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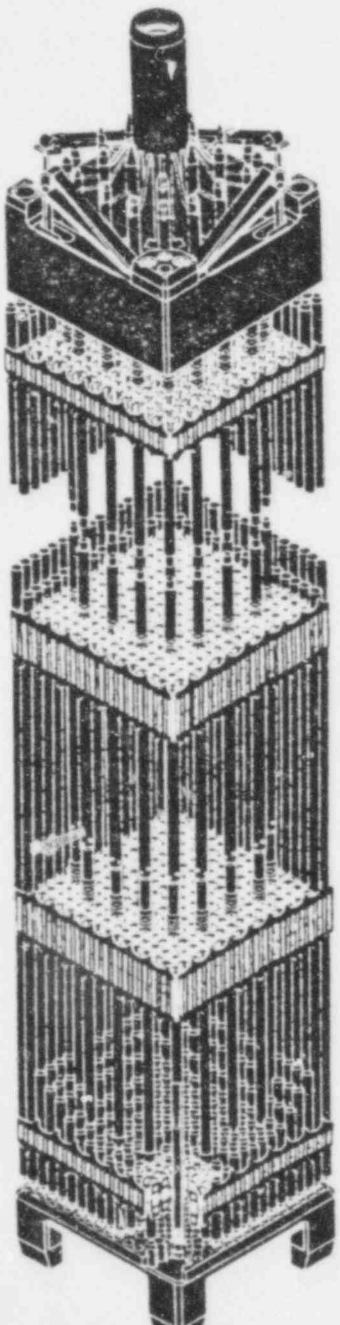
REFERENCE CORE REPORT 17 x 17



OPTIMIZED FUEL ASSEMBLY

VOLUME 2

Westinghouse Nuclear Energy Systems

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15.0 ACCIDENT ANALYSIS

15.0 ACCIDENT ANALYSES

15.0.1 GENERAL

This chapter addresses the representative initiating events listed on Table 15-1 of Regulatory Guide 1.70, Revision 3, the Standard Format and Content for Safety Analysis Reports, as they apply to a Westinghouse Pressurized Water Reactor.

Certain items of Table 15-1 in the guide warrant comment, as follows:

1. 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.

2. Item 6.2 - No instrument lines from the Reactor Coolant System boundary in the NSSS PWR design penetrate the Containment. (For the definition of the Reactor Coolant System boundary, refer to Section 5, ANSI-N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.)

15.0.2 CLASSIFICATION OF PLANT CONDITIONS

Since 1970 the ANS classification of plant conditions has been used to divide plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

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

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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.

15.0.2.1 <u>Condition I - Normal Operation and Operational Transients</u>

Condition I occurrences are those which are expected frequently or regularly in the course of normal plant operation, refueling, and maintenance. 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.

Typical Condition I events are as follows:

1. Steady state and shutdown operations

   a. Mode 1 - Power operation (> 5 to 100 percent of rated thermal power).

   b. Mode 2 - Startup ($K_{eff} \geq 0.99$, ≤ 5 percent of rated thermal power).

   c. Mode 3 - Hot standby ($K_{eff} < 0.99$, $T_{avg} \geq 350^{o}F$).

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d. Mode 4 - Hot shutdown (K_{eff} < 0.99, $200^oF \leq T_{AVG} \leq 350^oF$).

e. Mode 5 - Cold shutdown (K_{eff} < 0.99, $T_{avg} < 200^oF$).

f. Mode 6 - Refueling ($K_{eff} \leq 0.95$, $T_{avg} \leq 140^oF$).

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 oper.cion 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 the Technical Specifications.

3. Operational transients

a. Plant heatup and cooldown (up to 100^oF/hour for the reactor coolant system; 200^oF/hour for the pressurizer during cooldown and 100^oF/hour for the pressurizer during heatup).

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

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

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

15.0.2.2 Condition II - Faults of Moderate Frequency

At worst, a Condition II fault results in a 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 failure or reactor coolant system or secondary system overpressurization.

The following faults are included in this category:

1. Feedwater system malfunctions causing a reduction in feedwater temperature (Subsection 15.1.1).

2. Feedwater system malfunctions causing an increase in feedwater flow (Subsection 15.1.2).

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

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Subsection 15.1.4).

5. Loss of external load (Subsection 15.2.2).

6. Turbine trip (Subsection 15.2.3).

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

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

9. Loss of nonemergency A-C power to the station auxiliaries (Subsection 15.2.6).

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

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

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

13. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2).

14. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly) (Subsection 15.4.3).

15. Startup of an inactive reactor coolant loop at an incorrect temperature (Subsection 15.4.4).

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

17. Inadvertent operation of emergency core cooling system during power operation (Subsection 15.5.1).

18. Chemical volume control system malfunction that increases reactor coolant inventory (Subsection 15.5.2).

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

20. Failure of small lines carrying primary coolant outside containment (Subsection 15.6.2).

15.0.2.3 Condition III - Infrequent Faults

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude immediate resumption of the operation. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary. 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. The following faults are included in this category:

1. Minor steam system piping failures (Subsection 15.1.5).

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

3. Control rod misalignment - Single rod cluster control assembly withdrawal at full power) (Subsection 15.4.3).

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

5. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes, which actuate the emergency core cooling system (Subsection 15.6.5).

6. Waste gas system failure (Subsection 15.7.1).

7. Radioactive liquid waste system leak or failure (atmospheric release) (Subsection 15.7.2).

8. Liquid containing tank failure (Subsection 15.7.3).

15.0.2.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Plant design must be such as to preclude a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault must not cause a consequential loss of required functions of systems needed to mitigate the consequences of the fault including those of the emergency core cooling system and containment. The following faults have been classified in this category:

1. Steam system piping failure (Subsection 15.1.5).

2. Feedwater system pipe break (Subsection 15.2.8).

3. Reactor coolant pump rotor seizure (locked rotor) (Subsection 15.3.3).

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

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

6. Steam generator tube failure (Subsection 15.6.3).

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

8. Fuel handling accident (Subsection 15.7.4).

15.0.3 OPTIMIZATION OF CONTROL SYSTEMS

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

The system setpoints are derived by an analysis of the following control systems: rod control, steam dump, steam generator level, pressurizer pressure and pressurizer level.

15.0.4 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES

15.0.4.1 Design Plant Conditions

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

1. The guaranteed nuclear steam supply system thermal power output. This power output includes the thermal power generated by the reactor coolant pumps and is consistent with the license application rating described in Chapter 1.0.

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

Where initial power operating conditions are assumed in accident analyses, the "guaranteed nuclear steam supply system thermal power output" is assumed. Where demonstration of adequacy of the containment and

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engineered safety features are concerned, the "engineered safety features NSSS design rating" is assumed. Allowances for errors in the determination of the steady-state power level are made as described in Subsection 15.0.4.2. The values of pertinent plant parameters utilized in the accident analyses are given in Table 15.0-2. The thermal power values used for each transient analyzed are given in Table 15.0-3. In all cases where the ESF design rating is used in an analysis, the resulting transients and consequences are conservative compared to using the guaranteed NSSS thermal power rating.

15.0.4.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure noted above are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-8567 (Reference 1). 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 which the Improved Thermal Design Procedure is not employed, 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	$\pm$ 2% allowance for calorimetric error
2. Average reactor coolant system temperature	$\pm$ 4°F allowance for controller deadband and measurement error
3. Pressurizer pressure	$\pm$ 30 psi allowance for steady-state fluctuations and measurement error.

Table 15.0-3 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.

15.0.4.3 Power Distribution

The limiting conditions occurring during reactor transients are dependent on the core power distribution. The design of the core and the control system minimizes adverse power distribution through the placement of control rods and operating methods. In addition, the core power distribution is continuously monitored by the integrated protection system as descibed in Chapters 7 and 16. Audible alarms will be activated in the control room whenever the power distribution exceeds the limits assumed as initial conditions for the transients presented in this chapter.

For transients which may be DNB limited both the radial and axial peaking factors are of importance. The core thermal limits illustrated in Figures 15.0-1 and 15.0-1a are based on a reference axial power shape. The Low DNBR reactor trip setpoint is automatically adjusted for axial shapes differing from the reference shape by the method described in Section 4.4 and also described in Chapter 7 and 16. The radial peaking factor $F_{\Delta H}$ increases with decreasing power and with increasing rod insertion. The increase in $F_{\Delta H}$ resulting from decreasing reactor power and increased rod insertion is accounted for in the low DNBR reactor trip through measurement of power and control rod position.

For transients which may be overpower limited, the total peaking factor F_q is of importance. F_q is continuously monitored through the High Kw/ft reactor trip as described in Chapters 7 and 16 to assure that the limiting overpower conditions are not exceeded.

For overpower transients which are slow with respect to the fuel rod thermal time constant, fuel rod thermal evaluations are determined as discussed in Section 4.4. Examples of this are the uncontrolled boron dilution incident, which lasts many minutes, and the excessive load increase incident, which reaches equilibrium without causing a reactor trip. For overpower transients which are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical and rod cluster control assembly ejection incidents, which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is 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.

15.0.5 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 4.0.

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 highly on reactivity feedback effects. The values used for each accident are given in Table 15.0-3. Reference is made in that table to Figure 15.0-2 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event-by-event basis. Conservative combinations of parameters are used for a given transient to bound the effects of core life, although these combinations may not represent possible realistic situations.

15.0.6 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% of the rod cluster travel. For all accidents except the loss of flow events, the insertion time to dashpot entry is conservatively taken as 3.3 seconds. For the partial and complete loss of forced reactor coolant flow and locked rotor accidents (Subsections 15.3.1, 15.3.2, and 15.3.3/4), a time to dashpot entry of 2.7 seconds based on the thermal design flow rate was assumed. This assumption is discussed in Reference [2]. The time to dashpot is based on D-loop test results described in Reference [3]. The normalized rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 15.0-4.

Figure 15.0-5 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not an input into the point kinetics core model.

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

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figures 15.0-4 and 15.0-5. A total negative reactivity insertion following a trip of 4% $\Delta\rho$ 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 Section 4.3.

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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 15.0-6) is used in those transient analyses for which a point kinetics core model is used. Where special analyses required 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 (Figure 15.0-4) is used as code input.

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

A reactor trip signal acts to open eight trip breakers, two per channel set, 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 15.0-4.

Reference is made in Table 15.0-4 to the low DNBR trips shown in Figures 15.0-1 and 15.0-1a. These figures present the allowable reactor power as a function of the coolant loop inlet temperature and primary coolant pressure for N and N-1 loop operation (4 and 3-loop operation), for the design flow and power distribution, as described in Section 4.4.

The boundaries of operation defined by the low DNBR trip are represented as "protection lines" on this diagram. During operation with one loop out of service, the Integrated Protection System will automatically select setpoints for the Low DNBR trip consistent with the core limits for N-1 loop operation. 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 DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.82 for the thimble cell and 1.85 for the typical cell - see also Section 4.4). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value with the assumed axial and radial power distributions. The diagram shows that the DNB design basis is not violated 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); low DNBR (variable setpoint); high kw/ft (fixed setpoint).

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), 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 nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications, Chapter 16.0. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the plant technical specifications.

15.0.8 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX

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

15.0-5. The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the multiple 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.

15.0.9 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

The Westinghouse Nuclear Steam Supply System (NSSS) is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and the dynamic effects of the postulated accident. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17.0 discusses the quality assurance program which is implemented to ensure that the plant will be designed, constructed, and operated without undue risk to the health and safety of the general public. The incorporation of these features, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 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 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15 events, the operation of the non-safety-related rod control system, other than the reactor trip portion of the Control Rod Drive System (CRDS), is considered only if that action results in more severe consequences. 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 non-safety-related control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of Chapter 15 events.

15.0.10 FISSION PRODUCT INVENTORIES

15.0.10.1 Activities in the Core

The calculation of the core iodine fission product inventory is consistent with the inventories given in TID-14844 (Reference 4) 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 5). These inventories are given in Tables 15.0-7. The isotopes included in Table 15.0-7 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 15.0-8.

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.

15.0.10.2 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 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 15.0-7.

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15.0.11 RESIDUAL DECAY HEAT

15.0.11.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-ofcoolant accident per the requirements of Appendix K, 10 CFR 50.46, as described in References [6] and [7]. 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.

15.0.12 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 are listed in Table 15.0-3.

15.0.12.1 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 cladding using as input the nuclear power and the 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.

FACTRAN is further discussed in Reference [8].

15.0.12.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generators (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 neutron flux, low DNBR, high linear power (kW/ft), 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. ECCS, including the accumulators, is also modeled.

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

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

LOFTRAN is further discussed in Reference [9].

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15.0.12.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two or 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. 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 described in Reference [10].

15.0.12.4 WIT

WIT is a one-region neutron kinetics program with a single axial lump description of thermal kinetics making it useful in the analysis of transients in a heterogeneous reactor core consisting of fuel rods, fuel rod clad, and water moderator and coolant. The code is basically a core model and therefore generally useful for reactivity transients which terminate before significant effects occur from the remainder of the plant, i.e. transients shorter than the loop transit time, or subcritical events.

WIT is used in safety analysis of reactivity accidents from a subcritical condition.

WIT is further described in Reference [11].

15.0.12.5 THINC

The THINC Code is described in Section 4.4.

15.0.13 REFERENCES

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9. T. W. T. Burnett et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

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

11. D. B. Fairbrother and H. G. Hargrove, "WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, November, 1972.

TABLE 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

	N-Loop Operation	N-1 Loop Operation
Reactor core thermal power output (MWt)*	3411	2389
Thermal power generated by the reactor coolant pumps (MWt)	16	11
Guaranteed Nuclear Steam Supply System thermal power output (MWt)	3427	2400
Engineered Safety Features NSSS design rating (maximum calculated turbine rating) (MWt)	3581	2508

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

TABLE 15.0-2

VALUES OF PERTINENT PLANT PARAMETERS UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation	N-1 Loop Operation
Thermal output of nuclear steam supply system (MWt)	3427	2400
Reactor Core Thermal Power Output (MWt)	3411	2389
Core inlet temperature (^oF)	562.5	560.9
Reactor coolant average temperature (^oF)	591.6	586.8
Reactor coolant system pressure (psia)	2250	2250
Reactor coolant flow per loop (gpm)	97,100	103,400 (Active Loops) -29,100 (Inactive Loop)
Total reactor coolant flow (10^6 lb/hr)	143.4	104.0
Total steam flow from NSSS (10^6 lb/hr)	15.26	10.15
Steam pressure at steam generator outlet (psia)	1000	991
Maximum steam moisture content (%)	0.25	0.25
Feedwater temperature at steam generator inlet (^oF)	445	408
Average core heat flux ($Btu/hr\text{-}ft^2$)	197,200	138,100

* For accident analyses using the improved thermal design procedure.

SUMMARY O

FAULTS	COMPUTER CODES UTILIZED	KINETIC PARAMETERS ASSUMED: DELAYED NEUTRON FRACTION	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATIO
15.1 Increase in Heat Removal by the Secondary System					
- Feedwater System Malfunction Causing an Increase in Feedwater Flow	LOFTRAN	.0044	0.43	Minimum*	WRB-1
- Excessive Increase in Secondary Steam Flow	LOFTRAN	.0044/.0075	Figure 15.0-3 and 0.43	Maximum and Minimum*	WRB-1
- Accidental Depressurization of the Main Steam System	LOFTRAN	.0044	Function of Moderator Density, See Subsection 15.1-4 (Figure 15.1-11)	-2.2 pcm/^{o}F	W-3
- Steam System Piping Failure	THINC, LOFTRAN	.0044	Function of Moderator Density, See Subsection 15.1.5 (Figure 15.1-11)	See Section 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	.0044/.0075	Figure 15.0-3 and 0.43	Maximum*	WRB-1
- Loss of Non-Emergency A-C Power to the Station Auxiliaries	LOFTRAN	.0075	Figure 15.0-3	Maximum*	NA
- Loss of Normal Feedwater Flow	LOFTRAN	.0075	Figure 15.0-3	Maximum*	NA
- Feedwater System Pipe Break	LOFTRAN, FACTRAN	.0075	0.43	Maximum*	NA

613 054

TABLE 15.0-3

INITIAL CONDITIONS AND COMPUTER CODES USED

N	IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT (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 2400	388,400 281,100	562.5 560.9	2250 2250	1080 891	445 408
	Yes	3427 2400	388,400 281,100	562.5 560.9	2250 2250	1093 904	445 408
	No	0 (Subcritical)	382,400	557	2250	450	50
	No	0 (Subcritical)	382,400	557	2250	450	50
	Yes	3427 2400	388,400 281,100	562.5 560.9	2250 2250	1080 891	445 408
	NA	3581 2578	382,400 276,800	566.5 564.9	2280 2280	1122 916	447 410
	NA	3581 2578	382,400 276,800	566.5 564.9	2280 2280	1122 916	447 410
	NA	3581	382,400	562.5	2280	1116	445

613 055

FAULTS	COMPUTER CODES UTILIZED	KINETIC PARAMETERS ASSUMED DELAYED NEUTRON FRACTION	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATION
15.3 Decrease in Reactor Coolant System Flow Rate					
- Partial and Complete Loss of Forced Reactor Coolant Flow	LOFTRAN, THINC, FACTRAN	.0075	Figure 15.0-3	Maximum*	WRB-1
- Reactor Coolant Pump Shaft Seizure (Locked Rotor	LOFTRAN, FACTRAN	.0075	Figure 15.0-3	Maximum*	WRB-1
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	.0075	Refer to Subsection 15.4.1.2	Consistent with upperlimit shown on Figure 15.0-2	WRB-1
- Uncontrolled Rod Cluster Assemlby Bank Withdrawal at Power	LOFTRAN	.0075	Figure 15.0-3 and 0.43	Maximum and Minimum*	WRB-1
- Control Rod Mis-alignment	THINC, LOFTRAN	NA	NA	NA	WRB-1
- Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	THINC, LOFTRAN, FA[illegible]	NA	0.43	Minimum*	WRB-1
- Chemical and Volume Control System Mal-function that Results in a Decrease in Boron Concentration in the Reactor Coolant	[illegible]T	.0044	Figure 15.0-3	Minimum*	NA

613 056

TABLE 15.0-3 (Continued)

IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL INLET TEMPERATURE (^oF)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (^oF)
Yes	3427 2400	388,400 281,100	562.5 560.9	2250 2250	1080 891	445 408
No	3427 2400	382,400 276,800	566.5 564.9	2280 2280	1080 891	445 408
Yes	0	175,900	557	2250	NA	NA
Yes	3427, 2056, 343 2400, 343	388,400 281,100	562.5/560.3/557.6 560.9/557.6	2250 2250	1080/828/513 891/513	445, 387, 245 408, 245
Yes	3427	388,400	562.5	2250	NA	NA
Yes	2400	281,100	560.9	2250	891	408
NA	0 and 3427	NA	NA	NA	NA	NA

613 057

FAULTS	COMPUTER CODES UTILIZED	KINETIC PARAMETERS ASSUMED: DELAYED NEUTRON FRACTION	MODERATOR DENSITY ($\Delta\rho$/gm/cc)	DOPPLER	DNB CORRELATI[ON]
- 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	.0055/.0044	Refer to Subsection 15.4.8 (BOC, EOC)	Consistent with lower limit shown on Figure 15.0-2	NA
15.5 Increase in Coolant Inventory					
- Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA	NA
15.6 Decrease in Reactor Coolant Inventory					
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	LOFTRAN	.0044	Figure 15.0-3	Maximum*	WRB-1

* Reference Figure 15.0-2. Maximum refers to lower curve and minimum refers to upper curve.

NA - Not Applicable

BOC - Beginning of Cycle

EOC - End of Cycle

613 058

TABLE 15.0-3 (Continued)

IMPROVED THERMAL DESIGN PROCEDURE	INITIAL NSSS THERMAL POWER OUTPUT (MWt)	REACTOR VESSEL COOLANT FLOW (GPM)	VESSEL INLET TEMPERATURE (^{o}F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER VOLUME (ft^3)	FEEDWATER TEMPERATURE (^{o}F)
NA	3427	388,400	562.5	2250	1080	445
NA	3427 0	382,400 175,900	562.5 557	NA	NA	NA
NA	NA	NA	NA	NA	NA	NA
Yes	3427 2400	388,400 281,100	562.5 560.9	2250 2250	1080 891	445 408

613 059

TABLE 15.0-2a

VALUES OF PERTINENT PLANT PARAMETERS UTILIZED IN ACCIDENT ANALYSES*

	N-Loop Operation	N-1 Loop Operation
Thermal output of nuclear steam supply system (MWt)	3427	2400
Reactor core thermal power output (MWt)	3411	2389
Core inlet temperature (^oF)	562.5	560.9
Reactor coolant average temperature (^oF)	592.0	587.2
Reactor coolant system pressure (psia)	2250	2250
Reactor coolant flow per loop (gpm)	95,600	101,800 (Active Loops) -28,600 (Inactive Loop)
Total reactor coolant flow (10^6 lb/hr)	141.2	102.4
Total steam flow from NSSS (10^6 lb/hr)	15.27	10.16
Steam pressure at steam generator outlet (psia)	1018	1009
Maximum steam moisture content (%)	0.25	0.25
Feedwater temperature at steam generator inlet (^oF)	445	408
Average core heat flux (Btu/hr-ft^2)	197,200	138,100

*For accident analyses not using the improved thermal design procedure.

613 060

TABLE 15.0-4

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (sec)
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
Low DNBR	Variable, see Figures 15.0-1 and 15.0-1a	6.0*
High kw/ft	18 kw/ft	2.5
High Pressurizer Pressure	2410 psig	2.0
Low Pressurizer Pressure	1860 psig	2.0
Low Reactor Coolant Flow (from loop flow detectors)	87% loop flow	1.0
RCP Underspeed	93% of nominal speed	0.6
Turbine Trip	Not applicable	2.0

613 061

TABLE 15.0-4 (Continued)

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (sec)
Safety Injection Reactor Trip	Not applicable	2.0
Low Steam Generator Level	7.2% of narrow range level span	2.0
High Steam Generator Level - produces feedwater isolation and turbine trip	83.1% of narrow range level span	2.0

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* Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature in the coolant loops exceeds the trip setpoint until the rods are free to fall.

613 062

TABLE 15.0-5

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)

613 063

TABLE 15.0-5 (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

613 064

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1 Increase in Heat Removed by the Secondary System				
- Feedwater System Malfunction Causing an Increase in Feedwater Flow	Power range high flux, high steam generator level, Manual low DNBR, high kw/ft	High steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation valves	NA
- Excessive Increase Secondary Steam Flow	Power range high flux, Manual, low DNBR, high kw/ft	NA	Pressurizer self-actuated safety valves; steam generator safety valves	NA
- Accidental Depressurization of the Main Steam System	Low pressurizer pressure, Manual, SIS	Low pressurizer pressure, low compensated steam line pressure, Hi-1 containment pressure, Manual, low 4 T_{cold}	Feedwater isolation valves, Steamline stop valves	Auxiliary feed system Safety Injection System
- Steam System Piping Failure	SIS, low pressurizer pressure, Manual	Low pressurizer pressure, low compensated steamline pressure, Hi-1 containment pressure, Manual, low 4 T_{cold}	Feedwater isolation valves, Steamline stop valves	Auxiliary feed system; Safety Injection System
15.2 Decrease in Heat Removal by the Secondary System				
- Loss of External Electrical Load/ Turbine Trip	High Pressurizer pressure, low DNBR, low steam generator level, manual	Low steam generator level	Pressurizer safety valves, steam generator safety valves	Auxiliary feed system

TABLE 15.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
- Loss of Non-Emergency A-C Power to the Station Auxiliaries	Steam generator low level, Manual	Steam generator low level	Steam generator safety valves	Auxiliary feed system
- Loss of Normal Feedwater Flow	Steam generator low level, Manual	Steam generator low level	Steam generator safety valves	Auxiliary feed system
- Feedwater System Pipe Break	Steam generator low level, High Pressurizer Pressure, SIS, Manual, low DNBR	Hi-1 containment pressure, steam generator low level, low compensated steamline pressure	Steamline isolation valves, feedline isolation, Pressurizer safety valves, steam generator safety valves	Auxiliary feed system, Safety injection System
15.3 Decrease in Reactor Coolant System Flow Rate				
- Partial and Complete Loss of Forced Reactor Coolant Flow	Low flow, low RCP speed, 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

TABLE 15.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

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 s.p.), Manual	NA	NA	NA
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power range high flux, Hi pressurizer pressure, Manual, low DNBR, high kw/ft	NA	Pressurizer safety valves, steam generator safety valves	NA
- Control Rod Mis-alignment	Power range negative flux rate, Manual	NA	NA	NA
- Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Power range high flux, P-8, Manual	NA	NA	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, Manual, low DNBR, high kw/ft	NA	Low insertion limit annunciators for boration, VCT outlet isolation valves	NA

TABLE 15.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
- Spectrum of Rod Cluster Control Assembly Ejection Accidents	Power range high flux, High positive flux rate, Manual	NA	NA	NA
15.5 Increase in Reactor Coolant Inventory				
- Inadvertent Operation of ECCS During Power Operation	NA	NA	NA	NA
15.6 Decrease in Reactor Coolant Inventory				
- Inadvertent Opening of a Pressurizer Safety or Relief Valve	Pressurizer low pressure, Manual, low DNBR	Low pressurizer pressure	NA	Safety Injection System
- Steam Generator Tube Rupture	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety valves, steamline stop valves	Emergency Core Cooling System, Auxiliary Feedwater System, Emergency Power Systems

TABLE 15.0-6 (Continued)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
- Loss of Coolant Accident from Spectrum of Postulated Piping Breaks within the System	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

TABLE 15.0-7

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

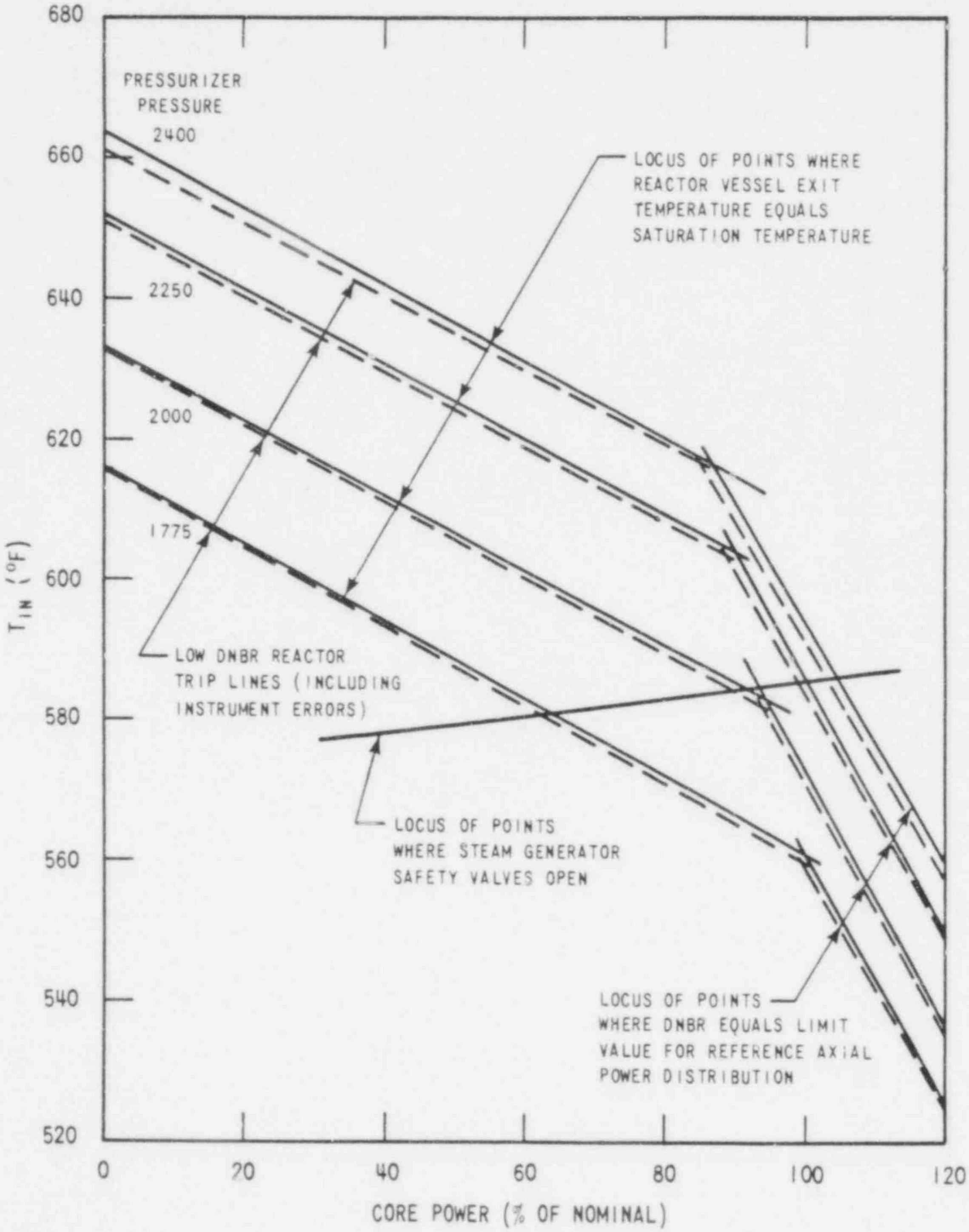
613 070

TABLE 15.0-8

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 .12% fuel defect level.
**Iodine spike assumed to be 500 times the equilibrium rate.

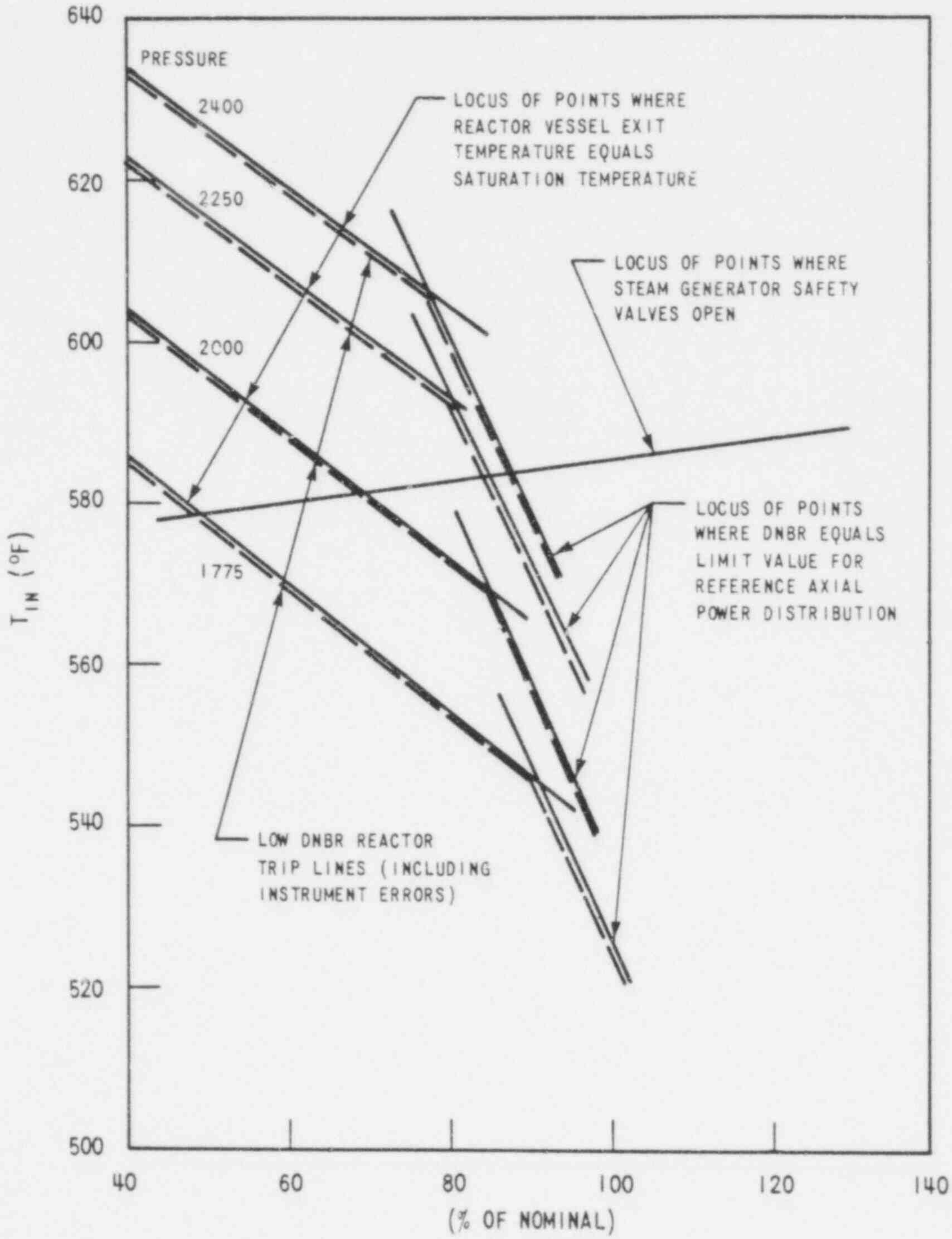


WCAP - 9500

Figure 15.0-1.

Illustration of Core Thermal Limits and DNB Protection (N Loop Operation)

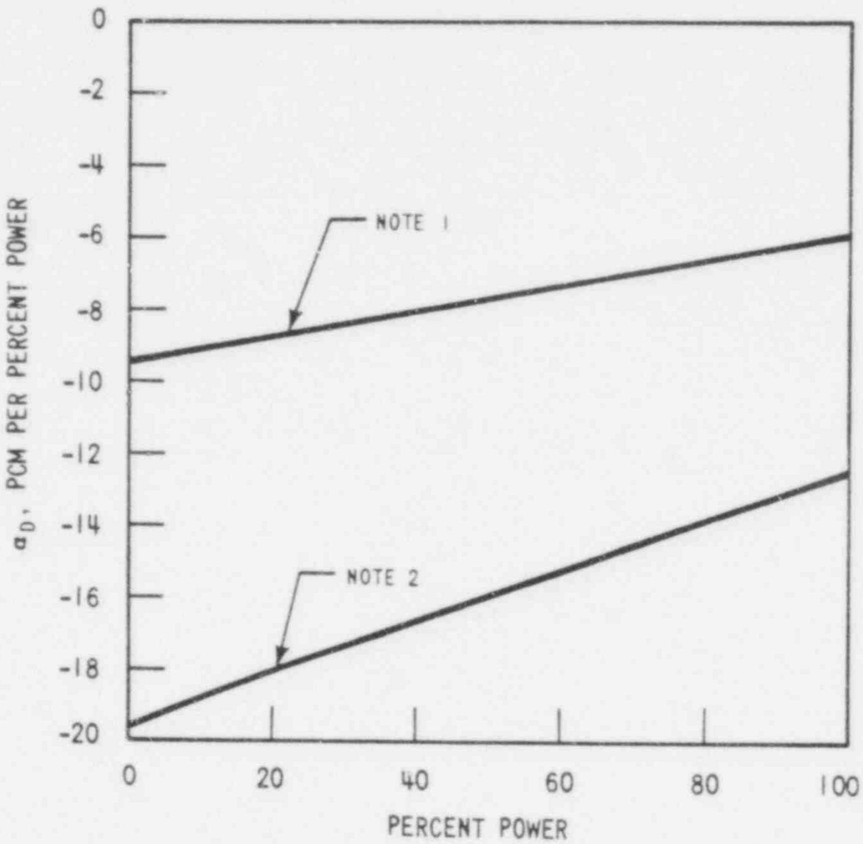
613 072



WCAP – 9500

Figure 15.0-1a.

Illustration of Core Thermal Limits and DNB Protection (N-1 Loop Operation)



NOTE 1 - UPPER CURVE, LEAST NEGATIVE DOPPLER ONLY POWER DEFECT = -0.78% $\Delta\rho$ (0 TO 100% POWER)

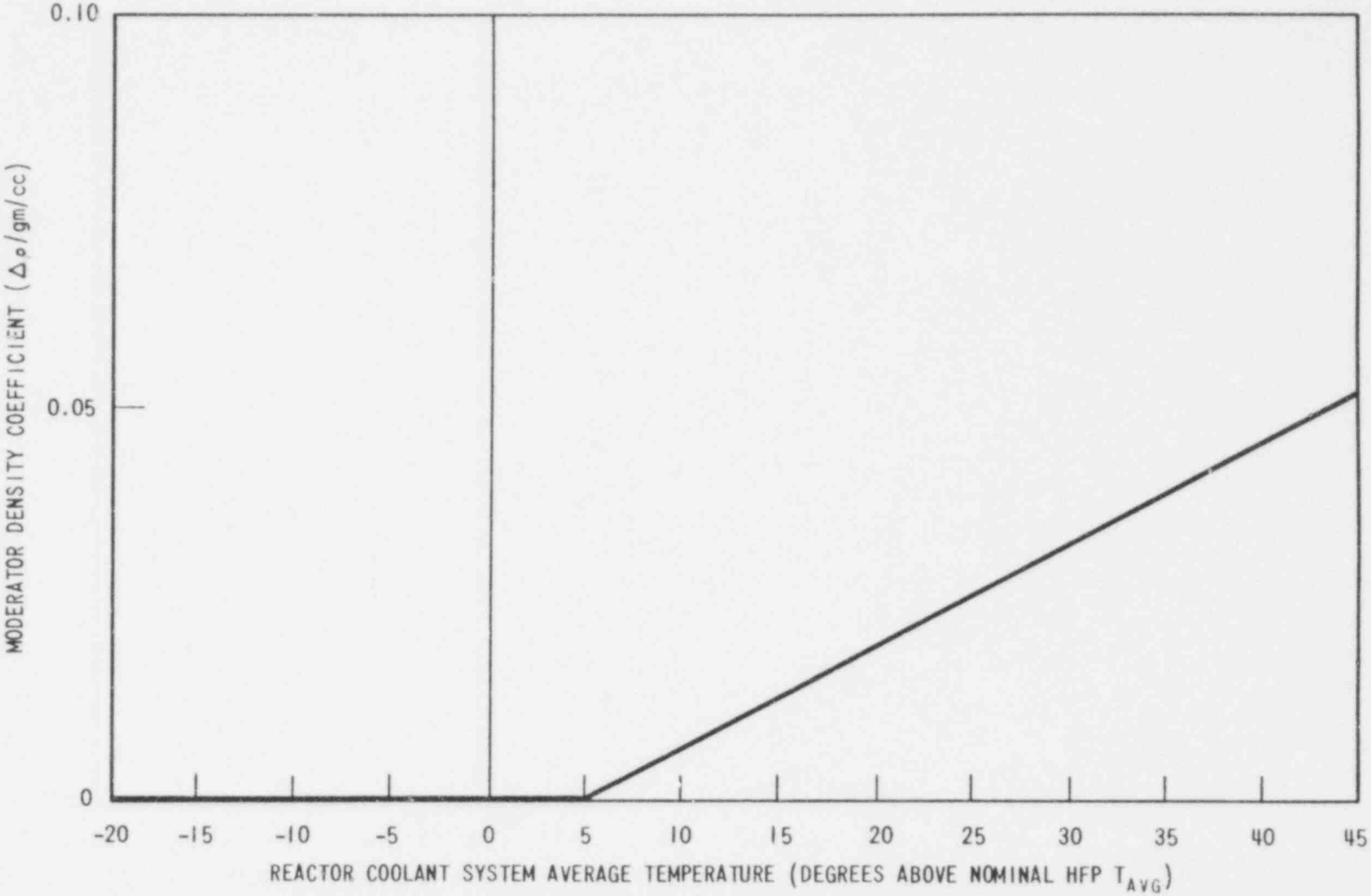
NOTE 2 - LOWER CURVE, MOST NEGATIVE DOPPLER ONLY POWER DEFECT = -1.6% $\Delta\rho$ (0 TO 100% POWER)

613 074

WCAP-9500

Figure 15.0-2.

Doppler Power Coefficient Used In Accident Analysis



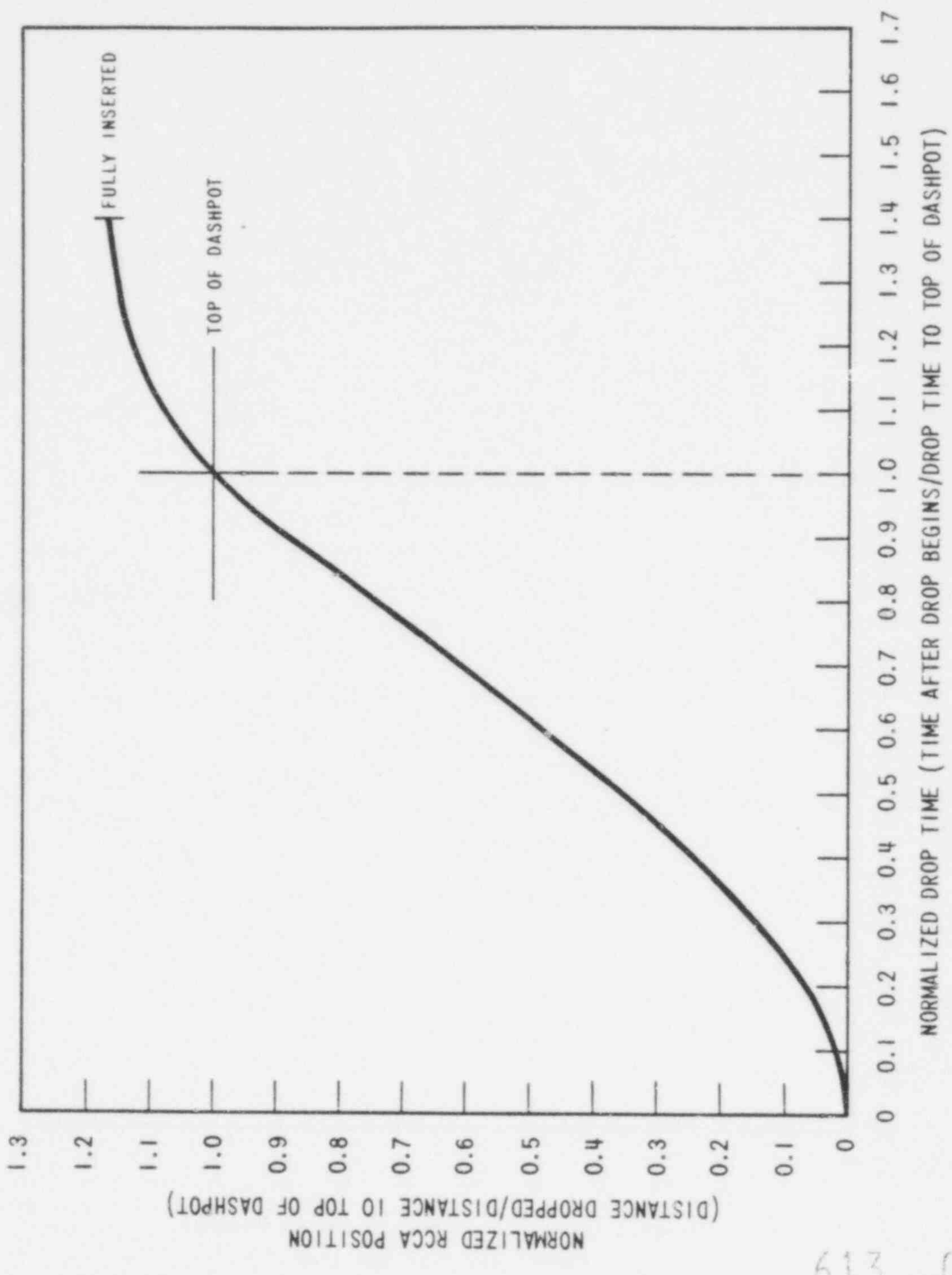
14,395-179

WCAP - 9500

Figure 15.0-3.
Minimum Moderator Density Coefficient Used
In Analysis

613 075

14,395-180



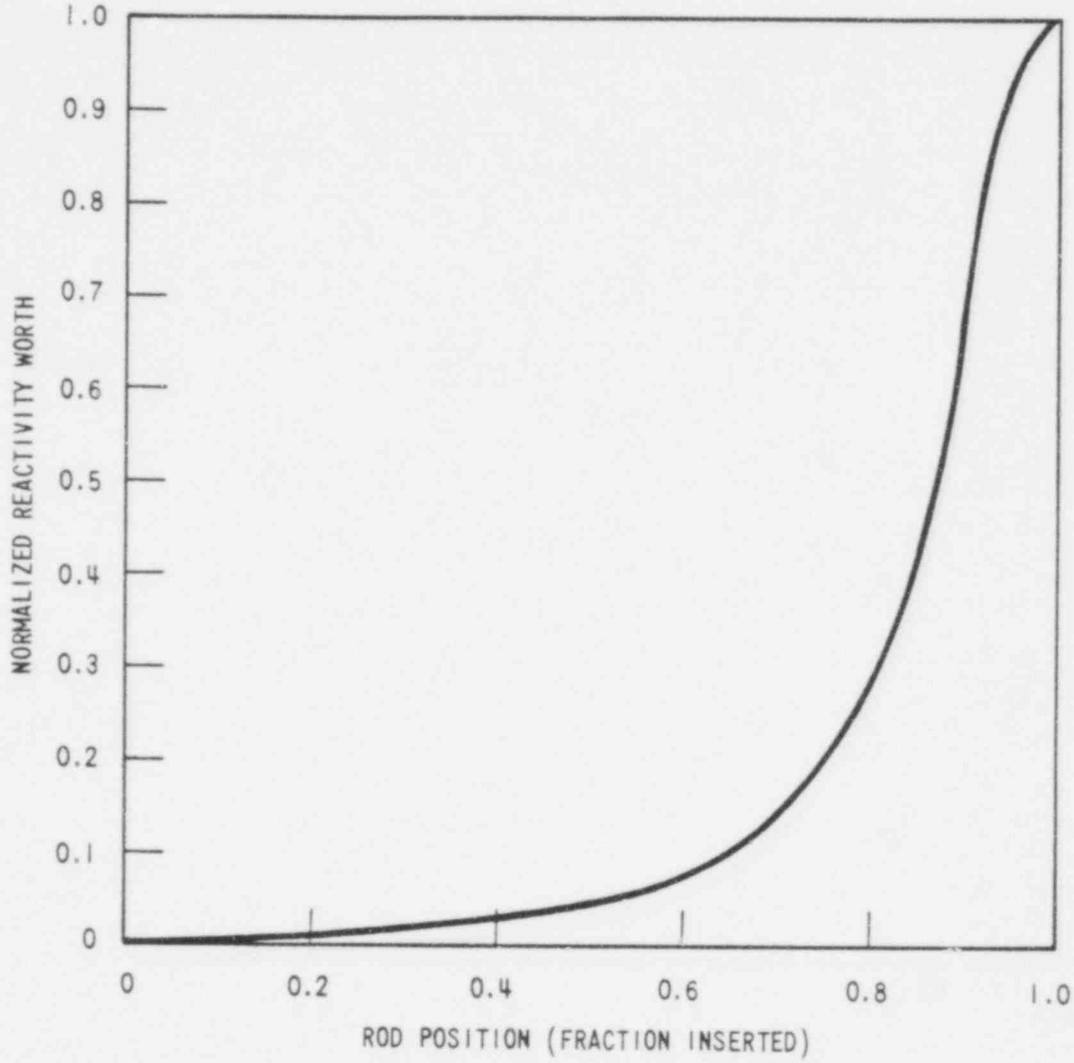
613 076

WCAP – 9500

Figure 15.0-4.

RCCA Position vs. Time to Dashpot

14,395-181

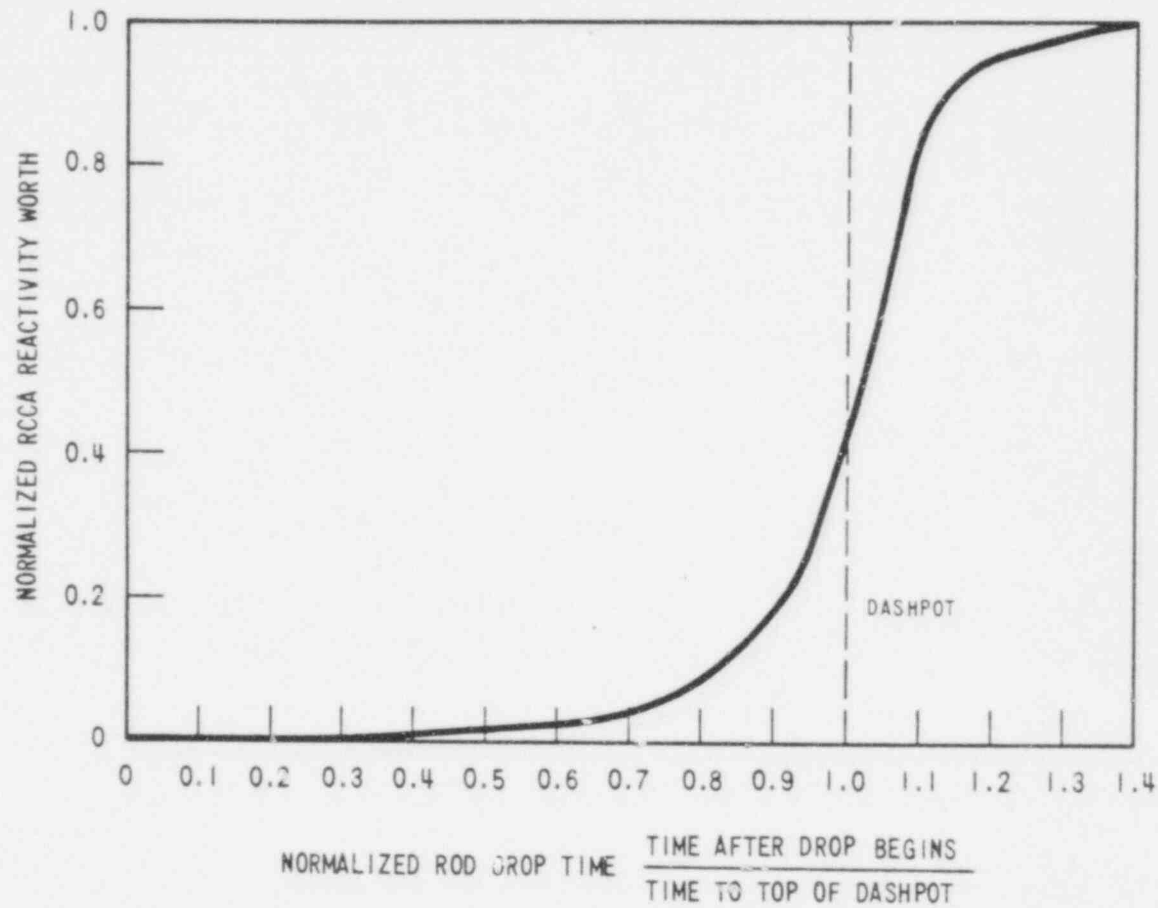


613 077

WCAP – 9500

Figure 15.0-5.

Normalized RCCA Reactivity Worth vs. Fraction Insertion



WCAP – 9500

Figure 15.0-6.

Normalized RCCA Bank Reactivity Worth vs. Normalized Drop Time

613 078

15.1

613 079

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. 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:

1. Feedwater system malfunction causing a reduction in feedwater temperature (Subsection 15.1.1).

2. Feedwater system malfunction causing an increase in feedwater flow (Subsection 15.1.2).

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

4. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the main steam system (Subsection 15.1.4).

5. Spectrum of steam system piping failures inside and outside containment (Subsection 15.1.5).

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 (Subsection 15.0.2).

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 reactor coolant system (RCS). The high neutron flux trip, low DNBR trip, and high Kw/ft trip prevent 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 heater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the reactor coolant system.

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

The net effect on the reactor coolant system due to a reduction in feedwater temperature is 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 Subsection 15.0.2).

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 Subsection 15.0.9 and listed in Table 15.0-6.

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

613 081

conditions are then used to perform 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. Simultaneous actuation of a low-pressure heater bypass and isolation of one string of low-pressure feedwater heaters.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Results

Opening of a low-pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 60°F, resulting in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load, due to opening of the low-pressure heater bypass valve, thus would result in a transient very similar (but of reduced magnitude) to that presented in Subsection 15.1.3 for an excessive increase in secondary steam flow incident, which evaluates the consequences of a 10 percent step load increase. Therefore, the results of this analysis are not presented.

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.

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15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Subsection 15.1.2), and the increase in secondary steam flow event (Subsection 15.1.3). Based on results presented in Subsections 15.1.2 and 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met. There are no radiological consequences of this event.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS CAUSING AN INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Causes and Accident Description

Additions 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. The high neutron flux trip, low DNBR trip, and high Kw/ft trip prevent 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 Reactor Coolant System due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of an excess of feedwater may cause a decrease in Reactor Coolant System 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 level trip, which closes all feedwater control and isolation valves and trips the main feedwater pumps.

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency (see Subsection 15.0.2).

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Plant systems and equipment which are available to mitigate the effects of the accident, are discussed in Subsection 15.0.9 and listed in Table 15.0-6.

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, the 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.

A control system malfunction or operator error is assumed to cause a feedwater control valve to open fully. 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 automatic control at full power.

Both of the above cases are analyzed for operation with four loops in service and for operation with three loops in service.

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

1. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 186 percent of nominal feedwater flow to one steam generator.

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2. For the feedwater control valve accident at zero load conditions, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 196 percent of the nominal full load value.

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

4. No credit is taken for the heat capacity of the Reactor Coolant System and steam generator thick metal in attenuating the resulting plant cooldown.

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

The at-power accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Normal reactor control systems and engineered safety feature systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high steam generator water level conditions. No single active failure will prevent operation of the reactor protection system. A discussion of ATWT considerations is presented in Reference [2].

Results

The calculated sequence of events for this accident are shown in Table 15.1-1.

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

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maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 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 conditions, 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 (with rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the manual control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event.

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

Transient results, (see Figures 15.1-1 and 15.1-2), show the core heat flux, pressurizer pressure, Tave and DNBR as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. The DNBR does not drop below the limit value. Figures 15.1-1a and 15.1-2a show the transient results with three loops in operation.

Following reactor trip and feedwater isolation, the plant will approach a stablized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control Reactor Coolant System 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 stablized condition will be in a time frame in excess of ten minutes following reactor trip.

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 normal value (i.e., the assumed high neutron flux trip point). 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.

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 DNBR's encountered for an excessive feedwater addition at power are at all times above the limit value; hence, the DNB design basis as described in Section 4.4 is met. 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 steam line 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 or a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5 (there are no pressure regulators whose malfunction could cause a steam flow transient).

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

1. Low DNBR.
2. High Kw/ft.
3. Power range high neutron flux.
4. Low pressurizer pressure.

An excessive load increase incident is considered to be an ANS Condition II event, fault of moderate frequency (See Subsection 15.0.2).

15.1.3.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using the LOFTRAN Code (Reference 1). The code simulates 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.

The above four cases are also analyzed for a 10 percent step load increase from 70 percent power, with three reactor coolant loops in service.

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. For all the cases, the least negative Doppler-only power coefficient curve of Figure 15.0-2 was used.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation. This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Uncertainties in initial conditions of reactor power, pressure, and reactor coolant system temperature are included in the limit DNBR as described in the WCAP.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

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

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

Results

The calculated sequence of events for the excessive load increase incident is shown on Table the limit value.

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

Figures 15.1-7 through 15.1-10 illustrate the transient assuming the reactor is in the automatic control mode. Both the minimum and maximum moderator feedback cases show that core power increases, thereby increasing the coolant average temperature and pressurizer pressure above their initial value. For both of these cases, the minimum DNBR remains above the limit value.

Figures 15.1-3a through 15.1-10a show the cases described above, but considering three reactor coolant loops in operation.

For all cases, the plant rapidly reaches a stablilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power. If the reactor trips, operating procedures would call for operator action to control Reactor Coolant System 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 stablized condition will be in a time frame in excess of ten minutes following reactor trip.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for most 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.

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 10 percent step load increase, the DNBR remains above the limit value; thus the DNB design basis as described in Section 4.4 is met. The plant reaches a stabilized condition rapidly following the load increase.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE CAUSING A DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

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 result from 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 Subsection 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 reactor coolant system causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied: assuming a stuck rod cluster control assembly, with offsite power available, and assuming a single failure in the engineered safety features there will be no consequential damage to the core or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event (See Subsection 15.0.2).

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

1. Safety injection actuation from any of the following:

   a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

   b. Low pressurizer pressure

   c. High-1 containment pressure

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2. A reactor trip from 1) DNB protection (low DNBR or high neutron flux), 2) low pressurizer pressure, or 3) 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 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. Excessive cooldown protection (low T_{cold} or low steamline pressure)

   b. Low pressurizer pressure

   c. High negative steam pressure rate in any loop

   d. High-2 containment pressure

Systems and equipment which are available to mitigate the effects of the accident are also discussed in Subsection 15.0.9 and listed in Table 15.0-6.

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 Reactor Coolant System temperature and pressure, during cooldown, and the effect of safety injection.

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2. Analyses to determine that there is no damage to the core or reactor coolant system.

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 15.1-11.

3. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the safety injection system. This corresponds to the flow delivered by one safety injection pump delivering its full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downsteam of the refueling water storage tank prior to the delivery of concentrated boric acid (2000 ppm) to the reactor coolant loops. This effect has been allowed for in the analysis.

4. The case studied is a steam flow of 269 lb/sec at 1200 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition. Cases analyzed in Subsection 15.1.3, excessive increase in secondary steam flow, bound

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a failure of a steam generator steam dump, safety, or relief valve from full power.

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

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

7. Cases are shown for four loops in operation and three loops in operation.

Results

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

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 15.1-13 and 15.1-14 show the transient results for a steam flow of 269 lb/sec at 1200 psia from one steam generator.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve.

Safety injection is initiated automatically by low pressurizer pressure. Operation of one SI pump is assumed. Boron solution at 2000 ppm enters the reactor coolant system providing sufficient negative reactivity to prevent core damage. The cooldown for the case shown in Figures 15.1-13 and 15.1-14 is more rapid than the case of steam release from all steam generators through one steam dump, relief, or safety valve. The calculated 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 will have a significant effect in slowing the cooldown.

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Figures 15.1-13a and 15.1-14a show the same parameters as Figures 15.1-13 and 15.1-14 only for the case with one loop out of service. The steam leak is assumed to occur on one of the loops which is in service. Safety injection is initiated automatically from a low pressurizer pressure safety injection signal.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit Reactor Coolant System pressure and pressurizer level by terminating safety injection flow and to control steam generator level and reactor coolant system coolant temperature using the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stablized condition will be in a time frame in excess of ten minutes following safety injection actuation.

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. For an accidental depressurization of the main steam system, the minimum DNBR remains well above the limiting value and no system design limits are exceeded. (The radiological consequences of this event are found on a plant specific basis).

15.1.5 SPECTRUM OF STEAM SYSTEM PIPING FAILURE INSIDE AND OUTSIDE CONTAINMENT

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 decreases. The energy removal from the reactor coolant system 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. The core is ultimately shut down by the boric acid delivered by the safety injection system.

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

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

2. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is met as stated in Section 4.4 for any rupture assuming the most reactive RCCA assembly stuck in its fully withdrawn position.

A major steamline rupture is classified as an ANS Condition IV event (See Section 15.0.2).

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

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

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

    a. Excessive cooldown protection (low T_{cold} or low steamline pressure)

    b. Low pressurizer pressure.

    c. High-1 containment pressure.

2. A reactor trip from 1) DNB protection (low DNBR or high neutron flux), 2) high linear heat flux, 3) low pressurizer pressure, or 4) 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 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. Excessive cooldown protection (low T_{cold} or low steamline pressure)

b. Low pressurizer pressure

c. High negative steam pressure rate in any loop

d. High-2 containment pressure.

Fast-acting isolation valves are provided in each steamline that will fully close within 5 seconds of actuation following a steamline isolation signal from the integrated protection system. An additional delay of 2.0 seconds is included for sensor and protection system delays. For breaks downsteam 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 10.0.

Table 15.1-2 lists the equipment required in the recovery from a high energy steamline rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details criteria. 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 Reactor Coolant System 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 analysis has been performed with four reactor coolant loops in operation and with three loops in operation (N-1 loop).

The following conditions were assumed to exist at the time of a main steamline 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 average coolant temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1-11. (The effect of power generation in the core on overall reactivity is shown in Figure 15.1-15).

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors 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 conditions 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 nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the conditions. These results verify conservatism: underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of high concentration boric acid (2000 ppm) solution corresponding to the most restrictive single failure in the safety injection portion of the emergency core cooling system (ECCS). The ECCS consists of three systems: 1) the passive accumulators, 2) the residual heat removal (low head safety injection system), and 3) the high head safety injection system. Only the high head system is modeled for the steamline break accident analysis.

   The modeling of the SI system in LOFTRAN is described in Reference [1]. The flow corresponds to that delivered by one SI 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 downsteam of the refueling water storage tank prior to the delivery of concentrated boric acid to the reactor coolant loops.

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

   The boric acid solution from the safety injection system is assumed to be uniformly delivered to the four reactor coolant loops. The boron in the loops is then delivered to the inlet plenum where the coolant (and boron) from each loop is mixed and delivered to the core. The stuck RCCA is conservatively assumed to be located in the core sector near the broken steam generator. Because the cold leg pressure is lowest in the broken loop due to larger loop flow and a larger loop pressure drop, more boron would actually be delivered to the core sector where the power is being generated, enhancing the effect of the boric acid on the transient. No credit was taken for this in the analysis. Furthermore, sensitivity studies have demonstrated that the transient is insensitive to boron worth or distribution.

For the cases where offsite power is assumed, the sequence of events in the SI 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 SI pump starts. In 12 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into the core before the 2000 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, a 10 second delay to start the standby diesel generators in addition to the time necessary to start the safety injection equipment (mentioned above) is included.

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 ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.4 ft^2 break. The following cases have been considered in determining the core power and reactor coolant system transients:

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a. Complete severance of the 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.

c. Case (a) and (b) above with one reactor coolant loop out of service.

6. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck 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 RCCA 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 Reactor Coolant System 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

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reactor coolant system 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.

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

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

9. Feedwater addition aggravates cooldown accidents like the steamline rupture. Therefore, the maximum feedwater flow is assumed. All the main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs, even though the plant is assumed to be in a hot standby condition. Full main and auxiliary feedwater flow is maintained for five seconds following the receipt of a feedwater isolation signal from the integrated protection system following safety injection actuation. An additional 2.0 second delay is added for sensor and protection system delays. During the first 5 seconds following the start of the transient, a feedwater isolation signal is generated (to close both the feedwater control and the feedwater isolation valves) to be sent to redundant valves with 5 second closure time. All the auxiliary feedwater is assumed to be pumped into the depressurizing steam generator.

10. The balance of plant assumptions used in the analysis are listed in Table 15.1-2a.

11. The effect of heat transferred from thick metal in the pressurizer and reactor vessel upper head is not included in the cases analyzed. Studies previously performed have shown that the heat transferred to the coolant from these latent sources is a net benefit in DNB and Reactor Coolant System energy when the effect of the extra heat on reactivity and peak power is considered.

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Results

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

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 15.1-16 through 15.1-18 show the reactor coolant system transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial no-load conditions (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 high containment pressure signals or by low steamline pressure 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 15.1-18, the core attains criticality with the RCCA's inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2000 ppm enters the reactor coolant system. A peak core power less than the nominal full power value is attained.

Figures 15.1-19 through 15.1-21 show the response of 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 standby diesel generator and 12 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 Reactor Coolant System is reduced by the decreased flow in the Reactor Coolant System. The power transient shown in Figure 15.1-19 is conservatively due to the underprediction of the feedback in the low flow condition. For the DNBR evaluation, a power and power shape analysis consistent with the fluid conditions was used.

Figures 15.1-16a through 15.1-18a show relevent parameters for case (a) assuming one reactor coolant loop is out of service when the steamline rupture occurs. Offsite power is available throughout the transient. As in case (a) with all loops in operation, steam is released from only one steam generator due to tripping of the fast acting steamline isolation valves by high containment pressure signals or low steamline pressure. The core attains criticality with the rod cluster control assemblies inserted (with the design shutdown assuming one stuck assembly) before boron solution at 2000 ppm enters the Reactor Coolant System from the safety injection system. The actuation and transport delays are taken into account as in case (a) with all loops in operation.

Figures 15.1-19a through 15.1-21a show time varying parameters for case (b) assuming one reactor coolant loop is out of service when the steamline rupture occurs with the loss of offsite electrical power at the time the safety injection signal is generated. The safeguards actuation and boron transport delays are taken into account as in case (b) with all loops in operation.

For all of these cases 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.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of the auxiliary feedwater flow and safety injection flow as described by plant operating procedures. The operating procedures would call for operator action to limit Reactor Coolant System pressure and pressurizer level by terminating safety injection flow and to control steam generator level and Reactor Coolant System coolant temperature using the 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 safety injection actuation.

The ability of the intact steam generators to remove residual energy from the Reactor Coolant System in the long term is demonstrated by the major rupture of a main feedwater line. The steamline break is less limiting with respect to cooldown without offsite power because temperatures are much lower, all of the auxiliary feedwater can be delivered to the steam generators, and the steam blowdown leaves a higher water inventory than the feedline blowdown. The feedline rupture demonstrates that the intact steam generators and auxiliary feedwater provide sufficient heat sink to remove long term heat following the transient.

Margin to Critical Heat Flux

A DNB analysis was performed for all of these cases. It was found that the DNB design basis as stated in Section 4.4 was met for all cases.

The maximum linear heat rate for the most limiting case remains less than 14 Kw/ft, which is less than the linear heat rate which results in fuel melting. There is no known fuel failure mechanism associated with this peak linear heat rate. Although no fuel failure mechanism has been identified, a very conservative assumption of 1 percent fuel failures has been recommended in Table 15.1-3 for use in the environmental consequences analysis.

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 15.1-3. 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:

1. 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.

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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 15.1-3.

15.1.6 REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

613 109

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

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

613 110

TABLE 15.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE
AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N LOOP	TIME (sec) N-1 LOOP
Excessive feedwater flow at full load	One main feedwater control valve fails fully open	0.0	0.0
	High steam generator water level signal generated	40.0	70.0
	Turbine trip occurs due to high steam generator level	43.0	73.0
	Minimum DNBR occurs	21.2	70.6
	Reactor trip occurs	45.0	75.0
	Feedwater isolation valves close	47.0	77.0
Excessive Increase in Secondary Steam Flow			
1. Manual Reactor Control (Minimum moderator feedback)	10% step load increase	0.0	0.0
	Equilibrium conditions reached (approximate time only)	150	200

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TABLE 15.1-1 (Continued)

ACCIDENT	EVENT	TIME (sec) N LOOP	TIME (sec) N-1 LOOP
2. Manual Reactor Control (Maximum moderator feedback)	10% step load increase	0.0	0.0
	Equilibrium conditions reached (approximate time only)	70	70
3. Automatic Reactor Control (Minimum moderator feedback)	10% step load increase	0.0	0.0
	Equilibrium conditions reached (approximate time only)	200	200
4. Automatic Reactor Control (Maximum moderator feedback)	10% step load increase	0.0	0.0
	Equilibrium conditions reached (approximate time only)	70	70
Accidental depressurization of the main steam system	Inadvertent opening of one main steam safety or relief valve	0.0	0.0

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TABLE 15.1-1 (Continued)

ACCIDENT	EVENT	TIME (sec) N LOOP	TIME (sec) N-1 LOOP
	Pressurizer empties	164.5	153.0
	2000 ppm boron reaches core	245.0	233.0
Steam System Piping Failure			
1. Case a (Plant initially at no load with offsite power)	Steamline ruptures	0.0	0.0
	Pressurizer empty	13.0	16.0
	Criticality attained	17.5	19.0
	2000 ppm boron reaches core	49.0	55.0
2. Case b (Same as Case a Except for loss of off-site power)	Steamline ruptures	0.0	0.0
	Pressurizer empty	14.5	18.0
	Criticality attained	20.0	21.5
	2000 ppm boron reaches core	54.0	63.0

TABLE 15.1-2

EQUIPMENT REQURED FOLLOWING A RUPTURE OF A MAIN STEAMLINE

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 low DNBR, high Kw/ft, and low reactor coolant pump speed 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). Controls for defeating automatic safety injection actuation during a cooldown and depressurization.
Safety injection system including the pumps, the refueling water storage 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.

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TABLE 15.1-2 (Continued)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Standby diesel generators and Class IE power distribution equipment.		
Service water and reactor plant component cooling water system.		
Containment spray system equipment.		
Auxiliary feedwater system including pumps, water supplies, piping, valves.		
Pressurizer and main steam safety valves.		

TABLE 15.1-2 (Continued)

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).		
Bypass feedwater control valves (trip closed feature).		
Main steamline stop valves (trip closed feature).		
Main steamline stop valve bypass valves (trip closed feature).		

613 116

TABLE 15.1-2 (Continued)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Steam generator blowdown isolation valves (automatic closure feature).		
Batteries (Class IE).		
Control room air conditioning.		
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.		

613 117

TABLE 15.1-2 (Continued)

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Post-Accident Monitoring System.		
Containment Atmosphere Recirculation System.		
ESFA and SI cubicle unit coolers.		

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TABLE 15.1-2a

BALANCE OF PLANT ASSUMPTIONS USED IN MAJOR RUPTURE OF A MAIN STEAM LINE

Item	Value Used in Analysis
Steam Line Stop Valve Closure Time	5.0 sec
Feedwater Isolation Valve Closure Time	5.0 sec

Table 10.1-1 gives the interface requirements for steam line and feedwater isolation valves.

613 119

TABLE 15.1-3

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

613 120

TABLE 15.1-3 (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.

613 121

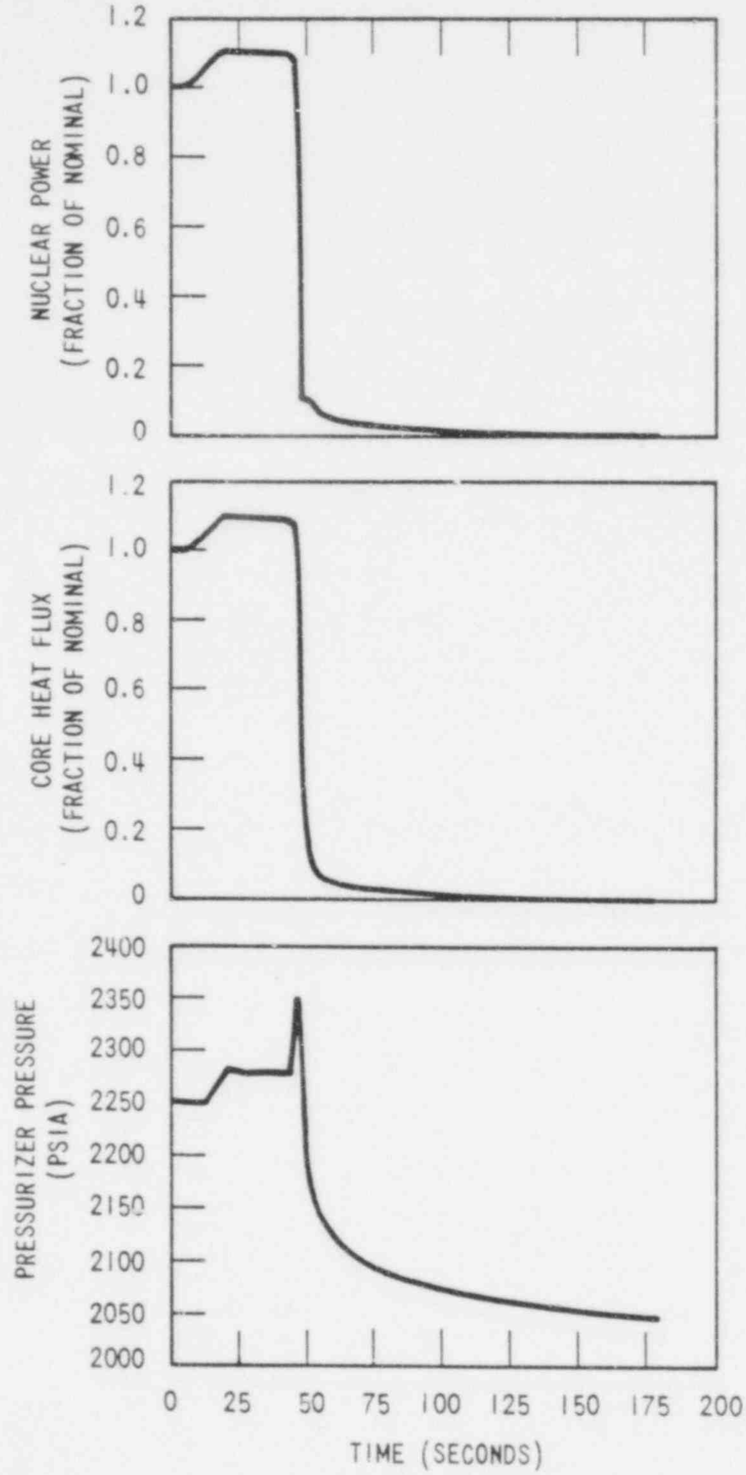

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Figure 15.1-1.

Transients for Feedwater Control Valve Malfunction

613 122

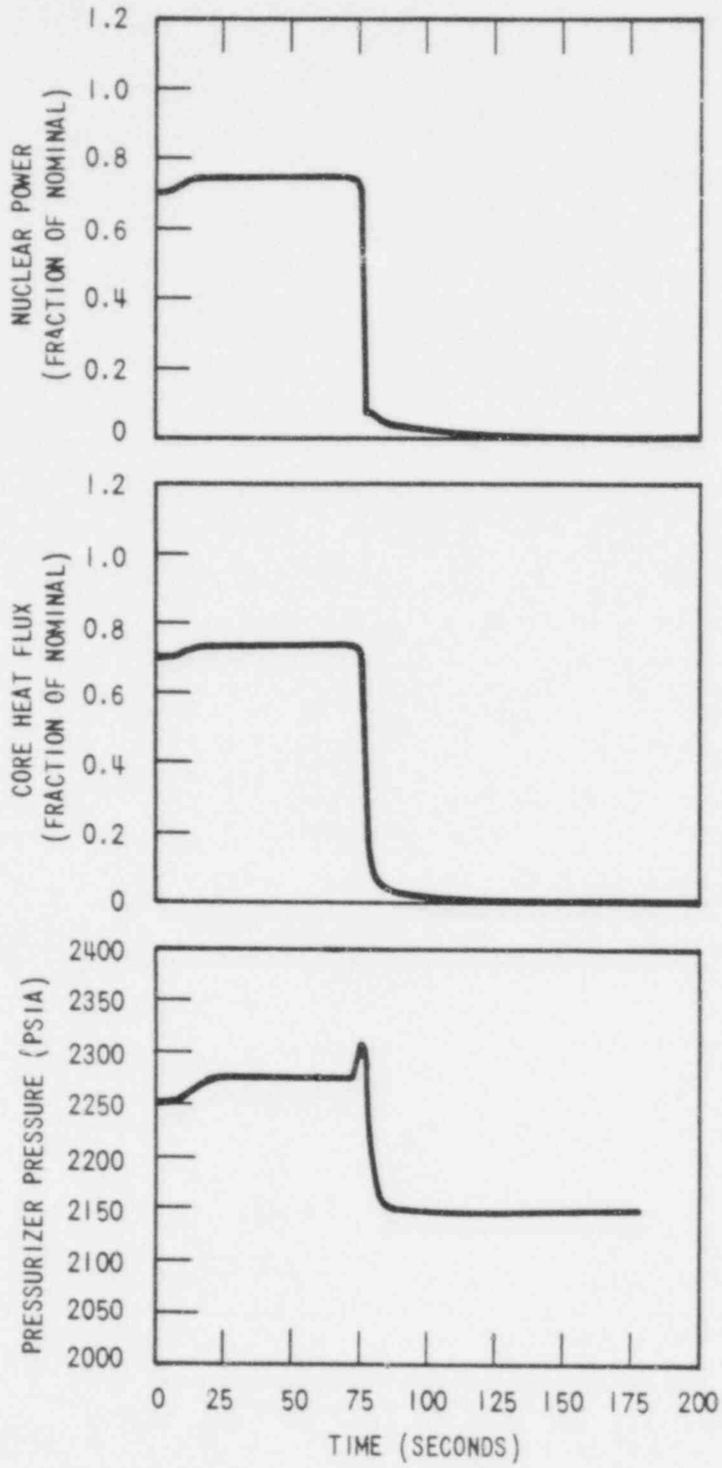

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Figure 15.1-1a.

Transients for Feedwater Control Valve Malfunction (N-1 Loop Operation)

613 123

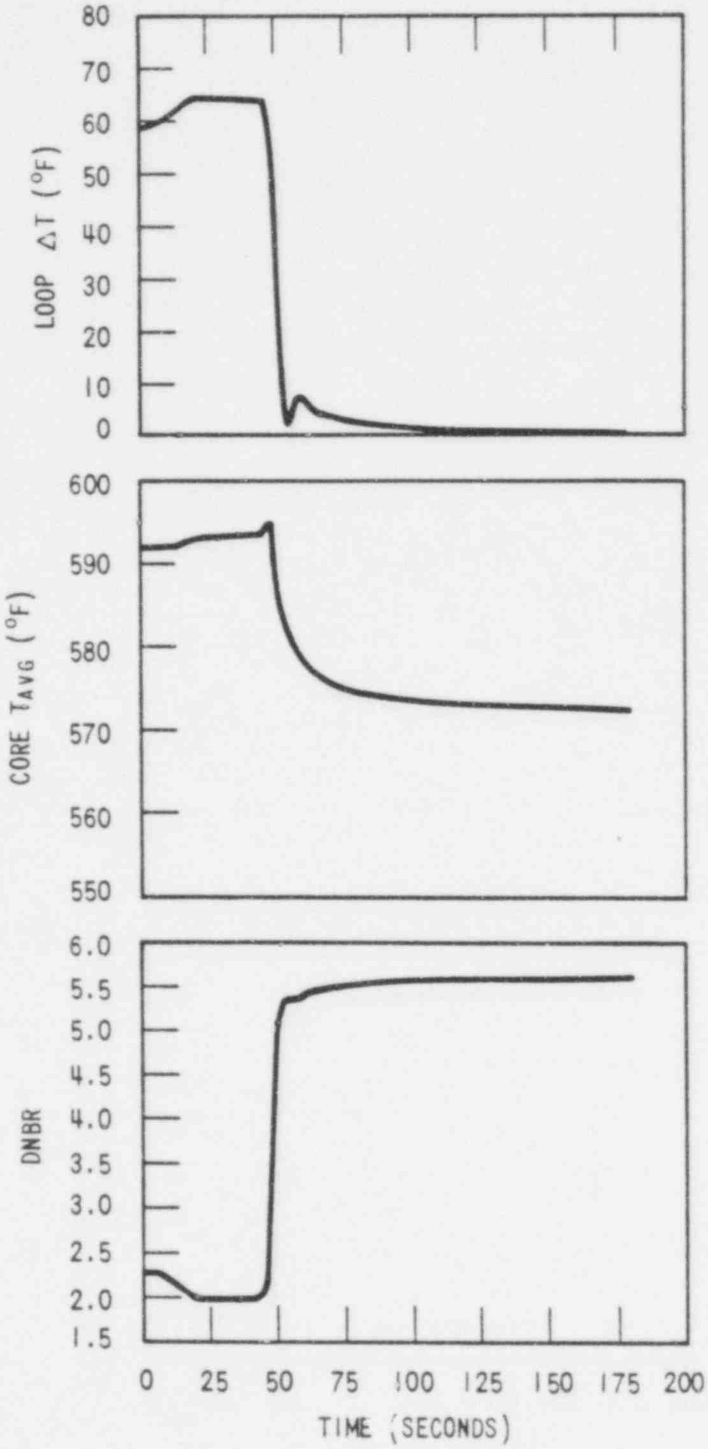

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Figure 15.1-2.

Transients for Feedwater Control Valve Malfunction

613 124

14,395-186

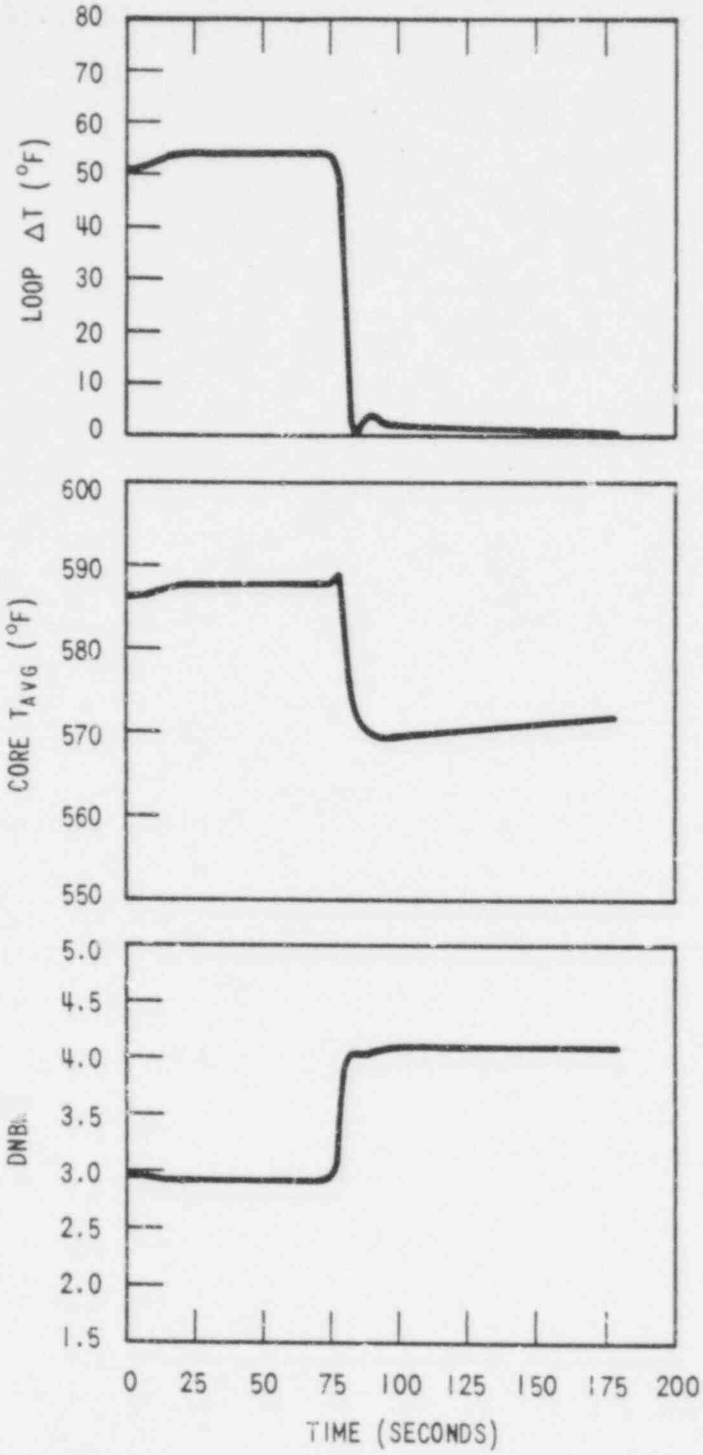

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Figure 15.1-2a.

Transients for Feedwater Control Valve Malfunction (N-1 Loop Operation)

613 125

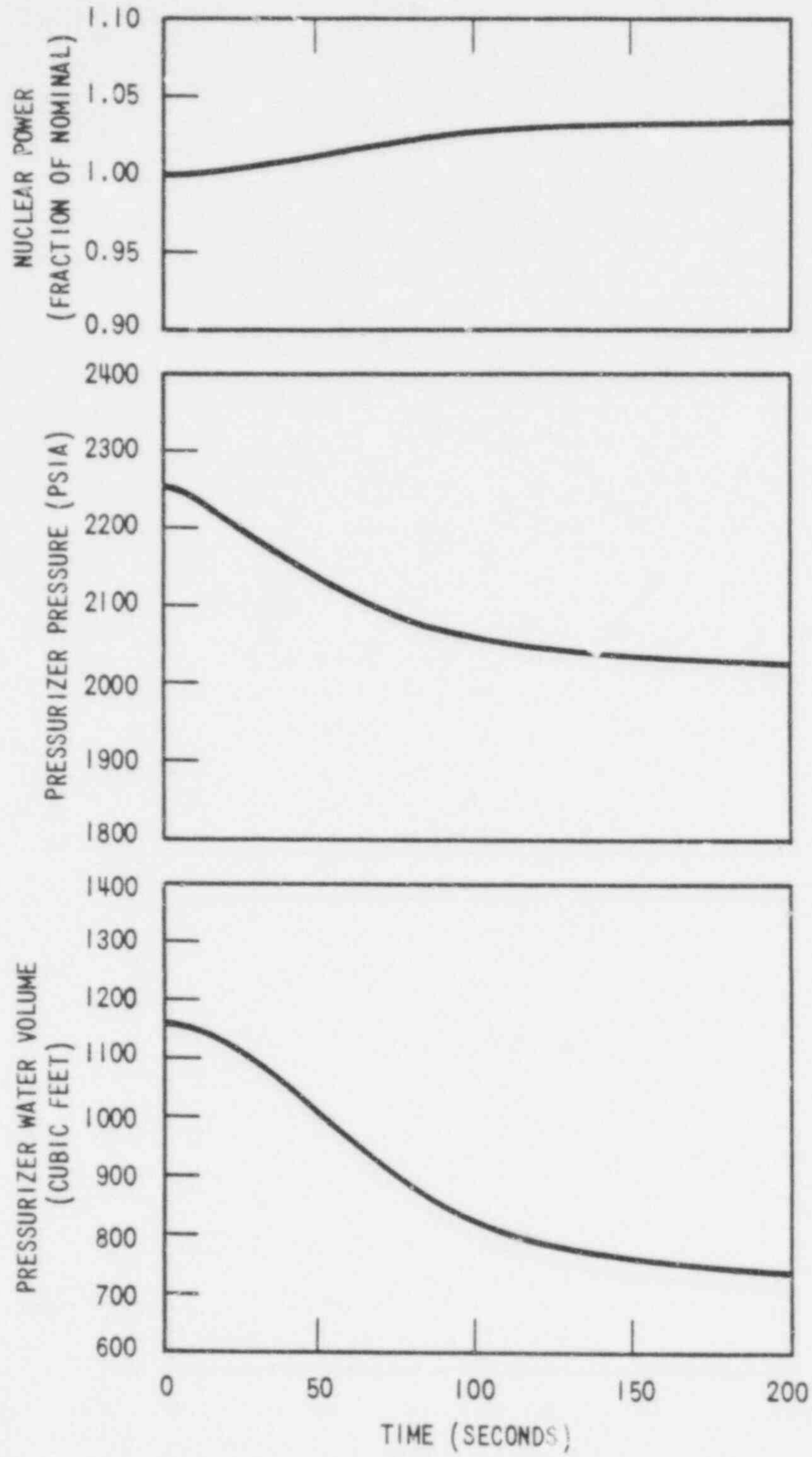

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Figure 15.1-3.

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

613 126

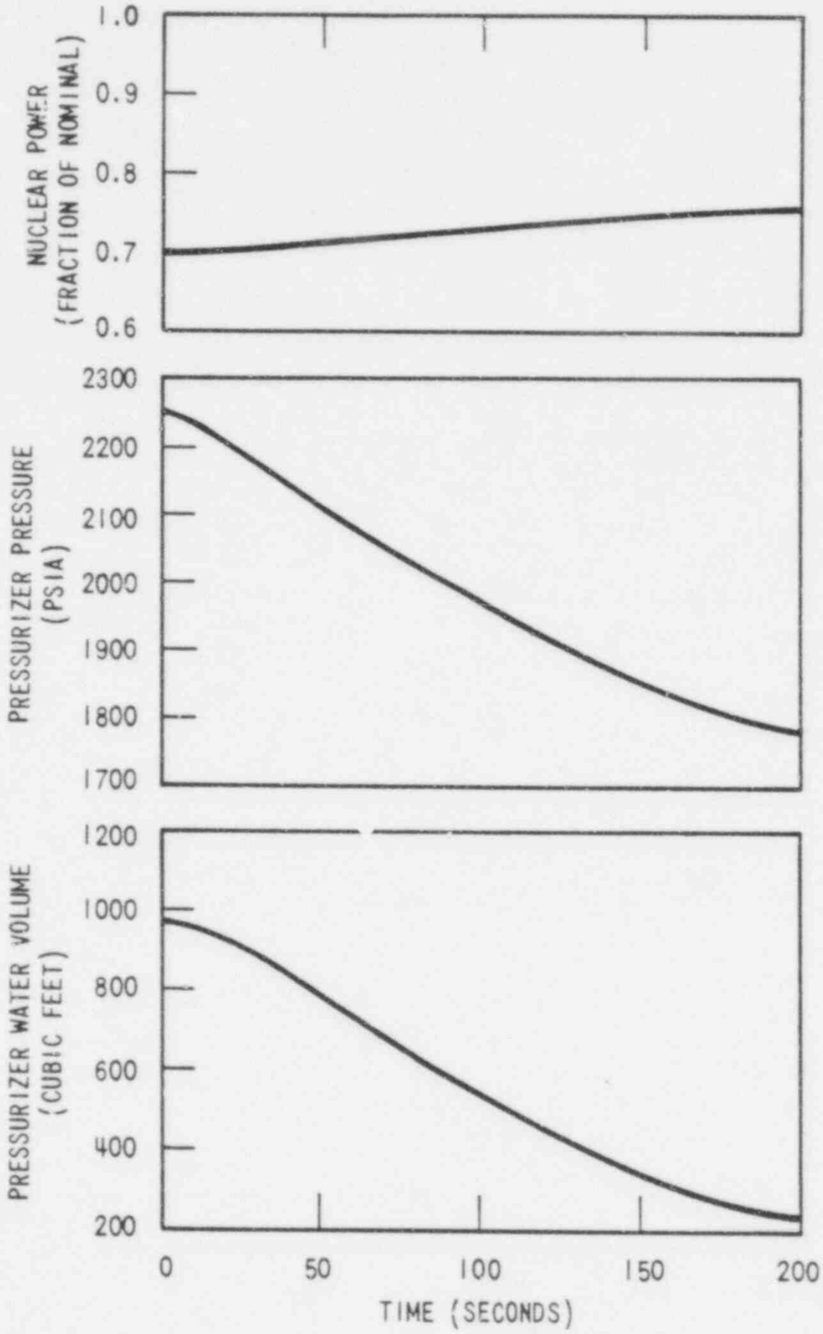

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Figure 15.1-3a.

