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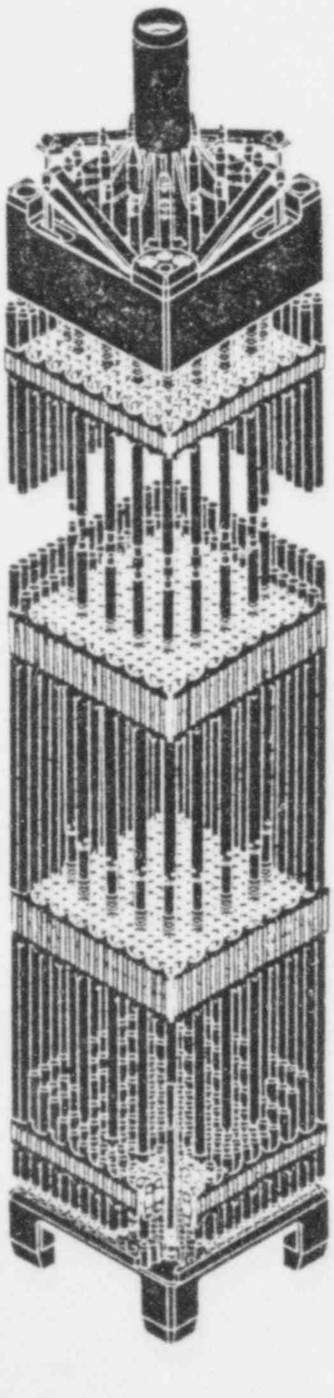

REFERENCE CORE REPORT

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OPTIMIZED FUEL ASSEMBLY

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15.0 ACCIDENT ANALYSIS

This chapter addresses the representative initiating events listed on pages 15-10, 15-11, and 15-12 of Regulatory Guide 1.70, Revision 3 as they apply.

Certain items in the guide warrant comment, as follows:

Items 1.3 and 2.1 - There are no pressure regulators in the Nuclear Steam Supply System (NSSS) pressurized water reactor (PWR) design whose malfunction or failure could cause a steam flow transient.

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.
4. Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

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15.0.1.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences 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 occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition 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 listed below:

1. Steady state and shutdown operations

   a. Power operation (>5 to 100 percent of rated thermal power)

   b. Startup ($K_{eff} \geq 0.99$ to ≥ 5 percent of rated thermal power)

   c. Hot standby (subcritical, Residual Heat Removal System isolated)

   d. Hot shutdown (subcritical, Residual Heat Removal System in operation)

   e. Cold shutdown (subcritical, Residual Heat Removal System in operation)

   f. Refueling

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2. Operation with permissible deviations

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

a. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service)

b. Radioactivity in the reactor coolant, due to leakage from fuel with cladding defects.

1) Fission products
2) Corrosion products
3) Tritium

c. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications

d. Testing as allowed by Technical Specifications

3. Operational transients

a. Plant heatup and cooldown (up to $100^{o}F$/hour for the Reactor Coolant System; $200^{o}F$/hour for the pressurizer during cooldown and $100^{o}F$/hour for the pressurizer during heatup)

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

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

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

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15.0.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. 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 or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

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

2. Feedwater system malfunctions that result in a increase in feedwater temperature (Section 15.1.2).

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

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

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

6. Turbine trip (Section 15.2.3).

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

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

9. Loss of nonemergency AC power to the station auxiliaries (Section 15.2.6).

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

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

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

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

14. Rod cluster control assembly misalignment (dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly) (Section 15.4.3).

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

16. Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant (Section 15.4.6).

17. Inadvertent operation of the Emergency Core Cooling System during power operation (Section 15.5.1).

18. Chemical and Volume Control System malfunction that increases reactor coolant inventory (Section 15.5.2).

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

20. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate Containment (Section 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequenlty during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although

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sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or Containment barriers. For the purposes of this report the following faults are included in this category:

1. Steam system piping failure (minor) (Section 15.1.15).

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

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

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

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

6. Gaseous Radwaste System leak or failure (Section 15.7.1).

7. Liquid Radwaste System leak or failure (Section 15.7.2).

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

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

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults 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 fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the Containment. For the purposes of this report, the following faults have been classified in this category:

1. Steam system piping failure (major) (Section 15.1.5).

2. Feedwater system pipe break (Section 15.2.8).

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

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

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

6. Steam generator tube failure (Section 15.6.3).

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

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

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 transient performance.

For each mode of plant oepration, 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 comprises an analysis of the following control systems: rod cluster control assembly, steam dump, steam generator, level, pressurizer pressure, and pressurizer level.

15.0.3 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES

15.0.3.1 Design Plant Conditions

Table 15.0.3-1 lists the principal power rating values which are assumed in analyses performed in this report. Two ratings are given:

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

2. The engineered safety features design rating. The NSSS supplied engineered safety features are designed for thermal power higher than the guaranteed value in order not to preclude realizaton of future potential power capability. This higher thermal power value is designated as the engineered safety features design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where the initial power operating conditions are assumed in accident analyses, the guaranteed NSSS thermal power output is assumed. Where demonstration of adequacy of the containment and engineered safety features are concerned, the engineered safety features design rating is assumed. Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The thermal power

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values used for each transient analyzed are given in Table 15.0.3-1. In all cases where the 3581 megawatt thermal (MWt) rating is used in an analysis, the resulting transients and consequences are conservative compared to using the 3427 MWt rating.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0.3-3.

15.0.3.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assum . The allowances on power, temperature, and pressure determined on a statistical basis and are included in the limit DNBR, as described in WCAP-8567 (Reference 10). This procedure is known as the "Improved Thermal Design Procedure," and is discussed more fully in Section 4.4.

For accidents which are not DNB limited, or in which the Improved Thermal Design Procedure is not employed the initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors were assumed in the analysis:

1. Core Power	±2 percent allowance for calorimetric error
2. Average Reactor Coolant System temperature	±4°F allowance for controller deadband and measurement error
3. Pressurizer pressure	±30 pounds per square inch (psi) allowance for steady state fluctuations and measurement error

Table 15.0.3-2 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the Improved Thermal Design Procedure.

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 control rods and operating instructions. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_q). The peaking factor limits are given in the Technical Specifications.

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 15.0.3-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.

The radial and axial power distributions described above are input to the THINC Code as described in Section 4.4.

For transients which may be overpower limited, the total peaking factor (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 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 C[illegible]cal and Volume Control System malfunction that results in a [illegible] in the boron concentration in the reactor coolant incident wh[illegible] many minutes, and the excessive increase in secondary ste[illegible] which may reach equilibrium without causing a reactor trip, th[illegible] fuel rod thermal evaluations are performed as discussed in Sect on 4.4. For overpower transients which are fast with respect to the fuel rod thermal time

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c[illegible]ant, for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical or low power startup and rod cluster control assembly ejection incidents which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation must be performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and rod power, a typical value at beginning-of-life for high power rods is approximately 5 seconds.

15.0.4 REACTIVITY COEFFICIENTS ASSUMED IN THE ACCIDENT ANALYSES

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

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of reactor coolant from cracks or ruptures in the Reactor Coolant System do not depend on reactivity feedback effects. The values are given in Table 15.0.3-2. Reference is made in that table to Figure 15.0.4-1 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life, 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 versus time of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel.

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The rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 15.0.5-1. The rod cluster control assembly insertion time to dashpot entry is taken as 3.05 seconds. Drop time testing requirements are dependent on the type of cluster control assemblies actually used in the plant and are specified in the plant Technical Specifications.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0.5-2 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 rod cluster control assembly negative reactivity insertion versus time is shown in Figure 15.0.5-3. The curve shown in this figure was obtained from Figures 15.0.5-1 and 15.0.5-2. A total negative reactivity insertion following a trip of 4 percent ΔK is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3.2-3. For Figures 15.0.5-1 and 15.0.5-2, the rod cluster control assembly drop time is normalized to 3.05 seconds, unless otherwise noted for a particular event.

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The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) 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 rod cluster control assembly position versus time of Figure 15.0.5-1 is used as code input.

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

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0.6-1.

Reference is made in that table to Overtemperature and Overpower ΔT trip shown in Figure 15.0.3-1. This figure presents the allowable Reactor Coolant Loop Average Temperature and ΔT for the design flow and power distribution, as described in Section 4.4, 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

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nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.47 for the thimble cell and 1.49 for the typical cell.) All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. 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 pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

The limit value, which was used as the DNBR limit for all accidents analyzed with the Improved Thermal Design Procedure (see Table 15.0.3-2), is conservative compared to the actual design DNBR value (1.31 for the thimble cell and 1.33 for the typical cell) required to meet the DNB design basis as discussed in Section 4.4.

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 plant Technical Specifications. During plant startup tests, it will be demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the Technical Specifications.

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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 15.0.7-1.

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

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the 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.

15.0.8 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

The NSSS is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions and dynamic effects of postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Reference[9] discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0.8-1 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI-N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

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In the analysis of the Chapter 15 events, control system action, is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of the Chapter 15 events.

15.0.9 FISSION PRODUCT INVENTORIES

15.0.9.1 Activities in the Core

The calculation of the core iodine fission product inventory is consistent with the inventories given in TID-14844 (Reference 1) and is based on a core power level of 3565 MWt. The fission product inventories for other isotopes which are important from a health hazards point of view are calculated using the data from NEDO-12154-1 (Reference 2). These inventories are given in Table 15.0.9-1. The isotopes included in Table 15.0-9-1 are the isotopes controlling from considerations of inhalation dose (iodines) and from external dose due to immersion (noble gases).

The Equilibrium Appearence rate of Iodines in the RCS due to conservative and realistic fuel defects are shown in Table 15.0.9-2.

The isotopic yields used in the calculations are from the data of NEDO-12154-1, utilizing the isotopic yield data for thermal fissioning of U-235 as the sole fissioning source. The change in fission product inventory resulting from the fissioning of other fissionable atoms has been reviewed. The results of this review indicated that inclusion of all fission source data would result in small (less than 10%) change in the isotopic inventories.

15.0.9.2 Activities in the Fuel Pellet Clad Gap

The fuel-clad gap activities were determined using the model given in Regulatory Guide 1.77. Thus, the amount of activity accumulated in the

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fuel-clad gap is assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life. The gap activities are given in Table 15.0.9-1.

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss of coolant accident per the requirements of Appendix K of 10CFR50.46 (Reference 3) as described in (References 4 and 5). These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following Loss of Coolant Accident

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

For example, consider the transient resulting from the postulated double ended break of the largest Reactor Coolant System pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative

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estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the Reactor Coolant System pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0.3-2.

15.0.11.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nulcear 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.

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

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FACTRAN is further discussed in Reference [6].

15.0.11.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam geneator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, Overtemperature ΔT, Overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The Emergency Core Cooling System, including the accumulators and upper head injection, 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 15.0.3-1. The core limits represents the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference [7].

15.0.11.3 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

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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., channelwise 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 [8].

15.0.11.4 THINC

The THINC Code is described in Section 4.4.

15.0.11.5 References

1. DiNunno, J. J., et al., "Calculation for Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

2. Meek, M. E. and Rider, B. F., "Summary of Fission Product Yields for U-235, U-238, Pu-239, and Pu-241 at Thermal Fission Spectrum and 14 Mev Neutron Energies," APED-5398, March 1968.

3. "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.

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4. 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.

5. Bordelon, F. M., et al, "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June 1974.

6. Hargrove H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO^2 Fuel Rod," WCAP-7908, June 1972.

7. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972.

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

9. "Westinghouse Nuclear Energy Systems Division Quality Assurance Plan," WCAP-8370-A.

10. H. Chelemer, et al, "Improved Thermal Design Procedure," WCAP-8567-P (Proprietary), July, 1975, and WCAP-8568 (Non-Proprietary), July 1975.

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TABLE 15.0.3-1

Nuclear Steam Supply System Power Ratings

Guaranteed NSSS thermal power output (MWt)	3427
Engineered safety features design rating (maximum calculated turbine rating) (MWt)	3581
Thermal power generated by the reactor coolant pumps (MWt)	16
Reactor core thermal power output (MWt)*	[illegible]

* Radiological consequences based on 3565 (MWt) power level.

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SUMMARY OF INITIAL COND

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED* MODERATOR TEMPERATURE ($\Delta K/^{o}F$)	MODERATOR DENSITY (ΔK/gm/cc)	DOPPLER	DNB CORRELATI
15.1 Increase in Heat Removal by the Secondary System					
- Feedwater System Mal-function Causing an Increase in Feedwater Flow	LOFTRAN	NA	0.43	Minimum*	WRB-1
- Excessive Increase in Secondary Steam Flow	LOFTRAN	NA	Figure 15.0.3-2 and 0.43	Maximum and Minimum*	WRB-1
- Accidental Depres-surization of the Main Steam System	LOFTRAN	NA	Function of Moderator Density, See Subsection 15.1-4 (Figure 15.1.4-1)	-2.2 pcm/^{o}F	W-3
- Steam System Piping Failure	THINC, LOFTRAN	NA	Function of Moderator Density, See Subsection 15.1.5 (Figure 15.1.4-1)	See Secti n 15.1.5	W-3
15.2 Decrease in Heat Removal by the Secondary System					
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	NA	Figure 15.0.3-2 and 0.43	Maximum* and minimum	WRB-1
- Loss of Non-Emer-gency A-C Power to the Station Auxiliaries	LOFTRAN	NA	Figure 15.0.3-2	Maximum*	NA
- Loss of Normal Feedwater Flow	LOFTRAN	NA	Figure 15.0.3-2	Maximum*	NA
- Feedwater System Pipe Break	LOFTRAN	NA	Figure 15.0.3-2	Maximum*	NA

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BLE 15.0.3-2

ITIONS AND COMPUTER CODES USED

ON	IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT[a] (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL INLET TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (°F)
	Yes	0 and 3427	387,600	590.8	2250	1100	440
	Yes	3427	387,600	590.8	2250	1100	440
	No	0 (Subcritical)	373,200	557	2250	483	42
	No	0 (Subcritical)	373,200	557	2250	483	42
	Yes	3427	387,600	590.8	2250	1100	440
	NA	3581	373,200	596.2	2280	1150	445
	NA	3581	373,200	596.2	2280	1150	445
	NA	3581	373,200	596.2	2280	1150	445

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		REACTIVITY COEFFICIENTS ASSUMED*			
FAULTS	COMPUTER CODES UTILIZED	MODERATOR TEMPERATURE ($\Delta K/^{o}F$)	MODERATOR DENSITY ($\Delta K/gm/cc$)	DOPPLER	D CORRE
15.3 Decrease in Reactor Coolant System Flow Rate					
- Partial and Complete Loss of Forced Reactor Coolant Flow	LOFTRAN, THINC, FACTRAN	NA	Figure 15.0.3-2	Maximum*	WR
- Reactor Coolant Pump Shaft Seizure (Locked Rotor	LOFTRAN, FACTRAN	NA	Figure 15.0.3-2	Maximum*	WRE
15.4 Reactivity and Power Distribution Anomalies					
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	TWINKLE, FACTRAN THINC	Refer to Section 15.4.1.2	Refer to Subsection 15.4.1.2	Consistent with upper limit shown on Figure 15.0.4-1	WRE
- Uncontrolled Rod Cluster Assemlby Bank Withdrawal at Power	LOFTRAN	NA	Figure 15.0.3-2 and 0.43	Maximum and Minimum*	WRE
- Control Rod Mis-alignment	THINC	NA	NA	NA	WRE
- Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	THINC, LOFTRAN, FACTRAN	NA	0.43	Minimum*	WRE
- Chemical and Volume Control System Mal-function that Results in a Decrease in Boron Concentration in the Reactor Coolant	NA	NA	NA	NA	NA

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TABLE 15.0.3-2 (Continued)

B ATION	IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT[a] (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL INLET TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATUR (°F)
-1	Yes	3427 and 2400	387,600	590.8	2250	1100	440
-1	No	3427 and 2400	373,200	594.8	2280	1100	440
-1	Yes	0	387,600	590.8	2250	NA	NA
-1	Yes	3427/2056/ 343	387,600	590.8/577.3/ 560.4	2250	1150/867/ 575	423/389/ 150
-1	Yes	3427	387,600	590.8	2250	NA	NA
-1	Yes	2400	281,300	580.7	2250	891	404
	NA	0 and 3427	NA	NA	NA	NA	NA

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FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED* MODERATOR TEMPERATURE ($\Delta K/^{o}F$)	MODERATOR DENSITY ($\Delta K/gm/cc$)	DOPPLER	DNB CORRELATI[illegible]
- Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Refer to Section 4.3	NA	NA	NA	NA
- Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	Refer to Subsection 15.4.8 min., max. feedback	Refer to Subsection 15.4.8 min., max. feedback	Consistent with lower limit shown on Figure 15.0.4-1	NA
15.5 Increase in Coolant Inventory					
- Inadvertent Operation of ECCS During Power Operation	LOFTRAN	NA	Figure 15.0.3-2	Minimum*	WRB-1
15.6 Decrease in Reactor Coolant Inventory					
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	LOFTRAN	NA	Figure 15.0.3-2	Maximum*	WRB-1

* See Figure 15.0.4-1. Maximum refers to lower curve and minimum refers to upper curve.

NA - Not Applicable

BOC - Beginning of Cycle

EOC - End of Cycle

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TABLE 15.0.3-2 (Continued)

N	IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT[a] (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL INLET TEMPERATURE (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (°F)
	NA	3427	387,600	590.8	2250	1100	440
	NA	0 and 3427	373,200	596.2	2220	NA	NA
	Yes	3427	387,600	590.8	2250	1100	440
	Yes	3427	387,600	590.8	2250	1100	440

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TABLE 15.0.3-3

Nominal Values of Pertinent Plant Parameters Utilized In The Accident Analyses*

Thermal output of NSSS (MWt)[a]	3427
Core inlet temperature (°F)	561.6
Vessel average temperature (°F)	590.8
Reactor Coolant System pressure (psia)	2250
Reactor coolant flow per loop (gpm)	96,900
Total Reactor Coolant flow (10^6 lb/hr)	143.4
Steam flow from NSSS (10^6 lb/hr)	15.14
Steam pressure at steam generator outlet (psia)	1000
Maximum steam moisture content (%)	0.25
Assumed feedwater temperature at steam generator inlet (°F)	440
Average core heat flux (Btu/hr-ft^2)	197,200

*For accident analyses using the Improved Thermal Design Procedure

[a]See Table 15.0.3-2

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TABLE 15.0.3-3a

Nominal Values of Pertinent Plant Parameters Utilized In The Accident Analyses*

Thermal output of NSSS (MWt)[b]	3427
Core inlet temperature (°F)	560.6
Vessel average temperature (°F)	590.8
Reactor Coolant System pressure (psia)	2250
Reactor coolant flow per loop (gpm)	93,300
Total Reactor Coolant flow (10^6 lb/hr)	138.3
Steam flow from NSSS (10^6 lb/hr)	15.15
Steam pressure at steam generator outlet (psia)	1000
Maximum steam moisture content (%)	0.25
Assumed feedwater temperature at steam generator inlet (°F)	440
Average core heat flux (Btu/hr-ft^2)	197,200

*For accident analyses not using the Improved Thermal Design Procedure

[b]See Table 15.0.3-2

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TABLE 15.0.6-1

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Power range neutron flux, high negative rate	3.5% 1 second	0.5
High neutron flux, P-8	85%	0.5
Overtemperature ΔT	Variable see Figure 15.0.3-1	6.0[a]
Overpower ΔT	Variable see Figure 15.0.3-1	6.0[a]
High pressurizer pressure	2410 psig	2.0
Low pressurizer pressure	1921 psig	2.0

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[a]Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

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TABLE 15.0.6-1 (Page 2)

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Low reactor coolant flow (From loop flow detectors)	87% loop flow	1.0
Undervoltage trip	68% nominal	1.5
Turbine trip	Not applicable	2.0
Low-low steam generator level	3.9% of narrow range level span	2.0
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	93.6% of narrow range level span	2.0

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TABLE 15.0.7-1

Determination of Maximum Overpower Trip Point - Power Range Neutron Flux Channel - Based on Nominal Setpoint Considering Inherent Instrument Errors

Variable	Accuracy of Measurement of Variable (% error)	Effect On Thermal Power Determination (% error) (Estimated)	(Assumed)
Calorimetric Errors in the Measurement of Secondary System Thermal Power:			
Feedwater temperature	± 0.5		
Feedwater pressure (small correction on enthalpy)	± 0.5	0.3	
Steam pressure (small correction on enthalpy)	± 2		
Feedwater flow	± 1.25	1.25	
Assumed Calorimetric Error (% of rated power)			± 2(a)
Axial power distribution effects on total ion chamber current			
Estimated Error (% of rated power)		3	
Assumed Error (% of rated power)			± 5(b)

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TABLE 15.0.7-1 (Continued)

Determination of Maximum Overpower Trip Point - Power Range Neutron Flux Channel - Based on Nominal Setpoint Considering Inherent Instrument Errors

Variable	Accuracy of Measurement of Variable (% error)	Effect On Thermal Power Determination (% error) (Estimated)	(Assumed)
Instrumentation channel drift and setpoint reproducibility			
Estimated Error (% of rated power)		1	
Assumed Error (% of rated power)			± 2(c)
Total assumed error in setpoint (a) + (b) + (c)			± 9

	Percent of Rated Power
Nominal Setpoint	109
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction	118

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TABLE 15.0.8-1 (Page 1)

Plant Systems and Equipment Available for
Transients and Accident Conditions

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1	INCREASE IN HEAT REMOVED BY THE SECONDARY SYSTEM				
	Feedwater system malfunctions that result in an increase in feedwater flow	Power range high flux, steam generator lo-lo level (Intact steam generators), manual	High steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	NA
	Excessive increase in secondary steam flow	Power range high flux Overtemperature ΔT, Overpower ΔT, manual	NA	Pressurizer self-actuated safety valves steam generator safety valves	NA
	Inadvertent opening of a steam generator relief or safety valve	Low pressurizer pressure, manual sis	Low pressurizer pressure, low compensated steam line pressure, hi-hi containment pressure, high negative steamline pressure rate, manual	Feedwater isolation valves, steam line stop valves	Auxiliary Feedwater System, Safety Injection System
	Steam system piping failure	SIS, low pressurizer pressure, manual	Low pressurizer pressure, low compensated steam line pressure, hi-hi containment pressure, high negative steamline pressure rate, manual	Feedwater isolation valves, steam line stop valves	Auxiliary Feedwater System, Safety Injection System

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TABLE 15.0.8-1 (Page 2)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	Loss of external electrical load/ turbine trip	High pressurizer pressure Overtemperature ΔT, lo-lo steam generator level, manual	Steam Generator lo-lo level	Pressurizer safety valves, steam generator safety valves	Auxiliary Feedwater System
	Loss of non-emergency AC power to the station auxiliaries	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator valves	Auxiliary Feedwater System
	Loss of normal feedwater flow	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator valves	Auxiliary Feedwater System
	Feedwater System pipe break	Steam generator lo-lo level, high pressurizer pressure, SIS, manual	High Containment pressure, steam generator lo-lo water level, low compensated steam line pressure	Steam line isolation valves, feedline isolation, pressurizer self-actuated safety valves, steam generator safety valves	Auxiliary Feedwater System, Safety Injection System
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE				
	Partial and complete loss of forced reactor coolant flow	Low flow, undervoltage, underfrequency, manual	NA	Steam generator safety valves	NA
	Reactor coolant pump shaft seizure (locked rotor)	Low flow, manual	NA	Pressurizer safety valves, steam generator safety valves	NA

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TABLE 15.0.8-1 (Page 3)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES				
	Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	Power range high flux (low setpoint), manual	NA	NA	NA
	Uncontrolled rod cluster control assembly bank withdrawal at power	Power range high flux, Overtemperature ΔT, high pressurizer pressure, manual	NA	Pressurizer safety valves, steam generator safety valves	NA
	Rod cluster control assembly misalignment	Power range negative flux rate, manual			
	Startup of an inactive reactor coolant loop at an incorrect temperature	Power range high flux, P-8, manual	NA	Low insertion limit annunciators for boration tion	NA
	Chemical and Volume Control System malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, Overtemperature ΔT,	NA	Low insertion limit annunciators for boration	NA
	Spectrum of rod cluster control assembly ejection accidents	Power range high flux, high positive flux rate, manual	NA	NA	NA

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TABLE 15.0.8-1 (Page 4)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.5	INCREASE IN REACTOR COOLANT INVENTORY				
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure, manual, safety injection trip	NA	NA	Safety Injection System
15.6	DECREASE IN REACTOR COOLANT INVENTORY				
	Inadvertent opening of a pressurizer safety or relief valve	Pressurizer low pressure, Overtemperature ΔT, manual	Pressurizer Low Pressure	NA	Safety Injection System
	Steam generator tube failure	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator shell side fluid operating system, steam generator safety and/or relief valves, steam line stop valves	Emergency Core Cooling System, Auxiliary Feedwater System Emergency Power System
	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System steam generator safety and/or relief valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

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TABLE 15.0.9-1

Iodine and Noble Gas Inventory in Reactor Core

and Fuel Rod Gaps*

Isotope	Core Activity (Curies)	Fraction of Activity in Gap** (%)	Gap Activity (Curies)
I-131	9.9×10^6	.10	9.9×10^5
I-132	1.4×10^7	.10	1.4×10^6
I-133	2.0×10^7	.10	2.0×10^6
I-134	2.2×10^7	.10	2.2×10^6
I-135	1.9×10^7	.10	1.9×10^6
Xe-131m	7.0×10^4	.10	7.0×10^3
Xe-133	2.9×10^6	.10	2.9×10^6
Xe-133m	1.9×10^7	.10	1.9×10^7
Xe-135	4.0×10^6	.10	4.0×10^6
Xe-135m	4.2×10^6	.10	4.2×10^6
Xe-138	1.6×10^7	.10	1.6×10^7
Kr-83m	1.2×10^6	.10	1.2×10^5
Kr-85	2.7×10^6	.10	2.7×10^5
Kr-85m	2.0×10^5	.10	2.0×10^4
Kr-87	4.9×10^6	.10	4.9×10^5
Kr-88	7.0×10^6	.10	7.0×10^5
Kr-89	8.7×10^6	.10	8.7×10^5

* Based on 650 days of operation

** NRC assumption in Regulatory Guide 1.25

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TABLE 15.0.9-2

Iodine Appearance Rates in Reactor Coolant

		I-131	I-132	I-133	I-134	I-135
Equilibrium Appearance Rate of Iodines in the RCS due to Fuel Defects (μgram/sec)	Conservative Case*	2.0(-2)	1.0E(-3)	5.2E(-3)	2.1E(-4)	1.7E(-3)
	Realistic Case	2.4E(-3)	1.2E(-4)	6.2E(-4)	2.5E(-4)	2.0E(-4)
Appearance Rate of Iodines in the RCS due to Iodine Spike (μgram/sec)**	Conservative Case	1.0E(-1)	5.0E(-1)	2.6E(0)	1.1E(-1)	8.5E(-1)
	Realistic Case	1.2E(0)	6.0E(-2)	3.1E(-1)	1.3E(-2)	1.0E(-1)

* Conservative case is based on 1.0% fuel defect level while realistic case is based on .15% fuel defect level.

** Iodine spike assumed to be 500 times the equilibrium rate

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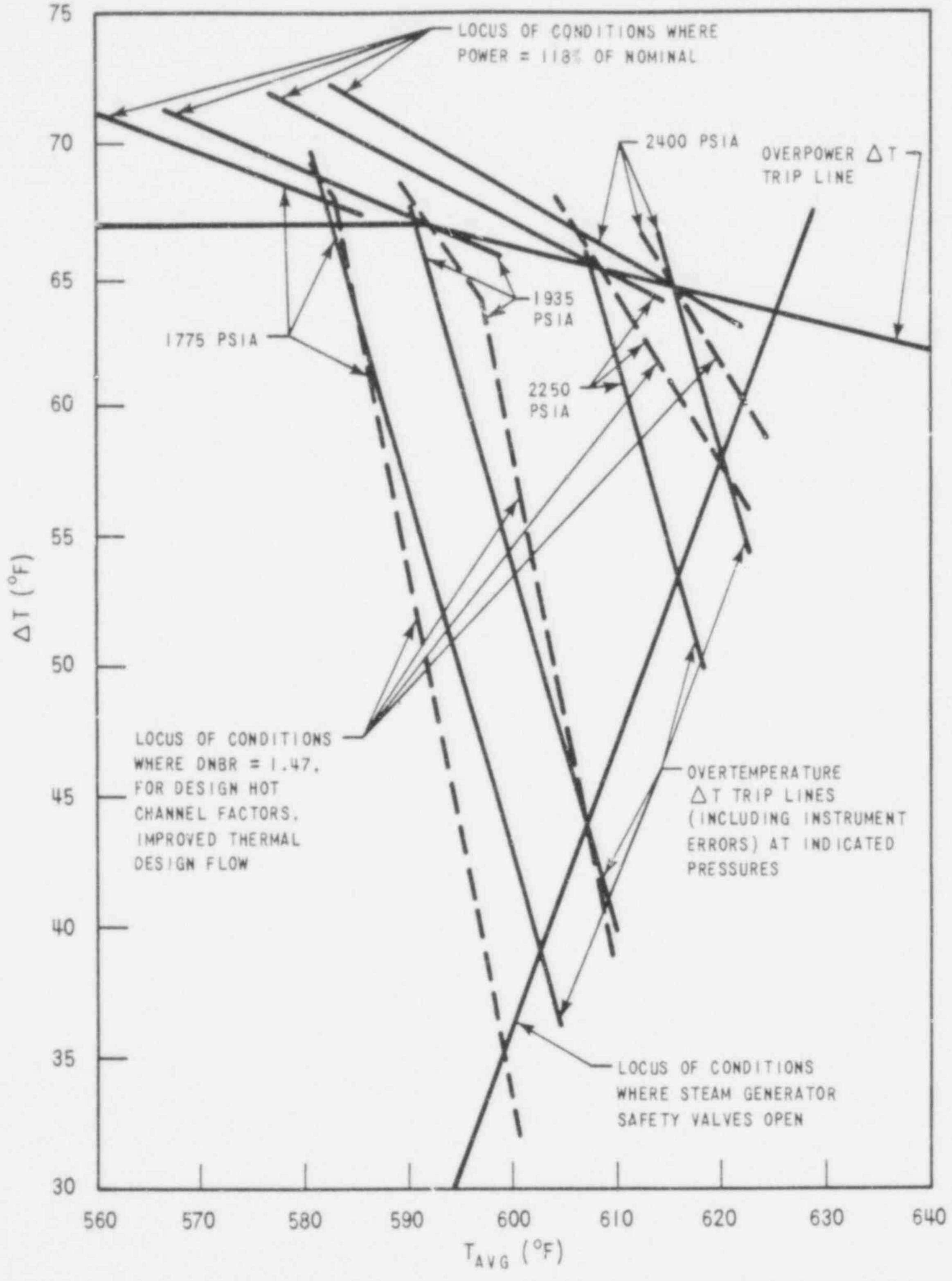

WCAP - 9500

Figure 15.0.3-1.

Illustration of Overtemperature and Overpower ΔT Protection

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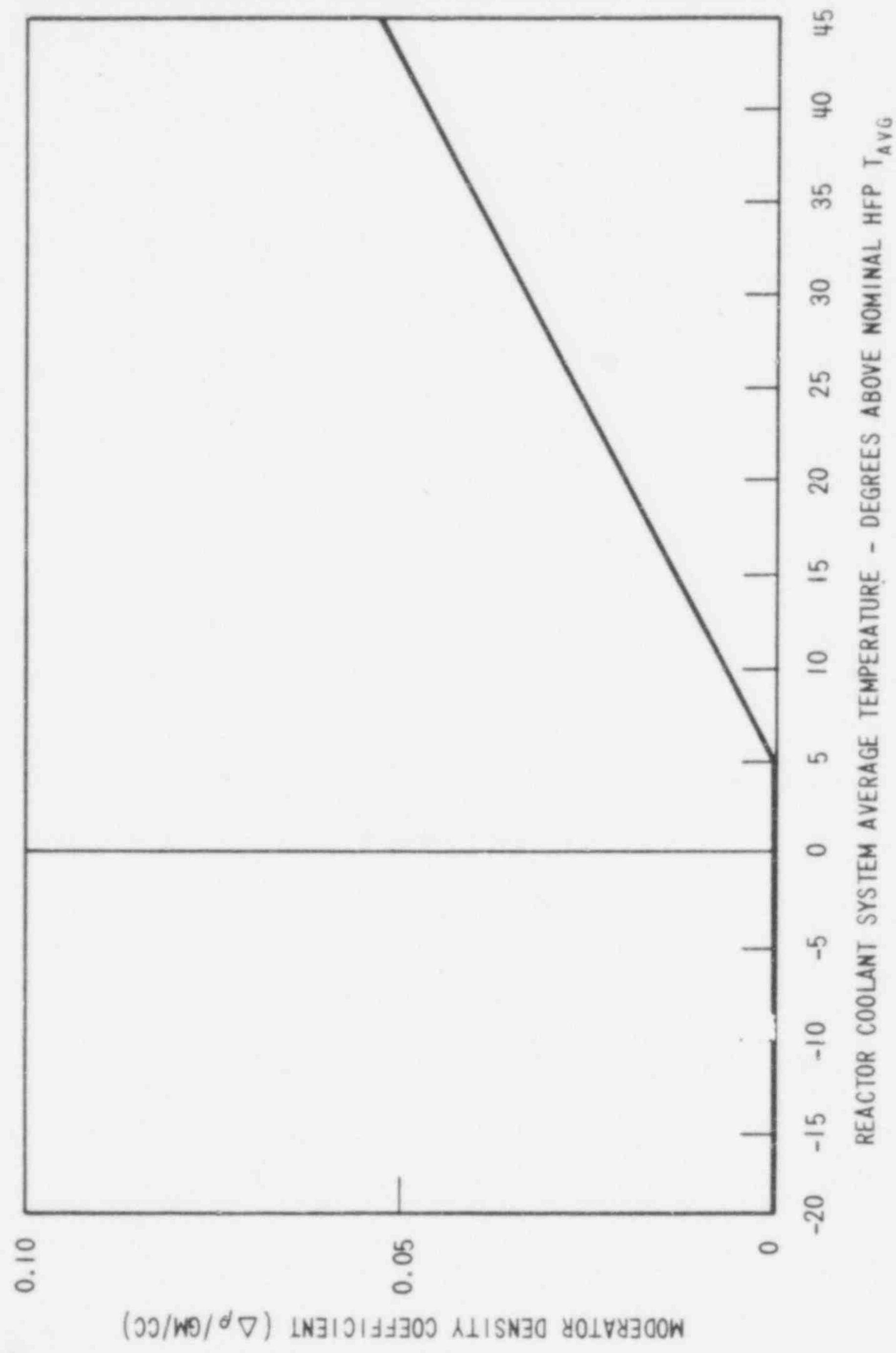

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WCAP – 9500

Figure 15.0.3-2.
Minimum Moderator Density Coefficient
Used in Analysis

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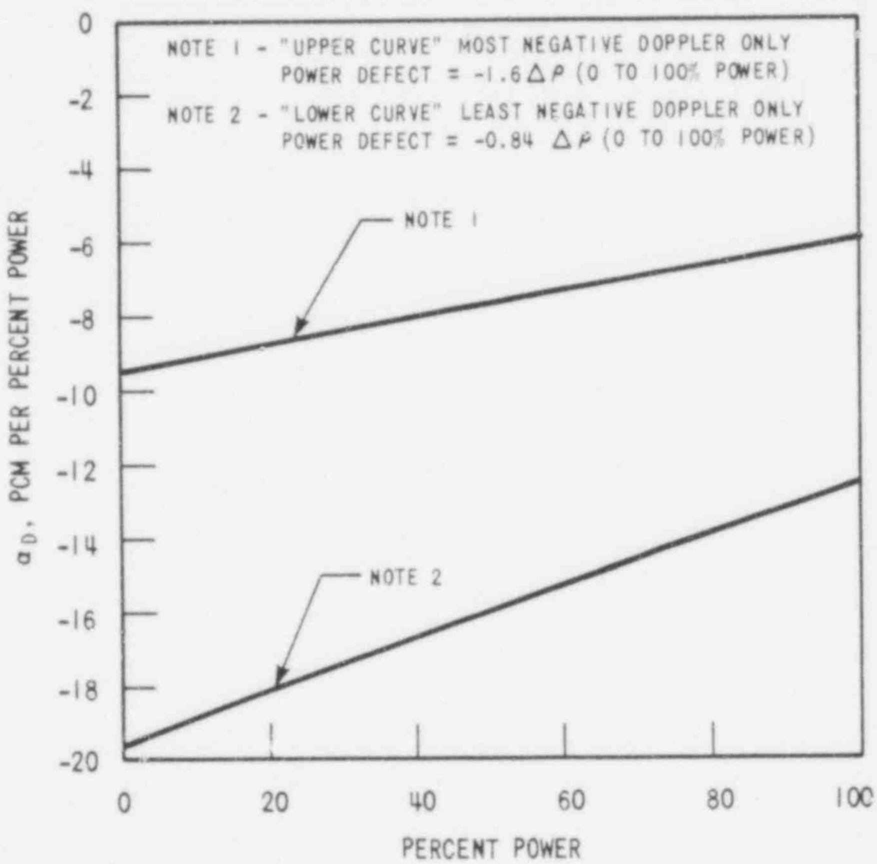

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Figure 15.0.4-1.

Doppler Power Coefficient Used in Accident Analysis

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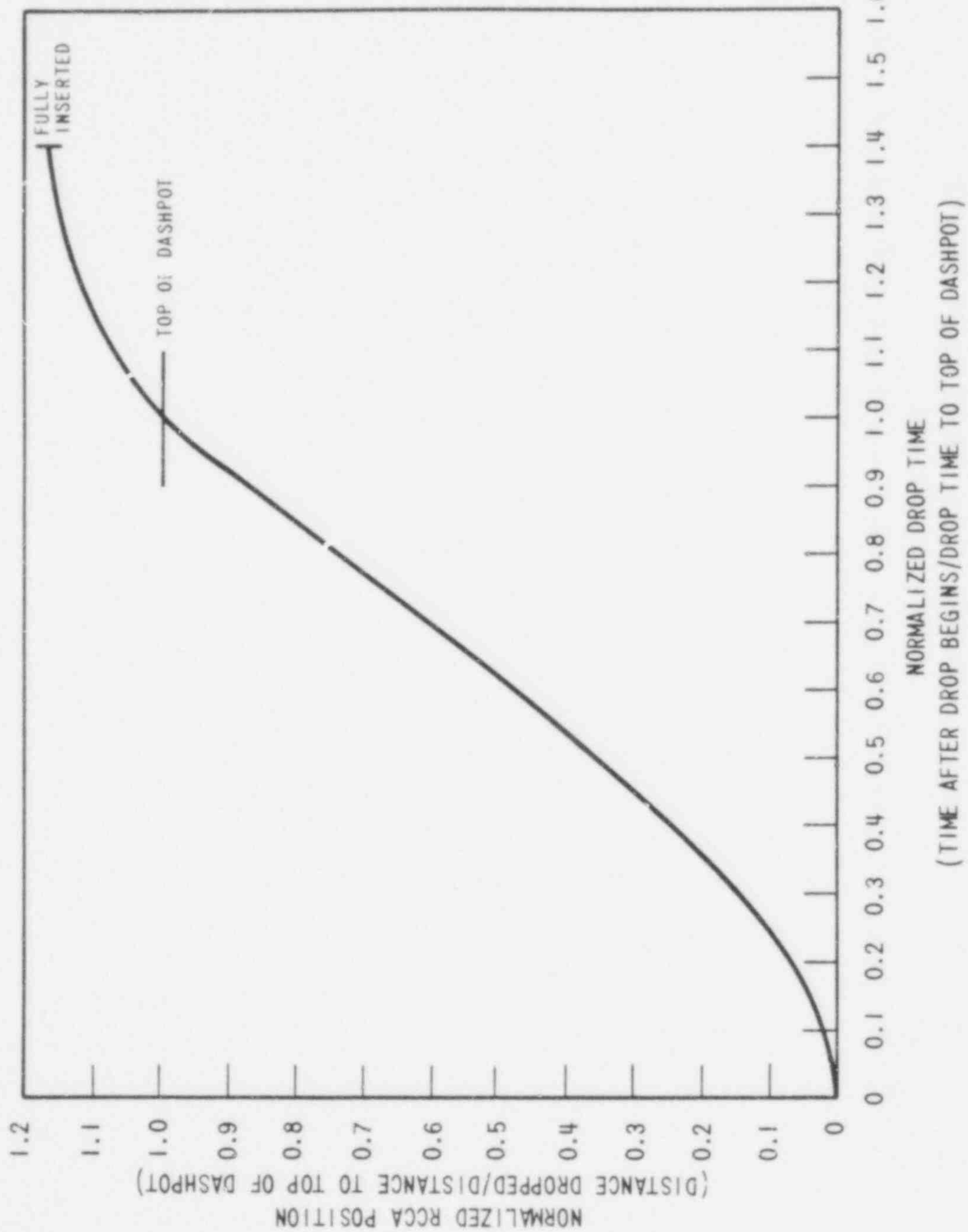

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Figure 15.0.5-1.

RCCA Position versus Time to Dashpot

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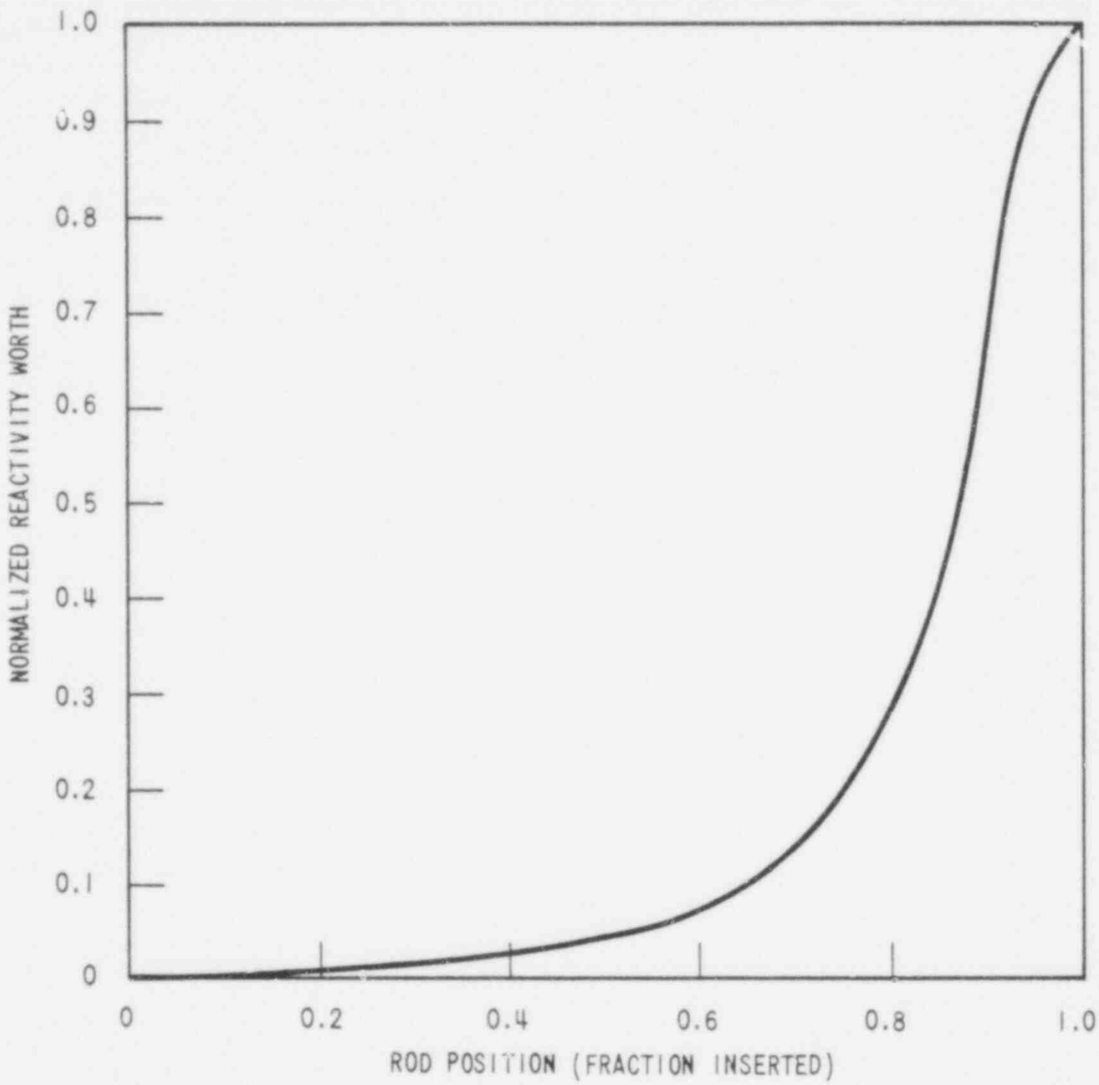

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WCAP – 9500

Figure 15.0.5-2.

Normalized Rod Worth versus Percent Inserted

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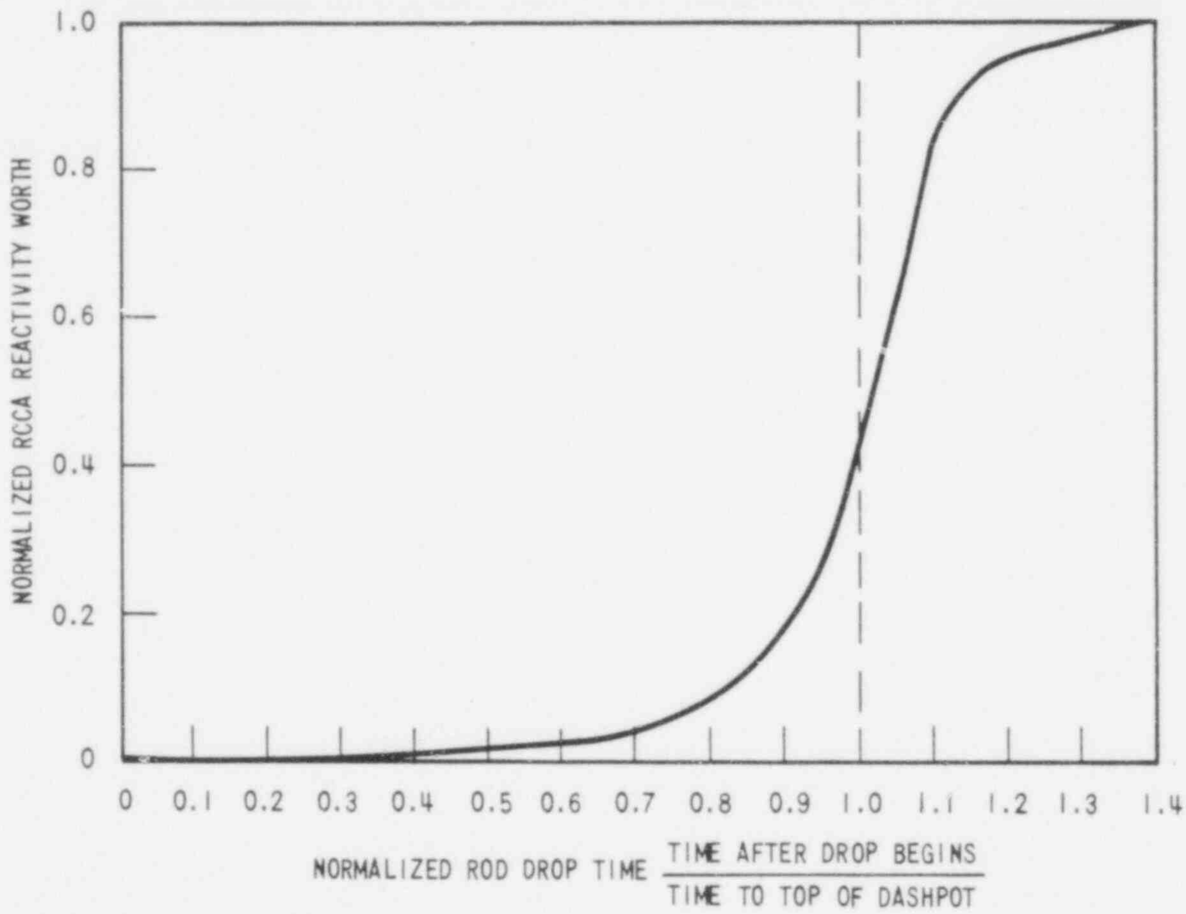

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WCAP - 9500

Figure 15.0.5-3.

Normalized RCCA Bank Reactivity Worth versus Normalized Drop Time

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15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

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

Discussions of the following Reactor Coolant System cooldown events are presented in this section:

1. Feedwater System malfunction causing a reduction in feedwater temperature.

2. Feedwater System malfunction causing an increase in feedwater flow.

3. Excessive increase in secondary steam flow.

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

5. Steam System piping failure.

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

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS CAUSING A REDUCTION IN FEEDWATER TEMPERATURE

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the

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Reactor Coolant System (RCS). The overpower-overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the limit value.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there could be a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

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

The net effect on the RCS due to a reduction in feedwater temperature would be similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

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

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Section 15.0.8 and listed in Table 15.0.8-1.

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15.1.1.2 Analysis of Effects and Consequences

This transient is analyzed by computing conditions at the feedwater pump inlet following opening of the heater bypass valve. These feedwater conditions are then used to recalculate a heat balance through the high pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

1. Plant initial power level corresponding to guaranteed NSSS thermal output.

2. Low pressure heater bypass valve opens, resulting in condensate flow splitting between the bypass line and the low pressure heaters; the flow through each path is proportional to the pressure drops.

3. Heater drain pumps trip; this increases the effect of the cold bypass flow.

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

Results

Opening of a low pressure heater byass valve and trip of the heater drain pumps causes a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than $1^{o}F$, resulting in a negligible increase in heat load on the primary system. The increased thermal load, due to opening of the low pressure heater bypass valve, thus would result in a transient very similar (but of greatly reduced magnitude) to that presented in Section 15.1.3 for an Excessive Load Increase Incident, which evaluates the consequences of a 10 percent step load increase. Therefore, the transient results of this analysis are not presented.

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15.1.1.3 Radiological Consequences

There will be no radiological consequences associated with a decrease in feedwater temperature event, and activity is contained within the fuel rods and reactor coolant system within design limits.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Section 15.1.2) and the increase in secondary steam flow event (Section 15.1.3). Based on results presented in Sections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 FEEDWATER SYSTEM MALFUNCTION CAUSING AN INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Causes and Accident Description

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

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

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater valves.

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

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.1.2.2 Analysis of Effects and Consequences

Method of Analysis

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

The system is analyzed to demonstrate plant behavior in the event that excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully, occurs. Two cases are analyzed as follows:

1. Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient.

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

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3. The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

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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 WCAP-8567.

2. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 149 percent of nominal feedwater flow to one steam generator.

3. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 206 percent of the nominal full load value for one steam generator.

4. For the zero load condition, feedwater temperature is at a conservatively low value of 70°F.

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

6. The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

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

Normal reactor control systems and Engineered Safety Systems are not required to function. The Reactor Protection System may function to trip the reactor due to an overpower condition. No single active failure will prevent operation of the Reactor Protection System. A discussion of ATWT considerations is presented in Reference [2].

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Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.4.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition," and therefore, the results of the analysis are not presented here. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent of nominal full power.

The full power case (maximum reactivity feedback coefficients, manual rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the automatic rod control mode results in a slightly less severe transient. The rod control system is, therefore, not required to function for an excessive feedwater flow event.

When the steam generator water level in the faulted loop reaches the high-high level setpoint, all feedwater isolation valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of the feedwater. In addition, a turbine trip is initiated.

Following turbine trip, the reactor will be tripped on a low-low steam generator water level signal in the intact steam generators. If the reactor were in the automatic control mode, the control rods would be inserted at the maximum rate following turbine trip, and the ensuing transient would then be similar to a loss of load (turbine trip event) as analyzed in Section 15.2.3.

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transient results, see Figures 15.1.2-1 and 15.1.2-2, show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. Following the turbine trip and feedwater isolation on the steam generator high-high level signal the reactor reaches a new stabilized condition at a reduced power level consistent with the reactivity parameters assumed to maximize the initial increase in code power. The reactor is tripped on low-low steam generator water level if no action is taken by the operator to terminate the reduced power operation.The DNB ratio does not drop below the limit value. Following the reactor trip, the plant approaches a stabilized condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant, hence the peak value does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). The peak fuel temperature will thus remain well below the fuel melting temperature.

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

The calculated sequence of events for this accident is shown in Table 15.1.2-1.

15.1.2.3 Radiological Consequences

There are minimal radiological consequences from this event. The high level signal causes a reactor and turbine trip and heat is removed from the secondary system through the steam generator power relief or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences will be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.2.4 Conclusions

The results of the analysis show that the DNB ratios encountered for an excessive feedwater addition at power are above the limit value, hence, no fuel or clad damage is predicted. Additionally, it has been shown that the reactivity insertion rate which occurs at no load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition analysis. The radiological consequences of this event will be less than the steamline break accident analyzed in Subsection 15.1.5.3.

15.1.3 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

15.1.3.1 Identification of Causes and Accident Description

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

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the 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; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

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

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1. Overpower ΔT
2. Overtemperature ΔT
3. Power range high neutron flux

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

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using LOFTRAN Code (Reference 1). The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

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

1. Reactor control in manual with minimum moderator reactivity feedback.

2. Reactor control in manual with maximum moderator reactivity feedback.

3. Reactor control in automatic with minimum moderator reactivity feedback.

4. Reactor control in automatic with maximum moderator reactivity feedback.

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and therefore, the least inherent transient capability. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. 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 taken for pressurizer heaters.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3. 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 WCAP-8567.

Normal reactor control systems and Engineered Safety Systems are not required to function. The Reactor Protection System is assumed to be operable; however, reactor trip is not encountered for many cases due to the error allowances assumed in the setpoints. No single active failure will prevent the Reactor Protection System from performing its intended function.

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

Results

Figures 15.1.3-1 through 15.1.3-4 illustrate the transient with the reactor in the manual control mode. As expected, for the minimum moderator feedback case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum moderator feedback, manually controlled case there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the limit value. For these cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

Figures 15.1.3-5 through 15.1.3-8 illustrate the transient assuming the reactor is in the automatic control mode and no reactor trip signals occur. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the limit value.

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

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

The calculated sequence of events for the excessive load increase incident is shown on Table 15.1.2-1.

15.1.3.3 Radiological Consequences

There will be no radiological consequences associated with this event and activity is contained within the fuel rods and reactor coolant system within design limits.

15.1.3.4 Conclusions

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

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

15.1.4.1 Identification of Causes and Accident Description

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

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 coefficient, the cooldown results in an insertion of positive reactivity.

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

> Assuming a stuck rod cluster control assembly, with offsite power available, and assuming a single failure in the Engineered Safety Features System there will be no return to criticality 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.

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

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

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

    a. Two-out-of-three low steamline pressure signals an any one loop

    b. Two-out-of-four pressurizer pressure signals.

    c. Two-out-of-three high containment pressure signals.

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

3. Redundant isolation of the main feedwater lines.

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4. Trip of the fast-acting steamline stop valves (designed to close in less than 5 seconds) on:

   a. Two-out-of-three low steamline pressure signals in any one loop.

   b. Two-out-of-three high-high containment pressure signals.

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

Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.1.4.2 Analysis of Effects and Consequences

Method of Analysis

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 cooldown, and the effect of safety injection.

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2. Analyses to determine that the reactor does not return to criticality.

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

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

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downstream of the boron injection tank isolation valves prior to the delivery of high concentration boric acid (20,000 parts per million (ppm)) to the reactor coolant loops. This effect has been accounted for in the analysis.

4. The case studied is a steam flow of 270 pounds per second at 1200 pounds per square inch absolute (psia) 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 conservative initial condition.

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

5. In computing the steam flow, the Moody Curve (Reference 3) for f(L/D) = 0 is used.

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

Results

The calculated time sequence of events for this accident is listed in Table 15.1.2-1.

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 15.1.4-2 and 15.1.4-3 show the transient results for a steam flow of 270 lb/sec at 1200 psia.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve. Safety injection is initiated automaticlly by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 20,000 ppm enters the RCS providing sufficient negative reactivity to maintain the reactor below criticality. 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 5 minutes, the neglected stored energy is likely to have a signi- ficant effect in slowing the cooldown.

15.1.4.3 Radiological Consequences

The inadvertent opening of a single steam dump relief or safety valve can result in steam release from the secondary system. If steam generator leakage exists coincident with the failed fuel conditions, some activity will be released. (The activity release and dose is provided on a plant specific basis).

15.1.4.4 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. Since the reactor does not return to criticality, a DNBR less than the limit value does not exist.

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure fails. 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 rod cluster control assembly (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 steamline rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be

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stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System.

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

Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10CFR100.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

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

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

The major rupture of a steamline is the most limiti... ...n transient and is analyzed at zero power with no decay heat. Decay ...at 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 here.

The following functions provide the protection for a steamline rupture:

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

   a. Two-out-of-three ...eamline pressure signals in any one loop

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b. Two-out-of-four pressurizer pressure signals.

c. Two-out-of-three high containment pressure signals.

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

3. Redundant isolation of the main feedwater lines.

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4. Trip of the fast acting steamline stop valves (designed to close in less than 5 seconds) on:

   a. Two-out-of-three low steamline pressure signals in any one loop.

   b. Two-out-of-three high-high containment pressure signals.

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

Fast-acting isolation valves are provided in each steamline; these valves will fully close within 10 seconds of a large break in the steamline. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. A description of steamline isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1.5-1 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since the requirements will vary depending upon postulated break location and details of balance of plant design and pipe rupture criteria as discussed elsewhere in this application. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

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 thermal and hydraulic behavior of the core following a steamline break. A detailed thermal and hydraulic digital-computer code, THINC, 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 steam break accident:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA 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. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The effect of power generation in the core on overall reactivity is shown in Figure 15.1.5-1.

   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 limiting statepoints for the cases analyzed.

   This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined 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 high concentration boric acid (20,000 ppm) solution corresponding to the most restrictive single failure in the Safety Injection System. The Emergency Core Cooling System consists of three systems: 1) the passive accumulators, 2) the Residual Heat Removal System, and 3) the Safety Injection System. Only the Safety Injection System is modeled for the steamline break accident analysis.

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The actual modeling of the Safety Injection System in LOFTRAN is described in Reference [1]. 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 boron injection tank isolation valves prior to the delivery of high concentration boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the Safety Injection System is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 10 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume contaning the low concentration borated water is swept before the 20,000 ppm reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 10 second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

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

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

b. Case (a) with loss of offsite power simultaneous with the steamline break and initiation of the safety injection signal. Loss of offsite power results in reactor coolant pump coastdown.

6. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly 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 two cases correspond to values determined from the respective transient analysis.

   Both 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. A spectrum of steamline breaks at various power levels has been analyzed in Reference [4].

7. In computing the steam flow during a steamline break, the Moody Curve (Reference 3) for f(L/D) = 0 is used.

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8. The Upper Head Injection (UHI) is simulated. The actuation pressure for the UHI is near the saturation pressure for the inactive coolant in the upper head. The insurge of cold UHI water keeps this inactive coolant from flashing and thus retarding pressure decrease. The effect of UHI is a faster pressure decrease which in turn permits more safety injection flow into the core. These effects are very small and results are not significantly affected.

These assumptions are discussed more fully in Reference [4].

Results

The calculated sequence of events for both cases analyzed is shown on Table 15.1.2-1.

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.

Core Power and Reactor Coolant System Transient

Figures 15.1.5-2 through 15.1-5-4 show the RCS transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial no-load condition (case a).

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steamlines by low steamline pressure signals, high-high containment pressure signals, or high negative steamline pressure rate signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steamline stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.1.5-3 the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) shortly before boron solution at 20,000 ppm enters the RCS. The continued addition of boron results in a peak core power significantly lower than the nominal full power value.

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 Safety Injection System. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the Safety Injection System due to changes in the RCS pressure. The Safety Injection System flow calculation includes the line losses in the system as well as the pump head curve.

Figures 15.1.5-5 through 15.1.5-7 show the salient parameters for case b, which corresponds to the case discussed above with additional loss of offsite power at the time the safety injection signal is generated. The Safety Injection System delay time includes 10 seconds to start the diesel in addition to 10 seconds to start the safety injection pump and open the valves. Criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full power value.

It should be noted that following a 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.

Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. It was found that both cases has a minimum DNBR greater than the limit value.

15.1.5.3 Radiological Consequences of a Postulated Steamline Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. Parameters used in both the realistic and conservative anslyses are listed in Table 15.1.5-2. These parameters are based on the Source Terms specified in ANSI N-237 Standard (March 1976), NUREG 0017, April 1976.

The primary and secondary coolant activities correspond to the specific activity limits given in the Technical Specifications. The primary coolant activities are 60.0 μCi/gm of dose equivalent I-131 due to a pre-existing iodine spike prior to the accident, and 100/E μCi/gm (conservatively assumed to be comprised entirely of noble gas activity).

The following conservative assumptions and parameters will be used to calculate the activity releases and offsite doses for the postulated steamline break:

Prior to the accident, an equilibrium activity of fission products exists in the primary and secondary systems caused by a primary to secondary leakage in the steam generators.

2. Offsite power is lost and the main steam condensers are not available for steam dump.

3. Eight hours after the accident the residual heat removal system starts operation to cool down the plant.

4. The total primary to secondary leakage is 1.0 gpm, with 0.347 (500 gal/day) in the defective steam generator and the rest divided equally between the three nondefective steam generators.

5. Defective fuel is 1 percent.

6. One percent of the total core fuel cladding is damaged.

7. After 8 hours, following the accident, no steam and activity are released to the environment.

8. No noble gas is dissolved in the steam generator water.

9. The iodine partition factor in the steam generators.

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.1$$

10. During the postulated accident iodine carryover from the primary side in the three good steam generators is diluted in the incoming feedwater.

15.1.5.4 Conclusions

The analysis has shown that the criteria stated in Subsection 15.1.5.1 are satisfied with the exclusion of the radiological criteria. The radiological assessments will be given on a plant specific basis. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the DNB design bases is met as stated in Section 4.4.

Parameters recommended for use in determining the amoumt of radioactivity released are given in Table 15.1.5-2.

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15.1.6 REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972.

2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

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

4. Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, Revision 1, (Proprietary), January, 1978, and WCAP-9227, Revision 1, (Non-Proprietary), January 1978.

TABLE 15.1.2-1 (Page 1)

Time Sequence of Event for Incidents Which Cause an Increase in Heat Removal By The Secondary System

Accident	Event	Time (sec.)
Excessive Feedwater flow at full power	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level signal generated	89.2
	Turbine trip occurs due to high-high steam generator level	92.2
	Minimum DNBR occurs	92.2
	Feedwater isolation valves close	96.2
	Low-low steam generator reactor trip setpoint reached in intact steam generators	168
Excessive Increase in Secondary Steam Flow		
1. Manual Reactor Control (Minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	100

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TABLE 15.1.2-1 (Page 2)

Time Sequence of Event for Incidents Which Cause an Increase in Heat Removal By The Secondary System

Accident	Event	Time (sec.)
2. Manual Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
3. Automatic Reactor Control (Minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
4. Automatic Reactor Control (Maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0

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TABLE 15.1.2-1 (Page 3)

Time Sequence of Event for Incidents Which Cause an Increase in Heat Removal By The Secondary System

Accident	Event	Time (sec.)
	Pressurizer empties	172
	20,000 ppm boron reaches core	176
Steam system piping failure		
1. Case a	Steamline ruptures	0
	Pressurizer empties	11
	Criticality attained	13
	20,000 ppm boron reaches core	14
2. Case b	Steamline ruptures	0
	Pressurizer empties	12
	Criticality attained	17
	20,000 ppm boron reaches core	24

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TABLE 15.1.5-1 (Page 1)

Equipment Required Following a Rupture of a Main Steam Line

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded).	Auxiliary Feedwater System including pumps, water supply, and system valves and piping (this system must be placed in service to supply water to operable steam generators no later than 10 minutes after the incident).	Steam generator power operated relief valves (can be manually operated locally). Control for defeating automatic safety injection actuation during a cooldown and depressurization.
Safety Injection System including the pumps, the refueling water storage tank, the boron injection tank, and the systems valves and piping.	Reactor Containment ventilation cooling units. Capability for obtaining a Reactor Coolant System sample.	Residual Heat Removal System including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the Reactor Coolant System in a cold shutdown condition.
Diesel generators and emergency power distribution equipment.		

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Equipment Required Following a Rupture of a Main Steam Line

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Essential Service Water System.		
Containment safeguards cooling equipment.		
Auxiliary Feedwater System including pumps, water supplies, piping and valves.		
Main feedwater control valves (trip closed feature).		
Bypass feedwater control valves (trip closed feature).		
Primary and secondary safety valves.		

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TABLE 15.1.5-1 (Page 3)

Equipment Required Following a Rupture of a Main Steam Line

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Circuits and/or equipment required to trip the main feedwater pumps.		
Main feedwater isolation valves (trip closed feature).		
Main steam line stop valves (trip closed feature).		
Main steam line stop valve bypass valves (trip closed (trip closed feature).		
Steam generator blowdown isolation valves (automatic closure feature).		

TABLE 15.1.5-1 (Page 4)

Equipment Required Following a Rupture of a Main Steam Line

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Batteries (Class 1E).		
Control Room ventilation.		
Control Room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting.		
Post Accident Monitoring System[a].		

[a]See Section 7.5 for a discussion of the Post Accident Monitoring System.

TABLE 15.1.5-2

Parameters to be Used in Analysis of Radiological Consequences of Steam Line Break Analysis

Parameter	Realistic Value	Conservative Value
Core Thermal Power	3565 MWt	3565 MWt
Offsite Power Availability	Available	Lost at Accident Initiation
Fraction of Core Power Produced in Rods Containing Defects	.0012	0.01*
Fraction of Fuel Rods whose Cladding fails as a result of the accident	0.0	0.01
Steam Generator Leak Rate prior to accident from all steam generators	.009 gpm	1.0 gpm
Fraction of activity in failed rods which is released to the coolant	N/A	N/A
Iodine Spike Release from fuel to coolant	See Table 15.0-8	See Table 15.0-8
Duration of release	4 hrs.	4 hrs.
Iodine inventory in secondary side prior to accident	4.5×10^{-5} μCi/gm DE I-131**	1.0 μCi/gm DE I-131
Steam Generator Leak Rate during accident from all steam generators	.009 gpm***	1.0 gpm

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TABLE 15.1.5-2 (Continued)

Parameters Used in Steam Line Break Analysis

Parameter		Realistic Value	Conservative Value
Integrated Feedwater Flow to non-defective steam generators (assumed to be at a constant rate)			
	0 - 2 hrs.	581, 505 lb.	581,505 lb.
	2 - 8 hrs.	1,066,473 lb.	1,066,473 lb.

*** Assumed to be independent of pressure differential across steam generator tubes.

** D. E. = dose equivalent.

* May be decreased to correspond to tech spec limit on maximum primary coolant activity.

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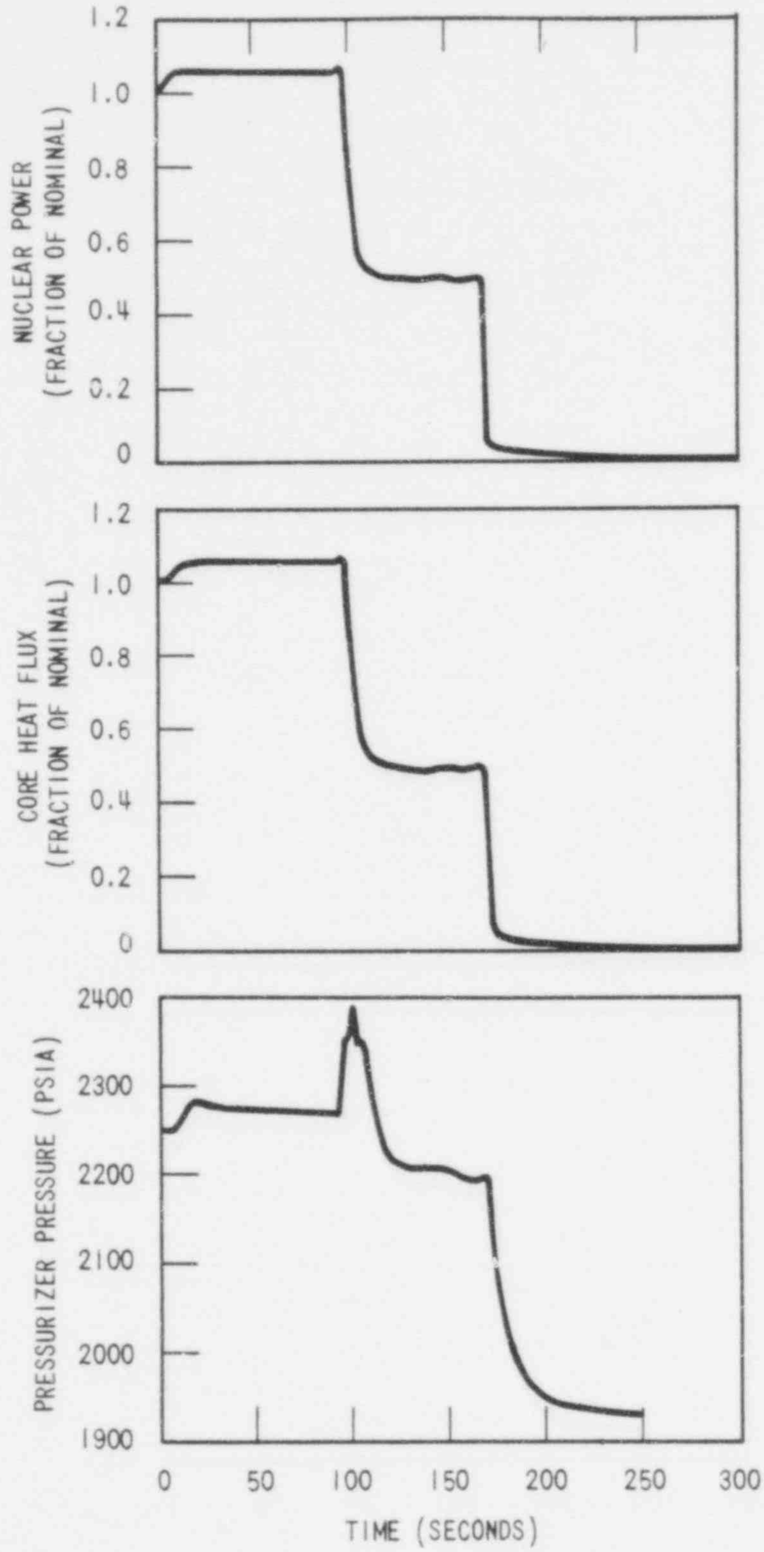

WCAP - 9500

Figure 15.1.2-1.

Feedwater Control Valve Malfunction

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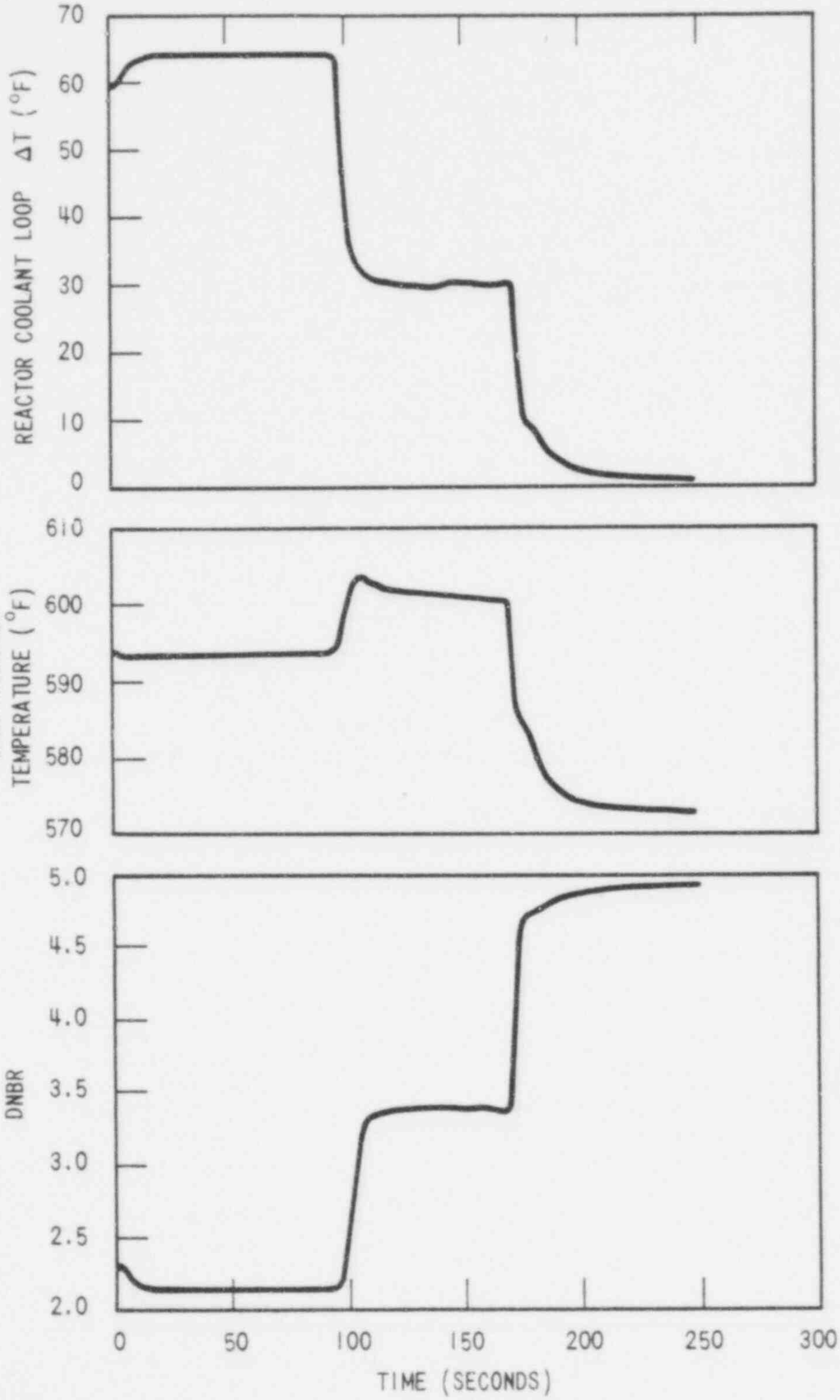

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Figure 15.1.2-2.

Feedwater Control Valve Malfunction

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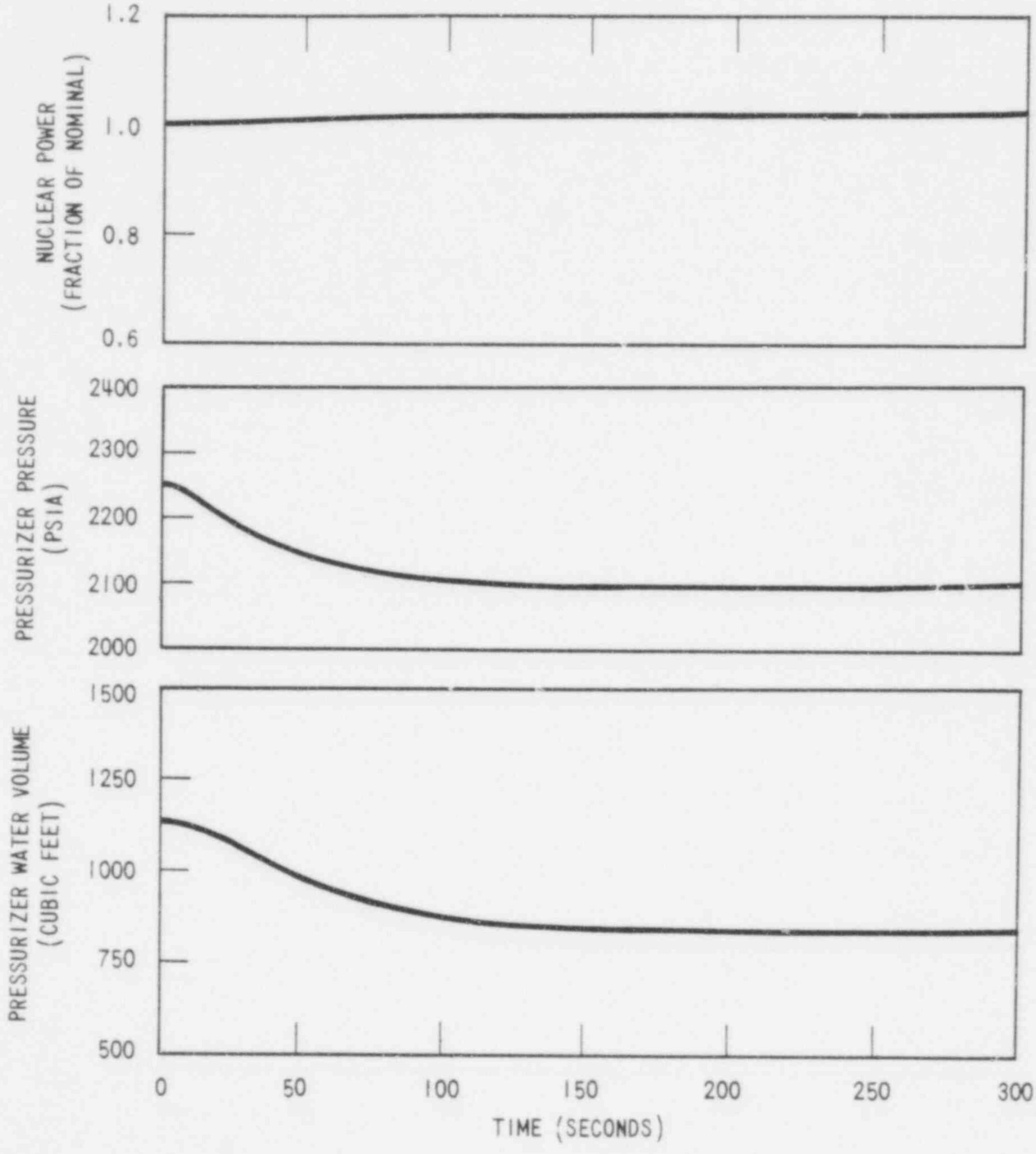

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Figure 15.1.3-1.

Ten Percent Step Load Increase, Minimum Moderator Feedback, Manual Reactor Control

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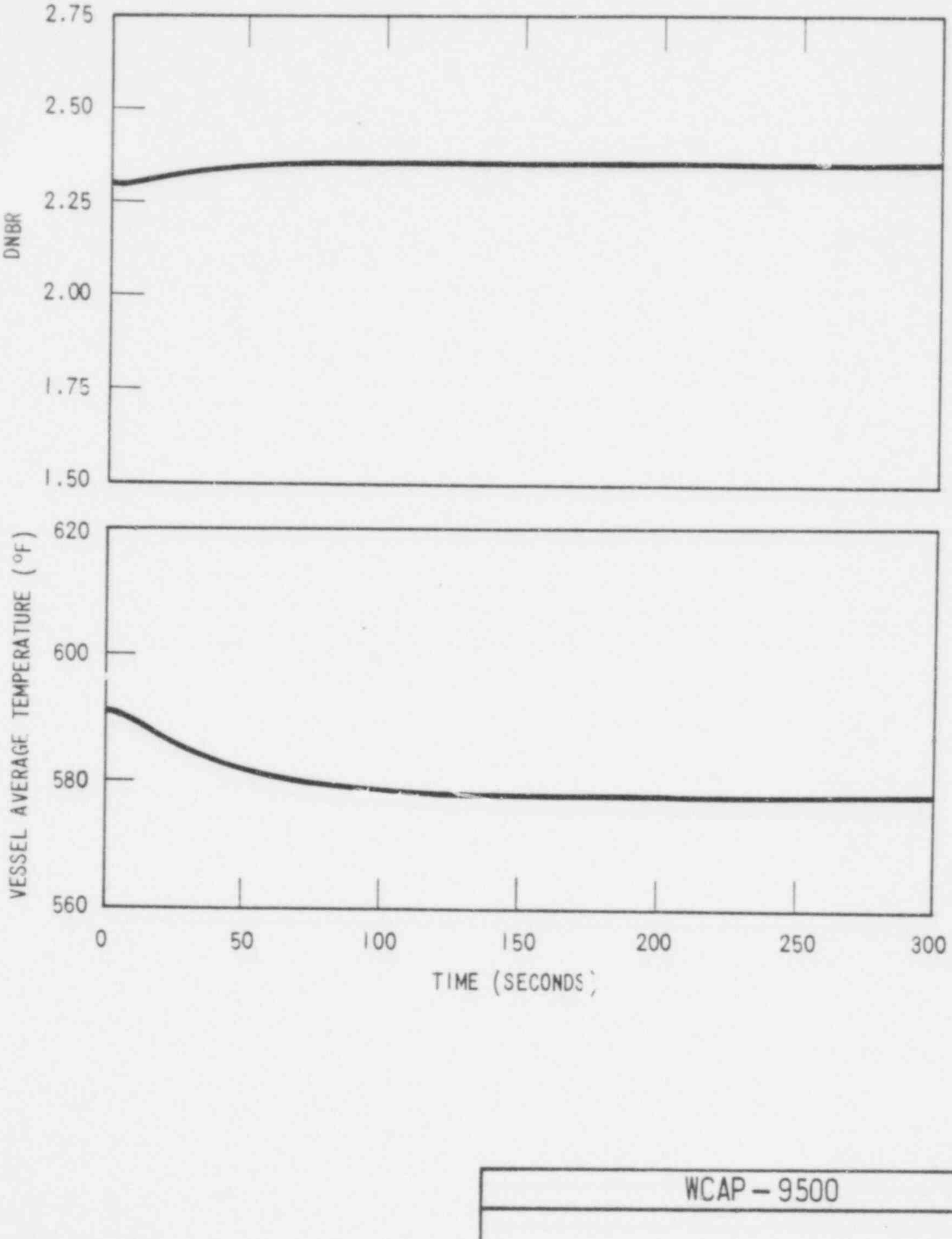

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Figure 15.1.3-2.

Ten Percent Step Load Increase, Minimum Reactivity Feedback, Manual Reactor Control

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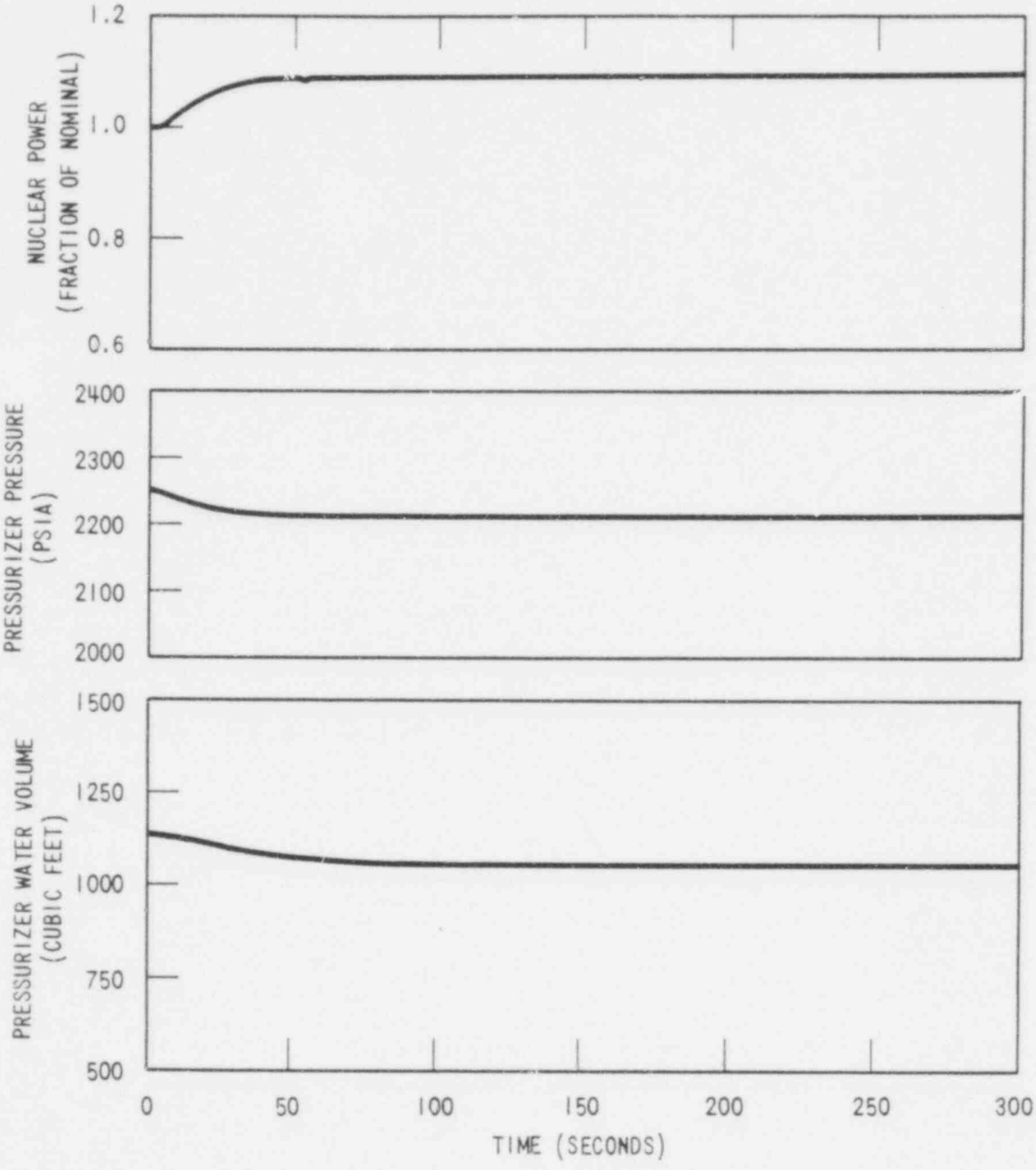

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Figure 15.1.3-3.

Ten Percent Step Load Increase, Maximum Reactivity Feedback, Manual Reactor Control

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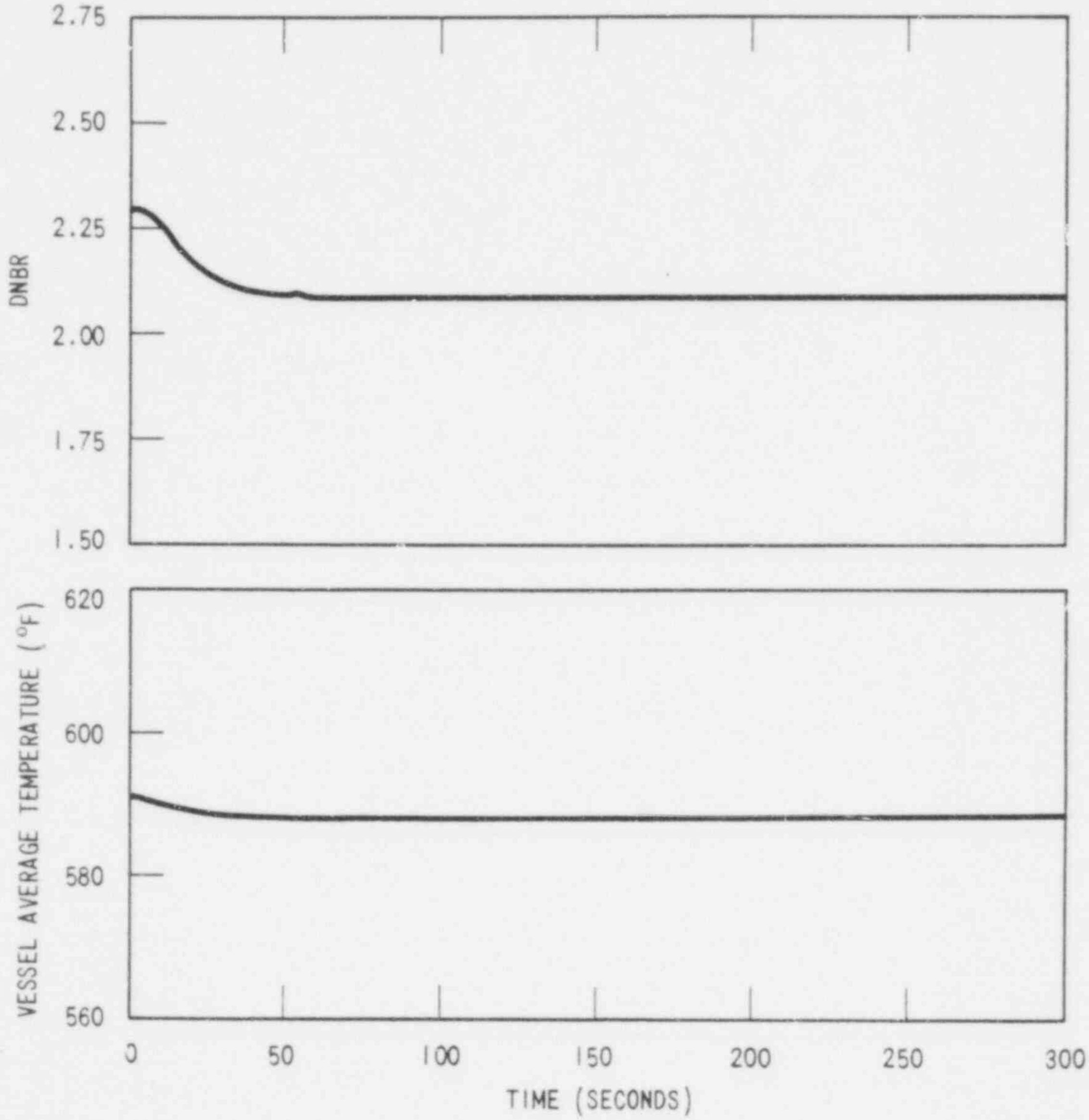

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Figure 15.1.3-4.

Ten Percent Step Load Increase, Maximum Reactivity Feedback, Manual Reactor Control

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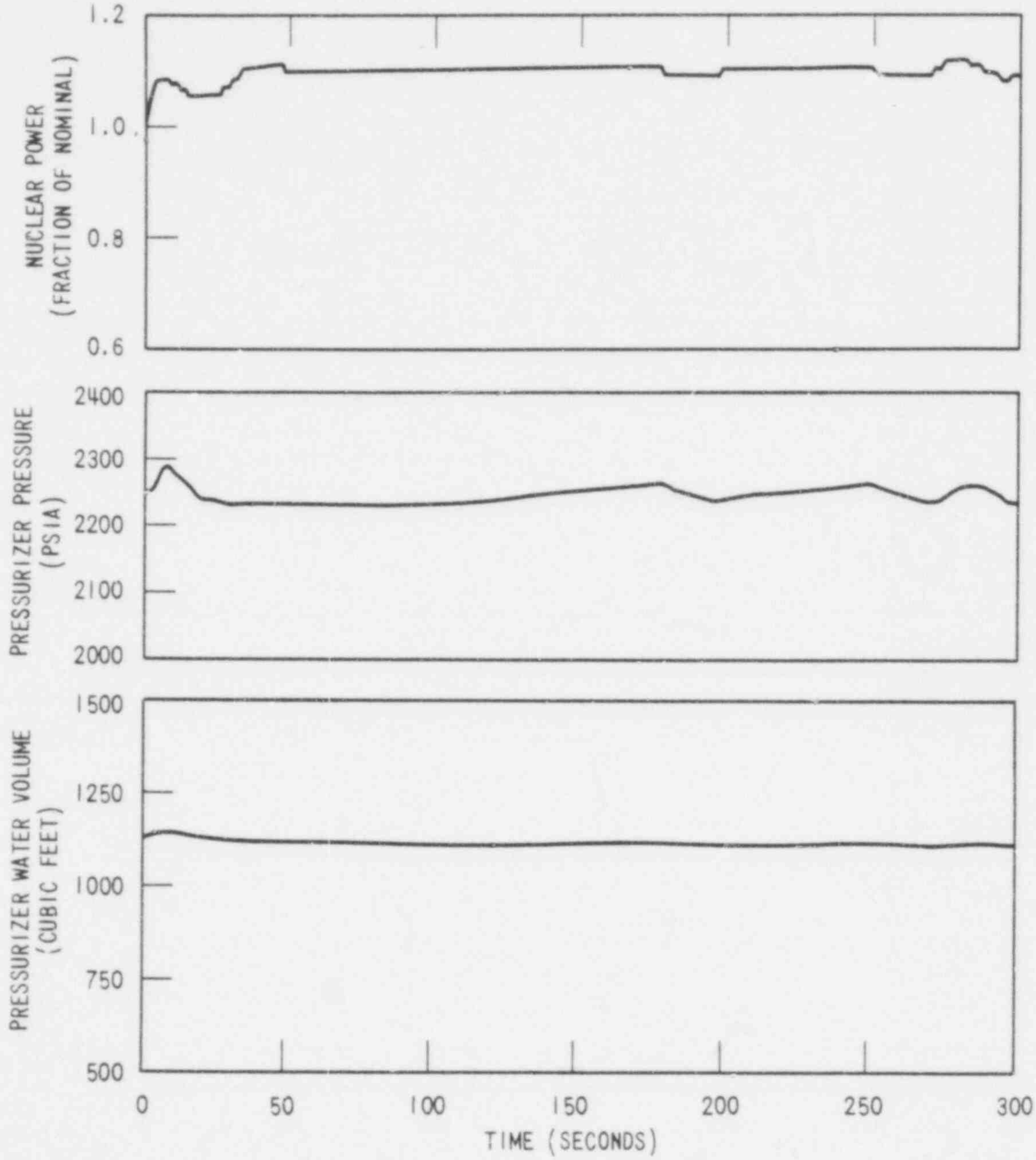

WCAP – 9500

Figure 15.1.3-5.

Ten Percent Step Load Increase, Minimum Reactivity Feedback, Automatic Reactor Control

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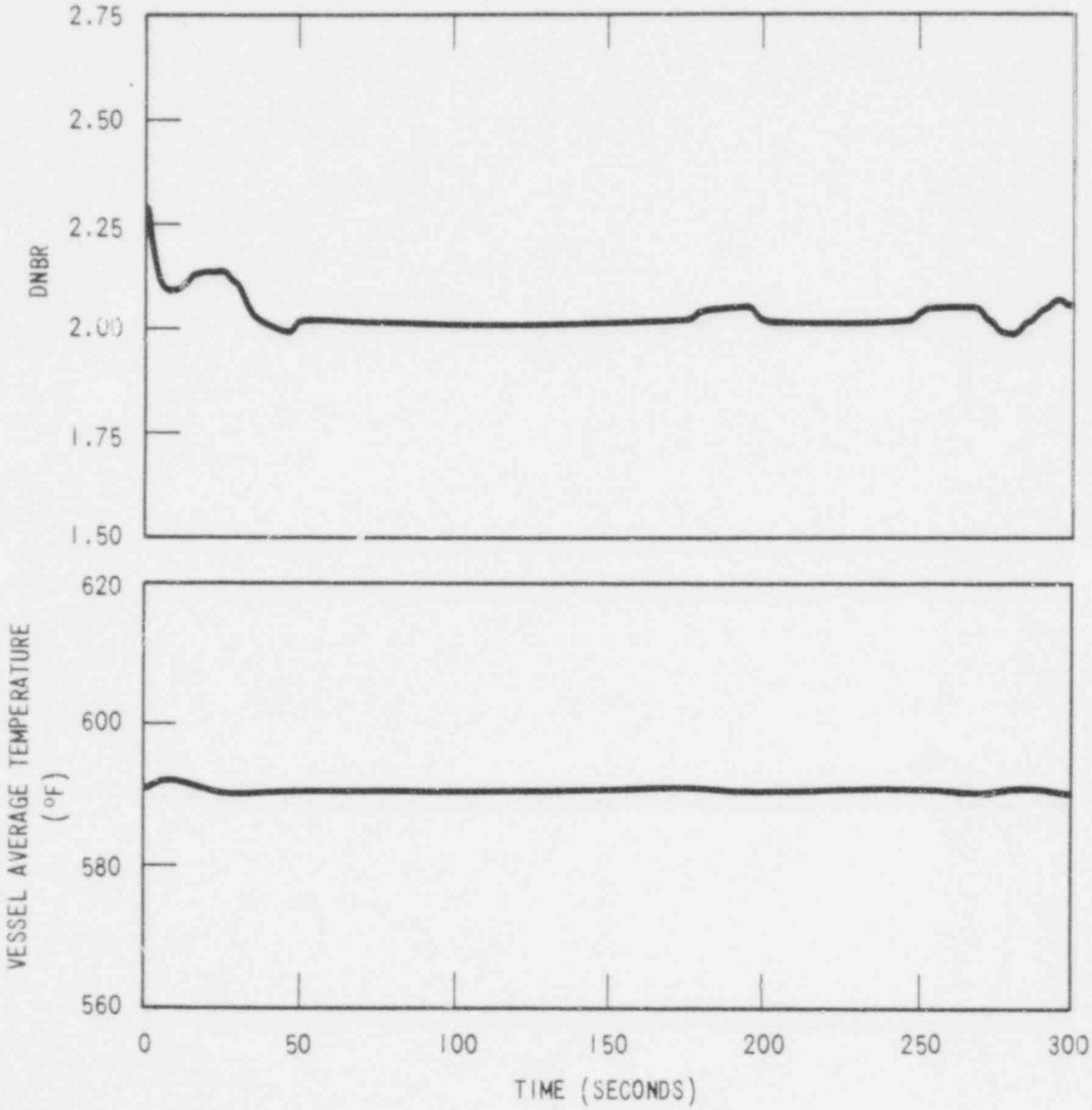

WCAP - 9500

Figure 15.1.3-6.

Ten Percent Step Load Increase, Minimum Reactivity Feedback, Automatic Reactor Control

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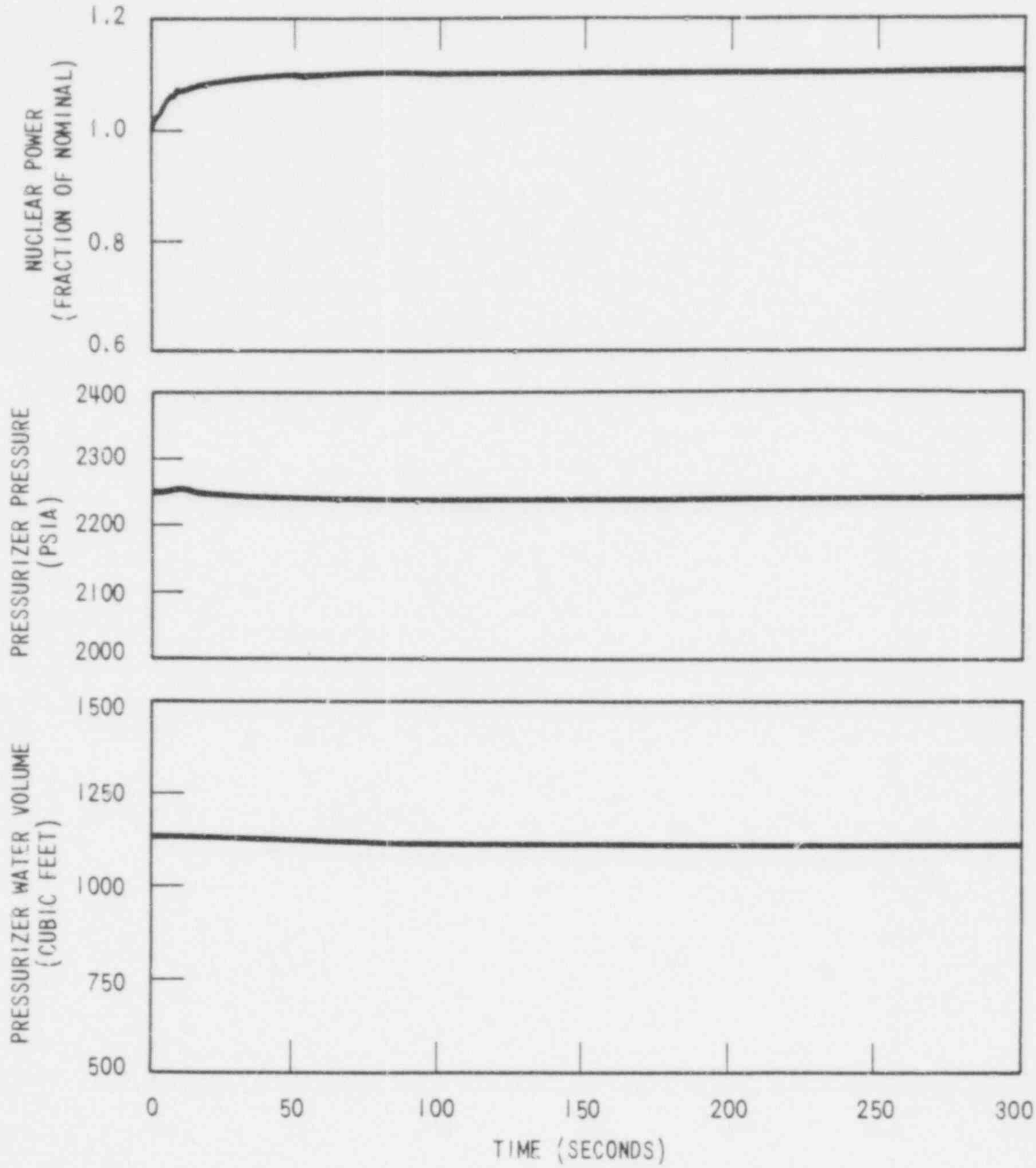

WCAP – 9500

Figure 15.1.3-7.

