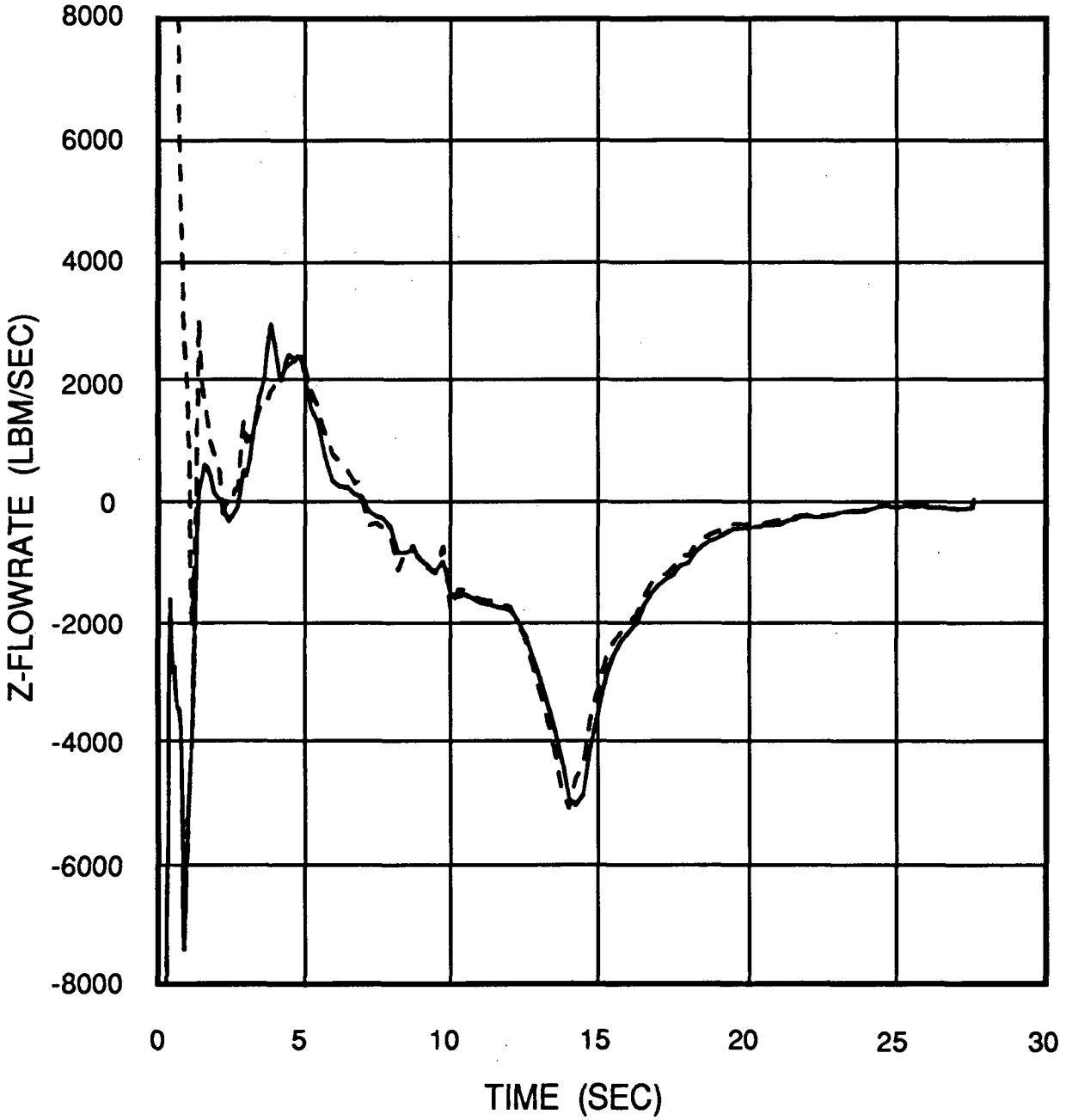


ZION STATION UFSAR

Figure 15.6-85

FLUID TEMPERATURE
DECLG (CD=0.8)

JULY 1993

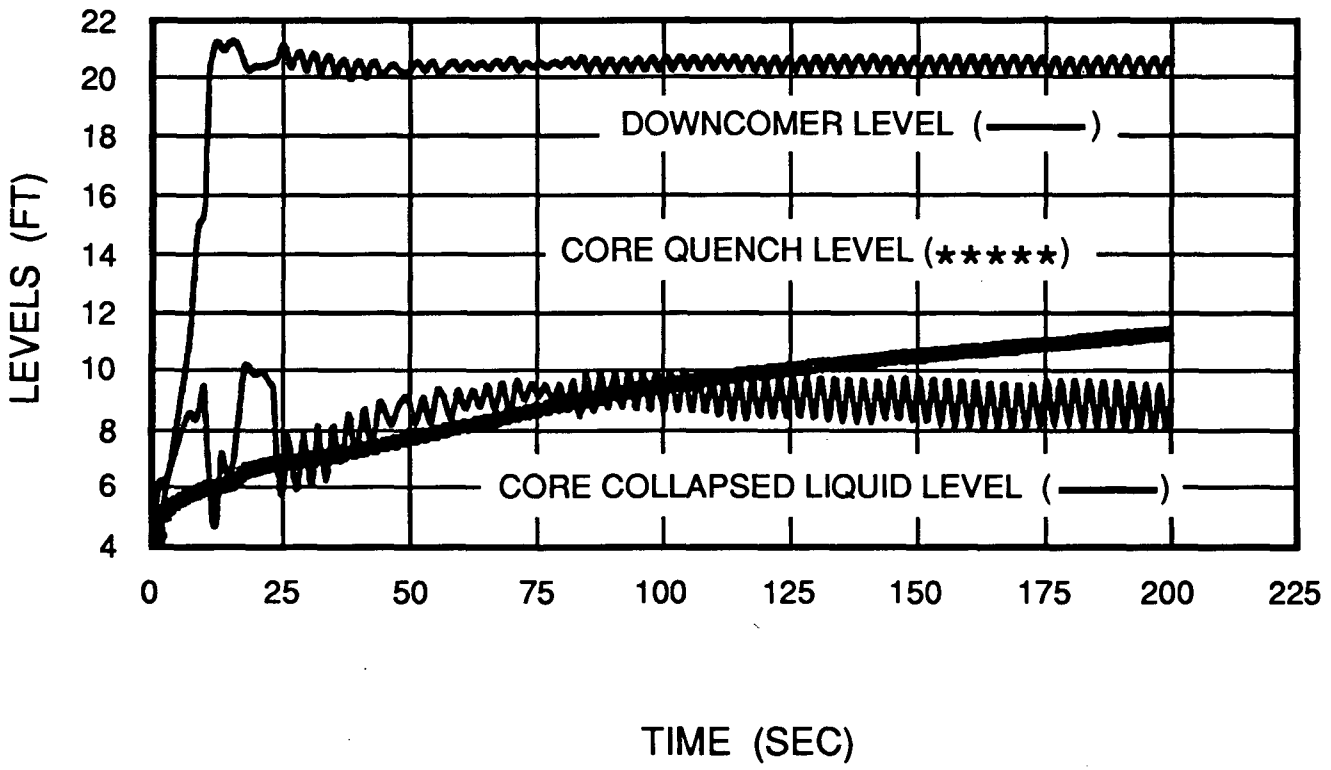


ZION STATION UFSAR

Figure 15.6-86

CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.8)

JULY 1993



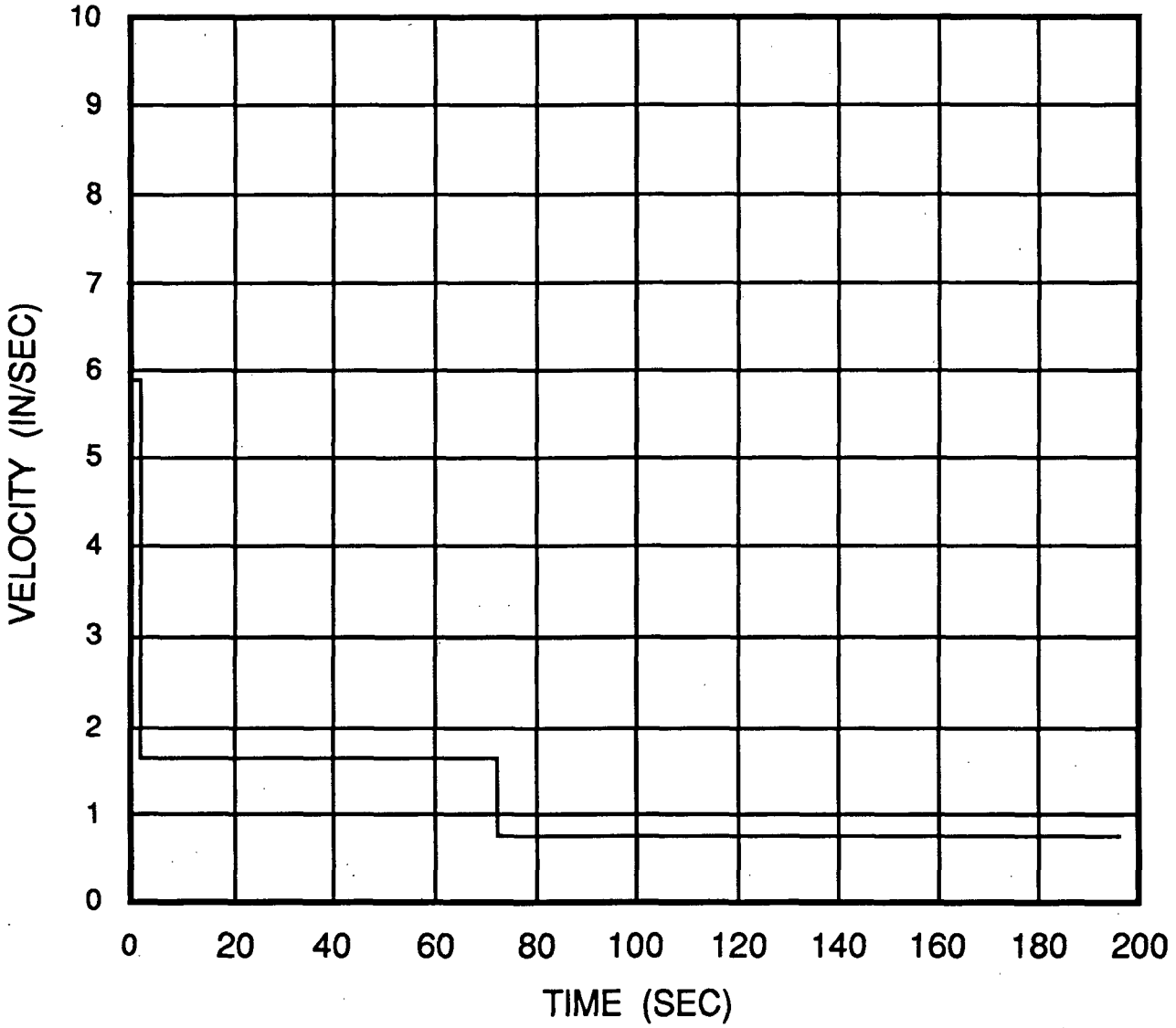
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 42.042 SECONDS FOR CD = 0.8

ZION STATION UFSAR

Figure 15.6-87
REFLOOD TRANSIENT
CORE AND DOWNCOMER LEVELS
DECLG (CD=0.8)

JULY 1993



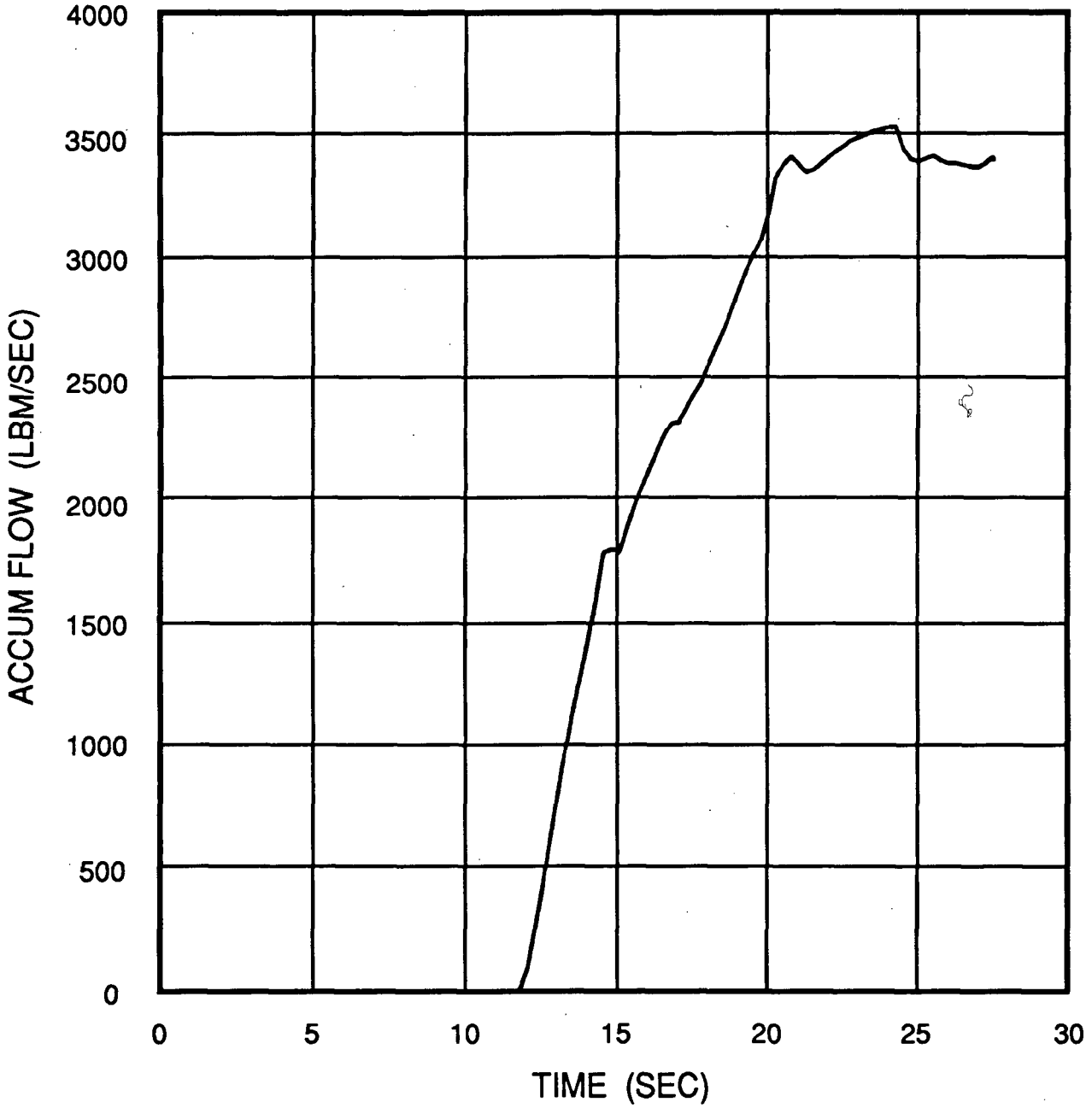
TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

BOC = 42.042 SECONDS FOR CD = 0.8

ZION STATION UFSAR

Figure 15.6-88
REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD=0.8)

JULY 1993

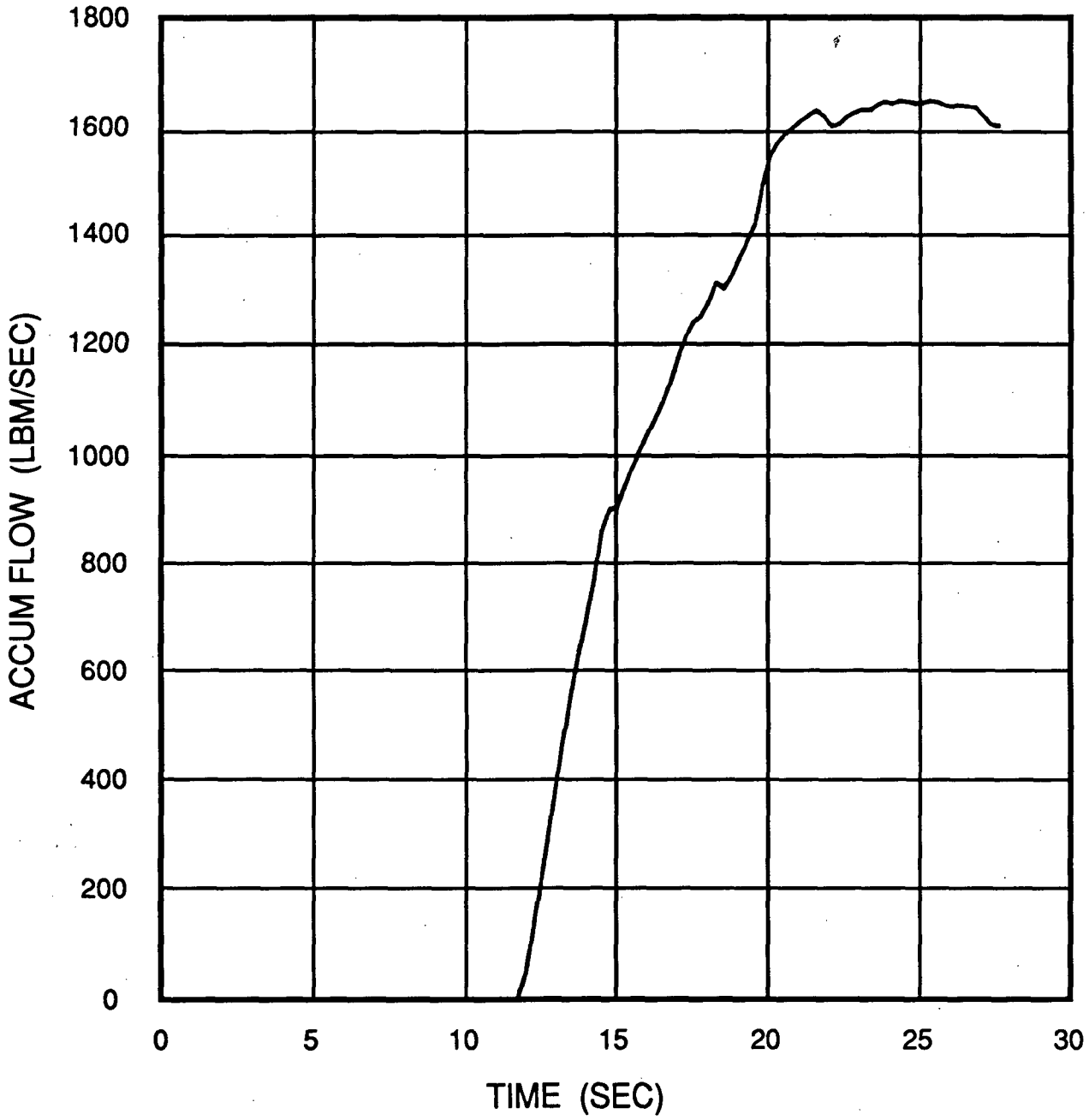


FLOW FROM TWO INTACT LOOP ACCUMULATORS

ZION STATION UFSAR

Figure 15.6-89
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.8)

JULY 1993

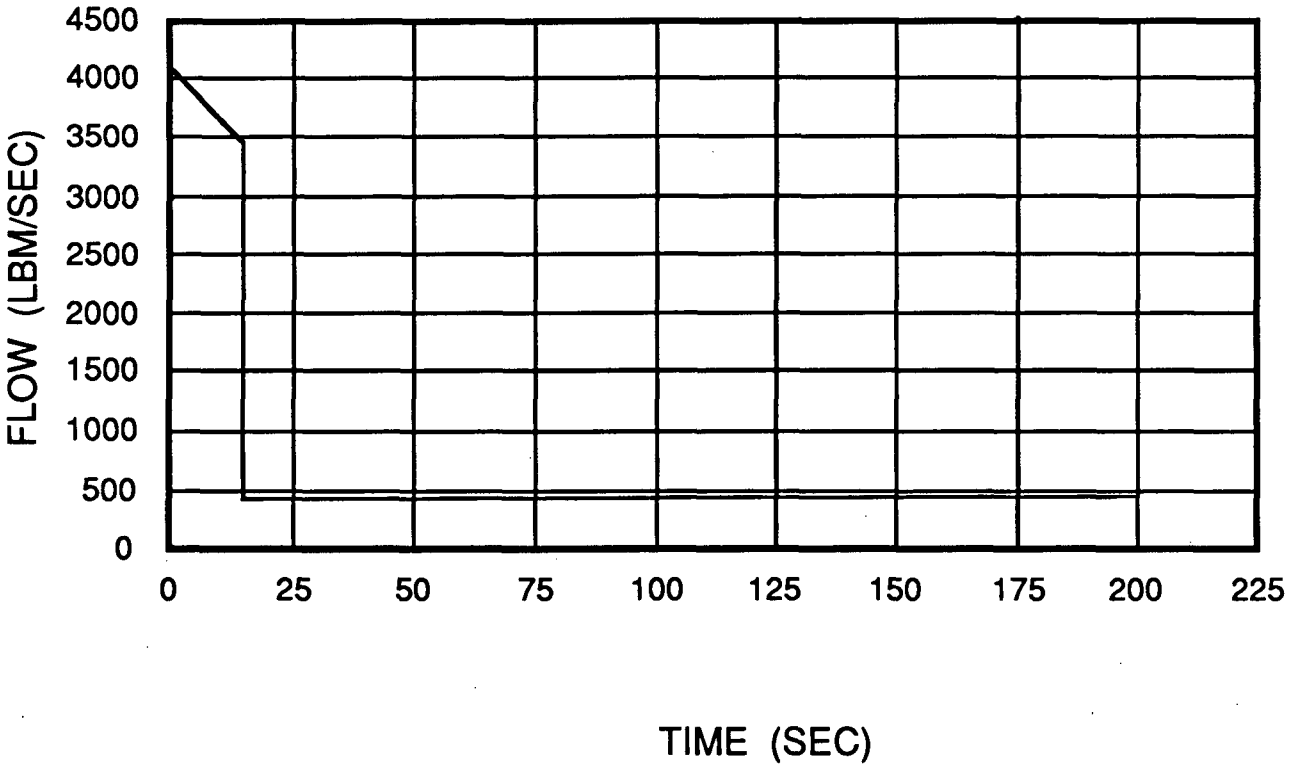


FLOW FROM ONE INTACT LOOP ACCUMULATOR

ZION STATION UFSAR

Figure 15.6-90
ACCUMULATOR FLOW
(BLOWDOWN)
DECLG (CD=0.8)

JULY 1993



TIME 0.0 IN BASH IS BOTTOM OF CORE RECOVERY (BOC) TIME

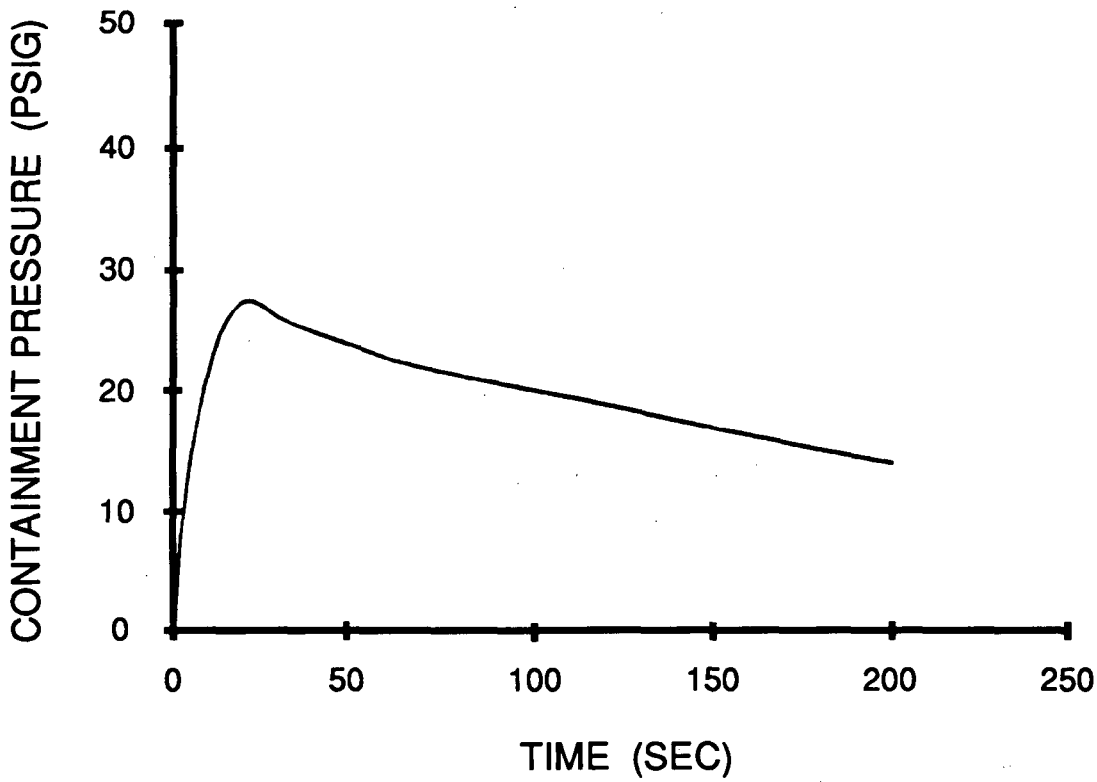
BOC = 42.042 SECONDS FOR CD = 0.8

ZION STATION UFSAR

Figure 15.6-91

PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.8)

JULY 1993

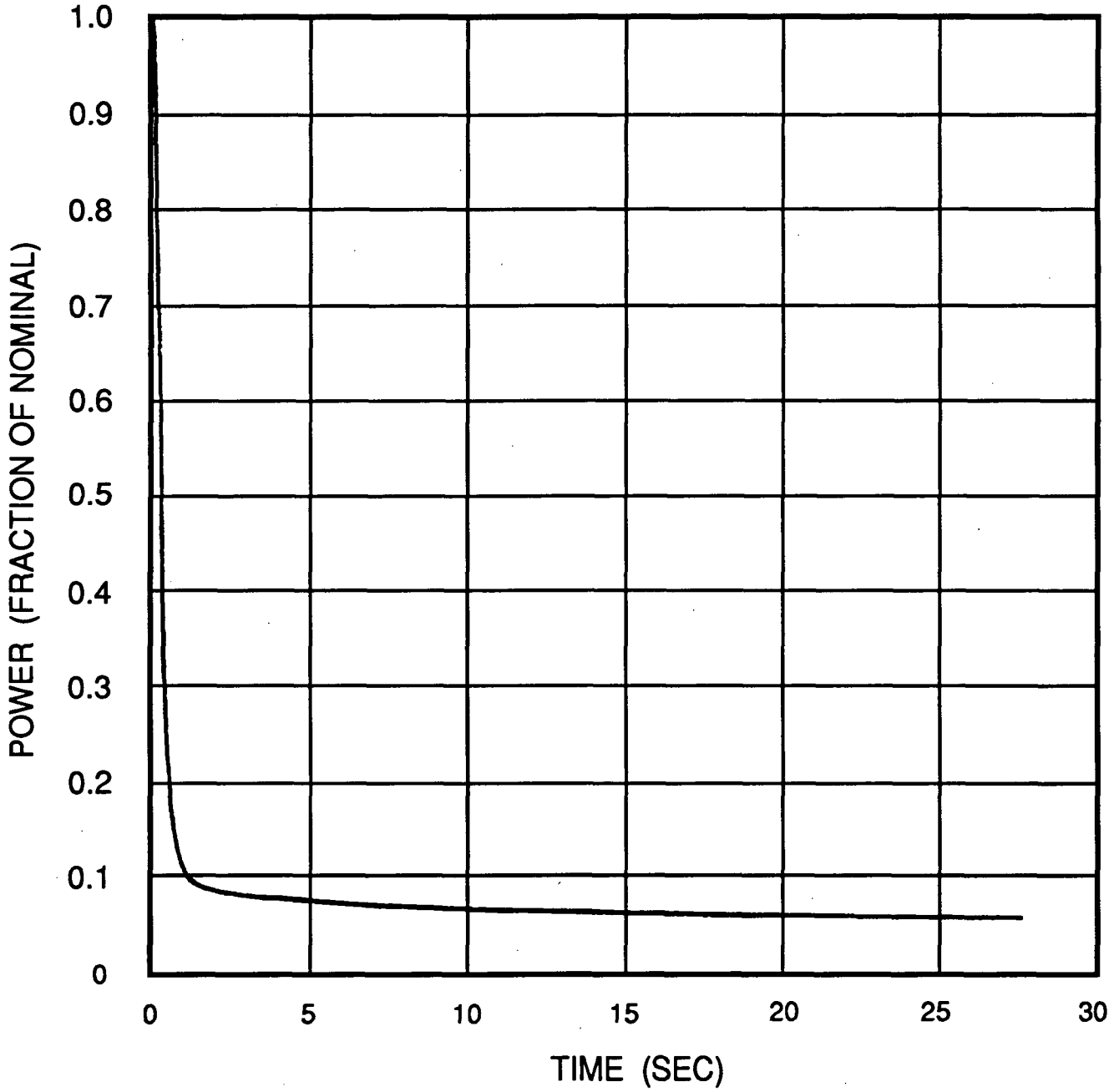


ZION STATION UFSAR

Figure 15.6-92

CONTAINMENT PRESSURE
DECLG (CD=0.8)

JULY 1993

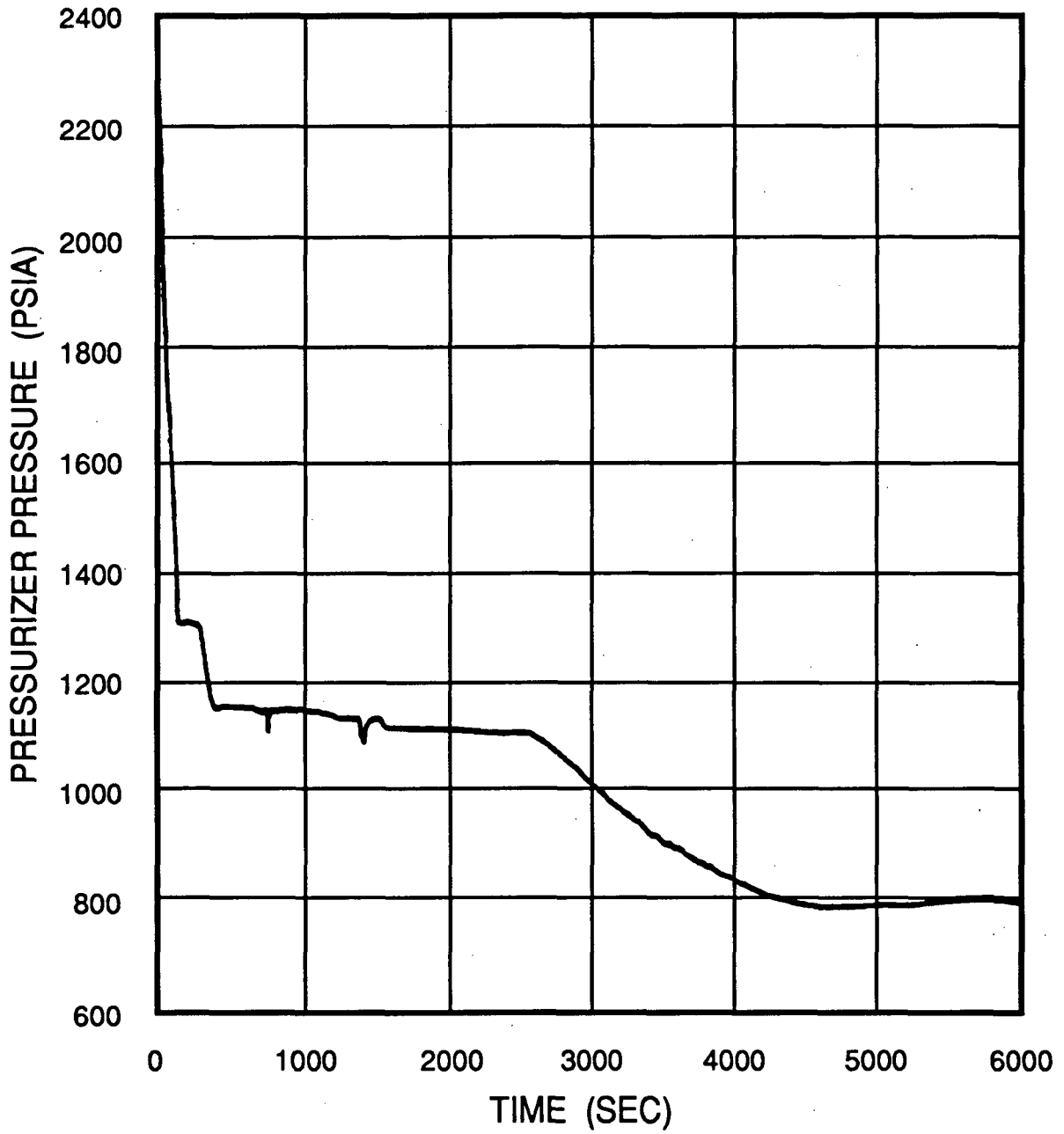


ZION STATION UFSAR

Figure 15.6-93

CORE POWER TRANSIENT
DECLG (CD=0.8)