Ten Percent Step Load Increase, Minimum Moderator Feedback, Manual Reactor Control (N-1 Loop Operation)

613 127

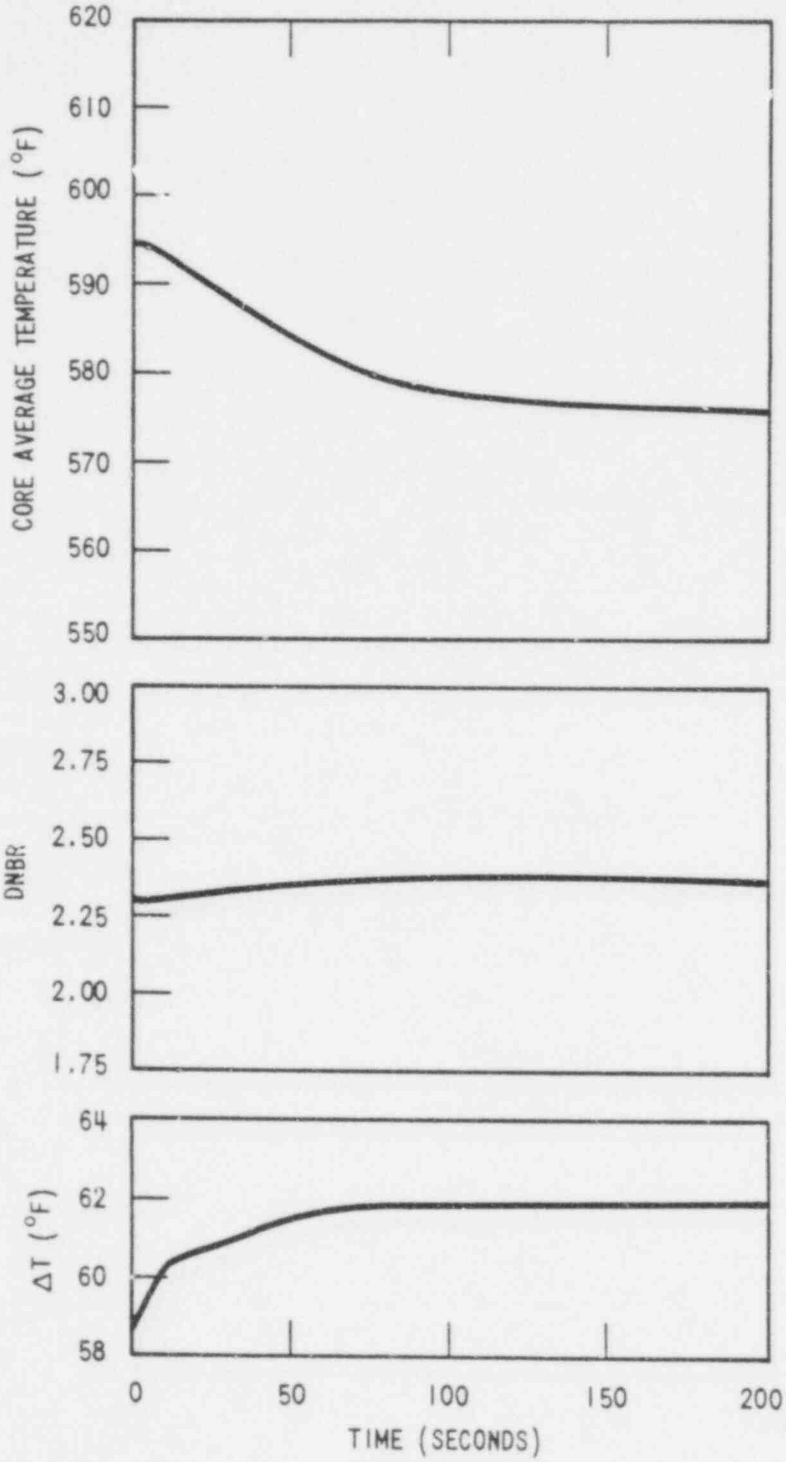

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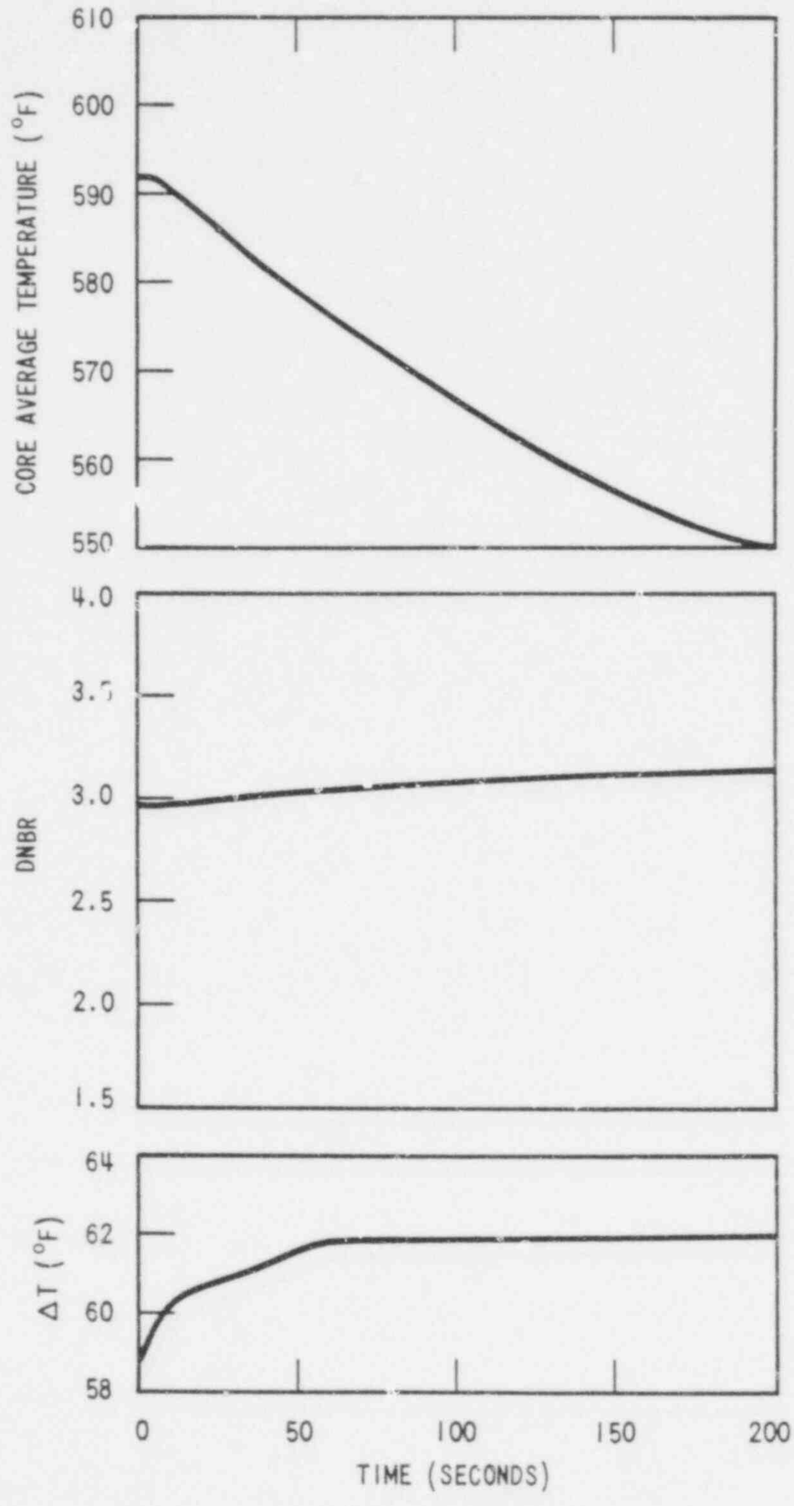
Figure 15.1-4.

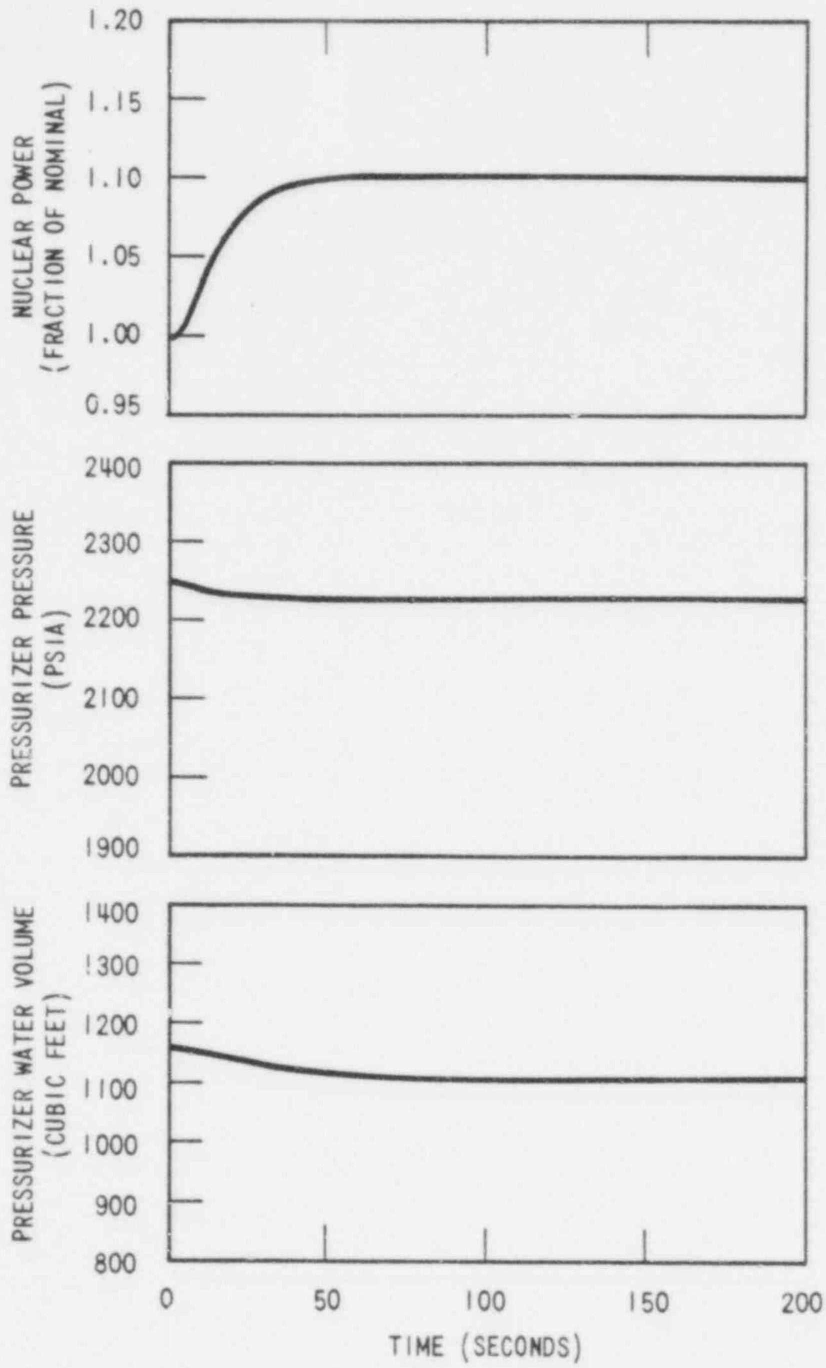
Ten Percent Step Load Increase, Minimum Moderator Feedback, Manual Reactor Control

613 128



WCAP-9500
Figure 15.1-4a. Ten Percent Step Load Increase, Minimum Moderator Feedback, Manual Reactor Control (N-1 Loop Operation)

14,395-189



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WCAP – 9500

Figure 15.1-5.

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

14,395-197

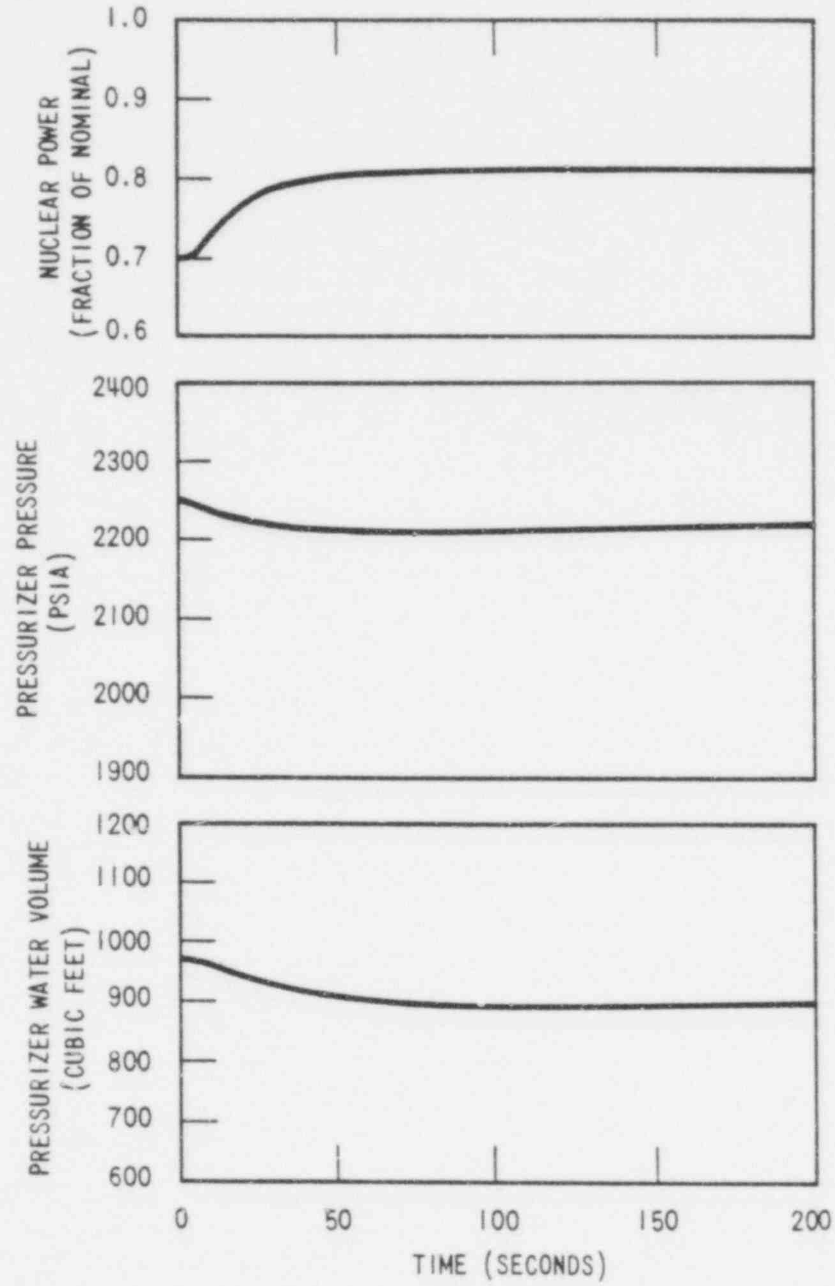

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Figure 15.1-5a.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Manual Reactor Control (N-1 Loop Operation)

613 131

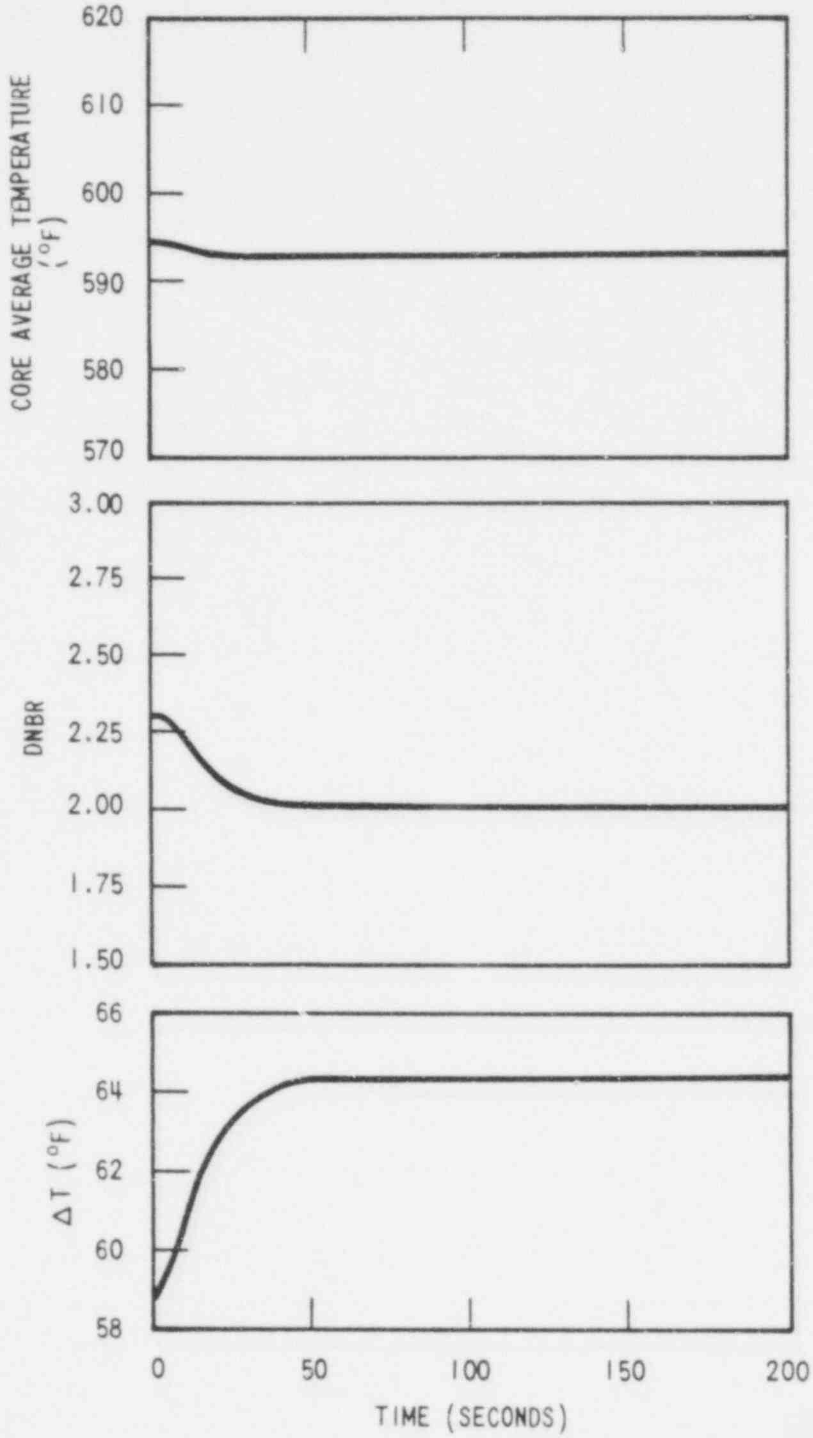

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Figure 15.1-6.

Ten Percent Step Lcad Increase, Maximum Moderator Feedback, Manual Reactor Control

813 152

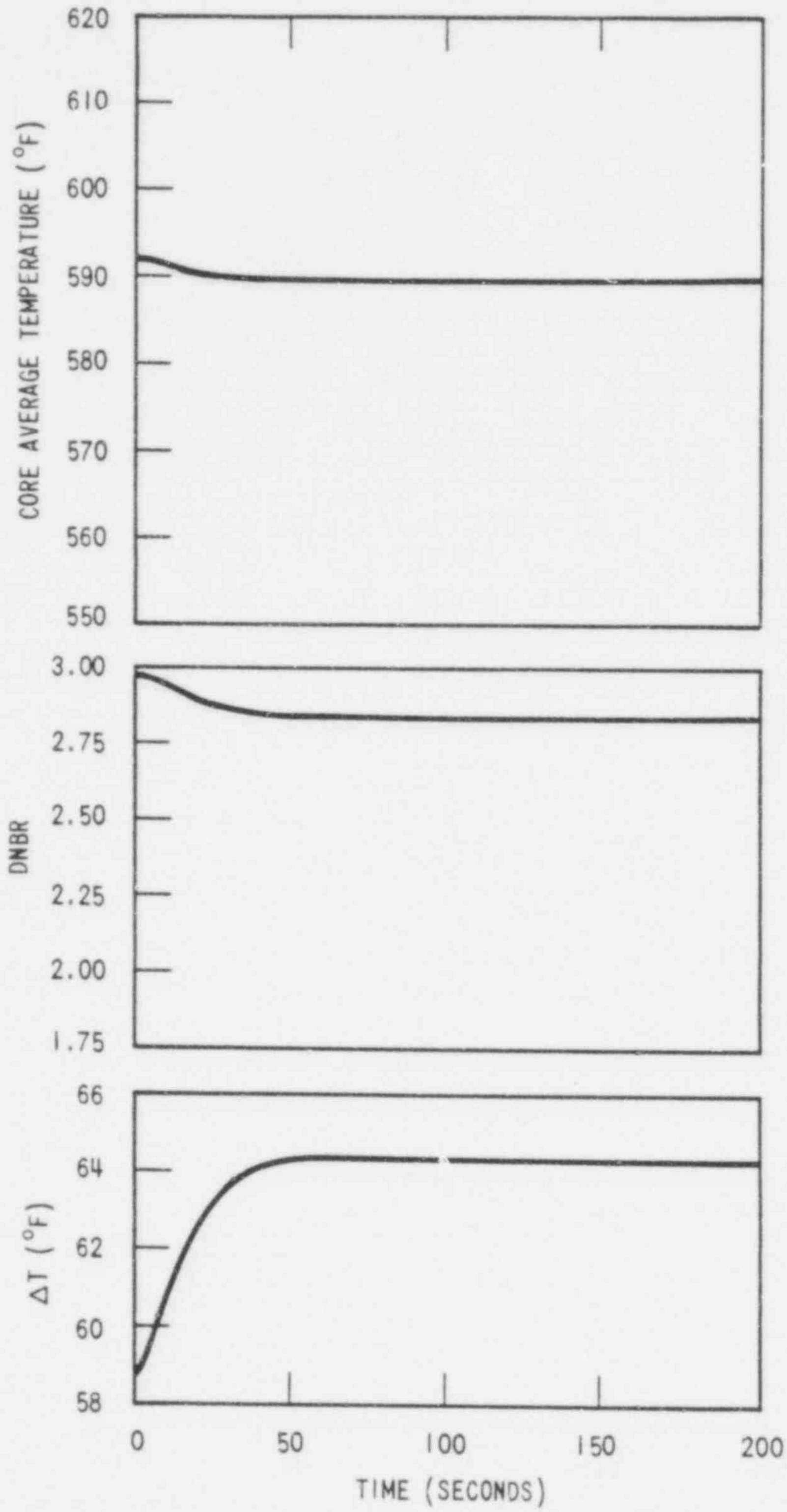

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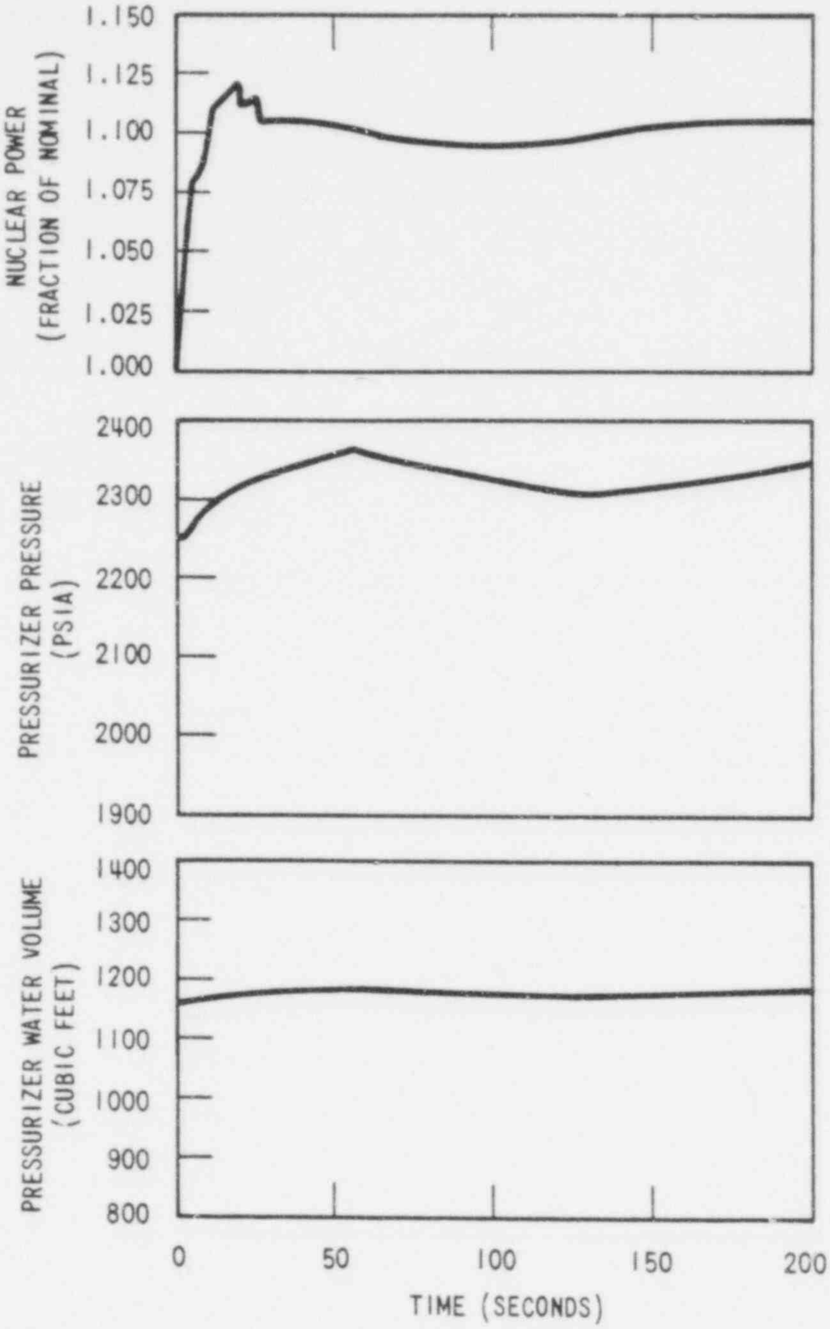
Figure 15.1-6a.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Manual Reactor Control (N-1 Loop Operation)

613 133



WCAP – 9500

Figure 15.1-7.

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

613 134

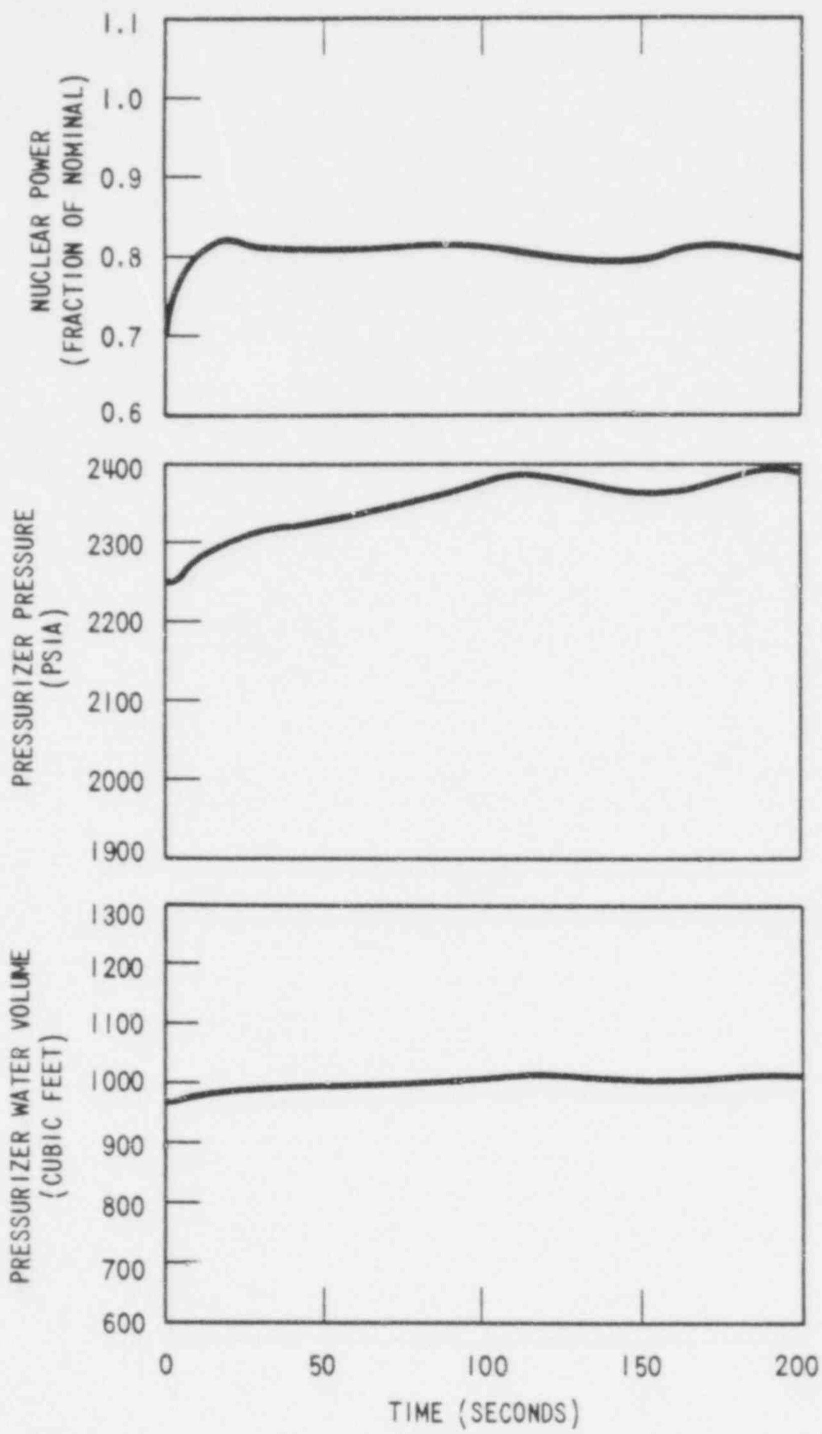


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Figure 15.1-7a.

Ten Percent Step Load Increase, Minimum Moderator Feedback, Automatic Reactor Control (N-1 Loop Operation)

613 135

14,395-192

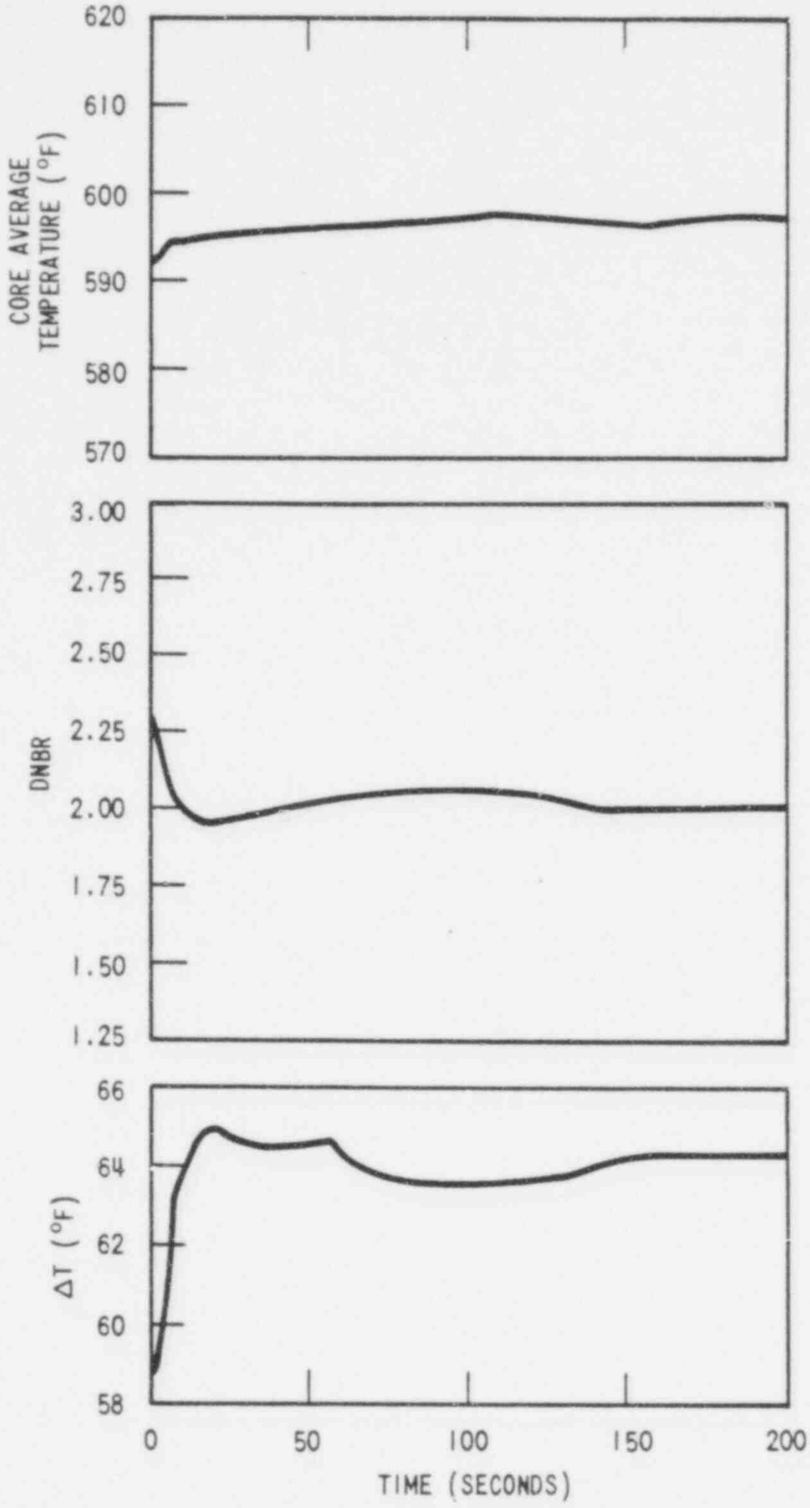

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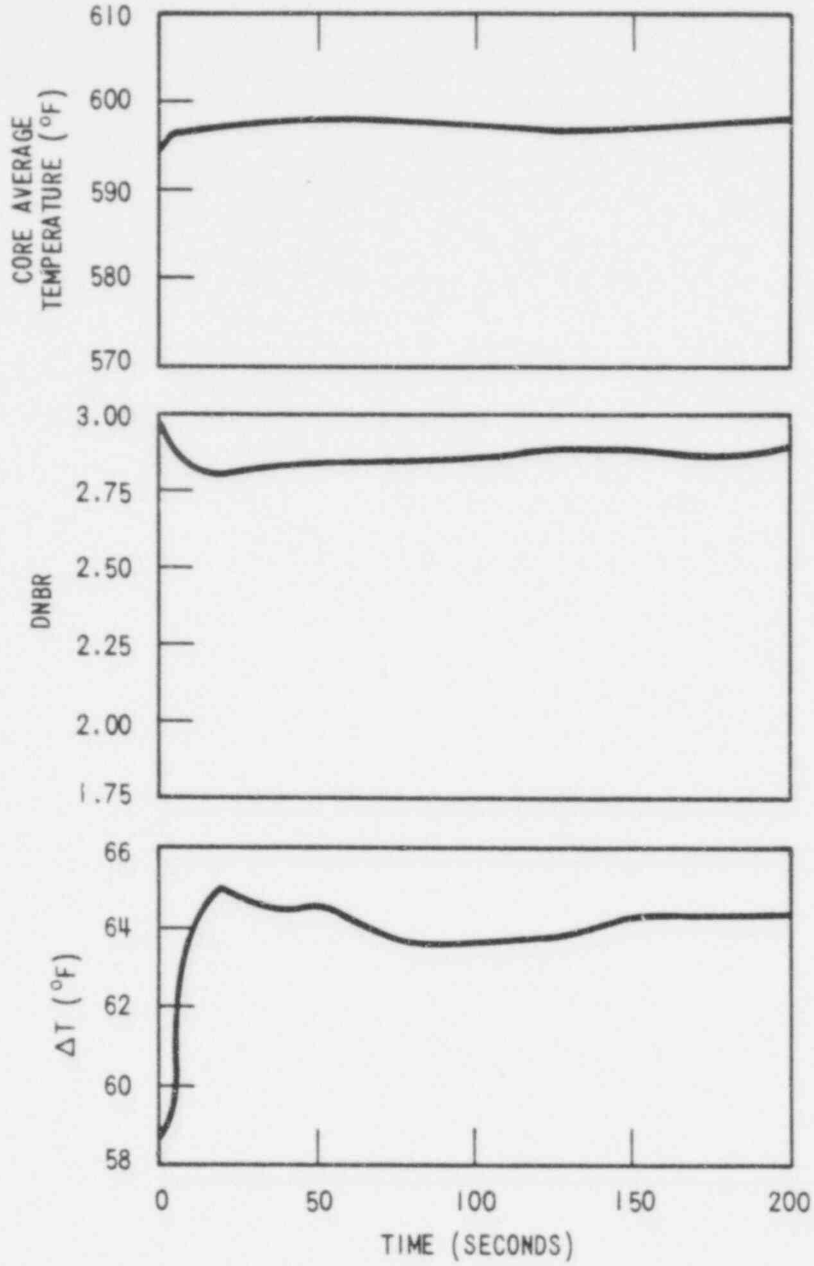
Figure 15.1-8.

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

613 136

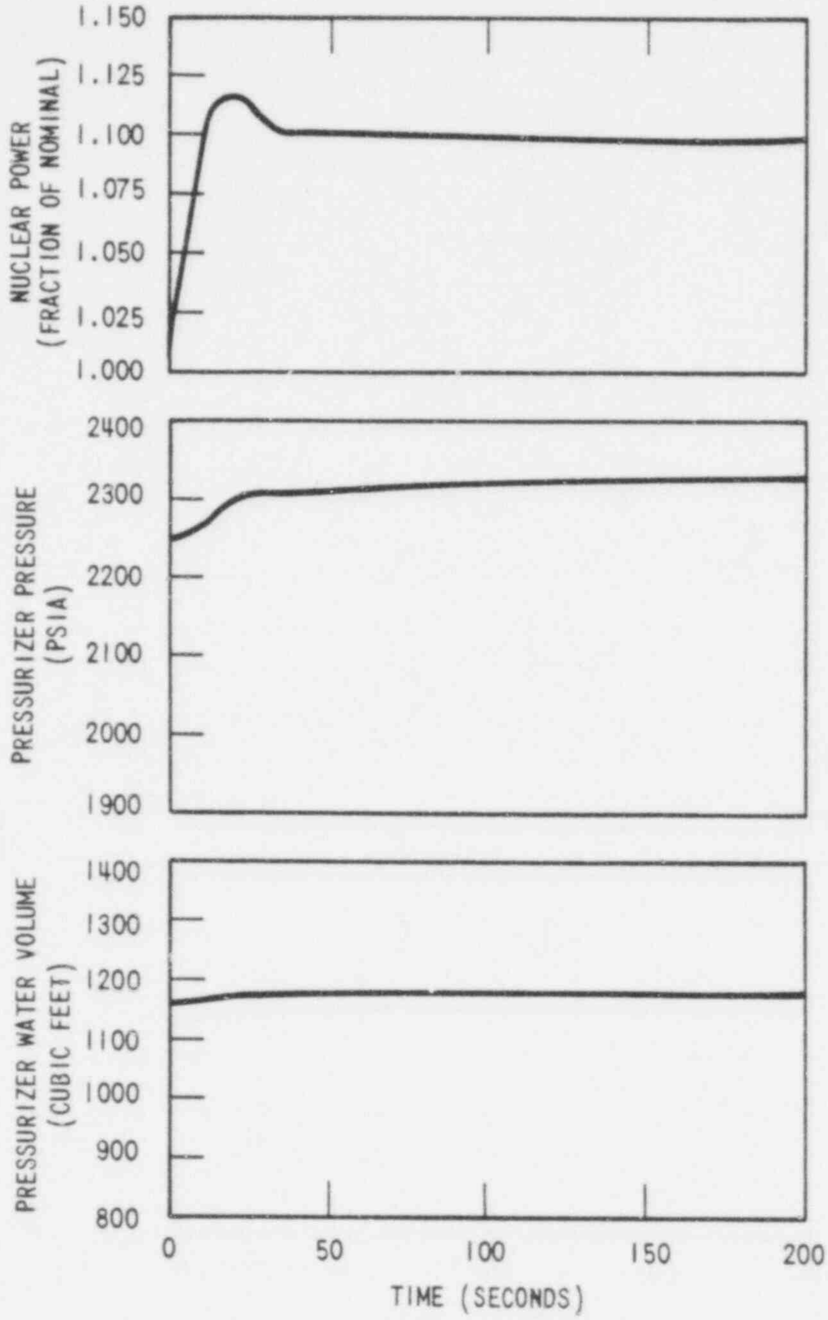


WCAP-9500

Figure 15.1-8a.

Ten Percent Step Load Increase, Minimum Moderator Feedback, Automatic Reactor Control (N-1 Loop Operation)

613 137

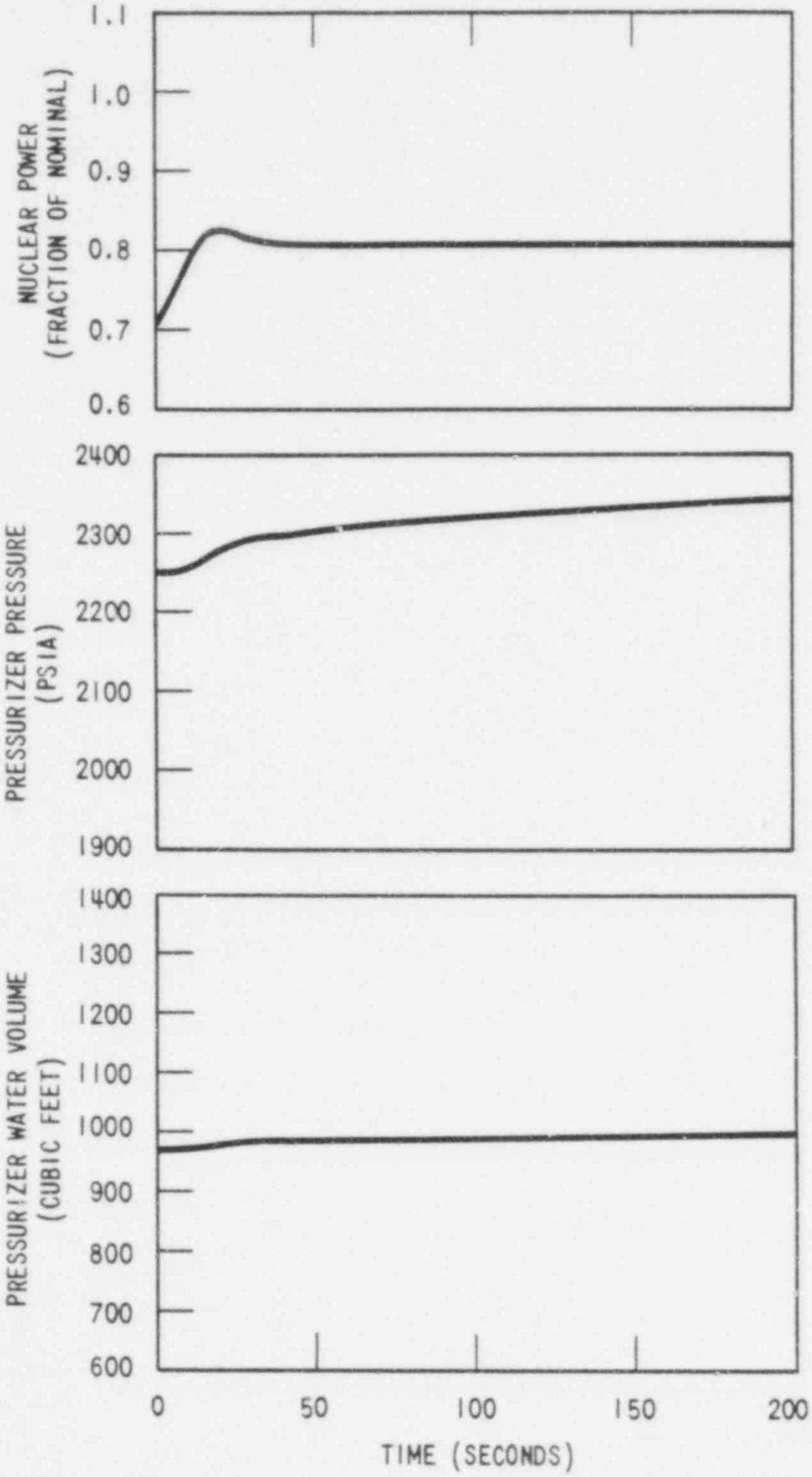


WCAP - 9500

Figure 15.1-9.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Automatic Rod Control

613 138



WCAP – 9500

Figure 15.1-9a.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Automatic Rod Control (N-1 Loop Operation)

613 139

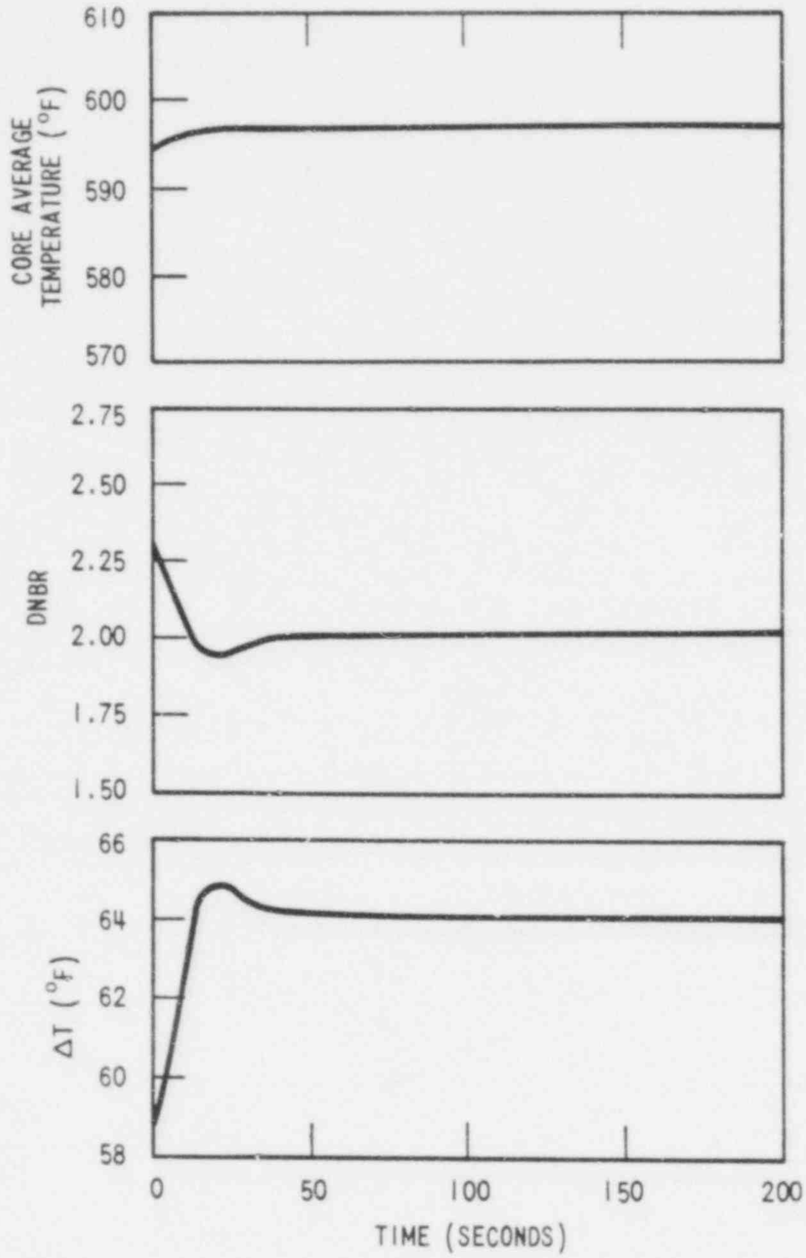


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Figure 15.1-10.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Automatic Rod Control

613 140

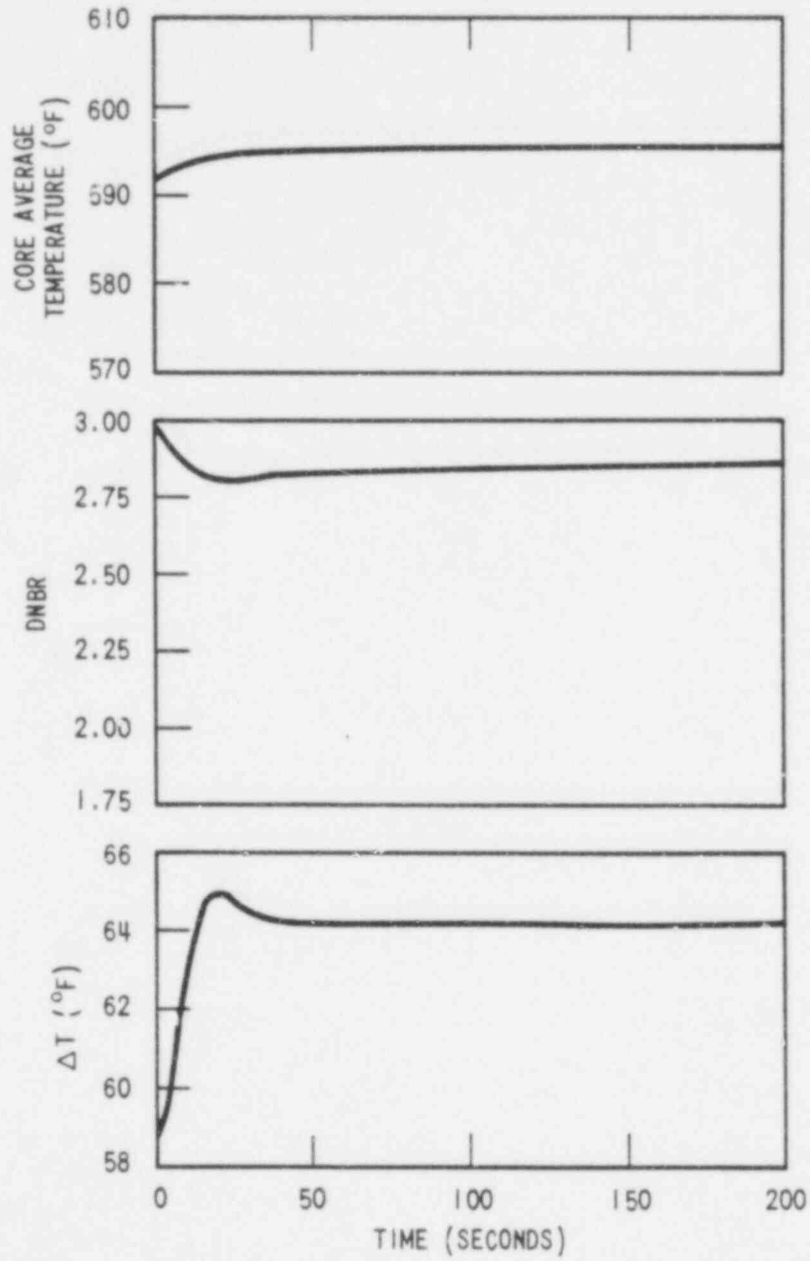

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Figure 15.1-10a.

Ten Percent Step Load Increase, Maximum Moderator Feedback, Automatic Rod Control (N-1 Loop Operation)

613 141

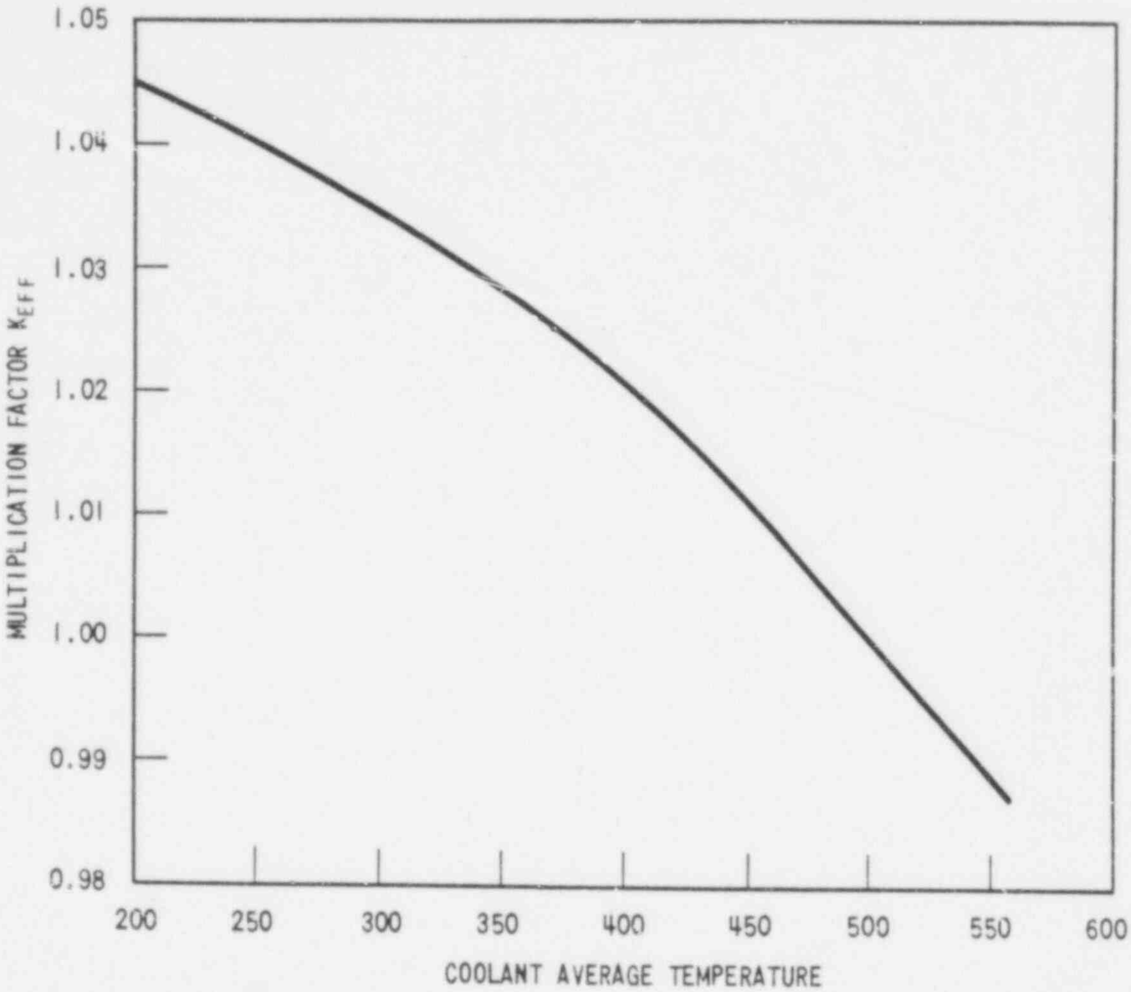

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Figure 15.1-11.

K_{EFF} vs. Coolant Average Temperature

613 142

Figure 15.1-12.

This Figure is Not Applicable

613 143

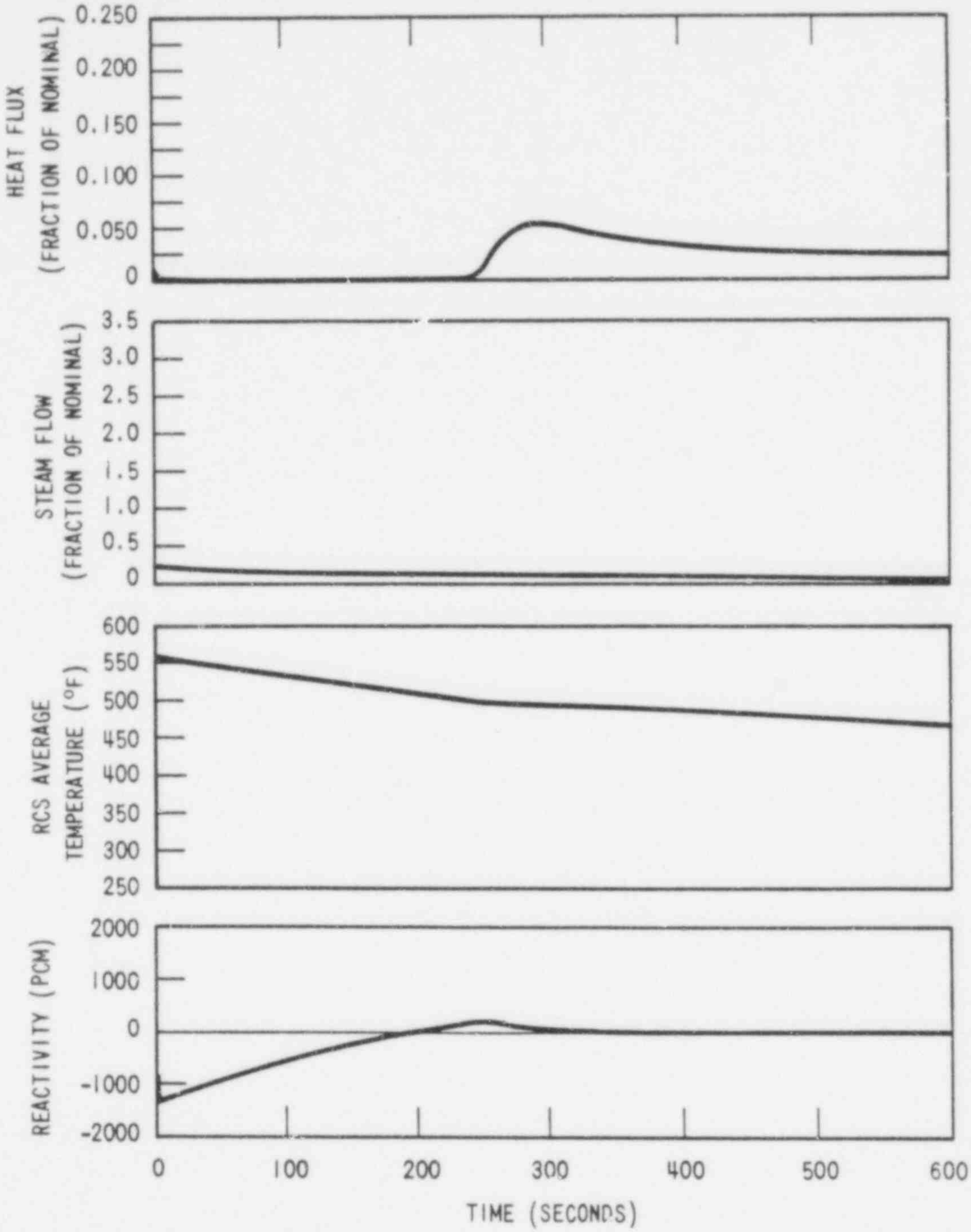

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Figure 15.1-13.

Failure of a Steam Generator Safety or Dump Valve - Heat Flux vs. Time, Steam Flow vs. Time, Average Temperature vs. Time, Reactivity vs. Time

613 144

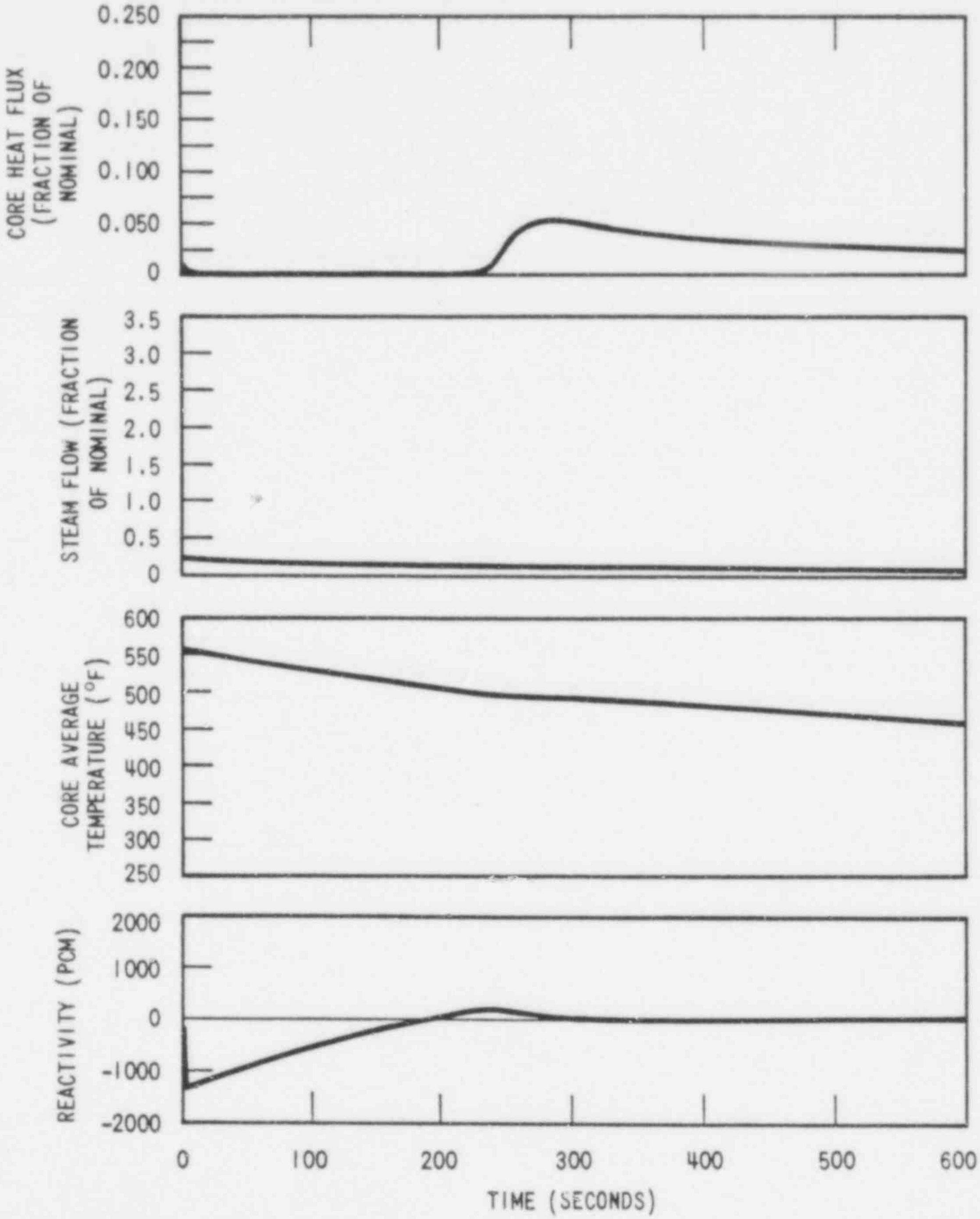

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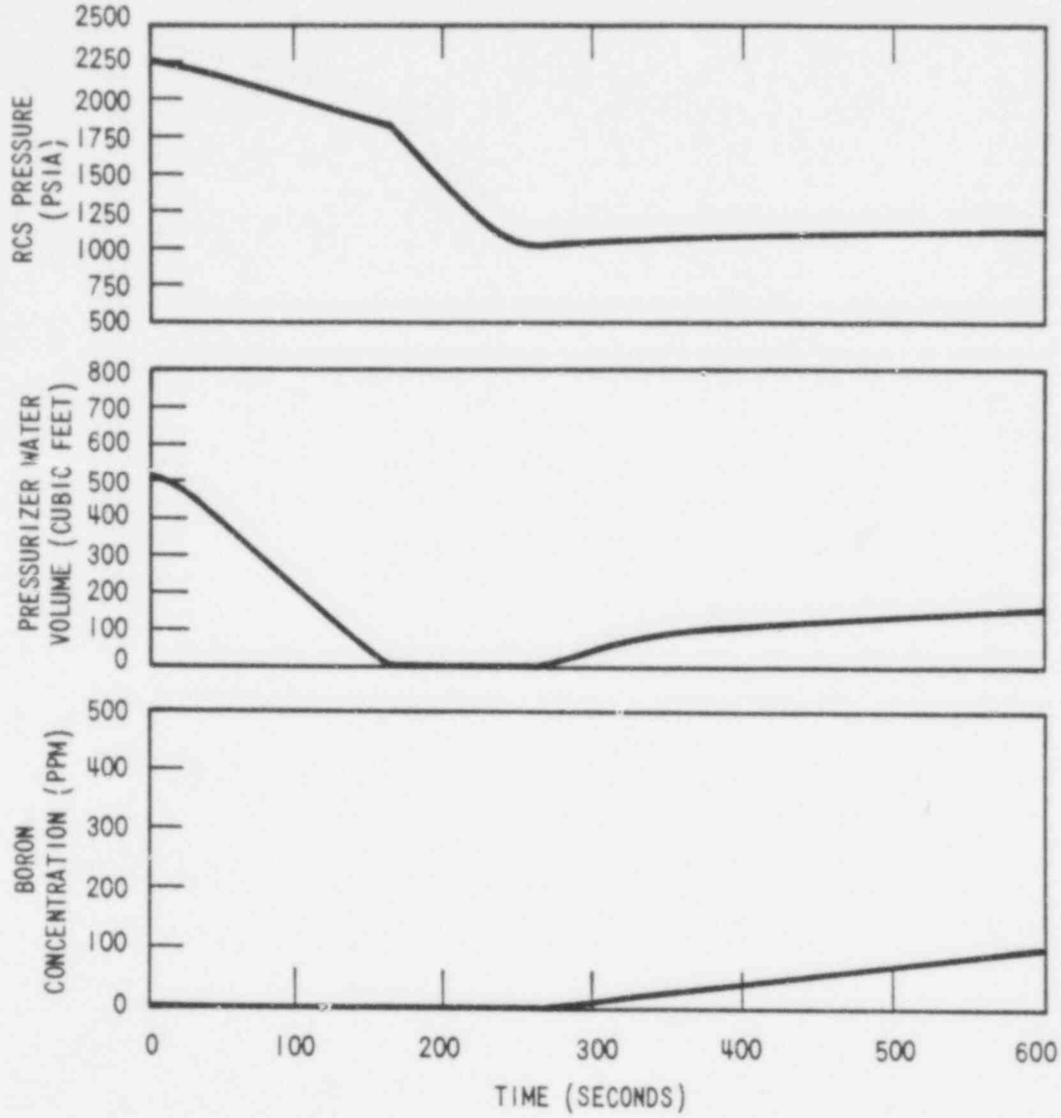
Figure 15.1-13a.

Failure of a Steam Generator Safety or Dump Valve - Heat Flux vs. Time, Steam Flow vs. Time, Average Temperature vs. Time, Reactivity vs. Time (N-1 Loop Operation)

613 145

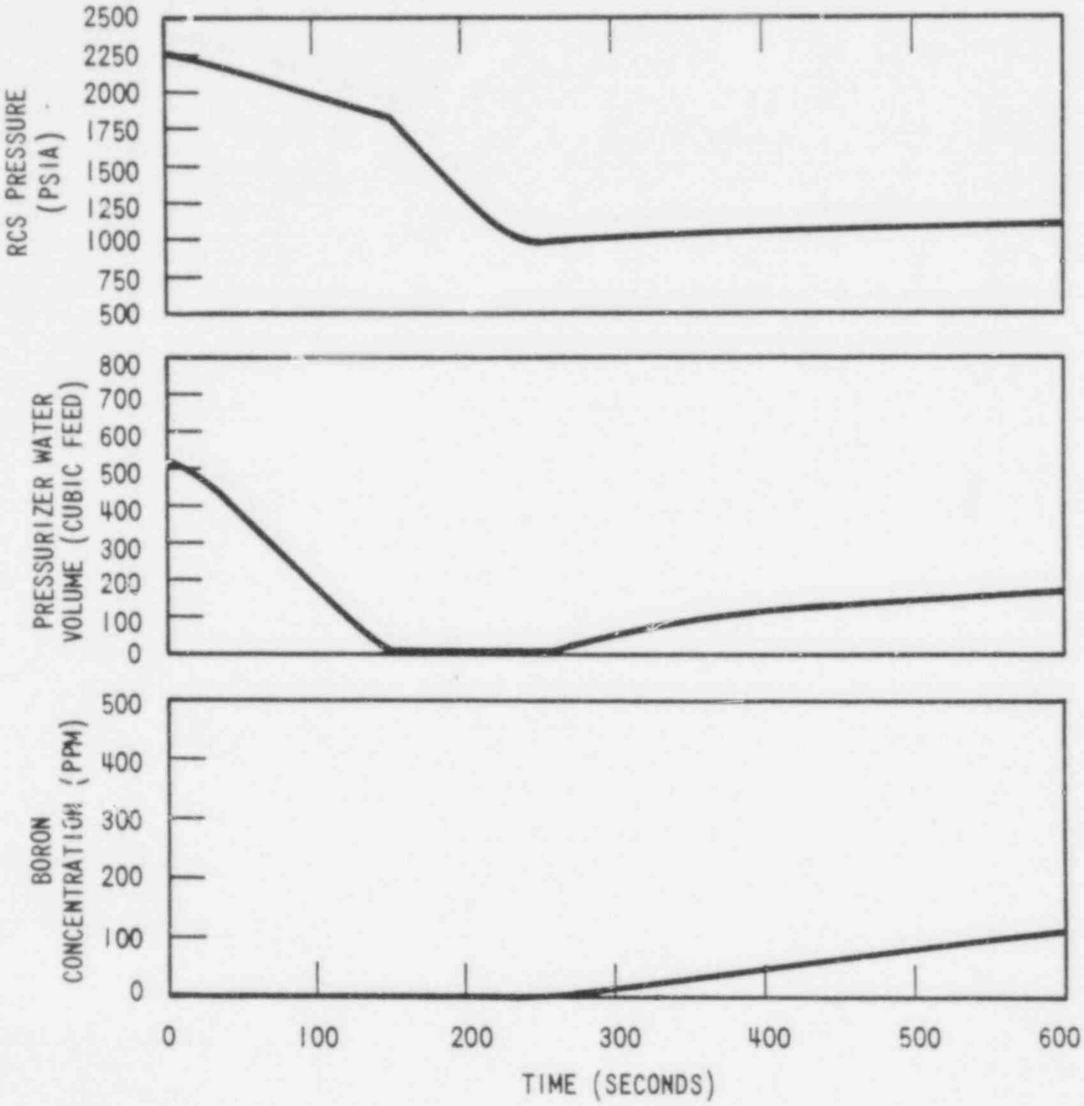


WCAP-9500

Figure 15.1-14.

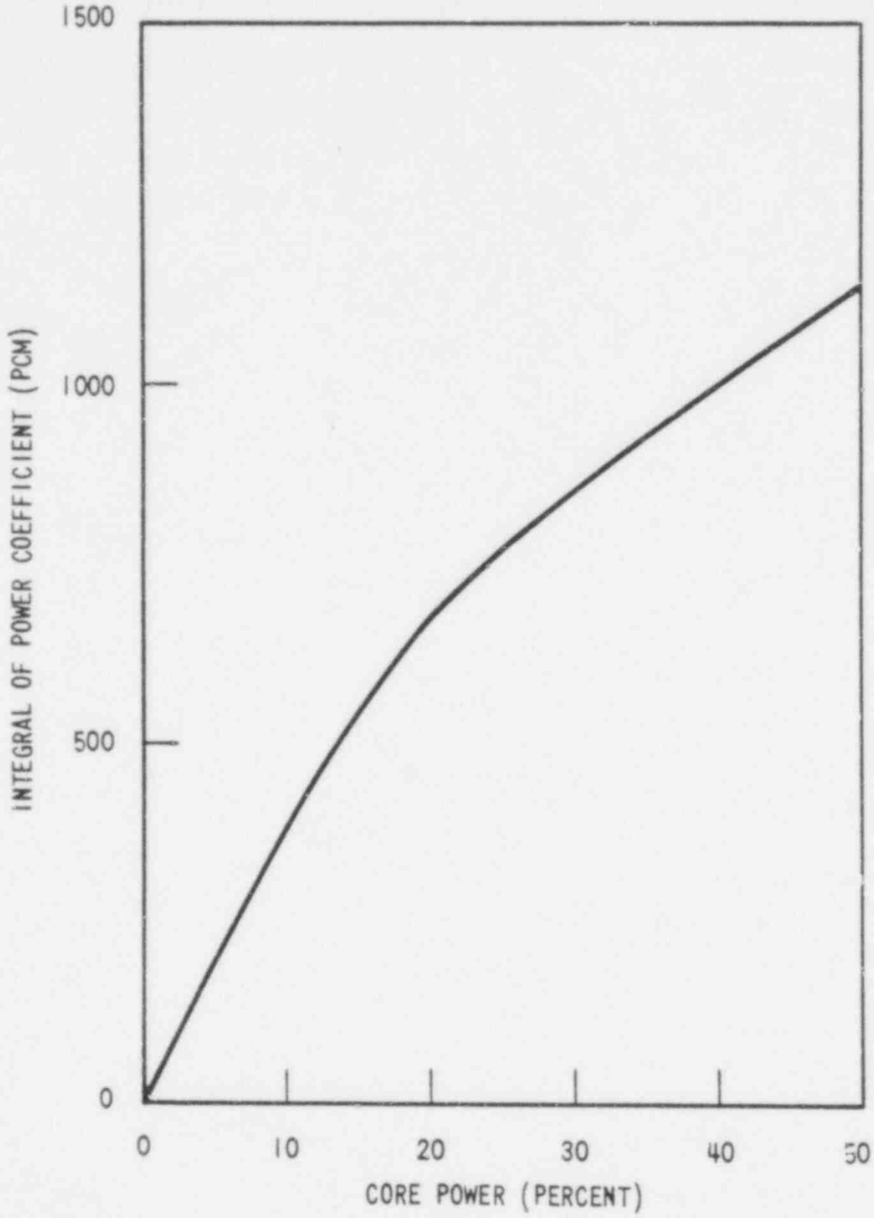
Failure of a Steam Generator Safety or Dump Valve - RCS Pressure vs. Time, Pressurizer Water Volume vs. Time, Boron Concentration vs. Time

613 146



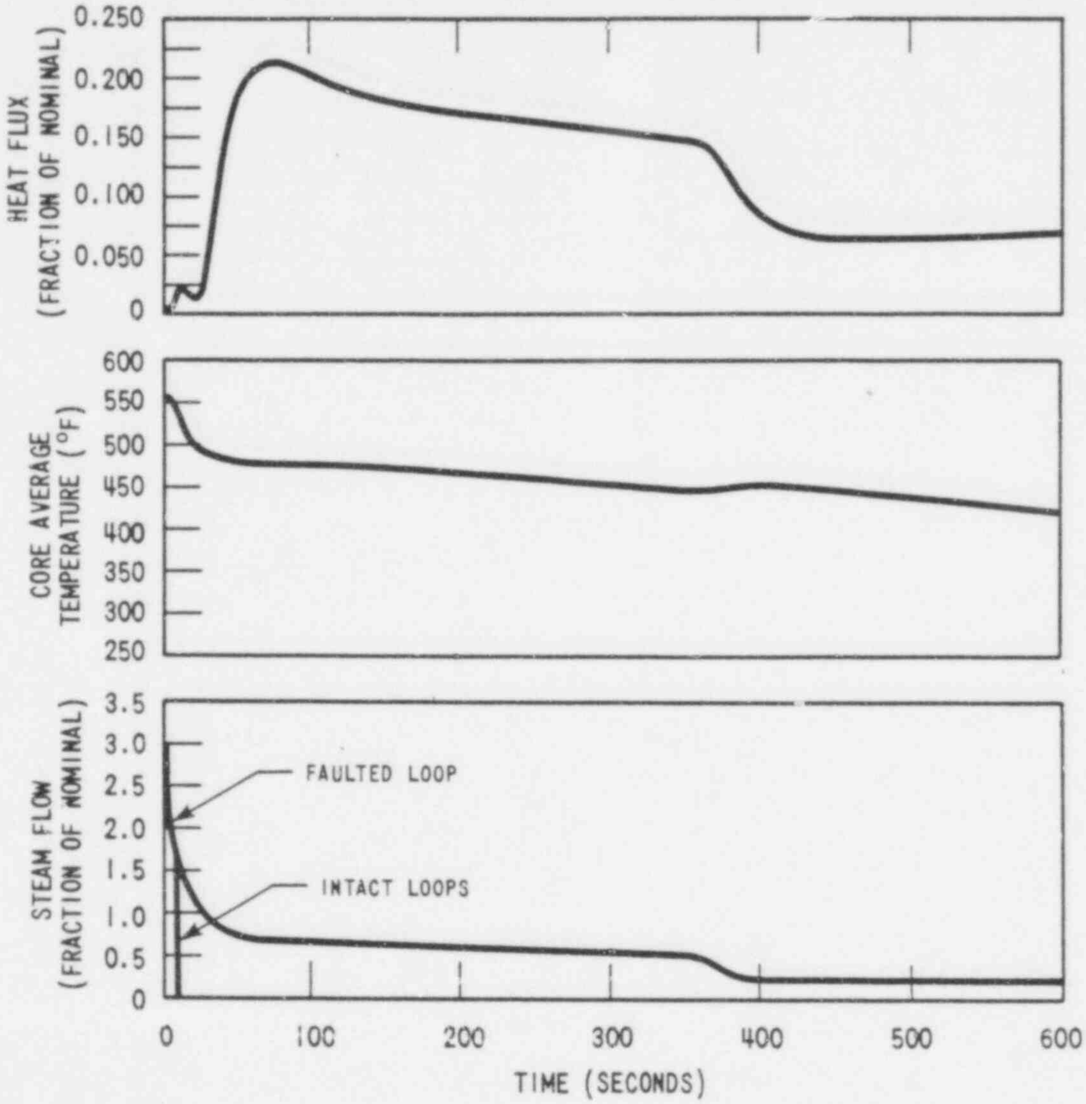
WCAP – 9500
Figure 15.1-14a. Failure of a Steam Generator Safety or Dump Valve - RCS Pressure vs. Time, Pressurizer Water Volume vs. Time, Boron Concentration vs. Time (N-1 Loop Operation)

613 147



WCAP - 9500
Figure 15.1-15. Doppler Power Feedback

613 148

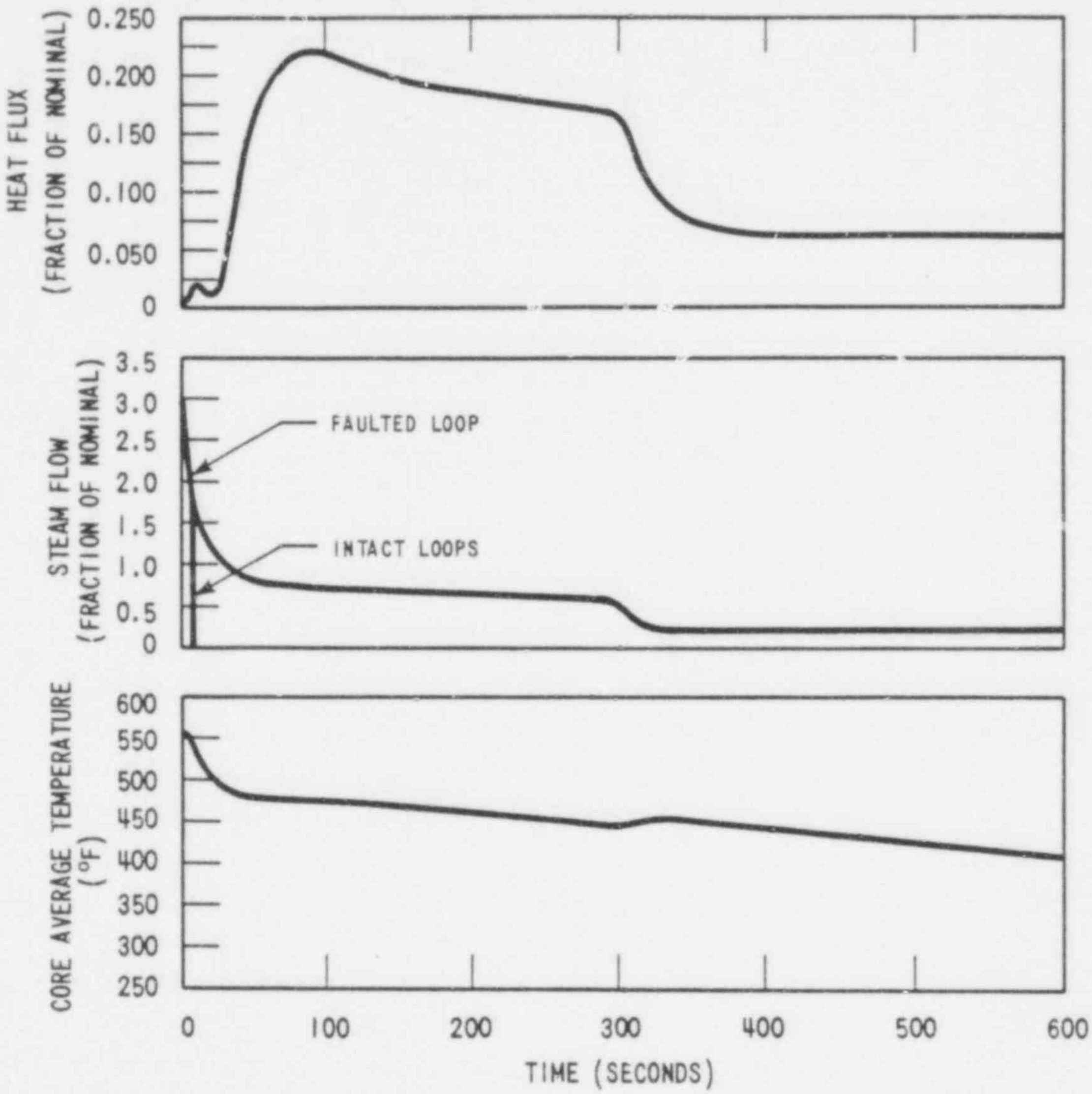


WCAP-9500

Figure 15.1-16.

1.4 FT^2 Steamline Rupture, Offsite Power Available - Heat Flux vs. Time, Average Temperature vs. Time, Steam Flow per Loop vs. Time

613 149

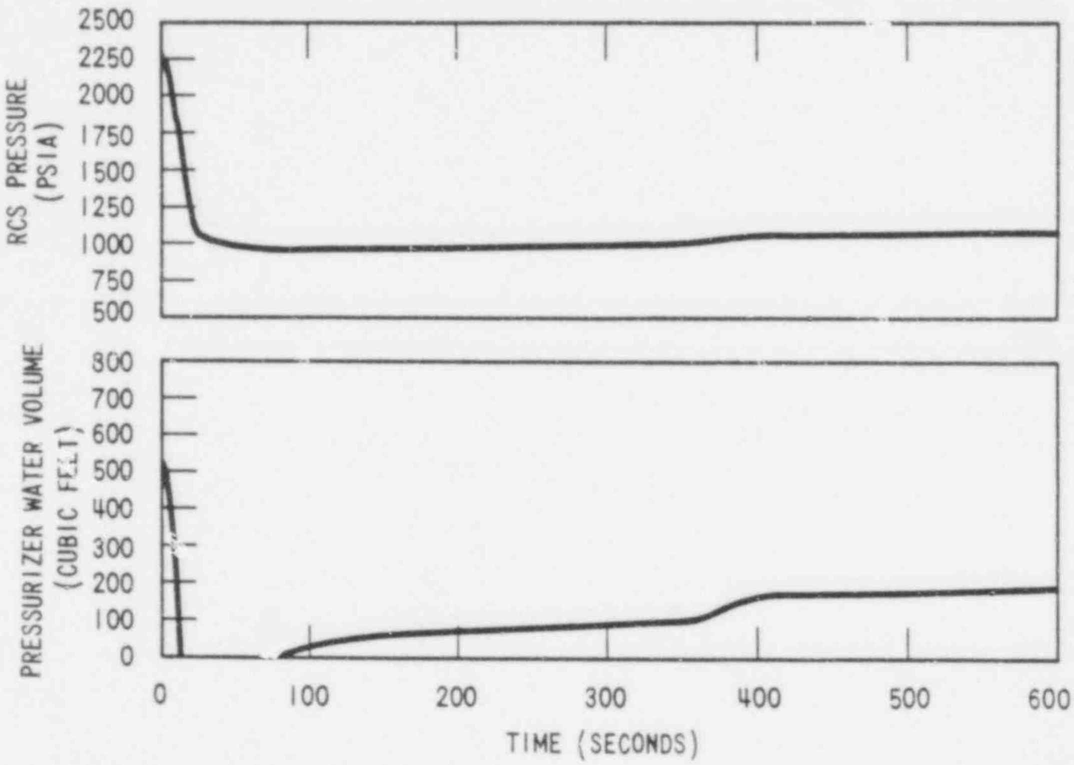


WCAP - 9500

Figure 15.1-16a.

1.4 FT^2 Steamline Rupture, Offsite Power Available - Heat Flux vs. Time, Average Temperature vs. Time, Steam Flow per Loop vs. Time (N-1 Loop Operation)

613 150



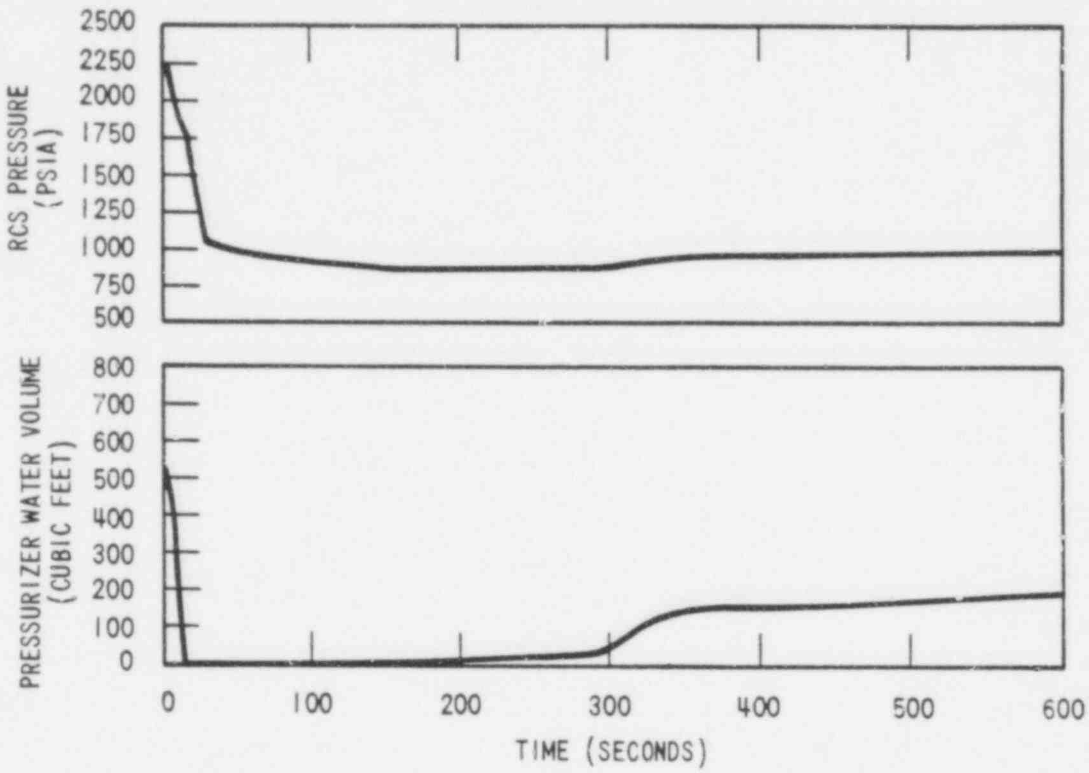
613 151

WCAP - 9500

Figure 15.1-17.