Ten Percent Step Load Increase, Maximum Reactivity Feedback, Automatic Reactor Control

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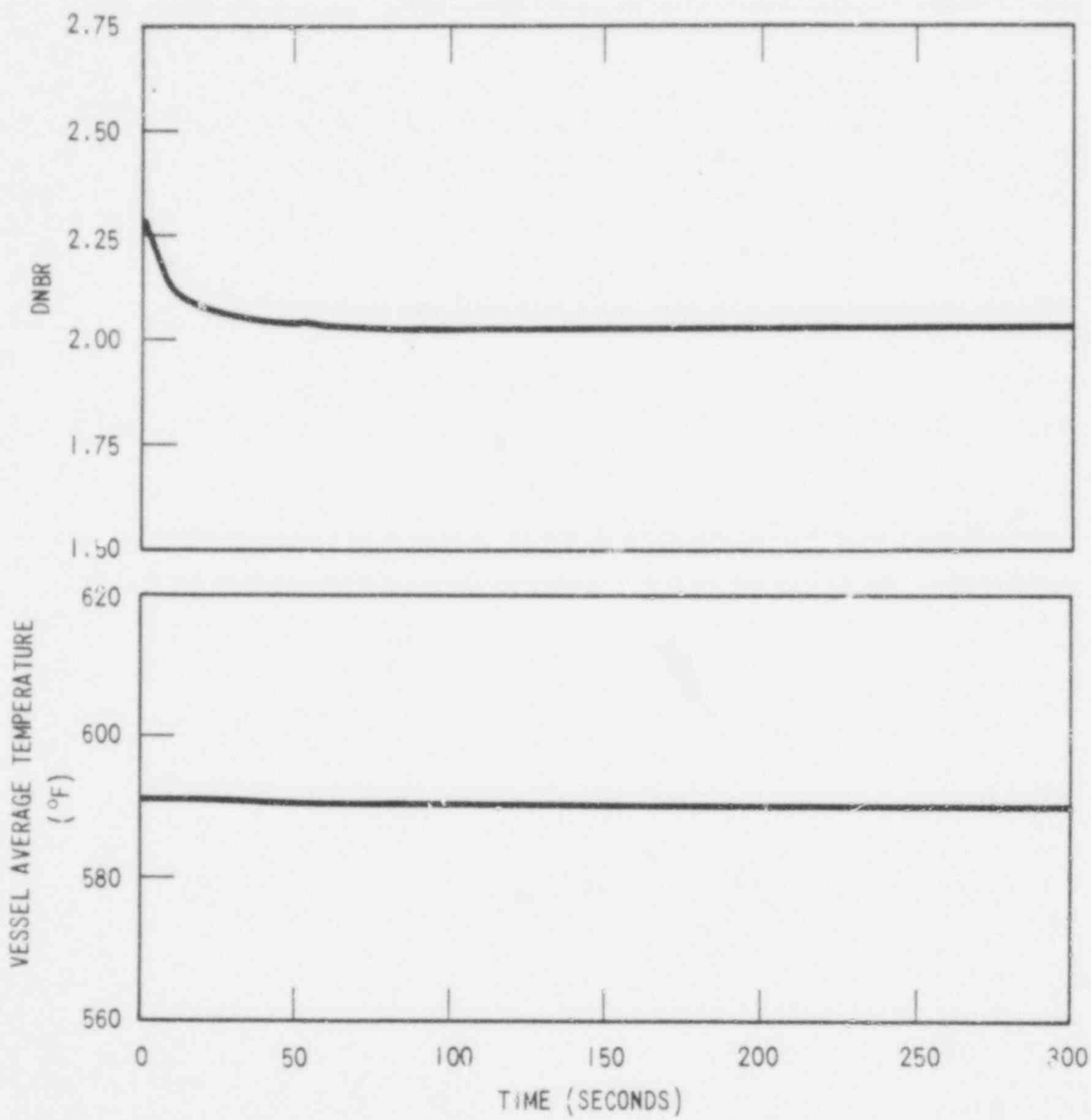

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WCAP - 9500

Figure 15.1.3-8.

Ten Percent Step Load Increase, Maximum Reactivity Feedback, Automatic Reactor Control

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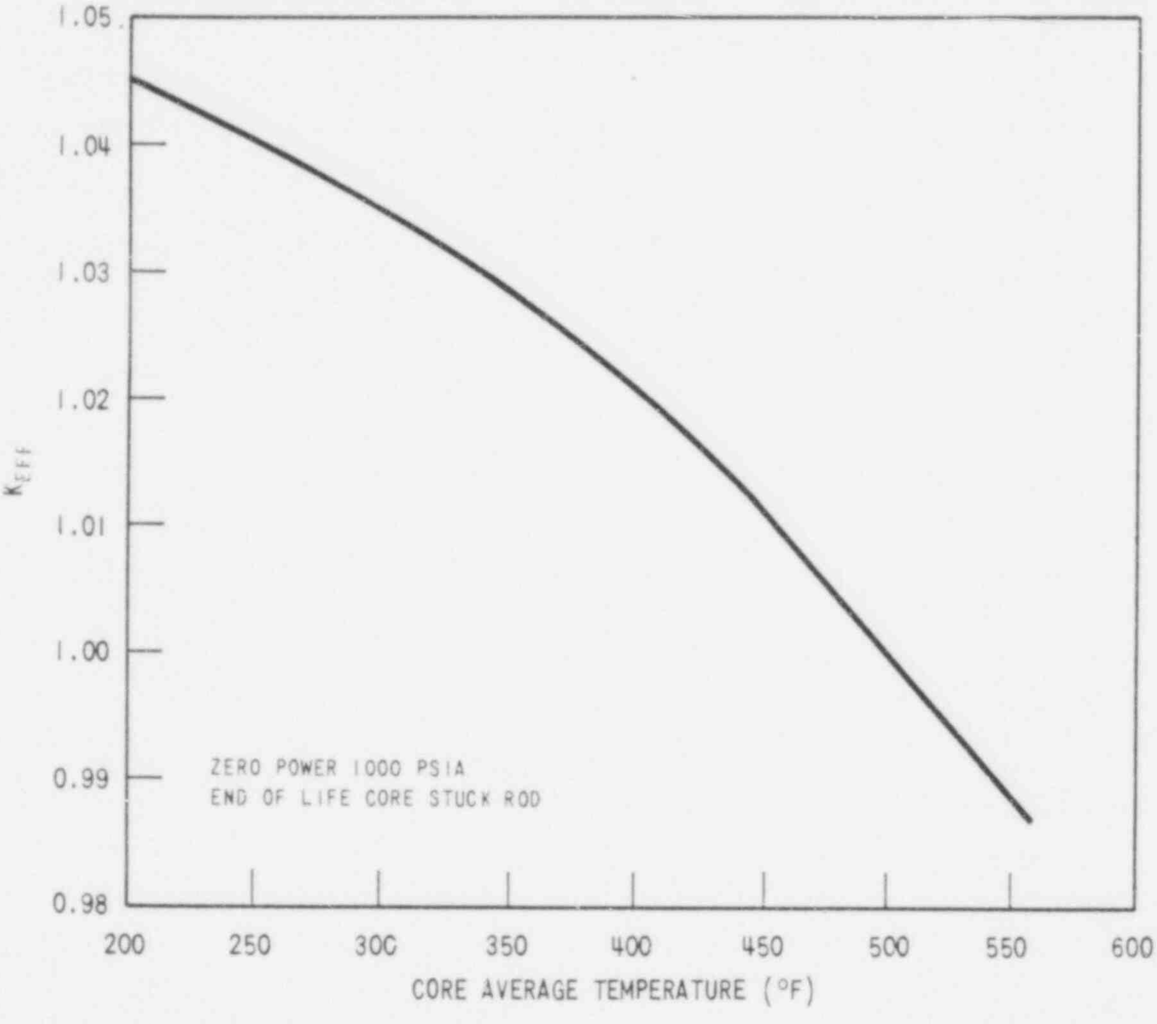

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Figure 15.1.4-1.

K_{EFF} vs. Temperature

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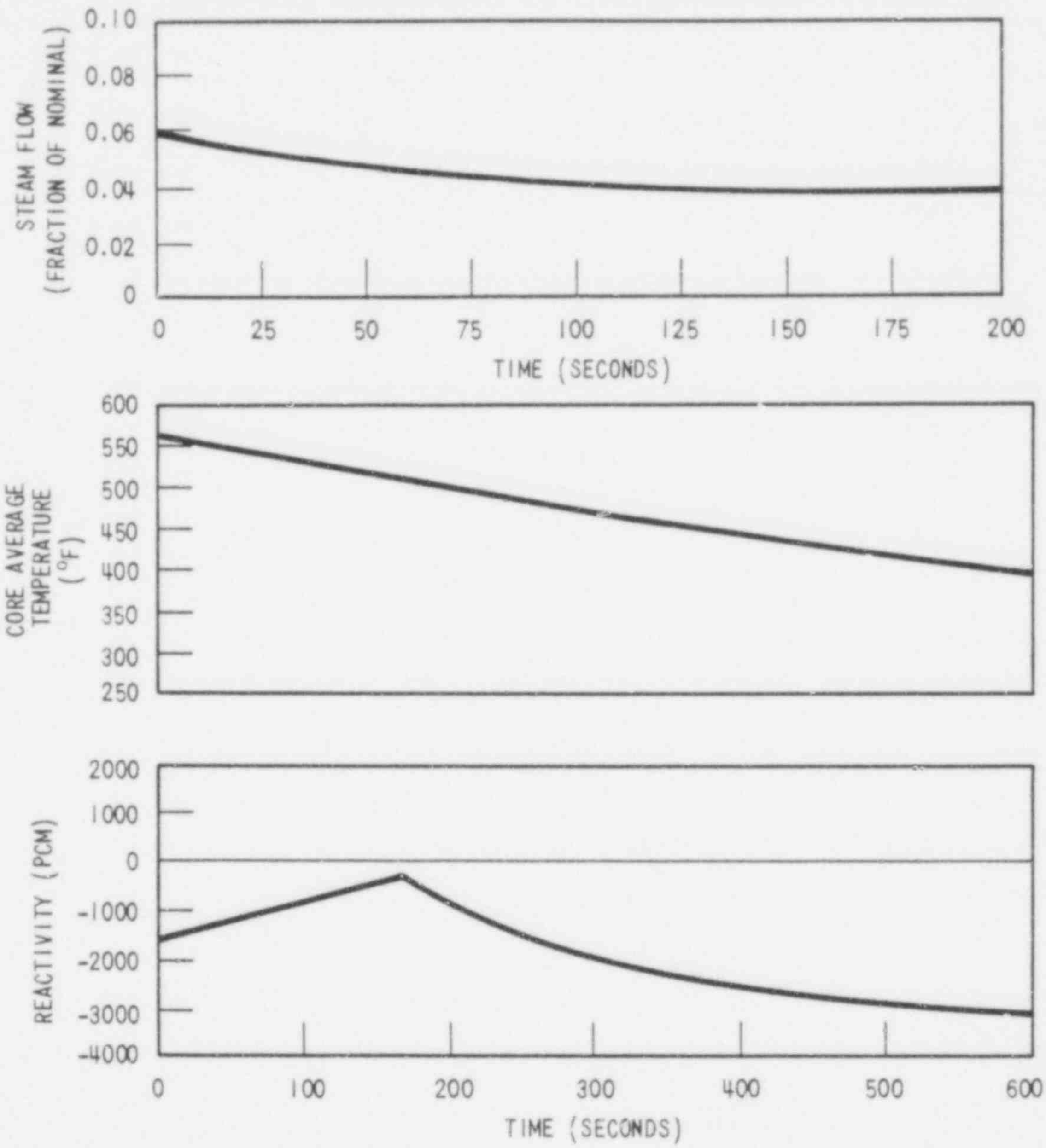

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WCAP – 9500

Figure 15.1.4-2.

Failure of a Steam Generator Safety or Dump Valve

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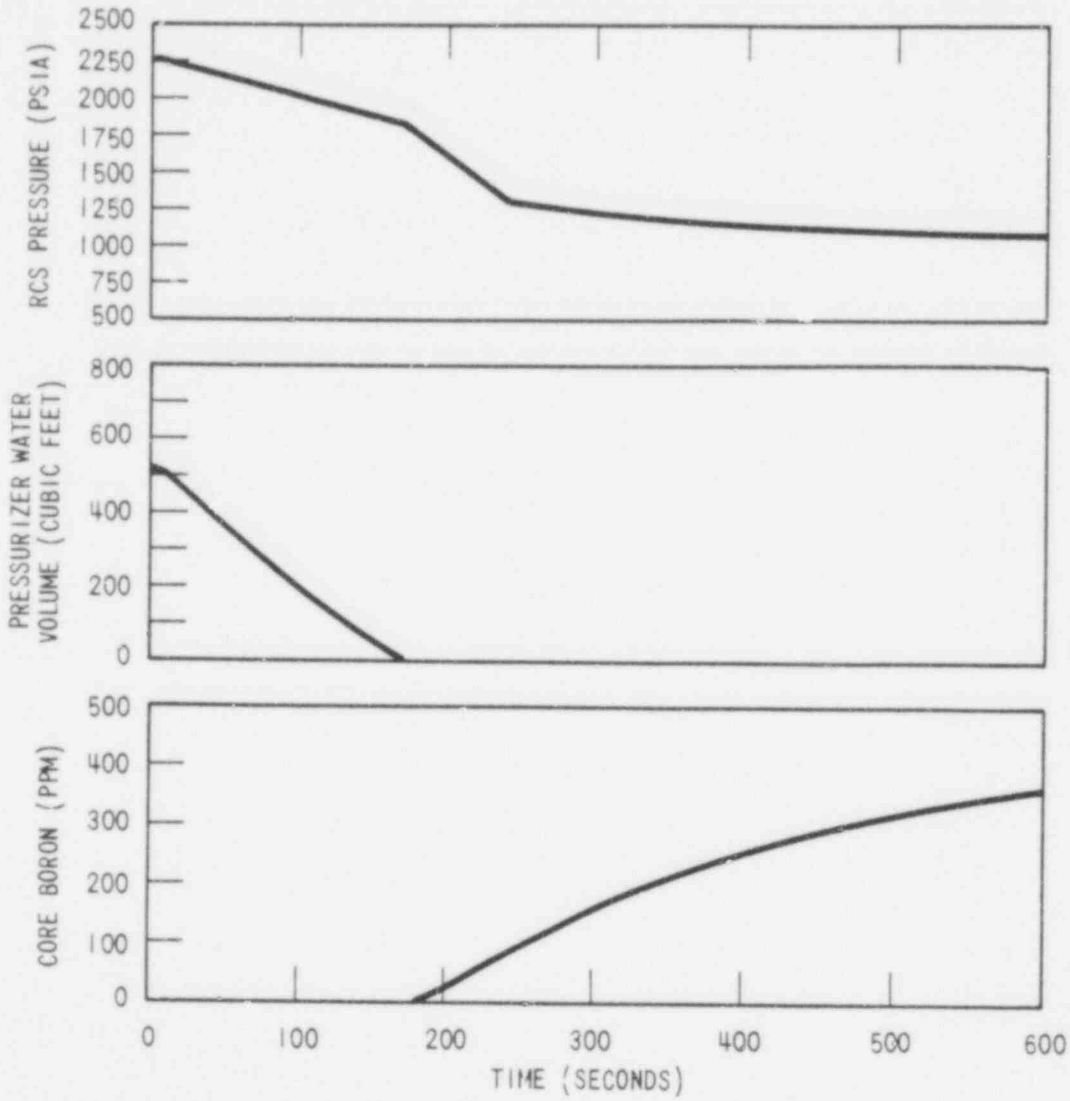

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Figure 15.1.4-3.

Failure of a Steam Generator Safety or Dump Valve

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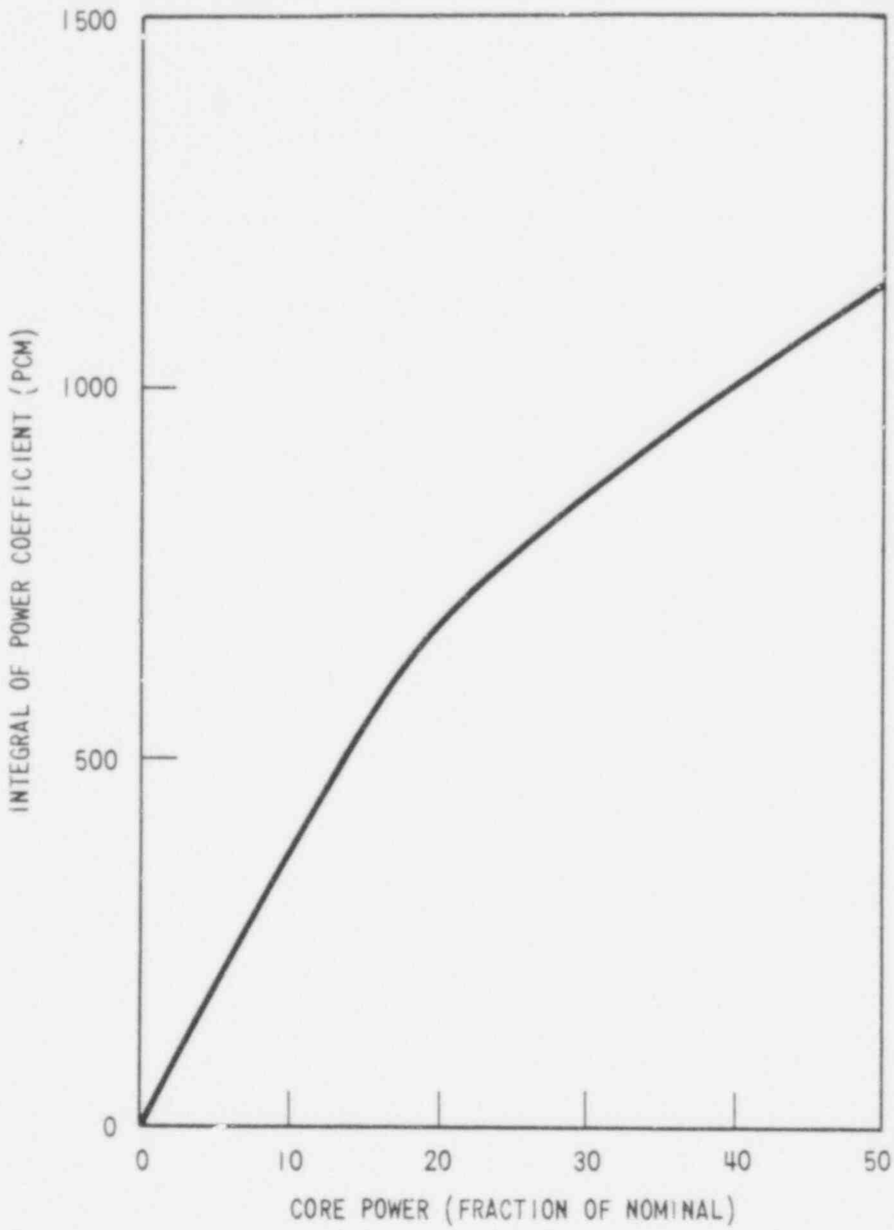

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Figure 15.1.5-1.

Doppler Power Feedback

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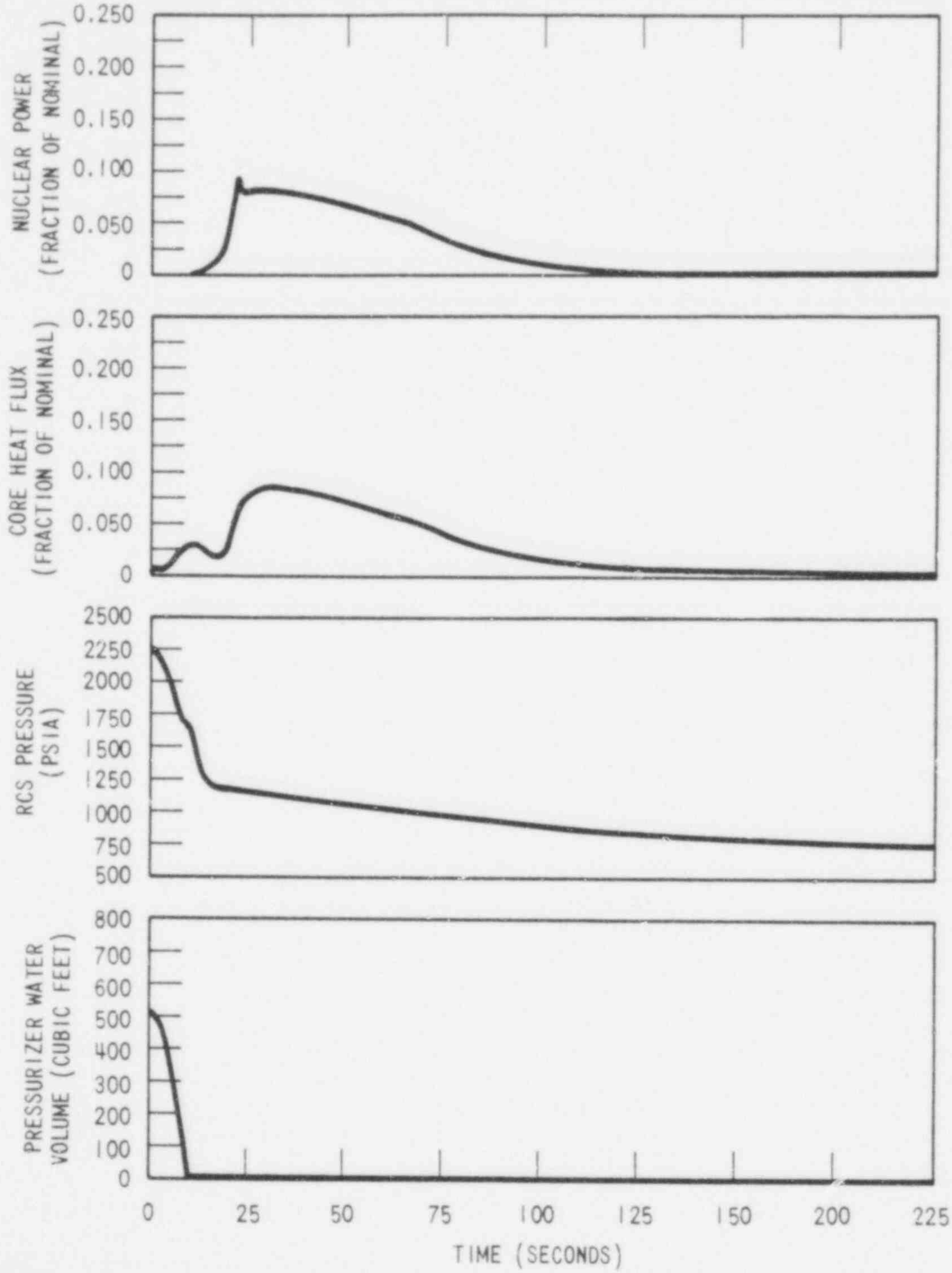

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Figure 15.1.5-2.

1.4 FT^2 Steamline Rupture,
Offsite Power Available

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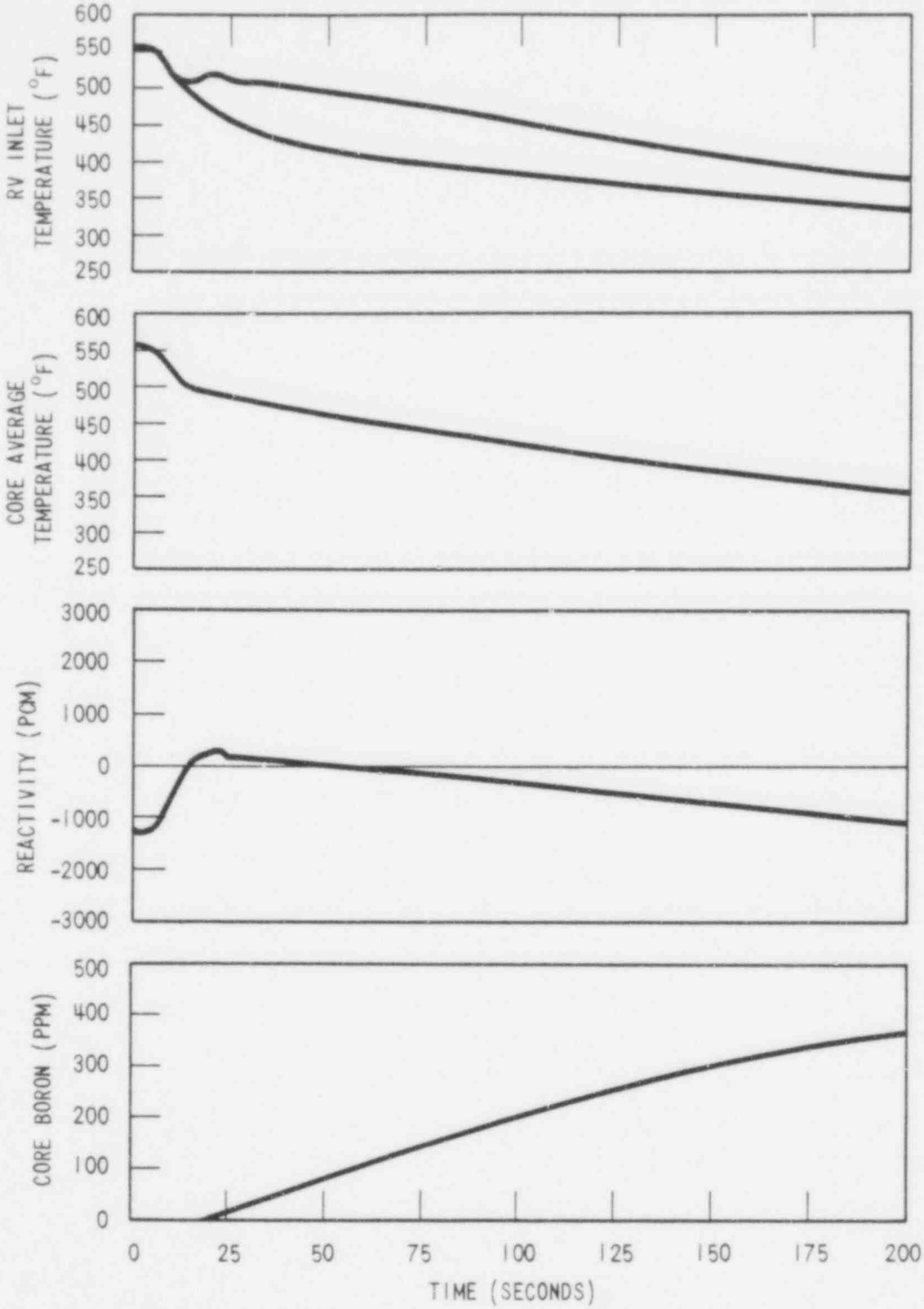

WCAP - 9500

Figure 15.1.5-3.

1.4 FT^2 Steamline Rupture, Offsite Power Available

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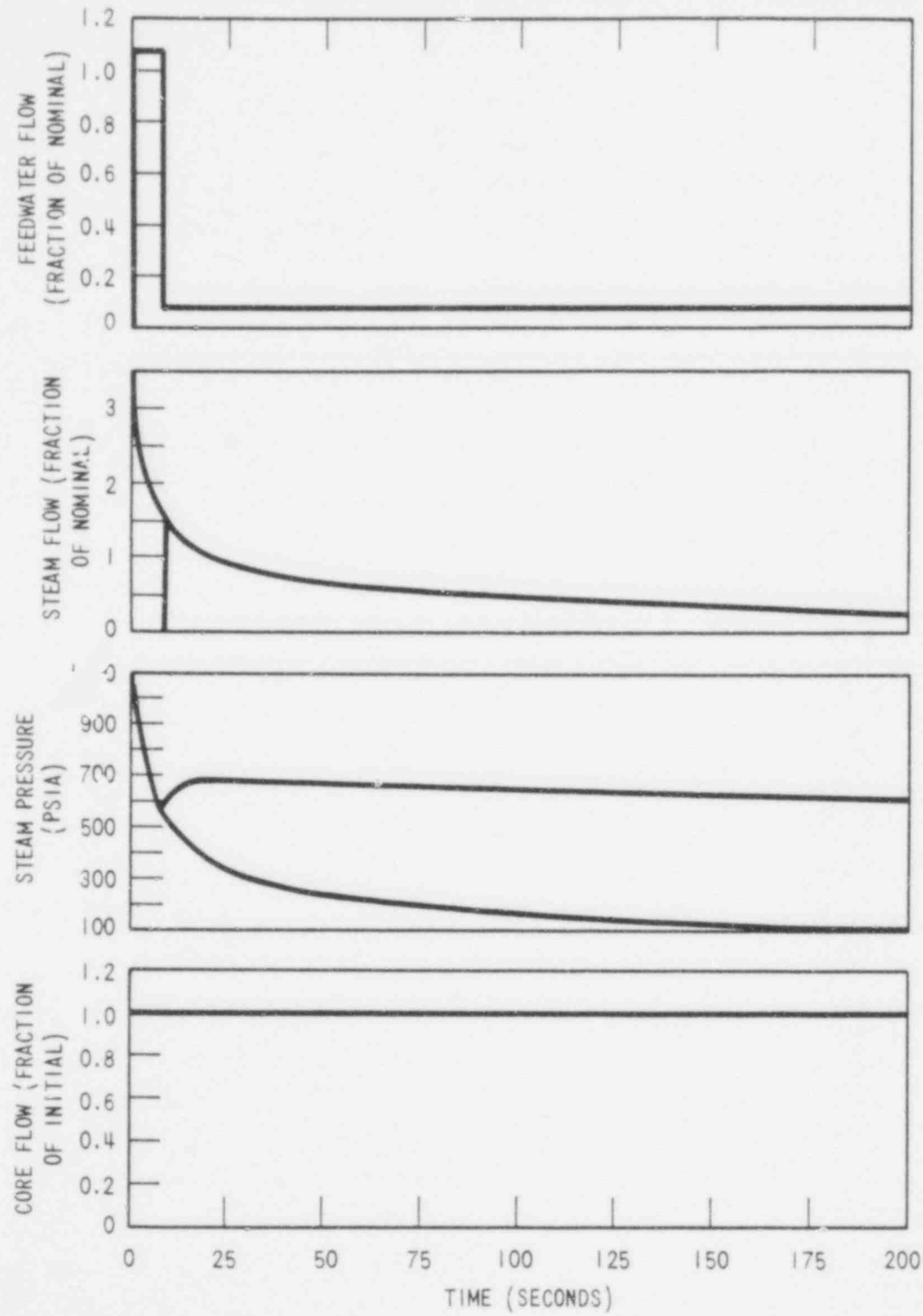
14,395-235



WCAP - 9500
Figure 15.1.5-4. 1.4 FT^2 Steamline Rupture, Offsite Power Available

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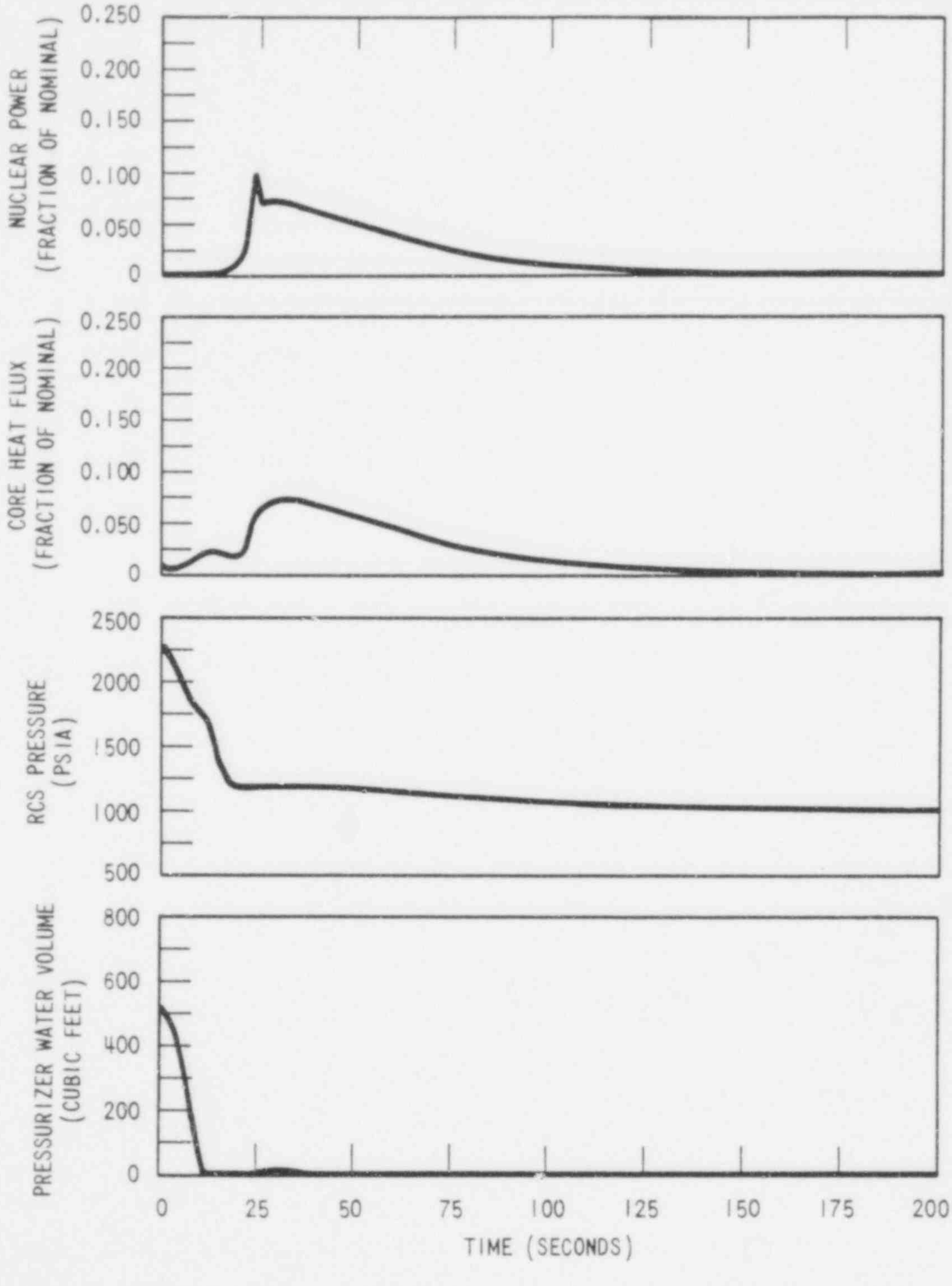

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Figure 15.1.5-5. 1.4 FT^2 Steamline Rupture, Offsite Power Not Available BLUE

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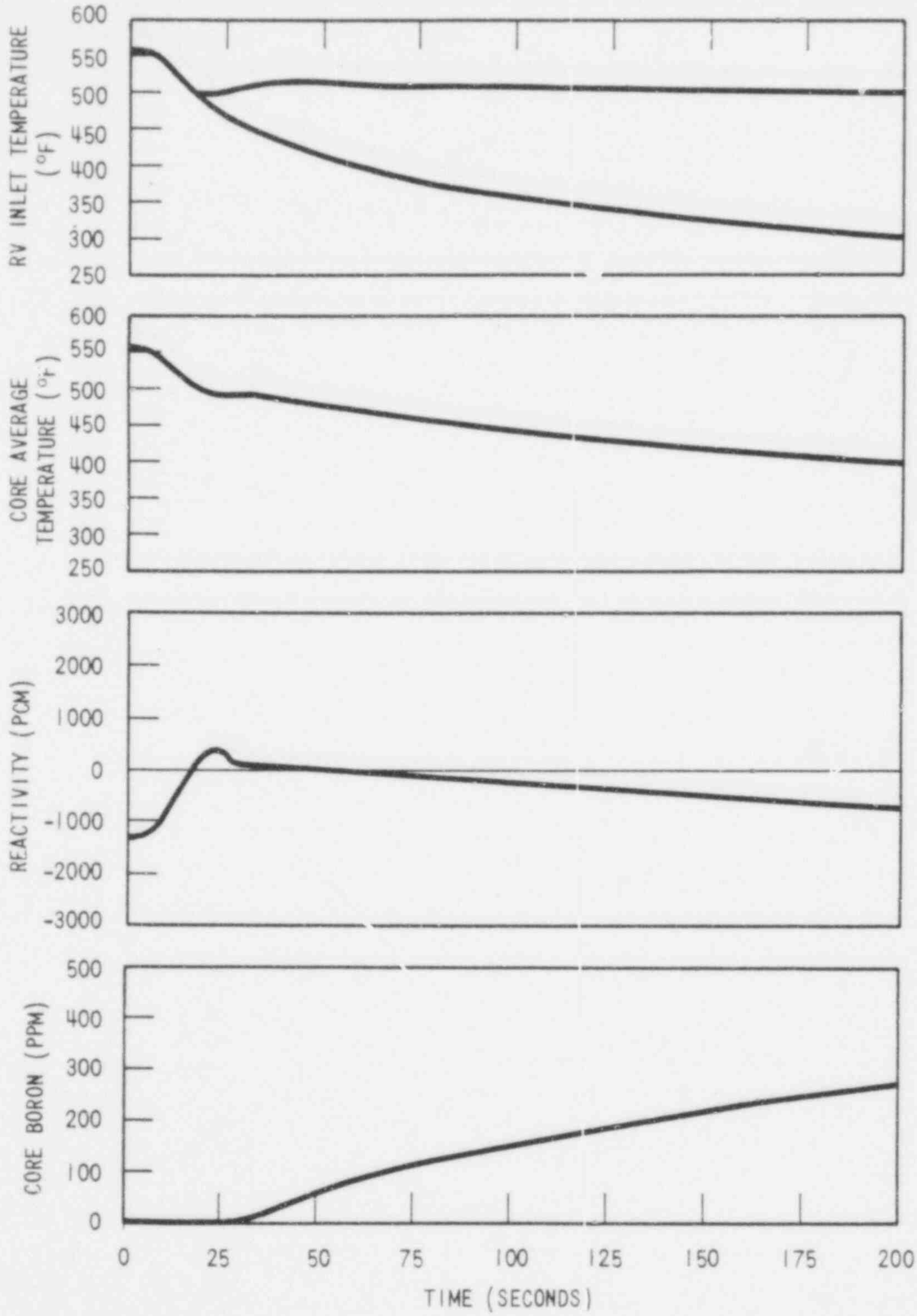

WCAP – 9500

Figure 15.1.5-6.

1.4 FT^2 Steamline Rupture, Offsite Power Not Available

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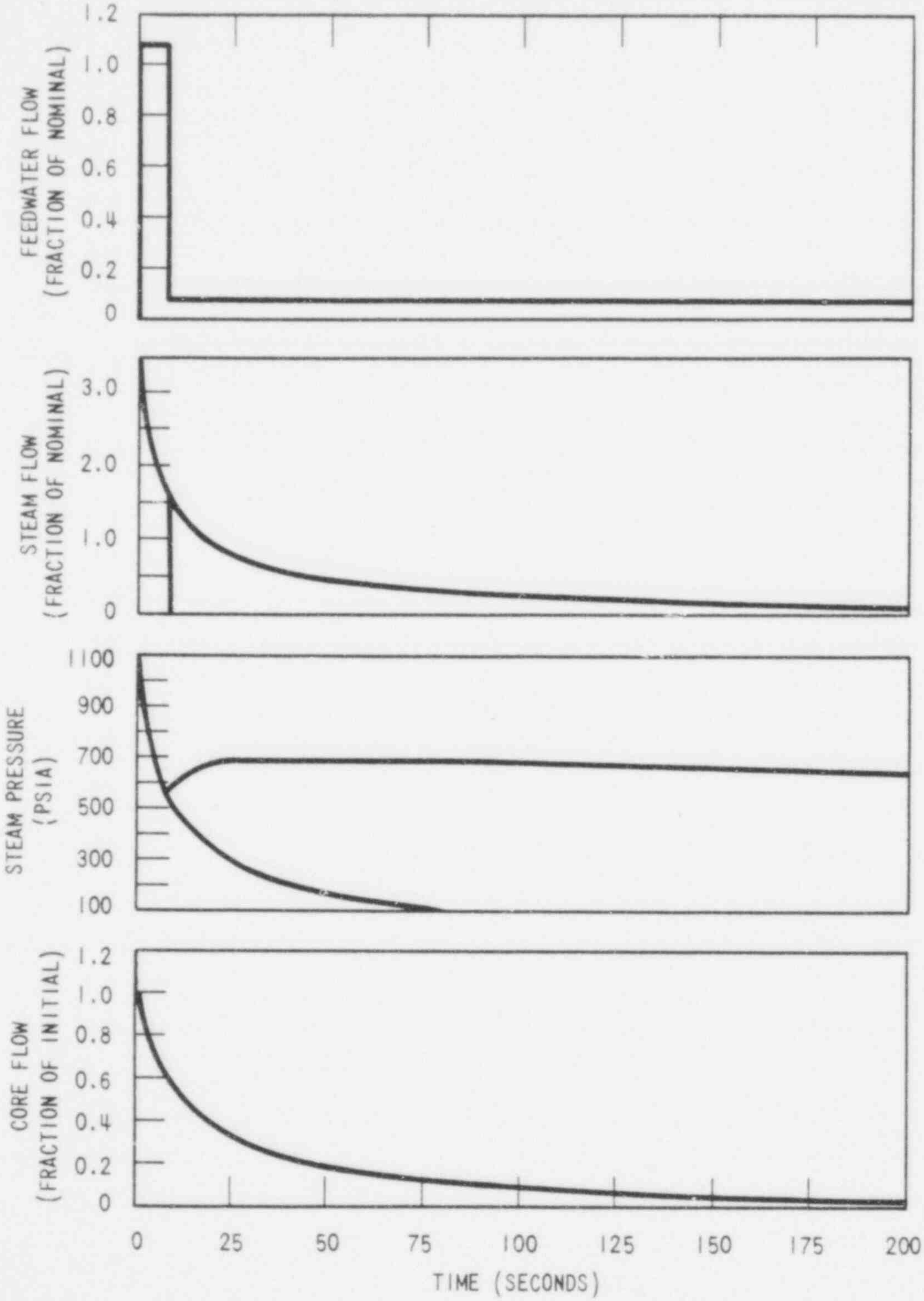

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Figure 15.1.5-7.

1.4 FT^2 Steamline Rupture, Offsite Power Not Available

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15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents nave 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 in Section 15.2:

1. Steam Pressure Regulator Malfunction
2. Loss of External Load
3. Turbine Trip
4. Inadvertent Closure of Main Steam Isolation Valves
5. Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip.
6. Loss of Non-Emergency A-C Power to the Station Auxiliaries
7. Loss of Normal Feedwater Flow
8. Feedwater System Pipe Break

The above items are considered to be ANS Conidtion II events, with the exception of a Feedwater System Pipe Break, which is considered to be an ANS Condition IV event. Section 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

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

There are no pressure regulators whose failure or malfunction could cause a steam flow transient.

15.2.2 LOSS OF EXTERNAL LOAD

15.2.2.1 Identification of Causes and Accident Description

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A major load loss on the plant can result from loss of external electrical load due to some electrical system disturbance. Offsite AC

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power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, the plant would be expected to trip from the Reactor Protection System if a safety limit were approached. With full load rejection capability plant operation would be expected to continue without a reactor trip. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure and overtemperature ΔT trips. Power and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6Hz. This resulting overfrequency is not expected to damage the sensors (Non-NSSS) in any way. However, it is noted that frequency testing of this equipment is required by the Technical Specifications. Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flowrate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition is not seen by safety related pump motors, Reactor Protection System equipment, or other safeguards loads. Safeguards loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor Protection System equipment is supplied from the inverters; the inverters are supplied from a DC bus energized from batteries or by a rectified AC voltage from safeguards buses.

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

The steam generator safety valve capacity is sized to remove the steam flow at the Engineered Safety Features Rating (105 percent of steam flow at rated power) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

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

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

A loss of external load event results in an NSSS transient that is less severe than a turbine trip event (see Section 15.2.3). Therefore, a detailed transient analysis is not presented for the loss of external load.

The primary-side transient is caused by a decrease in heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow. (Should feed flow not be reduced, a larger heat sink would be available and the transient would be less severe). Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves in approximately 0.3 seconds. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.1 seconds. Therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0.8-1.

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

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

Normal reactor control systems and Engineered Safety Systems are not required to function. The Auxiliary Feedwater System may, however, be automatically actuated following a loss of main feedwater; this will further mitigate the effects of the transient.

The Reactor Protection System may be required to function following a complete loss of external load to terminate core heat input and prevent DNB. Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will prevent operation of any system required to function. Refer to Reference [2] for a discussion of ATWT considerations.

15.2.2.3 Radiological Consequences

Loss of external load from full power would result in the operation of the steam dump system. This system keeps the main turbine generator operating to supply auxiliary electrical loads. Operation of the steam dump system results in bypassing steam to the condenser. If steam dumps are not available, steam generator safety and relief valvcs relieve to the atmosphere. Since no fuel damage is postulated for this transient the radiological releases will be less severe than those for the steam-line break accident analyzed in Subsection 15.1.5.3.

15.2.2.4 Conclusions

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

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the turbine stop valves close rapidly (typically 0.1 sec.) on loss of trip-fluid pressure actuated by one of a number of possible turbine trip signals. Turbine-trip initiation signals include:

1. Generator Trip
2. Low Condenser Vacuum
3. Loss of Lubricating Oil
4. Turbine Thrust Bearing Failure
5. Turbine Overspeed
6. Main Steam Reheat High Level
7. Manual Trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant primary system transient as described in Section 15.2.2.1 for the loss of external load event. A more severe transient occurs for the turbine trip event due to the more rapid loss of steam flow caused by the more rapid valve closure.

The automatic steam dump system would normally accomodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the Auxiliary Feedwater System to insure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control. See 15.2.2.1 for a further discussion of the transient.

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

A turbine-trip event is more limiting than loss of external load, loss of condenser vaccuum, and other turbine-trip events. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Section 15.2.3.2.

The plant systems and equipment available to mitigate the consequences of a turbine-trip are discussed in Section 15.0.8 and listed in Table 15.0.8.1.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. 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 to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program LOFTRAN (Reference 3). The program simulates the neutron kinetics, RCS pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant varibles including temperatures, pressures, and power level.

This accident is analyzed with the Improved Thermal Design Procedures as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3.

Major assumptions are summarized below:

1. Initial Operating Conditions - 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 WCAP-8567.

2. Moderator and Doppler Coefficients of Reactivity - the turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a least negative moderator temperature coefficient and the least negative Doppler coefficients. (See Figure 15.0.4-1).

3. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce and severity of the transient.

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

5. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the minimum and maximum reactivity feedback cases are analyzed:

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

   b. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

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

7. Reactor trip is actuated by the first Reactor Protection System trip setpoint reached. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, and low-low steam generator water level.

Except as discussed above, normal reactor coolant system and Engineered Safety Systems are not required to function. Several cases are presented in which pressurizer spray and power operated relief valves are assumed, but the more limiting cases where these functions are not assumed are also presented.

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The Reactor Protection System may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference [2].

Results

The transient responses for a turbine trip from full power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 15.2.3-1 through 15.2.3-8). The calculated sequence of events for the accident is shown in Table 15.2.3-1

Figures 15.2.3-1 and 15.2.3-2 show the transient responses for the total loss of steam load with a least negative moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip channel. The minimum DNBR remains well above the the limit value. Pressurizer pressure never attains a value sufficiently high to actuate the safety valves so the primary system pressure remains substantially below the 110% design value. The Steam Generator Safety Valves limit the secondary steam conditions to saturation at the safety valve setpoint.

Figures 15.2.3-3 and 15.2.3-4 show the response for the total loss of steam load with a large negative moderator temperature coefficient. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The pressurizer safety valves are not actuated for this case.

In the event that feedwater flow is not terminated at the time of turbine trip for this case, flow would continue under automatic control with the reactor at a reduced power. The operator would take action to

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terminate the transient and bring the plant to a stabilized condition. If no action were taken by the operator the reduced power operation would continue until the condenser hotwell was emptied. A low-low steam generator water level reactor trip would be generated along with auxiliary feedwater initiation signals. Auxiliary feedwater would then be used to remove decay heat with the results less severe than those presented in Section 15.2.7, Loss of Normal Feedwater Flow.

The turbine trip accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2.3-5 and 15.2.3-6 show the transients with a least negative moderator coefficient. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below the initial value throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

Figures 15.2.3-7 and 15.2.3-8 are the transients with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

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

15.2.3.3 Radiological Consequences

The radiological consequences resulting from atmospheric steam dump will be less severe than the steamline break event analyzed in Subsection 15.1.5.3 since no fuel damage is postulated to occur.

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15.2.3.4 Conclusions

Results of the analyses, including those in Reference [1], show that the plant design is such that a turbine trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the Reactor Protection System, i.e., the DNBR will be maintained above the the value. The applicable acceptance criteria as listed in Section 15.0.1 have been met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. The radiological consequences in this event will be less than the steam break event analyzed in Subsection 15.1.5.3.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

Inadvertent closure of the main steam isolation valves would result in a turbine trip. Turbine trips are discussed in Section 15.2.3.

15.2.5 LOSS OF CONDENSOR VACUUM AND OTHER EVENTS CAUSING A TURBINE TRIP

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

15.2.6 LOSS OF NON-EMERGENCY A-C POWER TO THE PLANT AUXILIARIES

15.2.6.1 Identification of Causes and Accident Description

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

This transient is more severe than the turbine trip event analyzed in Section 15.2.3 because for this case the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

(1) upon reaching one of the trip setpoints in the primary and secondary systems as result of the flow coastdown and decrease in secondary heat removal; or (2) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

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

1. Plant vital instruments are supplied from emergency DC power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

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

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

The Auxiliary Feedwater System is started automatically as follows:

Two motor drive auxiliary feedwater pumps are started on any of the following:

a. Low-low level in any generator
b. Any safety injection signal
c. Loss of offsite power
d. Trip of all main feedwater pumps
e. Manual actuation

One turbine drive auxiliary feedwater pump is started on any of the following:

a. Low-low level in any two steam generators
b. Loss of offsite power
c. Manual actuation

The motor driven auxiliary feedwater pumps are supplied power by the diesels and the turbine-driven pump utilizes steam from the secondary system. Both type pumps are designed to supply rated flow within one minute of the initiating signal even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the secondary steam to the atmosphere.

The pumps take suction from the auxiliary feedwater storage tank for delivery to the steam generators.

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.

A loss of non-emergency AC power to the station auxiliaries is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

A loss of AC power event, as described above, is a more limiting event than the turbine-trip-initiated decrease in secondary heat removal without loss of AC power, which analyzed in Section 15.2.3. However, a loss of AC power to the plant auxiliaries as postulated above could result in a loss of normal feedwater if the condensate pumps lose their power supply.

Following the reactor coolant pump coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of a-c power event is sufficient to remove residual heat from the core.

The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, Reactor Coolant System (RCS) including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the Engineered Safety Features design rating.

2. A conservative core residual heat generation based upon long term operation at the intial power level preceding the trip.

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation, following the reactor coolant pump coast-down.

4. Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

5. The worst single failure in the auxiliary feedwater system occurs.

6. Auxiliary feedwater is delivered to four steam generators.

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

The assumptions used in the analysis are essentially identical to the loss of normal feedwater flow incident (Section 15.2.7) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

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

Results

The transient response of the RCS following a loss of ac power is shown in Figures 15.2.6-1 and 15.2.6-2. The calculated sequence of events for this event is listed in Table 15.2.6-1.

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The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Section 15.3.2) i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay 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.

15.2.6.3 Radiological Consequences

A loss of nonessential AC power to plant auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power relief valves or safety valves. The parameters to be used in calculation of the radiological consequences of the loss of AC Power Analysis are summarized in Table 15.2.6-1. Since no fuel damage is postulated to occur from this transient, the radiological consequences will be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the Reactor Coolant System has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage. The radiological consequences of this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the Reactor Coolant System (RCS). Since the plant is tripped well before the steam generator heat transfer capabililty is reduced, the primary system variables never approach a DNB condition.

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

1. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

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

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

The Auxiliary Feedwater System is started automatically as discussed in Section 15.2.6.1. The steam driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators. The pumps take suction directly from the auxiliary feedwater storage tank for delivery to the steam generators.

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

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

A detailed anaysis using the LOFTRAN Code (Reference 3) is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the Engineered Safety Features design rating.

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

3. Reactor trip occurs on steam generator low-low level.

4. The worst single failure in the auxiliary feedwater system occurs.

5. Auxiliary feedwater is delivered to four steam generators.

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

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

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

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

An additional assumption made for the loss of normal feedwater evaluation is that only the pressurizer safety valves are assumed to function normally. Operation of the valves maintains peak RCS pressure at or below the actuation setpoint (2500 psia) throughout the transient.

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

Plant systems and equipment which are necessary to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. Normal reactor control systems are not required to function. The Reactor Protection System is required to function following a loss of normal feedwater as analyzed here. The Auxiliary Feedwater System is required to deliver a minimum auxiliary feedwater flowrate. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference [2].

Results

Figures 15.2.7-1 and 15.2.7-2 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, 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, at least two auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater pumps are such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS safety valves. Figure 15.2.7-1 shows that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2.3-1. As shown in Figures 15.2.7-1 and 15.2.7-2, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Radiological Consequences

If steam dump to the condenser is assumed to be lost, heat removal from the secondary system would occur through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur, radiological consequences resulting from this transient would be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.7.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves. The radiological consequences of this event would be less severe than the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.8 FEEDWATER SYSTEM PIPE BREAK

15.2.8.1 Identification of Causes and Accident Description

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

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

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

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

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

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

An Auxiliary Feedwater System is provided to assure that adequate feedwater will be available such that:

1. No substantial overpressurization of the RCS shall occur.

2. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

Refer to Section 10.4.9 for a description of the Auxiliary Feedwater System interfaces.

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

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. A number of cases of feedwater line break have been analyzed. Based on these analyses, it has been shown that the most limting feedwater line ruptures are the double ended rupture of the largest feedwater line, occurring at full power with and without loss of offsite power, with no credit taken for pressurizer control. These cases are analyzed below.

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

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

    a. High pressurizer pressure.

    b. Overtemperature ΔT.

    c. Low-low steam generator water level in any steam generator.

    d. Safety injection signals from any of the following:

        1) 2/3 low steam line pressure in any loop.

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2) 2/3 high containment pressure

(Refer to Chapter 7 for a description of the actuation system).

2. An Auxiliary Feedwater System to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10.4.9 for a description of the Auxiliary Feedwater System).

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 3) is performed in order to determine the plant transient following a feedwater line rupture.

The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The cases analyzed assume a double ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

1. The plant is initially operating at 102 percent of engineered safeguards power.

2. Initial power coolant average temperature is 4.0°F above the nominal value, and the initial pressurizer pressure is 30 psi above its nominal value.

3. No credit is taken for the pressurizer power operated relief valves or pressurizer spray.

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4. Initial pressurizer level is at the nominal programmed value plus 5 percent (error); initial steam generator water level is at the nominal value plus 5 percent in the faulted steam generator and at the nominal value minus 5 percent in the intact steam generators.

5. No credit is taken for the high pressurizer pressure reactor trip.

6. Main feedwater flow to all steam generators is assumed to be lost at the time the break occurs (all main feedwater spills out through the break).

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

8. A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

9. Reactor trip is assumed to be initiated when the low-low level trip setpoint minus 10 percent of the narrow range span in the faulted steam generator is reached.

10. The Auxiliary Feedwater System is actuated by the low-low steam generator water level signal. The Auxiliary Feedwater System is assumed to supply a total of 485 gallons per minute (gpm) to the three unaffected steam generators, including allowance for possible spillage through the main feedwater line break. A 60 second delay was assumed following the low-low level signal to allow time for startup of the emergency diesel generators and the auxiliary feedwater pumps. An additional 115 seconds was assumed before the feedwater lines were purged and the relatively cold (134°F) auxiliary feedwater entered the unaffected steam generators.

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

12. No credit is taken for charging or letdown.

13. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.

14. Conservative core residual heat generation is assumed based upon long term operation at the initial power level preceding the trip.

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

    a. High pressurizer pressure.
    b. Overtemperature ΔT.
    c. High pressurizer level.
    d. High Containment pressure.

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor driven auxiliary feedwater pumps, which then deliver auxiliary feedwater flow to the steam generators. The turbine drive auxiliary feedwater pump is initiated if the low-low steam generator water signal is reached in at least two steam generators. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes the main steam line isolation valves in all steam lines. This signal also gives a safety injection signal which initiates flow of borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Emergency operating procedures following a secondary system line rupture call for the following actions to be taken by the reactor operator:

1. Isolate feedwater flow spilling out the break of ruptured steam generator and align system so level in intact steam generators recovers.

2. Stop high head safety injection charging pumps if 1) wide range reactor coolant pressure is greater than 2000 psig, and is stable or increasing, 2) pressurizer water level is greater than 50 percent of span, and 3) steam generator narrow range level indication exists in at least one steam generator.

Isolating feedwater flow through the break allows additional auxiliary feedwater flow to be diverted to the intact steam generators.

Subsequent to recovery of level in the intact steam generators, the high head safety injection pumps will be turned off and plant operating procedures will be followed in cooling the plant to hot shutdown conditions.

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

No reactor control systems are assumed to function. The Reactor Protection System is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems assumed to function are the Auxiliary Feedwater System and the Safety Injection System. For the Auxiliary Feedwater System, the worst case configuration has been used, i.e., three intact steam generators receive auxiliary feedwater following the break. One motor driven auxiliary feedwater pump has been assumed to fail; the second motor driven pump together with the turbine driven pump delivers 485 gpm to the three intact steam generators allowing for spillage out of the break. Only one train of safety injection has been assumed to be available.

Following the trip of the reactor coolant pumps for the feedline rupture without offsite power, there will be a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Section 15.2.6, for the loss of AC power transient, to be sufficient to remove core decay heat following reactor trip. Pump coastdown characteristics are demonstrated in Sections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A detailed description and analysis of the Safety Injection System is provided in Section 6.3. The Auxiliary Feedwater System is described in Section 10.4.9.

Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2.8-1 through 15.2.8-8. Results for the case with offsite power available are presented in Figures 15.2.8-1 through 15.2.8-4. Results for the case where offsite power is lost are presented in Figures 15.2.8-5 through 15.2.8-8. The calculated sequence of events for both cases analyzed are listed in Table 15.2.3-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented in Figures 15.2.8-2 and 15.2.8-4 (with offsite power available) and Figures 15.2.8-6 and 15.2.8-8 (without offsite power) show that pressures in the RCS and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low-low steam generator water level. Pressure then decreases, due to the loss of heat input, until the Safety Injection System is actuated on low steam line pressure in the ruptured loop. Coolant expansion occurs due to reduced heat transfer capability in the steam generators; the pressurizer safety valves open to maintain primary pressure at an acceptable value. Addition of the safety injection flow aids in cooling down the primary and helps to ensure that sufficient fluid exists to keep the core covered with water.

Figures 15.2.8-1 and 15.2.8-5 show that following reactor trip, the plant remains subcritical.

RCS pressure will be maintained at the safety valve setpoint until safety injection flow is terminated by the operator, as mentioned in Section 15.2.8.2. The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the Auxiliary Feedwater System and makeup is provided by the Safety Injection System.

The major difference between the two cases analyzed can be seen in the plots of hot and cold leg temperatures, Figure 15.2.8-3 (with offsite power available) and Figure 15.2.8-7 (without offsite power). It is apparent from the initial portion of the transient (~200 seconds), that the case without offsite power results in higher temperatures in the hot leg. For longer times, however, the case with offsite power results in a more severe rise in temperature until the coolant pumps are turned off and the Auxiliary Feedwater System is realigned. The pressurizer fills for the case with power due to the increased coolant expansion resulting from the pump heat addition; hence, water is relieved for the case with power. As previously stated, however, the core remains covered with water for both cases.

15.2.8.3 Radiological Consequences

The feedwater line break with the most significant consequences would be one that occurred inside the containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the containment. Since no fuel failures are postulated, the radioactivity released would be less than that for the steamline break, as analyzed in Subsection 15.1.5.3. Furthermore, automatic isolation of the containment would further reduce any radiological consequences from this postulated accident.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed Auxiliary Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radiological doses from the postulated feedwater line rupture would be less than those previously presented for the postulated steam line break.

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15.2.9 REFERENCES

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

2. "Westinghouse Anticipated Transients Without Trip Analysis", WCAP-8330, August 1974.

3. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907, June 1972.

TABLE 15.2.3-1 (Page 1)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
Turbine Trip		
1. With Pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	Initiation of steam release from steam generator safety valves	7.0
	High pressurizer pressure reactor trip point reached	8.2
	Rods begin to drop	10.2
	Minimum DNBR occurs	11.5
	Peak pressurizer pressure occurs	12.0

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TABLE 15.2.3-1 (Page 2)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
2. With pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	Peak pressurizer pressure occurs	3.5
	Initiation of steam release from steam generator safety valves	8.0
	Low-low steam generator reactor trip point reached	86
	Rods begin to drop	88
	Minimum DNBR occurs	(1)

(1) DNBR does not decrease below its initial value.

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TABLE 15.2.3-1 (Page 3)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
3. Without pressurizer control (minimum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	4.0
	Rods begin to drop	6.0
	Initiation of steam release from steam generator safety valves	7.0
	Peak pressurizer pressure occurs	7.5
	Minimum DNBR occurs	(1)

(1) DNBR does not decrease below its initial value. 614 297

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TABLE 15.2.3-1 (Page 4)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
4. Without pressurizer control (maximum reactivity feedback)	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	3.9
	Rods begin to drop	5.9
	Initiation of steam release from steam generator safety valves	7.0
	Peak pressurizer pressure occurs	7.0
	Minimum DNBR occurs	(1)

(1) DNBR does not decrease below its initial value.

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TABLE 15.2.3-1 (Page 5)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
Loss of Non-Emergency AC Power	Main feedwater flow stops	10
	Low-low steam generator water level trip	60
	Rods begin to drop	62
	Reactor coolant pumps begin to coastdown	62
	Peak water level in pressurizer occurs	64
	Four steam generators begin to receive auxiliary feedwater from one motor driven auxiliary feedwater pump	121
	Core decay heat decreases to auxiliary feedwater heat removal capacity	$\backsimeq$ 1800

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TABLE 15.2.3-1 (Page 6)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
Loss of Normal Feedwater Flow	Main feedwater flow stops	10
	Low-low steam generator water level trip	60
	Rods begin to drop	62
	Peak water level in pressurizer occurs	64
	Four steam generators begin to receive auxiliary feedwater from two motor driven auxiliary feedwater pumps	121
	Core decay heat decreases to auxiliary feedwater heat removal capacity	< 1800

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TABLE 15.2.3-1 (Page 7)

Time S[illegible]ence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
Feedwater system pipe break		
1. With offsite power available	Main feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	31
	Rods begin to drop	33
	Auxiliary feedwater is delivered to intact steam generators	92
	Low steam line pressure setpoint reached in ruptured steam generator	240

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TABLE 15.2.3-1 (Page 8)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
	All main steam line isolation valves close	247
	Steam generator safety valve setpoint reached in intact steam generators	524
	Pressurizer water relief begins	1284
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	$\simeq$ 4000

TABLE 15.2.3-1 (Page 9)

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
2. Without offsite power	Main feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	31
	Rods begin to drop, power lost to the reactor coolant pumps	33
	Auxiliary feedwater is delivered to intact steam generators	92
	Low steam line pressure setpoint reached in ruptured steam generator	248

Time Sequence Of Events For Incidents Which Cause a Decrease In Heat Removal By The Secondary System

Accident	Event	Time (sec)
	All main steam line isolation valves close	255
	Steam generator safety valve setpoint reached in intact steam generators	559
	Core decay heat decreases to auxiliary feedwater heat removal capacity	$\simeq$ 1800

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TABLE 15.2.6-1

Parameters To Be Useo In Analysis Of The Radiological Consequences Of Loss Of AC Power Analysis

Parameter	Realistic Valve	Cons:rvative Valve
Core Thermal Power	3565 MWt	3565 MWt
Fraction of Core Power Produced in Rods Containing Defects	0.0012	0.01*
Fraction of Fuel Rods Whose Cladding Fails as a Result of the Accident	0.0	0.0
Total Steam Generator Leak Rate Prior to Accident	.009 gpm	1.0 gpm
Iodine Spike		
Release from Fuel to Coolant	See Table 15.0-8	See Table 15.0-8
Duration of Release	4 hrs	4 hrs
Total Steam Generator Leak Rate During Accident	.009 gpm	1.0 gpm**
Iodine Inventory in Secondary Coolant Prior to Accident	4.5×10^{-5} μCi/gm DE I-131***	1.0 μCi/gm DE I-131**
Duration of Plant Cooldown After Accident	8 hrs.	8 hrs.
Integrated Steam Release (assumed to be at a constant rate)		
0 - 2 hrs.	550,293 lb.	550,293 lb.
2 - 8 hrs.	1,405,802 lb.	1,405,802 lb.
Integrated Feedwater Flow (assumed to Heat at a constant rate)		
0 - 2 hrs.	779,432 lb.	779,432 lb.
2 - 8 hrs.	1,188,480 lb.	1,188,480 lb.

* May be decreased to correspond to tech spec limitation maximum primary coolant activity.

** 0.347 gpm in defective steam generator and 0.218 gpm per non-defective steam generator during accident and assumed to be independent of pressure differential across steam generator tubes.

*** D. E. = Dose Equivalent

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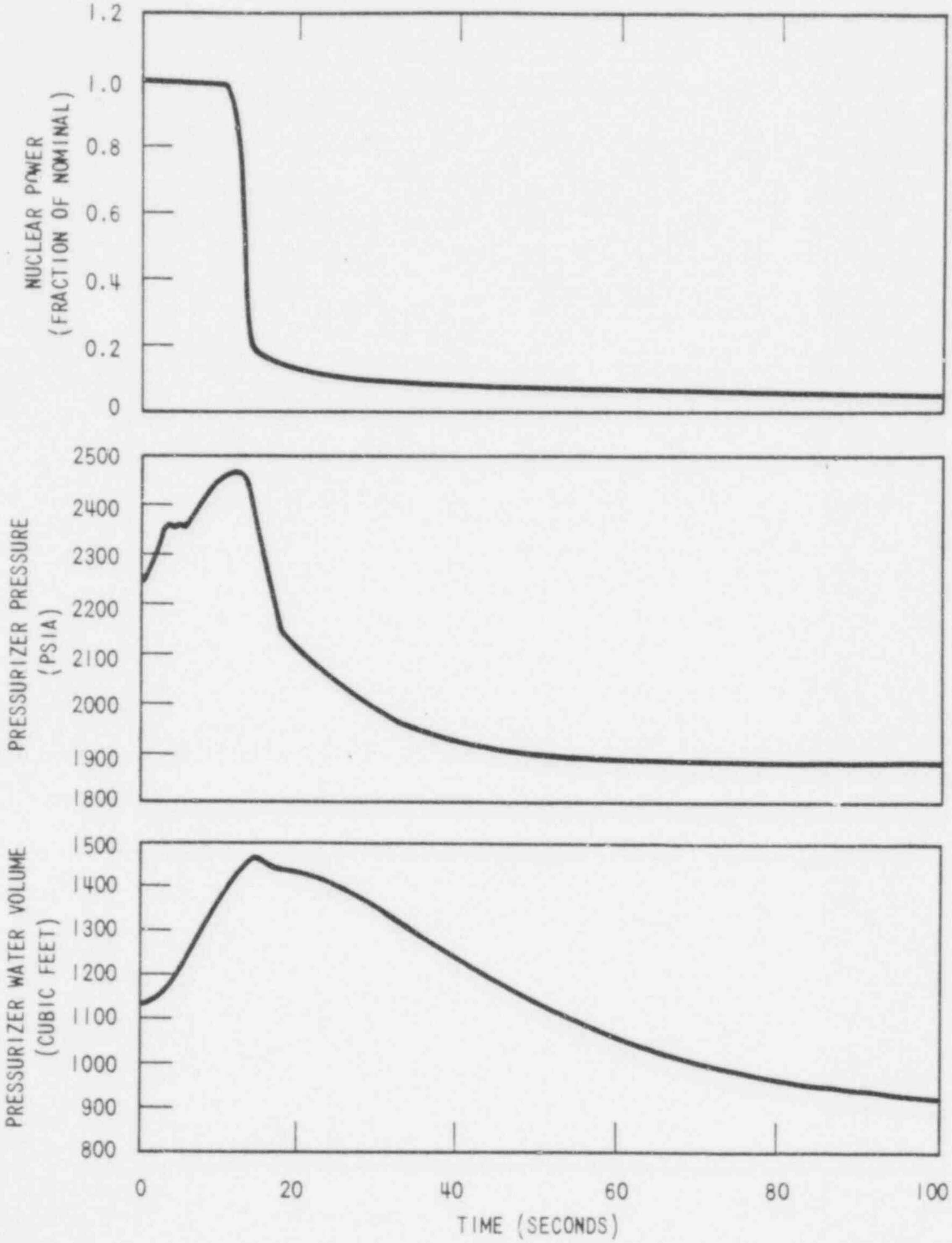

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Figure 15.2.3-1.

Turbine Trip Event With Pressurizer Spray and Power Operated Relief Valves, Minimum Reactivity Feedback BLUE

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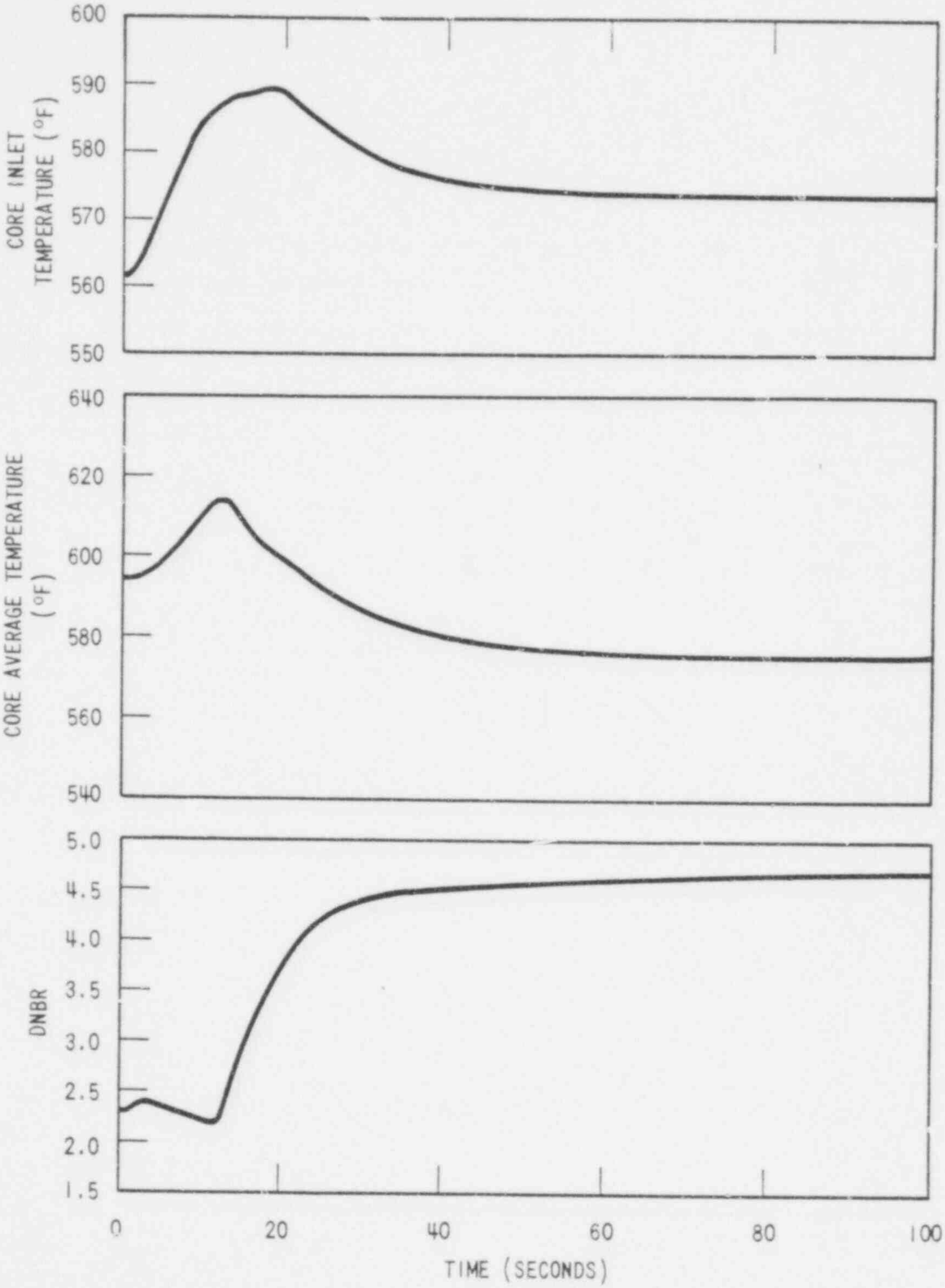

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Figure 15.2.3-2.

Turbine Trip Event With Pressurizer Spray and Power Operated Relief Valves, Minimum Reactivity Feedback BLUE

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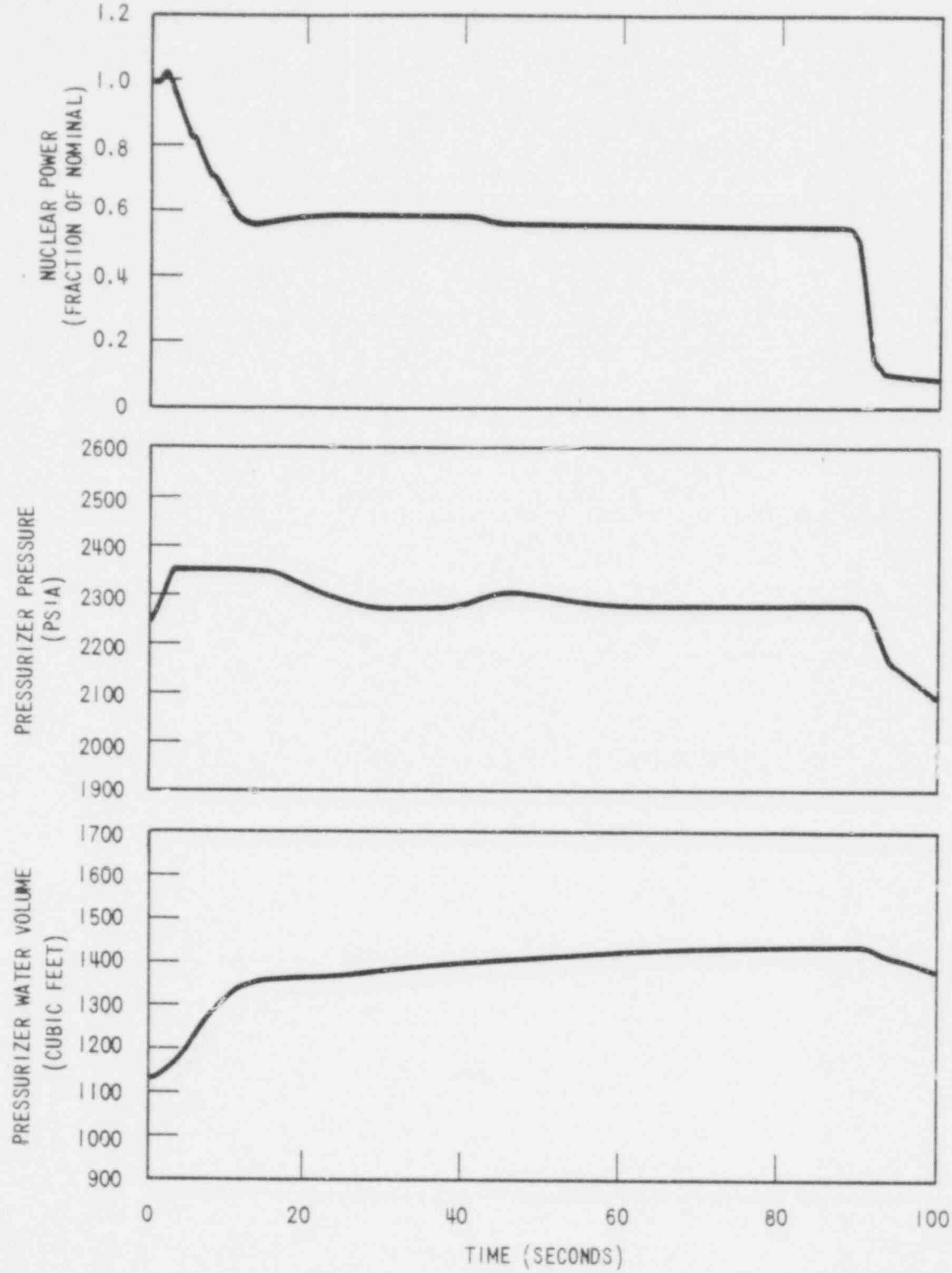

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Figure 15.2.3-3.

Turbine Trip Event With Pressurizer Spray and Power Operated Relief Valves, With Maximum Reactivity Feedback

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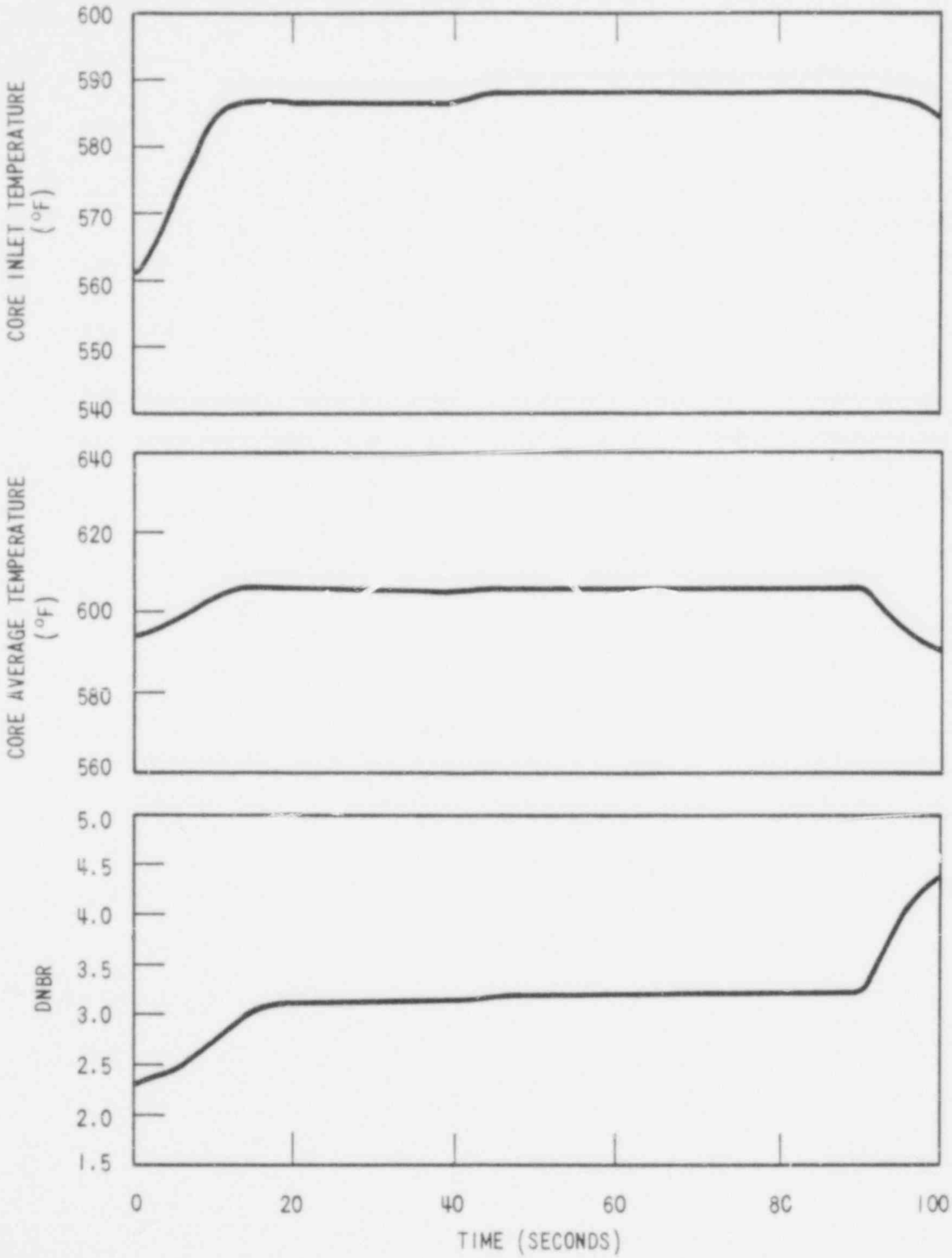

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Figure 15.2.3-4. BLUE Turbine Trip Event With Pressurizer Spray and Power Operated Relief Valves, With Ma:imum Reactivity Feedback

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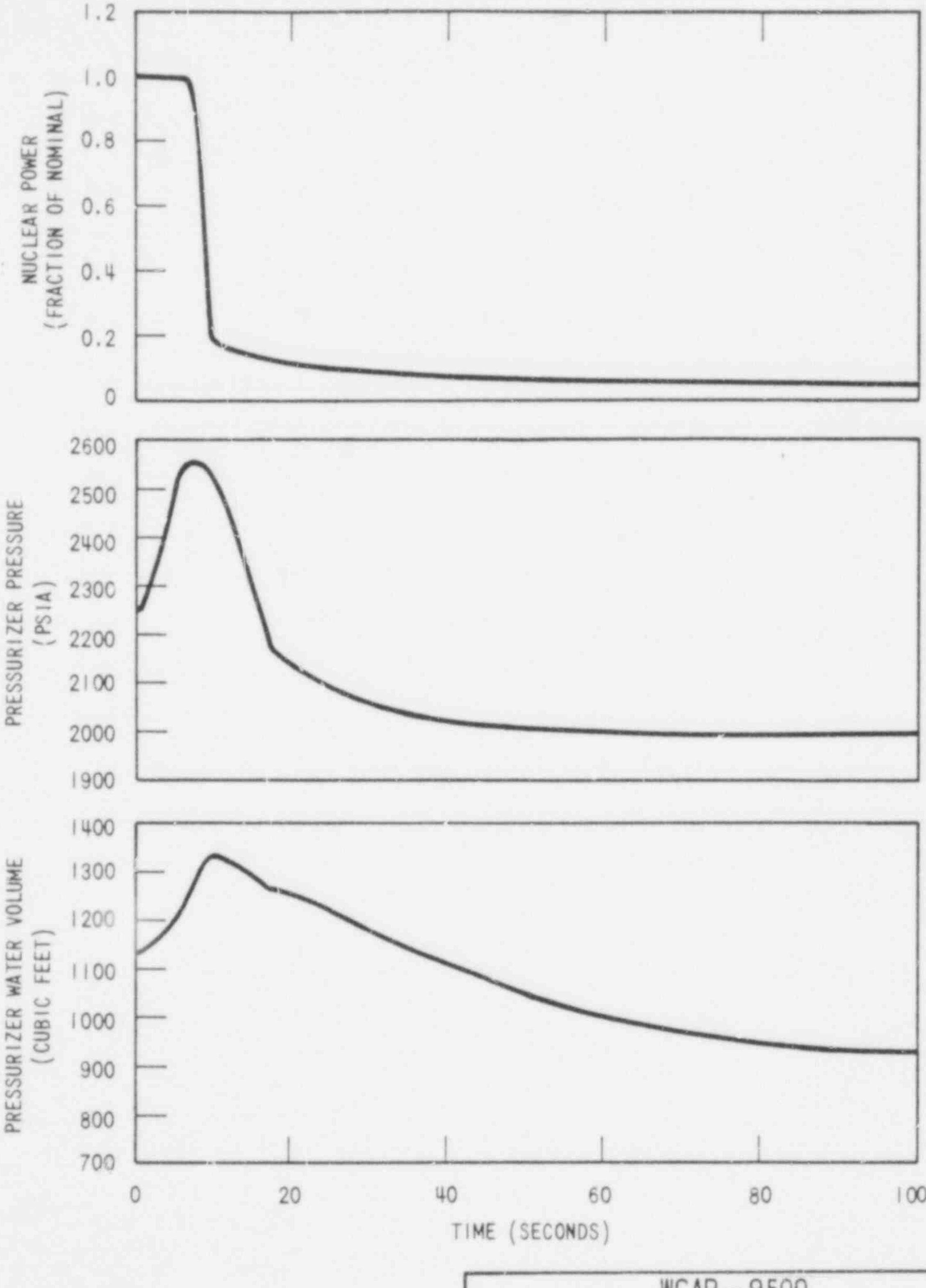

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Figure 15.2.3-5. BLUE

Turbine Trip Event Without Pressurizer Spray or Power Operated Relief Valves, Minimum Reactivity Feedback

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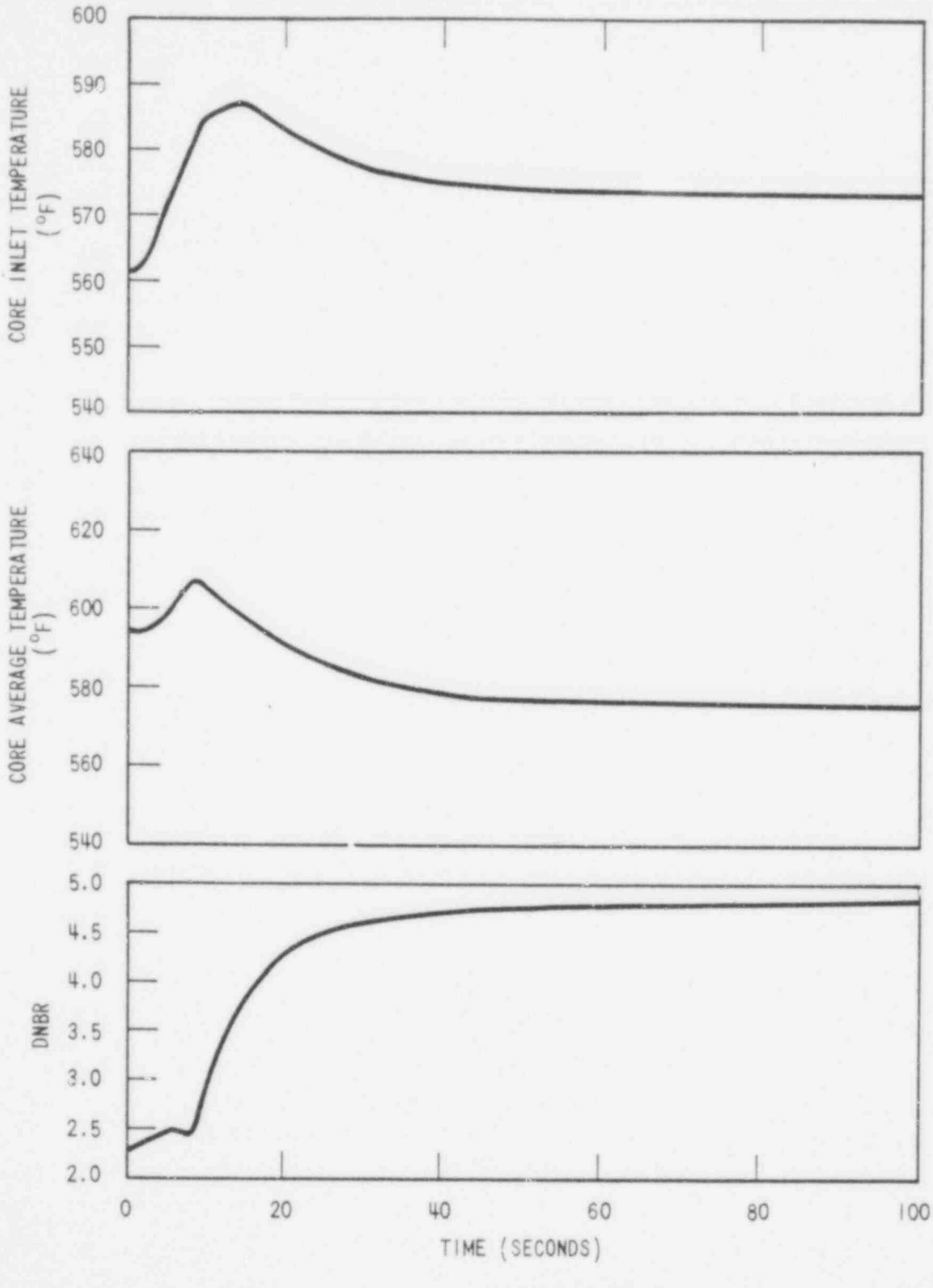

WCAP – 9500
Figure 15.2.3-6. BLUE Turbine Trip Event Without Pressurizer Spray or Power Operated Relief Valves, With Minimum Reactivity Feedback

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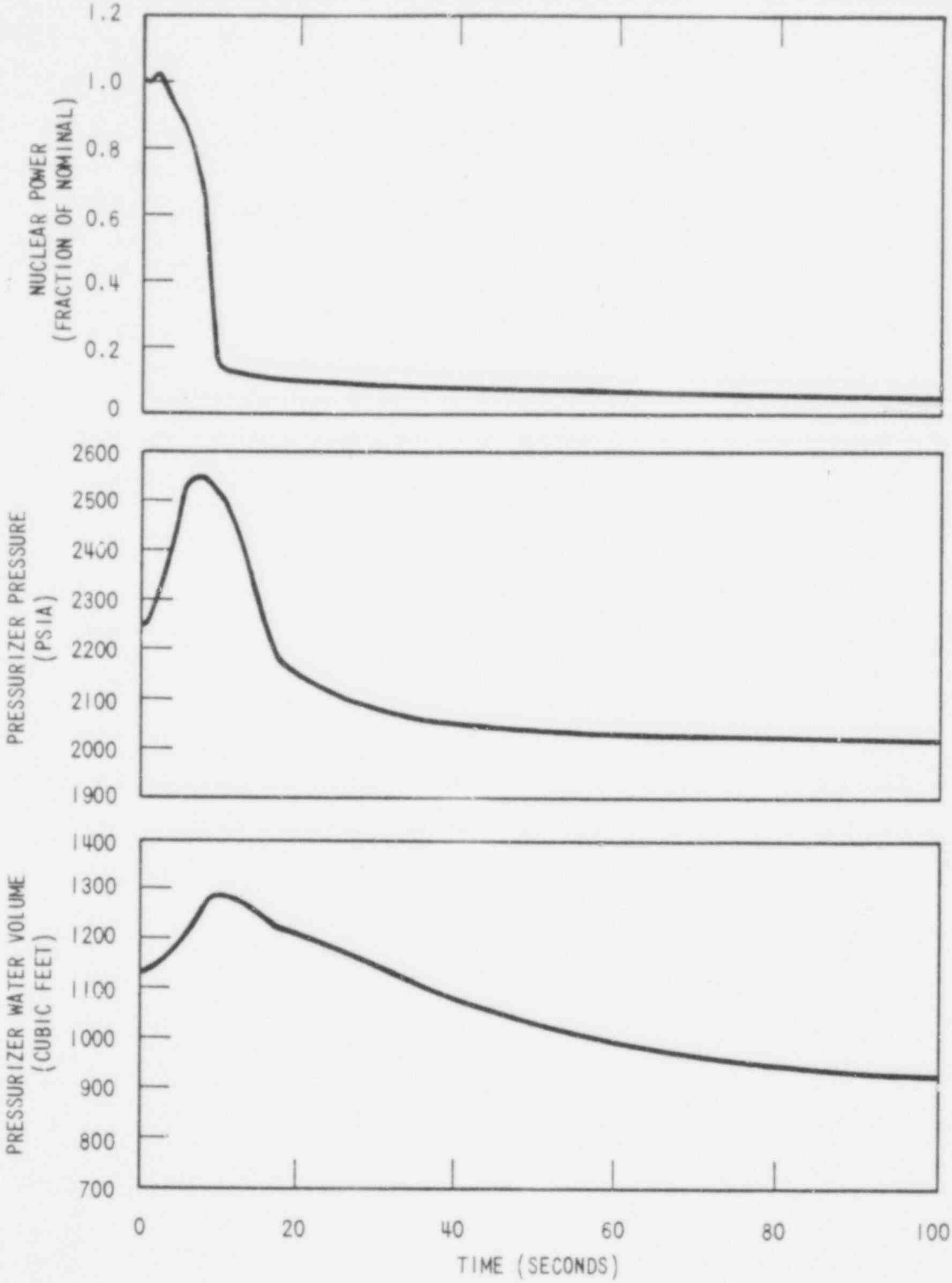
14.395-245



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WCAP – 9500

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Figure 15.2.3-7.

Turbine Trip Event Without Pressurizer Spray and Power Operated Relief Valves, With Maximum Reactivity Feedback

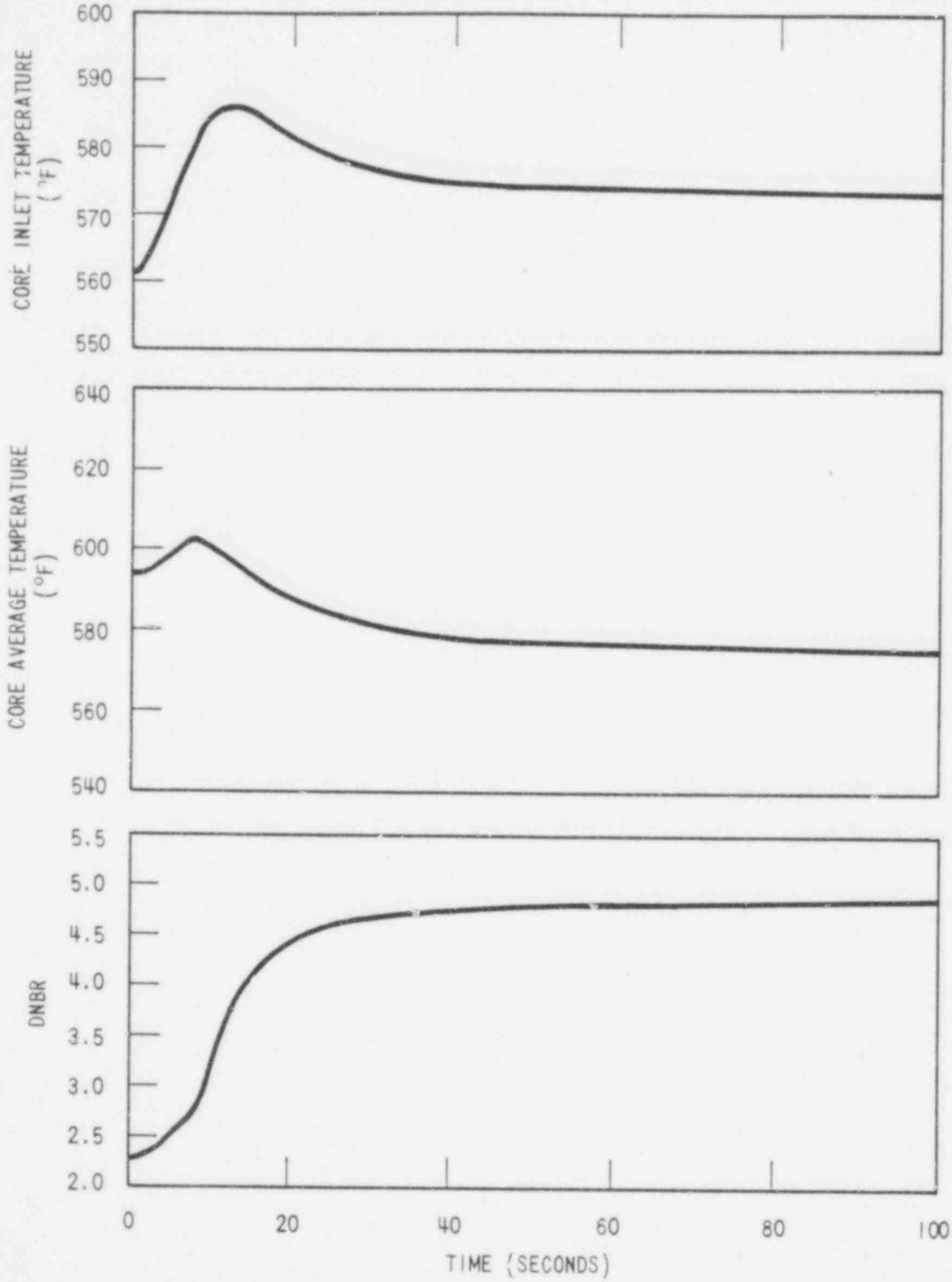

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Figure 15.2.3-8.

Turbine Trip Event Without Pressurizer Spray and Power Operated Relief Valves, With Maximum Reactivity Feedback

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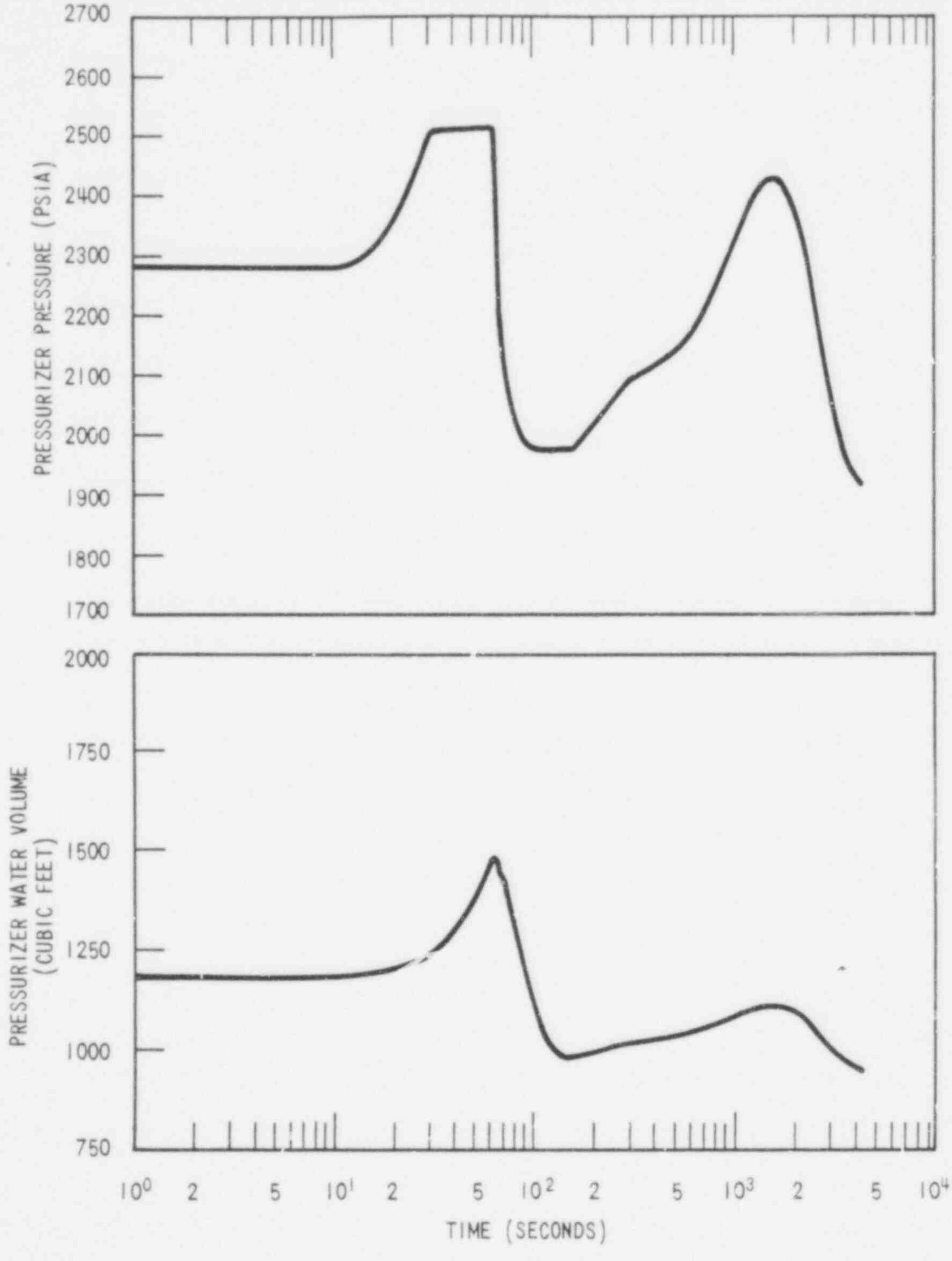

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Figure 15.2.6-1.