JULY 1993

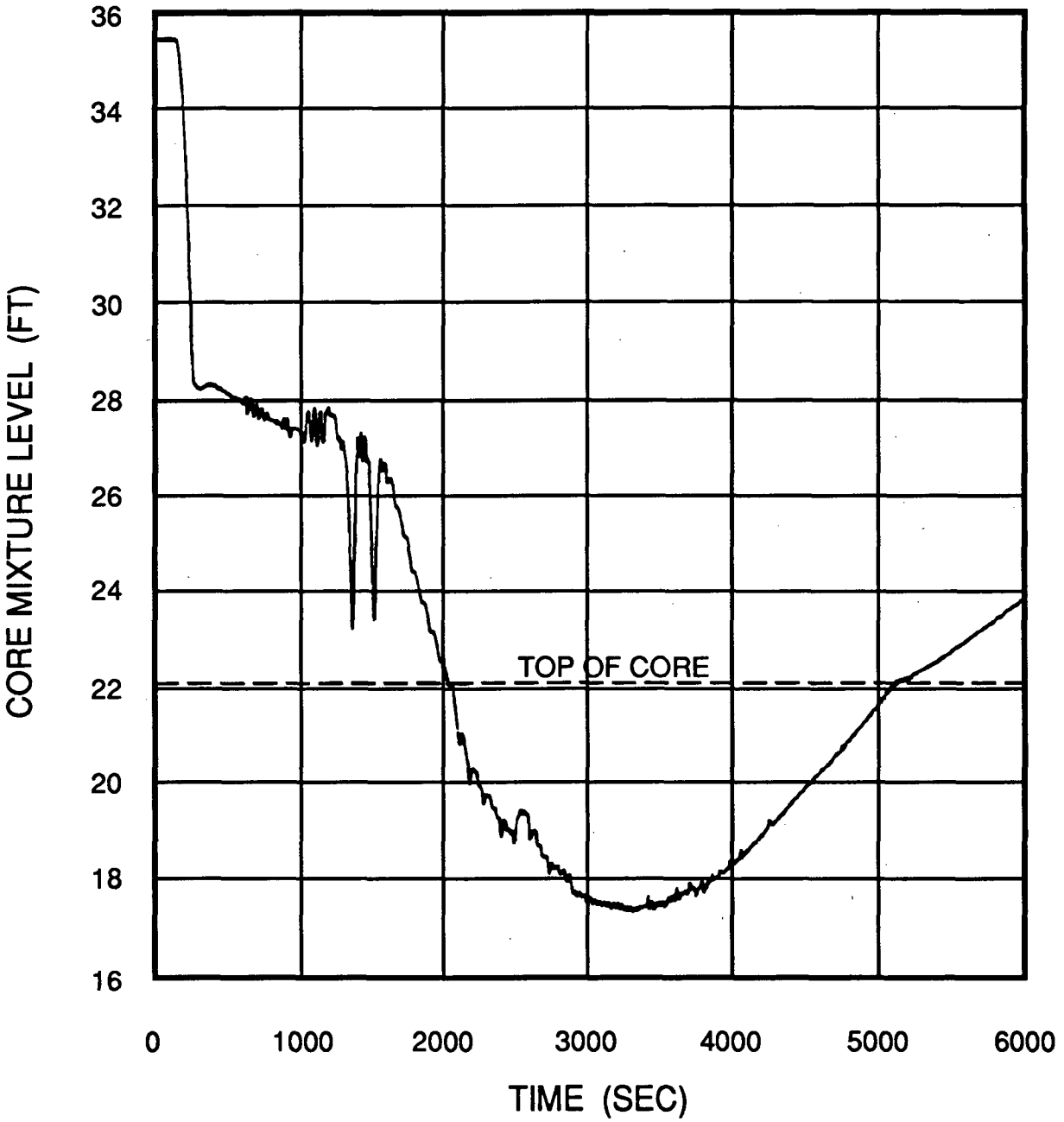


ZION STATION UFSAR

Figure 15.6-94

SMALL BREAK LOCA (2-INCH)
RCS DEPRESSURIZATION

JULY 1993

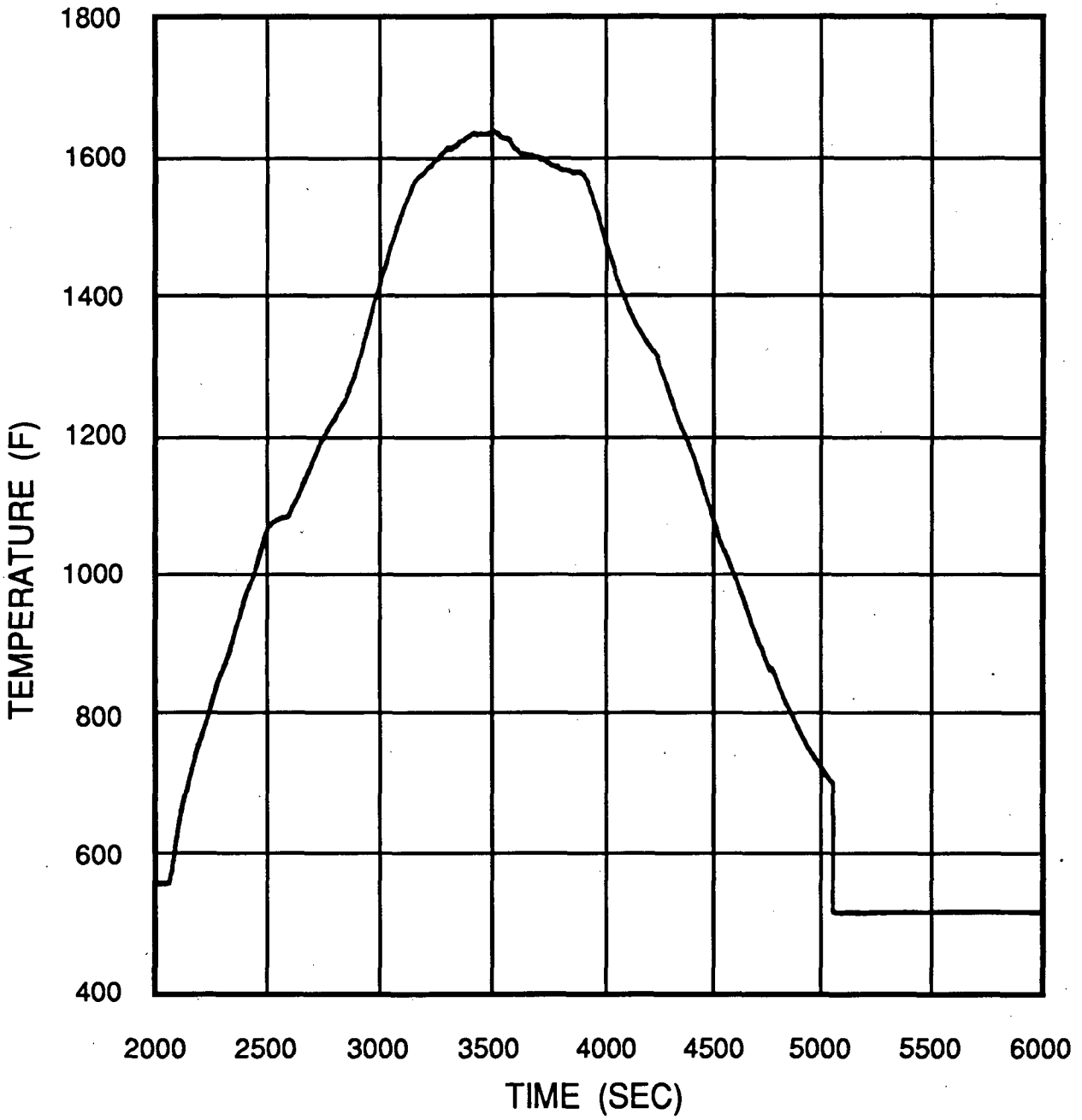


ZION STATION UFSAR

Figure 15.6-95

SMALL BREAK LOCA (2-INCH)
CORE MIXTURE LEVEL

JULY 1993

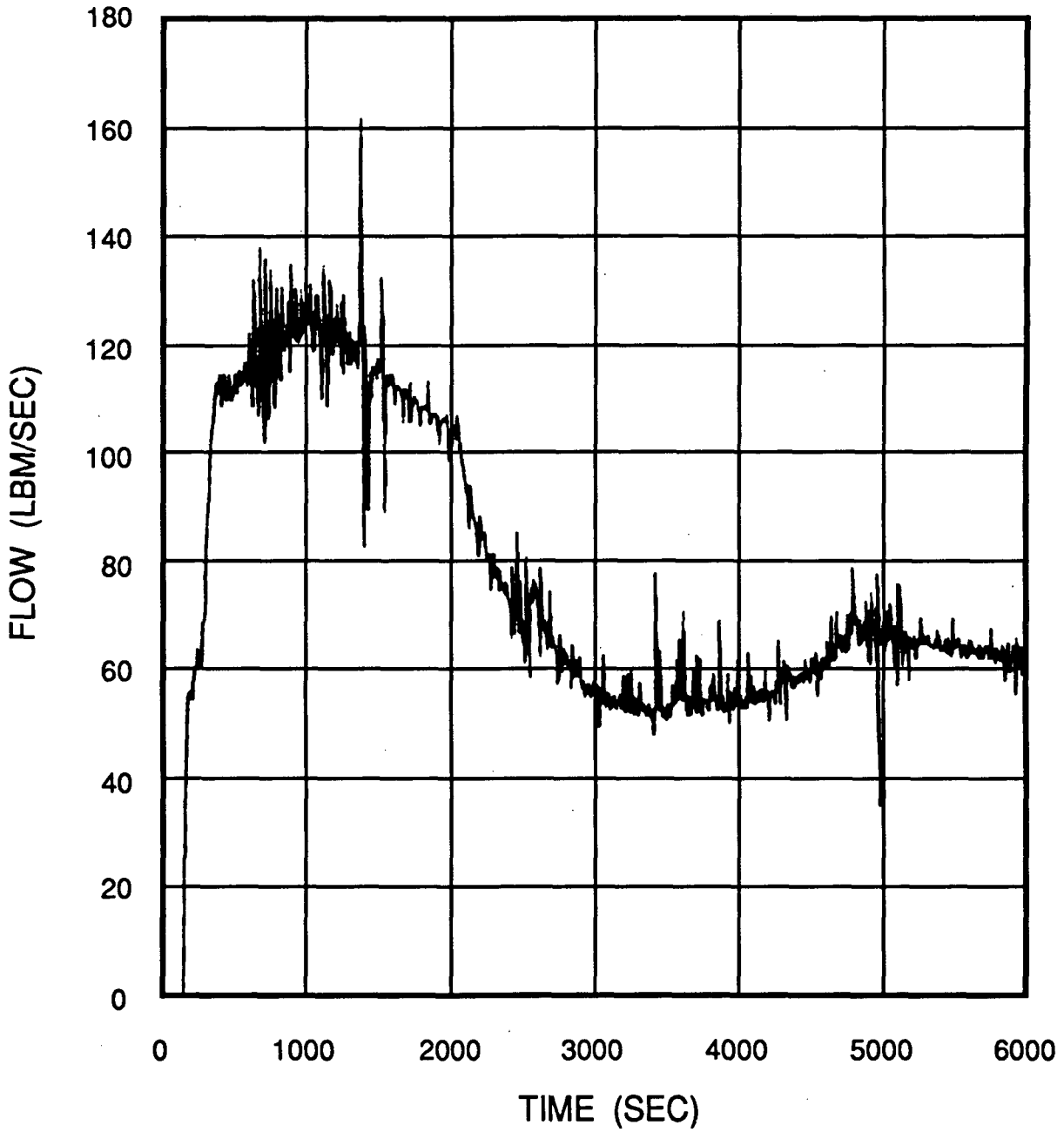


ZION STATION UFSAR

Figure 15.6-96

SMALL BREAK LOCA (2-INCH)
PEAK CLADDING TEMPERATURE

JULY 1993

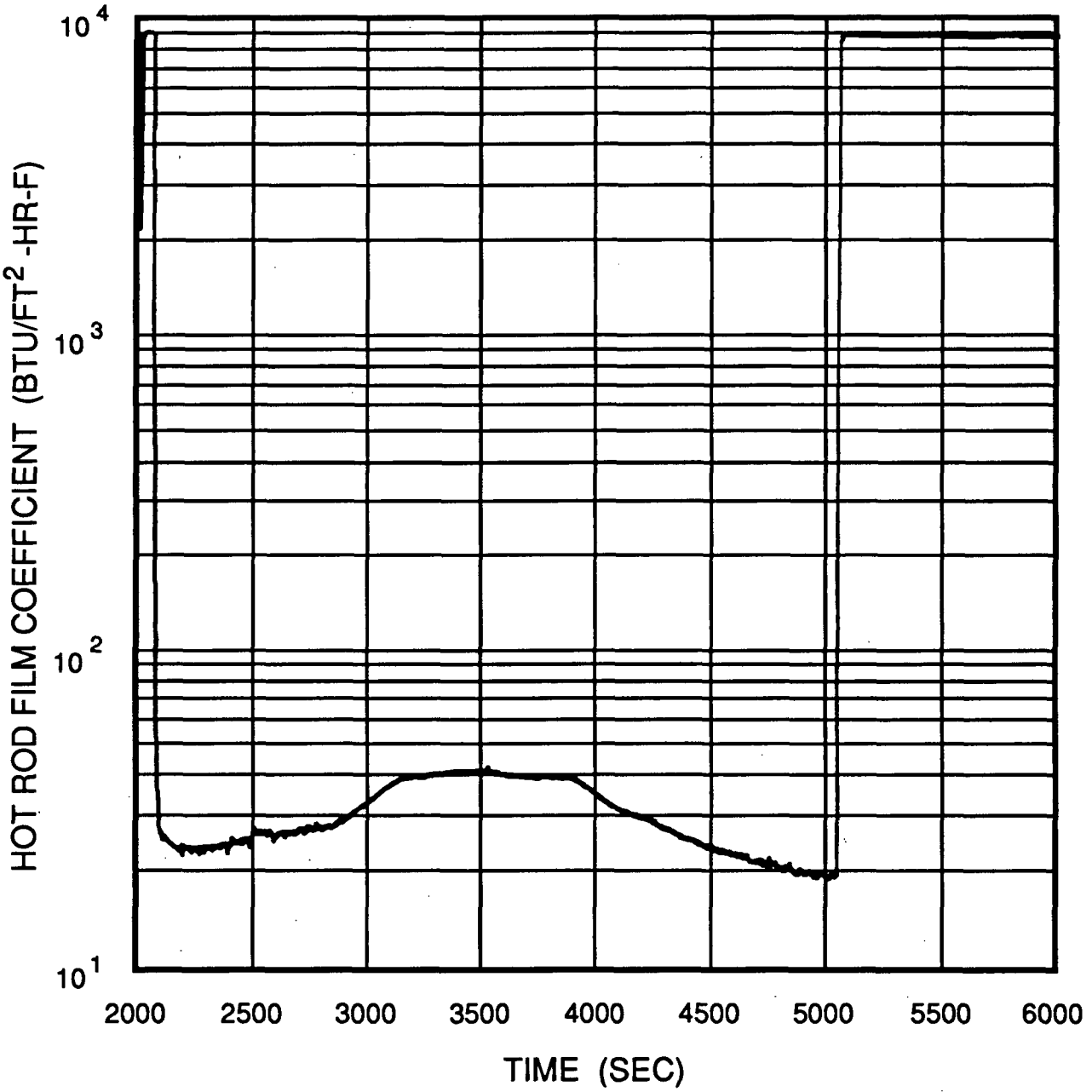


ZION STATION UFSAR

Figure 15.6-97

SMALL BREAK LOCA (2-INCH)
STEAM FLOWRATE

JULY 1993

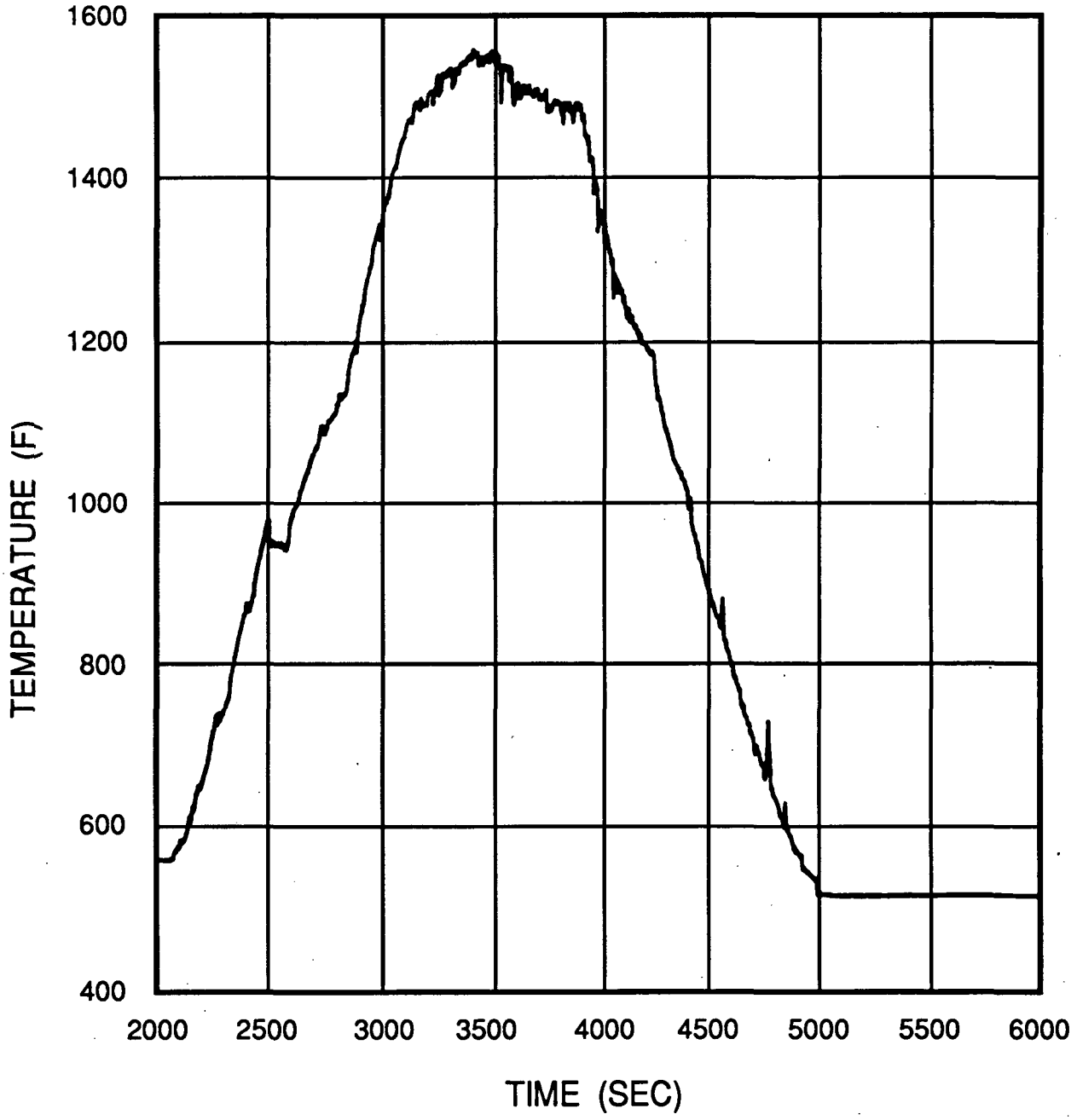


ZION STATION UFSAR

Figure 15.6-98

SMALL BREAK LOCA (2-INCH)
HOT ROD FILM COEFFICIENT

JULY 1993

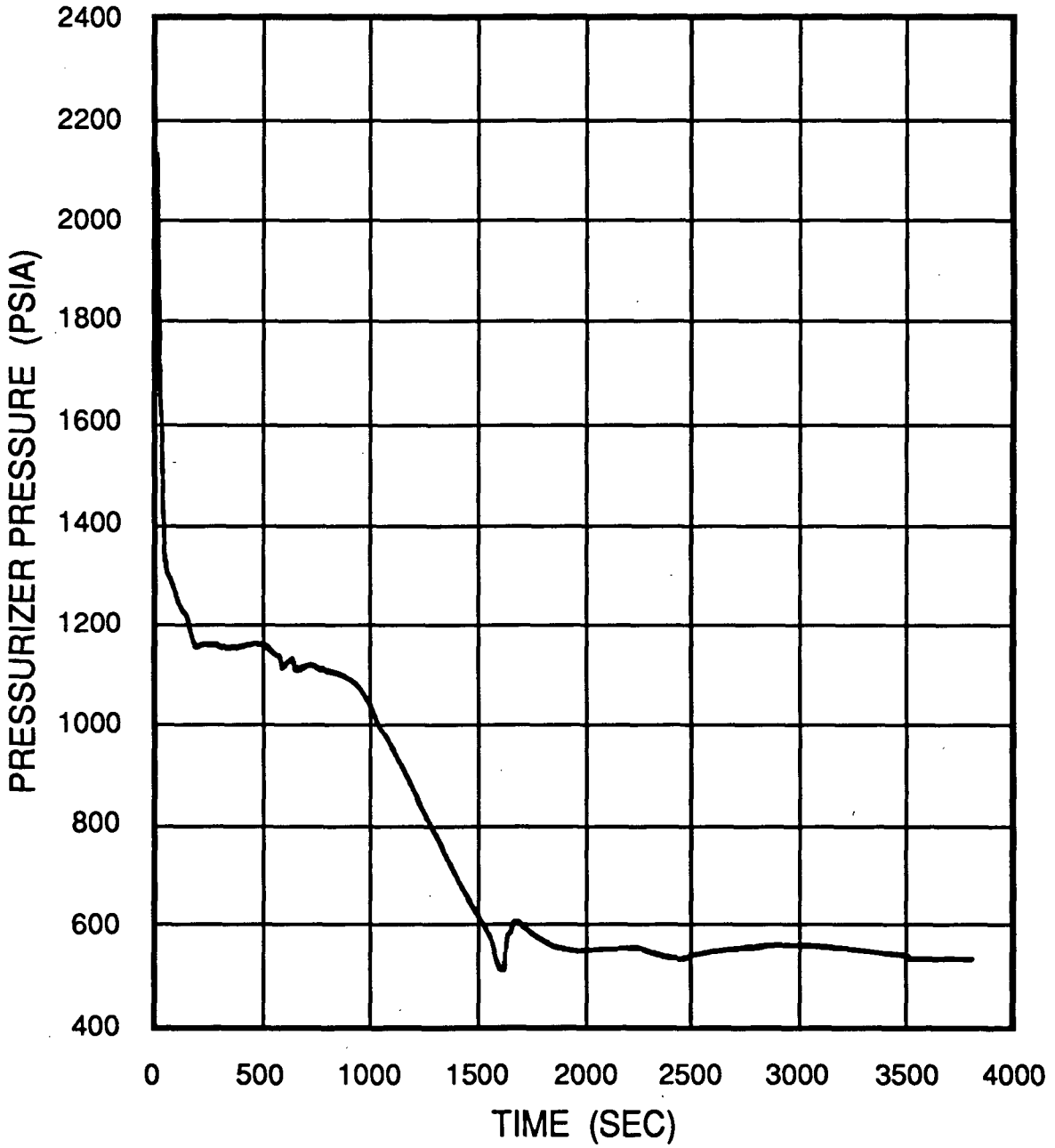


ZION STATION UFSAR

Figure 15.6-99

SMALL BREAK LOCA (2-INCH)
HOT SPOT FLUID TEMPERATURE

JULY 1993

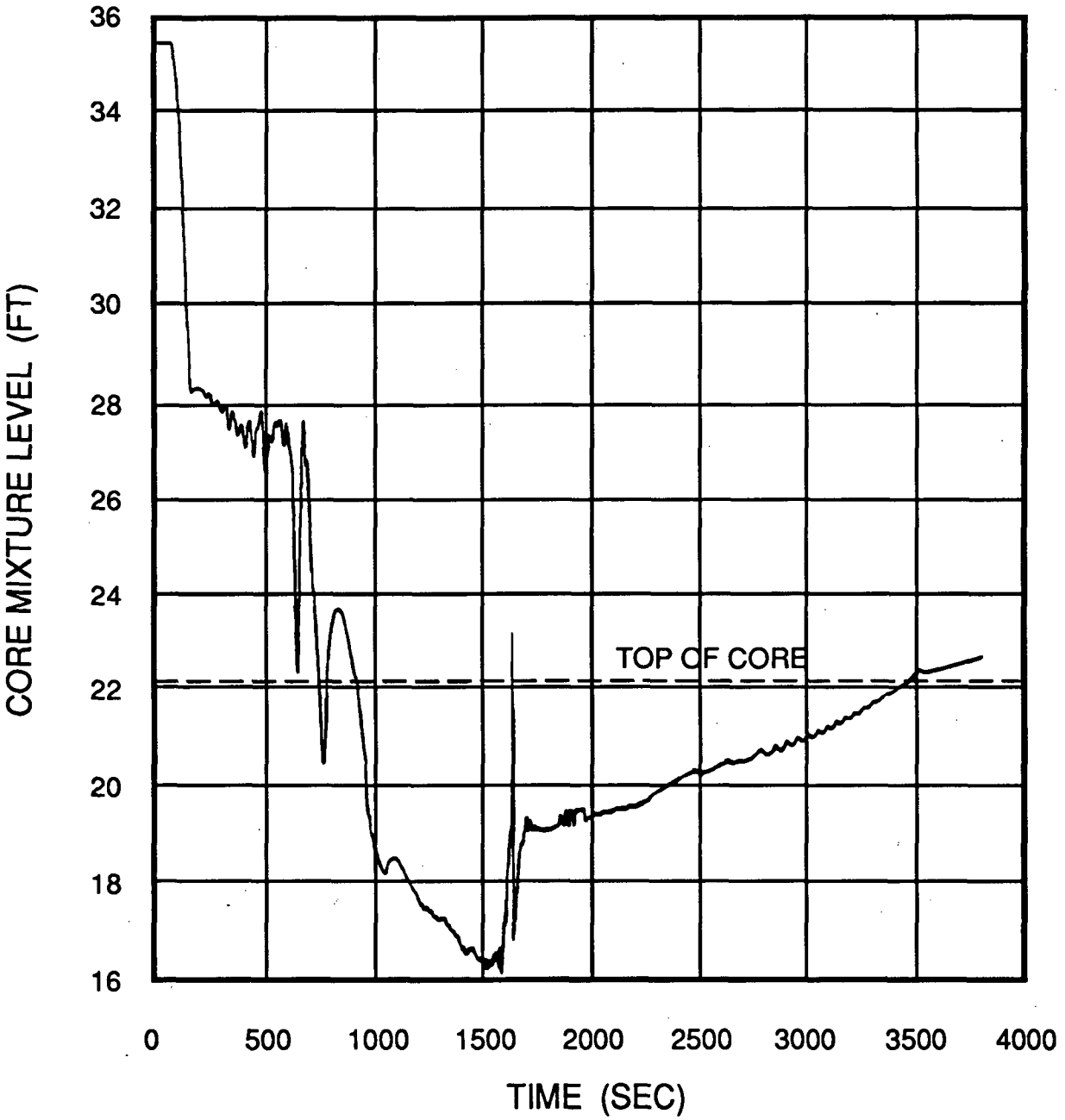


ZION STATION UFSAR

Figure 15.6-100

SMALL BREAK LOCA (3-INCH)
RCS DEPRESSURIZATION

JULY 1993

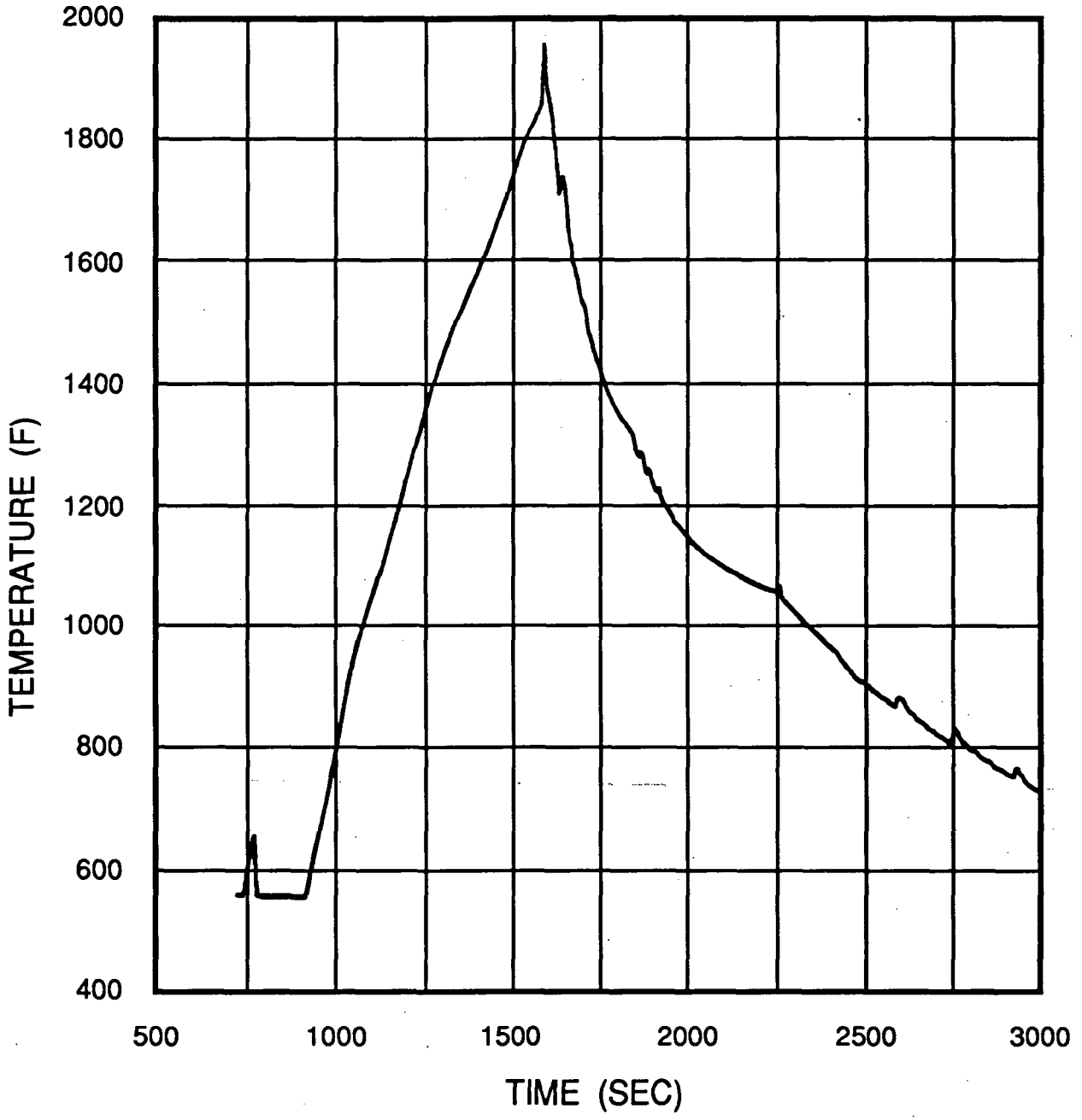


ZION STATION UFSAR

Figure 15.6-101

SMALL BREAK LOCA (3-INCH)
CORE MIXTURE LEVEL

JULY 1993

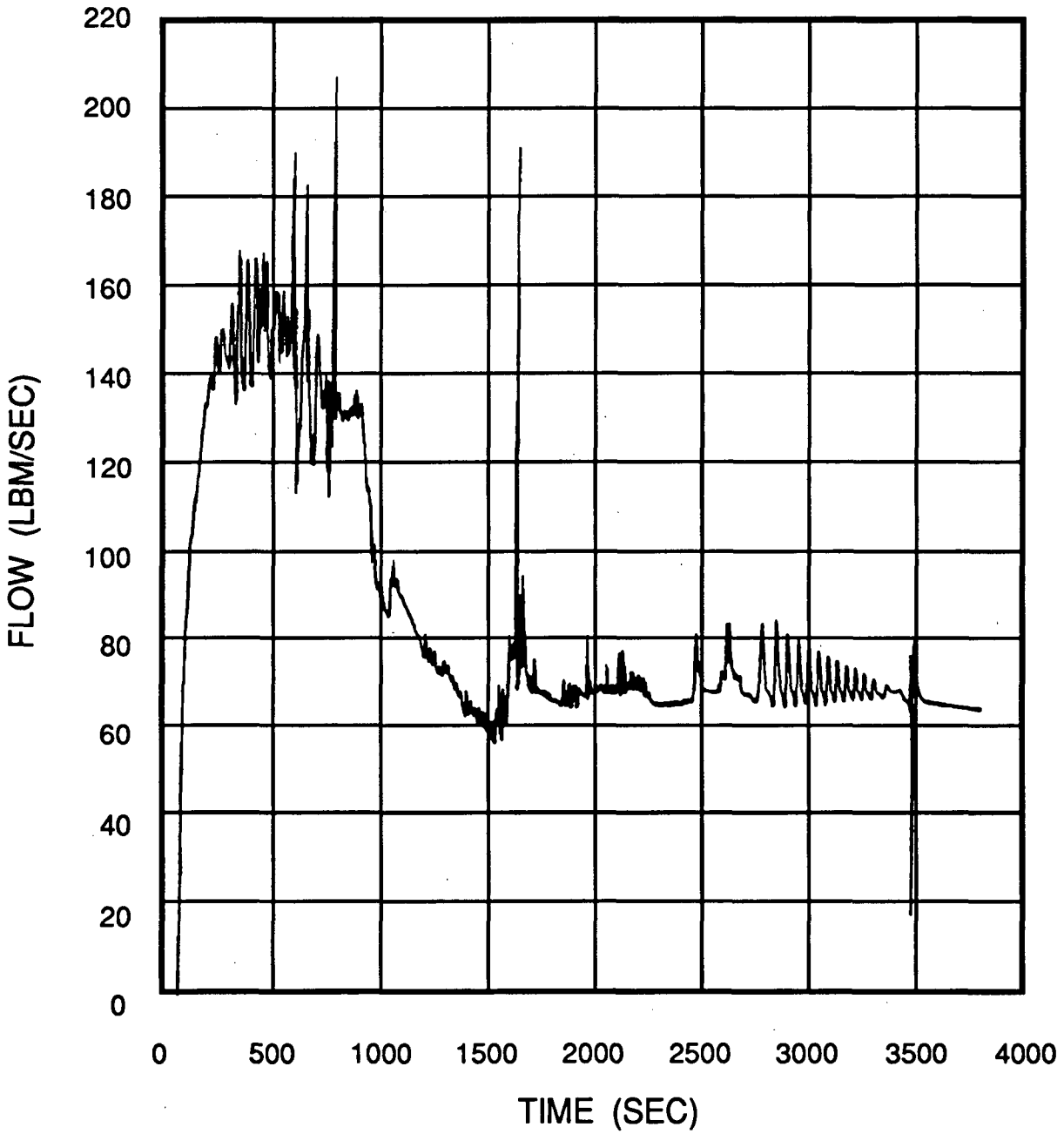


ZION STATION UFSAR

Figure 15.6-102

SMALL BREAK LOCA (3-INCH)
PEAK CLADDING TEMPERATURE

JULY 1993

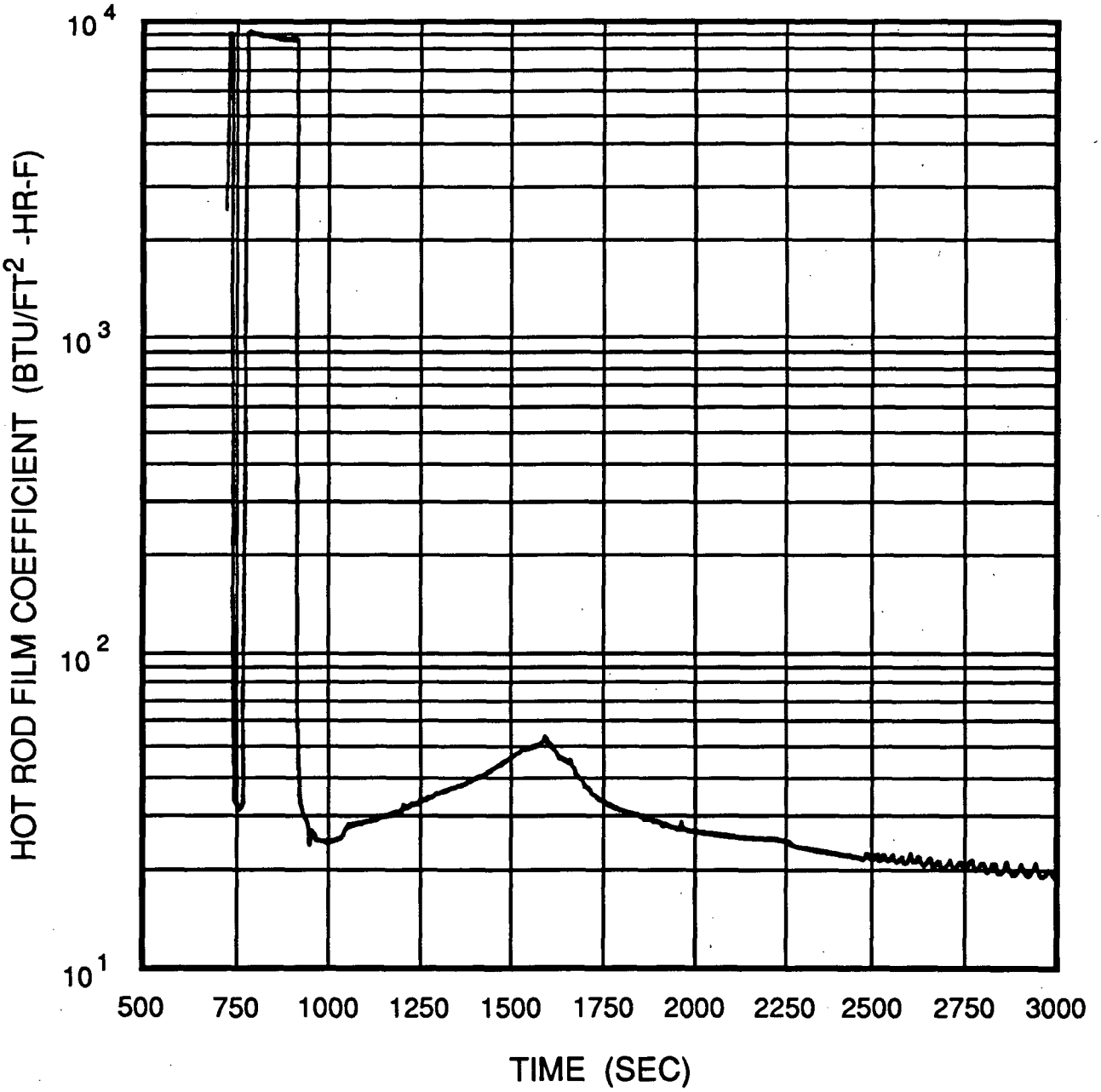


ZION STATION UFSAR

Figure 15.6-103

SMALL BREAK LOCA (3-INCH)
STEAM FLOWRATE

JULY 1993

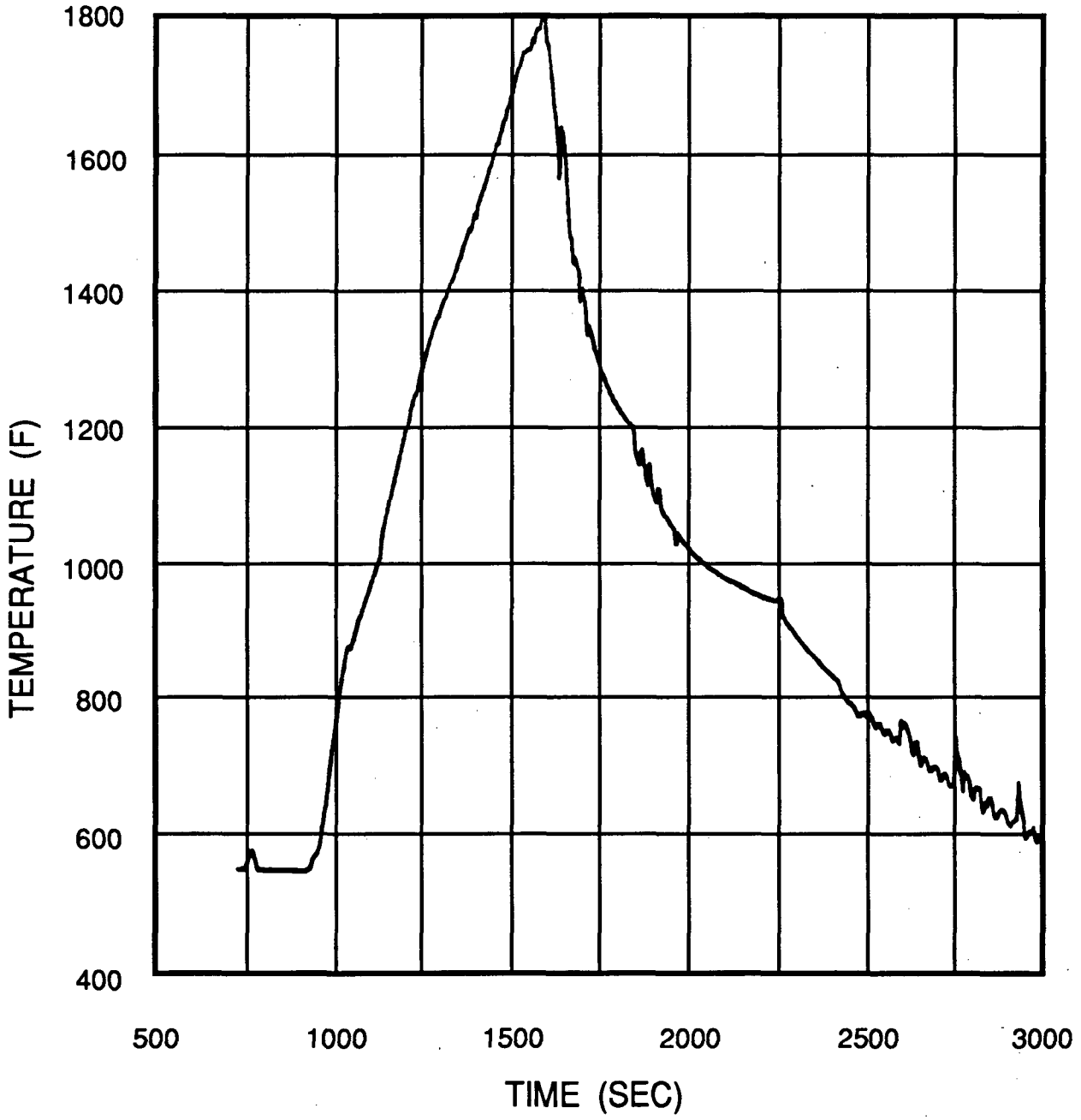


ZION STATION UFSAR

Figure 15.6-104

SMALL BREAK LOCA (3-INCH)
HOT ROD FILM COEFFICIENT

JULY 1993

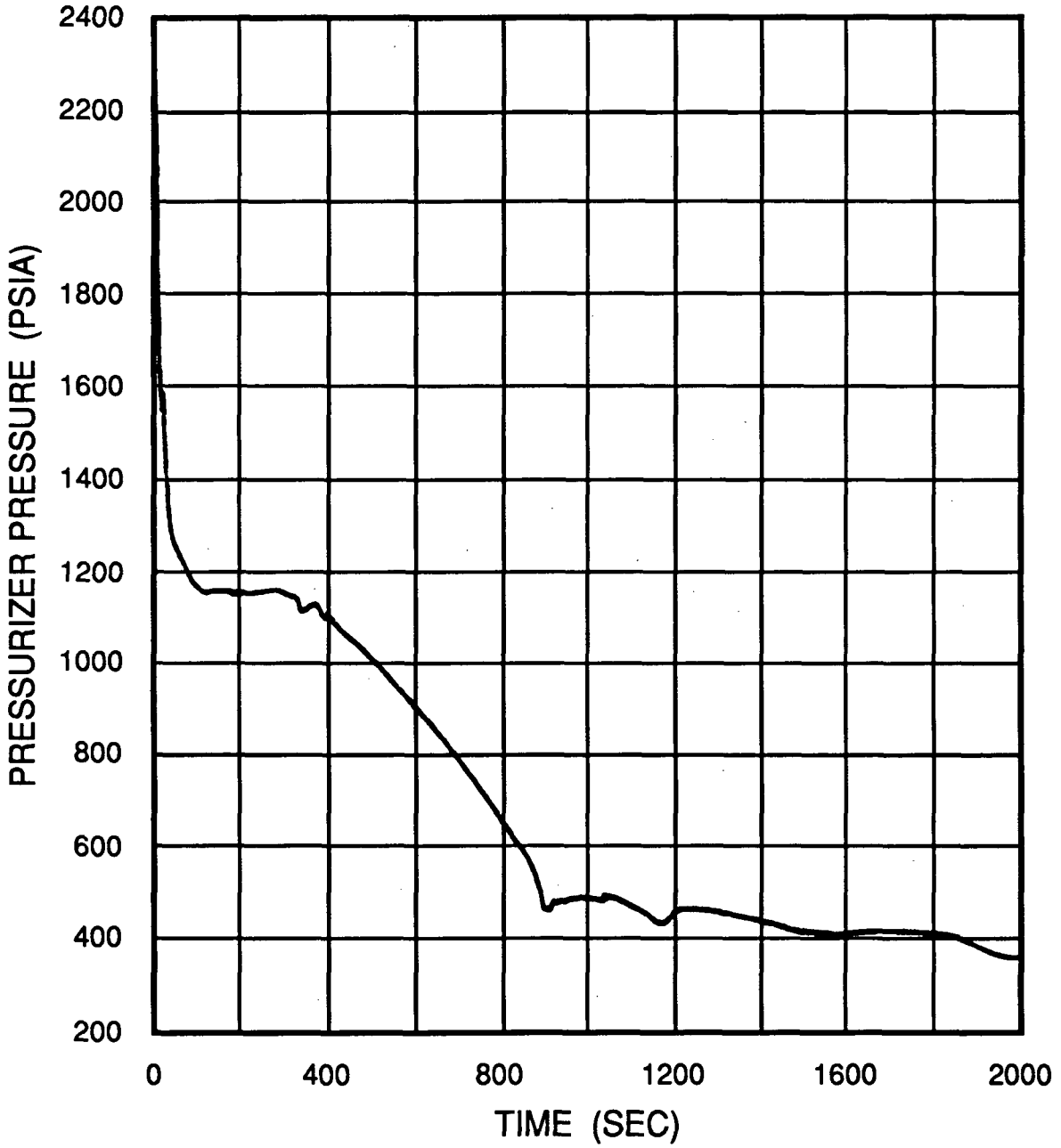


ZION STATION UFSAR

Figure 15.6-105

SMALL BREAK LOCA (3-INCH)
HOT SPOT FLUID TEMPERATURE

JULY 1993

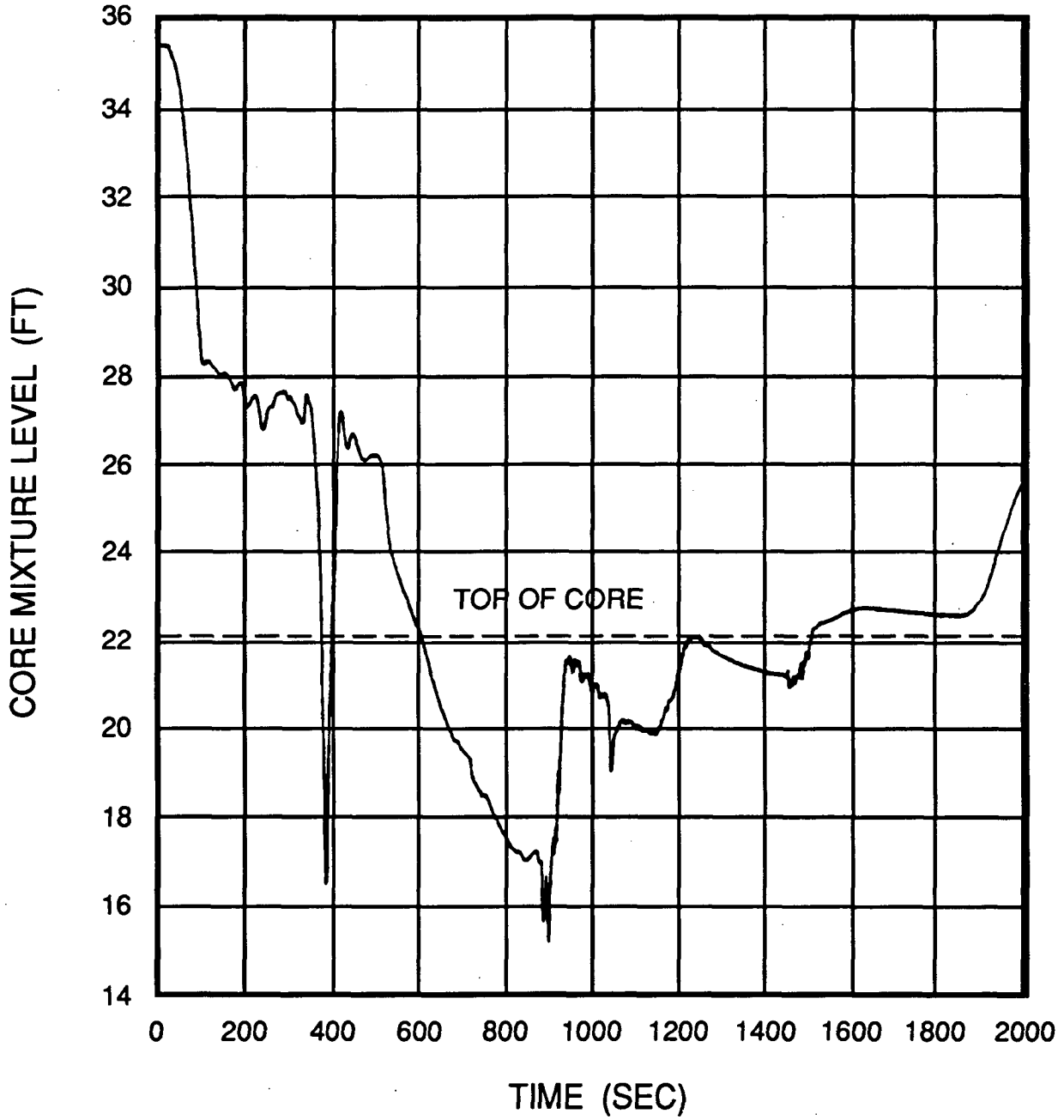


ZION STATION UFSAR

Figure 15.6-106

SMALL BREAK LOCA (4-INCH)
RCS DEPRESSURIZATION

JULY 1993

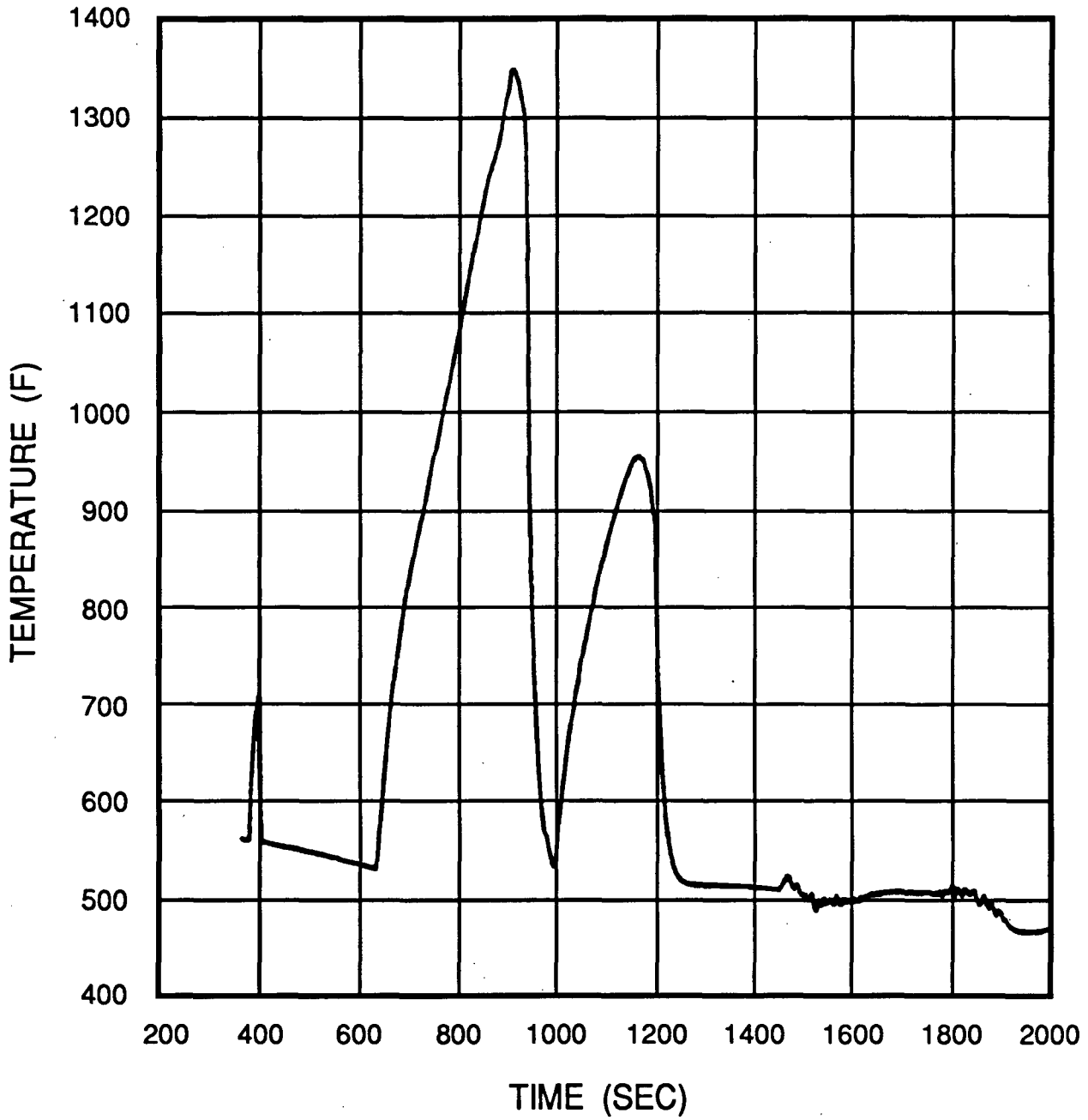


ZION STATION UFSAR

Figure 15.6-107

SMALL BREAK LOCA (4-INCH)
CORE MIXTURE LEVEL

JULY 1993

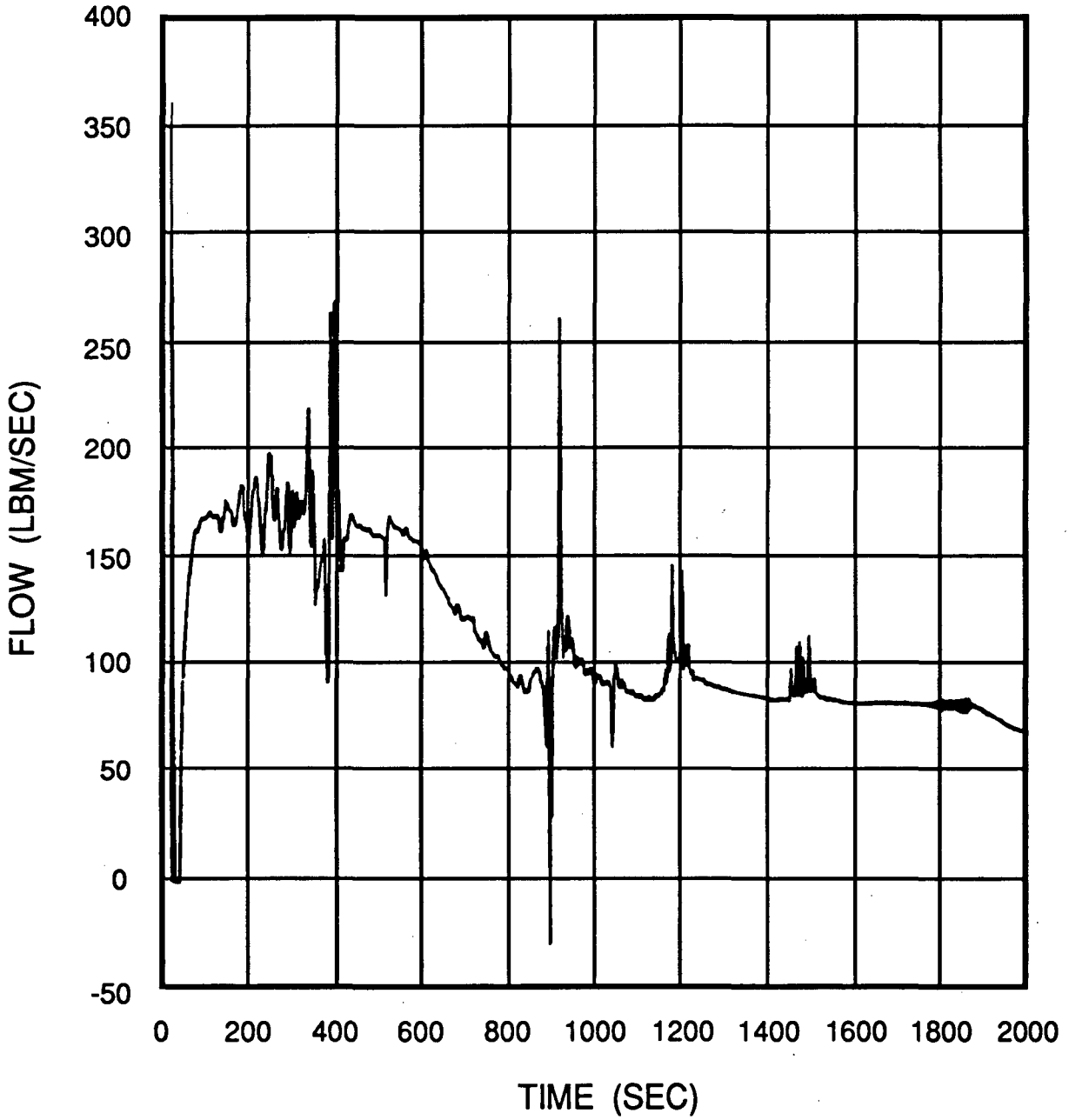


ZION STATION UFSAR

Figure 15.6-108

SMALL BREAK LOCA (4-INCH)
PEAK CLADDING TEMPERATURE

JULY 1993

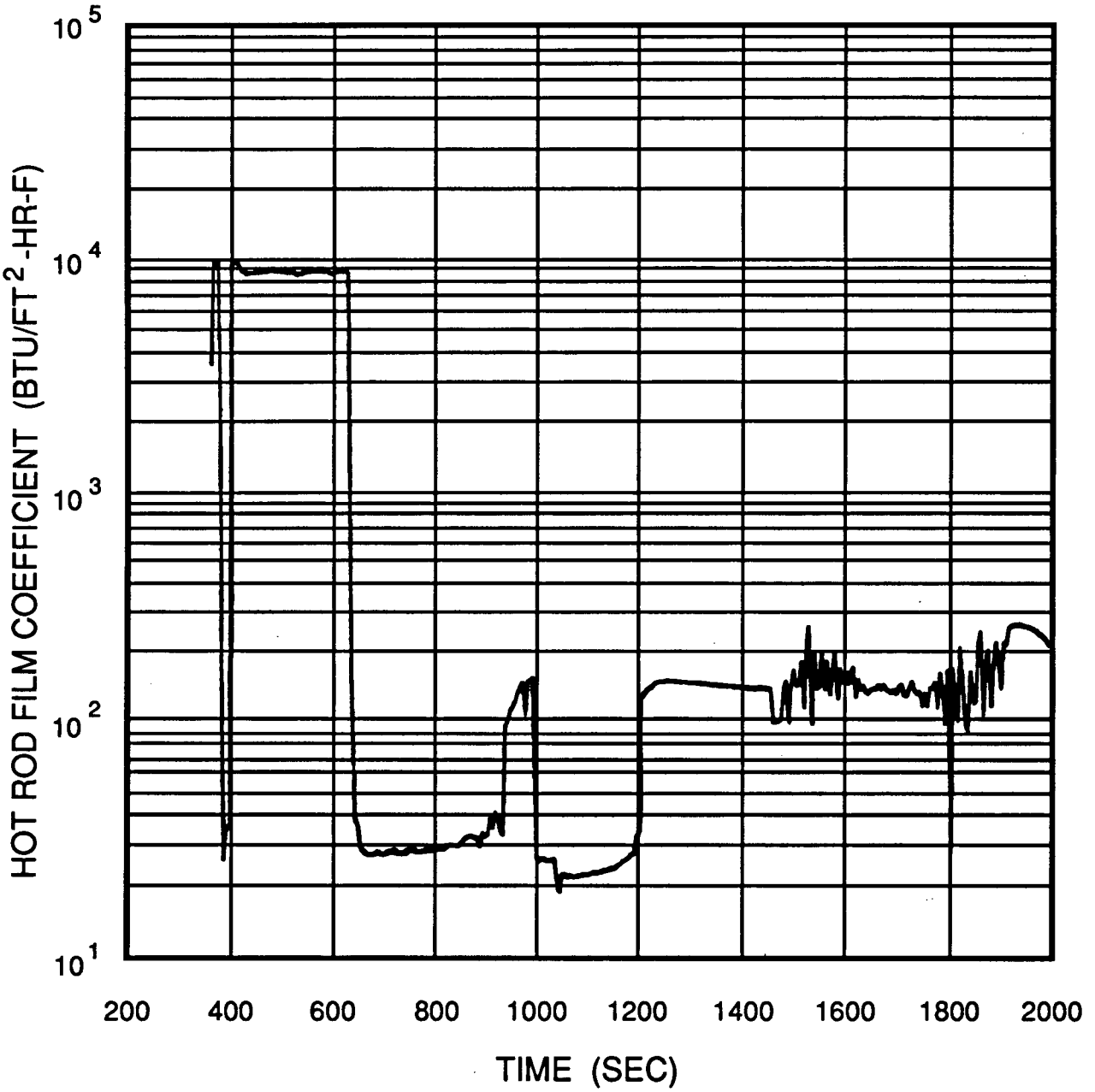


ZION STATION UFSAR

Figure 15.6-109

SMALL BREAK LOCA (4-INCH)
STEAM FLOWRATE

JULY 1993

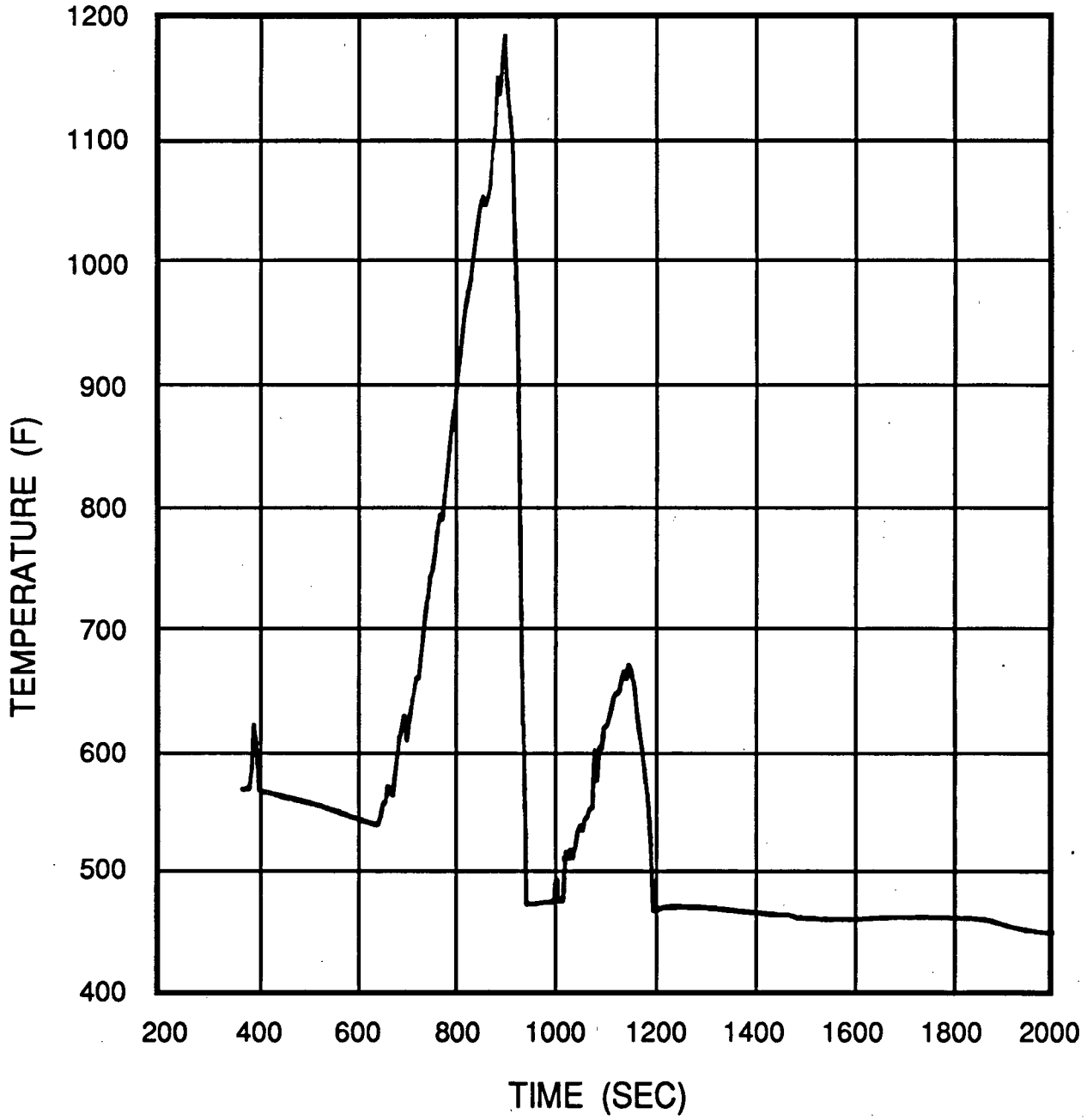


ZION STATION UFSAR

Figure 15.6-110

SMALL BREAK LOCA (4-INCH)
HOT ROD FILM COEFFICIENT

JULY 1993



ZION STATION UFSAR

Figure 15.6-111

SMALL BREAK LOCA (4-INCH)
HOT SPOT FLUID TEMPERATURE

JULY 1993

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

The fault conditions discussed in this section are more severe, but less probable, than those discussed in Sections 15.1, 15.2, 15.3, and 15.4. They may result in a breach of fission product barriers and a release to the environment.

These analyses demonstrate adequate provisions are included in the design of the plant and its engineered safety features. These provisions restrict potential exposure of the public to well below the limits of 10CFR100 guidelines for the following fault conditions:

1. Fuel Handling Accident,
2. Waste Liquid Release, and