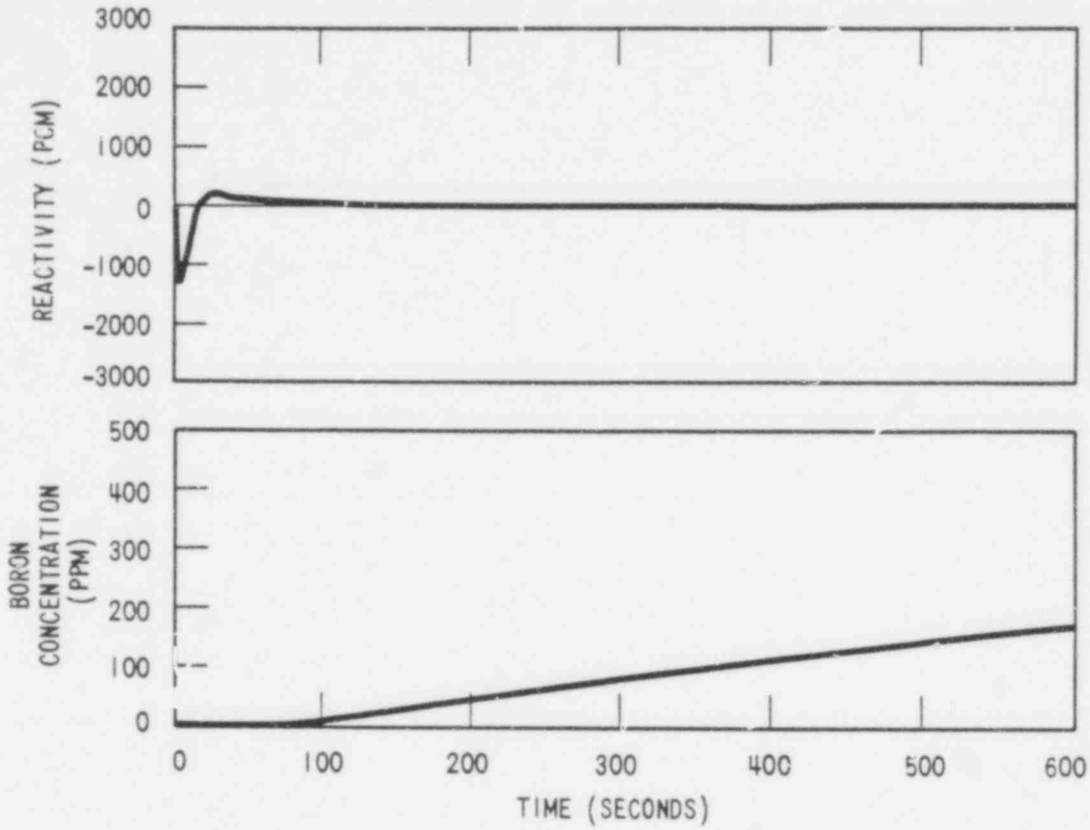
1.4 FT^2 Steamline Rupture, Offsite Power Available, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time

14,395-216



WCAP – 9500
Figure 15.1-17a. 1.4 FT^2 Steamline Rupture, Offsite Power Available, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time (N-1 Loop Operation)

613 152

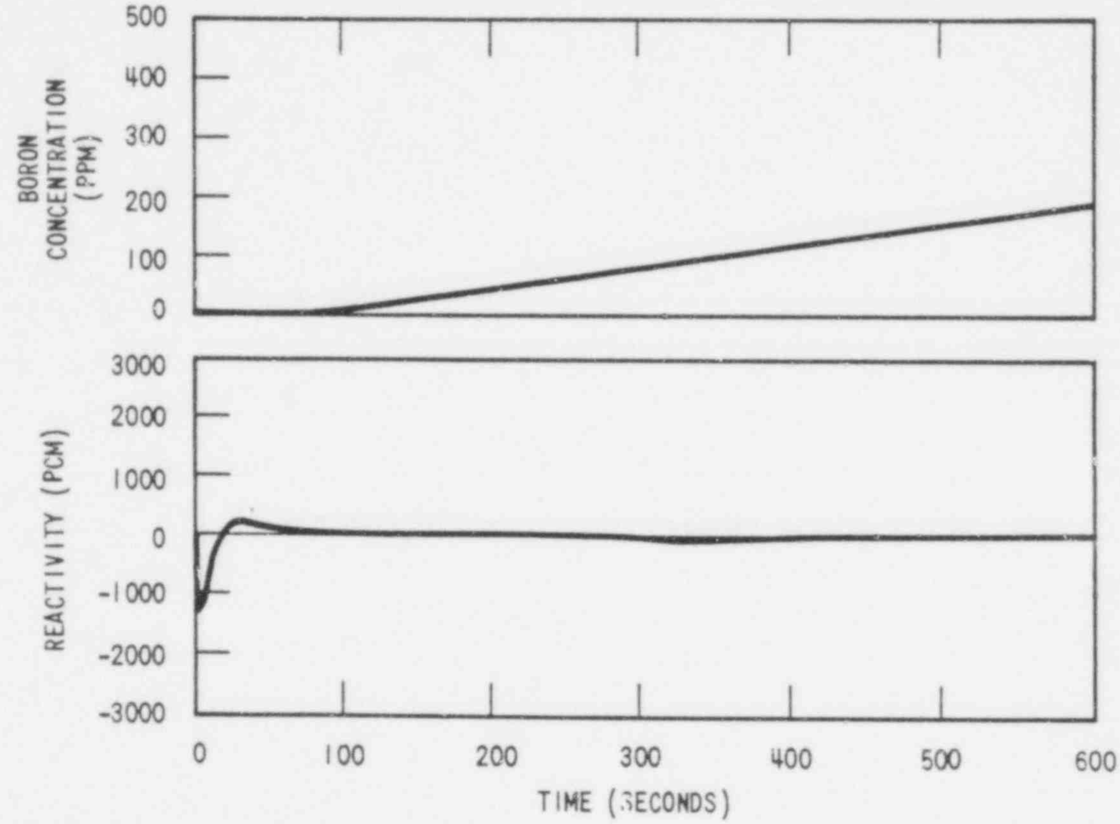


WCAP - 9500

Figure 15.1-18

1.4 FT^2 Steamline Rupture, Offsite Power Available, Boron Concentration vs. Time, Reactivity vs. Time

613 153

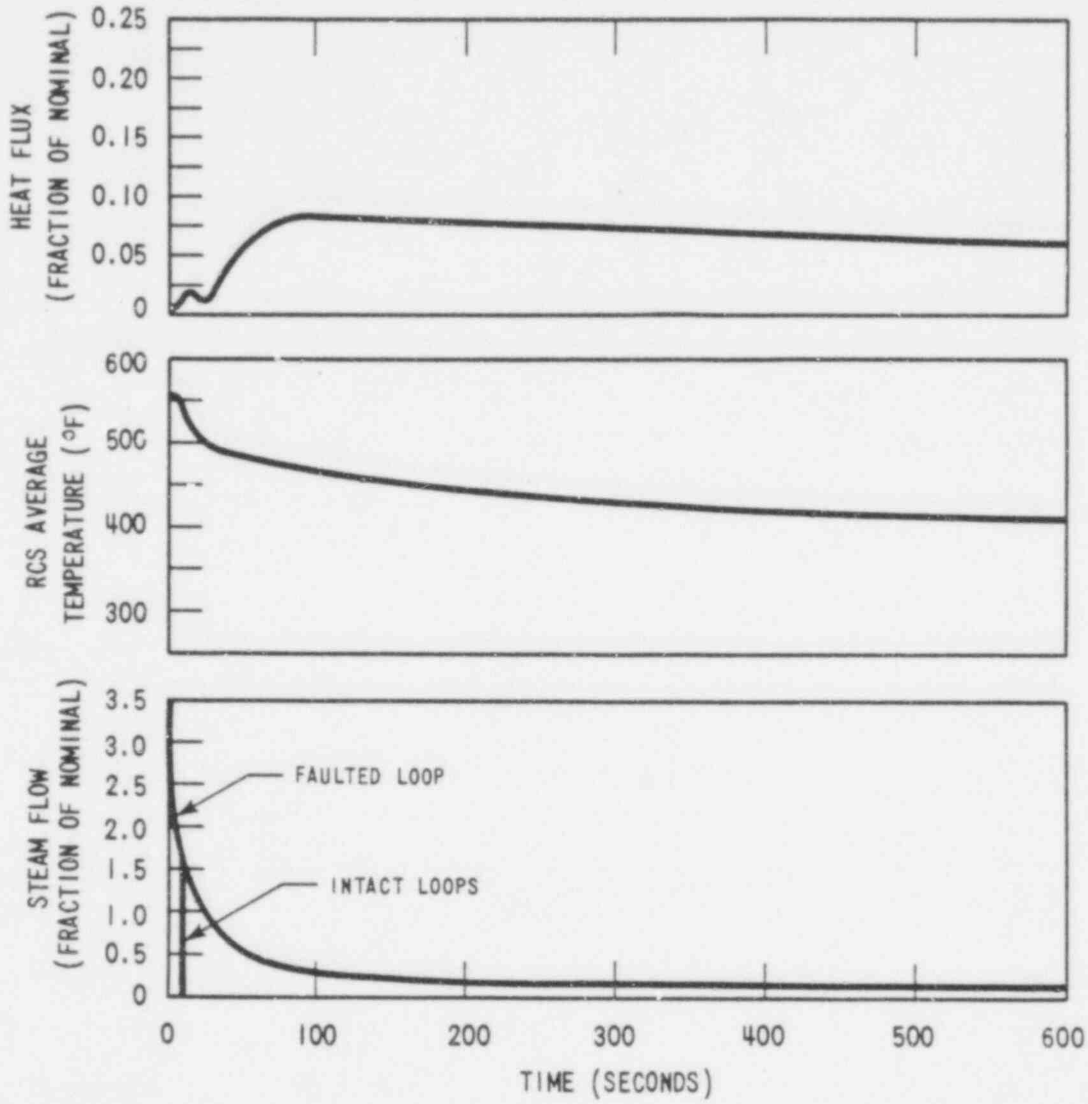


WCAP - 9500

Figure 15.1-18a.

1.4 FT^2 Steamline Rupture, Offsite Power Available, Boron Concentration vs. Time, Reactivity vs. Time (N-1 Loop Operation)

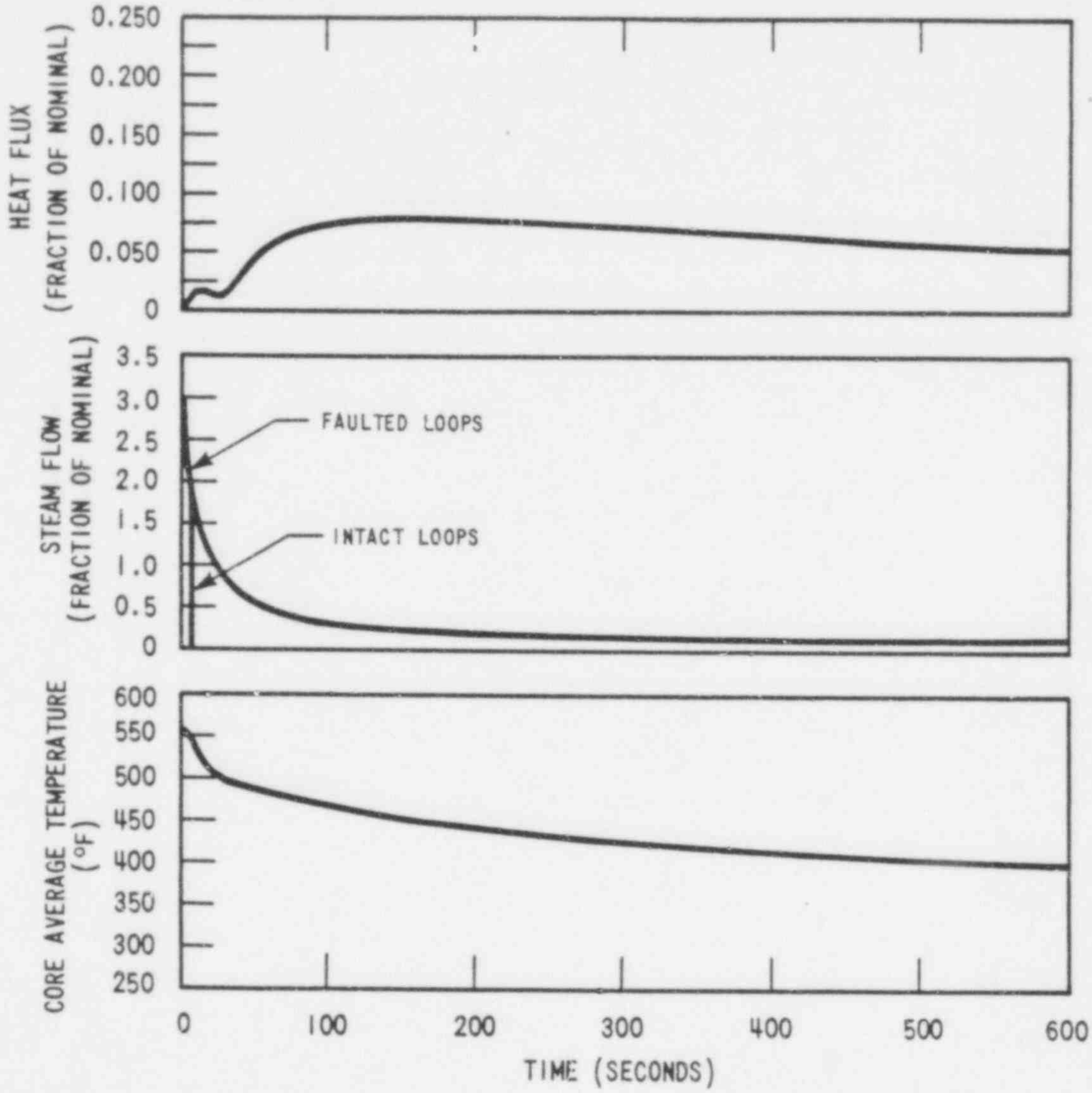
613 154



613 155

WCAP-9500
Figure 15.1-19. 1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Heat Flux vs. Time, Average Temperature vs. Time, Steam Flow per Loop vs. Time

14,395-218

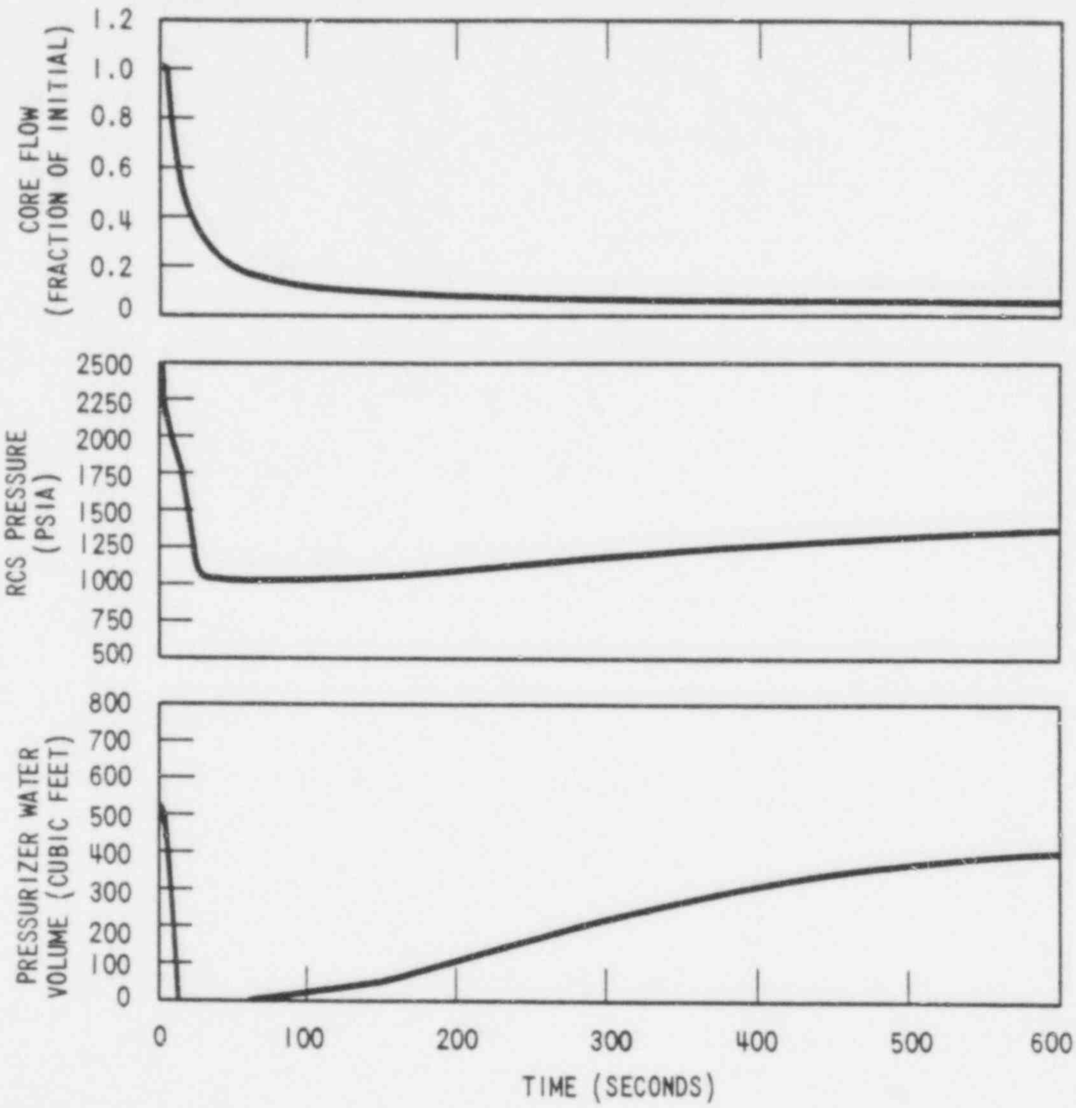


WCAP - 9500

Figure 15.1-19a.

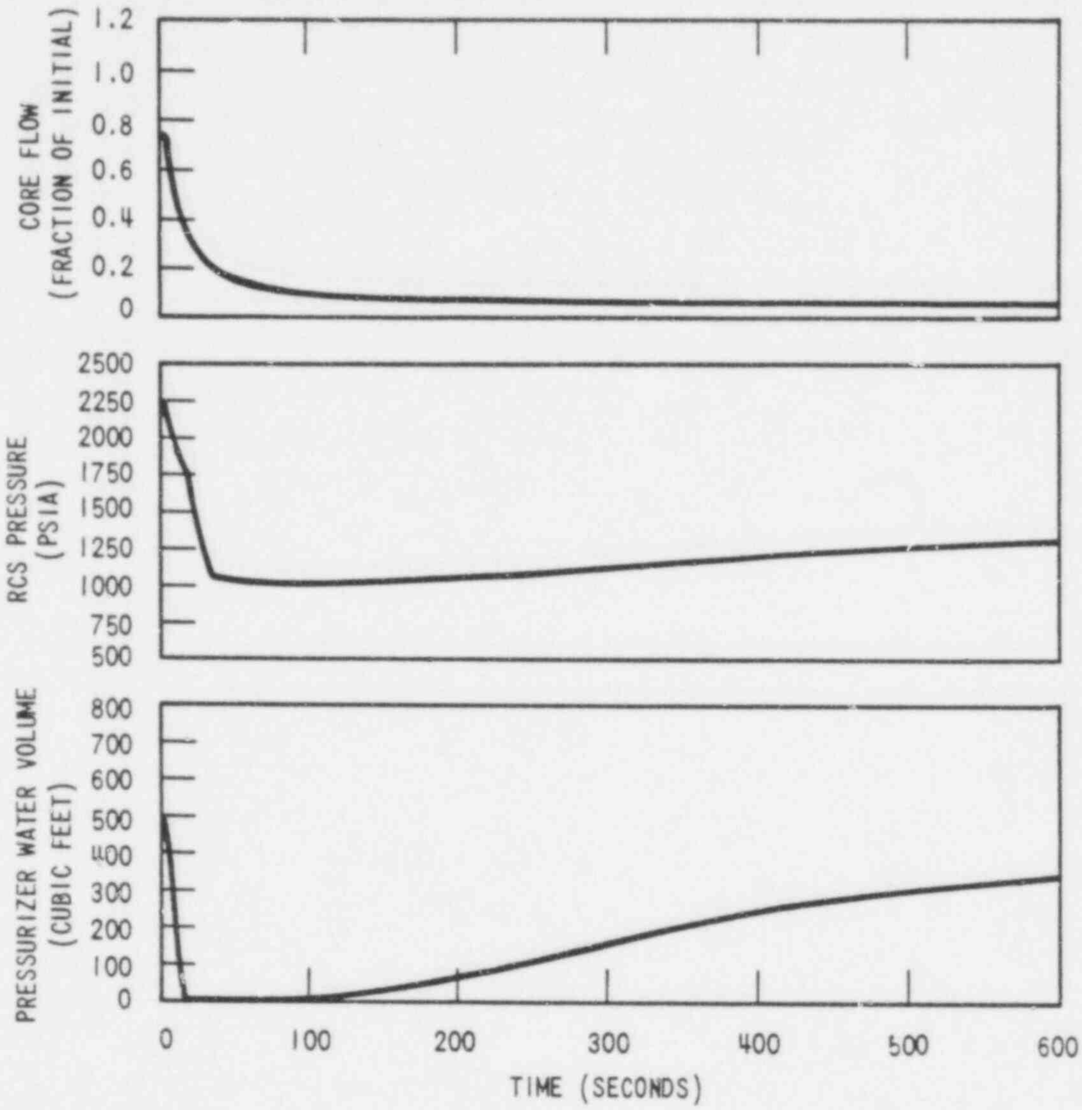
1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Heat Flux vs. Time, Average Temperature vs. Time, Steam Flow per Loop vs. Time (N-1 Loop Operation)

613 156



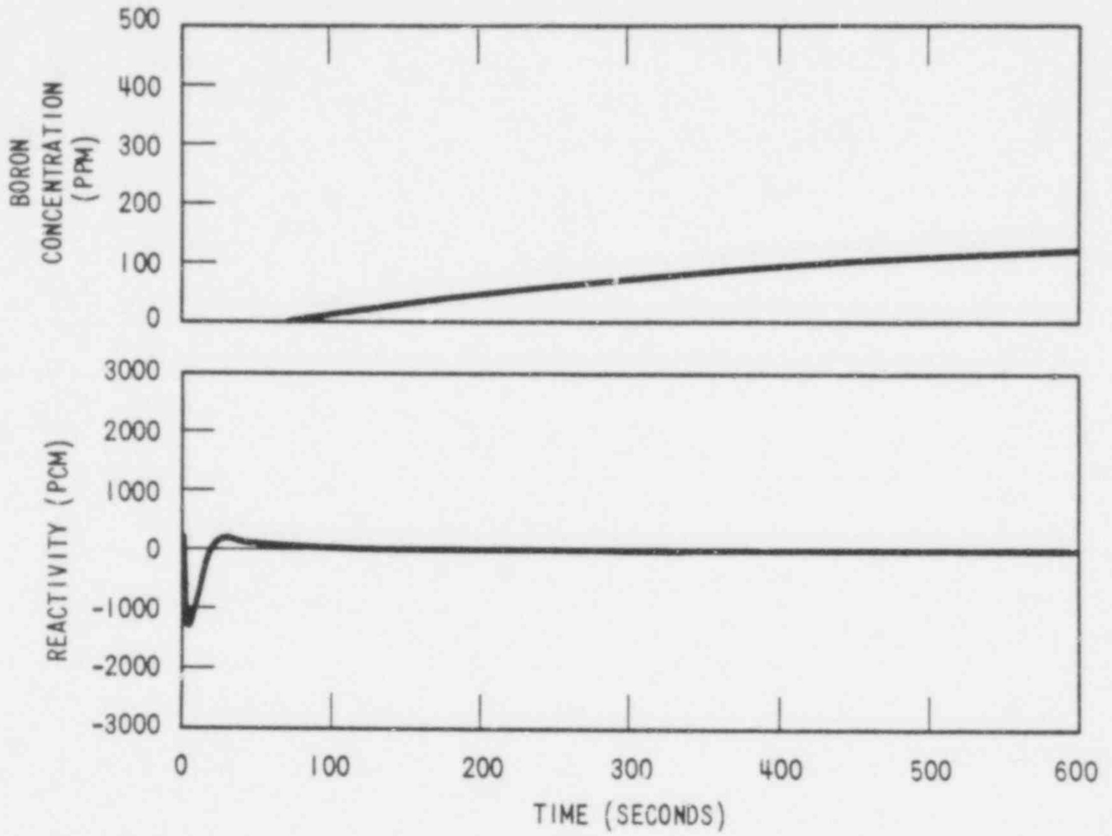
WCAP-9500
Figure 15.1-20. 1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Core Flow vs. Time, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time

613 157



WCAP - 9500
Figure 15.1-20a. 1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Core Flow vs. Time, RCS Pressure vs. Time, Pressurizer Water Volume vs. Time (N-1 Loop Operation)

613 158

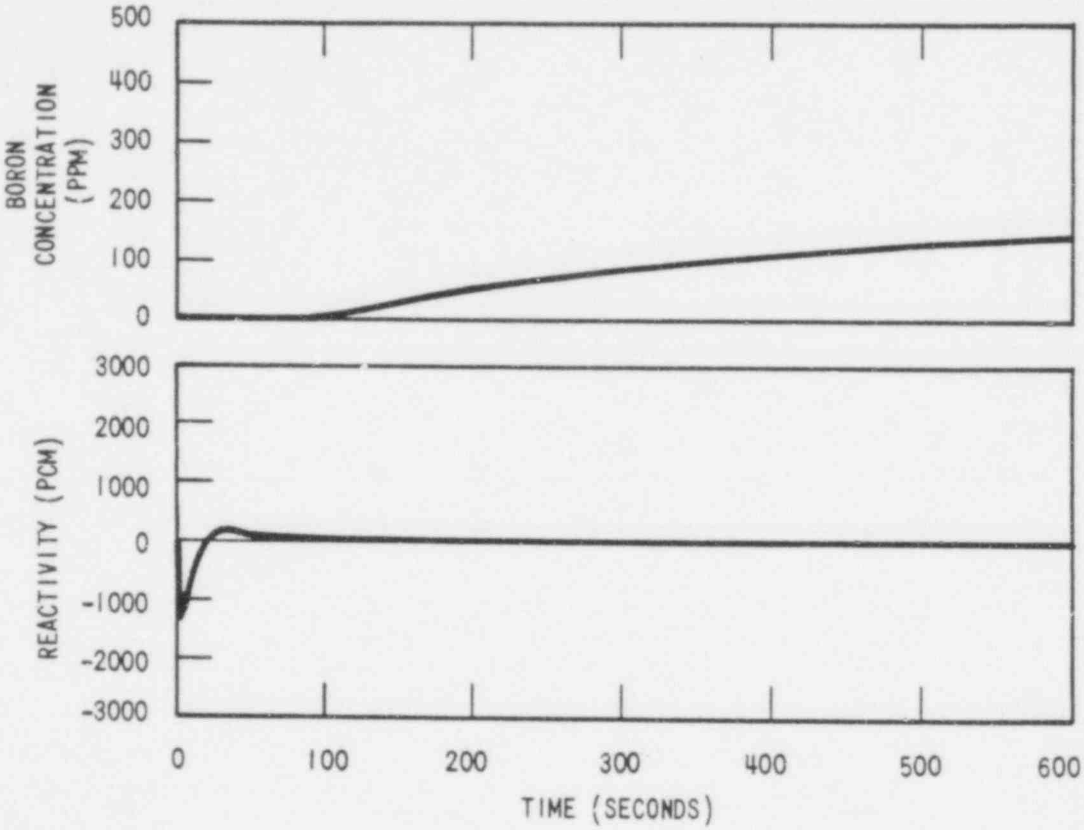


WCAP – 9500

Figure 15.1-21.

1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Boron Concentration vs. Time, Reactivity vs. Time

613 159



WCAP - 9500

Figure 15.1-21a

1.4 FT^2 Steamline Rupture, Offsite Power Not Available - Boron Concentration vs. Time, Reactivity vs. Time (N-1 Loop Operation)

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15.2

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15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

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

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

1. Steam pressure regulator malfunction or failure that results in decreasing steam flow (not applicable).

2. Loss of external electrical load (Subsection 15.2.2).

3. Turbine trip (Subsection 15.2.3).

4. Inadvertent closure of main steam isolation valves (Subsection 15.2.4).

5. Loss of condenser vacuum and other events causing a turbine trip (Subsection 15.2.5).

6. Loss of nonemergency AC power to the plant auxiliaries (Subsection 15.2.6).

7. Loss of normal feedwater flow (Subsection 15.2.7).

8. Feedwater system pipe break (Subsection 15.2.8).

The above items are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event.

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

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

15.2.2 LOSS OF EXTERNAL ELECTRICAL LOAD

15.2.2.1 Identification of Causes and Accident Description

A loss of external electrical load may occur due to some electrical system disturbance. Offsite AC power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite standby 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, no direct reactor trip signal would be generated. With full load rejection capability, plant operation would be expected to continue without a reactor trip. The plant would be expected to trip from the Reactor Protection System if a safety limit were approached. 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 low DNBR trips. Following a complete loss of load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz.

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Any increased frequency to the reactor coolant pump motors will result in a corresponding increase in flowrate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition does not effect other safety-related pump motors, Reactor Protection System equipment, or other safeguard loads. Safeguard loads are supplied from offsite power or, alternatively, from standby diesels. Reactor Protection System equipment is supplied from the 118 volt AC instrument power supply system, which in turn is supplied from the inverters; the inverters are supplied from a d-c bus energized from batteries or by a rectified AC voltage from safeguard 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 low DNBR 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 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, automatic rod control or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to relieve 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 load with the plant initially operating at the maximum calculated turbine load. The pressurizer safety valves and steam generator safety valves are 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]

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A loss of external load is classified as an ANS Condition II event, fault of moderate frequency. See Subsection 15.0.2.

A loss of external load event results in an NSSS transient that is less severe than a turbine trip event (see Subsection 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 the 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. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure. Therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

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

15.2.2.2 Analysis of Effects and Consequences

Method of Analsis

Refer to Subsection 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 Subsection 15.2.2.1.

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Normal reactor control systems and engineered safety feature 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 departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system 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 valves relieve to the atmosphere. Since no fuel damage is postulated for this transient the radiological releases will be less severe than those for the steamline break accident analyzed in Subsection 15.1.5.3.

15.2.2.4 Conclusions

Based on results obtained for the turbine trip event (Subsection 15.2.3) and considerations described in Subsection 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met. The radiological consequences of this event are not limiting.

15.2.3 TURBINE TRIP

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15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below 10 percent power) by a signal derived from the turbine auto-stop oil pressure (Westinghouse turbine) and turbine stop valves. The turbine stop valves close rapidly on loss of trip-fluid pressure actuated by one of a number of turbine trip signals. Turbine trip initiation signals include:

1. low condenser vacuum,

2. low bearing oil pressure,

3. turbine thrust bearing failure,

4. turbine overspeed,

5. DE-H DC power failure, and

6. 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 enable the steam dump system and, if above 10 percent power, trip the reactor. The loss of steam flow results in a rapid rise in secondary system temperature and pressure with a resultant primary system transient as described in Subsection 15.2.2.1 for the loss of external load event. The turbine trip event is analyzed because it results in the most rapid reduction in steam flow.

The automatic steam dump system would normally accommodate 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 is

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not available, the excess steam generation would be dumped to the atmosphere. Feedwater flow would be maintained by the Auxiliary Feedwater System to ensure 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 Subsection 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 Subsection 15.0.2).

The plant systems and equipment available to mitigate the consequences of a turbine trip are discussed in Subsection 15.0.9, and listed in Table 15.0-6.

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 without direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins; that is, the turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves and autostop oil pressure. The assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump. 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 variables including temperatures, pressures, and power level.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation. Plant characteristics and initial conditions are discussed in Subsection 15.0.4.

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. Cases with four loops in operation and with three loops in operation are considered.

2. Moderator and Doppler Coefficients of Reactivity

The turbine trip is analyzed with both a least negative moderator temperature coefficient and a large negative moderator temperature coefficient. The most negative Doppler power coefficient is used for all cases (see Figure 15.0-2).

3. Reactor Control

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

4. Steam Release

No credit is taken for the operati[illegible] steam dump system or steam generator power operated relief v[illegible]e. [illegible] steam generator pressure rises to the safety valve setpoint wh[illegible]ase through safety valves limits secondary steam pressure at the setpoint value.

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5. Pressurizer Spray and Power-Operated Relief Valves

Two cases for both the minimum and maximum moderator 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.The auxiliary feedwater flow would remove core decay heat following plant stabilization.

7. Reactor Trip

Reactor trip is actuated by the first Reactor Protection System trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, low DNBR and high pressurizer water level.

Except as discussed above, normal Reactor Control 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 to operate, but the more limiting cases where these functions are not assumed are also presented.

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 moderator feedback and two cases for maximum moderator feedback (Figures 15.2-1 through 15.2-8). For the minimum moderator feedback cases, the core has the least negative moderator coefficient of reactivity. For the maximum moderator feedback cases, the moderator temperature coefficient has its highest absolute value. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-1 and 15.2-2 show the transient responses for the turbine trip with minimum moderator feedback, 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 signal. The minimum DNBR remains well above the limit valve. The pressurizer safety valves are actuated, and maintain primary system pressure below 110 percent of the design value. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint.

Figures 15.2-3 and 15.2-4 show the responses for the total loss of steam load with maximum moderator feedback. 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 the 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 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.

The results would be 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 15.2-5 and 15.2-6 show the transients with minimum moderator feedback. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

Figures 15.2-7 and 15.2-8 are the transients with maximum moderator 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.

Figures 15.2-1a through 15.2-8a show the transient response for the conditions described above assuming operation with three loops in operation.

Following reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. 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.

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.

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, without a direct or immediate reactor trip, presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The DNBR remains above the limit value for all cases analyzed; thus, the DNB design-basis as described in Section 4.4 is met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. The radiological consequences of this event will be less than the steamline break event analyzed in Subsection 15.1.5.3.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

The inadvertent closure of the main steam isolation valves would result in a turbine trip and other consequences as discussed in Subsection 15.2.5.

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

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Subsection

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 Subsection 15.2.3 apply to loss of condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, as listed in Subsection 15.2.3.1, are covered by Subsection 15.2.3. Possible overfrequency effects due to a turbine overspeed condition are discussed in Subsection 15.2.2.1 and are not a concern for this type of event.

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE PLANT AUXILIARIES (LOSS OF OFFSITE POWER)

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency AC power may result in the loss of all power to the station 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 plant, or by a loss of the onsite AC distribution system.

This transient is more severe than the turbine trip event analyzed in Subsection 15.2.3 because for this case the decrease in heat removal by the secondary 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 due to: 1) turbine trip; 2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or 3) due to 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. 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 self actuated 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 the 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 emergency diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

The auxiliary feedwater system is started automatically as described below.

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

1. Low level in any steam generator.

2. Any safety injection signal.

3. Manual actuation.

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

1. Low level in any two steam generators.

2. Manual actuation.

Refer to Chapter 6 for interface criteria for the Auxiliary Feedwater System.

The motor driven auxiliary feedwater pumps are supplied by power from the ESF buses. The turbine-driven auxiliary feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. Both types of pumps are designed to start and supply rated flow within one minute of the initiating signal. The auxiliary pumps take suction from the condensate 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 Subsection 15.0.2 for a discussion of Condition II events.

A loss of nonemergency AC power event is a more limiting event than the turbine-trip-initiated decrease in secondary heat removal without loss of AC power, which was analyzed in Subsection 15.2.3. However, a loss of AC power to the plant auxiliaries as postulated above could also result in a loss of normal feedwater if the condensate pumps lose the 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 AC power event is sufficient to remove residual heat from the core.

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The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Subsection 15.0.9, and listed in Table 15.0-6.

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, 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.

The assumptions used in the analysis are as follows:

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. A heat transfer coefficient in the steam generator associated with RCS natural circulation, following the reactor coolant pump coastdown.

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

5. Auxiliary feedwater is delivered by only one auxiliary feed pump to two steam generators.

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

7. The initial reactor coolant average temperature is 4°F higher than the nominal value.

A second case is analized with three reactor coolant pumps initially operating. All of the assumptions above are applicable, except that the initial power level was assumed to be 72% of the engineered safety features design rating.

The assumptions used in the analysis are similar to the loss of normal feedwater flow incident (Subsection 15.2.7) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip, and only one auxiliary feedwater pump is conservatively assumed to deliver flow.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Results

The transient response of the RCS following a loss of AC power is shown in Figures 15.2-9 and 15.2-10 for 4 loops initially in operation, and in Figures 15.2-9a and 15.2-10a for three loops initially in operation.

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

are calculated sequence of events for this accident are listed in Table 15.2-1.

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 15.2-2. 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 RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The radiological consequences of this event are not limiting 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 RCS. Since the plant is tripped well before the steam generator heat transfer capability 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):

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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 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 the 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 Subsection 15.0.2 for a discussion of Condition II events.

Reactor trip on low water level in any steam generator provides protection for a loss of normal feedwater.

The Auxiliary Feedwater System is started automatically as discussed in Subsection 15.2.6.1. The motor-driven auxiliary feedwater pumps are supplied by power from the ESF buses. The turbine-driven auxiliary feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. The pumps take suction directly from the condensate 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 analysis 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 generator 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 level.

4. The worst single failure in the auxiliary feedwater system occurs (one of the three auxiliary feed pumps fails to start).

5. Auxiliary feedwater is delivered by two auxiliary feed pumps to two steam generators.

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

7. The initial reactor coolant average temperature is 4°F higher than the nominal value.

A second case is analyzed with three reactor coolant pumps initially operating. All of the assumptions above are applicable, except that the

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initial power level was assumed to be 72% of the engineered safety features design rating.

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 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) through the transient.

The assumptions used in the analysis are similar to the loss of AC power incident (Subsection 15.2.6) except that the reactor coolant pumps are assumed to continue to operate, and credit is taken for flow from two of the three auxiliary feedwater pumps.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

Plant systems and equipment which are available to mitigate effects of a loss of normal feedwater accident are discussed in Subsection 15.0.9 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The Reactor Protection System is required to function following a loss of normal feedwater as analyzed here. The Auxiliary Feedwater System is required to deliver a minimum auxiliary feedwater flow rate. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference [2].

Results

Figures 15.2-11 and 15.2-12 show the signif cant plant parameters following a loss of normal feedwater with four loops initially in operation. Figures 15.2-11a and 15.2-12a are for three loops initially in operation.

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 level trip, the 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 relief or safety valves. Figures 15.2-11 and 15.2-12 show that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-1.

As shown in Figures 15.2-11 and 15.2-12, the plant will slowly approach a stabilized condition at hot standby with auxiliary feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the auxiliary feed flow. The operating procedures would also 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 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.

15.2.7.3 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 system does not overpressurizer and 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. (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 Subsection 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.

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Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Subsection 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

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

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.

A major feedwater line rupture is classified as an ANS Condition IV event. See Subsection 15.0.2 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 limiting feedwater line ruptures is a double ended rupture of the largest feedwater line.

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Analyses have been performed at full power with and without loss of offsite power and at 72% power for three loops in operation with and without loss of offsite power. These cases are presented 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. Low steam generator water level in any steam generator,
    c. Safety injection signals from any of the following:
        1) 2/3 low steamline pressure in any one loop,
        2) 2/3 high containment pressure (Hi-1).

    (Refer to Chapter 7.0 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.

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:

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1. The plant is initially operating at 102 percent of the engineered safeguards design rating. (72% power for three loops in operation).

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

3. Initial pressurizer level is at the nominal programmed value plus 2 percent (error); initial steam generator water level is at the nominal value plus 5% in the faulted steam generator and at the nominal value minus 5% in the intact steam generators.

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

5. 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).

6. The full double-ended break area is assumed.

7. A conservative feedwater line 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.

8. Reactor trip is assumed to be actuated when the low level trip setpoint minus 10 percent of narrow range span in the ruptured steam generator is reached.

9. The auxiliary feedwater system is actuated by the low steam generator water level signal. The auxiliary feedwater system is assumed to supply a total of 470 gpm (320 gpm for 3 loops in operation) to

three intact steam generators, including allowance for spillage through the main feedwater line break. A 62-second delay was assumed fol- lowing the low level signal to allow time for startup of the standby diesel generators and the auxiliary feed pumps. An additional 630 seconds (842 for 3 loops in operation) was assumed before the feedwater lines were purged and the relatively cold (120°F) auxiliary feedwater entered the unaffected steam generators.

10. No credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

13. No credit is taken for charging or letdown.

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

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

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

    a. High pressurizer pressure.
    b. High pressurizer level.
    c. High containment pressure.

Receipt of a 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 driven auxiliary feedwater pump is initiated if the low steam generator water level signal is reached in at least two steam generators. Similiarly, 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.

Emergency operating procedures following a main feed line rupture require the following actions to be taken:

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

2. Turn off all reactor coolant pumps (if offsite power is still available), if SI pump operation has been verified and the wide range RCS pressure is decreasing and below 1550 psig, or component cooling water is lost.

Shutting off the reactor coolant pumps (action 2, above) serves to decrease the addition of energy (approximately 4.0 megawatts (MW) per pump) to the RCS. 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 plant operating procedures will be followed in cooling the plant to hot shutdown conditions.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.4.

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. The turbine driven auxiliary feedwater pump has been assumed to fail; the motor-driven pumps deliver 470 gpm (320 for N-1 loop operation) to the three steam generators. Only one train of safety injection has been assumed to be

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available which is consistent with the single failure criterion and minimum safeguards analysis. (The SI pumps have a 1500 psi cutoff head).

Following the trip of the reactor coolant pumps, 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 Subsection 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 Subsections 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 6.6.

Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2-13 through 15.2-24 for four loops initially in operation, and Figures 15.2-13a through 15.2-24a for three loops initially in operation. Results for the case with offsite power available are presented in Figures 15.2-13 through 15.2-18. Results for the case where offsite power is lost are presented in Figures 15.2-19 through 15.2-24. The calculated sequence of events for both cases analyzed are listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented in Figures 15.2-14 and 15.2-17 (with offsite power available) and Figures 15.2-20 and 15.2-23 (without offsite power) show that pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip occurs on low steam generator water level. Pressure then decreases, due to the

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loss of heat input, until the time at which the mass inventory in the intact steam generators is not sufficient to remove the core decay heat, and until steamline isolation and safety injection actuation occur. The pressurizer relief valves open to maintain RCS 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.

Reactor coolant system pressure will be maintained at the safety valve setpoint until safety injection flow is terminated. 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 pumps.

The major difference between the two cases analyzed can be seen in the plots of hot and cold-leg temperatures, Figures 15.2-15 and 15.2-16 (with offsite power available) and Figures 15.2-21 and 15.2-22 (without offsite power). It is apparent that for the initial transient (300 seconds), 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 due to the addition of pump heat.

The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition; hence, more water is relieved for the cases with power. As previously stated, however, the core remains covered with water for all 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, auto- matic 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.

15.2.9 REFERENCES

1. M. A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971.

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

3. T. W. T. Burnett et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

4. M. S. Balwin et al., "An Evaluation of Loss of Flow Accidents caused by Power System Frequency Transients in Westinghouse PWR's," WCAP-8424, Revision 1, June 1975.

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TABLE 15.2-1

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
Turbine Trip			
1. With pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feedwater flow	0.0	0.0
	High pressurizer pressure reactor trip point reached	8.4	-
	Initiation of steam release from steam generator safety valves	7.4	11.5
	Low SG level reactor trip point reached	-	36.0
	Rods begin to drop	10.4	38.0
	Minimum DNBR occurs	*	38.0
	Peak pressurizer pressure occurs	12.0	17.0
2. With pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0

*DNBR does not decrease below its initial value.

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TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR ACCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Initiation of steam release from steam generator safety valves	7.4	11.5
	Minimum DNBR occurs	*	*
	Peak pressurizer pressure occurs	8.5	6.8
	Low-steam generator level reactor trip point reached	58.7	43.4
	Rods begin to drop	60.7	45.4
3. Without pressurizer control (minimum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0
	High pressurizer pressure reactor trip point reached	5.5	7.9
	Rods begin to drop	7.5	9.9

*DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Initiation of steam release from steam generator safety valves	7.3	11.5
	Minimum DNBR occurs	*	*
	Peak pressurizer pressure occurs	8.9	11.5
4. Without pressurizer control (maximum moderator feedback)	Turbine trip, loss of main feed flow	0.0	0.0
	High pressurizer pressure reactor trip point reached	5.5	8.1
	Rods begin to drop	7.5	10.1
	Initiation of steam release from steam generator safety valves	7.4	11.5

*DNBR does not decrease below its initial value.

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TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Minimum DNBR occurs	*	*
	Peak pressurizer pressure occurs	8.4	11.5
Loss of Non-Emergency AC Power	Main feedwater flow stops	0.0	0.0
	Low steam generator water level trip	41.2	29.7
	Rods begin to drop	43.2	31.7
	Reactor coolant pumps begin to coastdown	43.2	31.7
	Two steam generators begin to receive auxiliary feedwater from one auxiliary feedwater pump	103.2	91.7
	Peak water level in pressurizer occurs	46.5	33.5

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TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1817	509
Loss of Normal Feedwater flow	Main feedwater flow stops	0.0	0.0
	Low steam generator water level trip	41.2	29.7
	Rods begin to drop	43.2	31.7
	Two steam generators begin to receive auxiliary feed from two auxiliary feedwater pumps	103.2	91.7
	Peak water level in pressurizer occurs	745.9	35.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	760.0	218.0

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TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
Feedwater System Pipe Break			
1. With Offsite Power Available	Main feedline rupture occurs	10	10
	Low steam generator level reactor trip setpoint reached in ruptured steam generator	14	13
	Rods begin to drop	16	15
	Auxiliary feedwater is delivered to intact steam generators	76	75
	Low steamline pressure setpoint reached in ruptured steam generator	106	98
	All main steamline isolation valves close	111	103

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TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Steam generator safety valve setpoint reached intact steam generators	632	830
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~5000	~4000
2. Without Offsite Power	Main feedline rupture occurs	10	10
	Low steam generator level reactor trip setpoint reached in ruptured steam generator	14	13
	Rods begin to drop, power lost to the reactor coolant pumps	16	15
	Low steamline pressure setpoint reached in ruptured steam generator	46	39
	All main steamline isolation valves close	51	44

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TABLE 15.2-1 (Continued)

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

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Auxiliary feedwater is delivered to intact steam generators	76	75
	Steam generator safety valve setpoint reached in intact steam generators	184	212
	Core decay heat decreases to auxiliary feedwater heat removal capacity	$\sim$600	$\sim$700

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TABLE 15.2-2

PARAMETERS TO BE USED IN ANALYSIS OF THE RADIOLOGICAL CONSEQUENCES OF LOSS OF AC POWER ANALYSIS

Parameter	Realistic Valve	Conservative 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 Feed· ter 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.

*** DE = Dose Equivalent

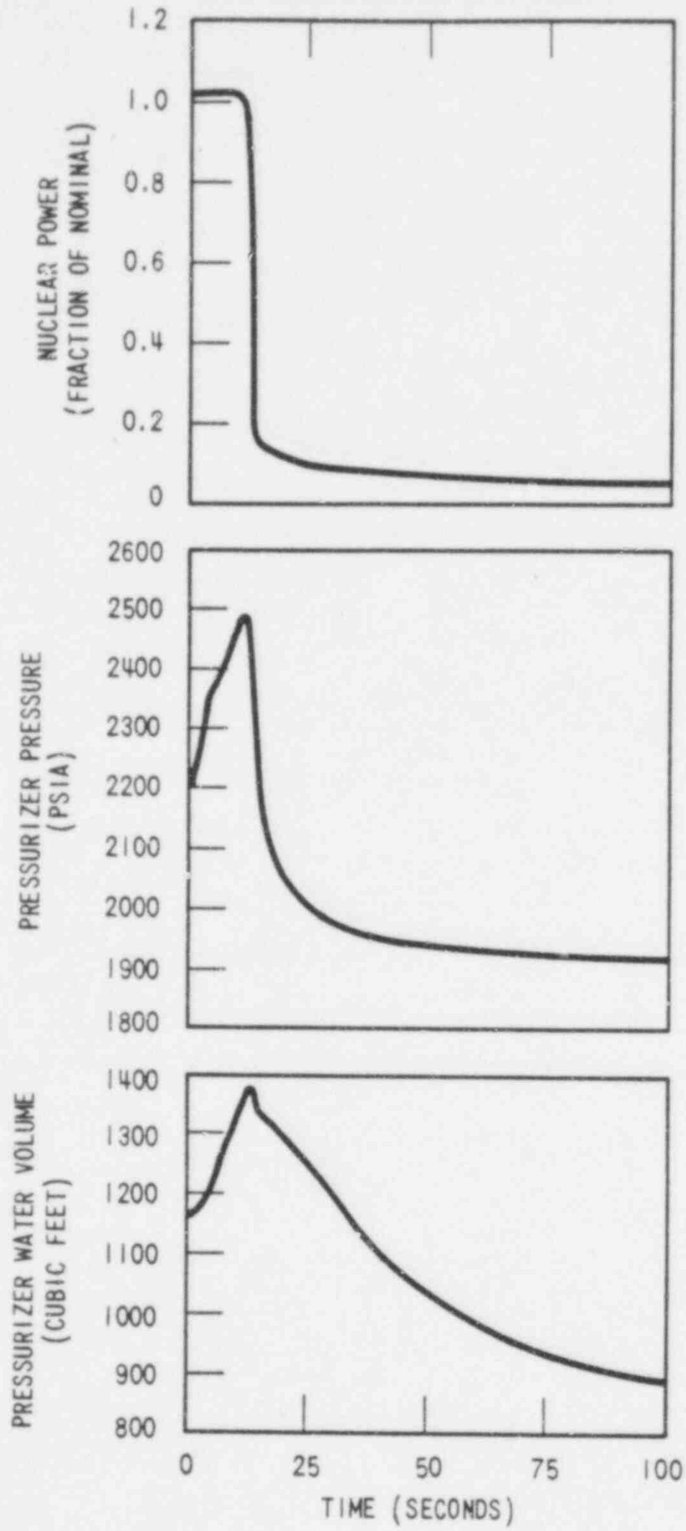

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Figure 15.2-1.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

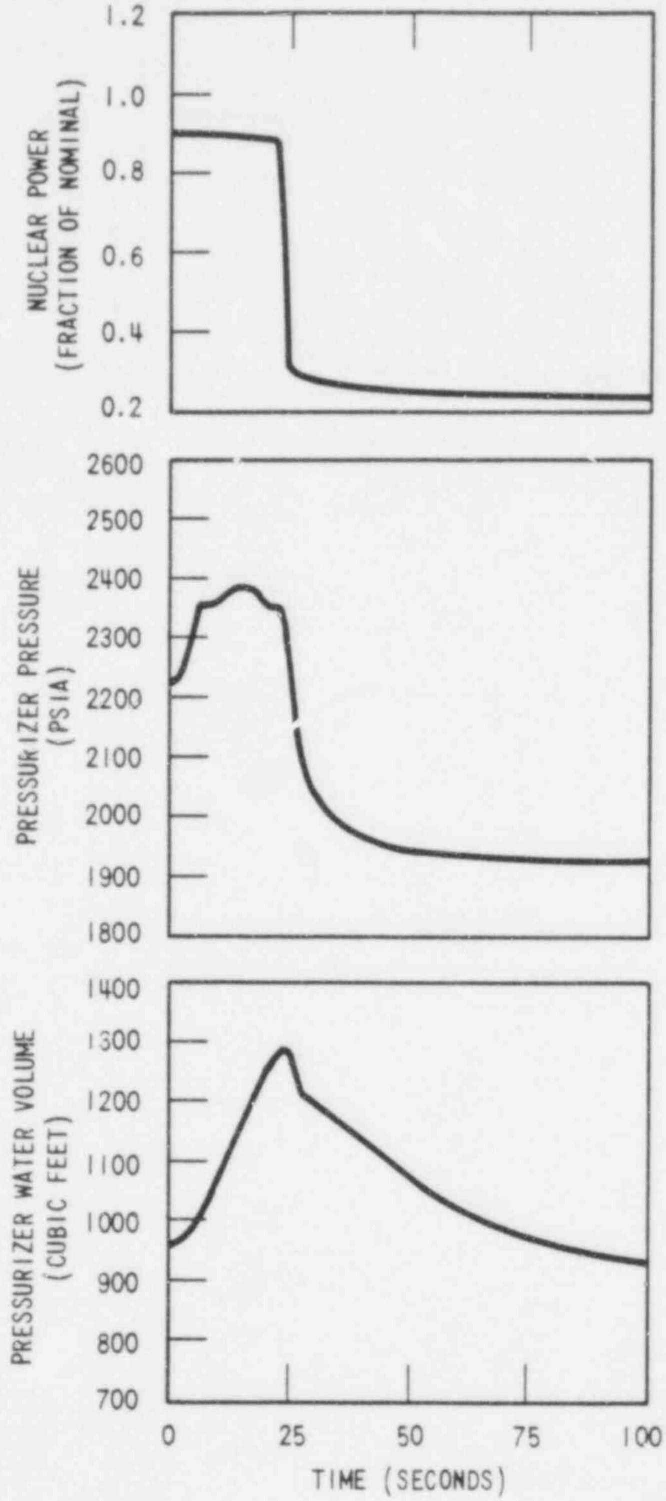

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Figure 15.2-1a.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback (N-1 Loop Operation)

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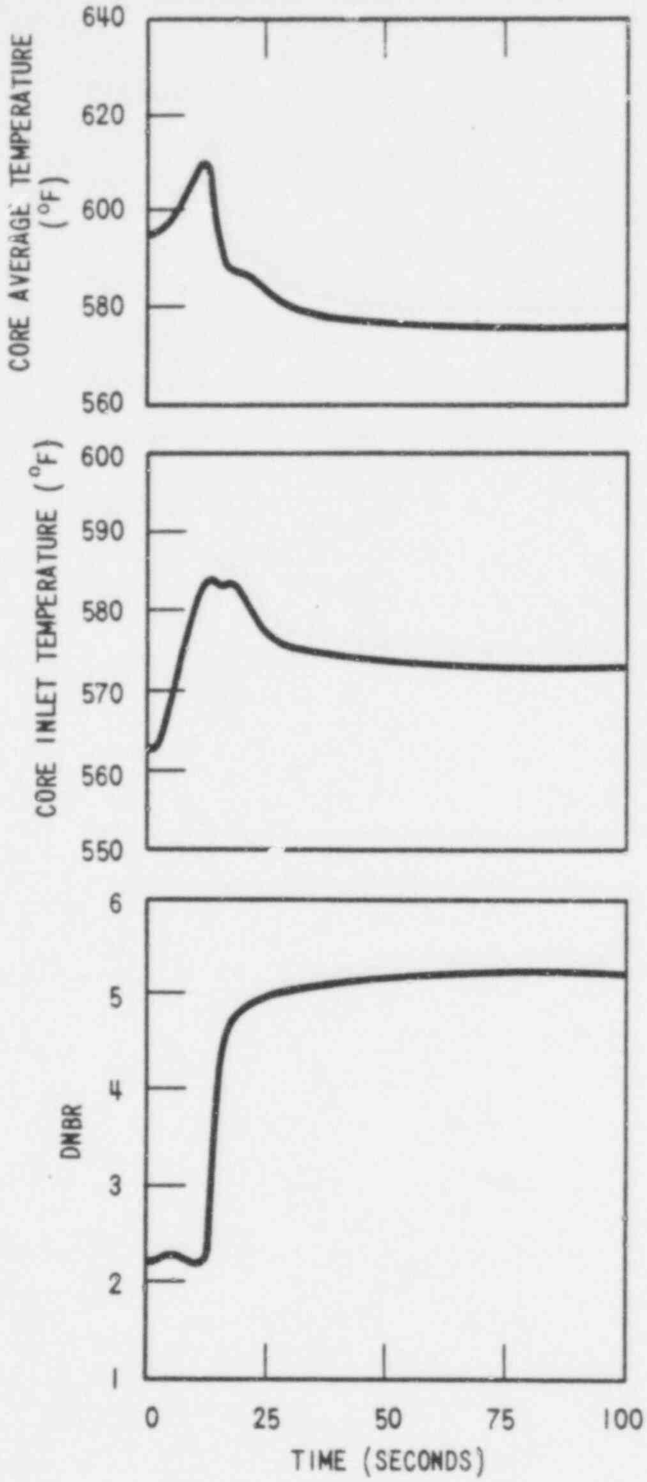

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Figure 15.2-2. Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

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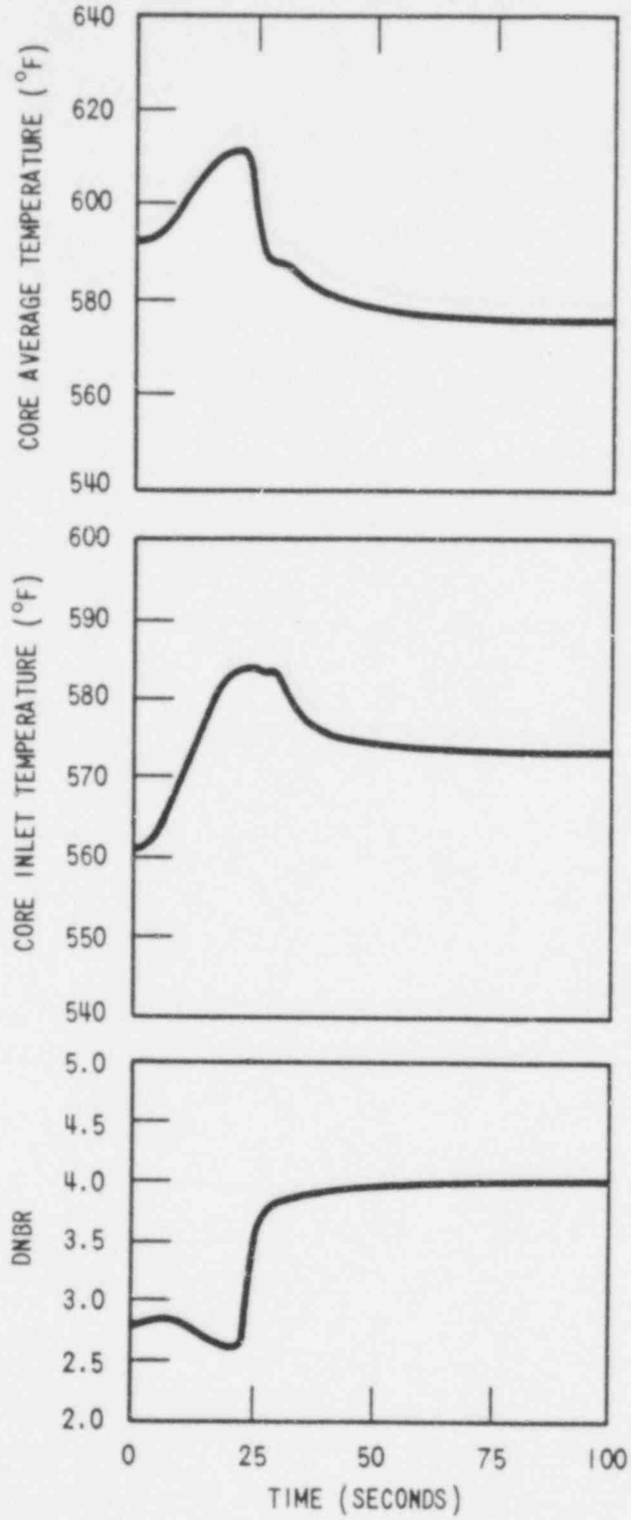

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Figure 15.2-2a.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback (N-1 Loop Operation)

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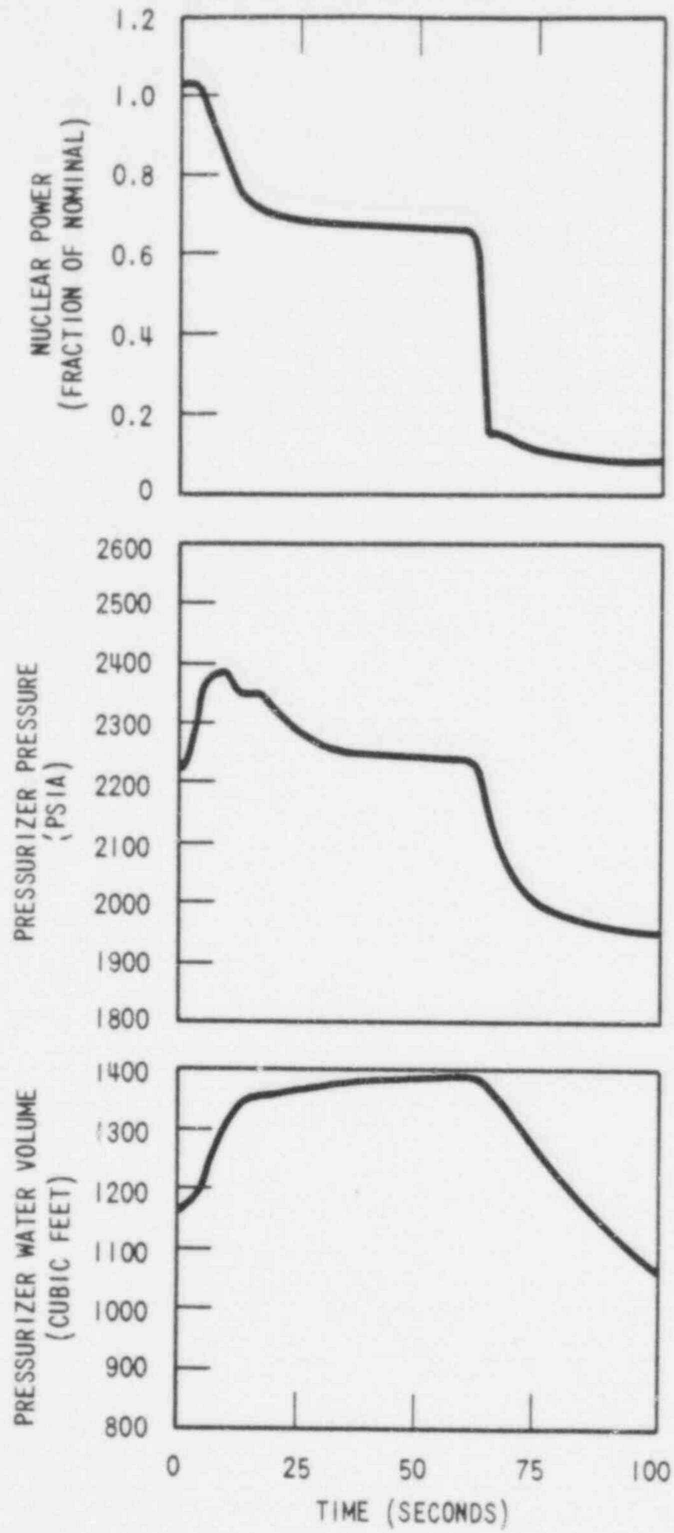

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Figure 15.2-3.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

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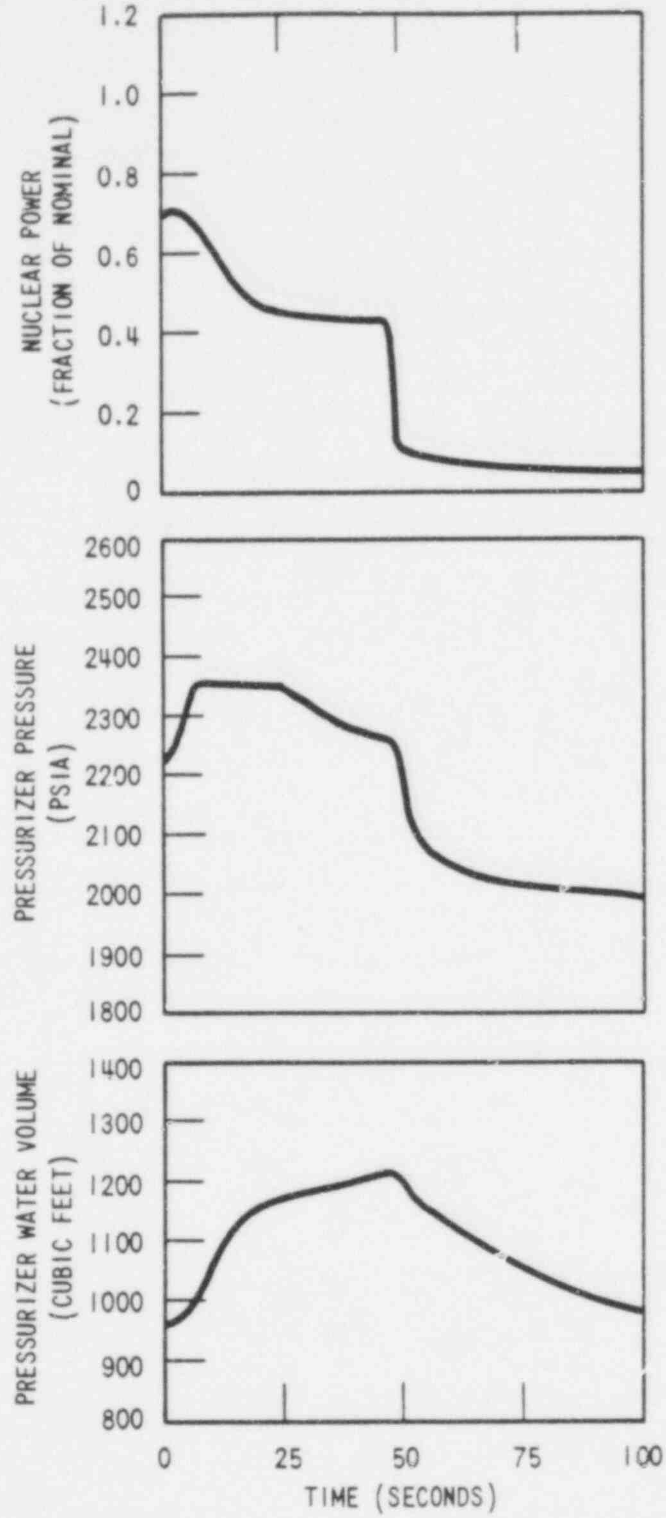

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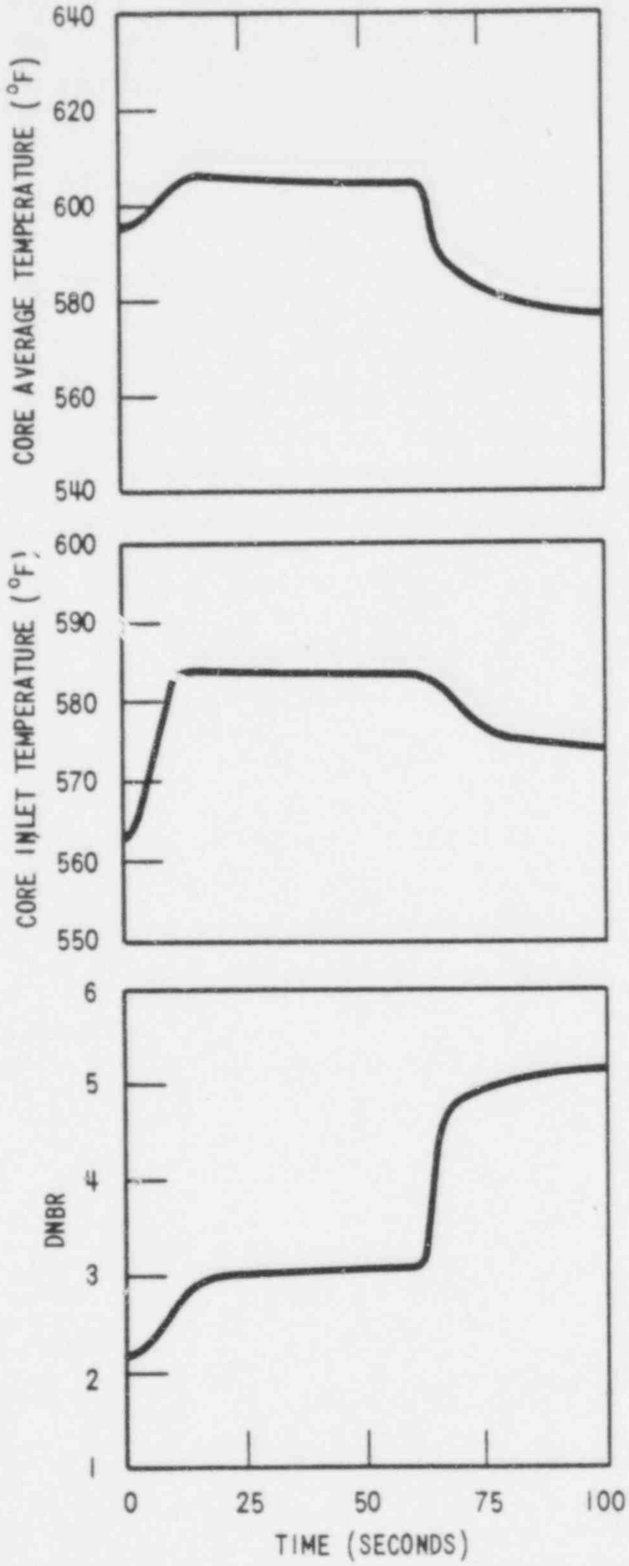
Figure 15.2-3a.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback (N-1 Loop Operation)

613 207

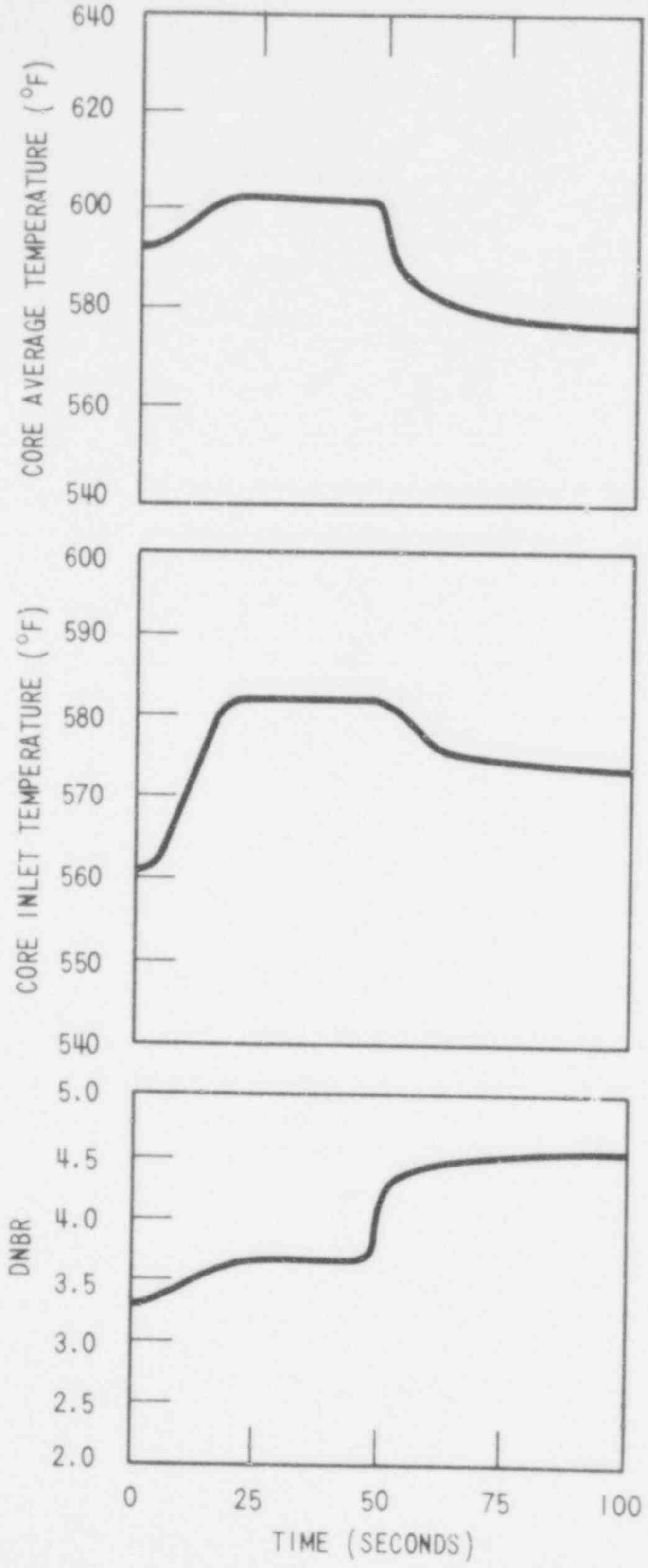


WCAP – 9500

Figure 15.2-4.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

613 288

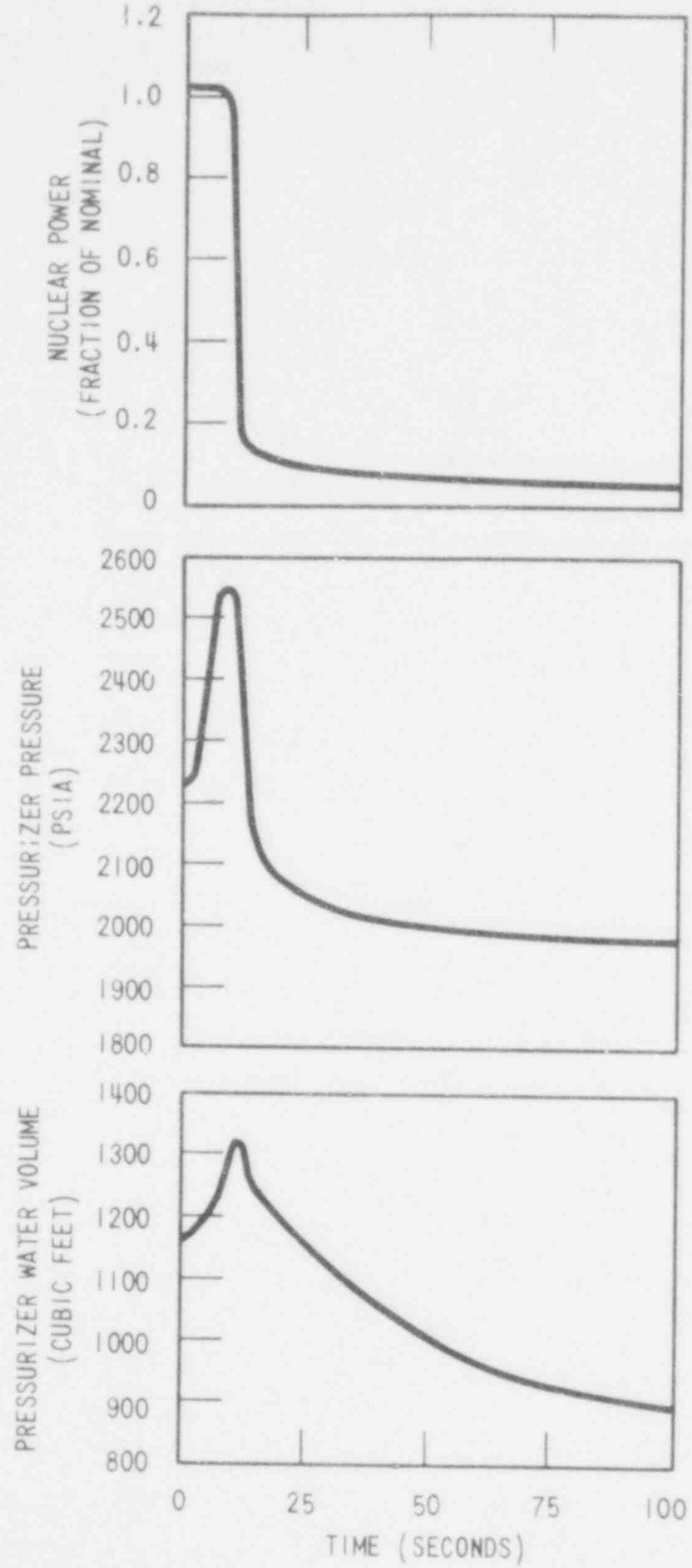


WCAP – 9500

Figure 15.2-4a.

Turbine Trip Accident With Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback (N-1 Loop Operation)

613 209

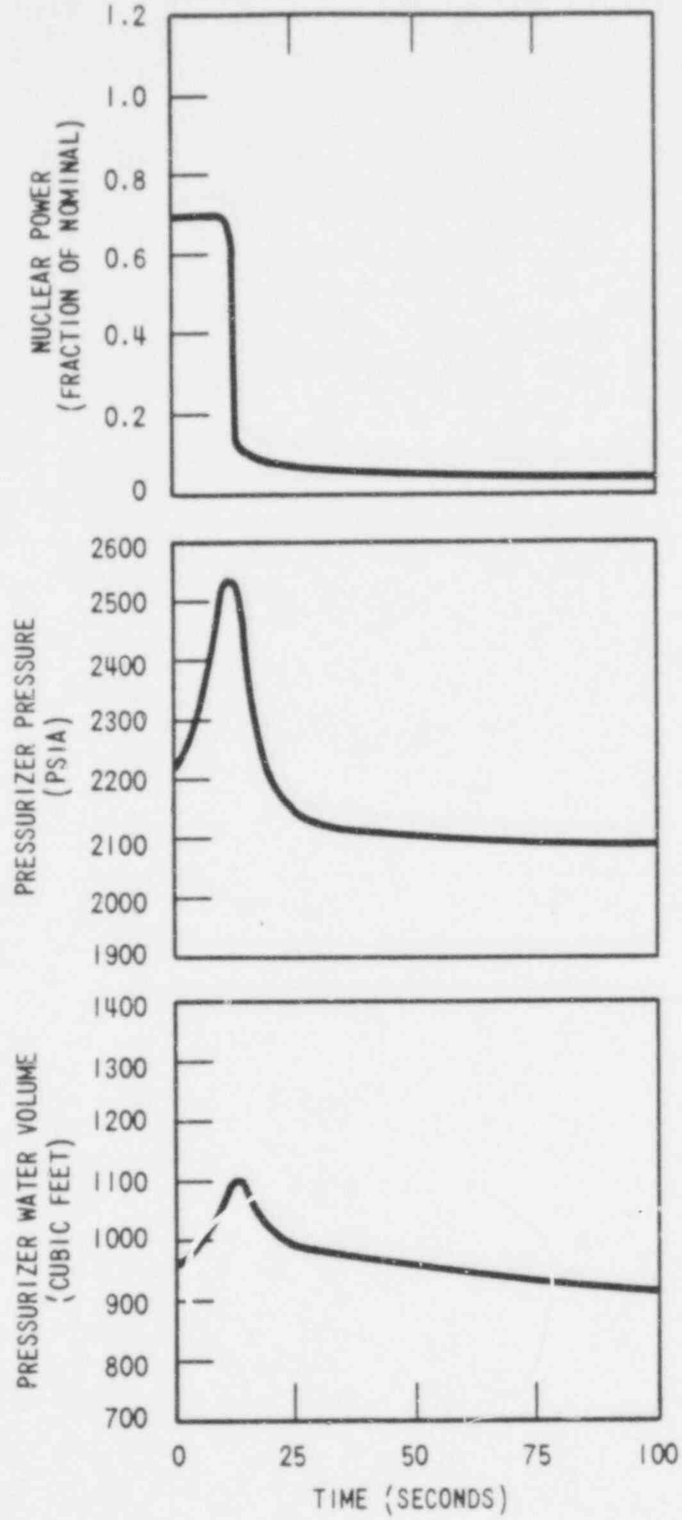


WCAP - 9500

Figure 15.2-5.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

613 210

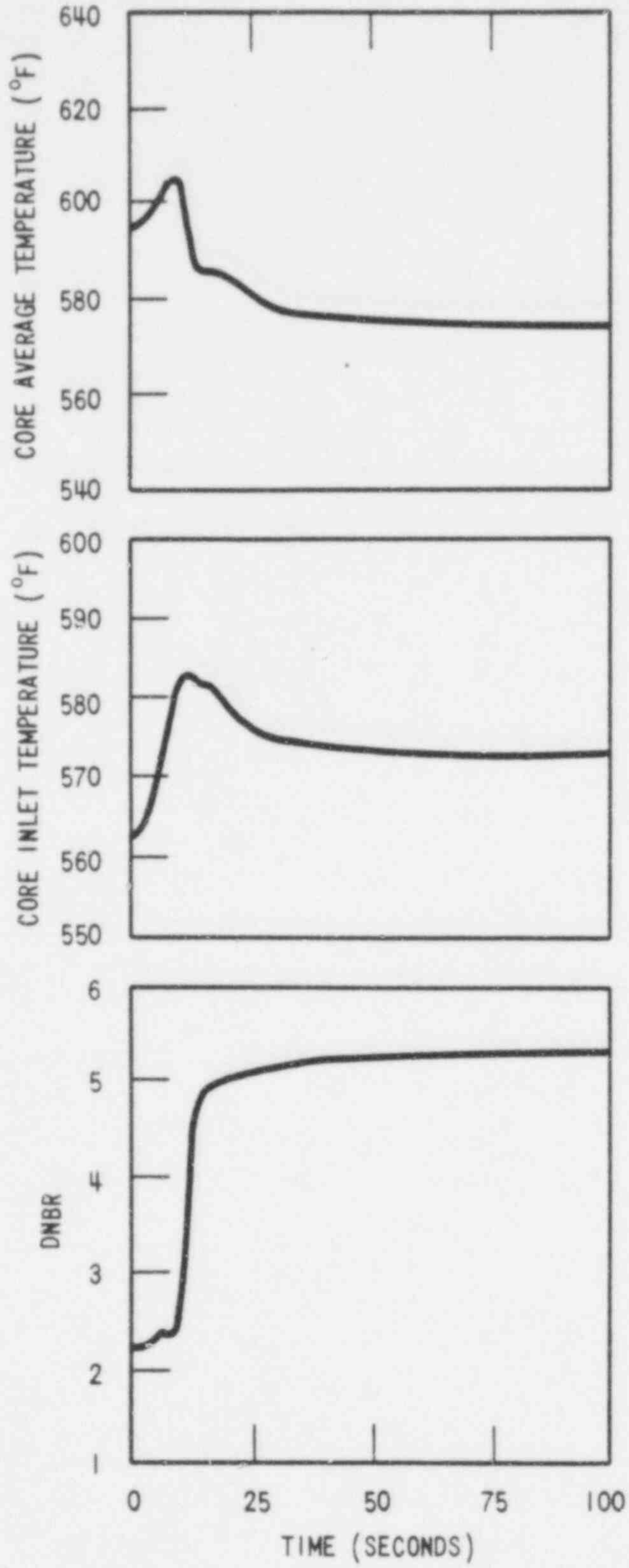


WCAP – 9500

Figure 15.2-5a.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback (N-1 Loop Operation)

613 211

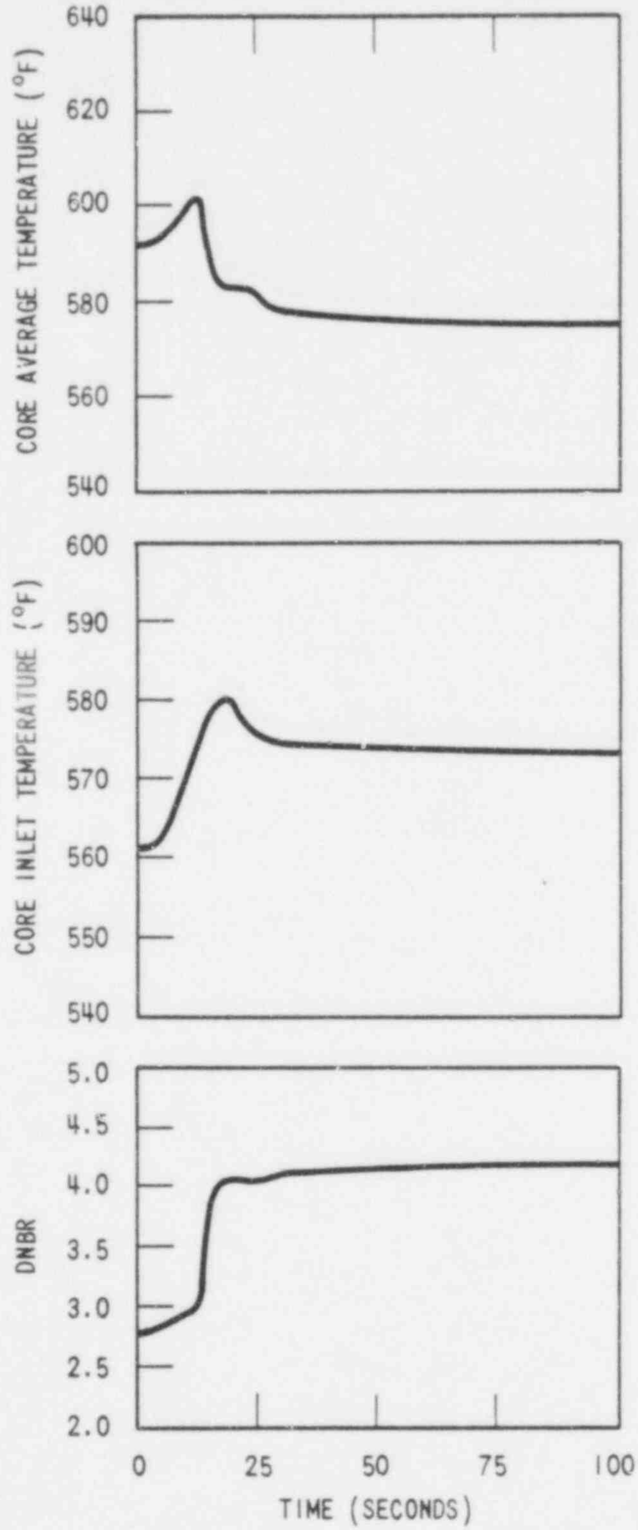


WCAP – 9500

Figure 15.2-6.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback

613 212

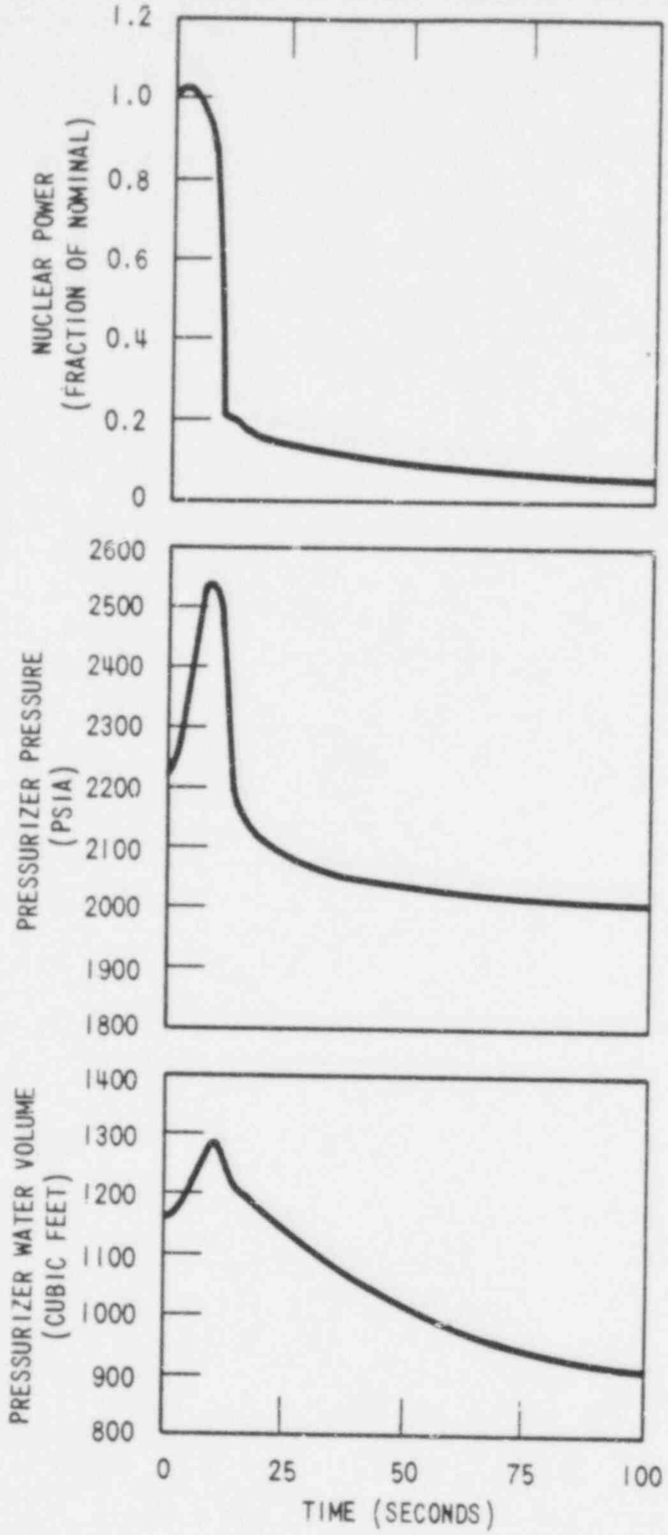


WCAP – 9500

Figure 15.2-6a.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Minimum Moderator Feedback (N-1 Loop Operation)

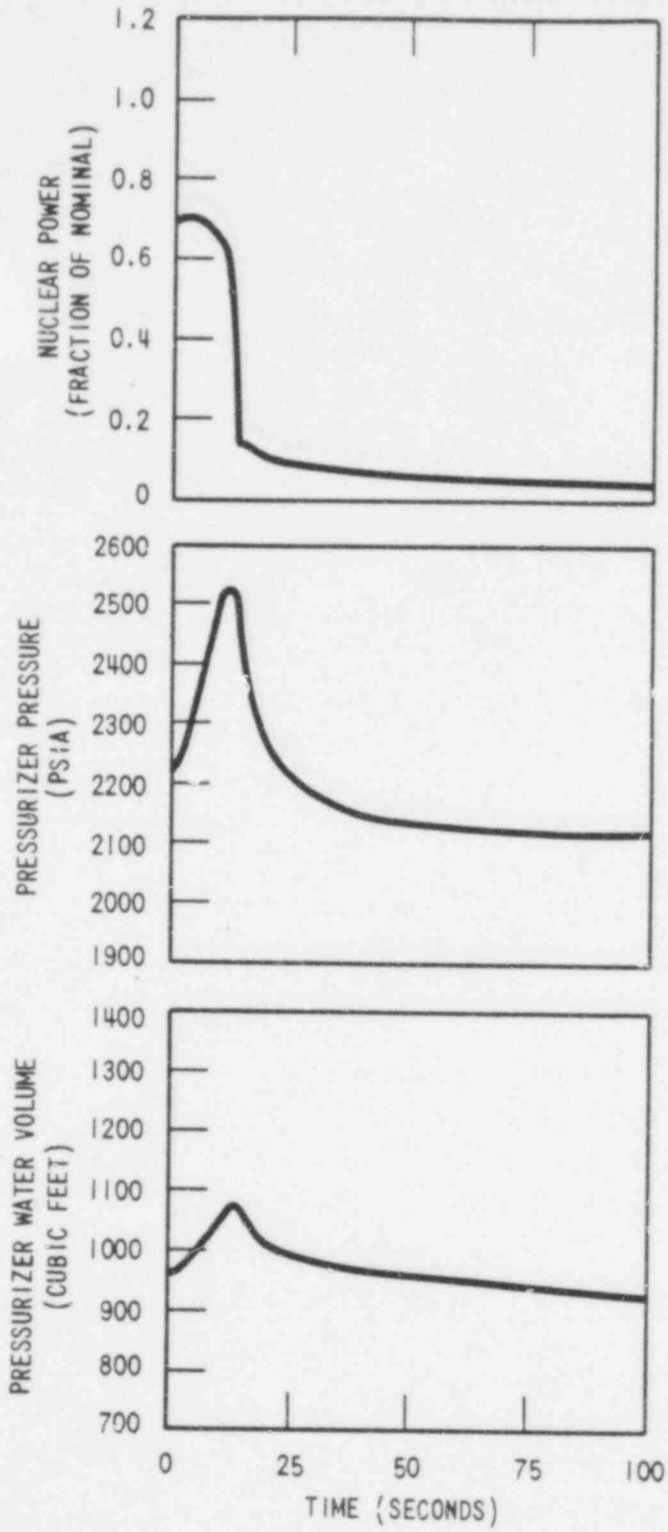
613 213



WCAP – 9500

Figure 15.2-7.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback



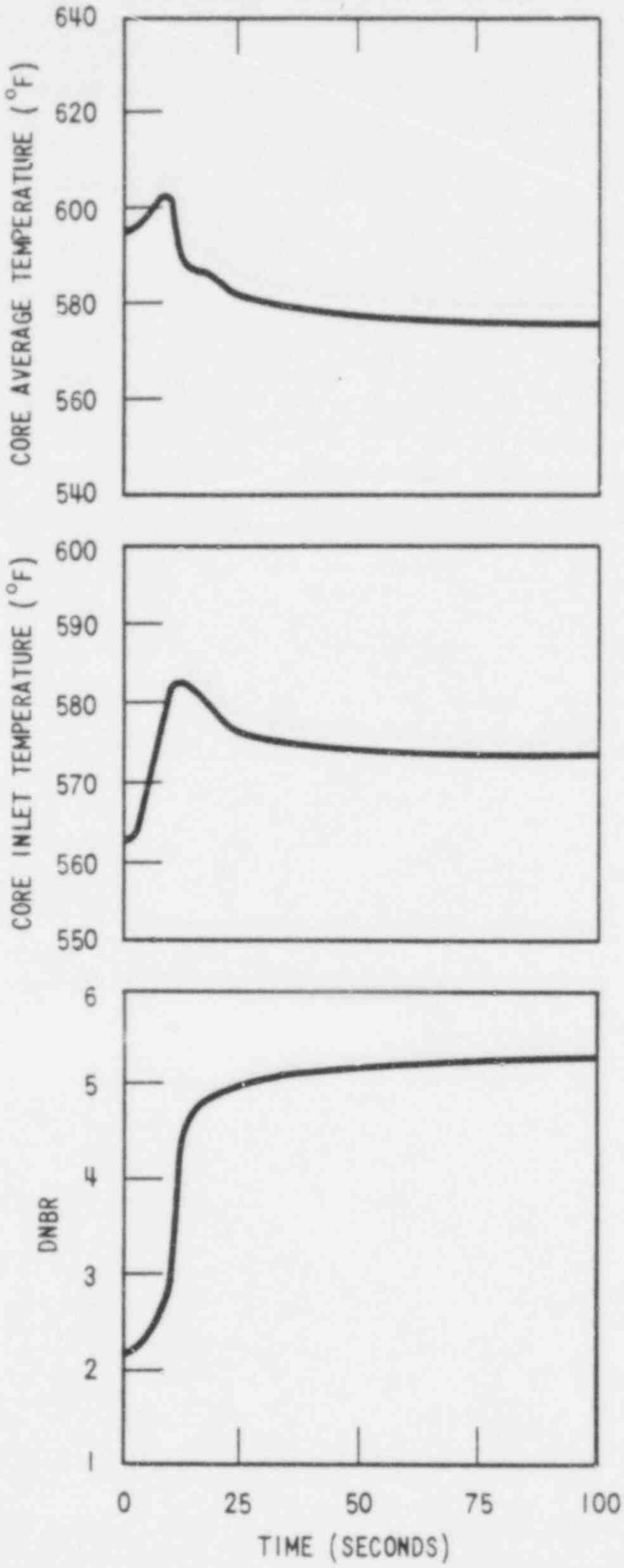
WCAP - 9500

Figure 15.2-7a.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback (N-1 Loop Operation)

613 215

14,395-67

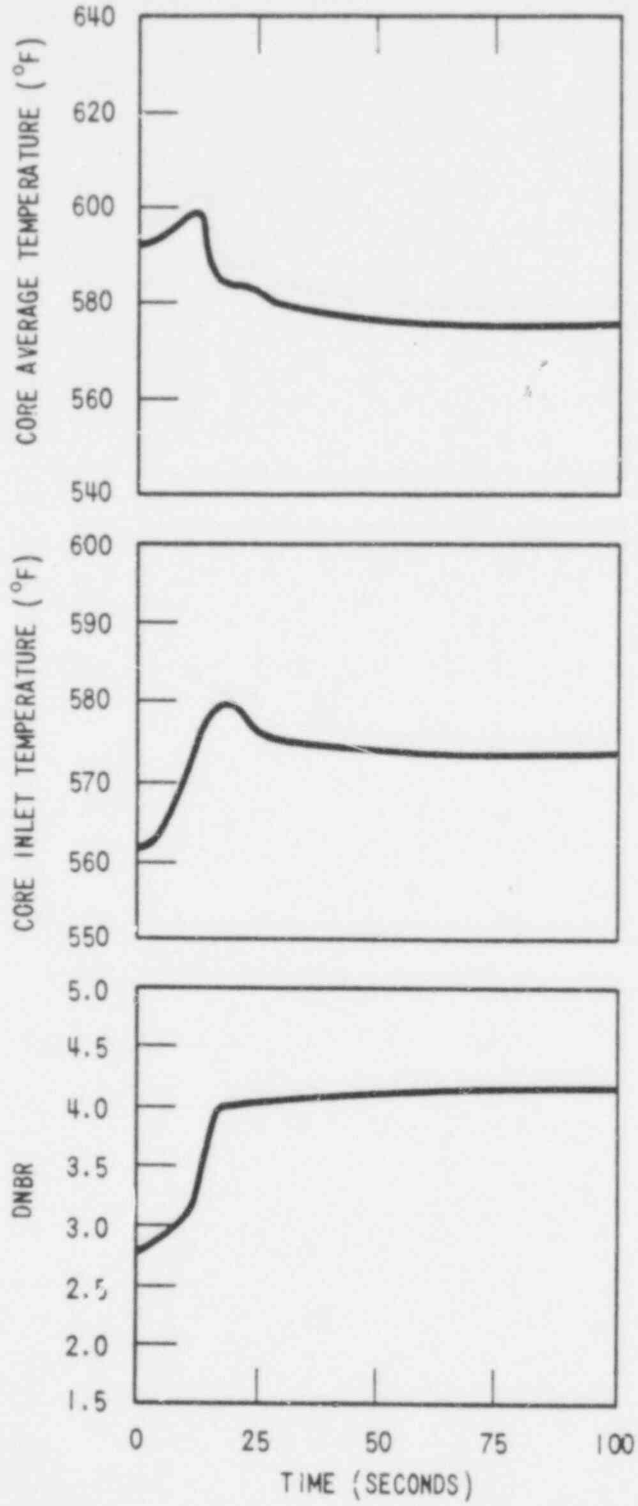


WCAP - 9500

Figure 15.2-8.

Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback

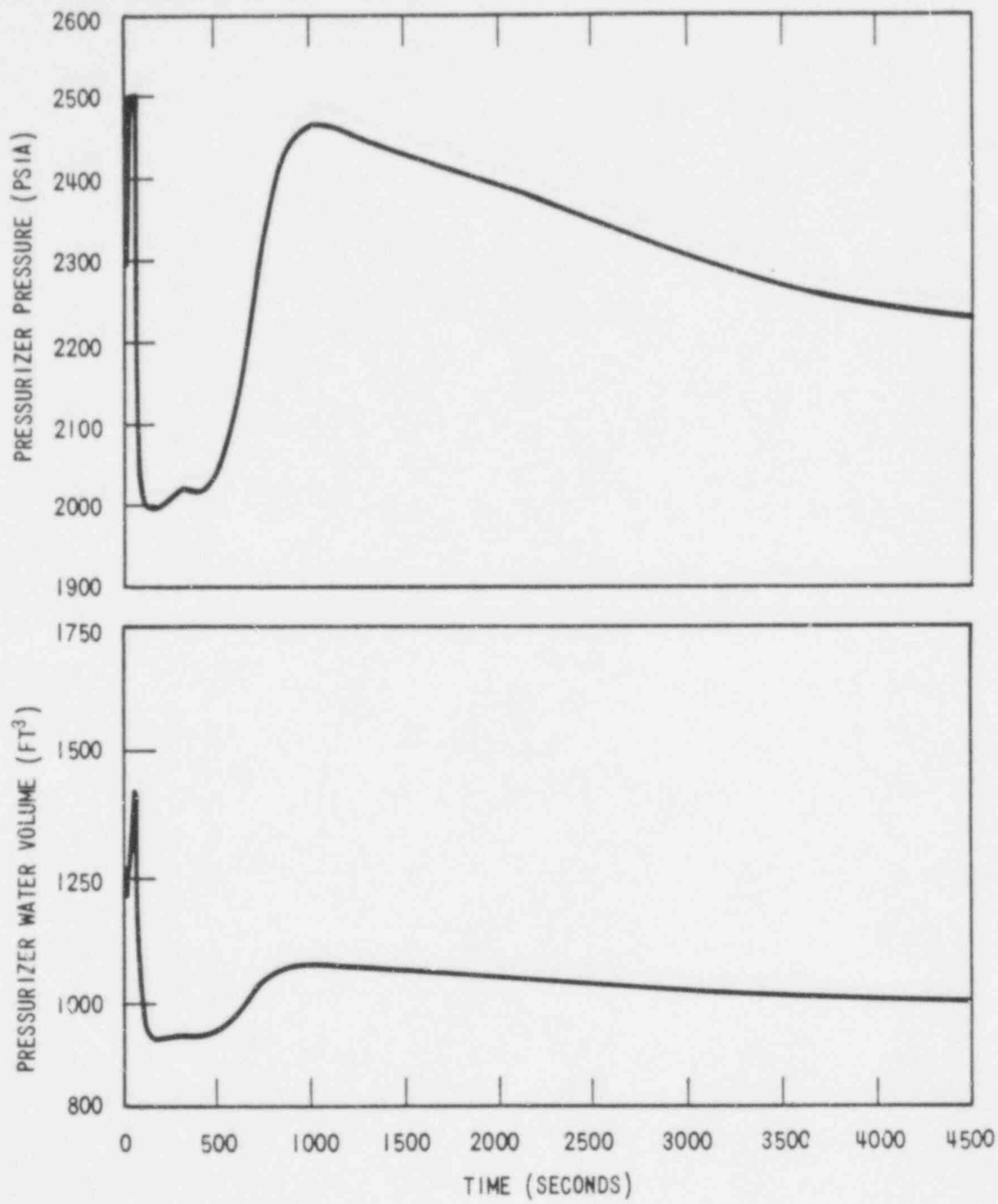
613 216



WCAP – 9500

Figure 15.2-8a.

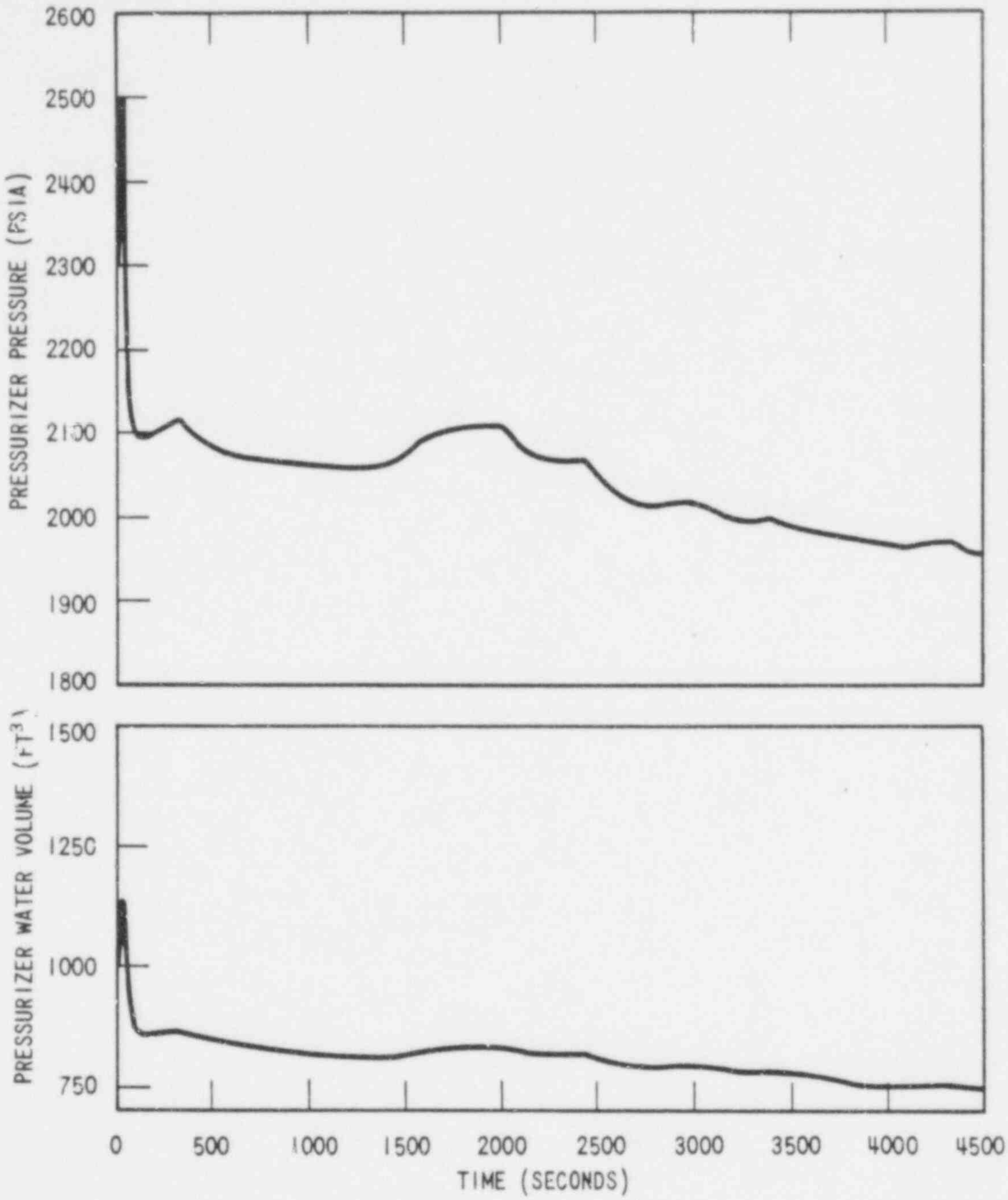
Turbine Trip Accident Without Pressurizer Spray and Power-Operated Relief Valves, Maximum Moderator Feedback (N-1 Loop Operation)



WCAP – 9500

Figure 15.2-9.

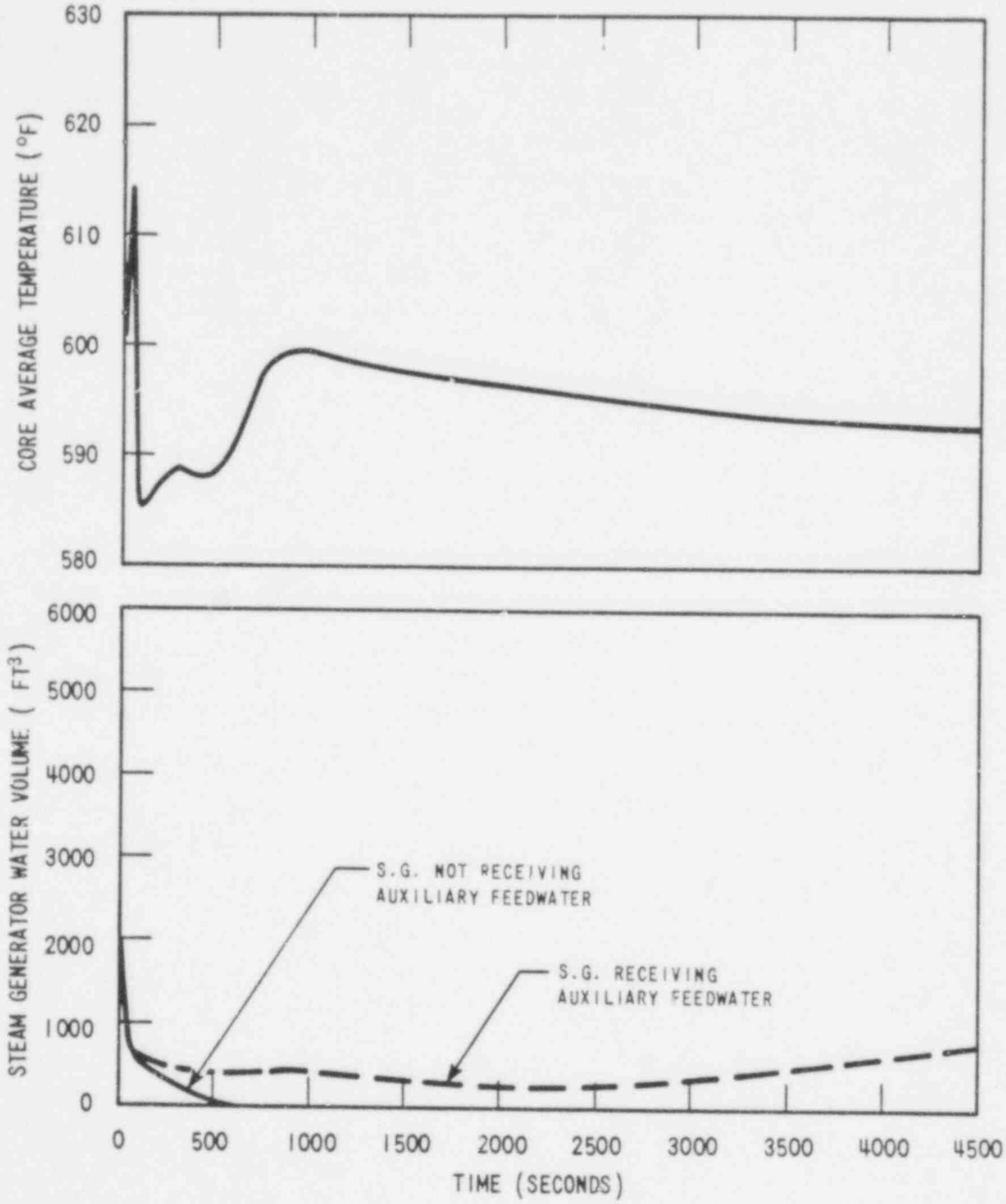
Pressurizer Pressure and Water Volume Transients for Loss of AC Power



WCAP-9500

Figure 15.2-9a.

Pressurizer Pressure and Water Volume Transients for Loss of AC Power (N-1 Loop Operation)

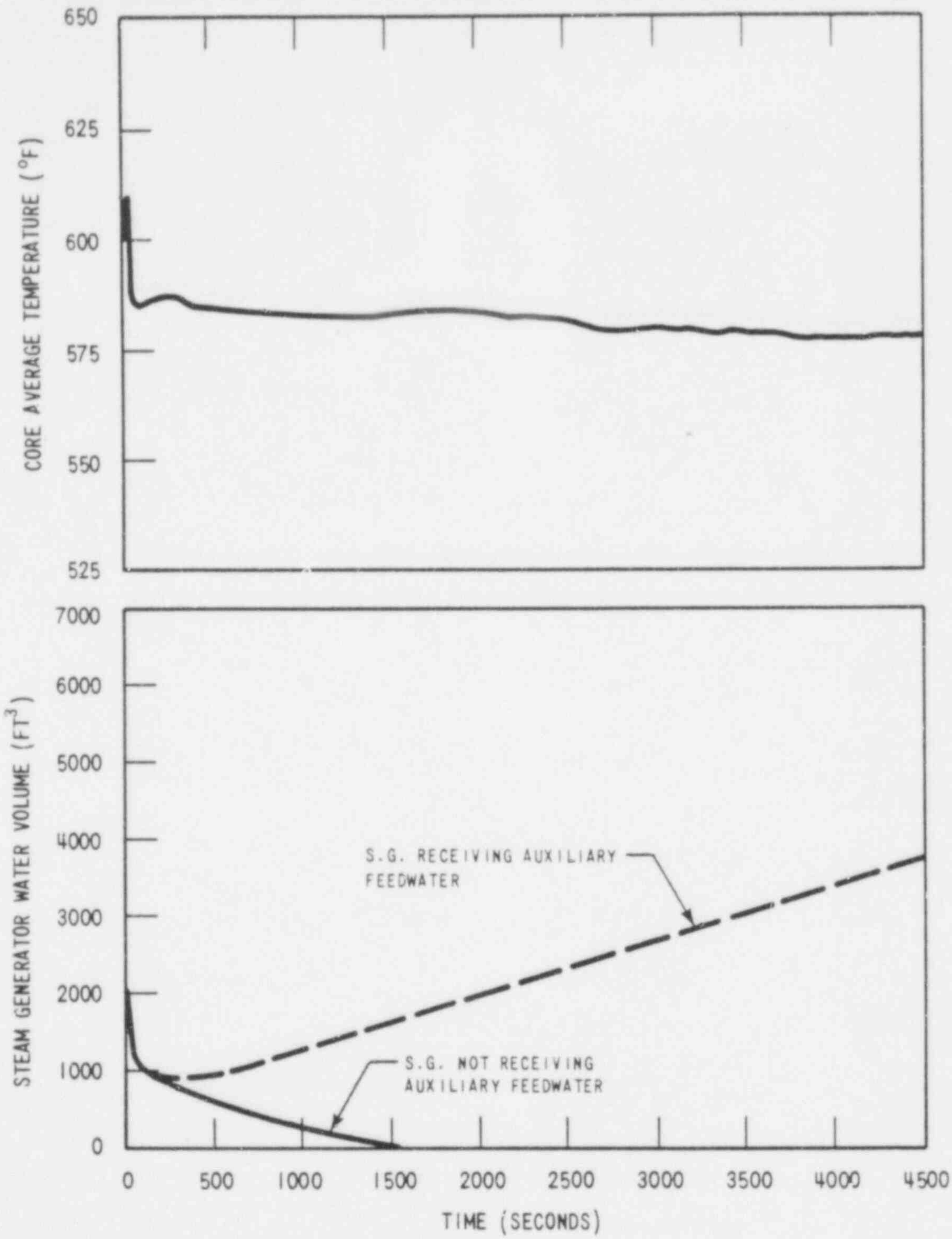


WCAP - 9500

Figure 15.2-10.

Core Average Temperature Transient and Steam Generator Water Volume Transient for Loss of AC Power

14,395-79

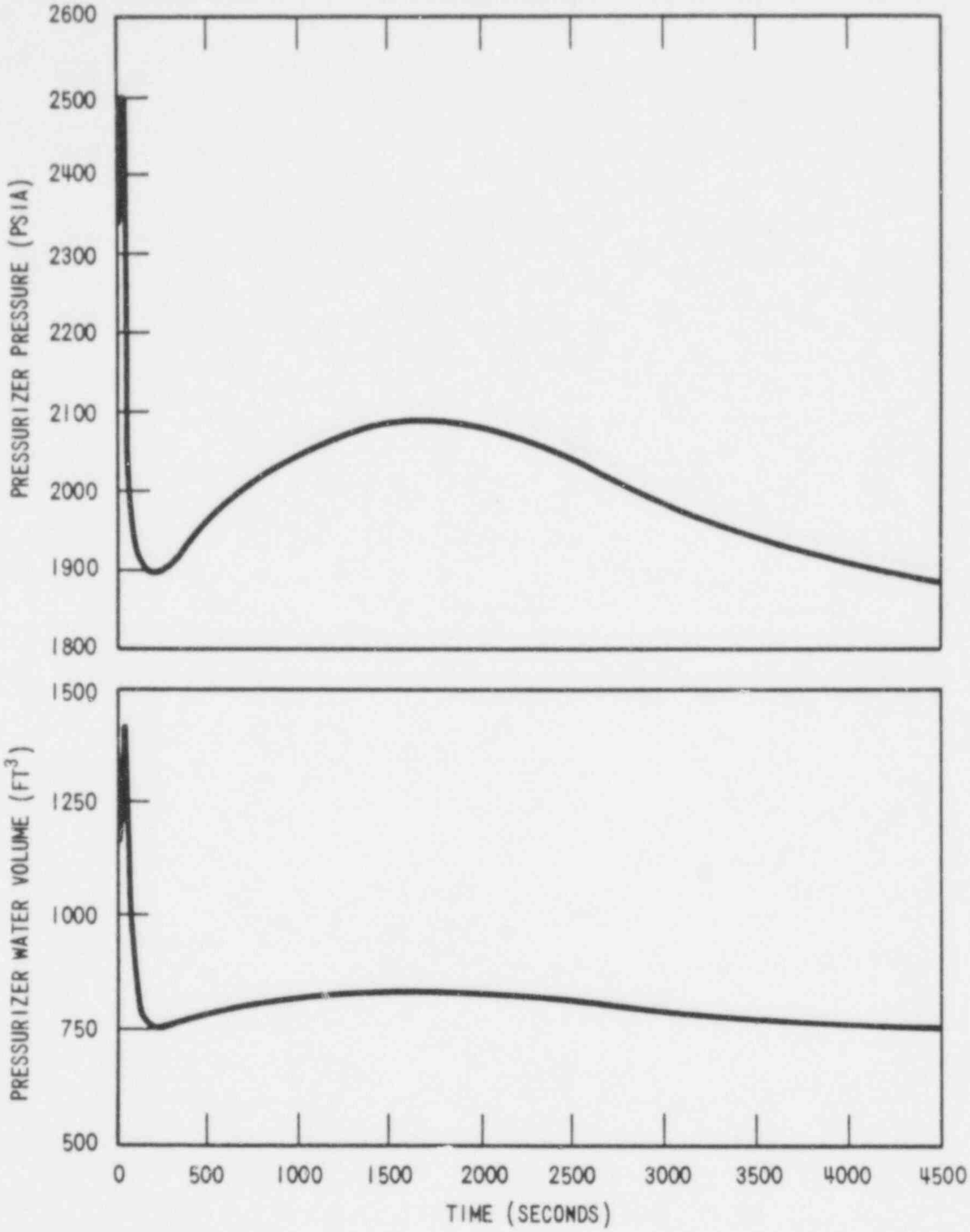


WCAP – 9500

Figure 15.2-10a.

Core Average Temperature Transient and Steam Generator Water Volume Transient for Loss of AC Power (N-1 Loop Operation)

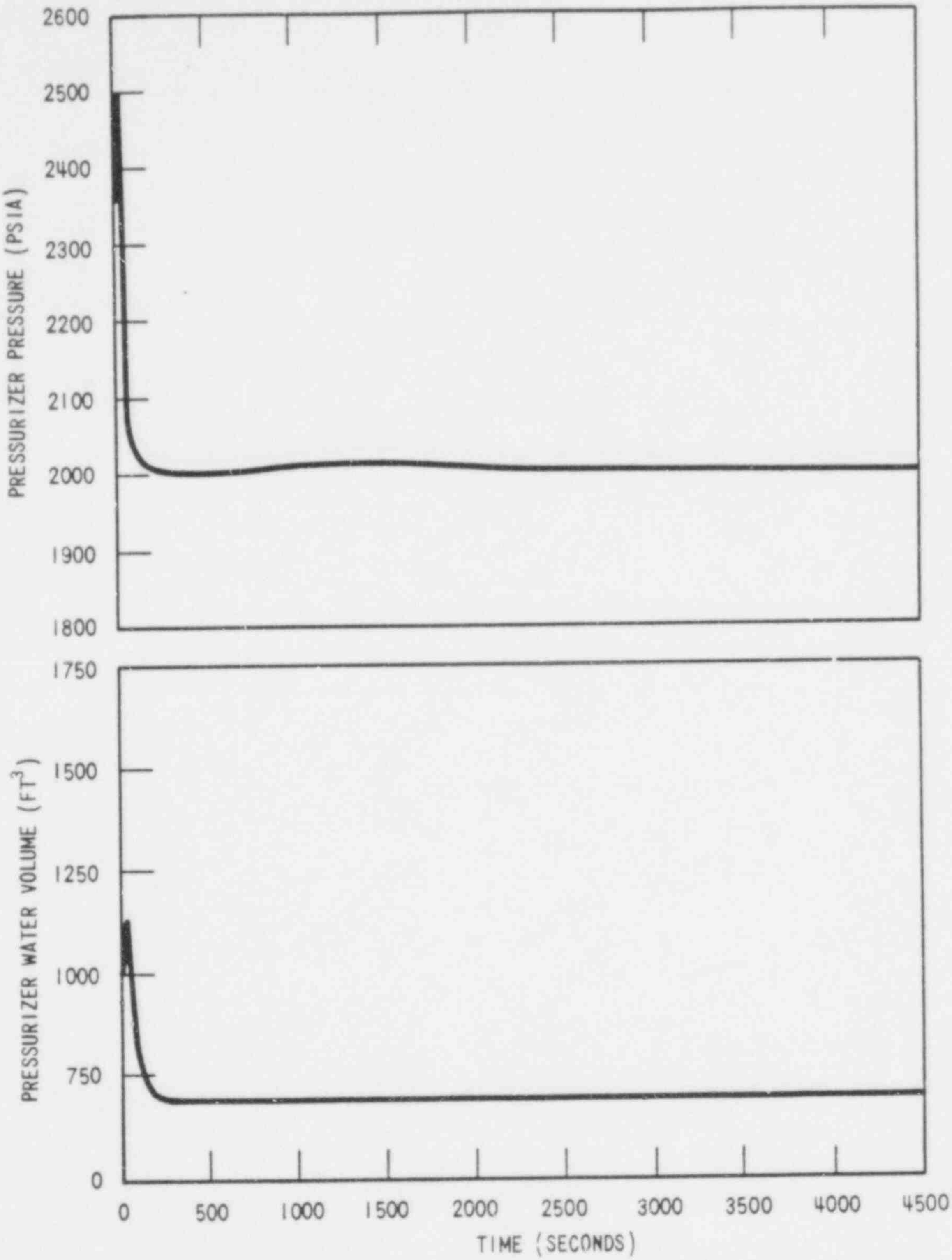
613 221



WCAP - 9500

Figure 15.2-11.

Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater



WCAP - 9500

Figure 15.2-11a.

Pressurizer Pressure and Water Volume Transients for Loss of Normal Feedwater (N-1 Loop Operation)

14,395-81

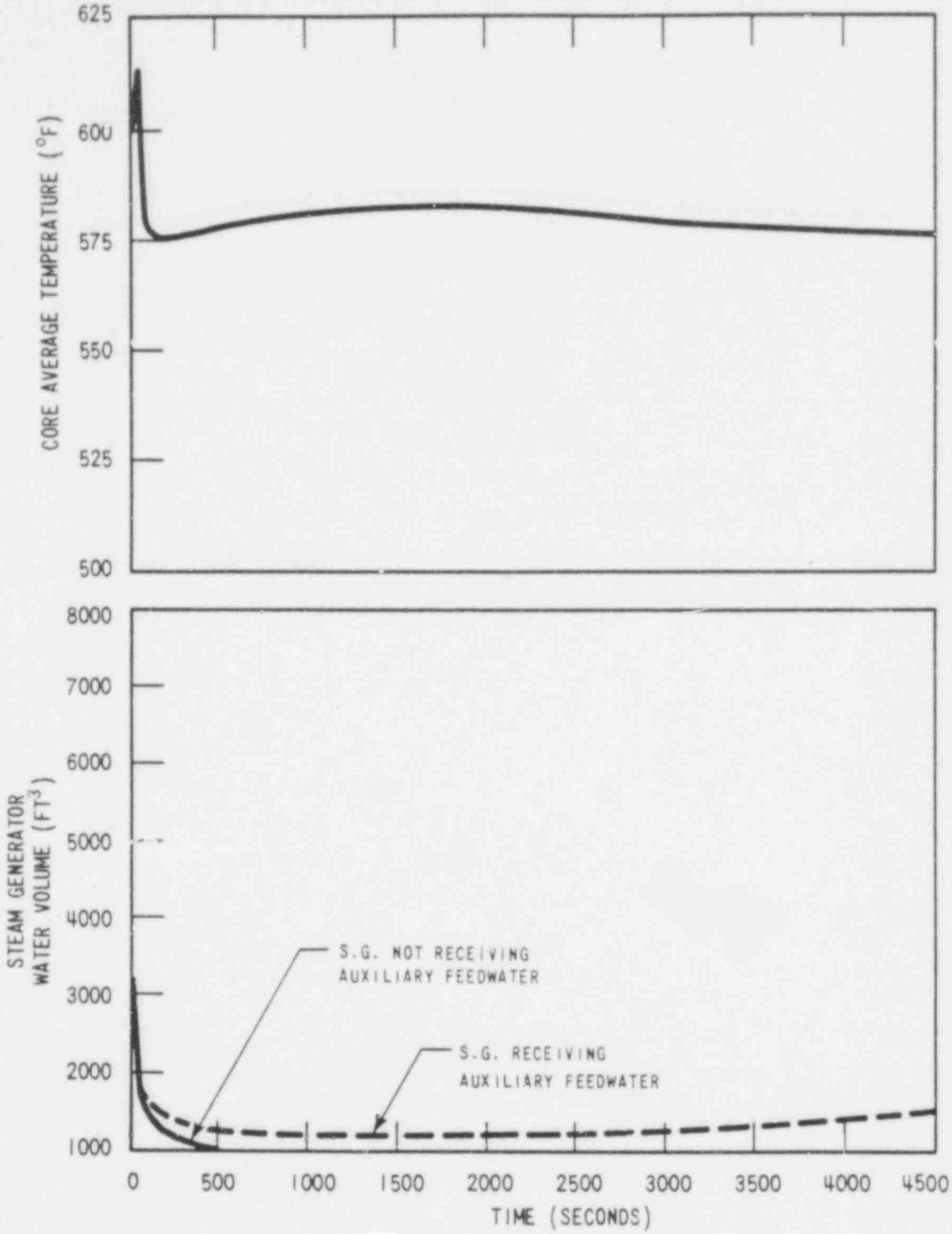


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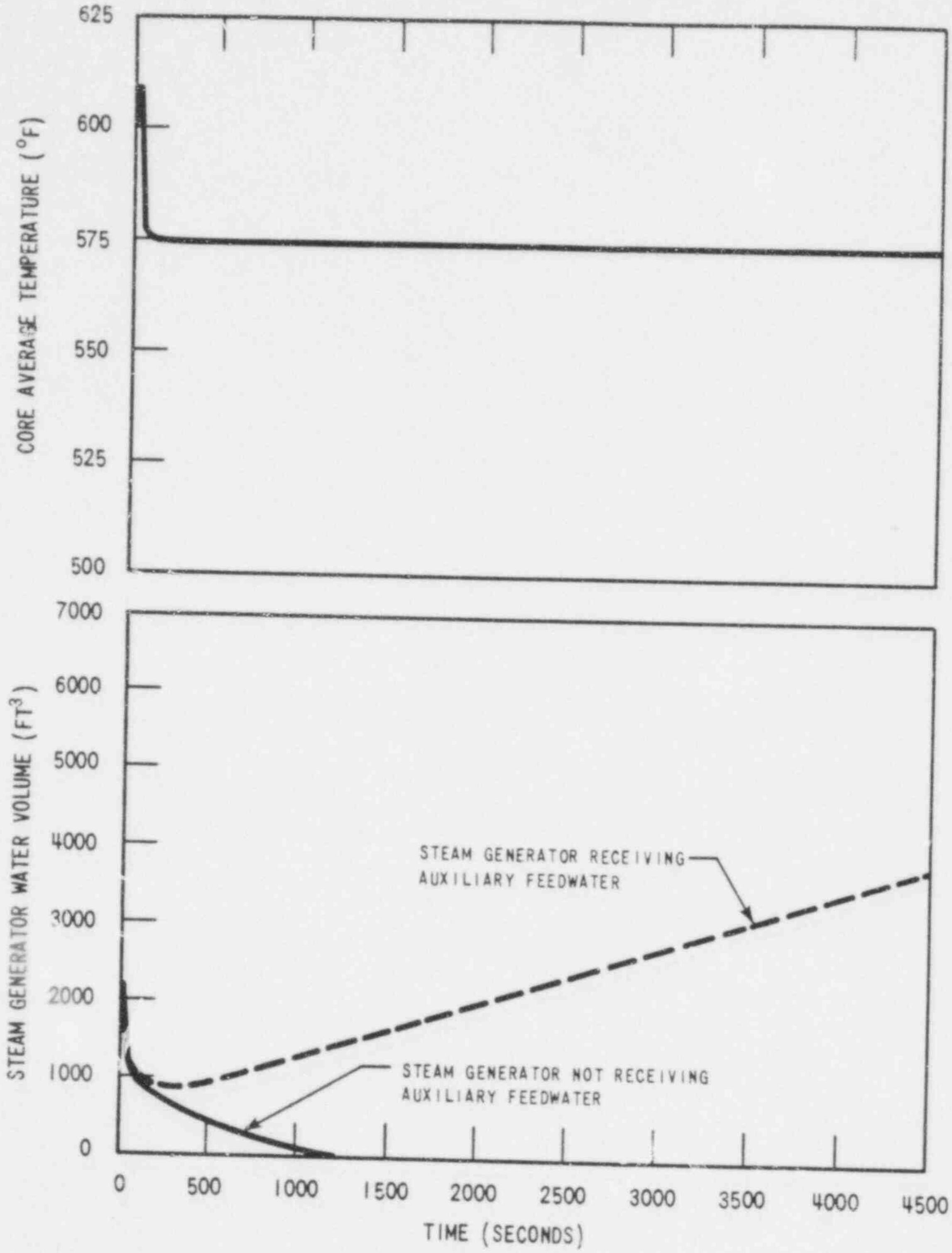
Figure 15.2-12.

Core Average Temperature Transient and Steam Generator Water Volume Transient for Loss of Normal Feedwater

6/13 224



WCAP – 9500
Figure 15.2-12a. Core Average Temperature Transient and Steam Generator Water Volume Transient for Loss of Normal Feedwater (N-1 Loop Operation)

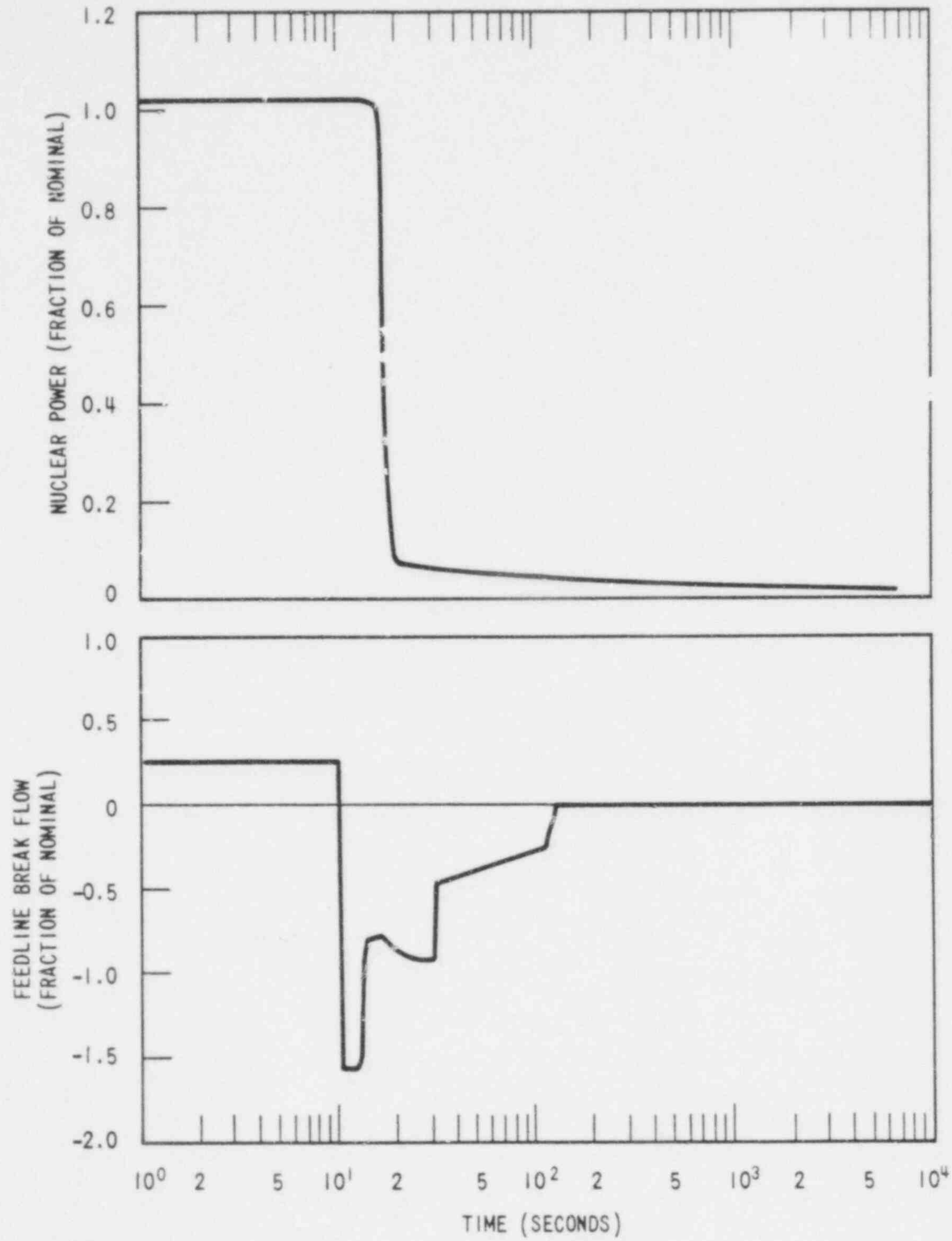


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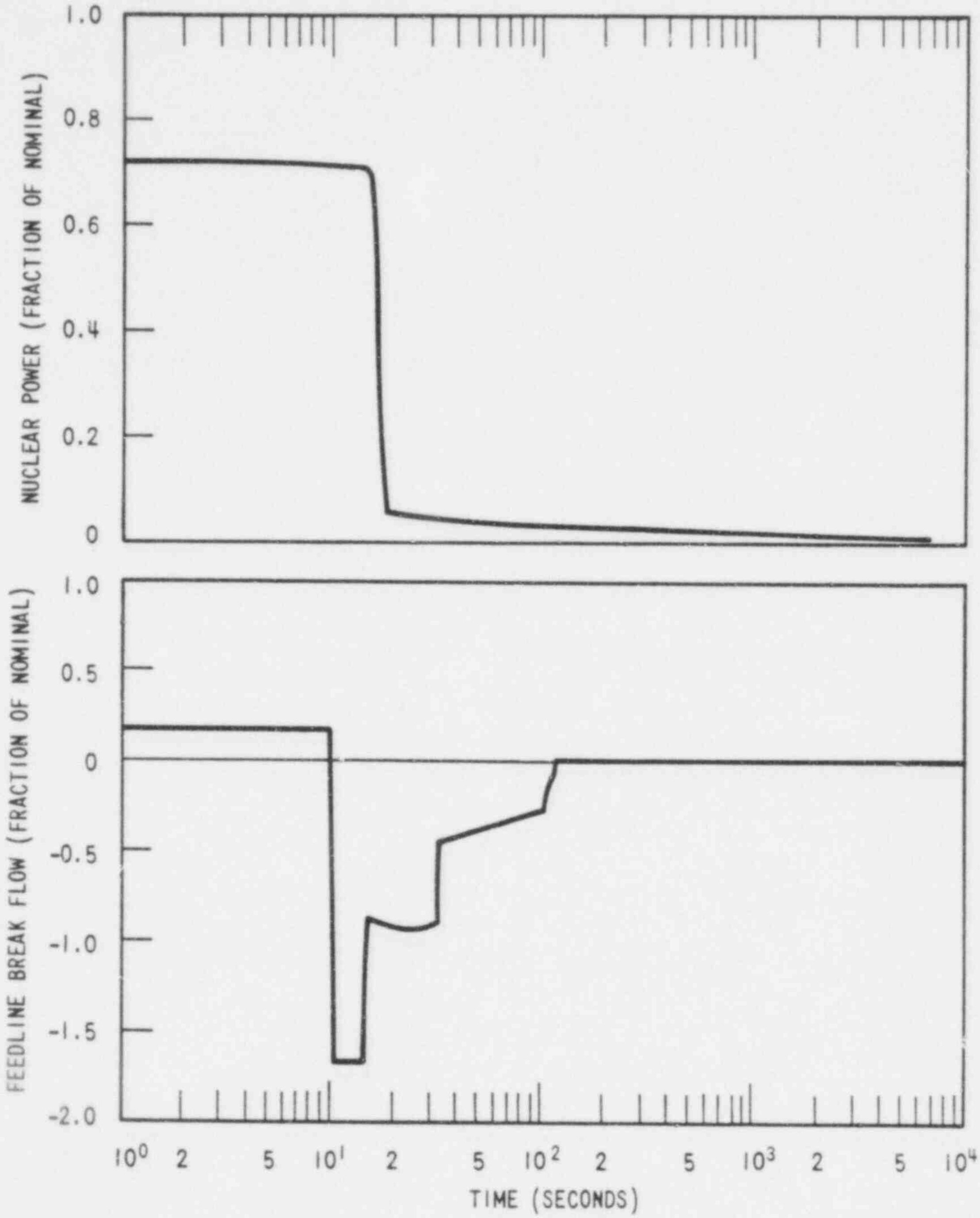
Figure 15.2-13.

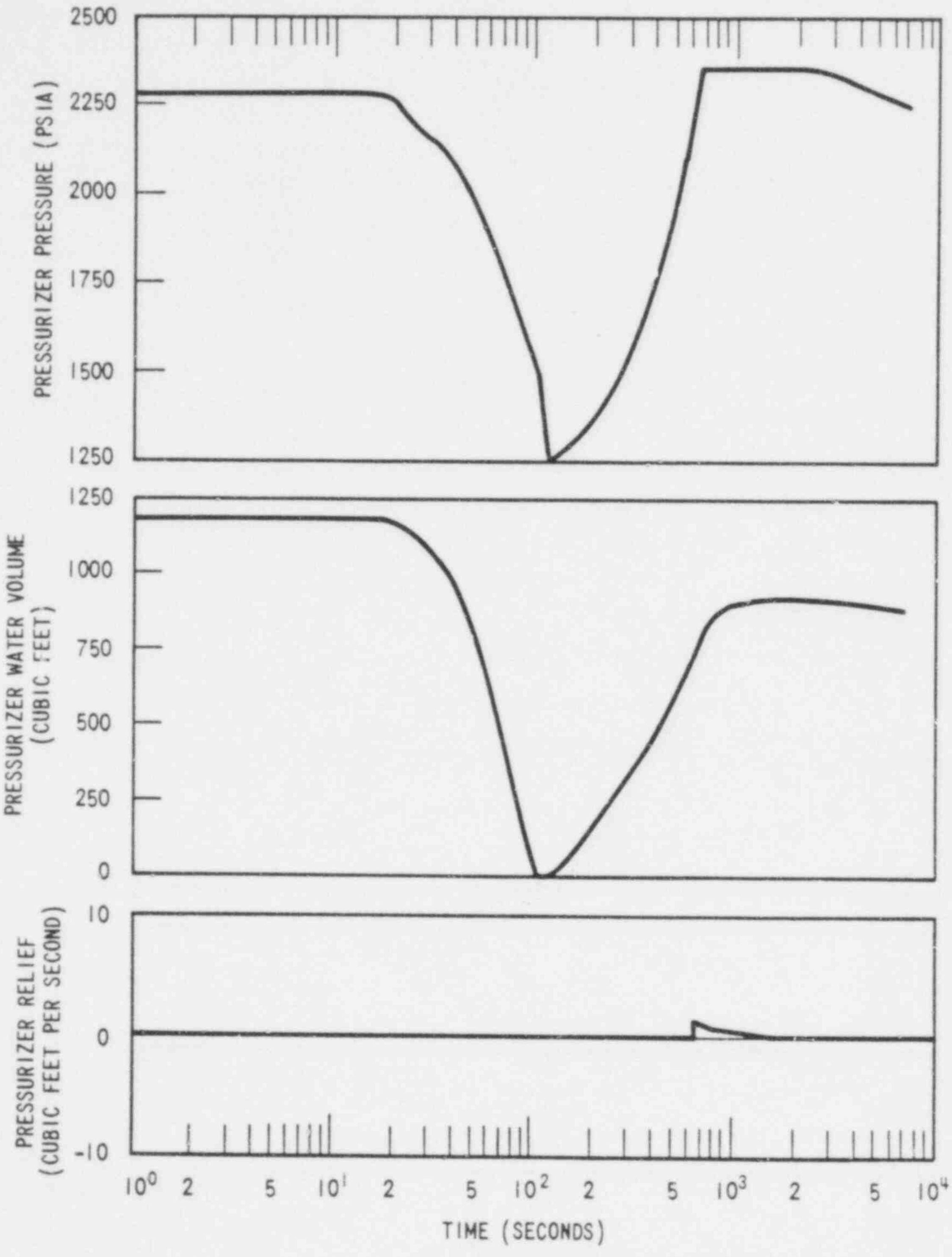
Nuclear Power Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available



WCAP-9500

Figure 15.2-13a.

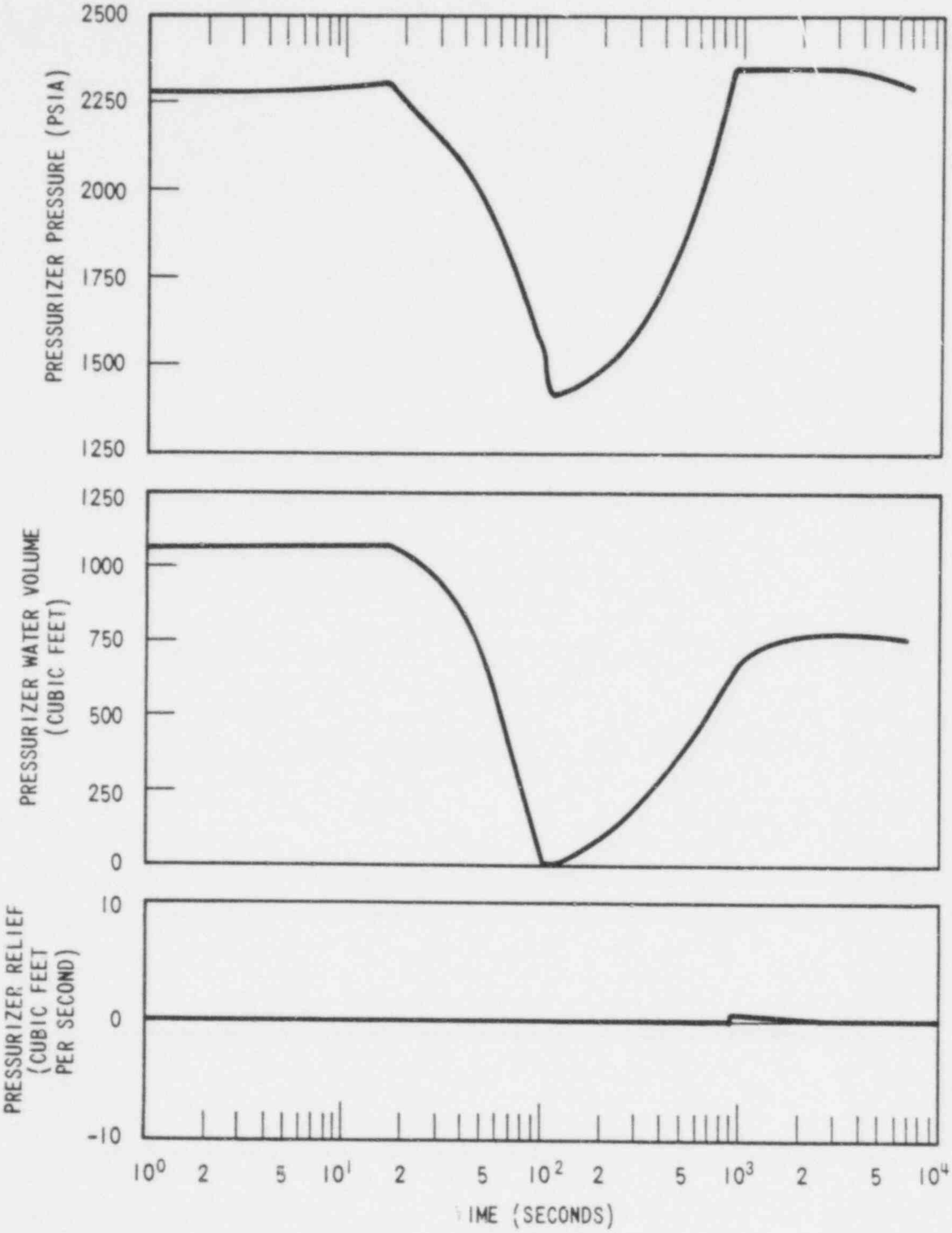
Nuclear Power Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)



WCAP – 9500

Figure 15.2-14.

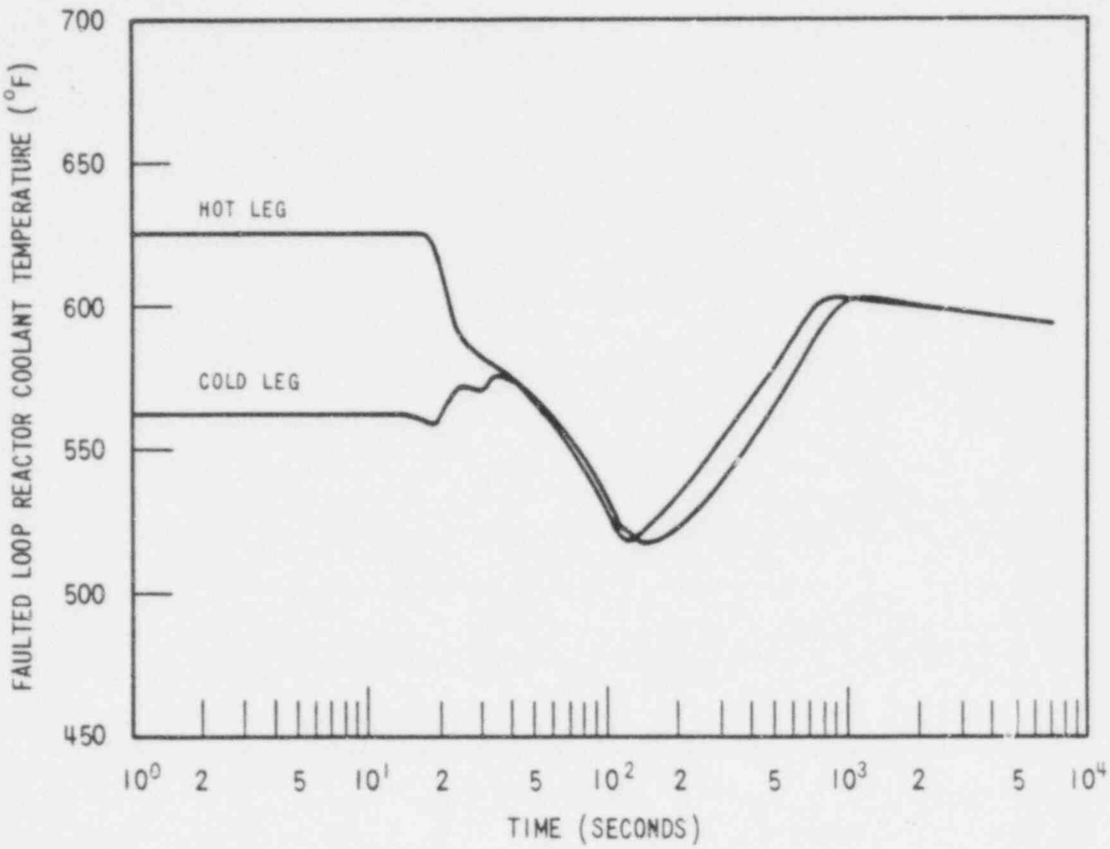
Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available



WCAP – 9500

Figure 15.2-14a.

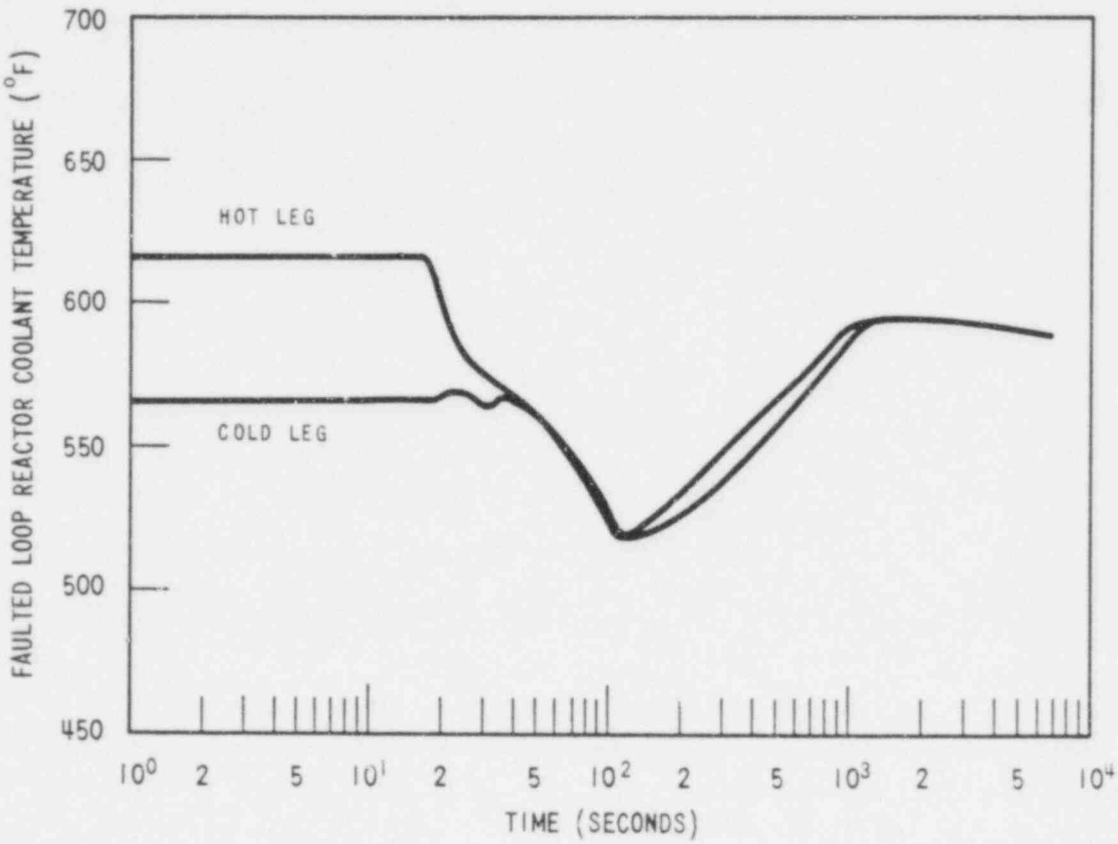
Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)



WCAP – 9500

Figure 15.2-15.

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture With Offsite Power Available

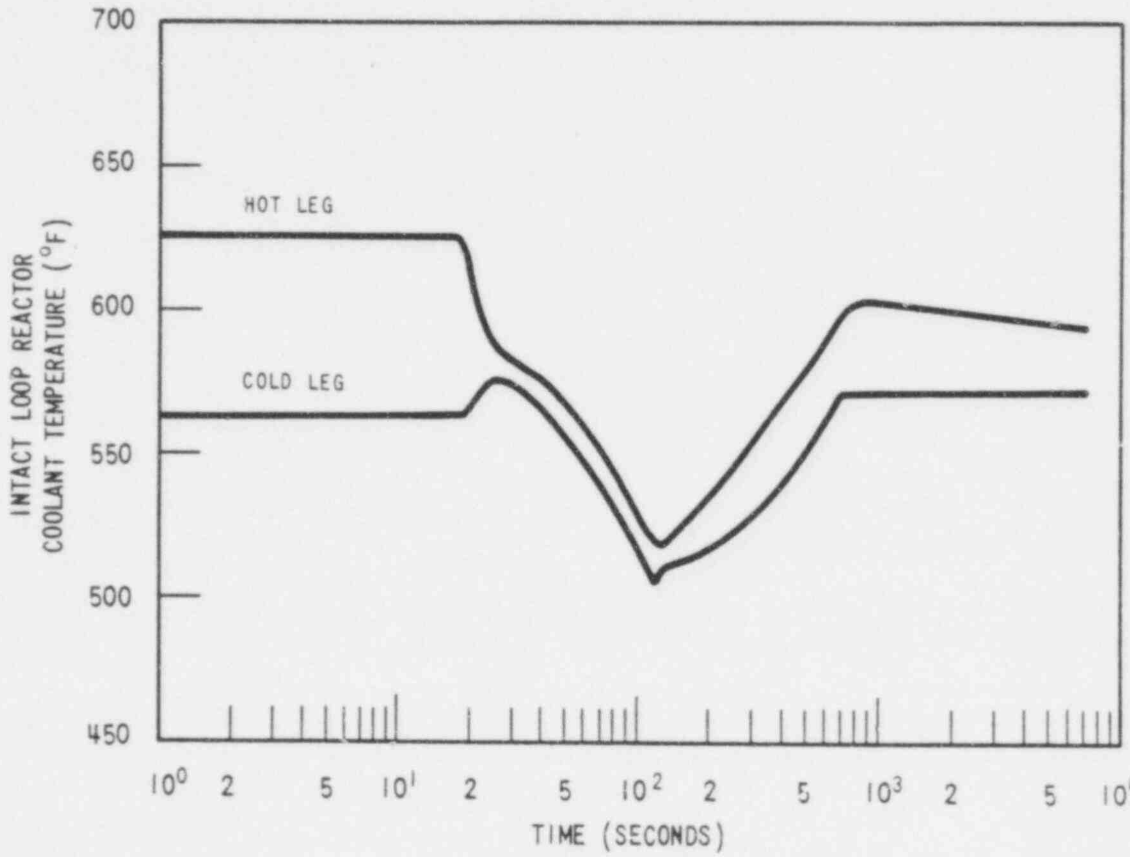


613 231

WCAP - 9500

Figure 15.2-15a.

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)



61 232

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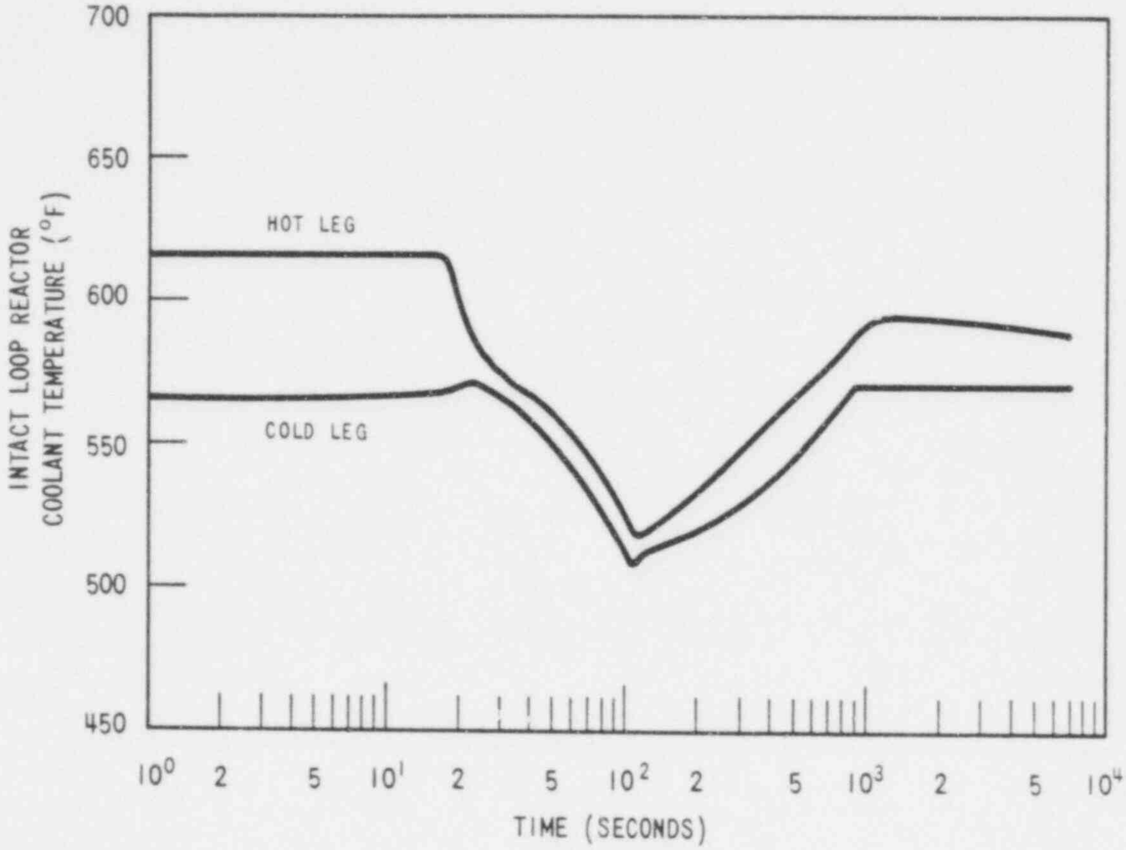
Figure 15.2-16.

Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture With Offsite Power Available



613 233

WCAP - 9500

Figure 15.2-16a.

Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)

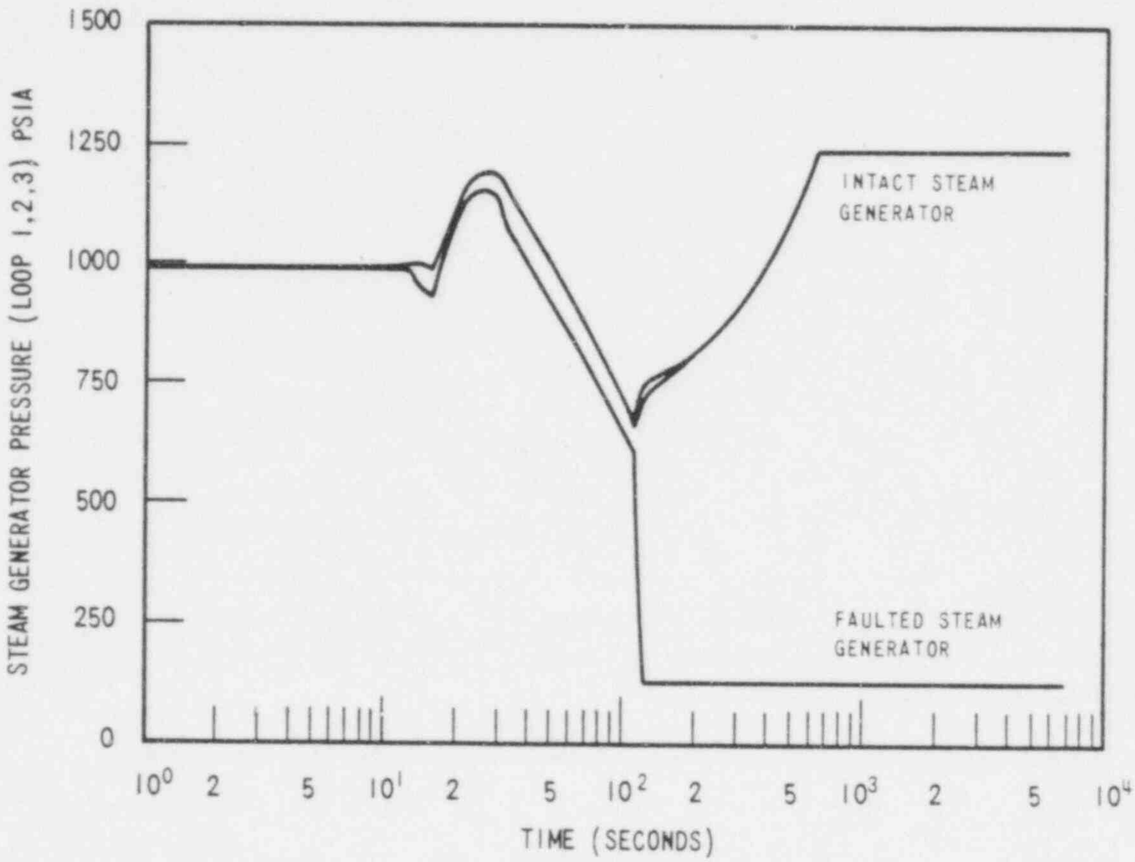


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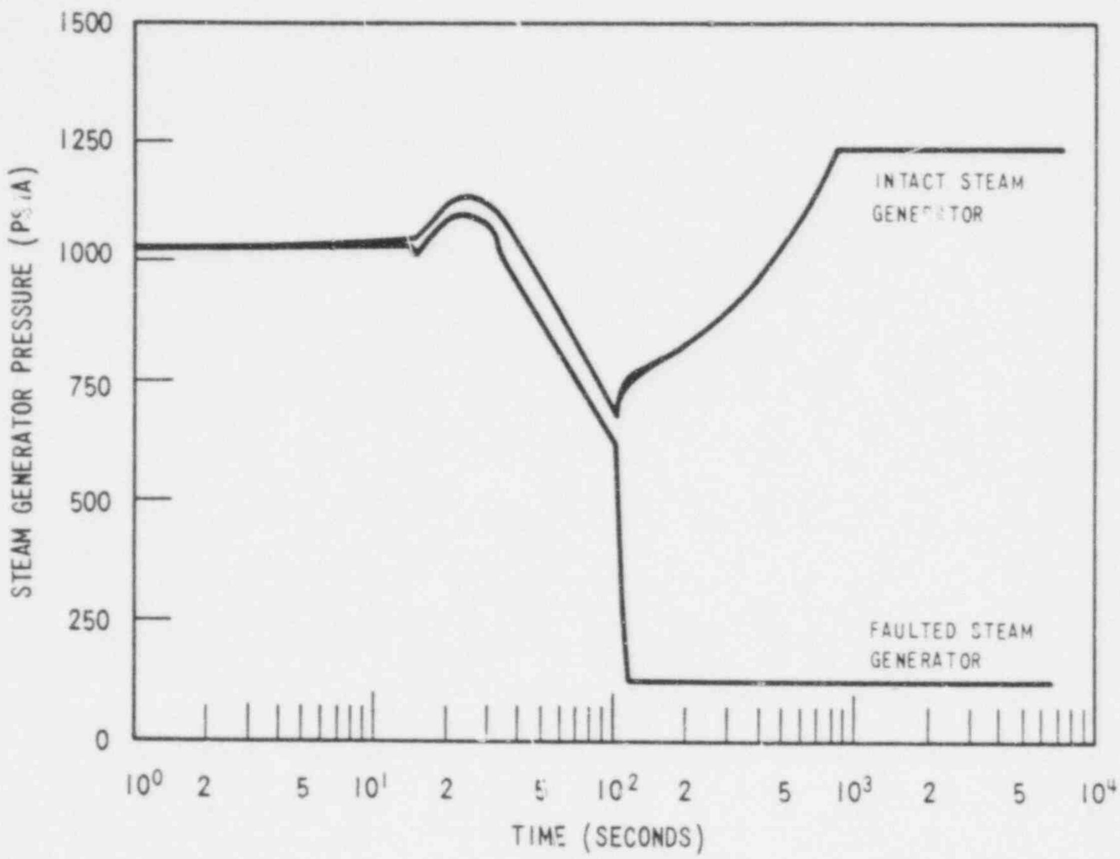
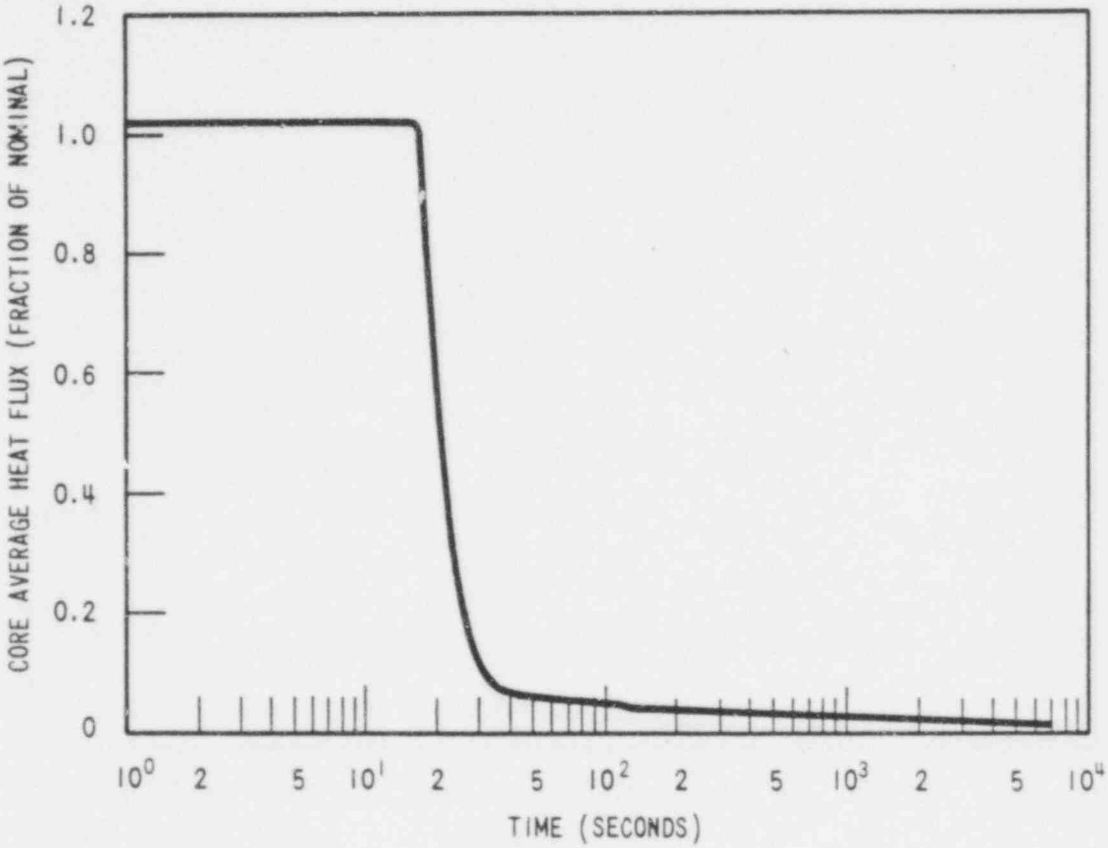
Figure 15.2-17.

Steam Generator Pressure Transients for Main Feedline Rupture with Offsite Power Available



613 235

WCAP – 9500
Figure 15.2-17a. Steam Generator Pressure Transients for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)



WCAP - 9500

Figure 15.2-18.

Core Average Heat Flux Transient, Transients for Main Feedline Rupture With Offsite Power Available

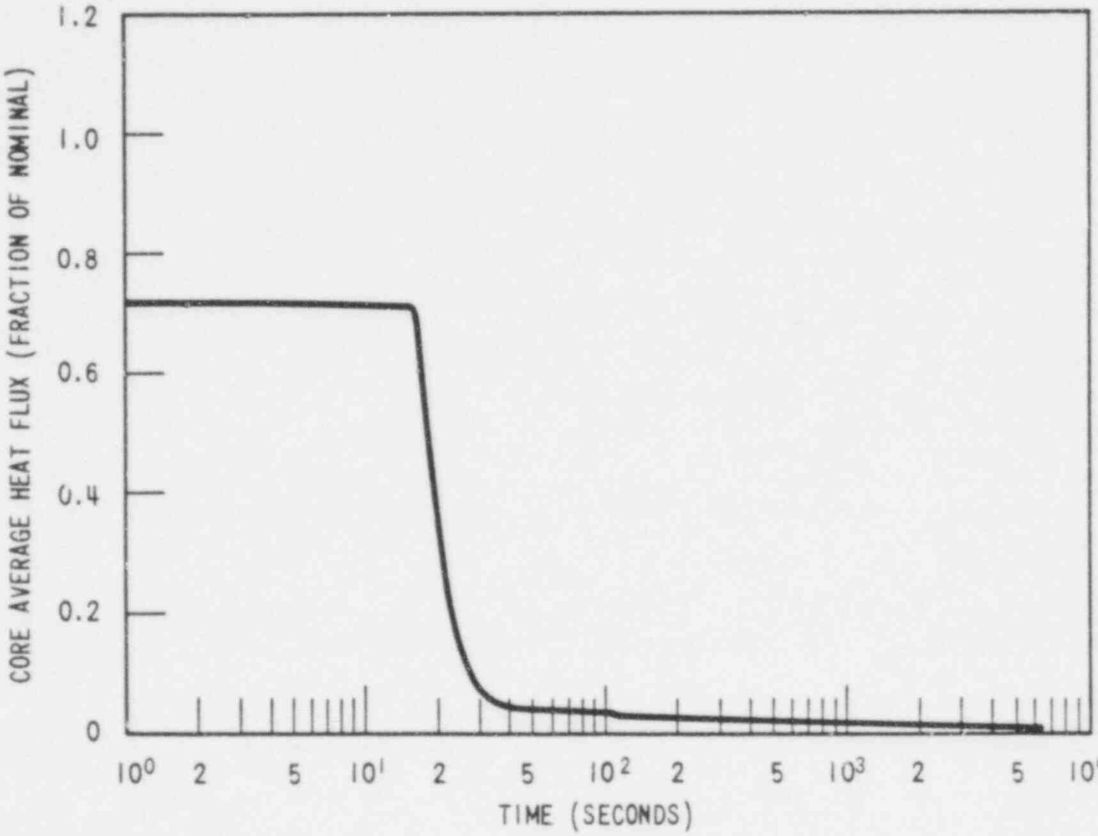


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Figure 15.2-18a.

Core Average Heat Flux Transient, Transients for Main Feedline Rupture With Offsite Power Available (N-1 Loop Operation)

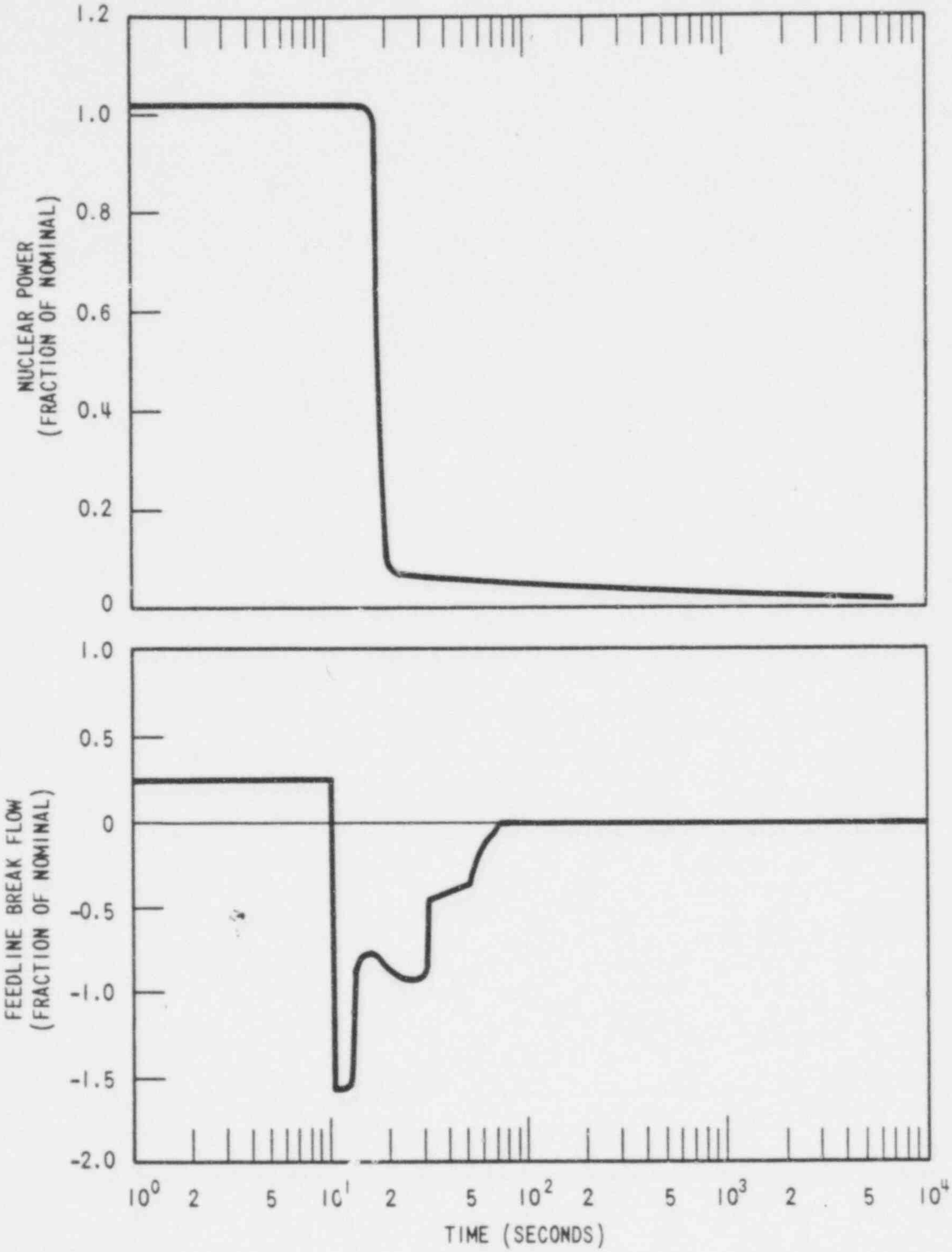

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Figure 15.2-19. Nuclear Power Transient, and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power

613 238

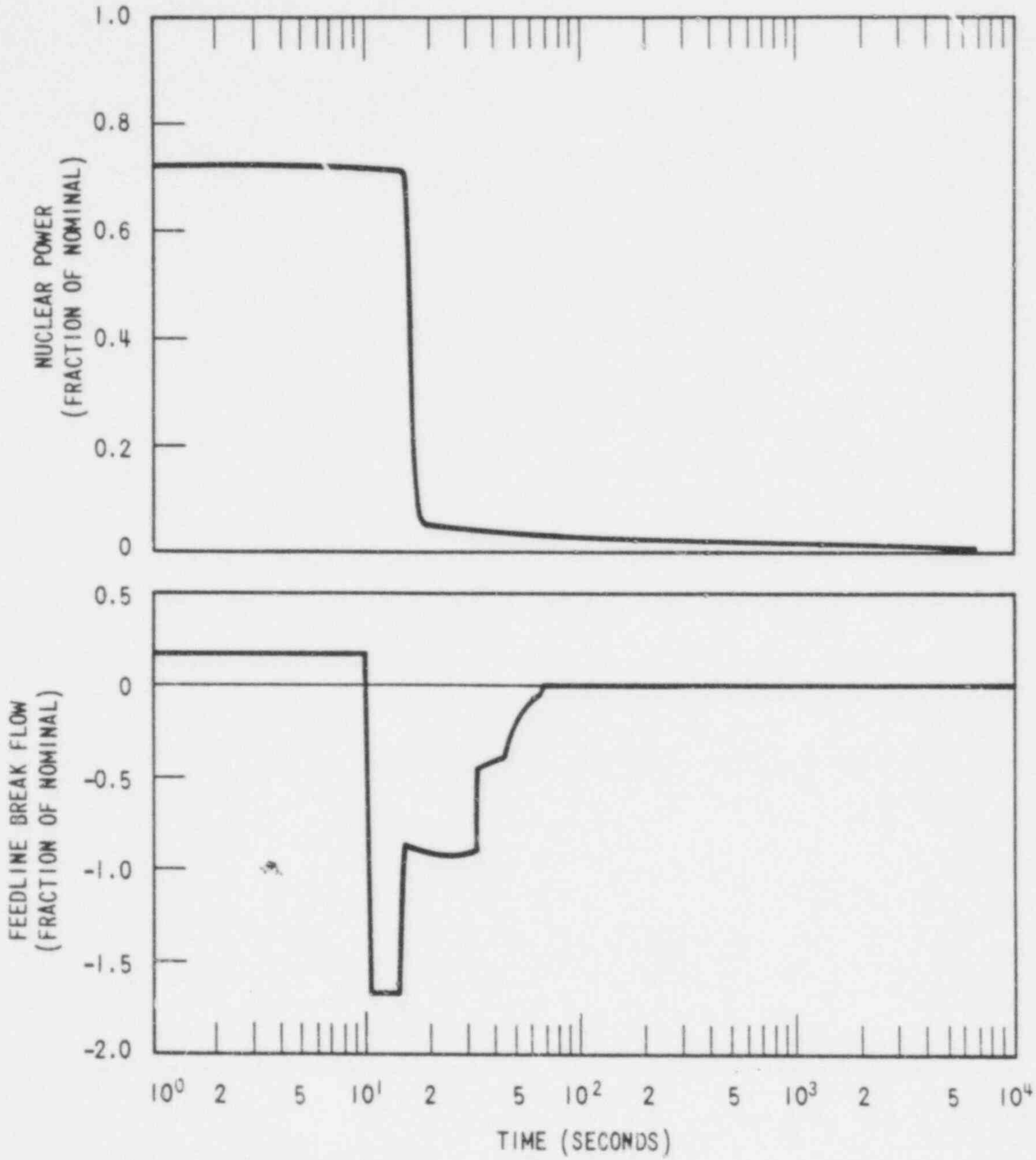

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Figure 15.2-19a.