Pressurizer Pressure and Water Volume Transients for Loss of Offsite Power

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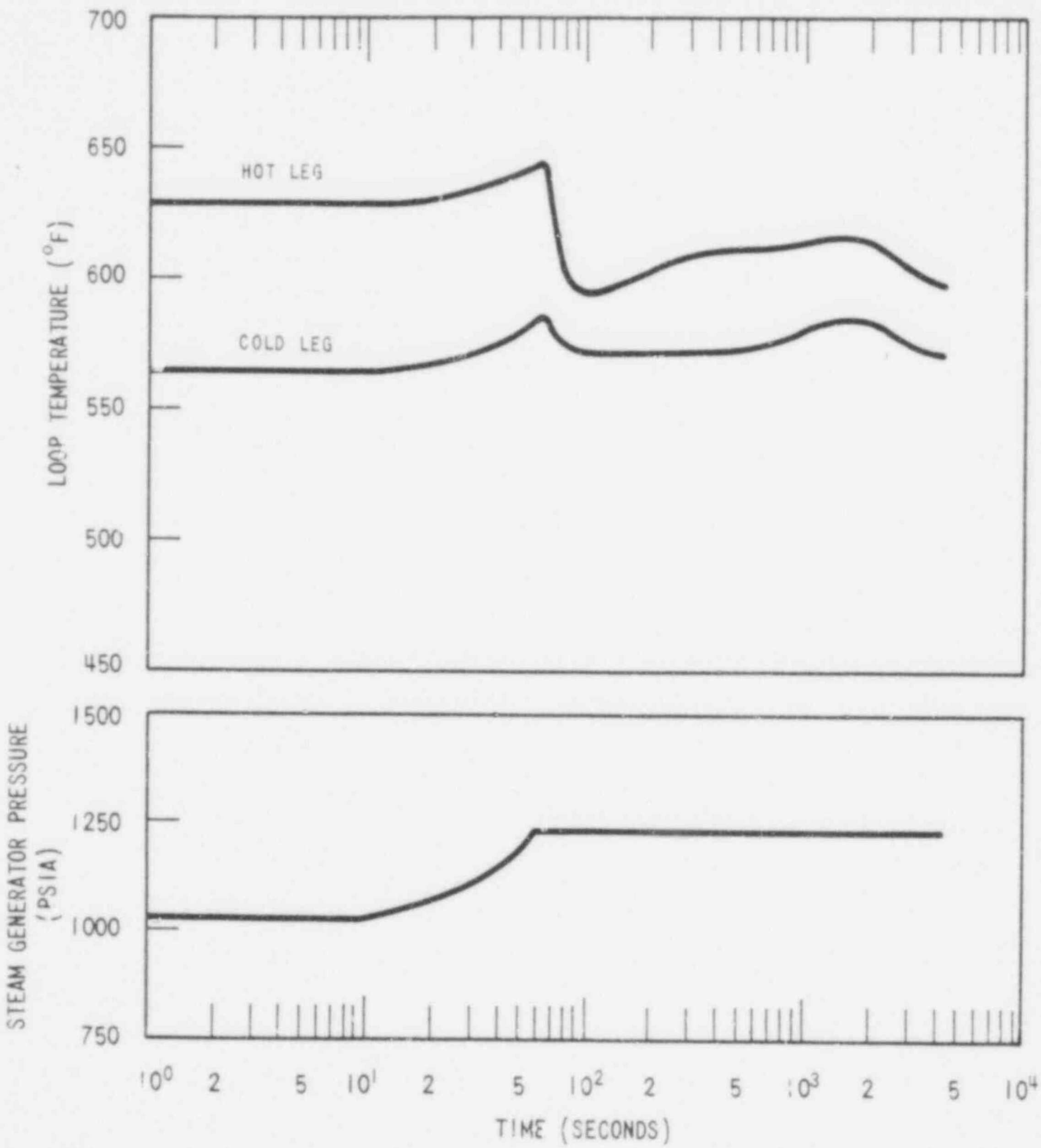

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Figure 15.2.6-2. BLUE

Core Average Temperature Transient and Steam Generator Pressure for Loss of Offsite Power

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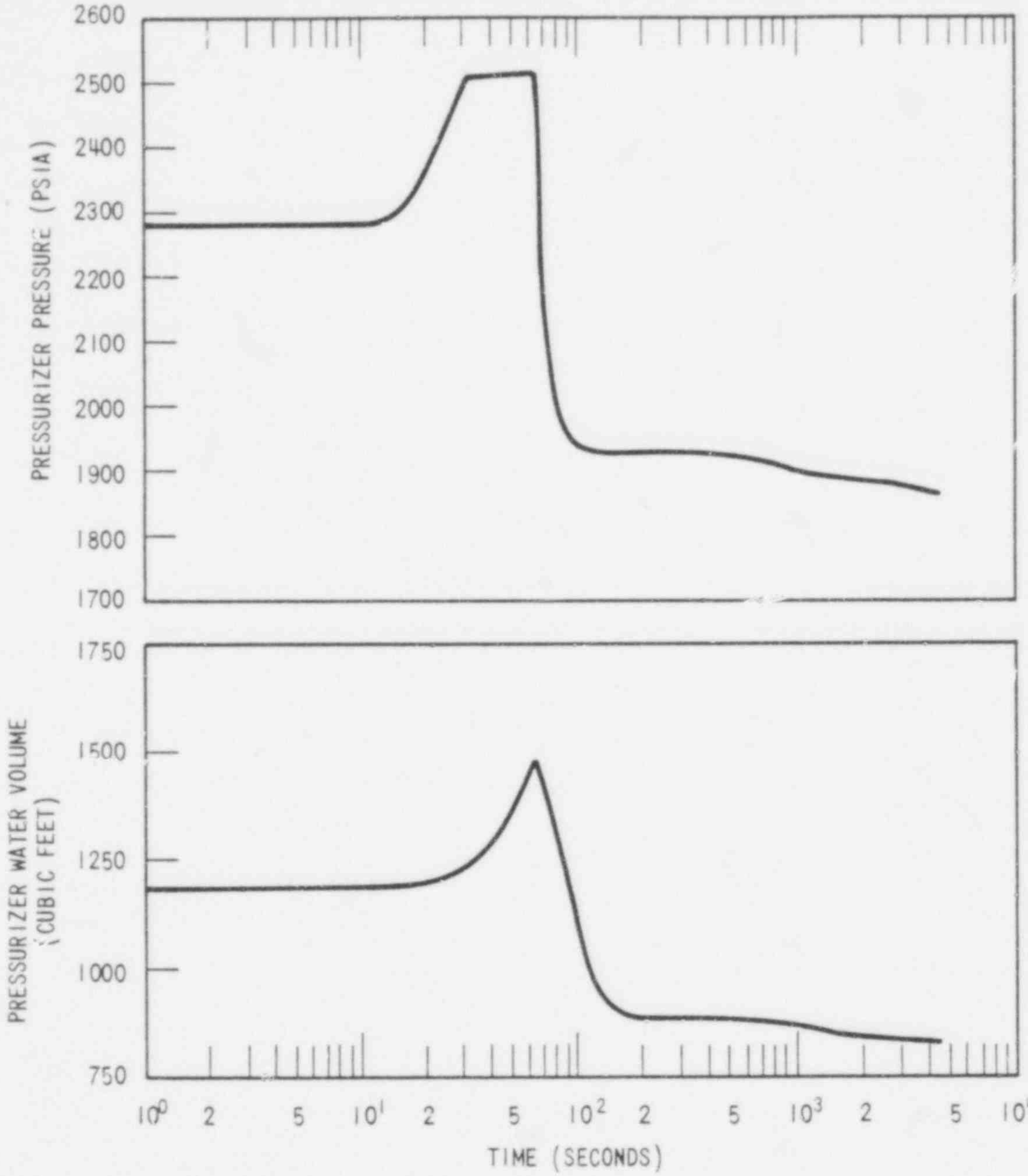

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Figure 15.2.7-1.

Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater

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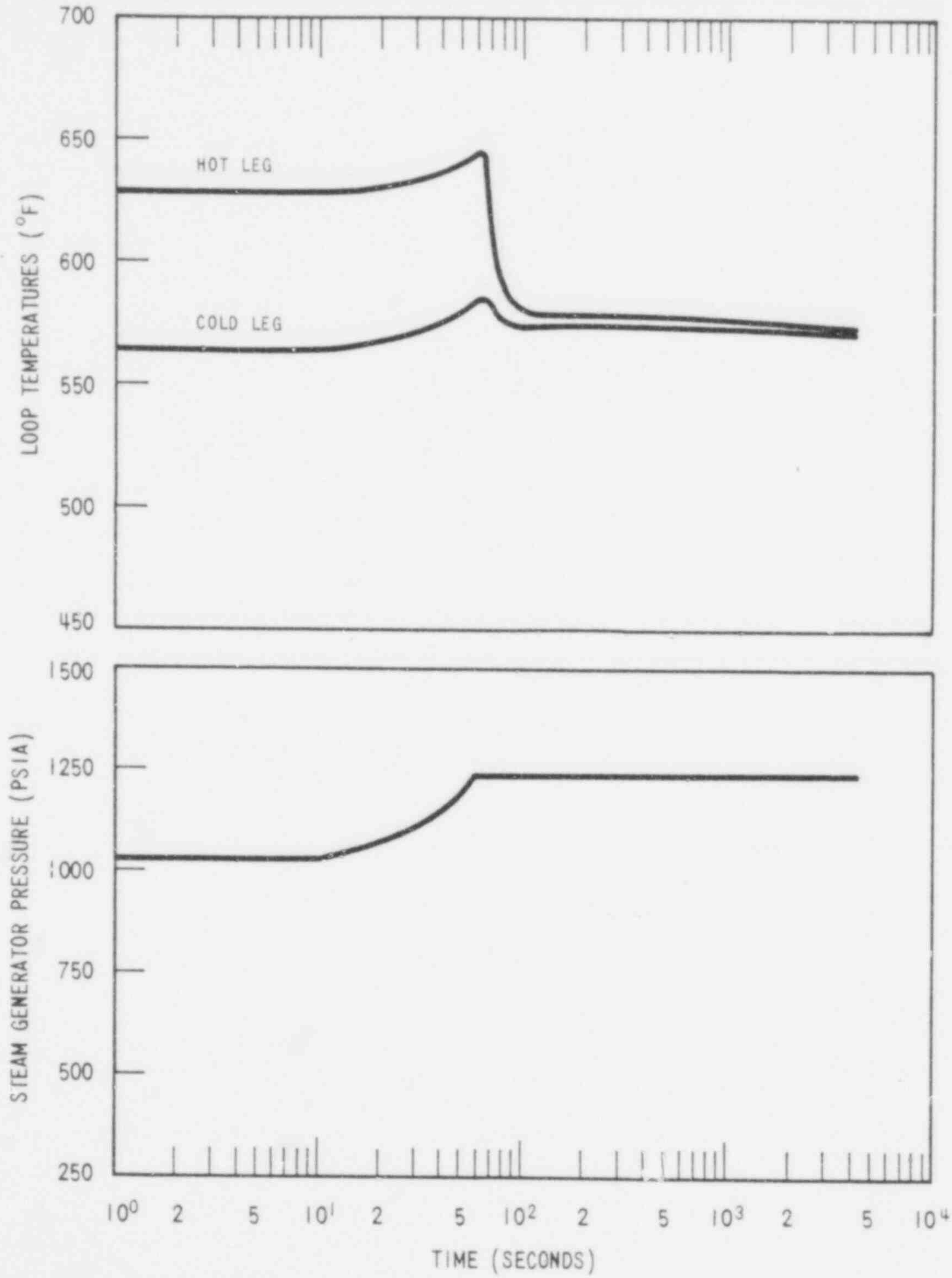

WCAP – 9500
BLUE Figure 15.2.7-2. Loop Temperatures and Steam Generator Pressure for Loss of Normal Feedwater

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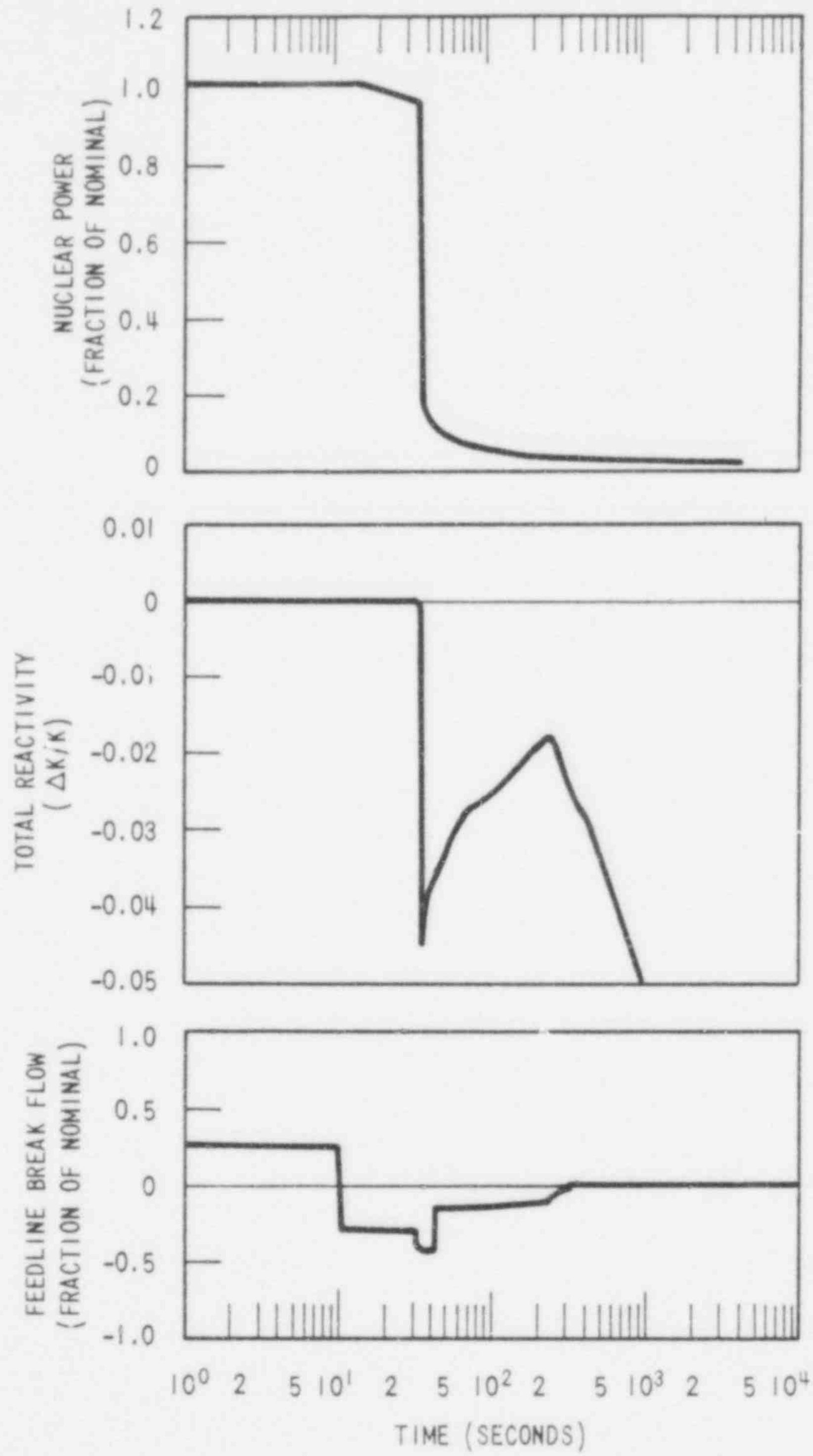

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Figure 15.2.8-1. BLUE Nuclear Power Transient, Total Core Reactivity Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available

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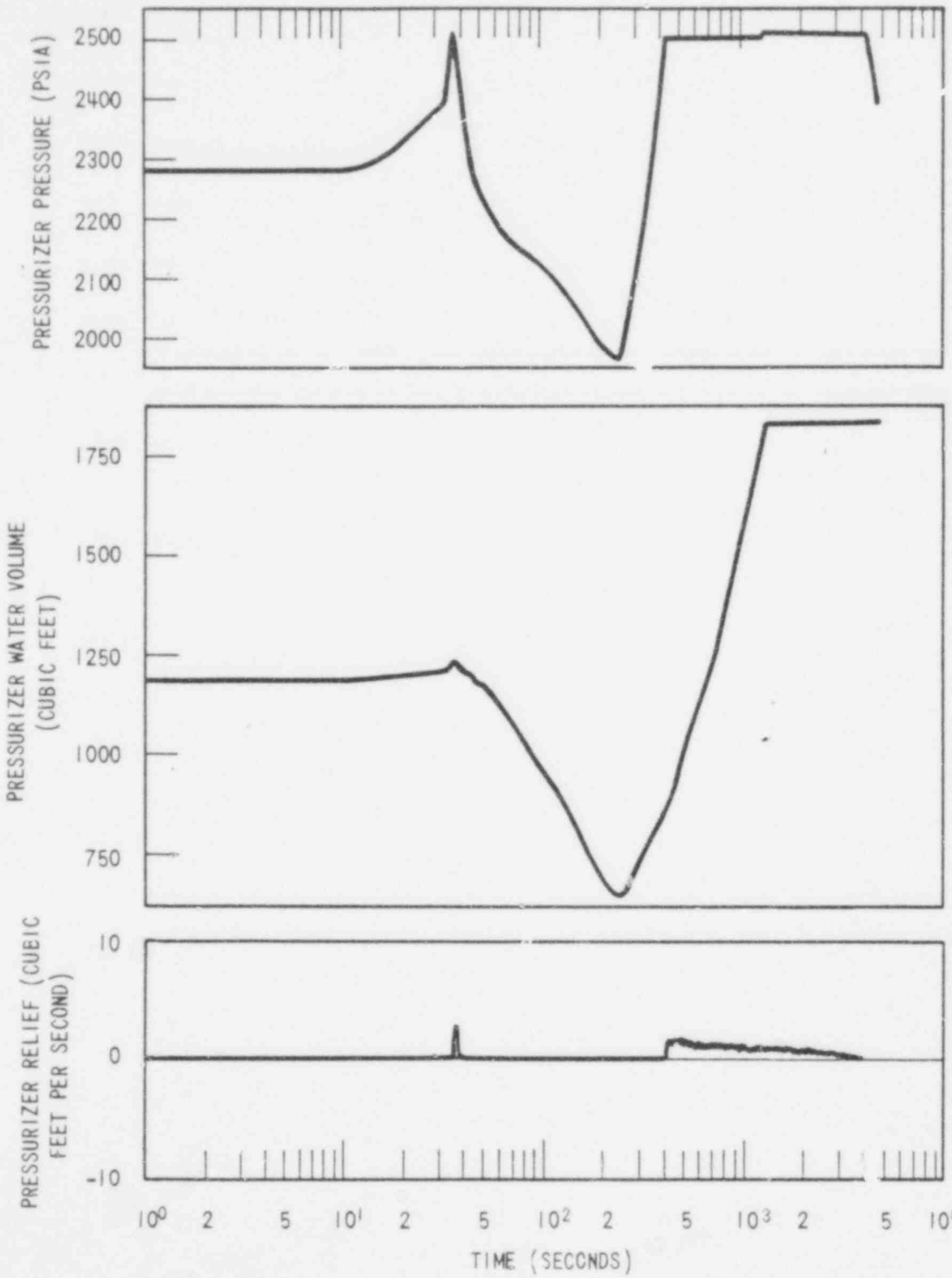
14,395-252



WCAP – 9500
BLUE Figure 15.2.8-2. Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available

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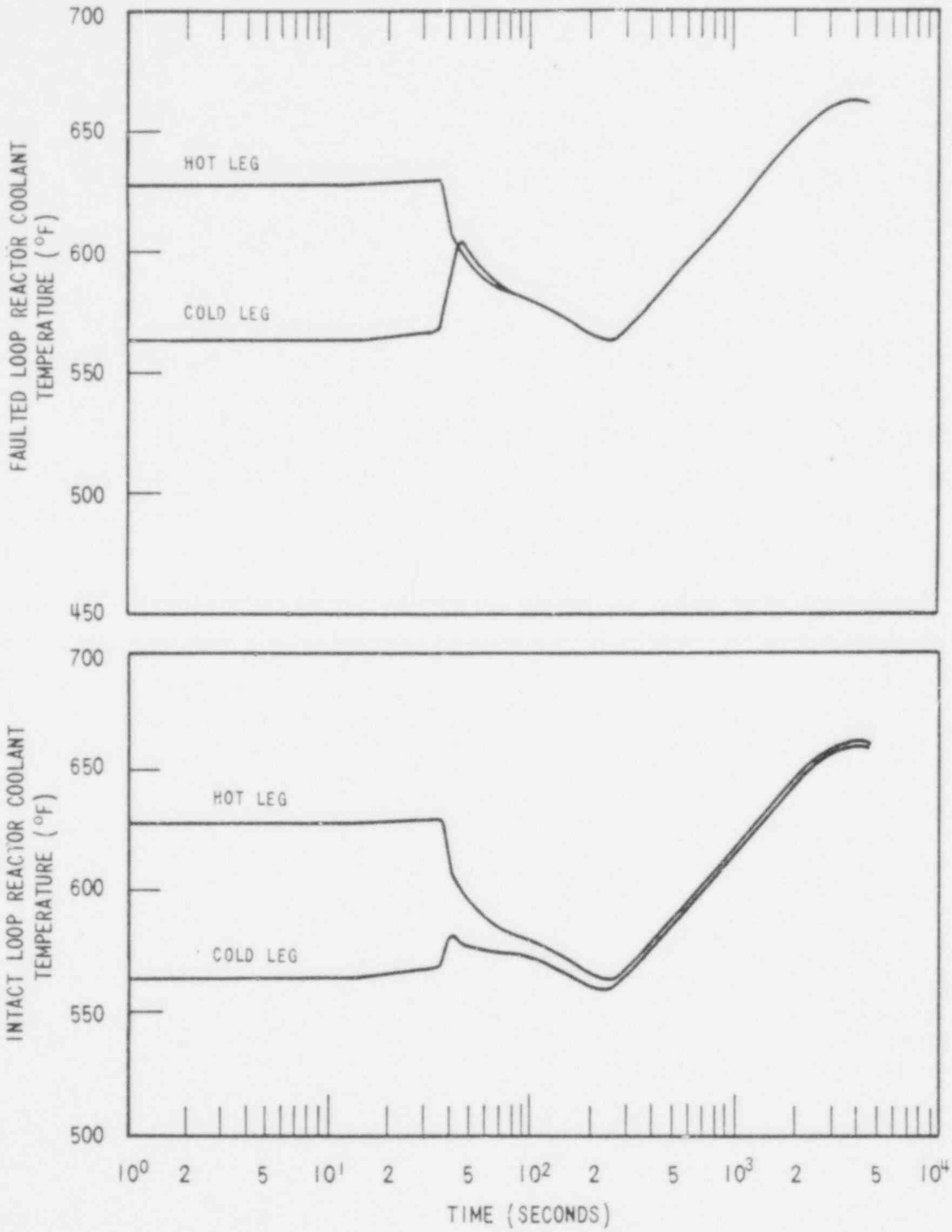

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Figure 15.2.8-3. BLUE

Reactor Coolant Temperature Transients for the Faulted and the Intact Loops for Main Feedline Rupture With Offsite Power Available

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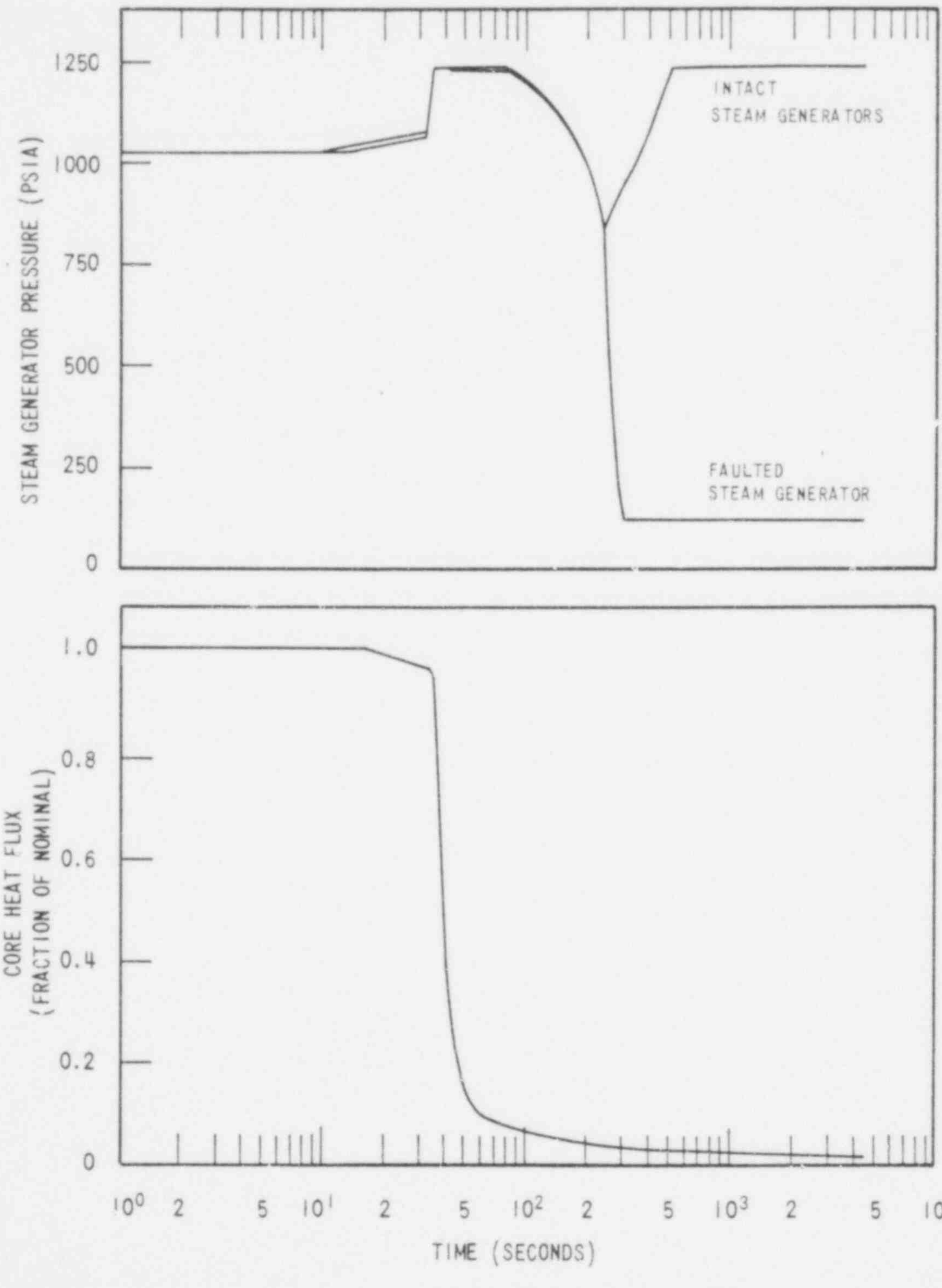

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Figure 15.2.8-4.

Steam Generator Pressure and Core Heat Flux Transients for Main Feedline Rupture With Offsite Power Available

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14,395-255

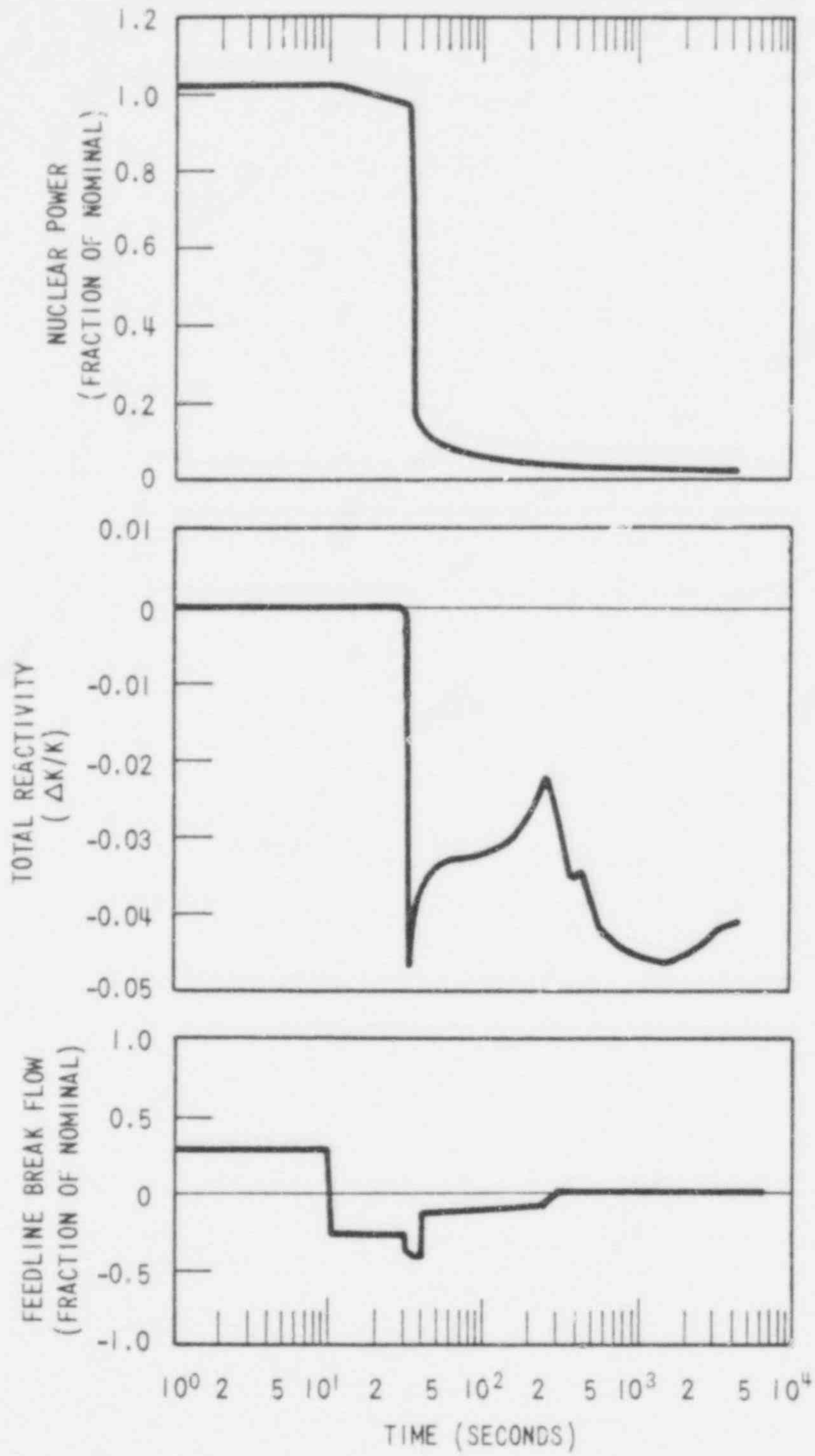

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Figure 15.2.8-5.

Nuclear Power Transient, Total Core Reactivity Transient, and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power Available

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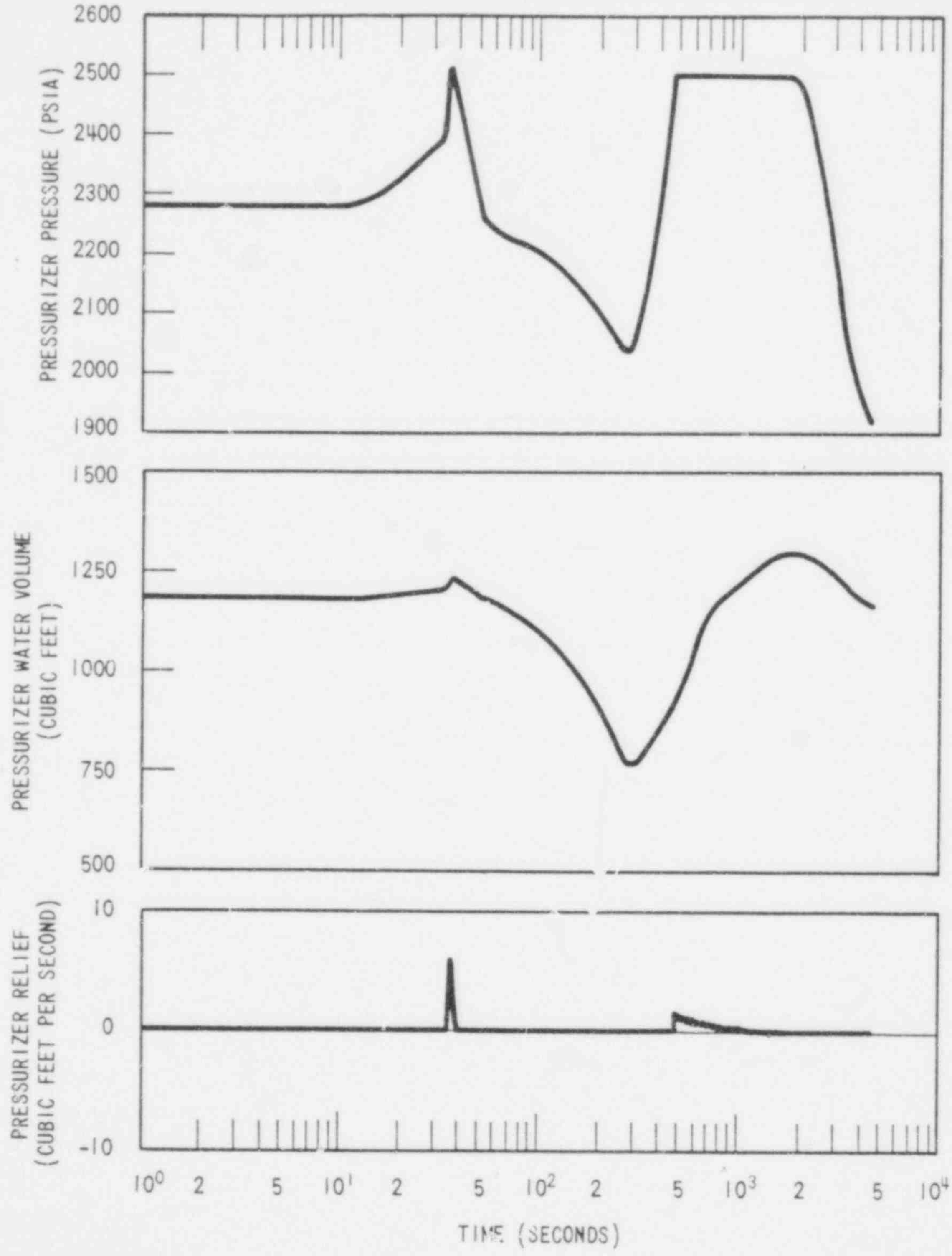

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Figure 15.2.8-6.

Pressurizer Pressure, Water Volume, and Relief Rate for Main Feedline Rupture Without Offsite Power Available

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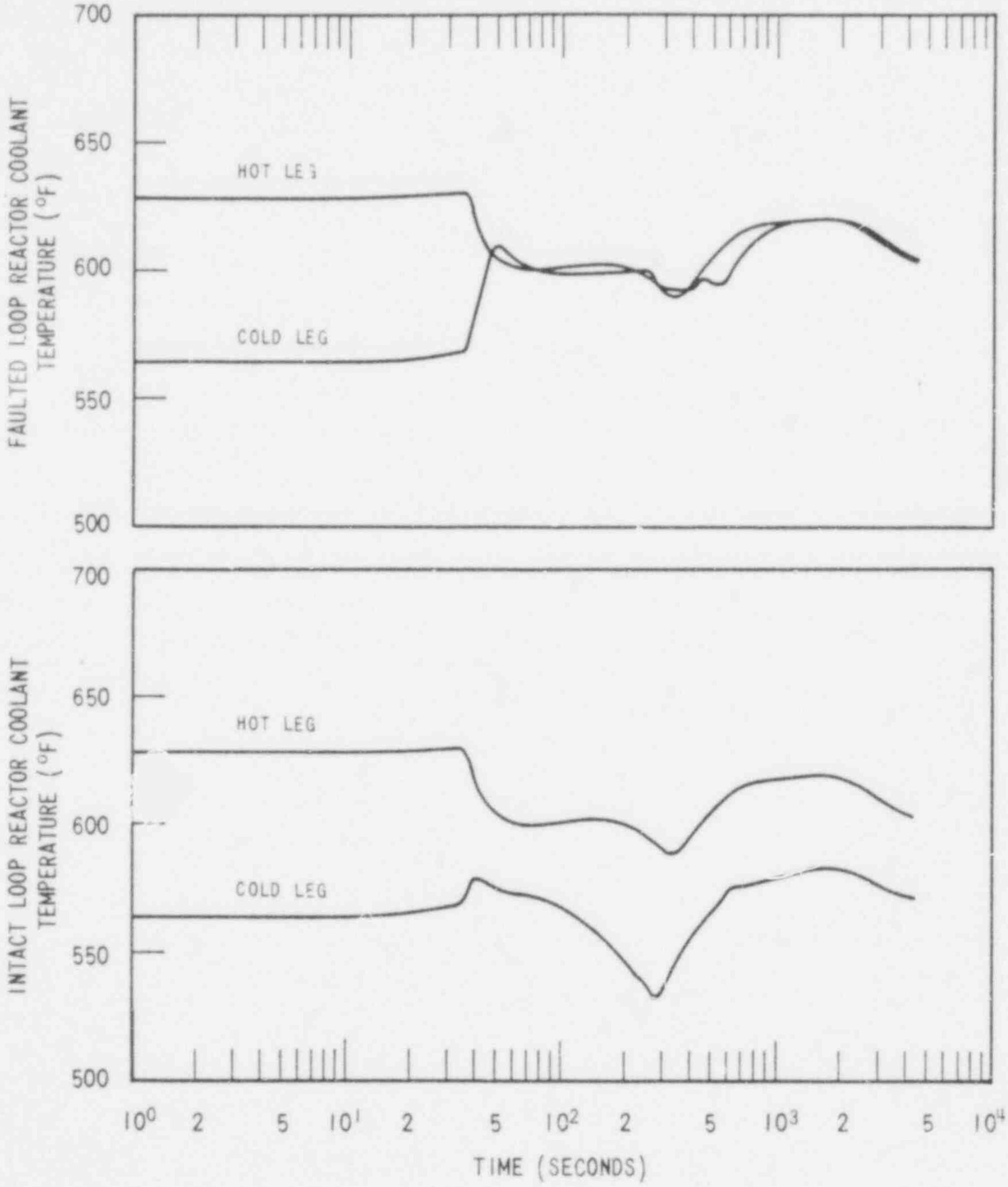

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Figure 15.2.8-7.

Reactor Coolant Temperature Transients for the Faulted and Intact Loops for Main Feedline Rupture Without Offsite Power Available

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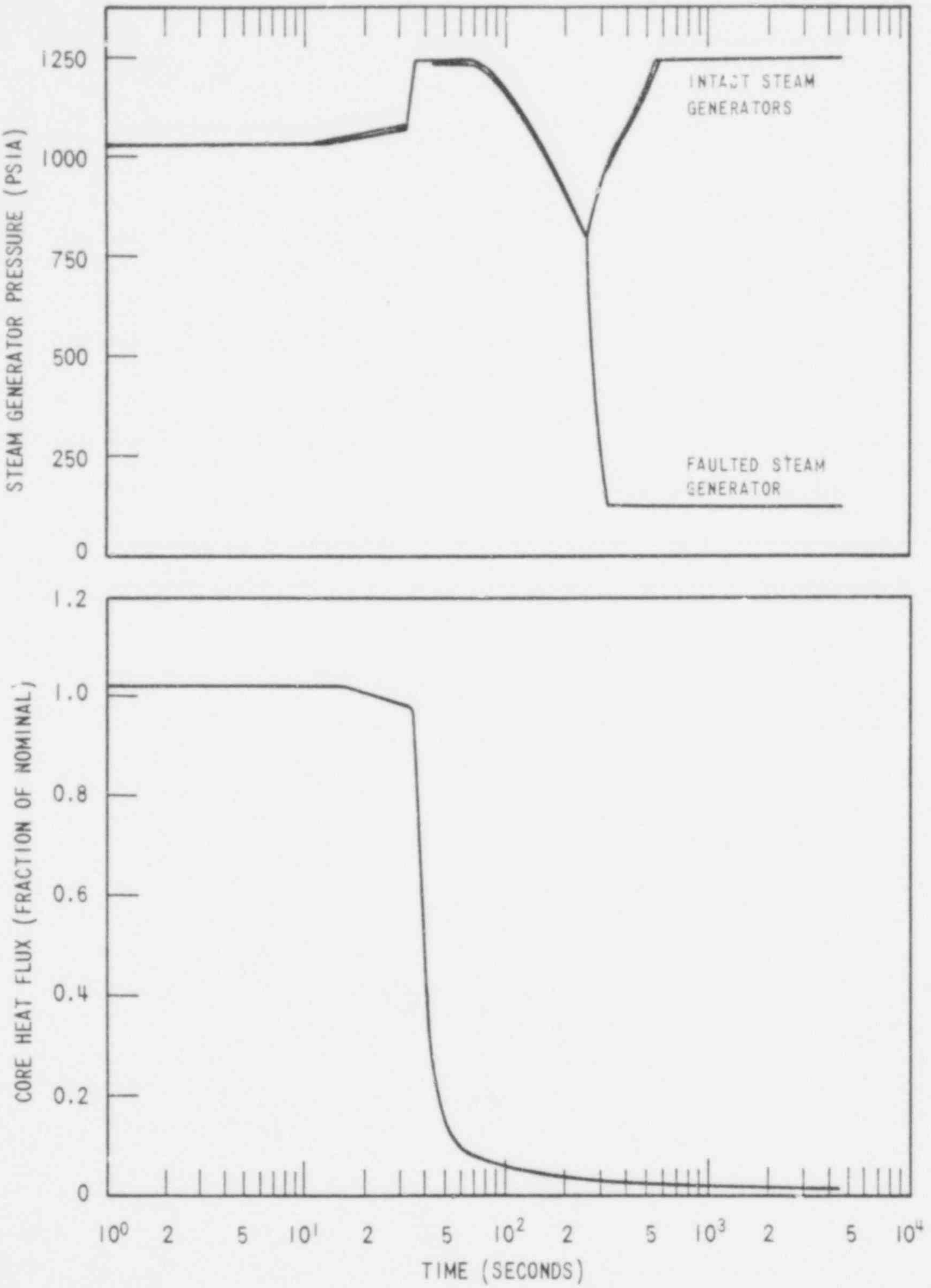

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Figure 15.2.8-8.

Steam Generator Pressure and Core Heat Flux Transients for Main Feedline Rupture Without Offsite Power Available

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15.3

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15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

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

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

1. Partial Loss of Forced Reactor Coolant Flow
2. Complete Loss of Forced Reactor Coolant Flow
3. Reactor Coolant Pump Shaft Seizure (Locked Rotor)
4. Reactor Coolant Pump Shaft Break

Item 1 above is considered to be an ANS Condition II event, item 2 an ANS Condition III event, and items 3 and 4 ANS Condition IV events. Section 15.0.1 contains a discussion of ANS classifications.

15.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Identification of Causes and Accident Description

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

Normal power for the pumps is supplied through individual buses connected to the generator and the offsite power system. When a generator trip occurs, the buses continue to be supplied from external power lines, and the pumps continue to supply coolant to the core.

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

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8 (Refer to Table 7.2.1-2 for a discussion of permissives), low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

Two cases have been analyzed:

1. Loss of one pump with four loops in operation.
2. Loss of one pump with three 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 time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. 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 (see Section 4.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

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This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3.

Initial Conditions

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 WCAP 8567.

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0.4-1). This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent power of 0.016 δk.

The least negative moderator temperature coefficient (see Figure 15.0.3-2) is assumed since this results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

Flow Coastdown

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

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

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Results

Figures 15.3.1-1 through 15.3.1-4 show the transient response for the loss of one reactor coolant pump with four loops in operation. Figure 15.3.1-4 shows the DNBR to be always greater than the limit value.

Figures 15.3.1-5 through 15.3.1-8 show the transient response for the loss of one reactor coolant pump with three loops in operation. The minimum DNBR is greater than the limit value, as shown in Figure 15.3.1-8.

For both cases analyzed, 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 tables for the two cases analyzed are shown on Table 15.3.1-1. The affected reactor coolant pump will continue to coast down, 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.

15.3.1.3 Radiological Consequences

A partial loss of reactor coolant flow from full load would result in a reactor and turbine trip. Assuming, that the condenser is not available, atmospheric steam dump may be required.

The radiological consequences resulting from atmospheric steam dump would be less severe than the steamline break event analyzed in Subsection 15.1.5.3 since fuel damage as a result of this transient is not postulated.

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15.3.1.4 Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

The radiological consequences of this event would be less than the steamline break event analyzed in Subsection 15.1.5.3.

15.3.2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.2.1 Identification of Causes and Accident Description

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

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator, and the offsite power system. Each pump is on a separate bus. When a generator trip occurs the buses continue to be supplied from external power lines and the pumps continue to supply coolant flow to the core.

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

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

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference [3] provides analyses of grid frequency disturbances and the resulting Nuclear Steam Supply System protection requirements which are generally applicable.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

15.3.2.2 Analysis of Effects and Consequences

Two cases have been analyzed:

1. Loss of four pumps with four loops in operation.

2. Loss of three pumps with three 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 time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. 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 (see Section 4.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

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

Results

Figures 15.3.2-1 through 15.3.2-4 show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3.2-4 shows the DNBR to be always greater than the limit value.

Figures 15.3.2-5 through 15.3.2-8 show the transient response for the loss of power to all reactor coolant pumps with three loops in operation. The reactor is again assumed to be tripped on undervoltage signal. The minimum DNBR is greater than the limit value, as shown in Figure 15.3.2-8.

For both cases analyzed, 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 two cases analyzed is shown on Table 15.3.1-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Section 15.2.6. With the reactor tripped, a stable plant condition would be attained. Normal plant shutdown may then proceed.

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15.3.2.3 Radiological Consequences

A complete loss of reactor coolant flow from full load results in a reactor and turbine trip. Assuming, that the condenser is not available, atmospheric steam dump would be required. The quantity of steam released would be the same as for a loss of offsite power.

Since fuel damage is not postulated, the radiological consequences resulting from atmospheric steam dump would be less severe than the steamline break analyzed in Subsection 15.1.5.

15.3.2.4 Conclusions

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

15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

15.3.3.1 Identification of Causes and Accident Description

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

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

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

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

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

Two cases are analyzed:

1. Four loops operating, one locked rotor.

2. Three loops operating, one locked rotor.

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most adverse steady

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state operating condition, i.e., maximum guaranteed steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature. Plant characteristics and initial conditions are further discussed in Section 15.0.3. With three loops operating, the maximum power level (including errors) allowed in that mode of operation is assumed.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 30 psi above nominal pressure (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 responses shown in Figures 15.3.3-2 and 15.3.3-6 are the responses at the point in the Reactor Coolant System having the maximum pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reached 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are full open at 2575 psia and their capacity for steam relief is as described in Section 5.4.

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 "not 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 3.0 times the average rod power (i.e., $F_Q = 3.0$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based 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 clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000

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BTU/hr-ft^2-^oF at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1800^oF (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left[\frac{-45,500}{1.986\ T}\right]$$

where:

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, ^oF

The reaction heat is 1510 cal/gm.

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

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

<u>Locked Rotor with Four Loops Operating</u>

The transient results for this case are shown in Figures 15.3.3-1 through 15.3.3-4. The results of these calculations are also summarized in Table 15.3.3-1. The peak Reactor Coolant System 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 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

Locked Rotor with Three Loops Operating

The transient results for this case are shown in Figure 15.3-3-5 through 15.3.3-6. The peak Reactor Coolant System pressure is slightly lower than for the previous case, but is still less than that which would cause stresses to exceed the faulted condition stress limits. The clad temperature transient is less severe than for the previous case.

The calculated sequence of events for the two cases analyzed is shown on Table 15.3.1-1. Figures 15.3.3-1 and 15.3.3-5 show the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The radiological consequences of a locked rotor accident will be analyzed on a plant specific basis. Westinghouse input to the assumptions to be used to perform the radiological evaluation are summarized in Table 15.3.3-2.

15.3.3.4 Conclusions

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

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2. Since the peak cald surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F the core will remain in place and intact with no loss of core cooling capability.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Identification of Casues and Accidents Description

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

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

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

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15.3.4.2 Radiological Consequences

The radiological consequences for a reactor coolant pump shaft break event would be similar to those from the locked rotor incident (Subsection 15.3.3).

15.3.4.3 Conclusions

The consequences of a reactor coolant pump shaft break are not greater than those calculated for the locked rotor accident (see Section 15.3.3). With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor conservatively assumed that DNB occurs at the beginning of the transient.

15.3.5 REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907, June 1972

2. Hargrove, H. G., "FACTRAN - A Fortran - IV Code for Thermal Transients in a UO_2 Fuel Rod", WCAP-7908, June 1972.

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

TABLE 15.3.1-1 (Page 1)

Time Sequence of Events for Incidents Which Result in a Decrease in Reactor Coolant System Flow

Accident	Event	Time (sec.)
Partial Loss of Forced Reactor Coolant Flow		
1. Four loops operating one pump coasting down	Coastdown begins	0.
	Low flow reactor trip	1.43
	Rods begin to drop	2.43
	Minimum DNBR occurs	3.80
2. Three loops operating, one pump coasting down	Coastdown begins	0.
	Low flow reactor trip	2.51
	Rods begin to drop	3.51
	Minimum DNBR occurs	4.70

Complete Loss of Forced Reactor Coolant Flow

Event	Four Loop operation	Three Loop operation
All operating pumps lose power and being coasting down	0.	0.

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TABLE 15.3.1-1 (Page 2)

Time Sequence of Events for Incidents Which Result in a Decrease in Reactor Coolant System Flow

Accident	Event	Time (sec.) Four Loop operation	Three Loop operation
	Reactor coolant pump undervoltage trip point reached	0.	0.
	Rods begin to drop	1.5	1.5
	Minimum DNBR occurs	3.8	3.7
Reactor Coolant Pump Shaft Seizure (Locked Rotor)			
	Rotor on one pump locks	0.	0.
	Low flow trip point reached	.03	.05
	Rods begin to drop	1.03	1.05
	Maximum RCS pressure occurs	4.0	4.8
	Maximum clad temperature occurs	3.7	4.1

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TABLE 15.3.3-1

Summary of Results for Locked Rotor Transients

	4 Loops Operating Initially	3 Loops Operating Initially
Maximum Reactor Coolant System Pressure (psia)	2570	2564
Maximum Clad Temperature (^{o}F) Core Hot Spot	2200	2331
Zr-H_2O reaction at core hot spont (% by weight)	1.4%	1.9%

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TABLE 15.3.3-2

Assumptions to be used for the Radiological Consequences of the Locked Rotor Accident

	EXPECTED	DESIGN
Power	3565	3565
Fraction of Fuel with Defects	0.0012*	0.01
Reactor Coolant Acitivity Prior to Accident	ANSI-N237	See SAR (Plant Specific)
Total Steam Generator Tube Leak Rate During Accident and Initial 8 Hours	0.009 gpm	1 gpm**
Activity Released to Reactor Coolant from Failed Fuel		
Noble Gas	None of core inventory	9% of gap inventory
Iodine	None of core inventory	9% of gap inventory
Iodine Partition Factor Prior to the Accident	0.1	0.1
Duration of Plant Cooldown by Secondary System After Accident, (hrs.)	8	8

* Per ANSI-N237, American National Standard Source Term Specification (March 1976).

** 0.347 gpm in defective steam generator and 0.218 gpm per non-defective steam generator during accident.

TABLE 15.3.3-2 (Continued)

	EXPECTED	DESIGN
Steam Release from 4 Steam Generators	***	561,979 lb (0-2 hr) 936,100 lb (2-8 hr)
Feedwater Flow to 4, Steam Generators	793,091 (0-2 hr) 1,024,438 (2-8 hr)	793,091 lb (0-2 hr) 1,024,438 lb (2-8 hr)

*** Condenser available, steam released through condenser off-gas system at 60 SCFM.

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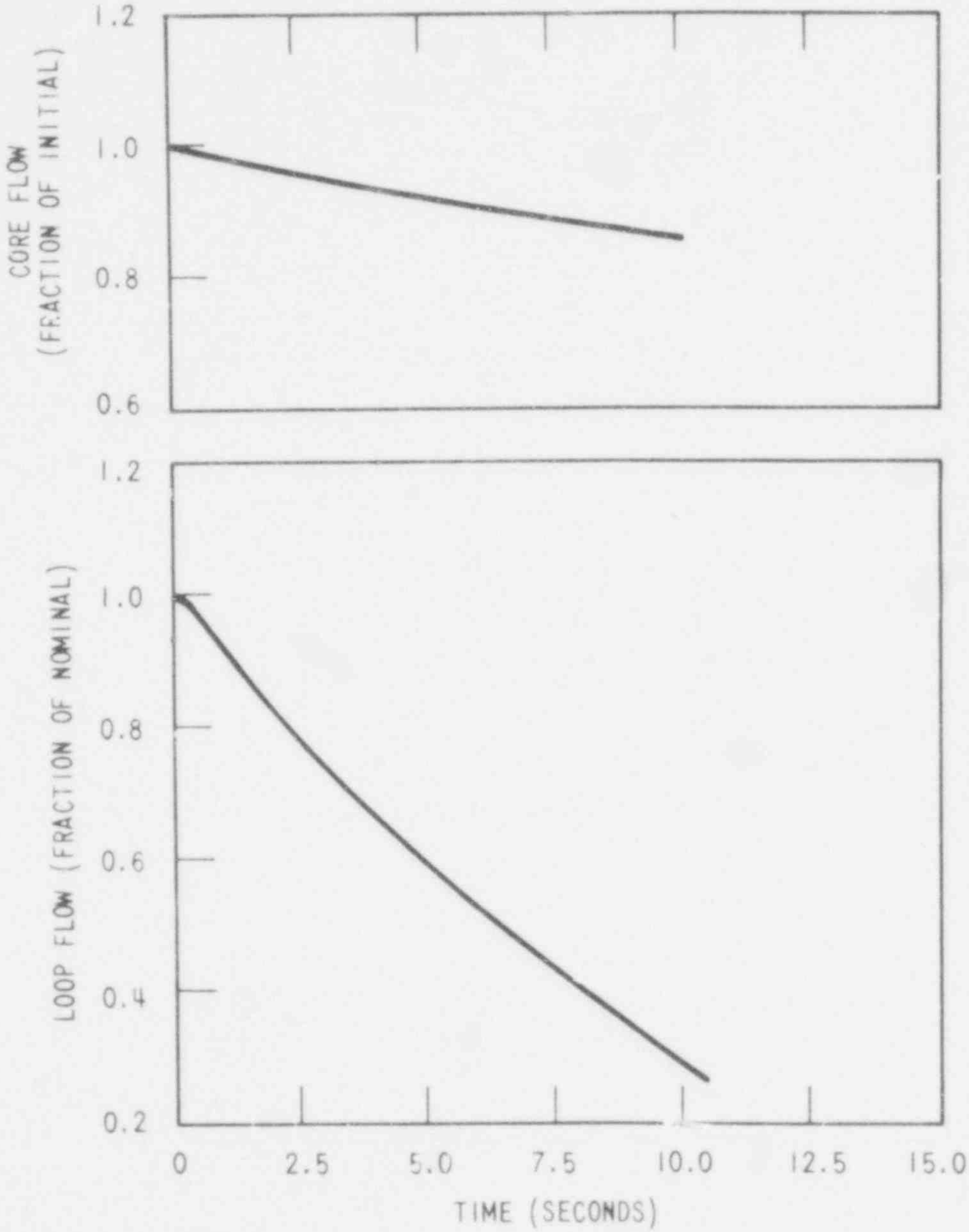

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Figure 15.3.1-1.

Flow Transients for Four Loops in Operation, One Pump Coasting Down

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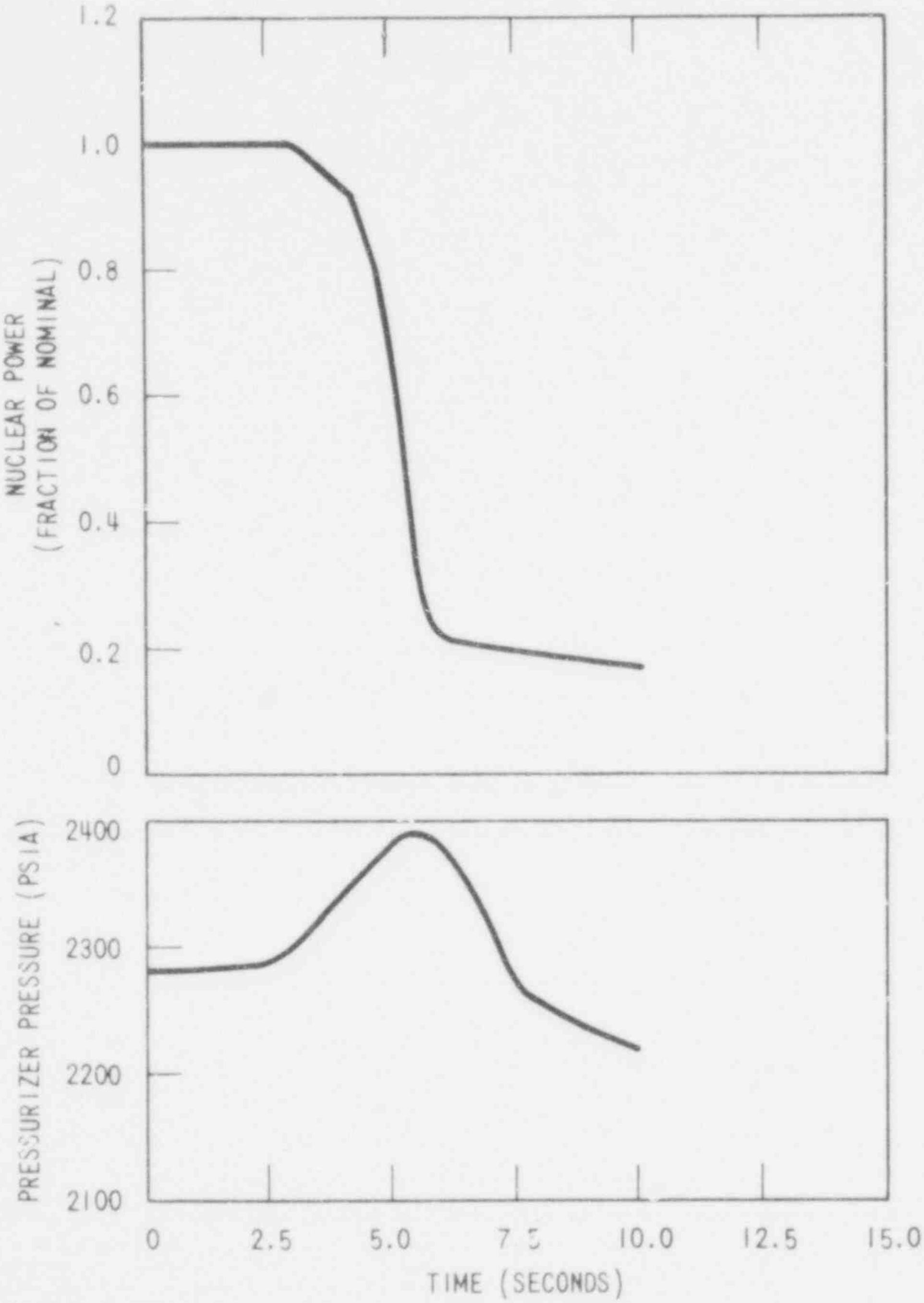

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Figure 15.3.1-2. BLUE

Nuclear Power and Pressurizer Pressure Transients for Four Loops in Operation, One Pump Coasting Down

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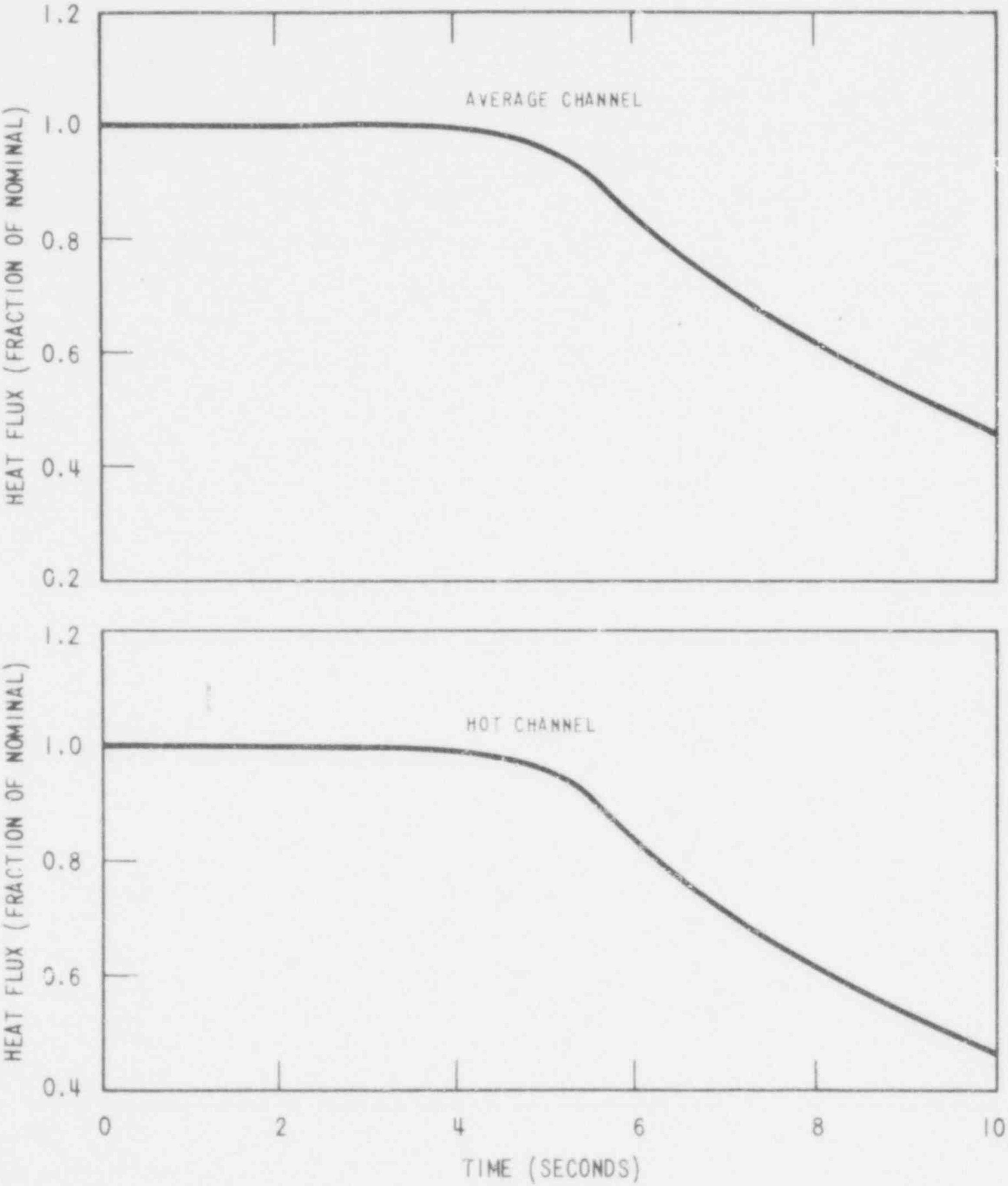

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Figure 15.3.1-3. BLUE

Average and Hot Channel Heat Flux Transients for Four Loops in Operation, One Pump Coasting Down

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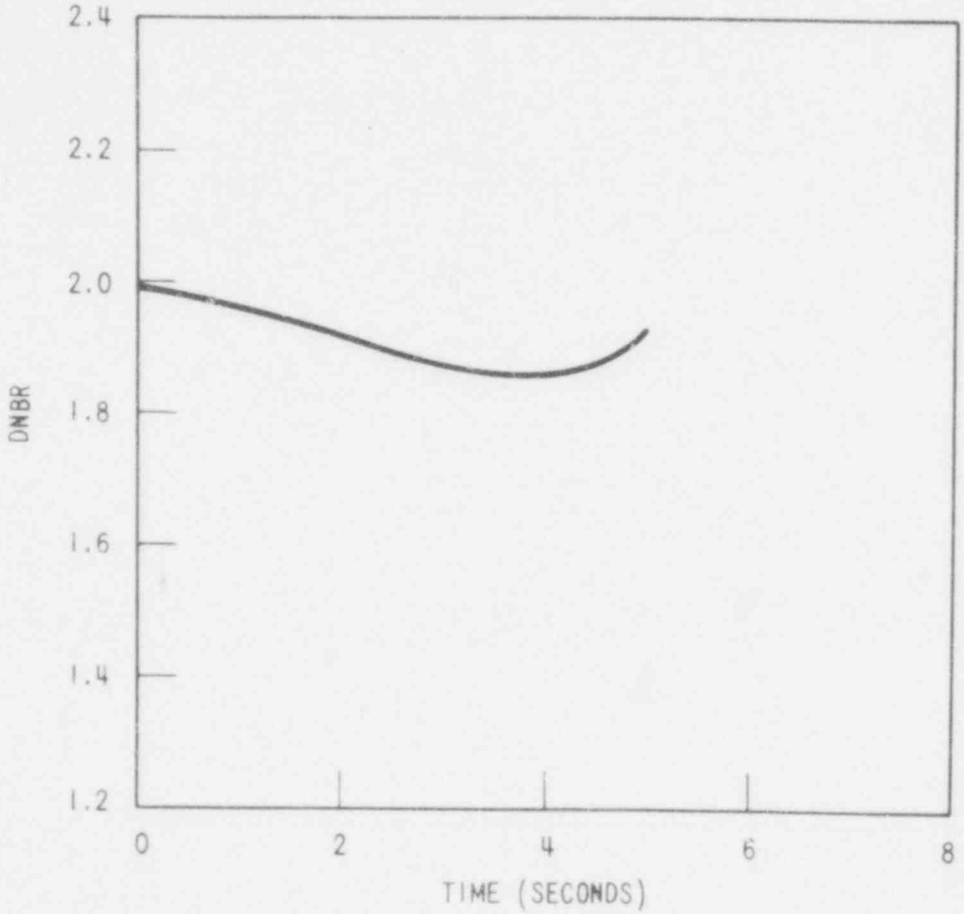

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Figure 15.3.1-4. BLUE

DNBR vs. Time for Four Loops in Operation, One Pump Coasting Down

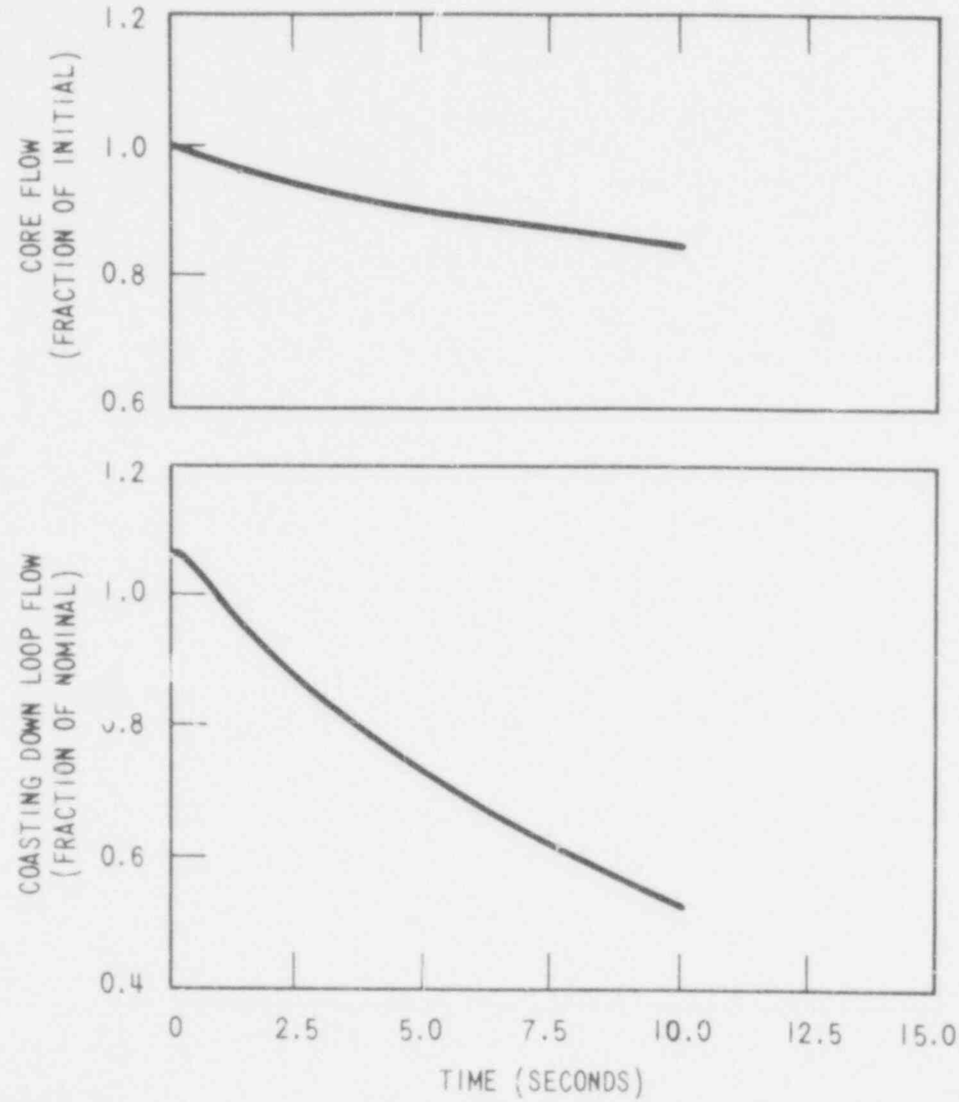

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Figure 15.3.1-5. BLUE

Flow Transients for Three Loops in Operation, One Pump Coasting Down

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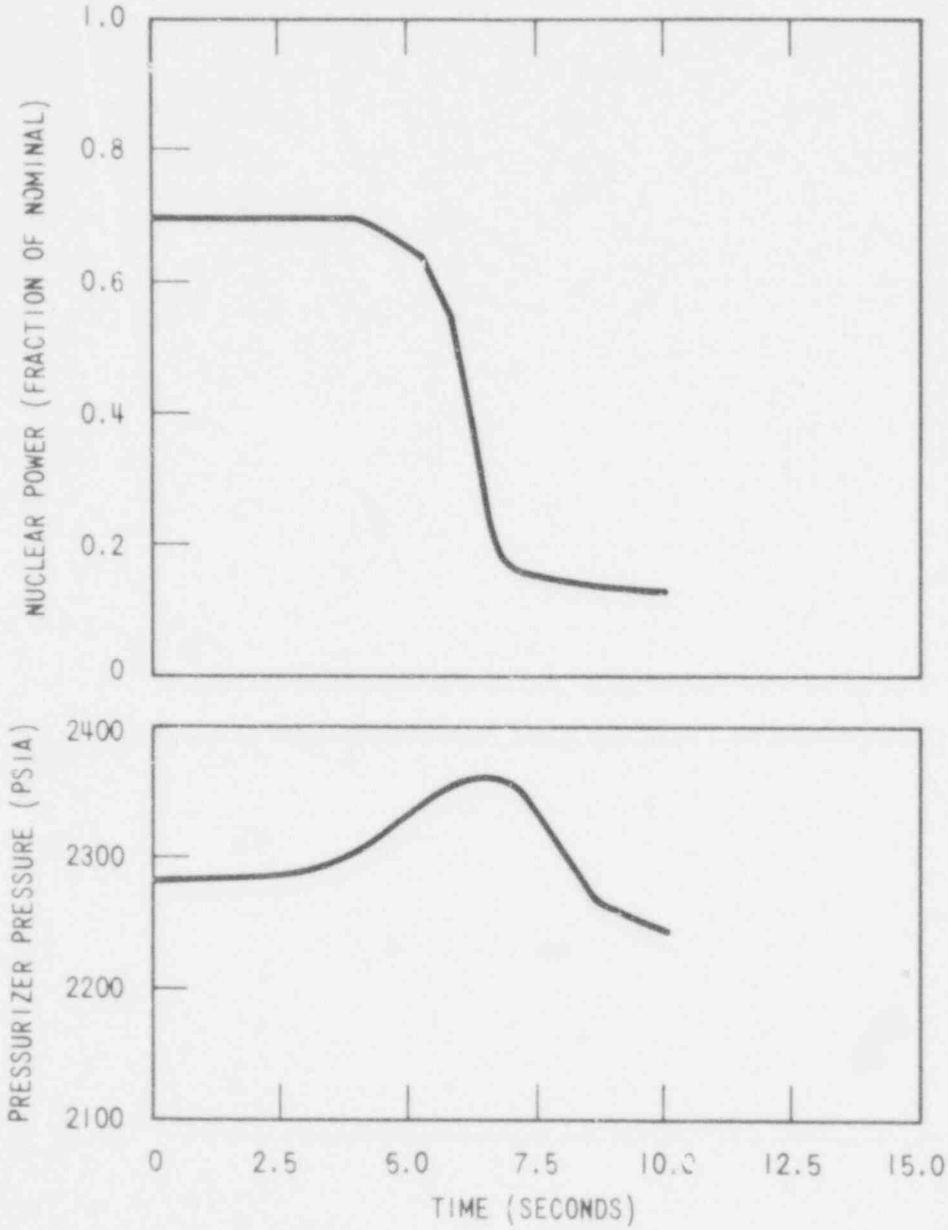

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Figure 15.3.1-6.

Nuclear Power and Pressurizer Pressure Transients for Three Loops in Operation, One Pump Coasting Down

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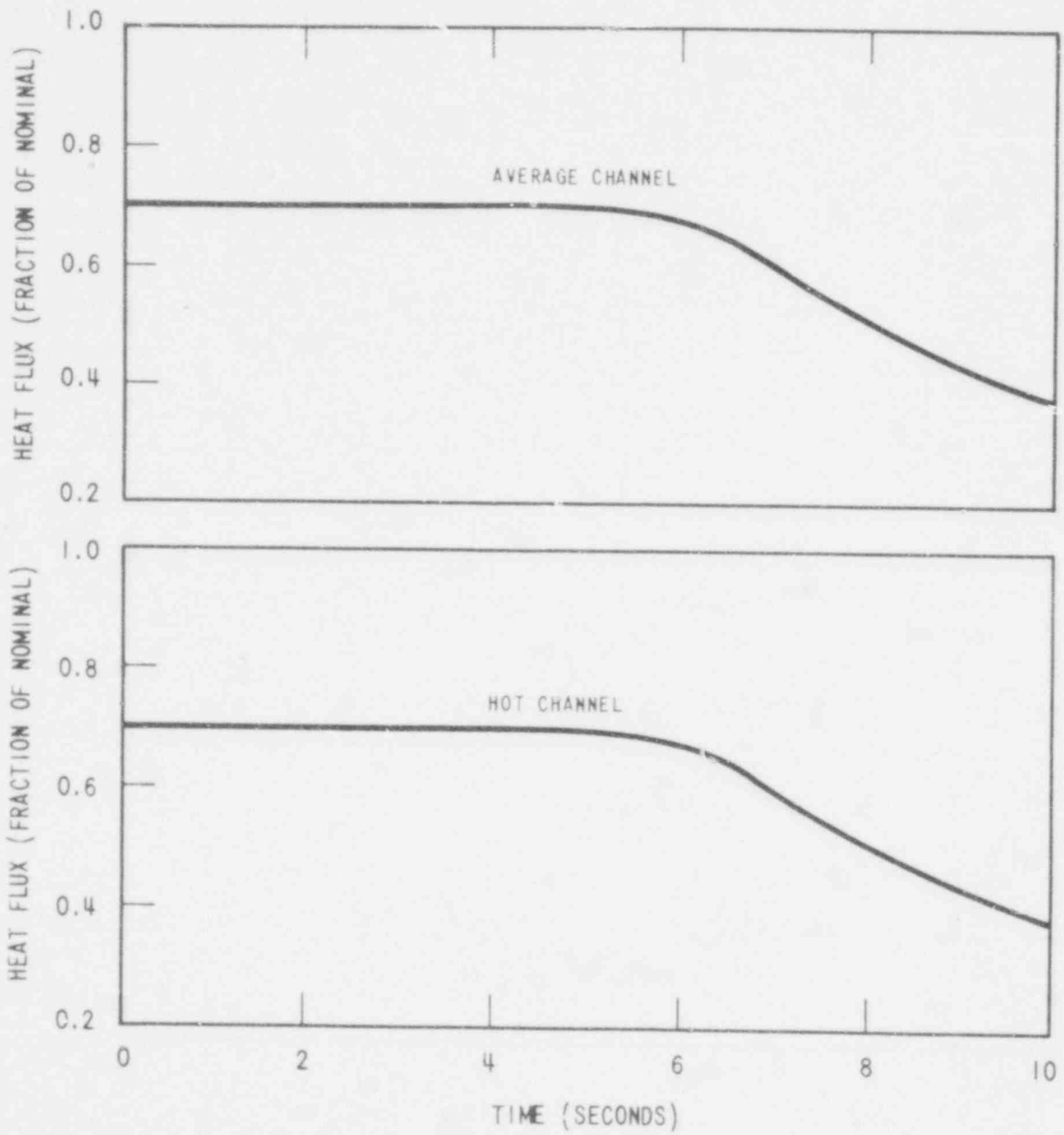

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Figure 15.3.1 7.

Average and Hot Channel Heat Flux Transients for Three Loops in Ope ·tion, One Pump Coasting Down

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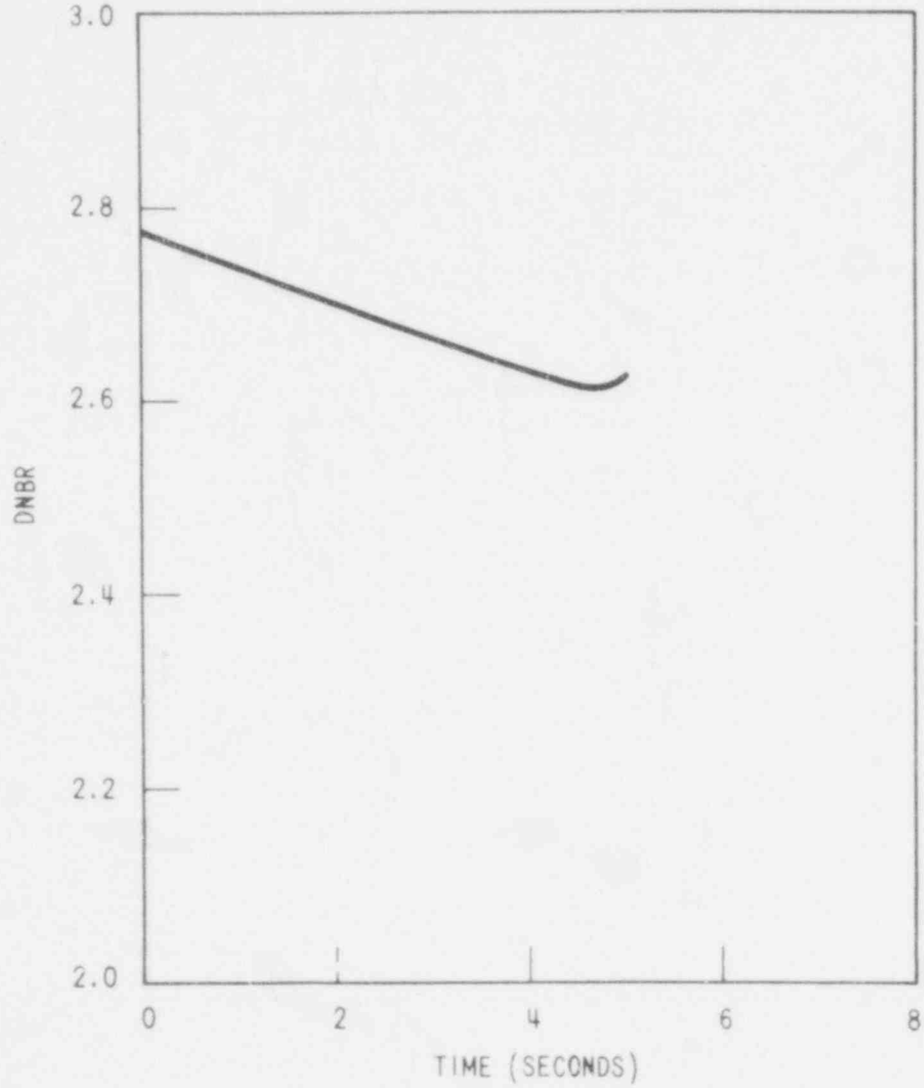

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Figure 15.3.1-8.

DNBR vs. Time for Three Loops in Operation, One Pump Coasting Down

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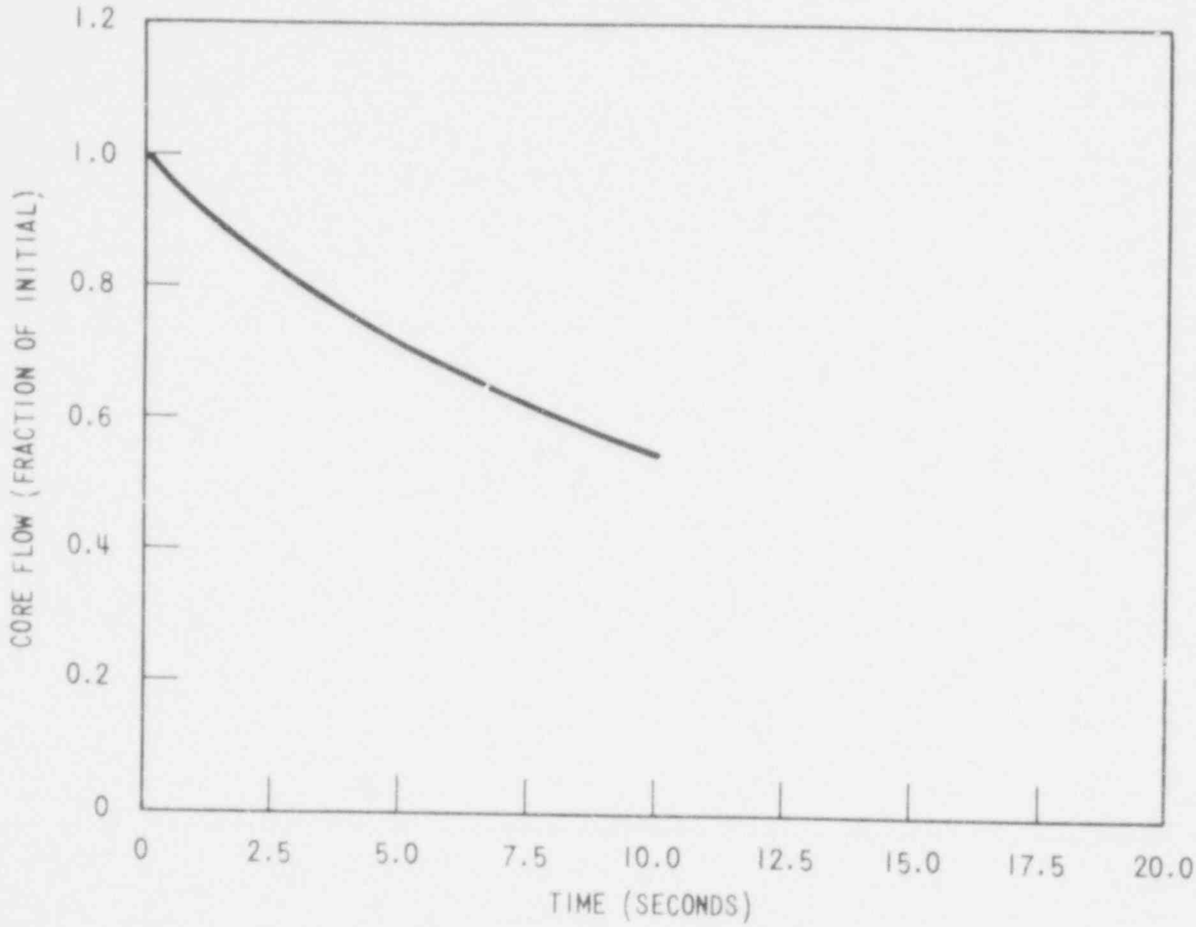

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Figure 15.3.2-1.

Core Flow Coastdown for Four Loops in Operation, Four Pumps Coasting Down

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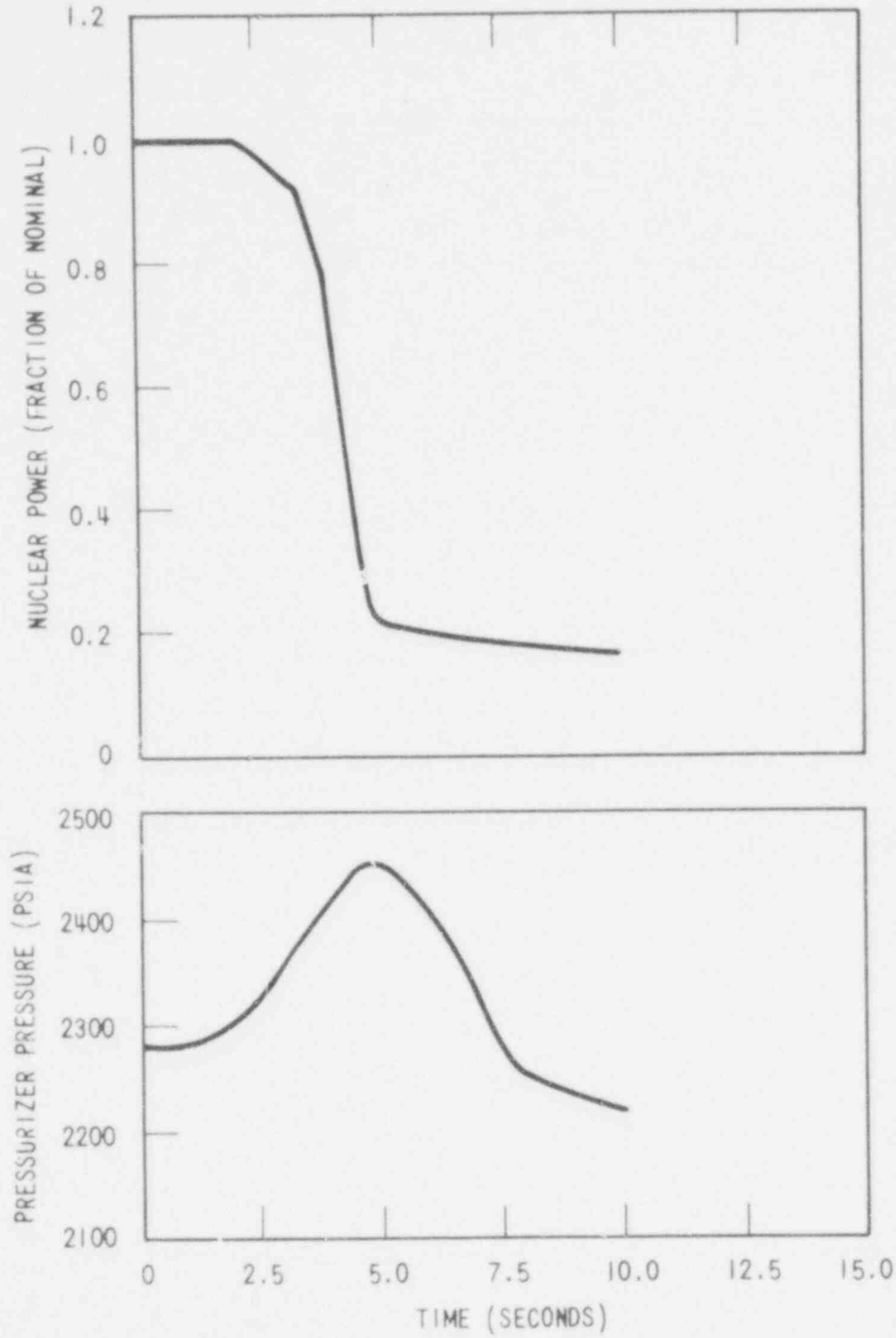

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Figure 15.3.2-2. BLUE

Nuclear Power and Pressurizer Pressure Transients for Four Loops in Operation, Four Pumps Coasting Down

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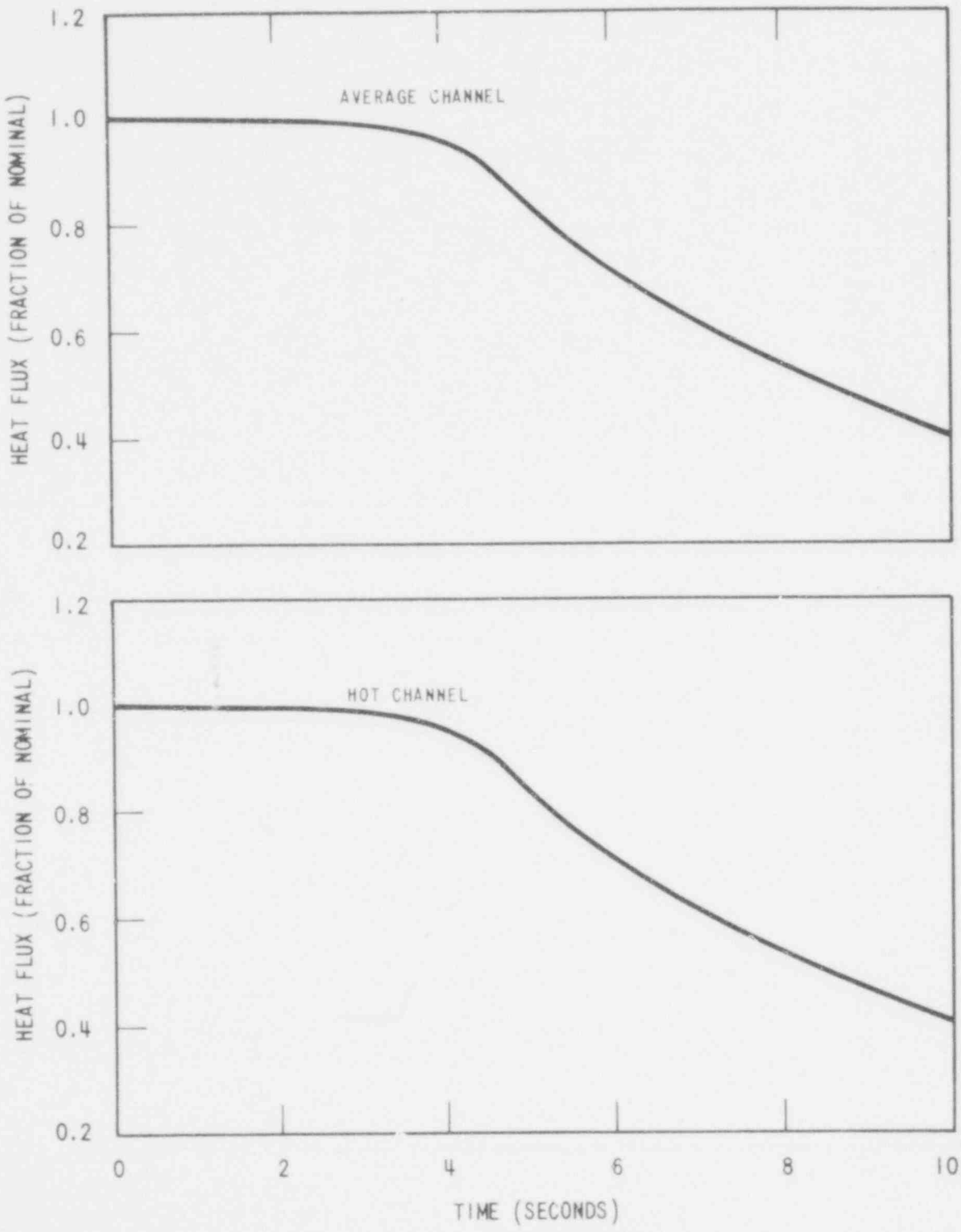

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Figure 15.3.2-3. BLUE

Average and Hot Channel Heat Flux Transients for Four Loops in Operation, Four Pumps Coasting Down

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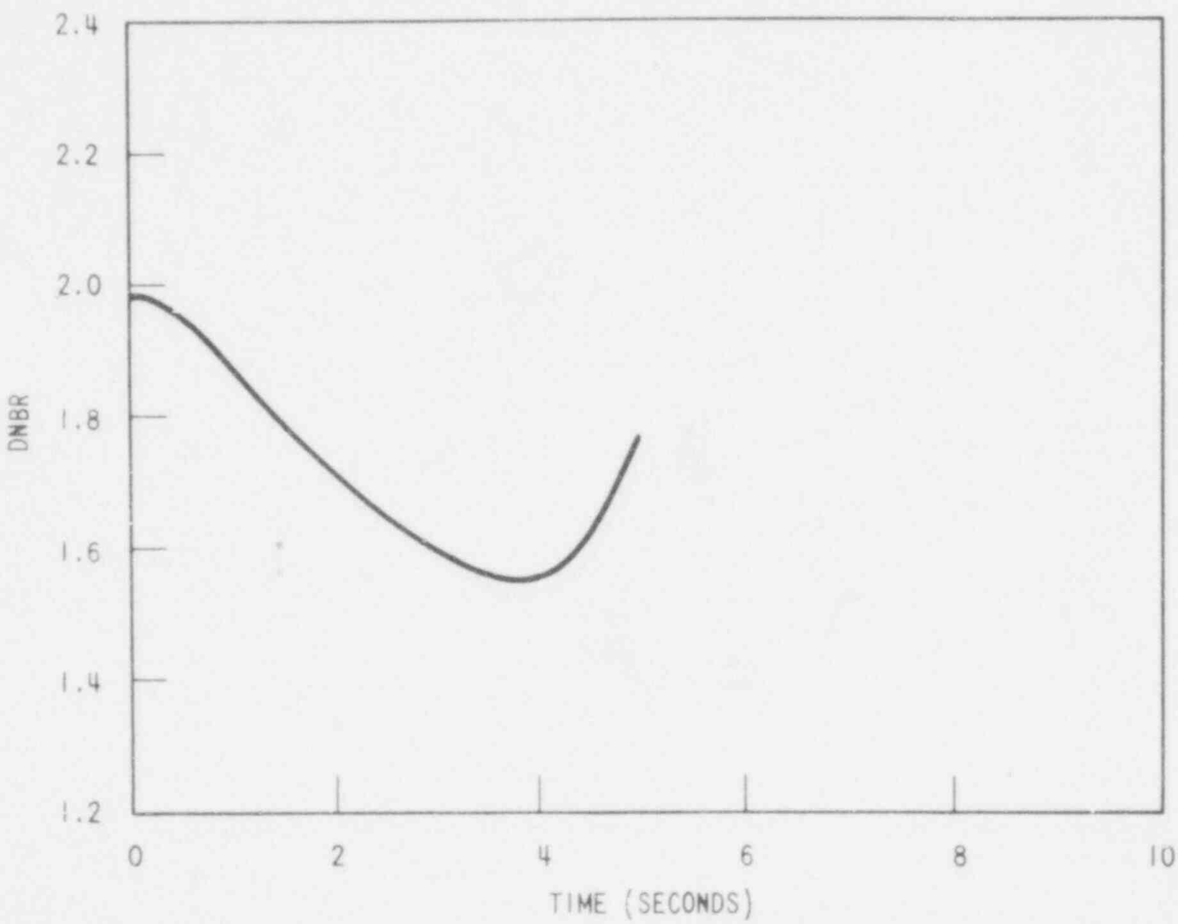

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Figure 15.3.2-4.

DNBR vs. Time for Four Loops in Operation, Four Pumps Coasting Down

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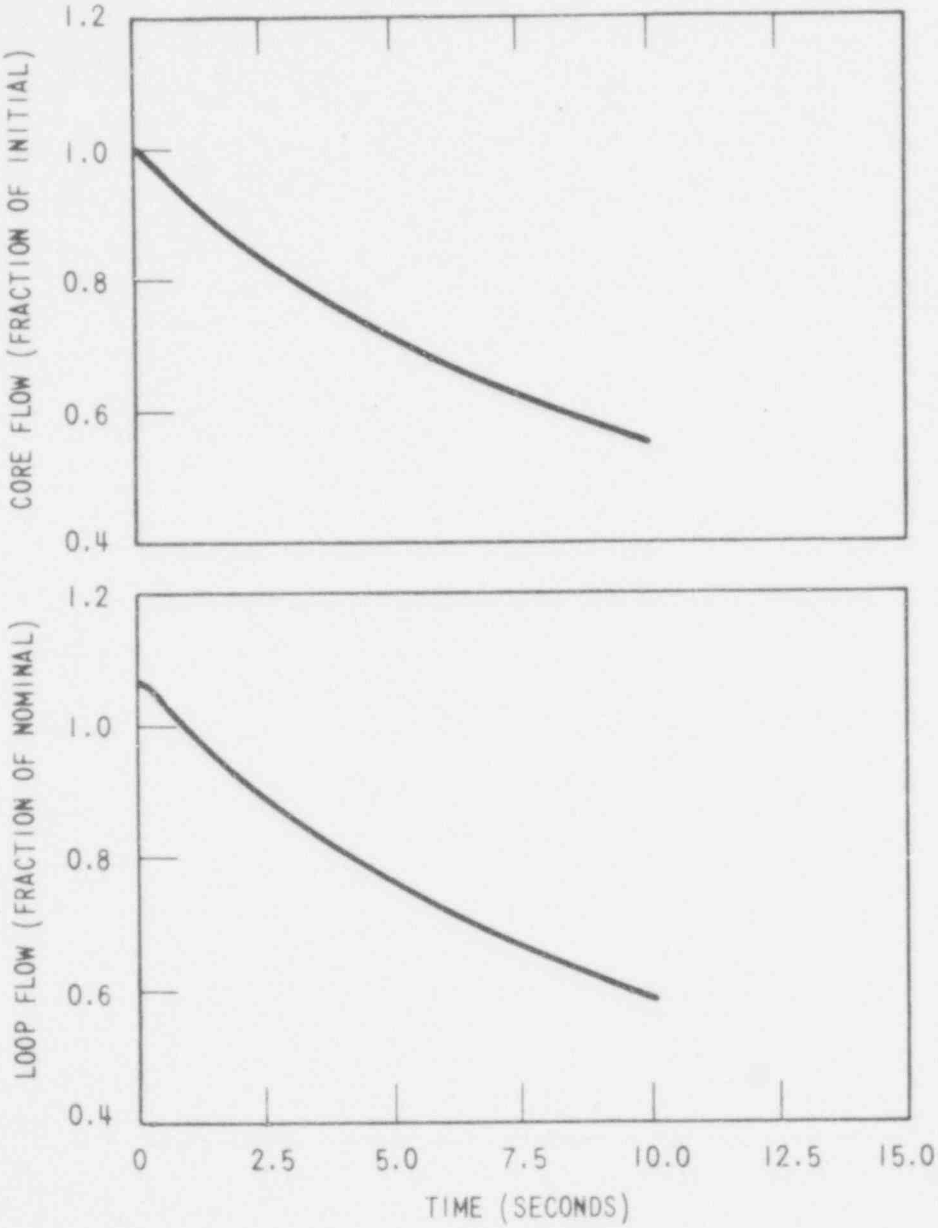

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Figure 15.3.2-5.