3. Waste Gas Release.

Analyses for limiting exposure to the public below the limits of 10CFR100 for additional fault conditions are provided as shown below:

1. Steam Generator Tube Rupture (Section 15.6.3),
2. Steam Pipe Rupture (Section 15.1.5), and
3. Rupture of Control Rod Mechanism Housing-Rod Control Cluster Assembly (RCCA) Ejection (Section 15.4.7).

The offsite exposures given in these analyses are based on calculational techniques used at the time the original FSAR was written. Although present day analyses use different assumptions and models, the resulting doses would probably be less than 10CFR100 limits due to the conservatism of the original values. Therefore, no re-analysis is required.

The effects of fuel densification on the analyses presented in this section are addressed in Appendix 4A.

15.7.1 Accidental Release of Radioactive Gases

15.7.1.1 Volume Control Tank Rupture Analysis

In the event that a rupture should occur in a volume control tank, caused by undetermined means, the two-hour integrated whole body dose at the site boundary during passage of the cloud of escaped gases would be 0.25 rem.

The integrated whole body dose is based on a release from the station with a wind velocity of one meter/second. This assumes inversion stability conditions, Pasquill "F", and allows for dilution in the wake of the building. The calculations also assume 100% release of the noble gas isotopes Xe-133, Xe-135, Kr-85, Kr-85m, Kr-87, and Kr-88 in the tank. The fission product inventory in the tank is based upon operation with defects in 1% of the fuel rods and the activity level in the tank at its maximum. Activity levels are listed in Appendix 15B, Table 15B.6-1.

The inhalation hazard at the site boundary from this accident is negligible due to low concentrations and low volatility of the halogens and other nongaseous isotopes for the temperature and pH of the fluid in the volume control tank. The normal temperature is about 130°F.

The calculations show that the general public will not be exposed to radiation hazards in excess of 10CFR100 limits from this accident.

15.7.1.2 Gas Decay Tank Rupture Analysis

The amount of gaseous activity contained within a gas decay tank is limited to an amount which would result in a whole body exposure of less than 500 mrem to any individual in an unrestricted area following a sudden failure of a tank. The gaseous radioactivity quantities are calculated utilizing the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release Due to a Waste Gas System Leak or Failure."

BTP ETSB 11-5 assumes only noble gases are present in the gas decay storage tanks. The activity limit is calculated for Xe-133 only. All other noble gases present in a gas decay storage tank are converted to Xe-133 equivalence utilizing correction factors obtained from Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I."

The activity is modeled to be released at ground level to the outside environment immediately after system failure. The short-term accident X/Q of 9.2×10^{-4} is used in the dose evaluation.