Nuclear Power Transient, and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)

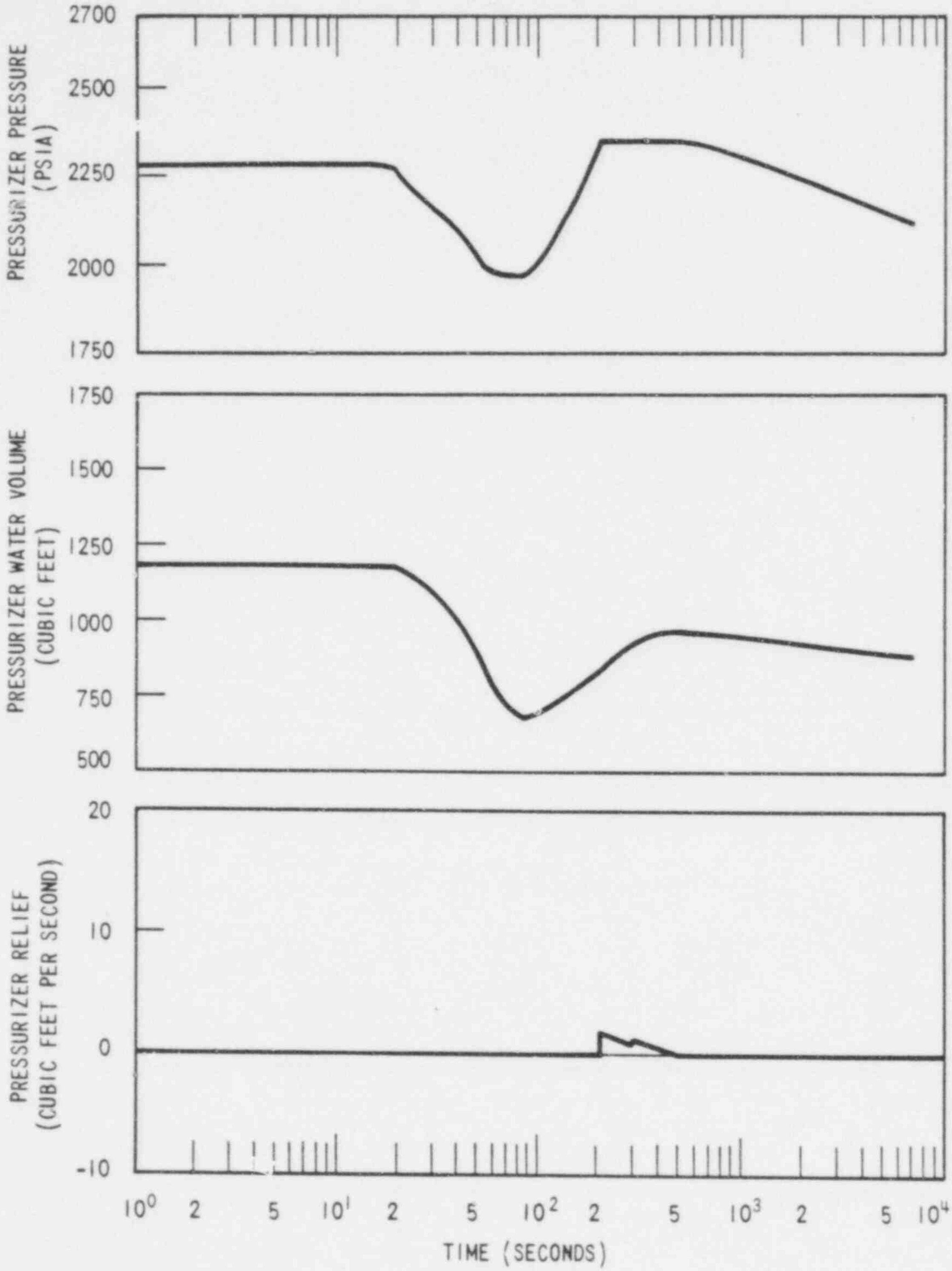

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Figure 15.2-20.

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

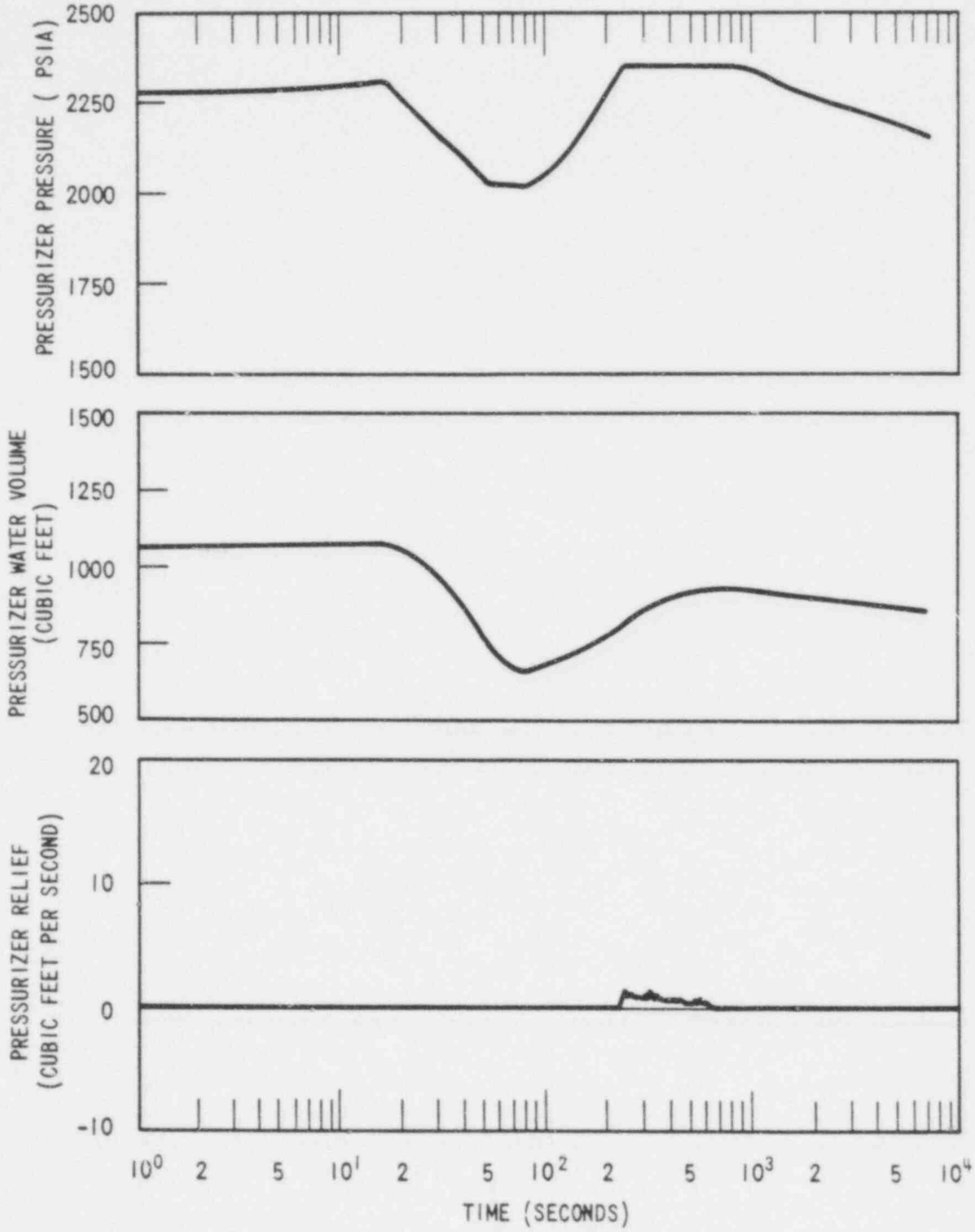

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Figure 15.2-20a.

Pressurizer Pressure, Water Volume, and Relief Rate for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)

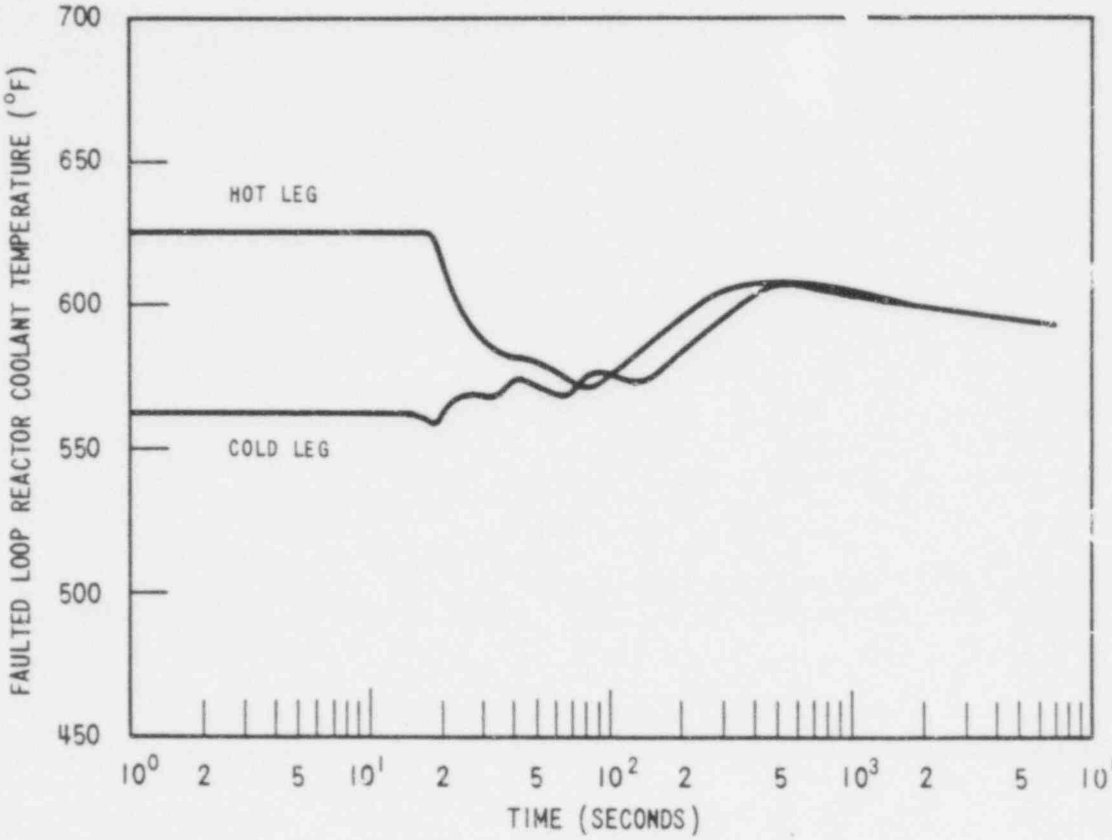

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Figure 15.2-21.

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture Without Offsite Power

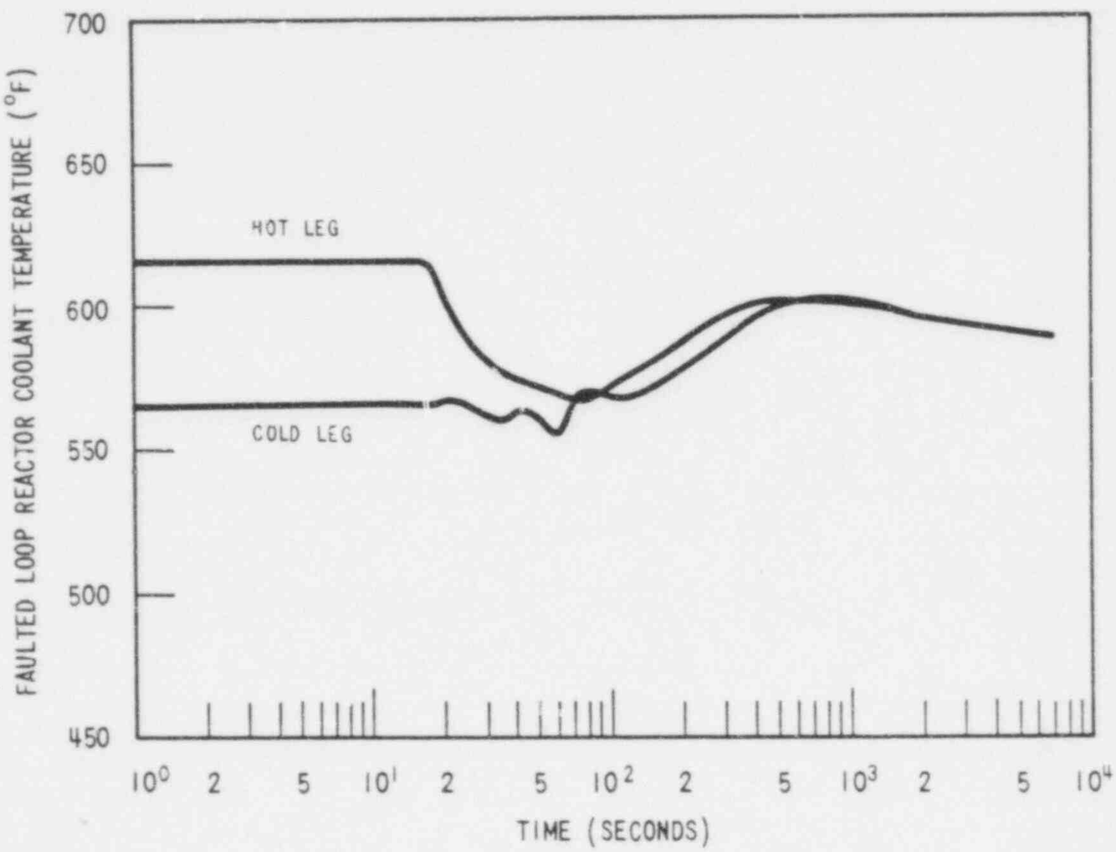

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Figure 15.2-21a.

Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)

613 243

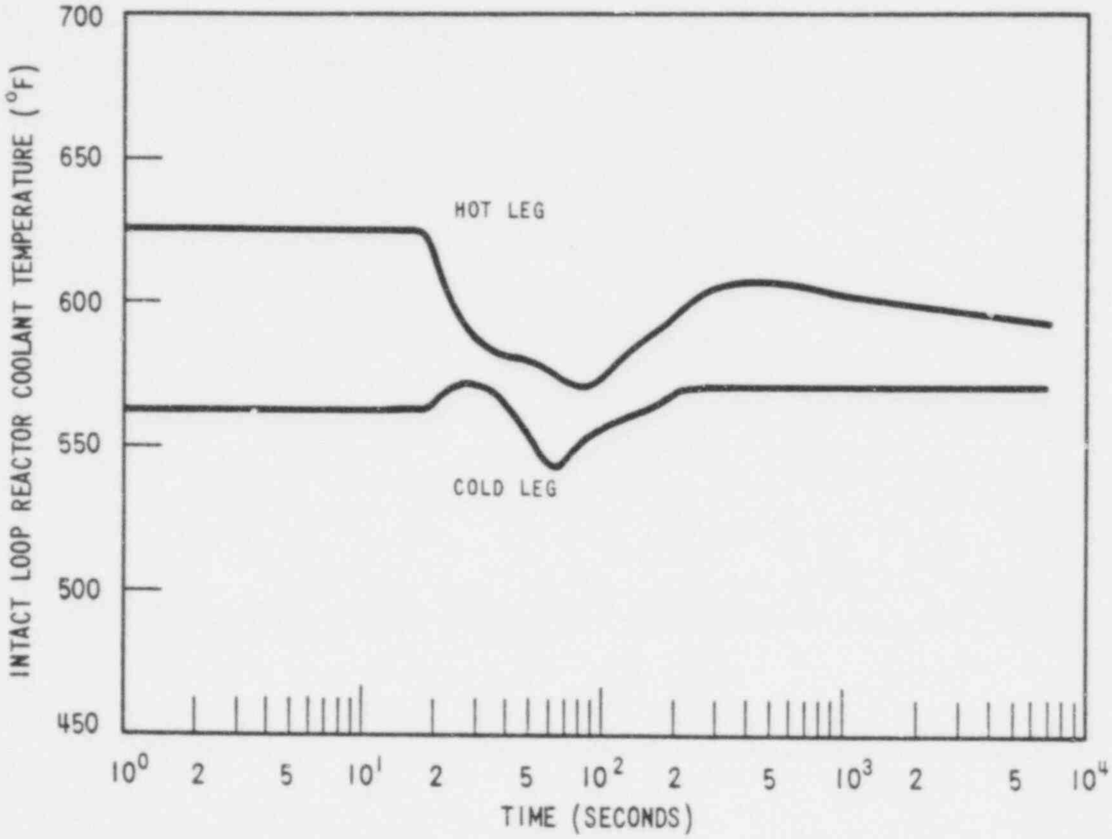

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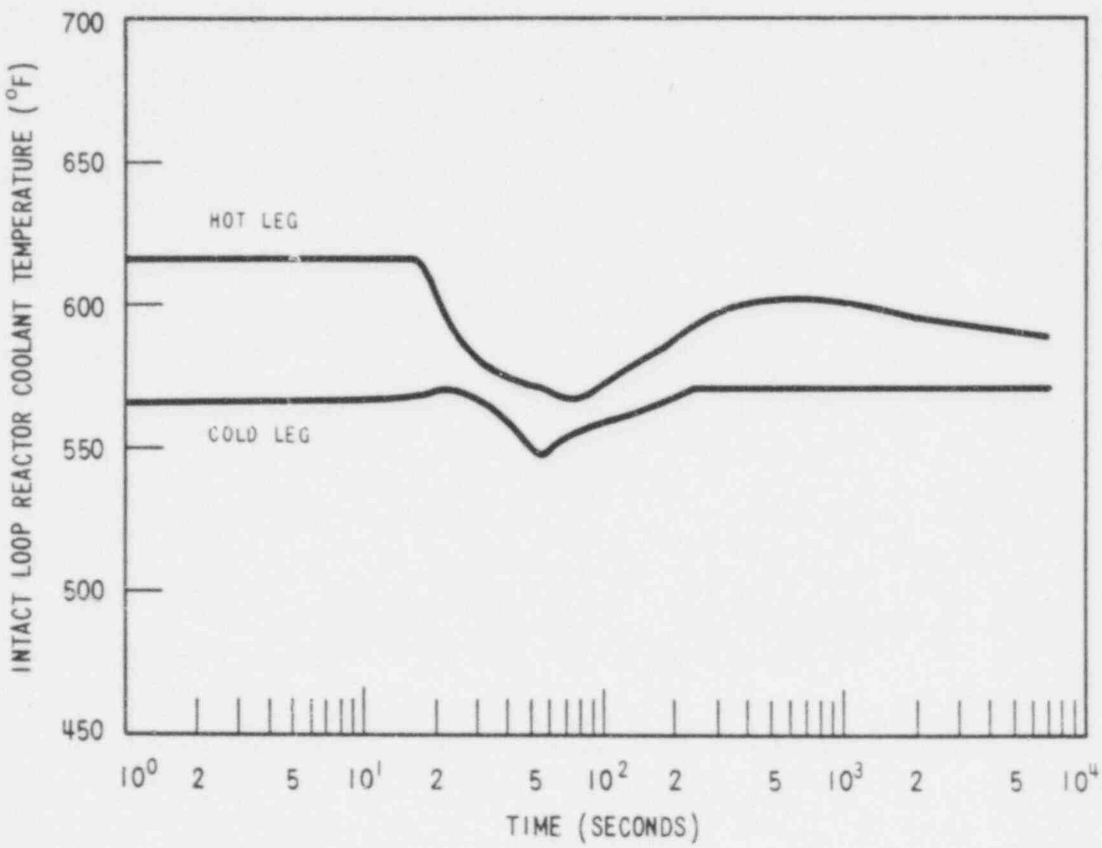
Figure 15.2-22.

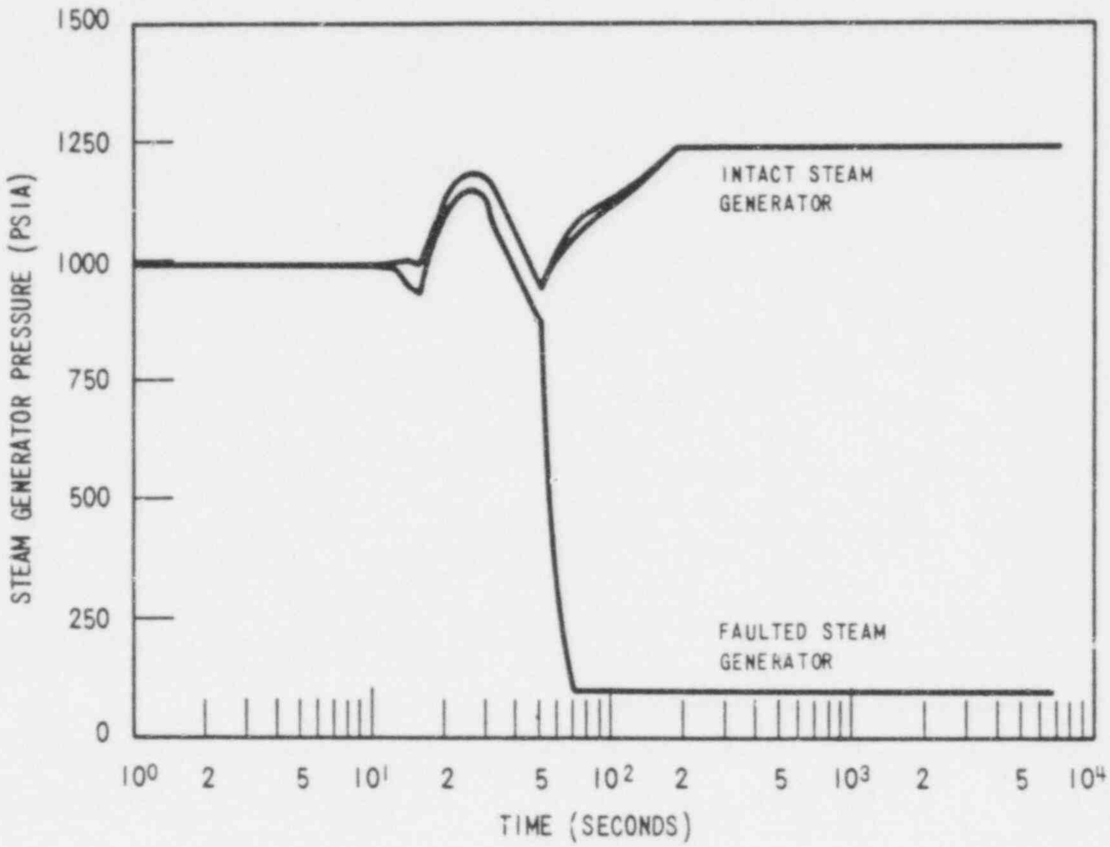
Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture Without Offsite Power



WCAP – 9500

Figure 15.2-22a.

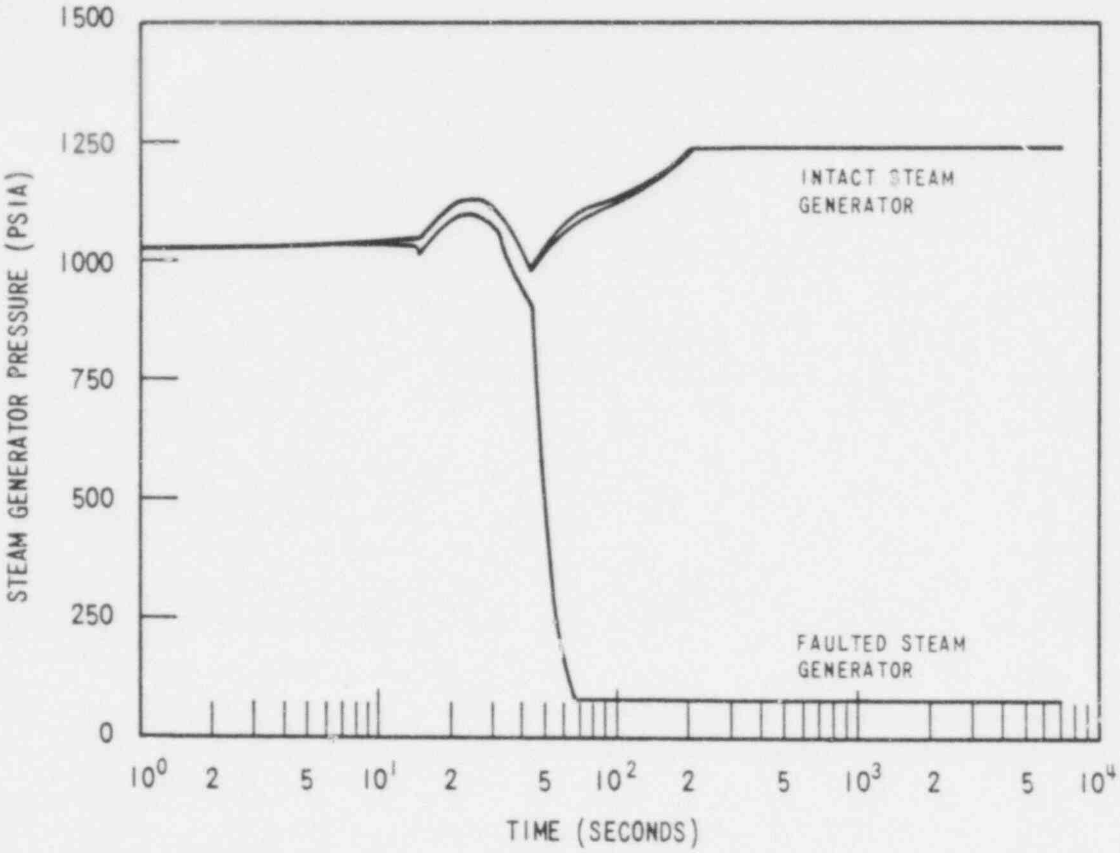
Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)



WCAP - 9500

Figure 15.2-23.

Steam Generator Pressure Transients for Main Feedline Rupture Without Offsite Power



WCAP – 9500

Figure 15.2-23a.

Steam Generator Pressure Transients for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)

14,395-95

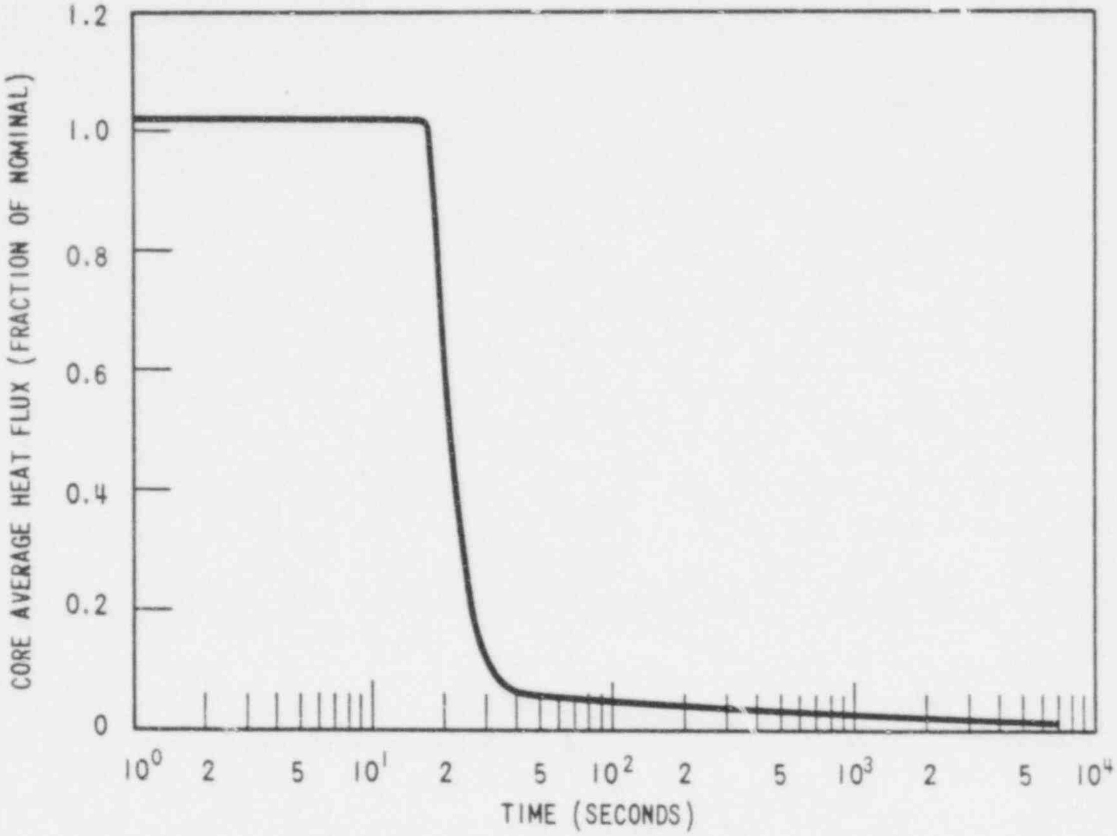


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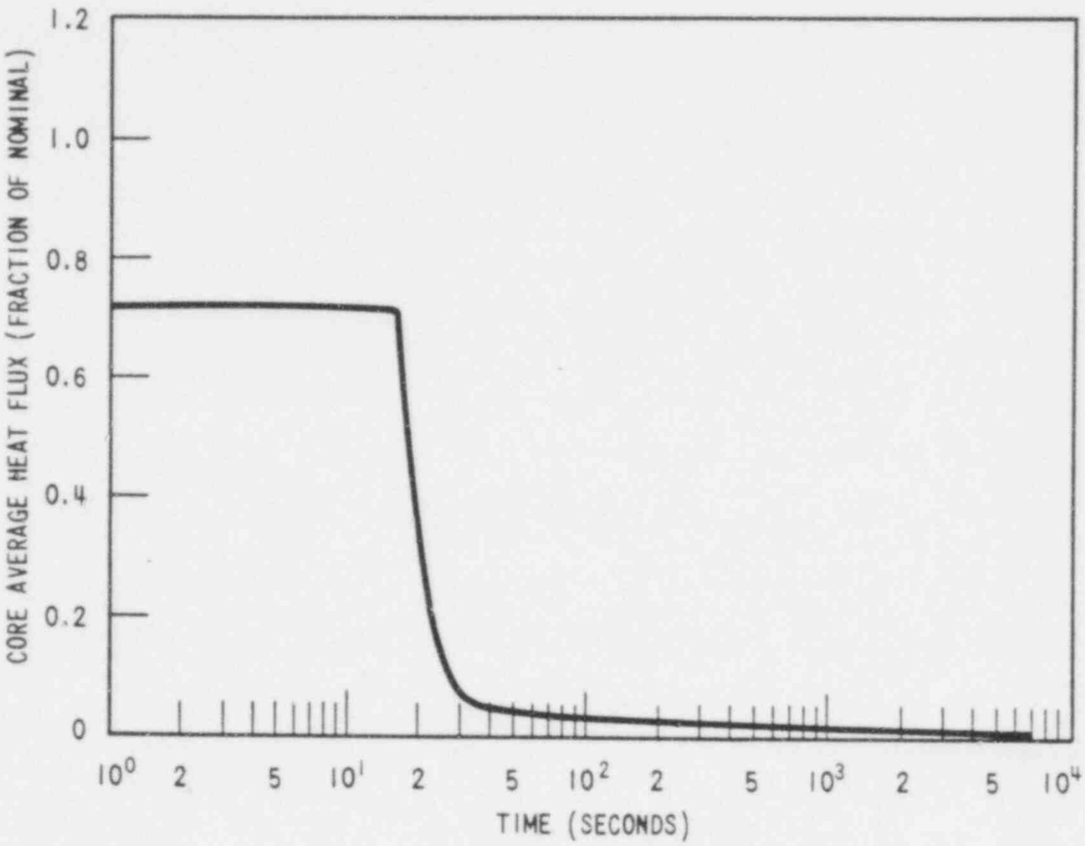
Figure 15.2-24.

C..e Average Heat Flux Transient for Main Feedline Rupture Without Offsite Power

613 248



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WCAP - 9500

Figure 15.2-24a.

Core Average Heat Flux Transient for Main Feedline Rupture Without Offsite Power (N-1 Loop Operation)

15.3

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. 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:

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 (see Subsection 15.0.2).

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 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.

Normal power for the reactor coolant pumps is supplied through individual buses connected to the generator. When a generator trip occurs, the buses are automatically transferred to an offsite power supply. The pumps will continue to supply coolant flow to the core. Following any

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turbine trip where there are not electrical faults or thrust bearing failures, which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds, thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.2.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of four low flow signals in any reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 10) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

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

<u>Method of Analysis</u>

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 (Reference 1) Code 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 (Reference 2) Code 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.

Initial Conditions

This accident is analized with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation. Plant characteristics and initial conditions are discussed in Subsection 15.0.4.

The initial margin in terms of power to the core thermal limits (Figure 15.0-1) is the governing parameter in the DNB evaluation of this event. For this analysis an initial core thermal margin consistent with the minimum allowed thermal margin as specified in the Technical Specifications was assumed.

Reactivity Coefficients

The most negative Doppler-only power coefficient is used (see Figure 15.0-2). This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent power of 0.016 $\Delta\rho$.

The least negative moderator temperature coefficient (see Figure 15.0-3) 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 Subsection 15.0.9 and listed in Table 15.0-5. 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 through 15.3-4 show the transient response for the loss of one reactor coolant pump with four loops initially in operation. Figure 15.3-4 shows the DNBR to be always greater than the limit value.

Figures 15.3-5 through 15.3-8 show the transient response for the loss of one reactor coolant pump with three loops initially in operation. The minimum DNBR is greater than the limit value, as shown in Figure 15.3-8.

For both cases analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel is not significantly 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 is shown on Table 15.3-1. 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. Following reactor trip, the plant will come to a stabilized condition at hot standby with one or more reactor coolant pumps in operation. Normal operating procedures may then be followed. 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.

15.3.1.3 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.

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, the DNB design-basis as described in Section 4.4 is met.

The radiological consequences of this event would be less thean 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 busses from a transformer connected to the generator. When a generator trip occurs, the busses are automatically transferred to an offsite power supply. The pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults or thrust bearing failures which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds, thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

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This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.2.

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

1. Low reactor coolant pump speed reactor trip.
2. Low reactor coolant loop flow.

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

The low reactor coolant pump speed trip is also provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 Hz/sec this trip function will protect the core from underfrequency events without requiring tripping of the RCP breakers. Refer to Chapter 7 for interface requirements concerning tripping of the RCP breakers for underfrequency events. Reference [3] provides analyses of grid frequency disturbances..

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 four 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 10) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is low enough this trip function will protect the core from underfrequency events. This effect is fully described in Reference [3].

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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 (Reference 1) Code 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 (Reference 2) Code 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 low reactor coolant pump speed.

Results

Figures 15.3-9 through 15.3-12 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 underspeed signal. Figure 15.3-12 shows the DNBR to be always greater than the limit value.

Figures 15.3-13 through 15.3-16 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 underspeed signal. The minimum DNBR is greater than the limit value, as shown in Figure 15.3-16.

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 cladding temperatures do not increase significantly above their respective initial values. The calculated sequence of events for the two cases analyzed are shown on Table 15.3-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Subsection 15.2.6. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed. 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.

15.3.2.3 <u>Radiological Consequences</u>

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 <u>Conclusions</u>

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, the DNB design-basis as described in Section 4.4 is met.

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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 Subsection 15.0.2.

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 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 state operating conditions, i.e., maximum steady-state power level, maximum steady state pressure, and maximum steady state coolant average temperature. Plant characteristics and initial conditions are further discussed in Subsection 15.0.4. 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 15.3-18 and 15.3-22 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 reaches 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 "hot spot" condition represent the upper limit with respect to cladding 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 level (i.e., F_Q at the initial core power = 3.0).

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 cladding 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 cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was .ssumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-^{o}F at the initiation of the transient. Thus t e large amount of energy stored in the fuel because of the small initial value is released to the cladding at the initiation of the transient.

Zirconium Steam Reaction

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

$$\frac{d\,(w^2)}{dt} = 33.3 \times 10^6 \ \exp \ \frac{(-45.500)}{1.986T} \qquad (15.3\text{-}1)$$

where:

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, ^{o}F

The reaction heat is 1510 cal/gm.

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

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.9 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Locked Rotor with Four Loops Operating

The transient results for this case are shown in Figures 15.3-17 through 15.3-20. The results of these calculations are also summarized in Table 15.3-2. 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 cladding surface temperature is considerably less than 2700°F. It should be noted that the cladding 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 15.3-21 through 15.3-24. The peak Reactor Coolant System pressure is slightly higher than for the previous case, but is still less than that which would cause stresses to exceed the faulted condition stress limits. The cladding temperature transient is more severe than for the previous case, but still well below the 2700°F limit.

The calculated sequence of events for the two cases analyzed is shown on Table 15.3-1. Figures 15.3-17 and 15.3-21 show that the core flow reaches a new equilibrium value by 10 seconds. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Following reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed to maintain a hot condition or to cool the plant to cold shutdown. 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.

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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.

15.3.3.4 Conclusions

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.

Since the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700^{O}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 Causes and Accident 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 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 15.0.2.

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 analysis conservatively assumes that DNB occurs at the beginning of the transient.

613 265

15.3.4.4 REFERENCES

1. T. W. T. Burnett, et al., "LOFTRAN Code Description, "WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

2. C. Hunin, "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

3. M. S. Balwin, et al., "An Evaluation of Loss of Flow Accidents caused by Power System Frequency Transinets in Westinghouse PWR's," WCAP-8424, Revision 1, June 1975.

613 266

TABLE 15.3-1

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	1.6
	Rods begin to drop	2.6
	Minimum DNBR occurs	3.6
2. Three loops operating, one pump coasting down	Coastdown begins	0.
	Low flow reactor trip	2.7
	Rods begin to drop	3.7
	Minimum DNBR occurs	4.5

ACCIDENT	EVENT	Four Loop Operation	Three Loop Operation
Complete Loss of Forced Reactor Coolant Flow			
	All operating pumps lose power and begin coasting down	0	0

TABLE 15.3-1 (Continued)

ACCIDENT	EVENT	Time (sec) Four Loop Operation	Time (sec) Three Loop Operation
	Reactor coolant pump underspeed trip point reached	1.0	1.0
	Rods begin to dron	1.6	1.6
	Minimum DNBR occurs	3.5	3.4
Reactor Coolant Pump Shaft Seizure (Locked Rotor)			
	Rotor on one pump locks	0	0
	Low flow trip point reached	.04	.04
	Rods begin to drop	1.04	1.04
	Maximum RCS nressure occurs	3.2	3.4
	Maximum cladding temperature occurs	3.3	3.6

613 268

TABLE 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

	4 LOOPS OPERATING INITIALLY	3 LOOPS OPERATING INITIALLY
Maximum Reactor Coolant System Pressure (psia)	2595	2610
Maximum Cladding Temperature (^{o}F) Core Hot Spot	2128	2216
$Zr-H_2O$ reaction at core hot spot (% by weight)	.10	.18

613 269

TABLE 15.3-3

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	9% of gap inventory
Iodine	None	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.

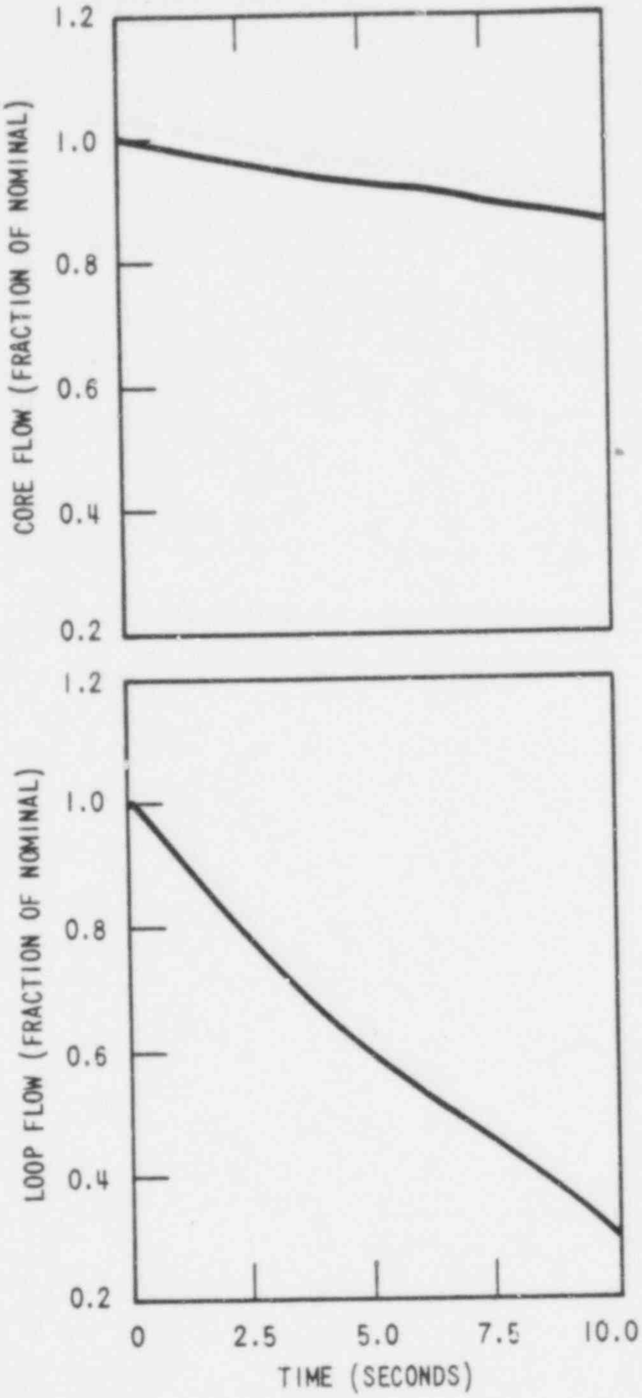
613 270

TABLE 15.3-3 (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.

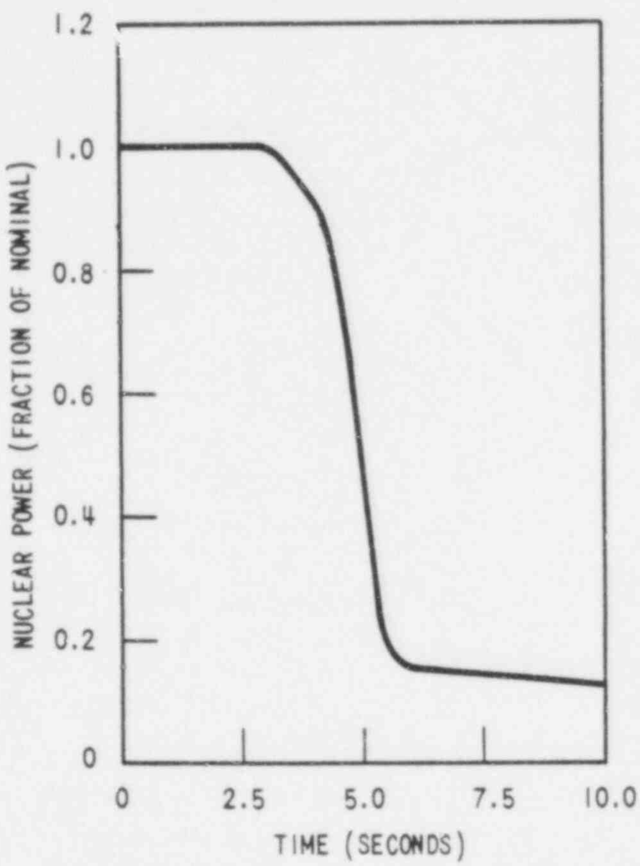
613 271



WCAP – 9500

Figure 15.3-1.

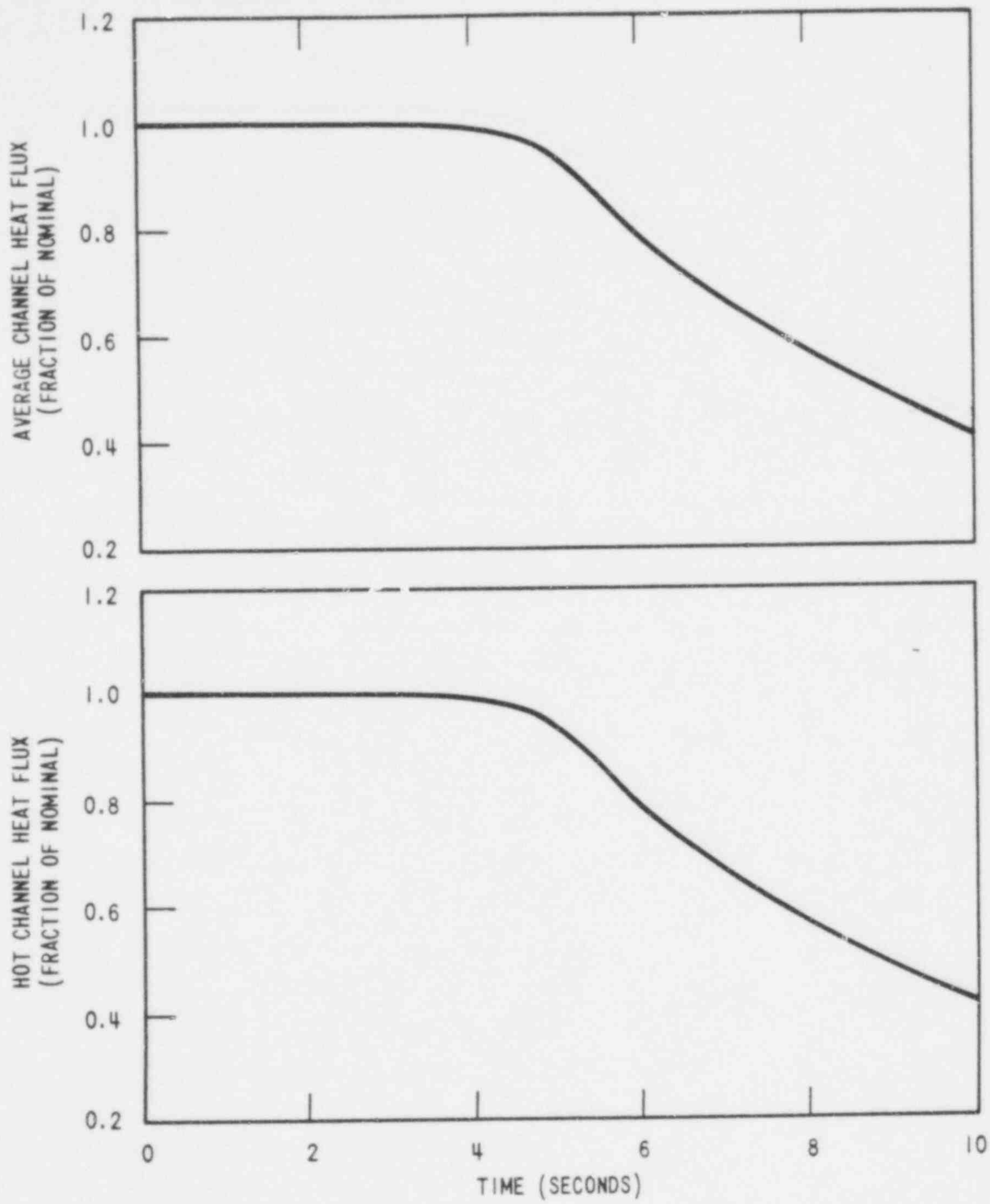
Flow Transients for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down



WCAP - 9500

Figure 15.3-2.

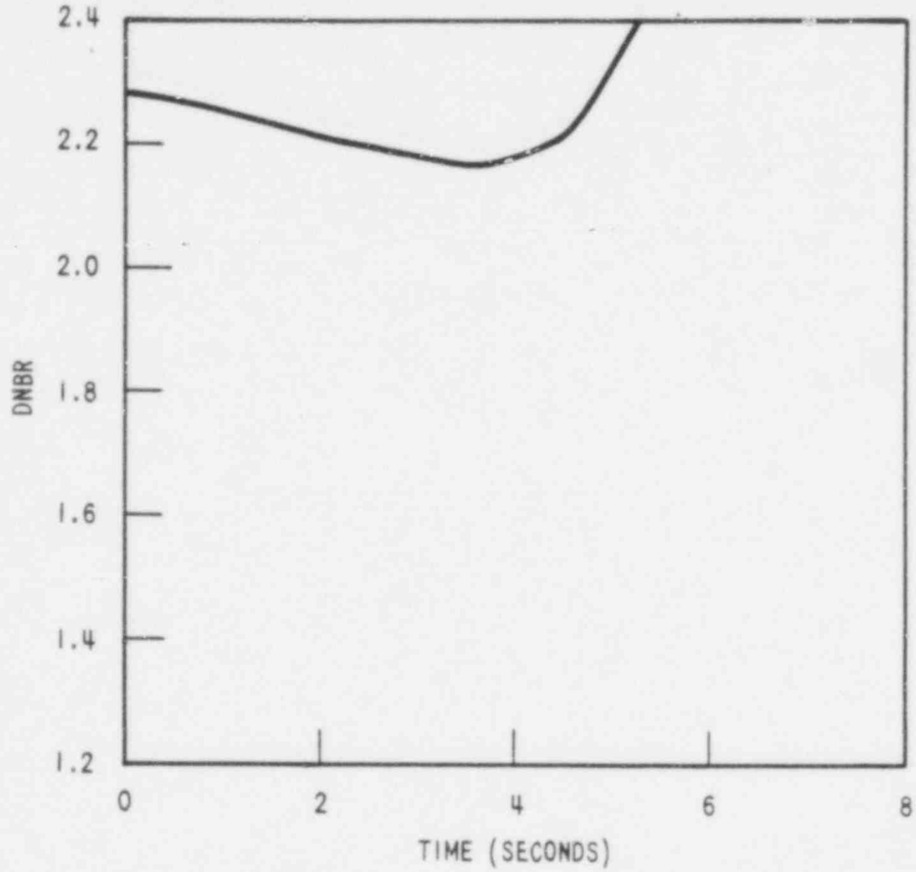
Nuclear Power Transient for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down



WCAP – 9500

Figure 15.3-3.

Average and Hot Channel Heat Flux Transient for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down

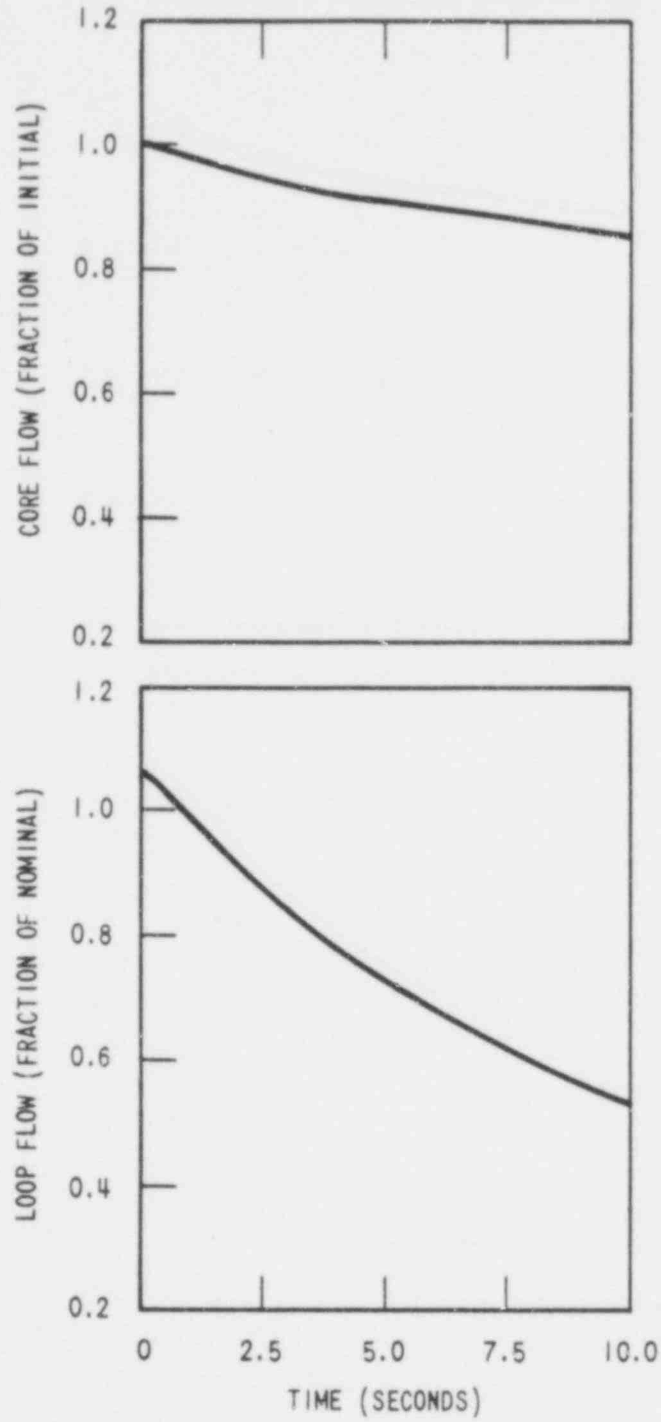


WCAP – 9500

Figure 15.3-4.

DNBR vs. Time for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down

14,395-118

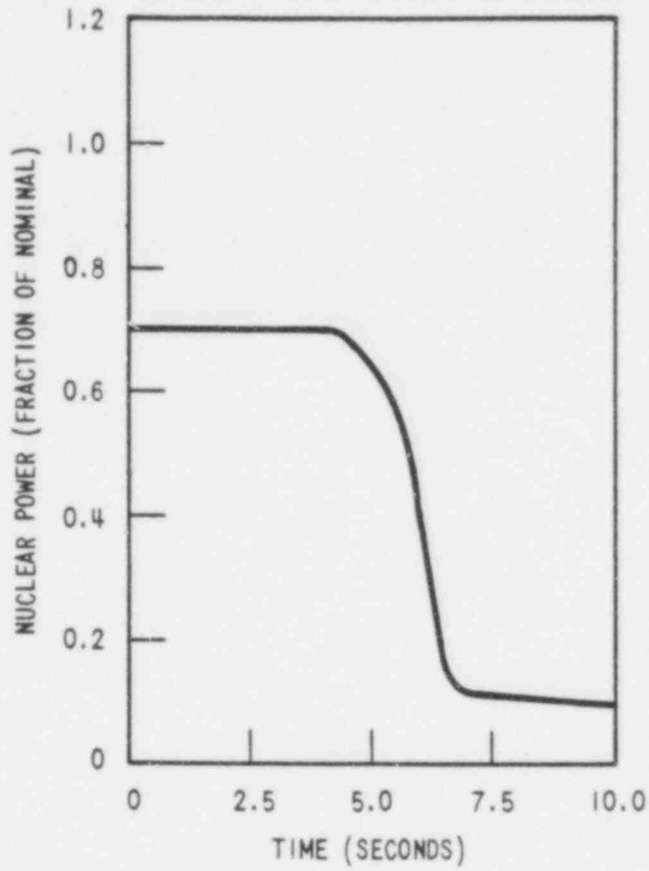


WCAP-9500

Figure 15.3-5.

Flow Transient for Partial Loss of Flow, Three Loops in Operation, One Pump Coasting Down

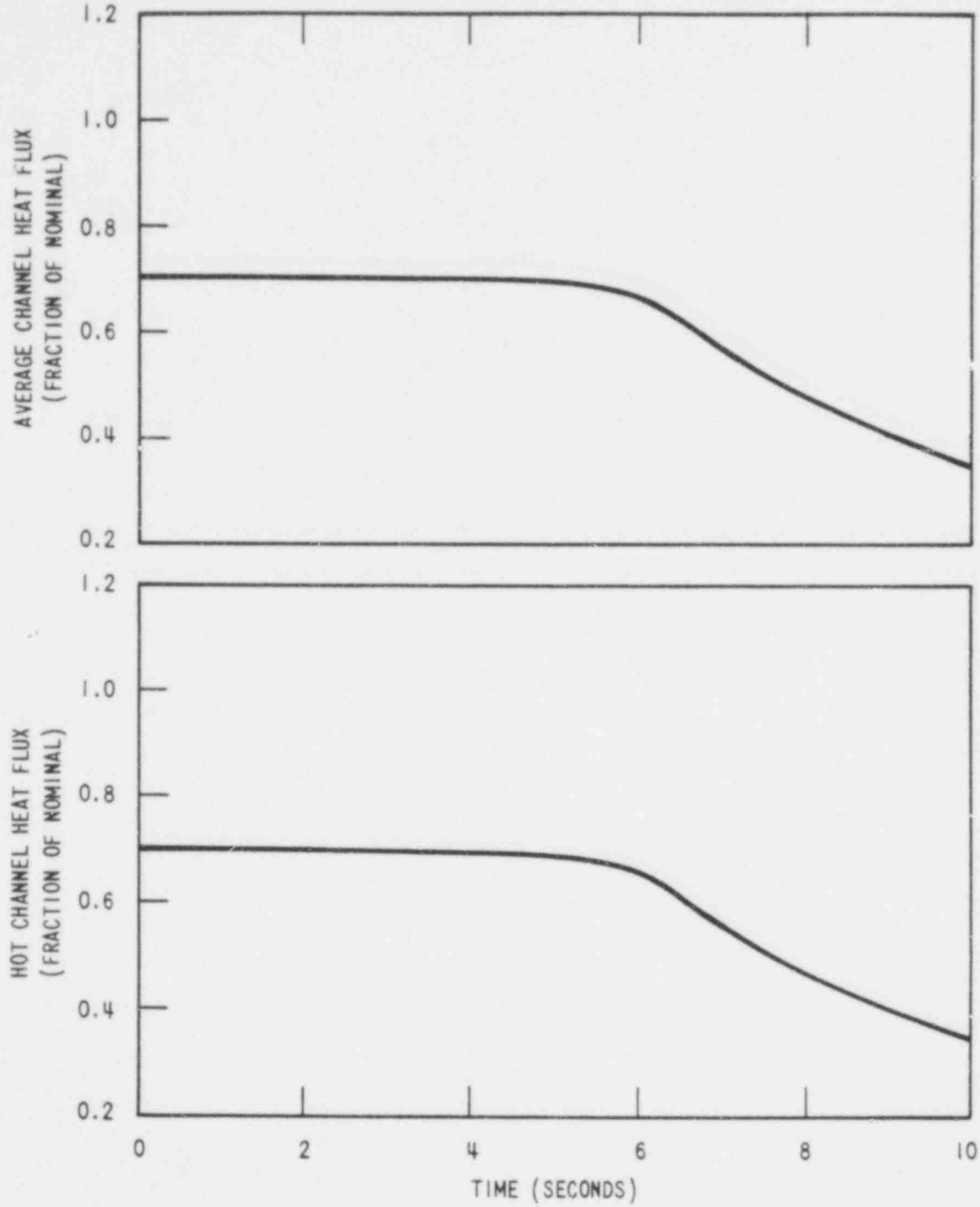
613 276



WCAP-950C

Figure 15.3-6.

Nuclear Power Transient for Partial Loss of Flow, Three Loops in Operation, One Pump Coasting Down

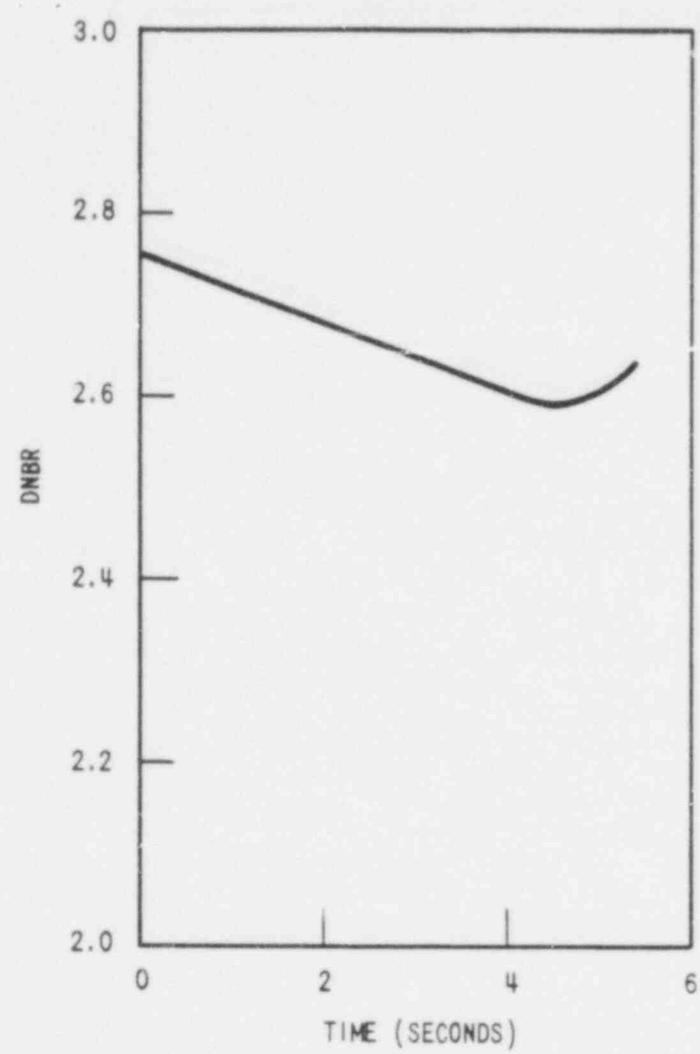


WCAP - 9500

Figure 15.3-7.

Average and Hot Channel Heat Flux Transients for Partial Loss of Flow, Three Loops in Operation, One Pump Coasting Down

613 297

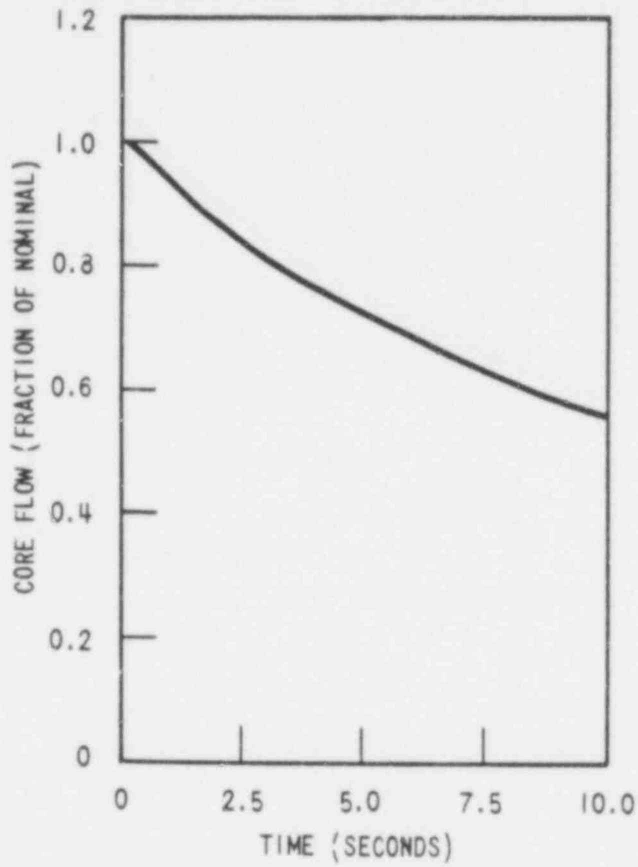


WCAP - 9500

Figure 15.3-8.

DNBR vs. Time for Partial Loss of Flow,
Three Loops in Operation,
One Pump Coasting Down

14,395-122

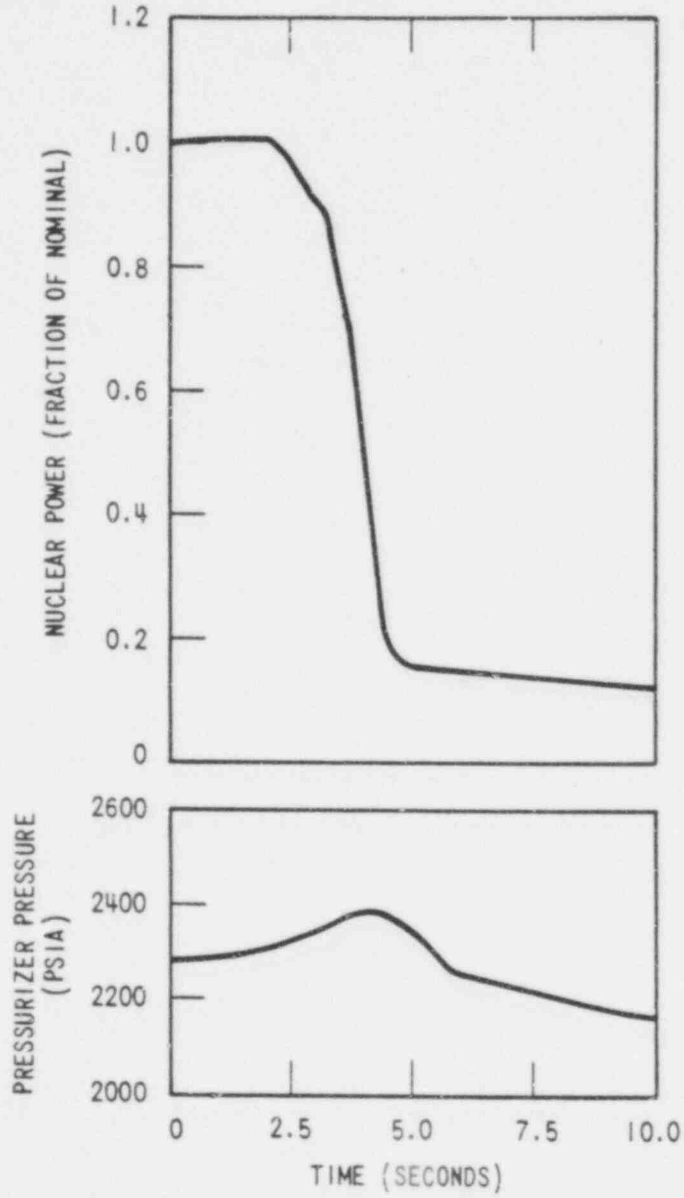


WCAP – 9500

Figure 15.3-9.

Core Flow Coastdown vs. Time for Four Loops in Operation, Four Pumps Coasting Down, Complete Loss of Flow

613 279

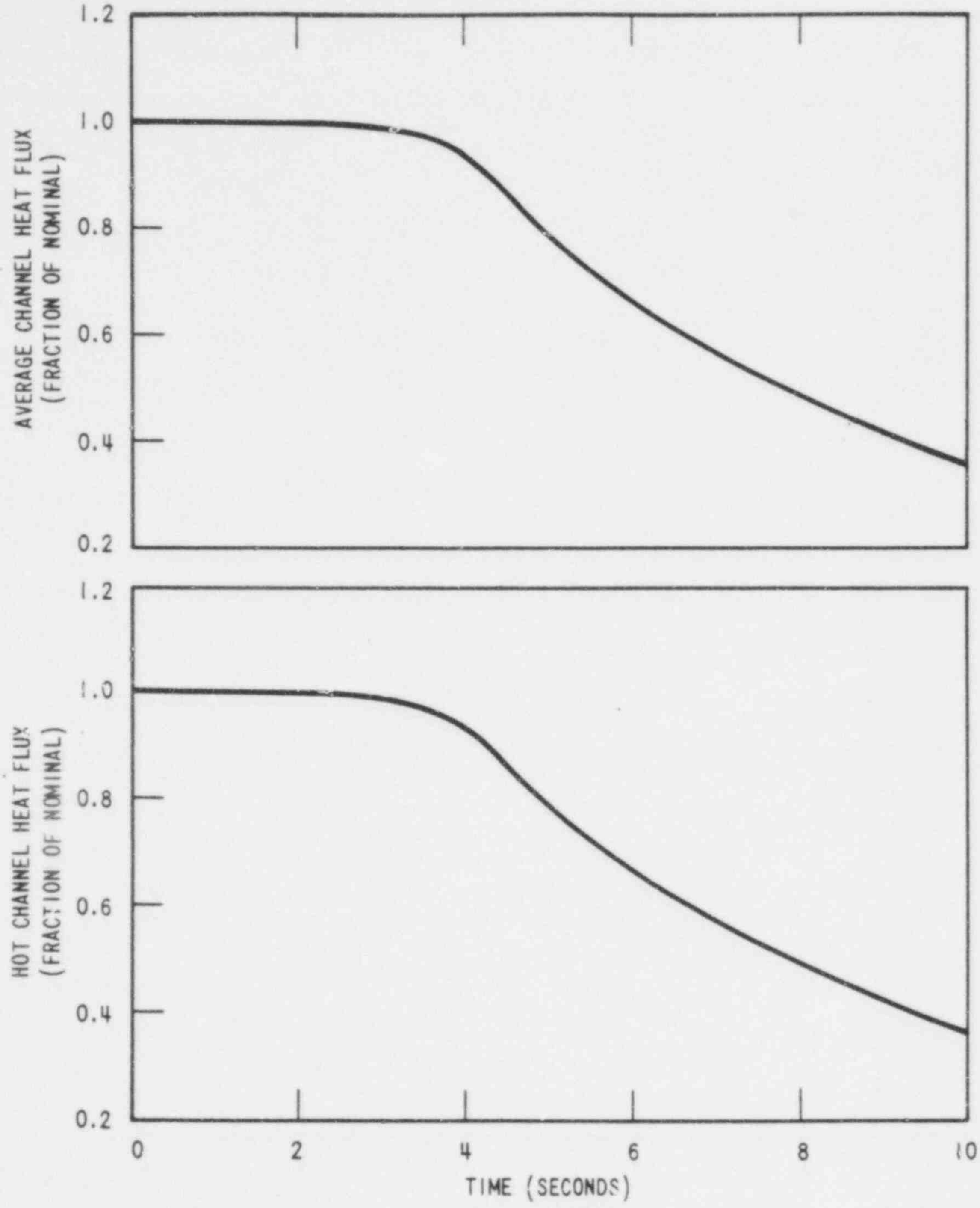


WCAP – 9500

Figure 15.3-10.

Nuclear Power Transient and Pressurizer Pressure Transient for Four Loops in Operation, Four Loops Coasting Down, Complete Loss of Flow

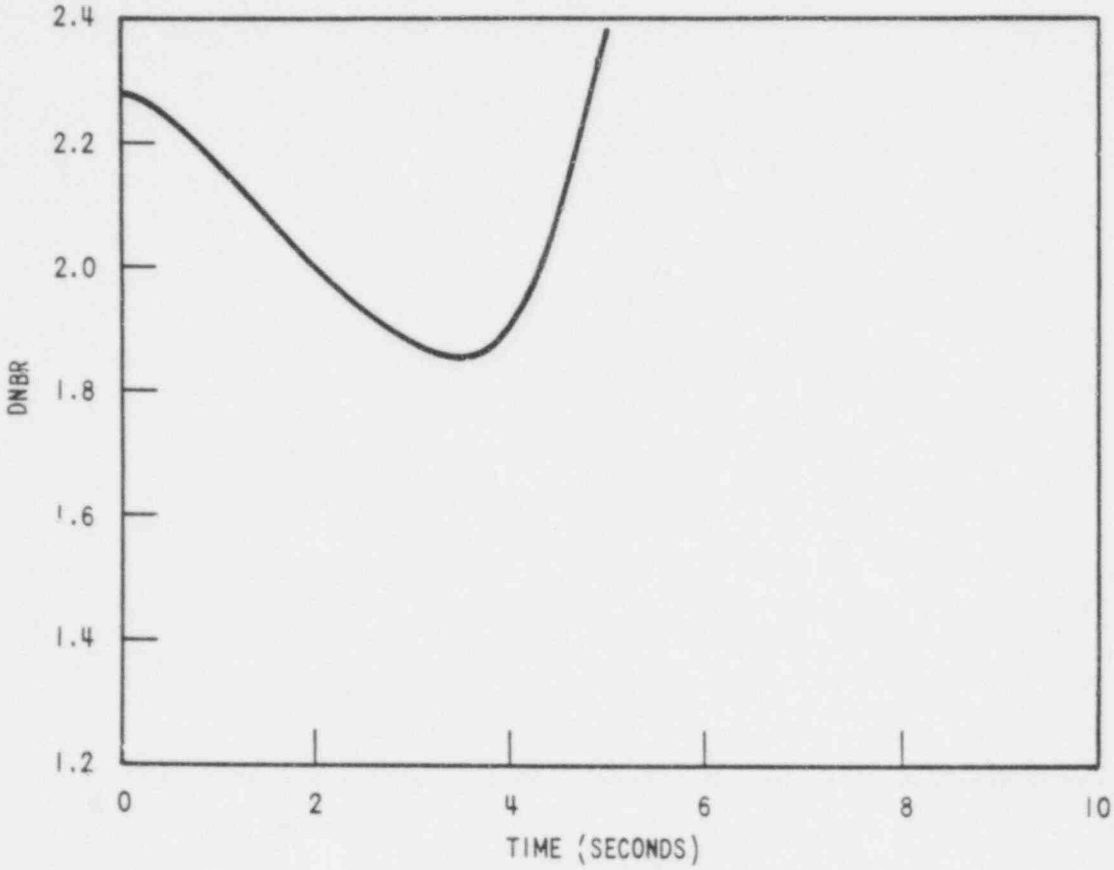
613 280



WCAP-9500

Figure 15.3-11.

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



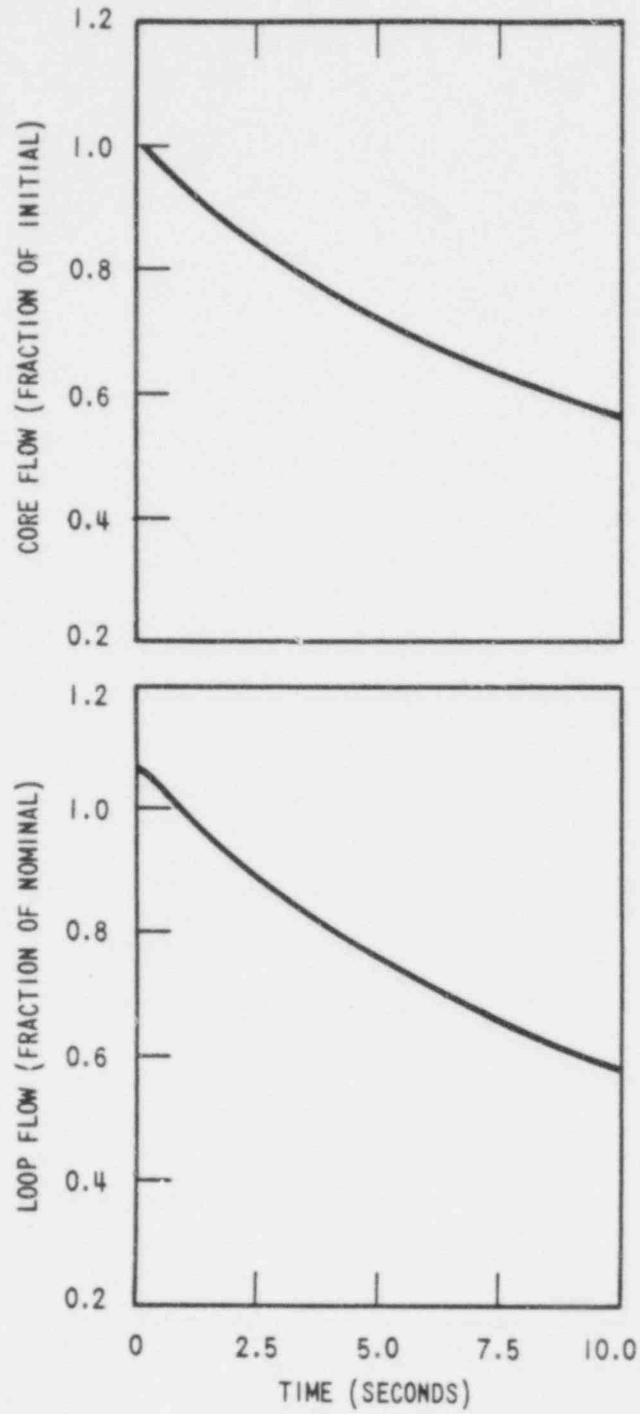
WCAP – 9500

Figure 15.3-12.

DNBR vs. Time for Four Loops in Operation, Four Pumps Coasting Down, Complete Loss of Flow

613 262

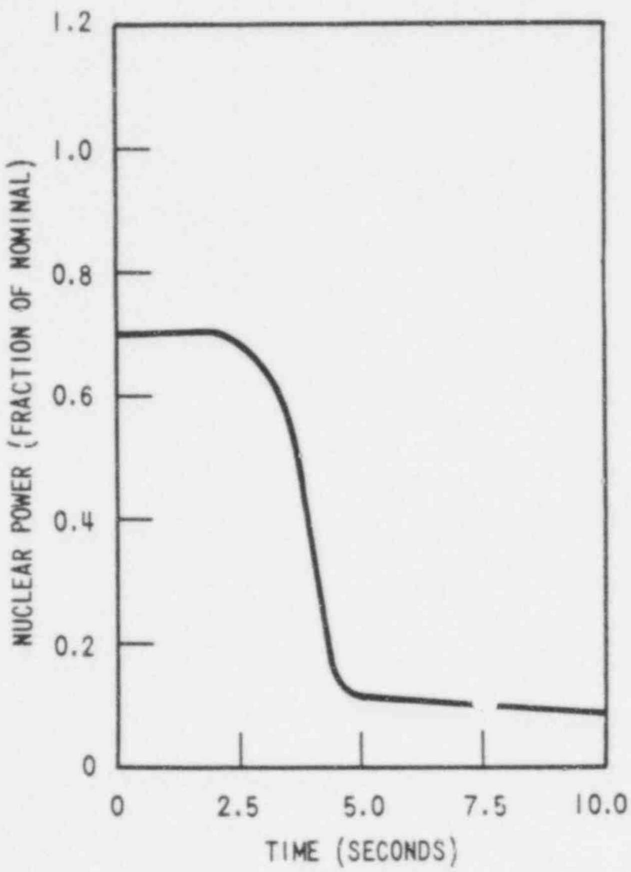
14,395-126



WCAP - 9500

Figure 15.3-13.
Flow Transients for Three Loops in Operation, Three Pumps Coasting Down, Complete Loss of Flow

615 285

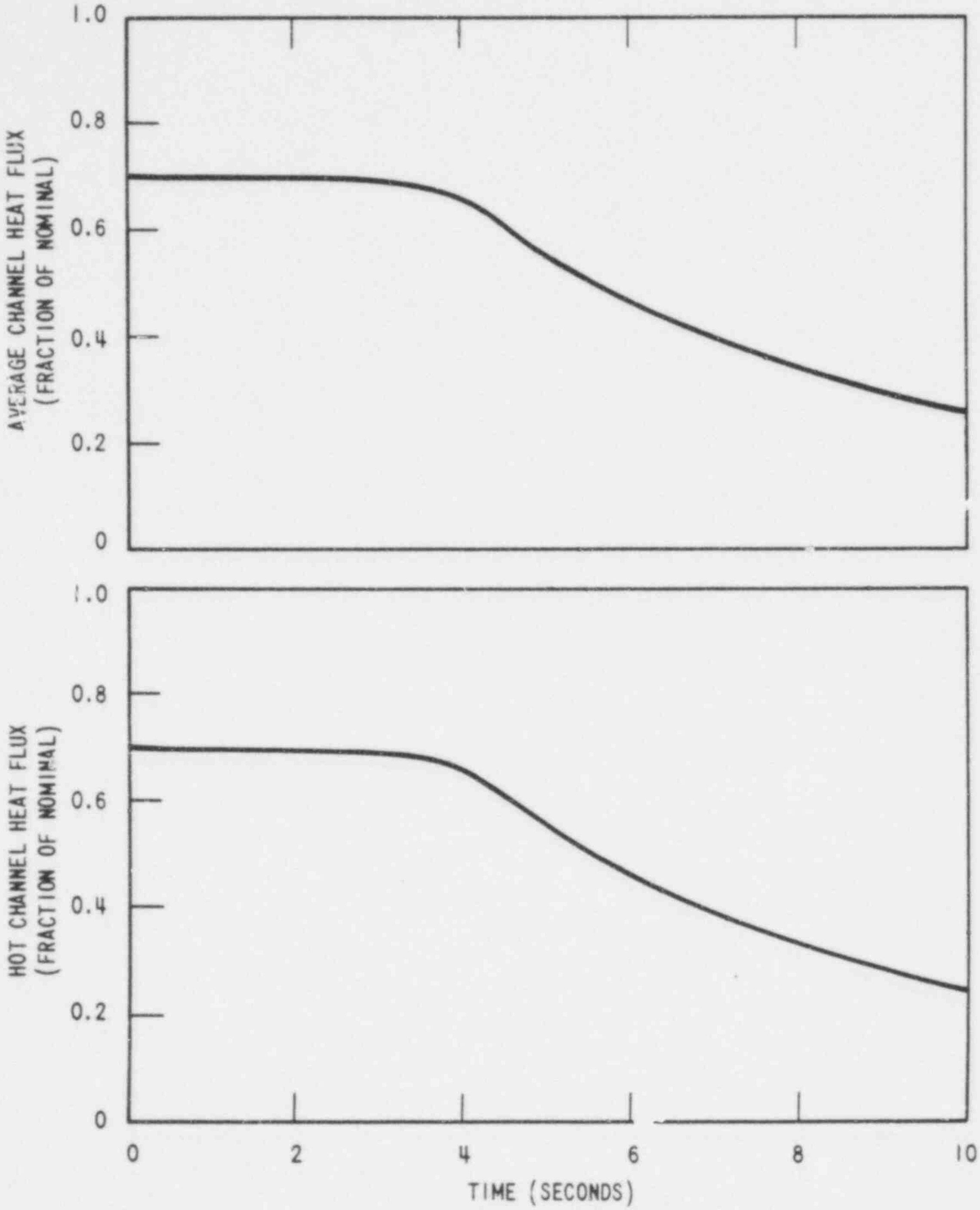


WCAP – 9500

Figure 15.3-14.

Nuclear Power Transient for Three Loops in Operation, Three Pumps Coasting Down, Complete Loss of Flow

613 284

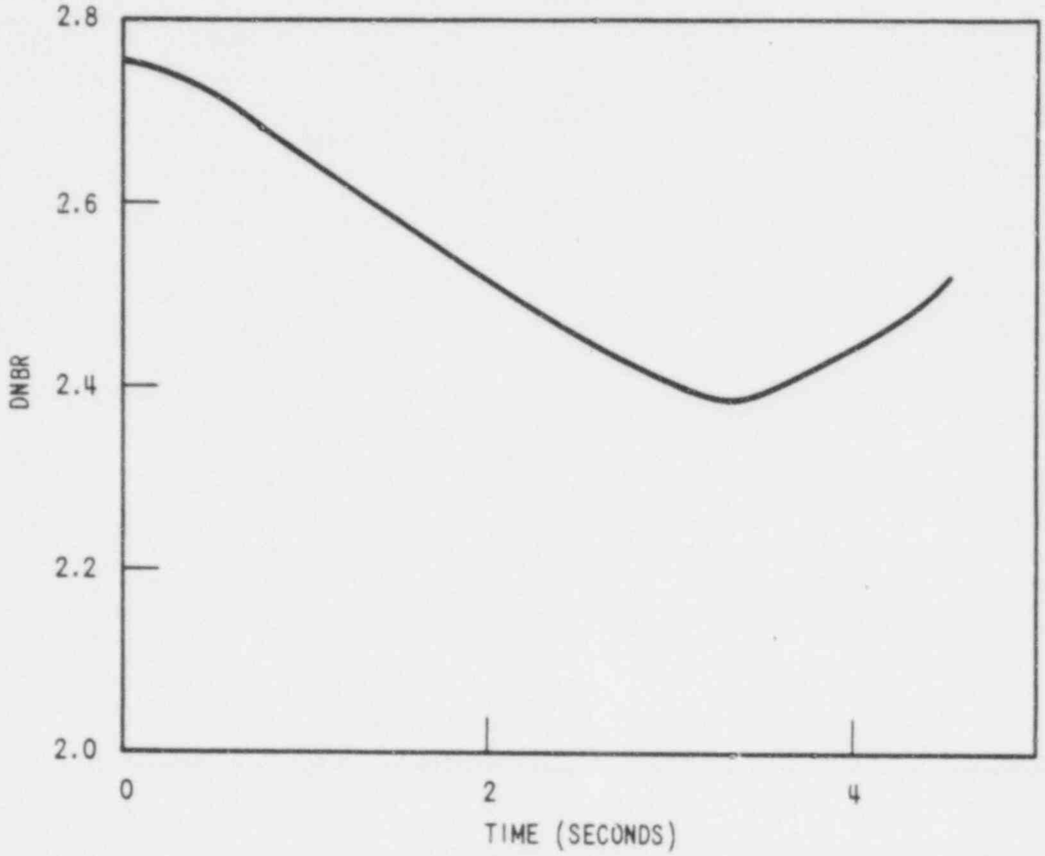


WCAP – 9500

Figure 15.3-15.

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

613 285

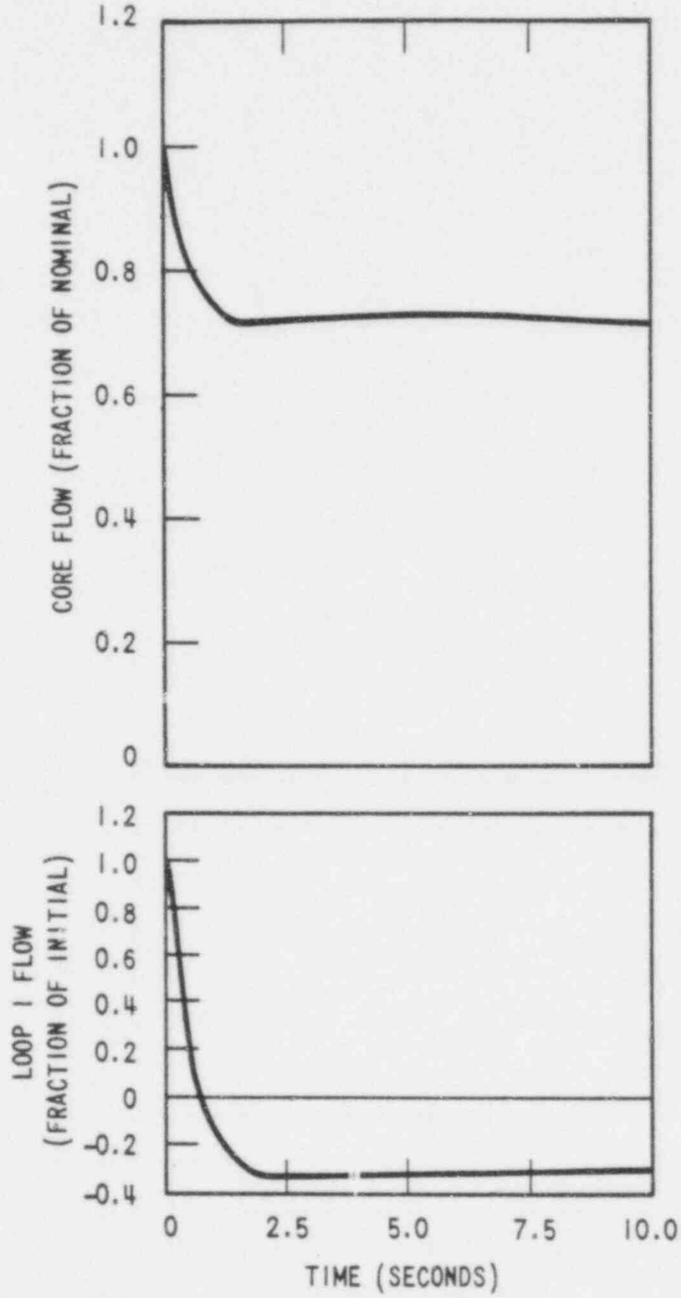


WCAP – 9500

Figure 15.3-16.