Flow Transients for Three Loops in Operation, Three Pumps Coasting Down

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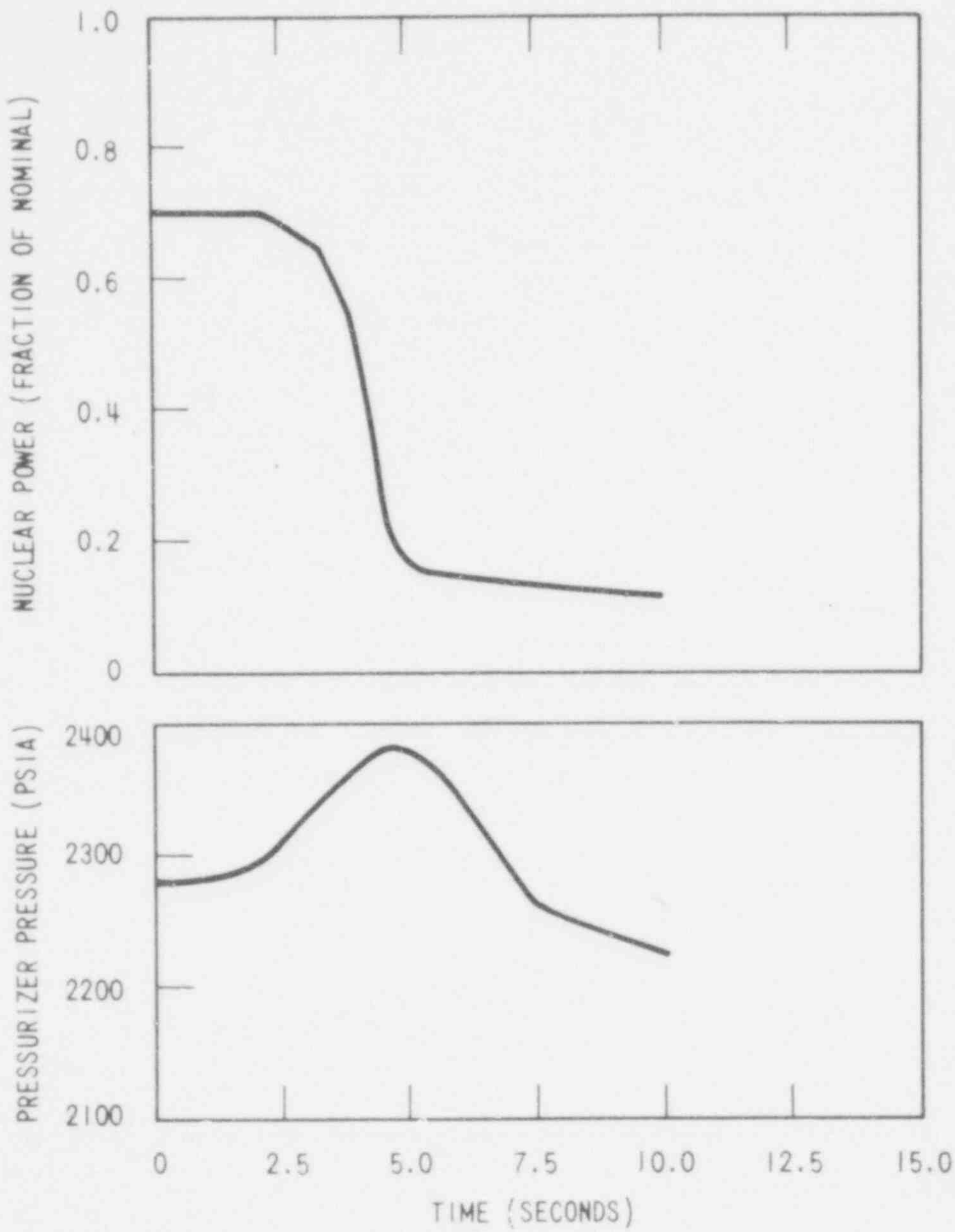

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Figure 15.3.2-6.

Nuclear Power and Pressurizer Pressure Transients for Three Loops in Operation, Three Pumps Coasting Down

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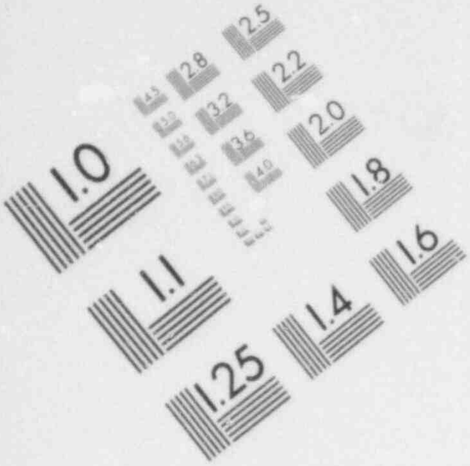

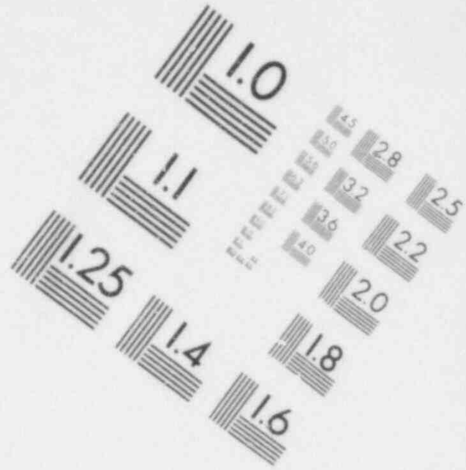

IMAGE EVALUATION
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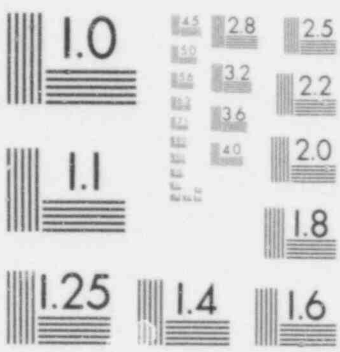

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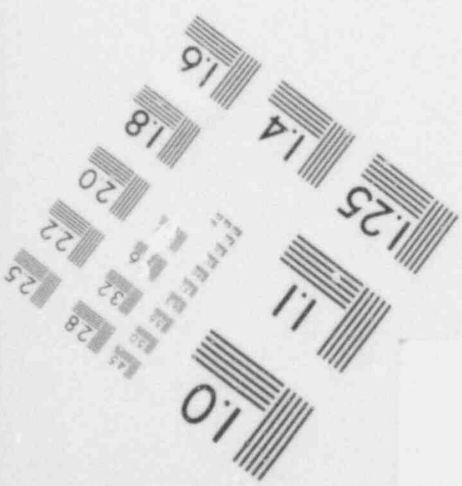

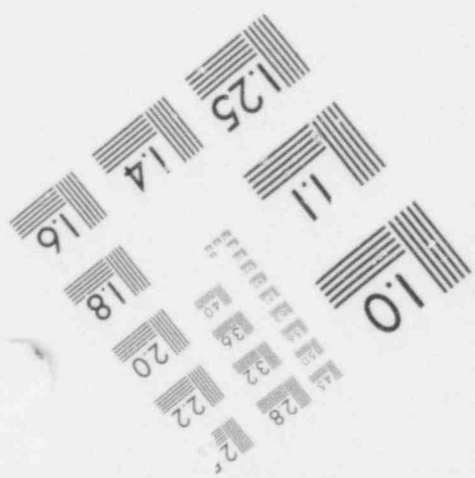

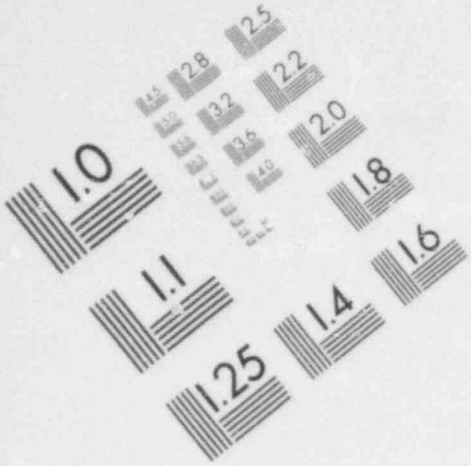

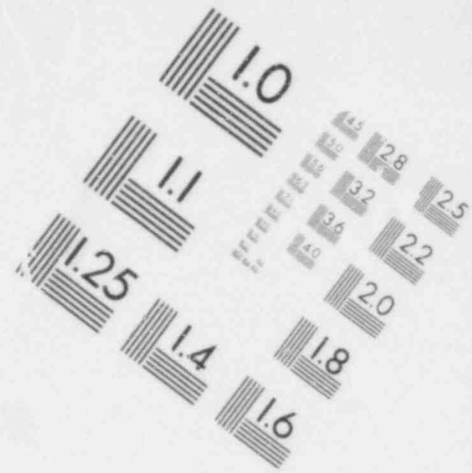

IMAGE EVALUATION
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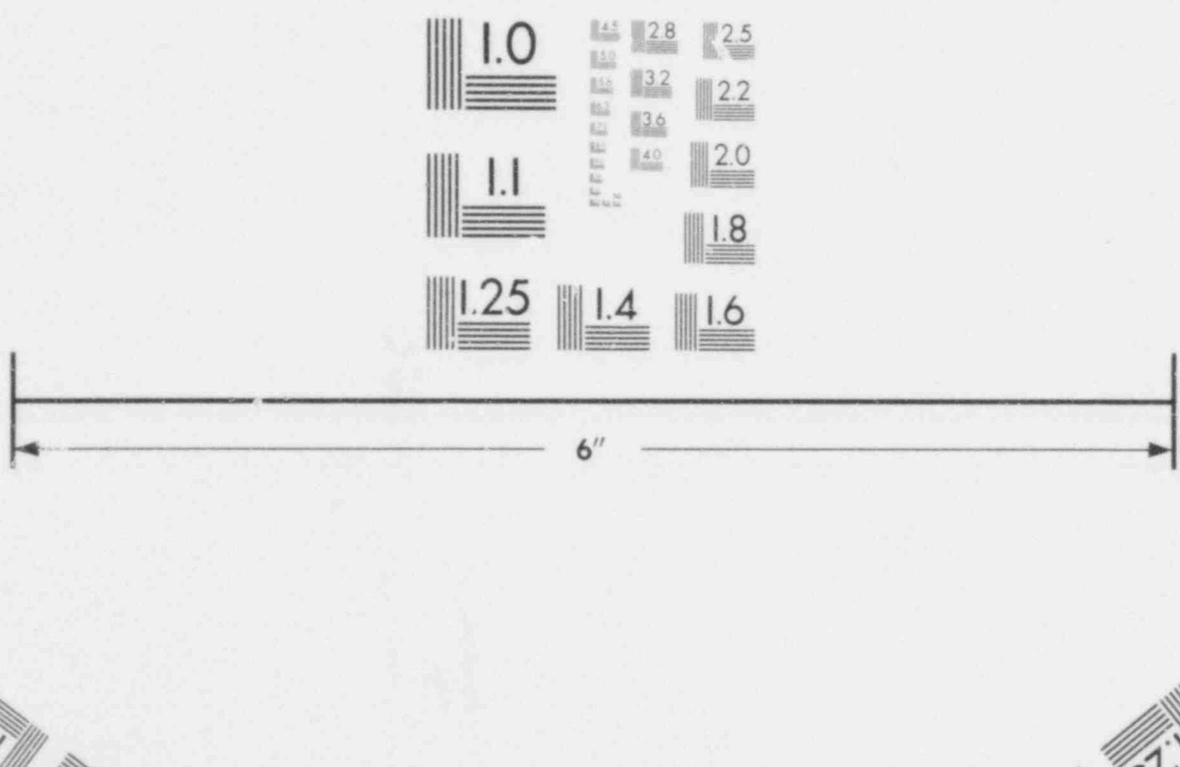

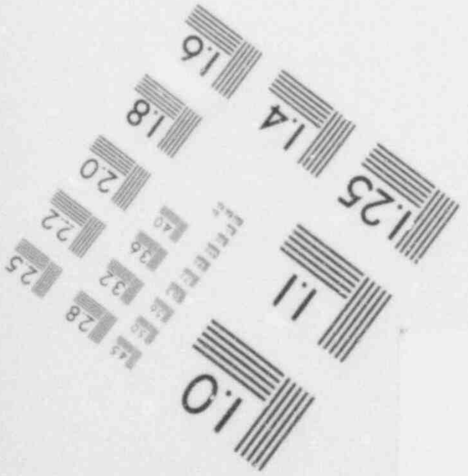

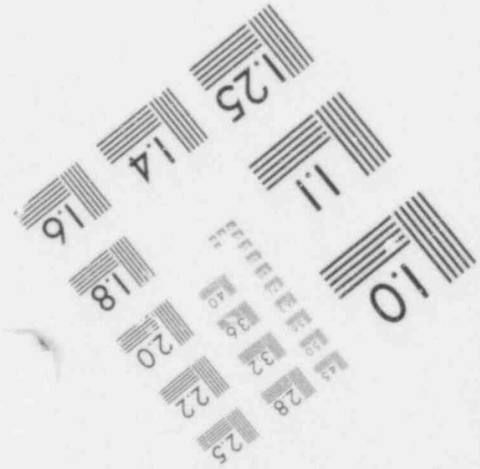

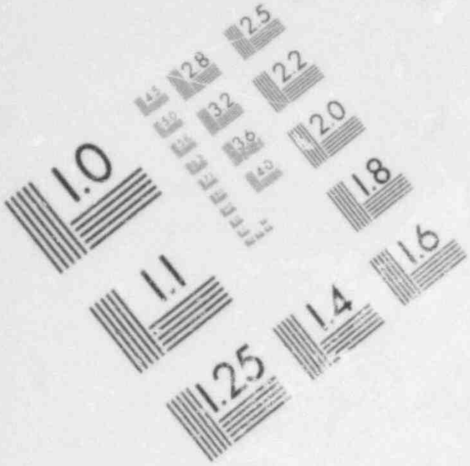

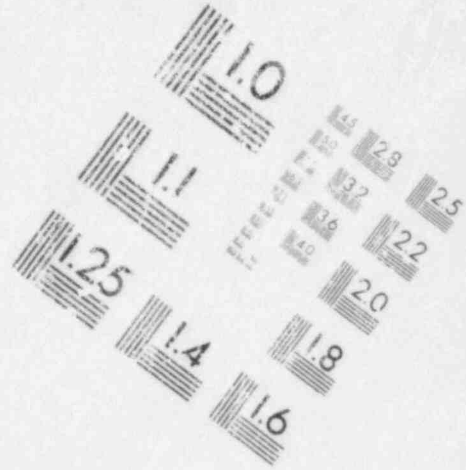

IMAGE EVALUATION TEST TARGET (MT-3)

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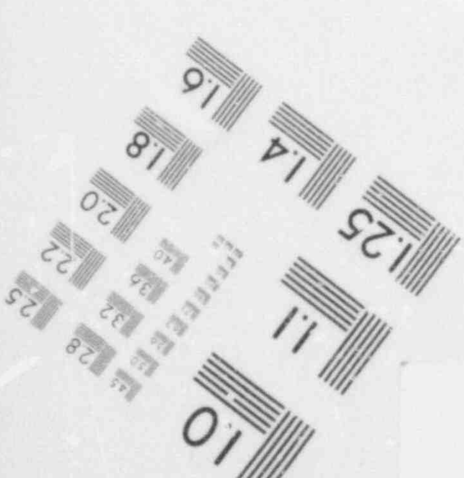

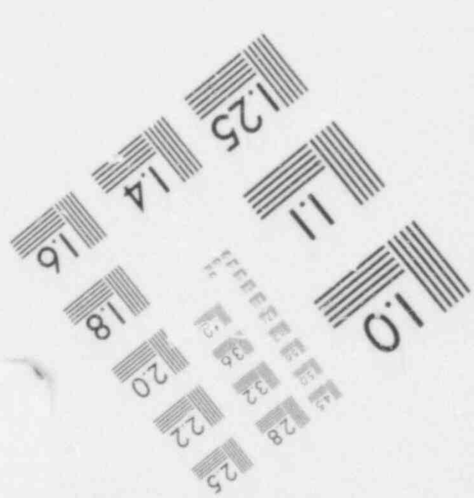

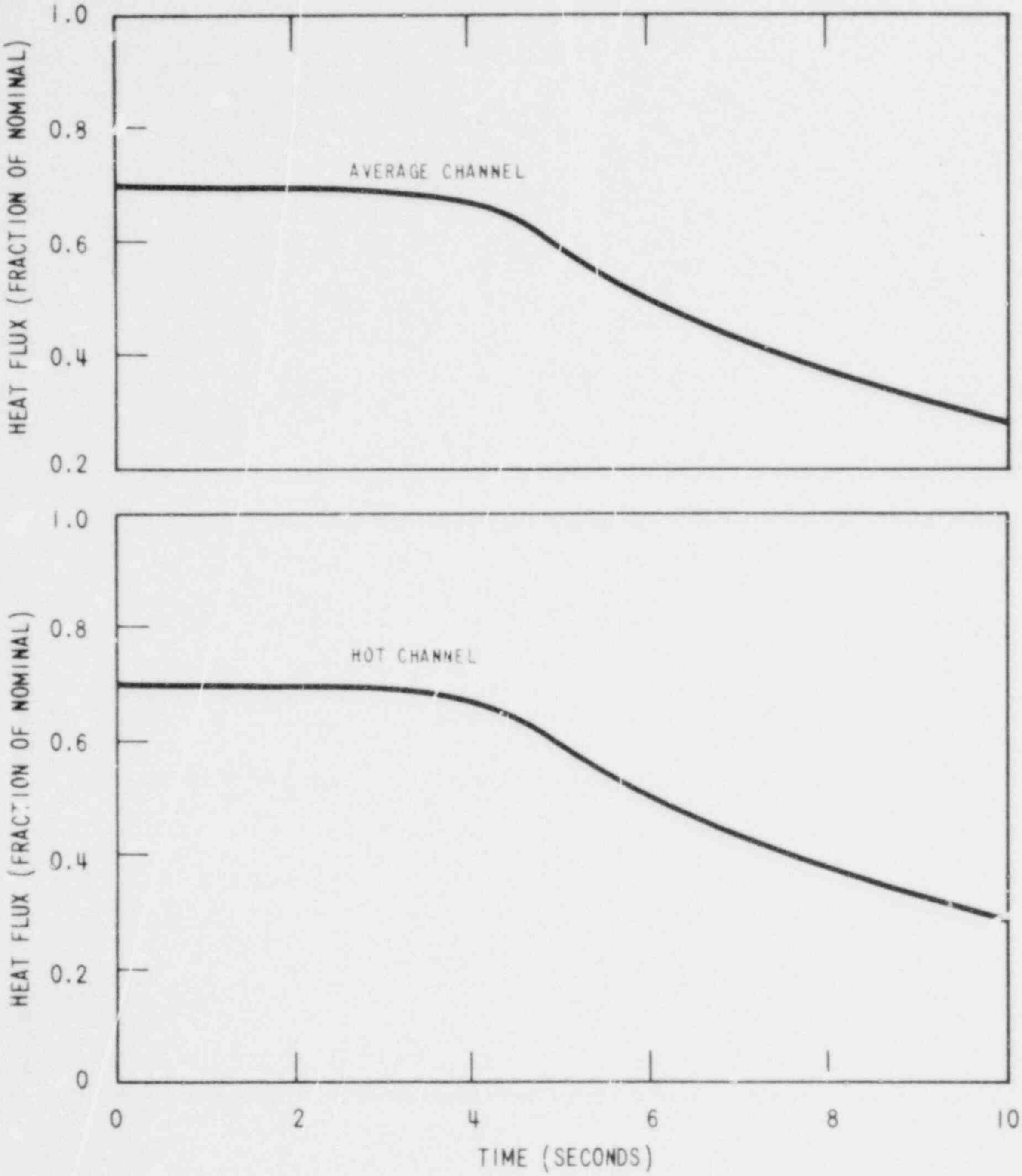

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Figure 15.3.2-7.

Average and Hot Channel Heat Flux Transients for Three Loops in Operation, Three Pumps Coasting Down

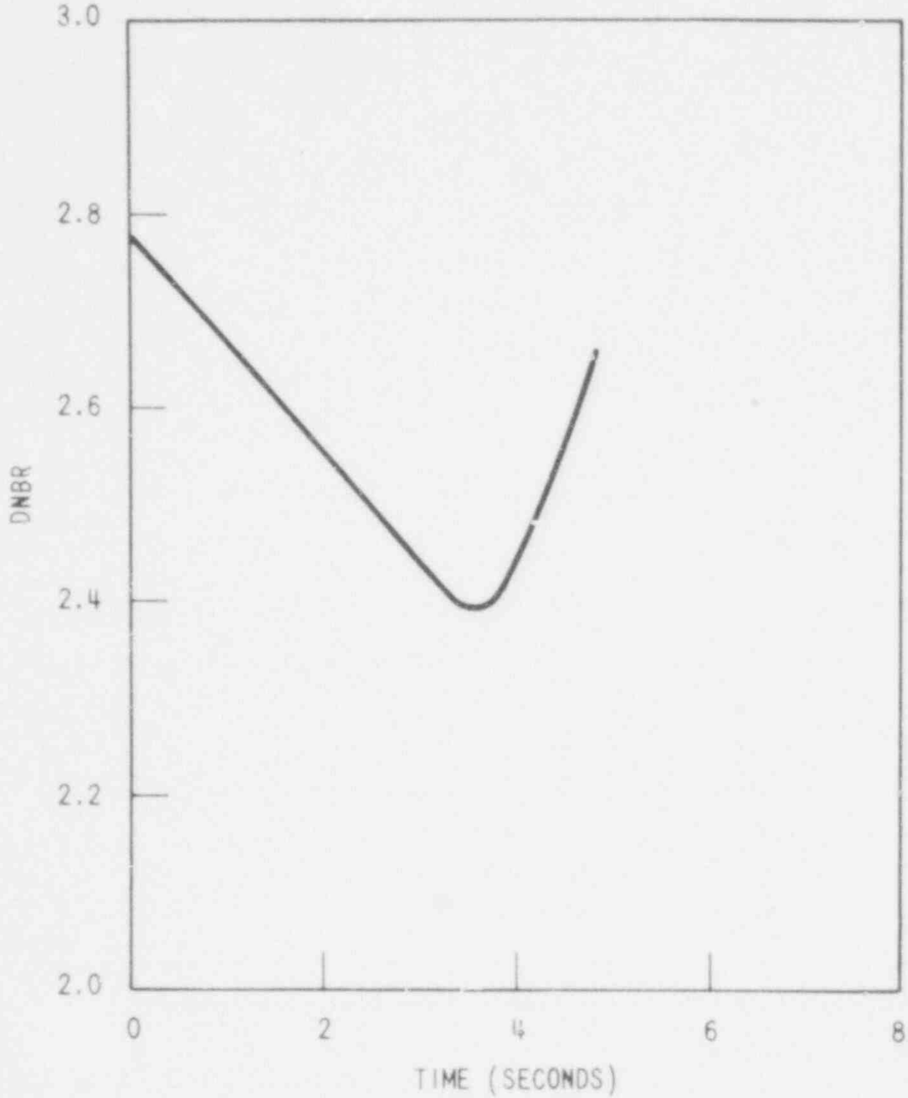

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Figure 15.3.2-8. BLUE DNBR vs. Time for Three Loops in Operation, Three Pumps Coasting Down

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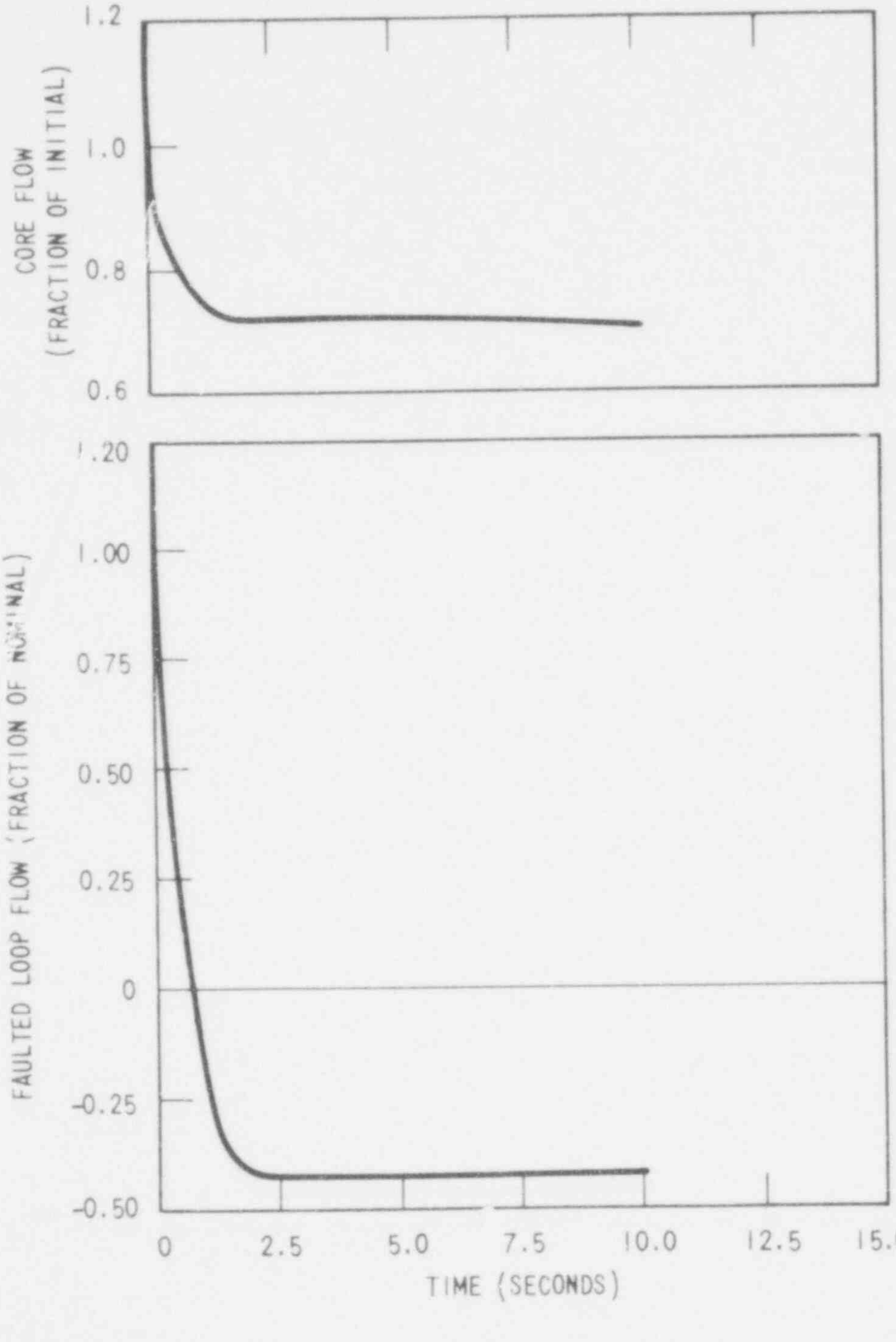

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Figure 15.3.3-1.

Flow Transients for Four Loops in Operation, One Locked Rotor

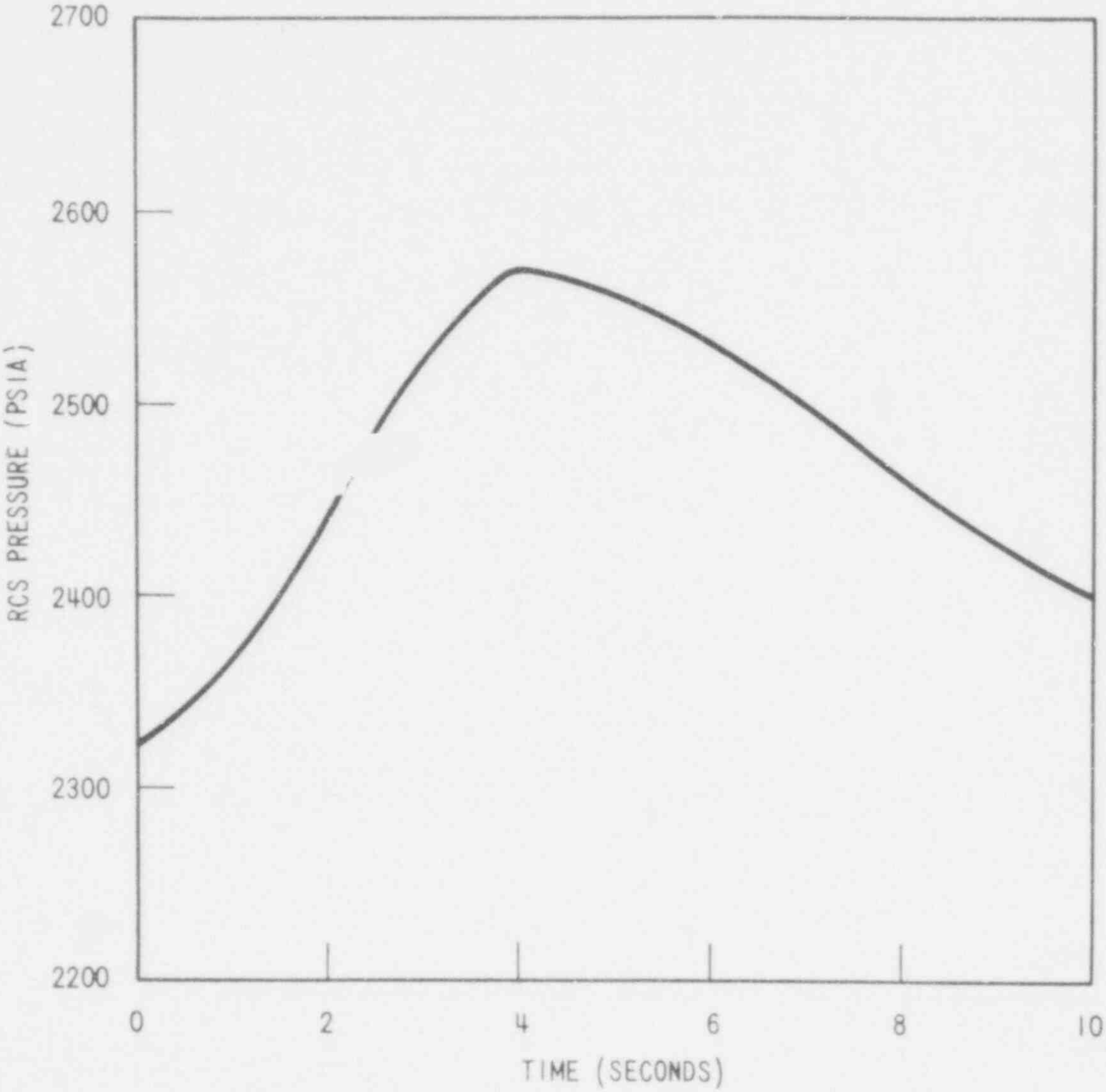

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Figure 15.3.3-2.

Peak Reactor Coolant Pressure for Four Loops in Operation, One Locked Rotor

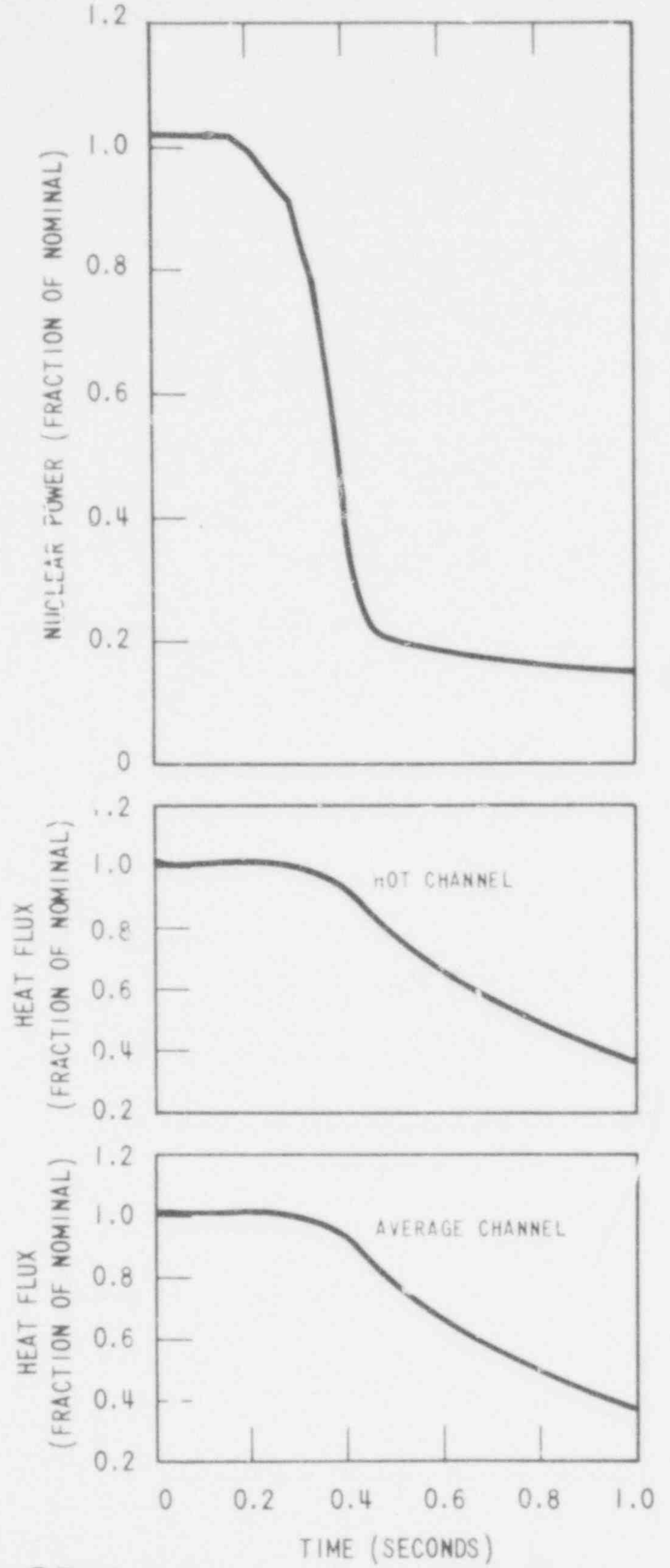

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Figure 15.3.3-3. BLUE

Nuclear Power, Average And Hot Channel Heat Flux Transients for Four Loops in Operation, One Locked Rotor

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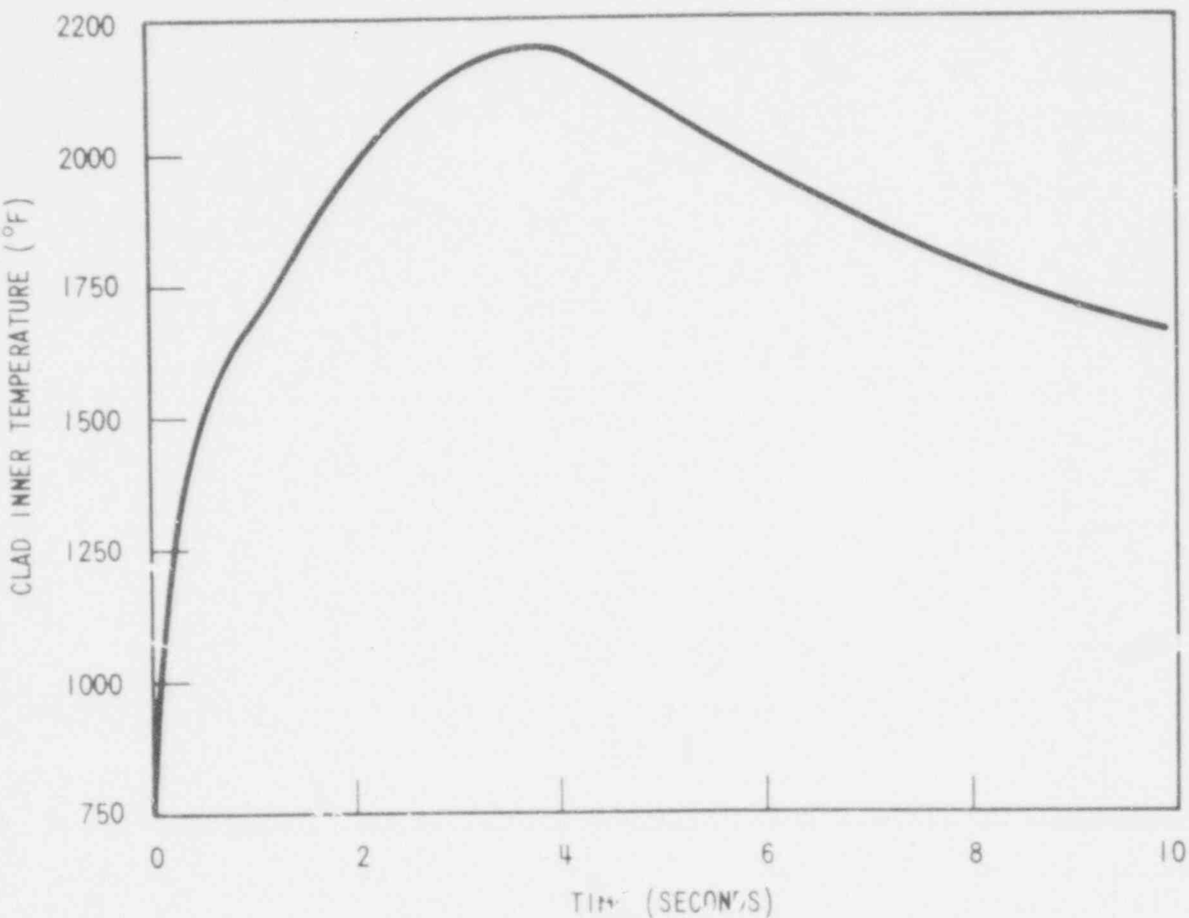

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Figure 15.3.3-4.

Maximum Clad Temperature at Hot Spot for Four Loops in Operation, One Locked Rotor

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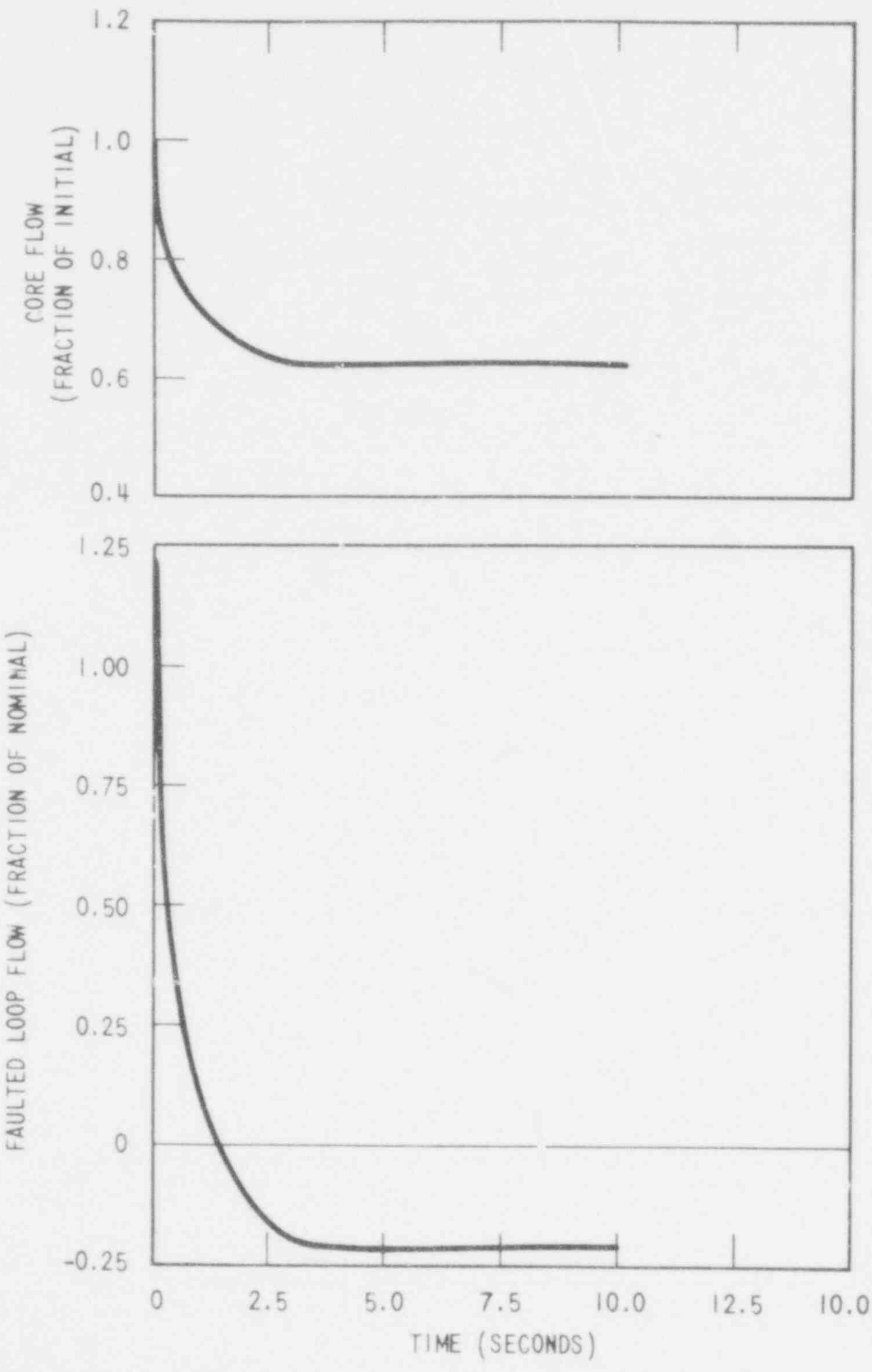

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Figure 15.3.3-5.

Flow Transients for Three Loops in Operation, One Locked Rotor

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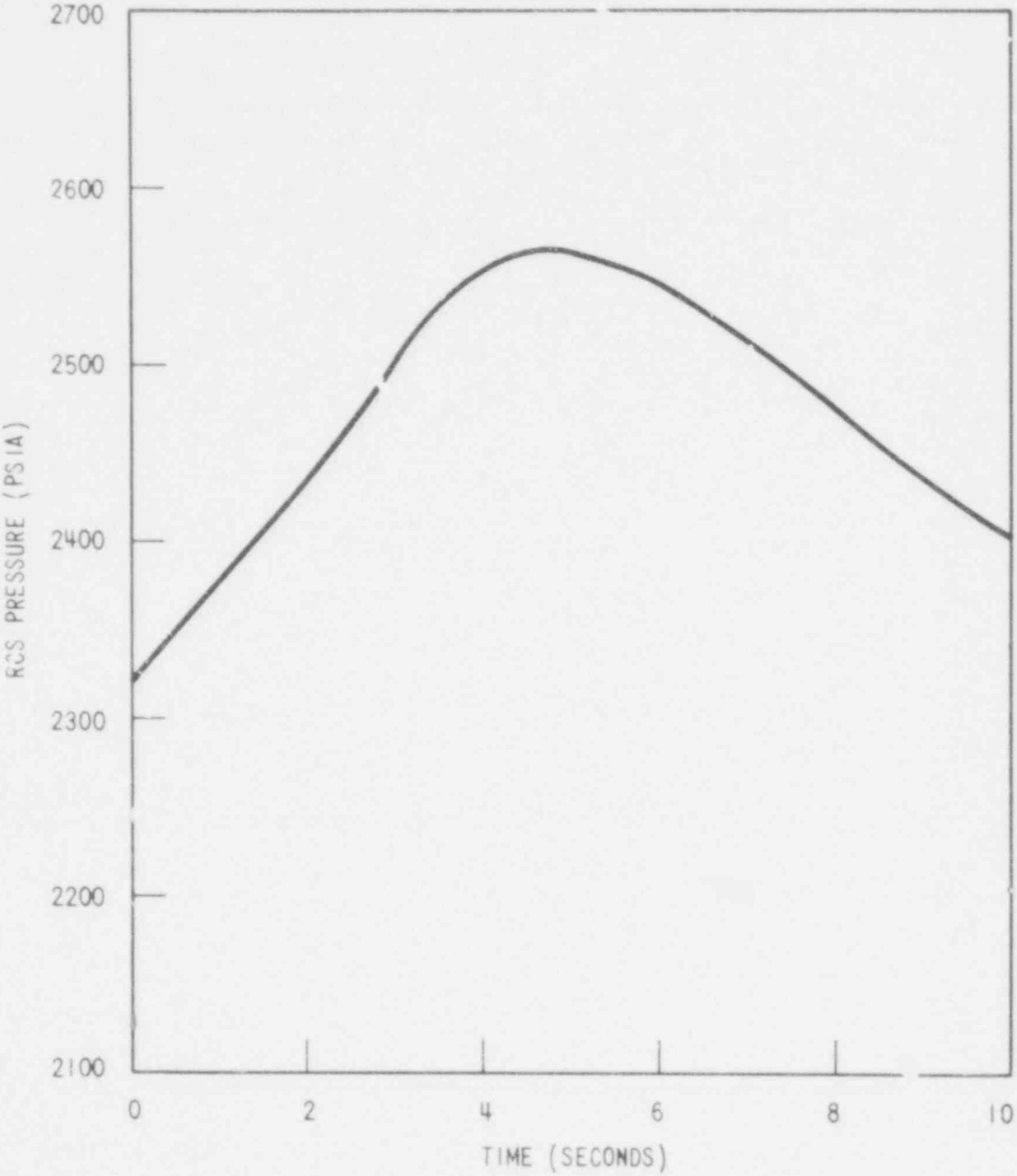

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Figure 15.3.3-6.

Peak Reactor Coolant Pressure for Three Loops in Operation, One Locked Rotor

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15.4

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. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following incidents are presented in Section 15.4:

1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition

2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

3. Rod Cluster Control Assembly Misalignment

4. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

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

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

7. Spectrum of Rod Cluster Control Assembly Ejection Accidents

Items 1, 2, 4, and 5 are considered to be ANS Condition II events, Item 6 and ANS Condition III event, and Item 7 an ANS Condition IV event. Item 3 entails both Condition II and III events. Section 15.0.1 contains a discussion of ANS classifications.

15.4.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA's resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor either subcritical, hot zero power or at power. The "at power" case is discussed in Section 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant").

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCA's from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequenti control banks having the maximum combined worth at maximum speed.

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

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The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power excursion of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the Reactor Protection System:

1. Source Range High Neutron Flux Reactor Trip - actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

2. Intermediate Range High Neutron Flux Reactor Trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

3. Power Range High Neutron Flux Reactor Trip (Low Setting) - actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated only after three of the four channels indicate a power level below this value.

4. Power Range High Neutron Flux Reactor Trip (High Setting) - actuates when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

5. High Nuclear Flux Rate Reactor Trip - actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

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

15.4.1.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the Uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods (TWINKLE (Reference 1)) to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). The average heat flux is next used in THINC (described in Section 4.4) for transient DNBR calculation.

This accident is analyzed using the Improved Thermal Design Procedure as described in WCAP-8567. 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 power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values as a function of power are used. See Section 15.0.4 and Table 15.0.3-2.

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2. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A highly conservative value is used in the analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a large fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10 percent increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position. See Section 15.0.5 for rod cluster control assembly insertion characteristics.

5. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.6.

6. 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 DNB analysis.

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

8. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to DNB.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.4.1-1 through 15.4.1-3 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 15.4.1-1 shows the neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4.1-3 shows the response of the average fuel and cladding temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limit value.

The calculated sequence of events for this accident is shown on Table 15.4.1-1. 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.

The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.4.1.3 Radiological Consequences

There will be no radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power start-up condition event since radioactivity is contained within the fuel rods and reactor coolant system within design limits.

15.4.1.4 Conclusions

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

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety

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valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the limit value.

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

The automatic features of the Reactor Protection System which prevent core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor trip if two-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 imbalance, coolant temperature and pressure to protect against 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 axial power imbalance to ensure that the allowable heat generation rate (kw/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any two-out-of four pressure channels which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from any two-out-of-three level channels when the reactor power is above approximately 10 percent (Permissive-7).

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

1. High neutron flux (one-out-of-four power range)

2. Overpower ΔT (two-out-of-four)

3. Overtemperature ΔT (two-out-of-four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is described in Chapter 7. Figure 15.0.3-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by an given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.47 for the thimble cell and 1.49 for the typical cell). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. 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 pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

The transient is analyzed by the LOFTRAN Code (Reference 3). This code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperature, pressures, and power level. The

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core limits as illustrated in Figure 15.0.3-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3.

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 WCAP 8567.

2. Reactivity Coefficients - Two cases are analyzed:

    a. Minimum Reactivity Feedback. A least negative moderator coefficient of reactivity 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.

    b. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

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

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

5. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks have the maximum combined worth at maximum speed.

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

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Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely offset the consequences of the accident. A discussion of ATWT considerations is presented in Reference [4].

Results

Figures 15.4.2-1 through 15.4.2-3 show the transient response for 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 and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 15.4.2-4 through 15.4.2-6. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.4.2-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the limit value.

Figures 15.4.2-8 and 15.4.2-9 shows the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power, respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the limit value.

The shape of the curves of minimum DNB ration versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.4.2-8, for example, it is noted that:

1. For high reactivity insertion rates (i.e., between $\sim 2 \times 10^{-4}$ δK/sec and 1.0×10^{-3} δK/sec) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNB ratio during the transient thus decreases with decreasing insertion rate.

2. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured Reactor Coolant System average temperature and pressure. This trip circuit is described in detail in Chapter 7; however, it is important in this context to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the Reactor Coolant System in response to power increases.

3. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (e.g., at $\sim 2 \times 10^{-4}$ δK/sec reactivity insertion rate).

   For reactivity insertion rates between $\sim 2 \times 10^{-4}$ δK/sec and $\sim 6 \times 10^{-5}$ δK/sec the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNB ratio) due to the fact that with lower insertion rates the power increase rate is

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slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4. For reactivity insertion rates less than 6×10^{-5} δK/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load of the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System, average temperature. This decrease in rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the overtemperature ΔT trip setpoint to be reached later with resulting lower minimum DNB ratios.

Figures 15.4.2-7, 15.4.2-8 and 15.4.2-9 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak clad centerline temperature will remain below the fuel melting temperature.

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 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 on Table 15.4.1-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.3 Radiological Consequences

The reactor trip causes a turbine trip, and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur, the radiological consequences associated with atmospheric steam release from this event would be less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value. The radiological consequences would be less severe than the steamline break accident analyzed in Subsection 15.1.5.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misalignment accidents include:

1. A dropped full length assembly;

2. A dropped full length assembly bank;

3. Statically misaligned full length assembly (See Table 15.4.3-1);

4. Withdrawal of a single full length assembly.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

Full length RCCA's are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCA's is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCA's of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped assembly, dropped assembly bank, and statically misaligned assembly events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod cluster control assembly (RCCA) from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidently dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10^{-4}/year-refer to Section 7.7.2.2) or multiple deliberate operator actions and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered so low that the limiting consequences may include slight fuel damage.

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Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant", and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

This selection of criterion is not in violation of GDC 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (Emphases have been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC 25.

A dropped assembly or assembly bank is detected by:

1. Sudden drop in the core power level as seen by the Nucelar Instrumentation System;
2. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;
3. Rod at bottom signal;
4. Rod deviation alarm;
5. Rod position indication.

Misaligned assemblies are detected by:

1. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;
2. Rod deviation alarm;
3. Rod position indicators.

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The resolution of the rod position indicator channel is ± 5 percent of span ($\pm$ 7.2 inches). Deviation of any assembly from its group by twice this distance (10 percent of span, or 14.4 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more rod position indicator channels should be ot of service, detailed operating instructions shall be followed to assure the alignment of the non-indicated assemblies. The operator is also required to take action as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

Plant systems and equipment which are available to mitigate the effects of the various control rod misoperations are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

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15.4.3.2 Analysis of Effects and Consequences

1. A dropped RCCA, dropped RCCA group, and statically misaligned RCCA.

Method of Analysis

Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC Code to calculate the DNBR.

Results

a. Dropped RCCA

A dropped RCCA typically results in a reactivity insertion of -150 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of an RCCA. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed.

b. Dropped RCCA Group

A dropped RCCA group typically results in a reactivity insertion of -1,200 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of a RCCA. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed to further cool down the plant.

c. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCAA is fully inserted, or where bank D is fully inserted with one RCCA

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fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the technical specifications may vary from time to time depending on a number of limiting criteria. 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 the RCCA misalignment shown in Table 15.4.3-1, with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case was 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 initial conditions are included in the limit DNBR as described in WCAP-8567.

DNB calculations have not been performed specifically for assemblies missing from other banks; however, power shape calculations have been done as required for the RCCA ejection 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 the RCCA misalignments shown in Table 15.4.3-1 with one RCCA fully inserted, the DNBR does not fall below the limit value. This case was analyzed assuming the initial reactor power,

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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 are included in the limit DNBR as described in WCAP-8567.

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 (as noted in Table 15.4-2) and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant technical specifications and operating instructions.

2. Single RCCA Withdrawal

Method of Analysis

Power distributions within the core are calculated using the computer codes as described in Table 4.1-2. The peaking factors are then used by THINC to calculate the minimum DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at beginning-of-life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

Results

For the single rod withdrawal event, two cases have been considered as follows:

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a. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR's than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the low DNBR trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5 percent.

b. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCA's in the controlling bank. The transient will then proceed in the same manner as Case a described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

The most limiting rod cluster control assembly misoperation, accidental withdrawal of a single RCCA, is predicted to result in less than 1% damage. The subsequent reactor and turbine trip would result in atmospheric steam dump, assuming the condenser was not available for use. The radiological consequences from this event would be no greater than the main steamline break event, analyzed in Subsection 15.1.5.

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15.4.3.4 Conclusions

For all cases of dropped single RCCA's or dropped banks, power decreases rapidly, and the reactor is tripped by the power range negative neutron flux rate trip. Thus, there is no reduction in the margin to core thermal limits, and the DNB design-basis as described in Section 4.4 is met.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value. Thus, the DNB design-basis as described in Section 4.4 is met.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core. The radiological consequences from these events would be no greater than the main steamline break event analyzed in Subsection 15.1.5.

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

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operatec ' power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

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Administrative procedures require that the unit be brought to a load of less than 25 percent of full power prior to starting the pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

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

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, (See Table 7.2.1-2 for a description of interlocks.) which has been previously reset for three loop operation.

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by three digital computer codes. The LOFTRAN Code (Reference 3) is used to calculate the loop and core low, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Reference 2) is used to calculate the core heat flux transient based on core flow and nulcear power from LOFTRAN. The THINC Code (Section 4.4) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Secion 15.0.3.

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

1. Initial reactor power pressure, and RCS temperatures are assumed to be at their nominal N-1 loop operation valves. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-8567.

2. Following initiation of startup of the idle pump, the inactive loop flow reverses and accelerates to its nominal flow value in approximately 28 seconds. This value is greater than the expected startup time, and is conservative for this analysis.

3. A conservatively large moderator density coefficient. (See Section 15.0.4)

4. A conservatively small (absolute value) negative Doppler power coefficient. (See Section 15.0.4)

5. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

6. The reactor trip is assumed to occur on low coolant flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 85% of rated power which corresponds to the nominal setpoint plus 9% for nuclear instrumentation errors.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.4.4-1 through 15.4.4-5. As shown in these curves, during the first part of the transient, the increase in

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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 limit value.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.4.4-1.

The calculated sequence of events for this accident is shown on Table 15.4.1-1 The transient results illustrated in Figures 15.4.4-1 through 15.4.4-5 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Radiological Consequences

There would be minimal radiological consequences associated with startup of an inactive reactor coolant loop at an incorrect temperature. Therefore, this event is not limiting. The reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postuated to occur from this transient, the radiological consequences associated with this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.4.4.4 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the limit value; thus, no fuel or clad damage is predicted.

15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

(Not applicable in PWR's)

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

15.4.6.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System via the reactor makeup portion of the Chemical and Volume Control System. Boron dilution is a manual operation under administrative control with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the Reactor Coolant System. The Chemical and Volume Control System 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 reactor water makeup control valve provides makeup to the Reactor Coolant System which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the Reactor Coolant System at pressure, at least one charging pump must be running in addition to a reactor makeup water pump.

The rate of addition of unborated makeup water to the Reactor Coolant System when it is not at pressure is limited by the capacity of the reactor makeup water pumps. Normally, only one reactor makeup water pump is operating while the other on standby. With the RCS at pressure, the maximum delivery rate is limited by the control valve.

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 on the control board.

In order to dilute two separate operations are required:

1. The operator must switch from the automatic makeup mode to the dilute mode;

2. The start button must be depressed.

Omitting either step would prevent dilution.

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

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

15.4.6.2 <u>Analysis of Effects and Consequences</u>

<u>Method of Analysis</u>

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

<u>Dilution During Refueling</u>

An uncontrolled boron dilution accident cannot occur during refueling as a result of a reactor coolant makeup system malfunction. This accident is prevented by administrative controls which isolate the Reactor Coolant System from the potential source of unborated water.

Valves 1NV186A, 1NV181A, 1NV231, 1NV244, and 1NV240 in the CVCS will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup to reach the reactor coolant system. Any makeup which is required during refueling will be borated water supplied from the refueling water storage tank by refueling water pump.

The most limiting alternate source of uncontrolled boron dilution would be the inadvertent opening of a valve in the Boron Thermal Regeneration System (BTRS). For this case highly borated RCS water is depleted of boron as it passes through the BTRS and is returned via the volume control tank. The following conditions are assumed for an uncontrolled boron dilution during refueling:

Technical Specifications require the reactor to be borated to a concentration equivalent to 5.0% Δ k at refueling. The maximum boron concentration to lose all shutdown margin is very conservatively estimated to be 1500 ppm.

Dilution flow is assumed to be the maximum capacity of the BTRS (100 gpm) with 0 ppm water returning to the RCS. This is assumed although normally this system is not operated at refueling conditions.

Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A minimum water volume (4722 ft^3) in the RCS is used. This is the minimum volume of the RCS for residual heat removal system operation.

Dilution During Cold Shutdown

Conditions at cold shutdown require the reactor to be shut down by at least 1.0% Δk. The maximum boron concentration required to meet this shutdown margin is very conservatively estimated to be 1572 ppm. The following conditions are assumed for an uncontrolled boron dilution during cold shutdown.

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Dilution flow is assumed to be the combined capacity of the two primary water makeup pumps with the coolant system depressurized (578 gpm). This is assumed although normally only one pump is in operation.

Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

A minimum water volume (5567 ft^3) in the Reactor Coolant System is used. This corresponds to the active volume of the Reactor Coolant System minus the pressurizer volume, while on the residual heat removal system.

Dilution During Hot Standby

Conditions at hot standby require the reactor to have available at least 1.30% Δk shutdown margin. The maximum boron concentration required to meet this shutdown margin is very conservatively estimated to be 1120 ppm. The following conditions are assumed for a continuous boron dilution during hot standby:

1. Dilution flow is assumed to be the combined capacity of the two primary water makeup pumps with the Reactor Coolant System at 135 psia and 350°F (approximately 570 gpm).

2. A minimum water volume (10427 ft^3) in the Reactor Coolant System is used. This volume corresponds to the active volume of the Reactor Coolant System minus the pressurizer volume.

Dilution During Startup

Conditions at startup are identical to the hot standby case with the exception of the dilution flow. The dilution flow is assumed to be a conservatively high charging flow rate (188 gpm) consistent with Reactor Coolant System operation at 2250 psia and 557°F.

Dilution During Full Power Operation

With the unit at power and the Reactor Coolant System at pressure, the dilution rate is limited by the capacity of the charging pumps (analysis is performed assuming all charging pumps are in operation although only one is normally in operation). The effective reactivity addition rate is a function of the reactor coolant temperature and boron concentration. The reactivity insertion rate calculated is based on a conservatively high value for the expected boron concentration at power (1500 ppm) as well as a conservatively high charging flowrate capacity (188 gpm).

The Reactor Coolant System volume assumed (10427 ft^3) corresponds to the active volume of the RCS excluding the pressurizer.

Results

Dilution During Refueling

For dilution during refueling, the minimum time required for the shutdown margin to be lost and the reactor to become critical is 101.6 minutes.

Dilution During Cold Shutdown

For dilution during cold shutdown, the minimum time required for the shutdown margin to be lost and the reactor to become critical is 3.4 minutes.

Dilution During Startup

For dilution during startup the minimum time required for the shutdown margin to be lost and the reactor to become critical is 29.0 minutes.

Dilution During Hot Standby

For dilution during hot standby, the minimum time required for the shutdown margin to be lost and the reactor to become critical is 11.5 minutes.

Dilution During Full Power Operation

1. With the reactor in automatic control, the power and temperature increase from boron dilution results in insertion of the rod cluster control assemblies and a decrease in the shutdown margin. The rod insertion limit alarms (low and low-low settings) provide the operation with adequate time (of the order of 78 minutes) to determine the cause of dilution, isolate the primary grade water source, and initiate reboration before the total shutdown margin is lost due to dilution.

2. With the reactor in manual control and if no operator action is taken, the power and temperature rise will cause the reactor to reach the overtemperature ΔT trip setpoint. The boron dilution accident in this case is essentially identical to a rod cluster control assembly withdrawal accident. The maximum reactivity insertion rate for boron dilution is approximately 1.11 pcm/sec and is seen to be within the range of insertion rates analyzed. Prior to the overtemperature ΔT trip, an overtemperature ΔT alarm and turbine runback would be actuated. There is adequate time available (of the order of 76.4 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the primary grade water sources and initiate reboration before the reactor can return to criticality.

15.4.6.3 Radiological Consequences

There would be minimal radiological consequences associated with a chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant event. The reactor trip causes a turbine-trip, and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage occurs from this transient, the radiological consequences associated with this event are less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.4.6.4 Conclusions

The result present above show that there is adequate time for the operator to manually terminate the source of dilution flow. Following termination of the dilution flow the reactor will be in a stable condition. The operator can then initiate reboration to recover the shutdown margin. The calculated sequence of events is shown on Table 15.4.1-1. The radiological consequences of this event would be less limiting than the steamline break event analyzed in Subsection 15.1.5.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

15.4.7.1 Identification of Causes and Accident Description

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

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of

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Condition I and Condition II transients. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

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

15.4.7.2 Analysis of Effects and Consequences

Method of Analysis

Steady state power distributions in the x-y plane of the core are calculated using the Computer codes as described in Table 4.1-2. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distributions in the x-y plan for a correctly loaded core assembly are also given in Chapter 4 based on enrichments given in that section.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (see Figures 15.4.7-1 to 15.4.7-5, inclusive).

Results

The following core loading error cases have analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange to two adjacent assemblies near the periphery of the core (see Figure 15.4.7-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.4.7-2 and 15.4.7-3).

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

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

Case C:

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4.7-4).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4.7-5).

15.4.7.3 Radiological Consequences

There are no radiological consequences associated with inadvertent loading and operation of a fuel assembly in an improper position since activity is contained with the fuel rods and reactor coolant system within design limits.

15.4.7.4 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

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

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

15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

15.4.8.1 Identification of Causes and Accident Description

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

15.4.8.1.1 Design Precautions and Protection

Certain features are intended to preclude the possibility of rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCA's, and minimizes the number of assemblies inserted at high power levels.

Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

1. Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

2. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed reactor coolant system.

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-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

4. The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel. This material exhibits

excellent notch toughness at all temperatures which 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 (and other) welds.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCA's inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCA's above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCA's is continuously indicated in the control room. An alarm will occur if a bank of RCCA's approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at low level alarm and emergency boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference [5]. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCA's are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident. See Section 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold or significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 6). Extensive tests of UO_2 zirconium clad fuel rods representative of those in Pressurized Water Reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a sightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 7) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marignally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) event for irradiated rods, did not occur below 300 cal/gm.

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

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

2. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2700°F).

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

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

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

A detailed discussion of the method of analysis can be found in Reference [5].

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 1), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one. two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described in the following) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in

0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel-and cladding transient heat transfer computer code, FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative pellet 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 the Bishop-Sandburg-Tong correlation (Reference 8) to determine the film boiling coefficient after DNB. The BST correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 15.0.11.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

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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 (section 4.4) 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 reactor coolant system and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4.8-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

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

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, 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 [5]. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full

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power rodded configurations and compared to values used in the analysis. It has been found that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature 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 reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kenetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis (Reference 5).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler defect used is given in Section 15.0.4. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning of life and 0.50 percent at end of life for the first cyclce. The accident is sensitive to β if the ejected rod worth is equal to or greater than β as in zero power transients. In order to allow for future cycles, pessimistic estimates of β of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4.8-1 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 3.05 seconds after the start of fall. The choice of such a conservative insertion rate means that 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.

The minimum design shutdown available for this plant at HZP may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, adverse xenon distribution conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCA's (one of which is the worst ejected rod) is to reduce the shutdown by about an additional one percent Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for a typical four-loop plant assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on the coincidence of low pressurizer pressure and level within one minute after the break. The reactor coolant pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the reactor coolant system temperature below no-lead by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than ten minutes after the break. The addition of high borated (20,000 ppm) safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

As discussed in Section 15.4.8.1.1, reactor protection for a rc ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the Reactor Trip System. No single failure of the Reactor Trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

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Results

Cases are presented for both beginning and end of life at zero and full power.

1. Beginning of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.25 percent Δk and 6.40 respectively. The peak hot spot clad average temperature was 2351°. 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.

2. Beginning of Cycle, Zero Power

   For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of .83 percent Δk and a hot channel factor of 11.0. The peak hot spot clad temperature reached 2654°F, the fuel center temperature was 4074°F.

3. End of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25 percent Δk and 6.40 respectively. This resulted in a peak clad temperature of 2172°F. The peak hot spot fuel temperature reached melting conservatively assumed at 4800°F. However, melting was restricted to less than 10% of the pellet.

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4. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and bank C at its insertion limit. The results were .905 percent Δk and 18.0 respectively. The peak clad and fuel center temperatures were 2696 and 4100°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 15.4.8-1. The nuclear power and hot spot fuel and clad temperature transients for the worst cases are presented in Figures 15.4.8-1 through 15.4.8-4. (Beginning of life full power and end of life zero power.)

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4.8-1 through 15.4.8-4, is presented in Table 15.4.1-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Section 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the Reactor Coolant System, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Section 15.6.5. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other loss of coolant accident to recover from the event.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based on a detailed three dimensional THINC analysis (Reference 5).

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 5). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the Reactor Coolant System.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. Since the 17 x 17 fuel design is also under-moderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

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15.4.8.3 Radiological Consequences of a Postulated Rod Ejection Accident

Two analyses of a postulated rod ejection accident will be performed: (1) a realistic analysis, and (2) an analysis based on Regulatory Guide 1.77 (May, 1974). The parameters used for each of these analyses are listed in Table 15.4.8-2.

Assumptions for Regulatory Guide 1.77 Analysis

The following conservative assumptions will be used in the Regulatory Guide 1.77 analysis of the release of radioactivit to the environment in the event of a postulated rod ejection accident:

1. Prior to the accident the plant is assumed to be operating at full power and the primary and secondary coolant correspond to the specific activity limits given in the Technical Specifications.

2. 100 percent of the noble gases and iodines in the cladding gaps of the fuel rods experiencing cladding damage (assumed to be 10 percent of the rods in the core) (Reference 5) is assumed released to the reactor coolant. Per Regulatory Guide 1.77, the gap activity consists of 10 percent of the total noble gases and 10 percent of the total radioactive iodine in the damaged rods at the time of the accident. The total core and fuel-clad gap activities are given in Table 15.0.9-1.

3. 50 percent of the iodines and 100 percent of the noble gases in the fuel that melts is assumed released to the reactor coolant. This is a very conservative assumption since only centerline melting could occur for a maximum time period of 6 seconds.

4. The fraction of fuel melting was conservatively assumed to be 0.25% of the core as determined by the following method:

   a. A conservative upper limit of 50 percent of the rods experiencing cladding damage may experience centerline melting (a total of 5 percent of the core).

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b. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the rod volume will actually melt (equivalent to 0.5 percent of the core that could experience melting).

c. A conservative maximum of 50 percent of the axial length of the rod will experience melting due to the power distribution (.5 of the 0.5 percent of the core = 0.25 percent of the core).

5. Instantaneous mixing occurs in the containment of all the noble gases and 50 percent of the iodine activity released from the coolant. It is assumed that 50% of the iodine activity released to the containment atmosphere immediately plates out on containment surfaces.

6. No credit is assumed for removal of iodine in the containment due to containment sprays.

7. The containment leaks for the first 24 hours at its design leak rate as specified in the technical specifications of 0.10 percent/day. Thereafter, the containment leak rate is 0.05 percent per day.

8. For the case of loss of offsite power, 58,600 pounds of steam are discharged from the secondary system through the relief valves the first 540 seconds following the accident. Steam dump is terminated after 540 seconds.

15.4.8.4 Conclusions

Even on a pessimistic basis, the analysis indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits. It is concluded that there is no danger of further consequential damage to the Reactor Coolant System. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to ten percent.

Parameters recommended for use in determining the radioactivity released to atmosphere for a rod ejection accident are given in Table 15.4.8-2. The Reactor Coolant System integrated break flow to Containment following a rod ejection accident is shown in Figure 15.4.8-5.

15.4.9 References

1. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kenetics Computer Code," WCAP-7979-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

2. Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

3. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972.

4. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

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.

6. Taxelius, T. G. (Ed), "Annual Report - SPERT Project, October, 1968, September, 1969," Idaho Nuclear Corporation IN-1370, June 1970.

7. Liimataninen, R. C. and Testa, F. J., "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, January - June 1966, p. 177, November 1966.

8. Bishop, A. A., Sanburg, 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 15.4.1-1 (Page 1)

Time Sequence of Events for Incident which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Conditon	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Peak average clad temperature occurs	12.8
	Peak average fuel temperature occurs	13.1
	Power range high neutron flux low setpoint reached	13.7
	Peak nuclear power occurs	13.9
	Rod begin to fall into core	14.22
	Peak heat flux occurs	16.5
	Minimum DNBR occurs	16.5
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0

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Time Sequence of Events for Incident which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
	Power range high neutron flux high trip point reached	1.79
	Rods begin to fall into core	2.29
	Minimum DNBR occurs	3.30
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (2 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	64.2
	Rods begin to fall into core	66.2
	Minimun DNBR occurs	67.1
Startup of an inactive reactor coolant loop at an incorrect temperature	Initiation of pump startup	1.0
	Power reaches P-8 trip setpoint	13.4
	Rods begin to drop	13.9
	Minimum DNBR occurs	15.0

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TABLE 15.4.1-1 (Page 3)

Time Sequence of Events for Incident which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
CVCS Malfunction that results in a decrease in the boron concentration in the reactor coolant		
1. Dilution during refueling	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	6096
2. Dilution during cold shutdown	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	201
3. Dilution during hot standby	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	689
4. Dilution during startup	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	1743

Time Sequence of Events for Incident which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
5. Dilution during full power operation		
a. Automatic reactor control	Dilution begins	0
	Shutdown margin lost	4680
b. Manual reactor control	Dilution begins	0
	Reactor trip setpoint reahced for overtemperature ΔT	97.4
	Rods begin to fall into core	99.4
	Shutdown is lost (if dilution continues after trip)	4585
Rod Cluster Control Assembly Ejection		
1. Beginning-of-Life, Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.14

TABLE 15.4.1-1 (Page 5)

Time Sequence of Events for Incident which Cause Reactivity and Power Distribution Anomalies

Accident	Event	Time (sec.)
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.11
	Peak heat flux occurs	2.20
	Peak clad temperature occurs	2.20
2. End-of-Life, Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.17
	Peak nuclear power occurs	0.21
	Rods begin to fall into core	0.68
	Peak clad temperature occurs	1.66
	Peak heat flux occurs	1.66
	Peak fuel temperature occurs	2.56

TABLE 15.4.3-1

MINIMUM CALCULATED DNBR FOR ROD CLUSTER CONTROL ASSEMBLY MISALIGNMENT

CASES ANALYZED	RADIAL POWER* PEAKING FACTOR ($F_{\Delta H}$)	MINIMUM DNBR
Bank D at insertion limit, D-12** fully withdrawn	1.63	***
Rod Cluster Control Assembly G-13 fully inserted	1.68	***
Rod Cluster Control Assembly D-12 fully inserted	1.67	***
Rod Cluster Control Assembly H-12 fully inserted	1.68	***
Rod Cluster Control Assembly F-10 fully inserted	1.66	***

*Values include 15% uncertainty allowance in $F_{\Delta H}$.

**Designations such as D-12 specify a core location; see Chapter 4.0.

***Minimum value greater than limit value (1.47 for thimble cell, 1.49 for typical cell); see Section 4.4.

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TABLE 15.4.8-1

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

TIME IN LIFE	BOL-HFP BEGINNING	BOL-HZP BEGINNING	EOL-HFP END	EOL-HZP END
Power Level, %	102	0	102	0
Ejected rod worth, %ΔK	0.25	0.83	0.25	0.905
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.07	1.30	3.80
Trip reactivity, %Δk	5.0	2.0	4.0	2.0
F_q before rod ejection	2.50		2.50	
F_q after rod ejection	6.40	11.0	6.40	18.0
Number of operational pumps	4	2	4	2
Max. fuel pellet average temperature, ^{o}F	4113	3530	3688	3533
Max. fuel center temperature, ^{o}F	4977	4074	4818	4100
Max. clad average temperature, ^{o}F	2351	2654	2130	2696
Max. fuel store energy, cal/gm	180	150	158	150
% Fuel Melt	<10%	0	<10%	0

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TABLE 15.4.8-2

Assumptions to be Used for Radiological Consequences for the Rod Cluster Control Assembly Ejection Accident

	REALISTIC ANALYSIS	REGULATORY GUIDE 1.77 ANALYSIS
Core thermal power	3565 MWt	3656 MWt
Reactor coolant activity prior to accident	ANS N237	See Table 15.0-12
Steam generator tube leakage rate during accident	0.009 gpm	1.0 gpm**
Failed fuel	0.0	10% of fuel rods in core
Activity released to reactor coolant from failed fuel and available for release		
Noble gases	None	10% of gap inventory
Iodines	None	10% of gap inventory
Melted fuel	None	0.25% of core
Activity released to reactor coolant from melted fuel and available for release		
Noble gases	None	0.25% of core inventory
Iodines	None	0.125% of core inventory
Iodine partition factor in steam generators	0.1	0.1
Iodine partition factor in condenser during accident	0.0001	NA

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TABLE 15.4.8-2 (Continued)

Assumptions to be Used for Radiological Consequences for the Rod Cluster Control Assembly Ejection Accident

	REALISTIC ANALYSIS	REGULATORY GUIDE 1.77 ANALYSIS
Plateout of iodine activity released to containment	50%	50%
Form of iodine activity in containment available for release		
Elemental iodine	91%	91%
Methyl iodine	4%	4%
Particulate iodine	5%	5%
Offsite power	Available	Lost
Steam dump from relief valves	0.0	58,600 lb
Duration of dump from relief valves	0.0	500 seconds

* American National Standard Source Term Specification N237 (assumes 100 lbs/day steam generator leakage)

** 0.347 in defective steam generator and 0.218 gpm per non-defective steam generator (during accident)

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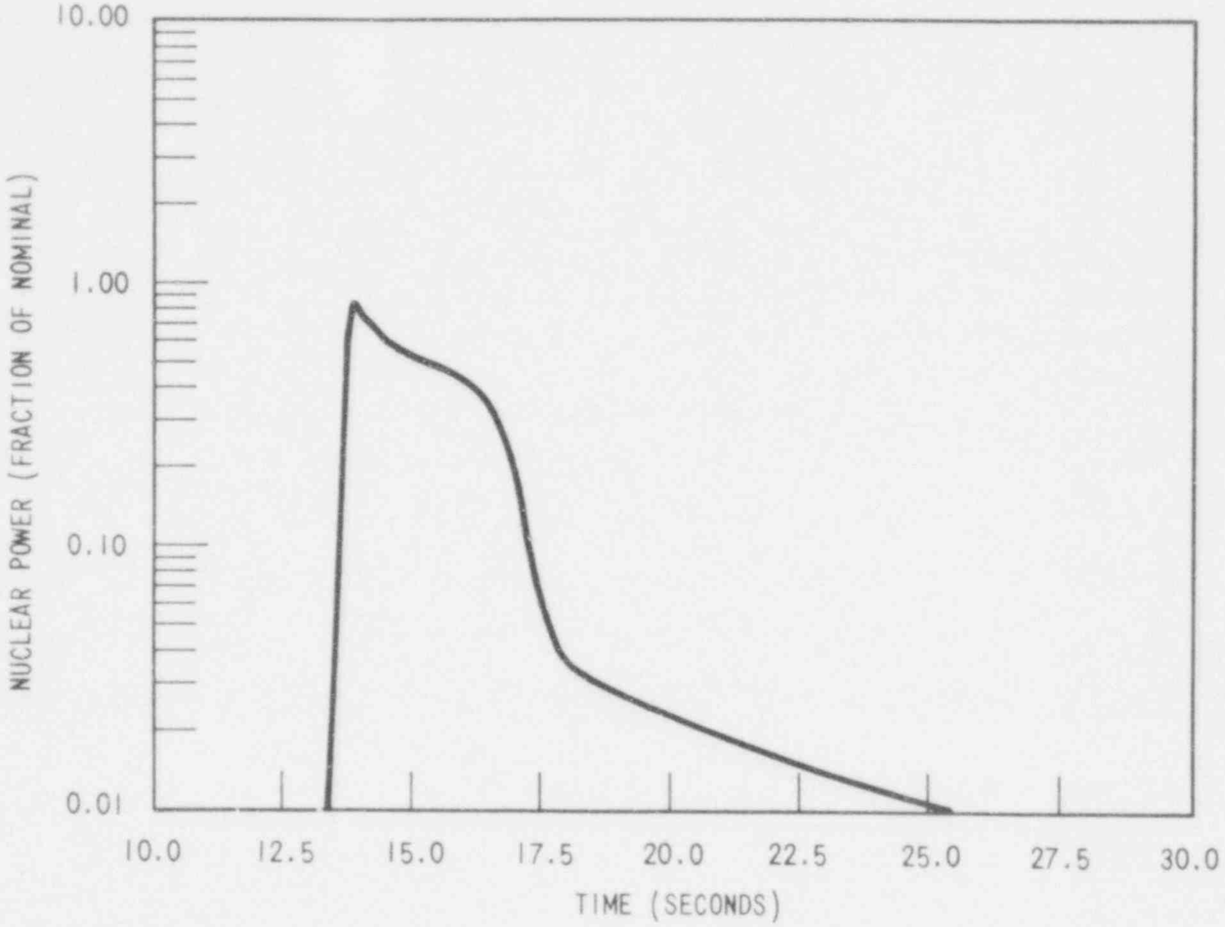

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Figure 15.4.1-1. BLUE

Neutron Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

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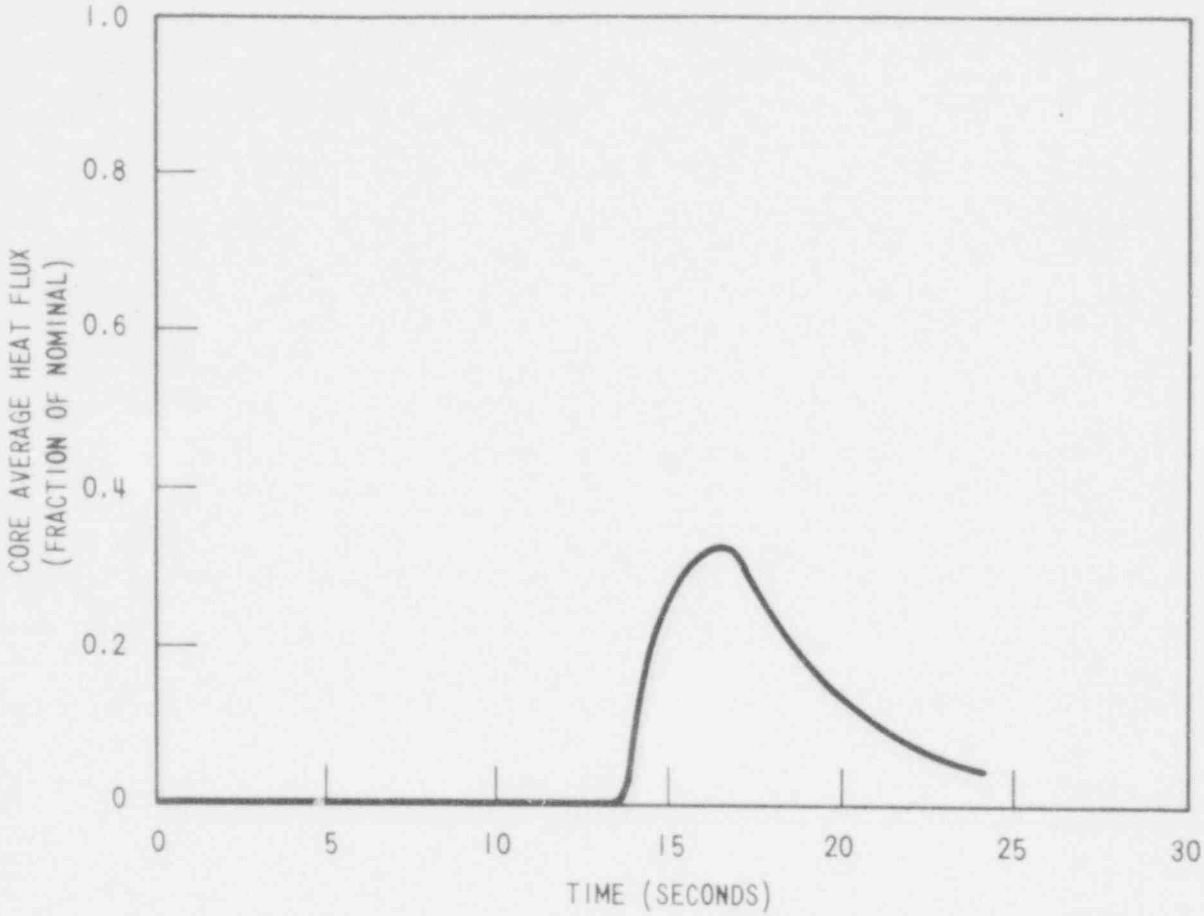

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Figure 15.4.1-2. BLUE

Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

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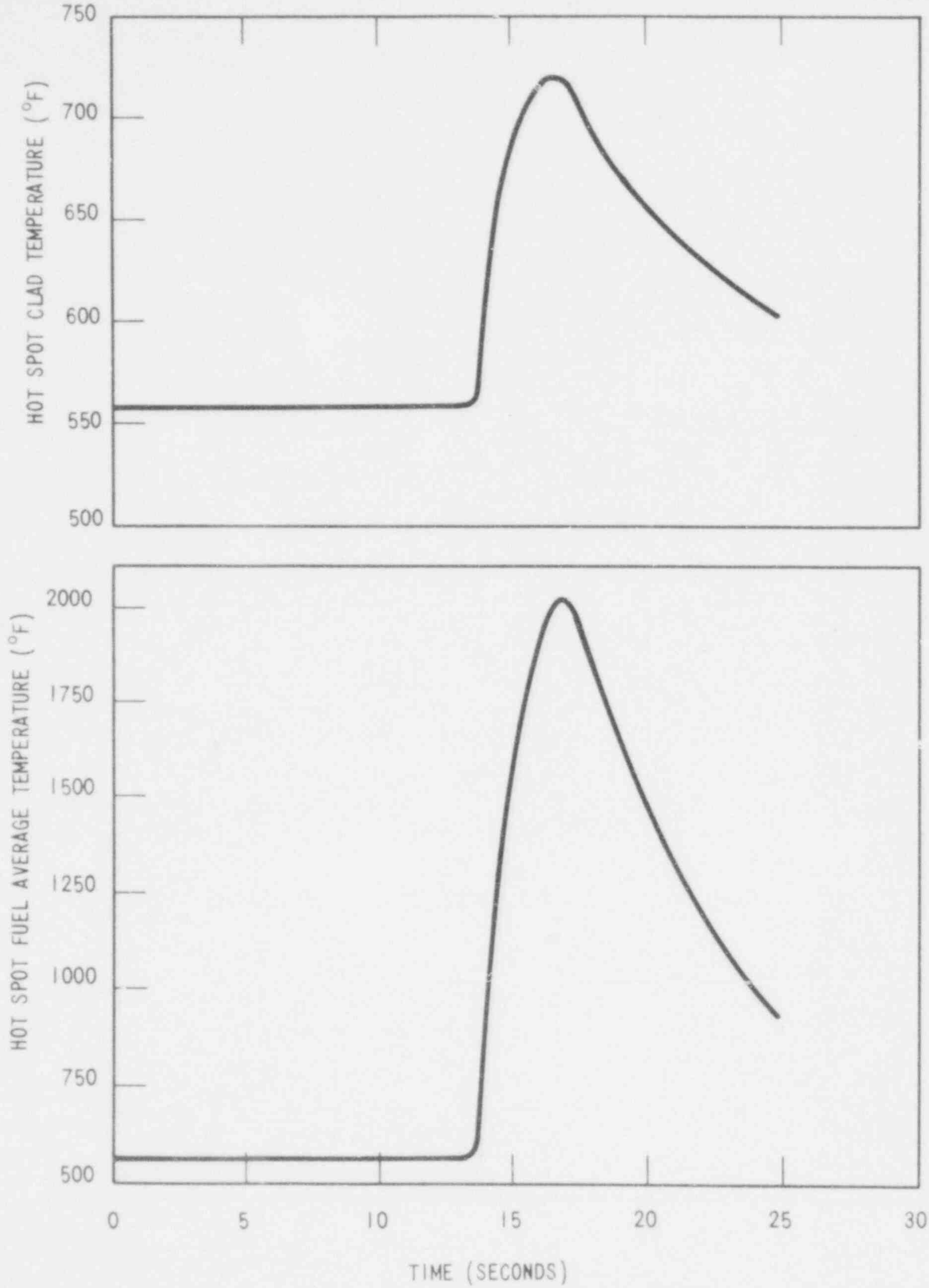

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Figure 15.4.1-3. BLUE

Fuel and Clad Temperature Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition

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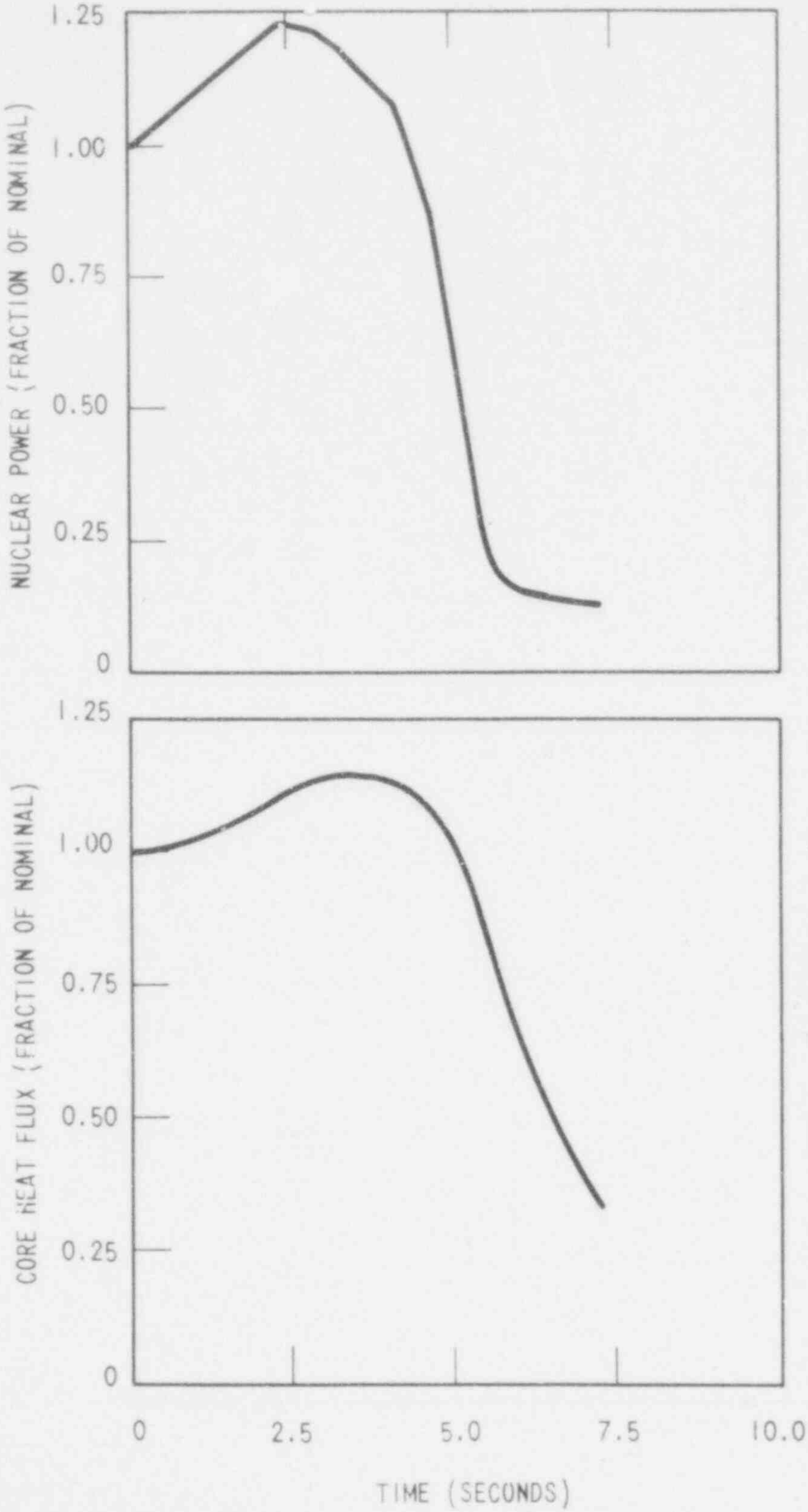

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Figure 15.4.2-1. BLUE

Uncontrolled RCCA Bank Withdrawal from Full Power With Minimum Reactivity Feedback (75 PCM/SEC Withdrawal Rate)

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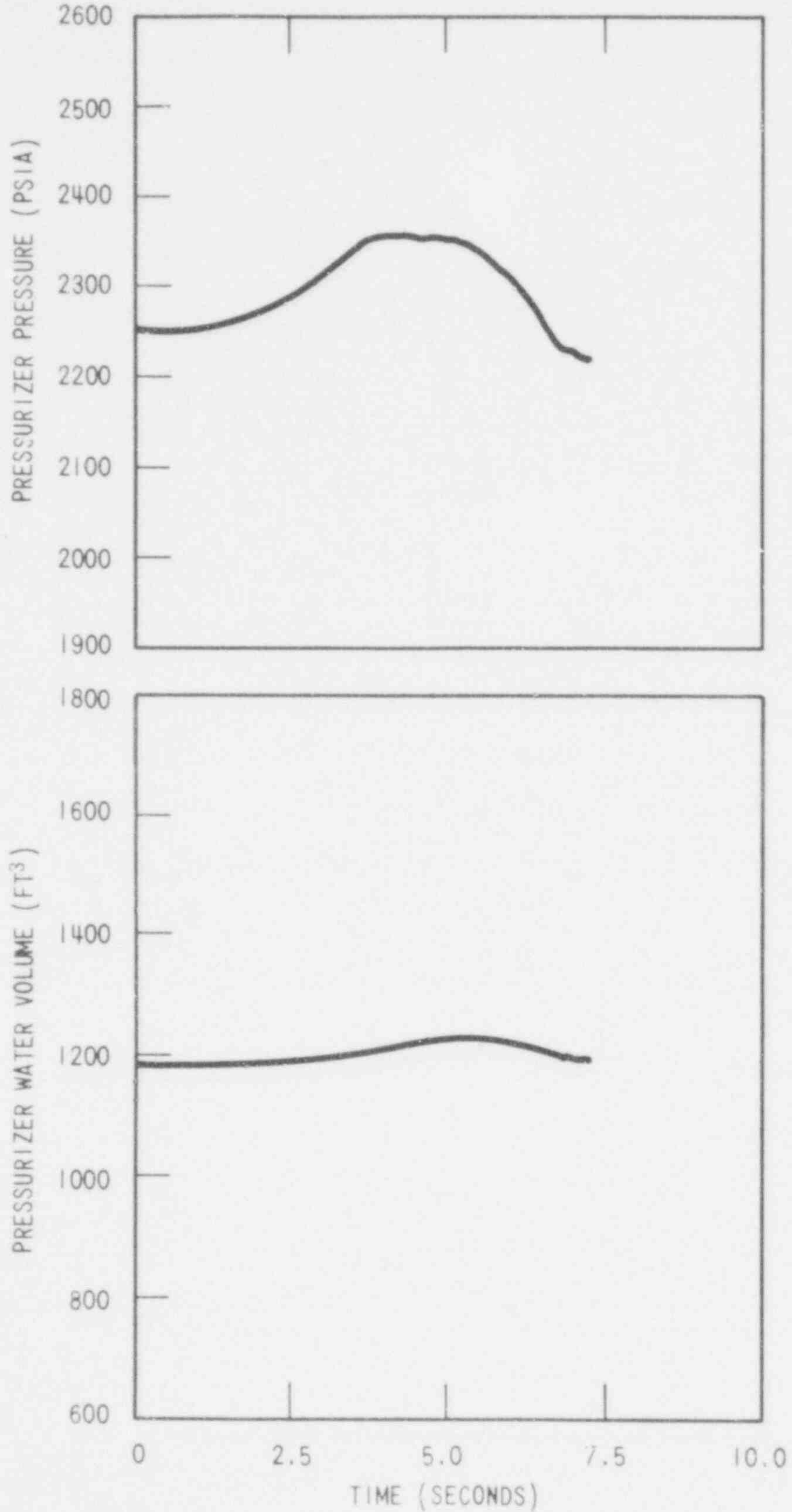

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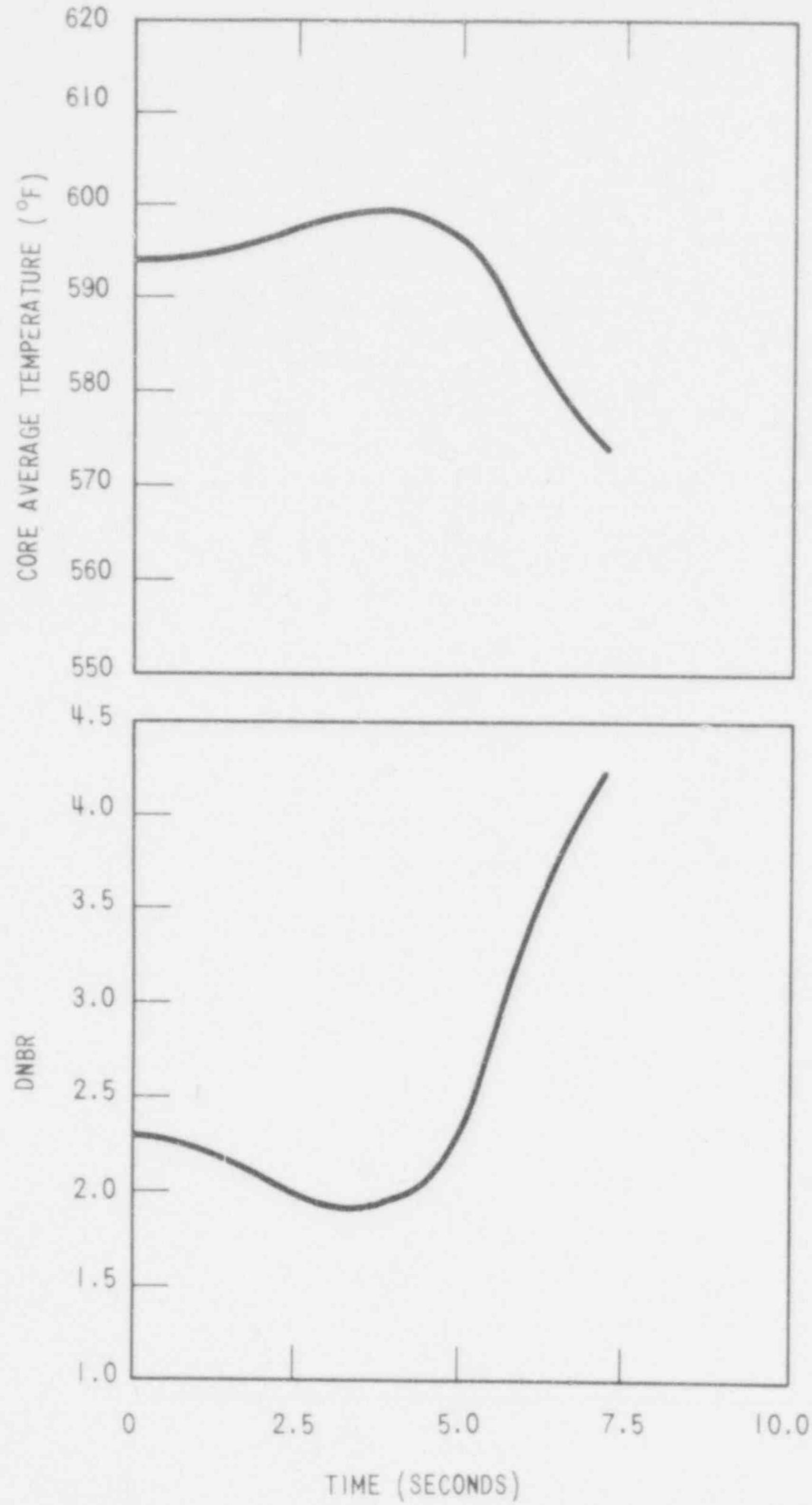
Figure 15.4.2-2. BLUE

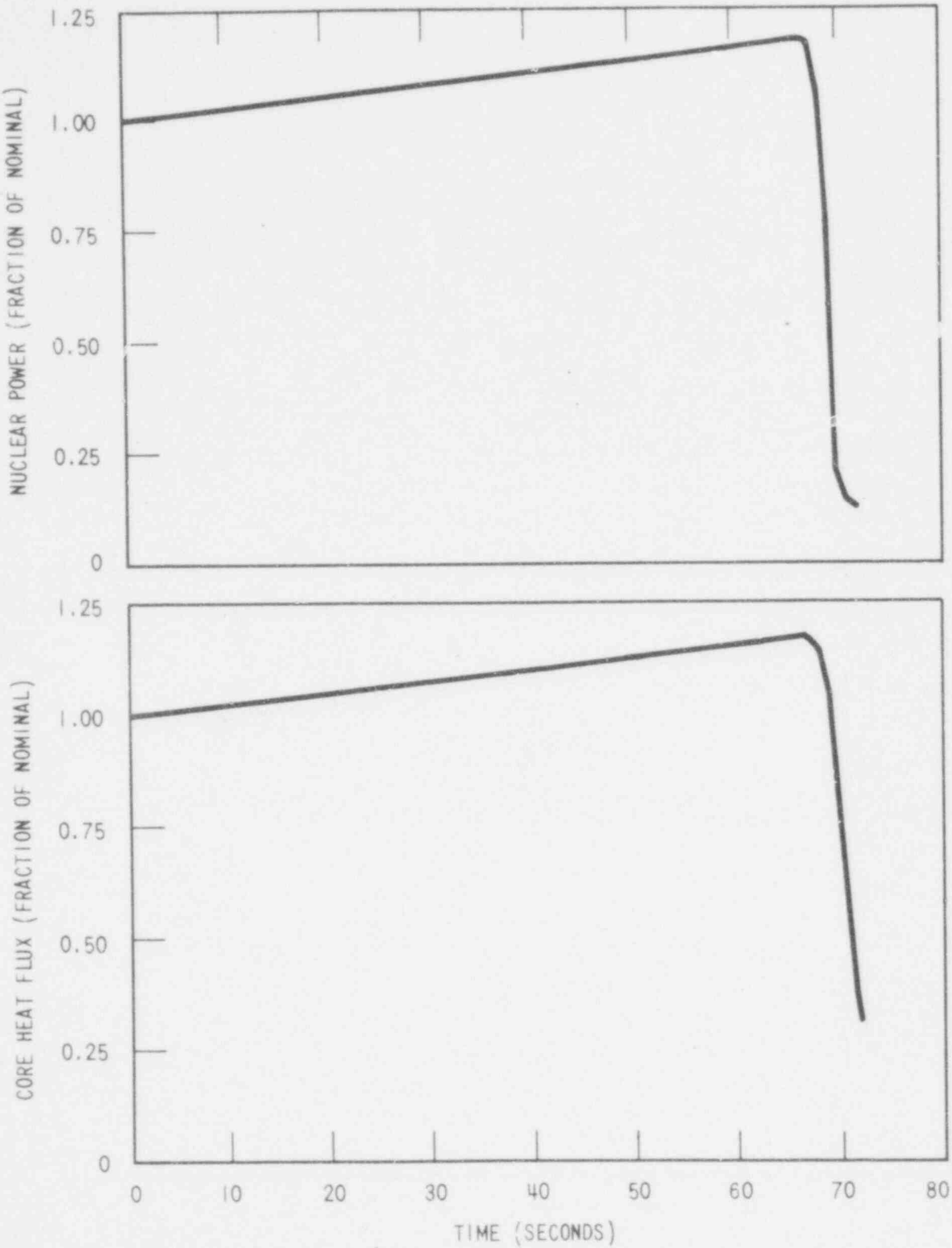
Uncontrolled RCCA Bank Withdrawal from Full Power With Minimum Reactivity Feedback (75 PCM/SEC Withdrawal Rate)



WCAP - 9500

Figure 15.4.2-3. BLUE

Uncontrolled RCCA Bank Withdrawal from Full Power With Minimum Reactivity Feedback (75 PCM/SEC Withdrawal Rate)



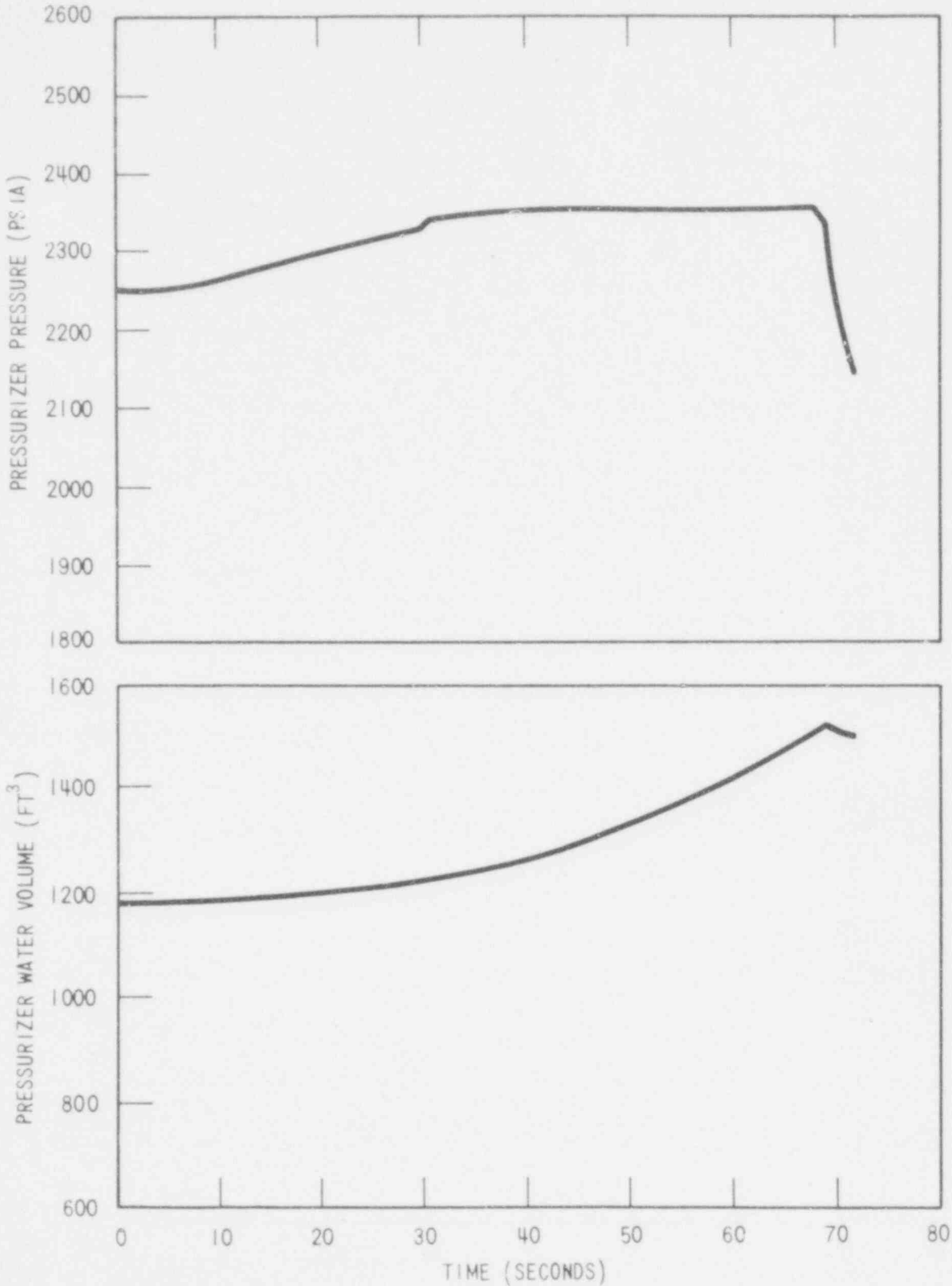
615 076

POOR ORIGINAL

WCAP – 9500

Figure 15.4.2-4. BLUE

Uncontrolled RCCA Bank Withdrawal from Full Power With Minimum Reactivity Feedback (2 PCM/SEC Withdrawal Rate)



POOR ORIGINAL

WCAP – 9500

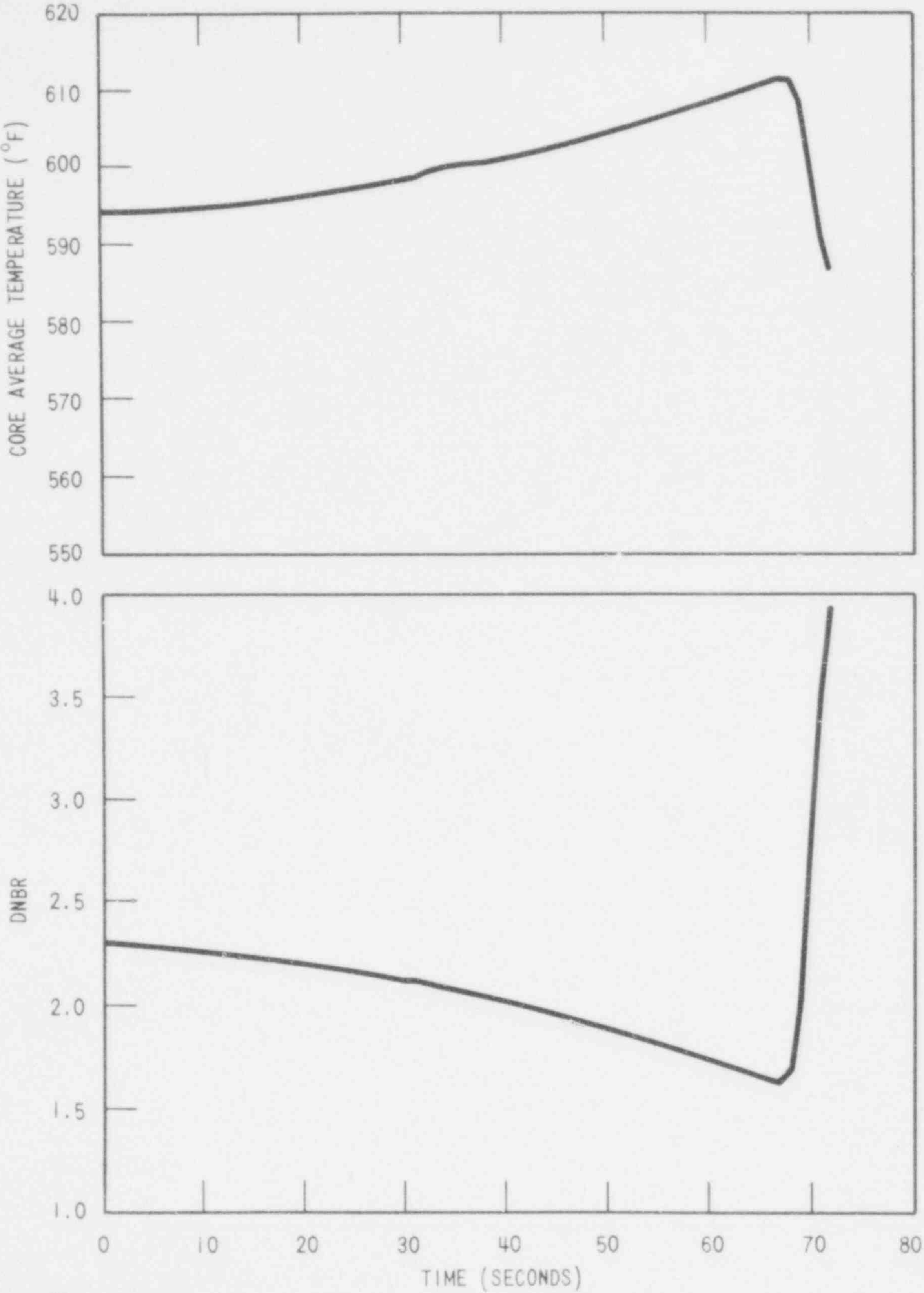
BLUE

Figure 15.4.2-5.