The amount of Xe-133 equivalent activity which would result in a whole body dose of 500 mrem at the site boundary (415 meters) is calculated to be 58,200 Curies.

No credible missile or singular event can be postulated which would cause the simultaneous failure of all six gas decay tanks. However, if all six tanks were to rupture, the worst possible dose at the site boundary would be 3 rem, which is less than the 25 rem limit set forth in 10CFR100.

15.7.1.3 Rupture of the Seismic Class II Components

The maximum amount of radioactivity which could be released from Seismic Class II components in the gaseous radwaste system is equivalent to 5970 curies of Xe-133. This is based on a failure in the system which allows the activity to escape from both the waste gas compressor and the contents of 4000 feet of one-inch diameter piping. Fuel defects on the order of 1% were assumed in the analysis. The whole body dose at the site boundary from the maximum released activity is 0.24 rem. The activity is assumed to be released at ground level to the outside environment immediately after system failure. The short-term accident χ/Q of 9.2×10^{-4} sec/m^3 was used in the dose evaluation.

The Zion plant is intended to operate within an anticipated range of failed fuel levels. This range has been chosen as representative of expected plant operation. The upper limit of the range is the design basis condition of fuel failure.

The assumptions made for these analyses are as follows:

1. Fuel failure levels of 0.1% to 1.0%, uniformly distributed throughout the core;

2. Minimum of 45-day gas decay tank holdup;

3. Steam generator leak rate of one gallon per minute, for a two-week period, 70% recycle of normally tritiated water, 6500 Mwt load;

4. Thyroid dose from I-131 assumes a cow resides at the nearest point on the site boundary;

5. Two-unit operation; and

6. Tritium calculations are based on 20-year peak plant inventory.

The results of these analyses are presented in Tables 15.7-1 and 15.7-2.

Based on the assumptions given above and assuming the design primary-to-secondary tube leakage in the steam generators of one gpm, the site boundary doses resulting from the pressure-relieving devices being full open for 30 minutes would be 20 mrem to the thyroid and negligible to the whole body.

15.7.2 Accidental Release of Radioactive Liquids

15.7.2.1 Normal Operation

The inadvertent release of a specific quantity of untreated liquid radioactive waste from the Zion Station is not considered to be a credible accident. All liquid waste to be released from the station must pass through one of two lake discharge tanks. The presence in one of these tanks of untreated waste and the subsequent discharge of this waste to the lake could occur only through a series of unusual and severe operator and supervisory personnel errors. These errors are several in number and their occurrence in the proper sequence to allow the release of untreated liquid waste is not credible.

The Liquid Waste Processing System is explained in Section 11.2.1 and is shown in Figures 11.2-1 thru 11.2-7. The processing method for each batch of liquid waste is determined after a sample of the batch has been analyzed. This sample is taken from the waste as collected in analysis tanks. The contaminated waste batches are processed through the demineralizers. The treated water is collected in the evaporator monitor tanks. A bypass line has been provided so that low activity level batches

of waste can be pumped directly to the evaporator monitor tanks. Through an improperly analyzed waste sample or a misalignment of valves, the possibility exists for an operator to accidentally pump a highly contaminated batch of liquid waste directly to one of the evaporator monitor tanks.

From the evaporator monitor tanks, processed liquid waste to be released is pumped to one of two lake discharge tanks. Before any transfer between these two tanks is permitted, the individual monitor tank is sampled, the sample analysis approved by the Non-Licensed Shift Supervisor, and the key locked valve switch between the monitor tank and the lake discharge tank actuated. In order to pump a highly contaminated batch of liquid waste to a lake discharge tank from an evaporator monitor tank, another sample would have to be improperly analyzed and the analysis approved. This, again, is unlikely. If the sample analysis were done incorrectly, the liquid waste could be sent to the Hold Up Tanks and processed through the Boric Acid Feed Demineralizer System or diluted before being transferred to a lake discharge tank.

Once the liquid waste batch is in one of the lake discharge tanks, additional steps must be taken before the waste can be released. The procedures that are involved in the release of a batch of liquid waste from the lake discharge tanks are explained in Section 11.2.1. These procedures consist of additional sample analysis, approval of the analysis by the senior operator or shift engineer and opening of a key locked valve.

If the sample analysis is improperly interpreted and approved, and the valve opened for discharge, the radiation monitor on the discharge line would, upon detecting an abnormally high radiation level, close the discharge valve and alarm in the Control Room. Thus, in order to pump a highly contaminated batch of liquid waste from a lake discharge tank into the circulating water discharge line, the radiation detector in the radioactive system discharge line would have to malfunction so that the valve in this line would not close automatically.

In summary, the release to the lake of a batch of highly contaminated liquid waste would require the following sequence of operator errors and equipment failures:

1. Improper analysis of a liquid waste sample taken from an analysis tank or a misalignment of valves allowing the highly contaminated waste batch to be pumped to an evaporator monitor tank without first being processed through the demineralizer;

2. Improper analysis of a waste sample taken from the evaporator monitor tank allowing the waste batch to be transferred to a lake discharge tank;

3. Improper analysis of a waste sample taken from the lake discharge tank allowing a locked closed valve in the radioactive system discharge line to be opened; and

4. Malfunction of the radiation monitor in the radioactive system discharge line allowing the waste batch to be released to the circulating water discharge line without the automatic closure of a valve which could stop this waste release.

This sequence of errors and malfunction is not credible.

15.7.3 Postulated Radioactive Releases Due to Liquid Tank Failures

15.7.3.1 Primary and Condensate Storage Tanks

The primary water storage tanks can receive processed water from the Liquid Radwaste System. The condensate storage tanks have the potential to receive contaminated water via the main condenser hotwell following a steam generator tube rupture event.

The radioactivity concentrations in each tank are limited to quantities listed in 10CFR20, Appendix B, Table 2, Column 2. This ensures that in the event of a tank rupture, resultant radionuclide concentrations at the nearest potable water supply and the nearest surface water supply in a restricted area will be less than the limits of 10CFR20. No dilution factors are taken into account.

Since the provisions of 10CFR20 are met, no potential health hazard exists in the event of a primary water storage tank rupture.

15.7.3.2 Rupture of the Seismic Class II Components

The evaluation of the potential consequences of a postulated release of all radioactive materials from the Seismic Class II components of the Liquid Radwaste System, concluded that the normal radioactivity level of liquids at these locations is too low to be of consequence. This analysis applies to a postulated double accident; such as a steam generator tube rupture followed by normal operation of Seismic Class I components and a failure of all Seismic Class II components. The major Seismic Class II components involved would be the 20,000-gallon blowdown monitor tanks and the 30,000-gallon lake discharge tanks. A conservatively estimated upper limit of radioactivity in these tanks is 1×10^{-4} mCi/ml, based on the primary system maximum activities listed in Table 15B.4-2 of Appendix 15B. These tanks are located in the basement of the Auxiliary Building (Seismic Class I) at El 542'. Consequently, a liquid spill resulting from a tank rupture would be confined by the basement walls and floor with essentially zero release to the environment.

15.7.4 Fuel Handling Accident in Containment and in the Fuel Building

Two separate analyses have been performed, one for a fuel handling accident which occurs in the Fuel Building and a second for a fuel handling accident which occurs in Containment.

15.7.4.1 Fuel Handling Accident in the Fuel Building

15.7.4.1.1 Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pool (SFP) floor and breaking of all the fuel rods, despite the administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.7.4.1.2 Method of Analysis

The following assumptions are postulated for a calculation of the fuel handling accident:

1. The accident occurs 100 hours following the reactor shutdown; i.e., the earliest time at which spent fuel would be first moved into the SFP;

2. The accident results in breakage of all fuel rods in an assembly;

3. The damaged assembly is the one that has operated at the highest power level in the core region to be discharged;

4. All of the activity in the fuel clad gap of the damaged rods is released to the pool. The gap fractions of the inventory are 10 percent of the noble gases and iodines except for Kr-85 and I-131 for which the gap fractions are 30 percent and 12 percent, respectively. These gap fractions are consistent with Regulatory Guide 1.25 (Reference 2), as adjusted by Reference 3;

5. The SFP water removes a large fraction of the iodine as the fuel assembly gases bubble to the surface of the pool. A pool scrubbing decontamination factor (DF) of 100 is assumed consistent with Regulatory Guide 1.25. Noble gases are not removed by the water;

6. All air from the Fuel Handling Building is routed through the Auxiliary Building Ventilation System charcoal absorber filters which remove 90 percent of the airborne iodine before the air stream is discharged to the environment. This filter DF of 10 is consistent with the NRC SER of October 6, 1992 for Zion Station. The filter efficiency in Technical Specifications is 95% for methyl iodine.

7. The Exclusion Area Boundary (EAB) X/Q is 9.2×10^{-4} sec/M^3 (Tables 15.6-29 and 30);

8. The breathing rate is 3.47×10^{-4} m^3/SEC (from Regulatory Guide 1.25);

9. The adult iodine inhalation thyroid dose conversion factors are from Regulatory Guide 1.25; and

10. The average gamma and beta energies per disintegration for the noble gases are given in Table 15B.3-3.

15.7.4.1.3 Fission Product Inventories

Fission product inventories in an assembly to be discharged are presented in Appendix 15B, Table 15B.3-3. The gap activities have been reduced to take into account the 100-hour decay of fission products. The significant isotopes after 100 hours are I-131, I-132, I-133, Kr-85, Xe-133m, Xe-133 and Xe-131m.

15.7.4.1.4 Iodine Decontamination Factors

An experimental test program (see Reference 1) was conducted to evaluate the extent of the iodine removal from the released gas which takes place as the gas rises through the body of solution in the SFP to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution.

In order to obtain all the necessary information regarding this mass transfer process, a number of small-scale tests were conducted using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry) and data collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression

for iodine decontamination factor (DF) in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large-scale tests were also performed with carbon dioxide. The small-scale carbon dioxide tests also resulted in a mathematical expression for DF in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full-size fuel assembly simulator was fabricated and placed in a deep pool for testing. The gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas and overall DFs were measured as a function of the total gas volume released. These measurements, combined with an analytical expression derived from small-scale tests with carbon dioxide, permitted an insitu measurement of the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large-scale gas releases, the DF for iodine was obtained, using the analytical expression from small-scale iodine testing:

$$DF = 7.3\ e^{0.313t/d}$$

where: t = rise time
d = effective bubble diameter

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the SFP solution and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

With consideration given to the total quantity of gas released from a full assembly, i.e., 6.9 standard cubic feet for the 15 x 15 array, the pool DF for iodine is indicated to be greater than 500 for the 22-foot depth over the fuel storage racks.

The analysis conservatively assumes a DF of 100 for pool scrubbing removal of iodine consistent with Regulatory Guide 1.25 (Reference 2).

15.7.4.1.5 Offsite Exposure

The noble gas released from the breakage of all the fuel rods in the dropped assembly is given in Table 15B.3-3. These quantities of gas discharged from the ventilation system vent under conditions which would yield the maximum ground concentration at the site boundary (415 meters), and a one meter/sec wind, result in a two-hour integrated whole body dose of 4.40 rem. This dose is less than the 25 rem given in 10CFR100.

The iodine isotopes released from the breakage of all the fuel rods in the dropped assembly are given in Table 15B.3-3. Taking into consideration the scrubbing effects of the fuel pool water and the presence of the charcoal filter in the ventilation system, these quantities of I-131, I-132 and I-133 released over a two-hour period under the meteorological conditions given previously would yield a thyroid dose at the site boundary of 32.36 rem. This dose is less than the 300 rem given in 10CFR100.

15.7.4.2 Fuel Handling Accident in Containment

15.7.4.2.1 Accident Description

The fuel handling accident in the Containment occurs during refueling, as a consequence of an undefined event, when a fuel assembly is dropped onto the top of the core. This evaluation references Unit 1, but is also applicable to Unit 2. Technical Specifications describes the Containment status during refueling operations.

The Cumulative Elapsed Time heading used in this section refers to the time that airborne activity has been released to the environment. The cumulative elapsed time, therefore, remains zero until the airborne radioactivity in the Containment Purge System reaches the first Containment Purge System isolation valve. Both the inboard and outboard isolation valves in the Containment Purge System will automatically close upon the detection of high radiation. The above assumption used to define the cumulative elapsed time is conservative because it is assumed that the outboard valve fails to close under a single failure.

The conservative analysis of the sequence of events and systems operation is as follows:

Event	Cumulative Elapsed Time
1. Fuel assembly is being handled by refueling equipment. The assembly drops onto the top of the core. Some of the fuel rods in both the dropped assembly and reactor core are damaged, resulting in the release of radioactive noble gases and gaseous iodine to the reactor coolant. A fraction of the gaseous radioactivity becomes airborne in the Containment, and partially mixes with the Containment atmosphere. The contaminated air enters the intake of the Containment Purge System and reaches the inboard isolation valve	0 seconds
2. The contaminated air in the purge line reaches the continuous air monitor detector 1(2)RE-PR09A, which measures the radioactivity and automatically triggers the electronic logic to initiate containment purge isolation valve closure.	31.5 seconds
3. The containment purge line valves close, 7 seconds (actual observed measured closure times are 5.5 to 7.0 seconds).	39 seconds

The above scenario is the most conservative, since the similar sequence of events associated with the System Particulate, Iodine, and Noble Gas (SPING) radiation monitor 1(2)RIA-PR40 would result in a release duration of 39 seconds.

15.7.4.2.2 Method of Analysis

The conservative input parameters, the initial conditions, and assumptions for this accident are as follows:

1. The radionuclidic activities at shutdown in the highest rated assembly are given in Appendix 15B, Table 15B.3-2;

2. The containment free volume is 2.736×10^6 ft^3;

3. The Containment Purge System is operating at a rate of 40,000 cfm;

4. There is no charcoal filtration system for radioiodine removal within the Containment Purge System;

5. The equivalence of all the gap activity of the highest rated fuel assembly is assumed to be released within the reactor vessel water volume upon impact of the dropped fuel assembly with the core;

6. The accident occurs 100 hours after plant shutdown;

7. The Exclusion Area Boundary (EAB) χ/Q is 9.2×10^{-4} sec/m^3 (Tables 15.6-29 and 30);

8. All the gap activity of the damaged rods is released and consists of 10% of the noble gases and iodines other than Kr-85 and I-131. The gap fractions for Kr-85 and I-131 are 30% and 12% respectively. The basis for the gap fractions is Regulatory Guide 1.25 as adjusted by Reference 3;

9. The noble gas and effective radioiodine reactor vessel water decontamination factors (DF) are 1.0 and 100, respectively (Regulatory Guide 1.25). The equivalent DF of 100 for iodines represents a DF of 133 for elemental iodine and a DF of 1 for organic iodine. The composition of the iodine form is 99.75% elemental iodine and 0.25% organic iodine;

10. The breathing rate is 3.47×10^{-4} m^3/sec (Regulatory Guide 1.25);

11. The airborne activity released into the Containment by the hypothetical accident is mixed with 4% of the containment free volume before being exhausted by the Containment Purge System;

12. The adult iodine inhalation thyroid dose conversion factors are from Regulatory Guide 1.25;

13. It is assumed that the containment internal cleanup (charcoal adsorber) filter units do not function in reducing the airborne radioiodine released from the pool before it is exhausted out of the Containment by the Containment Purge System;

14. The radionuclidic average gamma energies per disintegration are from Table 15.6-28;

15. All required radiation monitors are operating;

16. The analytical methods and associated assumptions used to evaluate the consequences of this accident are based on the conservative assumptions of Regulatory Guide 1.25; and

17. The containment equipment hatch is open;

18. All of the activity not released via the Containment Purge System is assumed to pass through the equipment hatch into the Fuel Handling Building and then to the environment by way of the Fuel Handling Building's ventilation system which is routed through the Auxiliary Building Ventilation System charcoal absorber filters having a 90% removal efficiency for elemental iodine and a 70% removal efficiency for organic iodine.

15.7.4.2.3 System Operation

During refueling, the Containment Purge System is normally operating. This system supplies 40,000 cfm of air through 32 supplying air outlets around the refueling pool at El 617'. Air is exhausted from the Containment through isolation valves, prefilters, HEPA filters and discharged to the atmosphere through the plant vent stack. Each Containment Purge Exhaust and Purge Supply System is equipped with two air-operated isolation valves in series (1(2)AOV-RV0001 through 4). These air-operated valves are designed to fail close on loss of air or electrical signal.

With the equipment hatch open, both the Containment and the Fuel Handling Building should be at the same pressure. As the fuel handling accident would not release a significant amount of energy, there should be no driving force to cause exfiltration of the activity from the Containment other than through the Containment Purge System.

RT-AR04A and RT-AR04B are designed for use during refueling. RT-AR04A and RT-AR04B will detect a fuel handling accident in sufficient time to isolate containment ventilation prior to a gas release from the containment following a fuel handling accident in containment. RT-AR04A and RT-AR04B are gamma detectors mounted on opposite sides of the refueling cavity at approximately EL 617'. These monitors fulfill the requirement as redundant monitors to the containment ventilation purge monitors. Credit for the containment SPING radiation monitor is not taken in the fuel handling accident in containment due to the SPING monitor response time to the postulated event.

RT-AR04A and RT-AR04B are powered from independent AC instrument power buses. RT-AR04A and RT-AR04B outputs are displayed on the radiation monitoring system display panel and alarm in the control room, upon receipt of a high radiation condition. When radiation levels exceed the setpoint, RT-AR04A and RT-AR04B independently provide actuation signal to close the containment purge valves, AOV-RV0001 through AOV-RV0004, and vent valves, AOV-RV0005 and AOV-RV0006. The sensitivity range for RT-AR04A and RT-AR04B is given in Table 12.3-4. RT-AR04A and RT-AR04B are de-energized when not in refueling mode. The monitor's automatic actuation functions are also defeated.

The design environmental operating temperature range is 65-120°F. Based on the system configuration, two independent detectors each controlling all four purge system valves, and separate power supplies, the containment Isolation System is single failure proof.

The purge exhaust pickup point is located 7'-0" from the containment wall at El 634' and Column Z-7. A 3/4-inch sample line located downstream of the containment purge exhaust fans, at El 636'-9" between Columns 23 and 24, and Rows L and M, takes continuous samples of the air being exhausted and routes the samples to the radiation detector 1RE-PR09A. This detector is located in the purge room of the Auxiliary Building at El 617' at approximately the same column and row location as the sample extraction position.

Radiation detector 1RE-PR09A signals are transmitted to bistable comparators. When radiation level signals to the bistable reach the setpoint, 5×10^{-4} μCi/cc, the bistable de-energizes a relay which performs the following functions:

1. Energizes high radiation light (red) locally and in the Control Room,
2. Energizes local and control room annunciator alarm, and
3. Energizes actuation of relays to close all four purge system valves (1AOV-RV0001 through 4), thereby isolating the Containment.

The sensitivity range of 1RE-PR09A is 10^{-6} to 10^{-2} μCi/cc. The design environmental operating temperature range is 65-120°F. This detector is Seismic Class I and is equivalent in design to a safety-related instrument. This radiation detector and its associated circuits are powered by an essential electrical bus.

There is an independent SPING radiation monitor for each unit, 1(2)RIA-PR40, which also measures radiation levels in the Containment and will, upon reaching a high radiation setpoint, initiate closure of all purge valves. For these monitors, continuous air samples are obtained by sampling lines controlled by two-way solenoid valves. The samples are taken at the refueling floor level (El 617') in the Containment at the Containment wall, approximately midway between the purge valves and the personnel hatch. The detectors are located in the Auxiliary Building (between columns 23 and 24, and rows P and R) at El 617'.

The detector signals are transmitted to bistable comparators. When radiation level signals to the bistable reach the setpoint, the bistable de-energizes a relay which performs the following functions:

1. Energizes high radiation light (red) locally and in the Control Room,
2. Energizes local and control room annunciator alarm, and
3. Energizes actuation of relays to close all four purge system valves (1AOV-RV0001 through 4), thereby isolating the Containment.

15.7.4.2.4 Offsite Exposure

The EAB radiological doses are calculated to be 102 rem to the thyroid and 0.6 rem to the whole body. Both calculated doses are well below the 10CFR100 guidelines of 300 rem to the thyroid and 25 rem whole body.

15.7.4.3 Loss of Refueling Cavity Water Level

15.7.4.3.1 Normal Operation

Borated water is used in the SFP, reactor vessel, fuel transfer canal, and refueling cavity during refueling for the following purposes:

1. Provides a means for removing the decay heat from irradiated fuel assemblies;

2. Maintains the irradiated fuel assemblies in a subcritical condition; and

3. Provides shielding from the radiation given off by irradiated fuel assemblies and reactor components.

During a postulated loss of level accident, the loss of decay heat removal from irradiated fuel assemblies could potentially result in fuel failures

During a postulated loss of level accident, the loss of decay heat removal from irradiated fuel assemblies could potentially result in fuel failures (and the release of radioactive contaminants) which could affect the public. Also very important is the loss of shielding because it affects the safety of plant personnel.

The water level in the SFP is maintained such that the radiation level at the surface of the water is approximately 2.5 mR/hr when an irradiated fuel assembly is being moved over the top of the spent fuel racks. During refueling, the dose level is maintained at an acceptable level while moving an irradiated fuel assembly over the reactor flange by ensuring the water level in the refueling cavity and the transfer canal is a minimum of 22 feet above the flange level. The level is normally maintained at El 615' (± 4") which is about 23'10" above the flange.

If a loss of water level in the refueling cavity or the SFP is discovered either visually, by the appropriate level alarms and/or level indication, or by an unexpected level increase in the drain collecting sumps in Containment or the Auxiliary Building or in the auxiliary building equipment drain tank, normal fuel movement is stopped. All assemblies being moved or in interim storage (RCC change fixture) are returned to the core or the SFP, whichever is quicker. Fuel inserts (RCCAs, thimble plugs, etc.) are reinserted into assemblies or lowered into safe positions below the level the water can drain to. The transfer cart is returned to the SFP side of the transfer canal and the transfer tube gate valve is closed. Makeup is then commenced to return the water level to normal. During breaks in fuel movement when personnel are not in immediate attendance, all fuel assemblies and inserts are placed in safe positions.

A total loss of refueling cavity water level during refueling is not considered a credible accident. While possible leakage paths do exist, plant design and operation preclude a failure which would result in a rapid loss of level resulting in fuel uncovery. Only a rapid loss of level could result in the uncovering of fuel assemblies as they are being moved to safe positions from unsafe ones. Paths which would allow a rapid loss of level could exist only if several coincident failures or operator errors were to occur coincident with the movement of assemblies.

15.7.4.3.2 Possible Sources of Water Loss

15.7.4.3.2.1 Spent Fuel Pool

The SFP is normally filled with borated water to around El 615'. Because the SFP is used primarily for semi-permanent storage of irradiated fuel assemblies, it is designed without piping connections which could be used to drain it. The SFP pump suction is taken from strainers at El 611'-8" and the return is piped to El 598', approximately eight feet above the top of the stored fuel assemblies. Rupture of any of this piping cannot cause the pool to be gravity drained to expose irradiated fuel assemblies. The pool

contains a level instrument which annunciates on the main control board when the water level falls below El 614'-8". The pool design also includes a leakoff system which can be used to determine leakage through the pool liner itself.

Makeup water to the SFP is normally supplied from the refueling water storage tank (RWST). Water can be pumped into the pool by the refueling water purification pump which has a design flowrate of 100 gpm. Alternate makeup can be provided from the holdup tanks to the pool via the fuel transfer canal and through the SFP gate. The holdup tank recirculation pump, which is a two speed pump with a capacity of 250/500 gpm, pumps water from any of the tanks into the transfer canal. In an emergency, unborated water can be supplied by connections made to the demineralized flush water system or by a fire hose from the Fire Protection System.

Access to the transfer canal from the SFP is provided by a removable gate the bottom of which is approximately 2'-3" above the top of stored fuel. An inflatable seal pressurized from instrument air is incorporated into the design of the gate to prevent leakage into the canal when it is empty. The gate is removed when the transfer canal is flooded during refueling.

15.7.4.3.2.2 Fuel Transfer Canal

The fuel transfer canal is normally empty, being filled only during refueling. The canal is connected to each unit's refueling cavity by a 20" diameter transfer tube at the ends of the canal. The tubes are centered at approximately El 578'6". Each tube is closed on its refueling cavity side by a bolted flange which is removed only during that unit's refueling operations. A manually operated sluice gate valve on the transfer canal side of each tube is also used to isolate the refueling cavity and the canal. The only other piping connection to the transfer canal is a 3-inch pipe from the hold up tanks which enters the canal at El 616'0" and discharges approximately one foot lower. Through this pipe, water can be pumped into the canal to provide alternate makeup to it and the SFP. The canal is normally filled with borated water from the SFP via a portable pump and made up to from the refueling cavity via the transfer tube or from the SFP via the pool's gate. The canal is emptied via a portable sump pump.

15.7.4.3.2.3 Refueling Cavity

The refueling cavity is only filled with water during refueling when the reactor head is removed. The cavity is normally filled from the RWST via the Residual Heat Removal (RHR) System by "overflowing" the Reactor Coolant System (RCS) as the head is lifted. The cavity is drained in the reverse manner. There are several possible leakage paths directly from the refueling cavity and the RCS. Refueling cavity level is monitored by a level meter in the Control Room and is checked daily per station procedure during fuel movements.

15.7.4.3.2.3.1 Refueling Cavity Drains

There are two drains in the refueling cavity, one at each end of the cavity. These drains are for removing the water which remains in the sections of the cavity lower than the reactor flange following drain down of the cavity. The upper internals laydown area is one foot below the reactor flange level and has a 1-inch drain line from it. This line is drained via a temporary connection to the containment sump. The lower internals laydown area is 10'1.5" lower than the refueling flange, but the cavity drops an additional 4'6" to accommodate the cavity upender and entrance to the transfer tube. This lower area has a 4-inch drain line in it. Drainage via this line is routed to either the reactor coolant drain tank or the containment sump via installed piping. Station procedures for filling the refueling cavity require that the valves on these lines be verified closed and, for the valves on the 4-inch lines to the reactor coolant drain tank and the containment sump, locked closed.

15.7.4.3.2.3.2 Refueling Cavity Boot Seal

Between the reactor vessel flange and the refueling cavity floor is a 1.89-inch gap to allow ventilation to cool the biological shield wall when the reactor is operating. During refueling when the cavity is to be filled, this gap is sealed by a molded rubber bladder boot. The boot is installed before the reactor head is removed and inflated to approximately 30 psi from a dedicated supply of nitrogen gas. The boot seal has been determined to have a minimum safety factor of 8 (see Reference 4). Tests have shown that the seal resists failure even when installed over a 2-inch gap and subjected to eight times the normal load of 25-feet water head with bladder pressures ranging from 0 to 45 psi. Failure of the boot seal is therefore considered extremely unlikely. Punctures of the seal would only cause localized leakage, but would not lead to gross failure.

15.7.4.3.2.3.3 Reactor Vessel Nozzle Inspection Covers

At the bottom of the cavity around the reactor vessel are eight openings which allow inspection of the reactor vessel nozzles and access to the excore neutron detector wells. These are covered during refueling by stainless steel covers which include full face gaskets. The design of the covers, which are each held down by approximately 20 bolts, is such that gross failure is highly unlikely.

15.7.4.3.2.3.4 Reactor Coolant System

The RCS has many piping connections to it. Physical failure of any of this piping or its valves is not considered credible as the conditions during refueling are much less stressful than those during normal reactor operation. Maintenance work, however, might breach the boundaries of the system.

The major (large diameter) piping consists mainly of closed loops which inherently prevent drainage out of the RCS. These loops may be opened for maintenance, but any openings are closed or isolated, typically using the loop stop isolation valves, prior to filling the refueling cavity. Once the cavity is flooded, work which requires breaking into these loops is delayed, if possible, until the cavity is drained.

A fluid-actuated mechanical seal in the appropriate reactor coolant loop nozzle can be used for isolation if an area must be worked on while the cavity is filled and cannot be isolated by inplace equipment. The seal is basically a solid plug with an inflatable seal around it and is designed to be inherently safe. The external pressurization supply is required only to release it, no external connections are needed to maintain it. Consequently, seal failure is unlikely. Even if the seal should somehow fail, water pressure will keep the central plug in place and this will limit leakage to an acceptable amount.

One large diameter piping connection which is not a loop is the connection between the containment recirculation sump and RHR System. Utilized during the recirculation phase of emergency core cooling, the sump is normally isolated from the RHR System by the SI-8811 valve on each train. This valve is electrically interlocked with the RH-8700 valve on the respective train to prevent its being opened when the RHR System is operating normally and RH-8700 valve is open. Failure of this interlock and the full opening of both of these valves will result in a 14-inch minimum diameter drainage path from the RCS (and the refueling cavity) to the containment recirculation sump.

Many smaller lines are connected to the reactor vessel and major RCS piping for instrumentation and sampling. Although these lines generally do not return to the RCS, and therefore, could result in drainage paths from it should they be opened for maintenance and/or their valves misaligned, their small size would limit any drainage to a very slow rate. In this category are the 58 guide tubes which extend from the bottom of the reactor vessel to the seal table for the Incore Flux Mapping System. These tubes extend through the seal table where they are sealed to the guide thimbles. Though the seals are part of the RCS pressure boundary when the reactor is operating, during refueling they are disconnected and replaced with temporary seals after the guide thimbles are withdrawn. Since the seal table is at the same level as the reactor vessel flange, leakage through the guide tubes could only result in drainage of the refueling cavity down to the flange.

15.7.4.3.3 Method of Analysis

The conservative input parameters and the initial conditions and assumptions for this accident are as follows:

1. Refueling operations are being conducted as described in Section 9.1.4;

2. The refueling cavity, fuel transfer canal and SFP are initially filled with borated water to around El 615′ level;

3. The transfer tube between the refueling cavity and the fuel transfer canal is open and the gate between the transfer canal and the SFP is removed;

4. The loss of level is discovered by the alarming of the level instrumentation in the SFP when the water level falls below El 614′8";

5. Four irradiated fuel assemblies are assumed to be in the refueling cavity; two in the RCC change fixture, one raised in the refueling cavity upender, and one raised in the mast over the core;

6. Makeup to the refueling cavity is not available at the time of the accident;

7. All fuel handling equipment continues to operate throughout the accident; and

8. Dose rates were calculated using the Quad Mod G computer code developed by Oak Ridge National Laboratories.

15.7.4.3.4 Consequences of a Loss of Level

A loss of water level in the refueling cavity and SFP has two main effects. The first effect which occurs immediately is the decrease in shielding as the level falls. This can become very significant especially as an assembly approaches being uncovered. The second effect, which can be the most dangerous with regard to limiting the accident, is the loss of decay heat removal capability if the level falls enough. Failure to remove an irradiated fuel assembly's decay heat may lead to fuel failure and radioactive contaminant release.

15.7.4.3.4.1 Exposure of a Fuel Assembly

The reactor vessel and the SFP were designed so that drainage through any of their normally connected piping cannot result in the uncovering of any "stored" irradiated fuel assemblies. Therefore a loss of level in the refueling cavity or SFP, in itself, will not result in the uncovering of irradiated fuel assemblies and possible radioactive releases, although radiation exposure to plant personnel may increase.

During refueling operations, however, the uncovering of irradiated fuel assemblies is possible. When a fuel assembly is being moved in the mast of the manipulator crane, its top is within nine feet of the surface of the

water. Even when assemblies are in interim storage positions such as the RCC change fixture or upright in the upender, approximately 12 inches of the top of the assembly may be uncovered if the cavity level should fall to the reactor flange level. Because of this, all fuel assemblies are to be returned to the core or the SFP when an uncontrolled loss of level is indicated or during breaks in fuel movement when personnel are not in immediate attendance.

As previously noted, the only likely sources of leakage are in the RCS or the refueling cavity. However, with the SFP gate and the transfer tube open to allow the movement of fuel, any loss of water in the refueling cavity will also effect the SFP and transfer canal. It is beneficial to have the SFP gate and transfer tube open, since more water must be drained from the combined volume to decrease the water level a certain amount than if they were isolated. This allows more time to move the fuel assemblies into safe positions. Although the decreased level in the SFP would result in increased dose rates, once all assemblies are in safe positions, the transfer tube gate valve can be closed to isolate the leak, and the pool refilled to reduce the dose rates.

Upon detection of a loss of level, the cavity level will be at El 614'8". The top of a fuel assembly while it is being moved by the manipulator crane is at El 606'4". Therefore, in order to uncover a fuel assembly which is being moved, the level in the refueling cavity must drop by 8'4". With the refueling cavity, fuel transfer canal and SFP connected, more than 256,000 gallons of water must be drained to lower the water level this amount.

The four fuel assemblies which are out of the core at the time the level starts falling would typically require about 18 minutes 35 seconds to be placed in safe positions, but after 15 minutes 25 seconds all the assemblies would be below El 606'4". (See Table 15.7-3 for a representative time analysis). The final position of the fuel assemblies would be as follows:

1. The fuel assembly from the upender is placed in the SFP;
2. The fuel assembly in the manipulator crane mast is placed in the core;
3. One of the assemblies in the RCC change fixture is placed in the core; and
4. The other assembly in the RCC change fixture is horizontal in the transfer cart by the SFP upender.

In this configuration, all assemblies are safe from being uncovered although the assembly in the transfer cart is not in an actual storage position. Placing this last assembly in storage would extend the time an assembly is at El 606'4" thus increasing the risk of exposure. After the assembly in the manipulator crane has been lowered into the core, the transfer tube gate valve can be closed isolating the transfer canal and SFP from further drainage. Refilling the pool can then be initiated. The

assembly in the transfer cart can be moved to a storage position in the pool when the water level has increased to an appropriate depth.

Since the shortest time needed to move the four assemblies to below El 606'4" on route to safe positions is 15 minutes 25 seconds, exposure of an assembly will occur if the water drains below El 606'4" within that time. For this to happen, the combined refueling cavity, transfer canal and SFP would have to drain at an average of more than 16,600 gpm.

Of the possible drainage paths from the refueling cavity, most are too small to result in this large a drainage rate. As an example, if the 1-inch drain from the upper internals laydown area should be opened, the flow rate through it with a 25-foot water depth (actually El 615'0"-590'1.5") would initially be 108 gpm and if the 4-inch drain from the cavity upender well were opened, the flow rate with a 39-foot water depth (actually El 615'0"-576'6") would initially be 1990 gpm. Even the 2-inch loop drain line from the lowest point in the RCS piping, the crossover leg, would initially have a flowrate of only about 510 gpm with a 42'3" head (El 615'0"-572'9").