DNBR vs. Time for Three Loops in Operation, Three Pumps Coasting Down, Complete Loss of Flow

613 286



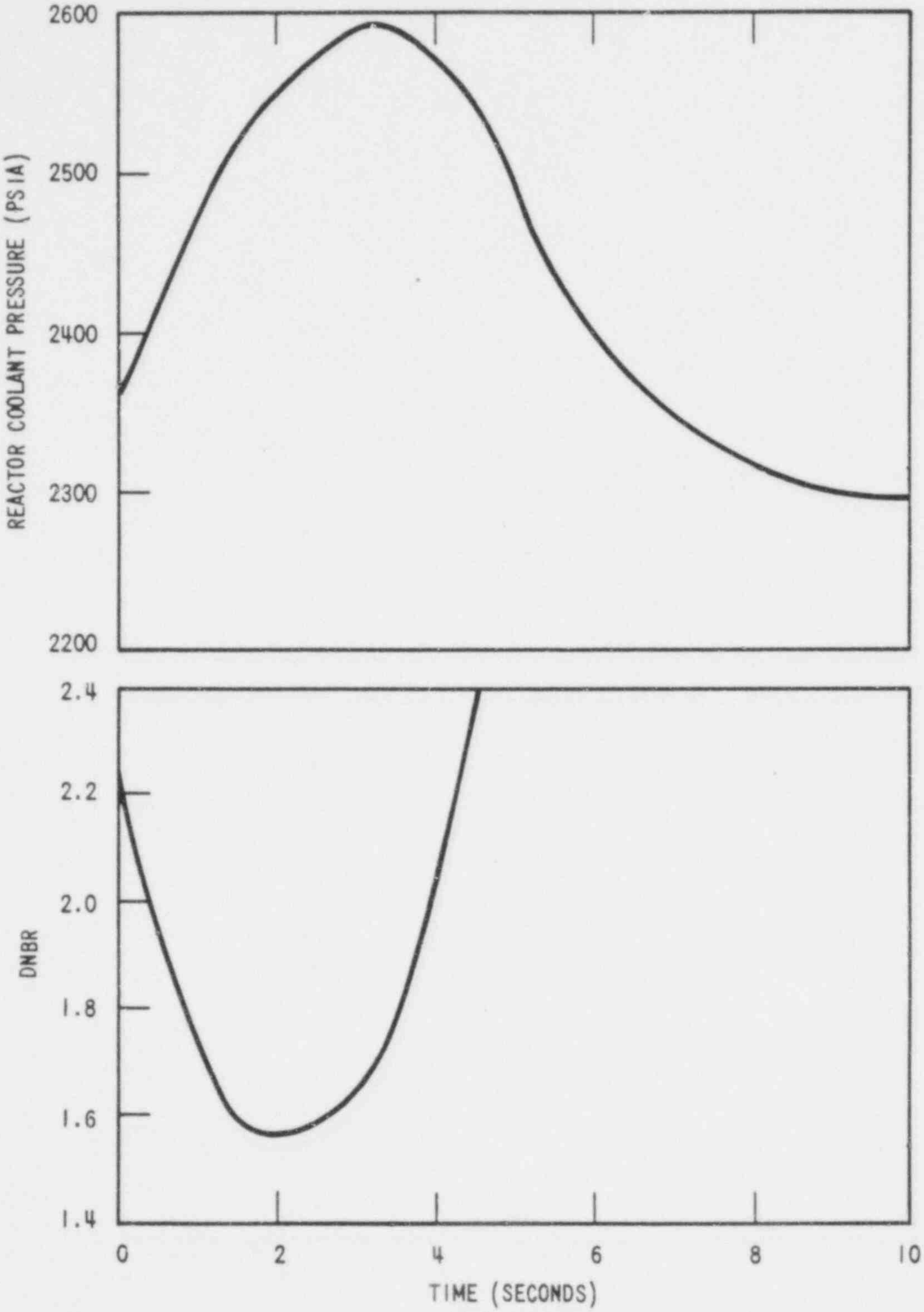
WCAP - 9500

Figure 15.3-17.

Flow Transients for Four Loops in Operation, One Locked Rotor

613 287

14,395-131



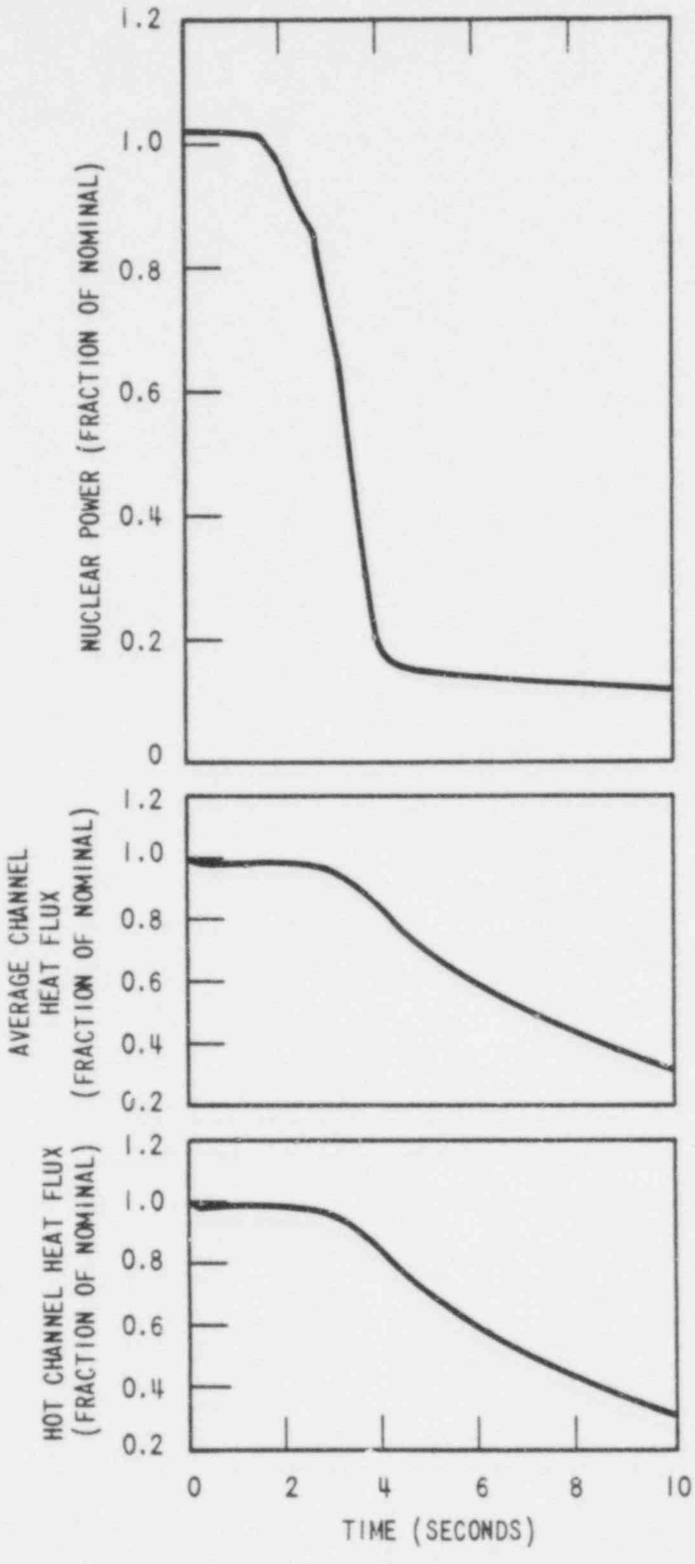
WCAP — 9500

Figure 15.3-18.

Reactor Coolant System Pressure Transient and DNBR Transient for Four Loops in Operation, One Locked Rotor

613 288

14,395-132



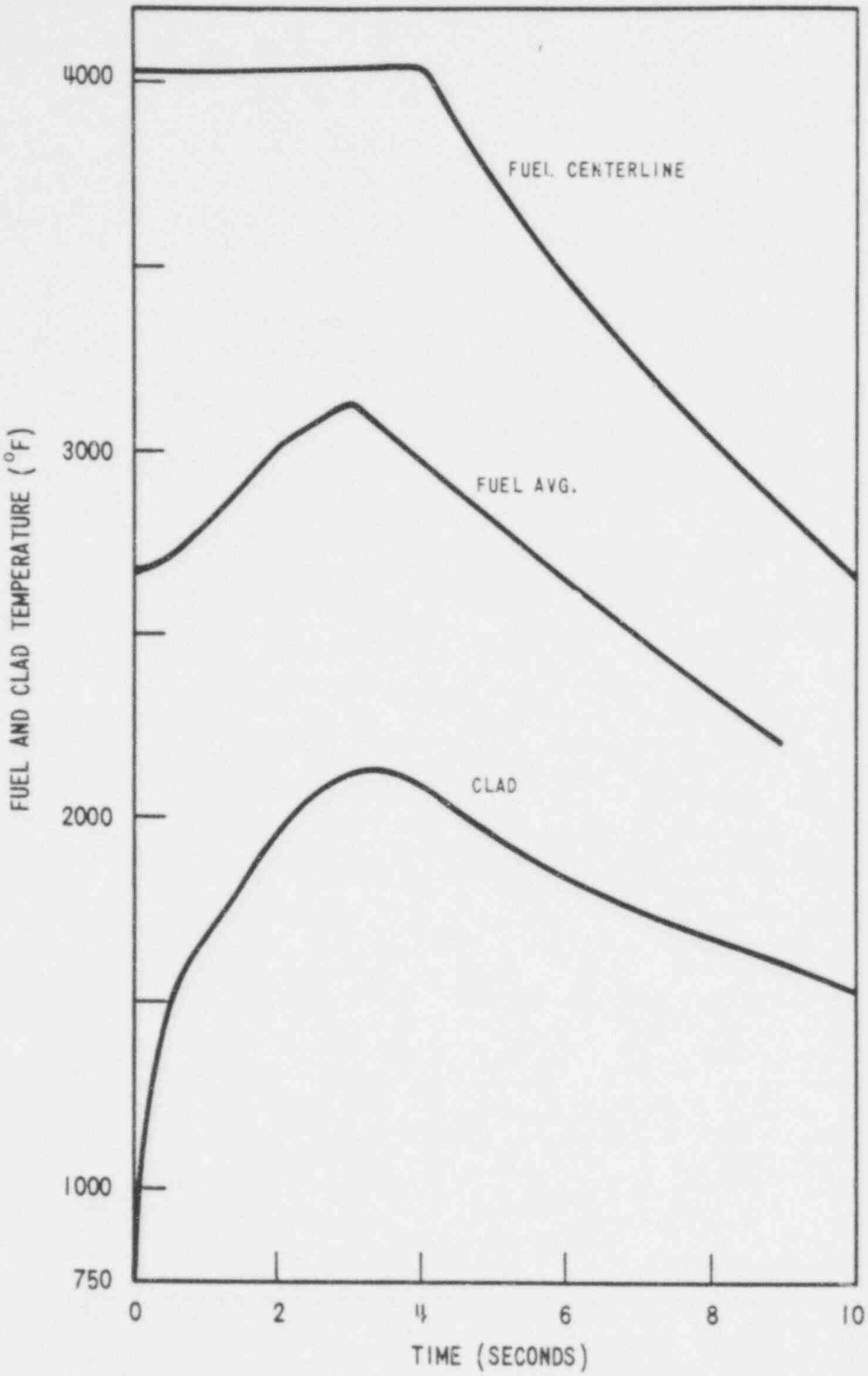
WCAP – 9500

Figure 15.3-19.

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

613 289

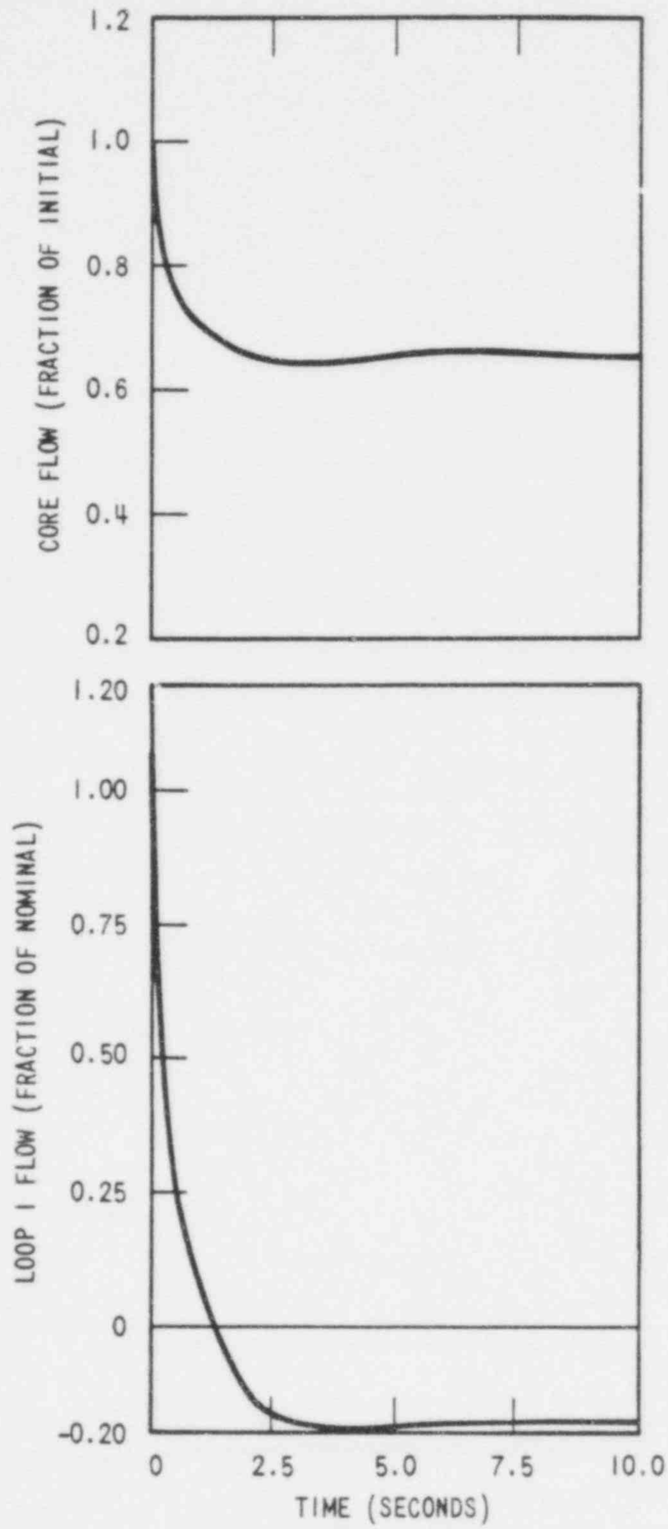
14,395-133



WCAP – 9500
Figure 15.3-20. Maximum Clad, Fuel Centerline, and Fuel Average Temperature at Hot Spot for Four Loops in Operation, One Locked Rotor

613 290

14,395-134



WCAP – 9500

Figure 15.3-21.

Flow Transients for Three Loops in Operation, One Locked Rotor

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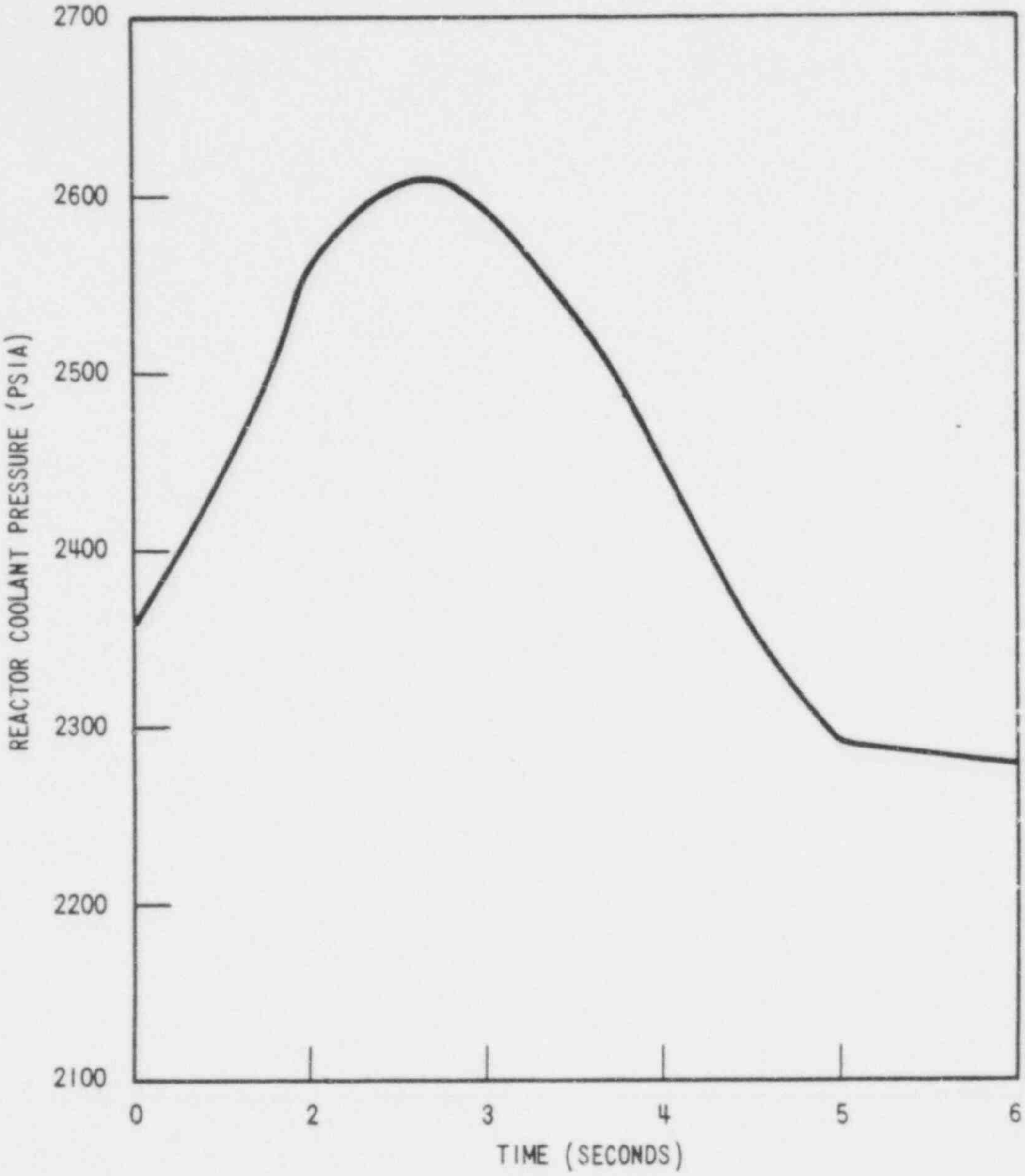

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Figure 15.3-22.

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

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14,395-136

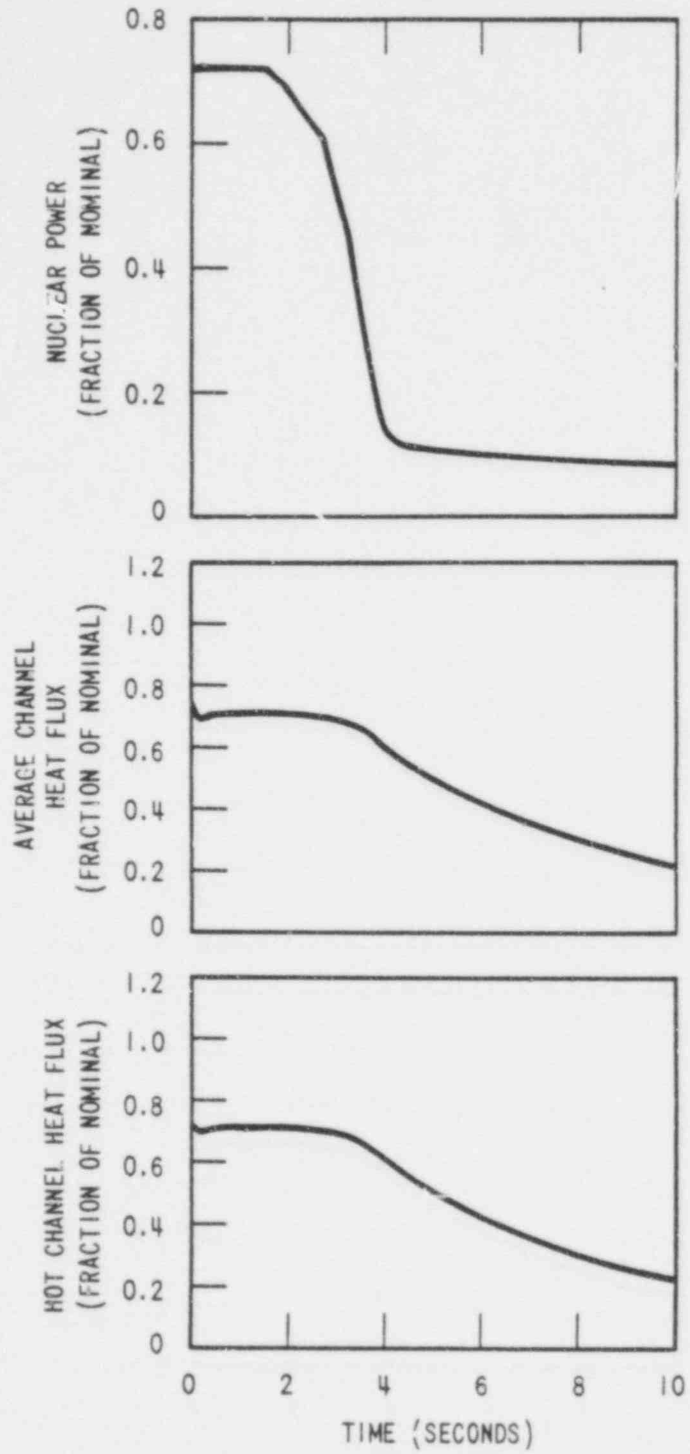

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Figure 15.3-23.

Nuclear Power Transient, Average and Hot Channel Heat Flux Transients for Three Loops in Operation, One Locked Rotor

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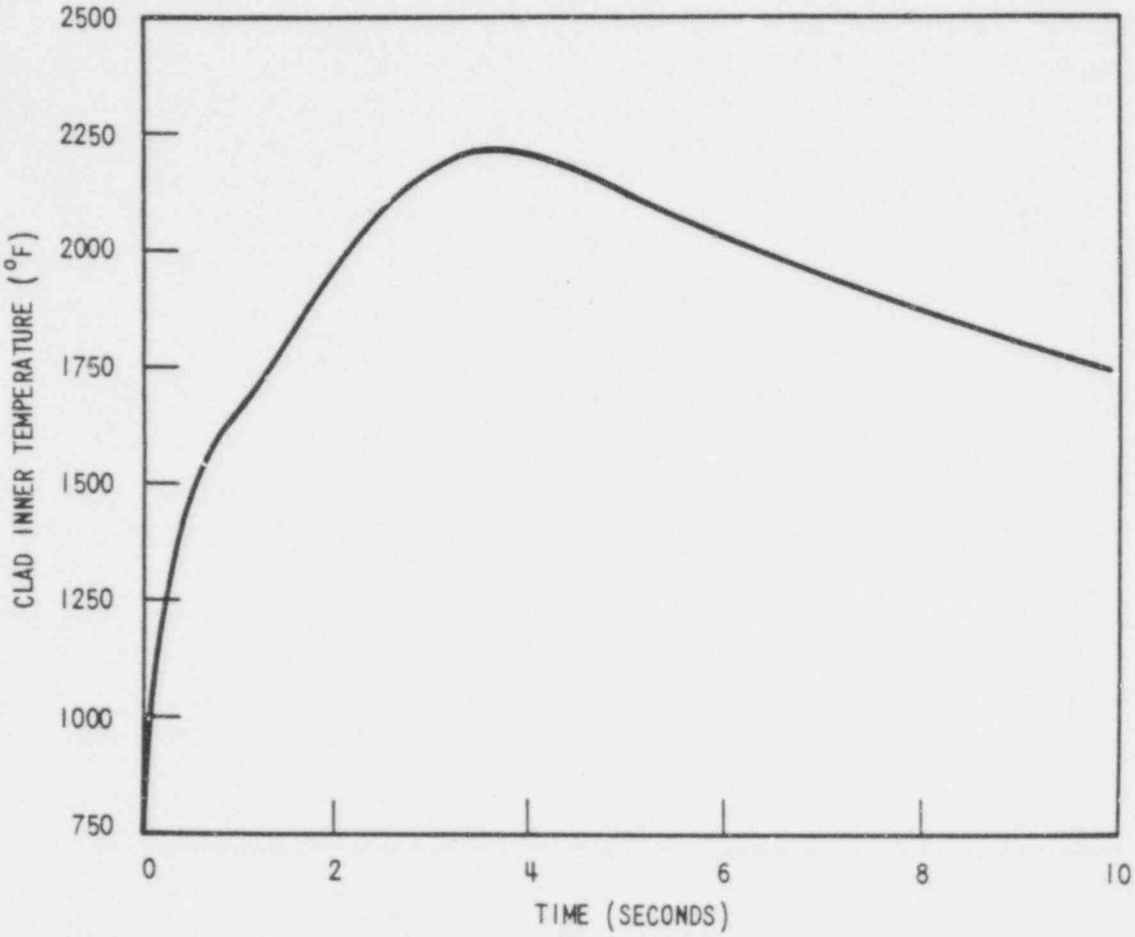

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Figure 15.3-24.

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

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15.4

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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 RCCA 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 this section:

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

2. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2).

3. Rod cluster control assembly misalignment (Subsection 15.4.3).

4. Startup of an inactive reactor coolant pump at an incorrect temperature (Subsection 15.4.4).

5. Malfunction or failure of the flow controller in a BWR (not applicable) (Subsection 15.4.5).

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

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

8. Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8).

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9. Spectrum of rod drop accident in a BWR (not applicable) (Section 15.4.9).

Items 1, 2, 4, and 6 above are considered to be ANS Condition II events, item 7 an ANS Condition III event, and item 8 an ANS Condition IV event. Item 3 entails both Condition II and III events. See Subsection 15.0.2.

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 withdrawal accident is defined as the 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 coolant or rod control systems. This could occur with the reactor either subcritical, at hot zero power or at power. The "at power" case is discussed in Subsection 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 Subsection 15.4.6).

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

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analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

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

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

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 out of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three out of the four channels indicate a power level below this value.

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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% of full power. This trip function may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three out of the four channels indicate a power level below this value.

4. Power range high neutron flux reactor trip (high setting)

Actuated 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 departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed using a

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spatial neutron kinetics code, 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 with the Improved Thermal Design Procedure as described in WCAP-8567. Plant characteristics and initial conditions are discussed in Subsection 15.0.4.

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 (least negative) values as a function of power are used. See Subsection 15.0.3, and Table 15.0-3.

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 slightly positive value, obtained by adjusting the initial boron concentration in the nuclear code, 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 larger fuel to 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 RCCA release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25% to 35%. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.6 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 highest combined worth at maximum speed (72 steps/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 sequential control banks in their highest 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). The 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 lowest initial flow minimizes the resulting DNBR. This assumption is conservative compared to assuming N or N-1 loops in operation.

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Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.9 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.4-1 through 15.4-3 show the transient behavior for the uncontrolled RCCA bank withdrawal, with the accident terminated by reactor trip at 35% of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region.

Figure 15.4-1 shows the neutron flux transient. The neutron flux does not overshoot the nominal full power value.

The energy release and the flux temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 15.4-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 15.4-3 shows the response of the hot spot fuel and cladding temperature. The hot spot fuel average temperature increases to a value lower than the nominal full power hot spot value. The minimum DNBR at all times remains above its limit value.

Since only the reactor coolant pumps were assumed to be in operation, these results are conservative compared to assuming N or N-1 loops in operation.

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

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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 which is always greater than the limit value. Thus, the DNB design-basis as described in Section 4.4 is met.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel cladding 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 (a fault of moderate frequency) as defined in Subsection 15.0.2.

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

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1. Power range neutron flux instrumentation actuates a reactor trip if two-out-of-four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four N-16 power signals exceed a low DNBR setpoint. This setpoint is automatically varied with axial power shape, $F_{\Delta H}$, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four high kw/ft channels exceed high kw/ft setpoint. The kw/ft calculation is automatically varied with axial power shape and F_{xy} 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-four level channels when the reactor power is above approximately 10% (Permissive-10).

6. Intermediate range neutron flux instrumentation actuates a reactor trip if one-out-of-two channels exceed an overpower setpoint.

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. Low DNBR (two-out-of-four).

3. High kw/ft (two-out-of-four).

The manner in which the Low DNBR trip provides protection over the full range of Reactor Coolant System conditions is described in Chapter 7. Figure 15.0-1 and 15.0-1a presents allowable reactor power as a function of coolant loop inlet temperature and power as a function of primary coolant pressure for the design flow and power distribution as described in Section 4.4. The boundaries of operation defined by the low DNBR trip are represented as "protection lines" on this diagram. During operation with one loop out of service, the Integrated Protection System will automatically select setpoints for the Low DNBR trip consistent with the core limits for N-1 loop operation. 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 DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.82 for the thimble cell and 1.85 for the typicall cell - see also Section 4.4). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value with the assumed axial and radial power distributions. The diagram shows that the DNB design basis is not violated 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); low DNBR (variable setpoint); high kw/ft (fixed setpoint).

15.4.2.2 Analysis of Effects and Consequences

Method of Analysis

This 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 temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation. Plant characteristics and initial conditions are discussed in Subsection 15.0.4.

In order to obtain conservative results for an uncontrolled rod withdrawal at power accident, 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. Reactivity Coefficients - two cases are analyzed:

   a. Minimum Reactivity Feedback. A least negative moderator temperature coefficient of reactivity and a least negative Doppler only power coefficient of reactivity (Figure 15.0-2) are assumed.

   b. Maximum Reactivity Feedback. A conservatively large negative moderator temperature coefficient and a most negative Doppler-only power coefficient are assumed.

3. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. For the N-1 loop cases, reactor trip is actuated when neutron flux reaches the P-8 trip point, conservatively assumed to be at 85% of nominal power. The low DNBR trip includes 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 having the maximum combined worth at maximum speed combined with boron dilution occurring at the maximum dilution rate.

6. Cases are analyzed for operation with four loops in service and for operation with three loops in service (N-1 loop operation).

The effect of RCCA movement on the axial core power distribution is accounted for by the axial power shape measurement as described in Chapter 7.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.9, and listed in Table 15.0-6. 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 15.4-4 through 15.4-6 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 15.4-7 through 15.4-9. Reactor trip on low DNBR 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 15.4-10 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

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provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and low DNBR trip channels. The minimum DNBR is never less than the limit value.

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

Figures 15.4-4a through 15.4-12a show similar results for RCCA bank withdrawal transients during plant operation with one loop out of service (N-1 loop operation).

The shape of the curves of minimum DNBR 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 15.4-11, for example, it is noted that:

1. For high reactivity insertion rates (i.e., between approximately 2.7 x 10^{-4} $\Delta\rho$/sec and 7.5 x 10^{-4} $\Delta\rho$/sec) reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature. As the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux.

2. The Low DNBR reactor trip initiates a trip when measured reactor coolant system parameters approach the DNB lines shown in Figure 15.0-1. As the reactivity insertion rate decreases below ~ 2.7 x 10^{-4} δk/sec, the rise in the reactor coolant temperature results in a trip initiated by the Low DNBR reactor trip.

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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% of its nominal value (i.e., the high neutron flux trip point 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 low DNBR reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118% of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will remain below the fuel melting temperature.

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

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

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.

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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 low DNBR 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. Thus, the DNB design-basis as described in Section 4.4 is met. 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 MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

1. A dropped RCCA,

2. A dropped RCCA bank,

3. Statically misaligned RCCA (see Table 15.4-2), and

4. Withdrawal of a single RCCA.

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Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal. 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 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 Subsection 15.0.2. However, the single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. A single RCCA in the control bank could be withdrawn since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures (probability for single random failure is on the order of 10^{-4}/year (refer to Subsection 7.7) or multiple serious operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of

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conditions is low. The limiting consequences may include slight fuel damage. Since this is 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 in accordance with General Design Criterion 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." (Emphasis has been added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that criterion established for the single rod withdrawal at power is appropriate and in accordance with General Design Criterion 25.

A dropped RCCA or RCCA bank is detected by:

1. A sudden drop in the core power level as seen by the nuclear instrumentation system,

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, or

5. Rod position indication.

Misaligned RCCA's are detected by:

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

2. Rod deviation alarm, or

3. Rod position indicators.

The resolution of the rod position indicator system is less than ±7.5 inches. Deviation of any RCCA from its group by twice this distance (~15 inches) will not cause power distributions worse than design limits. The deviation alarm alerts the operator to rod deviations with respect to group demand position in excess of 12 steps. If the rod deviation alarm is not operable, the operator is 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 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 low DNBR 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 Subsection 15.0.9 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

A dropped RCCA, dropped RCCA group, statically misaligned RCCA, and a single RCCA withdrawal are analyzed in the following paragraphs.

Method of Analysis for Dropped or Misaligned RCCA

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

Dropped RCCA Results

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.

Following the dropped rod event, the operator may manually retrieve the dropped rod following approved retrieval procedures.

Dropped RCCA Group Results

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 group. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed.

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

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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 the incident.

Statically Misaligned RCCA Results

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which bank D is fully inserted with one RCCA fully withdrawn; a 12-foot misalignment error. 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 15.4-2, 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 15.4-2 with one RCCA fully inserted, the 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 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 the operator realigns the RCCA group following approved procedures.

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.

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Single RCCA Withdrawal Results

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

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in 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.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCA's in the controlling bank. The transient will then proceed in the same manner as Case 1 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%

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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.

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 accident 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 operated at power, and assuming the secondary side of the steam generator in the inactive loop is not

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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.

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 (a fault of moderate frequency) as defined in Subsection 15.0.2.

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 flow, 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 nuclear power from LOFTRAN. The THINC Code (see 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.

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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 Subsection 15.0.4.

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 full flow value in approximately 28 seconds. This time is much greater than the expected startup time and is conservative for this analysis.

3. A conservatively large moderator density coefficient.

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

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 loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 85 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

7. The initial margin in terms of power to the core thermal limits (Figure 15.0-1) is the governing parameter in the DNB evaluation of this event. For this analysis an initial core thermal margin consistent with the minimum allowed thermal margin as specified in the Technical Specifications was assumed.

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Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.9 and listed in Table 15.0-6. 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 15.4-13 through 15.4-17. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the 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 15.4-13.

The calculated sequence of events for this accident is shown on Table 15.4-1. The transient results illustrated in Figures 15.4-13 through 15.4-17 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal 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.

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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 postulated 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 limiting DNBR; 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.

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

15.4.6.1 Identification of Causes and Accident Description

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. A boric acid blend system is provided to permit matching of the boron concentration of the 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.

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The opening of the primary 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 RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

The rate of addition of unborated makeup water to the reactor coolant system when it is not at pressure is limited by the capacity of the primary water supply pumps or the Boron Thermal Regeneration System. Normally, only one primary water supply pump is operating while the other is on standby. With the RCS at pressure, the maximum delivery rate is 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.

Information on the status of the reactor coolant makeup and Boron Thermal Regeneration System 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. The signals initiating those alarms will also cause the closure of control valves terminating the addition to the reactor coolant system.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.2.

15.4.6.2 Analysis of Effects and Consequences

Method of Analysis

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

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considered in this analysis. The hot shutdown case is bounded by the analysis for cold shutdown and hot standby. Table 15.4-1 contains the time sequence of events for this accident.

Dilution During Refueling

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 RCS from the potential source of unborated water.

Valves FCV-110B, FCV-111B, 8354, 8355, and 8361 in the CVCS (see Figure 9.3-3) will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup water 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 the RHR pumps. See the Technical Specifications for actual requirements.

Dilution During Cold Shutdown, Hot Standby and Startup

For an uncontrolled boron dilution during subcritical initial conditions, the transient will be terminated by the Source Range High Neutron Flux Reactor Trip. The trip is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint (always below 10^{-5} of Nominal Power). Trip actuation results in three simultaneous protective actions:

1. A minimum of -.7% $\Delta k/k$ worth of trip reactivity is inserted into the core to assure adequate shutdown margin. The .7% shutdown worth is always available whenever there is dilution capability (CVCS valves listed in discussion of dilution during refueling are not locked out).

2. Valves LCV-113A and LCV-112B (shown on Figure 9.3-1 sheet 3) automatically isolate the sources of dilution. Thus operator action (response) is not needed to terminate boron dilution from a cold shutdown, hot standby, or startup condition.

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3. Valves LCV-112C and 113B will automatically open to allow suction from the refueling water storage tank.

For the analysis of boron dilution from cold shutdown, hot standby, and startup conditions a digital computer code is used. The WIT-6 Code (Reference 5) is used to calculate the reactivity and nuclear power transient. The code includes the simulation of core thermal and hydraulic feedback equations. Thermal and hydraulic feedback, however, is negligible since dilution is automatically stopped before significant power levels are achieved. Thus the transient is primarily a function of the reactivity insertion rate. The reactivity insertion rate is conservatively calculated for each plant operating mode and assumed to be constant over the entire transient. WIT-6 calculates the time at which the source range setpoint is reached and the integral reactivity insertion following CVCS valve closure (dilution stops). The trip worth is then compared with the integral reactivity insertion, due to boron dilution, to assure subcriticality following reactor trip and CVCS valve closure.

In order to obtain conservative results from WIT-6 for the boron dilution transient, the following assumptions are made:

1. Reactor trip is assumed to be initiated by the source range high neutron flux. The setpoint is at its highest preselected value, 10^{-5} of nominal power.

2. The reactor is assumed to be shutdown by 1% Δk. Normal shutdown is at least 1.3% Δk.

3. The initial power level is assumed to be below the power level expected for any 1% shutdown condition.

4. The total trip worth is .7% Δk. This value corresponds to the worth of the initially withdrawn rods less an allowance for a stuck rod.

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Figure 15.4-18 shows minimum shutdown margin versus reactivity insertion rate following reactor trip and CVCS valve closure.

Dilution During Cold Shutdown

The following conditions are assumed for an uncontrolled boron dilution during cold shutdown:

1. The Reactor Coolant System is filled with borated water (approximately 1600 ppm) which assures at least a 1% shutdown margin.

2. Dilution flow is assumed to be the combined capacity of the reactor makeup water pump and the boron thermal regeneration system with the coolant system depressurized (approximately 470 gpm).

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

4. A minimum water volume (5866 ft^3) in the Reactor Coolant System is used. This corresponds to the coolant volume of the reactor vessel and residual heat removal loop.

Dilution During Hot Standby

The following conditions are assumed for an uncontrolled dilution during hot standby:

1. The Reactor Coolant System is filled with borated water (approximately 1620 ppm) which corresponds to at least a 1.3% shutdown margin.

2. Dilution flow is the combined capacity of the reactor makeup water pump and the boron thermal regeneration system with the coolant system at 2250 psia (approximately 631 gpm).

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3. A minimum water volume (10450 ft^3) in the Reactor Coolant System is used. This corresponds to the active volume of the reactor coolant system with one reactor coolant loop in operation and reverse flow in the inactive loops. The reverse flow in the inactive loops assures adequate mixing in the RCS.

Dilution During Startup

The following conditions are assumed for an uncontrolled dilution during startup:

1. Initial boron concentration and dilution flow are the same as hot standby.

2. A minimum water volume (10450 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 at Power

With the unit at power and the Reactor Coolant System at pressure, the dilution rate is limited by the combined capacity of the reactor makeup water pumps and the BTRS system (analysis is performed assuming the charging pumps and BTRS system are 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 dilution flow rate (671 gpm). The Reactor Coolant System volume assumed (10450 ft^3) corresponds to the active volume of the RCS minus the pressurizer volume. The analysis is also applicable to N-1 loop operation, since reverse flow in the inactive loop still assures adequate mixing between loops. Thus the active volumes will be the same, which results in reactivity addition rates and dilution times being nearly the same for three and four loop operating conditions.

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Results

For dilution during refueling:

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 control which isolates the RCS from the potential source of unborated water.

For dilution during cold shutdown:

For dilution during cold shutdown a peak reactivity insertion rate of ~ 3.8 pcm/sec results in no complete loss of shutdown margin (see Figure 15.4-18) with boron dilution automatically terminated.

For dilution during hot standby:

For dilution during hot standby a peak reactivity insertion rate of 2.7 pcm/sec results in no complete loss of shutdown margin (see Figure 15.4-18) with boron dilution automatically terminated.

For dilution during startup:

For dilution during startup a peak reactivity insertion rate of 2.7 pcm/sec results in no complete loss of shutdown margin (see Figure 15.4-18) with boron dilution automatically terminated.

For 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 31 minutes) to determine

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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 low DNBR 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 2.79 pcm/sec and is seen to be within the range of insertion rates analyzed. Prior to the low DNBR trip, a low DNBR alarm and turbine runback would be actuated. There is adequate time available (of the order of 31 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

No fuel damage occurs. The radiological consequences of this event would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

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The results presented above show that for all cases, either the dilution flow is automatically terminated, or 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.

For all cases, the reactor will be in a stable condition following termination of the dilution flow. The operator can then initiate reboration to recover the shutdown margin using the CVCS. If the reactor has tripped, operating procedures will also call for operator action to control 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.

15.4.7 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 errors 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 an identification number and loaded in accordance with a core loading diagram. After core loading, the identification numbers are verified for every assembly in the core.

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 Subsection 15.0.2.

15.4.7.2 Analysis of Effects and Consequences

Method of Analysis

Steady state power distribution in the x-y plane of the core are calculated using 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 plane for a correctly loaded core assembly are also given in Chapter 4.0 based on enrichments given in that section.

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For each core loading error analyzed, the percent deviation (from assembly average power) between the predicted detector readings for a normally loaded core and the perturbed core loadings (cases A, B, C and D) are shown for all incore detector locations (see Figures 15.4-21 through 15.4-25, inclusive).

Results

The following core loading error cases have been 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 15.4-21).

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 15.4-22 and 15.4-23).

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 15.4-24).

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Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4-25).

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 within 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 and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by adminstrative 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

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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 in the Westinghouse pressurized water reactors are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCA's, and minimizes the number of RCCA's 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.

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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.

Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the 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 required 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 [6]. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

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Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a complete RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings. The full length control rod drive mechanism is described in Subsection 3.9.4.

Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not be bent because of the rigidity of multiple adjacent housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the

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missile shield it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing were short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since 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.

Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is classified as an ANS Condition IV incident. See Subsection 15.0.2. Due to the extremely low probability of an RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

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Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 7). Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thesholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 8) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

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

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel.

2. Average clad temperature at the hot spot below the temperature at which cladding 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, and

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

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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 [9].

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 10), 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 rod worth and hot channel factor. Further description of TWINKLE appears in Subsection 15.0.12.

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Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal heat flux times the design hot channel factor (F_Q). 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 11) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNBR is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Subsection 15.0.12.

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System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in LOFTRAN (Reference 3). This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

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 15.4-3 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. The computer codes as described in Table 4.1-2 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 to provide worst case results.

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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 [9]. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power rodded configurations and compared to values used in the analysis. Experience has shown 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 kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis (Reference 9).

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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 previously, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady state computer code with a Doppler weighting factor of 1.0. The Doppler defect used is given in Subsection 15.0.5. The Doppler weighting factor will increase under accident conditions.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning-of-life and 0.50 percent at end-of-life for the first cycle. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff} as in zero power transients. In order to allow for future cycles, pessimistic estimates of β_{eff} 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 15.4-3 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to

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release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dash- pot does not occur until 3.3 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This conservatism is important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end-of-life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, and adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of two stuck RCCA's (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1 percent $\Delta\rho$. 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. ECCS is actuated on low pressurizer pressure are level within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (approximately 1200 psi depending on the system temperature) in about 8 minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2 percent $\Delta\rho$ due to the pressure coefficient. Adequate shutdown margin for cooldown is available for more than 10 minutes after the break. The addition of borated water by safety injection flow starting 1 minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

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Reactor Protection

As discussed in Subsection 15.4.8.1.1, reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the Reactor Trip System. No single failure of the Reactor Trip System will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

15.4.8.2.2 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 $\Delta\rho$ and an F_Q of 6.40, respectively. The peak hot spot clad average temperature was 2344^{o}F. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900^{o}F. However, melting was restricted to less than 10 percent 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 0.827 percent $\Delta\rho$ and a hot channel factor of 11.1. The peak hot spot clad temperature reached 2659^{o}F, the fuel center temperature was 4150^{o}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 $\Delta\rho$ and 6.40, respectively. This resulted in a peak clad temperature of 2125^oF. The peak hot spot fuel center temperature reached melting at 4800^oF. However, melting was restricted to less than 10 percent of the pellet.

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 with banks B and C at their insertion limits. The results were 0.91 percent $\Delta\rho$ and 18.1 respectively. The peak cladding and fuel center temperatures were 2700^oF and 4169^oF. 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 15.4-3. Results for the cases with N-1 loops in operation are presented in Table 15.4-3a. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning-of-life, full power and end-of-life, zero power) are presented in Figures 15.4-26 through 15.4-29.

The calculated sequence of events for the worst case rod ejection accidents, as shown in Figures 15.4-26 through 15.4-29, is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously in Subsection 15.4.8.2.2, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents are discussed in Subsection 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.

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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 9). Although limited fuel melting at the hot spot was predicted for the full power cases, it is highly unlikely that melting will occur since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

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 9). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. 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

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relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences 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 15.4-4.

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 radioactivity 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 9) 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 15.0-7.

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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).

   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.

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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

Conservative analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the fission product release, as a result of a number of fuel rods entering DNB, is limited to less than 10 percent of the fuel rods in the core. The radiological consequences of the rod ejection accident will be provided on a plant specific basis.

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS IN A BWR

This is not applicable.

15.4.10 REFERENCES

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

2. C. Hunin, "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

3. T. W. T. Burnett et al., "LOFTRAN Code Description," WCAP-7907, June 1972. Also supplementary information in letter from T. M. Anderson, NS-TMA-1802, May 26, 1978 and NS-TMA-1824, June 16, 1978.

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

5. D. B. Fairbrother and H. G. Hargrove, "WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, November 1972.

6. T. W. T. Burnett, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969.

7. T. G. (Ed) Taxelius, "Annual Report - Spert Project, October 1968, September 1969," Idaho Nuclear Corporation 1N-1370, June 1970.

8. R. C. Liimataninen and F. J. Testa, "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," ANL-7225, January-June 1966, p. 177, November 1966.

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

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

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

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TABLE 15.4-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS 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 Condition (Applies to N or N-1 loop in operation)	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	13.7
	Peak nuclear power occurs	13.9
	Rods begin to fall into core	14.2
	Minimum DNBR occurs	16.5
	Peak heat flux occurs	16.5
	Peak average clad temperatur occurs	16.6
	Peak average fuel temperature occurs	16.9

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TABLE 15.4-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
Uncontrolled RCCA Bank Withdrawal at Power			
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0.0	0.0
	Power range high neutron flux high trip point reached	1.8	7.1
	Rods begin to fall into core	2.3	7.6
	Minimum DNBR occurs	3.3	8.1
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (7 pcm/sec)	0	0
	Low DNBR reactor trip signal initiated	14.9	338.4

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TABLE 15.4-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

ACCIDENT	EVENT	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Rods begin to fall into core	16.9	340.4
	Minimum DNBR occurs	17.2	341.0
Startup of an inactive reactor coolant loop at an incorrect temperature (N-1 loop in operation)	Initiation of pump startup	0.0	
	Power reaches P-8 trip setpoint	12.2	
	Rods begin to drop	12.7	
	Minimum DNBR occurs	13.4	
Dilution during full power operation			
a. Automatic reactor control	Dilution begins	0.0	
	Shutdown margin lost		1860
b. Manual reactor control	Dilution begins	0.0	

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TABLE 15.4-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

ACCIDENT	EVENT	TIME (sec)
	Reactor trip setpoint for low DNBR	33
	Rods begin to fall into core	35
	Shutdown margin is lost (if dilution continues after trip)	1860
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
	Rods begin to fall into core	0.55
	Peak fuel temperature occurs	2.39
	Peak heat flux occurs	2.42
	Peak clad temperature occurs	2.42

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TABLE 15.4-1 (Continued)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

ACCIDENT	EVENT	TIME (sec)
2. End-of-Life, Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.18
	Peak nuclear power occurs	0.21
	Rods begin to fall into core	0.68
	Peak clad temperature occurs	1.53
	Peak heat flux occurs	1.53
	Peak fuel temperature occurs	2.49

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TABLE 15.4-2

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	***

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*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 (1.82 for thimble cell, 1.85 for typical cell); see Section 4.4.

TABLE 15.4-3

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	0.827	0.25	0.91
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	11.1	6.40	18.1
Number of operational pumps	4	2	4	2
Max. fuel pellet average temperature, ^{o}F	4144	3574	3718	3569
Max. fuel center temperature, ^{o}F	4978	4150	4833	4169
Max. clad average temperature, ^{o}F	2349	2659	2125	2700
Max. fuel stored energy, cal/gm	182	152	160	152
% Fuel Melt	<10%	0	<10%	0

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TABLE 15.4-3a

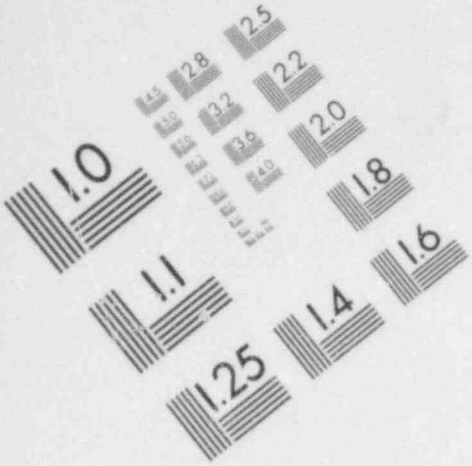
PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT FOR N-1 LOOP OPERATION

TIME IN LIFE	BEGINNING	END
Power Level, %	0.70	0.70
Ejected rod worth, %ΔK	0.25	0.25
Delayed neutron fraction, %	0.55	0.44
Feedback reactivity weighting	1.30	1.30
Trip Reactivity, %Δk	5.0	4.0
F_q before rod ejection	2.93	2.93
F_q after rod ejection	6.40	6.40
Number of operational pumps	3	3
Max. fuel pellet average temperature, ^{O}F	3493	3163
Max. fuel center temperature, ^{O}F	4880	4505
Max. clad average temperature, ^{O}F	1669	1570
Max. fuel stored energy, cal/gm	148	132
% fuel melt	0	0

TABLE 15.4-4

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



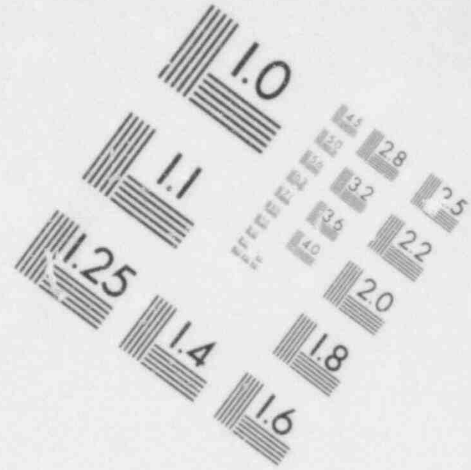
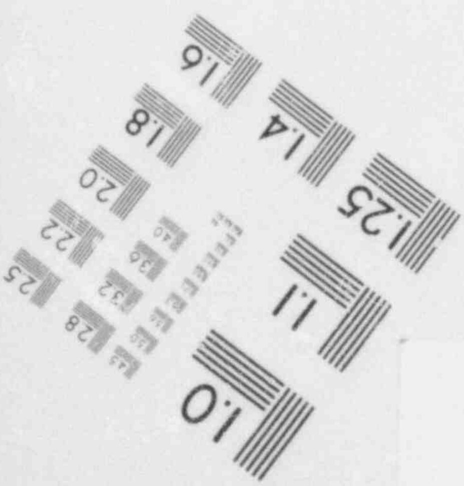
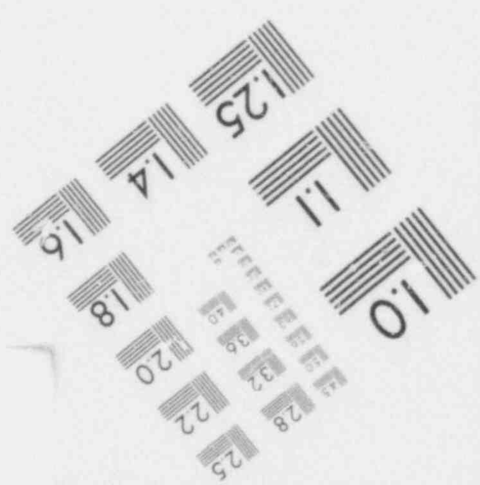
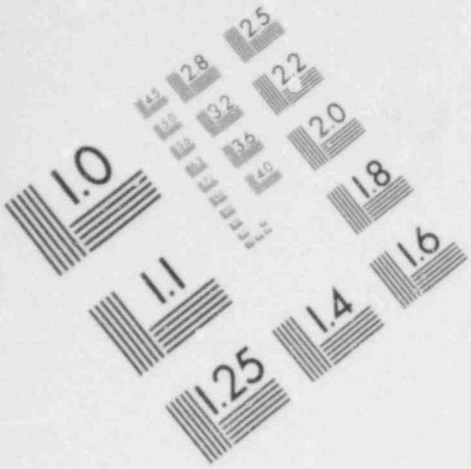


IMAGE EVALUATION
TEST TARGET (MT-3)

6″







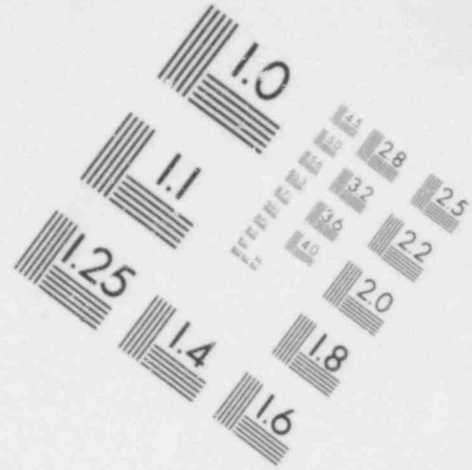
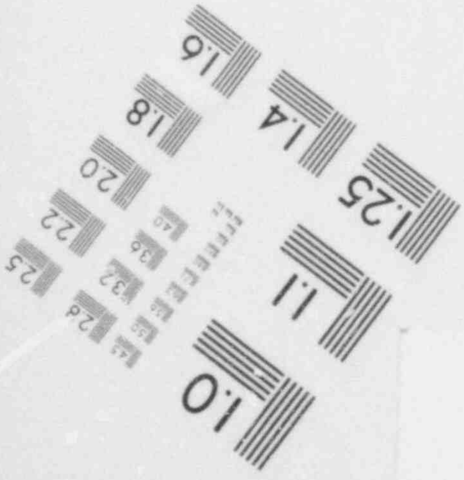


IMAGE EVALUATION
TEST TARGET (MT-3)



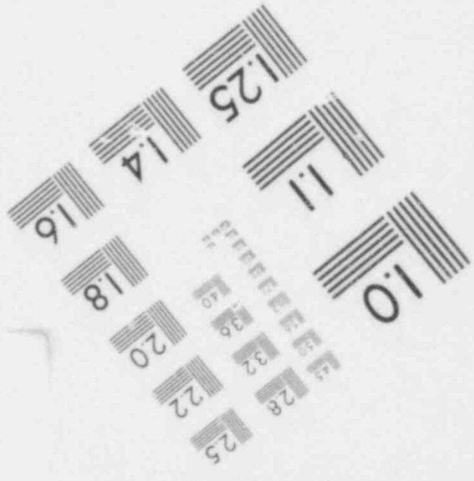


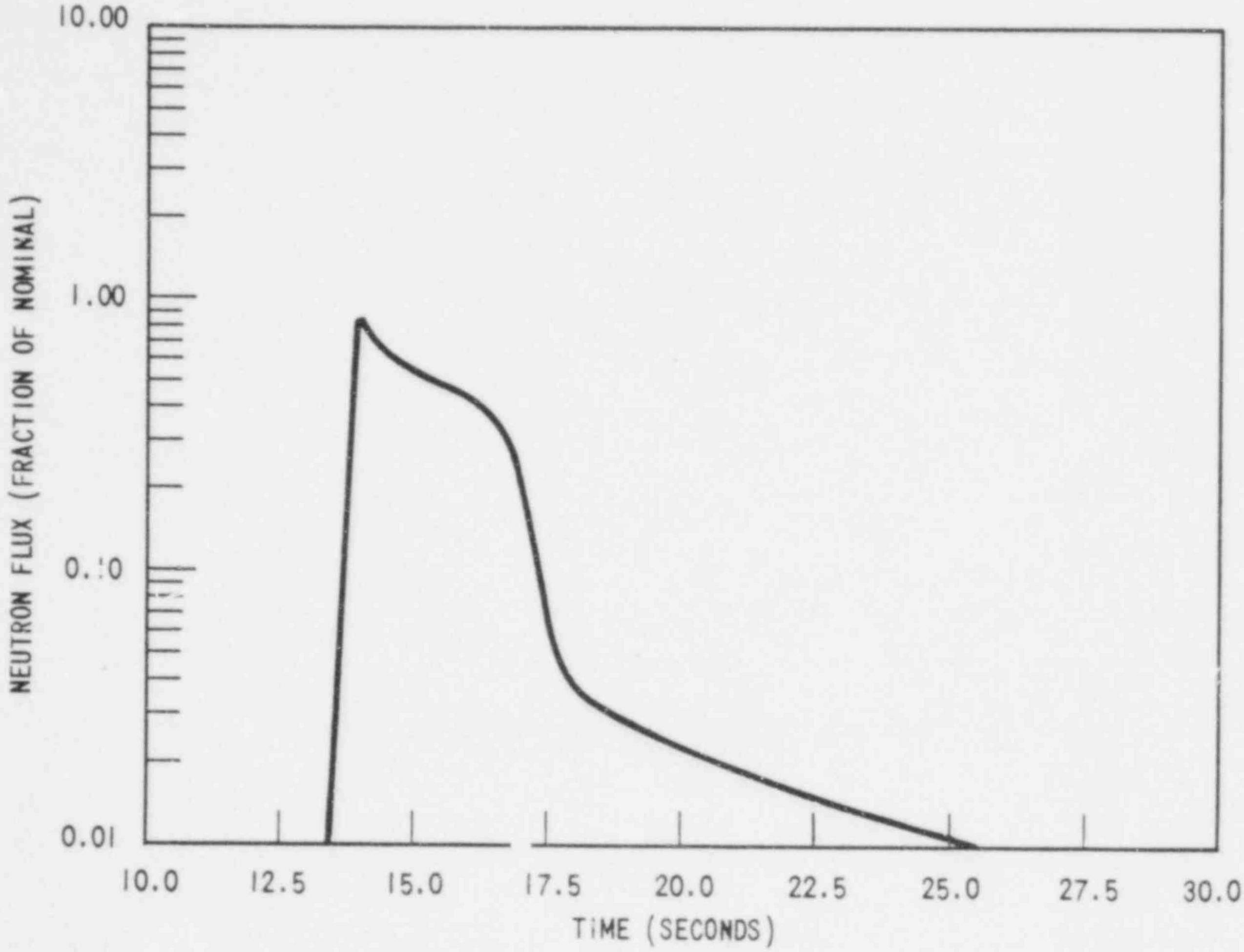
TABLE 15.4-4 (Cont'd)

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)

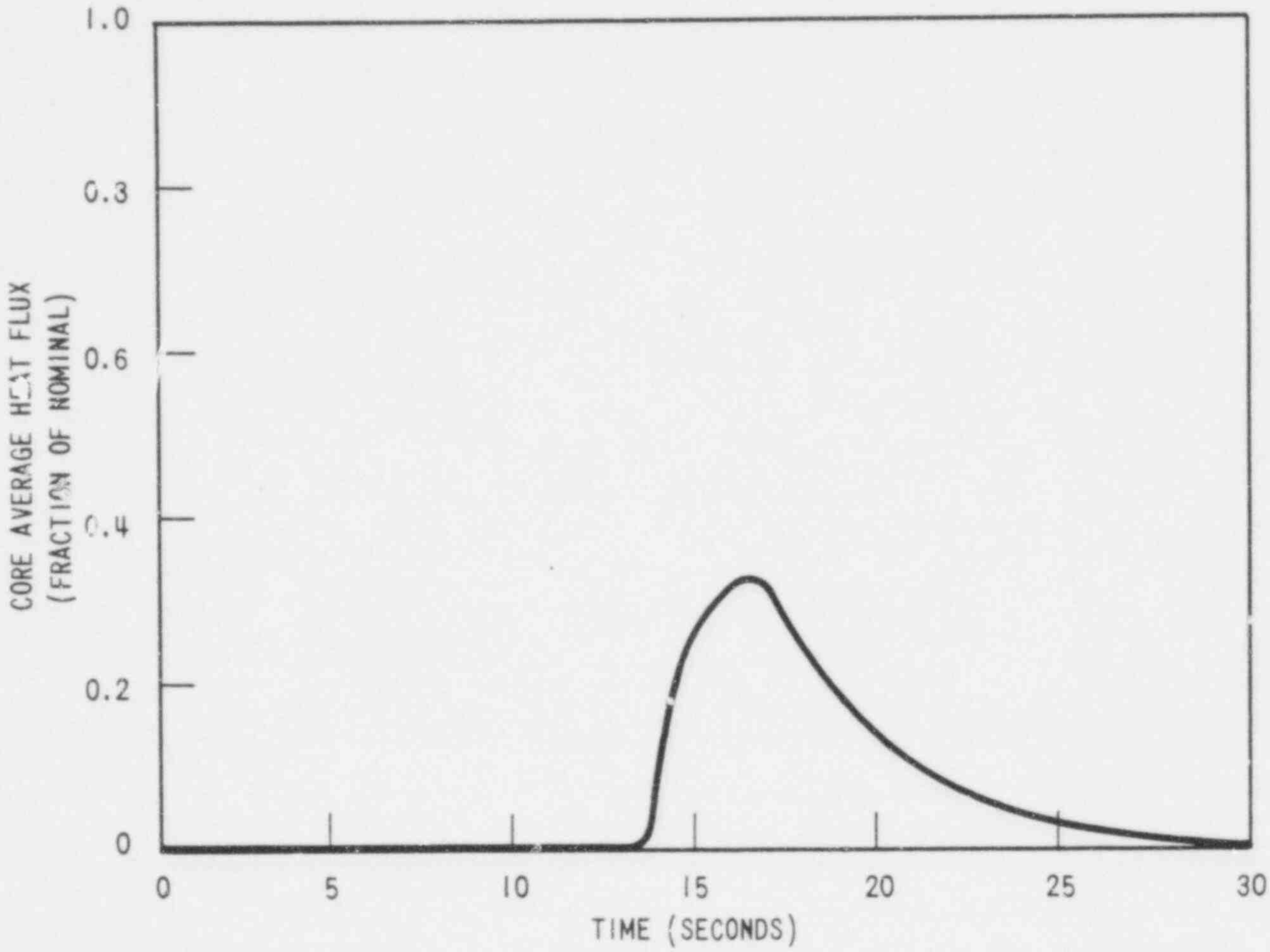


614 002

WCAP - 9500

Figure 15.4-1.

Neutron Flux Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition

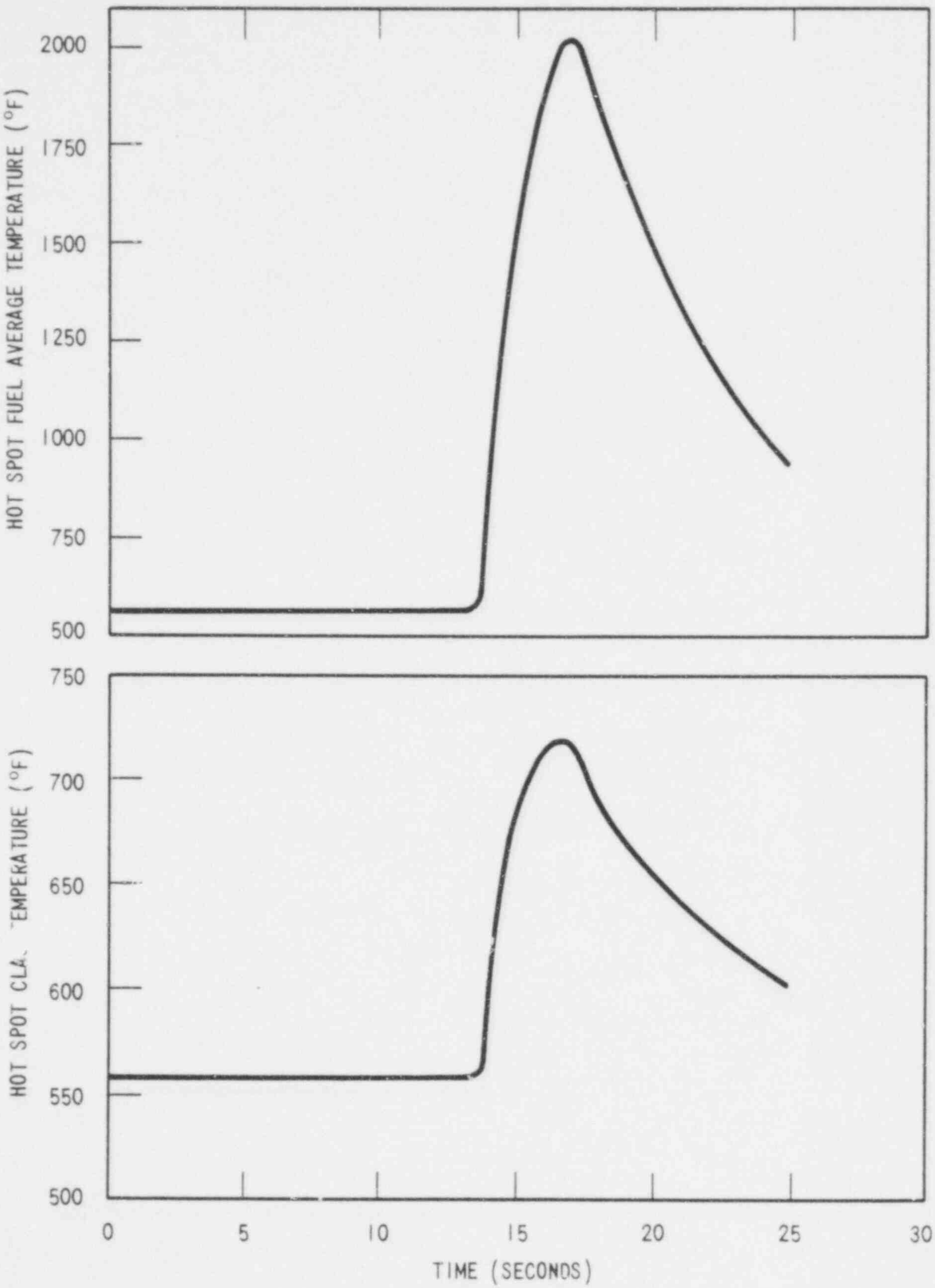


614 003

WCAP-9500

Figure 15.4-2.

Thermal Flux Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition

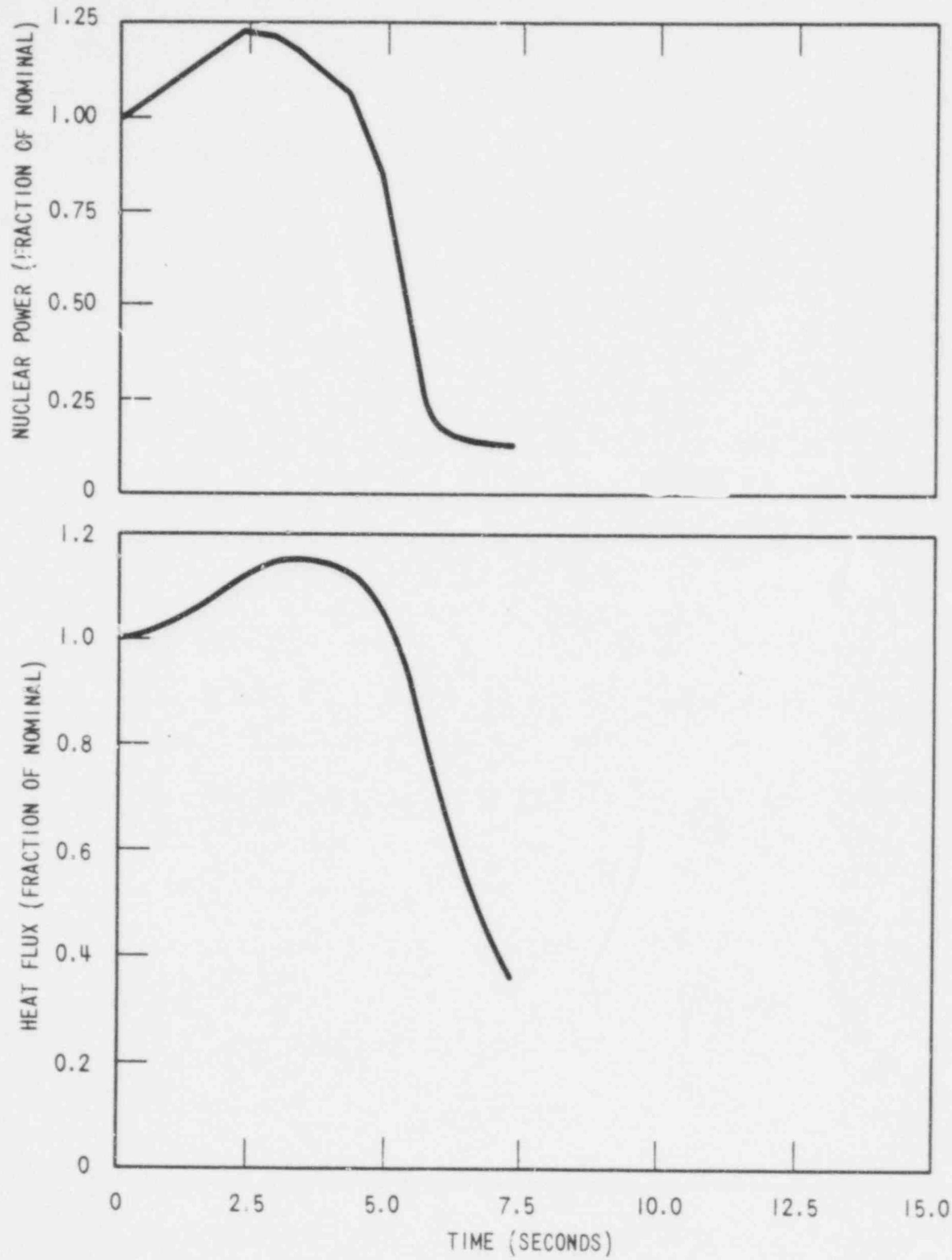


WCAP – 9500

Figure 15.4-3.

Fuel and Clad Temperature Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition

614 004

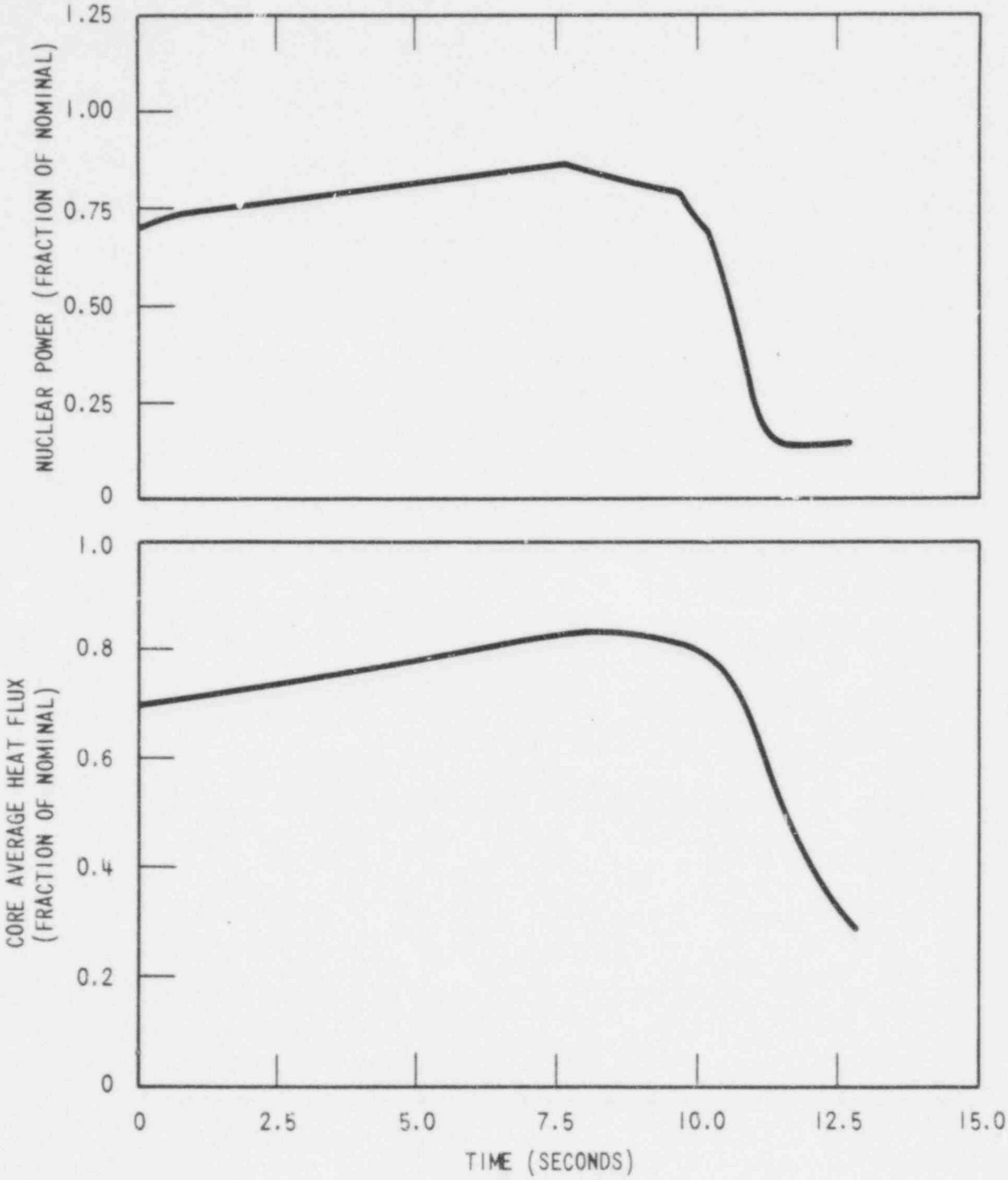


614 005

WCAP-9500

Figure 15.4-4.

Nuclear Power Transient and Heat Flux Transient for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

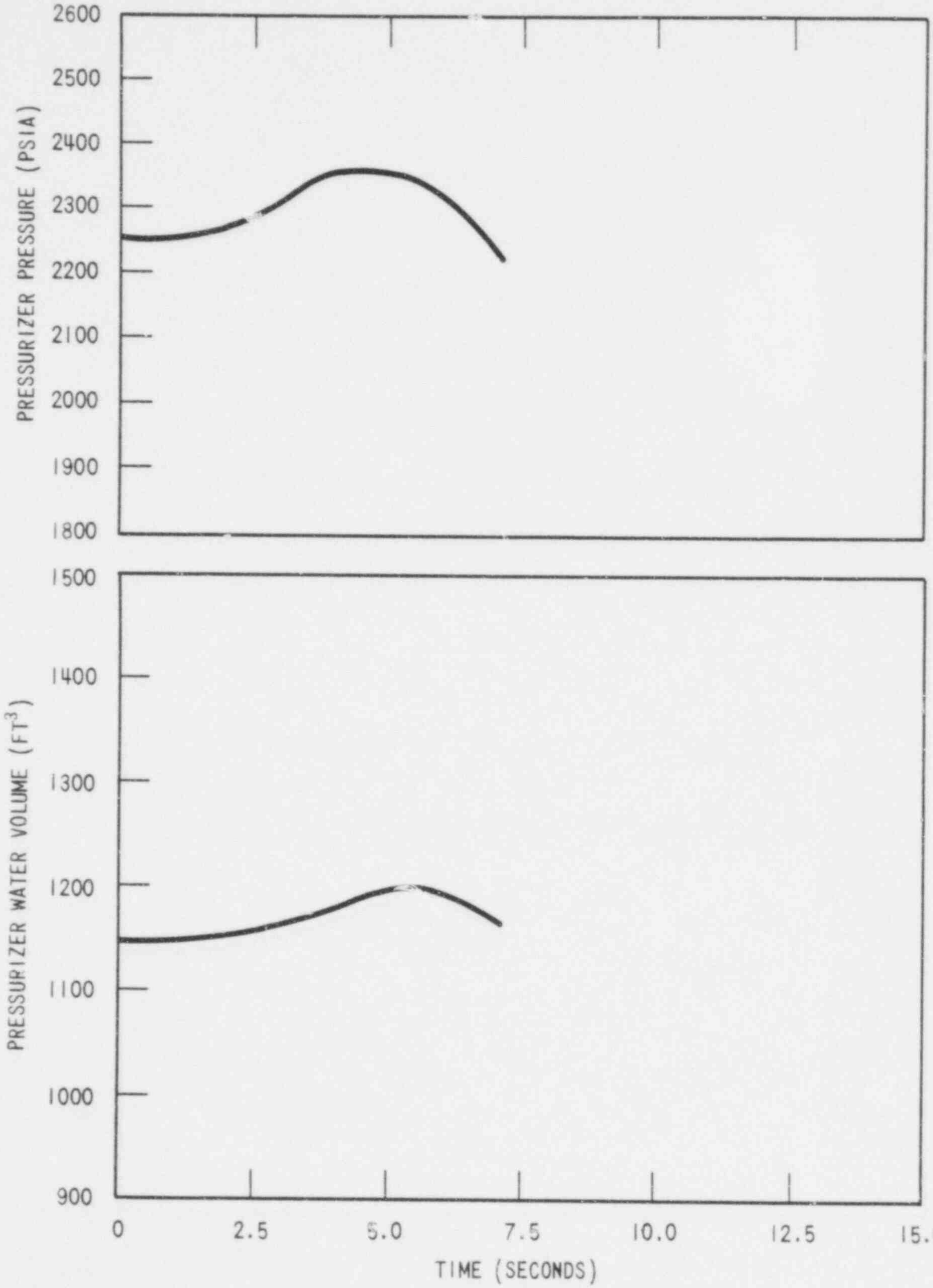


614 006

WCAP – 9500

Figure 15.4-4a.

Nuclear Power Transient and Heat Flux Transient for Uncontrolled Rod Withdrawal from 70% Power With Maximum Feedback and 75 PCM/SEC Withdrawal Rate (N-1 Loop Operation)

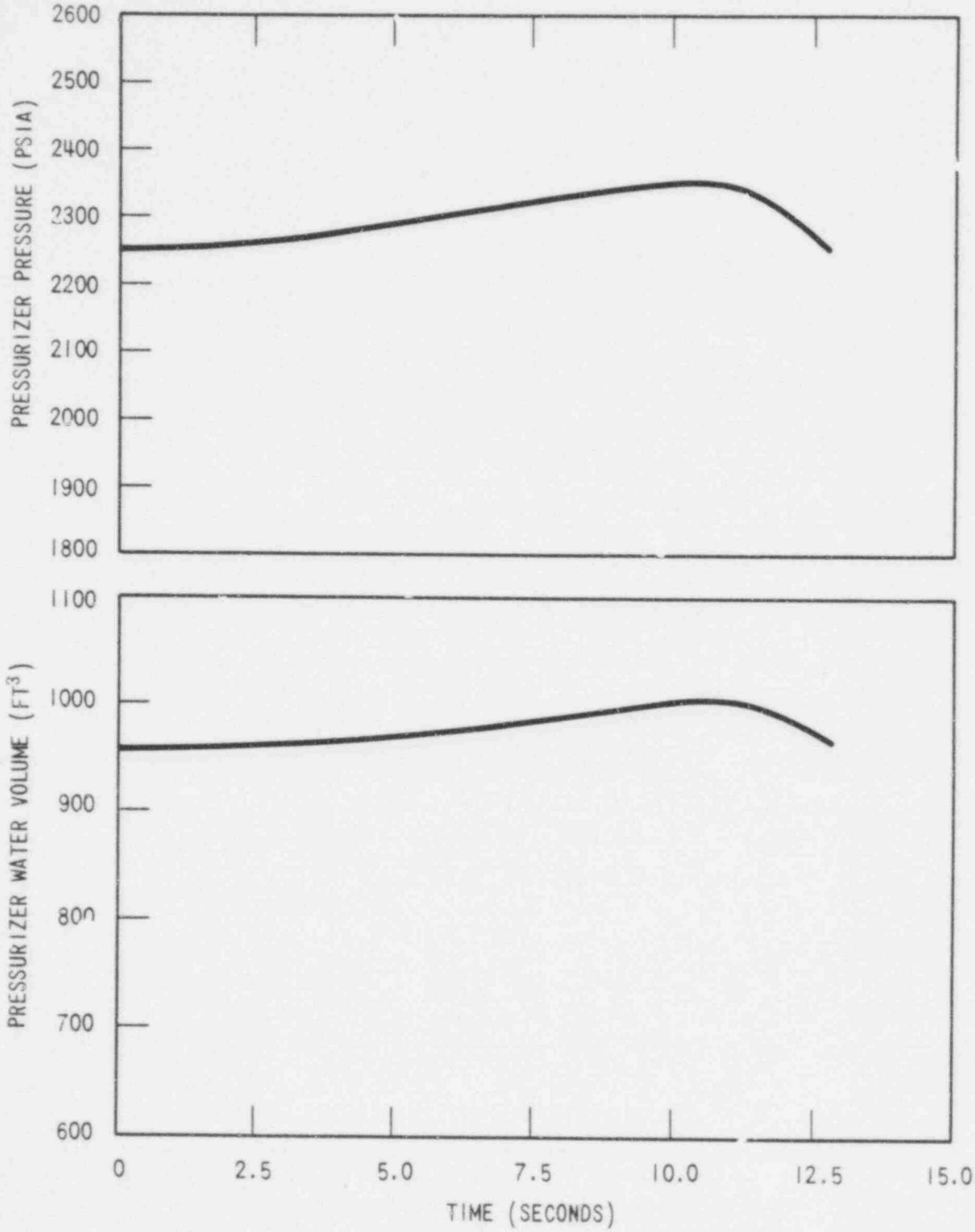


WCAP - 9500

Figure 15.4-5.

Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

614 007

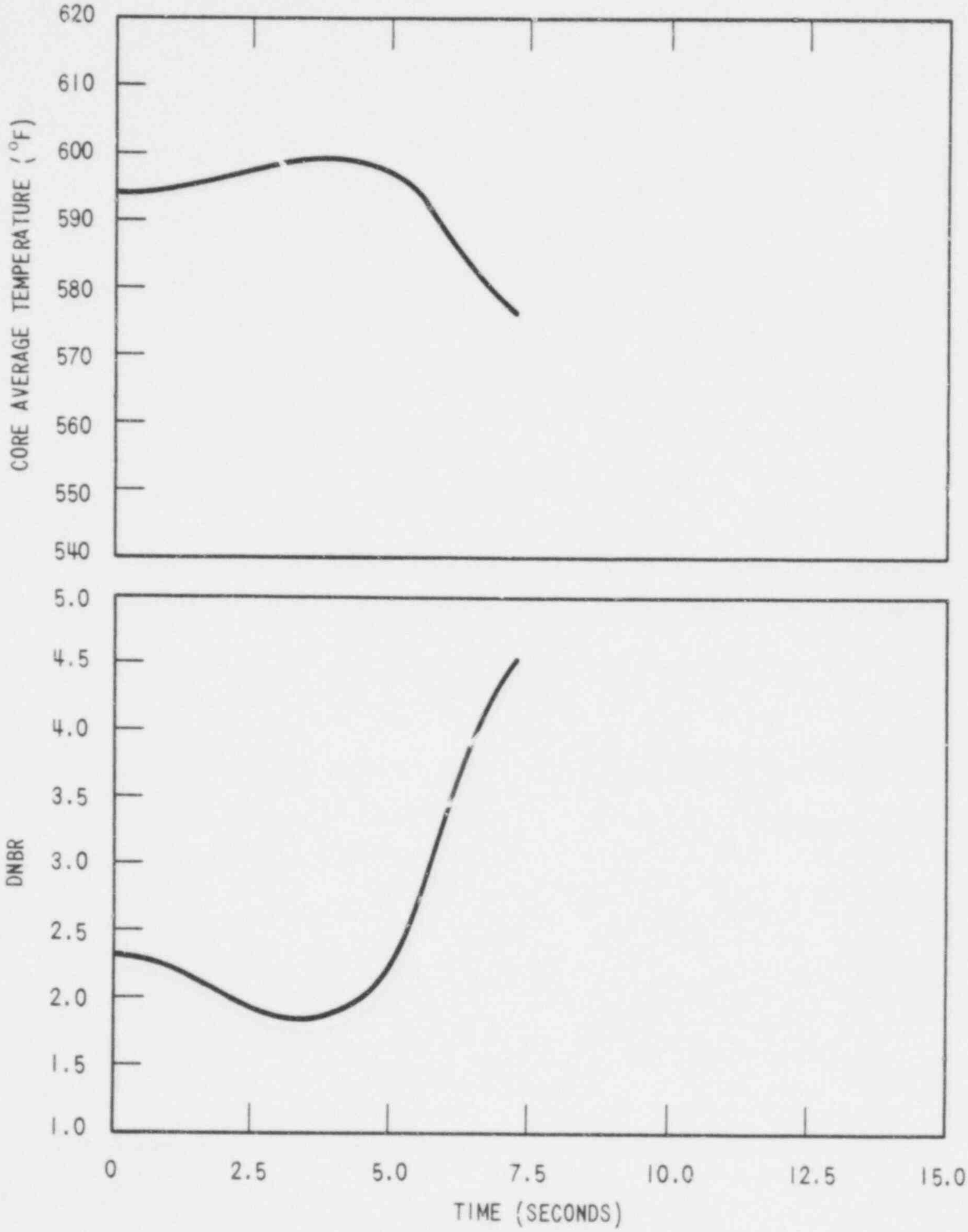


WCAP – 9500

Figure 15.4-5a.

Pressurizer Pressure and Water Volume Transients for Uncontrolled Rod Withdrawal from 70% Power With Maximum Feedback and 75 PCM/SEC Withdrawal Rate (N-1 Loop Operation)

614 008

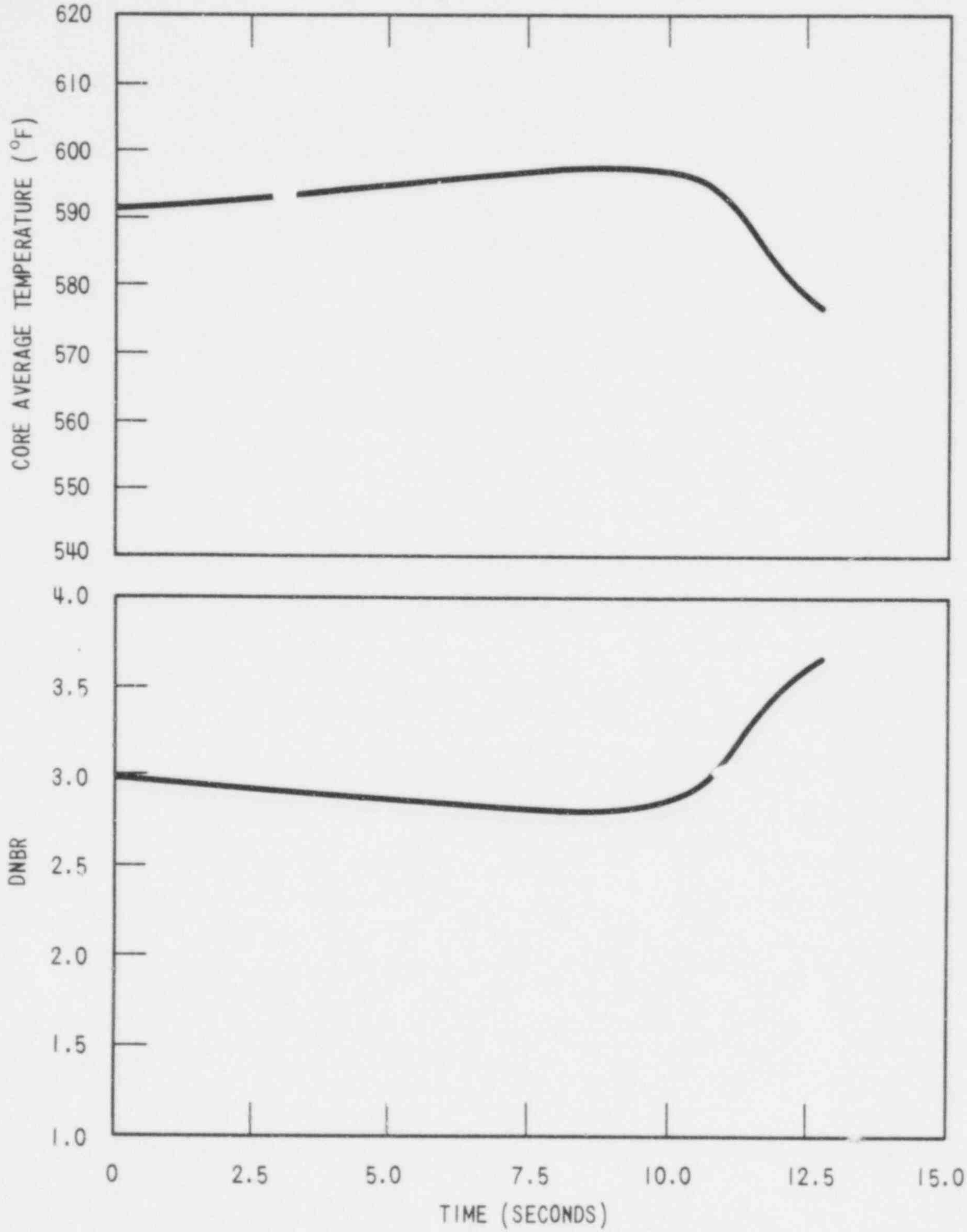


WCAP - 9500

Figure 15.4-6.