Uncontrolled RCCA Withdrawal from Full Power With Minimum Reactivity Feedback (3 PCM/SEC Withdrawal Rate)

615 077

14,395-289



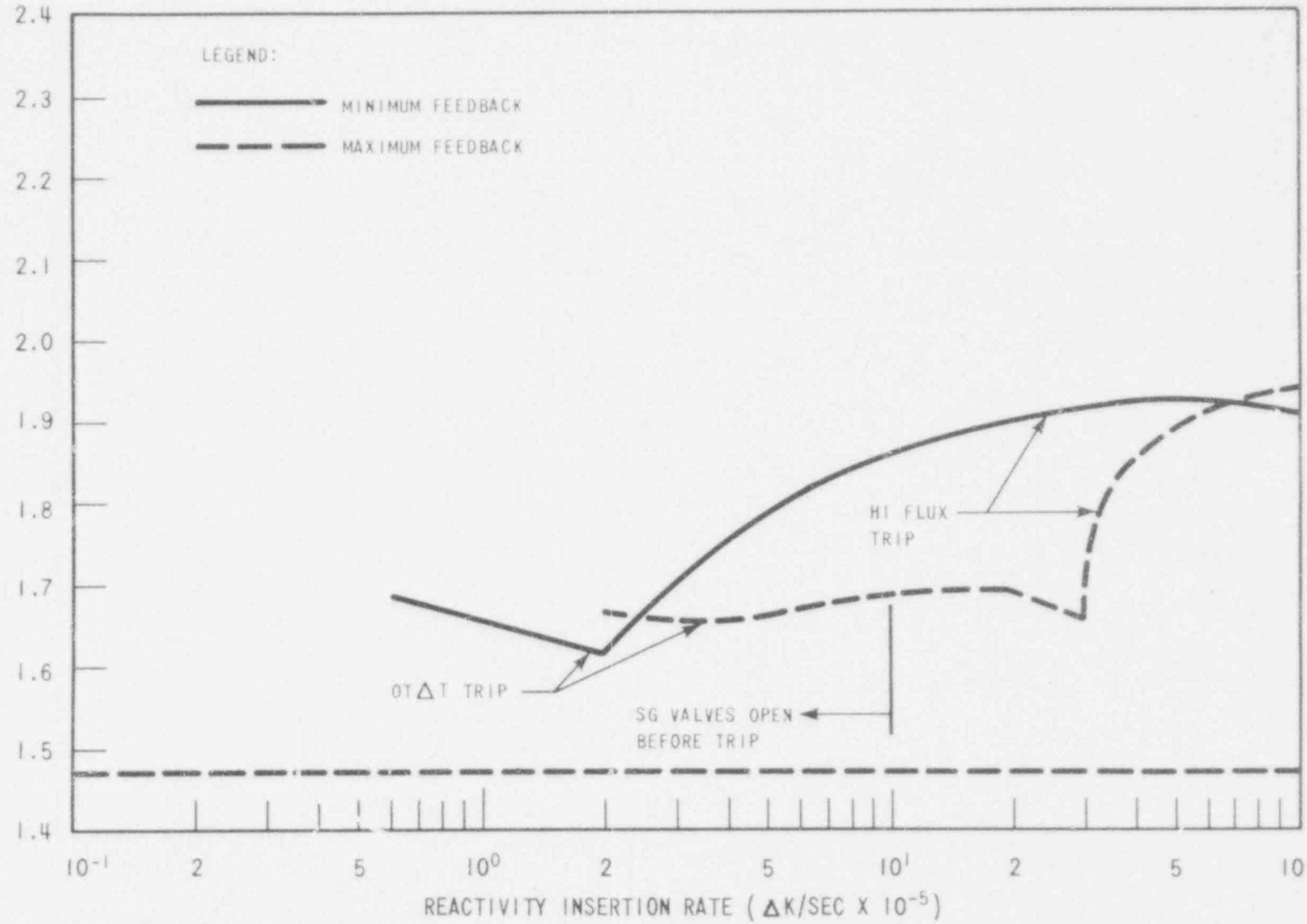
POOR ORIGINAL

WCAP - 9500

Figure 15.4.2-6. BLUE

Uncontrolled RCCA Bank Withdrawal from Full Power With Minimum Reactivity Feedback (2 PCM/SEC Withdrawal Rate)

615 078



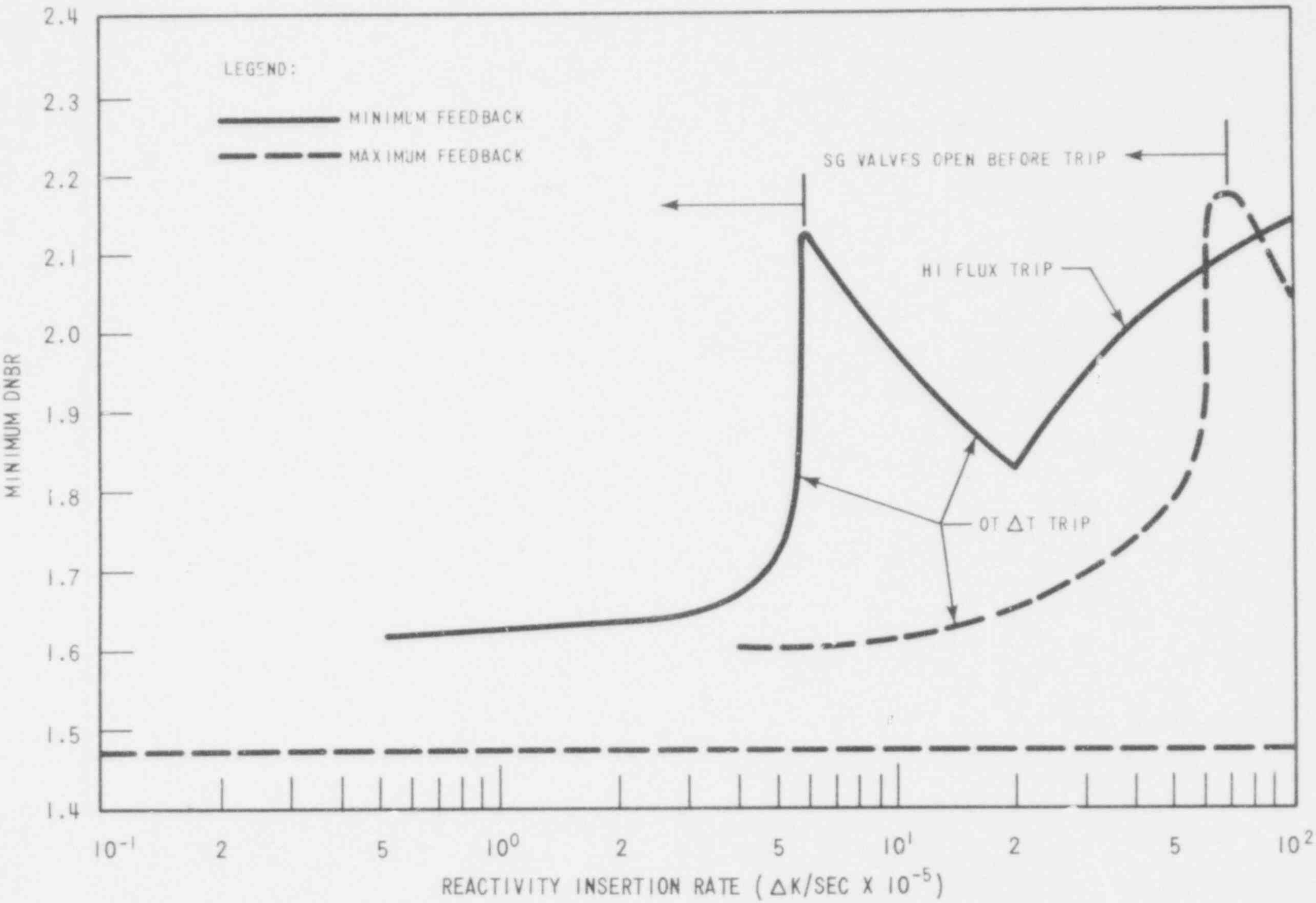
14,395-290

615 079

WCAP-9500

Figure 15.4.2-7. BLUE

Minimum DNBR vs. Reactivity Insertion Rate;
Rod Withdrawal from 100% Power



14,395-291

POOR ORIGINAL

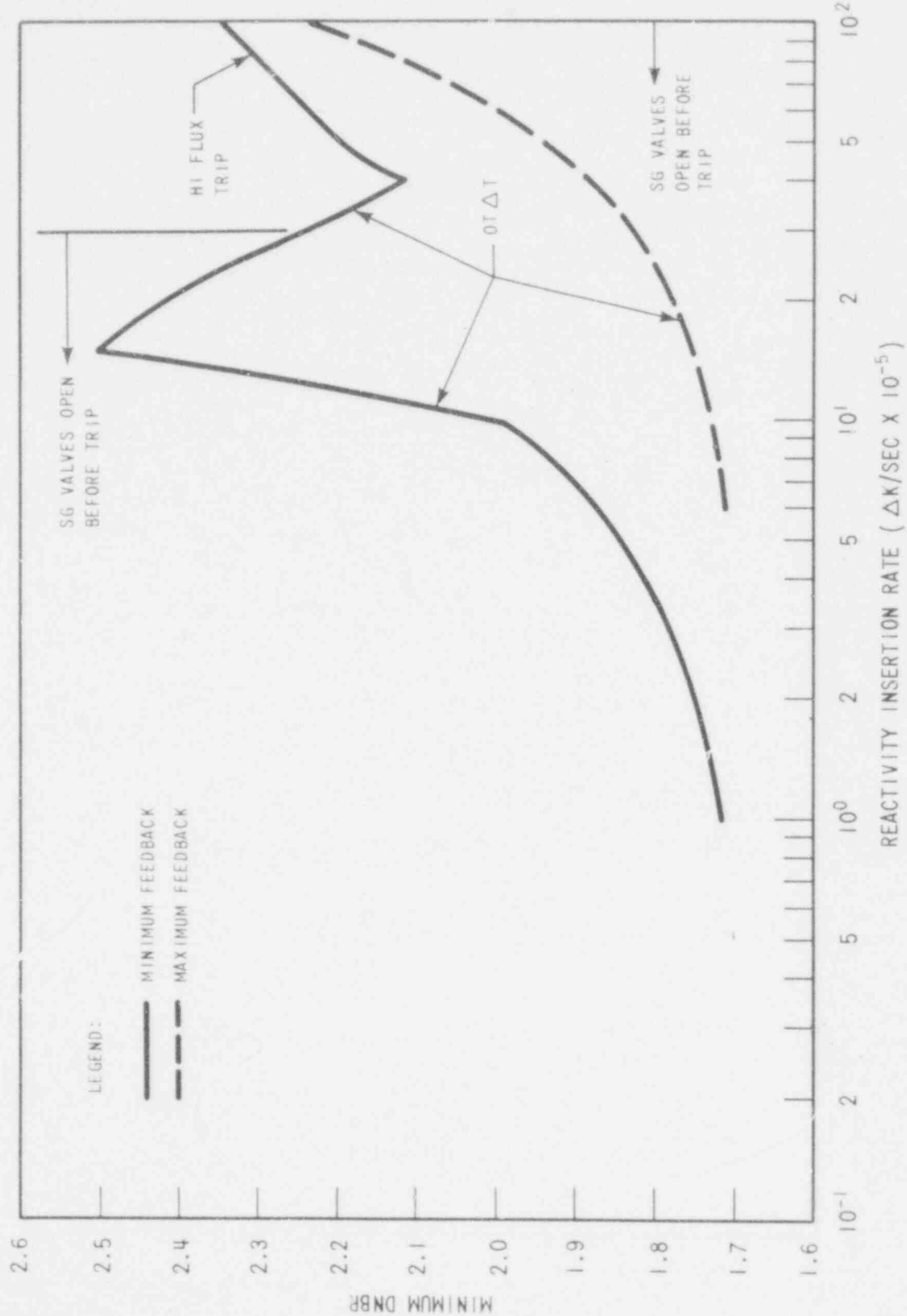
615 080

WCAP-9500

Figure 15.4.2-8. BLUE

Minimum DNBR vs. Reactivity Insertion Rate;
Rod Withdrawal from 60% Power

14,395-292



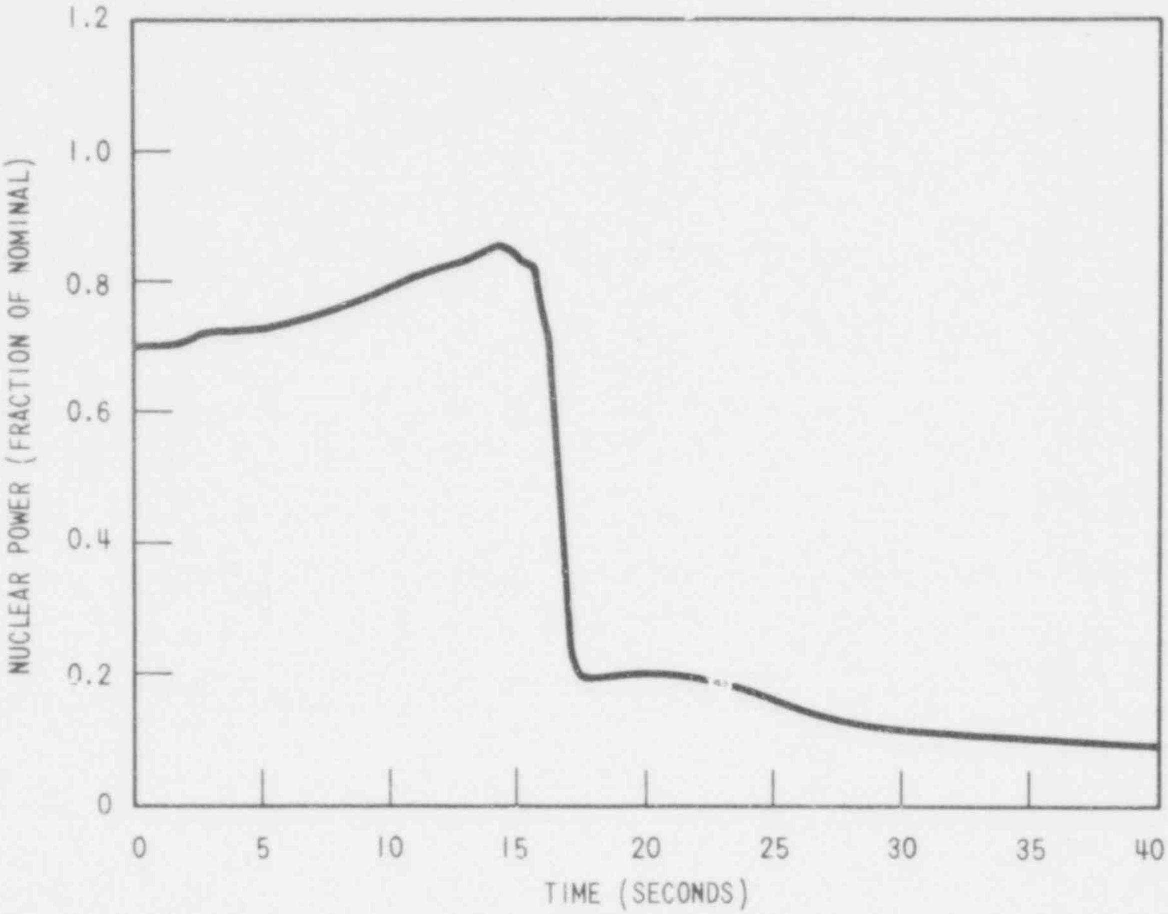
POOR ORIGINAL

WCAP – 9500

Figure 15.4.2-9. BLUE

Minimum DNBR vs. Reactivity Insertion Rate; Rod Withdrawal from 10% Power

615 081



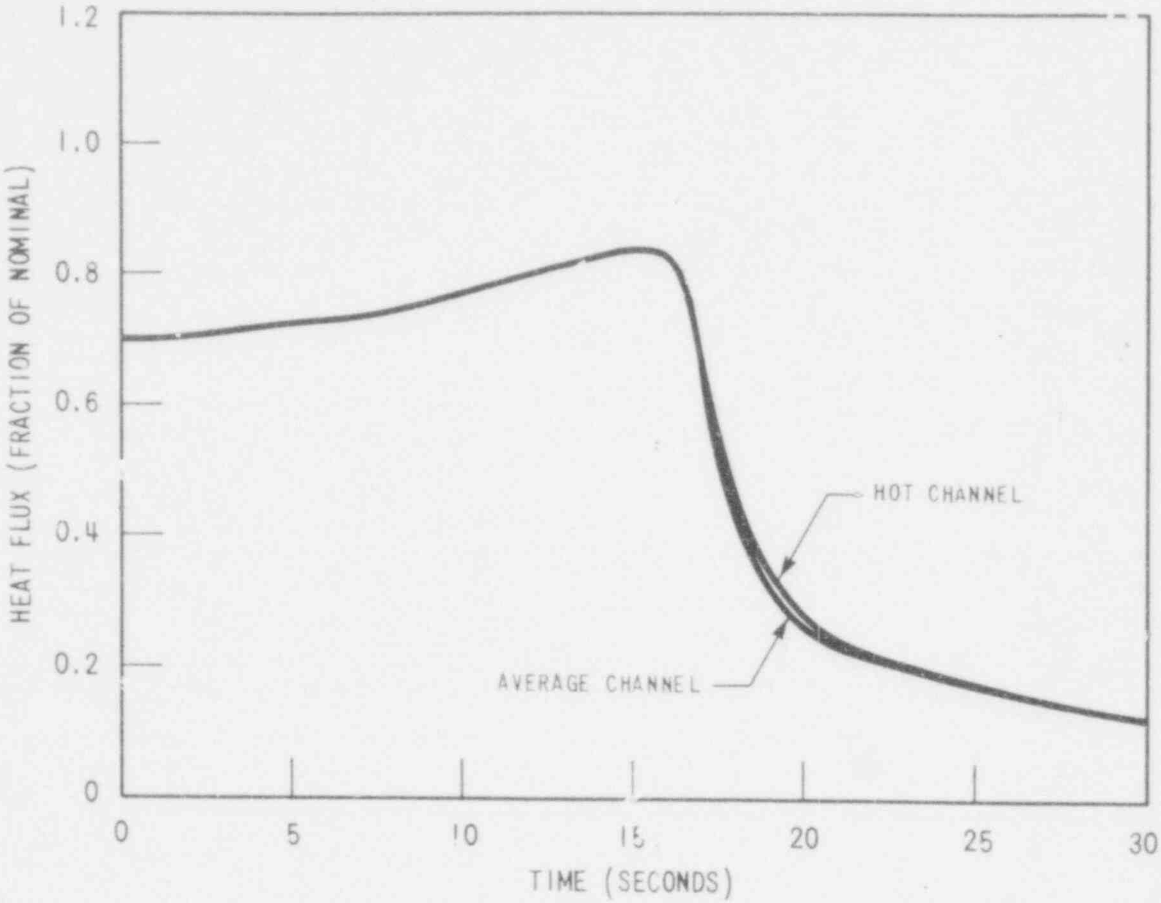
POOR ORIGINAL

WCAP – 9500

Figure 15.4.4-1.

Impropr Startup of an Inactive Reactor Coolant Pump

615 082

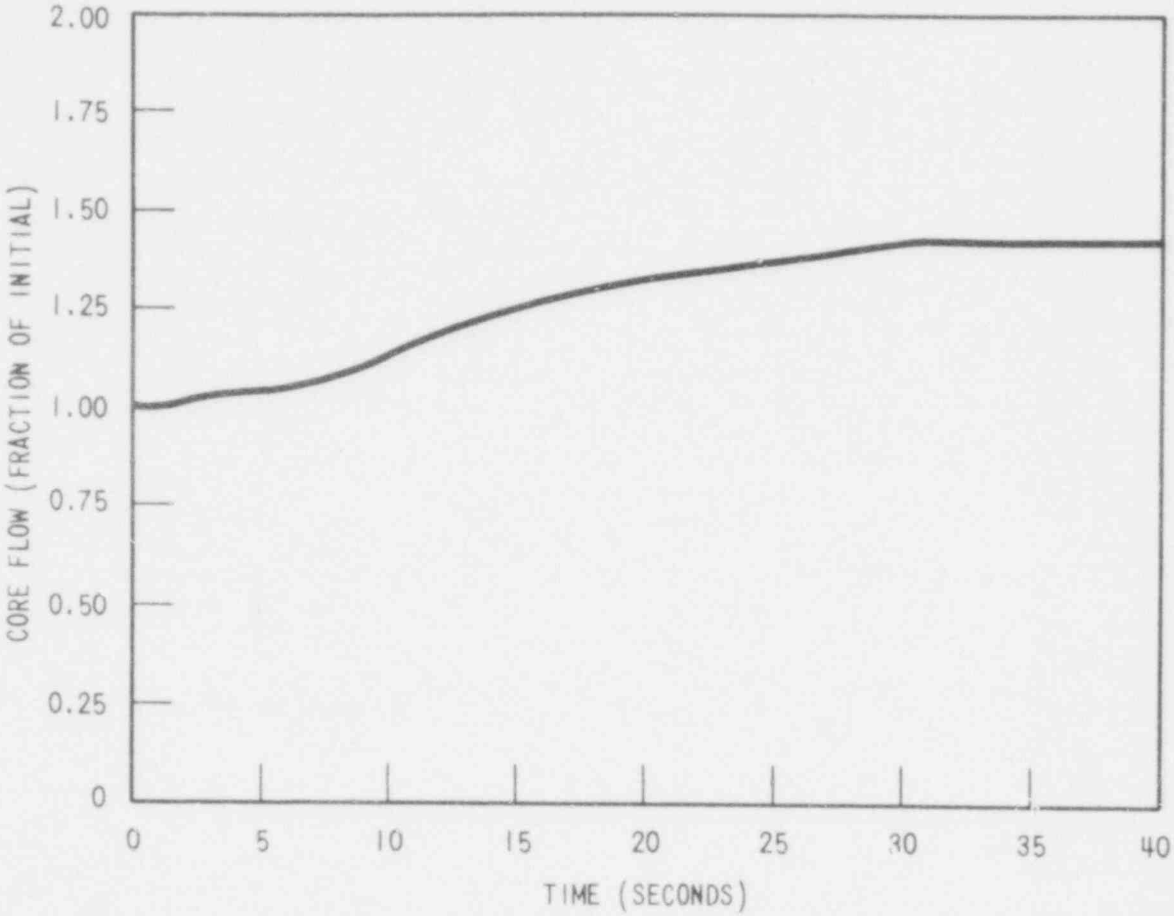


WCAP – 9500

Figure 15.4.4-2. BLUE

Improper Startup of an Inactive Reactor Coolant Pump

615 083



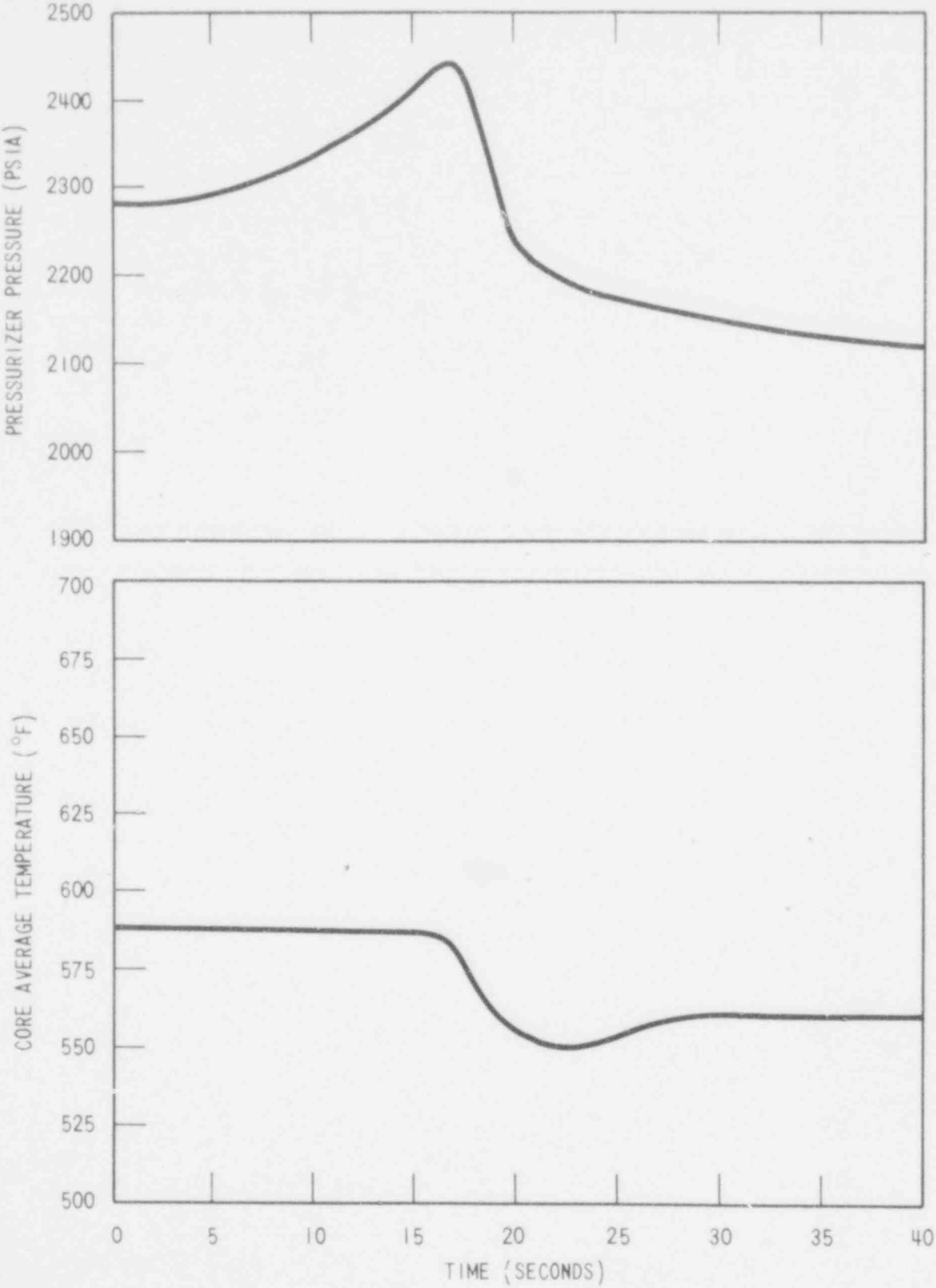
WCAP - 9500

Figure 15.4.4-3. BLUE

Improper Startup of an Inactive Reactor Coolant Pump

615 084

14,395-296



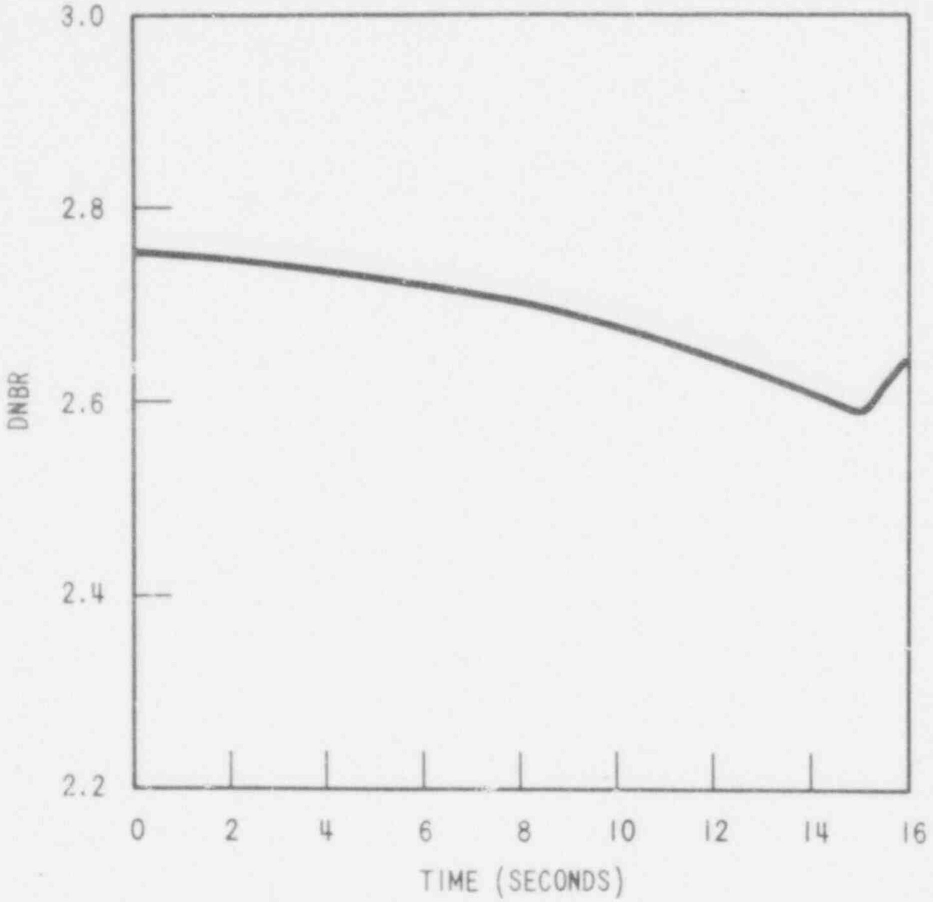
POOR ORIGINAL

WCAP – 9500

Figure 15.4.4-4. BLUE

Improper Startup of an Inactive Reactor Coolant Pump

615 085

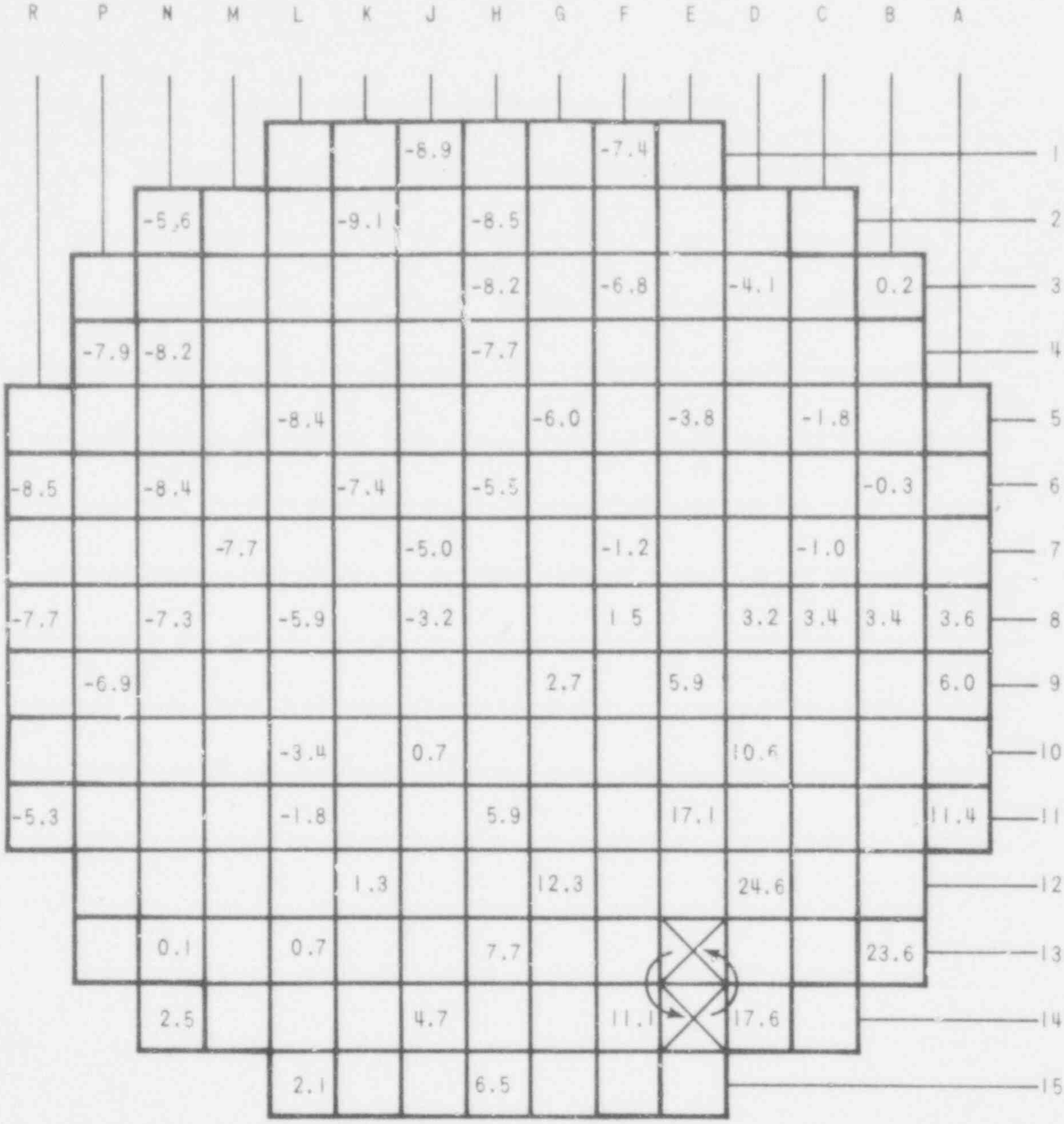


615 080

WCAP – 9500

Figure 15.4.4-5. BLUE

Improper Startup of an Inactive Reactor Coolant Pump



THE NUMBERS REPRESENT DEVIATION
FROM ASSEMBLY AVERAGE POWER

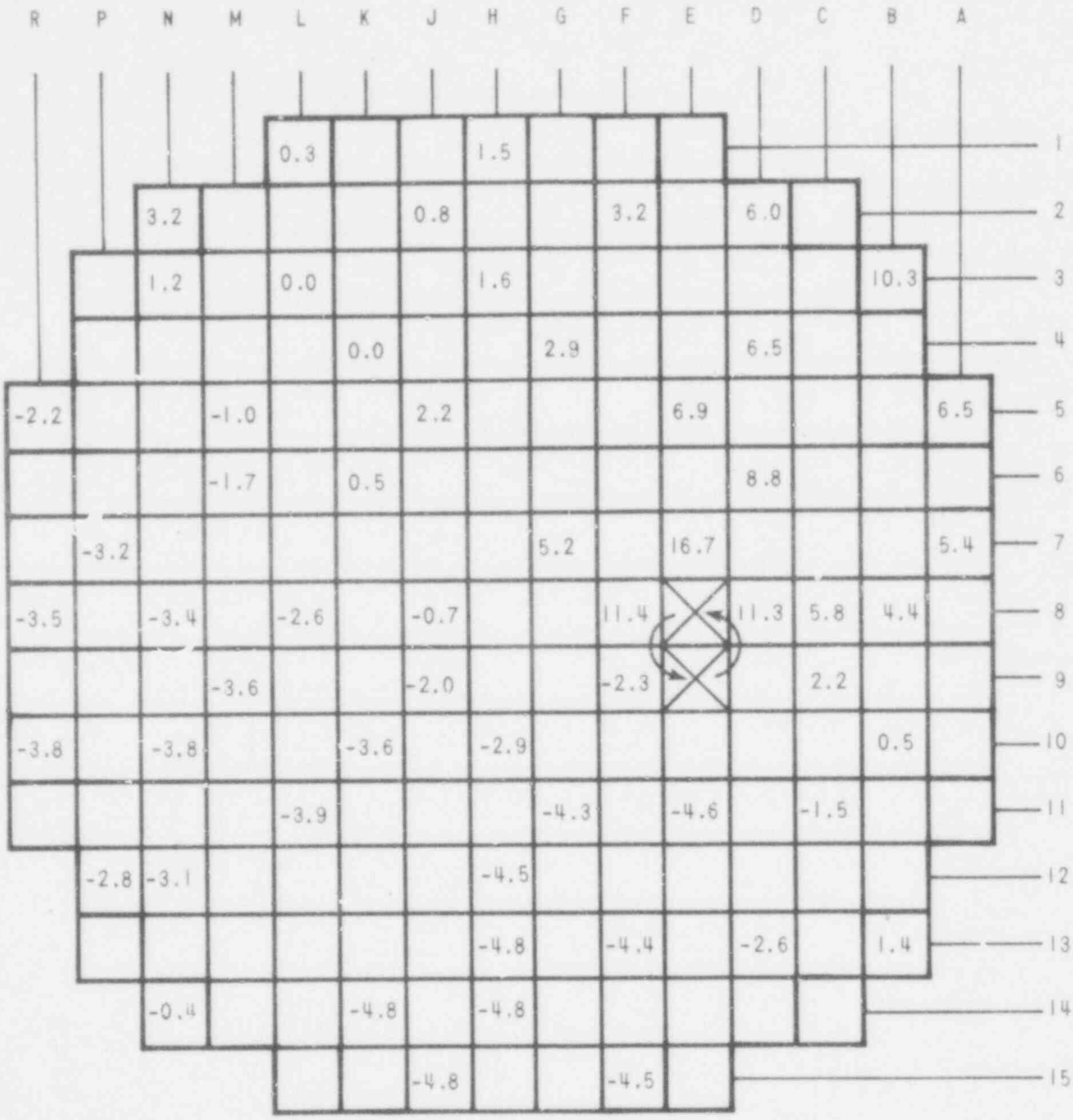
CASE A

615 087

WCAP - 9500

Figure 15.4.7-1. BLUE

Interchange Between Region 1 and Region 3 Assembly



CASE B-1

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

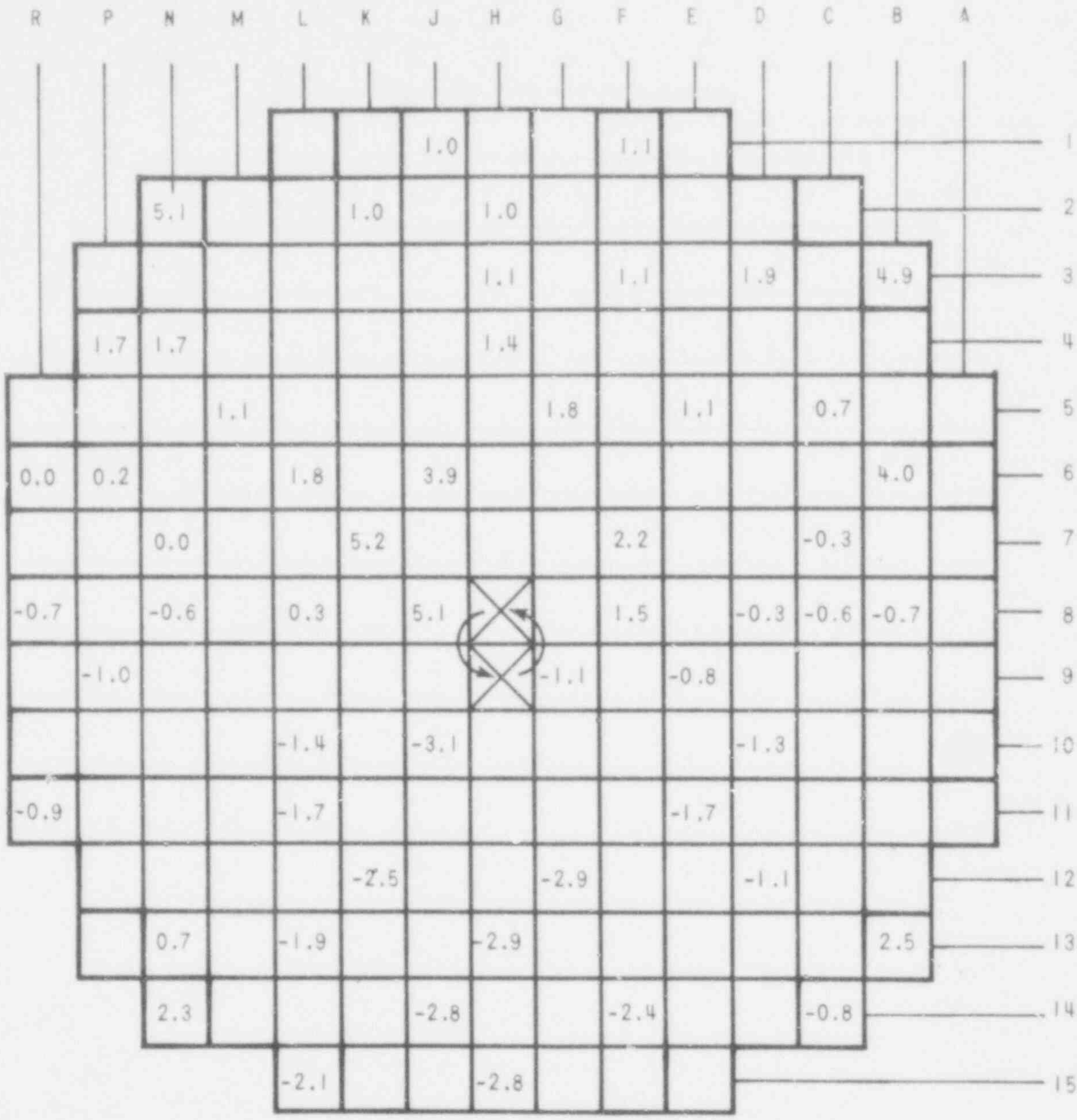
POOR ORIGINAL

WCAP – 9500

Figure 15.4.7-2. BLUE

Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Retained by the Region 2 Assembly

615 088



CASE B-2

THE NUMBERS REPRESENT THE PERCENT DEVIATION
FROM ASSEMBLY AVERAGE POWER

POOR ORIGINAL

WCAP – 9500

Figure 15.4.7-3. BLUE

Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Transferred to the Region 1 Assembly

615 089

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6	X		5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

POOR ORIGINAL

615 090

WCAP – 9500

Figure 15.4.7-4. BLUE

Enrichment Error: A Region 2 Assembly Loaded into the Core Central Position

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11			-14					
2			0.4			-9.2		-12							
3								-12		-14		-15		-13	
4		3.2	1.2					-11							
5					-1.5				-12		-15		-16		
6	9.8		7.1			-1.6		-8.0						-16	
7				9.2			-2.3			-12			-14		
8	20.0		17.8		10.3		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0				23.6			1.9			-8.6				-13
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

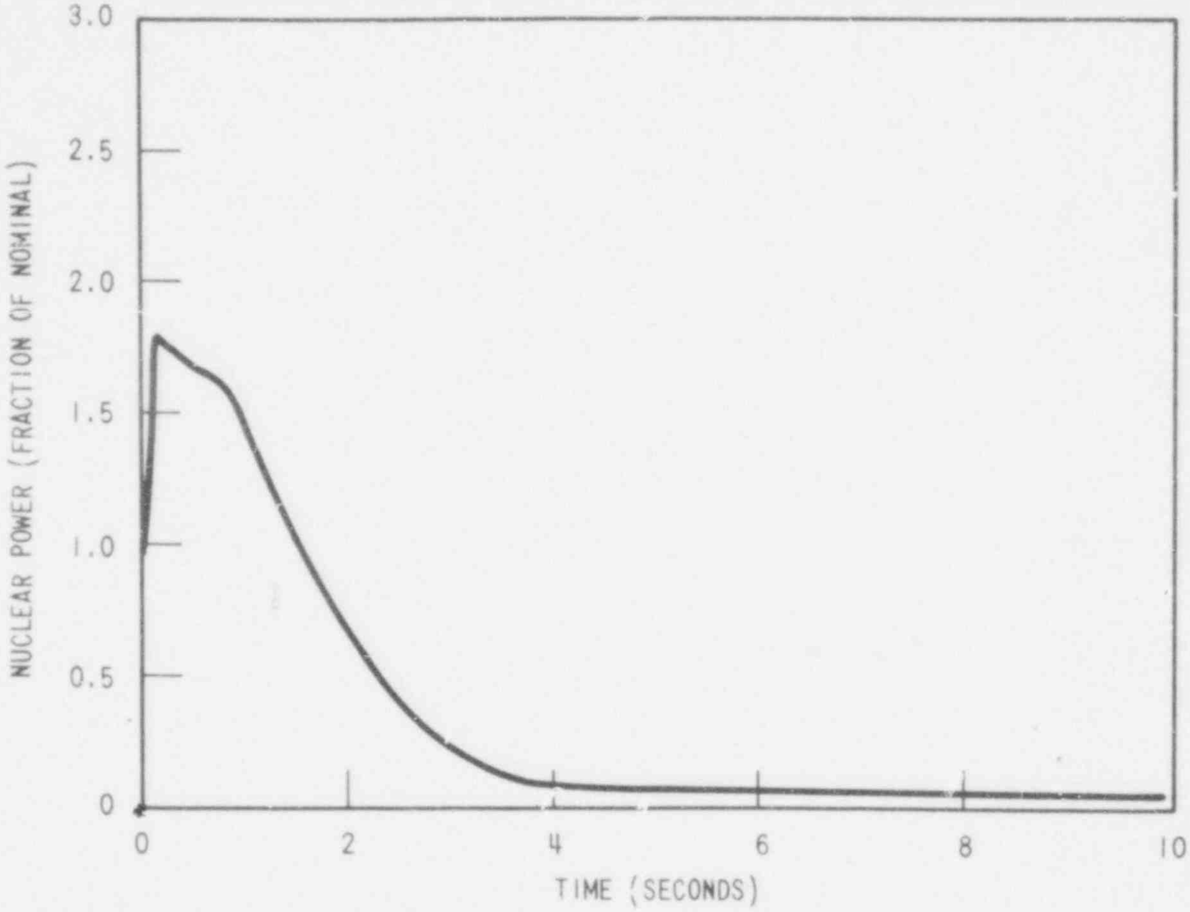
THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

WCAP – 9500

Figure 15.4.7-5. BLUE

Loading a Region 2 Assembly into a Region 1 Position Near Core Periphery

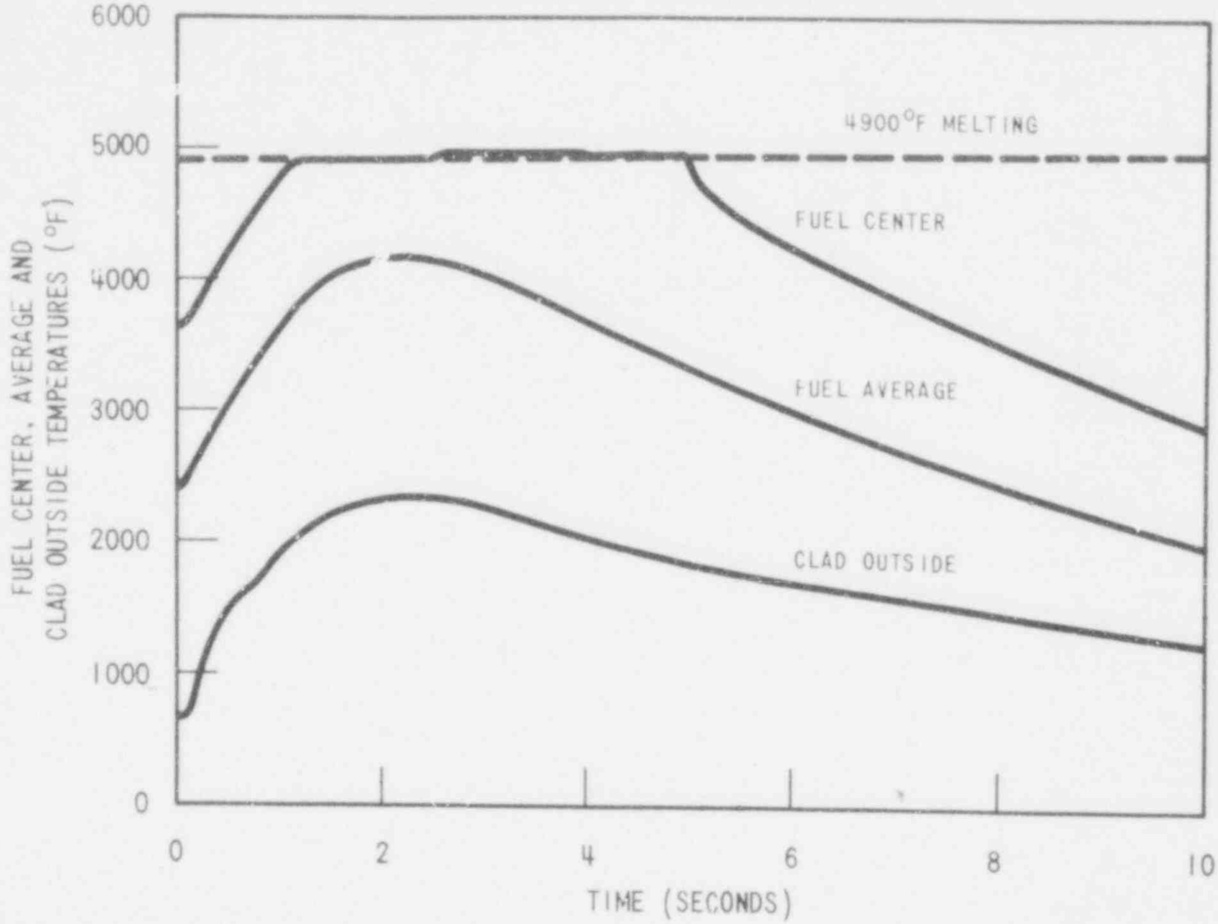
615 091



POOR ORIGINAL

WCAP – 9500
Figure 15.4.8-1. BLUE Nuclear Power Transient, BOL HFP Rod Ejection Accident

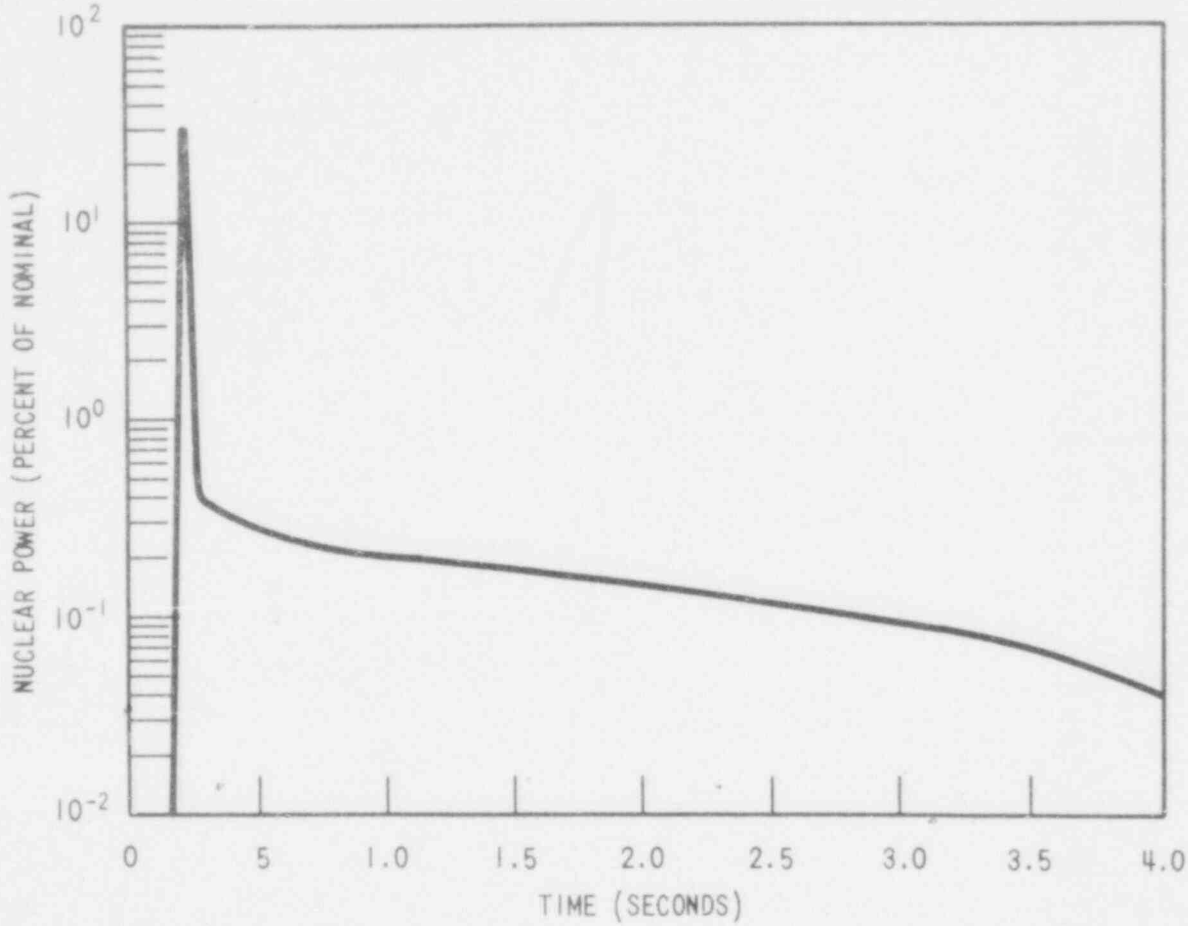
015 092



POOR ORIGINAL

615 093

WCAP – 9500
Figure 15.4.8-2. BLUE Hot Spot Fuel and Clad Temperature vs. Time, BOL HFP Rod Ejection Accident



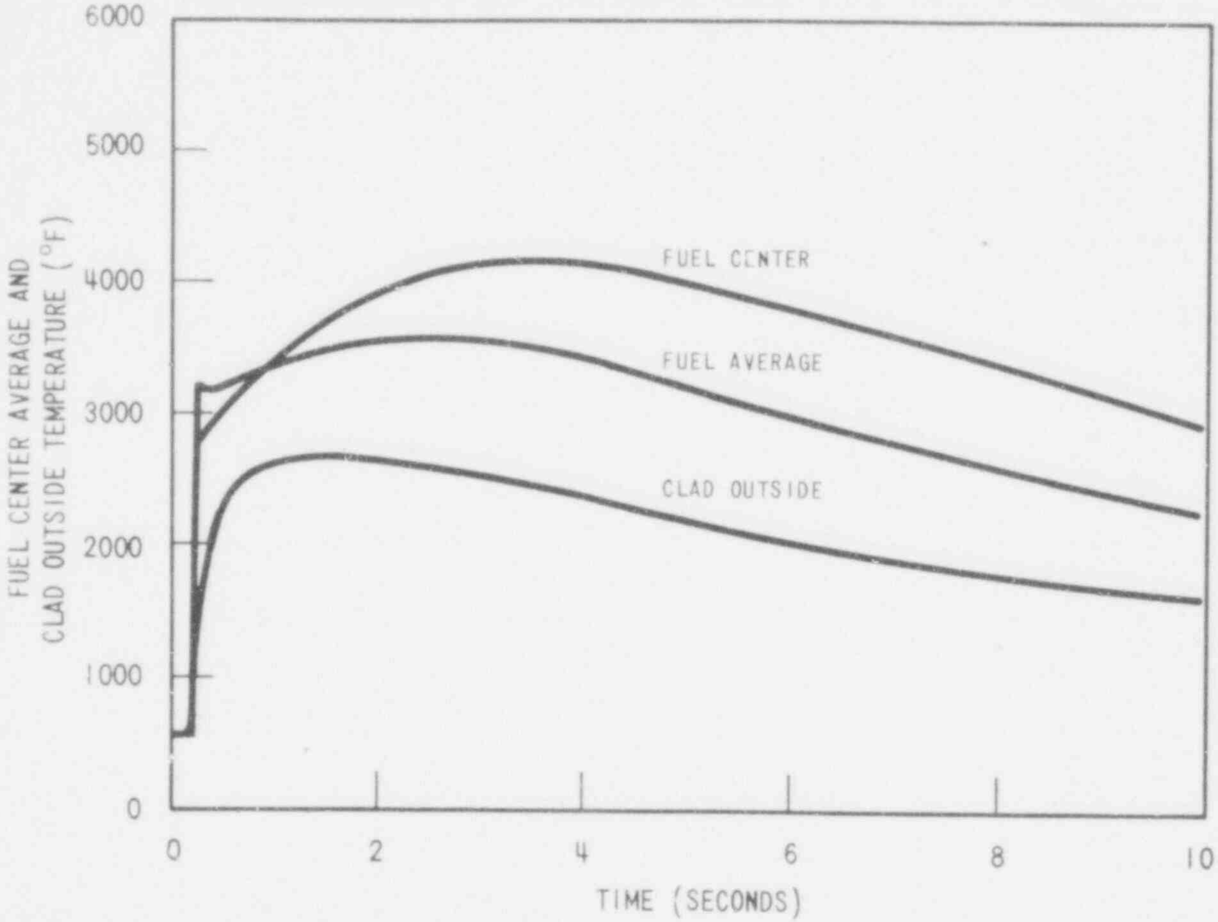
POOR ORIGINAL

615 094

WCAP – 9500

Figure 15.4.8-3. BLUE

Nuclear Power Transient
EOL HZP Rod Ejection Accident



WCAP – 9500

Figure 15.4.8-4. BLUE

Hot Spot Fuel and Clad Temperatures vs. Time, EOL HZP and Rod Ejection Accident

15.5

POOR ORIGINAL

615 096

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

1. Inadvertent Operation of Emergency Core Cooling System During Power Operation

2. Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

3. A number of BWR Transients (Not applicable)

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Section 15.0.1 contains a discussion of ANS classifications.

15.5.1 INADVERTENT OPERATION OF EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the boron injection tank from the charging pumps and the valves isolating the boron injection tank from the injection header then automatically open. The charging pumps then force highly concentrated (20,000 ppm) boric acid solution from the boron injection tank, through the header and injection line and into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the Reactor Coolant System is at normal pressure. The passive injection system and the low head system also provide no flow at normal Reactor Coolant System pressure.

A Safety Injection System (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal, the operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

If the Reactor Protection System does not produce an immediate trip as a result of the spurious SIS signal, the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. Pressurizer pressure and water level drop. Load will decrease due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the Reactor Protection System low pressure trip or by manual trip.

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

Recovery from this second case is made in the same manner as described for the case where the SIS signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. The time at which reactor trip occurs is of little concern for this transient. At lower loads coolant contraction will be slower resulting in a longer time to trip.

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

15.5.1.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the Safety Injection System is analyzed by employing the detailed digital computer program LOFTRAN (Reference 1). The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the Safety Injection System. The program computes pertinent plant variables including temperatures, pressures, and power level.

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

A typical transient is presented representing minimum reactivity feedback. Results with maximum reactivity feedback are similar except that the transient is slower. For calculational simplicity, zero injection line purge volume was assumed in this analysis, thus the boration transient begins immediately when the appropriate valves are opened.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3.

The major assumptions are as follows:

1. Initial Operating Conditions

   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 WCAP 8567.

2. Moderator and Doppler Coefficients of Reactivity

A least negative moderator temperature coefficient was used. A low (absolute value) Doppler power coefficient was assumed.

3. Reactor Control

The reactor was assumed to be in manual control.

4. Pressurizer Heaters

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

5. Boron Injection

At time zero two charging pumps inject 20,000 ppm borated water into the cold leg of each loop.

6. Turbine Load

Turbine load was assumed constant until the governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

7. Reactor Trip

Reactor trip was initiated by low pressurizer pressure.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0.8-1. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.5.1-1 through 15.5.1-3 show the transient response to inadvertent operation of ECCS during power operation. Neutron flux starts decreasing immediately due to boron injection but steam flow does not decrease until later in the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. When the low pressure trip setpoint is reached, the reactor trips and control rods start moving into the core. DNBR increases throughout the transient.

The calculated sequence of events is shown on Table 15.5.1-1. After reactor trip, pressure and temperature slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. Recovery from this accident is discussed in Section 15.5.1.1.

15.5.1.3 Radiological Consequences

There are minimal radiological consequences associated with inadvertent ECCS operation. If the SIS signal results in a reactor trip, the reactor trip causes a turbine trip and heat is removed from the secondary system through the steam generator power relief valves or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences associated with atmosphere steam release from this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

15.5.1.4 Conclusions

Results of the analysis show that spurious safety injection without immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System.

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DNB ratio is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the Reactor Coolant System.

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

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

An increase in reactor coolant inventory which results from the addition of cold, unborated water to the Reactor Coolant System is analyzed in Section 15.4.6, Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant. An increase in reactor coolant inventory which results from the injection of highly borated water into the Reactor Coolant System is analyzed in Section 15.5.1, Inadvertent Operation Emergency Core Cooling System during Power Operation.

15.5.3 A NUMBER OF BWR TRANSIENTS

(Not applicable)

15.5.4 REFERENCES

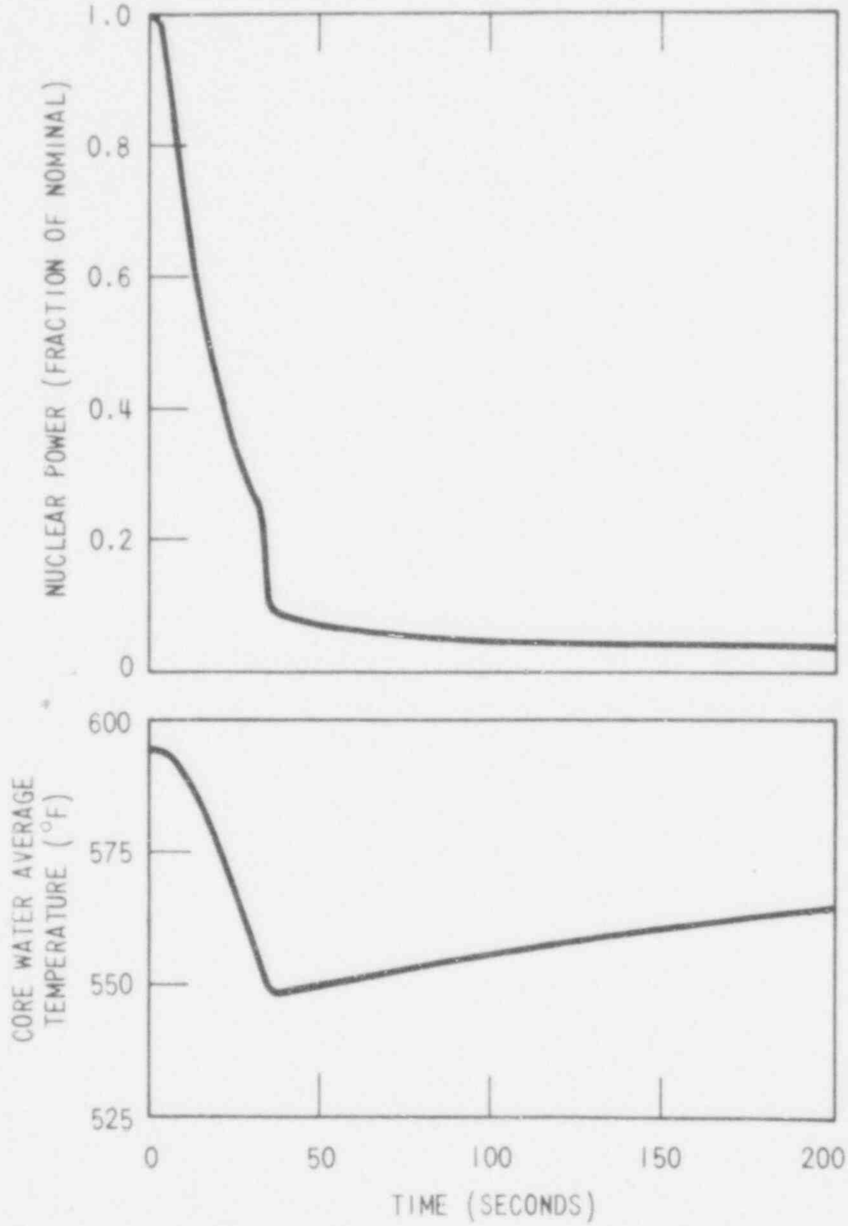
1. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907, June 1972.

615 102

TABLE 15.5.1-1

TIME SEQUENCE OF EVENTS FOR INCREASE IN REACTOR COOLANT INVENTORY EVENTS

Accident	Event	Time (Sec.)
Inadvertent Actuation of ECCS During Power Operation	Spurious SI signal generated; two Charging Pumps begin injecting borated water	0
	Turbine throttle valve wide open, load begins to drop with steam pressure	21
	Low pressurizer pressure reactor trip setpoint reached	28
	Control Rod Motion Begins	30



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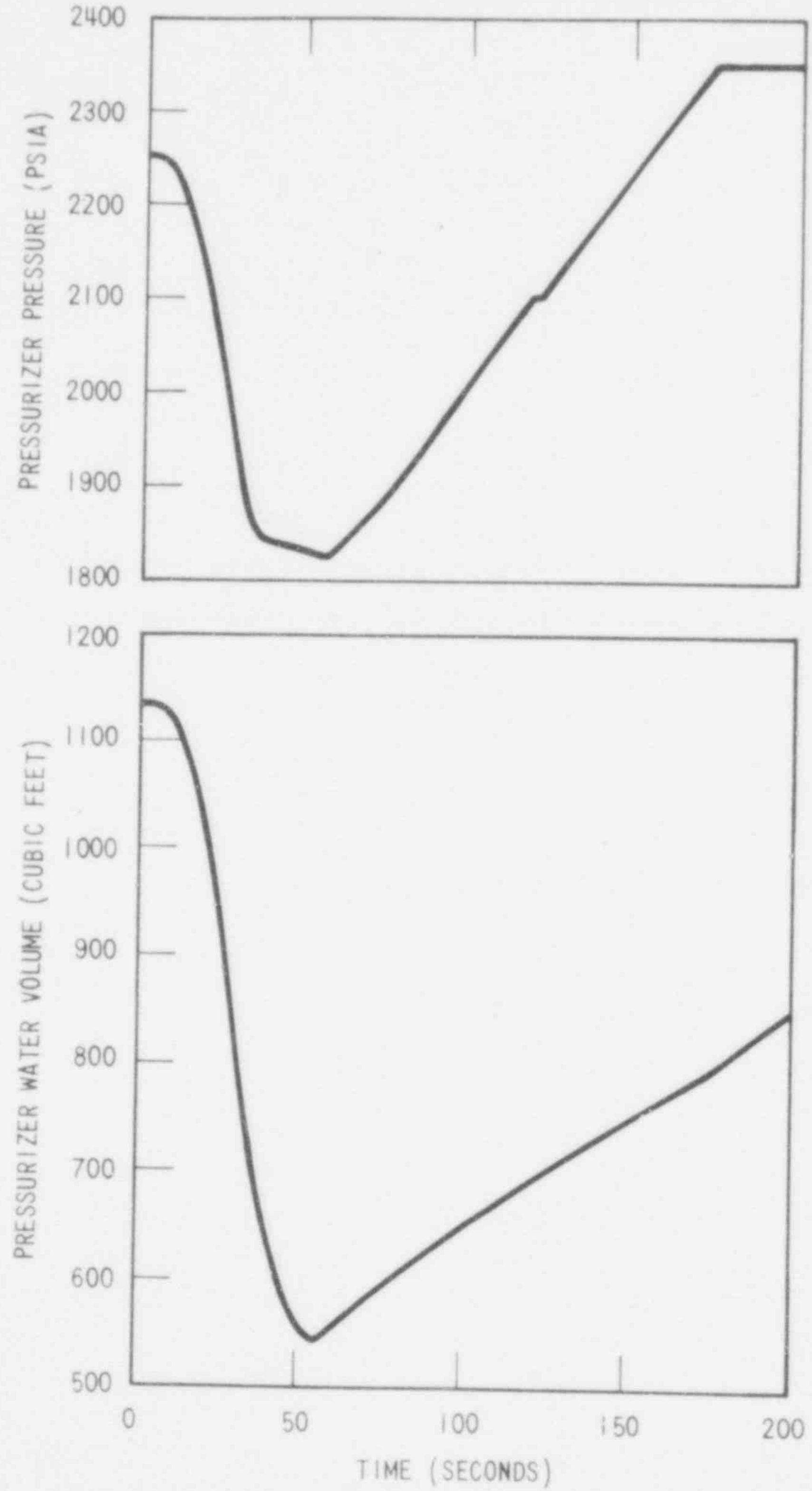
WCAP – 9500

Figure 15.5.1-1. BLUE

Inadvertent Actuation of ECCS During Power Operation

14,395-303



POOR ORIGINAL

WCAP – 9500

Figure 15.5.1-2. BLUE

Inadvertent Operation of ECCS
During Power Operation

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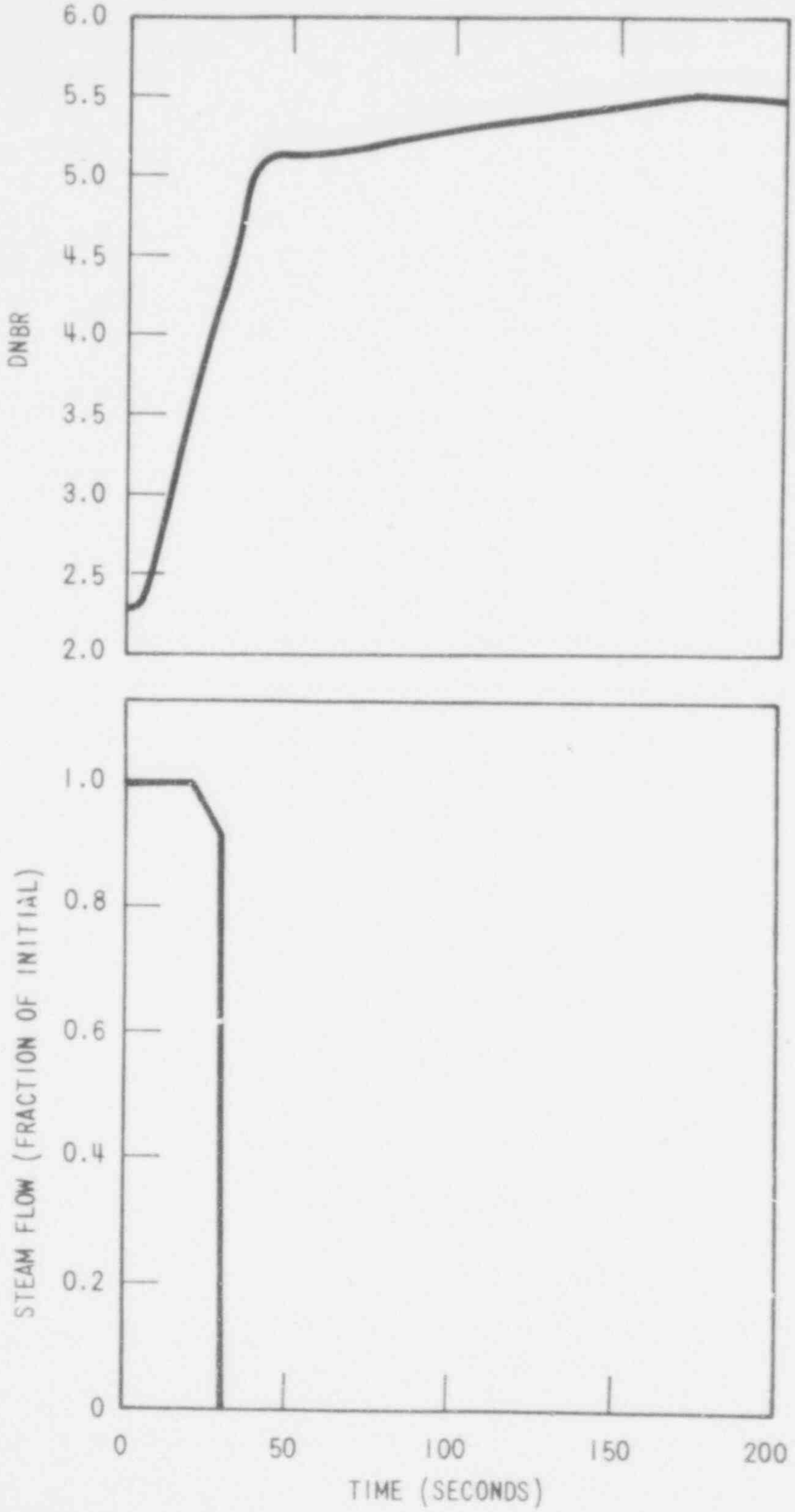

POOR ORIGINAL

WCAP - 9500

Figure 15.5.1-3. BLUE

Inadvertent Operation of ECCS During Power Operation

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15.6

POOR ORIGINAL

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15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory as discussed in this section are as follows:

1. Inadvertent opening of a pressurizer safety or relief valve.

2. Failure of small lines carrying primary coolant outside containment.

3. Steam Generator Tube Rupture.

4. BWR Piping failure outside containment (Not applicable).

5. Loss-of-Coolant Accident resulting from a spectrum of postulated piping breaks within the Reactor Coolant Pressure Boundary.

6. A number of BWR transients (Not applicable).

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY OR RELIEF VALVE

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow-rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the Reactor Coolant System are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing Reactor Coolant System pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At this time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain

POOR ORIGINAL

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BLUE

the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following Reactor Protection System signals:

1. Overtemperature ΔT

2. Pressurizer low pressure

An inadvertent opening of a pressurizer safety valve is classified as an ANS Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN (Reference 1). The code simulates the neutron kinetics, Reactor Coolant System, 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.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Section 15.0.3.

In order to give conservative results in calculating the DNBR during the transient, the following assumptions are made:

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 WCAP 8567.

2. A least negative moderator coefficient of reactivity is assumed. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

3. A large (absolute value) Doppler coefficient of reactivity such that the resultant amount of positive feedback is conservatively high in order to retart any power decrease due to moderator reactivity feedback.

Plant systems and equipment which are necessary to mitigate the effects of a Reactor Coolant System depressurization caused by an inadvertent safety valve opening are discussed in Section 15.0.8 and listed in Table 15.0.8-1.

Normal reactor control systems are not required to function; however, the rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst-case assumption; if the reactor were in manual control, an earlier trip could occur on low pressurizer pressure. The Reactor Protection System functions to trip the reactor on the appropriate signal. No single active failure will prevent the Reactor Protection System from functioning properly.

Results

The system response to an inadvertent opening of a pressurizer safety or relief valve is shown on Figures 15.6.1-1 and 15.6.1-2. Figure 15.6.1-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The pressure decay transient and average temperature transient following the accident are given in Figure 15.6.1-2. Pressure drops more rapidly while core heat generation is reduced via the trip, and would then slow once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6.1-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6.1-1.

15.6.1.3 Radiological Consequences

An inadvertent opening of a pressurizer safety or relief valve releases primary coolant to the pressurizer relief tank; however, even assuming a direct release to the containment atmosphere, the radiological consequences of this event would be substantially less than that of a LOCA (Subsection 15.6.5) because less primary coolant is released and the activity is lower as fuel damage is not predicted as a result of this event.

15.6.1.4 Conclusions

The results of the analysis show that the pressurizer low pressure and the overtemperature ΔT Reactor Protection System signals provide adequate protection against the RCS depressurization event. No fuel or clad damage is predicted for this accident. The radiological consequences of this event would be substantially less than that of the LOCA analyzed in Subsection 15.6.5.

15.6.2 FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

15.6.2.1 Identification of Causes and Accident Description

The accident results from a break in small lines such as a sample line connected to the primary coolant system and penetrating the containment. Ruptures of small cross-sectional lines will cause explusion of the coolant at a rate which can be accommodated by a charging pump which would maintain an operational water level in the pressurizer, permitting the operator to conduct an orderly shutdown. The release contains the radionuclide concentration of the primary coolant.

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The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system (RCS) through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is adequate to sustain pressurizer level and a pressure of 2250 psia for a break through a 0.375 inch dimater hole. This break results in a loss of approximately 17.5 lb/sec, and, due to the use of a 0.245 inch restriction, is the maximum flow available for all reactor coolant sample line breaks outside of the containment. In addition, all such lines meet the requirements of General Design Criterion 55 of Appendix A 10 CFR 50. There are no instrument lines which pass through the containment and connect directly to the RCS. A failure of a small line carrying primary coolant outside containment is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.2 for a discussion of Condition II events.

15.6.2.2 Analysis of Effects and Consequences

Since this event does not result in a leakage rate greater than the capacity of a charging pump and pressurizer level does not decrease, normal shutdown procedures can be employed. There are no significant consequences to the reactor or its essential auxiliary systems.

15.6.2.3 Radiological Consequences

There could be moderate radioactive releases from the failure of a small line carrying primary coolant outside containment. This accident will be evaluated in the Applicant's SAR. The primary coolant activity that would be used in the small line break analysis is 60 μCi of dose equivalent I-131 resulting from a preexisting iodine spike.

15.6.3 STEAM GENERATOR TUBE RUPTURE

There is a large effort currently underway to examine several aspects of the steam generator tube rupture analysis, specifically:

1. Revisions to emergency operating procedures, including E3, for steam generator tube rupture and

2. Corresponding modifications to analytical models.

As a result the analytical results and accompanying text will be provided in an amendment to this document.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable.

15.6.5 LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 square foot (ft^2). This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, in that it is an infrequent fault which may occur during the life of the plant.

The Acceptance Criteria for the LOCA is described in 10CFR50.46 as follows:

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1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

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

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

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

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

These criteria were established to provide signficant margin in Emergency Core Cooling System (ECCS) performance following a LOCA.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

15.6.5.2 Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the LOCA analysis for boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

b. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of Large Break LOCA Transient

The sequence of events following a large break LOCA are presented in Figure 15.6.5-1.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel continues to be transferred to the reactor coolant. 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. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction depending on the ralative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the Auxiliary Feedwater System. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the Reactor Coolant System pressure falls below approximately 1250 psia the upper head injection accumulators begin to inject borated water directly into the reactor upper head region. This water is directed from the upper head directly to all but 8 peripheral assemblies in the core via the RCC guide tubes and UHI support columns. This flow provides additional core cooling during the blowdown phase of the transient. A detailed description of the interactions of UHI water and those effects on the blowdown and subsequent reflood transients is given in Reference [3].

When the RCS depressurizers to 400 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the bypassing of emergency core cooling water injected into the RCS are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated From the later stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process. Pumped safety injection flows are provided in Table 15.6.5-6.

Continued operation of the ECCS pumps supplies water during long term cooling. Core temperatures have been reduced to long term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the engineered safety features sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The Containment Spray System continues to operate to further reduce containment pressure. Approximately 24 hours after initiation of the LOCA the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small Break LOCA Transient

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

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.25 lb/sec.

Should a larger break occur. depressurization of the Reactor Coolant System causes fluid to flow to the Reactor Coolant System from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences o. accident are limited in two ways:

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

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

Before the break occurs the plant is in an equlibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between The Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressure. When the Reactor Coolant System depressurizes to the upper head accumulator setpoint pressure, the upper head accumulator begins injecting borated water into the reactor vessel upper head. A description of the operation of upper head injection during small break transients can be found in WCAP 8479 Rev. 2 (Reference 2).

When the RCS depressurizes to 400 psia, the cold leg accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initialization of the accident and effects of pump coastdown are included in the blowdown analyses.

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15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10CFR50.

Large Break LOCA Evaluation Model

The analysis of a large break LOCA Transient is divided into three phases: 1) blowdown, 2) refill, and 3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the Containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in References [2]or[3]. 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 differences between the approved non-UHI Westinghouse Appendix K Model nd the model used for these analyses are reported in WCAP-8479, Revision 2 (Reference 2). The thermal analyses reported in this section were performed with an upper head fluid temperature of T_{cold}. The UHI accumulator pressure setpoint ensures UHI activation prior to upper head fluid flashing. The SATAN-VI, POWLOCTA, WREFLOOD, LOTIC, and LOCTA-IV codes which are used in the LOCA analysis are described in detail in References [2] through [7]. These 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 SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and refill while the POWLOCTA calculates the average channel axial temperature distribution during this period of the transient. The WREFLOOD

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computer code is used to calculate the thermal-hydraulic transient during the reflood phases of the accident. The LOTIC computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod duirng the three phases.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary sytems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown and refill phases, these data are transferred to the WREFLOOD code. Also at the end-of-blowdown and refill phases, the mass and energy release rates during blowdown are transferred to the LOTIC code for use in the determination of the containment pressure response during these phases of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI and POWLOCTA codes, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate, the coolant pressure and temperature, and the quench front height during the reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the Containment through the break. 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 transient pressure computed by the LOTIC code is input to the WREFLOOD code. For the analyses presented in this report, the containment pressure was conservatively assumed to be 0.0 psig. Except for the 1.0 DECLG perfect and imperfect mixing breaks which utilize a constant backpressure conservatively assumed to he 0.5 psig. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel cladding temperature and metal-water reaction of the hottest rod in the core. Dynamic steam cooling is included in the LOCTA-IV calculation as described in Reference [8].

Schematic representation of the computer code interfaces is given in Figure 15.6.5-2.

The large break analysis was performed with the NRC approved Westinghouse UHI ECCS Evaluation Model (Reference 2).

Small Break LOCA Evaluation Model

The WFLASH program used in the analysis of the small break LOCA is an extension of the FLASH-4 code [9] developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the RCS.

The RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied through the system. A detailed description of WFLASH is given in Reference [10].

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular it enables a proper calculation of the behavior of the loop seal during a loss of coolant transient.

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Clad thermal analyses are performed with the LOCTA-IV code (Reference 7) which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history from the WFLASH hydraulic calculations as input.

15.6.5.3.2 <u>Input Parameters and Initial Conditions</u>

Table 15.6.5-1 lists important input parameters and initial conditions used in the analysis.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (refer to Reference 2). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a signficant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K, 10CFR50.46.

The worst break (C_D = 0.6) was run with a variation in UHI accumulate volume delivery for the perfect (1020 ft^3) and imperfect mixing case (860 ft^3) assumptions. The delivered volume considered in the analysis encompasses the volume delivery band (156 ft^3) associated with UHI delivery uncertainties at a 95 percent probability level.

Cases presented herein provide the results of a conservative application of this range of values. In addition, UHI volume deliveries for each case presented herein will differ somewhat due to variation in UHI flowrate during the time of isolation valve closure and be dependent on discharge coefficient assumed.

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The imperfect mixing case was analyzed to develop a low delivery volume since the upper head drains earlier in the transient and subsequently voids the lower plenum and core, thereby representing a conservative case. The imperfect mixing case was also run at a higher pressure (1300) than the perfect mixing case (1200) to allow for a ±50 psi, uncertainty in accumulator setpoint pressure. Similarly, the high pressure for the imperfect mixing case represents the most conservative case since the smaller accumulator volume would be delivered in a shorter amount of time and earlier in the blowdown transient, thereby providing for a longer core heatup time.

15.6.5.3.3 Results

Large Break Results

Based on the results of the LOCA sensitivity studies, (Reference 2) the limiting large break was found to be the double ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.6.5-2 through 15.6.5-5.

Figures 15.6.5-3 through 15.6.5-77 present the parameters of principal interest from the large break ECCS analyses. For all cases analyzed transients of the following parameters are presented:

Figures 15.6.5-3 through Figures 15.6.5-17	The following quantities are presented at the clad burst location and at the hot spot (location of maximum clad temperature) both on the hottest fuel rod (hot rod):

1. fluid quality
2. mass velocity
3. heat transfer coefficient.

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The heat transfer coefficient shown is calculated by the LOCTA IV code.

Figures 15.6.5-18 through Figures 15.6.5-42 — The system pressure shown is the calculated pressure in the core. Core flowrates, and core void fraction are also presented.

Figures 15.6.5-43 through Figures 15.6.5-52 — These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The nodal notation of the figures is defined in Table 15.6.5-7.

Figures 15.6.5-53 through Figures 15.6.5-62 — These figures show the core reflood transient.

Figures 15.6.5-63 through Figures 15.6.5-72 — These figures show the Emergency Core Cooling System flowrates for all cases analyzed. Both UHI and cold leg accumulators are included in the figures. As described earlier the cold leg accumulator delivery during blowdown is discarded until the end of bypass is calculated. Cold leg accumulator flow, however, is established in refill-reflood calculations. The cold leg accumulator flow assumed is the sum of that injected in the intact cold legs.

Figures 15.6.5-73 through Figures 15.6.5-77 — These figures show the total cold leg accumulator mass injection prior to end of bypass, accumulator mass spilled out break, calculated bypass deficit, and vessel inventory.

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The maximum clad temperature calculated for a large break is 2195°F which is less than the Acceptance Criteria limit of 2200°F of 10CFR50.46. The maximum local metal-water reaction is 6.9 percent, which is well below the embrittlement limit of 17 percent as required by 10CFR50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Small Break Results

As noted previously, the calculated peak cladding temperature resulting from a small break LOCA is much less than that calculated for a large break. Since the major change in input parameters dut to the smaller optimized fuel rod is a slight increase in core flow area, the small break ECCS results with optimized fuel would not be significantly different than the standard 17x17 small break ECCS results. The maximum calculated peak cladding temperature for this plant configuration with standard 17x17 fuel was 1511°F for a 6 inch diameter break. This is much less than both the worst case large break peak clad temperature and the acceptance criteria limit of 2200°F of 10 CFR 50.46. A fuel specific analysis was not performed due to the almost 700°F difference between the standard fuel analysis and the acceptance criteria.

15.6.5.4 Radiological Consequences of a Postulated Loss-of-Coolant Accident

Two analyses will be performed: 1) a realistic analysis, and 2) an analysis based on Regulatory Guide 1.4, Revision 2. The parameters to be used for each of these analyses are listed in Table 15.6.5-1. The radiological consequences of a LOCA will be evaluated on a plant specific basis.

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Fission Produc. Release to the Containment

The radiological assessment will be based on the conservative fission product release given in Regulatory Guide 1.4.

Thus, a total of 100 percent of the noble gas core inventory and 25 percent of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, will be assumed that 91 percent is in elemental form, 4 percent in methyl form and 5 percent in particulate form. The total core noble gas and iodine inventories are given in Table 15.0.9-1.

.5.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable.

15.6.7 REFERENCES

1. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7907 June 1972.

2. Young, M. Y., Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection, WCAP-8479, (Westinghouse Proprietary), and WCAP-8480, January, 1975.

3. Bordelon, F. M., H. W. Massie, and T. A. Borden, "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, (Non-Proprietary) July 1974.

4. Bordelon, F. M., et. al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant," WCAP-8302, (Proprietary) June 1974, and (Non-Proprietary) June 1974.

5. Kelly, R. D., et. al., "Calculated Model for Core Reflooding (Proprietary) June 1974, and (Non-Proprietary) June 1974.

6. Hsieh, T., and Raymond, M., "Long Term Ice Condenser Containment LOTIC Code Supplement 1," WCAP-8355 Supplement 1, May 1975, WCAP-8354 (Proprietary), July 1974.

7. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-9220 (Proprietary) February, 1979, and WCAP-9221 (Non-Proprietary) February, 1978.

8. Eicheldinger, C., "Westinghouse ECCS Evaluation Model, February, 1978 Version," WCAP-9220 (Proprietary) February, 1979, and WCAP-9221 (Non-Proprietary) February, 1978.

9. Porsching, T. A., et. al., 'FLASH-4: A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant," WAPD-TM-84; Bettis Atomic Power Laboratory, March 1969.

10. Exposito, V. J., K. Kesavan and B. A. Maul, "WFLASH-A FORTRAN IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2, (Proprietary) July 1974, and WCAP-8261, Revision 1, (Non-Proprietary) July 1974.

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TABLE 15.6.1-1

Time Sequence of Events for Incidence Which Cause a Decrease in Reactor Coolant Inventory

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve open fully	0.0
	Low pressurizer pressure reactor trip setpoint reached	32.8
	Rods begin to drop	34.8
	Minimum DNBR occurs	35.0

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TABLE 15.6.5-1

Input Parameters Used in the ECCS Analysis

Core Power* (mwt)	3411
Peak Linear Power (Includes 102% factor) kW/ft Large Break	 12.88
Total Peaking Factor, F_Q	2.32
Axial Peaking Factor, F_Z	1.495
Power Shape Large Break	 Chopped Cosine
Fuel Assembly Array	17 x 17 (optimized)
Cold Leg Accumulator Water Volume (nominal) (Ft^3/accumulator)	1050
Cold Leg Accumulator Tank Volume (nominal) (Ft^3/accumulator)	1350
Cold Leg Accumulator Gas Pressure (minimum) (psia)	400
UHI Accumulator injected volume Perfect mixing case (ft^3) Imperfect mixing case (ft^3)	 1020 860
UHI Accumulator initial pressure Perfect mixing case (psia) Imperfect mixing case (psia)	 1200 1300
Safety Injection Pumped Flow	See Table 6
Containment Parameters	0 psig (see text for additional details)
Initial Loop Flow (lb/sec)	9961
Vessel Inlet Temperature (^oF)	561.6
Vessel Outlet Temperature (^oF)	620.0
Reactor Coolant Pressure, psia	2280
Steam Pressure (psia)	1000
Steam Generator Tube Plugging Level (%)	0

Notes:

*Two percent is added to this power to account for calorimetric error.