The size of the opening needed to result in an average drainage rate of 16,600 gpm for a level drop from El 614'8" to El 606'4" is about 150 square inches at the reactor flange level, 129 square inches at the centerline of the nozzles, or 108 square inches at the lowest point in the RCS. This corresponds to approximately 11.6% of the refueling cavity boot seal failing completely, the failure of a 12.8-inch internal diameter pipe at the reactor nozzle or the failure of an 11.7-inch internal diameter pipe at the lowest point in the RCS.

15.7.4.3.4.2 Decrease of Radiation Shielding

Gamma exposure rates are reduced by a factor of two for every 3.84 inches of water shielding,i.e., 3.84 inches of water is one half thickness. Water is used to shield plant personnel from the radiation given off by irradiated fuel assemblies and reactor components. The major reactor components which must be shielded are the upper and lower internals. Calculation of the exposure rates for each of the internals was performed given the following conditions:

1. Cavity water level at El 615'8" (normal refueling level);

2. Cavity water level at El 602'7" (at the top of the lower internals' thermal shield when in its stand);

3. Cavity water level at El 599'3.5" (one meter below the top of lower internals' thermal shield when in its stand);

4. Cavity water level at El 596'0" (two meters below the top of the lower internals' thermal shield when in its stand);

5. Cavity water level at El 592'8.5" (three meters below the top of the lower internals' thermal shield when in its stand); and

6. Cavity water level at El 591'1.5" (at the reactor flange level - 3.5 meters below the top of the lower internals' thermal shield when in its stand).

Discussions of the resultant exposure rates from the upper internals, lower internals, reactor core (lower internals in place), and the SFP from these reduced water levels are presented in the following sections.

15.7.4.3.4.2.1 Upper Internals

The upper internals are removed every refueling to allow access to the fuel assemblies in the core. Because the upper internals are situated above the core, only the lower support plate is highly activated. Radiation exposure calculations for the previously mentioned conditions were done for the following personnel locations:

1. On the bridge of the manipulator crane;
2. At the railing around the refueling cavity at the closest point to the lower internals laydown area; and
3. In the polar crane cab at the closest point to the lower internals laydown area.

When the water level is normal, the exposure rate caused by the upper internals at each of the three points is negligible. With the water level at the lowest point, the exposure rate received from the upper internals is 195 mR/hr at point 1, 85.3 mR/hr at point 2, and 25.8 mR/hr at point 3. For a complete summary of the exposure rates at each point for the different water level conditions, see Table 15.7-4.

15.7.4.3.4.2.2 Lower Internals

The lower internals are normally removed for the inservice inspection of the reactor vessel every ten years. Because of its close proximity to the core, the lower internals is highly activated with the thermal shield as the dominant source. Exposure rates from the lower internals were calculated for the following points:

1. On the bridge of the manipulator crane;
2. At the railing around the refueling cavity at the closest point to the lower internals;
3. Behind the missile barrier around the steam generators;
4. In the polar crane cab at the closest point to the lower internals; and
5. At the containment upender control panel.

The exposure rate received at each of the five points from the lower internals is negligible when the water level is normal. With the water level at its lowest point, the exposure rate received from the lower internals is 40.8 R/hr at point 1, 83.7 R/hr at point 2, and 900 mR/hr at point 4. The exposure rates at points 3 and 5 are negligible for all conditions. A complete summary of the exposure rates at each point for the different water level conditions is included in Table 15.7-5.

15.7.4.3.4.2.3 Reactor Core

The fuel which is in the reactor core, though highly activated, would provide only a small radiation exposure for most accidents. The top of a fuel assembly in the core is at approximately El 579'10". For all drainage paths outside of the RCS, the water level at the most would drop only to the reactor flange level. Fuel assemblies in the core would therefore remain covered by 11'3" of water (actually El 591'1.5"-579'10") which is 35 (135"/3.84") half thicknesses. The resultant exposure rate is relatively insignificant when compared to the exposure rate which results from exposure of the upper internals while positioned in the upper internals stand.

For leakage from the RCS however, the water level could drop as low as the bottom of the nozzles. In this condition, the exposure rate from the fuel assemblies becomes significant as only 35.5 inches of water would remain above the top of the fuel. The shielding afforded by this depth of water only reduces the gamma exposure rate by a factor of about 600.

15.7.4.3.4.2.4 Spent Fuel Pool

As mentioned earlier, with the transfer tube and SFP gate open, a loss of water level in the refueling cavity will also cause a loss of level in the SFP. Starting from a worst case condition, all fuel assemblies can be placed in safe positions in 18 minutes 35 seconds. Assuming the transfer tube gate valve takes an additional 2 minutes 30 seconds to close, the SFP would drain for 21 minutes 5 seconds. With a 16,600 gpm leak, the SFP level would drop about 11'5" to El 603'3" in this time. Since the top of a stored fuel assembly is at El 590'-$1^{1}/_{10}$", this loss would leave 13'-$1^{3}/_{4}$" above the fuel. The top of a typical fuel assembly has been measured at 1000 R/hr on contact. The 13'-$1^{3}/_{4}$" of water is 41.1 (157.75"/3.84") half thicknesses, which results in a gamma exposure rate of 4.2×10^{-7} mR/hr at the water surface.

Even if a catastrophic leak should occur in the refueling cavity, the SFP level can, at the most, only drop to El 592'4", the level of the bottom of the SFP gate, if the transfer tube gate valve was left open. At this level, 2'-$2^{9}/_{10}$" (El 592'4"-590'-$1^{1}/_{10}$") of water would remain above the fuel assemblies. This depth of water is 7.01 (26.9"/3.84") half thicknesses, which results in a gamma exposure rate of 7.78 R/hr at the surface of the water.

During periods of normal reactor operation, the transfer tube gate valves are closed and their flanges installed. The transfer canal is usually drained. The SFP gate is in place and its seal inflated. If the gate and/or its seal should have a catastrophic failure, the SFP would lose 6'11" of level to fill and equalize levels with an empty transfer canal. Assuming the water level in the spent fuel pool starts at its lowest normal level (El 614'8"), the level would fall to El 607'9". The fuel assemblies in the pool would remain covered by 17'-$7^{9}/_{10}$" (El 607'9"-590'-$1^{1}/_{10}$") of water which is 55.18 (211.9"/3.84") half thicknesses and would yield a final gamma exposure rate of 2.45×10^{-11} mR/hr.

The dose rates in this section are based on the dose generated by one assembly and are intended to be approximate. Actual doses would depend upon the number and activity of the fuel assemblies as well as the geometry and content of the shielding materials.

15.7.5 Spent Fuel Cask Drop Accidents

Reference Appendix 9A for incident analysis.

15.7.6 References, Section 15.7

1. D. Malinowski, M.J. Bell, E.R. Duhn, Topical Report - "Radiological Consequences of a Fuel Handling Accident," WCAP-7518, December 1971.

2. Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 23, 1972.

3. Federal Register / Vol 53, No. 39 / Monday, February 29, 1988 / pages 6040 through 6043.

4. "Experimental Verification of the RPV/Cavity Liner Annulus Seal for Zion Nuclear Generating Facility," Impell Corporation, Report 03-590-1098, February 1985.

TABLE 15.7-1

DESIGN ESTIMATES OF ANNUAL SITE BOUNDARY DOSES
0.1 % Failed Fuel

	Method	Major Isotope	MPC (μCi/ml)	% of MPC*
1.	Containment purge	Xe-133	3×10^{-7}	6.3×10^{-5}
		Xe-135	1×10^{-7}	2.2×10^{-6}
		Kr-85	3×10^{-7}	1.46×10^{-6}
		H-3	2×10^{-7}	1×10^{-3}
		Whole Body Dose Rate = 0.055 mrem/yr		
2.	Auxiliary Bldg Ventilation System	Xe-133	3×10^{-7}	12.94×10^{-6}
		Xe-135	1×10^{-7}	4.0×10^{-7}
		Kr-85	3×10^{-7}	2.66×10^{-7}
		H-3	2×10^{-7}	8.0×10^{-5}
		Whole Body Dose Rate = 0.005 mrem/yr		
3.	Condenser Air Ejector (during steam generator leak)	Xe-133	3×10^{-7}	3.73×10^{-4}
		Xe-135	1×10^{-7}	3.6×10^{-5}
		Kr-85	3×10^{-7}	8×10^{-6}
		Kr-88	2×10^{-8}	1.2×10^{-4}
		H-3	2×10^{-7}	Negligible
		I-131	1×10^{-10}	1.31×10^{-7}
		Whole Body Dose Rate = 0.05 mrem/yr		
		Thyroid Dose Rate = 0.04 mrem/yr		
4.	Gas Decay Tanks	Kr-85	3×10^{-7}	2.4×10^{-4}
		Xe-133	3×10^{-7}	2.67×10^{-3}
		Whole Body Dose Rate = 0.005 mrem/yr		

* Individual isotopic MPCs are not used to establish our releases.

TABLE 15.7-2

DESIGN ESTIMATES OF ANNUAL SITE BOUNDARY DOSES
1.0 % Failed Fuel

Method	Major Isotope	MPC (μCi/ml)	% of MPC*
1. Containment purge	Xe-133	3×10^{-7}	2.58×10^{-4}
	Xe-135	1×10^{-7}	8×10^{-6}
	Kr-85	3×10^{-7}	5.33×10^{-6}
	H-3	2×10^{-7}	1.6×10^{-3}
	Whole Body Dose Rate = 0.1 mrem/yr		
2. Auxiliary Bldg Ventilation System	Xe-133	3×10^{-7}	12.94×10^{-5}
	Xe-135	1×10^{-7}	4.0×10^{-6}
	Kr-85	3×10^{-7}	2.66×10^{-6}
	H-3	2×10^{-7}	8.0×10^{-4}
	Whole Body Dose Rate = 0.05 mrem/yr		
3. Condenser Air ejector (during steam generator leak)	Xe-133	3×10^{-7}	3.73×10^{-3}
	Xe-135	1×10^{-7}	3.6×10^{-4}
	Kr-85	3×10^{-7}	8×10^{-5}
	Kr-88	2×10^{-8}	1.2×10^{-3}
	H-3	2×10^{-7}	Negligible
	I-131	1×10^{-10}	1.31×10^{-6}
	Whole Body Dose Rate = 0.5 mrem/yr		
	Thyroid Dose Rate = .4 mrem/yr		
4. Gas Decay Tanks	Kr-85	3×10^{-7}	2.4×10^{-3}
	Xe-133	3×10^{-7}	2.67×10^{-4}
	Whole Body Dose Rate = 0.05 mrem/yr		

* Individual isotopic MPCs are not used to establish our releases.

TABLE 15.7-3 (1 of 2)

TIME ANALYSIS FOR REPOSITIONING FOUR SPENT FUEL ASSEMBLIES

Initial Conditions:

One fuel assembly in the manipulator crane mast over the core.
One fuel assembly upright in transfer cart in cavity upender.
Two fuel assemblies in the RCC change fixture.
T=0 at receipt of alarm indicating low water level at El 614'8".

A. Assembly in Cavity Upender

Lower Cavity Upender and Transfer Cart	:35
Send Transfer Cart to Spent Fuel Pool	2:45
Raise transfer Cart in SFP Upender	:37
Lower Fuel Handling tool and Latch Fuel Assembly	1:15
Remove Fuel Assembly from Transfer Cart	:41
Lower SFP Upender and Transfer Cart	:37
Return Transfer Cart to Refueling Cavity	2:45
Raise Transfer Cart in Cavity Upender	:35
Total	9:50*

B. Assembly in Manipulator Crane Mast (concurrent with A. above)

Insert Fuel Assembly in Mast into Core	3:10
Withdraw Mast from Core	1:50
Move Manipulator Crane from Core to RCC Change Fixture	1:39
Lower Mast into RCC Change Fixture and Latch Assembly	:46
Raise Fuel Assembly into Mast	:46
Move Manipulator Crane over Cavity Upender	:14
Total	8:25

(further movement must await arrival of transfer cart)

C. Last Assembly from RCC Change Fixture (subsequent to A. and B. above)

Lower Fuel Assembly in Mast into Cavity Upender	1:15
Raise Mast out of Cavity Upender	:41
Move Manipulator Crane over RCC Change Fixture	:14
Remove Last Assembly from RCC Change Fixture and Insert into Core	6:35
Total	8:45*

* Critical Path Time

TABLE 15.7-3 (2 of 2)

TIME ANALYSIS FOR REPOSITIONING FOUR SPENT FUEL ASSEMBLIES

D. First Assembly from RCC Change Fixture (concurrent with C. above)

Lower Cavity Upender and Transfer Cart	:35
Send Transfer Cart to SFP	2:45
Total	3:20

(This assembly can be left in this position as further movement may result in increased time an assembly is at El 606'4" level)

Final Conditions:

One Fuel Assembly in the SFP.
Two Fuel Assemblies in the Core.
One Fuel Assembly in Horizontal Transfer Cart at SFP Upender.

Critical Path Operations

Total Critical Path Time (A and C)	18:35
- Time to Insert Assembly Fully into Core	3:10
= Last Time Fuel Assembly at the El 606'4" Level	15:25

Basis (average of actual fuel movements)

Cavity Upender	to raise or lower	0:35 min
SFP Upender	to raise or lower	0:37 min
Transfer System (Cart)	cavity to SFP	2:45 min
Manipulator Crane	place fuel from mast into upender	1:15 min
	withdraw fuel from upender into mast	0:41 min
	place or withdraw fuel from/to mast to/from RCC fixture	0:46 min
	place fuel from mast into core	3:10 min
	withdraw fuel from core to mast	1:50 min
	travel upender to core	1:25 min
	travel core to RCC fixture	1:39 min
Total time	from upender into core	7:08 min
	from RCC fixture into core	6:35 min

TABLE 15.7-4

UPPER REACTOR INTERNALS RADIATION EXPOSURE CALCULATION RESULTS

Case	Water Level El	Radiation Exposure Rate from Upper Internals (mR/hr)		
		Refueling Bridge	Cavity Railing	Polar Crane Cab
Case 1:	Normal	Negligible	Negligible	Negligible
Case 2:	602'7"	0.548	Negligible	0.011
Case 3:	599'3.5"	147	6.25	6.61
Case 4:	596'0"	180	34.9	13.6
Case 5:	592'8.5"	191	58.8	19.6
Case 6:	591'1.5"[1]	195	85.3	25.8

[1] Water at Reactor Vessel Flange Level

TABLE 15.7-5

LOWER REACTOR INTERNALS RADIATION EXPOSURE CALCULATION RESULTS

Case	Water Level El	Radiation Exposure Rate from Thermal Shield (R/hr)				
		Refueling Bridge	Cavity Railing	Behind Missile Barrier	Polar Crane Cab	Upender Control Panel
Case 1:	Normal	Negligible	Negligible	Negligible	Negligible	Negligible
Case 2:	602'7"[1]	1.3	13.5	Negligible	.9	Negligible
Case 3:	599'3.5"[2]	16.1	41.8	Negligible	.9	Negligible
Case 4:	596'0"[3]	25.5	60.5	Negligible	.9	Negligible
Case 5:	592'8.5"[4]	32.4	73.0	Negligible	.9	Negligible
Case 6:	591'1.5"[5]	40.8	83.7	Negligible	.9	Negligible

(1) Water at top of thermal shield

(2) Water 1 meter below top of thermal shield

(3) Water 2 meters below top of thermal shield

(4) Water 3 meters below top of thermal shield

(5) Water at Reactor Vessel Flange

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

APPENDIX 15A: VARIOUS ASPECTS OF ECCS PERFORMANCE

This Appendix has been deleted.

See information provided in Section 15.6.5.

APPENDIX 15B: RADIATION SOURCES

NOTE: This document was retyped for clarity in the 1992 UFSAR Update.

APPENDIX 15B

RADIATION SOURCES

This appendix presents the quantities of radioactive isotopes present in the core, fuel rod gap, coolant, volume control tank, gas decay tank and recirculated sump water. A brief discussion of the derivations is also provided.

15B.1 ACTIVITIES IN THE CORE

The total core activity calculation is consistent with TID 14844[1] and data from ORNL-2127.[2] Numerical values for isotopes which are important as health hazards are given in Table 15B.2-1.

15B.2 ACTIVITIES IN THE FUEL ROD GAP

The computed gap activities are based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. For the purposes of this analysis, the fuel pellets were divided into five concentric rings each with a release rate dependent on the mean fuel temperature within that ring. The diffusing isotope is assumed present in the gas gap when it has diffused to the boundary of its ring.

The diffusion coefficient, D′, for Xe and Kr in UO_2, varies with temperature through the following expression:

$$D'(T) = D'(1673) \exp \frac{-E}{R} \left(\frac{1}{T} - \frac{1}{1673}\right)$$

where

E	=	activation energy
D′(1673)	=	diffusion coefficient at 1673°K = 1×10^{-11} sec $^{-1}$
T	=	temperature in °K
R	=	gas constant

The above expression is valid for temperatures above 1100°C. Below 1100°C fission gas release occurs mainly by two temperature independent phenomena, recoil and knockout, and is predicted by using D′ at 1100°C. The value used for D′ (1673°K), based on data at burnups greater than 10^{19} fissions/cc, accounts for possible fission gas release by other mechanisms and pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes is assumed to be the same as for Xe and Kr. Toner and Scott[3] observed that iodine diffuses in UO_2 at

about the same rate as Xe and Kr and has about the same activation energy. Data surveyed and reported by Belle[4] indicates that iodine diffuses at slightly slower rates than do Xe and Kr.

For a full core cycle at 3391 MWt, the above analysis results in a pellet-clad gap activity of less than 3% of the dose equivalent equilibrium core iodine inventory. The noble gas activity present in the pellet-clad gap and assumed release to the containment is about 2.5% of the core inventory.

The percentage of the total core activity present in the gap for each isotope is also listed in Table 15B.2-1.

The core temperature distribution used in this analysis, based on hot channel factors, $F_{\Delta H}$ = 1.70 and F_q = 2.82, is presented in Table 15B.2-2.

15B.3 FUEL HANDLING SOURCES

The source term for the Fuel Handling Accident is based on the equivalent of one complete fuel assembly being damaged so as to release the gap inventory of noble gases and iodines. The damaged fuel rods are all assumed to have been operating at the maximum radial peaking factor of 1.65. The gap fractions are conservatively assumed to be those from Regulatory Guide 1.25[5] (10 percent for all nuclides except for Kr-85, which has a gap fraction of 30 percent) with the exception of I-131, which has an assumed gap fraction of 12 percent consistent with Reference 6. Assumptions used in calculating the source term of the Fuel Handling Accident are summarized in Table 15B.3-1. Based on the End-Of-Life (EOL) core fission product inventories presented in Table 15.6-32 and correcting for the licensed power rating of 3250 MWt, the resulting Fuel Handling Accident in containment source term at time of shutdown is given in Table 15B.3-2. The corresponding source term for the Fuel Handling Accident in the Fuel Building is given in Table 15.3-3. As discussed in Section 15.7.4, the Fuel Handling Accident analysis also takes into consideration radioactive decay during the 100 hours between shutdown and the time of the accident.

15B.4 REACTOR COOLANT FISSION PRODUCT ACTIVITIES

The parameters used in the calculation of the reactor coolant fission product inventories together with the pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are summarized in Table 15B.4-1; while the results of the calculations are presented in Table 15B.4-2. In these calculations the defective fuel rods were assumed to be present at the initial core loading and were uniformly distributed throughout the core. Thus, the fission product escape rate coefficients were based upon the average fuel temperature. The calculations were performed for the prevailing temperature downstream of the regenerative heat exchanger. The coolant density correction of 0.733 should be made in order to obtain the correct activities of the reactor operating temperature.

The fission product activities in the reactor coolant during operation with small cladding defects (fuel rods containing pinholes or fine cracks) in the 1% of the fuel rods were computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{wi}}{dt} = Dv_i N_{C_i} - (\lambda_i + Rn_i + \frac{B'}{B_o - tB'}) N_{wi}$$

for daughter nuclides in the coolant,

$$\frac{dN_{wj}}{dt} = Dv_j N_{C_j} - (\lambda_j + Rn_j + \frac{B'}{B_o - tB'}) N_{wj} + \lambda_i N_{wi}$$

where:

N	=	population of nuclide
D	=	fraction of fuel rods having defective cladding
R	=	purification flow, coolant system volumes per sec.
B_o	=	initial boron concentration, ppm
B′	=	boron concentration reduction rate by feed and bleed, ppm per sec
n	=	removal efficiency of purification cycle for nuclide
λ	=	radioactive decay constant
v	=	escape rate coefficient for diffusion into coolant

Subscript C refers to core

Subscript w refers to coolant

Subscript i refers to parent nuclide

Subscript j refers to daughter nuclide

15B.5 REACTOR COOLANT TRITIUM SOURCES

GENERAL DISCUSSION

During the fissioning of uranium, tritium atoms are generated in the fuel at a rate of approximately 8×10^{-5} atoms per fission (1.05×10^{-2} curries/mwt - day). Other sources of tritium include neutron reactions with boron (in the coolant for shim control), neutron reactions with lithium (utilized in the coolant for pH control, and produced in the coolant neutron reactions with boron), and by neutron reactions with naturally occurring deuterium in light water.

A. Release of Ternary Produced Tritium

The tritium formed by ternary fission in uranium fueled reactors, can be retained in the fuel, accumulate in the void between the fuel and cladding, react with cladding material (zirconium tittide), or diffuse through the cladding into the coolant. Operating experience at the Shippingport reactor (zirconium clad) indicated that less than 1% of the ternary produced tritium is released to the reactor coolant. In order to insure adequate sizing of liquid waste treatment facilities, WNES conservatively assumes that 30% of the ternary produced tritium is released to coolant. This assumption then requires that the waste treatment system be sized to process approximately 4 reactor coolant system volumes in addition to normal reactor plant liquid wastes. Anticipated ternary tritium losses to the reactor coolant is 1%.

B. Tritium Produced from Boron Reactions

The neutron reactions with boron resulting in the production of tritium are:

B^{10} (n, 2α) T
B^{10} (n, α), Li^7 (n, n α) T
B^{11} (n, T) Be^9
B^{10} (n, d) Be^9 (n, α) T

Of the above reactions, only the first two contribute significantly to the tritium production. The B^{11} (n, T) Be^9 reaction has a threshold of 14 Mev and a cross section of ~ 5 mb, since the number of neutrons produced at this energy are less than 10^9 n/cm^2-sec the tritium produced from this reaction is negligible. The B^{10} reaction may be neglected, since Be^9 has been found to be unstable.

C. Tritium Produced from Lithium Reactions

The neutron reactions with lithium resulting in the production of tritium are:

Li^7 (n, nα) T
Li^6 (n, α) T

In the WNES designed reactors, lithium is used to maintain the reactor coolant pH at ~ 9.5. The reactor coolant is maintained at a maximum level of 2.2 ppm lithium. A cation demineralizer is included in the Chemical and Volume Control System to remove the excess lithium produced in the B^{10} (n, α) Li^7 reactions.

The Li^6 (n, α) T reaction is controlled by limiting the Li^6 impurity in the lithium used in the reactor coolant and in lithiating the demineralizers to less than 0.0001 parts of Li^6.

This limitation has been in effect on WAPD designed reactors since 1962.

D. Tritium Production from Deuterium Reactions

Since the amount of naturally occurring deuterium is less than 0.00015 the tritium produced from this reaction is negligible; less than 1 curie per year.

E. Tritium Sources from the Reactor Employing Ag-In-Cd Absorber Rods

Basic Assumptions and Plant Parameters:

1.	Core thermal power	3391 MWt
2.	Plant load factor	0.8
3.	Core volume	1153 ft^3
4.	Core volume fractions	
	a. UO_2	0.3023
	b. Zr + SS	0.1035
	c. H_2O	0.5942
5.	Initial reactor coolant boron level	
	a. Initial cycle	840 ppm
	b. Equilibrium cycle	1200 ppm
6.	Reactor coolant volume	12,760 ft^3
7.	Reactor coolant transport times	
	a. In-core	0.77 sec
	b. Out-of-core	10.87 sec
8.	Reactor coolant peak lithium level	
	(99.9% pure Li^7)	2.2 ppm
9.	Core averaged neutron fluxes:	n/cm^2 -sec
	a. $E > 6$ Mev	2.91×10^{12}
	b. $E > 5$ Mev	7.90×10^{12}
	c. 3 Mev $\leq E \leq$ 6 Mev	2.26×10^{13}

c. 3 Mev $\leq$ E $\leq$ 6 Mev 2.26×10^{13}

d. 1 Mev $\leq$ E $\leq$ 5 Mev 5.31×10^{13}

e. E < 0.625 ev 2.26×10^{13}

10. Neutron reaction cross-sections

a. B^{10} (n, 2 α) T: σ (1 Mev $\leq$ E $\leq$ 5 Mev) = 31.6 mb (spectrum weighted)

σ (E > Mev) = 75 mb

b. Li^{7} (n, nαV) T: σ (3 Mev $\leq$ E $\leq$ 5 Mev) = 39.1 mb (spectrum weighted)

σ (E > 6 Mev) = 400 mb

11. Fraction of ternary tritium diffusing through zirconium cladding

a. Design value 0.30

b. Expected value 0.01

c. Li^{6} (n,nα) T: σ (E < 0.625 ev) = 675 barns

15B.6 VOLUME CONTROL TANK ACTIVITIES

The 400 ft^3 volume control tank is assumed to contain 160 ft^3 of liquid and 240 ft^3 of vapor. Table 15B.6-1 lists the activities in the volume control tank with clad defects in 1% of the fuel rods.

15B.7 GAS DECAY TANK ACTIVITIES

Refer to Section 15.7.1.2.

15B.8 ACTIVITY IN RECIRCULATED SUMP WATER

Table 15B.8-1 lists the concentration of iodine and noble gas isotopes in the recirculation loop at time zero after the design basis loss-of-coolant accident based on the following assumptions:

Power level	3391 MWt
Reactor coolant volume	12,760 ft^3
Emergency cooling water volume	46,900 ft^3
Iodine gap activity picked up by containment sump water	100%

Percent of core iodine activity present in fuel* rod gaps	3%

15B.9 REFERENCES

(1) DiNunno, J.J., et. al, "Calculation of Distance Factors for Power and Test Reactor Sites," TID 14844, March 1962.

(2) Bloneke, J.O. and Todd, Mary F., "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," (4 volumes) August and December 1957.

(3) D.F. Toner and J.L. Scott, "Fission Product Release for UO_2," Nuclear Safety Volume III No. 2, December 1961.

(4) J. Belle, Uranium Dioxide: Properties and Nuclear Applications Naval Reactors Division of Reactor Development, USAEC, 1961.

(5) Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 23, 1972.

(6) Federal Register / Vol 53, No. 39 / Monday, February 29, 1988 / pages 6040 through 6043.

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* Under TID 14844 assumptions, 50% of the core iodine is assumed to be released from the fuel rods and picked up by the recirculated water.

TABLE 15B.2-1

CORE AND GAP ACTIVITIES

Assumptions: Operation at 3391 MWt for 500 days
Temperature Distribution Specified in Table 15B.2-2

Isotope	Curies in the Core ($X 10^7$)	Percent of Core Activity in the Gap	Curies in the Gap ($X 10^5$)
I-131	8.35	2.3	19.2
I-132	12.75	0.26	3.3
I-133	19.09	0.79	15.1
I-134	23.01	0.16	3.8
I-135	17.05	0.43	7.5
Xe-133	19.02	1.85	35.2
Xe-133m	0.52	1.27	0.66
Xe-135	5.26	0.54	2.84
Xe-135m	5.25	0.086	0.45
Kr-85	0.095	21.57	2.05
Kr-85m	3.76	0.29	1.09
Kr-87	7.22	0.20	1.44
Kr-88	10.27	0.29	2.98

TABLE 15B.2-2

CORE TEMPERATURE DISTRIBUTION

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
14.5	2500

TABLE 15B.3-1

Assumptions Used in Calculating the Source Term for a Fuel Handling Accident

Power Level (MWt)	3250
Lead Rod Burnup (MWD/MTU)	60,000
Number of Failed Fuel Rods	204 (1 assembly)
Number of Fuel Assemblies in the Core	193
Assembly Radial Peaking Factor	1.65
Fuel-rod Inventory Released to Gap, %	
Iodine-131	12
Other Iodine	10
Krypton-85	30
Other Noble Gases	10

TABLE 15B.3-2

ACTIVITIES IN THE HIGHEST RATED POWER ASSEMBLY
WITH AN ASSEMBLY RADIAL PEAKING FACTOR OF 1.65
CURIES AT TIME OF REACTOR SHUTDOWN

ISOTOPE	TOTAL CURIES	FRACTION IN FUEL-CLADDING GAP	FUEL-CLADDING GAP CURIES
I-131	7.62×10^5	0.12	9.14×10^4
I-132	1.13×10^6	0.10	1.13×10^5
I-133	1.44×10^6	0.10	1.44×10^5
I-134	1.79×10^6	0.10	1.79×10^5
I-135	1.46×10^6	0.10	1.46×10^5
Kr-85m	2.21×10^5	0.10	2.21×10^4
Kr-85	7.70×10^3	0.30	2.31×10^3
Kr-87	4.16×10^5	0.10	4.16×10^4
Kr-88	6.31×10^5	0.10	6.31×10^4
Xe-133m	4.05×10^4	0.10	4.05×10^3
Xe-133	1.54×10^6	0.10	1.54×10^5
Xe-135	4.44×10^5	0.10	4.44×10^4

Table 15B.3-3

Source Term Information for the
Fuel Handling Accident in the Fuel Building

ISOTOPE	TOTAL CURIES/ASSEMBLY (100 HOURS AFTER SHUTDOWN)	FRACTION IN FUEL-CLADDING GAP	E_B (MEV)	E_γ (MEV)
I-131	3.302×10^5	0.12		
I-132	2.740×10^5	0.10		
I-133	3.280×10^4	0.10		
Kr-85	6.577×10^3	0.30	0.253	0.002
Xe-131m	4.981×10^3	0.10		0.163
Xe-133	6.233×10^5	0.10	0.102	0.081
Xe-133m	1.168×10^4	0.10		0.233

TABLE 15B.4-1

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES

1.	Core thermal power, max. calculated, MWt	3391
2.	Fraction of Fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, ft^3	12,760
4.	Reactor coolant average temperature, °F	570
5.	Purification flow rate (normal), gpm	75
6.	Effective cation demineralizer flow, gpm	7
7.	Volume control tank volumes	
	a. Vapor, ft^3	270
	b. Liquid, ft^3	130
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, La, Ce, Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs-134, 136, 137, Y-90, 91 and Mo-99	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for Cs-134, 136, 137, Y-90, 91 and Mo-99	10.0
11.	Volume control tank noble gas stripping fraction (closed system):	

Isotope	Stripping Fraction
Kr-85	2.3×10^{-5}
Kr-85m	2.7×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}

TABLE 15B.4-1 (Continued)

Isotope	Stripping Fraction
Xe-133	1.6×10^{-2}
Xe-133m	3.7×10^{-2}
Xe-135	1.8×10^{-1}
Xe-135m	8.0×10^{-1}
Xe-138	1.0

TABLE 15B.4-2

REACTOR COOLANT MAXIMUM ACTIVITIES
DOWNSTREAM OF REGENERATIVE HEAT EXCHANGER

Activation Products	μc/cc (150°F)
Mn-54	7.6×10^{-4}
Mn-56	2.8×10^{-2}
Co-58	2.4×10^{-2}
Fe-59	10.0×10^{-4}
Co-60	7.3×10^{-4}

Non-Volatile Fission Products (Continuous Full Power Operation)

	μc/cc (150°F)		μc/cc (150°F)
Br-84	4.24×10^{-2}	I-133	3.59
Rb-88	3.36	Te-134	2.59×10^{-2}
Rb-89	8.53×10^{-2}	I-134	4.75×10^{-1}
Sr-89	3.73×10^{-3}	Cs-134	2.21×10^{-1}
Sr-90	1.15×10^{-4}	I-135	1.84
Y-90	2.27×10^{-4}	Cs-136	1.22×10^{-1}
Sr-91	1.77×10^{-3}	Cs-137	1.11
Y-91	6.29×10^{-3}	Cs-138	7.33×10^{-1}
Mo-99	4.64	Ba-140	4.05×10^{-3}
I-131	2.34	La-140	1.58×10^{-3}
Te-132	2.54×10^{-1}	Ce-144	2.97×10^{-4}
I-132	8.28×10^{-1}	Pr-144	2.97×10^{-4}

Gaseous Fission Products	μc/cc (150°F)
Kr-85	4.44
Kr-85m	2.145
Kr-87	1.19
Kr-88	3.36
Xe-133	2.40×10^{2}
Xe-133m	2.63
Xe-135	6.87
Xe-138	5.52×10^{-1}

TABLE 15B.5-1

TRITIUM PRODUCTION IN THE REACTOR COOLANT
(Curies/Year)

Tritium Source	Total Produced	Released to the Coolant: Design Value	Released to the Coolant: Expected Value
Ternary Fissions	10,420	3126	104
Burnable Poison Rods (Initial Cycle)	922	277	9
Soluble Poison Boron			
(Initial Cycle)	378	378	378
(Equilibrium Cycle)	525	525	525
Li-7 Reaction	11	11	11
Li-6 Reaction	6	6	6
Deuterium Reaction	1	1	1
Totals Initial Cycle	11,738	3799	509
Totals Equilibrium Cycle	10,963	3669	647

TABLE 15B.6-1

VOLUME CONTROL TANK ACTIVITIES

Assumptions are given previously under reactor coolant activity.

Isotope	Activity (Curies)
Vapor	
Kr-85	1.56×10^{1}
Kr-85m	6.34×10^{1}
Kr-87	2.32×10^{1}
Kr-88	1.01×10^{2}
Xe-133	1.23×10^{4}
Xe-133m	1.34×10^{2}
Xe-135	2.80×10^{2}
Xe-135m	8.71×10^{-1}
Xe-138	3.90×10^{0}

TABLE 15B.7-1

GAS DECAY TANK ACTIVITIES

Assumptions: Volume of the tank immaterial to this calculation.

Clad Defects in 1% of fuel rods.

Operation at 3391 MWt for 280 days.

Tank contains entire gaseous activity stripped off from the reactor coolant system.

Reactor Coolant System Volume is 12,760 ft^3.