Core Average Temperature Transient and DNBR vs. Time for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 75 PCM/SEC Withdrawal Rate

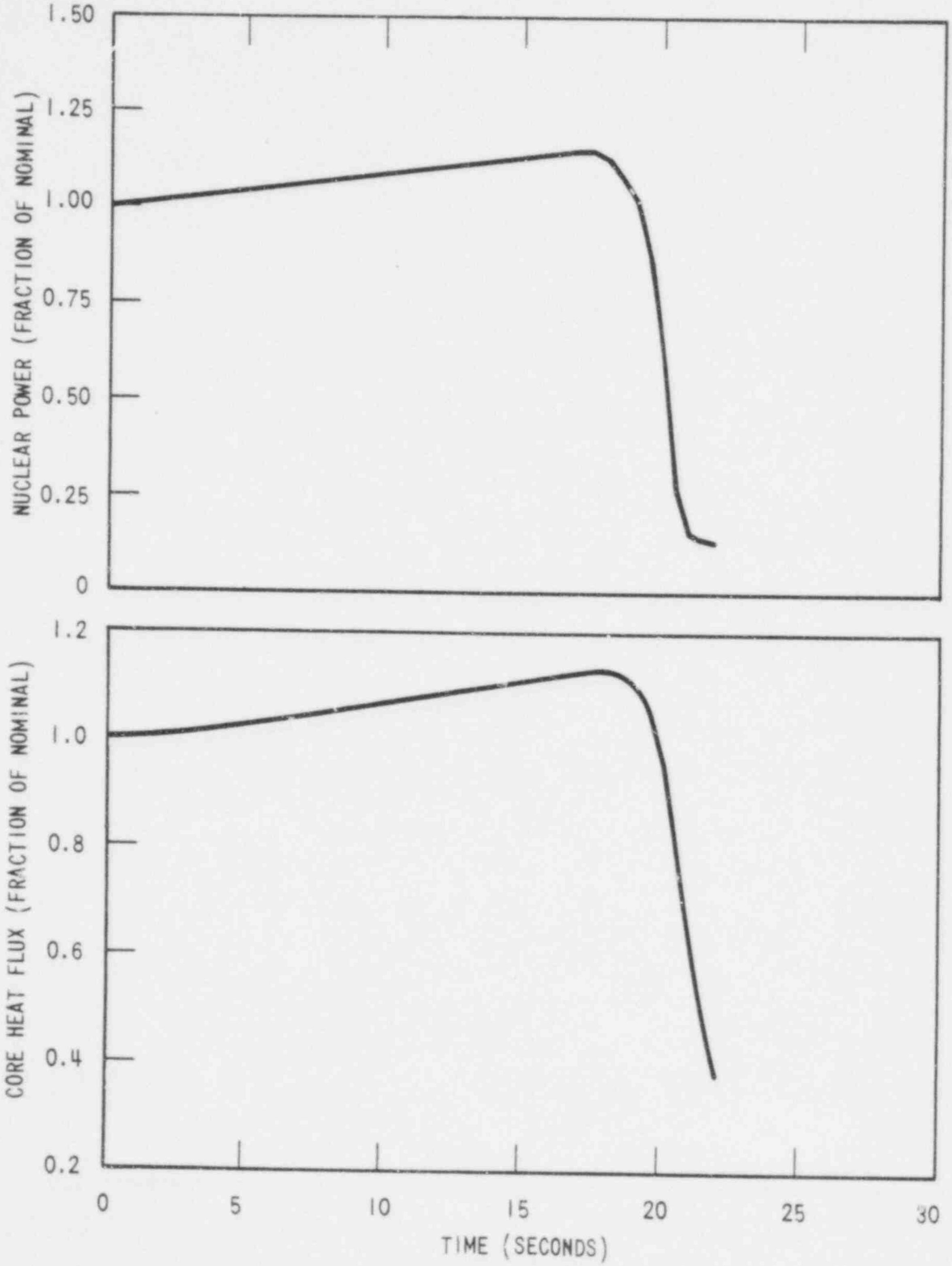
614 009



WCAP – 9500

Figure 15.4-6a.

Core Average Temperature Transient and DNBR vs. Time for Uncontrolled Rod Withdrawal from 70% Power With Maximum Feedback and 75 PCM/SEC Withdrawal Rate (N-1 Loop Operation)



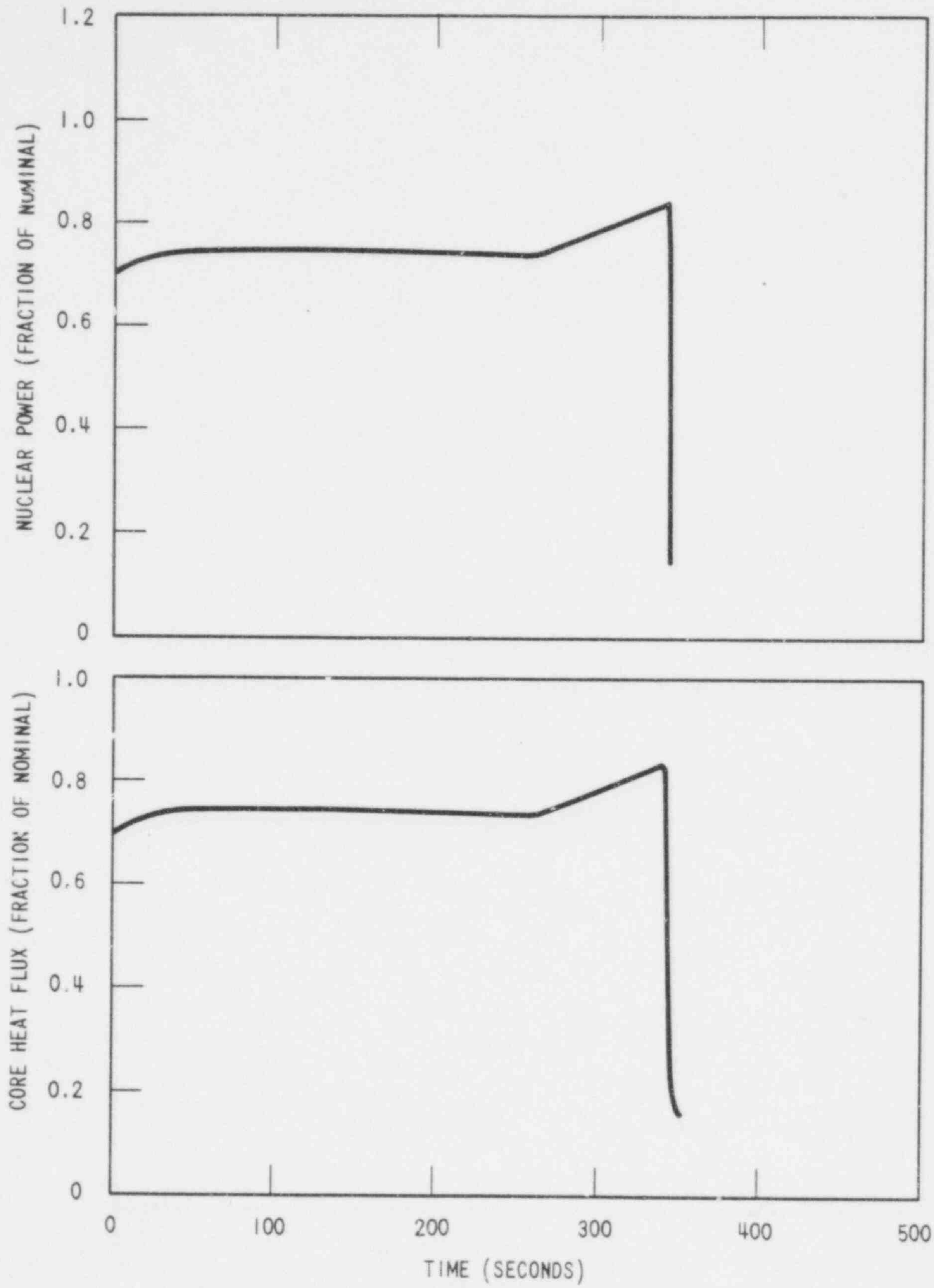
WCAP - 9500

Figure 15.4-7.

Nuclear Power Transient and Heat Flux Transients for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 7 PCM/SEC Withdrawal Rate

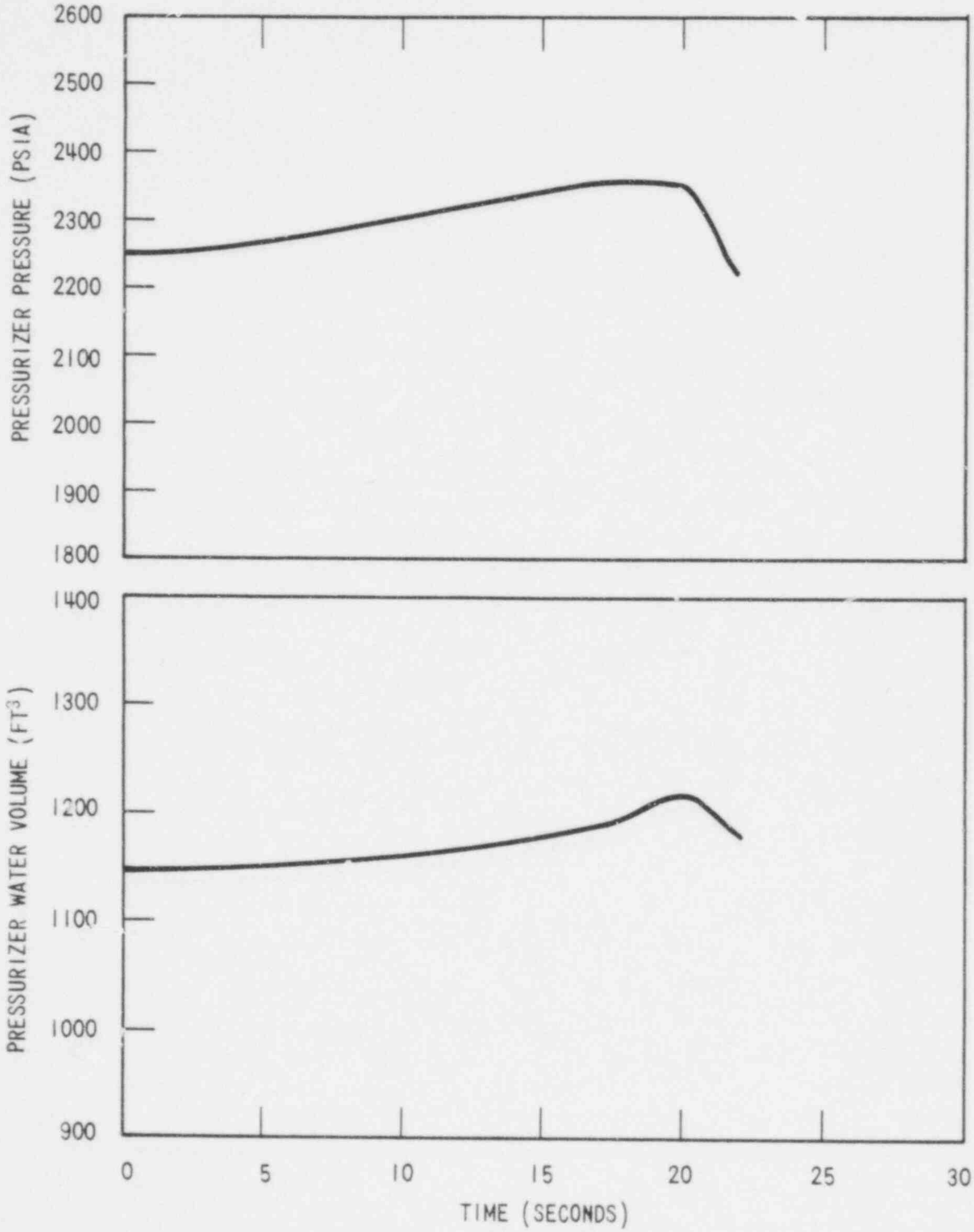
614 011

14,395-153



WCAP – 9500
Figure 15.4-7a. Nuclear Power Transient and Heat Flux Transient for Uncontrolled Rod Withdrawal from 70% Power With Maximum Feedback and 7 PCM/SEC Withdrawal Rate (N-1 Loop Operation)

614 12

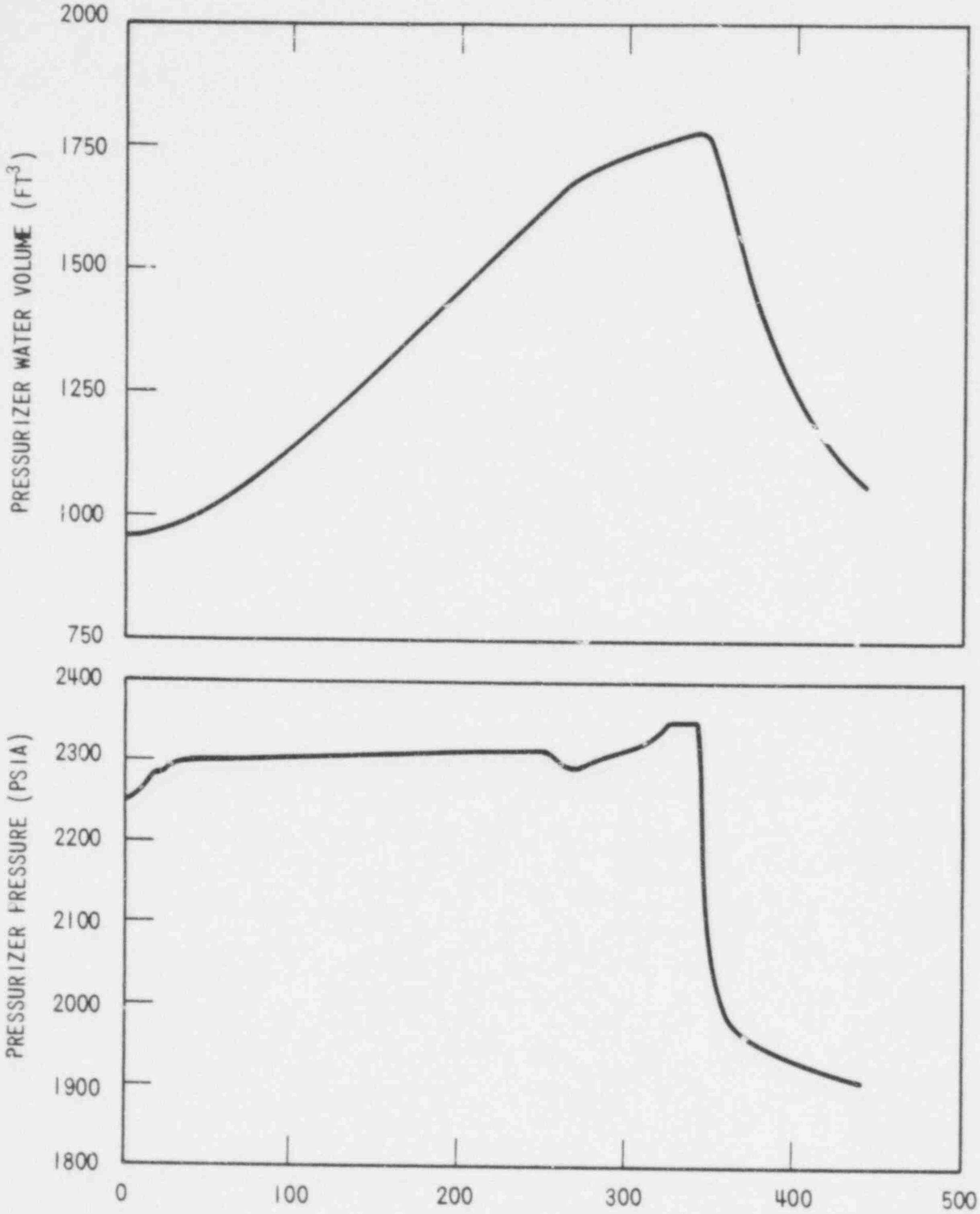


WCAP – 9500

Figure 15.4-8.

Pressurizer Pressure and Water Volume Transients for Uncotnrolled Rod Withdrawal from Full Power With Minimum Feedback and 7 PCM/SEC Withdrawal Rate

614 013



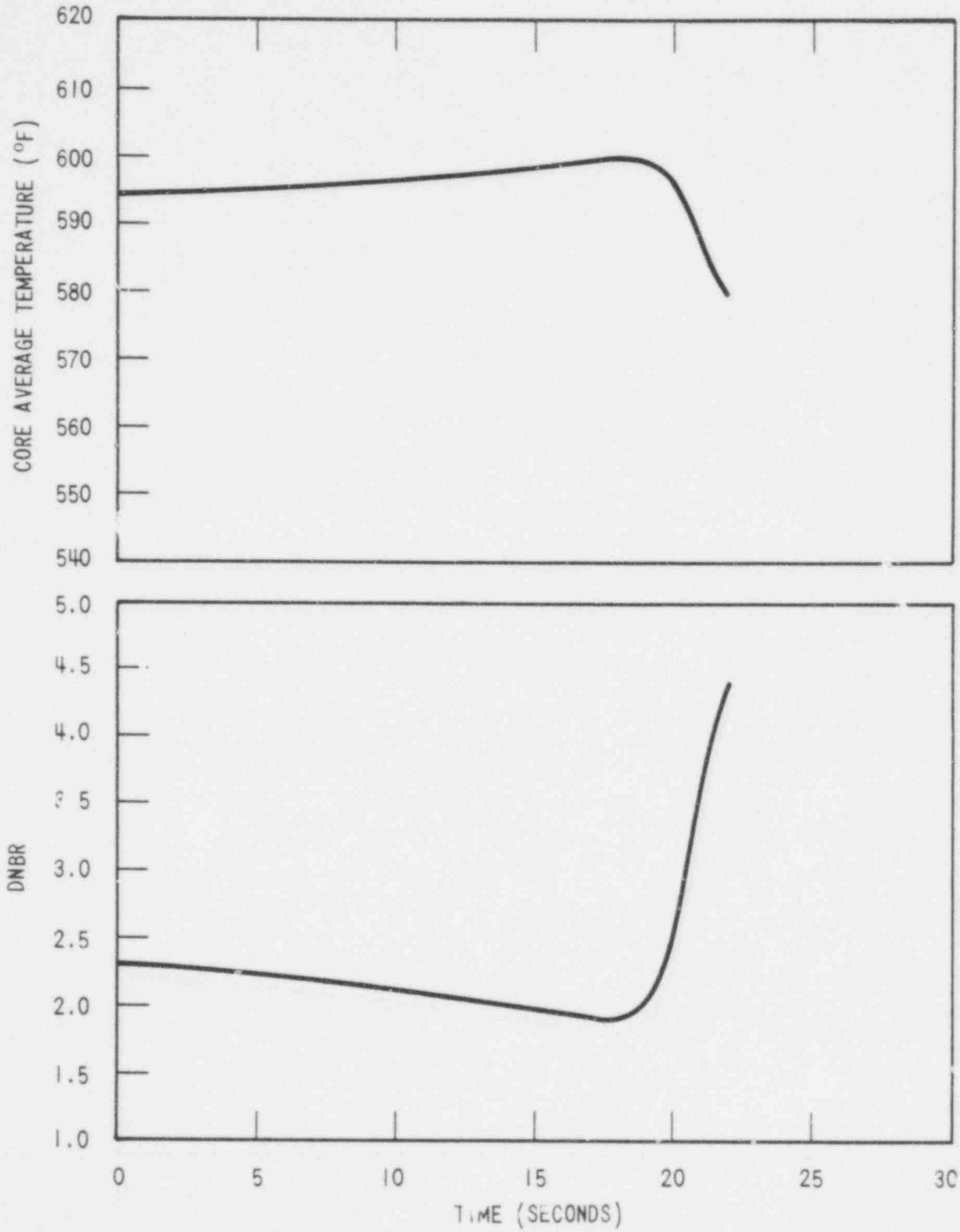
WCAP – 9500

Figure 15.4-8a.

Pressurizer Pressure and Water Volume Transients for Uncotnrolled Rod Withdrawal from 70% Power With Maximum Feedback and 7 PCM/SEC Withdrawal Rate (N-1 Loop Operation)

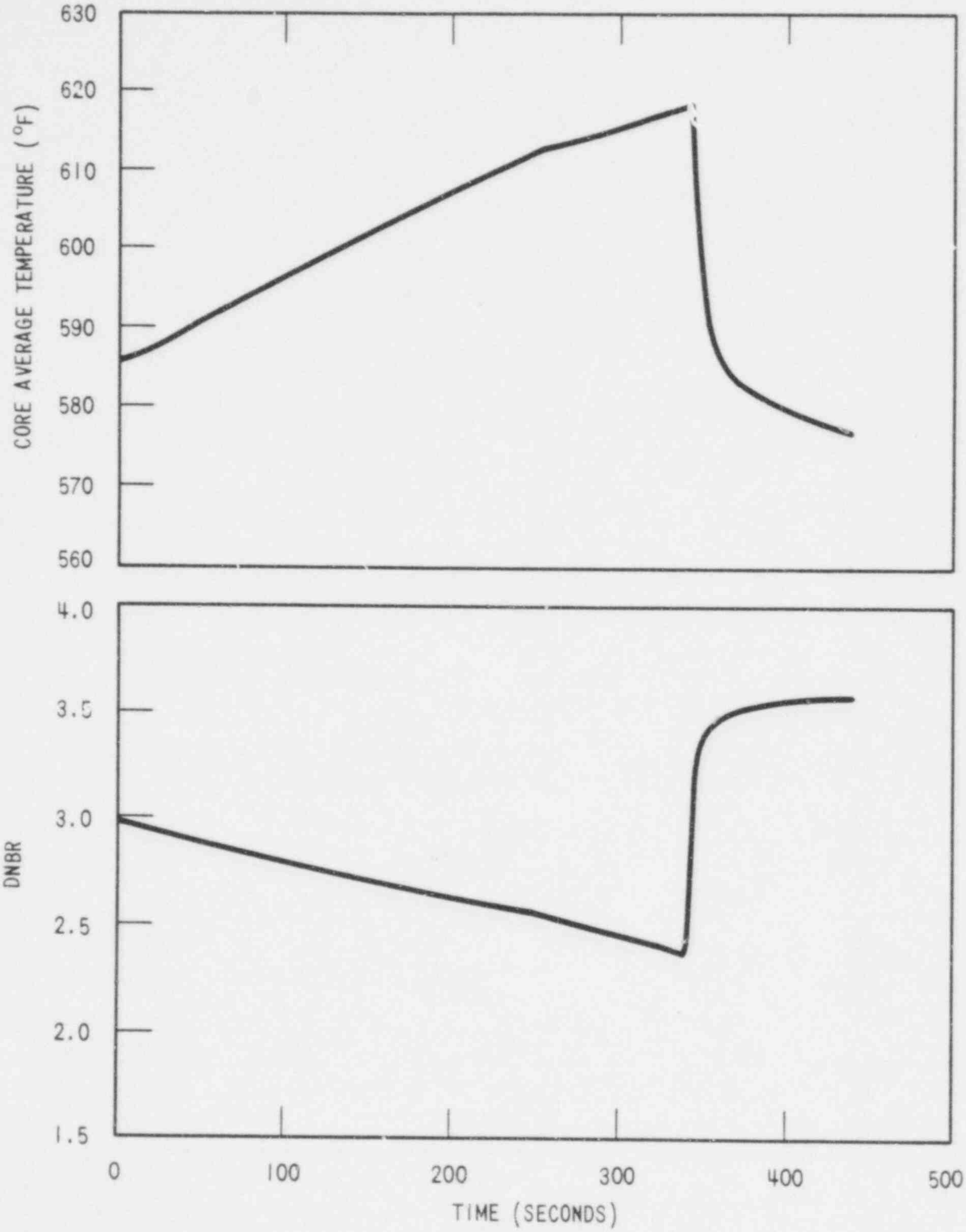
614 014

14,395-146



WCAP – 9500
Figure 15.4-9. Core Average Temperature Transient and DNBR vs. Time for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 1 PCM/SEC Withdrawal Rate

614 015

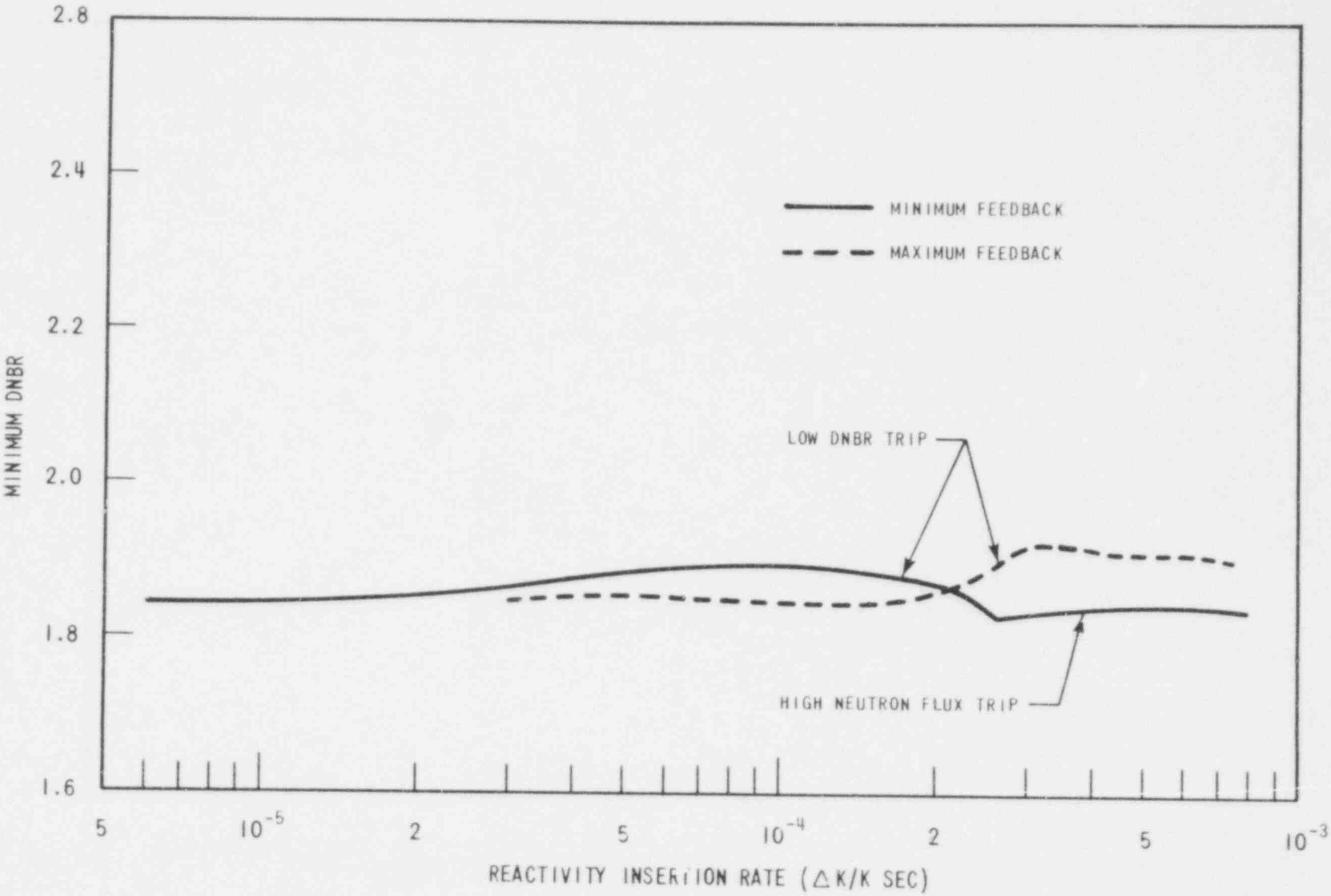


614 016

WCAP – 9500

Figure 15.4-9a.

Core Average Temperature Transient and DNBR vs. Time for Uncontrolled Rod Withdrawal from 70% Power With Maximum Feedback and 7 PCM/SEC Withdrawal Rate (N-1 Loop Operation)



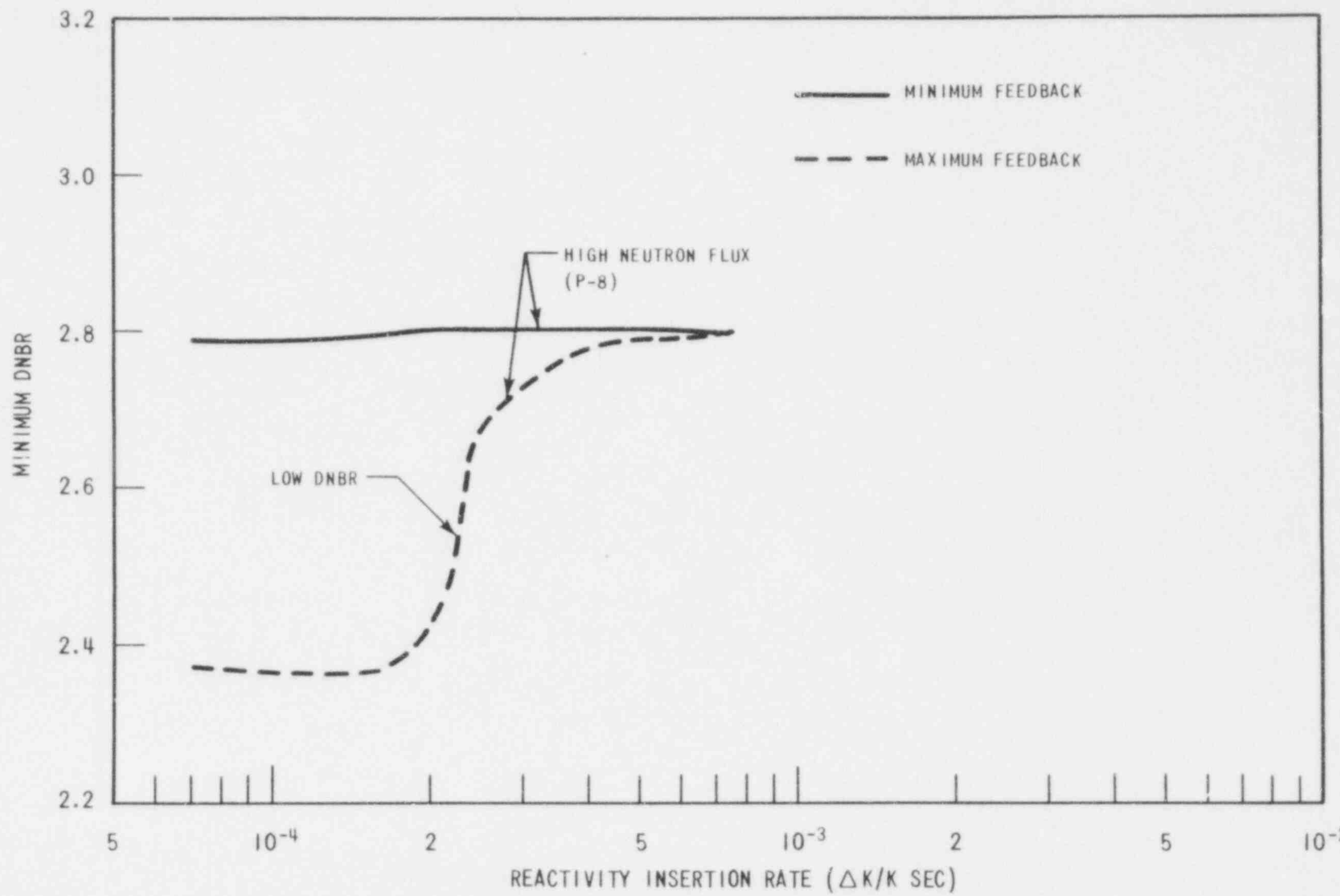
614 017

WCAP - 9500

Figure 15.4-10.

Rod Withdrawal at Power (100% Power)

14,395-147

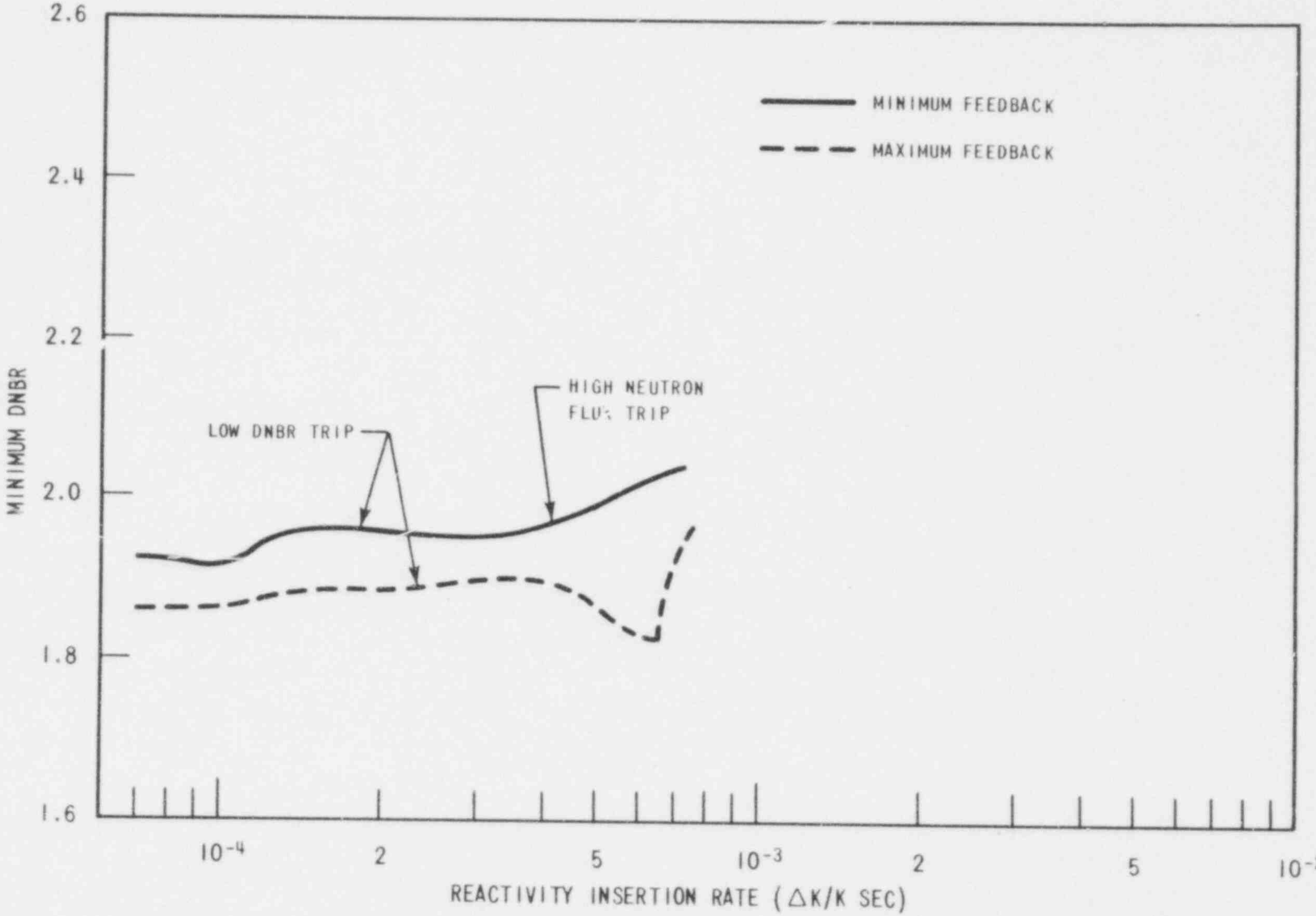


614 018

WCAP-9500

Figure 15.4-10a.
Rod Withdrawal at Power (70% Power)
(N-1 Loop Operation)

14,395-156



14,395-148

614 019

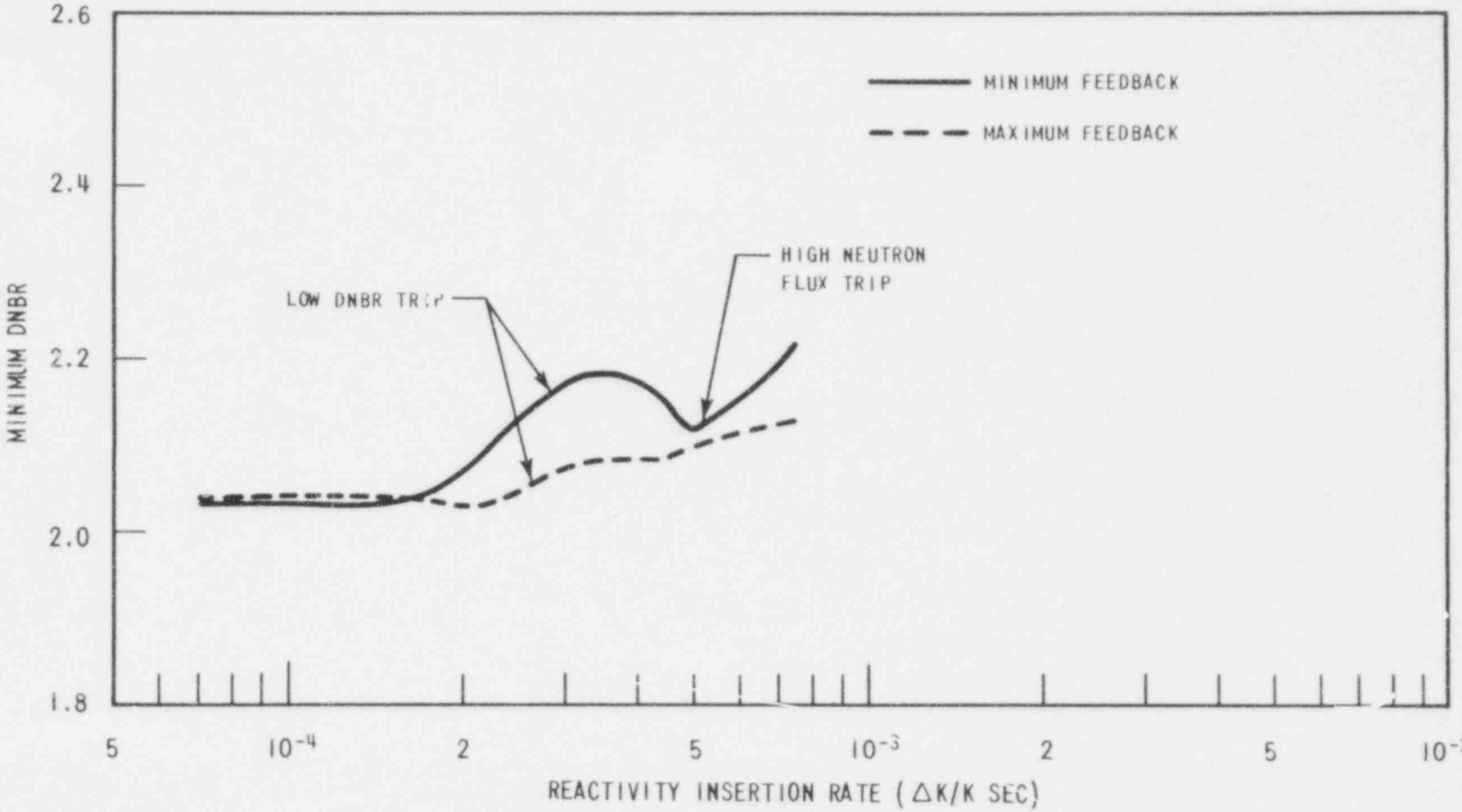
WCAP-9500

Figure 15.4-11.
Rod Withdrawal at Power (60% Power)

Figure 15.4-11a

Is Not Applicable

614 021



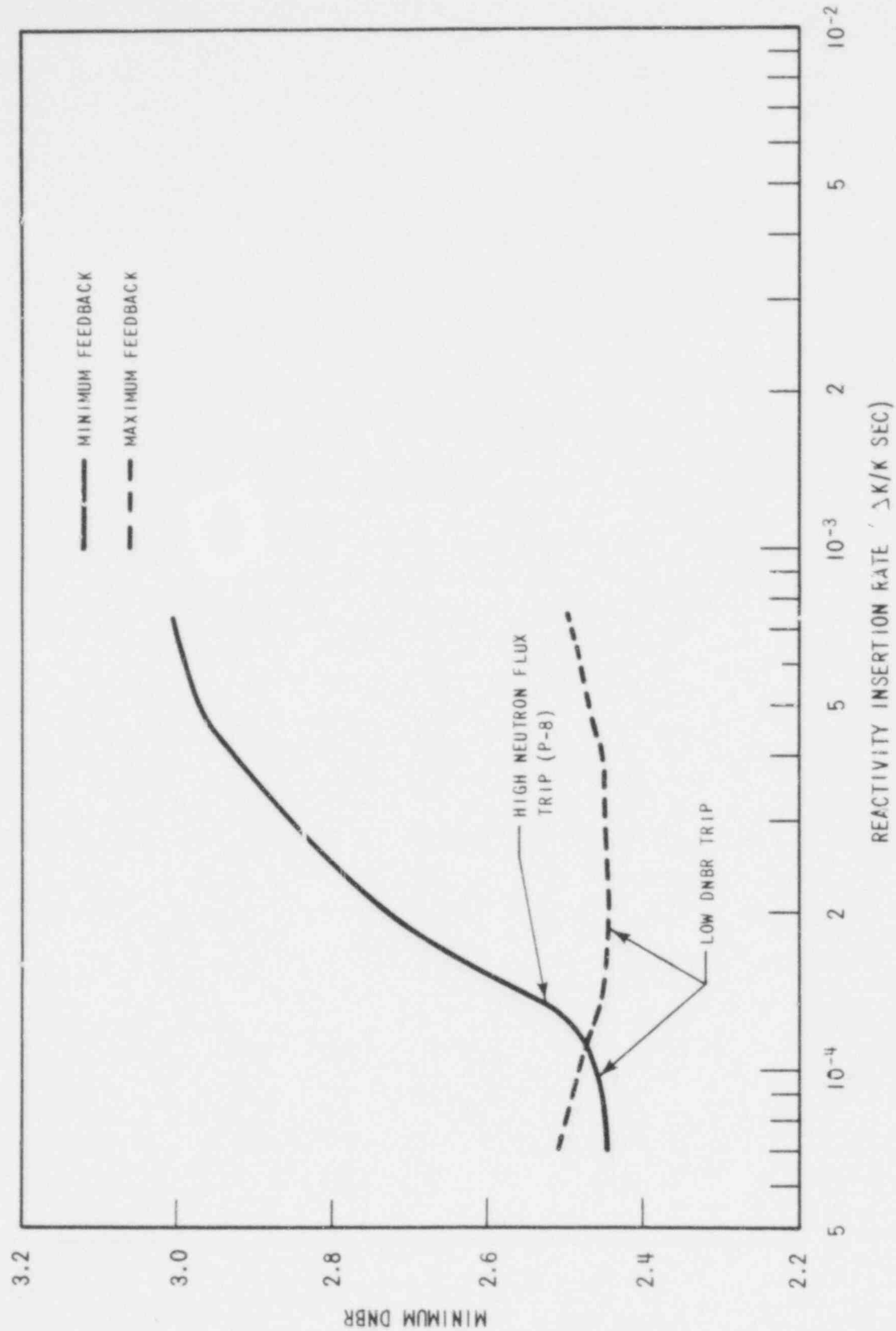
WCAP-9500

Figure 15.4-12.

Rod Withdrawal at Power (10% Power)

14,395-149

14,395-158

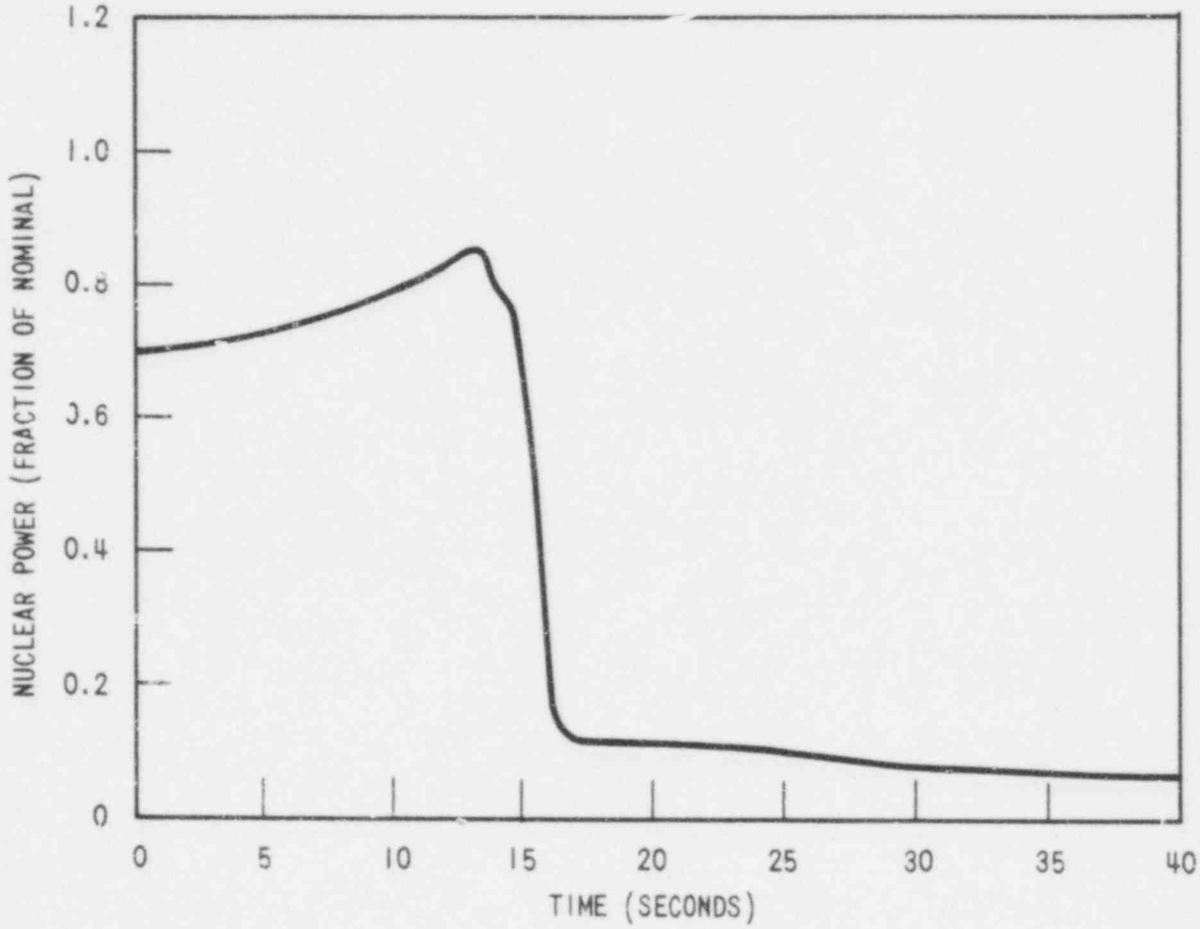


WCAP – 9500

Figure 15.4-12a.

Rod Withdrawal at Power (10% Power)
(N-1 Loop Operation)

614 022



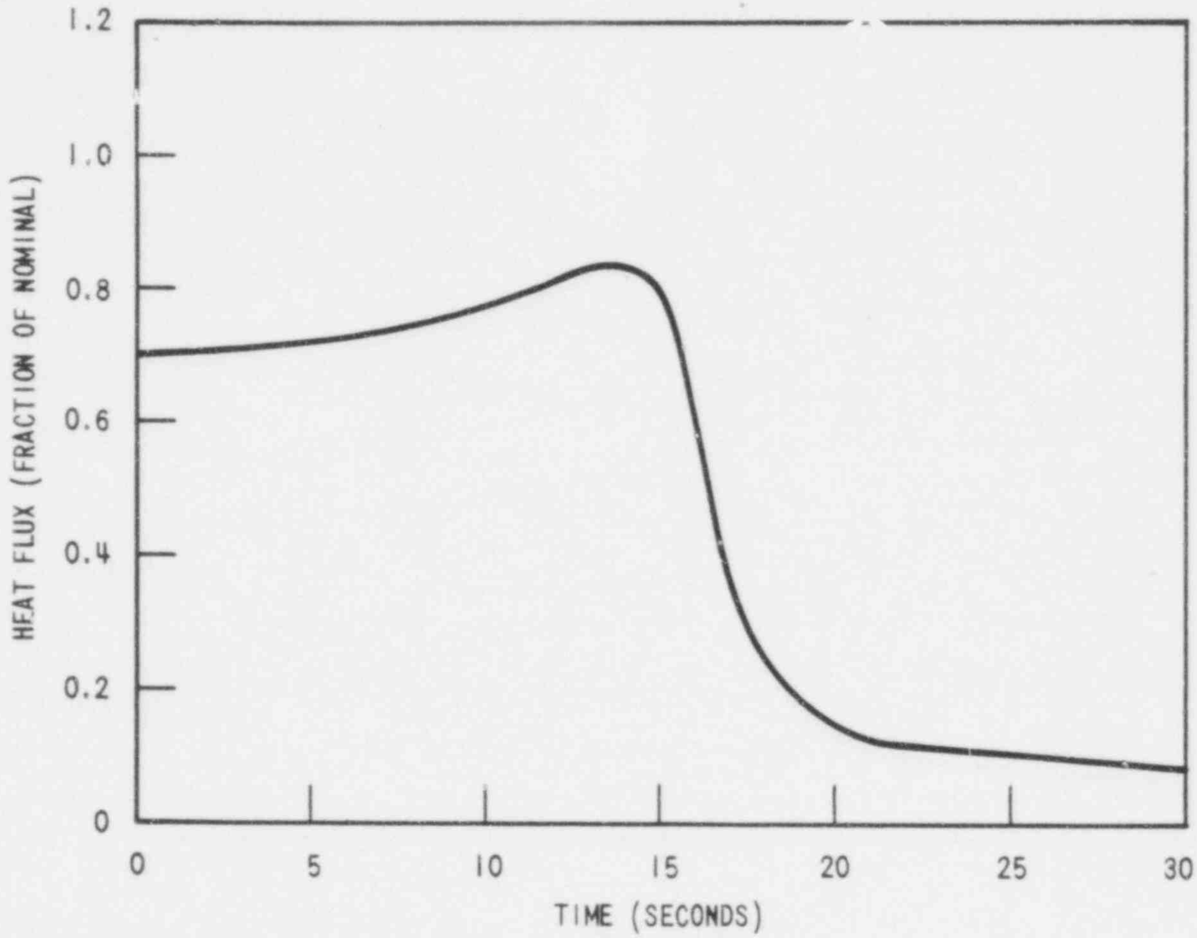
WCAP - 9500
Figure 15.4-13. Improper Startup of an Inactive Reactor Coolant Pump

614 023

Figures 15.4-13a Through 15.4-25a

Are Not Applicable

614 024

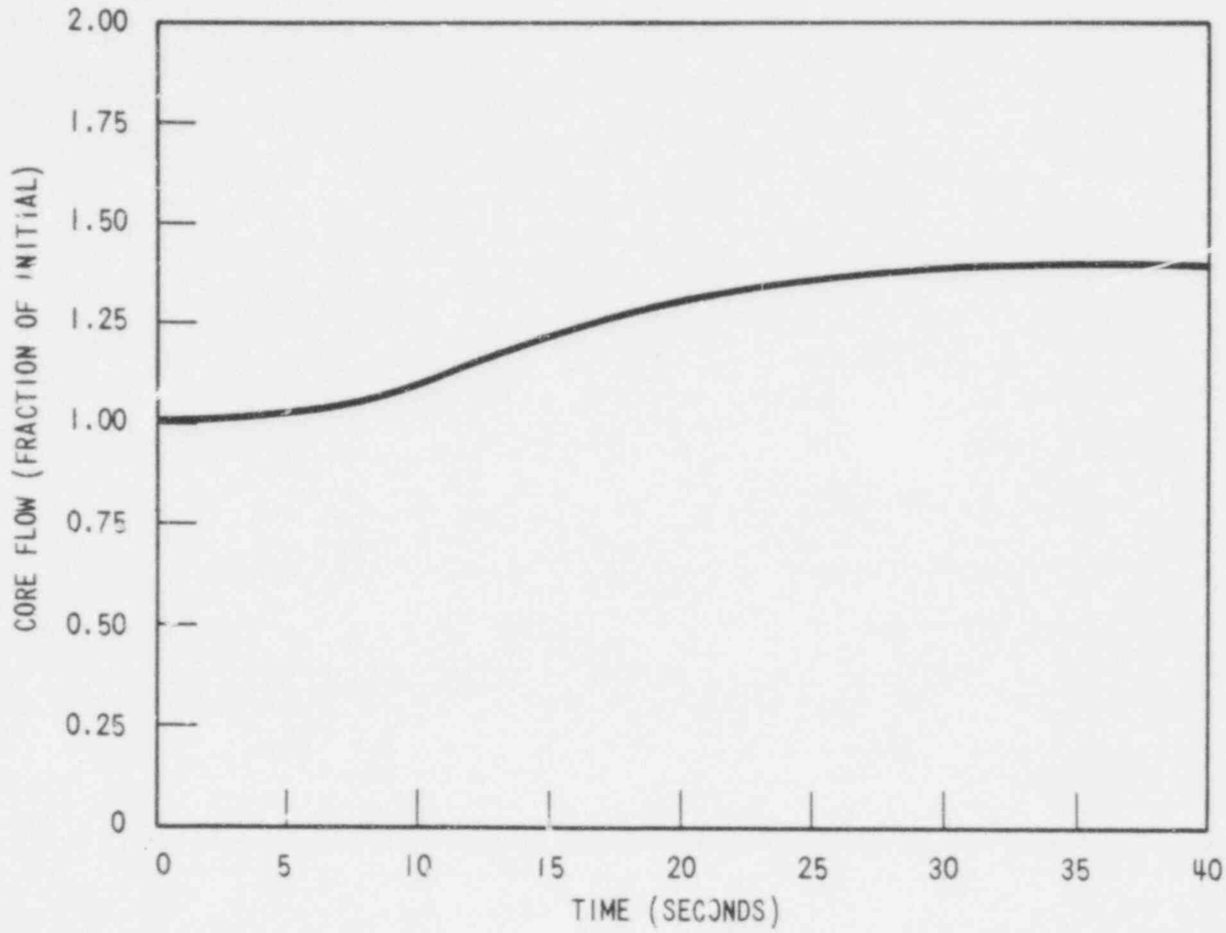


WCAP - 9500

Figure 15.4-14.

Improper Startup of an Inactive Reactor Coolant Pump

614 025

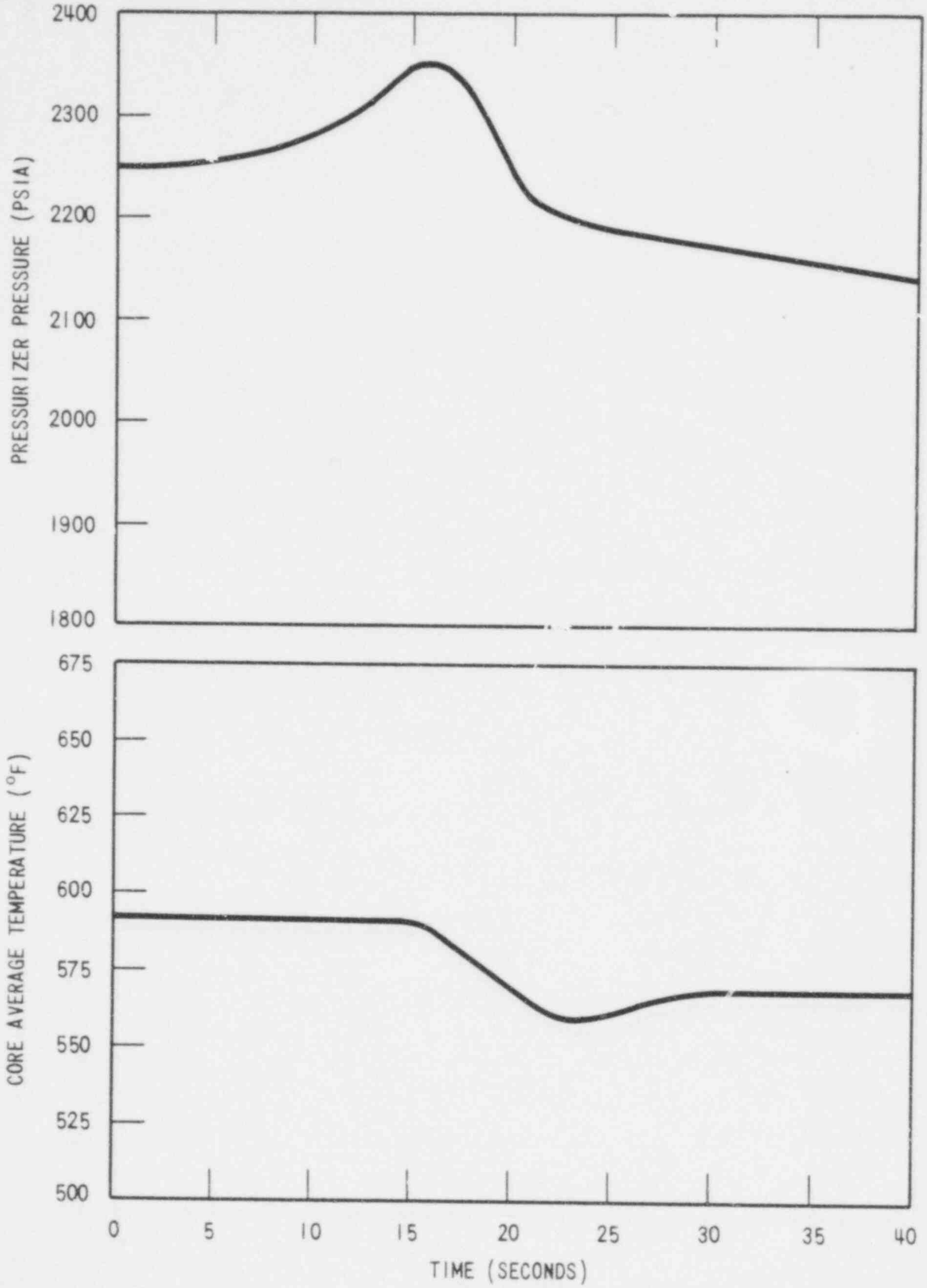


WCAP - 9500

Figure 15.4-15.

Improper Startup of an Inactive Reactor Coolant Pump

514 026

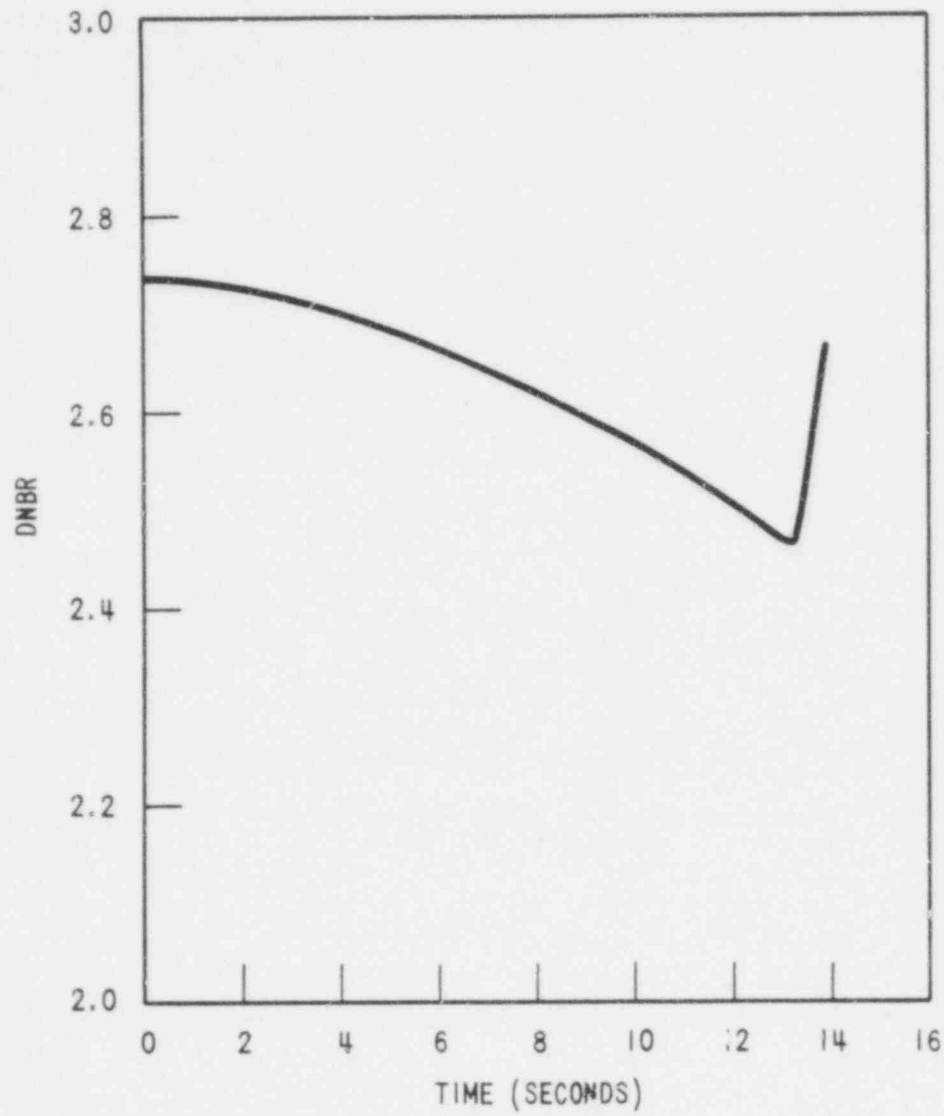


WCAP – 9500

Figure 15.4-16.

Improper Startup of an Inactive Reactor Coolant Pump

614 027

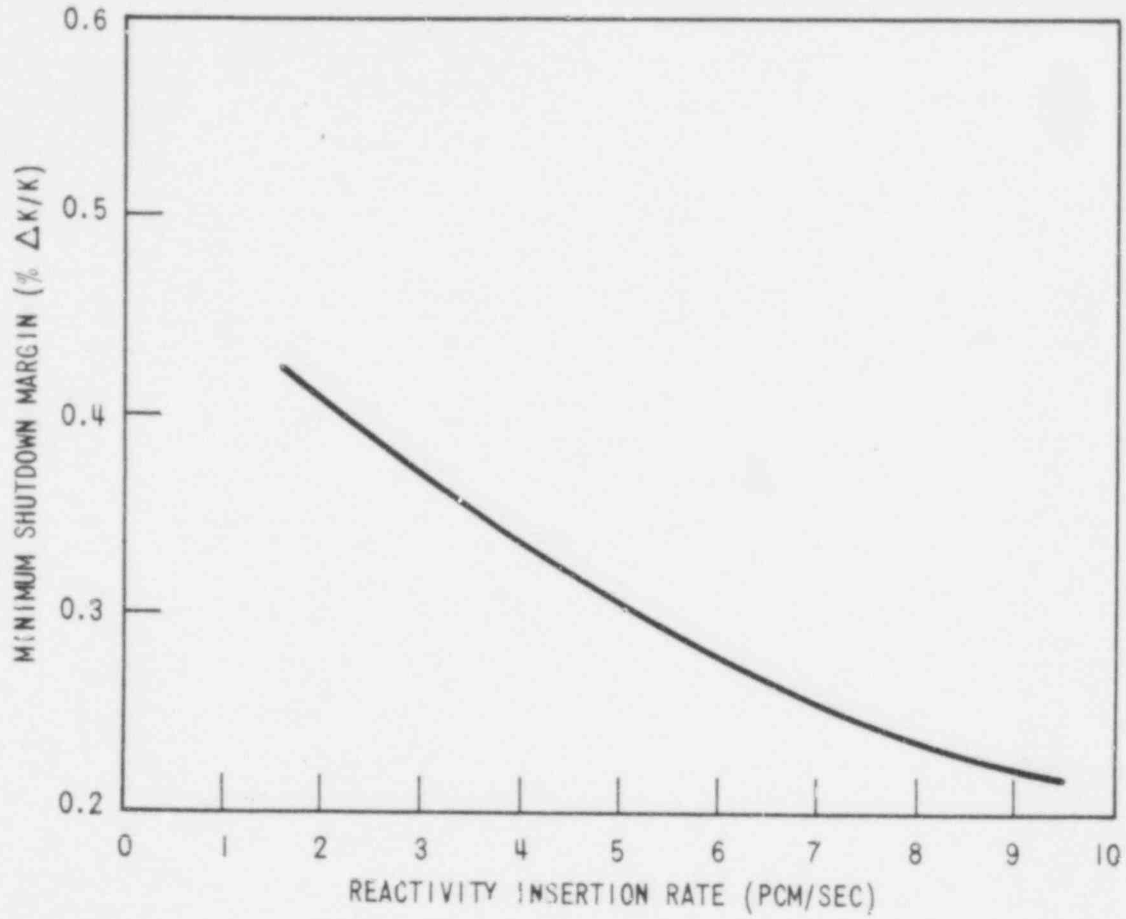


614 021

WCAP-9500

Figure 15.4-17.

Improper Startup of an Inactive Reactor Coolant Pump



514 029

WCAP – 9500

Figure 15.4-18.

Uncontrolled Boron Dilution from Shutdown and Zero Power Conditions

Figures 15.4-19 Through 15.4-20

These Figures are Not Applicable

614 030

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7						23.6	
14			2.5				4.7			11.1		17.6			
15					2.1			6.5							

CASE A

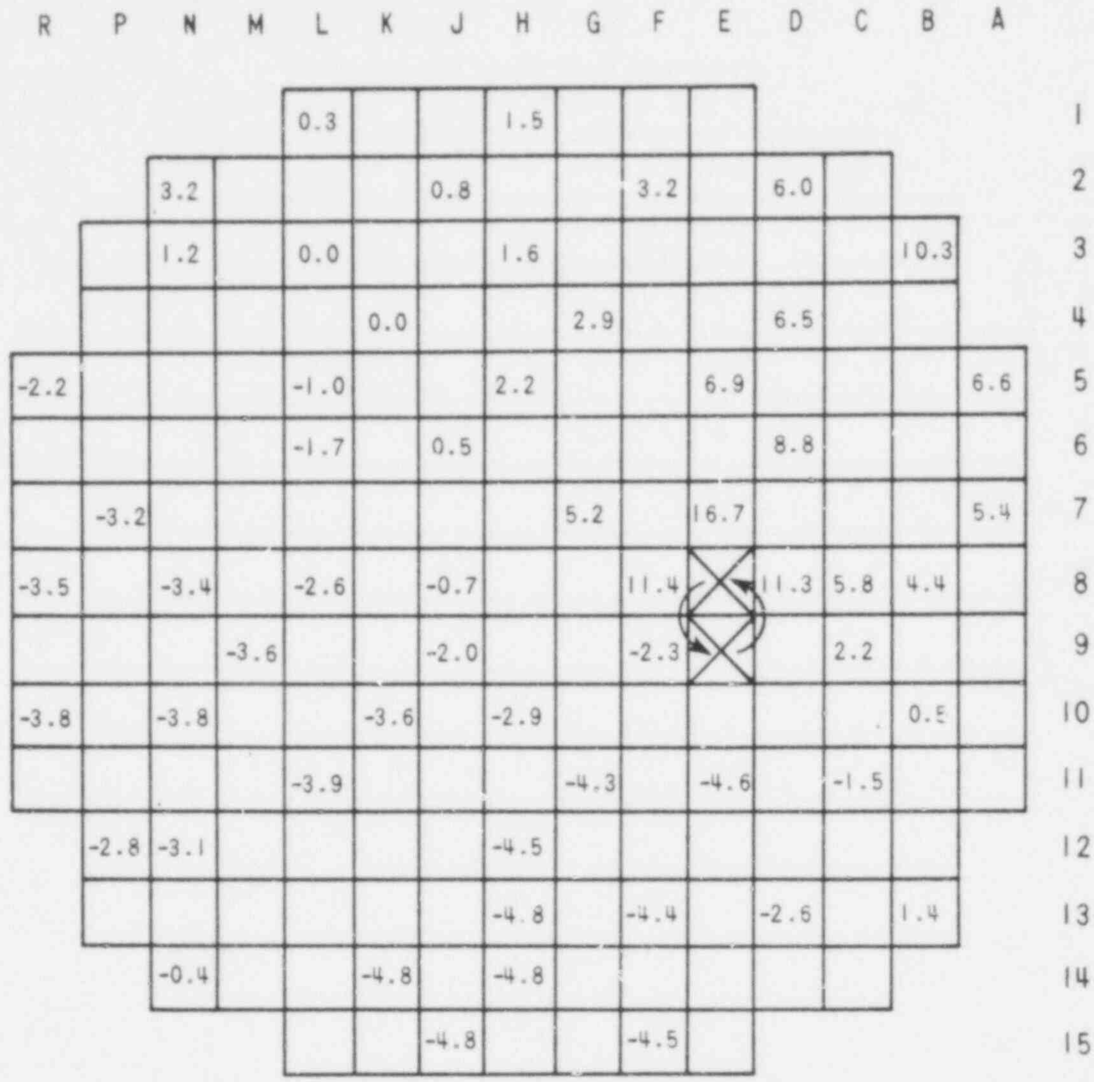
THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

WCAP – 9500

Figure 15.4-21.

Interchange Between Region 1 and Region 3 Assembly

614 031



CASE B-1

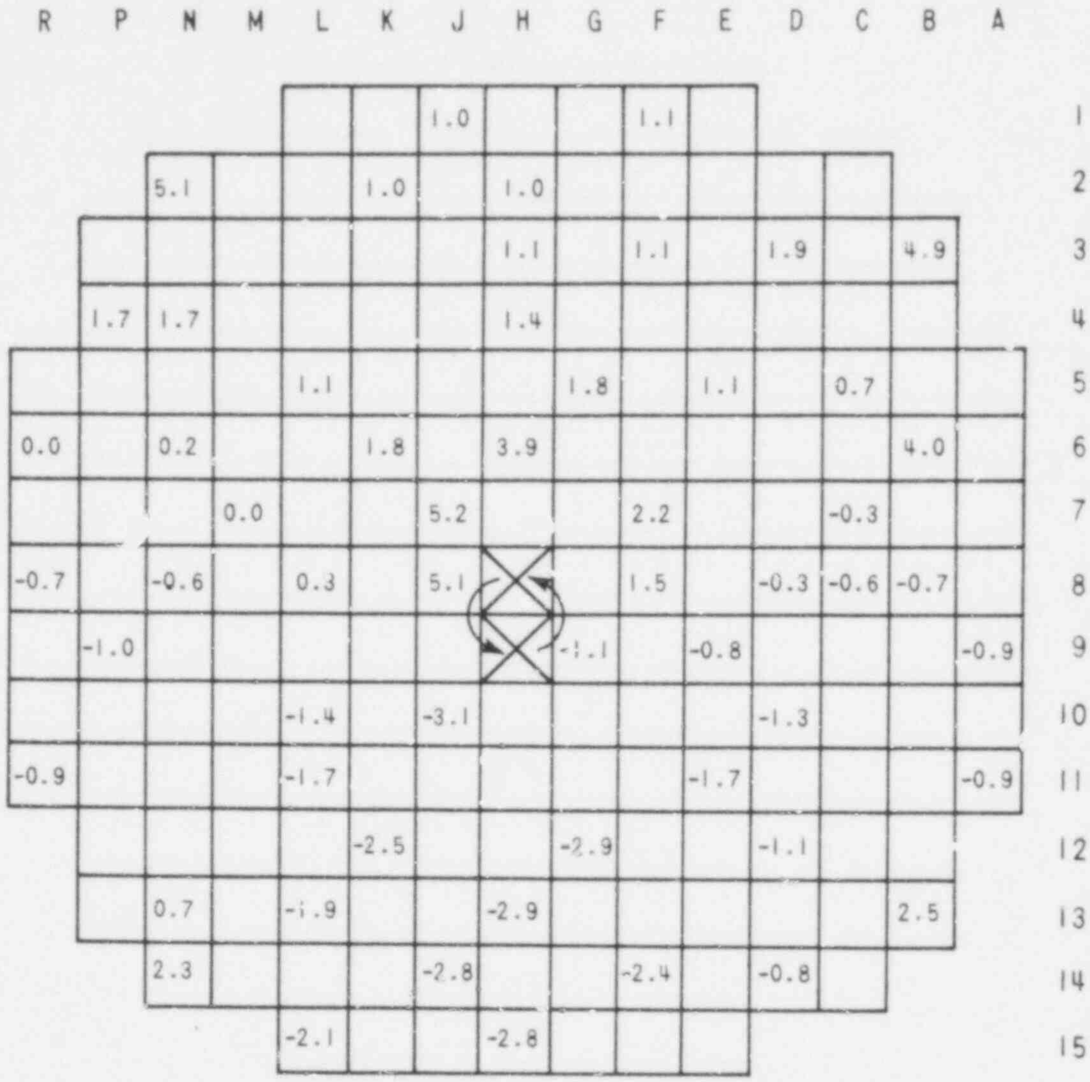
THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

614 032

WCAP - 9500

Figure 15.4-22.

Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Retained by the Region 2 Assembly



CASE B-2

THE NUMBERS REPRESENT THE PERCENT
DEVIATION FROM ASSEMBLY AVERAGE POWER

614 033

WCAP - 9500
Figure 15.4-23. Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Transferred to the Region 1 Assembly

	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

614 034

WCAP – 9500

Figure 15.4-24.

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		10.8		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0		X		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

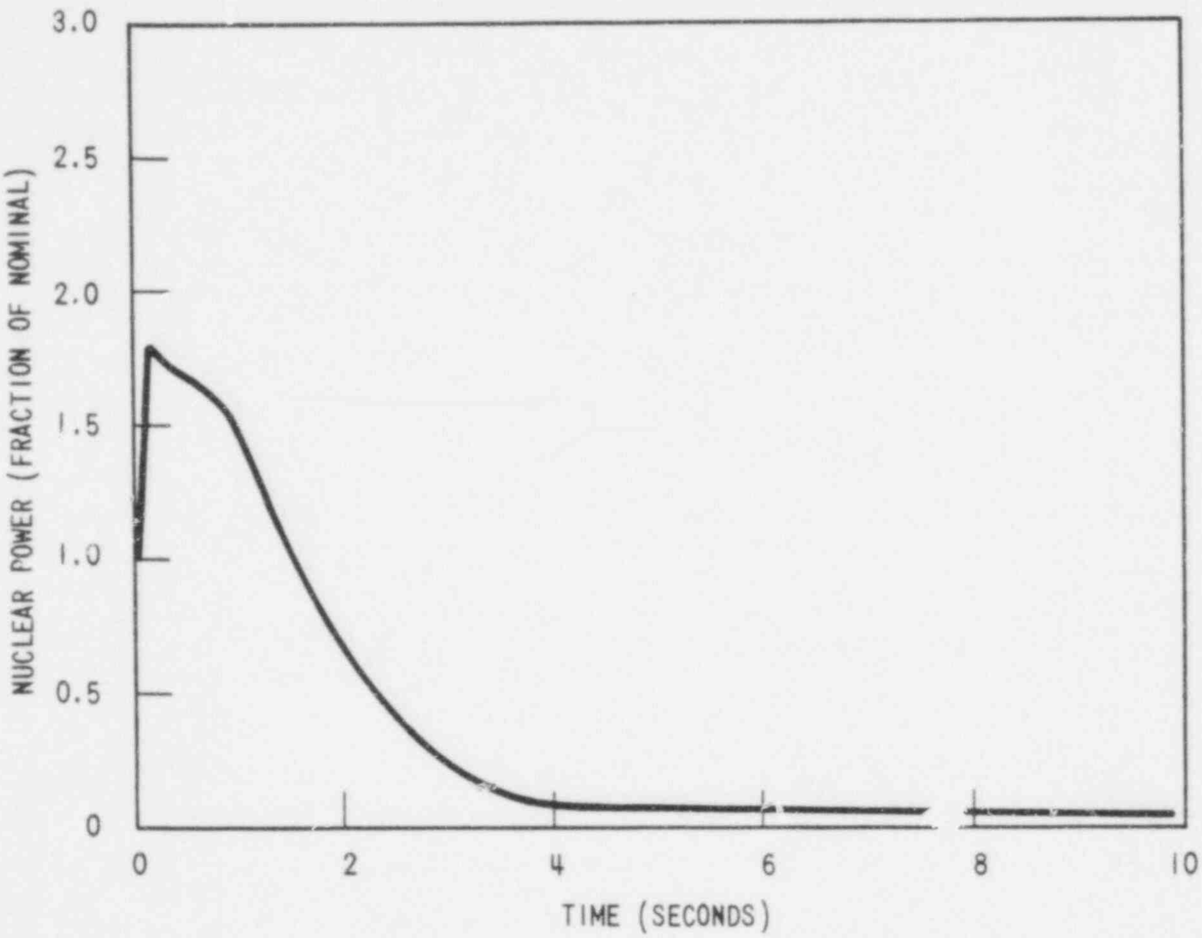
THE NUMBERS REPRESENT THE PERCENT DEVIATION FROM ASSEMBLY AVERAGE POWER

614 035

WCAP – 9500

Figure 15.4-25.

Loading a Region 2 Assembly into a Region 1 Position Near Core Periphery

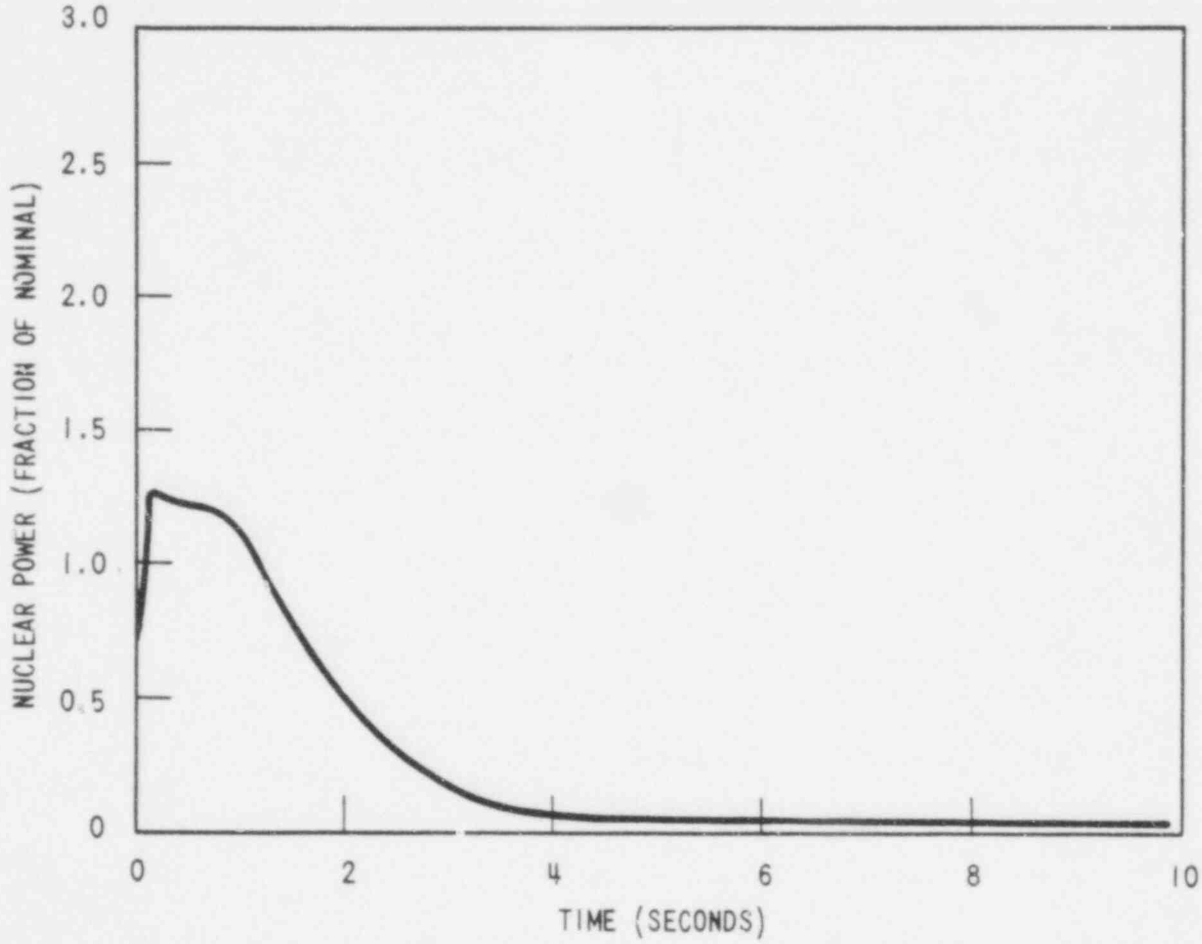


614 036

WCAP - 9500

Figure 15.4-26.

Nuclear Power Transient, BOL HFP Rod Ejection Accident

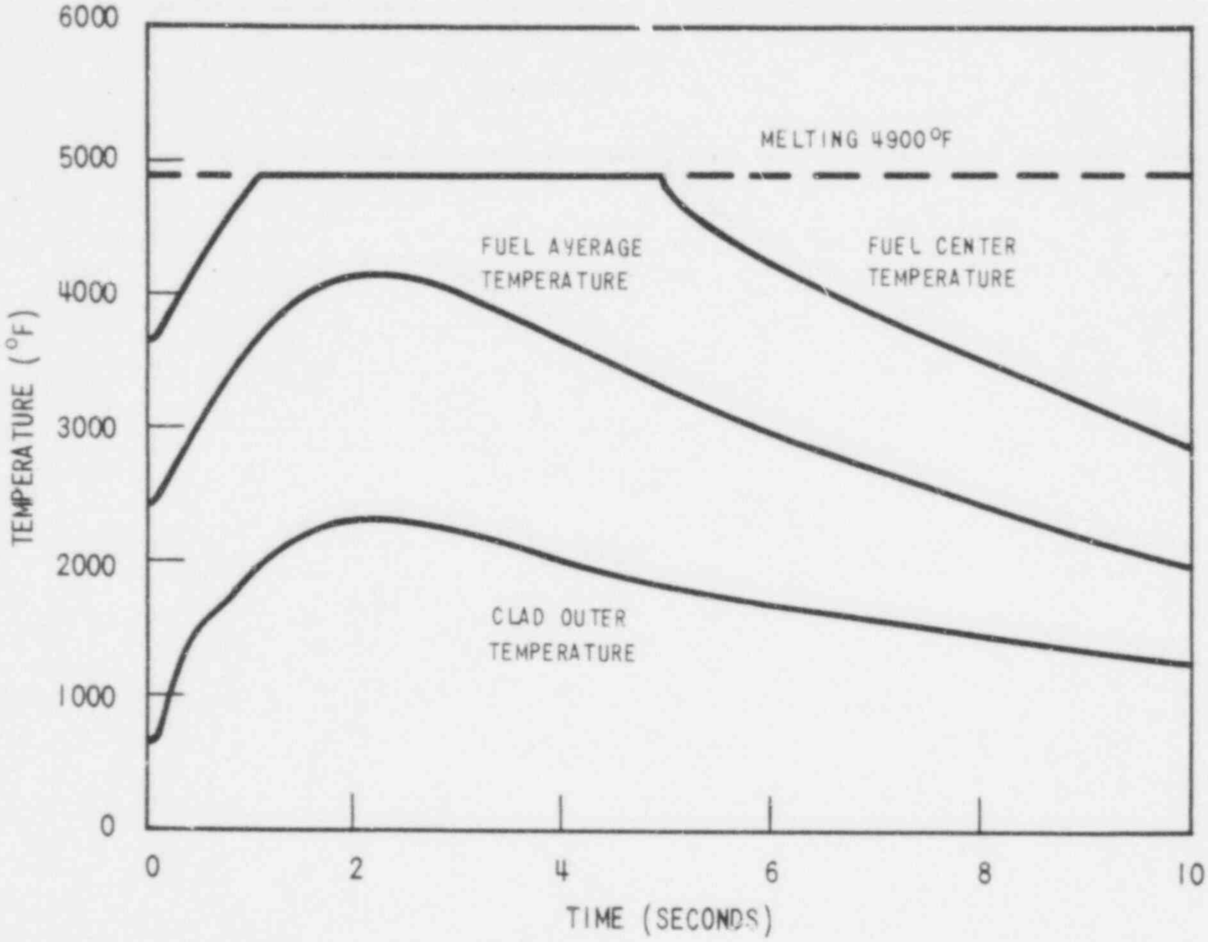


WCAP – 9500

Figure 15.4-26a.

Nuclear Power Transient BOL 70% Power Rod Ejection Accident (N-1 Loop Operation)

614 037

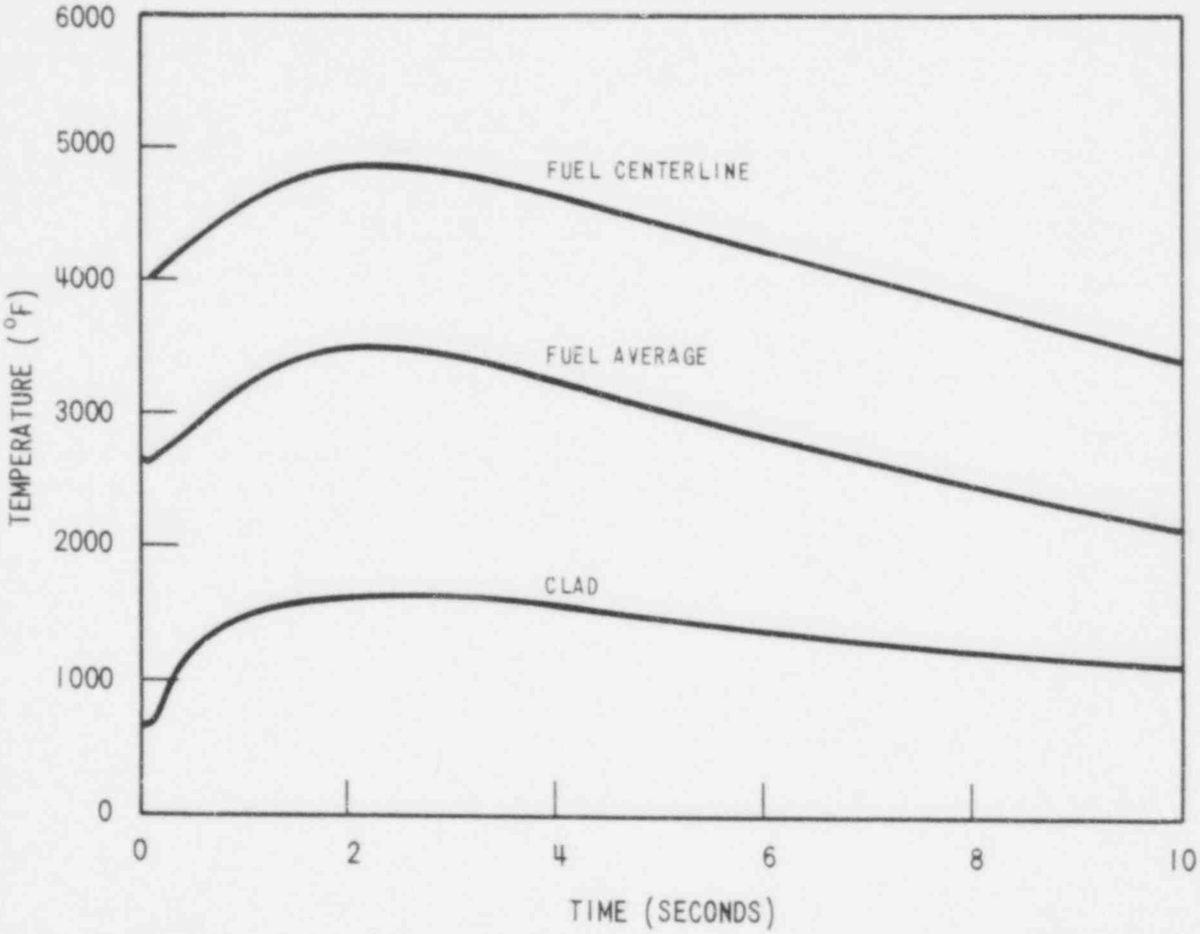


614 038

WCAP - 9500

Figure 15.4-27.