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TABLE 15.6.5-2

Large Break LOCA Time Sequence of Events

Perfect Mixing

	C_D = 1.0 DECLG (sec)	C_D = 0.8 DECLG (sec)	C_D = 0.6 DECLG (sec)	C_D = 0.4 DECLG (sec)
Start	0.0	0.0	0.0	0.0
Reactor Trip Signal	0.8	0.8	0.8	0.8
UHI Accumulator Injection	3.1	3.4	4.2	6.3
Safety Injection Signal	4.8	4.9	5.0	5.3
Cold Leg Accumulator Injection	12.3	14.9	17.4	22.8
UHI Accumulator Injection Complete	21.6	23.4	25.5	29.7
Pump Injection	29.8	29.9	30.0	30.3
End of Bypass	49.7	49.0	69.5	66.3
Bottom of Core Recovery	107.3	109.9	104.0	112.0
Cold Leg Accumulator Empty	112.4	116.1	118.6	128.4

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TABLE 15.6.5-3

Large Break LOCA Time Sequence of Events

Imperfect Mixing

	C_D = 1.0 DECLG (sec)
Start	0.0
Reactor Trip Signal	0.8
UHI Accumulator Injection	2.2
Safety Injection Signal	4.8
Cold Leg Accumulator Injection	13.4
UHI Accumulator Injection Complete	19.4
Pump Injection	29.8
End of Bypass	48.8
Bottom of Core Recovery	70.2
Cold Leg Accumulator Empty	112.2

TABLE 15.6.5-4

Large Break LOCA Results Fuel Cladding Data

Perfect Mixing

	C_D = 1.0 DECLG (sec)	C_D = 0.8 DECLG (sec)	C_D = 0.6 DECLG (sec)	C_D = 0.4 DECLG (sec)
RESULTS				
Peak Clad Temperature (°F)	2195	2147	2103	2000
Peak Clad Temperature Location (ft)	7.25	5.5	7.5	5.75
Local Zr/H_2O Reaction (max), (%)	6.9	6.2	5.9	2.5
Local Zr/H_2O Location, (ft)	7.5	7.5	7.5	6.0
Total Zr/H_2O Reaction, (%)	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time, (sec)	72.8	68.1	68.0	82.8
Hot Rod Burst Location, (ft)	5.5	5.75	5.75	6.0

TABLE 15.6.5-5

Large Break LOCA Results Fuel Cladding Data

Imperfect Mixing

RESULTS	$C_D = 1.0$ DECLG
Peak Clad Temperature (^oF)	1992
Peak Clad Temperature Location (ft)	7.5
Local Zr/H_2O Reaction (max), (%)	4.0
Local Zr/H_2O Location, (ft)	7.5
Total Zr/H_2O Reaction, (%)	<0.3
Hot Rod Burst Time, (sec)	108.2
Hot Rod Burst Location, (ft)	6.25

TABLE 15.6.5-6

Safety Injection Pumped Flow

Pressure (psia)	SI Flow (lb/sec)
14.7	493.2
34.7	437.2
54.7	378.6
74.7	315.5
114.7	198.4
214.7	100.2
614.7	81.2
1014.7	58.5
3014.7	0.0

TABLE 15.6.5-7

Nodal Representation of Core Transients

Large Break

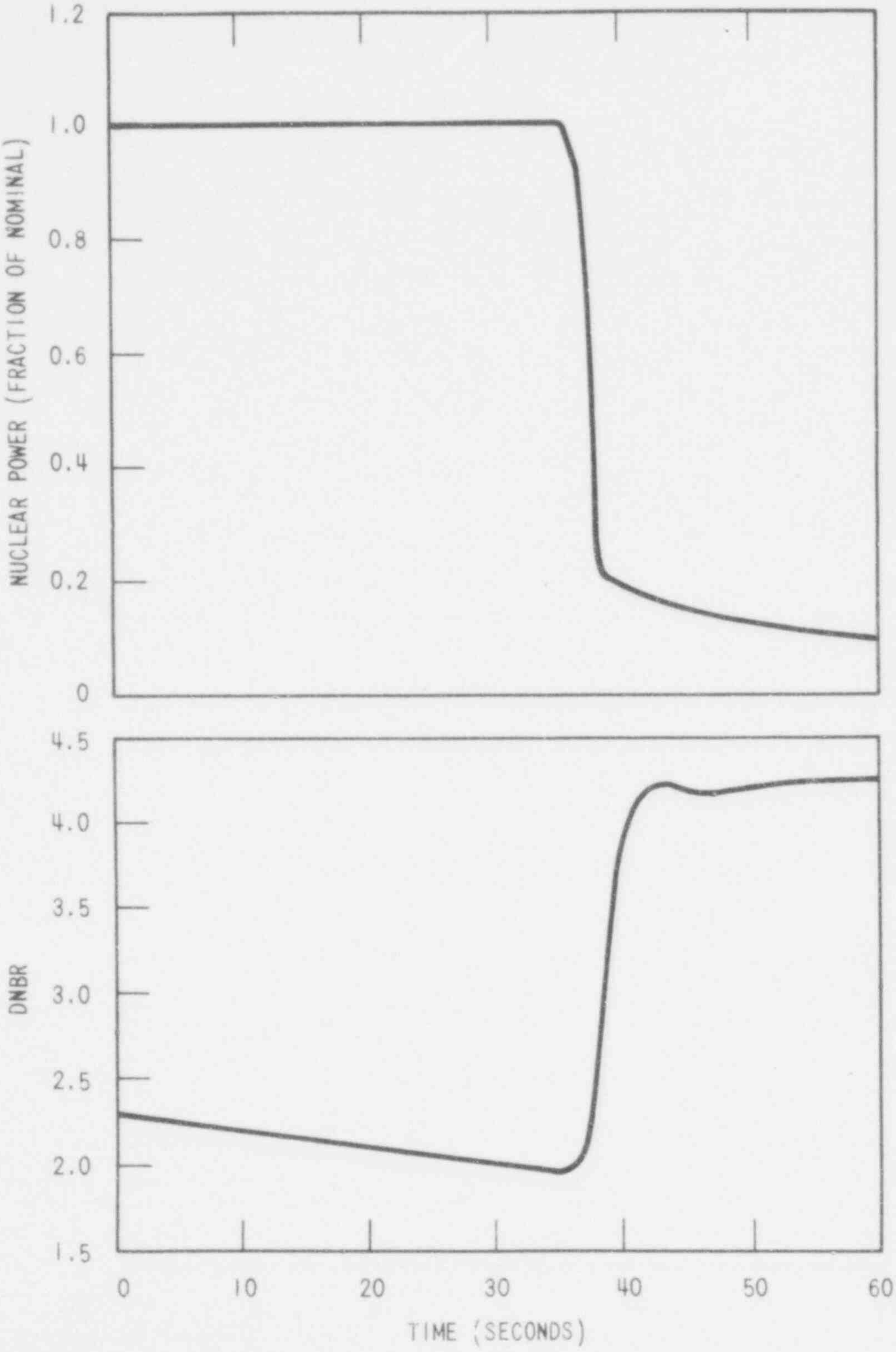
Node	Elevation from Bottom of Core (ft.)	"Node"	Elevation from Bottom of Core (ft.)
1	0.0	11	6.75
2	1.5	12	7.00
3	3.0	13	7.25
4	4.0	14	7.50
5	5.0	15	7.75
6	5.5	16	8.0
7	5.75	17	9.0
8	6.0	18	10.5
9	6.25	19	12.0
10	6.50		

*Applicable to the nodes in Figures 15.6.5-3 through 15.6.5-52.

POOR ORIGINAL

BLUE

615 135



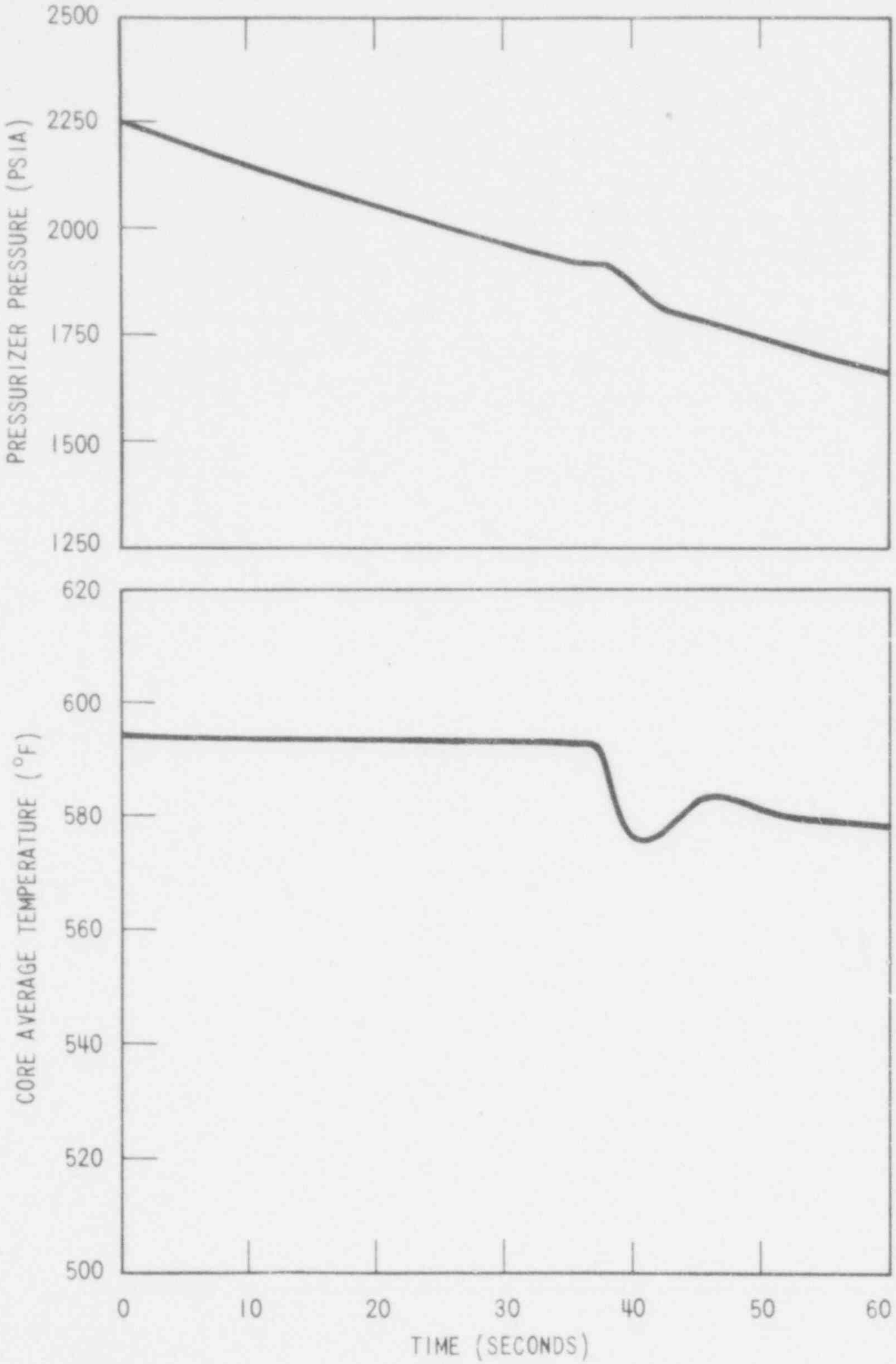
POOR ORIGINAL

615 136

WCAP - 9500

Figure 15.6.1-1. BLUE

Inadvertent Opening of a Pressurizer Safety Valve



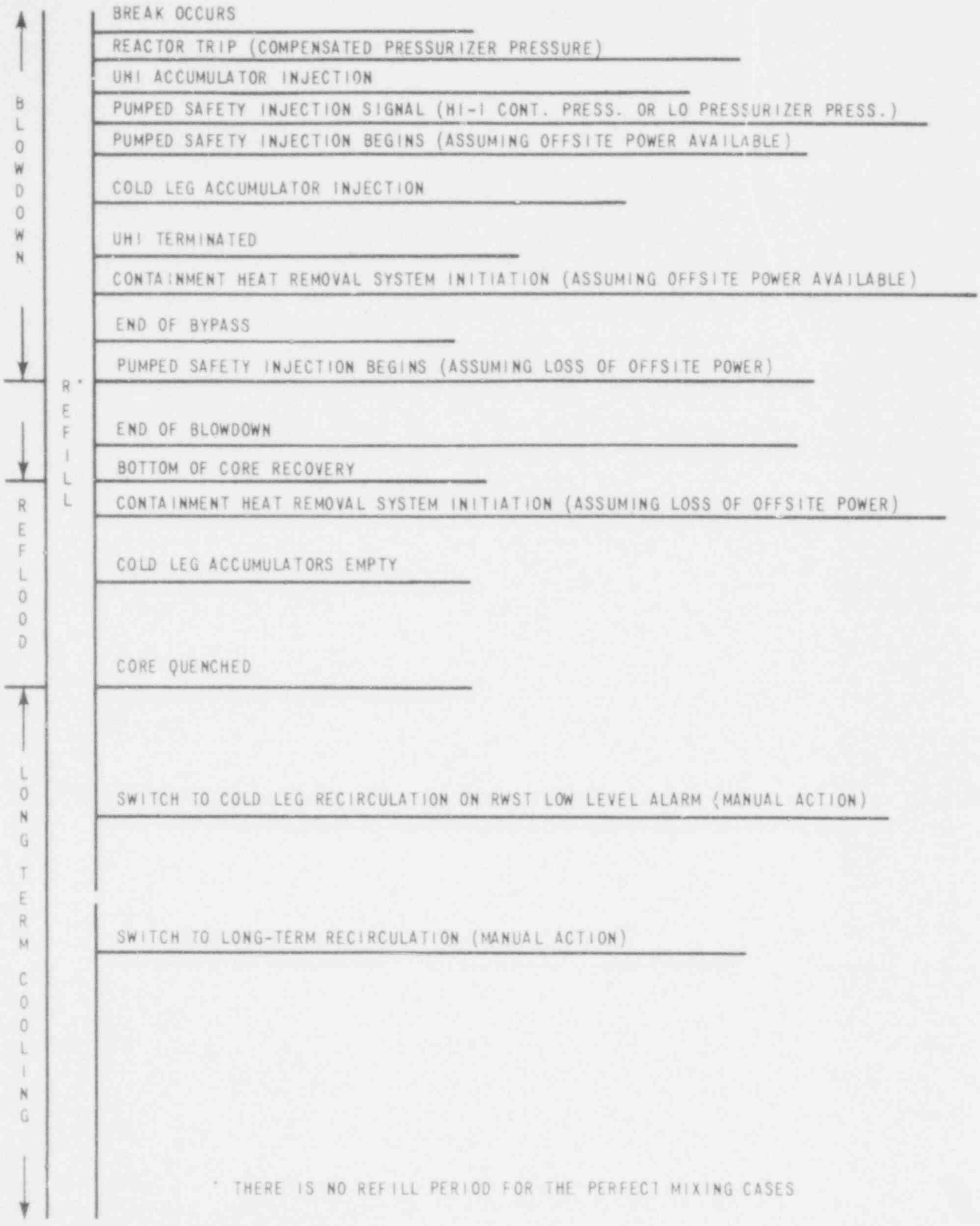
POOR ORIGINAL

WCAP – 9500

Figure 15.6.1-2. BLUE

Inadvertent Opening of a Pressurizer Safety Valve

615 137



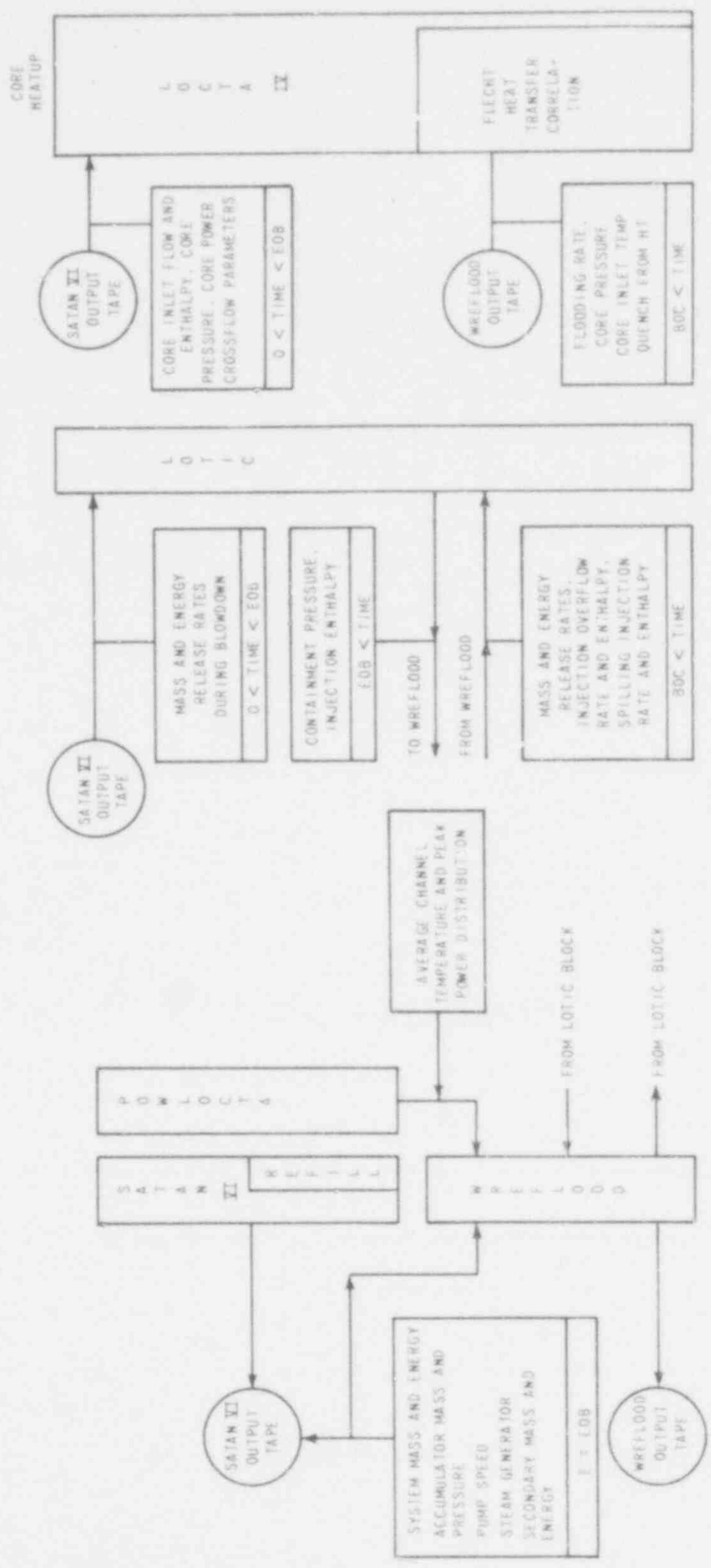
POOR ORIGINAL

615 138

WCAP – 9500

Figure 15.6.5-1. BLUE

Sequence of Events for Large Break Loss-of-Coolant Analysis



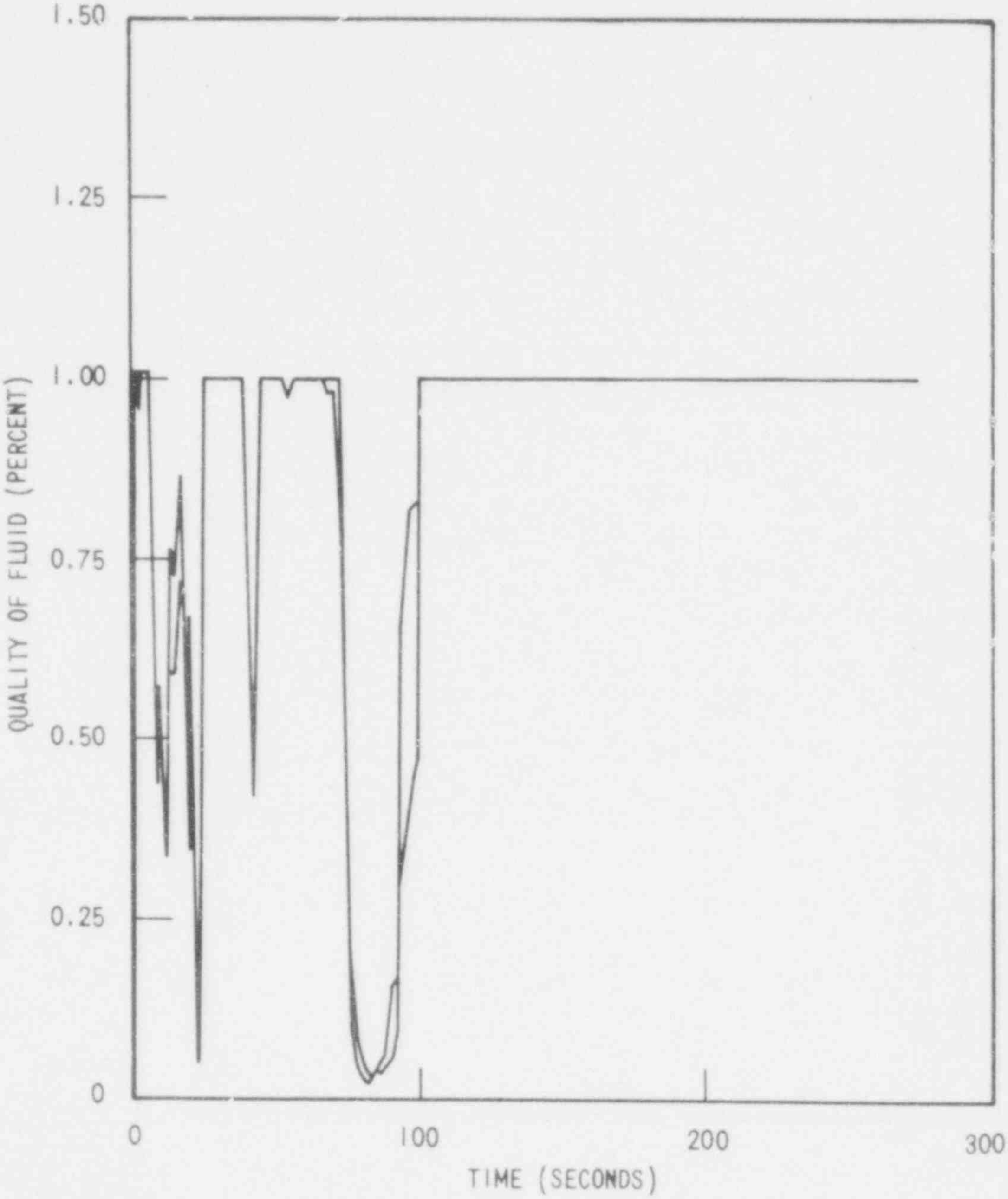
WCAP – 9500

Figure 15.6.5-2. BLUE

Code Interface Description for Large Break Model

615 139

POOR ORIGINAL

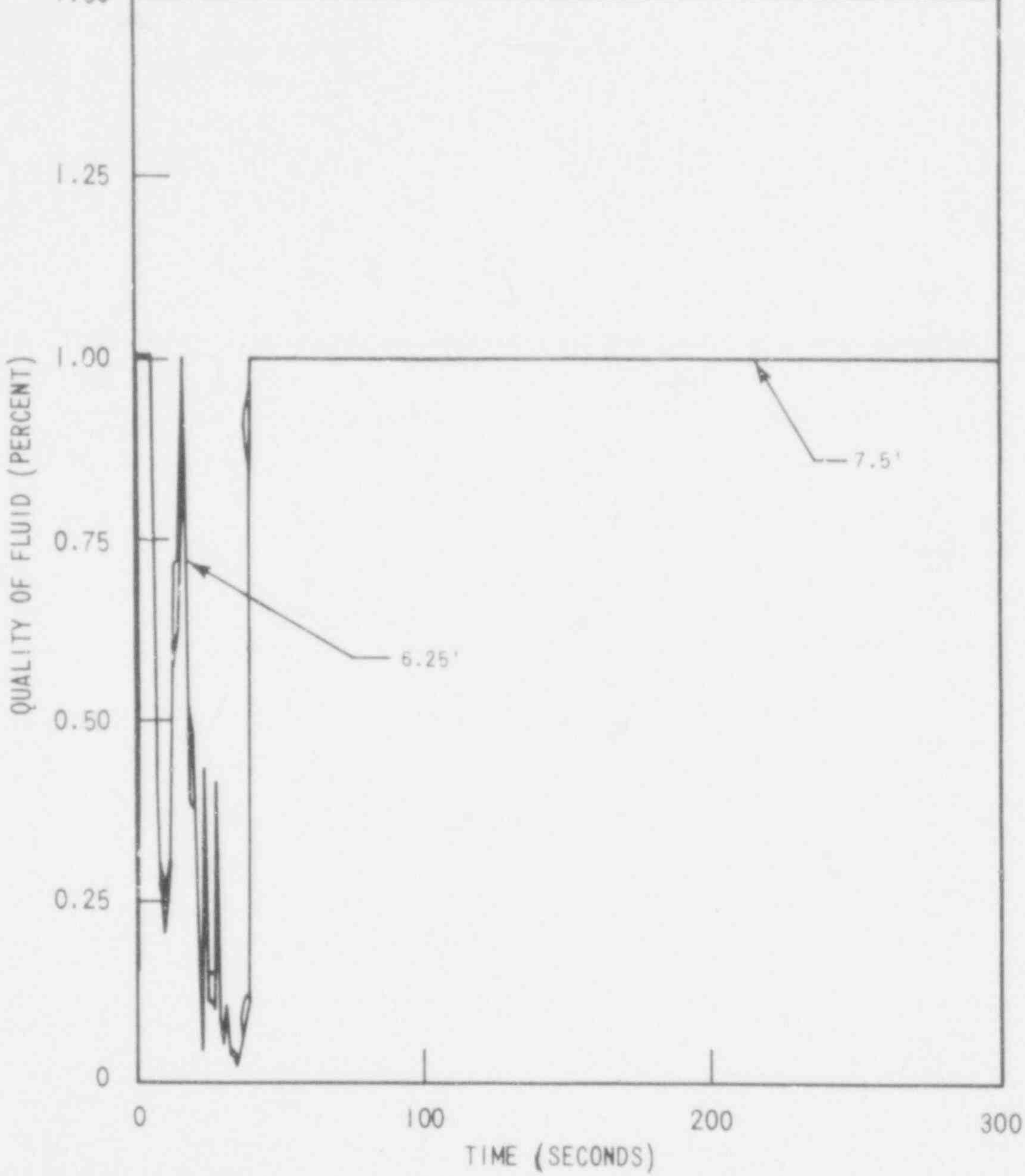


WCAP - 9500

Figure 15.6.5-3. BLUE

Fluid Quality - DECLG (C_D = 1.0)

615 140



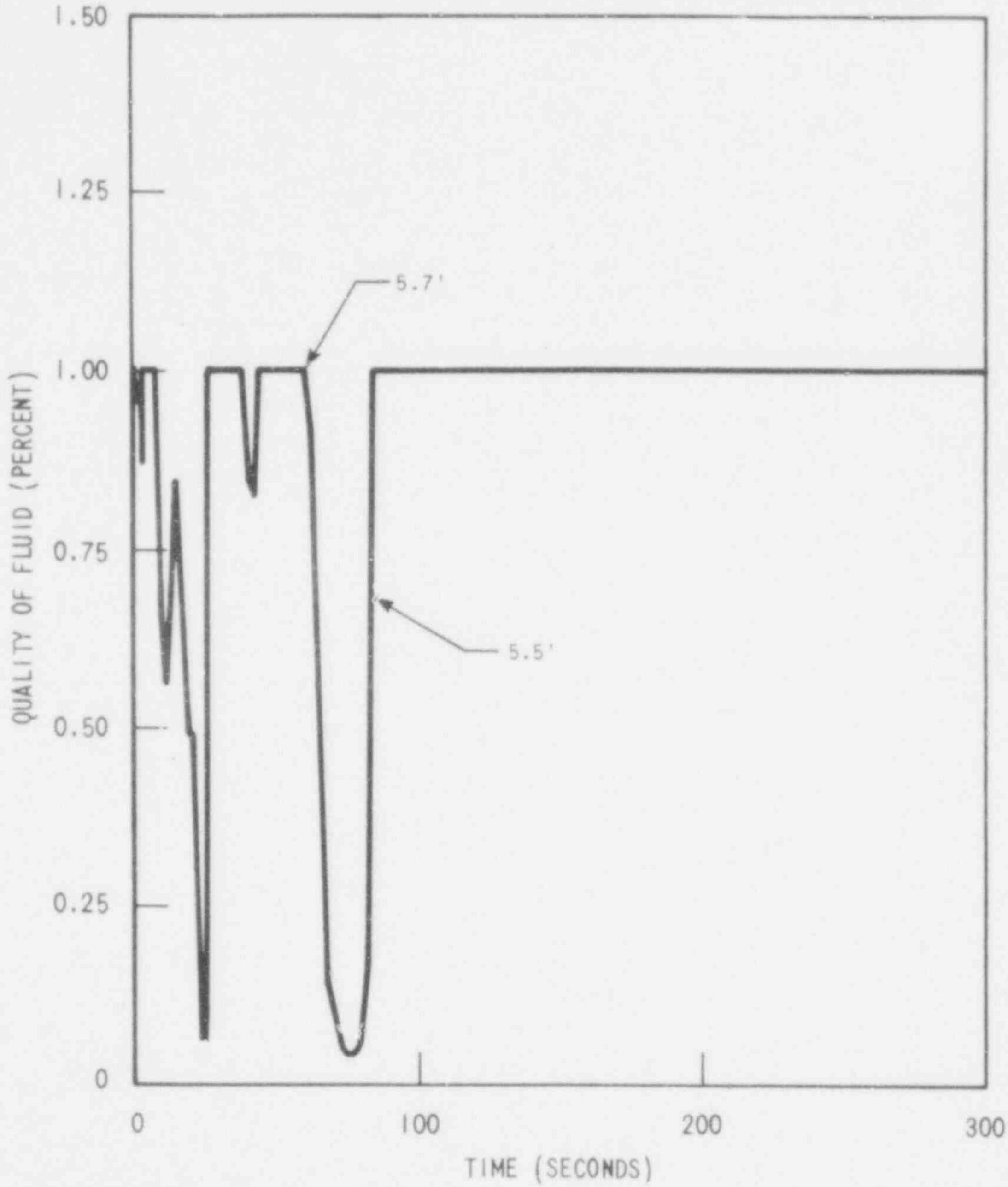
POOR ORIGINAL

WCAP - 9500

Figure 15.6.5-4. BLUE

Fluid Quality - DECLG (C_D = 1.0)

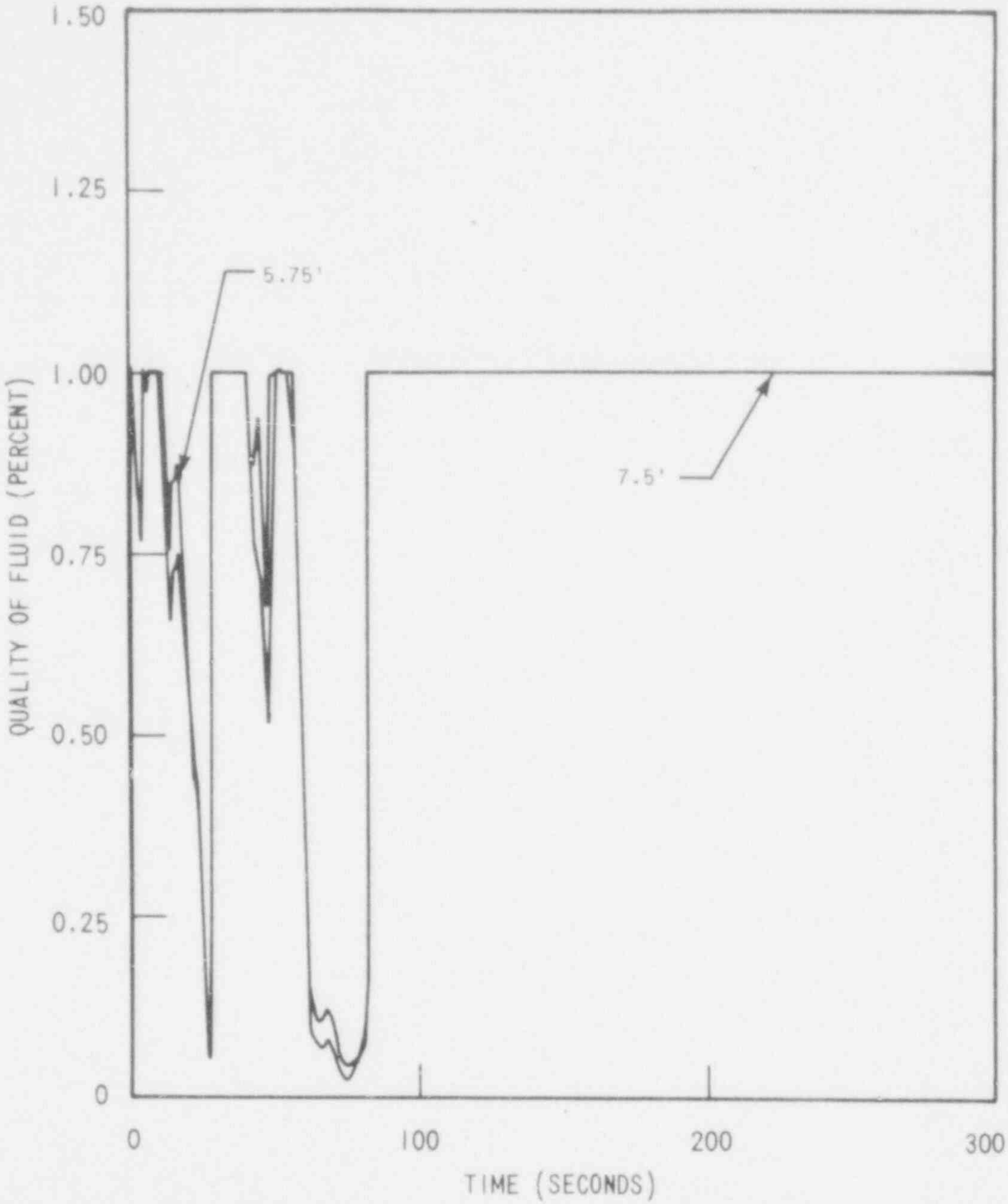
616 141



POOR ORIGINAL

WCAP - 9500
Figure 15.6.5-5. BLUE Fluid Quality – DECLG (C_D = 0.8)

615 142



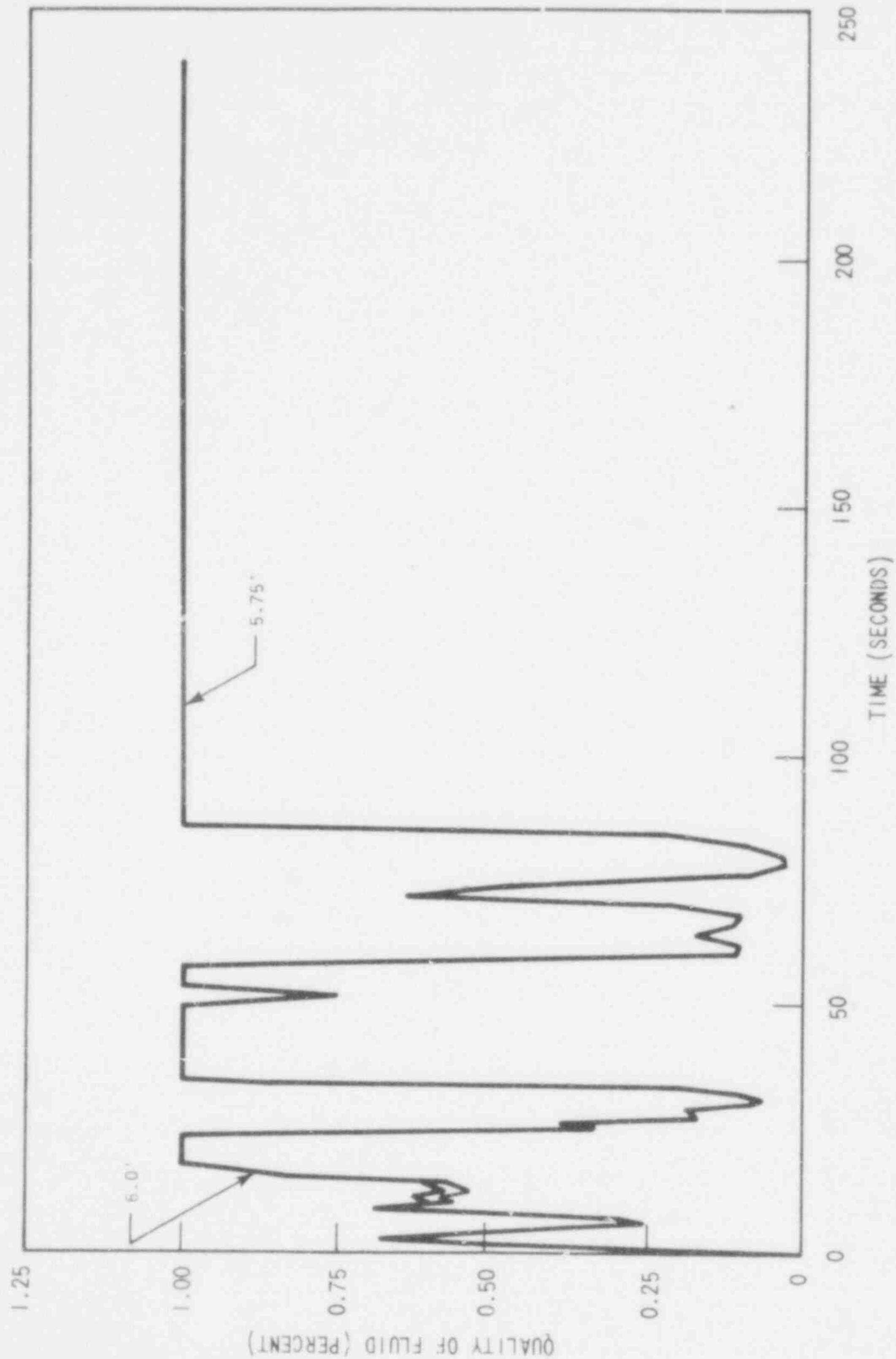
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-6. BLUE

Fluid Quality – DECLG (C_D = 0.6)

615 143

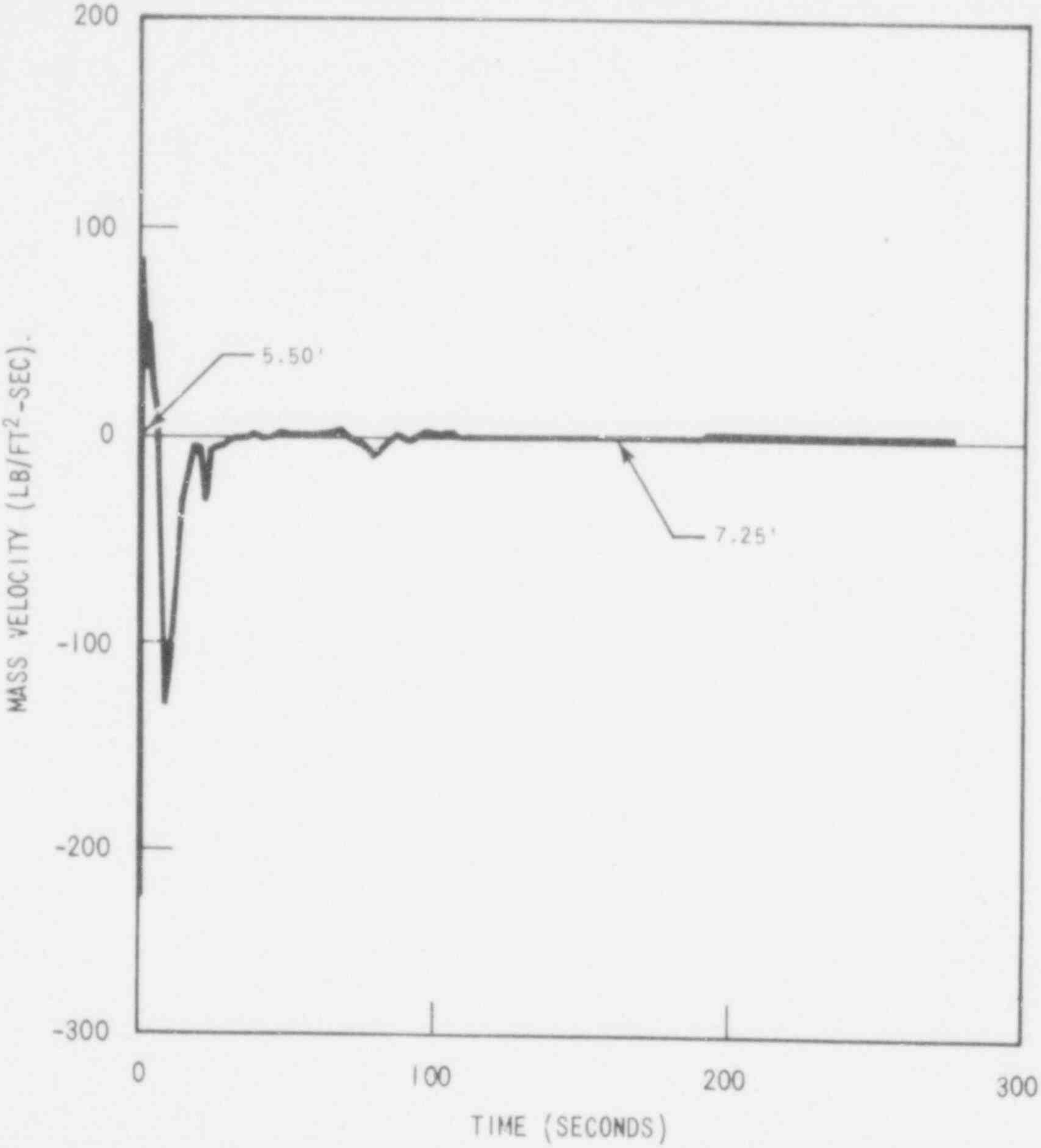


144

WCAP - 9500

Figure 15.6.5-7. BLUE

Fluid Quality – DECLG (C_D = 0.4)



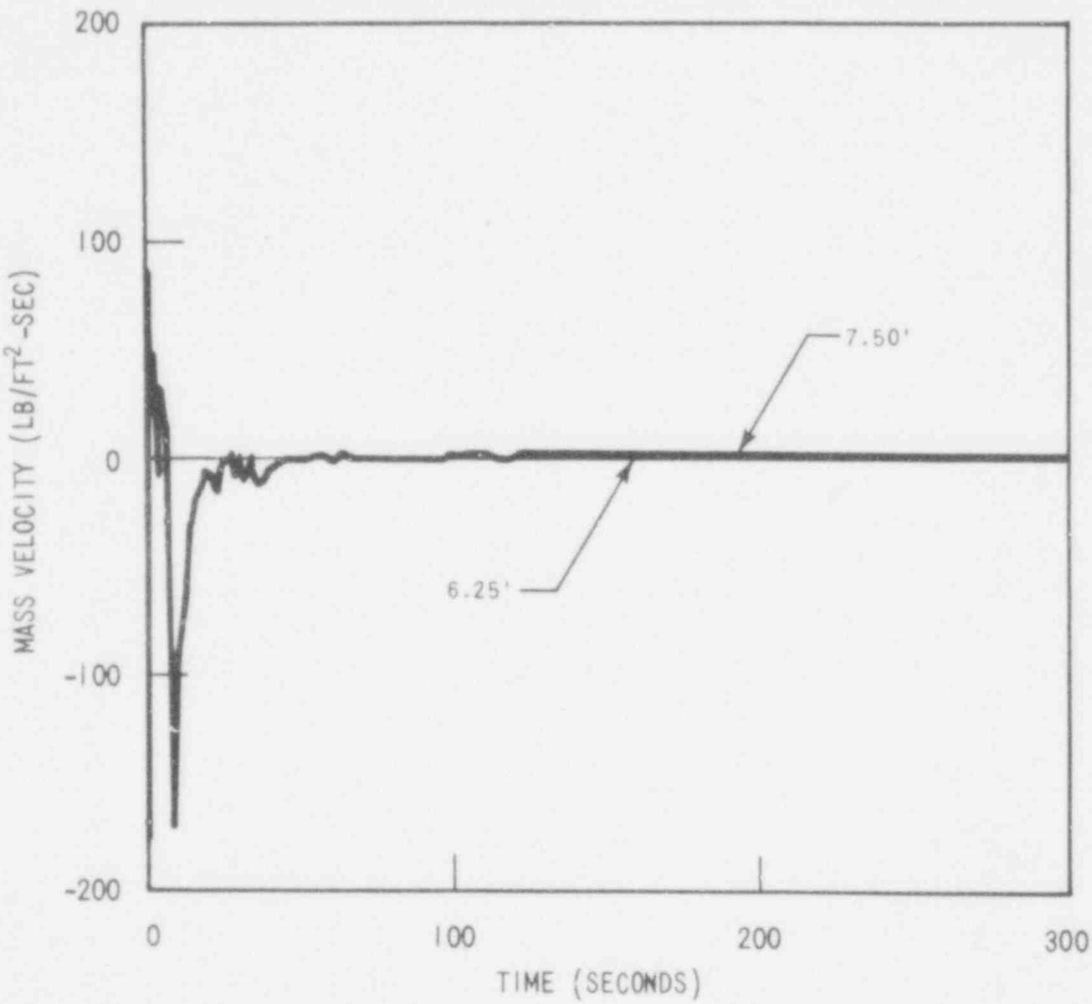
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-8. BLUE

Mass Velocity – DECLG (C_D = 1.0)

615 145



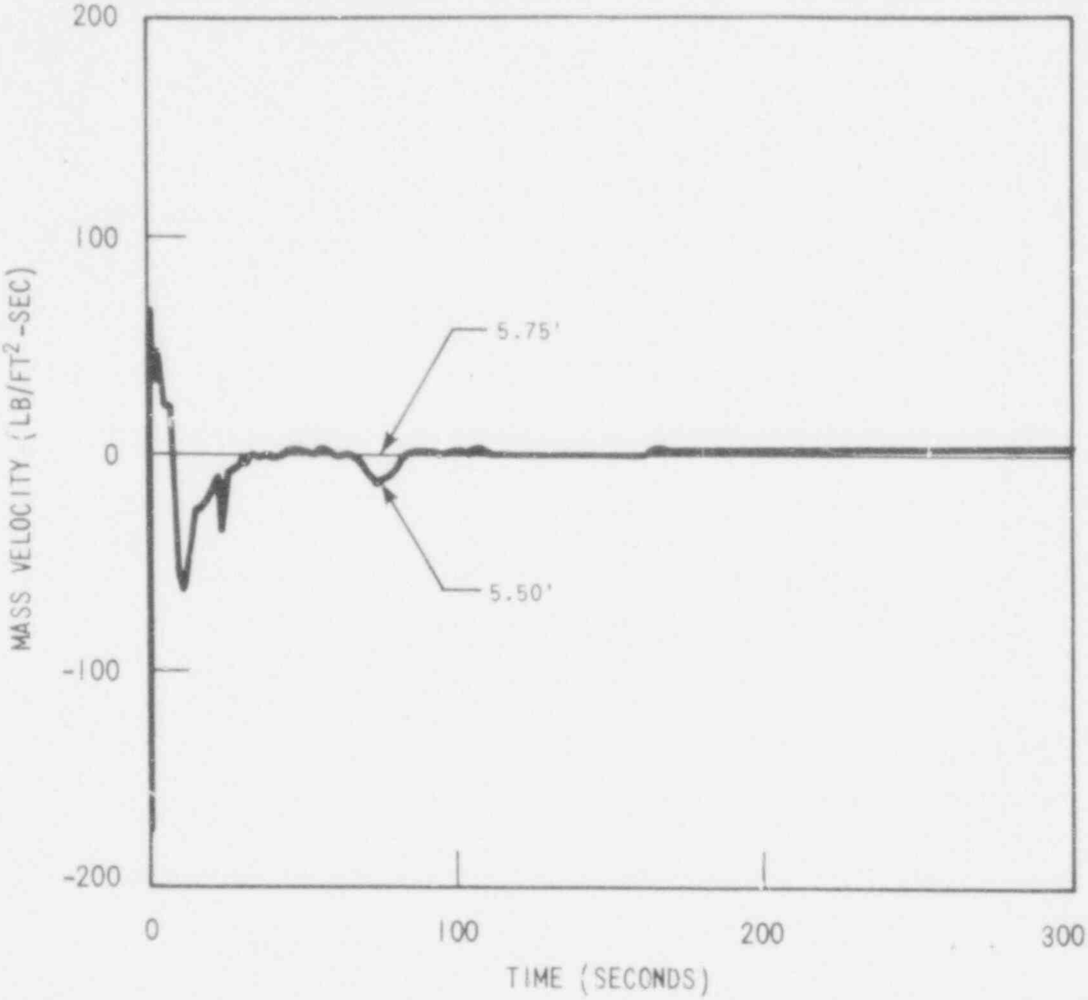
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-9. BLUE

Mass Velocity – DECLG (C_D = 1.0)

615 146

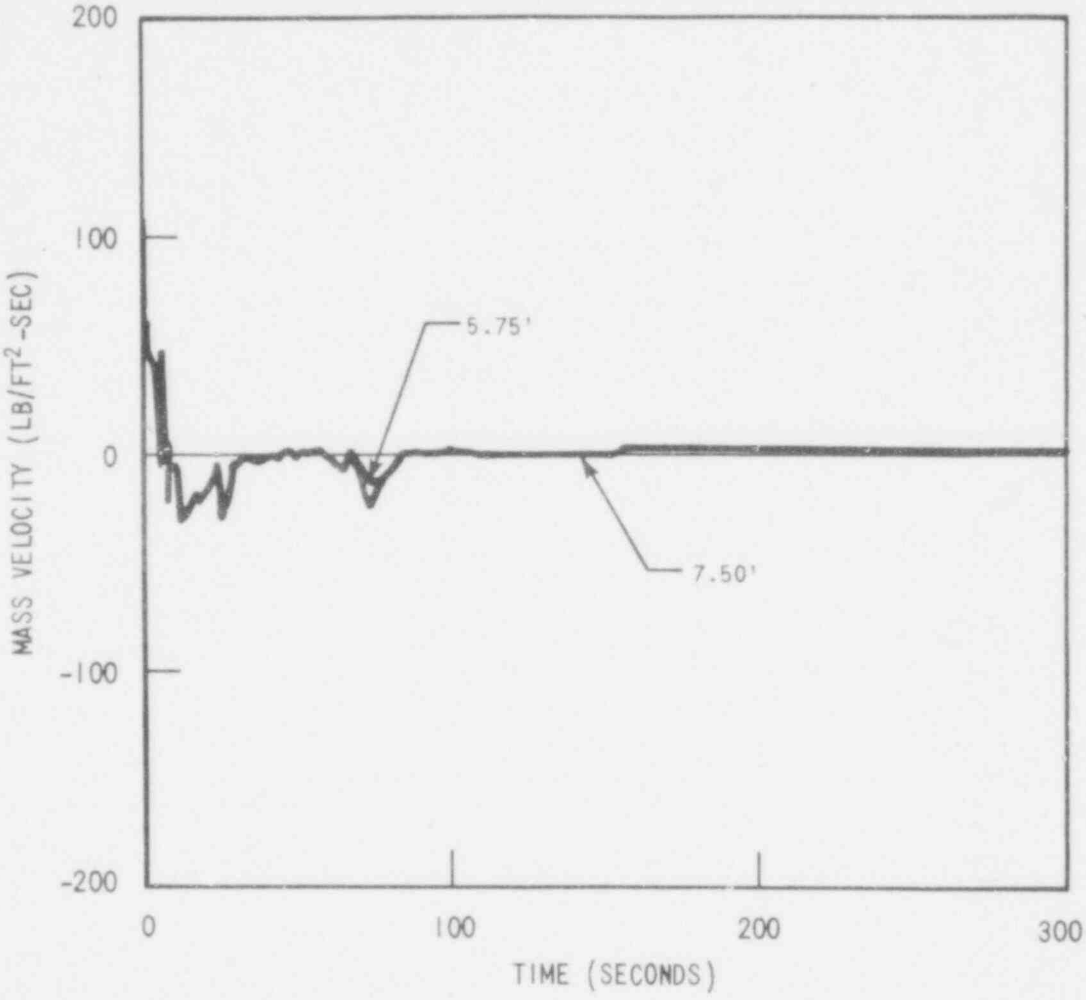


615 147

WCAP – 9500

Figure 15.6.5-10. BLUE

Mass Velocity – DECLG (C_D = 0.8)



POOR ORIGINAL

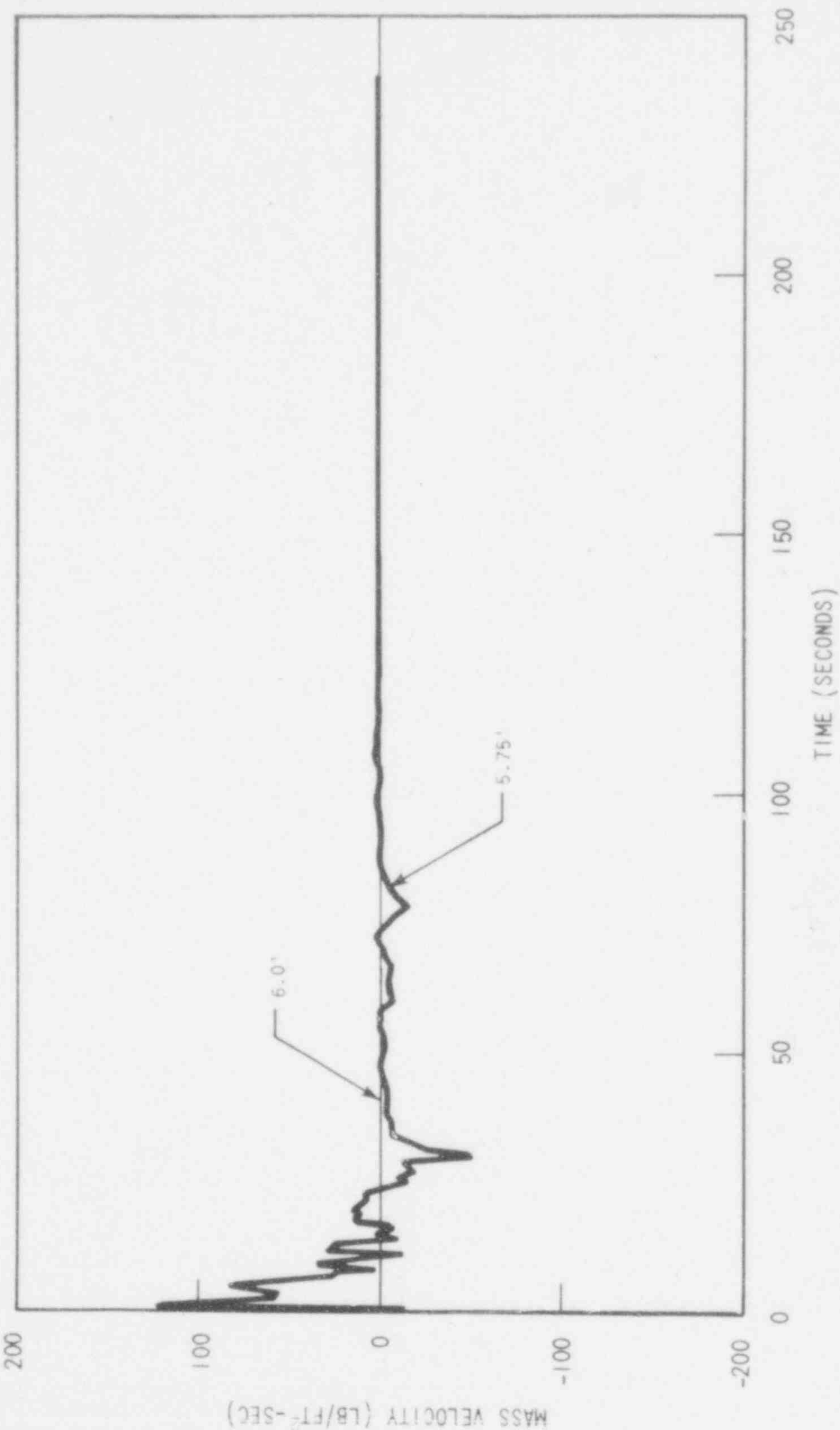
615 148

WCAP - 9500

Figure 15.6.5-11. BLUE

Mass Velocity - DECLG (C_D = 0.6)

14,395-364



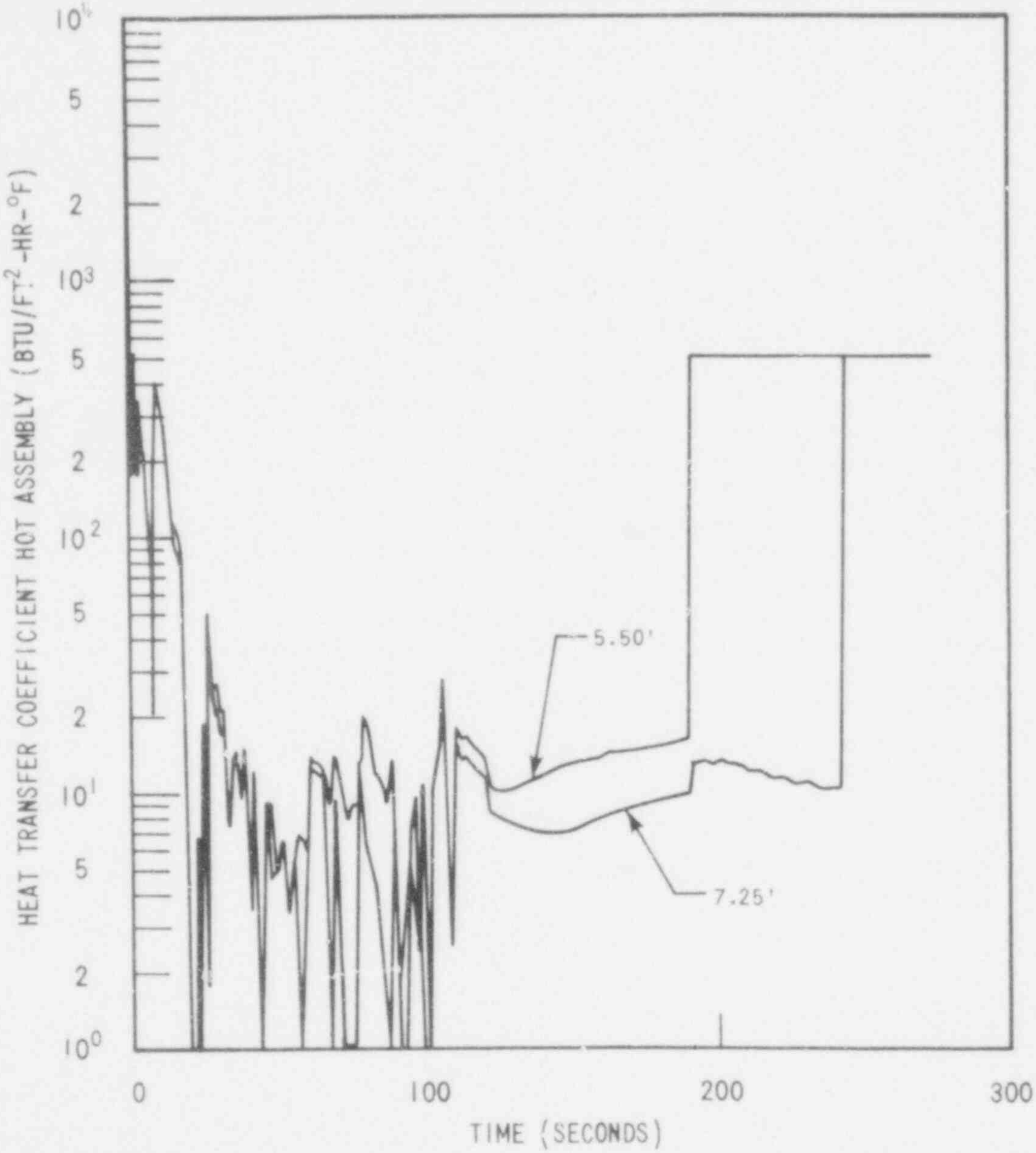
WCAP - 9500

Figure 15.6.5-12. BLUE

Mass Velocity – DECLG (C_D = 0.4)

POOR ORIGINAL

615 149



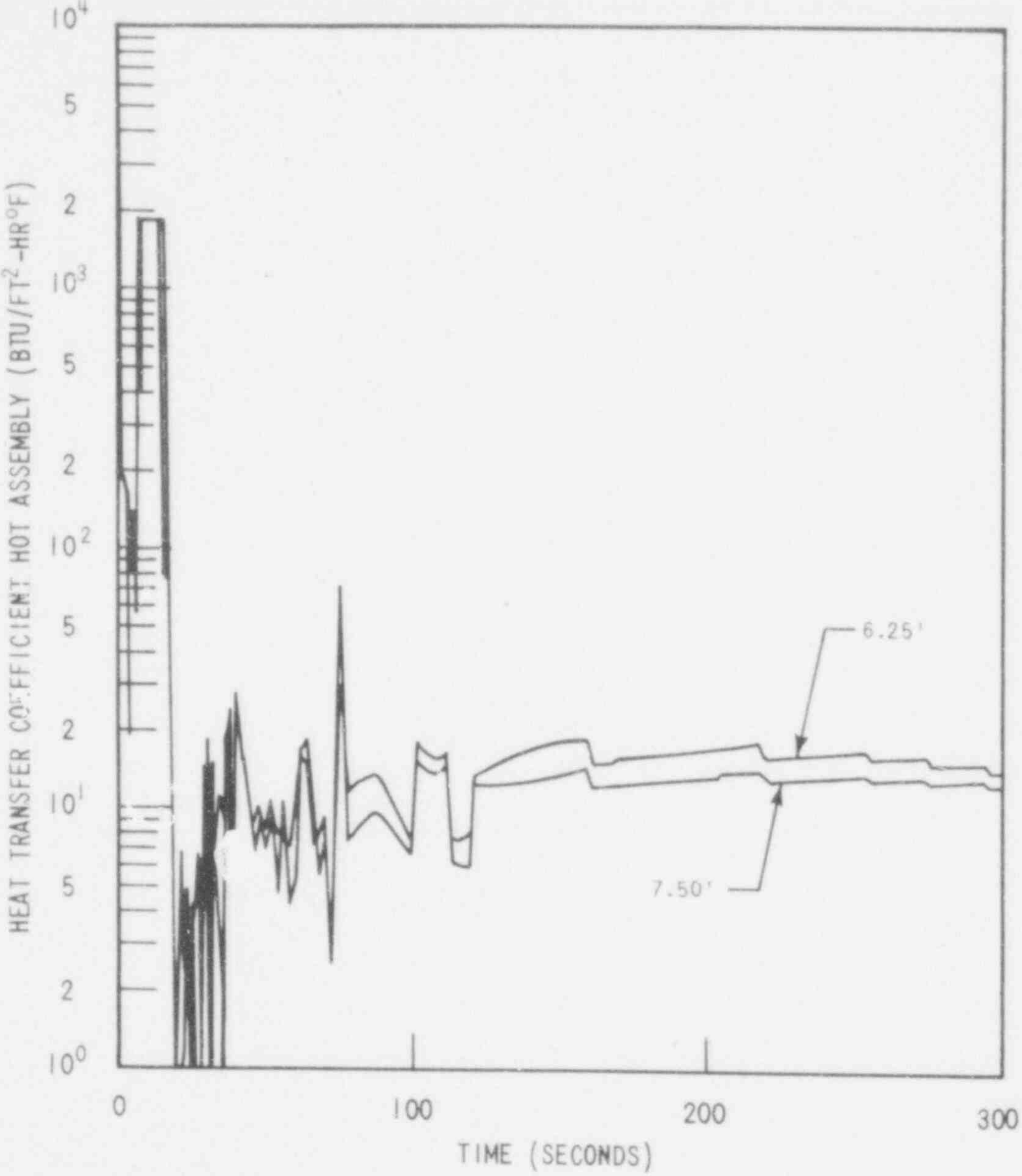
POOR ORIGINAL

615 150

WCAP-9500

Figure 15.6.5-13. BLUE

Heat Transfer Coefficient
DECLG (C_D = 1.0)

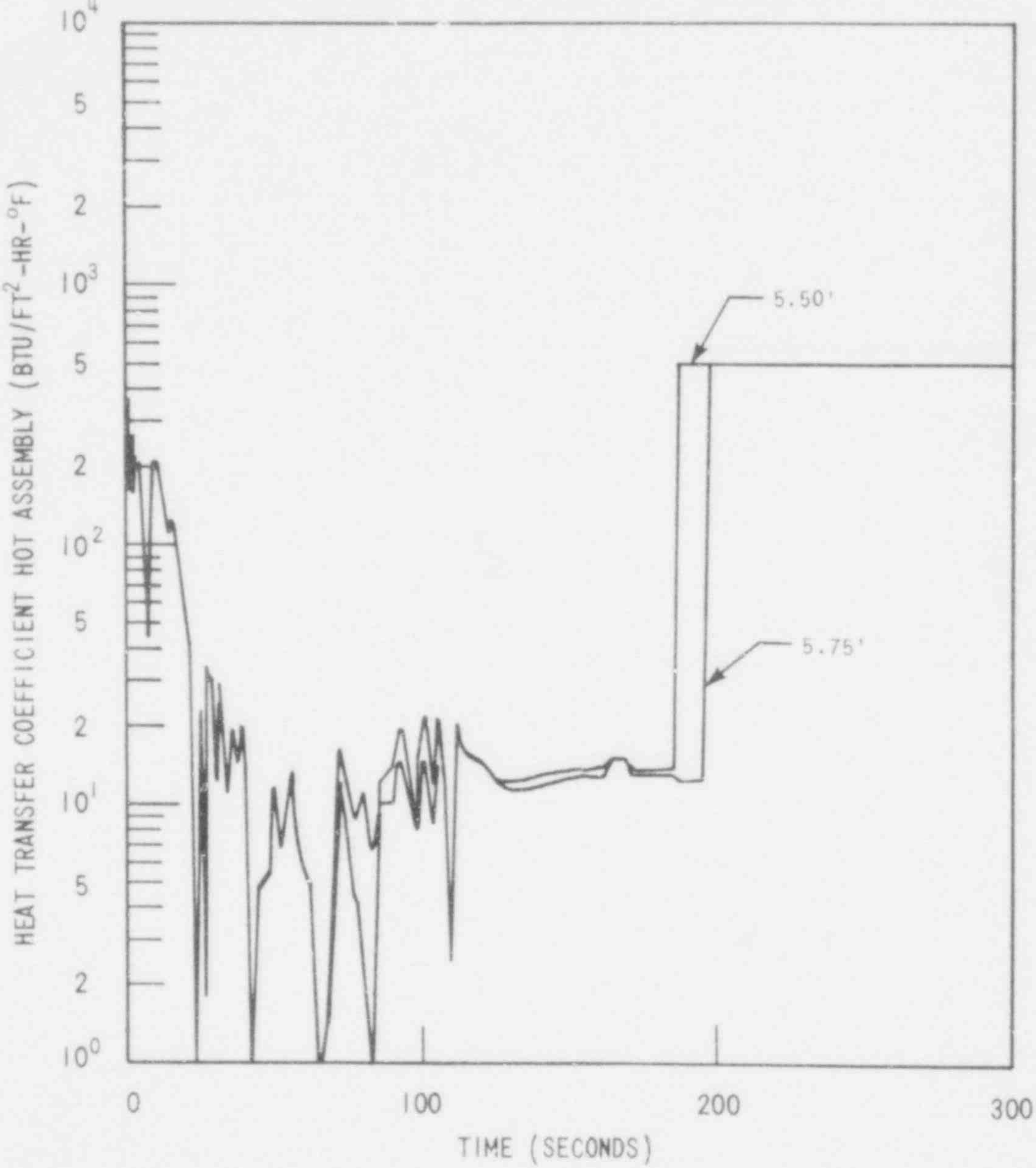


WCAP - 9500

Figure 15.6.5-14. BLUE

Heat Transfer Coefficient
DECLG (C_D = 1.0)

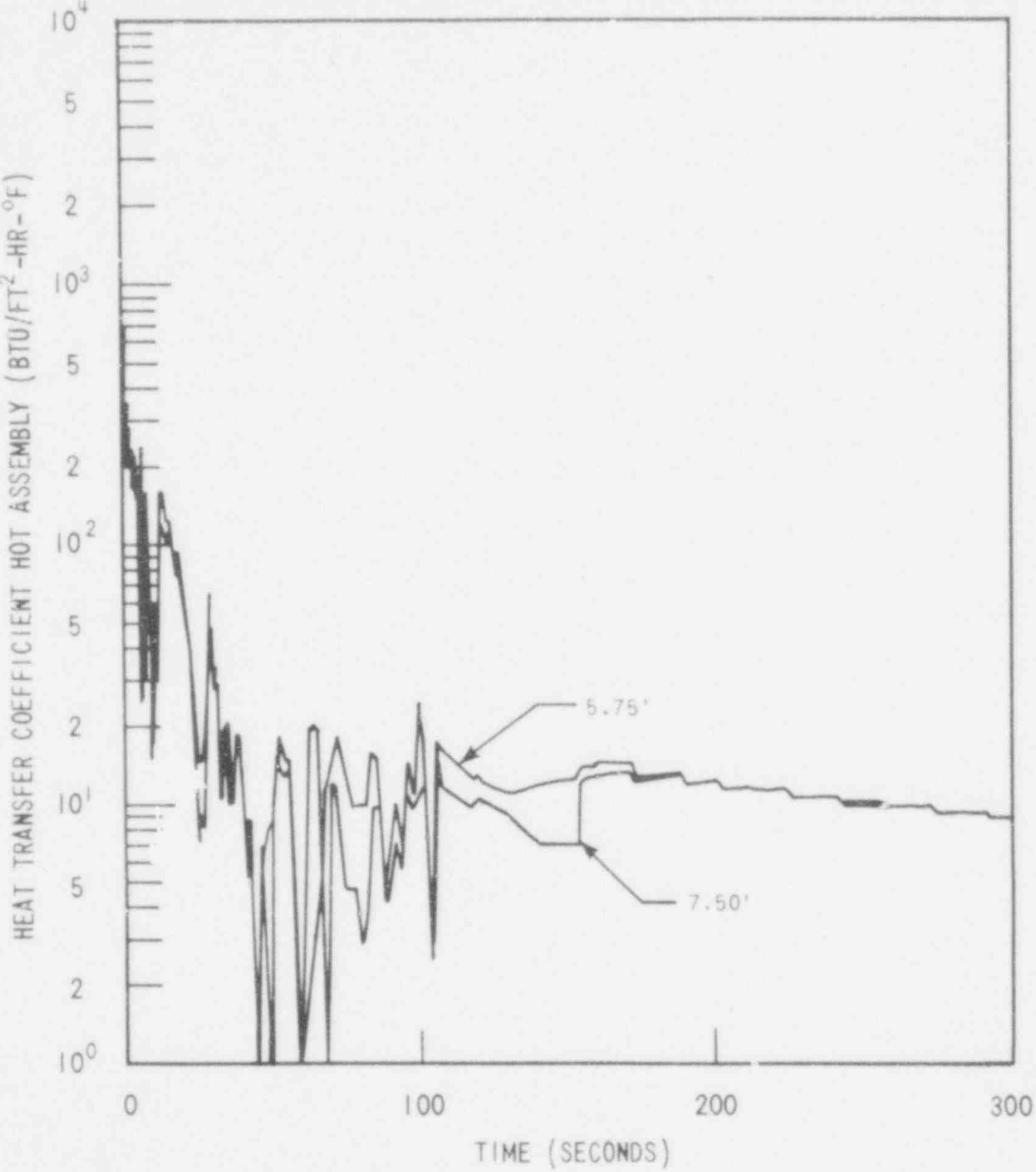
14,395-367



POOR ORIGINAL

WCAP - 9500
Figure 15.6.5-15. BLUE Heat Transfer Coefficient DECLG (C_D = 0.8)

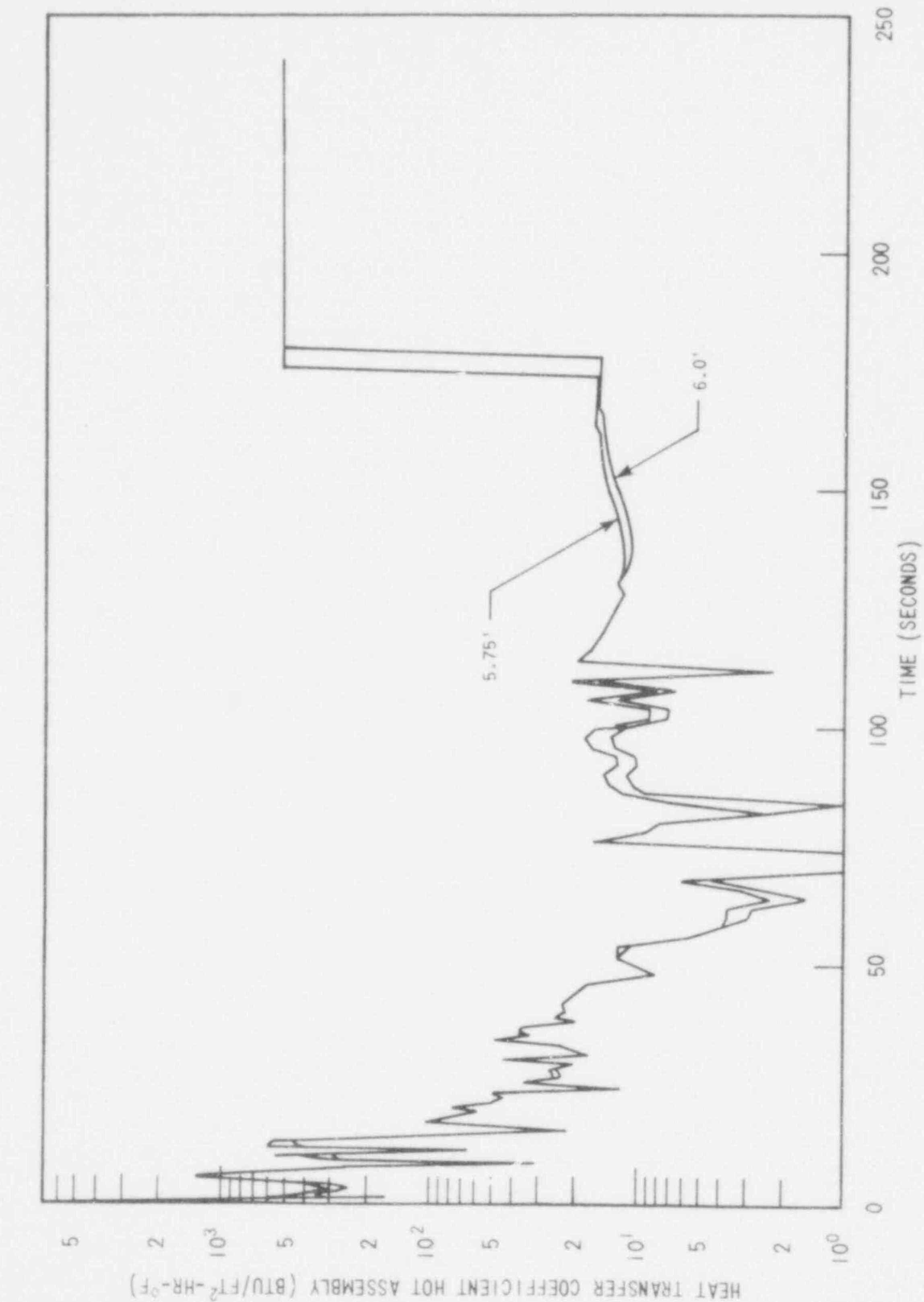
615 152



WCAP – 9500

Figure 15.6.5-16. BLUE

Heat Transfer Coefficient DECLG (C_D = 0.6)

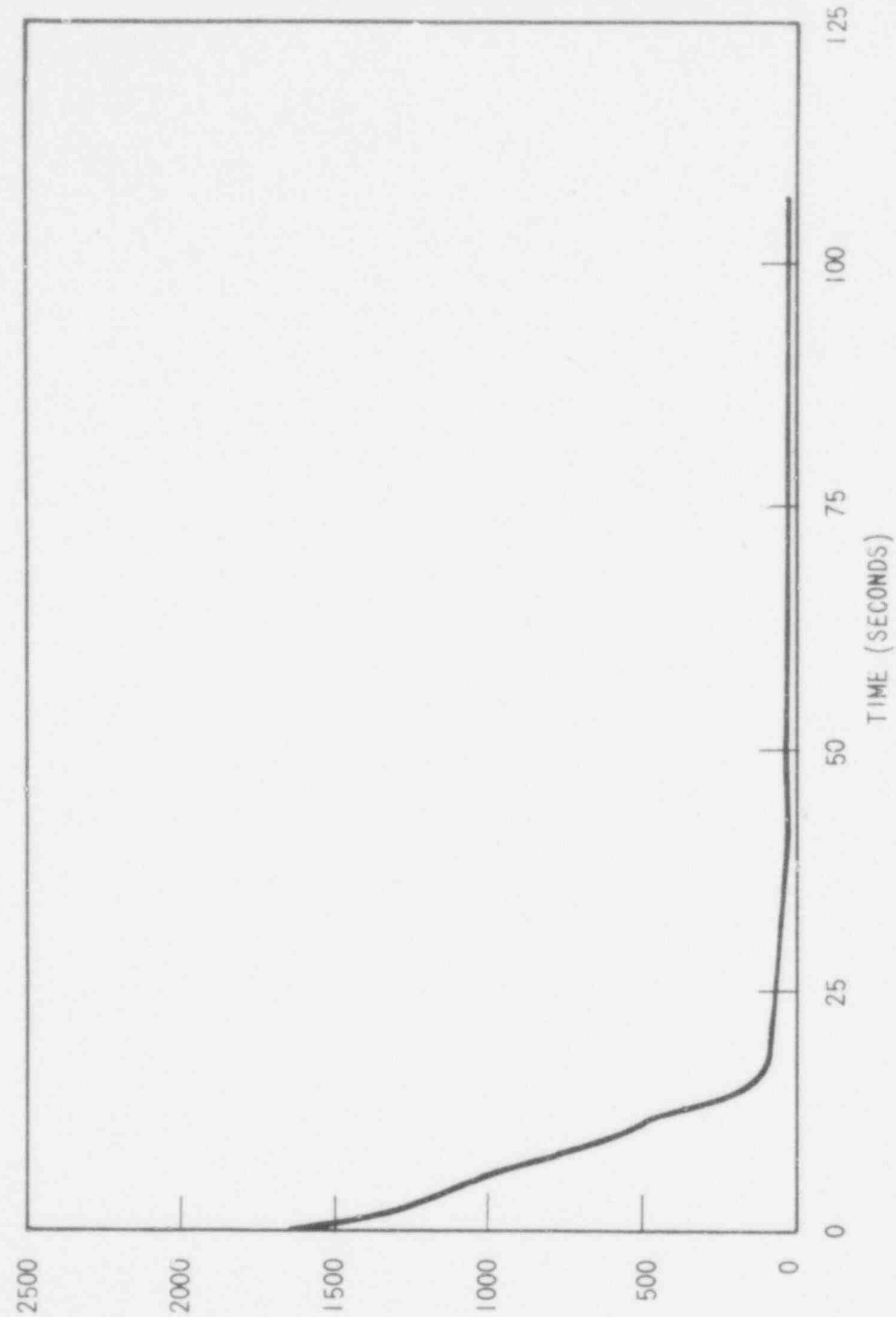


WCAP – 9500

Figure 15.6.5-17. BLUE

Heat Transfer Coefficient
DECLG (C_D = 0.4)

615 154

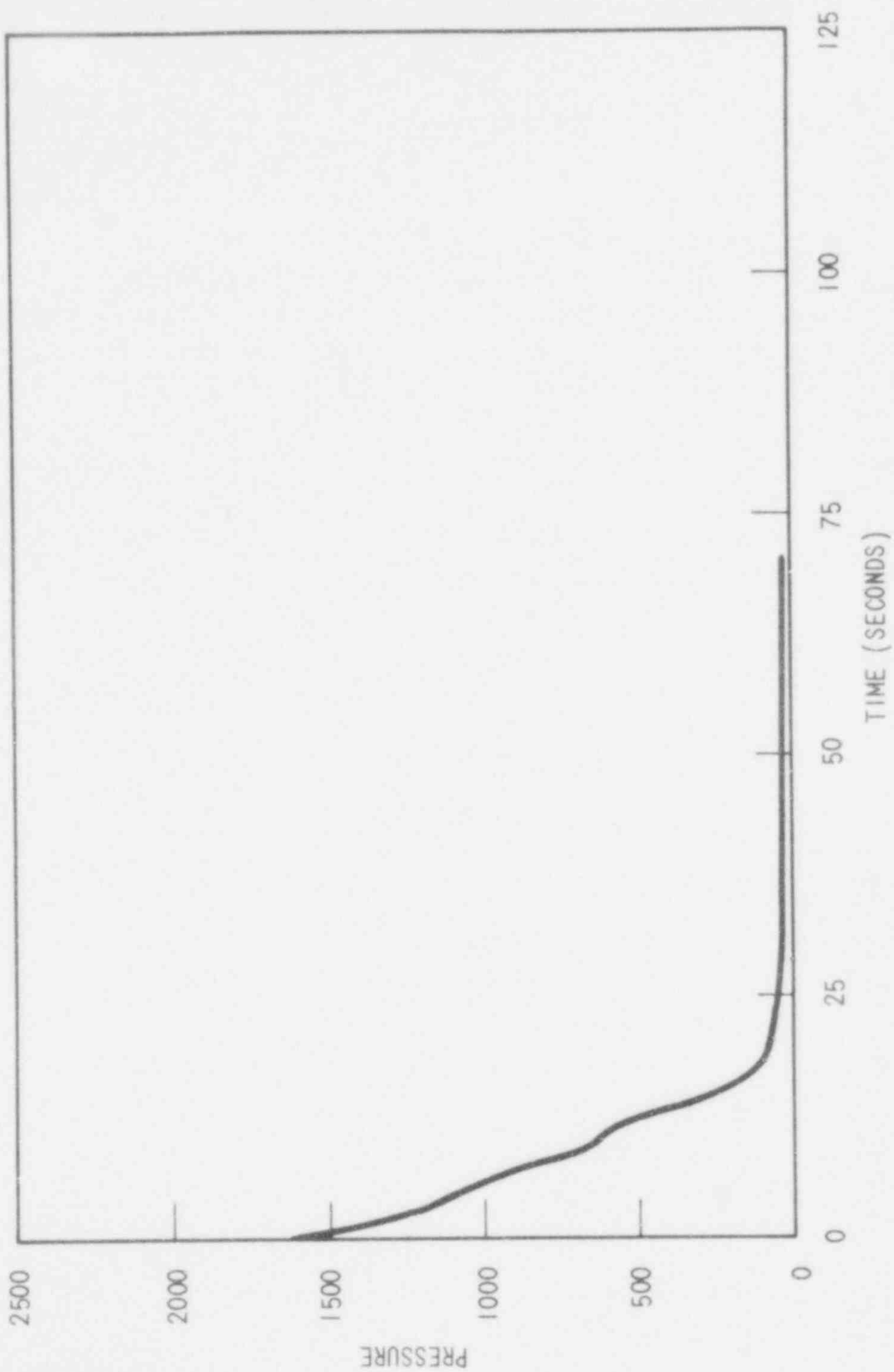


615 155

WCAP - 9500

Figure 15.6.5-18. BL JE

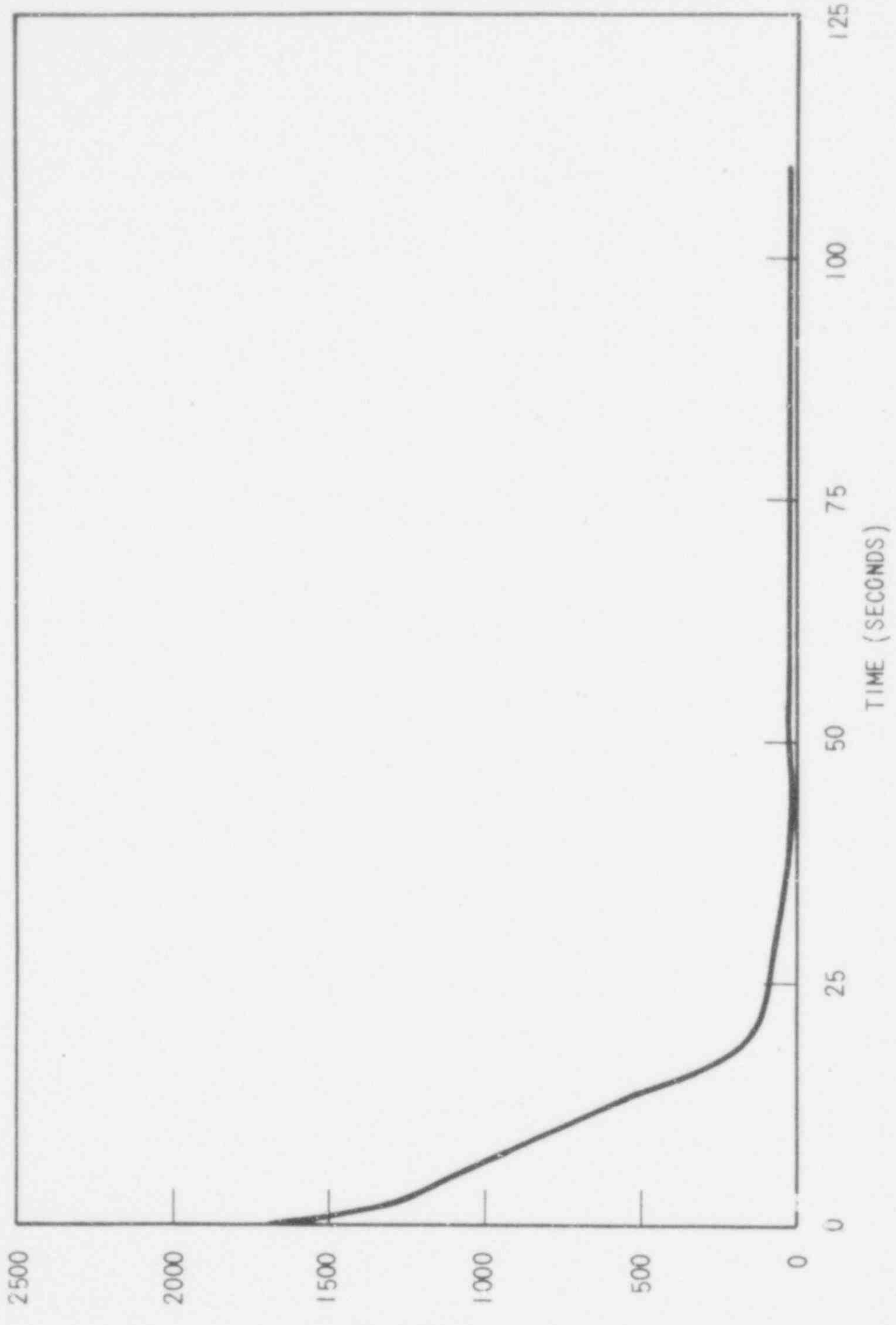
Core Pressure – DECLG (C_D = 1.0)



POOR ORIGINAL

WCAP - 9500
Figure 15.6.5-19. BLUE Core Pressure – DECLG (C_D = 1.0)

615 156



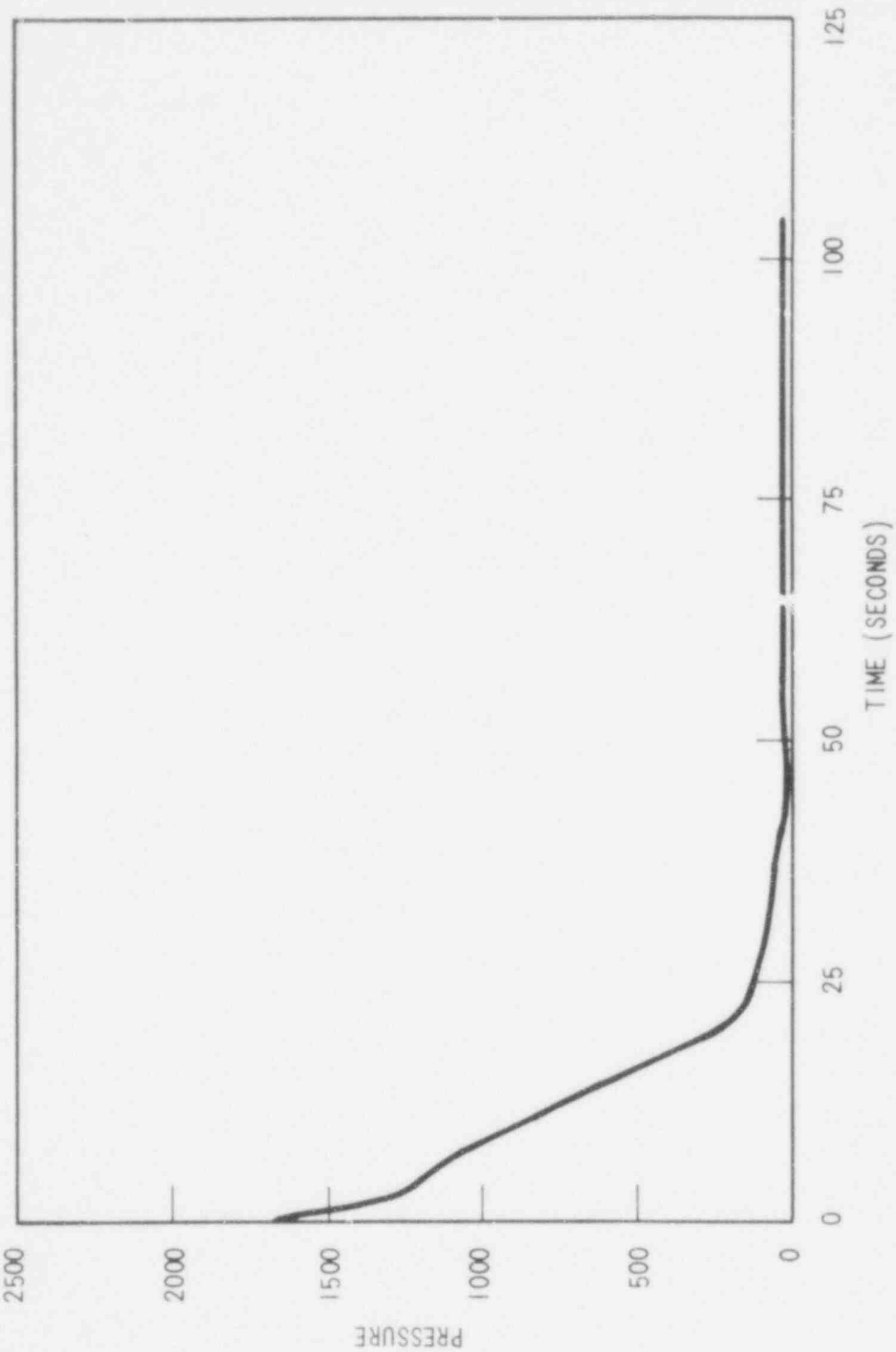
615 157

WCAP – 9500

Figure 15.6.5-20. BLUE

Core Pressure – DECLG (C_D = 0.8)

14,395-373



POOR ORIGINAL

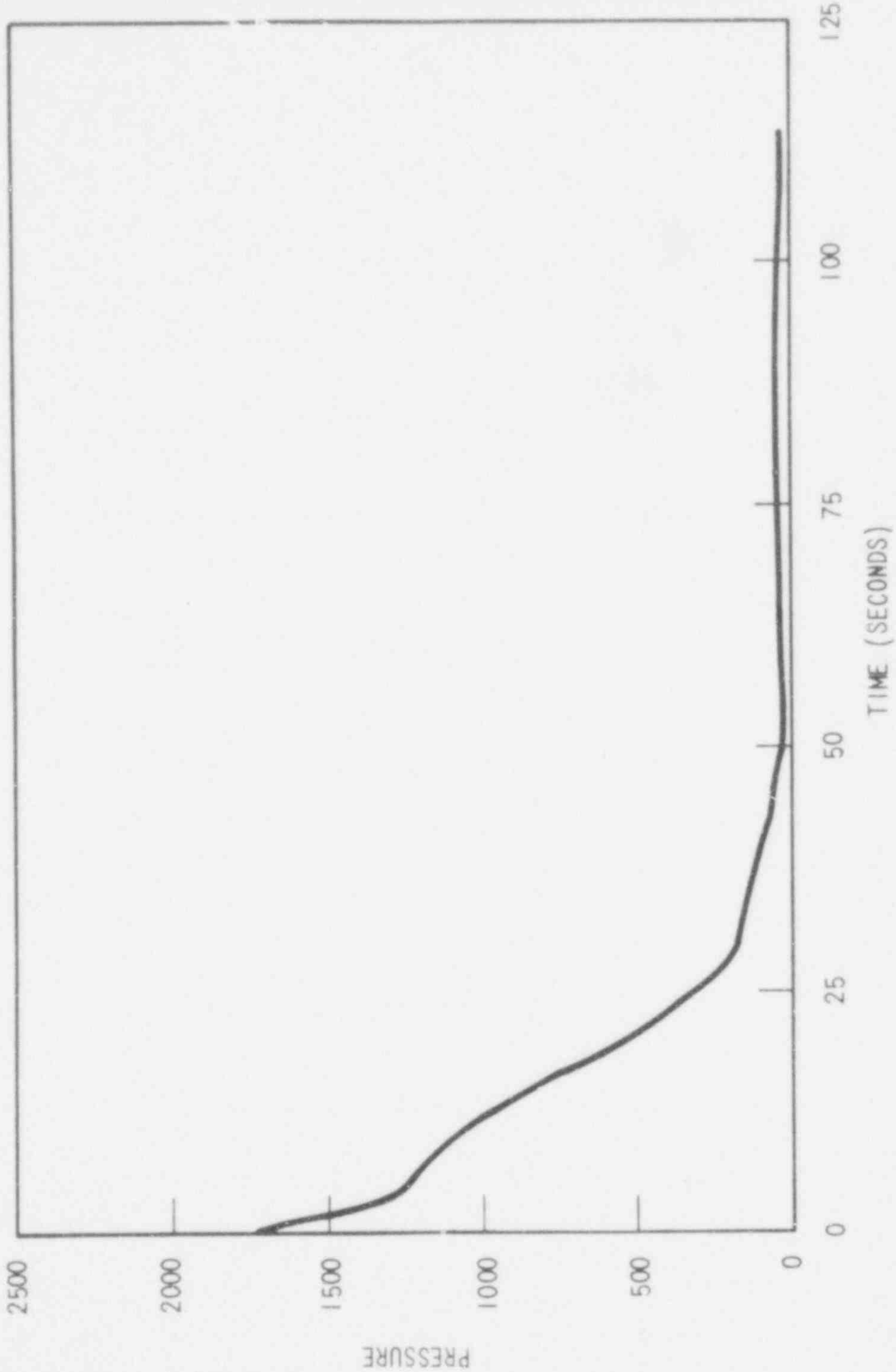
615 158

WCAP – 9500

Figure 15.6.5-21. BLUE

Core Pressure – DECLG (C_D = 0.6)

14,395-374



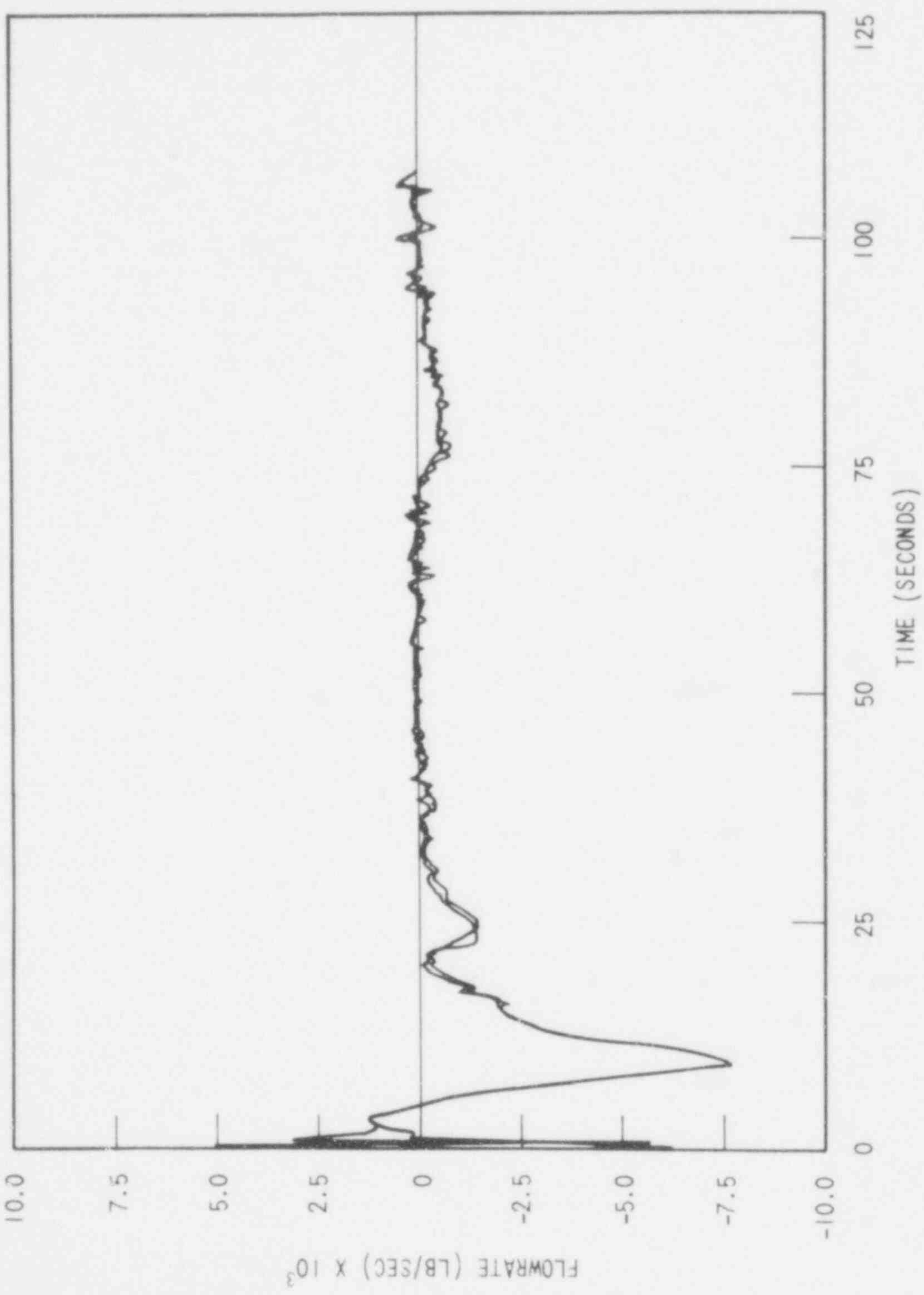
POOR ORIGINAL

615 159

WCAP – 9500

Figure 15.6.5-22. BLUE

Core Pressure – DECLG (C_D = 0.4)