Isotope	Total Activity Curies
Kr-85	5.52×10^3
Kr-85m	5.53×10^2
Kr-87	3.15×10^2
Kr-88	9.76×10^2
Xe-133	8.09×10^4
Xe-133m	8.43×10^2
Xe-135	2.47×10^3
Xe-138	1.62×10^2

TABLE 15B.8-1

CONCENTRATION OF IODINE
ISOTOPES IN THE RECIRCULATION LOOP

Isotope	Recirculation Loop Concentration (μc/cc)
I-131	1.3×10^3
I-132	3.3×10^2
I-133	1.5×10^3
I-134	3.9×10^2
I-135	7.6×10^2

The radiation sources circulating in the residual heat removal loop are shown in Table 15B.8-2 and are used for whole body radiation doses in the auxiliary building.

The radioactivity in the containment also would be additional source of radiation to the auxiliary building following a loss-of-coolant accident.

TABLE 15B.7-1

GAS DECAY TANK ACTIVITIES

(This Table Has Been Intentionally Deleted)

APPENDIX 15C: HYDROGEN PURGE SYSTEM

APPENDIX 15C

HYDROGEN PURGE SYSTEM

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APPENDIX 15C

15C.1 HYDROGEN PURGE SYSTEM

The Hydrogen Purge System consists of two separate 100% capacity fans each fed from one of two separate lines from the Containment. Either fan may be fed from the main 42-inch purge line or from the pressure and vacuum relief line. The fans and lines are Seismic Class I and redundant power-operated components are supplied from separate essential busses. The charcoal filter units are Seismic Class I and consist of roughing, HEPA, and charcoal filters in series, with a rated flow of 400 cfm. One charcoal filter unit is provided for each Hydrogen Purge System fan.

15C.1.1 Hydrogen Purge System Components

The Hydrogen Purge System exhaust ventilation fans, each of which can provide a flow rate of 360 cfm, were sized based on the following assumptions from the first hydrogen production analysis:

1. The hydrogen generation rate in the Containment after a LOCA is such that the hydrogen level reaches 3.0% in 55 days.

2. The purge rate was selected to match the hydrogen production rate at 55 days so that the hydrogen level will slowly decrease from 3.0% as the production rate decreases.

3. The entire quantity of hydrogen required to be removed from the Containment at 55 days can be purged in a period of one hour. This allows the operator to choose the optimum purge period as a function of the prevailing meteorological conditions on each day that the Hydrogen Purge System is operated.

A second hydrogen production analysis was performed based on the assumptions given in Safety Guide 7. Appendix 15C shows that the design of the purge fans, as described above, was sufficient to remove the hydrogen generated in the Safety Guide 7 analysis.

The Hydrogen Purge System exhaust charcoal filters have the following parameters:

Airflow	400 ft^3/min
Depth of Charcoal Bed	2 inches
Face Velocity	40 ft/min
Efficiency	Methyl iodide removal efficiency $\geq$ 99% at 70% RH

The ventilation fans and the charcoal filters are the only two components unique to the Hydrogen Control Systems for the Zion Station.

15C.2 CONTROLLED CONTAINMENT VENTING AFTER A LOSS-OF-COOLANT ACCIDENT

15C.2.1 Introduction and Summary

The Hydrogen Thermal Recombiners are the licensing basis hydrogen removal system. Proper operation of the Thermal Recombiners will ensure that containment purging will not be required and will therefore eliminate radiological doses caused by purging. The Hydrogen Purge System could be used as a backup system to the Thermal Recombiners and therefore, the purge analysis has been provided for completeness. The exposures calculated for the containment purge analysis are based on calculational techniques using assumptions given in Safety Guide 7 which were in use at the time the original FSAR was written. Although present day analysis uses different assumptions and models, it is believed that exposures resulting from any purging that might be required will be less than 10CFR100 limits and that the values given here are still conservative. It should be noted that the containment purge analysis does not take credit for any hydrogen removal from the Thermal Recombiners.

The general purposes of this analysis are:

1. To determine and report the amounts of hydrogen which may be generated after a postulated LOCA based on the conservative assumptions given in Safety Guide 7;

2. To demonstrate that controlled containment venting, which is provided as a backup to the Thermal Recombiners, can be used as a method to cope with the hydrogen generated.

To achieve these purposes, extremely conservative assumptions concerning the generation and reduction of hydrogen in containment are used. The assumptions given in Safety Guide 7 are used for the hydrogen generation analysis (see Section 15.6.5.6.4) and an extremely conservative venting scheme is assumed for the Hydrogen Purge System.

Successful operation of the Thermal Recombiners should eliminate the need for containment purging as a method to reduce hydrogen concentrations. However, if purging should be required, under the assumptions used, the hydrogen concentration reaches the flammable limit of 4.1% at the end of 36 days. Venting is begun on day 30 when the hydrogen concentration is still below the Safety Guide 7 limit of 4.0% by volume in the Containment.

The Containment is vented on a daily basis for a period of one hour. A venting system flow rate of approximately 360 cfm is required.

In summary, if venting is carried out in the manner analyzed (without the benefit of operator discretion, without the reduction offered by the charcoal filters, and with other conservative assumptions), the resulting thyroid doses are 144.0 rem at the site boundary and 33.8 rem at the low population zone. The whole body doses are 1.940 rem at the site boundary and 0.460 rem at the low population zone. A full summary of doses is given in Section 15C.2.4. The doses actually expected from a controlled venting operation are much lower than those given above, since many factors of

conservatism have been combined in the analysis. Further discussion of this point is given in Section 15C.2.5.

On the basis of the magnitude of the calculated doses given above, it is concluded that controlled venting is an entirely practicable and satisfactory method of maintaining a limit on the hydrogen concentration in the Containment following a postulated LOCA.

15C.2.2 Meteorological Data

The same meteorological parameters are used in the analysis of post-LOCA containment venting as are used in the analysis of the initial environmental consequences of the LOCA as presented in Chapter 15. These meteorological parameters are discussed in detail in Section 15.6.5.5. For long term dose calculations, site dispersion factors were calculated by taking a time averaging of the χ/Q values given in Table 15.6-29. The assumptions used in obtaining the values presented in this table are conservative, especially for the two-hour period immediately after the LOCA, and do not represent the most favorable meteorological condition to be expected at the site. Containment venting, since it is done at the discretion of the operator, can be accomplished under the most advantageous meteorological conditions. This is substantiated, at least on a preliminary basis, by the data presented in Chapter 2.

Table 15.C.2-1 presents the time averaged χ/Q values to be used in this analysis.

15C.2.3 Purge Scheme

For the purpose of this analysis, it is assumed that purging will begin on Day 30 when the hydrogen volume is 3.82%. An average purge rate will be maintained such that this level is not exceeded. Purges will be carried out daily for a period of one hour except on days when the stability class or wind speed are unfavorable. Day 30 was selected as the starting point for purge because of the following factors:

1. If events occur as expected after a DBA, the hydrogen level will not reach 3.82% and no purging will be required;

2. This level allows sufficient margin below the lower flammable limit (4.1%);

3. Because of the low rates of production of hydrogen, approximately 36 days are required before the concentration reaches 4.1%, the lower flammability limit. This results in a very high probability of being able to vent under more favorable meteorological conditions than those assumed in this analysis; and

4. From the standpoint of minimizing the doses, the purge start is best delayed as long as is feasible after the accident (keeping factor 3 above in mind).

The purge rate was selected to match the rate of production at the time of initiation of the purge, so that the hydrogen level will slowly decrease from 3.82% as the production rate decreases. This approach will minimize the offsite doses. The required equivalent continuous purge rate, obtained from the production rate at 30 days (Figure 15.6-61) is approximately 14 cfm. Assuming the purge system will run one hour per day, the actual purge system flow rate required is approximately 336 cfm.

15C.2.4 Determination of Offsite Doses

The amount of activity released to the environs as the result of the Containment is a function of the amount of activity present in the Containment at the inception of the purge and the purge rate. The expression for the activity present in the Containment at any time, t, after the beginning of the purge is given below:

$$Q(t) = Q_o e^{-(\lambda_d + \lambda_p)\, t}$$

where:

Q_o = activity present in the Containment at the beginning of the purge period, curies
λ_d = radiological decay constant, $Days^{-1}$
λ_p = continuous purge rate constant, $Days^{-1}$

The activity released by purging the Containment is given by:

$$Q_R = \int_o^t \lambda_p\, Q(t)\, dt = \frac{Q_o\, \lambda_p}{\lambda_d + \lambda_p} \left(1 - e^{-(\lambda_d + \lambda_p)\, t} \right)$$

If purging is continued indefinitely, the amount released is:

$$Q_R = \frac{Q_o\, \lambda_p}{\lambda_d + \lambda_p}$$

This expression applies to each isotope present in the Containment. If the purge is not continuous, but is carried out on a regular scheduled series of "puffs", it can be represented in both the dose equation and the hydrogen analysis by an equivalent continuous purge rate. The equivalent continuous purge rate constant is determined from the actual flow rate of the purge system and the time schedule as follows:

$$\lambda_p = \left(\frac{F}{V}\right) f_p$$

where:

F = purge flow rate, 336 cfm
V = containment volume, $2.86 \times 10^6 ft^3$
f_p = average fraction of time that purging occurs

The isotopes of interest are I-131 (inhalation dose) and Kr-85 (whole body dose). For Kr-85, which has a long half-life, the activity present in the Containment after 30 days would be essentially the same as the activity introduced into the Containment at the time of the LOCA. The I-131 activity in the Containment after 30 days was computed considering only the fraction that is in the organic form, since the spray system will reduce the contribution of elemental iodine to a negligible proportion during the delay period. The fraction of I-131 assumed to be in the organic form is 2.5% of the core inventory.

The equation used for the inhalation dose is:

$$D_{INH} = Q_R \cdot B \cdot DCF \cdot \frac{\chi}{Q}$$

The average breathing rate of 2.32×10^{-4} m^3/sec was used, since the exposures are long term averages. The dose conversion factor (DCF) is 1.48×10^6 rem per curie for I-131.

The equation for the semi-infinite spherical model was used for the whole body dose as follows:

$$D_{WB} = 0.246\ Q_R \cdot E \cdot \frac{\chi}{Q}$$

A summary of the doses due to containment purging, assuming the largest possible quantities of krypton and iodine are released to the environs, are given in Table 15C.2-2. The doses are listed for a purge system with and without charcoal filters. An iodine removal efficiency of 99% was assumed for the charcoal filter.

15C.2.5 Factors of Conservatism

In order to identify more clearly the degree of conservatism incorporated in this analysis, the following summary of the assumptions used is given.

These doses are conservative because of the use of the following assumptions:

1. Conservative rate of hydrogen production based on Safety Guide No. 7;
2. Halogen release fractions from TID-14844;
3. Fraction of organic iodine of 2.5% of core inventory;
4. Large contingency on aluminum inventory;

5. "Accident" meteorology equivalent to that used to determine initial offsite doses due to LOCA;
6. No cleanup of organic iodine - internally or externally;
7. Ground level release;
8. Uncorrected semi-infinite spherical model for whole body dose; and
9. Long term exposure dose.

The use of more realistic (less conservative) meteorological parameters would certainly lower the doses given in Table 15C.2-2. Also, the use of more realistic parameters for the other variables would also reduce the doses given in Table 15C.2-2 or eliminate entirely the need for "controlled" venting.

TABLE 15C.2-1

SITE DISPERSION FACTORS APPLICABLE DURING CONTAINMENT VENTING

Distance (Meters)	Time Averaged x/Q for 30 day Period (sec/m^3)
300	4.33×10^{-5}
400	3.5×10^{-5}
600	2.58×10^{-5}
1000	1.59×10^{-5}
1600	8.3×10^{-6}
3000	2.0×10^{-6}

TABLE 15C.2-2

DOSES RECEIVED AS A RESULT OF CONTAINMENT PURGING

	30-Day Dose at Site Boundary (415 m)	30-Day Dose at Low Population Zone Boundary (1610 m)
Without Charcoal Filters		
Whole Body Dose (rem)	1.94	0.46
Thyroid Dose (rem)	144.0	33.8
With Charcoal Filters		
Whole Body Dose (rem)	1.94	0.46
Thyroid Dose (rem)	1.44	0.34

Appendix 15D: Containment Pressure Analysis

Note: This Containment Pressure Analysis incorporates VANTAGE 5 fuel, increased Containment Spray delay time and decreased service water flow to the Reactor Containment Fan Coolers.

Appendix 15D

15D.1 Containment Pressure Analysis

Section 15D.2 provides the VANTAGE 5 fuel design basis analysis performed for the Zion peak containment pressure analysis. Section 15.6.5.4.4.1 through 15.6.5.4.6.5 provide the historical perspective and the evolution of the Zion peak containment pressure analysis design basis. The assumptions/data used in these historical analyses are provided for information purposes to explain the evolution of the containment analysis.

15D.2 Peak Containment Pressure

This section documents the containment analysis performed for Zion to incorporate the increase in the Containment Spray delay time and a decreased service water flow to the Marlo Coil RCFCs. (The Marlo Coil Fan Coolers replaced the original Westinghouse Fan Coolers around 1986.) This analysis is based on the limiting case of a Double Ended Pump Suction (DEPS) break which generates a peak pressure of 46.79 psig at 118.9 seconds. The calculated peak pressure demonstrates there is still margin to the Zion design pressure limit of 47.0 psig.

15D.3 Input Assumptions

The following input assumptions were used for the design basis analysis.

15D.3.1 Containment Initial Conditions

Maximum Pressure	15.7 psia
Maximum Temperature	120.0°F
Minimum Relative Humidity	20%
Minimum Volume	2.715 E+06 cubic feet

Appendix 15D (Continued)

15D.3.2 Containment Spray

Spray delay time (from Containment Hi-Hi)	110 seconds
Containment Hi-Hi pressure setpoint	24.8 psig
Spray flow capacity	5230 gpm (two pumps total)
Spray flow during recirculation	1769 gpm (1 RHR pump)

15D.3.3 Fan coolers

Fan Cooler delay time (from Containment Hi)	58 seconds
Containment Hi pressure setpoint	6.3 psig

Fan Cooler Heat removal capacity per cooler.

	Containment Temperature (°F)	Heat Transfer Rate (Btu/hr)
	120	0.0
	160	34.07E+06
	200	54.93E+06
	240	78.00E+06
	271	96.20E+06
	300	117.27E+06
	500	117.27E+06

The above fan cooler capacity is based on an assumed service water inlet temperature of 80°F and flow rate of 1500 gpm.

15D.3.4 Refueling Water Storage Tank

Time to Empty	2013 seconds

Appendix 15D (Continued)

15D.3.5 Equipment Assumed Operable

Fan Coolers	3
Containment Spray Pumps	2
ECCS Trains	2

Note: Zion has a unique ECCS design and the above combination of operable equipment assumed is based on the worst single active failure criteria. The single failure for this case is the loss of 1 CS pump which is consistent with a maximum ECCS case and conservative for short term results. For the recirculation phase, only 1 RHR train is assumed available which is consistent with a minimum ECCS case and conservative for long term results.

15D.3.6 Passive Heat Sinks

UFSAR Tables 15.6-11 & 15.6-17

15D.3.7 Mass & Energy Release

UFSAR Tables 15.6-18 & 15.6-19

(A quench front of 8 feet (Case 1) is assumed)

15D.3.8 References

1) ComEd (NFS) calculation, "Zion LOCA Containment Integrity Analysis," PSA-Z-95-03. A. Patterson, 3/8/95.

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16. TECHNICAL SPECIFICATIONS

16.1 PRELIMINARY TECHNICAL SPECIFICATIONS

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

16.2 PROPOSED FINAL TECHNICAL SPECIFICATIONS

The Technical Specifications for Zion Station are issued by the NRC as Appendix A to the Operating License. Because of their evolving nature during original definition, and their subsequent revision during plant operations, the Technical Specifications are placed in a separate document.

16.3 COMPONENT LISTS

This chapter presents information associated with various Technical Specification surveillance requirements. The component lists denoted herein help to clarify the equipment required to be tested in accordance with surveillance requirements.

16.3.1 Reactor Protection System Instrument Numbers

The reactor protection system instrument numbers associated with Technical Specifications denote the primary sensors for the various protection channels and are specified in Table 16.3-1.

16.3.2 Engineered Safeguards System Instrument Numbers

The engineered safeguards system instrument numbers associated with Technical Specifications denote the primary sensors for the various safeguards channels and are specified in Table 16.3-2.

16.3.3 Engineered Safeguards Equipment Actuation Test

The engineered safeguards equipment actuation tests associated with Technical Specifications are specified in Table 16.3-3. This table denotes engineered safeguards components, which are either actuated or continuity-checked to the final actuating device.

16.3.4 Reactor Containment Fan Cooler Dampers

The reactor containment fan cooler dampers associated with Technical Specifications are specified in Table 16.3-4. This table denotes the accident position of each required containment fan cooler damper.

16.3.5 Containment Spray System Components

The contaminated spray system components associated with Technical Specifications are specified in Table 16.3-5.

16.3.6 Auxiliary Feedwater Pump and Supply System

The auxiliary feedwater pump and supply system components associated with Technical Specifications are specified in Table 16.3-6.

16.3.7 Centrifugal Charging Pump System

The centrifugal charging pump system components associated with Technical Specifications are specified in Table 16.3-7.

16.3.8 Safety Injection Pump System

The safety injection pump system components associated with Technical Specifications are specified in Table 16.3-8.

16.3.9 Residual Heat Removal Pump System

The residual heat removal system components associated with Technical Specifications are specified in Table 16.3-9.

16.3.10 Accumulator Tanks

The accumulator tank components associated with Technical Specifications are specified in Table 16.3-10.

16.3.11 Component Cooling Water Pump System

The component cooling pump system components associated with Technical Specifications are specified in Table 16.3-11.

16.3.12 Service Water Pump System

The service water pump system components associated with Technical Specifications are specified in Table 16.3-12.

16.3.13 Hydrogen Control System

The hydrogen control system components associated with Technical Specifications are specified in Table 16.3-13.

16.3.14 Isolation Seal Water System

The isolation seal water system components associated with Technical Specifications are specified in Table 16.3-14.

16.3.15 Penetration Pressurization System

The penetration pressurization system components are specified in Table 16.3-15.

16.3.16 Containment Isolation Valves

The containment isolation valves associated with Technical Specifications are specified in Table 16.3-16. This table specifically denotes containment isolation valves that (1) receive a Phase A or Phase B signal, (2) are manually operated, or (3) are main steam isolation valves.

16.3.17 4160-V Engineered Safeguard Bus Main, Reserve, and Standby Feeds

The 4160-V engineered safeguard bus main, reserve, and standby feeds associated with Technical Specification 4.15.1.D.1 are specified in Table 16.3-17.

16.3.18 Failed Fuel Monitoring Instruments

The failed fuel monitoring instruments associated with Technical Specification 4.19.1 are specified in Table 16.3-18.

16.3.19 Fire Suppression Water System

The fire suppression water system components associated with Technical Specification 4.21.2 are specified in the Fire Protection Report.

16.3.20 Boric Acid System (per unit) Relocated Technical Specifications

Boric Acid system operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-20.

16.3.21 NIS Detector Temperature Control Relocated Technical Specifications

NIS Detector operability and surveillance requirements that have been relocated from the Technical Specifications are provided in Table 16.3-21.

16.3.22 Pressurizer Safety Valve Relocated Technical Specifications

Pressurizer Safety Valve operability requirements that have been relocated from the Technical Specifications are provided in Table 16.3-22.

16.3.23 Reactor Head Vent System Relocated Technical Specifications

Reactor Head Vent system operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-23.

16.3.24 RCS Pressurization and Integrity Relocated Technical Specifications

RCS Pressurization and Integrity operability requirements that have been relocated from the Technical Specifications are provided in Table 16.3-24.

16.3.25 (Not Used)

16.3.26 RCS Structural Integrity Relocated Technical Specifications

RCS Structural Integrity operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-26.

16.3.27 RCS Chemistry Relocated Technical Specifications

RCS Chemistry limit requirements that have been relocated from the Technical Specifications are provided in Table 16.3-27.

16.3.28 Accident Monitoring Instrumentation Relocated Technical Specifications

Accident Monitoring Instrumentation operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-28.

16.3.29 Penetration Pressurization System Relocated Technical Specifications

Penetration Pressurization system operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-29.

16.3.30 Refueling Operations, Communications, Fuel Handling SRO Relocated Technical Specifications

Refueling Operations, Communications, and Fuel Handling SRO requirements that have been relocated from the Technical Specifications are provided in Table 16.3-30.

16.3.31 Refueling Equipment, Spent Fuel Cooling System, and Fuel Inspection Program Relocated Technical Specifications

Refueling Equipment, Spent Fuel Cooling system, and Fuel Inspection Program requirements that have been relocated from the Technical Specifications are provided in Table 16.3-31.

16.3.32 Plant Radiation Monitoring Instrumentation Relocated Technical Specifications

Plant Radiation Monitoring Instrumentation operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-32.

16.3.33 Failed Fuel Monitoring Relocated Technical Specifications

Failed Fuel Monitoring instrumentation operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-33.

16.3.34 Shock Suppressor Relocated Technical Specifications

Shock Suppressor operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-34.

16.3.35 Sealed Source Contamination Relocated Technical Specifications

Sealed Source Contamination requirements that have been relocated from the Technical Specifications are provided in Table 16.3-35.

16.3.36 Reactor Coolant System Pressure Isolation Valve Relocated Technical Specifications

Reactor Coolant System Pressure Isolation Valve allowable leakage limits that have been relocated from the Technical Specifications are provided in Table 16.3-36.

16.3.37 Boundary Doors for Flood Protection Relocated Technical Specifications

Boundary Doors for Flood Protection requirements that have been relocated from the Technical Specifications are provided in Table 16.3-37.

16.3.38 Containment Ventilation System Relocated Technical Specifications

Containment Ventilation system operability and surveillance testing requirements that have been relocated from the Technical Specifications are provided in Table 16.3-38.

TABLE 16.3-1 (1 of 2)

REACTOR PROTECTION SYSTEM INSTRUMENT NUMBERS

	REACTOR CHANNEL DESCRIPTION		DEVICE DESIGNATION
1.	Manual Reactor Trip		NA
2.	Power Range Neutron Flux (Low setpoint, high setpoint, high rate and negative rate)		NC-41, NC-42, NC-43, NC-44
3.	Source Range Nuclear Flux		NC-31, NC-32
4.	Intermediate Range Flux		NC-35, NC-36
5.	ΔT (overpower and overtemperature)	a.	Temperature: TE-411A, TE-411B, TE-421A, TE-421B, TE-431A, TE-431B, TE-441A, TE-441B, TE-410A, TE-410B, TE-420A, TE-420B, TE-430A, TE-430B, TE-440A, TE-440B
		b.	Pressure: PT-455, PT-456, PT-457, PT-458
		c.	Flux: NC-41, NC-42, NC-43, NC-44
6.	Pressurizer Pressure (high and low)		PT-455, PT-456, PT-457, PT-458
7.	Pressurizer High Level		LT-459, LT-460, LT-461
8.	Primary Coolant Flow		FT-414, FT-415, FT-416, FT-424, FT-425, FT-426, FT-434, FT-435, FT-436, FT-444, FT-445, FT-446
9.	RCP Undervoltage	UNIT I	447(KP)/142, 447(KP)/143, 447(KP)/144, 447(KP)/145
		UNIT II	447(KP)/242, 447(KP)/243, 447(KP)/244, 447(KP)/245
10.	RCP Underfrequency	UNIT I	81x142, 81x143, 81x144, 81x145
		UNIT II	81x242, 81x243, 81x244, 81x245
11.	RCP Breaker Trip	UNIT I	RCP1A, RCP1B, RCP1C, RCP1D
		UNIT II	RCP2A, RCP2B, RCP2C, RCP2D

TABLE 16.3-1 (2 of 2)

REACTOR PROTECTION SYSTEM INSTRUMENT NUMBERS

REACTOR CHANNEL DESCRIPTION	DEVICE DESIGNATION
12. Low Steam Generator Level in coincidence with steam flow/feed flow mismatch	a. Level: LT-517, LT-518, LT-527, LT-528 LT-537, LT-538, LT-547, LT-548
	b. Feed Flow: FT-510, FT-511, FT-520, FT-521 FT-530, FT-531, FT-540, FT-541
	c. Steam Flow: FT-512, FT-513, FT-522, FT-523 FT-532, FT-533, FT-542, FT-543
13. Low-Low Steam Generator Level	LT-517, LT-518, LT-519, LT-527, LT-528, LT-529 LT-537, LT-538, LT-539, LT-547, LT-548, LT-549
14. Safety Injection	See Table 16.3-2.
15. Turbine Trip	a. Auto Stop Oil Dump: AST-3, AST-4, AST-5
	b. Stop Valve Close: 33T1/IO, 33T2/IO 33T3/IO, 33T4/IO
16. Automatic Reactor Trip Logic	N.A.
PERMISSIVES	
P-6	NC-35, NC-36
P-7	Nuclear Flux: NC-41, NC-42, NC-43, NC-44 Turbine Pressure: PT-505, PT-506
P-8	NC-41, NC-42, NC-43, NC-44
P-10	NC-41, NC-42, NC-43, NC-44

TABLE 16.3-2 (1 of 3)

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENT NUMBERS

CHANNEL DESCRIPTION	DEVICE DESIGNATION
I. SAFETY INJECTION	
1. Manual Actuation	N.A.
2. Automatic Actuation	N.A.
3. Low Pressurizer Pressure	PT-455, PT-456, PT-457
4. High Steamline Differential Pressure	PT-514, PT-524, PT-534, PT-544 PT-515, PT-525, PT-535, PT-545 PT-516, PT-526, PT-536, PT-546
5. High Steamline Flow in Coincidence with Low-Low-T_{avg} or Low Steamline Pressure	a. Flow: FT-512, FT-513, FT-522, FT-523 FT-532, FT-533, FT-542, FT-543
	b. Temperature: TE-411A, TE-411B, TE-421A, TE-421B TE-431A, TE-431B, TE-441A, TE-441B TE-410A, TE-410B, TE-420A, TE-420B TE-430A, TE-430B, TE-440A, TE-440B
	c. Pressure: PT-516, PT-526, PT-536, PT-546
6. High Containment Pressure	PT-CS19, PT-CS20, PT-CS21, PT-CS22
II. CONTAINMENT SPRAY	
1. Manual Actuation	N.A.
2. Automatic Actuation	N.A.
3. High-High Containment Pressure	PT-CS19, PT-CS20, PT-CS21, PT-CS22

TABLE 16.3-2 (2 of 3)

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENT NUMBERS

CHANNEL DESCRIPTION	DEVICE DESIGNATION
III. CONTAINMENT ISOLATION	
A) Phase A	
1. Manual Actuation	N.A.
2. Safety Injection	Section I of this table.
B) Phase B	
1. Manual Actuation	N.A.
2. Automatic Actuation	N.A.
3. High-High Containment Pressure	PT-CS19, PT-CS20, PT-CS21, PT-CS22
IV. STEAM LINE ISOLATION	
1. Manual Actuation	N.A.
2. Automatic Actuation	N.A.
3. High-High Containment Pressure	PT-CS19, PT-CS20, PT-CS21, PT-CS22
4. High Steamline Flow in Coincidence with Low-Low T_{avg} or Low Steamline Pressure	a. Flow: FT-512, FT-513, FT-522, FT-523 FT-532, FT-533, FT-542, FT-543
	b. Temperature: TE-411A, TE-411B, TE-421A, TE-421B TE-431A, TE-431B, TE-441A, TE-441B TE-410A, TE-410B, TE-420A, TE-420B TE-430A, TE-430B, TE-440A, TE-440B
	c. Pressure: PT-516, PT-526, PT-536, PT-546

TABLE 16.3-2 (3 of 3)

ENGINEERED SAFEGUARDS SYSTEM INSTRUMENT NUMBERS

CHANNEL DESCRIPTION		DEVICE DESIGNATION
V. AUXILIARY FEEDWATER		
1. Manual		N.A.
2. Automatic		N.A.
3. Steam Generator Water Level Low-Low		LC-517B, LC-527B, LC-537B, LC-547B LC-518B, LC-528B, LC-538B, LC-548B LC-519B, LC-529B, LC-539B, LC-549B
4. Undervoltage - RCP Busses Start Turbine Driven Pump	UNIT I UNIT II	447(KP)-142,447(KP)-143,447(KP)-144,447(KP)-145 447(KP)-242,447(KP)-243,447(KP)-244,447(KP)-245
5. SI Start Motor- and Turbine-Driven Pumps		See Section I of this Table.
6. Station Blackout		
a. Start Motor-Driven Pumps	UNIT I UNIT II	427(CV-7)-142, 427(CV-7)-143, 427(CV-7)-144 427(CV-7)-242, 427(CV-7)-243, 427(CV-7)-244
b. Start Turbine-Driven Pumps	UNIT I UNIT II	447(KP)-142,447(KP)-143,447(KP)-144,447(KP)-145 447(KP)-242,447(KP)-243,447(KP)-244,447(KP)-245
7. Secondary Undervoltage Protection System	UNIT I	427-1(27D)-147, 427-1(27D)-148, 427-1(27D)-149 427-2(27D)-147, 427-2(27D)-148, 427-2(27D)-149
	UNIT II	427-1(27D)-247, 427-1(27D)-248, 427-1(27D)-249 427-2(27D)-247, 427-2(27D)-248, 427-2(27D)-249
PERMISSIVES		
P-11		Pressurizer pressure - PT-455, PT-456, PT-457
P-12		Temperature TE-411A, TE-411B, TE-421A, TE-421B TE-431A, TE-431B, TE-441A, TE-441B TE-410A, TE-410B, TE-420A, TE-420B TE-430A, TE-430B, TE-440A, TE-440B

TABLE 16.3-3 (1 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
I. Steamline Isolation		
HOV-MS0001		X
HOV-MS0002		X
HOV-MS0003		X
HOV-MS0004		X
II. Safeguards Actuation		
A) Feedwater Isolation		
FCV-510		X
FCV-510A		X
FCV-520		X
FCV-520A		X
FCV-530		X
FCV-530A		X
FCV-540		X
FCV-540A		X
MOV-FW0018		X
MOV-FW0016		X
MOV-FW0017		X
MOV-FW0019		X
B) Diesel Starts		
Diesel 0 Start	X	
Diesel 1A Start (2A)	X	
Diesel 1B Start (2B)	X	

TABLE 16.3-3 (2 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

	COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
C)	Reactor Trip Breaker		
	A Trip Breaker	X	
	B Trip Breaker	X	
	A Bypass Breaker		X
	B Bypass Breaker		X
D)	Aux Feedwater Pump Start		
	Aux FW Pump 1A Aux Oil Pump (2A)	X	
	Aux FW Pump 1B Aux Oil Pump (2B)	X	
	Aux FW Pump 1A (2A)	X	
	Aux FW Pump 1B (2B)	X	
	Aux FW Pump 1C Oil Pump (2C)	X	
	FCV-MS57	X	
E)	Safety Injection		
	Charging Pump 1B Lube Oil Pump (2B)	X	
	Charging Pump 1B (2B)	X	
	Safety Injection Pump 1A (2A)	X	
	Safety Injection Pump 1B (2B)	X	
	Residual Heat Removal Pump 1B (2B)	X	
	MOV-RH8716B	X	
	MOV-RH8700B	X	
	Component Cooling Pump OE	X	
	Component Cooling Pump OD	X	
	Component Cooling Pump OC	X	
	Charging Pump 1A Lube Oil Pump (2A)	X	
	Charging Pump 1A (2A)	X	

TABLE 16.3-3 (3 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
Residual Heat Removal Pump 1A (2A)	X	
MOV-RH8700A	X	
MOV-SI8803A	X	
MOV-SI8808C * OPEN	X	
MOV-SI8803B	X	
MOV-SI8808D * OPEN	X	
MOV-LCV112B	X	
MOV-LCV112D	X	
MOV-VC8105	X	
MOV-SI8808A * OPEN	X	
MOV-SI8804A	X	
MOV-SI9011B ** CLOSED	X	
MOV-SI8923B	X	
MOV-SI8806 * OPEN	X	
MOV-SI8802 * OPEN	X	
MOV-SI8800D	X	
MOV-SI8923A	X	
MOV-SI8800A	X	
MOV-SI9010A	X	
AOV-SI8880	X	
MOV-RH8716C	X	
MOV-RH9000 ** CLOSED	X	

TABLE 16.3-3 (4 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
MOV-RH8703 * OPEN	X	
MOV-RH8716A	X	
Component Cooling Pump OA	X	
Component Cooling Pump OB	X	
AOV-SI8870A	X	
MOV-LCV112C	X	
MOV-LCV112E	X	
MOV-VC8106	X	
MOV-SI8808B * OPEN	X	
MOV-SI9010B		
MOV-SI8807B	X	
MOV-SI8804B	X	
MOV-SI8809A ** OPEN	X	
AOV-SI8870B	X	
MOV-SI8800B	X	
MOV-SI8812B * OPEN	X	
MOV-SI8812A * OPEN	X	
MOV-SI9011A ** CLOSED	X	
MOV-SI8807A	X	
MOV-SI8809B ** OPEN	X	
MOV-SI8800C	X	
F) Emergency Fan Coolers		
Vent Fan 1A low speed (2A)	X	

TABLE 16.3-3 (5 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

	COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
	Vent Fan 1B low speed (2B)	X	
	Vent Fan 1C low speed (2C)	X	
	Vent Fan 1D low speed (2D)	X	
	Vent Fan 1E low speed (2E)	X	
G)	Service Water Pump Starts and System Isolation		
	Service Water Pump 1A (2A)	X	
	Service Water Pump 1B (2B)	X	
	MOV-SW0001	X	
	MOV-SW0100	X	
	OMOV-SW0006	X	
	OMOV-SW0005	X	
	OMOV-SW0008	X	
	OFCV-SW54	X	
	Service Water Pump 1C (2C)	X	
	MOV-SW0002	X	
	MOV-S00115	X	
	OMOV-SW0007	X	
	OMOV-SW0010	X	
	OMOV-SW0009	X	
H)	Containment Isolation Phase "A" & Isolation Valve Seal Water		
	AOV-VC8149A	X	
	AOV-VC8152	X	

TABLE 16.3-3 (6 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
AOV-DT9159A	X	
AOV-DT9160A	X	
AOV-DT9157	X	
LCV-DT1003	X	
Reactor Coolant Drain Pump 1A (2A)	X	
Reactor Coolant Drain Pump 1B (2B)	X	
FCV-SS9354A	X	
FCV-SS9355A	X	
FCV-SS9356A	X	
FCV-SS9357A	X	
FCV-IA01A		X
FCV-PR24A	X	
FCV-FP08	X	
FCV-SS02	X	
FCV-SS03	X	
FCV-SS04	X	
FCV-SS05	X	
ACV-VC8149C	X	
FCV-PR19B	X	
FCV-PR20B	X	
FCV-PR21B	X	
FCV-PR22B	X	
FCV-PR23B	X	
FCV-VF01B (A)	X	
FCV-VN02B (A)	X	

TABLE 16.3-3 (7 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
FCV-RV112	X	
FCV-RV114	X	
FCV-WD17B	X	
FCV-SA01B	X	
AOV-RV0002	X	
AOV-RV0004	X	
FCV-IW09	X	
FCV-IW11	X	
FCV-IW13	X	
FCV-IW15	X	
FCV-IW17	X	
AOV-RC8026	X	
AOV-RC8029	X	
AOV-RC8033	X	
AOV-CC9437	X	
FCV-PR19A	X	
FCV-PR20A	X	
AOV-BD0003	X	
AOV-BD0005	X	
AOV-BD0007	X	
AOV-VC9149B	X	
AOV-VC8153	X	
MOV-VC8100		X
FCV-IA01B		X

TABLE 16.3-3 (8 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

	COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
I)	Containment Ventilation Isolation		
	FCV-RV97	X	
	FCV-RV98	X	
	FCV-RV99A	X	
	AOV-RV0005	X	
	FCV-RV91	X	
	FCV-RV92	X	
	FCV-RV93A	X	
	FCV-RV100	X	
	FCV-RV101	X	
	FCV-RV102A	X	
	FCV-RV94	X	
	FCV-RV95	X	
	FCV-RV96A	X	
	FCV-RV103	X	
	FCV-RV104	X	
	FCV-RV105A	X	
	FCV-RV99B	X	
	FCV-RV93B	X	
	FCV-RV102B	X	
	AOV-RV0006	X	
	FCV-RC96B	X	
	FCV-RC105B	X	
	FCV-RV93C	X	
	FCV-RV0001	X	

TABLE 16.3-3 (9 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
FCV-RV0003	X	
OSV-PV11A	X	
OSV-PV11B	X	
FCV-RV93D	X	
FCV-RV96C	X	
FCV-RV96D	X	
FCV-RV99C	X	
FCV-RV99D	X	
FCV-RV102C	X	
FCV-RV102D	X	
FCV-RV105C	X	
FCV-RV105D	X	
OSV-OV20	X	
OSV-OV24	X	
III. Containment Spray		
A) Spray Actuation		
Containment Spray Pump 1A (2A)	X	
MOV-CS0002	X	
MOV-CS0008	X	
MOV-CS0003	X	
Containment Spray Pump 1B (2B)	X	
MOV-CS0004	X	
MOV-CS0009	X	
MOV-CS0005	X	

TABLE 16.3-3 (10 of 10)

ENGINEERED SAFEGUARDS EQUIPMENT ACTUATION TEST

	COMPONENTS	DEVICE ACTUATION	CONTINUITY CHECK TO FINAL ACTUATION DEVICE
	Containment Spray Pump 1C (2C) Starting Circuit	X	
	MOV-CS0008	X	
	MOV-CS0010	X	
	MOV-CS0007	X	
B)	Containment Isolation Phase "B"		
	MOV-CC9413A		X
	MOV-CC685		X
	MOV-CC9414		X
	MOV-CC9413B		X
	MOV-CC9438	X	

TABLE NOTATIONS:

* Valves which have been designated as required to be stroked during REFUELING OUTAGE: These valves are placed in the proper positions for ECCS operation and have their power removed. Removal of power has been found to be acceptable to satisfy spurious valve actuation criteria until analysis determines a final acceptable solution. These valves do not change position throughout LOCA. Stroking at a REFUELING interval will insure their OPERABILITY. Energization of these valves is permissible to support other testing or plant evolutions such as heatup or cooldown.

** These valves are placed in proper position with power removed as specified above. These valves are required to be shifted when going from cold to hot leg injection. Quarterly testing is kept to insure OPERABILITY of these valves. These valves may also be energized to support plant testing or operations.

TABLE 16.3-4

REACTOR CONTAINMENT FAN COOLER DAMPERS

COMPONENT NAME	ACCIDENT POSITION	COMPONENT NUMBER
Containment Fan Cooler-1A/2A		RV001-1A(2A)
Accident Inlet Damper	Open	FCV-RV91
Normal Flow Inlet Damper	Closed	FCV-RV92
Accident Outlet Damper	Open	FCV-RV93A
Accident Outlet Damper	Open	FCV-RV93B
Accident Outlet Damper	Open	FCV-RV93C
Accident Outlet Damper	Open	FCV-RV93D
Containment Fan Cooler-1B/2B		RV002-1B(2B)
Accident Inlet Damper	Open	FCV-RV94
Normal Flow Inlet Damper	Closed	FCV-RV95
Accident Outlet Damper	Open	FCV-RV96A
Accident Outlet Damper	Open	FCV-RV96B
Accident Outlet Damper	Open	FCV-RV96C
Accident Outlet Damper	Open	FCV-RV96D
Containment Fan Cooler-1C/2C		RV003-1C(2C)
Accident Inlet Damper	Open	FCV-RV97
Normal Flow Inlet Damper	Closed	FCV-RV98
Accident Outlet Damper	Open	FCV-RV99A
Accident Outlet Damper	Open	FCV-RV99B
Accident Outlet Damper	Open	FCV-RV99C
Accident Outlet Damper	Open	FCV-RV99D
Containment Fan Cooler-1D/2D		RV001-1D(2D)
Accident Inlet Damper	Open	FCV-RV100
Normal Flow Inlet Damper	Closed	FCV-RV101
Accident Outlet Damper	Open	FCV-RV102A
Accident Outlet Damper	Open	FCV-RV102B
Accident Outlet Damper	Open	FCV-RV102C
Accident Outlet Damper	Open	FCV-RV102D
Containment Fan Cooler-1E/2E		RV002-1E(2E)
Accident Inlet Damper	Open	FCV-RV103
Normal Flow Inlet Damper	Closed	FCV-RV104
Accident Outlet Damper	Open	FCV-RV105A
Accident Outlet Damper	Open	FCV-RV105B
Accident Outlet Damper	Open	FCV-RV105C
Accident Outlet Damper	Open	FCV-RV105D

TABLE 16.3-5

CONTAINMENT SPRAY SYSTEM COMPONENTS

COMPONENT NAME	COMPONENT NUMBER
Containment Spray Pump 1A (2A) (Motor Driven)	CS001-1A (2A)
Containment Spray Pump 1A (2A) Discharge Isolation Valve	MOV-CS0002
Spray Additive to 1A (2A) Eductor Stop Valve	MOV-CS0008 *
Containment Spray Pump 1A (2A) Header Stop Valve	MOV-CS0003
1A (2A) RHR Train to 1A (2A) Header Stop Valve	MOV-CS0049 #
Containment Spray Pump 1B (2B) (Motor Driven)	CS002-1B (2B)
Containment Spray Pump 1B (2B) Discharge Isolation Valve	MOV-CS0004
Spray Additive to 1B (2B) Eductor Stop Valve	MOV-CS0009 *
Containment Spray Pump 1B (2B) Header Stop Valve	MOV-CS0005
1B (2B) RHR Train to 1B (2B) Header Stop Valve	MOV-CS0050 #
Containment Spray Pump 1C (2C) (Diesel Driven)	CS003-1C (2C)
Containment Spray Pump 1C (2C) Discharge Isolation Valve	MOV-CS0006
Spray Additive to 1C (2C) Eductor Stop Valve	MOV-CS0010 *
Containment Spray Pump 1C (2C) Header Stop Valve	MOV-CS0007

* Iodine Removal only

\# Containment Spray Recirculation Phase System only

TABLE 16.3-6

AUXILIARY FEEDWATER PUMP AND SUPPLY SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Auxiliary Feedwater Pump 1A (2A) (turbine-driven)	FW004
Auxiliary Feedwater Pump 1B (2B) (motor driven)	FW005
Auxiliary Feedwater Pump 1C (2C) (motor-driven)	FW006
Condensate Storage Tank-1 (2)	SC001
Auxiliary Feedwater Pump 1A (2A) service water supply valve	MOV-SW0102
Auxiliary Feedwater Pump 1B (2B) service water supply valve	MOV-SW0101
Auxiliary Feedwater Pump 1B (2B) service water supply valve	MOV-SW0104
Auxiliary Feedwater Pump 1C (2C) service water supply valve	MOV-SW0103
Auxiliary Feedwater Pump 1C (2C) service water supply valve	MOV-SW0105

TABLE 16.3-7

CENTRIFUGAL CHARGING PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Centrifugal Charging Pump-1A (2A)	VC006-1A (2A)
Centrifugal Charging Pump-1B (2B)	VC006-1A (2B)
Charging Header Isolation Valves	MOV-VC8105 MOV-VC8106
Recirculation Flow Isolation Valves	MOV-VC8110 MOV-VC8111
Volume Control Tank Outlet Isolation Valves	MOV-VC112B MOV-VC112C
Emergency Suction Valves From RWST	MOV-VC112D MOV-VC112E
ECCS Charging Header Discharge Isolation Valves	MOV-SI8803A MOV-SI8803B

TABLE 16.3-8

SAFETY INJECTION PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Safety Injection Pump - 1A(2A)	SI003-1A (2A)
Safety Injection Pump - 1B(2B)	SI004-1B (2B)
Safety Injection Pumps' Suction Valve	MOV-SI8806

TABLE 16.3-9

RESIDUAL HEAT REMOVAL PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Residual Heat Removal Pump-1A (2A) Residual Heat Removal Pump-1B (2B)	RH001-1A (2A) RH002-1B (2B)
Residual Heat Removal Pump-1A (2A) Residual Heat Removal Pump-1B (2B)	RH003-1A (2A) RH004-1B (2B)
Recirculation Sump to RHR Pump Suction Valves	MOV-SI8811A MOV-SI8811B
RWST to RHR Pump Suction Valves	MOV-RH8700A MOV-RH8700B
Isolation Valves from Reactor Coolant System to RHR Pumps	MOV-RH8701 MOV-RH8702
Residual Heat Removal Pumps' Suction Valves	MOV-SI8812A MOV-SI8812B

TABLE 16.3-10

ACCUMULATOR TANKS

COMPONENT NAME	COMPONENT NUMBER
Accumulator Tank - 1A (2A)	SI005
Accumulator Tank - 1B (2B)	SI006
Accumulator Tank - 1C (2C)	SI007
Accumulator Tank - 1D (2D)	SI008
Accumulator Tank - 1A (2A) Isolation Valve	MOV-SI8808A
Accumulator Tank - 1B (2B) Isolation Valve	MOV-SI8808D
Accumulator Tank - 1C (2C) Isolation Valve	MOV-SI8808B
Accumulator Tank - 1D (2D) Isolation Valve	MOV-SI8808C
Accumulator Tank - 1A (2A) Check Valve	SI8948A
Accumulator Tank - 1B (2B) Check Valve	SI8948D
Accumulator Tank - 1C (2C) Check Valve	SI8948B
Accumulator Tank - 1D (2D) Check Valve	SI8948C

TABLE 16.3-11

COMPONENT COOLING WATER PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Component Cooling Pump-OC (Standby)	OCC005-OC
Component Cooling Pump-OD (Unit 1)	OCC006-OD
Component Cooling Pump-OE (Unit 1)	OCC007-OE
Component Cooling Heat Exchanger - #0 (Standby)	OCC001-#0
Component Cooling Heat Exchanger - #1 (Unit 1)	1CC001-#1
Component Cooling Pump-OB (Unit 2)	OCC003-OB
Component Cooling Pump-OA (Unit 2)	OCC004-OA
Component Cooling Heat Exchanger - #2 (Unit 2)	2CC001-#2

TABLE 16.3-12 (1 of 2)

SERVICE WATER PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Unit 1:	
Service Water Pump-1A	1SW001-1A
Service Water Pump-1B	1SW002-1B
Service Water Pump-1C	1SW003-1C
Containment Fan Cooler Service Water Inlet Valves	1MOV-SW0003
Containment Fan Cooler Service Water Inlet Valves	1MOV-SW0004
Containment Fan Cooler Service Water Inlet Valves	1MOV-SW0005
Containment Fan Cooler Service Water Inlet Valves	1MOV-SW0006
Containment Fan Cooler-1A Service Water Outlet Valve	1MOV-SW0007
Containment Fan Cooler-1B Service Water Outlet Valve	1MOV-SW0010
Containment Fan Cooler-1C Service Water Outlet Valve	1MOV-SW0009
Containment Fan Cooler-1D Service Water Outlet Valve	1MOV-SW0008
Containment Fan Cooler-1E Service Water Outlet Valve	1MOV-SW0011

TABLE 16.3-12 (2 of 2)

SERVICE WATER PUMP SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Unit 2:	
Service Water Pump-2A	2SW001-2A
Service Water Pump-2B	2SW002-2B
Service Water Pump-2C	2SW003-2C
Containment Fan Cooler Service Water Inlet Valves	2MOV-SW0003
Containment Fan Cooler Service Water Inlet Valves	2MOV-SW0004
Containment Fan Cooler Service Water Inlet Valves	2MOV-SW0005
Containment Fan Cooler Service Water Inlet Valves	2MOV-SW0006
Containment Fan Cooler-2A Service Water Outlet Valve	2MOV-SW0011
Containment Fan Cooler-2B Service Water Outlet Valve	2MOV-SW0010
Containment Fan Cooler-2C Service Water Outlet Valve	2MOV-SW0009
Containment Fan Cooler-2D Service Water Outlet Valve	2MOV-SW0008
Containment Fan Cooler-2E Service Water Outlet Valve	2MOV-SW0007

TABLE 16.3-13

HYDROGEN CONTROL SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Hydrogen Recombiner 1 (2)	RV050

TABLE 16.3-14

ISOLATION SEAL WATER SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Isolation Valve Seal Water Tank	IW001
Isolation Valve Seal Water Header #1	IW031-1/2"-ER
Isolation Valve Seal Water Header #2	IW064-3/4"-X1N
Isolation Valve Seal Water Header #3	IW011-3/4"-AAR
Isolation Valve Seal Water Header #4	IW063-1/2"-EN
Isolation Valve Seal Water Header #5	IW013-3/4"X-1R
Seal Water Header #1 Isolation Valve	IW0192
Seal Water Header #2 Isolation Valve	IW0194
Seal Water Header #3 Isolation Valve	IW0190
Seal Water Header #4 Isolation Valve	IW0193
Seal Water Header #5 Isolation Valve	IW0191

TABLE 16.3-15

PENETRATION PRESSURIZATION SYSTEM

COMPONENT NAME	COMPONENT NUMBER
Penetration Pressurization Air Compressor #1	1PP001-#1
Penetration Pressurization Air Compressor #0	0PP001-#0
Penetration Pressurization Air Compressor #2	2PP001-#2

TABLE 16.3-16 (1 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
I. PHASE "A"		
AOV-BD0001	Blowdown from Steam Generator B	≤ 60
AOV-BD0002	Blowdown from Steam Generator B	≤ 60
AOV-BD0003	Blowdown from Steam Generator D	≤ 60
AOV-BD0004	Blowdown from Steam Generator D	≤ 60
AOV-BD0005	Blowdown from Steam Generator C	≤ 60
AOV-BD0006	Blowdown from Steam Generator C	≤ 60
AOV-BD0007	Blowdown from Steam Generator A	≤ 60
AOV-BD0008	Blowdown from Steam Generator A	≤ 60
FCV-BD17	Common Steam Generator Blowdown Line	≤ 60
AOV-CC9437	Cooling Water Return from Excess Letdown Hx	≤ 60
AOV-DT9157	Nitrogen to Reactor Coolant Drain Tank*	≤ 60
AOV-DT9159A & B	Reactor Coolant Drain Tank to Gas Analyzer* (AOV-DT9159A Body to Bonnet Joint Leak Rate Test)	≤ 60
AOV-DT9160A & B	Reactor Coolant Drain Tank to Waste Gas	≤ 60
AOV-DT9170 & LCV-DT1003	Reactor Coolant Drain Tank Pump Discharges* (LCV-DT1003 Packing Leak Test)	≤ 60
FCV-FP08	Fire Protection to Containment	≤ 60
FCV-IA01A & B	Instrument Air Supply to Containment*	≤ 60
FCV-PR19A & B	Reactor Vessel Leak Detection Sample	≤ 60
FCV-PR20A & B	Reactor Vessel Leak Detection Sample	≤ 60
FCV-PR21A & B	Reactor Vessel Leak Detection Sample	≤ 60
FCV-PR22A & B	Reactor Vessel Leak Detection Sample	≤ 60
FCV-PR23A & B	Reactor Vessel Leak Detection Sample	≤ 60
FCV-PR24A & B	Containment Air Particulate & Gas Monitor Inlet*	≤ 60
SOV-PR25A**	Containment Air Sample*	≤ 60
SOV-PR26A	Containment Air Sample*	≤ 60

TABLE 16.3-16 (2 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
SOV-PR25B**	Containment Air Sample*	≤ 60
SOV-PR26B	Containment Air Sample*	≤ 60
SOV-PR25C**	Containment Air Sample*	≤ 60
SOV-PR26C	Containment Air Sample*	≤ 60
SOV-PR25D**	Containment Air Sample*	≤ 60
SOV-PR26D	Containment Air Sample*	≤ 60
AOV-RC8025 & RC8026	Pressurizer Relief Tank to Gas Analyzer	≤ 60
AOV-RC8028 & RC8029	Primary Water to Pressurizer Relief Tank* (AOV-RC8029 Packing Leak Rate Test)	≤ 60
AOV-RC8033	Nitrogen to Pressurizer Relief Tank*	≤ 60
AOV-RV0001 & RV0002	Containment Purge Supply*	< 7
AOV-RV0003 & RV0004	Containment Purge Exhaust*	< 7
AOV-RV0005 & RV0006	Containment Vent Isolation* (AOV-RV0005 Packing Leak Rate Test)	< 7
FCV-RV111 & RV112	Heating Water Supply to Containment* (FCV-RV112 Packing Leak Test)	≤ 60
FCV-RV113 & RV114	Heating Water Return from Containment	≤ 60
FCV-SA01A & B	Service Air Supply to Containment* (FCV-SA01A Packing Leak Test)	≤ 60
AOV-SI8880	Nitrogen to Accumulators*	≤ 60
FCV-SS02	Steam Generator Blowdown Sample	≤ 60
FCV-SS03	Steam Generator Blowdown Sample	≤ 60
FCV-SS04	Steam Generator Blowdown Sample	≤ 60
FCV-SS05	Steam Generator Blowdown Sample	≤ 60
AOV-SS9354A & B	Pressurizer Steam Sample	≤ 60
AOV-SS9355A & B	Pressurizer Liquid Sample	≤ 60
AOV-SS9356A & B	Reactor Coolant Hot Leg Sample	≤ 60
AOV-SS9357A & B	Accumulator Sample	≤ 60
MOV-VC8100	RCP Seal Water Return	≤ 60

TABLE 16.3-16 (3 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
MOV-VC8105 & VC8106	Charging to Regenerative Heat Exchanger* (MOV-VC8105 Packing Leak Test)	≤ 60
AOV-VC8152 & VC8153	Letdown from Regenerative Heat Exchanger	≤ 60
FCV-VN02A & B	Aux. FW Pump Steamline Drain from Containment	≤ 60
FCV-VF01A & B	Hydrogen Recombiner Return to Containment	≤ 60
FCV-WD17A & B	Discharge from Containment Sump Pumps	≤ 60
II. PHASE "B"		
MOV-CC9413A & B	Cooling Water Supply to RCPs* (MOV-CC9413A Packing Leak Test)	≤ 60
MOV-CC9414	Cooling Water Return from RCP Oil Coolers*	≤ 60
MOV-CC9438 & CC685	Cooling Water Return from RCP Thermal Barriers*	≤ 60
III. MANUALLY OPERATED		
BD-0009	Blowdown From Steam Generator B	N.A.
BD-0010	Blowdown From Steam Generator B	N.A.
BD-0011	Blowdown From Steam Generator D	N.A.
BD-0012	Blowdown From Steam Generator D	N.A.
BD-0013	Blowdown From Steam Generator C	N.A.
BD-0014	Blowdown From Steam Generator C	N.A.
BD-0015	Blowdown From Steam Generator A	N.A.
BD-0016	Blowdown From Steam Generator A	N.A.
CA0062	Chemical Addition to Feedwater	N.A.
CA0063	Chemical Addition to Feedwater	N.A.
CA0061	Chemical Addition to Feedwater	N.A.
CA0064	Chemical Addition to Feedwater	N.A.
CC-9499	Cooling Water Supply to Excess Letdown Hx	N.A.
CC-9486	Cooling Water Supply to RCPs (Check)	N.A.

TABLE 16.3-16 (4 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
CC-9500	Cooling Water Supply to Excess Ltd. Hx (Check)	N.A.
CS-0005	Containment Spray Header Isolation (Check)*	N.A.
CS-0009	Containment Spray Header Isolation (Check)*	N.A.
CS-0013	Containment Spray Header Isolation (Check)*	N.A.
CS-0037	Containment Spray Pump A Recirc*	N.A.
CS-0038	Containment Spray Header Drain*	N.A.
CS-0040	Containment Spray Pump B Recirc*	N.A.
CS-0041	Containment Spray Header Drain*	N.A.
CS-0043	Containment Spray Pump C Recirc*	N.A.
CS-0044	Containment Spray Header Drain*	N.A.
CS-0052	Containment Pressure Sensor Isolation	N.A.
CS-0053	Containment Pressure Sensor Isolation	N.A.
CS-0054	Containment Pressure Sensor Isolation	N.A.
CS-0055	Containment Pressure Sensor Isolation	N.A.
CS-0056	Containment Pressure Sensor Isolation	N.A.
CS-0057	Containment Pressure Sensor Isolation	N.A.
CS-0058	Containment Pressure Sensor Isolation	N.A.
CS-0059	Containment Pressure Sensor Isolation	N.A.
DW-0030	Demineralized Flushing Water to Containment* (Packing Leak Test)	N.A.
DW-0038	Demineralized Flushing Water to Containment	N.A.
DT-9158	Nitrogen to Reactor Coolant Drain Tank (Check)*	N.A.
PP-0101	Penetration Pressurization Header Isolation*	N.A.
PP-0102	Penetration Pressurization Header Isolation*	N.A.
PP-0103	Penetration Pressurization Header Isolation*	N.A.
PP-0104	Penetration Pressurization Header Isolation*	N.A.

TABLE 16.3-16 (5 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
PR-0029	Containment Air Sample Return (Check)*	N.A.
PR-0030	Containment Air Sample Return*	N.A.
RC-8045	Nitrogen to Pressurizer Relief Tank* (Packing Leak Rate Test)	N.A.
RC-8047	Nitrogen to Pressurizer Relief Tank*	N.A.
RC-8079	Relief Valve Header to Pressurizer Relief Tank (Check)*	N.A.
SF-0010	Purification Pump to Refueling Cavity	N.A.
OSF-0011(U1)	Refueling Cavity to Purification Pump	N.A.
OSF-0012(U2)	Refueling Cavity to Purification Pump	N.A.
SF-8767	Refueling Cavity to Purification Pump* (Body to Bonnet Joint Leak Rate Test)	N.A.
SF-8787	Purification Pump to Refueling Cavity* (Body to Bonnet Joint Leak Rate Test)	N.A.
2SI-0003(U2)	SIS Test Line Grab Sample Stop	N.A.
SI0007	Nitrogen to Accumulators* (Packing Leak Rate Test)	N.A.
SI8857	Nitrogen to Accumulators Relief*	N.A.
1PI-933 Root Valve (U1)	Root Valve (U1) SIS Test Line Press. Inst. Root	N.A.
SI-8933	Nitrogen to Accumulators (Check)*	N.A.
SI-8957A & B	Residual Heat Loop Return (Checks)	N.A.
SI-8961	Accumulator Test Line Isolation*	N.A.
SI-0245	Accumulator Test Line Isolation*	N.A.
AOV-SI8888**	Accumulator Test Line Isolation	≤60
SI-9032	High Head Safety Injection (Check)	N.A.
VC-8224	Reactor Coolant Loop Fill Header (Check)	N.A.
VC-8246	Charging to Regenerative Heat Exchanger (Check)	N.A.
VC-8369A	RCP Seal Water Supply	N.A.
VC-8372A	RCP Seal Water Supply	N.A.
VC-8369B	RCP Seal Water Supply	N.A.

TABLE 16.3-16 (6 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
VC-8402A	Charging to Regen. Hx* (Packing Leak Test)	N.A.
VC-8372B	RCP Seal Water Supply	N.A.
VC-8369C	RCP Seal Water Supply	N.A.
VC-8372C	RCP Seal Water Supply	N.A.
VC-8369D	RCP Seal Water Supply	N.A.
VC-8372D	RCP Seal Water Supply	N.A.
VC-8368A	RCP Seal Water Supply (Check)	N.A.
VC-8368B	RCP Seal Water Supply (Check)	N.A.
VC-8368C	RCP Seal Water Supply (Check)	N.A.
VC-8368D	RCP Seal Water Supply (Check)	N.A.
VC-8480A & B	RC Loop Fill Header* (VC-8480A Packing Leak Test)	N.A.
VC-8402B	Charging to Regen. Hx* (Packing Leak Test)	N.A.
VC-8403	Charging to Regen. Hx* (Packing Leak Test)	N.A.
IV. OTHER		
MOV-CS0002	Containment Spray Header Isolation*	N.A.
MOV-CS0003***	CS Hdr. Isolation* (Packing and Body to Bonnet Joint)	N.A.
MOV-CS0004	Containment Spray Header Isolation*	N.A.
MOV-CS0005***	CS Hdr. Isolation* (Packing and Body to Bonnet Joint)	N.A.
MOV-CS0006	Containment Spray Header Isolation*	N.A.
MOV-CS0007***	CS Hdr. Isolation* (Packing and Body to Bonnet Joint)	N.A.
MOV-FW0016	Feedwater to Steam Generator B	N.A.
MOV-FW0017	Feedwater to Steam Generator C	N.A.
MOV-FW0018	Feedwater to Steam Generator A	N.A.
MOV-FW0019	Feedwater to Steam Generator D	N.A.
MOV-FW0050	Aux Feed to Steam Generator B	N.A.
MOV-FW0051	Aux Feed to Steam Generator B	N.A.
MOV-FW0052	Aux Feed to Steam Generator C	N.A.
MOV-FW0053	Aux Feed to Steam Generator C	N.A.
MOV-FW0054	Aux Feed to Steam Generator A	N.A.
MOV-FW0055	Aux Feed to Steam Generator A	N.A.

TABLE 16.3-16 (7 of 9)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
MOV-FW0056	Aux Feed to Steam Generator D	N.A.
MOV-FW0057	Aux Feed to Steam Generator D	N.A.
MOV-MS0006	Steam to Auxiliary Feedwater Pump	N.A.
MOV-RH8701**	Residual Heat Loop Outlet	N.A.
MOV-RH9000	Hot Leg Safety Injection	N.A.
MOV-SI8803A & B	ECCS Charging Header Discharge Isolation Valves	N.A.
MOV-SI8802	Cold Leg Safety Injection	N.A.
MOV-SI8809A & B	Residual Heat Removal to Loops	N.A.
AOV-SI8870A	Hi Head SI Header Leakoff	≤ 60
MOV-SI9011A & B	Hot Leg Safety Injection	N.A.
MOV-SW0001**	Service Water to Fan Coolers	N.A.
MOV-SW0002**	Service Water to Fan Coolers	N.A.
MOV-SW0003**	Service Water to Fan Coolers	N.A.
MOV-SW0004**	Service Water to Fan Coolers	N.A.
MOV-SW0005**	Service Water to Fan Coolers	N.A.
MOV-SW0006**	Service Water to Fan Coolers	N.A.
MOV-SW0007	Service Water Return From Fan Coolers	N.A.
MOV-SW0008	Service Water Return From Fan Coolers	N.A.
MOV-SW0009	Service Water Return From Fan Coolers	N.A.
MOV-SW0010	Service Water Return From Fan Coolers	N.A.
MOV-SW0011	Service Water Return From Fan Coolers	N.A.
MOV-CS0049	Containment Spray From A RHR Loop	≤60
MOV-CS0050	Containment Spray From B RHR Loop	≤60
MOV-SI8811A	Cont. Recirc. Sump to A RHR Train Isolation	≤60

TABLE 16.3-16 (8 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
MOV-SI8811B	Cont. Recirc. Sump to B RHR Train Isolation	≤60
HCV-VC0182***	Charging to Regen. Hx* (Packing Leak Test)	N.A.
V. MAIN STEAM ISOLATION VALVES		
HOV-MS0001	Main Steam Isolation Valve - Loop 1	≤ 5
HOV-MS0002	Main Steam Isolation Valve - Loop 2	≤ 5
HOV-MS0003	Main Steam Isolation Valve - Loop 3	≤ 5
HOV-MS0004	Main Steam Isolation Valve - Loop 4	≤ 5
FCV-MS82	MSIV Bypass Valve - Loop 1	N.A.
FCV-MS83	MSIV Bypass Valve - Loop 2	N.A.
FCV-MS84	MSIV Bypass Valve - Loop 3	N.A.
FCV-MS85	MSIV Bypass Valve - Loop 4	N.A.
MOV-MS0005	Main Steam to Aux. Feed Pump	N.A.
MOV-MS0011	Main Steam to Aux. Feed Pump	N.A.
MOV-MS0017	Main Steam Relief	N.A.
MS0014	Main Steam Relief	N.A.
MS0015	Main Steam Relief	N.A.
MS0016	Main Steam Relief	N.A.
MS0017	Main Steam Relief	N.A.
MS0018	Main Steam Relief	N.A.
MS0054 or MS0175	Main Steam Bypass	N.A.
MOV-MS0018	Main Steam Relief	N.A.
MS0019	Main Steam Relief	N.A.
MS0020	Main Steam Relief	N.A.
MS0021	Main Steam Relief	N.A.

TABLE 16.3-16 (9 of 9)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (seconds)
MS0022	Main Steam Relief	N.A.
MS0023	Main Steam Relief	N.A.
MS0055 or MS0176	Main Steam Bypass	N.A.
MOV-MS0019	Main Steam Relief	N.A.
MS0024	Main Steam Relief	N.A.
MS0025	Main Steam Relief	N.A.
MS0026	Main Steam Relief	N.A.
MS0027	Main Steam Relief	N.A.
MS0028	Main Steam Relief	N.A.
MS0056 or MS0177	Main Steam Bypass	N.A.
MOV-MS0020	Main Steam Relief	N.A.
MS0029	Main Steam Relief	N.A.
MS0030	Main Steam Relief	N.A.
MS0031	Main Steam Relief	N.A.
MS0032	Main Steam Relief	N.A.
MS0033	Main Steam Relief	N.A.
MS0057 or MS0178	Main Steam Bypass	N.A.

TABLE NOTATIONS:

* Indicates that Appendix J Leak Rate Testing is required.

** R.G. 1.97 Clarifications

--- SOV-PR25A, SOV-PR25B, SOV-PR25C, and SOV-PR25D have EQ position indication, but do not have independent power supplies. To ensure positive valve position indication for R.G. 1.97, these four SOV are normally administratively controlled closed and de-energized.

--- AOV-SI8888 has positive valve position indication for R.G. 1.97 by administratively controlling the valve normally closed and air-isolated. AOV-SI8888 is not EQ.

--- MOV-RH8701 has positive valve position indication for R.G. 1.97 by administratively controlling the valve normally closed and de-energized. MOV-RH8701 is not EQ.

--- MOV-SW0001, MOV-SW0002, MOV-SW0003, MOV-SW0004, MOV-SW0005, and MOV-SW0006 only have local control and indication. The SW valves are not EQ, but are accessible for local control during accident events. MOV-SW0001 and MOV-SW0002 are normally open, but get a signal to open during accident events. Except for this open signal, these six SW valves are post-accident service valves which do not change position during or following an accident. These valves are only rarely re-positioned. Any change to the normal lineup of these SW valves is administratively controlled. No additional considerations are made for R.G. 1.97 as is documented in UFSAR Section 7.5.3, Reference 2.

*** Indicates that position indication and stroke time testing per Containment Isolation Technical Specifications are not required. These valves are not required to close on an Isolation signal. The 10CFR50 Appendix J tested boundary in the packing gland and/or body to bonnet joint.

TABLE 16.3-17

4160-V ENGINEERED SAFEGUARDS BUS MAIN, RESERVE, AND STANDBY FEEDS

BUS	MAIN FEED BUS	RESERVE FEED BUS	STANDBY FEED DIESEL
Unit 1			
147	142	241	0
148	143	241	1A
149	144	241	1B
Unit 2			
247	242	141	0
248	243	141	2A
249	244	141	2B

TABLE 16.3-18

FAILED FUEL MONITORING INSTRUMENTS

INSTRUMENT NO.	TYPE	LOCATION
Unit 1		
1RTPR18	Scintillation Detector	Directly above volume control tank
1RTPR27	Scintillation Detector	CVCS Letdown Line
Unit 2		
2RTPR18	Scintillation Detector	Directly above volume control tank
2RTPR27	Scintillation Detector	CVCS Letdown Line

- Table 16.3-19 was Intentionally Deleted -

16.3-20

BORIC ACID SYSTEM (per unit)
RELOCATED ZION TECH SPEC 3/4.2.1.F

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
One Boric Acid System shall be OPERABLE per unit. The Boric Acid System shall consist of the following: 1. At least one boric acid storage tank containing at least 20,400 gallons of 3.2% to 3.8% by weight boric acid solution at a temperature of at least 60 degrees fahrenheit. 2. At least one OPERABLE boric acid transfer pump, and 3. At least one associated flow path from the boric acid storage tank to the suction of the charging pumps. APPLICABILITY: MODES 1,2,3,and 4. ACTION: With the Boric Acid System inoperable, restore the system to OPERABLE status within 72 hours or be in at least MODE 3 borated to the cold shutdown boron concentration within the next 6 hours; restore the Boric Acid System to OPERABLE status within the next 7 days or be in MODE 5 within the next 30 hours.	1. The Boric Acid System shall be demonstrated OPERABLE: a. At least once per 7 days by: 1. Verifying boric acid tank volume, 2. Verifying the concentration of the boric acid solution, and 3. Verifying the boric acid tank solution temperature. b. At least once per 92 days the boric acid transfer pumps shall be functionally tested.