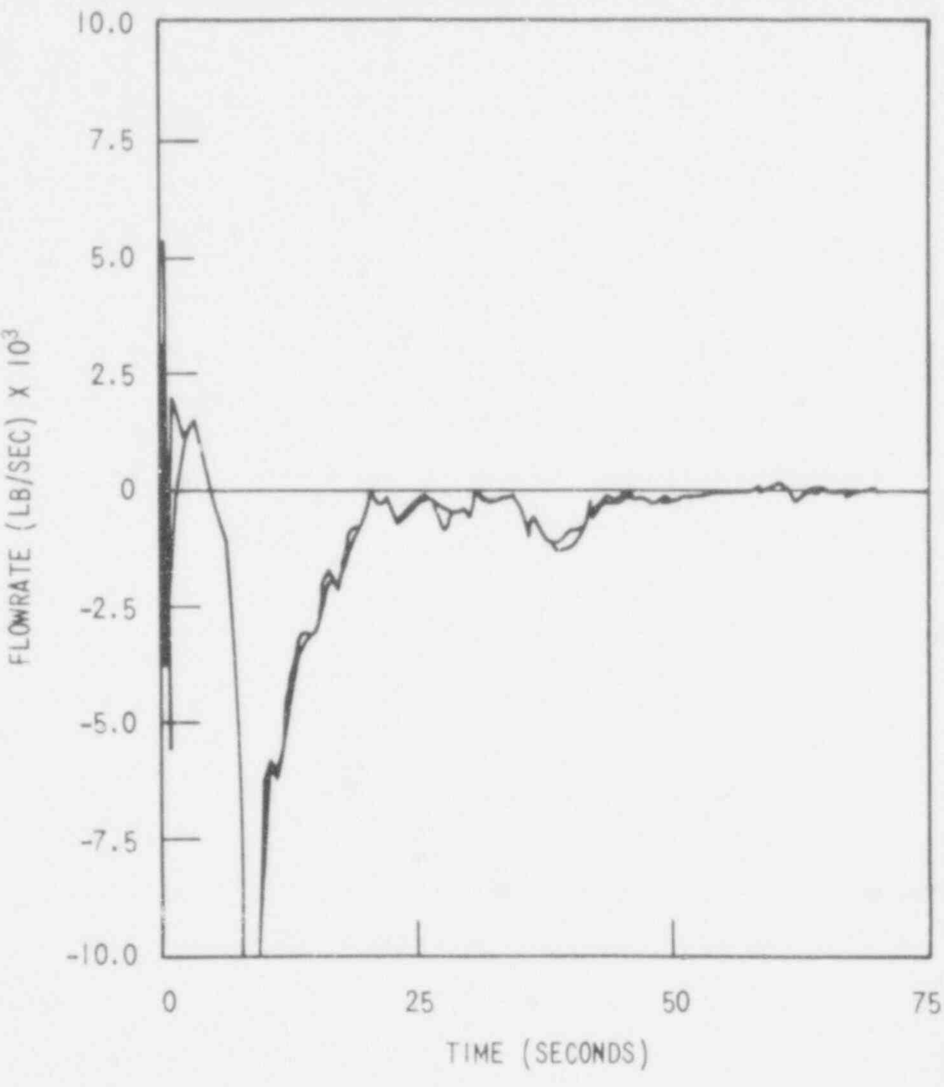


615 160

WCAP – 9500

Figure 15.6.5-23. BLUE

Flowrate at Lower Half and Midplane of Core, C_D = 1.0 DECLG, Perfect Mixing



POOR ORIGINAL

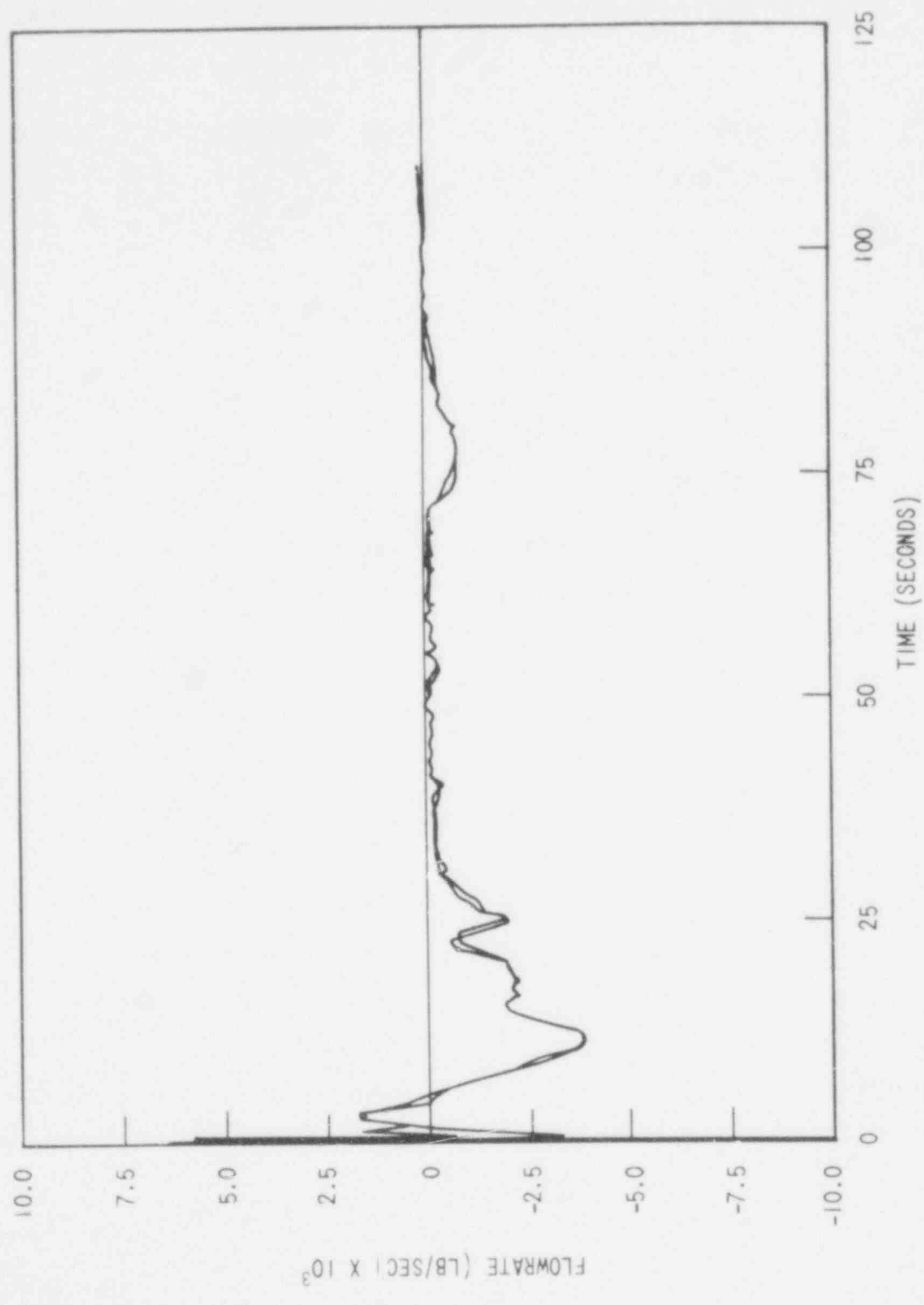
615 161

WCAP – 9500

Figure 15.6.5-24. BLUE

Flowrate at Lower Half and Midplane of Core, C_D = 1.0 DECLG, Imperfect Mixing

14,395-377



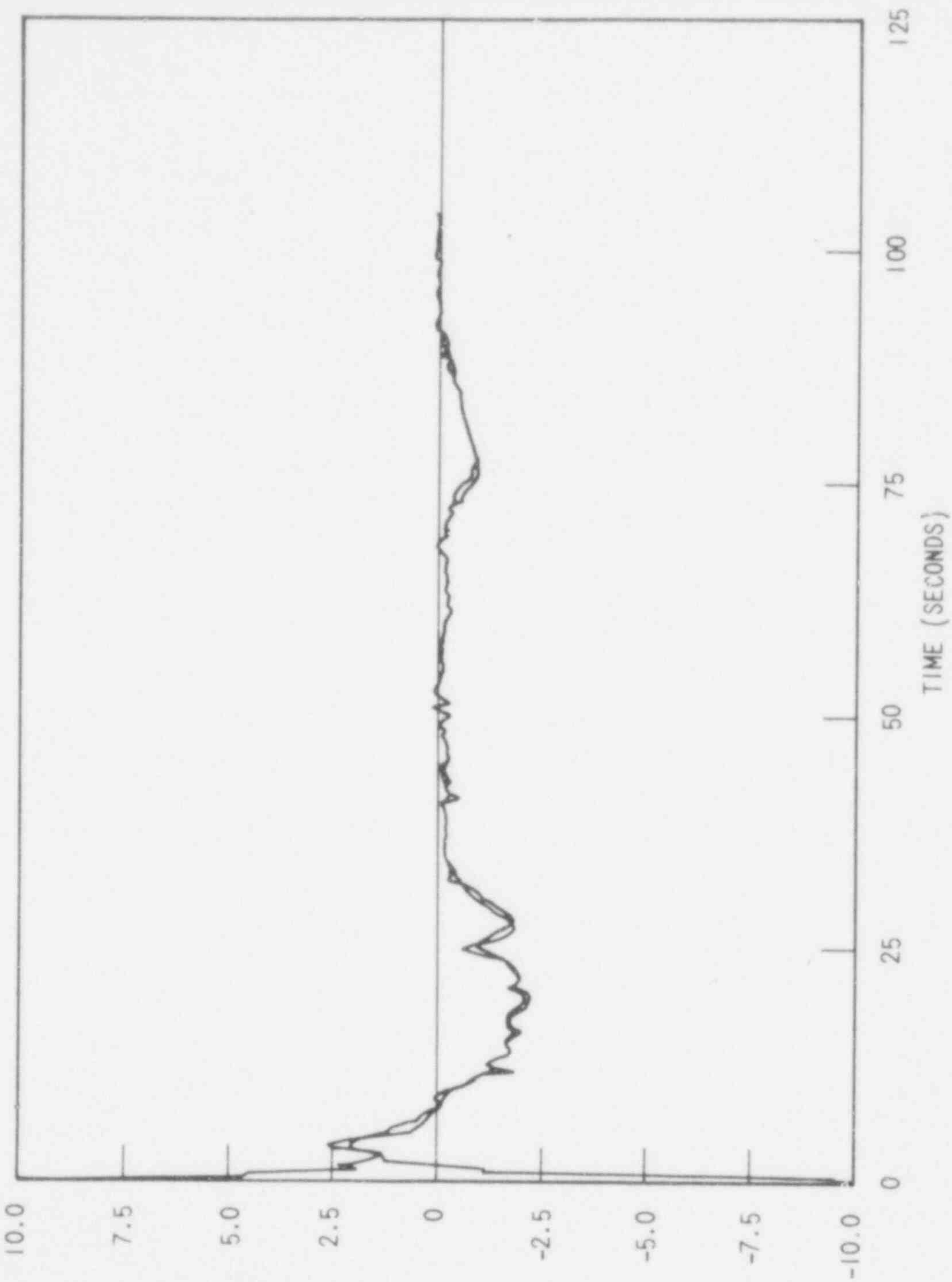
POOR ORIGINAL

WCAP - 9500

Figure 15.6.5-25. BLUE

Flowrate at Lower Half and Midplane of Core, C_D = 0.8 DECLG, Perfect Mixing

615 162



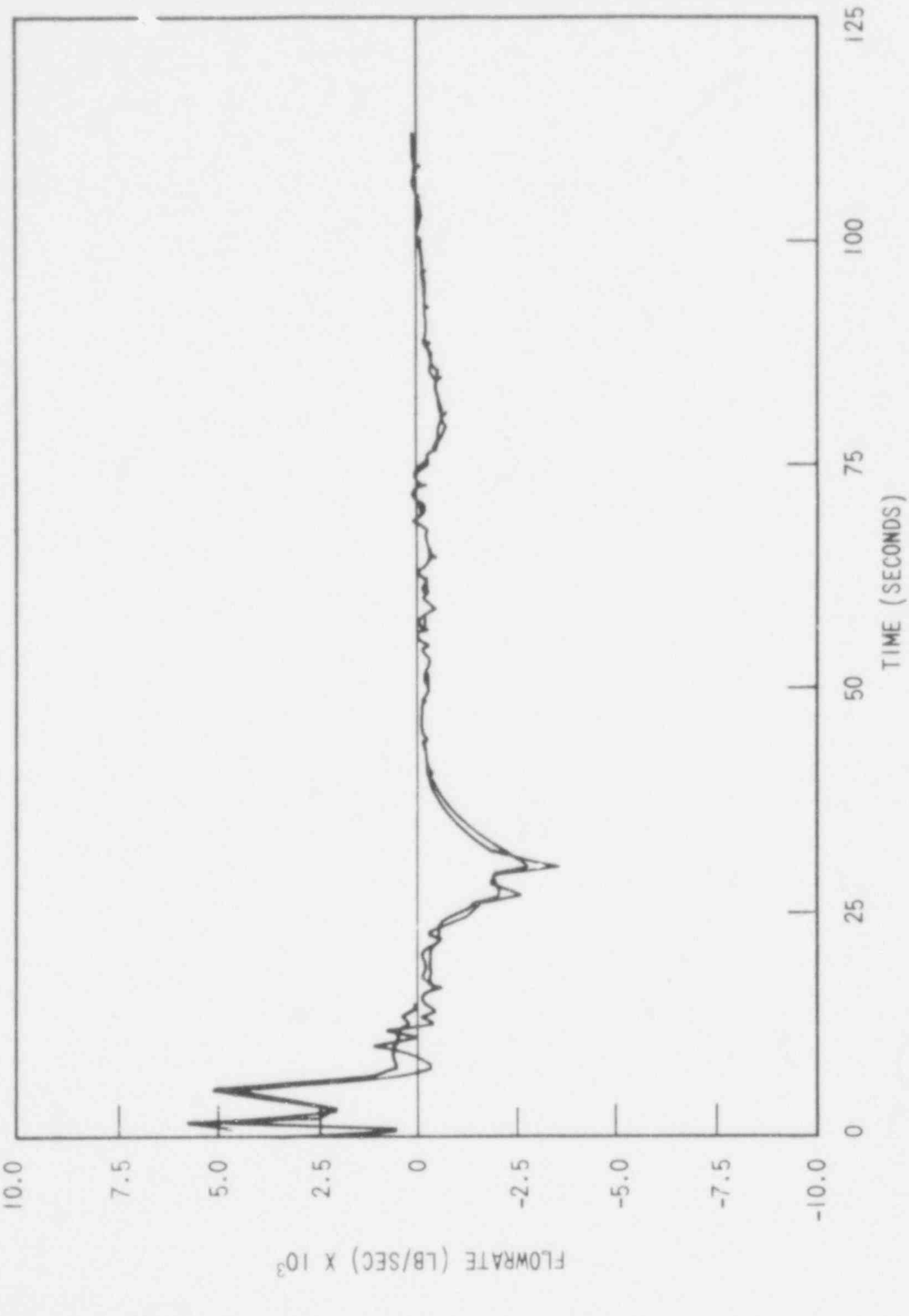
POOR ORIGINAL

615 163

WCAP – 9500

Figure 15.6.5-26. BLUE

Flowrate at Lower Half and Midplane of Core, C_D = 0.6 DECLG, Perfect Mixing



615 164

WCAP – 9500

Figure 15.6.5-27. BLUE

Flowrate at Lower Half and Midplane of Core, C_D = 0.4 DECLG, Perfect Mixing

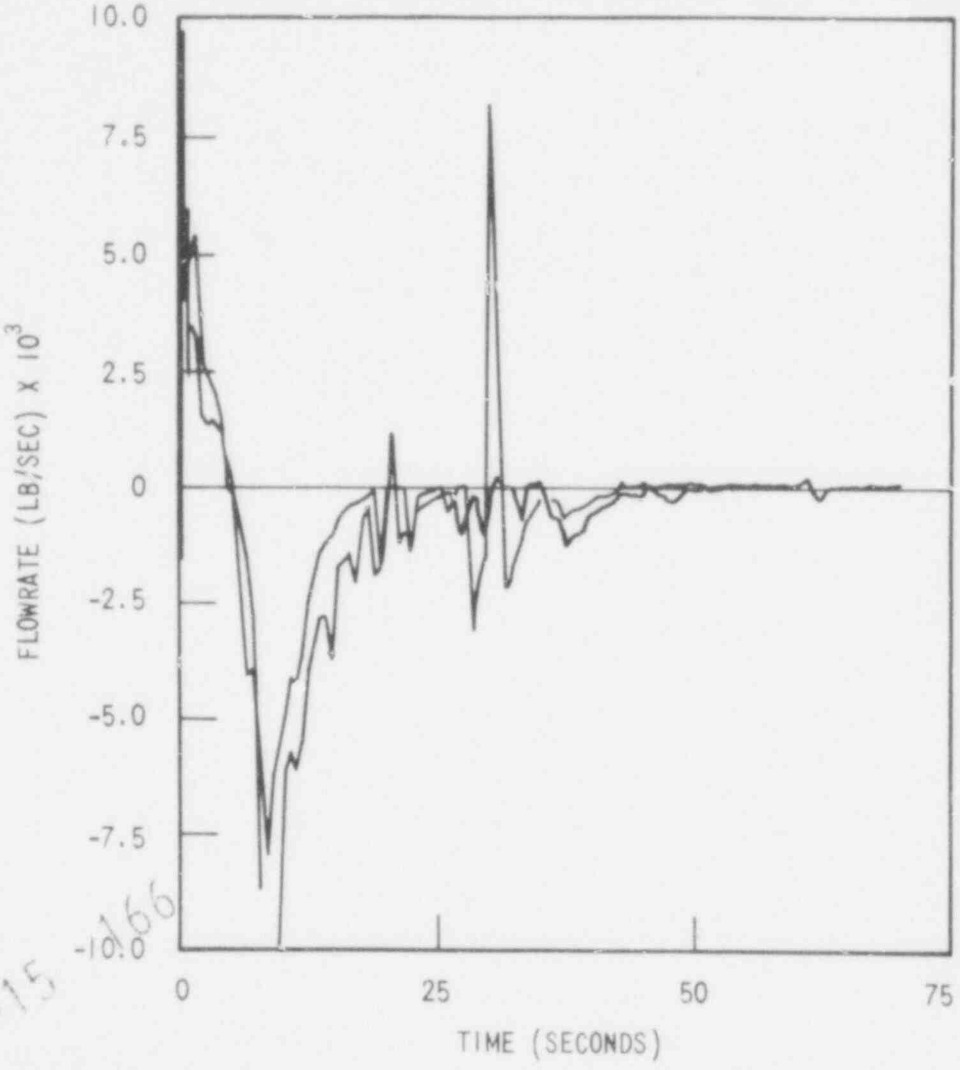
14,395-380

615 165

WCAP – 9500

Figure 15.6.5-28. BLUE

Flowrate at Upper Half and Top of Core, C_D = 1.0 DECLG, Perfect Mixing



515 166

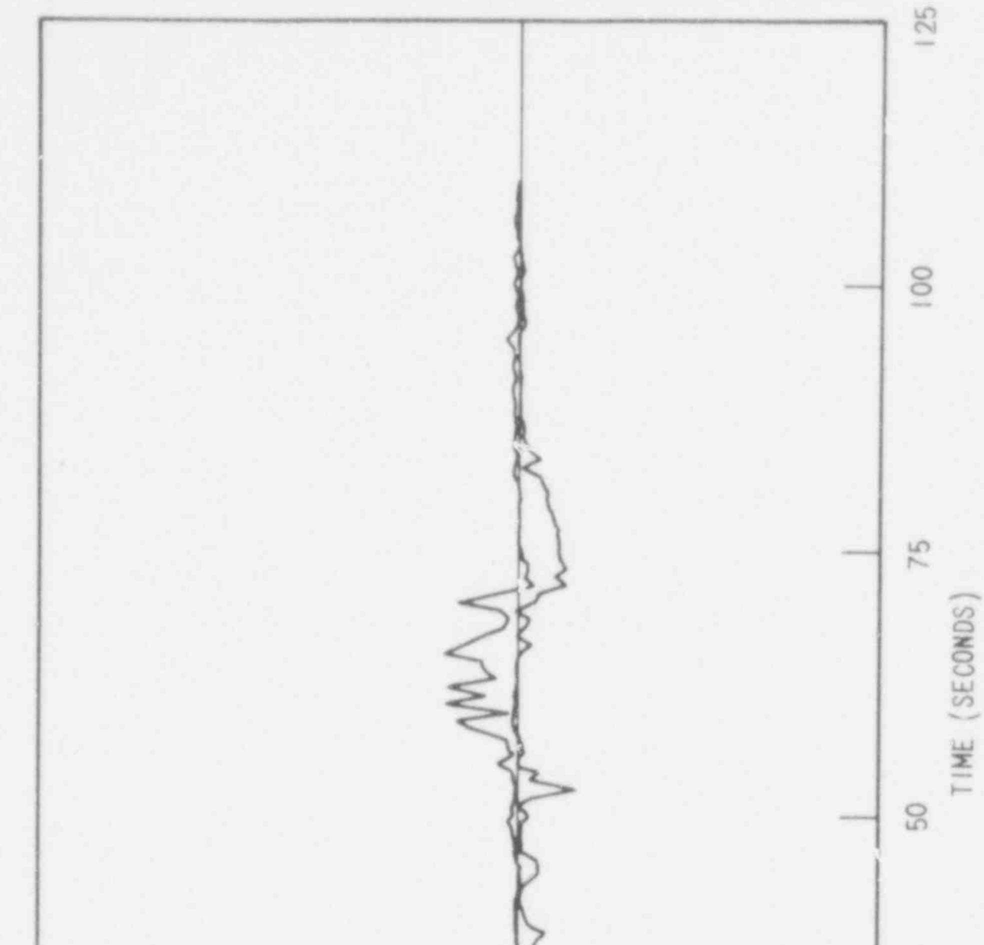
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-29. BLUE

Flowrate at Upper Half and Top of Core, C_D = 1.0 DECLG, Imperfect Mixing

14,395-382



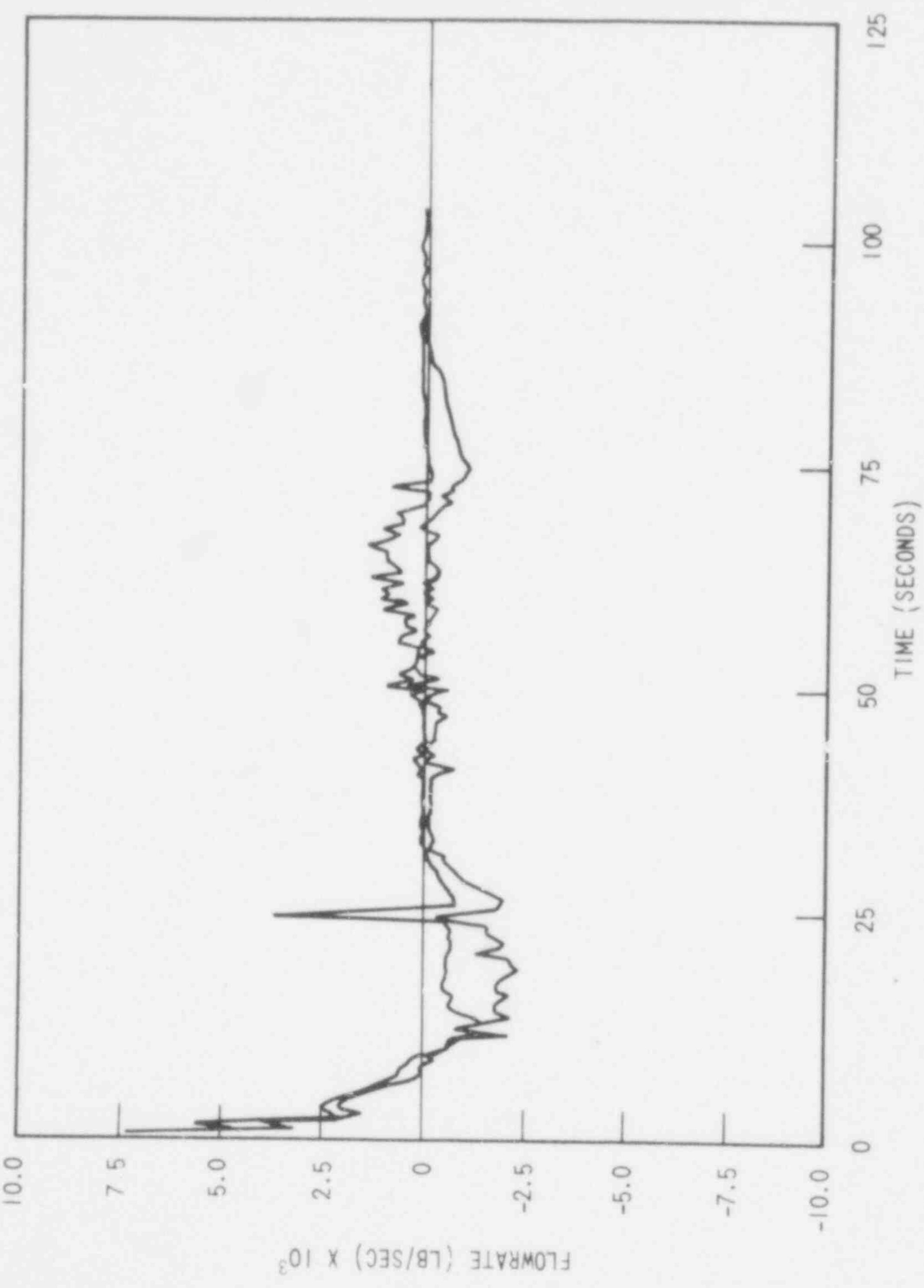
POOR ORIGINAL

615 167

WCAP – 9500

Figure 15.6.5-30. BLUE

Flowrate at Upper Half and Top of Core, C_D = 0.8 DECLG, Perfect Mixing



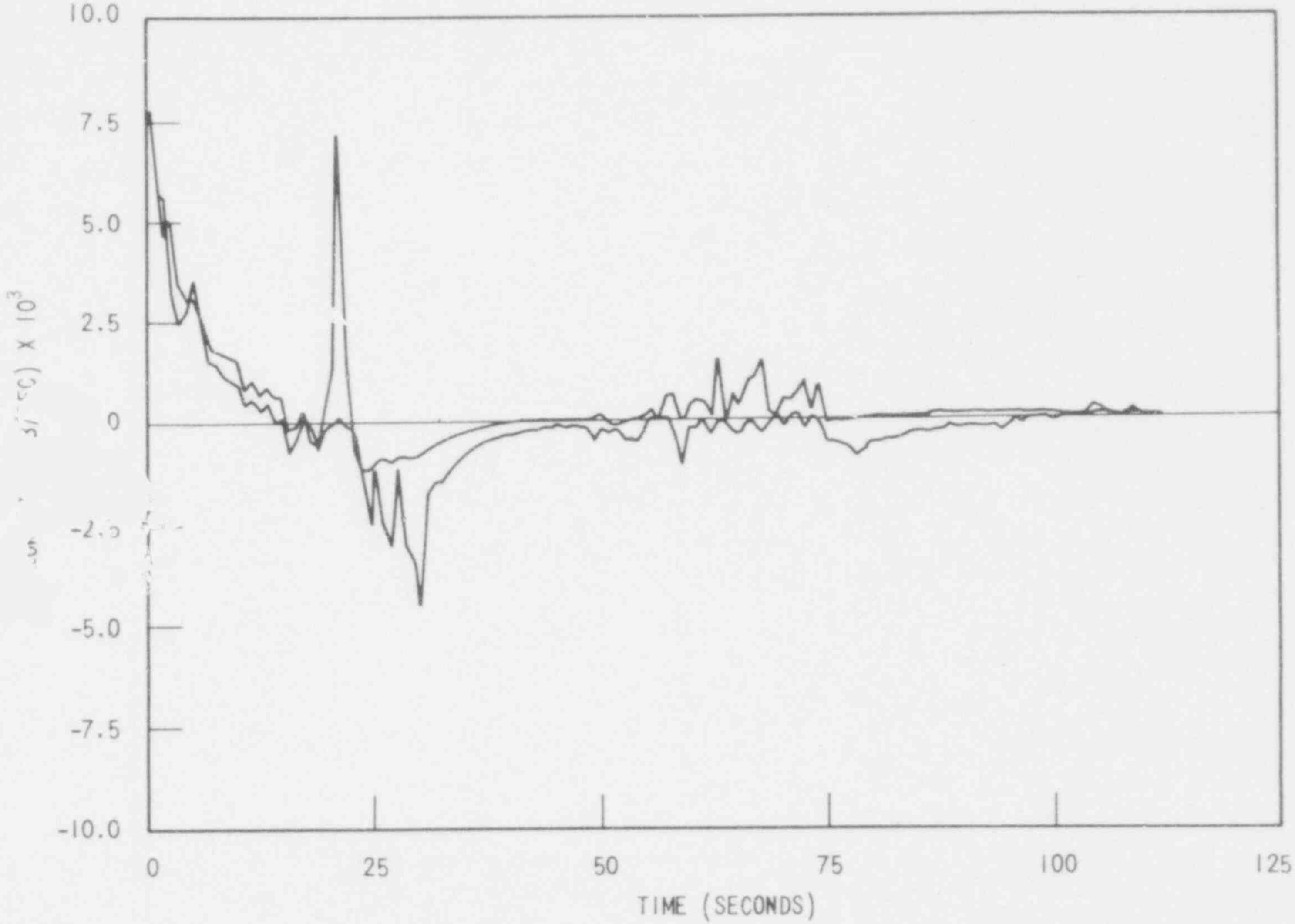
615 168

WCAP – 9500

Figure 15.6.5-31. BLUE

Flowrate at Upper Half and Top of Core, C_D = 0.6 DECLG, Perfect Mixing

615 169

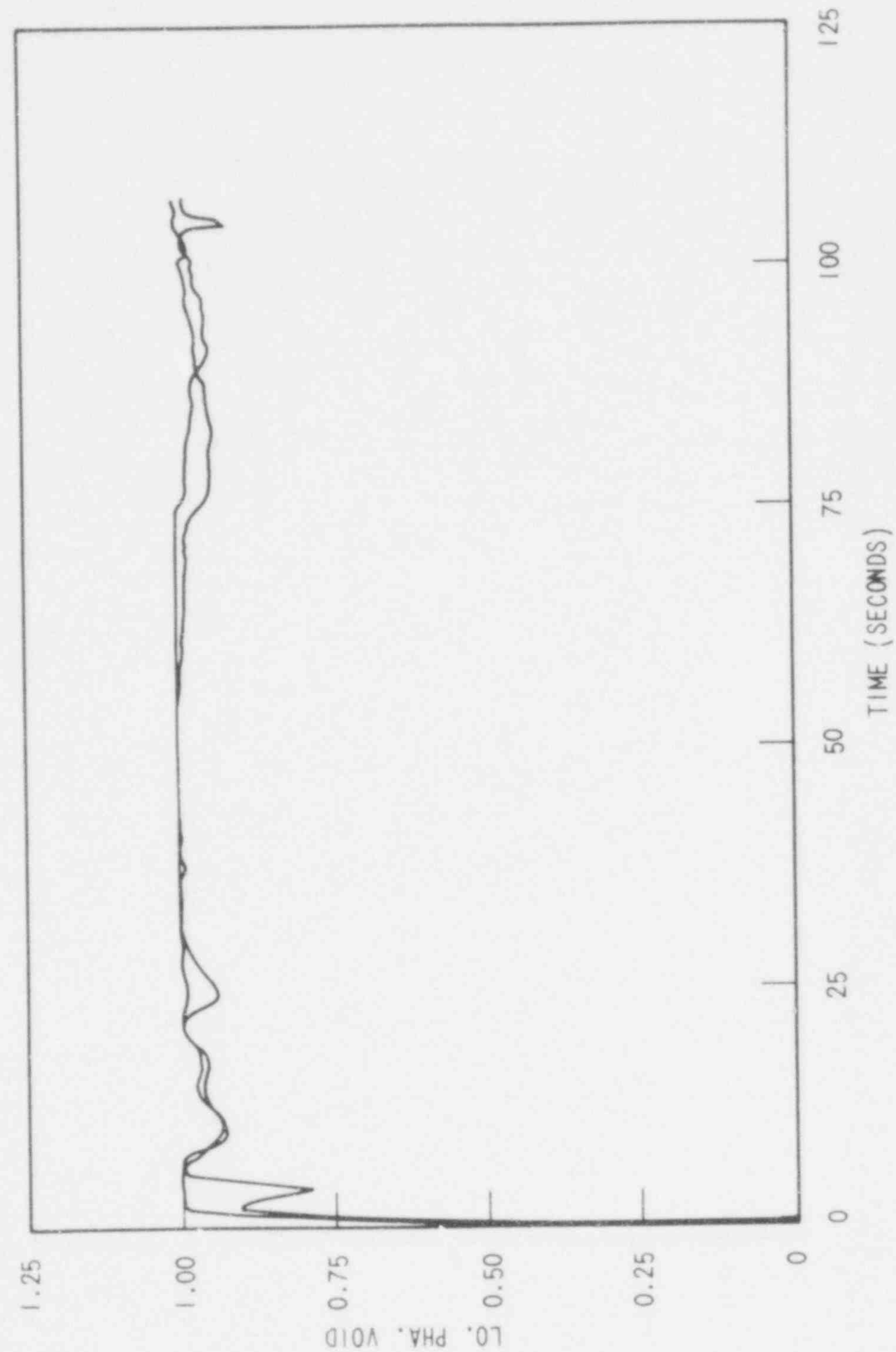


WCAP-9500

Figure 15.6.5-32. BLUE
Flowrate at Upper Half and Top of Core,
C_D = 0.4 DECLG, Perfect Mixing

14,395-384

14,395-385

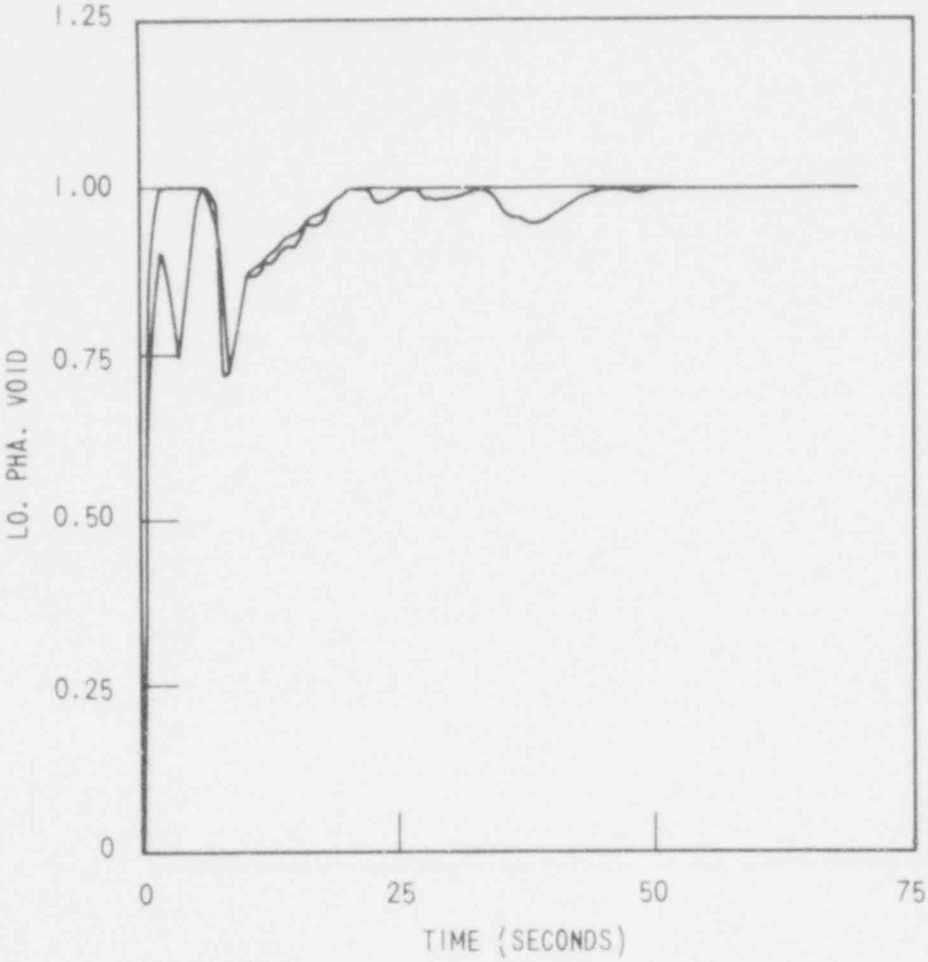


WCAP – 9500

Figure 15.6.5-33. BLUE

Void Fraction in Lower Half of Core, C_D = 1.0 DECLG, Perfect Mixing

615 170



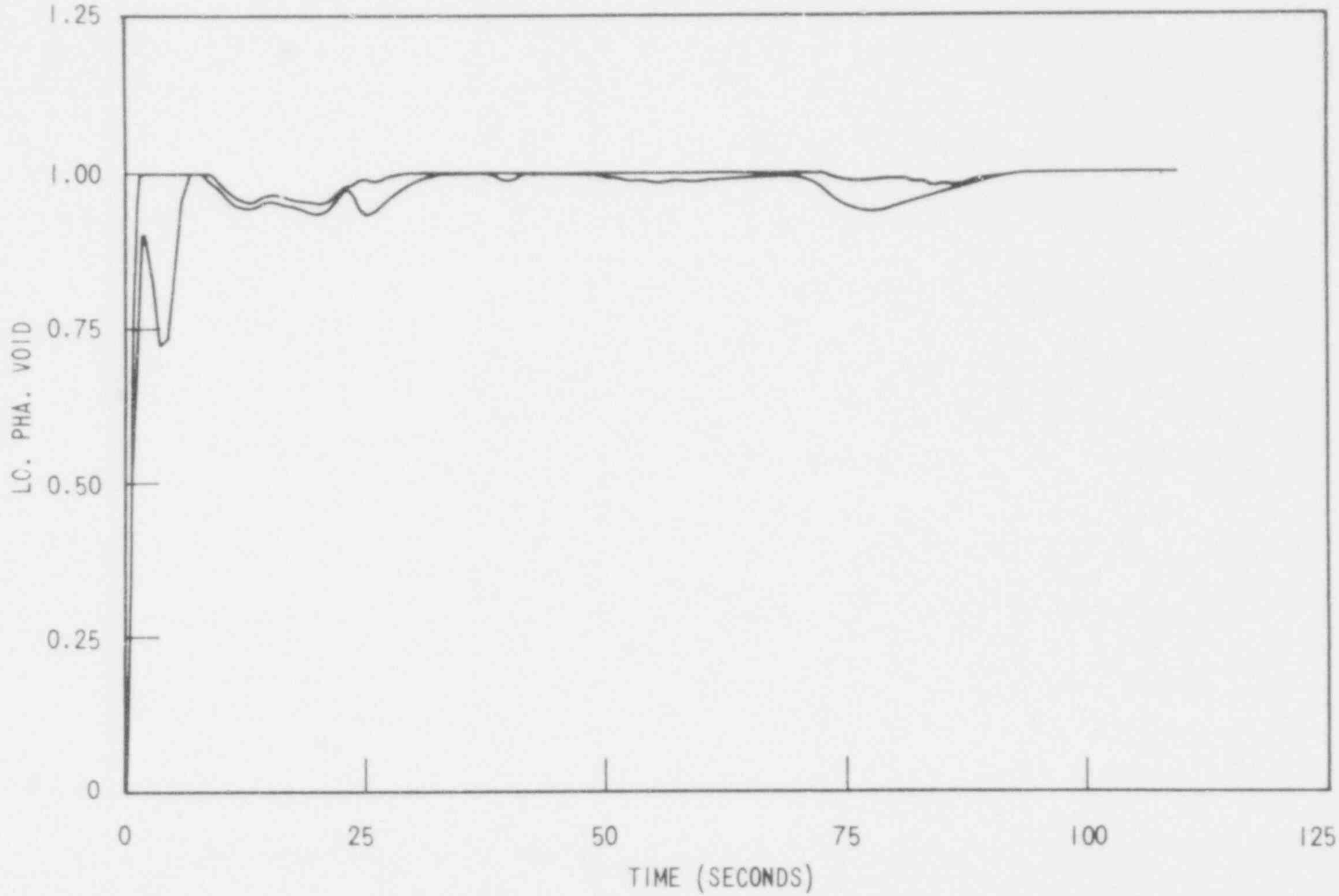
WCAP – 9500

Figure 15.6.5-34. BLUE

Void Fraction in Lower Half of Core, C_D = 1.0 DECLG, Imperfect Mixing

615 171

615 172



WCAP - 9500

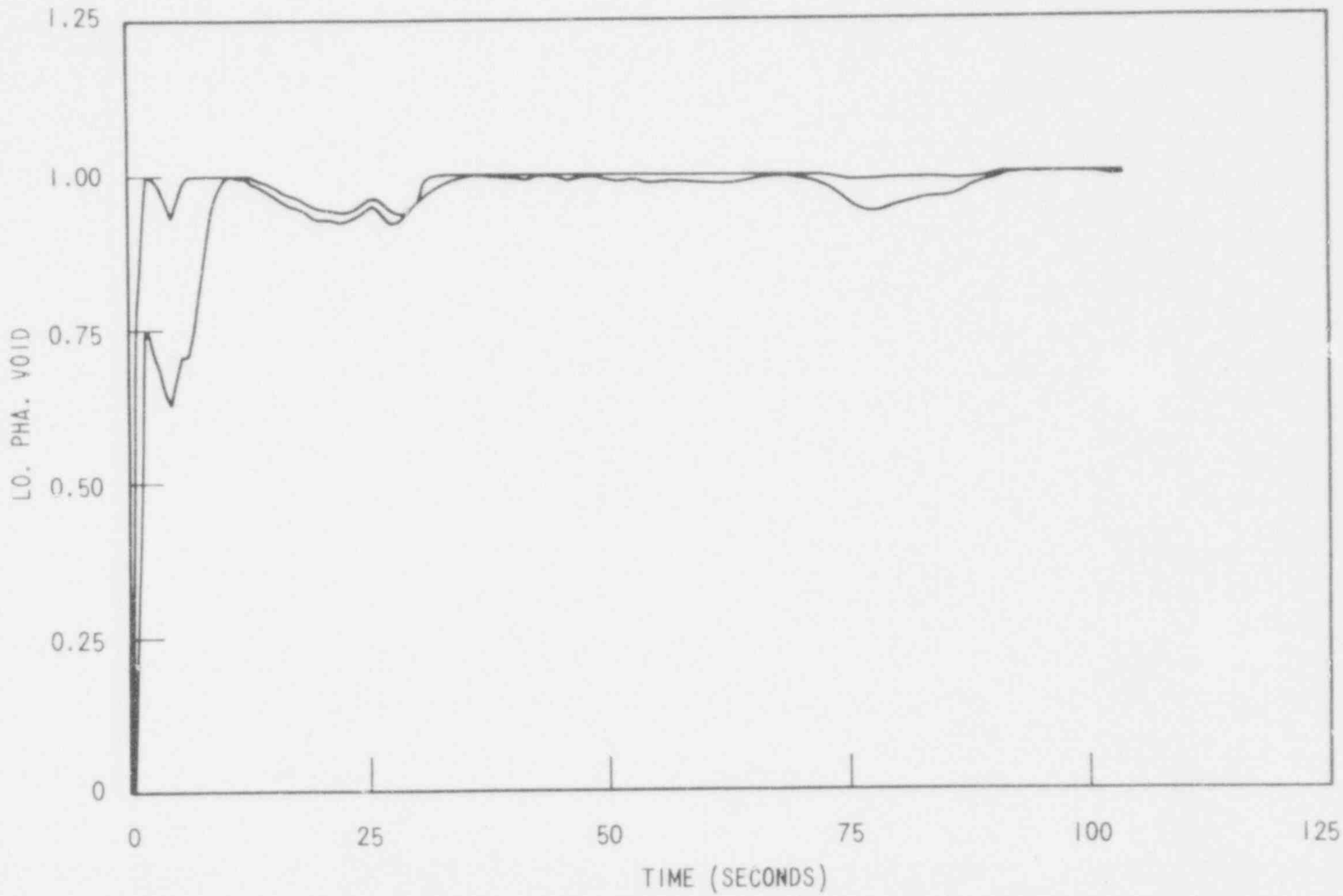
Figure 15.6.5-35. BLUE

Void Fraction in Lower Half of Core, C_D = 0.8 DECLG, Perfect Mixing

14,395-387

POOR ORIGINAL

615 173

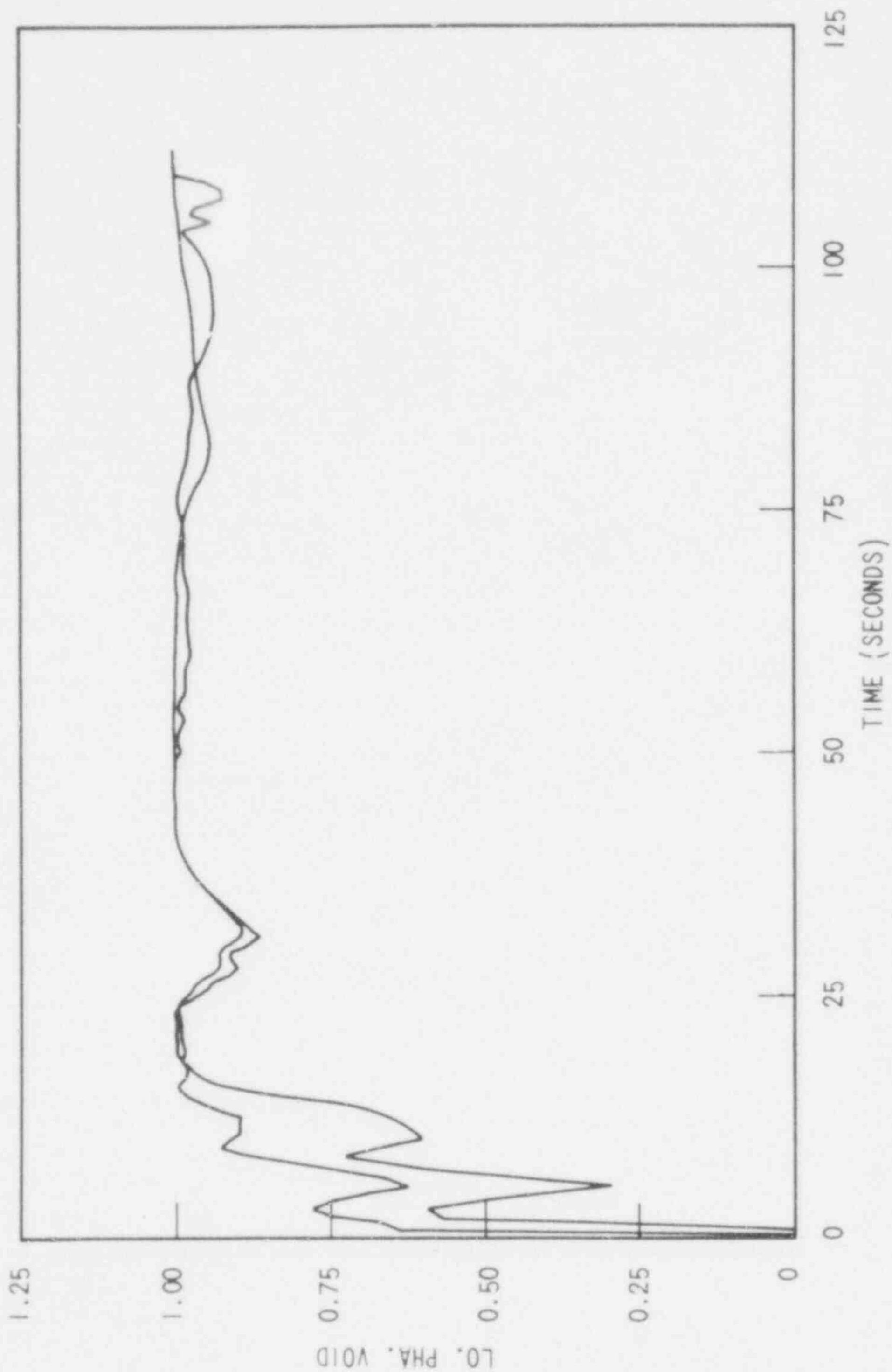


WCAP-9500

Figure 15.6.5.36. BLUE

Void Fraction in Lower Half of Core, C_D = 0.6 DECLG, Perfect Mixing

14,395-388



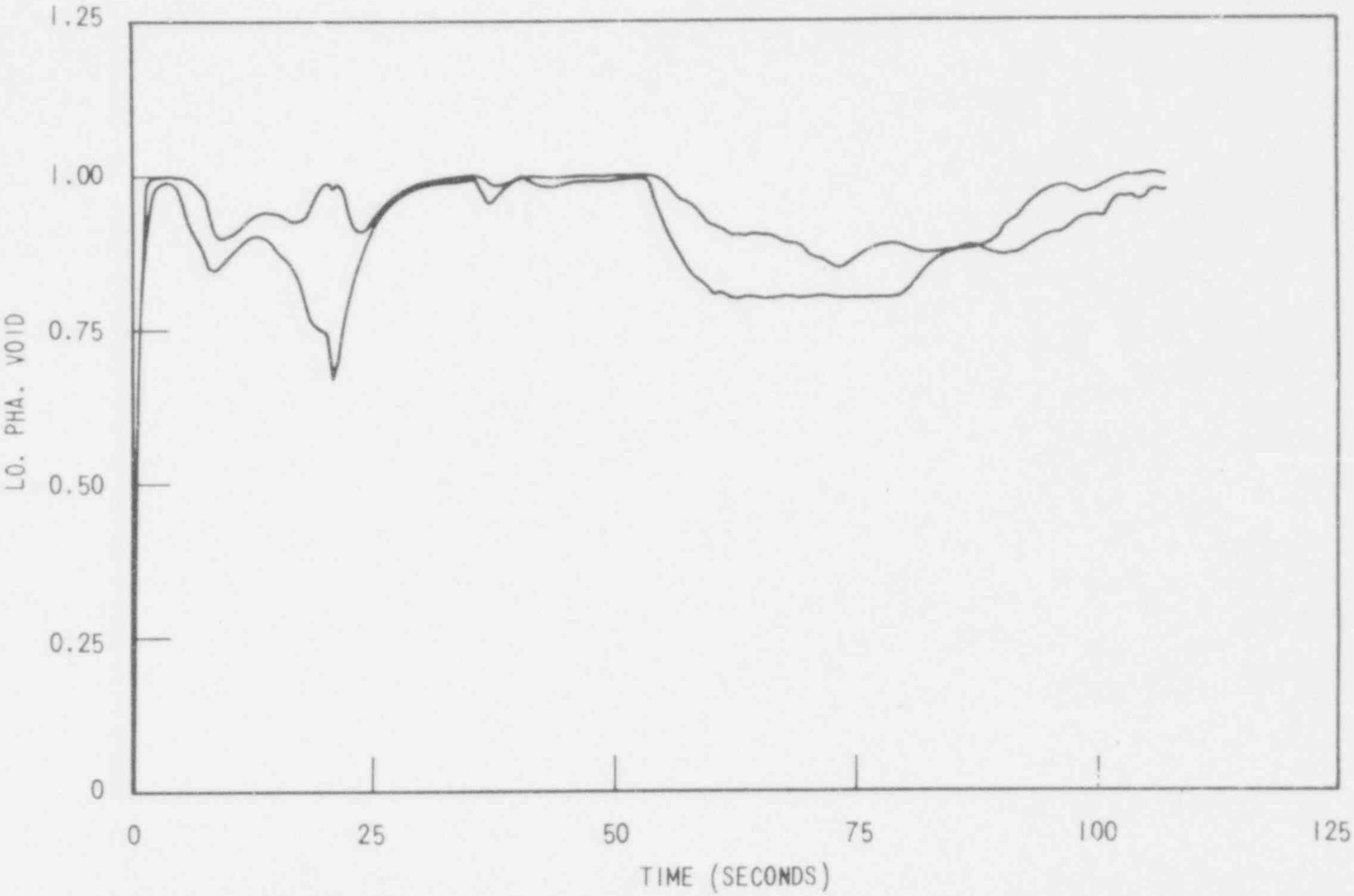
POOR ORIGINAL

WCAP – 9500
Figure 15.6.5-37. BLUE Void Fraction In Lower Half of Core, C_D = 0.4 DECLG, Perfect Mixing

615 174

POOR ORIGINAL

615 175

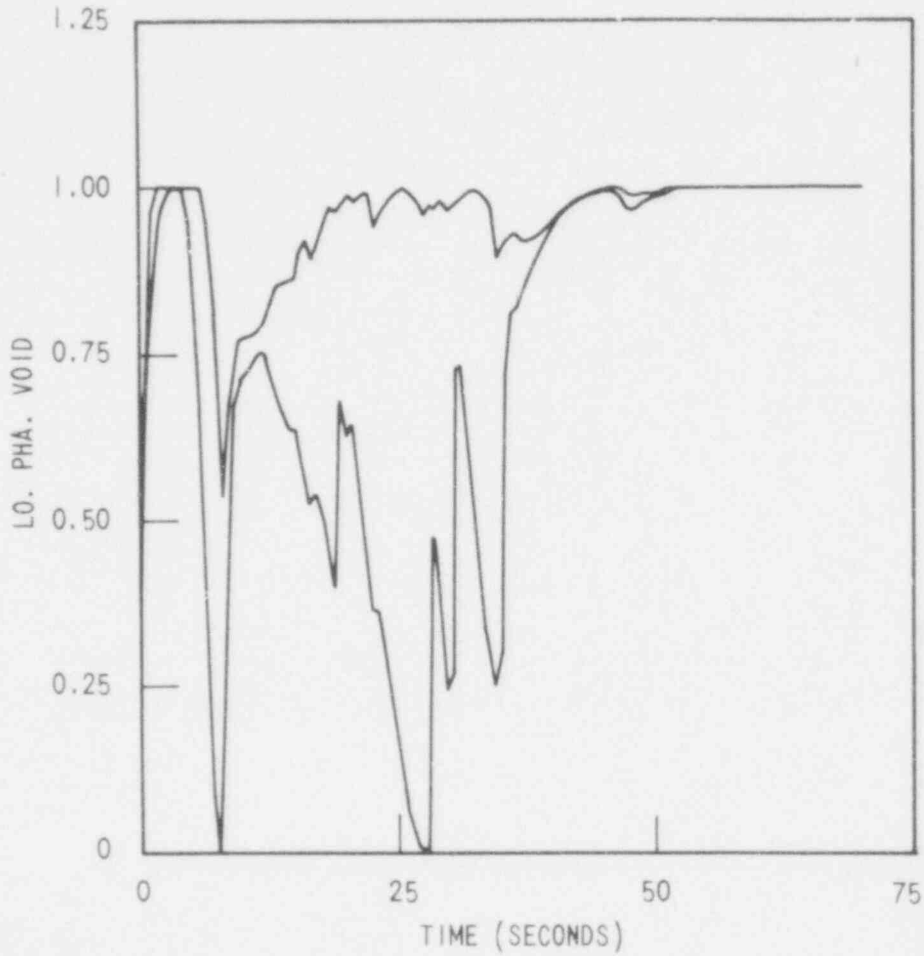


WCAP-9500

Figure 15.6.5-38. BLUE

Void Fraction in Upper Half of Core, C_D = 1.0 DECLG, Perfect Mixing

14,395-390



POOR ORIGINAL

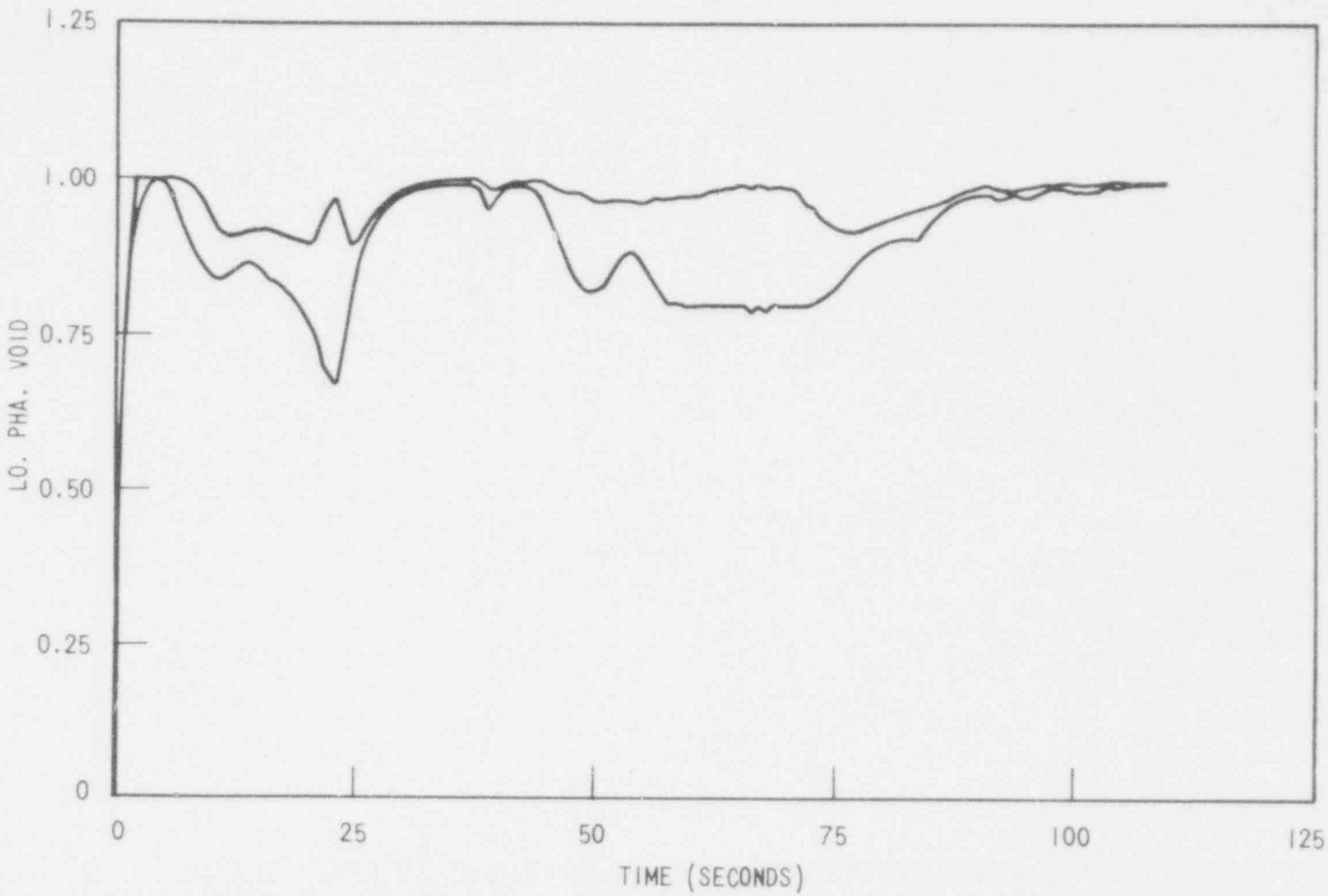
WCAP--9500

Figure 15.6.5-39. BLUE

Void Fraction in Upper Half of Core, C_D = 1.0 DECLG, Imperfect Mixing

615 176

615 177



WCAP - 9500

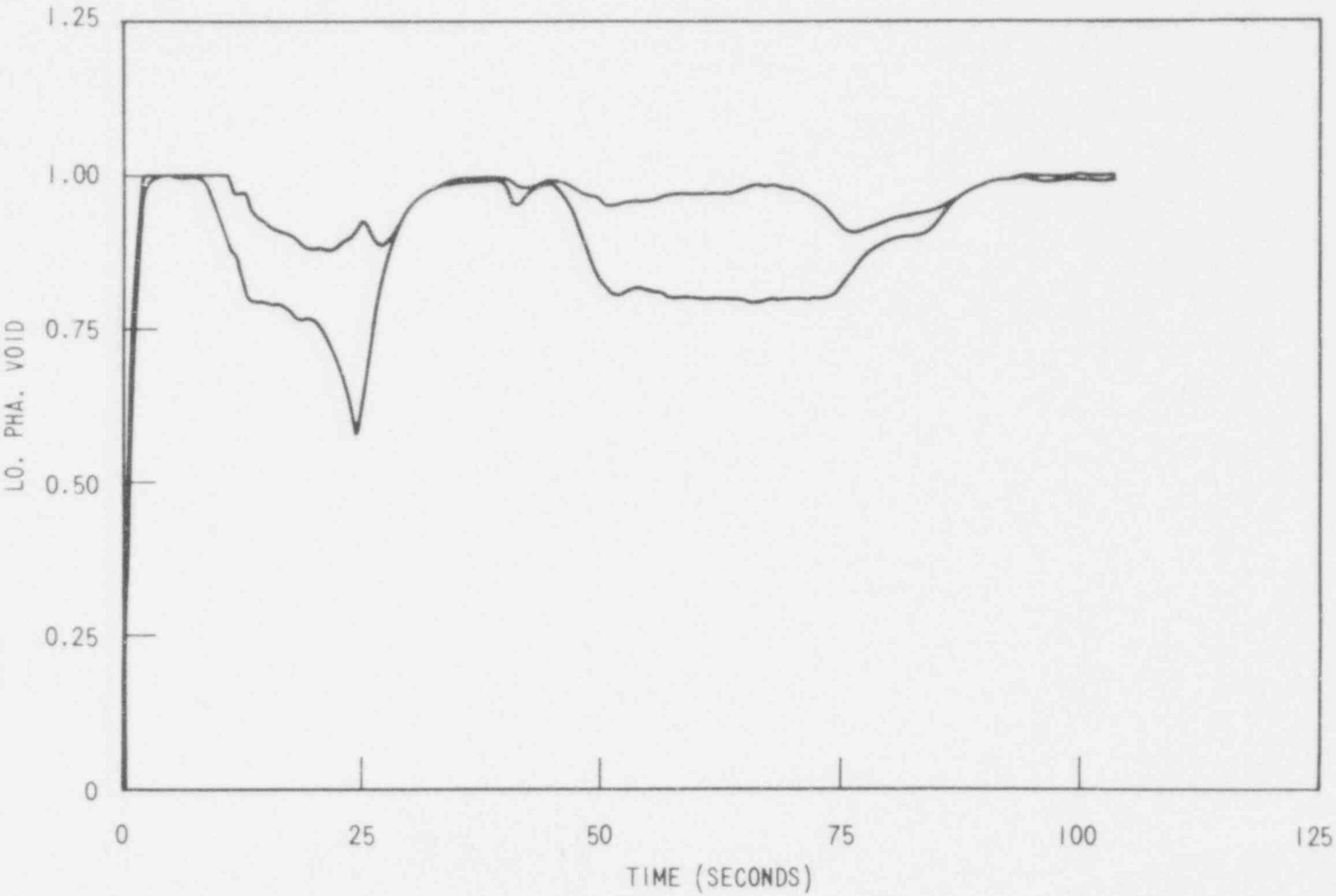
Figure 15.6.5-40. BLUE

Void Fraction in Upper Half of Core, C_D = 0.8 DECLG, Perfect Mixing

14,395-392

POOR ORIGINAL

615 178

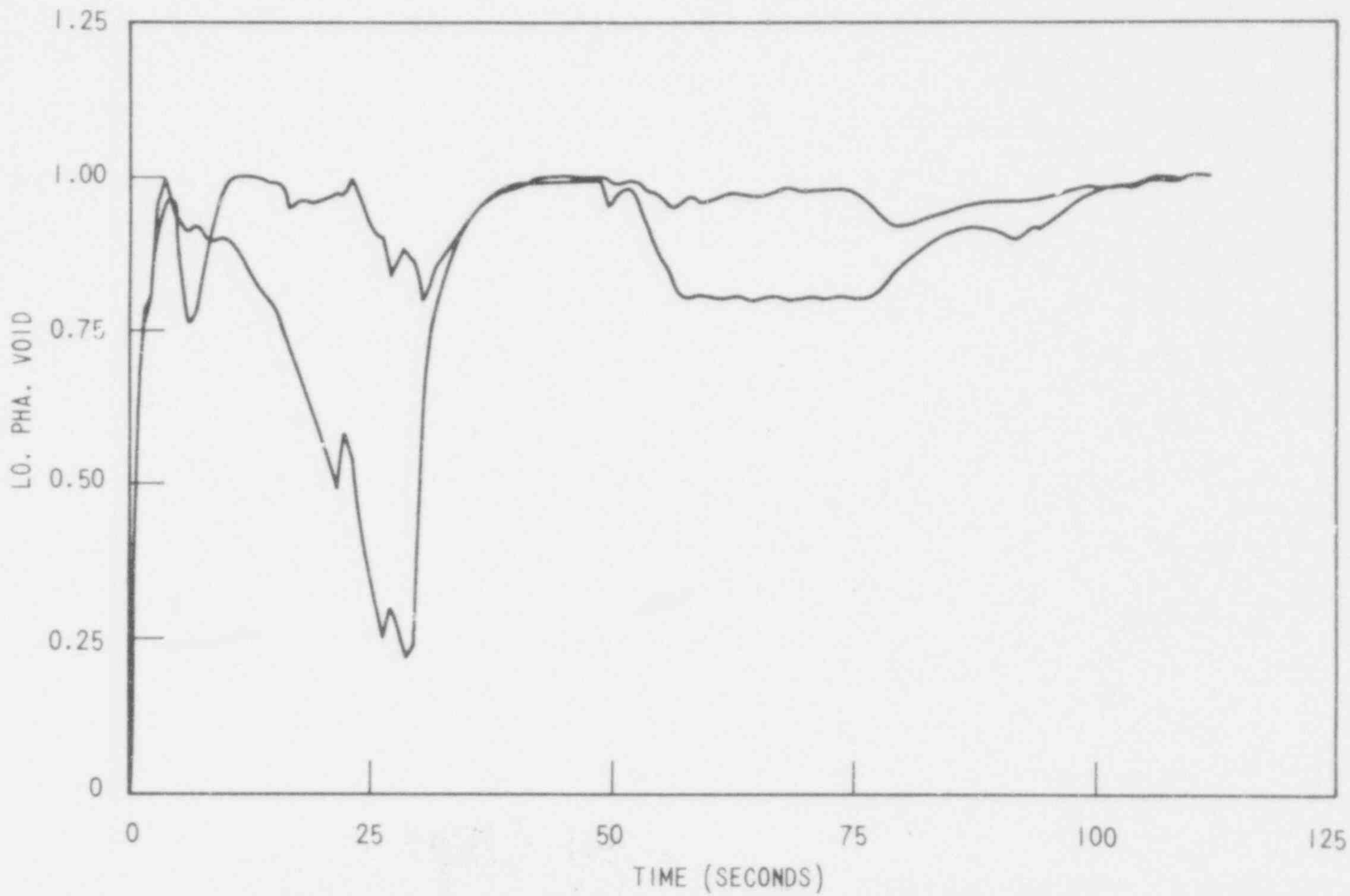


WCAP-9500

Figure 15.6.5-41. BLUE

Void Fraction in Upper Half of Core,
C_D = 0.6 DECLG, Perfect Mixing

14,395-396



WCAP-9500

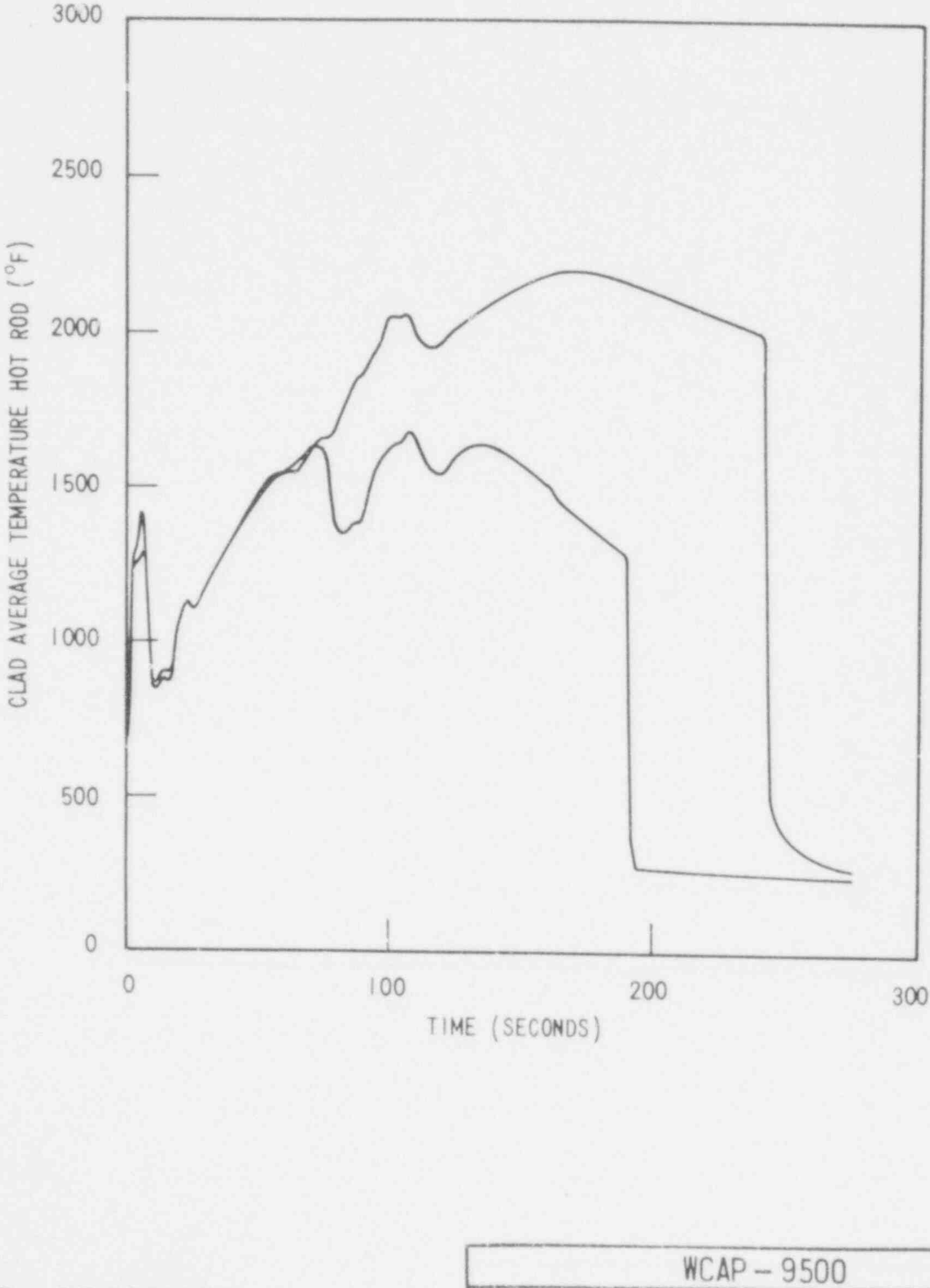
Figure 15.6.5-42. BLUE

Void Fraction in Upper Half of Core, C_D = 0.4 DECLG, Perfect Mixing

POOR ORIGINAL

615 179

14,395-394

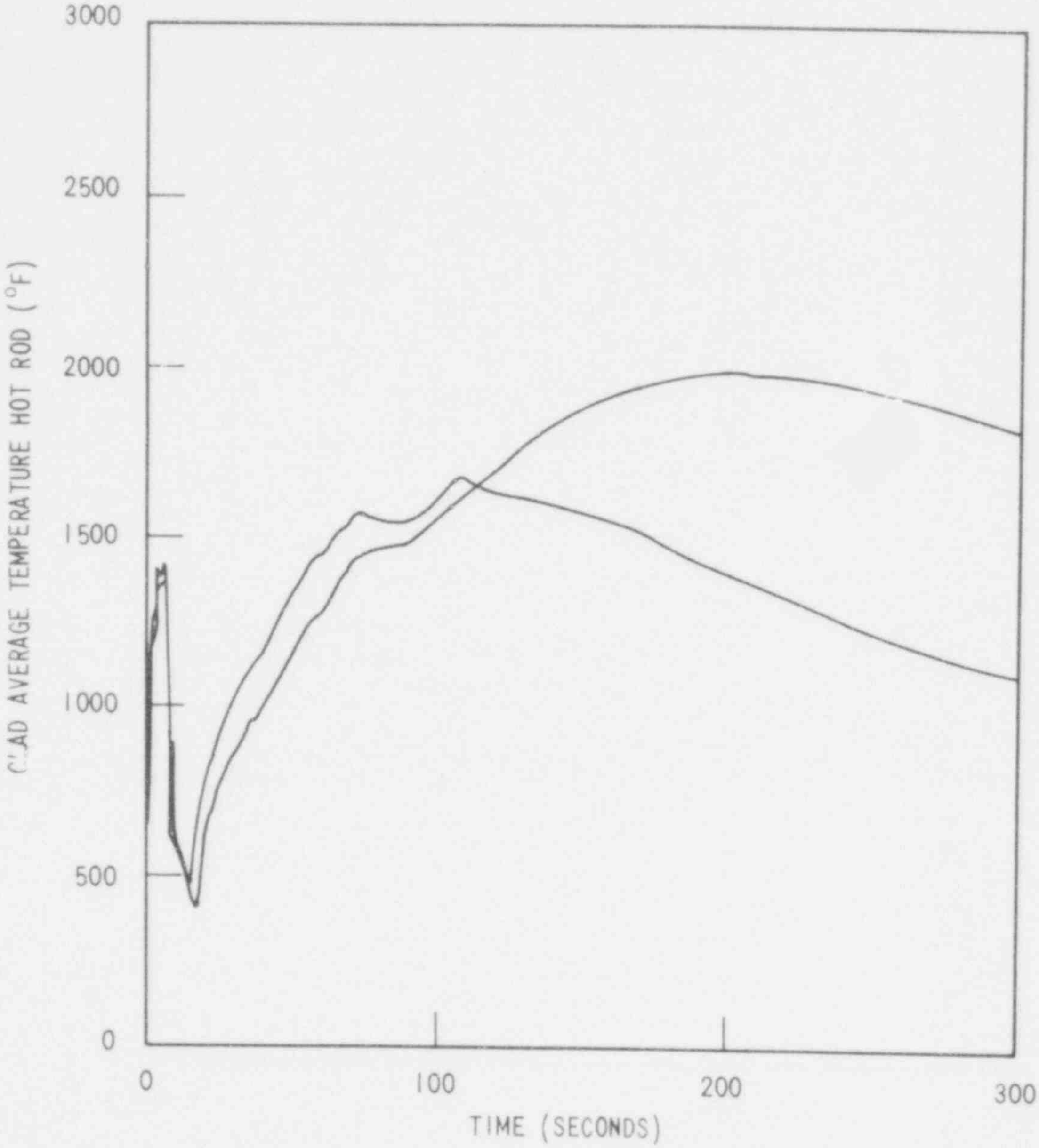


180

WCAP – 9500

Figure 15.6.5-43. BLUE

Peak Clad Temperature – Nodes 9 and 13, C_D = 1.0 DECLG, Perfect Mixing



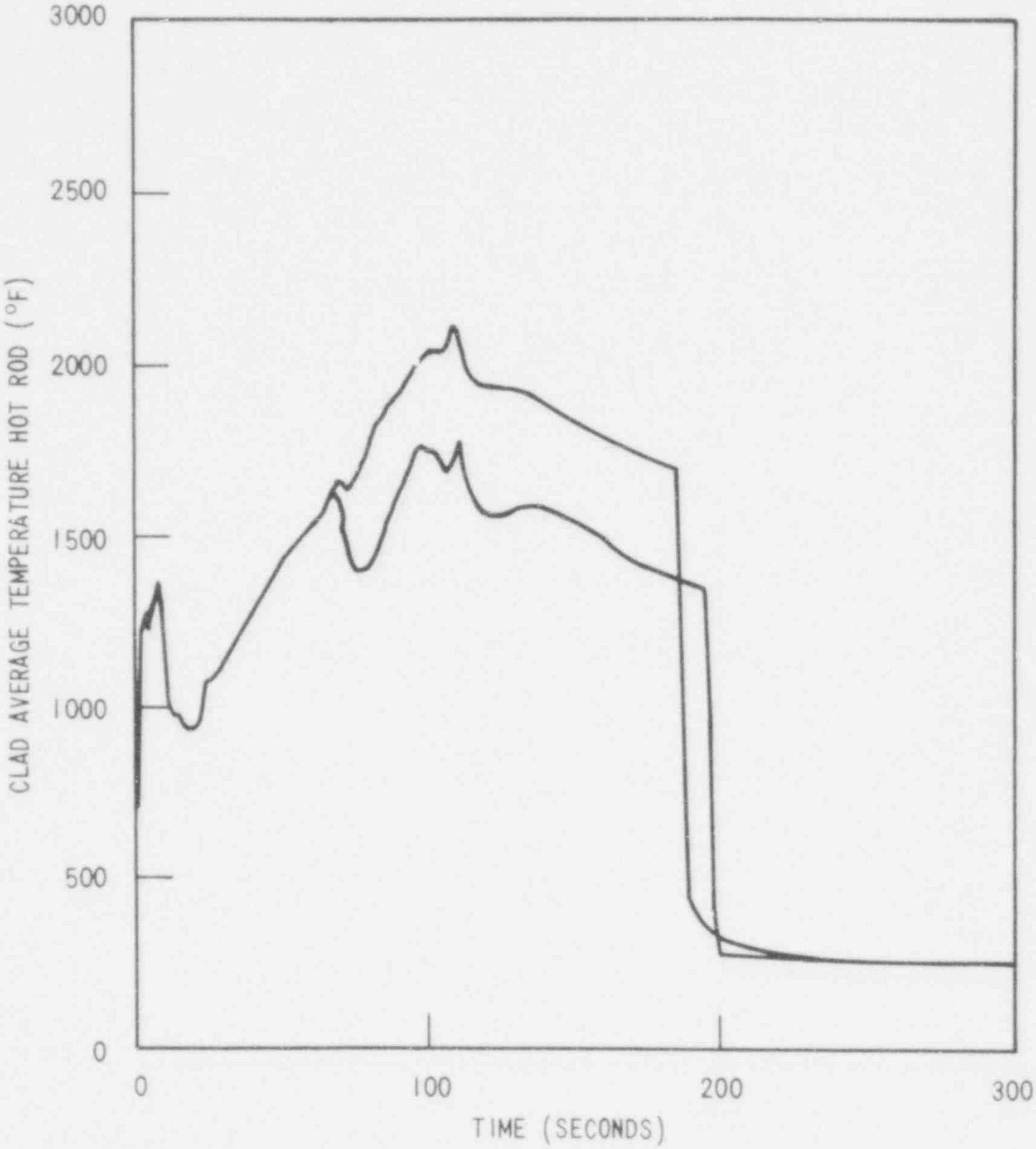
POOR ORIGINAL

615 181

WCAP - 9500

Figure 15.6.5-44. BLUE

Peak Clad Temperature – Nodes 13 and 14, C_D = 1.0 DECLG, Imperfect Mixing



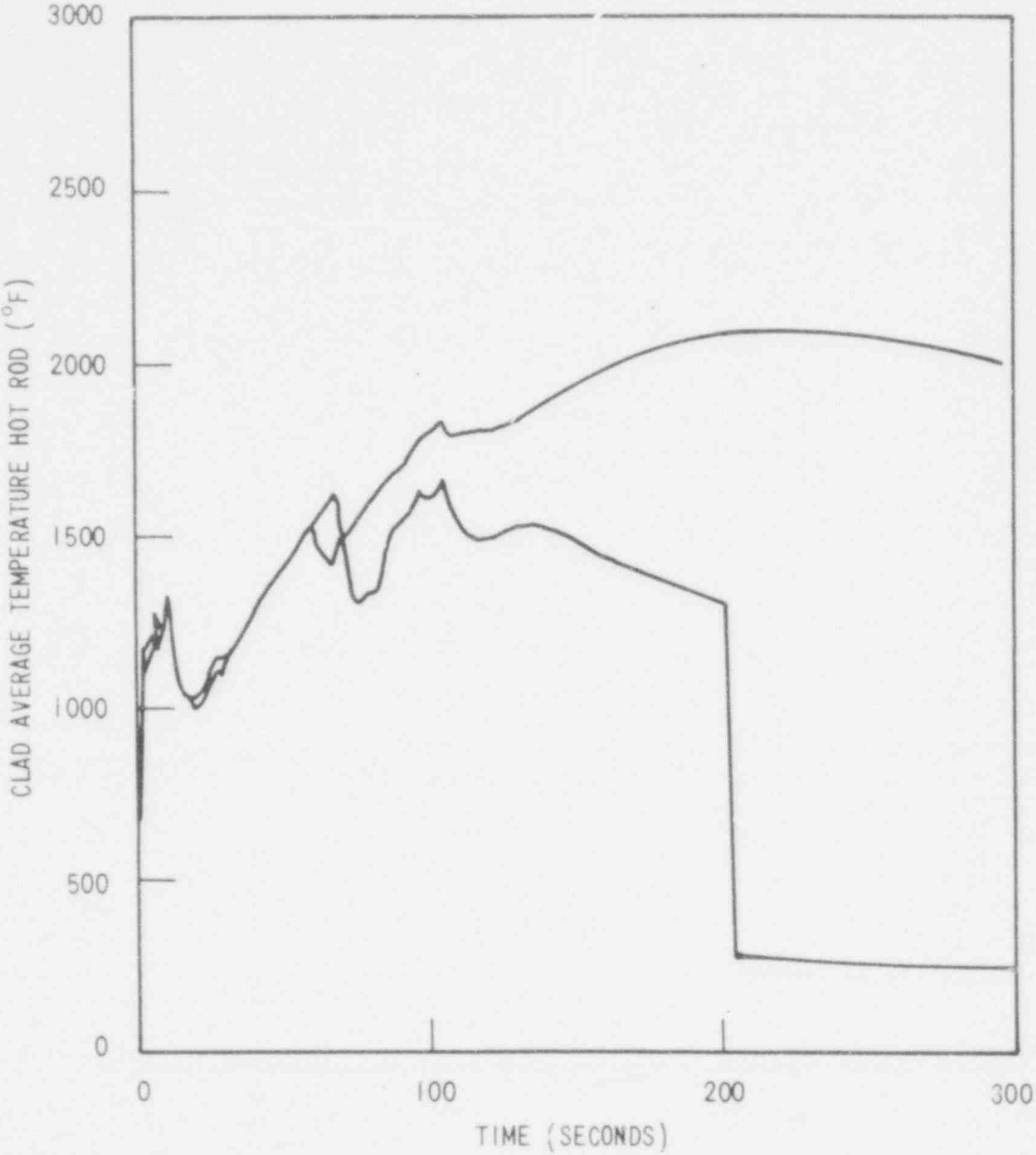
POOR ORIGINAL

615 182

WCAP – 9500

Figure 15.6.5-45. BLUE

Peak Clad Temperature – Nodes 13 and 14, C_D = 0.8 DECLG, Perfect Mixing

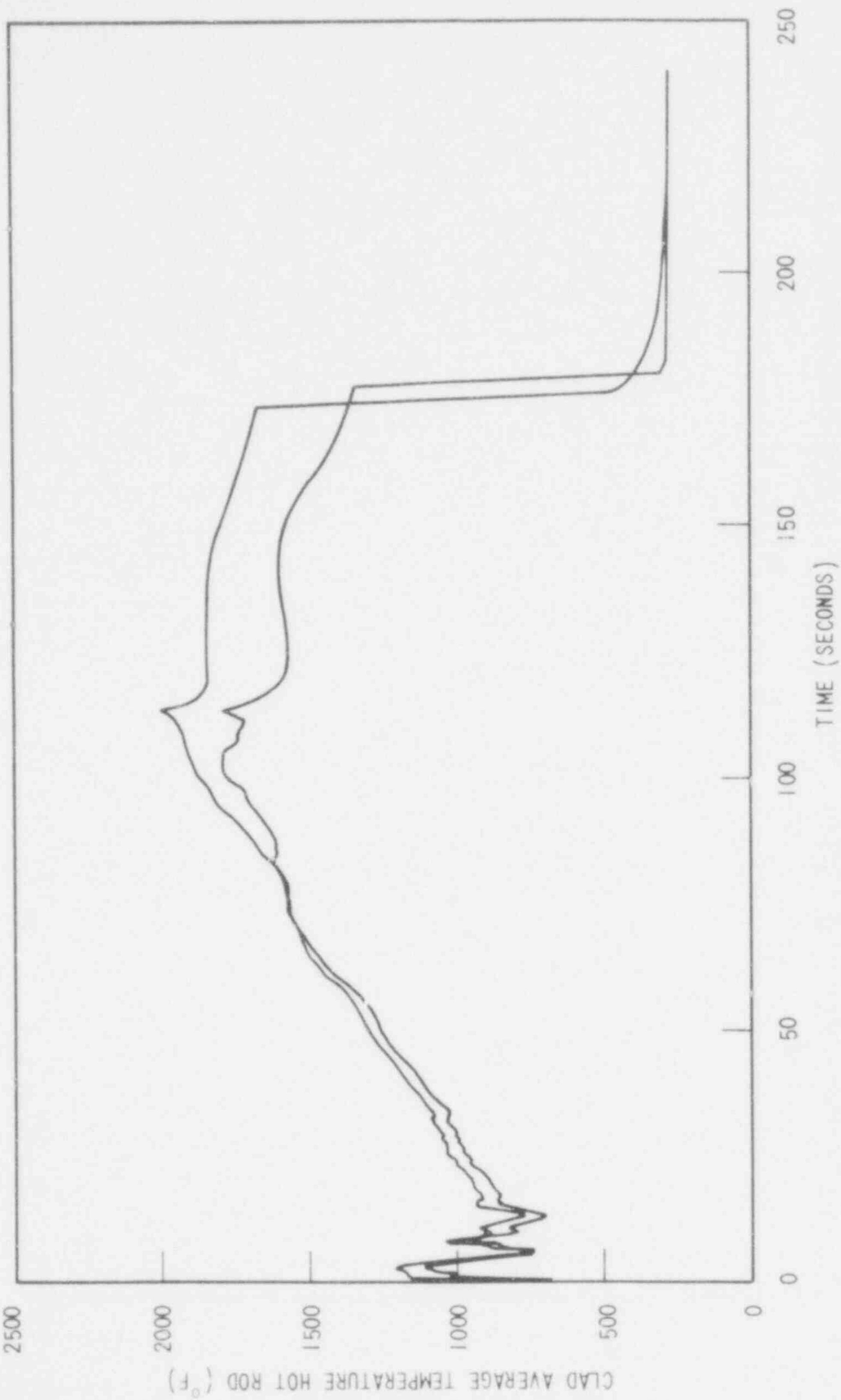


WCAP – 9500

Figure 15.6.5-46. BLUE

Peak Clad Temperature – Nodes 13 and 14, C_D = 0.6 DECLG, Perfect Mixing

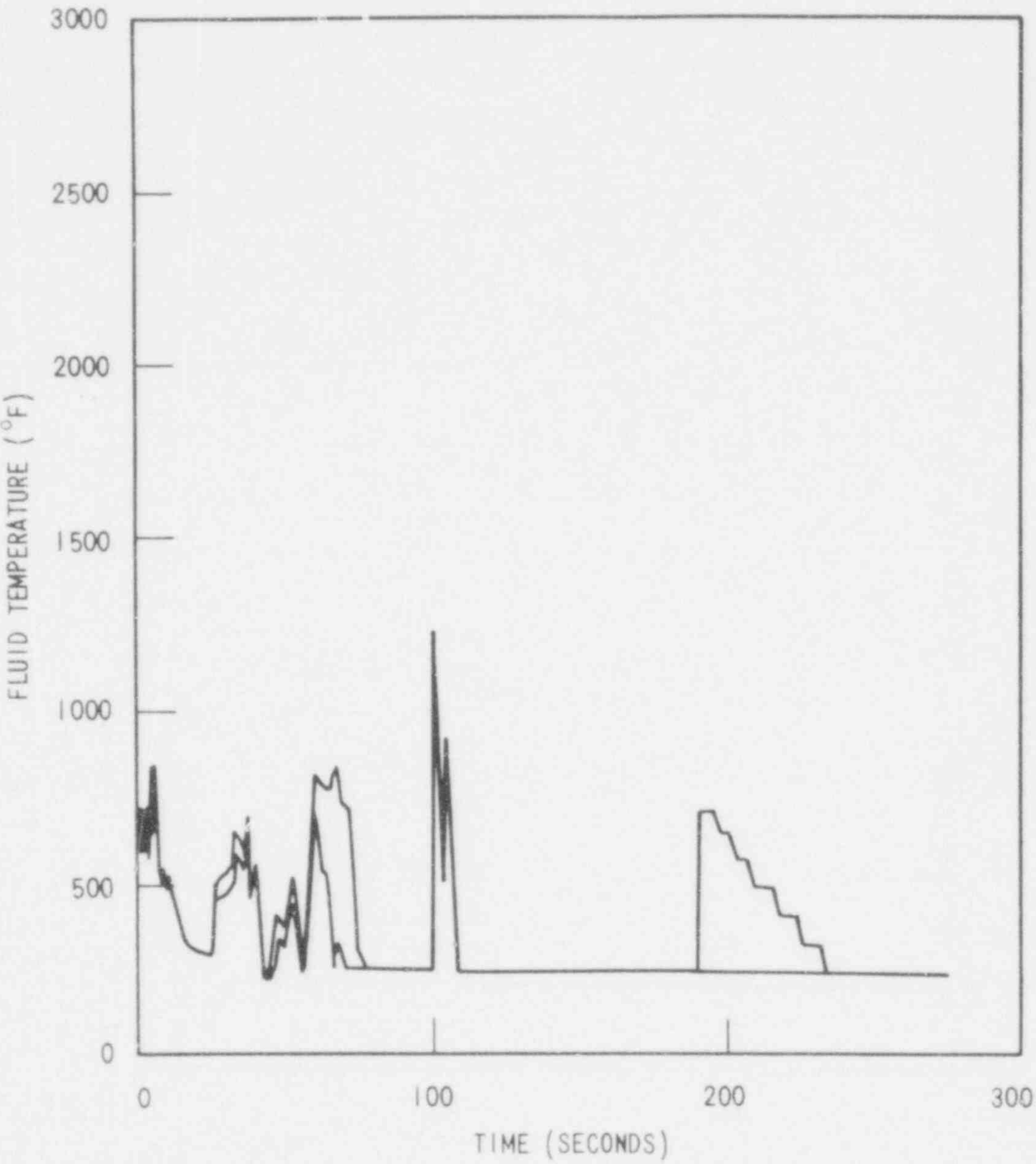
615 183



POOR ORIGINAL

615 184

WCAP - 9500
Figure 15.6.5-47. BLUE Peak Clad Temperature – Nodes 8 and 14, C_D = 0.4 DECLG, Perfect Mixing