TABLE 16.3-21

NIS DETECTOR TEMPERATURE CONTROL
RELOCATED ZION TECH SPEC 3/4.2.2.C.3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
One of the two reactor cavity ventilation fans (Unit 1 1RV012-1A, 1RV013-1B) or (Unit 2 2RV012-2A, 2RV013-2B) shall be operating whenever Tavg is greater than 145°F.	Reactor cavity ventilation fan operation shall be verified at least once per shift.

TABLE 16.3-22

PRESSURIZER SAFETY VALVES
RELOCATED ZION TECH SPEC 3/4.3.1.C.1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
At least one code safety valve shall be operable whenever the vessel is closed, except during hydrostatic tests. APPLICABILITY: RCS Temperature ≤ 200° F. ACTION: N/A	Refer to Technical Specifications.

TABLE 16.3-23

REACTOR VESSEL HEAD VENT SYSTEM
RELOCATED ZION TECH SPEC 3/4.3.1.G

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
At least one reactor vessel head vent path consisting of at least two valves in series powered from an emergency bus shall be OPERABLE and closed.* APPLICABILITY: Modes 1,2,3, and 4. ACTION: With both reactor vessel head vent paths inoperable, startup and/or power operation may continue provided the inoperable vent paths are maintained closed with power removed from the valve actuator of all the valves in the inoperable vent paths; restore at least one inoperable vent path to OPERABLE status within 30 days, or, be in MODE 3 within 6 hours and in MODE 5 within the following 30 hours. * Power may be removed from the valves to prevent inadvertent actuation.	Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by: 1. Verifying all manual isolation valves in each vent path are in the open position, 2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room while in MODE 5 or MODE 6, and 3. Verifying flow through the reactor vessel head vent paths while in MODE 5 or MODE 6.

TABLE 16.3-24

RCS PRESSURIZATION AND INTEGRITY
RELOCATED ZION TECH SPEC 3.3.2.C and 3.3.2.D

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
1. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the primary and secondary coolant is below 70°F. 2. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F. APPLICABILITY: N/A ACTION: N/A	Not applicable

TABLE 16.3-25

(Not Used)

TABLE 16.3-26 (1 of 4)

RCS STRUCTURAL INTEGRITY
RELOCATED ZION TECH SPEC 3/4.3.4

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
The structural integrity of the primary system boundary shall be maintained at a level comparable to the original acceptance standards throughout the life of a unit. Weld repairs shall be made to the original acceptance levels.	A. General The baseline inspection and all subsequent In-Service Inspections shall be done by individual contracts. As each contract period nears expiration, a review of the previous inspections shall be made in order to correlate past and future inspections. During this negotiation period, reviews will be conducted to take advantage of improvements in technology and to utilize the latest developed equipment for the inspection procedures. As always, decisions in regard to new inspection techniques shall be made to comply with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55, except where specific relief has been granted by the Commission pursuant to 10CFR50, Section 50.55.

TABLE 16.3-26 (2 of 4)

RCS STRUCTURAL INTEGRITY
RELOCATED ZION TECH SPEC 3/4.3.4

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	B. In-service Inspection and Testing 1. In-Service Inspection and Testing activities shall be incorporated into other specified surveillance requirements. The In-Service Inspection program shall be written in accordance with requirements of the ASME Section XI In-Service Inspection Code and will include the utilization of the procedures developed for the 3-1/3, 6-2/3, and 10 year In-Service Inspection. These procedures shall be used to perform baseline examination of the planned In-Service Inspection areas and shall also be used as a basis for comparing the baseline examination of the welds in the system to future Inservice Inspection Results as required in A. above. 2. Inspection following a Refueling After a Reactor Coolant System is closed following opening for refueling, all accessible pressure-retaining components, piping, and/or valves of the reactor coolant pressure boundary shall be visually examined pursuant to A. above for evidence of reactor coolant leakage while the system is under a test.

TABLE 16.3-26 (3 of 4)

RCS STRUCTURAL INTEGRITY
RELOCATED ZION TECH SPEC 3/4.3.4

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	3. Should certain ASME Section XI Code requirements as described in A. above be discovered to be impractical due to unforeseen reasons during the process of performing inspections, tests, or during review, relief will be requested from the specific Section XI Code requirements at that time. No additional reporting requirements other than those specified by Section XI ASME Boiler and Pressure Vessel Code are required.

TABLE 16.3-26 (4 of 4)

RCS STRUCTURAL INTEGRITY
RELOCATED ZION TECH SPEC 3/4.3.4

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	C. Materials Irradiation Surveillance Specimen Inspection (per unit). Specimen capsules to be used in the reactor vessel material surveillance program shall be withdrawn during the refueling period either immediately preceding or following the Effective Full Power Years (EFPY) of the unit life as follows:

UNIT 1

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
T	REMOVED (1.16)
U	REMOVED (3.52)
X	REMOVED (5.17)
Y	8.5
Z	32
W,S,V	STANDBY

UNIT 2

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
U	REMOVED (1.27)
T	REMOVED (3.56)
Y	8.5
X	13
W,S,V,Z	STANDBY

TABLE 16.3-27 (1 of 3)

RCS CHEMISTRY
RELOCATED ZION TECH SPEC 3/4.3.5

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
The Reactor Coolant System chemistry shall be maintained within the limits given below. APPLICABILITY: Modes 1,2,3,4,5, and 6. ACTION: Modes 1,2,3, and 4. a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least Mode 3 within the next six hours and in Mode 5 within the following 30 hours. b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least Mode 3 within six hours and in Mode 5 within the following 30 hours.	The Reactor Coolant System chemistry parameters specified below shall be analyzed at least once per 72 hours.

TABLE 16.3-27 (2 of 3)

RCS CHEMISTRY
RELOCATED ZION TECH SPEC 3/4.3.5

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
Modes 5 and 6 a. With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce pressurizer pressure to less than or equal to 500 psig, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.	

TABLE 16.3-27 (3 of 3)

RCS CHEMISTRY
RELOCATED ZION TECH SPEC 3/4.3.5

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

* Limit not applicable with Tavg less than or equal to 250° F.

TABLE 16.3-28 (1 of 3)

ACCIDENT MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.8.9

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
The accident monitoring instrumentation channels shown below shall be OPERABLE. APPLICABILITY: Modes 1,2,and 3. ACTION: a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least MODE 4 within the next 12 hours. b. With the number of OPERABLE accident monitoring channels less than the Minimum Operable Channels, restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in at least MODE 4 within the next 12 hours. c. The provisions of Technical Specification 3.0.4 are not applicable.	Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check and instrument channel calibration operations at the frequencies shown in the Table below.

TABLE 16.3-28 (2 of 3)

ACCIDENT MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.8.9

INSTRUMENT (PARAMETER)	TOTAL NO. OF CHANNELS	REQUIRED NO. OF CHANNELS	MINIMUM OPERABLE CHANNELS
1. PZR PORV Position Indicator*	1/valve	1/valve*	1/valve*
2. PZR PORV Block Valve Position Indicator	1/valve	1/valve*	1/valve*
3. PRZ Safety Valve Position Indicator** (Primary: Temperature Detectors)	3-(1/valve)	1/valve	1/valve
4. PZR Safety Valve Position Indicator (Backup: Acoustic Monitors)	3-(1/valve)	1/valve	N/A

* Not required if the PZR PORV Block Valve is closed and de-energized to comply with Technical Specifications.

** Direct indication of PZR Safety Valve Position - NUREG 0578, Item 2.1.3.a.

TABLE 16.3-28 (3 of 3)

ACCIDENT MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.8.9

INSTRUMENT (PARAMETER)	CHANNEL CHECK	CHANNEL CALIBRATION
1. PZR PORV Position Indicator	M	R
2. PZR PORV Block Valve Position Indicator	M	R
3. PZR Safety Valve Position Indicator** (Primary: Temperature Detectors)	M	R
4. PZR Safety Valve Position Indicator (Backup: Acoustic Monitors)	M	R

** Direct indication of PZR Safety Valve Position - NUREG 0578, Item 2.1.3.a

TABLE 16.3-29 (1 of 2)

PENETRATION PRESSURIZATION SYSTEMS
RELOCATED ZION TECH SPEC 3/4.9.2

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
Penetration Pressurization Systems A. The penetration (electrical and mechanical) and the liner weld channels shall be OPERABLE. APPLICABILITY: Modes 1,2,3, and 4. ACTION: a. With one header of the nitrogen system or one air pressurization system not OPERABLE, restore the inoperable system to OPERABLE status within four days or be in at least Mode 3 within the next four hours and in Mode 5 within the following 48 hours.* b. With air consumption in excess of 0.2% of the containment volume per day at 47 psig (950 SCFH) immediately initiate repairs.* c. With a portion of the weld channel pressurization system inoperable and not repairable by any practical means, disconnect that portion of the system.* * Technical Specification 3.0.4 does not apply.	Penetration Pressurization Systems A. Penetration and Liner Weld Channels 1. The penetrations and liner weld channels shall be demonstrated OPERABLE at least once per 31 days by verifying the system is pressurized to ≥ 47 psig. 2. The air flow to the penetrations and liner weld channels shall be checked at least once per 31 days.

TABLE 16.3-29 (2 of 2)
PENETRATION PRESSURIZATION SYSTEMS
RELOCATED ZION TECH SPEC 3/4.9.2

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. At least two of the three penetration pressurization air compressors shall be OPERABLE. APPLICABILITY: Modes 1,2,3, and 4. ACTION: a. With two of the three penetration pressurization air compressor systems inoperable, make an effort to restore at least one inoperable compressor and also verify the OPERABILITY of the Nitrogen Supply System.* b. With three penetration pressurization air compressor systems inoperable, make an effort to restore at least two inoperable compressors and also verify the OPERABILITY of both the Nitrogen Supply System and the Instrument Air System. * The provisions of Technical Specification 3.0.4 are not applicable.	B. Penetration Pressurization Air Compressors 1. The Penetration Pressurization air compressors shall be load tested at least once per each REFUELING OUTAGE to ±10% of design pressure and capacity. 2. The Nitrogen Supply System, when required by Action a, shall be demonstrated OPERABLE daily by verifying sufficient capacity exists to maintain pressure in the penetration and liner weld channels for 24 hours. 3. The Instrument Air System, when required by Action b, shall be demonstrated OPERABLE daily by verifying the Instrument Air System header pressure ≥90 psig.

TABLE 16.3-30

REFUELING OPERATIONS
COMMUNICATIONS, FUEL HANDLING SRO
RELOCATED ZION TECH SPEC 3/4.13.1.A.5&6

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
1. Direct communication between the control room and containment shall be OPERABLE.	1. Communication between the control room and the containment shall be verified before any alteration of the reactor core begins.
2. A licensed fuel handling foreman or licensed senior reactor operator shall be present at the reactor cavity during any movement of fuel within the containment.	
3. No loads heavier than the weight of a single spent fuel assemble plus the tool for moving that assembly shall be carried over fuel stored in the spent fuel pool. The spent fuel tool, the burnable poison tool, the rod cluster control changing fixture and the thimble plug shall not be carried at heights greater than two feet over fuel stored in the spent fuel pool.	

TABLE 16.3-31
REFUELING EQUIPMENT
SPENT FUEL COOLING SYSTEMS
FUEL INSPECTION PROGRAM
RELOCATED ZION TECH SPEC 3/4.13.5, 3/4.13.7, and 3/4.13.8

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
Refueling Equipment OPERABILITY:	
1. At least one of the spent fuel pit cooling system trains shall be OPERABLE.	1. At least one of the two spent fuel pit cooling system trains shall be tested and verified to be OPERABLE immediately prior to the CORE ALTERATIONS.
2. A fuel inspection program shall be established to provide such information as can be determined from inspections performed on discharged fuel. This program shall include the removal and examination of any special test assemblies.	2. Inspection of the fuel will include the following items: A. Results of tests for failed fuel. B. Results of on-site visual and non-destructive examinations on lead burnup, special test assemblies and those typical of various discharge exposures. C. Results of off-site examinations, if performed, including extent of fuel densification, fission gas generation, clad creepdown characteristics, etc.
3. The fuel transfer system and manipulator crane operability shall be verified. All interlocks shall be checked and a load test equivalent to the weight of a fuel assembly shall be made prior to refueling.	

TABLE 16.3-32 (1 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.14

1. Radiation Monitoring Instrumentation

A. The radiation monitoring instrumentation shown in the Table below shall be OPERABLE.

APPLICABILITY: As indicated in Table.

ACTION:

a. With one or more radiation monitoring channels inoperable, take the ACTION shown in the Table.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

1. Radiation Monitoring Instrumentation

A. Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and FUNCTIONAL TEST operations for the applicable MODES at the frequencies shown in the Table below.

TABLE 16.3-32 (2 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.14

Instruments	Min Channels Operable	Action #	Applicable Modes
1. Area Monitors			
A. Fuel Storage Pool Area			
1. OR-0005	1	24	All
2. OR-AR13	1	21	During Fuel Handling Operations
B. Control Room			
1. OR-0001	1	24	All
C. Technical Support Center			
1. ORE-AR31	1	24	All
2. ORE-AR32	1	24	All
3. ORE-AR33	1	24	All
D. Auxiliary Building Area			
1. OR-AR04	1	24	All
2. OR-AR08	1	24	All
3. OR-AR09	1	24	All
4. OR-AR10	1	24	All
5. OR-AR11	1	24	All
6. OR-0006	1	24	All
2. Process Monitors			
A. Containment			
1. Reactor Leak Detection			
a. 1R-PR12A	1	28	1,2,3
b. 1R-PR12B	1	28	1,2,3
c. 2R-PR12A	1	28	1,2,3
d. 2R-PR12B	1	28	1,2,3
B. Component Cooling			
1. OR-PR07	1	26	All
2. 1R-0017	1	26	All
3. 2R-0017	1	26	All

TABLE 16.3-32 (3 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION
RELOCATED ZION TECH SPEC 3/4.14

Instruments	Min Channels Operable	Action #	Applicable Modes
C. Failed Fuel			
1. 1R-PR18 or 1R-PR27	1	30	1,2
2. 2R-PR18 or 2R-PR27	1	30	1,2
D. Service Water			
1. OR-PR06	1	27	All
2. OR-PR08	1	27	1,2,3,4
3. OR-PR09	1	27	1,2,3,4
4. 1R-PR08	1	27	1,2,3,4
5. 2R-PR08	1	27	1,2,3,4
E. Steam Generator Blowdown			
1. 1R-019	1	26	1,2,3,4
2. 2R-019	1	26	1,2,3,4
F. Gas Monitor			
1. 1R-PR15	1	26	1,2,3,4
2. 2R-PR15	1	26	1,2,3,4
3. OR-PR02B	1	26	1,2,3,4
G. Technical Support Center			
1. OR-PR32A (Channel 1)	1	23	1,2,3,4
2. OR-PR32C (Channel 3)	1	23	1,2,3,4
3. OR-PR32E (Channel 5)	1	23	1,2,3,4
4. OR-PR32G (Channel 7)	1	23	1,2,3,4

TABLE 16.3-32 (4 of 8)
PLANT RADIATION MONITORING INSTRUMENTATION
TABLE NOTATION

Action 21: With the number of channels OPERABLE less than the minimum number required, stop all movement of fuel within the spent fuel pool and crane operation with loads over the spent fuel pool.

Action 23: With the number of OPERABLE channels less than the minimum number required, within 2 hours initiate and maintain the ventilation system in the recirculation mode of operation.

Action 24: With the number of OPERABLE channels less than the minimum required, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

Action 26: With the number of OPERABLE channels less than the minimm number required, perform a grab sample analysis at least once per shift. If the inoperable channel is not returned to OPERABLE status within 30 days, conduct a Station Review to determine a plan of action to restore the channel to operability.

Action 27: With the number of OPERABLE channels less than the minimum required, effluent via this pathway may continue provided the gross radioactivity level (beta/gamma or isotopic) is determined at least once per day. If the inoperable channel is not returned to OPERABLE within 30 days conduct a Station Review to determine a plan of action to restore the channel to operability.

Action 28: With the number of channels OPERABLE less than the minimum required, perform manual sampling of the containment atmosphere at least once per shift and perform at least three of the five acceptable monitoring requirements provided in Technical Specifications

Action 30: With the number of channels OPERABLE less than the minimum required, initiate an alternate method (if feasible) of monitoring the appropriate parameter(s) within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE within 7 days of the event, or

2) Conduct a Station Review within 14 days to determine a plan of action to restore the channel to OPERABLE status.

TABLE 16.3-32 (5 of 8)

PLANT RADIATION MONITORING
RELOCATED ZION TECH SPEC 3/4.14

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
1. Area Monitors				
A. Fuel Storage Pool Area				
1) 0R-0005	D	M^3	R	N/A
2) 0R-AR13	D	M^3	R	Q
B. Control Room				
1) 0R-0001	D	M	R	Q
C. Technical Support Center				
1) 0RE-AR31	D	M	R	Q
2) 0RE-AR32	D	M	R	Q
3) 0RE-AR33	D	M	R	Q
D. Auxiliary Building Area				
1) 0R-AR04	D	M	R	Q
2) 0R-AR08	D	M	R	Q
3) 0R-AR09	D	M	R	Q
4) 0R-AR10	D	M	R	Q
5) 0R-AR11	D	M	R	Q
6) 0R-0006	D	M	R	Q
2. Process Monitors				
A. Containment				
1. Reactor Leak Detection				
a) 1R-PR12A	D	N/A	R	Q
b) 1R-PR12B	D	N/A	R	Q
c) 2R-PR12A	D	N/A	R	Q
d) 2R-PR12B	D	N/A	R	Q

TABLE 16.3-32 (6 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
TABLE NOTATION

Instrument	Channel Check	Source Check	Channel Calibration(1)	Channel Functional Test (2)
B. Component Cooling				
1. 0R-PR07	D	M	R	Q
2. 1R-0017	D	M	R	Q
C. Failed Fuel				
1. 1R-PR18	D	N/A	R	Q
OR				
2. 1R-PR27	D	N/A	R	Q
3. 2R-PR18	D	N/A	R	Q
OR				
4. 2R-PR27	D	N/A	R	Q
D. Service Water				
1. 0R-PR06	D	M	R	Q
2. 0R-PR08	D	M	R	Q
3. 0R-PR09	D	M	R	Q
4. 1R-PR08	D	M	R	Q
5. 2R-PR08	D	M	R	Q

TABLE 16.3-32 (7 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
TABLE NOTATION

Instrument	Channel Check	Source Check	Channel Calibration(1)	Channel Functional Test (2)
E. Steam Generator Blowdown				
1. 1R-019	D	M	R	Q
2. 2R-019	D	M	R	Q
F. Gas Monitors				
1. 1R-PR15	D	M	R	Q
2. 2R-PR15	D	M	R	Q
3. 0R-PR02B	D	M	R	Q
G. Technical Support Center				
1. 0R-PR32A (Channel 1)	D	M	R	Q
2. 0R-PR32C (Channel 3)	D	M	R	Q
3. 0R-PR32E (Channel 5)	D	M	R	Q
4. 0R-PR32G (Channel 7)	N/A	N/A	R	Q

TABLE 16.3-32 (8 of 8)

PLANT RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
TABLE NOTATION

NOTE:

1. CHANNEL CALIBRATION shall include performance of a CHANNEL FUNCTIONAL TEST.
2. The CHANNEL FUNCTIONAL TEST shall also demonstrate that, any automatic isolation occurs and that local and remote annunciation (if installed) occurs, if any of the following conditions exist:
   a. Instrument indicates measured levels greater than the alarm setpoint.
   b. Circuit failure
   c. Instrument indicates a downscale failure
   d. Instrument controls not set in the "operate" mode.
3. Daily during fuel handling operations or load handling operations in or near the spent fuel pool.
4. Daily when purging the containment during fuel handling operations.
5. Within 72 hours prior to commencing refueling operations.

TABLE 16.3-33 (1 of 2)
FAILED FUEL MONITORING
RELOCATED ZION TECH SPECS 3.19

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
FAILED FUEL MONITORING Applicability: Applies to periodic testing and calibration of failed fuel monitoring instrumentation. Objectives: To detect failed fuel. Specification: 1. The instrumentation, 1RT-PR27 and 2RT-PR27, shall be operable during reactor power operation except as noted in Specifications 2 and 3 below.	FAILED FUEL MONITORING Applicability: Applies to periodic testing and calibration of failed fuel monitoring instrumentation. Objectives: To establish the testing and calibration frequencies of failed fuel monitoring instrumentation. Specification: 1. The instrumentation, 1RT-PR27 and 2RT-PR27, shall be functionally daily tested using a check source. If the check source is masked by the radioactivity being monitored, normal operation will be checked by noting the instrument fail indication condition instead of the check source reading. In addition, during the period in which the check source is being masked, once per week the activity will be compared to a manual sample and shall be within the following agreement. a. For RT-PR18, an area survey within 25%. b. For RT-PR27, an Iodine analysis will be run and compared to RT-PR27. The results shall agree within the accuracy of the method utilized for the Iodine analysis. The monitor shall be recalibrated if the results are outside this span.

TABLE 16.3-33 (2 of 2)
FAILED FUEL MONITORING
RELOCATED ZION TECH SPECS 3.19

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
2. Instrumentation may be taken out of service for short periods of time for testing; however, the requirements of Specification 3 below will apply.	2. The instrumentation shall be calibrated initially electronically. The calibration shall be verified during refueling operations using the following methods. A. For RT-PR18, at least two different levels of radiation shall be used. B. For RT-PR27 at least two known concentrations of radioactive materials shall be used.
3. When the required failed fuel monitoring instruments are out of service, a radioactive iodine analysis of the reactor coolant system shall be done on the unit whose detectors are inoperable.	3. An iodine analysis of the reactor coolant system shall be done once per shift on a unit whose failed fuel instrumentation is inoperable.
4. The provisions of Technical Specification 3.0.3 are not applicable.	

TABLE 16.3-34 (1 of 12)

SHOCK SUPPRESSORS
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION

3.22. Shock Suppressors (Snubbers)

1. Mechanical Snubbers

A. All safety related mechanical snubbers listed in station procedures shall be OPERABLE.

APPLICABILITY: MODES 1,2,3, and 4.

(MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.)

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.22.1.A.3 on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENT

4.22. Shock Suppressors (Snubbers)

1. Mechanical Snubbers

A. Each mechanical snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

1. Visual Inspection - Mechanical Snubbers

The visual inspection shall be performed in accordance with the following schedule.

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period *#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

* The inspection interval shall not be lengthened more than one step at a time.

TABLE 16.3-34 (2 of 12)
SHOCK SUPPRESSERS
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.1.A. 2. Visual Inspection Acceptance Criteria - Mechanical Snubbers Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE. 3. Functional Tests - Mechanical Snubbers At least once per 18 months during shutdown, a representative sample (10% of the total of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.22.1.A.4, an additional 10% of that type of snubber shall be

TABLE 16.3-34 (3 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.1.A.3. (Continued) functionally tested. The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories: a. The first snubber away from each reactor vessel nozzle, b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.), and c. Snubbers within 10 feet of the discharge from a safety relief valve. Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.* * Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

TABLE 16.3-34 (4 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.1.A.3. (Continued) In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria. For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

TABLE 16.3-34 (5 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.1.A (Continued) 4. Functional Test Acceptance Criteria - Mechanical Snubbers The mechanical snubber functional test shall verify that: a. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Drag force shall not have increased more than 50% since the last functional test. b. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression. c. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

TABLE 16.3-34 (6 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION

3.22. Shock Suppressors (Snubbers) (Continued)

2. Hydraulic Snubbers

A. All safety related hydraulic snubbers listed in station procedures shall be OPERABLE.

APPLICABILITY: MODES 1,2,3, and 4

(MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.)

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.22.2.A.3 on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENT

4.22. Shock Suppressors (Snubbers)

2. Hydraulic Snubbers

A. Each hydraulic snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

1. Visual Inspections - Hydraulic Snubbers

All hydraulic snubbers shall be visually inspected in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period *
0	18 months ± 25%
1,2	12 months ± 25%##
3,4	6 months ± 25%
5-8	124 days ± 25%
9-14	62 days ± 25%
15,16	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

* The inspection interval shall not be lengthened more than one step at a time.

TABLE 16.3-34 (7 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.2.A. Hydraulic Snubbers (Continued) 2. Visual Inspection Acceptance Criteria - Hydraulic Snubbers Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

TABLE 16.3-34 (8 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.2.A. (Continued) 3. Functional Test - Hydraulic Snubbers (except large bore) Once each refueling cycle, a representative sample of at least 10 hydraulic snubbers shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional ten snubbers shall be so tested until no more failures are found or all units have been tested. The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories: a. The first snubber away from each reactor vessel nozzle, b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.), and c. Snubbers within 10 feet of the discharge from a safety relief valve.

TABLE 16.3-34 (9 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.2.A.3. (Continued) Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.* In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria. For snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported

TABLE 16.3-34 (10 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.2.A.3. (Continued) by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.
	* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.
	4. Large Bore Hydraulic Snubber - Functional Testing Once each refueling cycle, a representative sample of at least 1 Large Bore Hydraulic Snubber shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. Large bore hydraulic snubbers have rated full load capacity of greater than 50,000 lbs. The representative sample selected for functional testing shall be a different steam generator snubber than has been previously functionally tested during this testing cycle.

TABLE 16.3-34 (11 of 12)
SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.22.2.A.4. (Continued) For snubber(s) found inoperable, an engineering evaluation shall be performed on components which are supported by the snubber(s). The purpose of the engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service. For each unit and subsequent unit found inoperable, an additional two snubbers shall be so tested until no more failures are found or all units have been tested. 5. Functional Test Acceptance Criteria - Hydraulic Snubber The hydraulic snubber functional test shall verify that: a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression. b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

TABLE 16.3-34 (12 of 12)

SHOCK SUPPRESSER
RELOCATED ZION TECH SPECS 3.22

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
4.22. Shock Suppressors (Continued)	4.22. Shock Suppressors (Continued)
3. Snubber Services Life Monitoring	3. Snubber Service Life Monitoring A record of the service life of each mechanical and hydraulic snubber, the date at which designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required. Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 16.3-35 (1 of 2)

SEALED SOURCE CONTAMINATION
RELOCATED ZION TECH SPECS 3.24

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
1. Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material, or 5 microcuries of other alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination. APPLICABILITY: At all times ACTION: a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either: 1. Decontaminate and repair the sealed source, or 2. Dispose of the sealed source in accordance with Commission Regulations. b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.	1. Test Requirements - Each sealed source shall be tested for leakage and/or contamination by: a. The licensee, or b. Other persons specifically authorized by the Commission or an Agreement State. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample. 2. Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below. a. Sources Subject to Handling - At least once per 6 months for all sealed sources containing radioactive materials: 1. With a half-life greater than 30 days (excluding Hydrogen 3), and 2. In any form other than gas.

TABLE 16.3-35 (2 of 2)

SEALED SOURCE CONTAMINATION
RELOCATED ZION TECH SPECS 3.24

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.24.2 b. Stored or Continuously Shielded Sources - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use. Sealed sources contained in shielded devices such as radiation monitors are considered to be stored unless they are removed from the shielded mechanism. c. Startup sources and fission detectors Each sealed startup source and fission detector shall be tested within 31 days prior to being installed and following repair or maintenance to the source. 3. Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination, if the contamination could have resulted from source leakage.

TABLE 16.3-36 (1 of 2)

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES
RELOCATED ZION TECH SPECS TABLE 3.3.3-1

Valve Number	Valve Size	Function	Allowable Leakage[b]
SI9012A*	2"	Safety Injection Cold Leg, Loop A	(a)
SI9012B*	2"	Safety Injection Cold Leg, Loop D	(a)
SI9012C*	2"	Safety Injection Cold Leg, Loop C	(a)
SI9012D*	2"	Safety Injection Cold Leg, Loop B	(a)
SI9004C*	2"	Safety Injection Hot Leg, Loop B	(a)
SI9004D*	2"	Safety Injection Hot Leg, Loop C	(a)
SI8905B*	4"	Safety Injection Hot Leg, Loop A	(a)
SI8905A*	4"	Safety Injection Hot Leg, Loop D	(a)
SI9001A*	8"	RHR/Safety Injection Cold Leg, Loop B	(a)
SI9001B*	8"	RHR/Safety Injection Cold Leg, Loop C	(a)
SI9001C*	8"	RHR/Safety Injection Cold Leg, Loop D	(a)
SI9001D*	8"	RHR/Safety Injection Cold Leg, Loop A	(a)
SI8949C*	8"	Safety Injection Hot Leg, Loop B	(a)
SI8949D*	8"	Safety Injection Hot Leg, Loop C	(a)
RH8949A*	8"	RHR/Safety Injection Hot Leg, Loop A	(a)
RH8949B*	8"	RHR/Safety Injection Hot Leg, Loop B	(a)
SI9002A**	8"	Residual Heat Removal Cold Leg, Loop B	(a)
SI9002B**	8"	Residual Heat Removal Cold Leg, Loop C	(a)
SI9002C**	8"	Residual Heat Removal Cold Leg, Loop D	(a)
SI9002D**	8"	Residual Heat Removal Cold Leg, Loop A	(a)
RH8736A**	8"	Residual Heat Removal Hot Leg, Loop A	(a)
RH8736B**	8"	Residual Heat Removal Hot Leg, Loop D	(a)
SI8948A**	10"	Accumulator A Cold Leg, Loop A	(a)
SI8948B**	10"	Accumulator C Cold Leg, Loop C	(a)
SI8948C**	10"	Accumulator D Cold Leg, Loop D	(a)
SI8948D**	10"	Accumulator B Cold Leg, Loop B	(a)
SI8956A**	10"	Accumulator A Cold Leg, Loop A	(a)
SI8956B**	10"	Accumulator C Cold Leg, Loop C	(a)
SI8956C**	10"	Accumulator D Cold Leg, Loop D	(a)
SI8956D**	10"	Accumulator B Cold Leg, Loop B	(a)
MOV-RH8701	14"	Residual Heat Removal Suction, Loop A	(a)
MOV-RH8702	14"	Residual Heat Removal Suction, Loop A	(a)

(NOTES PERTAINING TO THE TABLE APPEAR ON FOLLOWING PAGE)

TABLE 16.3-36 (2 of 2)

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES
RELOCATED ZION TECH SPECS TABLE 3.3.3-1

(a) Allowable Leakage Rates

1) Leakage rates ≤ than 1.0 gpm are acceptable.

2) Leakage rates > 1.0 gpm but ≤ 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum leakage rate of 5.0 gpm by ≥ 50%.

3) Leakage rates > 1.0 gpm but ≤ 5.0 gpm are unacceptable if the latest measured leakage rate exceeds the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum leakage rate of 5.0 gpm by ≥ 50%.

4) Leakage rates > 5.0 gpm are unacceptable.

(b) Test Conditions

1) Test differential pressure shall not be < 150 psid.

* Test whenever Reactor Coolant System pressure has decreased to 1850 psig.

** Test whenever Reactor Coolant System pressure has decreased to 700 psig.

TABLE 16.3-37

BOUNDARY DOORS FOR FLOOD PROTECTION
RELOCATED ZION TECH SPECS TABLE 6.8-1

In the event of the possibility of flooding, all doors listed in the Table below shall be verified closed.

Door Number	Door Zone	Location
1	1	Exterior from Service Building vestibule
5		Loading from Service Building storeroom
16	5B	Exterior from Turbine Building Track Door - Unit #1
20	4	Exterior from Service Building East Wall
69	5A	Exterior from Turbine Building West Wall - Unit #1
70	3	Exterior from Turbine Building East Wall Between Units
71	8	Exterior from Turbine Building East Wall Unit #2
72	9B	Exterior from Turbine Building West Wall Unit #2
73	9A	Exterior from Turbine Building West Wall Unit #2
277	12B	Exterior to car shed - rolling steel door
278	12A	Exterior from car shed West Wall
316	10	Exterior from Cribhouse South Wall
318	11	Exterior from Cribhouse North Wall
341	6	Exterior Radwaste Annex - Rolling Steel Door
342	19B	Exterior from Isolation Safety Valve Enclosure 1E South Side Unit #1
343	19D	Exterior from Isolation Safety Valve Enclosure 1W South Side Unit #1
344	19E	Exterior from Isolation Safety Valve Enclosure 1W North Side Unit #1
345	14A	Exterior from Isolation Safety Valve Enclosure 2E South Side Unit #2
346	14B	Exterior from Isolation Safety Valve Enclosure 2E North Side Unit #2
347	14D	Exterior from Isolation Safety Valve Enclosure 2W North Side Unit #2
348	14C	Exterior from Isolation Safety Valve Enclosure 2W South Side Unit #2
	14C	Exterior from Containment Escape Hatch Enclosure Unit 1
	19C	Exterior from Containment Escape Hatch Enclosure Unit 2

TABLE 16.3-38

CONTAINMENT VENTILATION SYSTEM
RELOCATED TECH SPEC 3/4.9.6

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
1. Containment Purge Supply and Exhaust isolation valves shall be opened only for safety-related reasons and shall be limited to a maximum opening of 50 degrees.	1. Verify at least once per 18 months that the Containment Purge and Exhaust isolation valves can not be opened greater than 50 degrees.

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17. QUALITY ASSURANCE

17.0 GENERAL QUALITY ASSURANCE PROGRAM DESCRIPTION

A comprehensive quality assurance program was used in the design and construction of the Zion Station (References 1, 2, and 3). The program assures that the control of quality has been performed and documented for each phase of material selection, fabrication, installation, and/or erection in accordance with the approved specifications and drawings. This program has been in effect since 1968. The program relates principally to the reactor coolant and safety systems, the containment, and other components necessary for the safety of the nuclear portion of the plant including the augmented quality requirements (Section 19 of the QAM). However, procedures are followed to assure appropriate levels of quality for the balance of the plant. The basic quality areas covered by the ComEd program are the same as those presented in 10CFR50 Appendix B and in ANSI N45.2-1971.

The Quality Assurance Program is implemented in accordance with the current, NRC approved revision to the ComEd Quality Assurance Topical Report, CE-1-A.

17.1 REFERENCES, CHAPTER 17

1. Zion Station PSAR, Quality Assurance Plan, Docket Numbers 50-295 and 50-304.

2. Westinghouse Quality Assurance Plan, as filed in Docket Number 50-327, Sequoyah Nuclear Plant.

3. Commonwealth Edison Quality Assurance Program, dated 1975 (Topical Report No. CE-1-A).