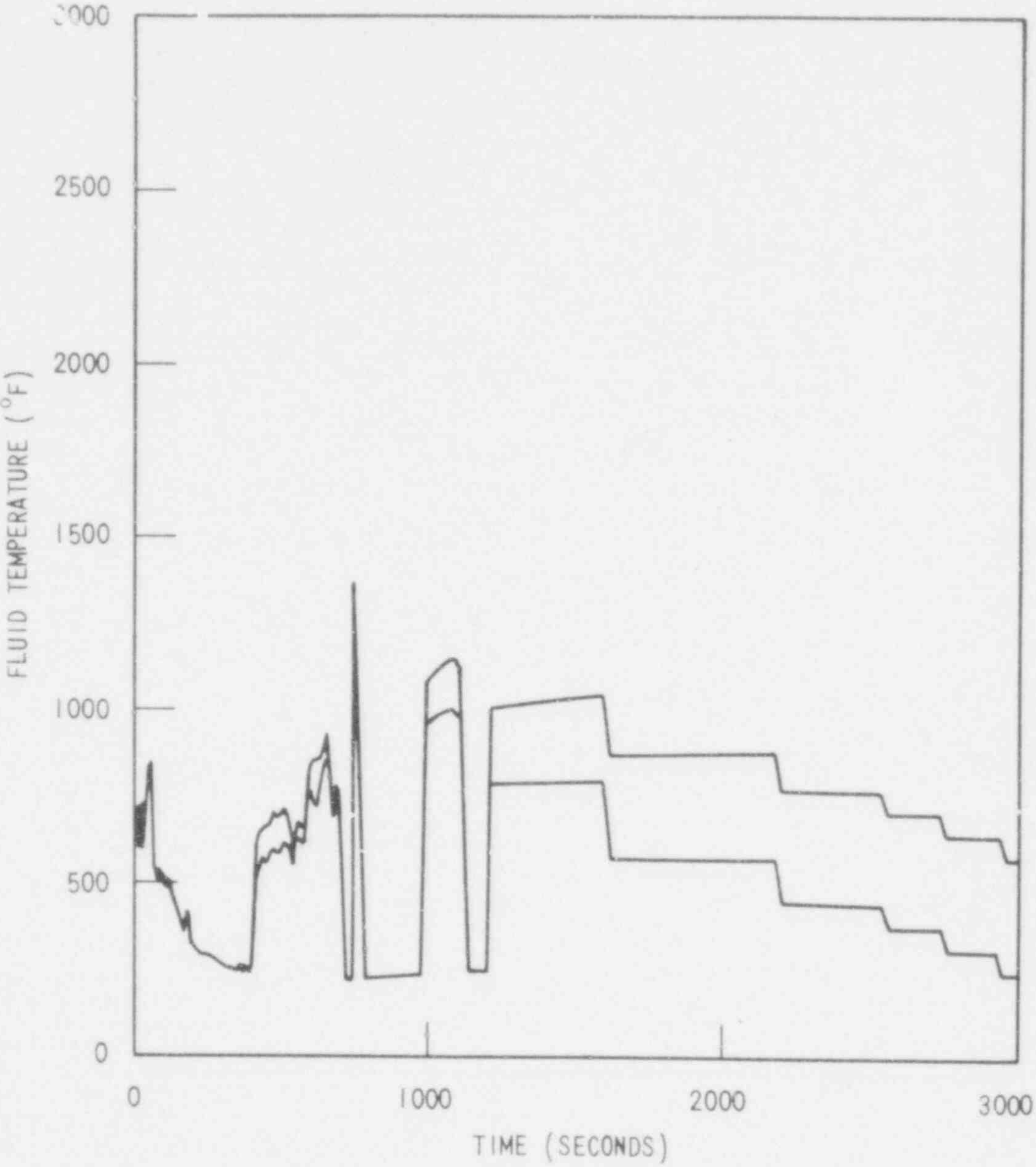


WCAP – 9500

Figure 15.6.5-48. BLUE

Fluid Temperature – Nodes 13 and 14, C_D = 1.0 DECLG, Perfect Mixing

615 185



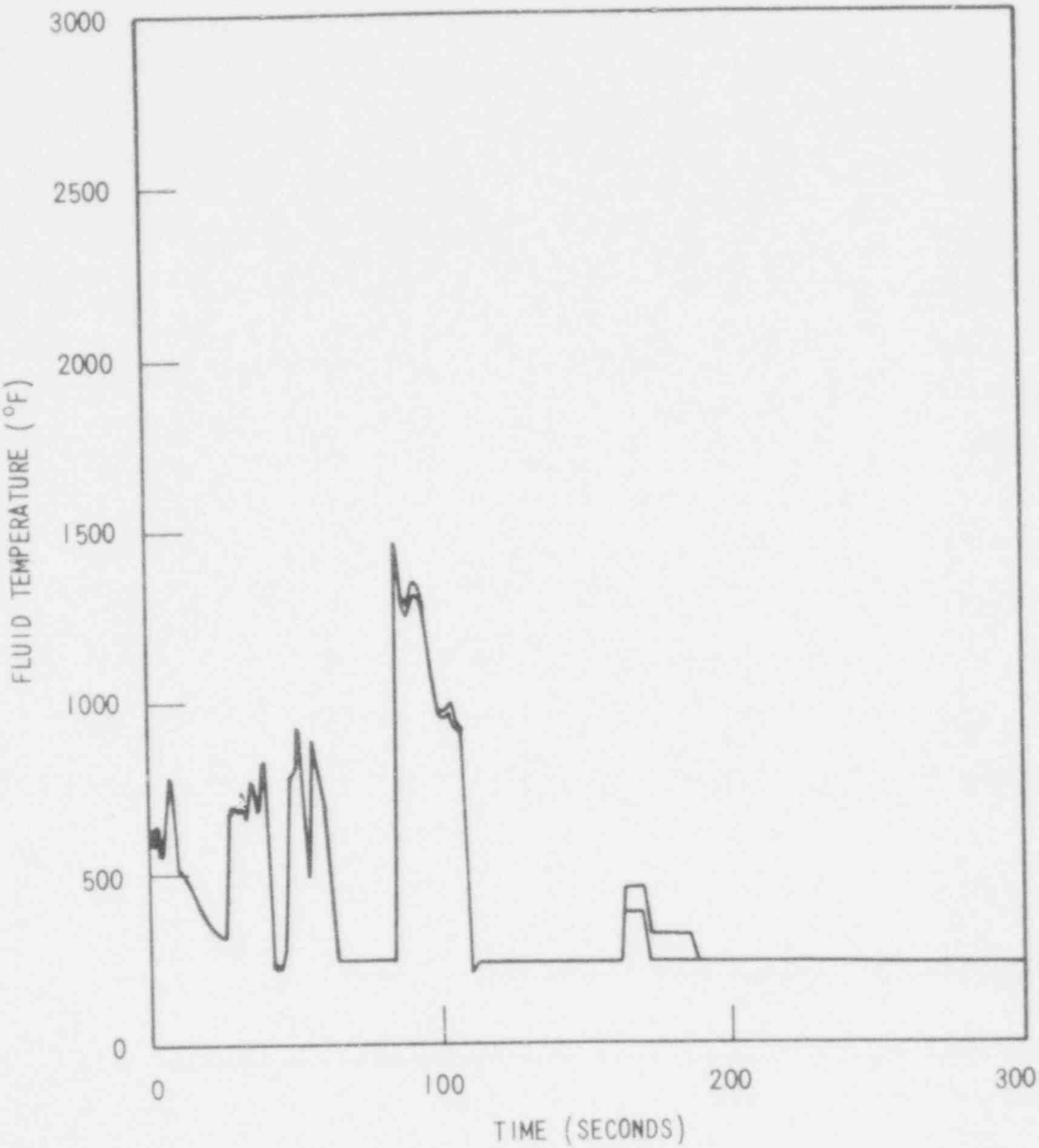
POOR ORIGINAL

615 186

WCAP – 9500

Figure 15.6.5-49. BLUE

Fluid Temperature – Nodes 9 and 13, C_D = 1.0 DECLG, Imperfect Mixing



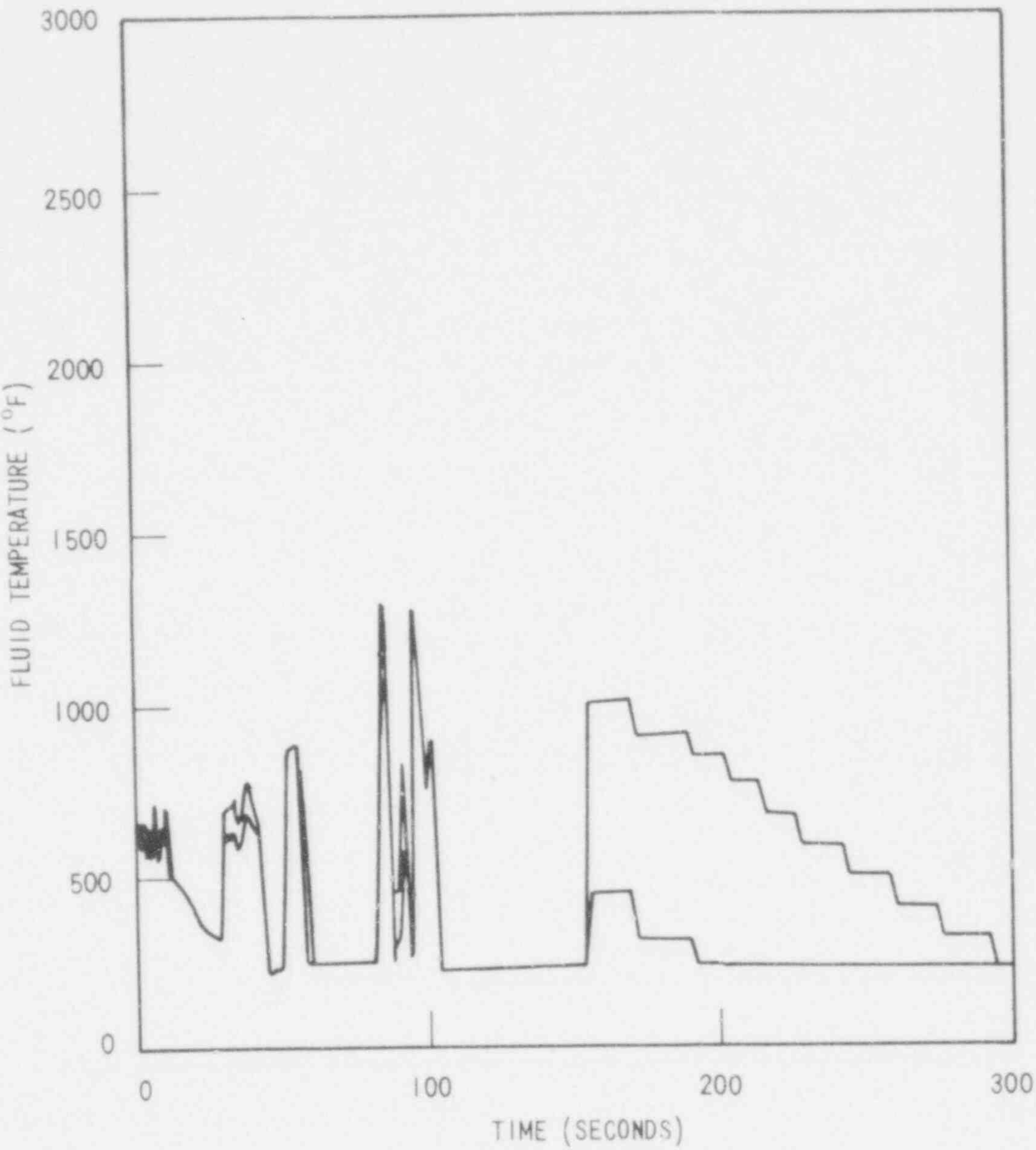
POOR ORIGINAL

615 187

WCAP – 9500

Figure 15.6.5-50. BLUE

Fluid Temperature – Nodes 13 and 14, C_D = 0.8 DECLG, Perfect Mixing



POOR ORIGINAL

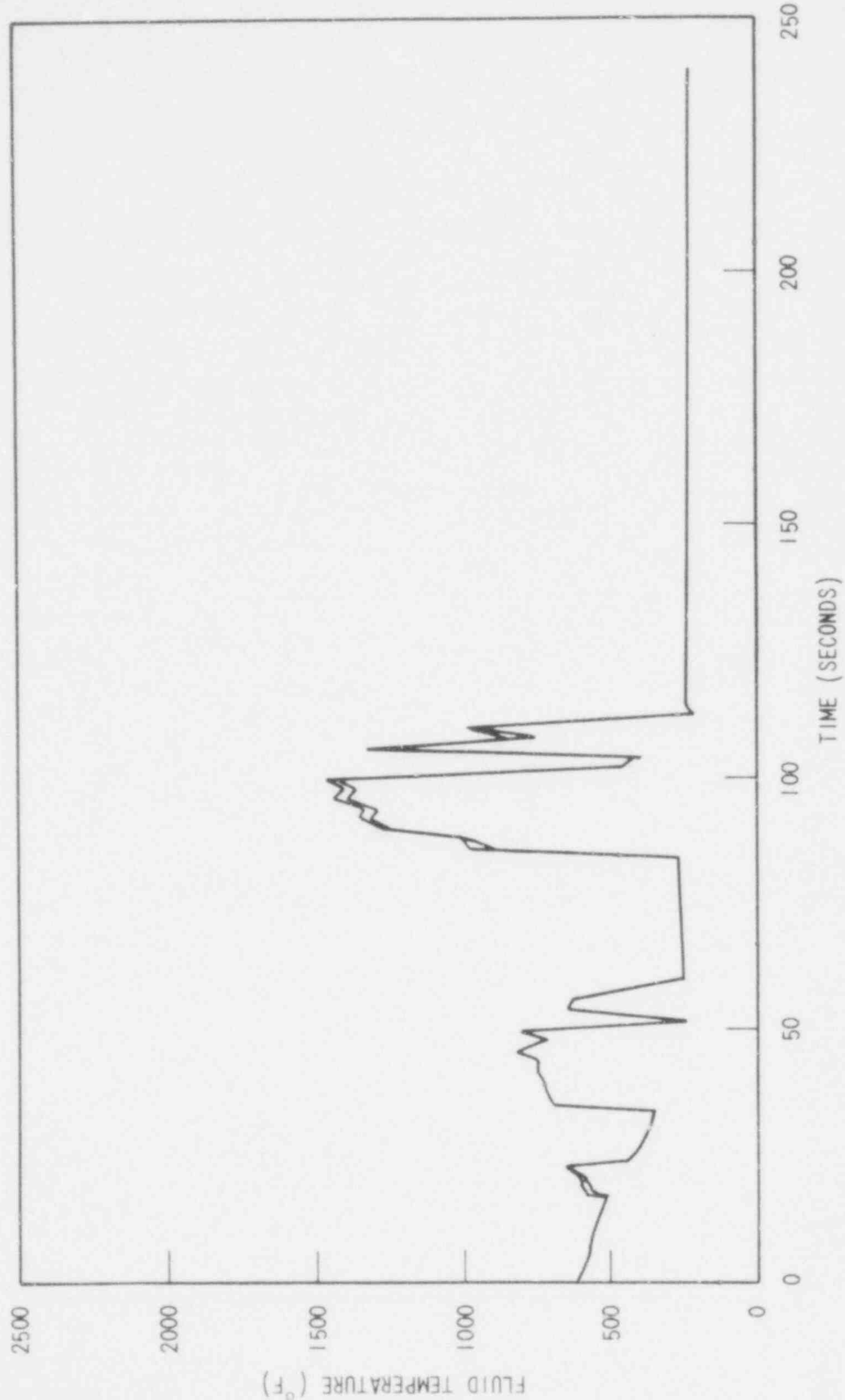
WCAP - 9500

Figure 15.6.5-51. BLUE

Fluid Temperature – Nodes 13 and 14, C_D = 0.6 DECLG, Perfect Mixing

615 188

14,395-404



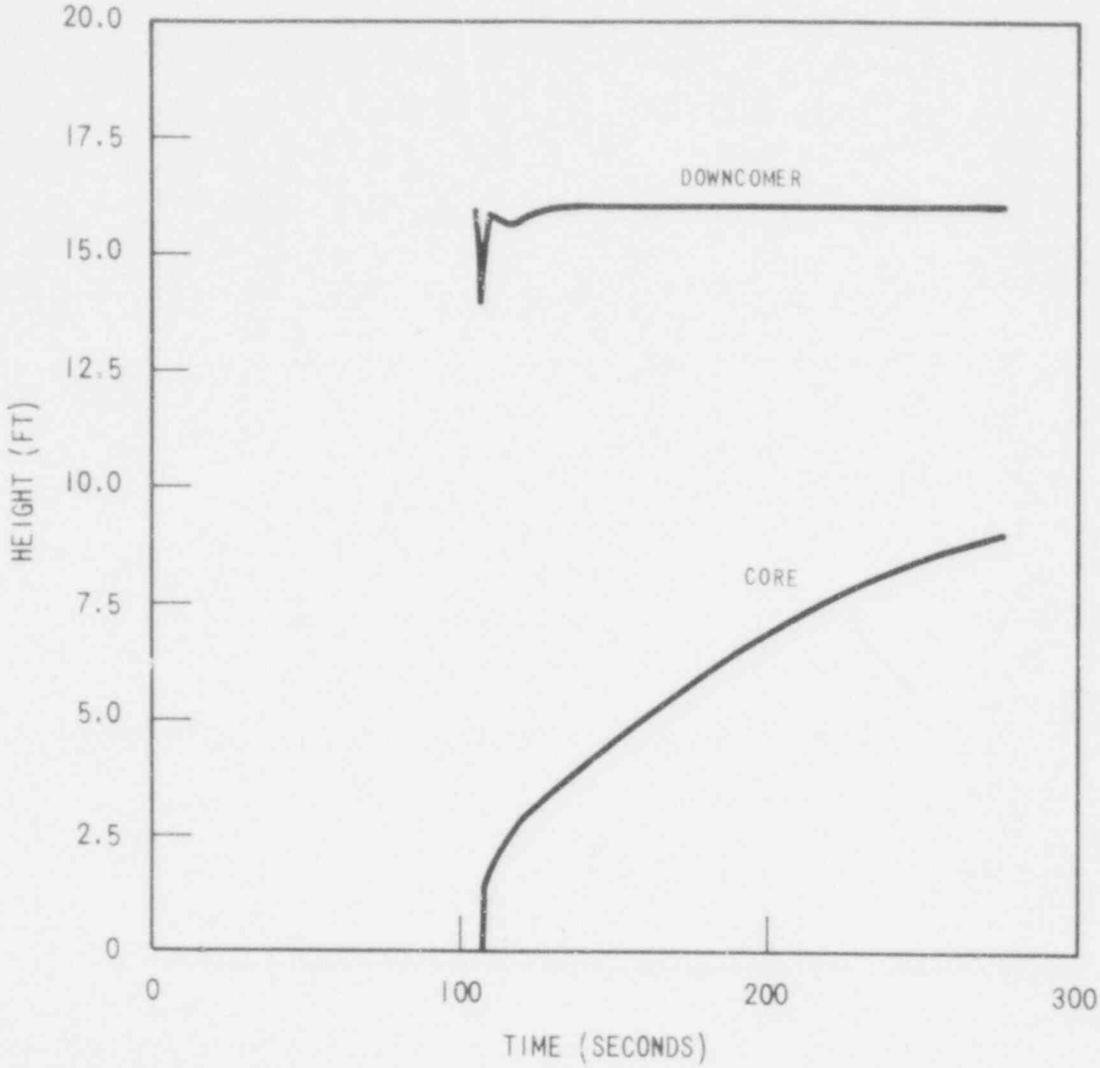
POOR ORIGINAL

615 189

WCAP – 9500

Figure 15.6.5-52. BLUE

Fluid Temperature – Nodes 8 and 14, C_D = 0.4 DECLG, Perfect Mixing



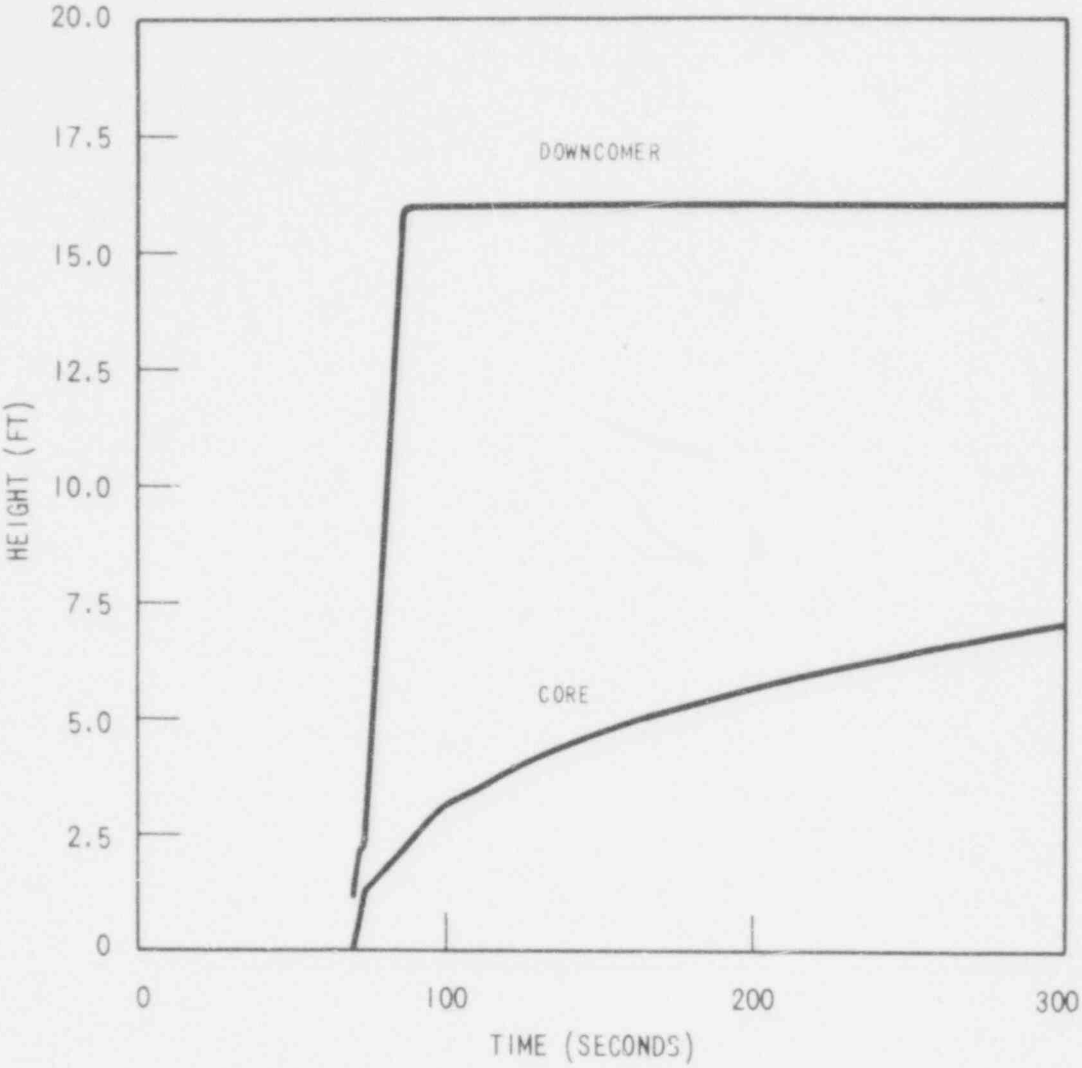
POOR ORIGINAL

615 190

WCAP - 9500

Figure 15.6.5-53. BLUE

Reflood Transient,
C_D = 1.0 DECLG, Perfect Mixing



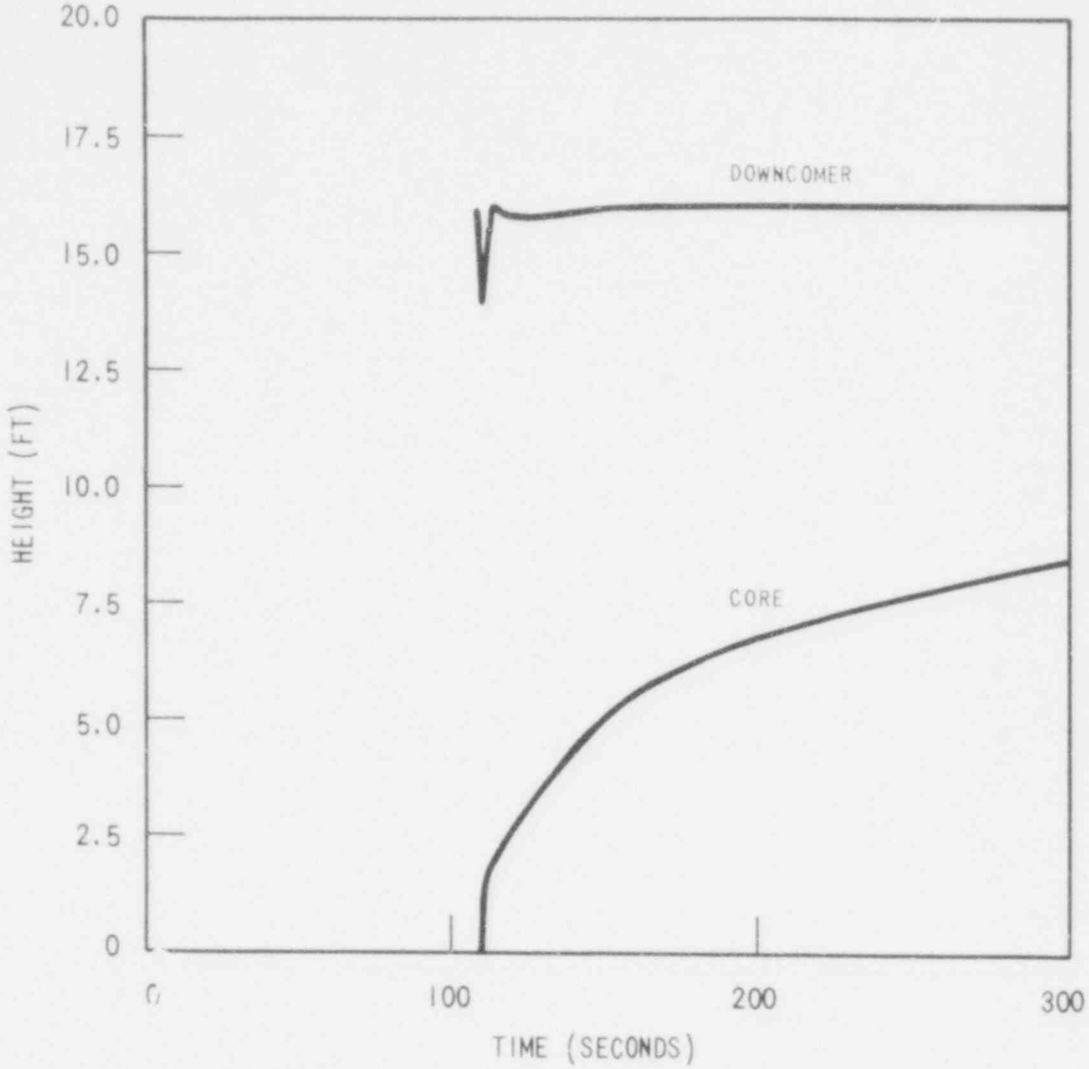
POOR ORIGINAL

615 191

WCAP – 9500

Figure 15.6.5-54. BLUE

Reflood Transient,
C_D = 1.0 DECLG, Imperfect Mixing



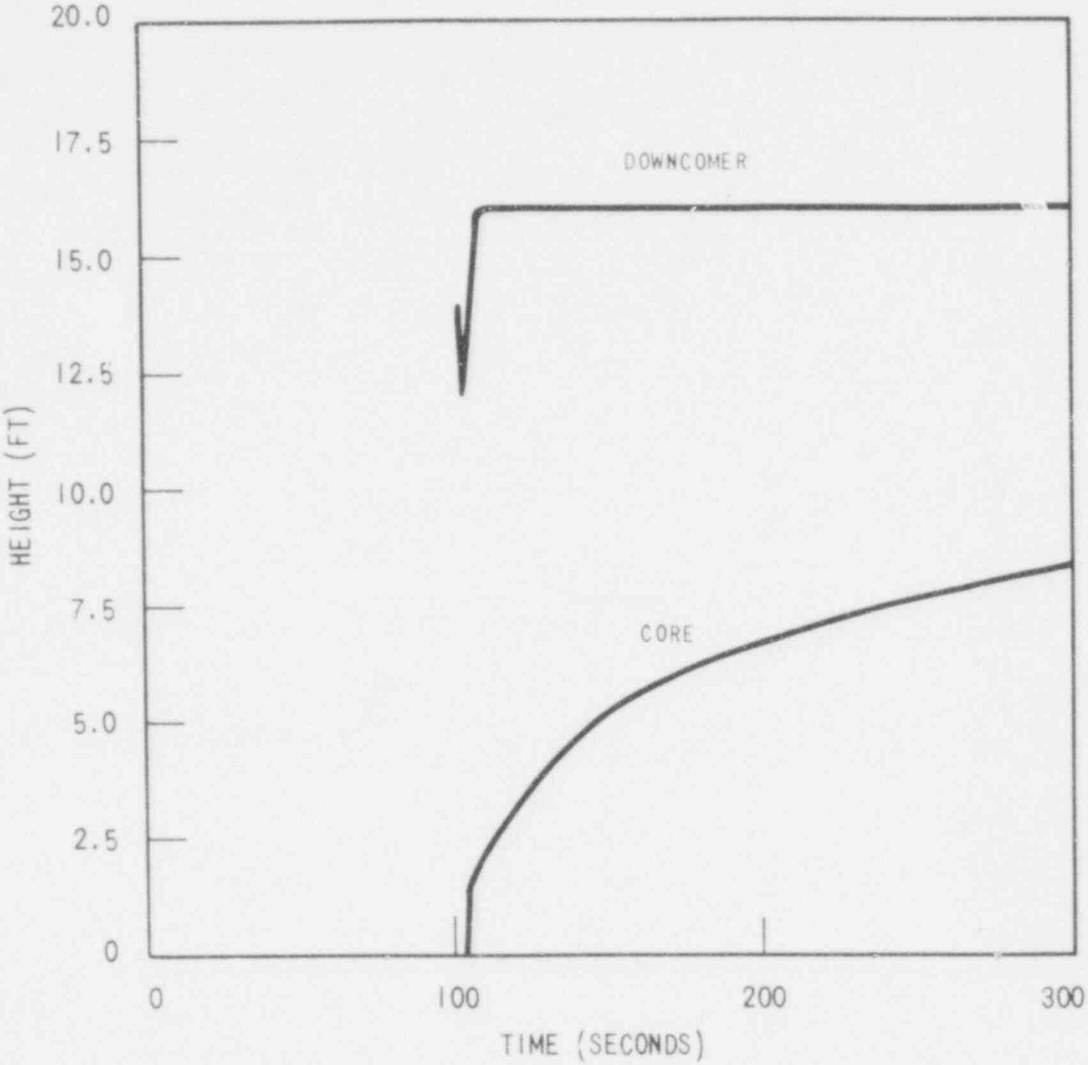
POOR ORIGINAL

615 192

WCAP – 9500

Figure 15.6.5-55. BLUE

Reflood Transient,
C_D = 0.8 DECLG, Perfect Mixing



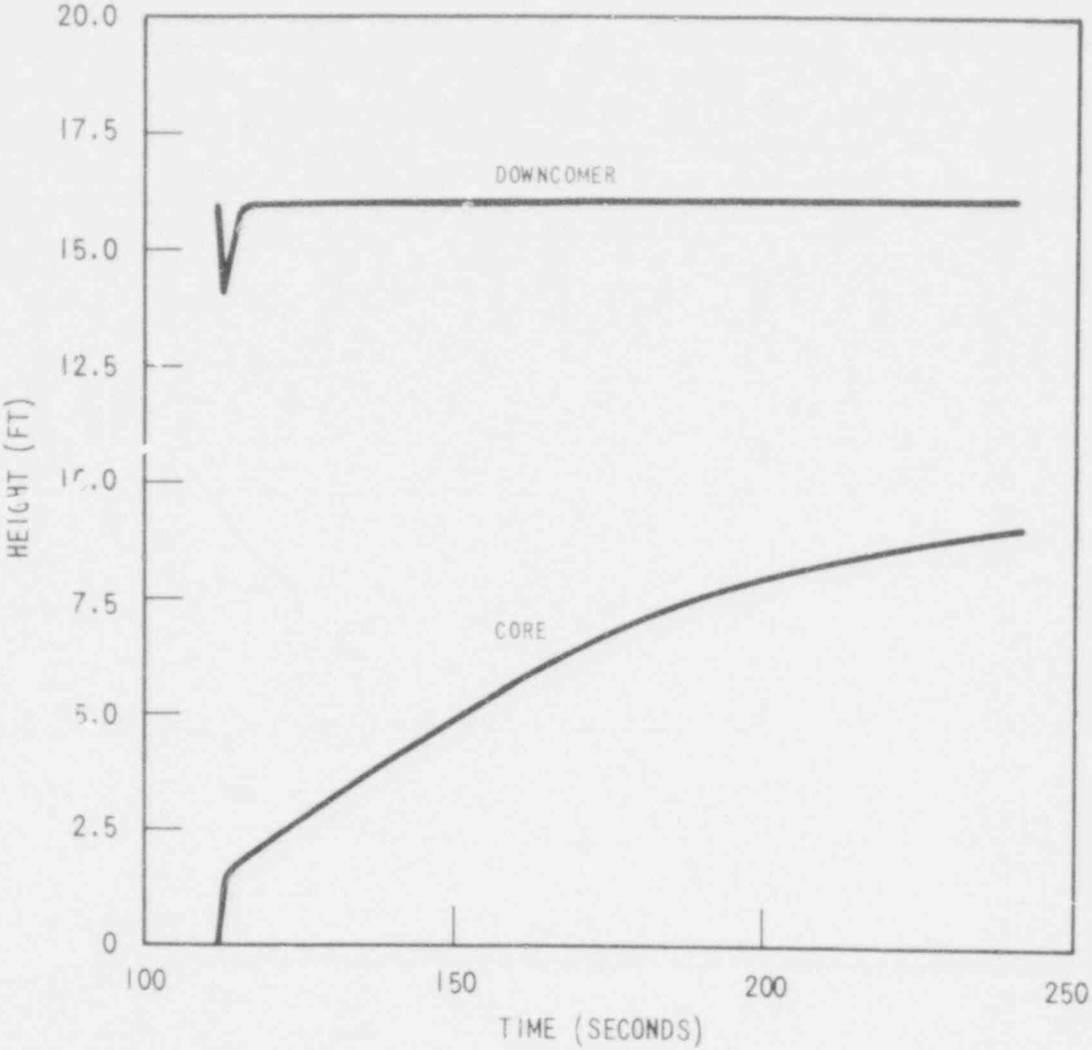
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-56. BLUE

Reflood Transient,
C_D = 0.6 DECLG, Perfect Mixing

615 193



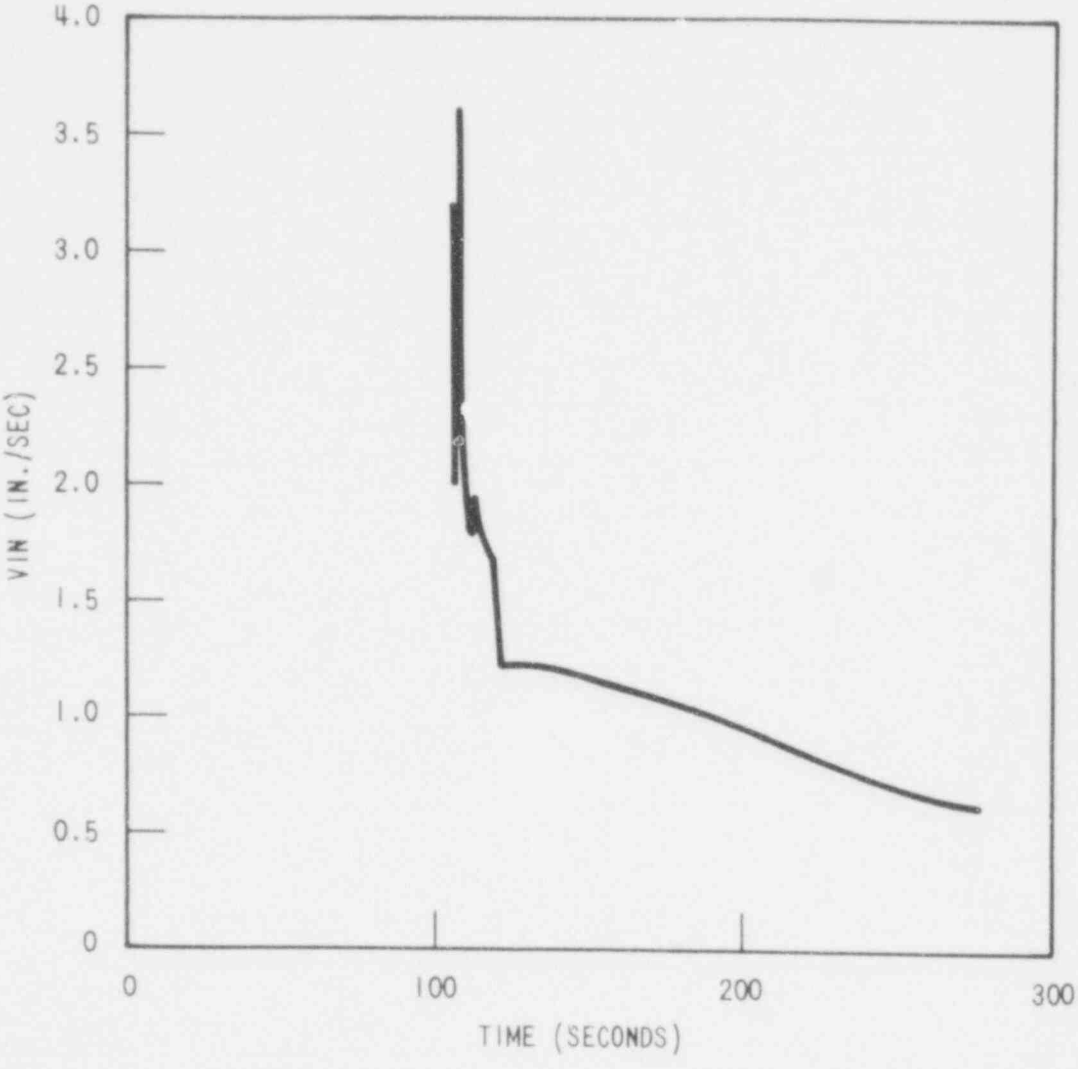
POOR ORIGINAL

615 194

WCAP - 9500

Figure 15.6.5-57. BLUE

Reflood Transient,
C_D = 0.4 DECLG, Perfect Mixing



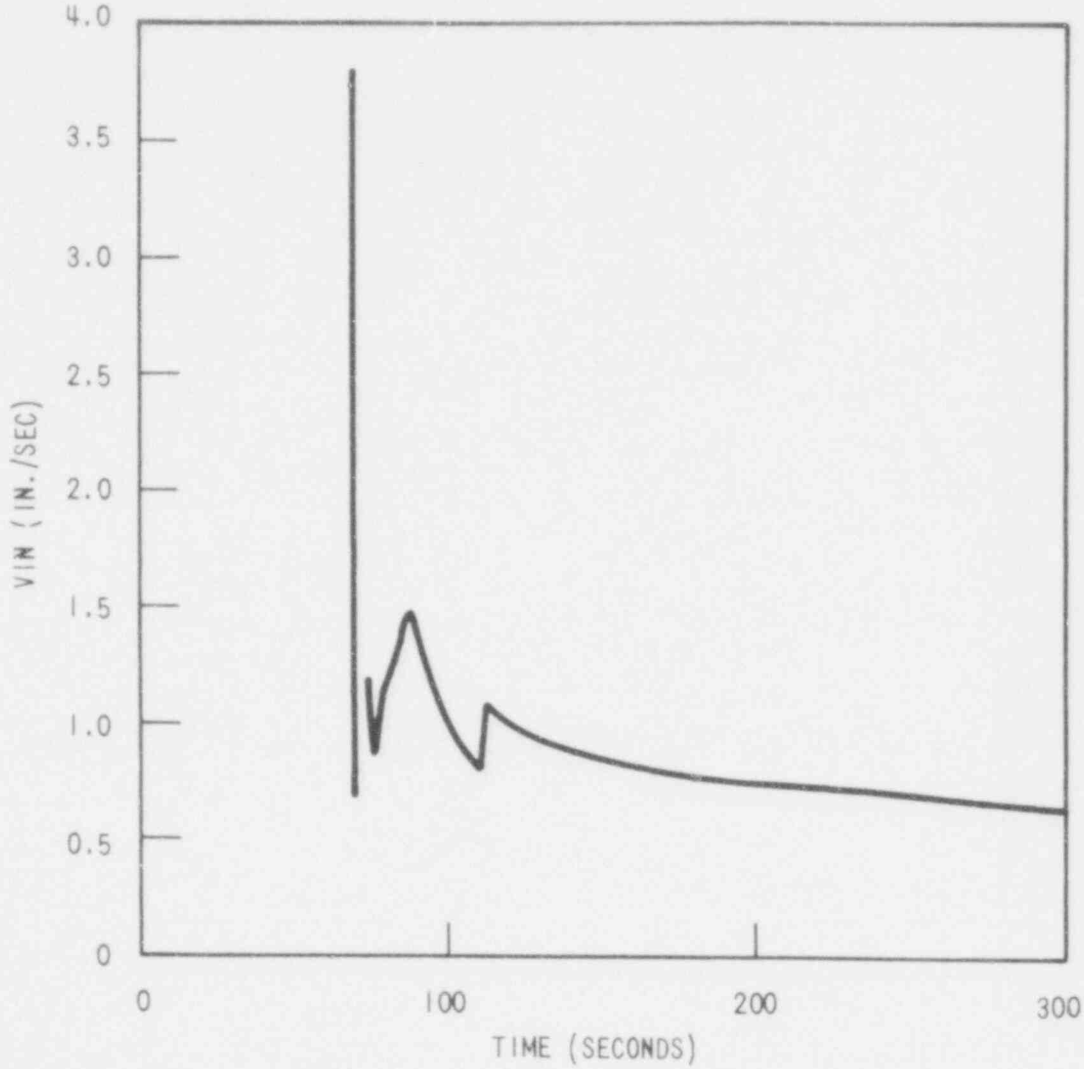
POOR ORIGINAL

615 195

WCAP – 9500

Figure 15.6.5-58. BLUE

Reflooding Rate,
C_D = 1.0 DECLG, Perfect Mixing



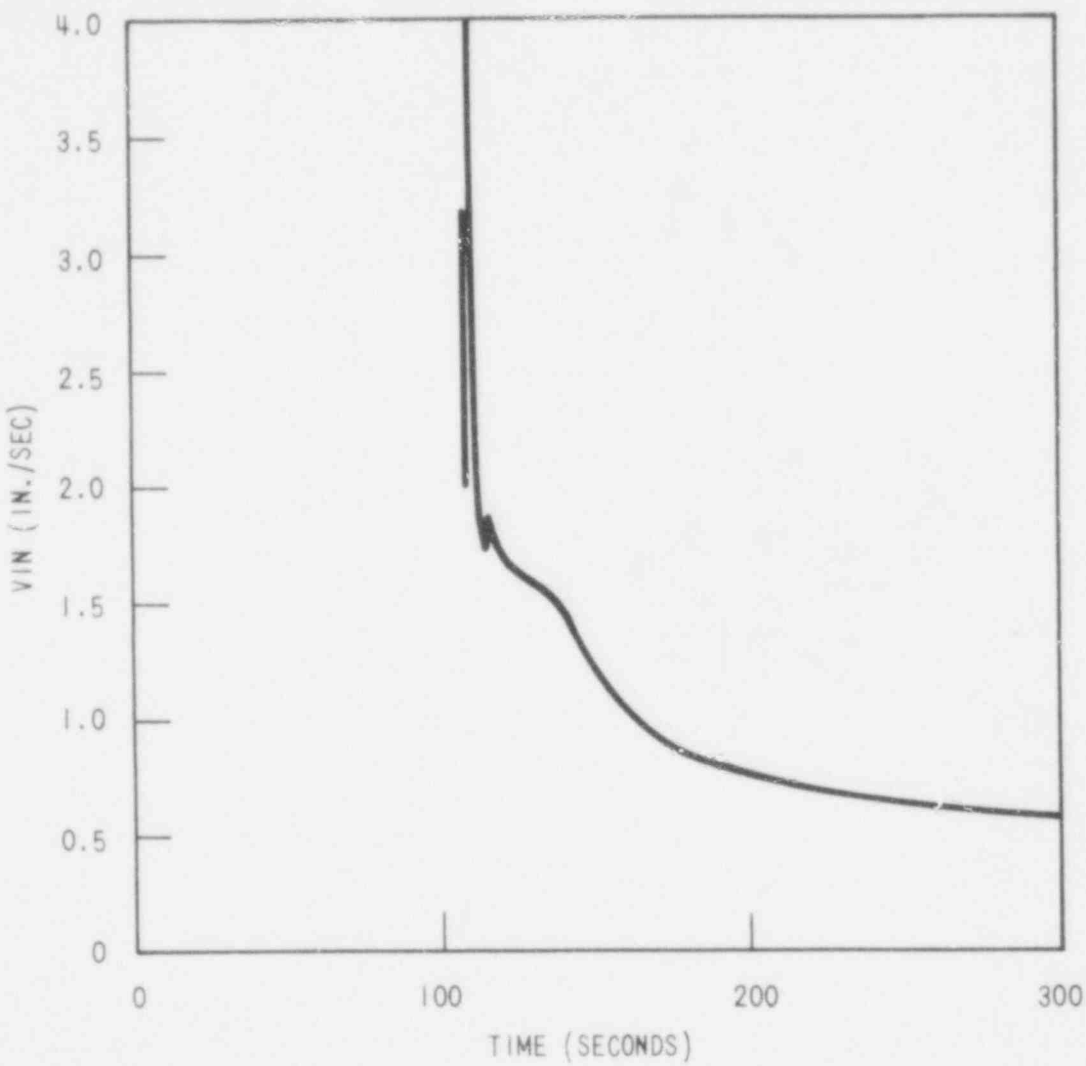
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-59. BLUE

Reflooding Rate,
C_D = 1.0 DECLG, Imperfect Mixing

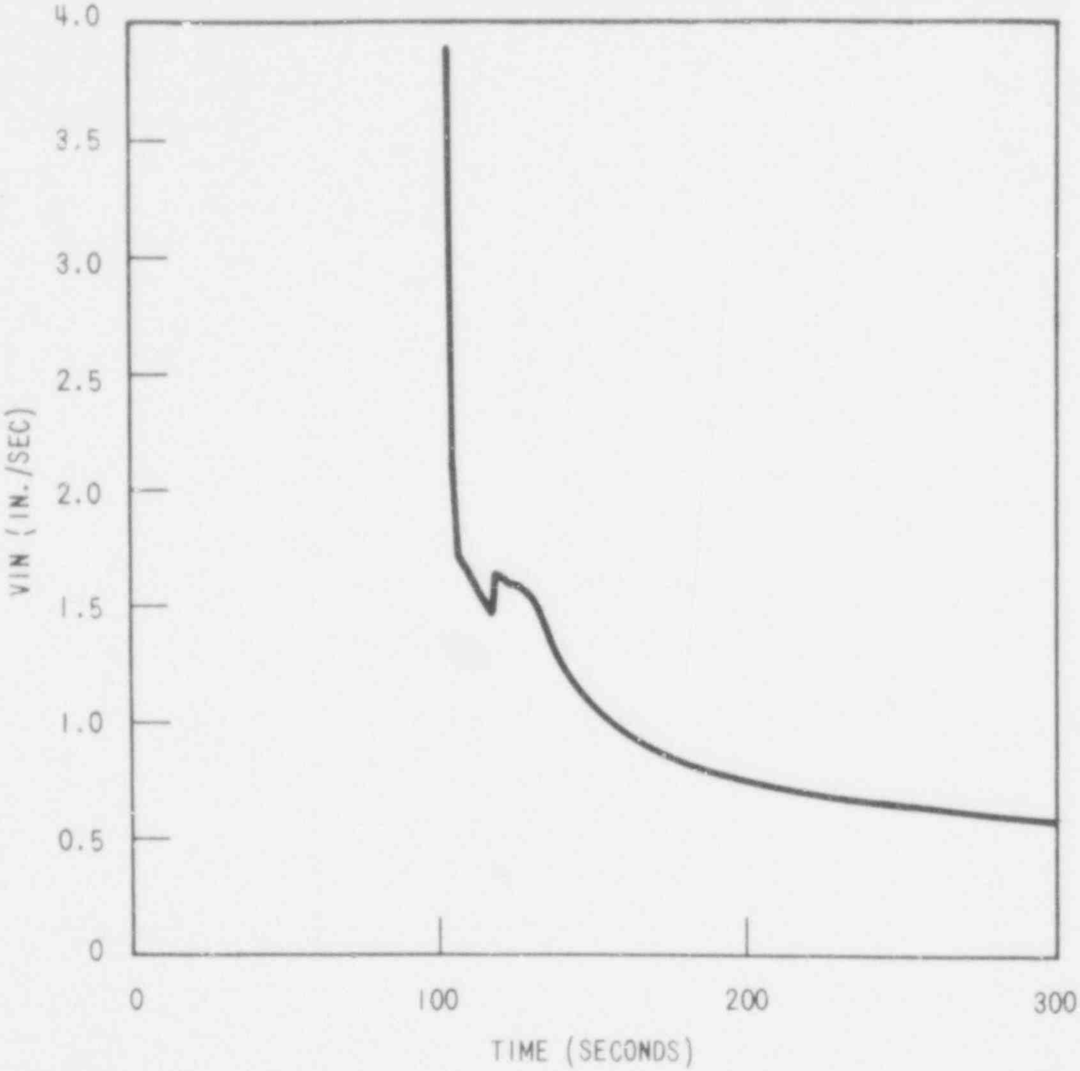
615 196



615 197

POOR ORIGINAL

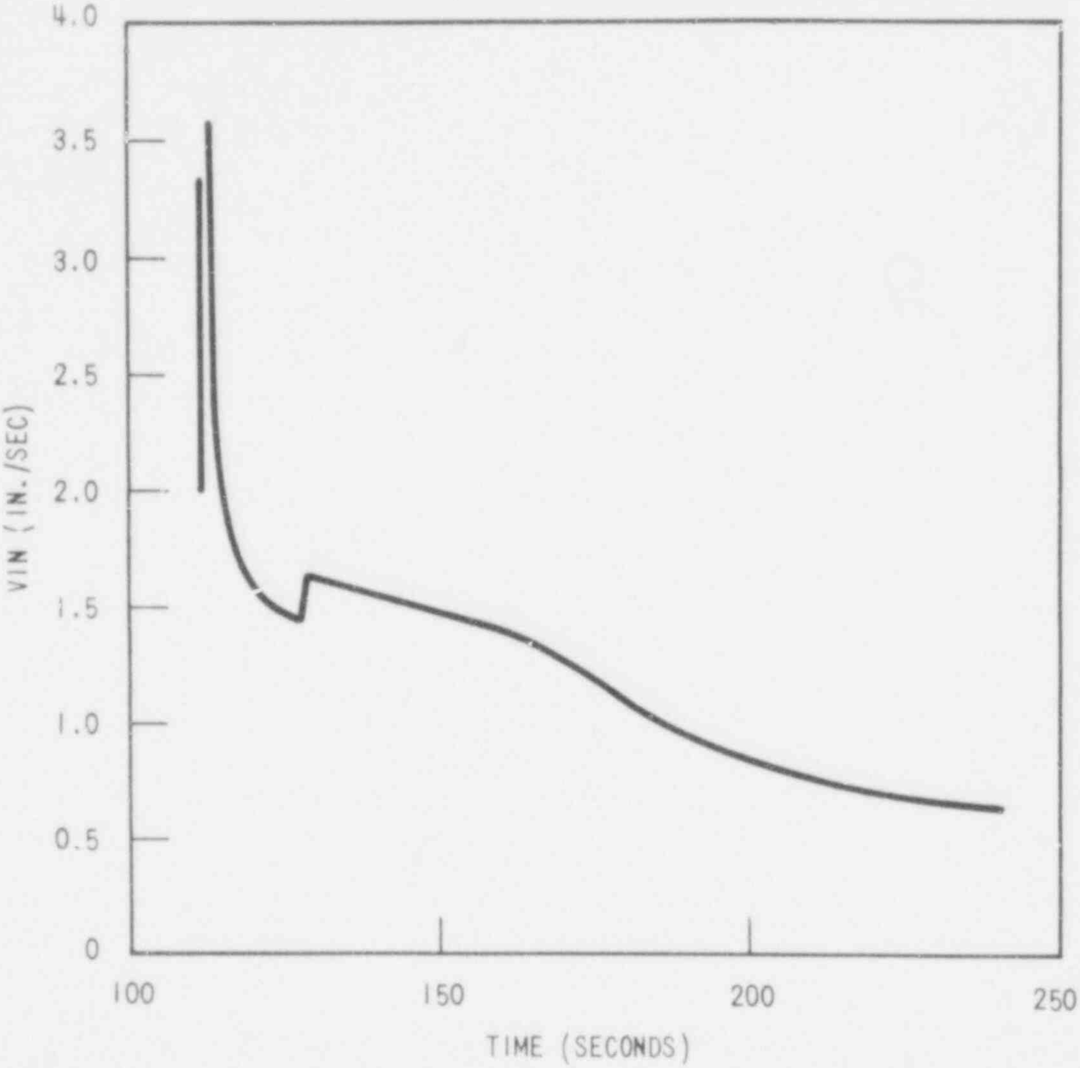
WCAP - 9500
Figure 15.6.5-60. BLUE Reflooding Rate, C_D = 0.8 DECLG, Perfect Mixing



POOR ORIGINAL

615 198

WCAP - 9500
Figure 15.6.5-61. BLUE Reflooding Rate, C_D = 0.6 DECLG, Perfect Mixing



POOR ORIGINAL

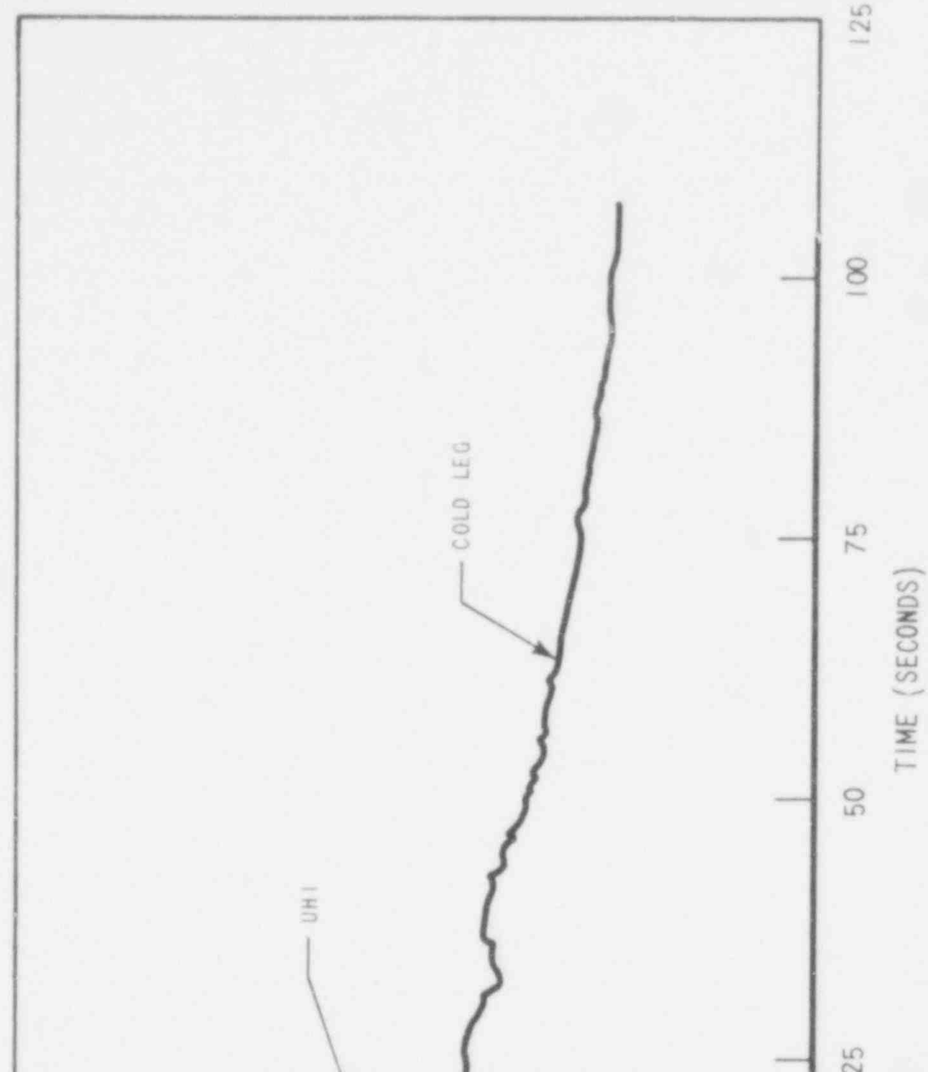
615 199

WCAP – 9500

Figure 15.6.5-62. BLUE

Reflooding Rate,
C_D = 0.4 DECLG, Perfect Mixing

14,395-415



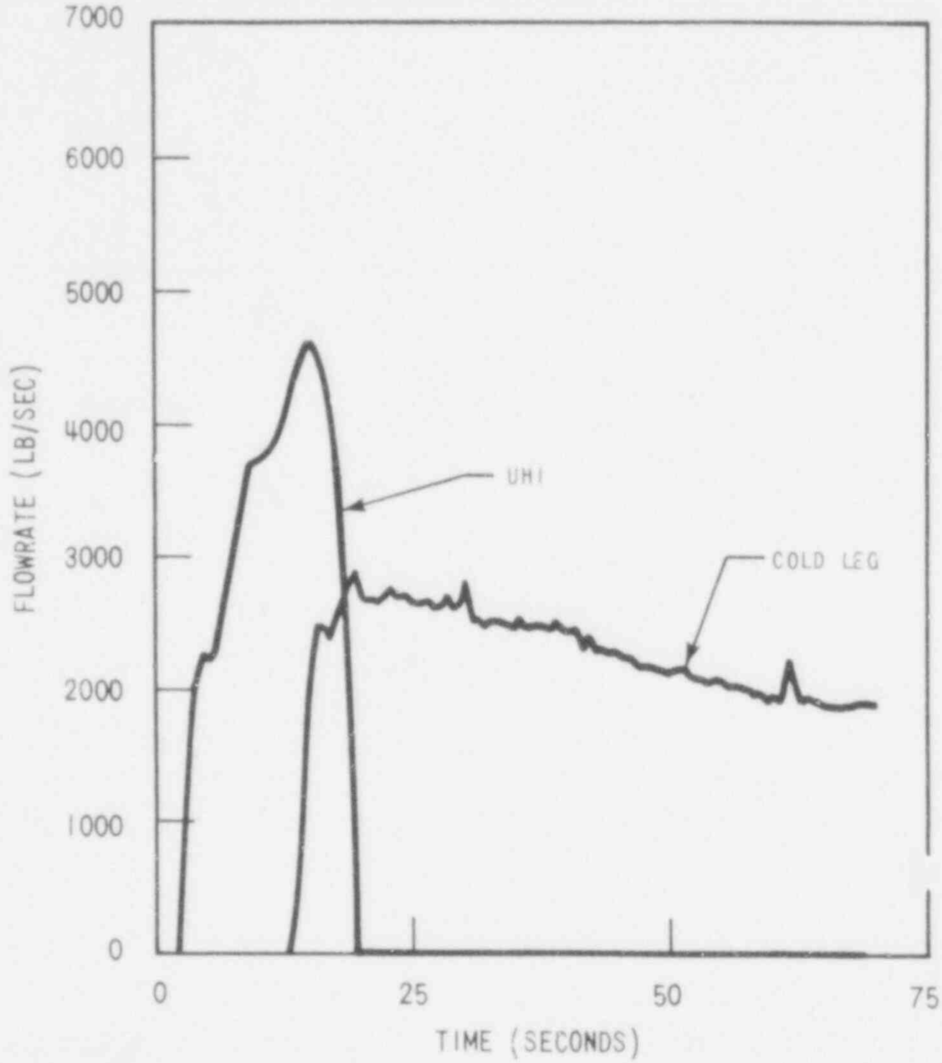
615 200

WCAP – 9500

Figure 15.6.5-63. BLUE

Accumulator Flowrates,
C_D = 1.0 DECLG, Perfect Mixing

POOR ORIGINAL

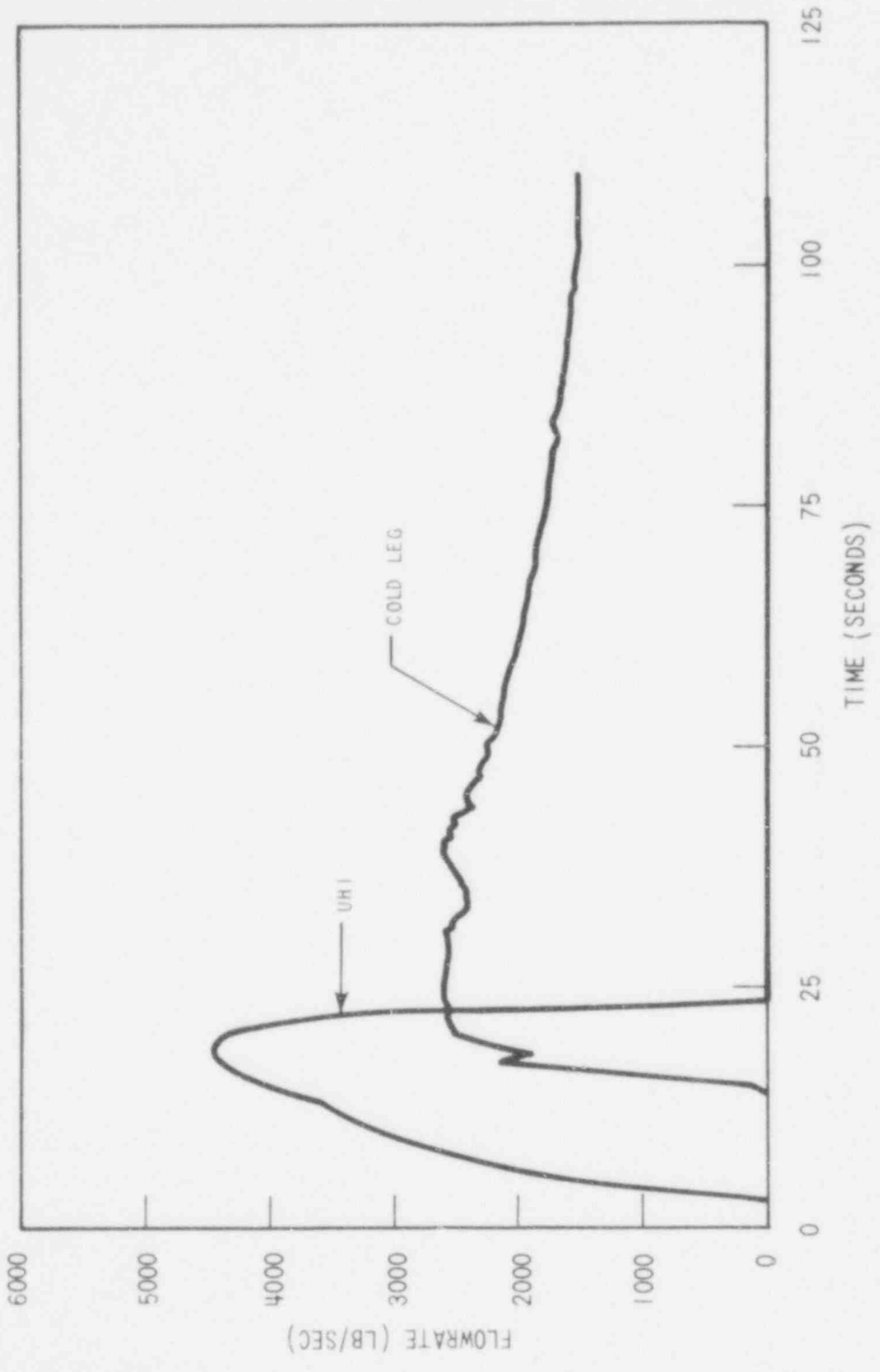


615 201

WCAP – 9500

Figure 15.6.5-64. BLUE

Accumulator Flowrates
C_D = 1.0 DECLG, Imperfect Mixing

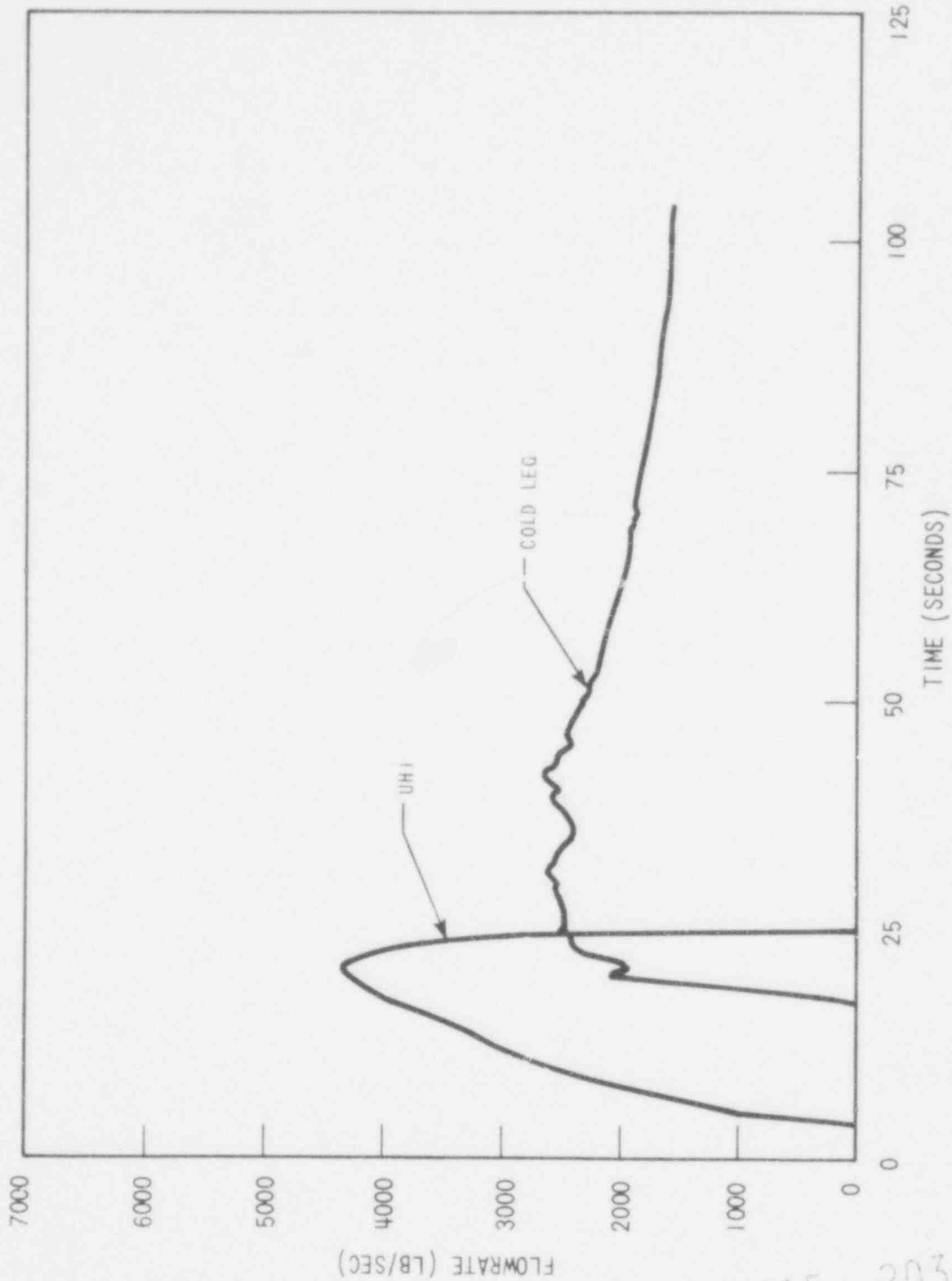


615 202

WCAP – 9500

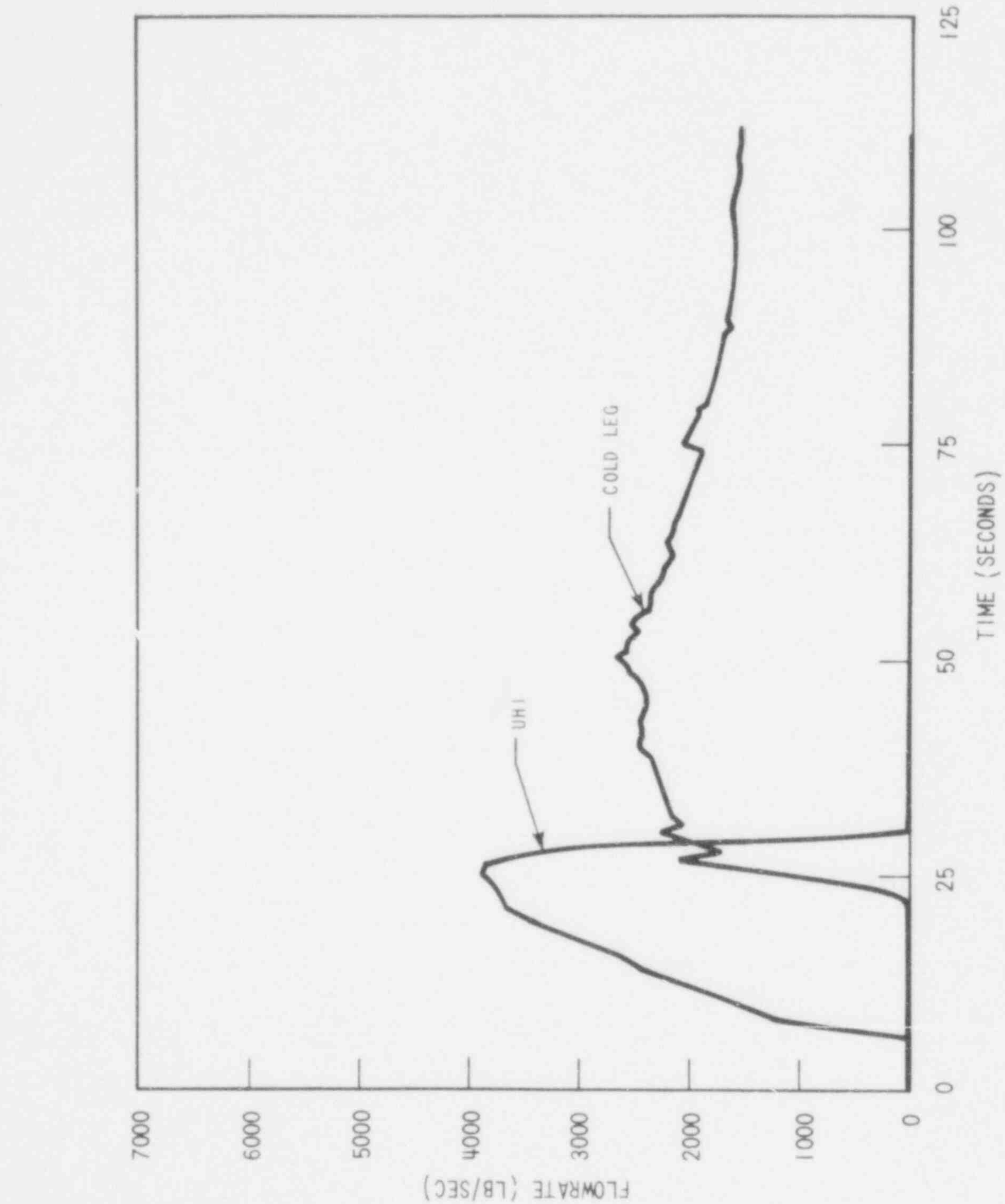
Figure 15.6.5-65. BLUE

Accumul:tor Flowrates, C_D = 0.8 Dr.CLG Perfect Mixing



615 203

WCAP – 9500
Figure 15.6.5-66. BLUE Accumulator Flowrates, C_D = 0.6 DECLG, Perfect Mixing



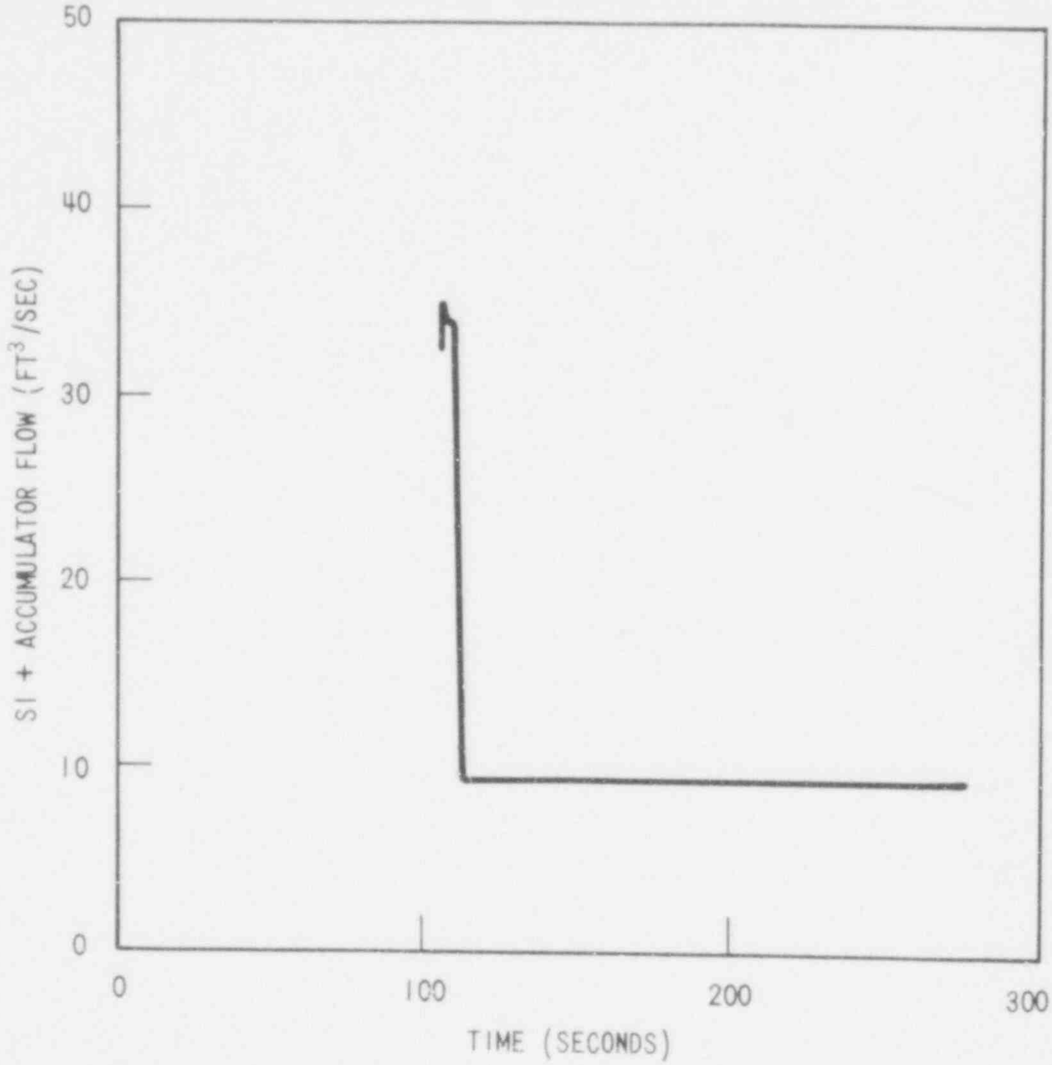
POOR ORIGINAL

615 204

WCAP – 9500

Figure 15.6.5-67. BLUE

Accumulator Flowrates,
C_D = 0.4 DECLG, Perfect Mixing



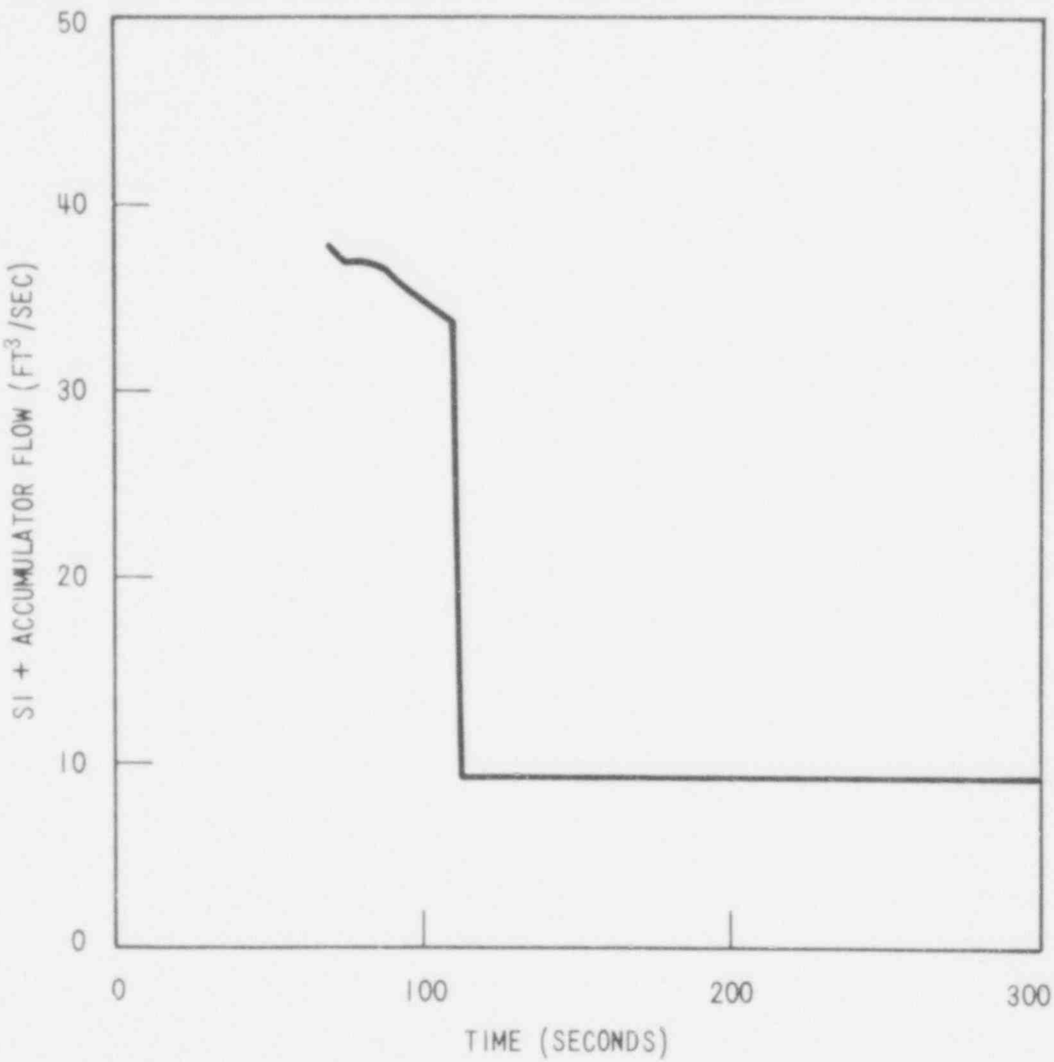
POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-68. BLUE

SI + Accumulator Flow,
C_D = 1.0 DECLG, Perfect Mixing

615 205



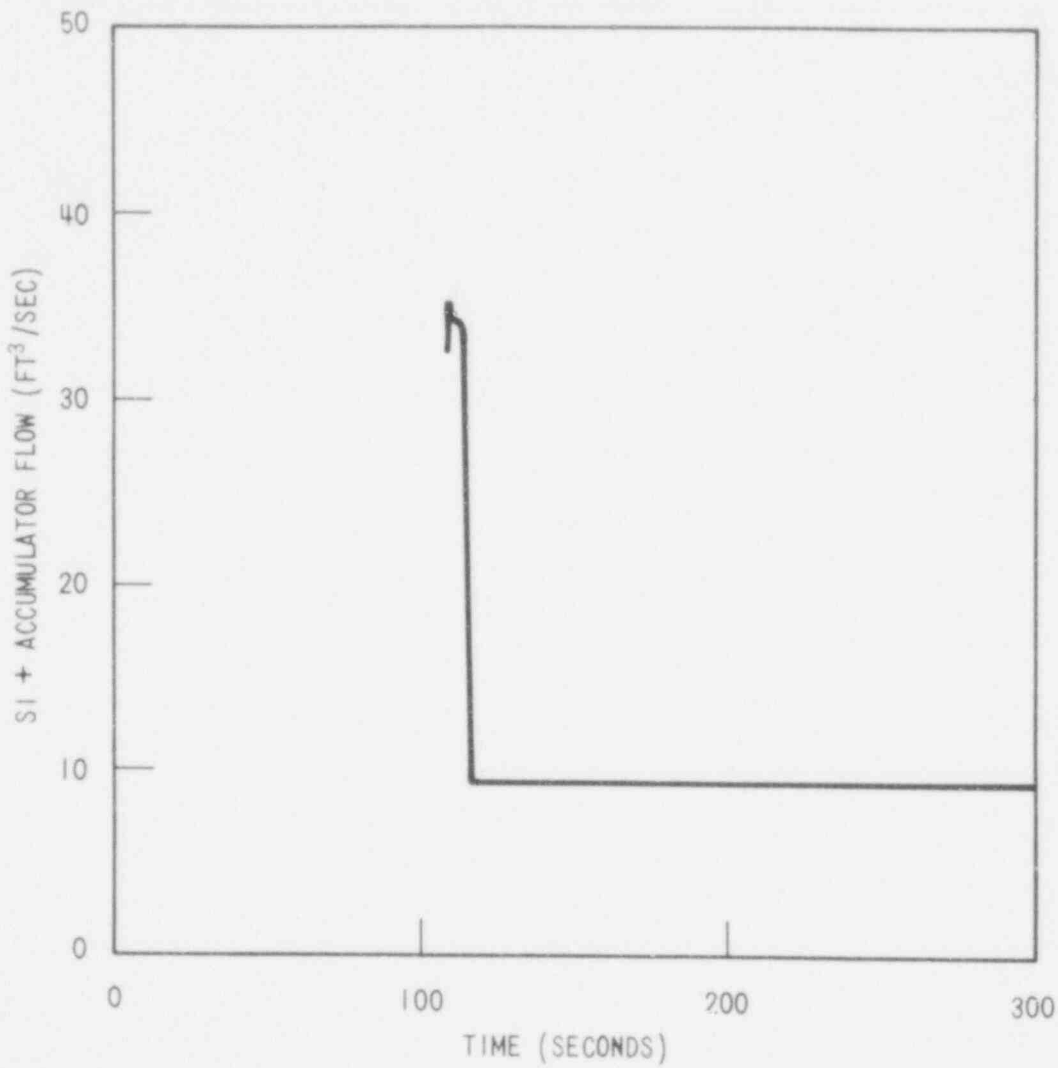
615 206

POOR ORIGINAL

WCAP – 9500

Figure 15.6.5-69. BLUE

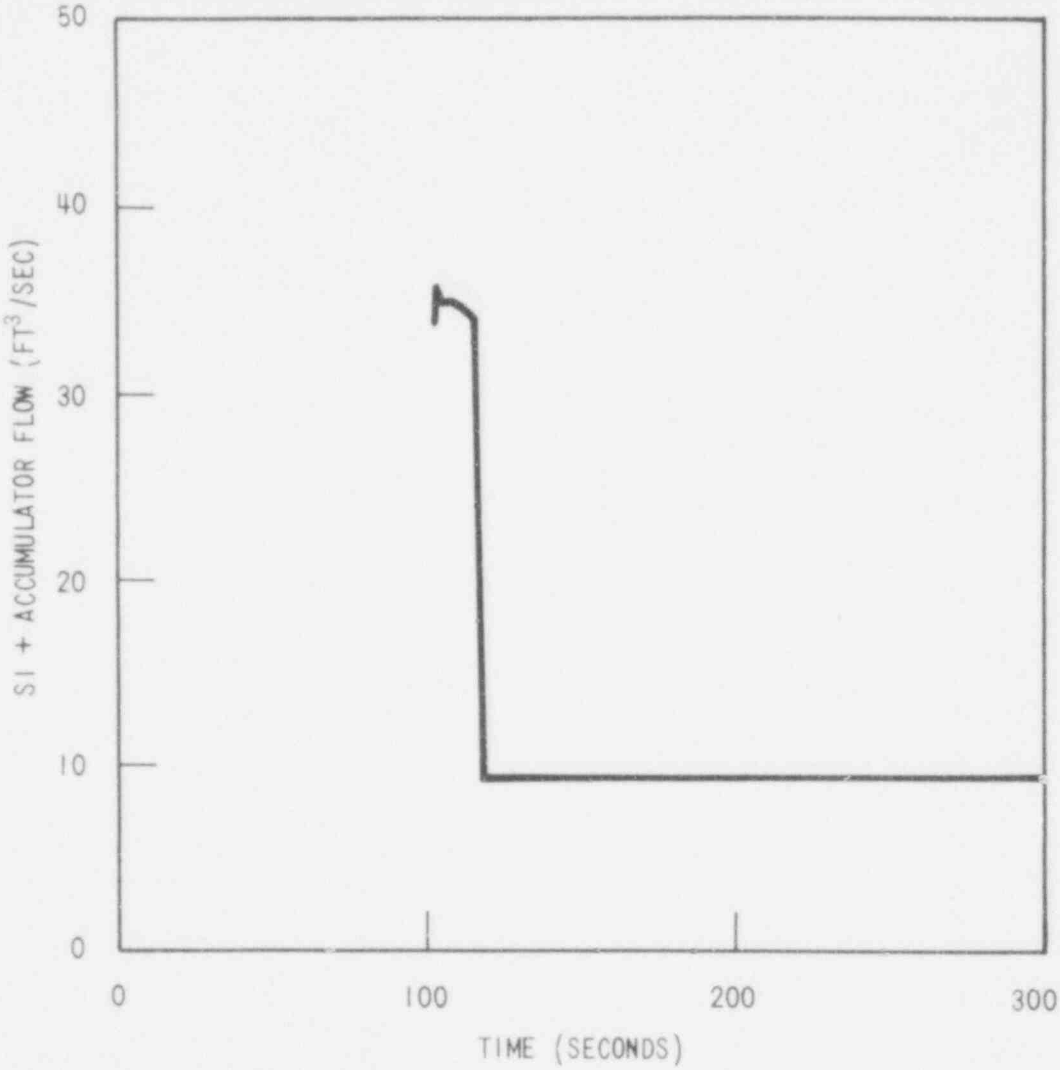
SI + Accumulator Flow,
C_D = 1.0 DECLG, Imperfect Mixing



POOR ORIGINAL

615 207

WCAP - 9500
Figure 15.6.5-70. BLUE SI + Accumulator Flow, C_D = 0.8 DECLG, Perfect Mixing



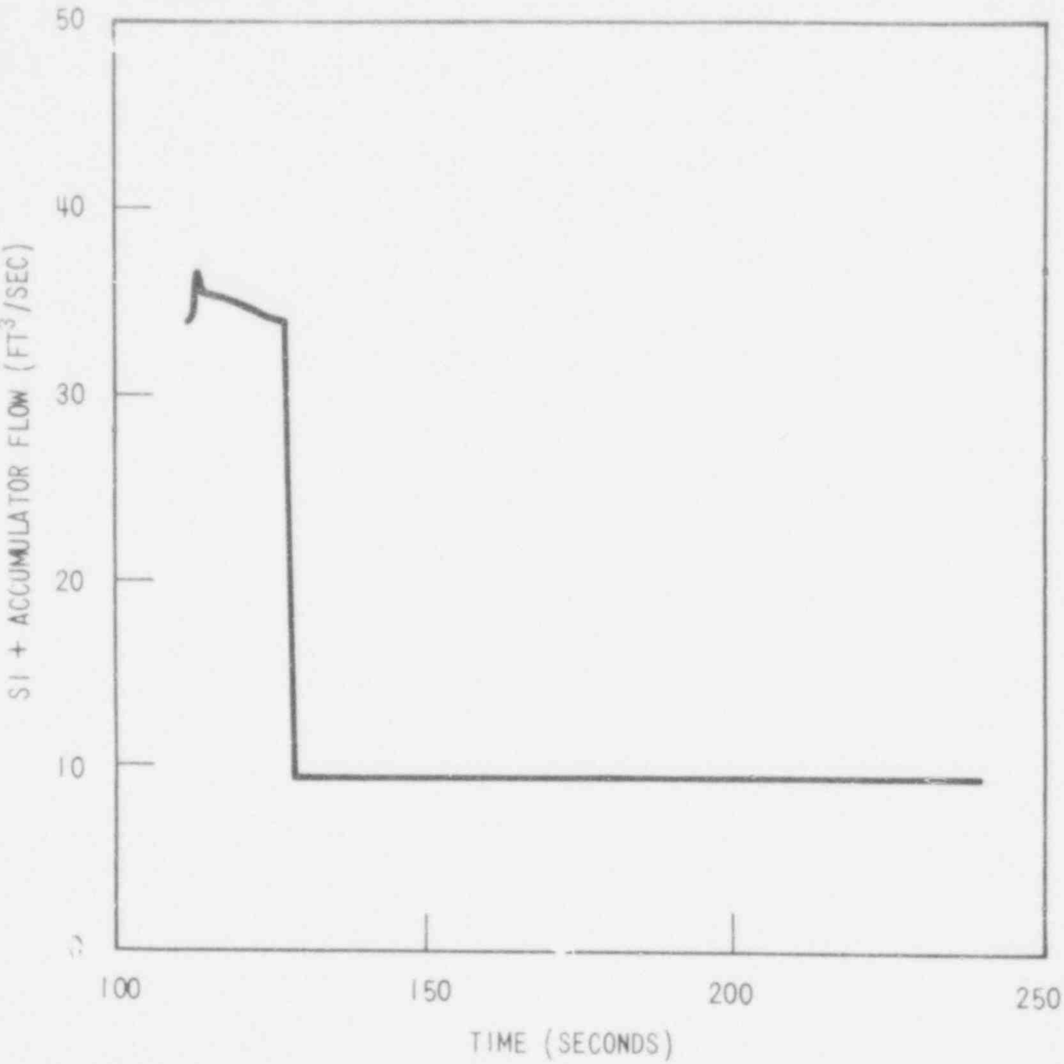
615 208

POOR ORIGINAL

WCAP - 9500

Figure 15.6.5-71. BLUE

SI + Accumulator Flow,
C_D = 0.6 DECLG, Perfect Mixing



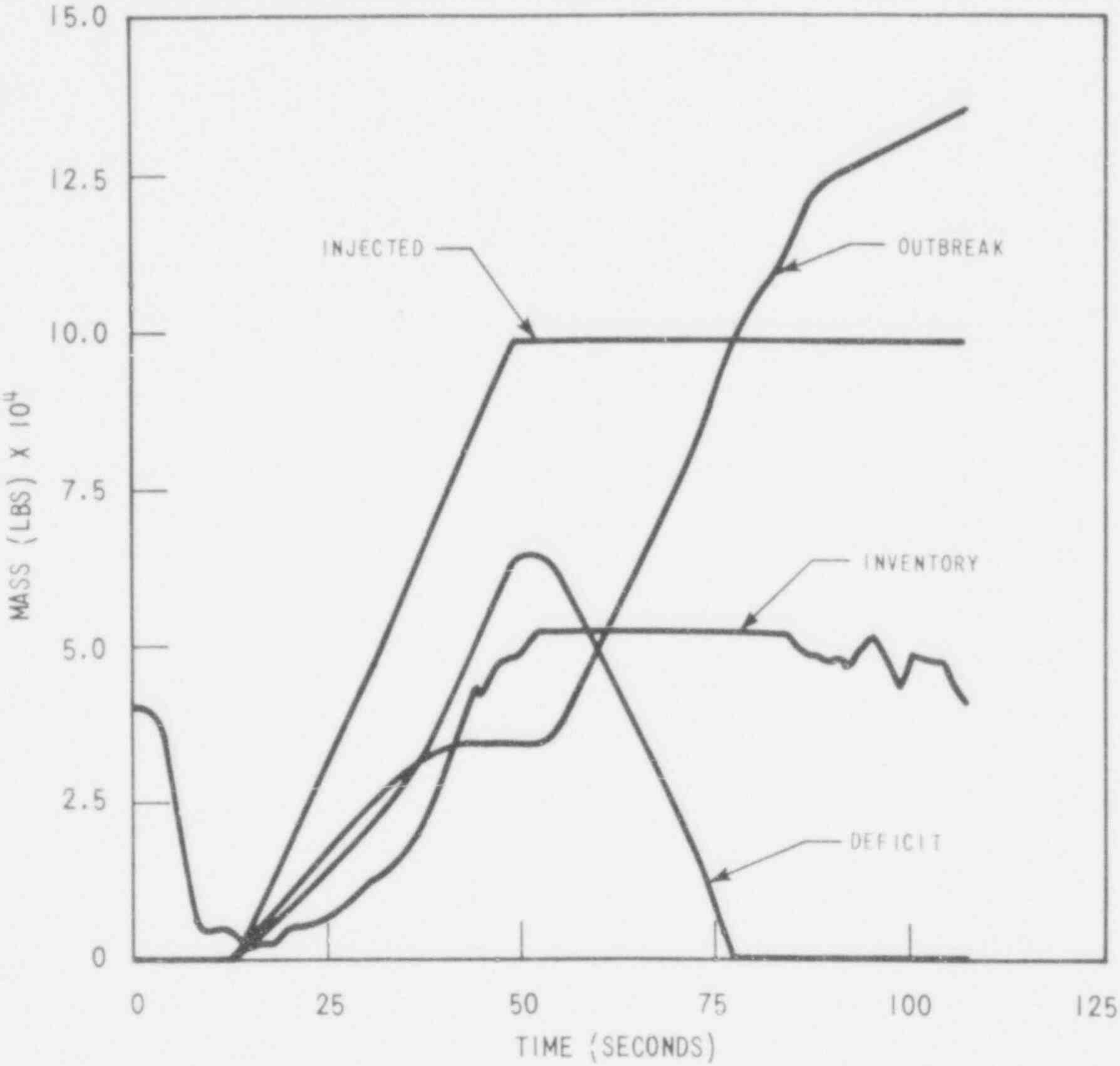
POOR ORIGINAL

WCAP - 9500

Figure 15.6.5-72. BLUE

SI + Accumulator Flow, C_D = 0.4 DECLG, Perfect Mixing

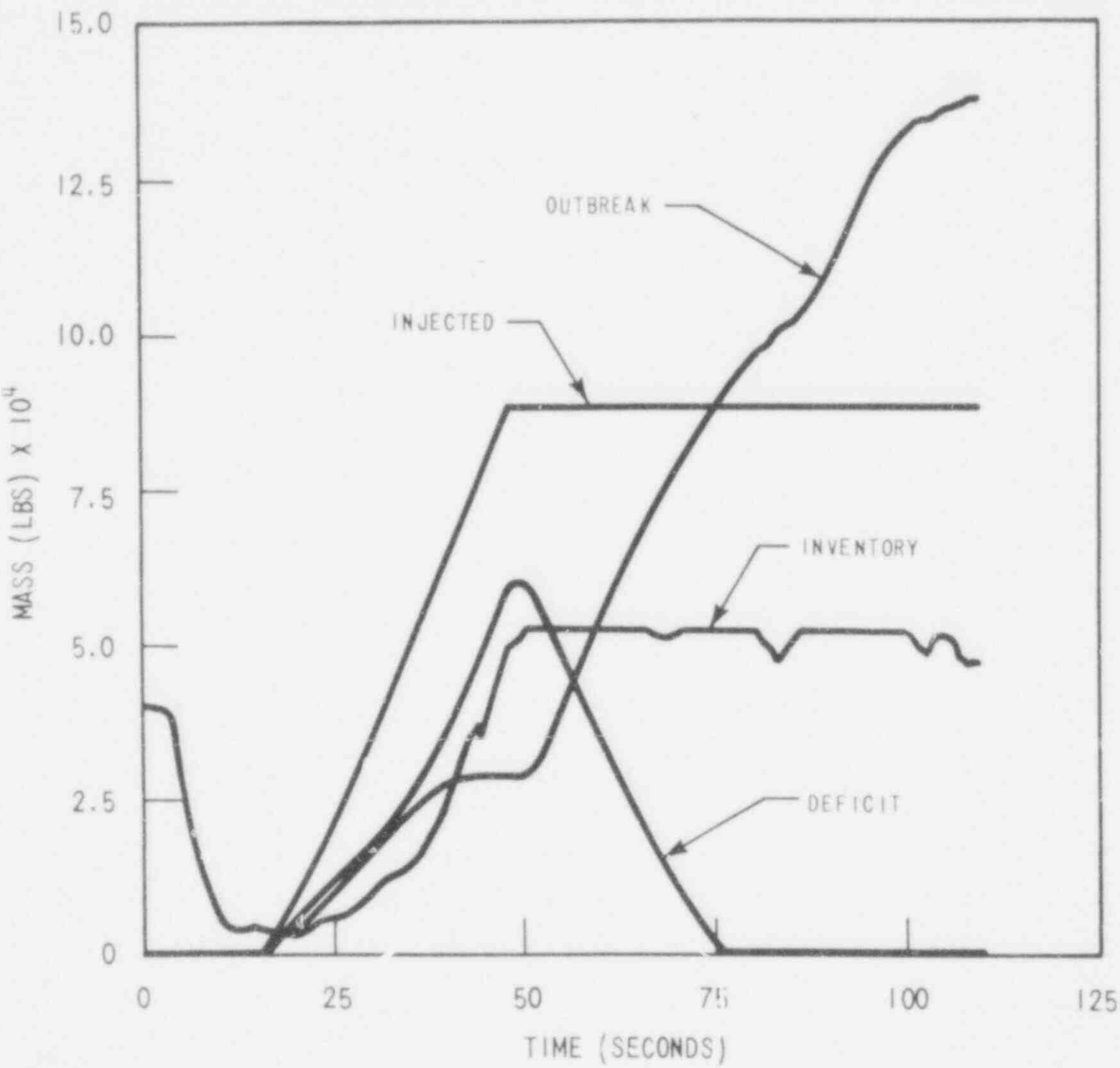
615 209



WCAP-9500

Figure 15.6.5-73. BLUE

Vessel Mass Inventory,
C_D = 1.0 DELCG, Perfect Mixing



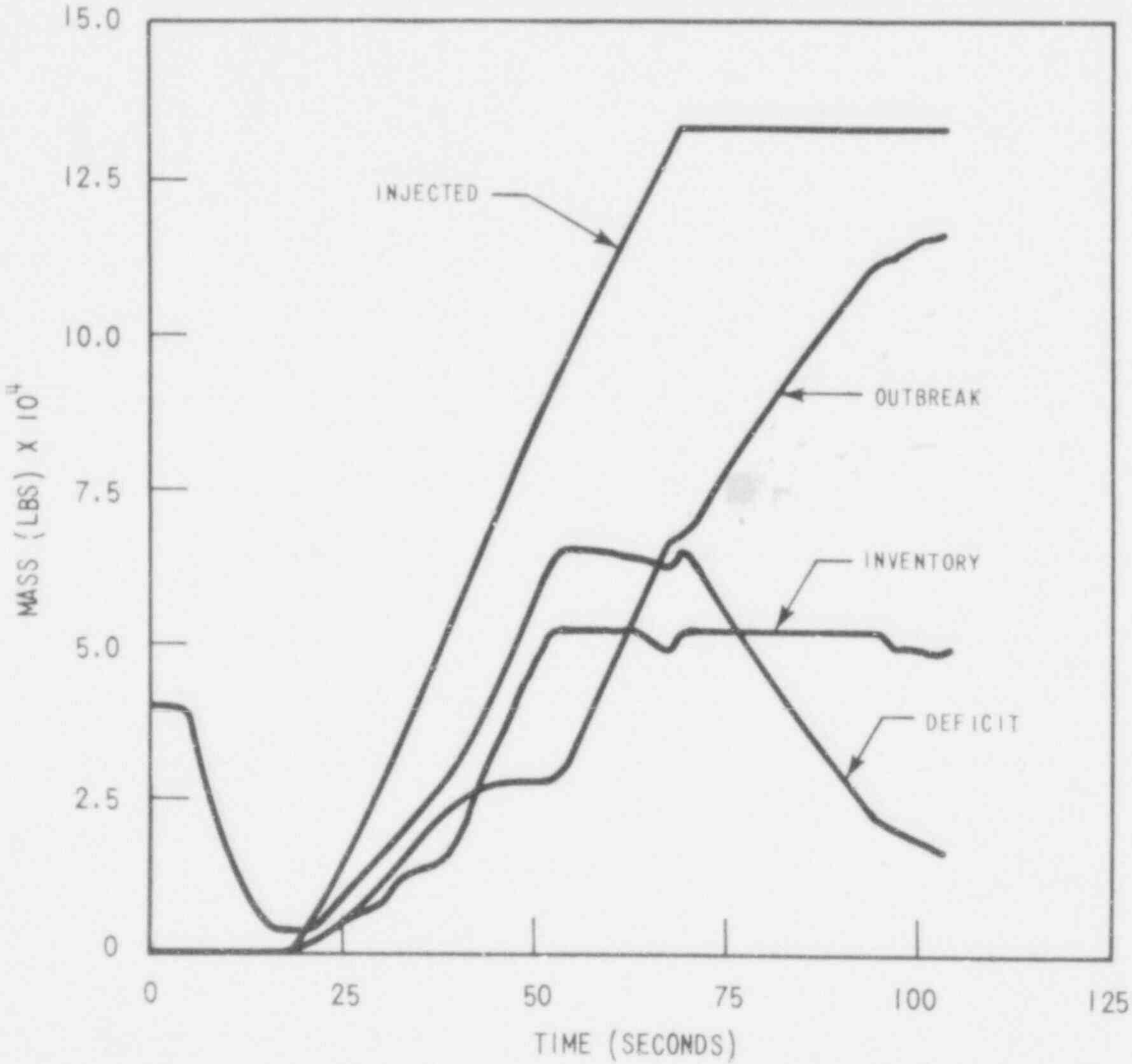
POOR ORIGINAL

615 211

WCAP – 9500

Figure 15.6.5-75. BLUE

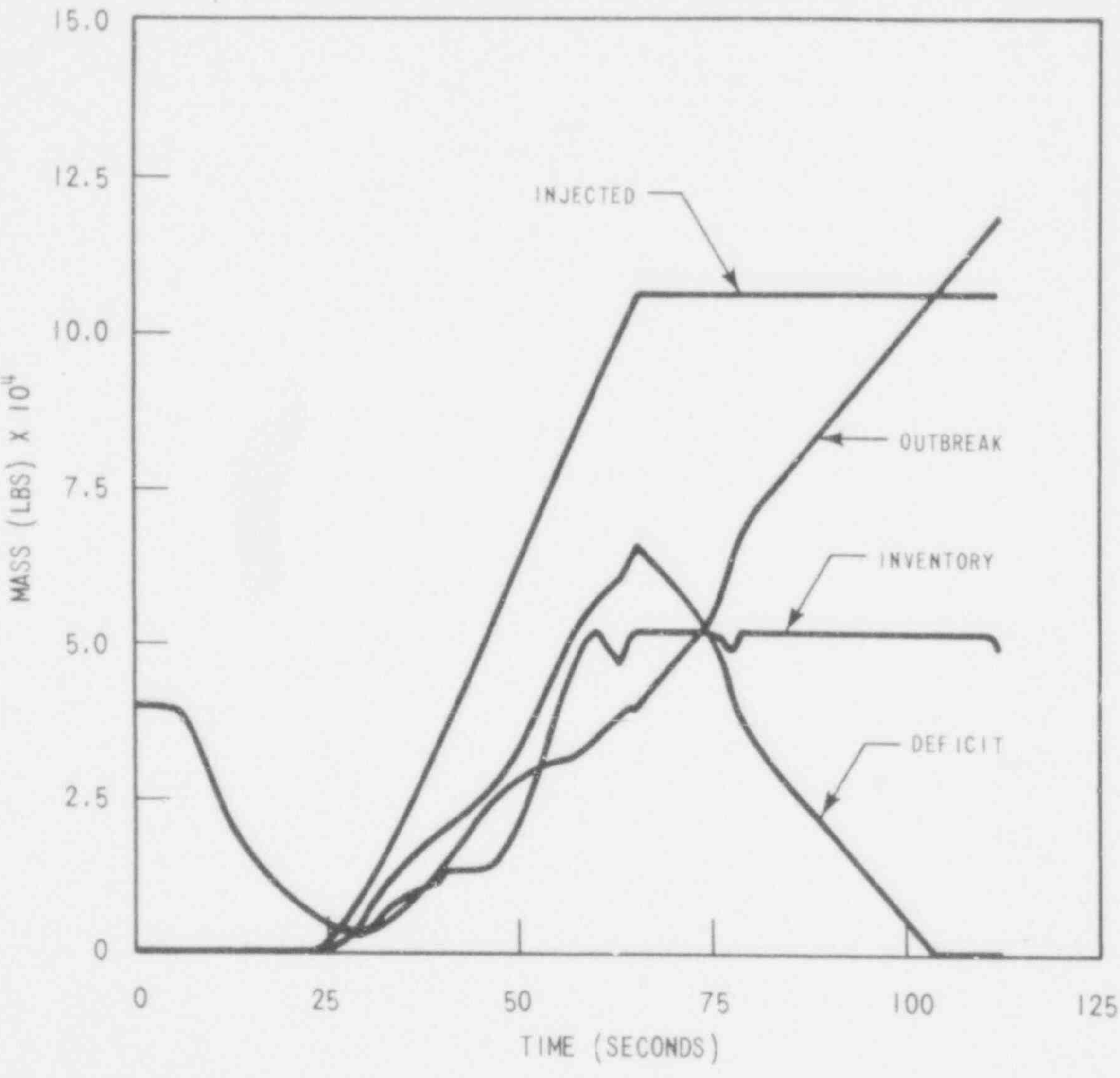
Vessel Mass Inventory,
C_D = 0.8 DECLG, Perfect Mixing



615 212

POOR ORIGINAL

WCAP - 9500
Figure 15.6.5-76. BLUE Vessel Mass Inventory, C_D = 0.6 DECLG, Perfect Mixing



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POOR ORIGINAL

WCAP - 9500

Figure 15.6.5-77. BLUE

Vessel Mass Inventory,
C_D = 0.4 DECLG, Perfect Mixing

15.7

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15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

15.7.1 RADIOACTIVE GAS WASTE SYSTEM LEAK OR FAILURE

Text and further discussion could be provided by Westinghouse on a plant specific basis.

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

Text and further discussion could be provided by Westinghouse on a plant specific basis.

15.7.3 POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES

Text and further discussion could be provided by Westinghouse on a plant specific basis.

15.7.4 FUEL HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the postulated rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.7.4.2 Analysis of Effects and Consequences

The fuel assembly from the core region discharged which has the peak inventory is the assembly assumed to be dropped. The assembly inventory is determined assuming maximum full power operation at the end of core life immediately preceding shutdown. The gap model discussed in Regulatory Guide 1.25 (May 1972) is used to determine the fuel-cladding gap activities. Thus, 10 percent of the total assembly iodines and noble gases, except for 30 percent for Kr-85, are assumed to be in the

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fuel-cladding gap. The remainder of the assumptions used to determine the gap activity of the assembly are listed in Table 15.7-1. The radial peaking factor given in this table is from Regulatory Guide 1.25. The total assembly and fuel-cladding activities at the time of reactor shutdown are given in Table 15.7-2.

15.7.4.3 Radiological Consequences of a Postulated Fuel Handling Accident

Two analyses of a postulated fuel handling accident will be performed: 1) a realistic analysis, and 2) an analysis based on Regulatory Guide 1.25. The parameters used for each of these analyses are listed in Table 15.7-3.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

Text and further discussion could be provided by Westinghouse on a plant specific basis.

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TABLE 15.7-1

Nuclear Characteristics of Peak Inventory Discharged Assembly Used In Radiological Consequences of a Fuel Handling Accident

Core power	3565 MW(t)
Number of assemblies	193
Radial peaking factor	1.65
Maximum fuel rod pressurization	$\leq$ 1200 psia

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BLUE

TABLE 15.7-2

Noble Gas and Iodine Activities Released as a Result Of a Fuel Handling Accident*

	Assembly Activity (Ci)	Fraction of Activity in Gap (%)	Gap Activity (Ci)
Kr-85	1.8 x E(+3)	.3	5.4 x E(+2)
Xe-131m	6.5 x E(+2)	.1	6.5 x E(+1)
Xe-133m	1.2 x E(+4)	.1	1.2 x E(+3)
Xe-133	1.5 x E(+5)		1.5 x E(+4)
Xe-135m	7.8 x E(-1)	.1	7.8 x E(-2)
Xe-135	2.6 x E(+2)	.1	2.6 x E(+1)
I-130	1.1 x E(+1)	.1	1.1 x E(0)
I-131	7.4 x E(+4)	.1	7.4 x E(+3)
I-132	6.2 x E(+4)	.1	6.2 x E(+3)
I-133	7.5 x E(+3)	.1	7.5 x E(+2)
I-135	5.1 x E(0)	.1	5.1 x E(-1)

* These values are based on the following assumptions per Regulatory Guide 1.25

Gap inventory of 314 fuel rods in discharge region
Radial peaking factor of 1.65
Accident occurs 100 hours after shutdown

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TABLE 15.7-3

Parameters Used in Fuel Handling Accident Analyses

	REALISTIC ANALYSIS	REGULATORY GUIDE 1.25 ANALYSIS
Time between plant shutdown and accident	26.5 days*	100 hours
Maximum fuel rod pressurization	$\leq$ 1200 psia	$\leq$ 1200 psia
Minimum water depth between top of damaged fuel rods and pool surface	$\geq$ 23 feet	$\geq$ 23 feet
Damage to fuel assembly	One row of rods (17) ruptured	All rods ruptured
Fuel assembly activity	Average of fuel assemblies in core region discharged	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods	Gap activity in ruptured rods**
Radial peaking factor	1.0	1.65
Form of iodine activity release to spent fuel pool		

TABLE 15.7-3 (Continued)

Parameters Used in Fuel Handling Accident Analyses

	REALISTIC ANALYSIS	REGULATORY GUIDE 1.25 ANALYSIS
elemental iodine	100%	99.75%
methyl iodine	0.0%	0.25%
Decontamination factor in spent fuel pool		
elemental iodine	760	133
methyl iodine	-	1
noble gases	1	1

* Time to transfer one-half of the fuel assemblies in the core region discharged during refueling, based on Westinghouse PWR operating experiences.

** 10% of the total radioactive iodine and 10% of the total noble gases, except for 30% for Kr-85, in the damaged rods at the time of the accident.

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15.8

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15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

A discussion of Anticipated Transients without Scram (ATWS) is presented in Reference [1] .

15.8.1 REFERENCES

1. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

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