Hot Spot Fuel and Clad Temperature vs. Time, BOL HFP Rod Ejection Accident

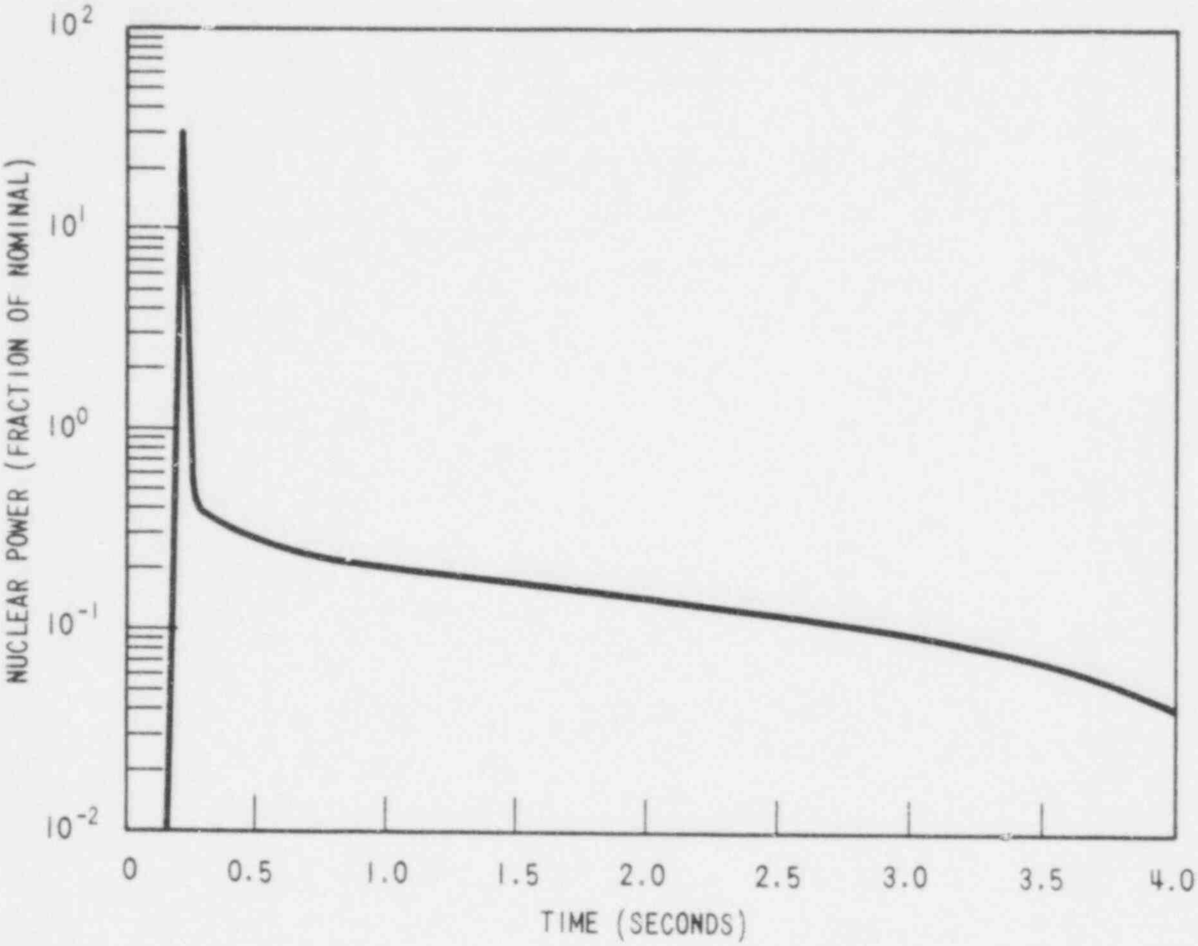


WCAP - 9500

Figure 15.4-27a.

Hot Spot Fuel and Clad Temperature vs. Time, BOL, 70% Power Rod Ejection Accident (N-1 Loop Operation)

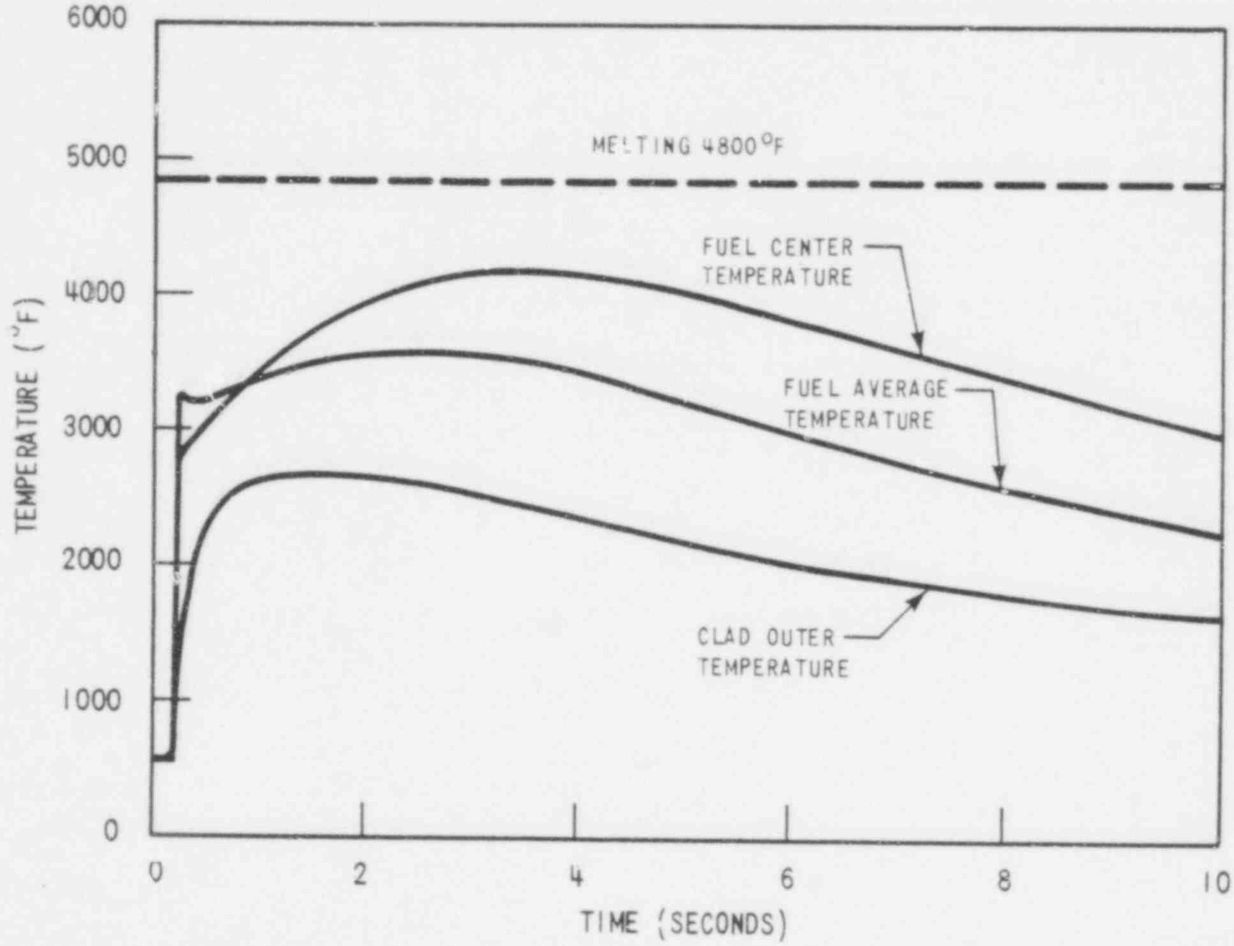
614 039



WCAP - 9500

Figure 15.4-28.

Nuclear Power Transient,
EOL HZP Rod Ejection Accident



614 041

WCAP – 9500

Figure 15.4-29.

Hot Spot Fuel and Clad Temperature vs. Time, EOL HZP Rod Ejection Accident

15.5

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Several events have been postulated which could cause an increase in reactor coolant inventory. Discussion of the following events is presented in this section:

1. Inadvertent operation of the emergency core cooling system during power operation.

2. Chemical and volume control system malfunction that increases reactor coolant inventory.

3. BWR Transients (Not applicable).

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. (See Subsection 15.0.2.)

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 Identification of Causes and Accident Description

Spurious emergency core cooling system (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

Following the actuation signal, the safety injection pumps will start taking suction from the refueling water storage tank (RWST). The safety injection pumps have a cutoff head of about 1500 psi and consequently provide no flow at normal RCS pressure.

An SIS signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates SIS will also produce a reactor trip. If a reactor trip is generated by

614 043

the spurious SIS signal, the plant is automatically brought to the hot shutdown condition. Since the safety injection pumps have a shutoff head less than operating pressure in the reactor coolant system, their actuation has no effect on the transient.

If the reactor protection system does not produce an immediate trip as a result of the spurious SIS signal, the reactor will continue to operate at power. Since the safety injection pumps have a cutoff head of about 1500 psi, they will not provide borated water following their actuation at normal RCS pressure.

15.5.1.2 Analysis of Effects and Consequences

No analysis is performed for this incident since the inadvertent actuation of the safety injection pumps does not result in a system transient or affect the transient which would result if the reactor would be tripped.

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

Spurious safety injection without immediate reactor trip has no effect on the Reactor Coolant System.

If a reactor trip is generated by the spurious safety injection signal, a normal shutdown can be commenced without boration from the safety injection pumps because of the cutoff head of about 1500 psi. The radiological consequences would be less severe than the steamline break event analyzed in Subsection 15.1.5.3.

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 Subsection 15.4.6. An increase in reactor coolant inventory which results from the injection of highly borated water into the reactor coolant system is analyzed in Subsection 15.5.1.

15.5.3 BWR TRANSIENTS

This is not applicable.

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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 (Subsection 15.6.1)

2. Failure of small lines carrying primary coolant outside containment. (Subsection 15.6.2)

3. Steam generator tube rupture (Subsection 15.6.3)

4. BWR piping failure outside containment (not applicable) (Subsection 15.6.4)

5. Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (Subsection 15.6.5)

6. BWR transients (not applicable) (Subsection 15.6.6)

Items 1 asnd 2 above are considered to be ANS Condition II events and items 3 and 5 are considered to be ANS Condition IV events. See Subsection 15.0.1.

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 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 that 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 the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following Reactor Protection System signals:

1. Low DNBR,
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 Subsection 15.0.2)

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, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the Improved Thermal Design Procedure as described in WCAP-8567. Initial operating conditions are assumed at values consistent with steady-state N and N-1 loop operation. Plant characteristics and initial conditions are discussed in Subsection 15.0.4.

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 temperature 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 most negative Doppler only power coefficient is assumed (Figure 15.0-2) so that the resultant amount of positive feedback is conservatively high in order to retard any power decrease due to moderator reactivity feedback.

4. Cases are analyzed considering four loops in operation and three loops in operation.

Plant systems and equipment which are available to mitigate the effects of the Reactor Coolant System depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.9 and listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode in order to hold the core at full power longer and thus delay the trip. This is a worst case assumption; if the reactor were in manual control, 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.

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Results

The system response to an inadvertent opening of a pressurizer safety valve is shown on Figures 15.6-1 through 15.6-3. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on pressurizer pressure. The pressure decay transient following the accident is given in Figure 15.6-2. Pressure drops more rapidly after core heat generation is reduced via the trip, and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-3. The DNBR remains above the limit valve throughout the transient. Figures 15.6-1a through 15.6-3a show the transient for the case with three reactor coolant loops in service (N-1 loop operation).

Following reactor trip, RCS pressure will continue to fall until flow through the inadvertently opened valve is terminated. Automatic actuation of the Safety Injection System may occur if the pressure falls to the low pressurizer pressure SI setpoint. RCS pressure will stabilize following operator action to terminate flow to the inadvertently opened valve; normal operating procedures may then be followed. 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 stabilize the plant will be in a time frame in excess of ten minutes following reactor trip.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6-1.

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 low DNBR reactor protection system signals provide adequate protection against the RCS depressurization event. The DNBR remains above the limit valve throughout the transient; thus, the DNB design-basis as described in Section 4.4 is met. 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.

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 diameter 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

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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 SPFCTRUM 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 (see Section 15.0.1).

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary (see Section 5.2) 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 [2] as follows:

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.

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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. Reference [3] presents a recent study in regard to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

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:

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

2. 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 15.6-4.

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 RCS depressurizers to 600 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 2250 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.

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.

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Description of Small Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long term recirculation.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10CFR50 [2].

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 Reference [4]. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes which are used in the LOCA analysis are described in detail in References [5] through [8]; code modifications are specified in References [9] through [11]. 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

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code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The COCO 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 phase, these data are transferred to the WREFLOOD code. Also at the end-of-blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the Containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and 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 WREFLOOD and COCO codes are interactively linked. 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 clad temperature and metal-water reaction of the hottest rod in the core.

The large break analysis was performed with the February 1978 version of the evaluation model which includes modifications delineated in References [16, 17, 18 and 19].

The analysis in this section was performed with the upper head fluid temperature equal to the reactor coolant system cold leg fluid temperature, achieved by increasing the upper head cooling flow (24).

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 [12] 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 [13].

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 multi-node 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.

Clad thermal analyses are performed with the LOCTA-IV code [8] 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.

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Figure 15.6-48 presents the hot rod power shape utilized to perform the small break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height and also local power is maximized in the upper regions of the reactor core (10' to 12'). This power shape is skewed to the top of the core with the peak local power occurring at the 10.5' core elevation.

This is limiting for the small break analysis because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two phase mixture height, remains low. The peak clad temperature occurs above 10 feet.

Schematic representations of the computer code interfaces are given in Figures 15.6-5 and 15.6-6.

The small break analysis was performed with the October 1975 version of the Westinghouse ECCS Evaluation Model (refer to References 8, 13, 14, and 15).

15.6.5.3.2 <u>Input Parameters and Initial Conditions</u>

Table 15.6-5 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature. The effect of using the cold leg temperature in the reactor vessel upper head is described in Reference [24]. In addition, the analysis in this section utilized the upflow barrel-baffle methodology described in Reference [20].

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The bases used to select the numerical values that are input parameters to the anal .s have been conservatively determined from sensitivity studies (refer to References [21], [22], and [23]). 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.

15.6.5.3.3 Results

Large Break Results

Based on the results of the LOCA sensitivity studies, (References [21], [22], and [23]) 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 15.6-1 and 15.6-3.

The mass and energy release data for the break resulting in the highest calculated peak clad temperature are presented in Section 6.2.1.5.

Figures 15.6-7 through 15.6-33 present the parameters of principal interest from the large break ECCS analyses. For all cases analyzed transients of the following parameters are presented:

1. Hot spot clad temperature.

2. Coolant pressure in the reactor core.

3. Water level in the core and downcomer during reflood.

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4. Core reflooding rate.

5. Thermal power during blowdown.

The Containment pressure transient resulting from a LOCA is presented in Section 6.2.1.5.

For the limiting break analyzed, the following additional transient parameters are presented:

1. Core flow during blowdown (inlet and outlet).

2. Core heat transfer coefficients.

3. Hot spot fluid temperature.

4. Mass released to Containment during blowdown.

5. Energy released to Continment during blowdown.

6. Fluid quality in the hot assembly during blowdown.

7. Mass velocity during blowdown.

8. Accumulator water flow rate during blowdown.

9. Pumped safety injection water flow rate during reflood.

The maximum clad temperature calculated for a large break is 1991°F which is less than the Acceptance Criteria limit of 2200°F of 10CFR50.46. The maximum local metal-water reaction is 3.81 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

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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 clad temperature resulting from a small break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies Reference [21] the limiting small break was found to be less than a 10 inch diameter rupture of the RCS cold leg. Therefore, a range of small break analyses are presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6-1 and 15.6-4.

Figures 15.6-34 through 15.6-47 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed the following transient parameters are presented:

1. RCS pressure.

2. Core mixture height.

3. Hot spot clad temperature.

4. Core power after reactor trip.

5. Pumped safety injection.

For the limiting break analyzed, the following additional transient parameters are presented:

1. Core steam flow rate.

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2. Core heat transfer coefficient.

3. Hot spot fluid temperature.

The maximum calculated peak clad temperature for all small breaks analyzed is 1792°F. These results are well below all Acceptance Criteria limits of 10CFR50.46 and in all cases are not limiting when compared to the results presented for large breaks.

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. The radiological consequences of a LOCA will be evaluated on a plant specific basis.

Fission Product 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 15.0-7.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable.

15.6.7 REFERENCES

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18. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, November 1, 1978.

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21. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974.

22. Salvatori, R., "Westinghouse ECCS - Plan Sensitivity Studies, WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974.

23. Johnson, W. J., Massie, H. W. and Thompson, C. M., "Westinghouse ECCS-Four Loop Plant (17x17) Sensitivity Studies", WCAP-8565-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

24. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2030, January 1979.

614 067

TABLE 15.6-1 (Sheet 1 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0	0.0
	Low Pressurizer Pressure reactor trip setpoint reached	41.3	35.4
	Rods begin to drop	43.3	37.4
	Minimum DNBR occurs	43.6	38.2
Large break LOCA			
1. DECLG C_D = 0.8	Start	0.0	
	Reactor trip signal	0.83	
	Safety injection signal	1.1	
	Accumulator injection begins	14.1	
	End-of-bypass	23.8	
	End-of-bkowdown	26.7	

TABLE 15.6-1 (Sheet 2 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Pump injection begins	26.1	
	Bottom of core recovery	34.4	
	Accumulator empty	51.3	
2. DECLG $C_D = 0.6$	Start	0.0	
	Reactor trip signal	0.84	
	Safety injection signal	1.3	
	Accumulator injection begins	16.4	
	End-of-bypass	24.51	
	End-of-blowdown	24.53	
	Pump injection begins	26.3	
	Bottom of core recovery	34.7	
	Accumulator empty	52.6	

614 069

TABLE 15.6-1 (Sheet 3 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
3. DECLG $C_D = 0.4$	Start	0.0	
	Reactor trip signal	0.88	
	Safety injection signal	1.6	
	Accumulator injection begins	21.2	
	End-of-bypass	33.2	
	End-of-blowdown	37.0	
	Pump injection begins	26.6	
	Bottom of core recovery	44.3	
	Accumulator empty	60.7	
Small break LOCA			
1. 3 inch	Start	0.0	
	Reactor trip signal	27.3	
	Top of core uncovered	695	

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TABLE 15.6-1 (Sheet 4 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Accumulator injection begins	N/A	
	Peak clad temperature occurs	1426	
	Top of core covered	2295	
2. 4 inch	Start	0.0	
	Reactor trip signal	16.9	
	Top of core uncovered	330	
	Accumulator injection begins	802	
	Peak clad temperature occurs	790	
	Top of core covered	1190	
3. 6 inch	Start	0.0	
	Reactor trip signal	11.2	
	Top of core uncovered	126	

614 071

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	TIME (sec) N Loop	TIME (sec) N-1 Loop
	Accumulator injection begins	349	
	Peak clad temperature occurs	235	
	Top of core covered	365	

614 072

TABLE 15.6-2

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

Licensed core power[a], (MWt)	3411
Peak linear power, includes 102% factor (kW/ft)	12.88
Total peaking factor, F_Q^T	2.32
Axial peaking factor, F_Z	1.451
Power shape	
Large break	Chopped cosine
Small break	See Figure 15.6-48
Fuel assembly array	Optimized 17x17
Accumulator water volume, nominal (ft^3/accumulator)	1200
Accumulator tank volume, nominal (ft^3/accumulator)	1650
Accumulator gas pressure, minimum (psia)	600
Safety injection pumped flow	See Figures 15.6-21 and 15.6-47
Containment parameters	See Section 6.2
Initial loop flow (lb/sec)	9984
Vessel inlet temperature (^oF)	560.7
Vessel outlet temperature (^oF)	643.3
Reactor coolant pressure (psia)	2250
Steam pressure (psia)	988
Steam generator tube plugging level (%)	0

[a]Two percent is added to this power to account for calorimetric error.

614 073

TABLE 15.6-3

LARGE BREAK LOCA RESULTS FUEL CLADDING DATA

Results	C_D = 0.8 DEC1G	C_D = 0.6 DECLG	C_D = 0.4 DECLG
Peak clad temperature (^oF)	1964	1991	1707
Peak clad temperature location (ft)	7.5	7.5	7.5
Local Zr/H_2O reaction, maximum (%)	3.45	3.81	1.07
Local Zr/H_2O location (ft)	7.5	7.5	7.5
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (sec)	52.8	52.4	N/A
Hot rod burst location (ft)	6.0	6.25	N/A

614 074

TABLE 15.6-4

SMALL BREAK LOCA RESULTS FUEL CLADDING DATA

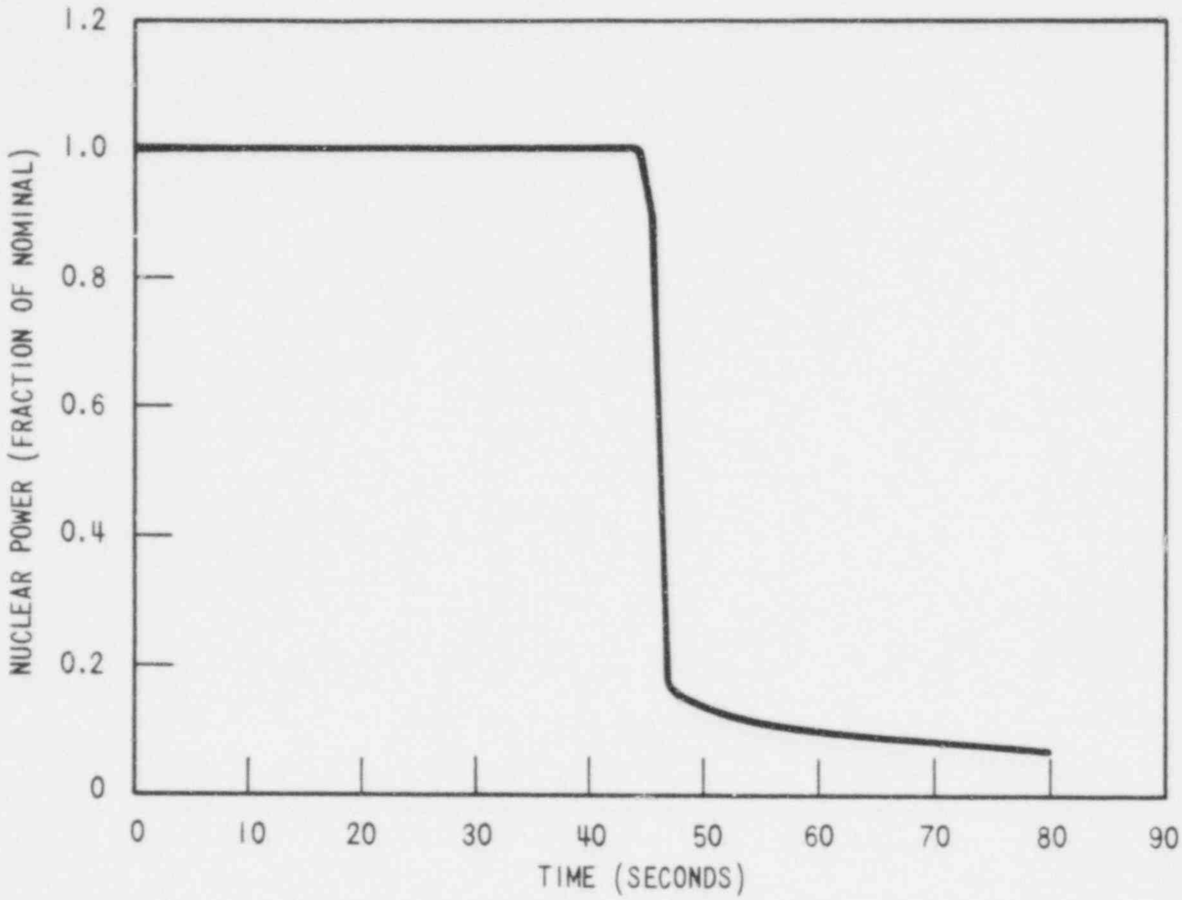
Results	3 Inch	4 Inch	6 Inch
Peak clad temperature (^{o}F)	1594	1792	1588
Peak clad temperature location (ft)	11.5	11.25	11.0
Local Zr/H_2O reaction, maximum (%)	1.25	1.56	0.37
Local Zr/H_2O location (ft)	11.25	11.25	11.0
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (sec)	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A

614 075

TABLE 15.6-5

INPUT PARAMETERS TUBE USED IN RADIOLOGICAL CONSEQUENCES OF LOCA ANALYSES

	Realistic Analysis	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt	3565 MWt
Activity released to containment and available for release		
noble gases	Inventory in one RCS volume	100% of core inventory
iodines	Inventory in one RCS volume	25% of core inventory
Plateout of iodine activity released to containment	50%	50%
Form of iodine activity in primary containment available for release		
element iodine	91%	91%
methyl iodine	4%	4%
particulate iodine	5%	5%

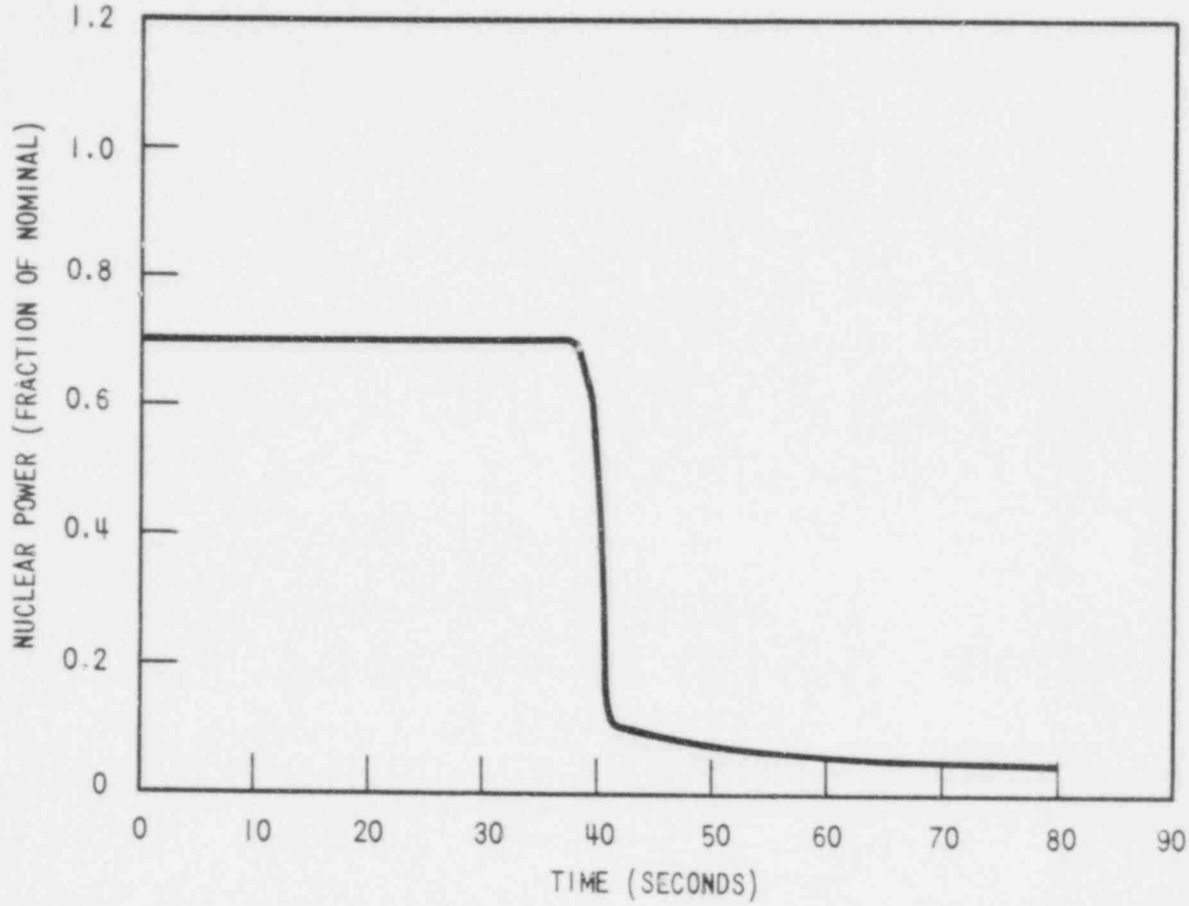


614 077

WCAP - 9500

Figure 15.6-1.

Nuclear Power Transient for Inadvertent Opening of a Pressurizer Safety Valve

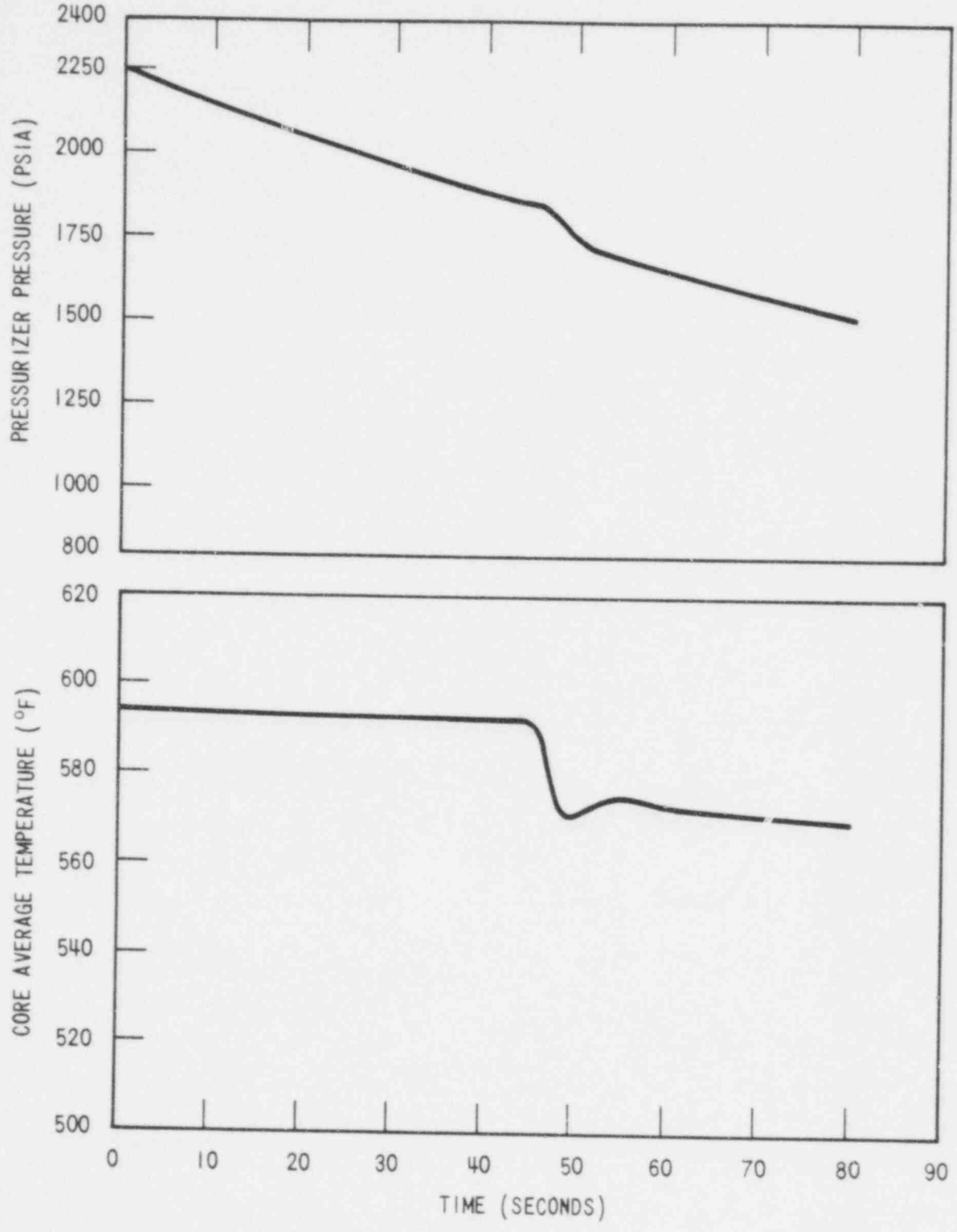


614 078

WCAP - 9500

Figure 15.6-1a.

Nuclear Power Transient for Inadvertent Opening of a Pressurizer Safety Valve (N-1 Loop Operation)



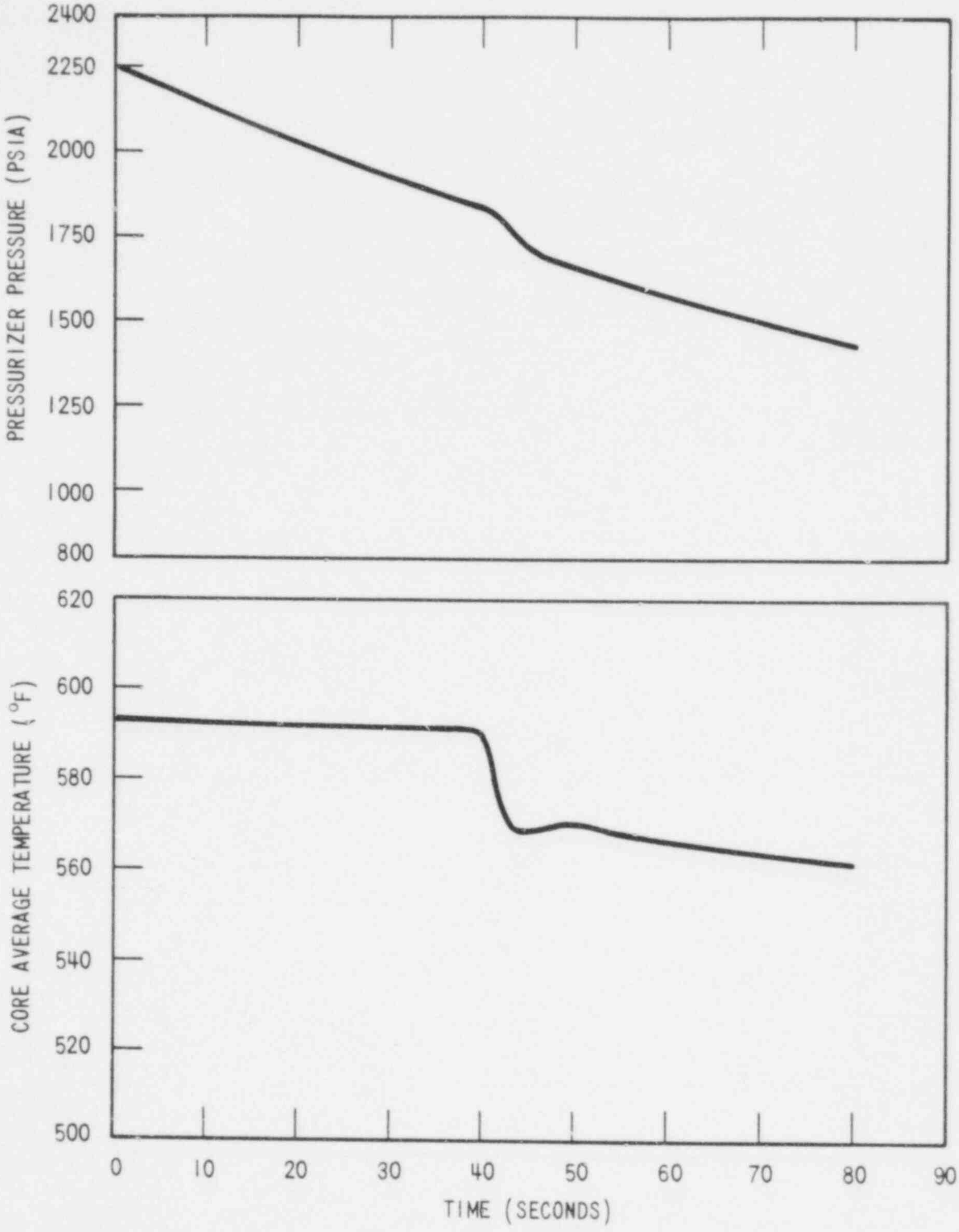
614 079

WCAP – 9500

Figure 15.6-2.

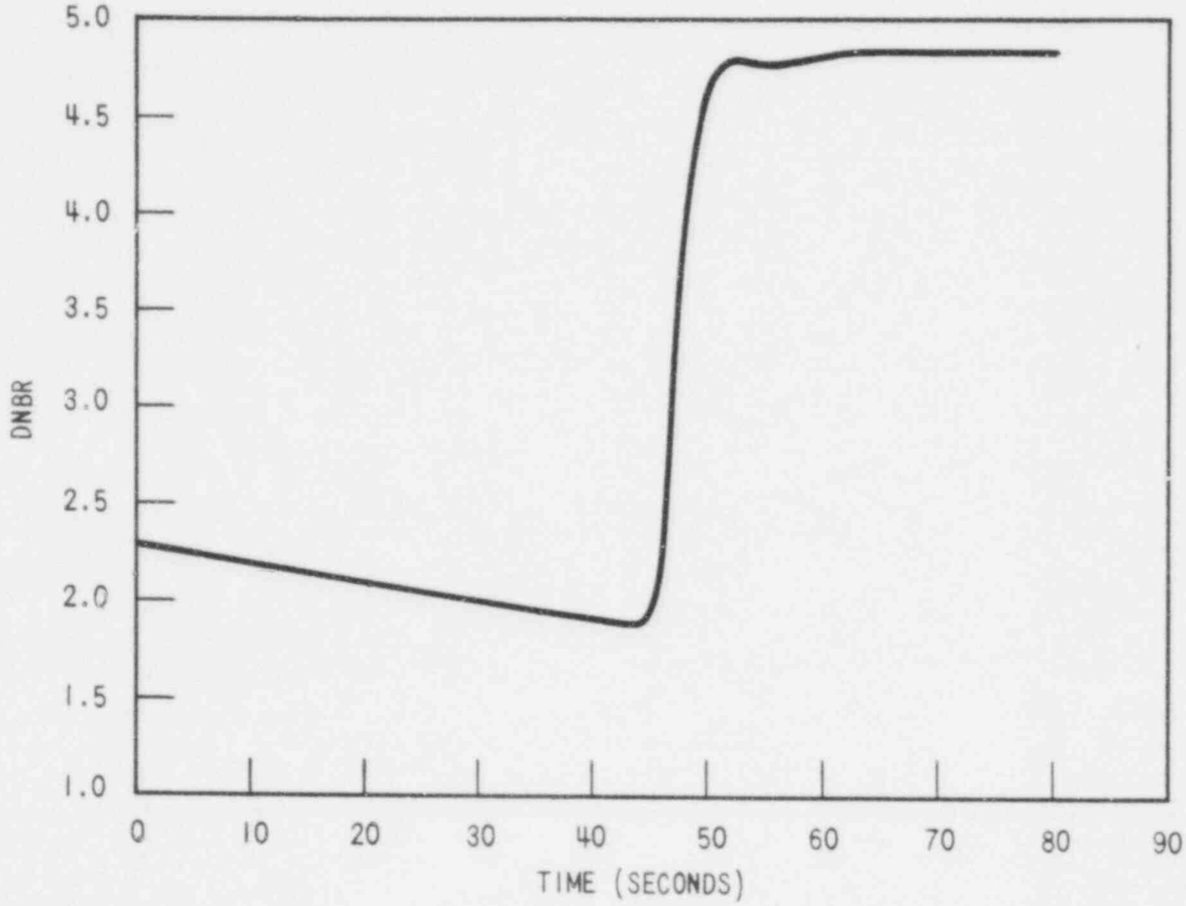
Pressurizer Pressure Transients and Core Average Temperature Transient for Inadvertent Opening of a Pressurizer Safety Valve

14,395-112



614 080

WCAP – 9500
Figure 15.6-2a. Pressurizer Pressure Transients and Core Average Temperature Transient for Inadvertent Opening of a Pressurizer Safety Valve (N-1 Loop Operation)

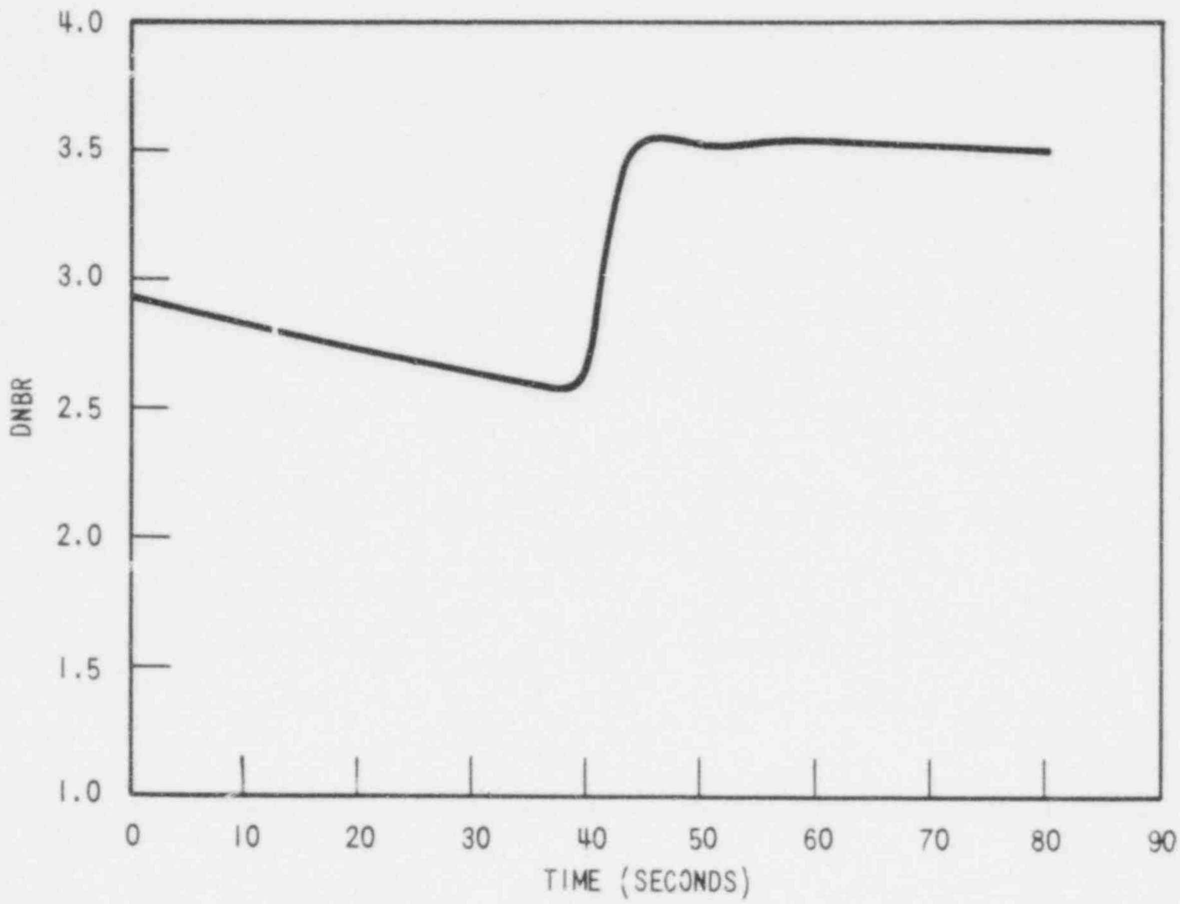


614 081

WCAP - 9500

Figure 15.6-3.

Transient for Inadvertent Opening of a Pressurizer Safety Valve

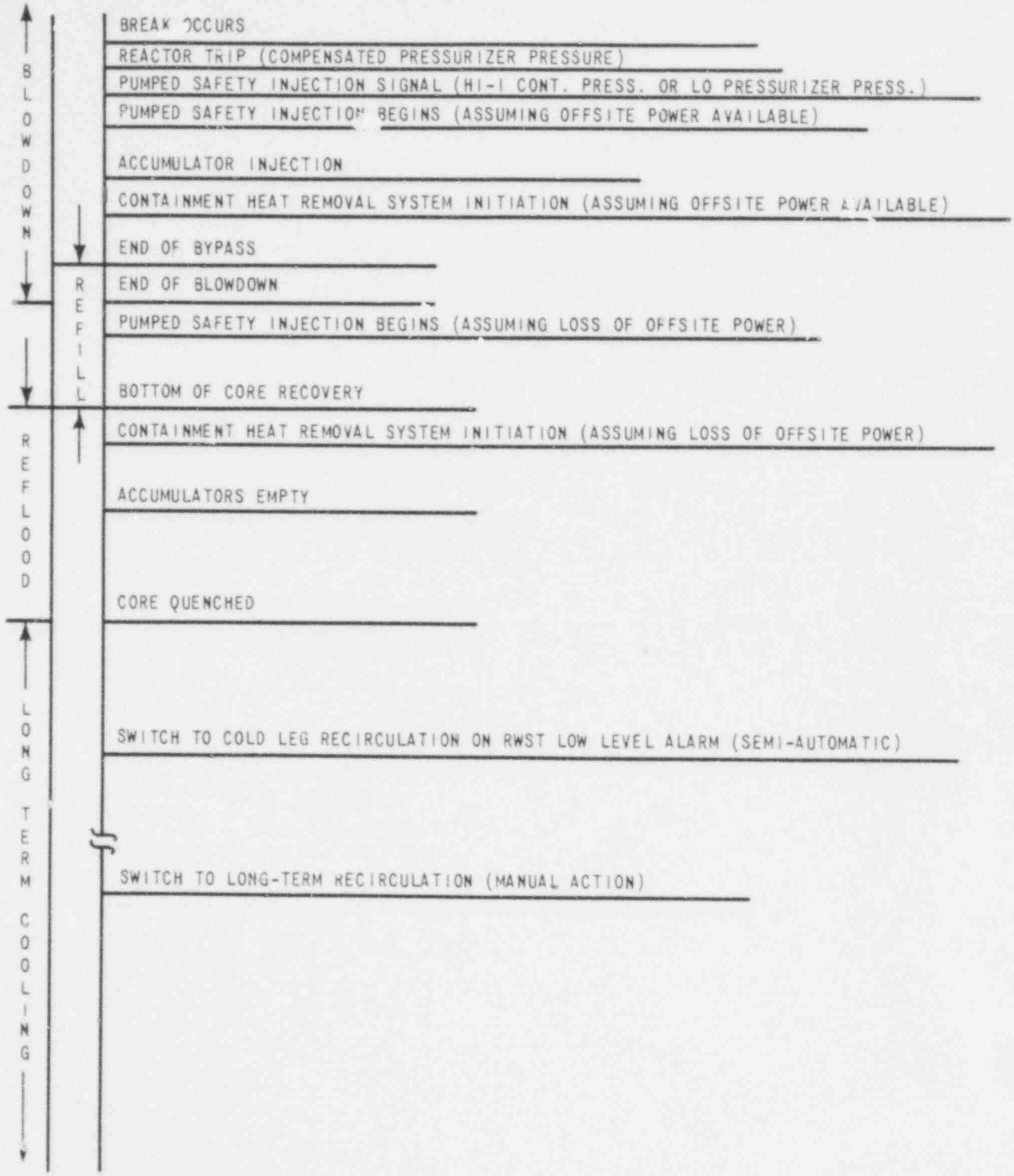


614 082

WCAP – 9500

Figure 15.6-3a.

Transient for Inadvertent Opening of a Pressurizer Safety Valve (N-1 Loop Operation)

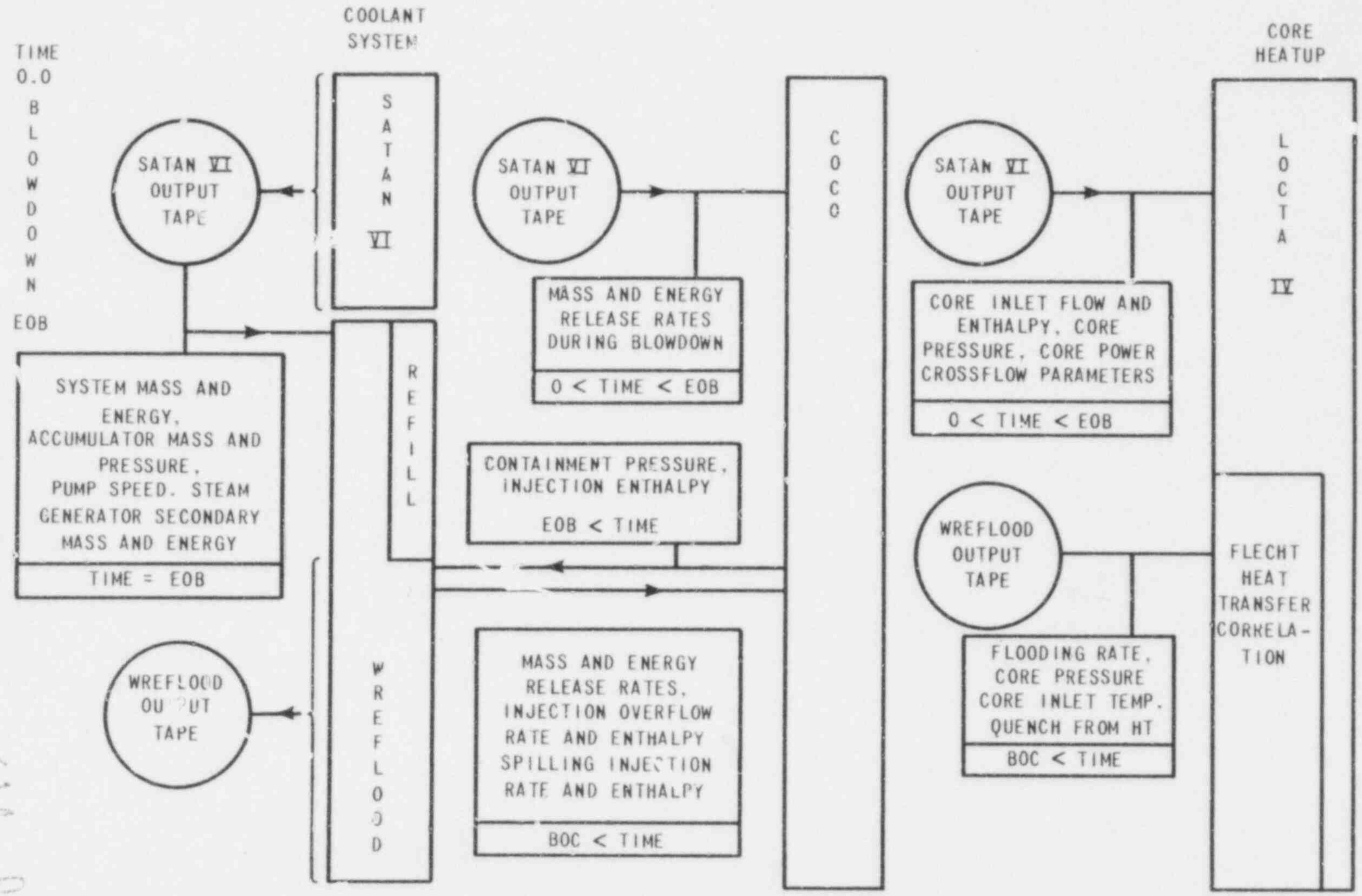


614 083

WCAP – 9500

Figure 15.6-4.

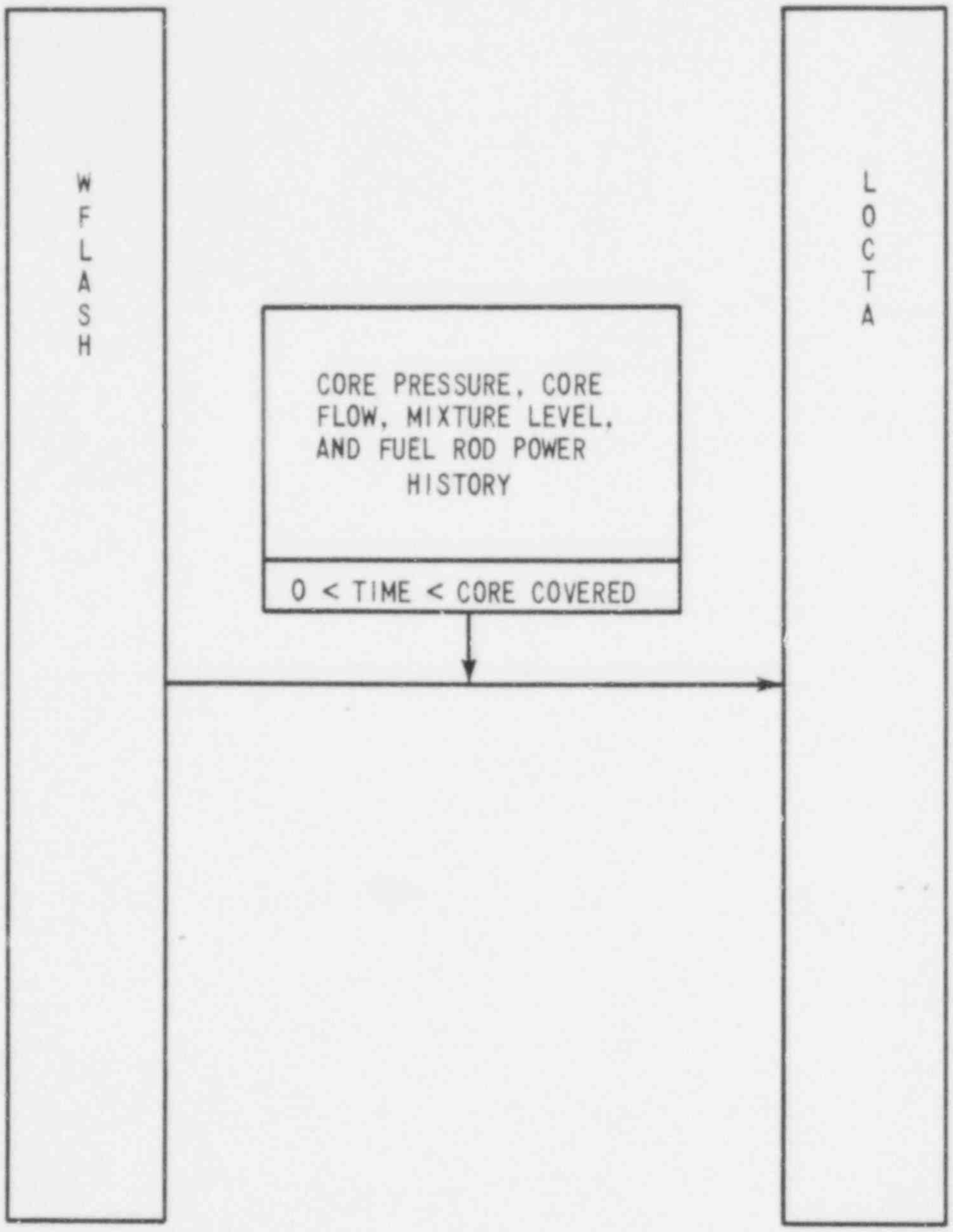
Sequence of Events for Large Break LOCA Analysis



WCAP - 9500

Figure 15.6-5.
Code Interface Description for Large Break Model

614 084



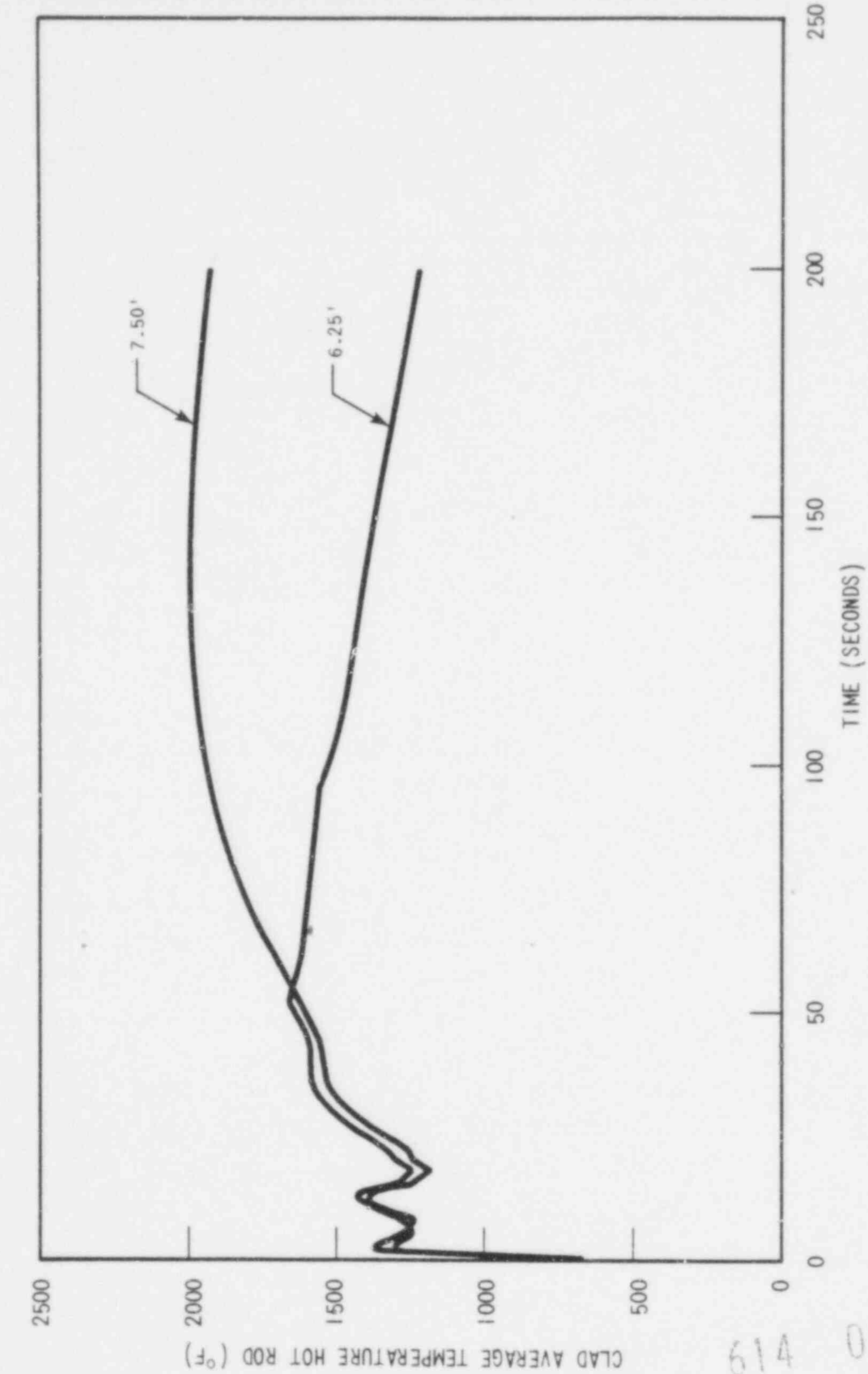
614 085

WCAP – 9500

Figure 15.6-6.

Code Interface Description for Small Break Model

14,395-311



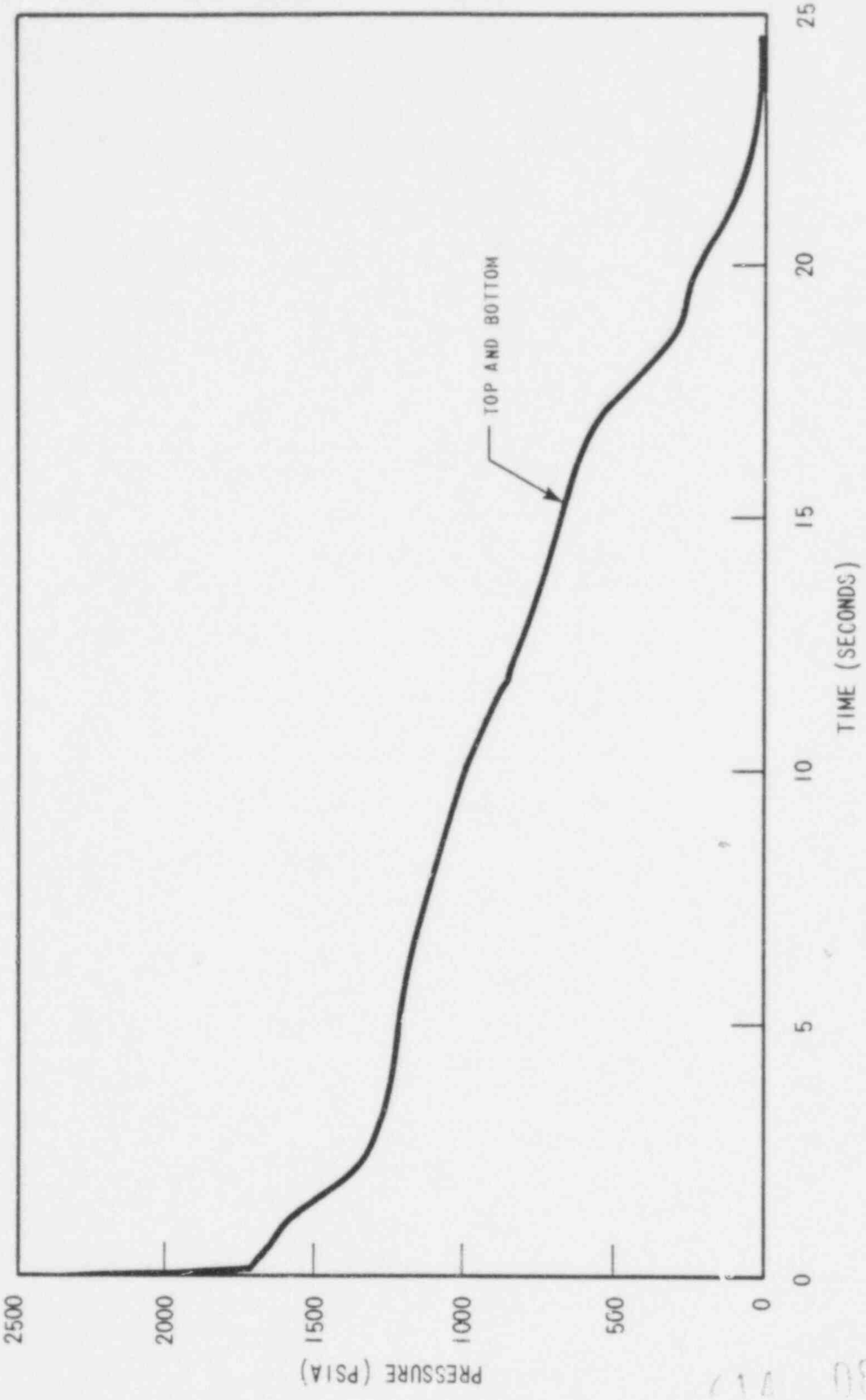
614 086

WCAP - 9500

Figure 15.6-7.

Peak Clad Temperature
DECLG (C_D = 0.6)

14,395-312

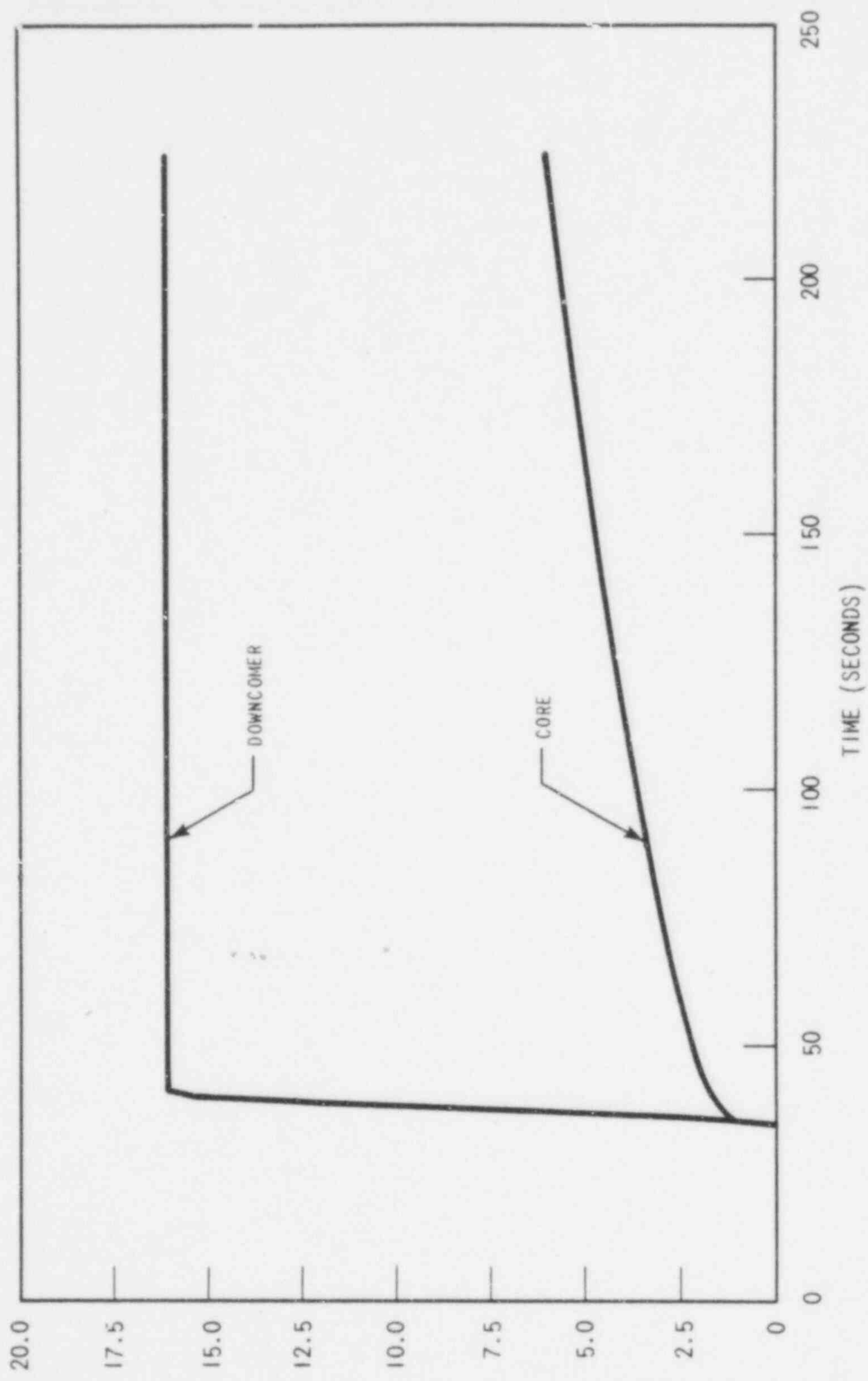


WCAP – 9500

Figure 15.6-8.

Core Pressure DECLG (C_D = 0.6)

14,395-313



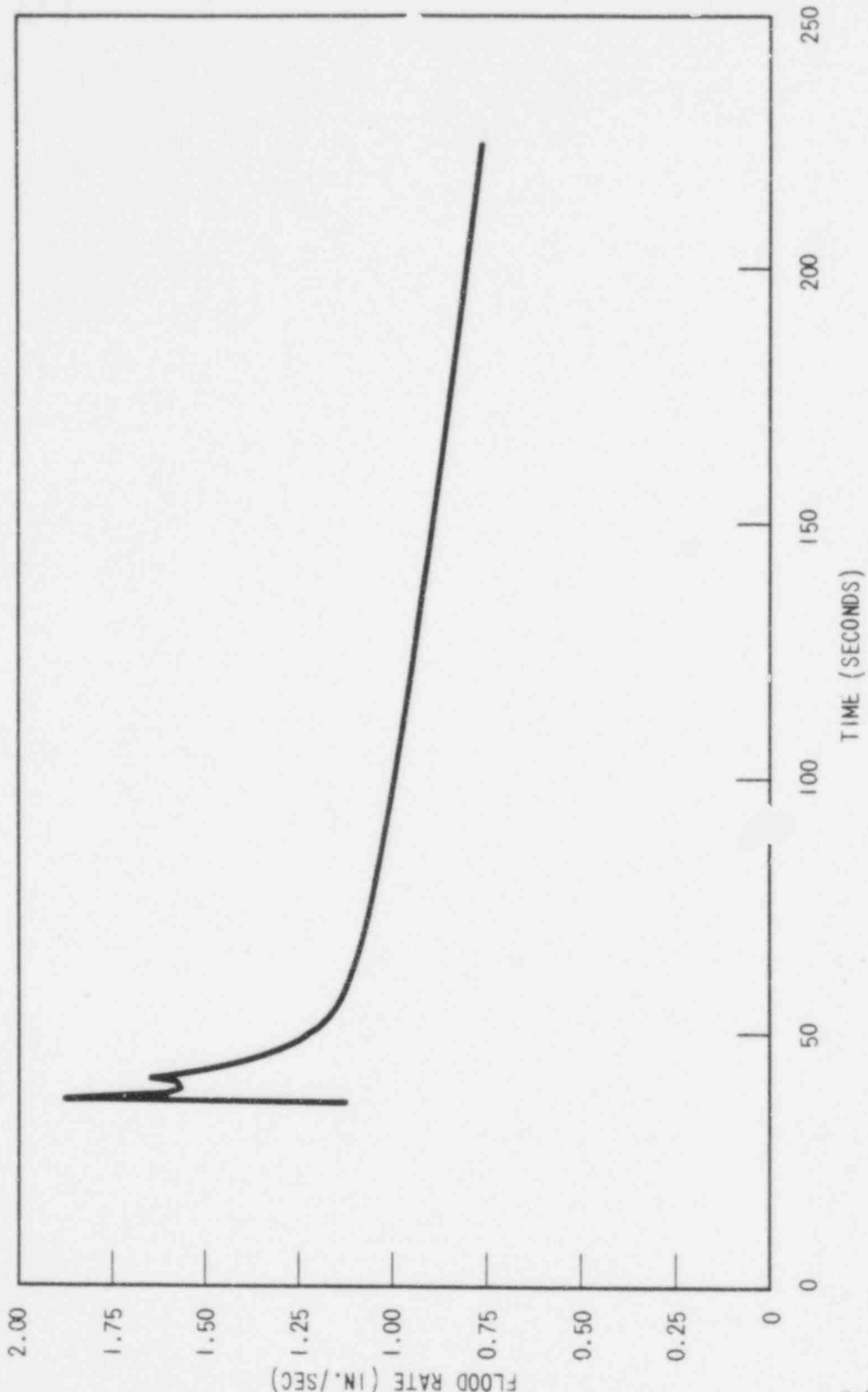
614 088

WCAP - 9500

Figure 15.6-9.

Downcorner and Core Water Levels During Reflood -- DECLG (C_D = 0.6)

14,395-314



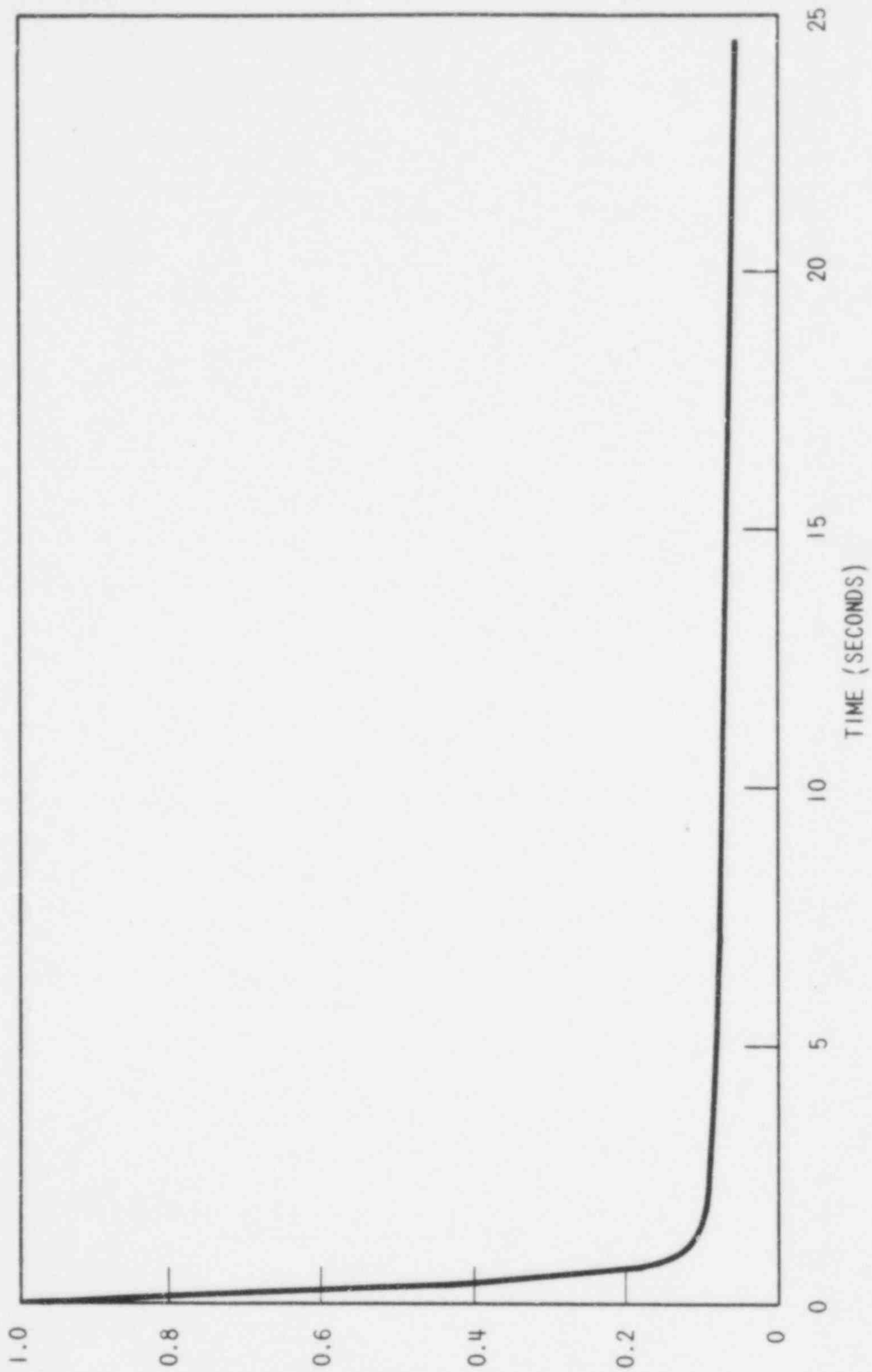
614 089

WCAP-9500

Figure 15.6-10.

Core Inlet Velocity During Reflood DECLG (C_D = 0.6)

14,395-315

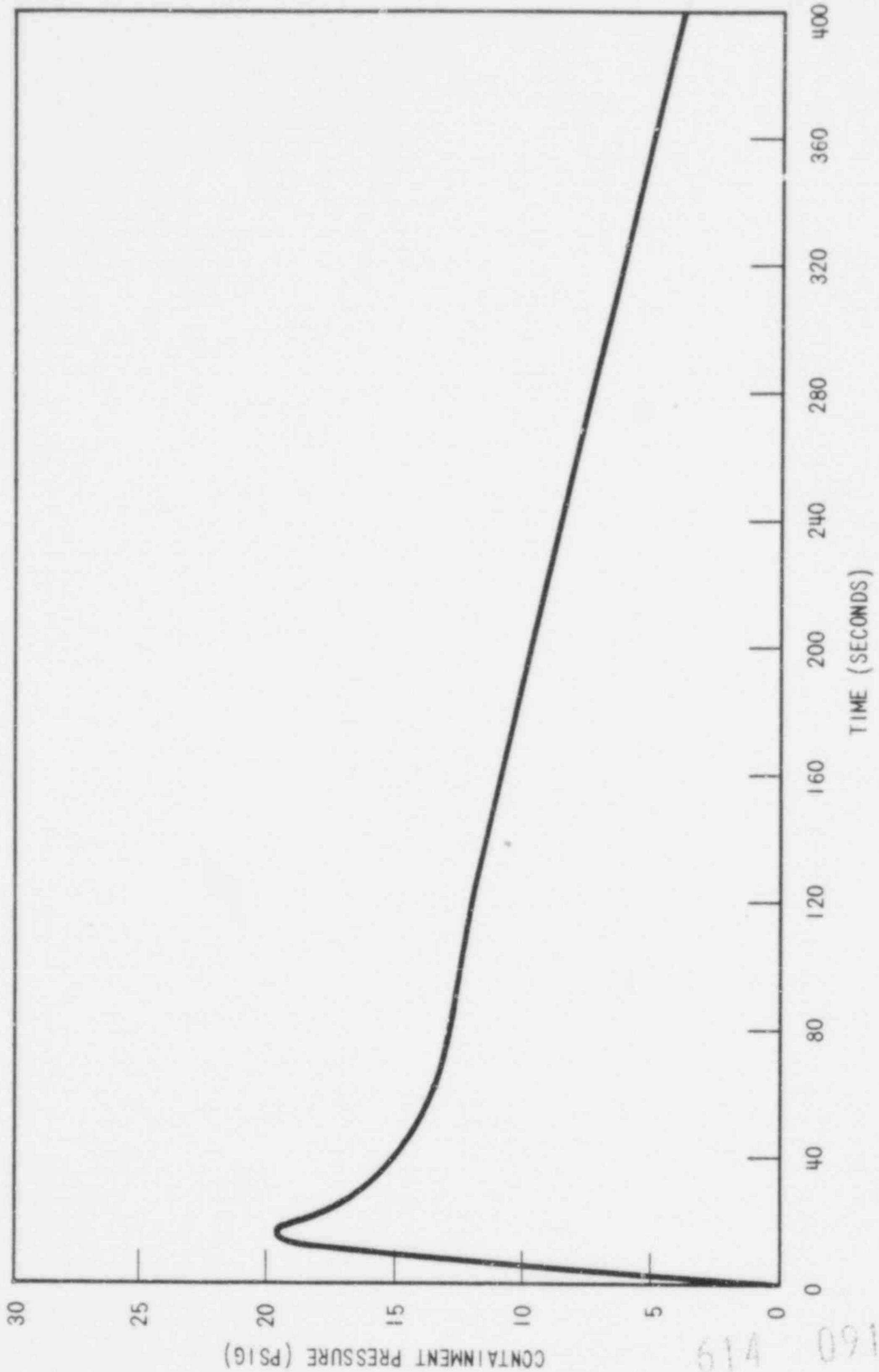


614 090

WCAP - 9500

Figure 15.6-11.

Core Power Transient
DECLG (C_D = 0.6)



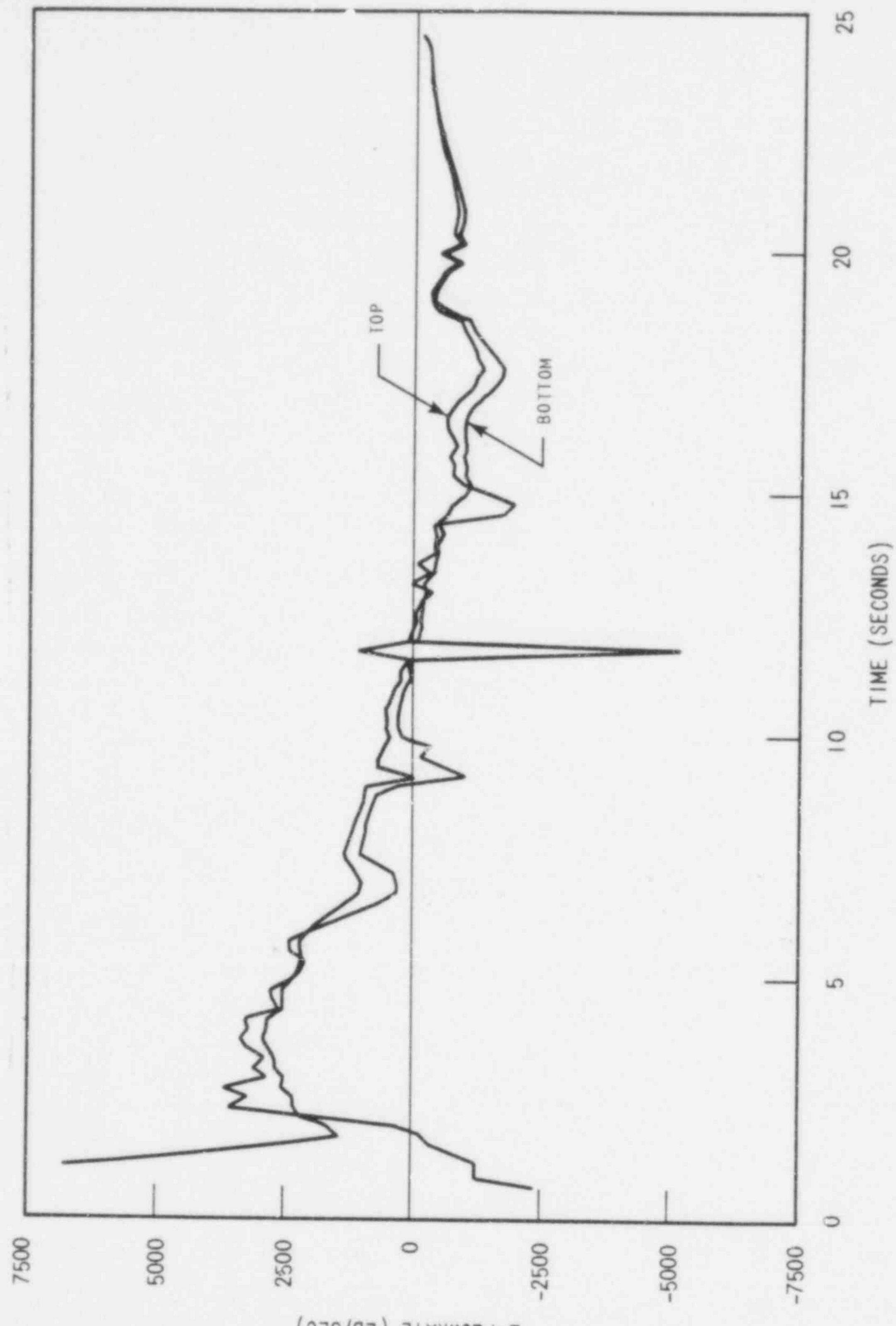
614 091

WCAP - 9500

Figure 15.6-12

Containment Pressure DECLG (C_D = 0.6)

14,395-317

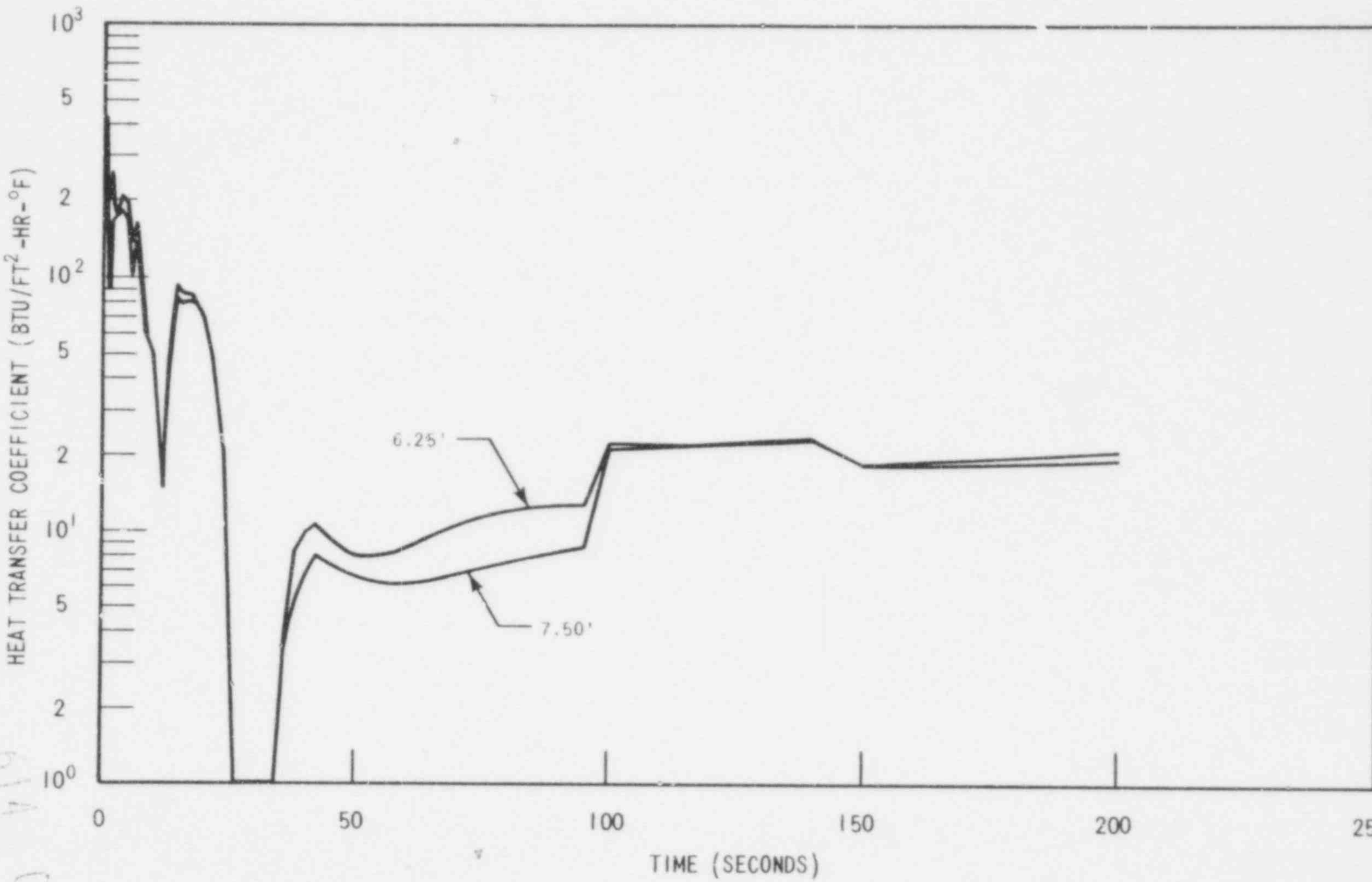


614 092

WCAP – 9500

Figure 15.6-13.

Core Flow-Top and Bottom - DECLG (C_D = 0.6)



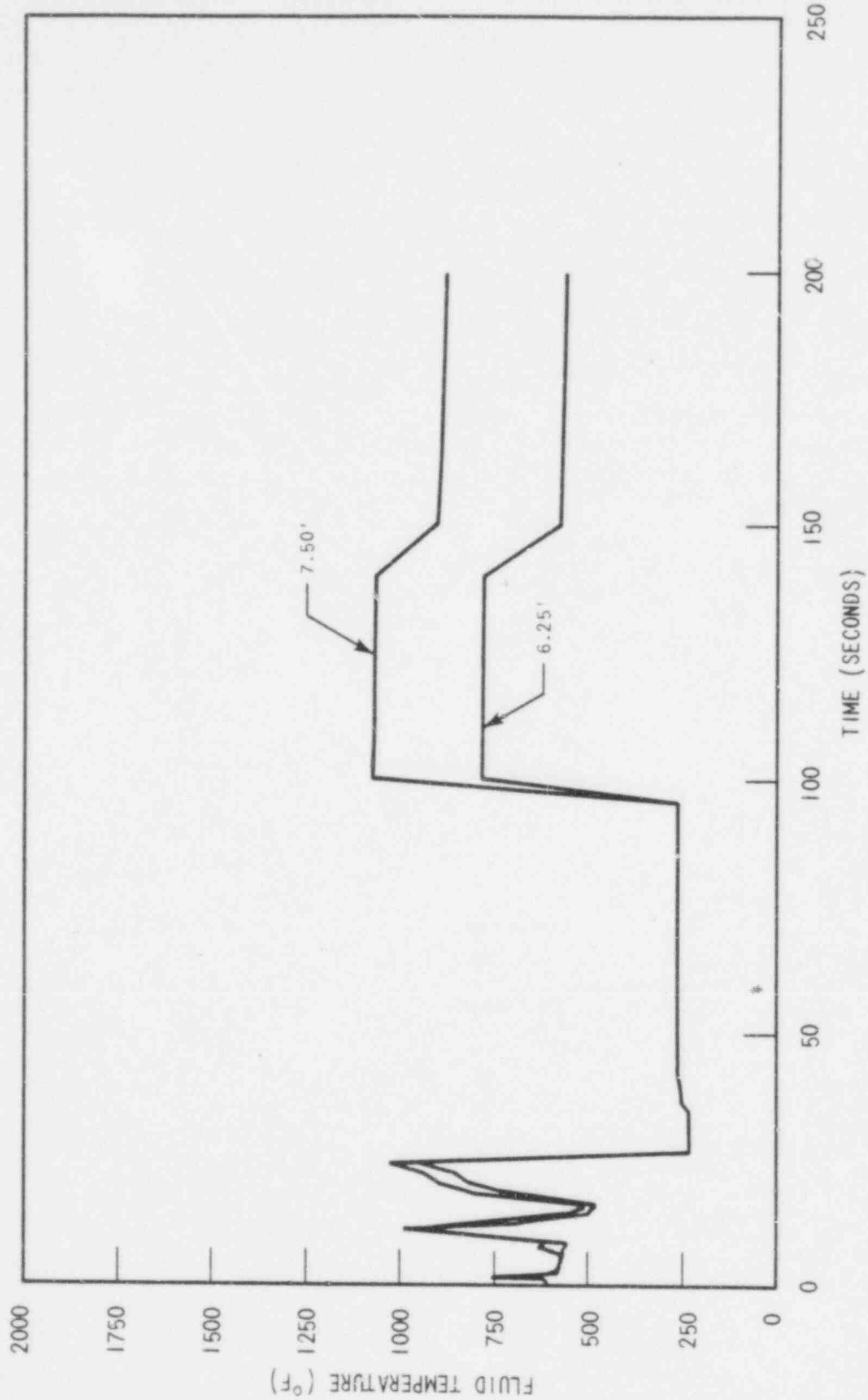
614 093

WCAP - 9500

Figure 15.6-14.
Core Heat Transfer
Coefficient - DECLG (C_D = 0.6)

14,395-318

14,395-319

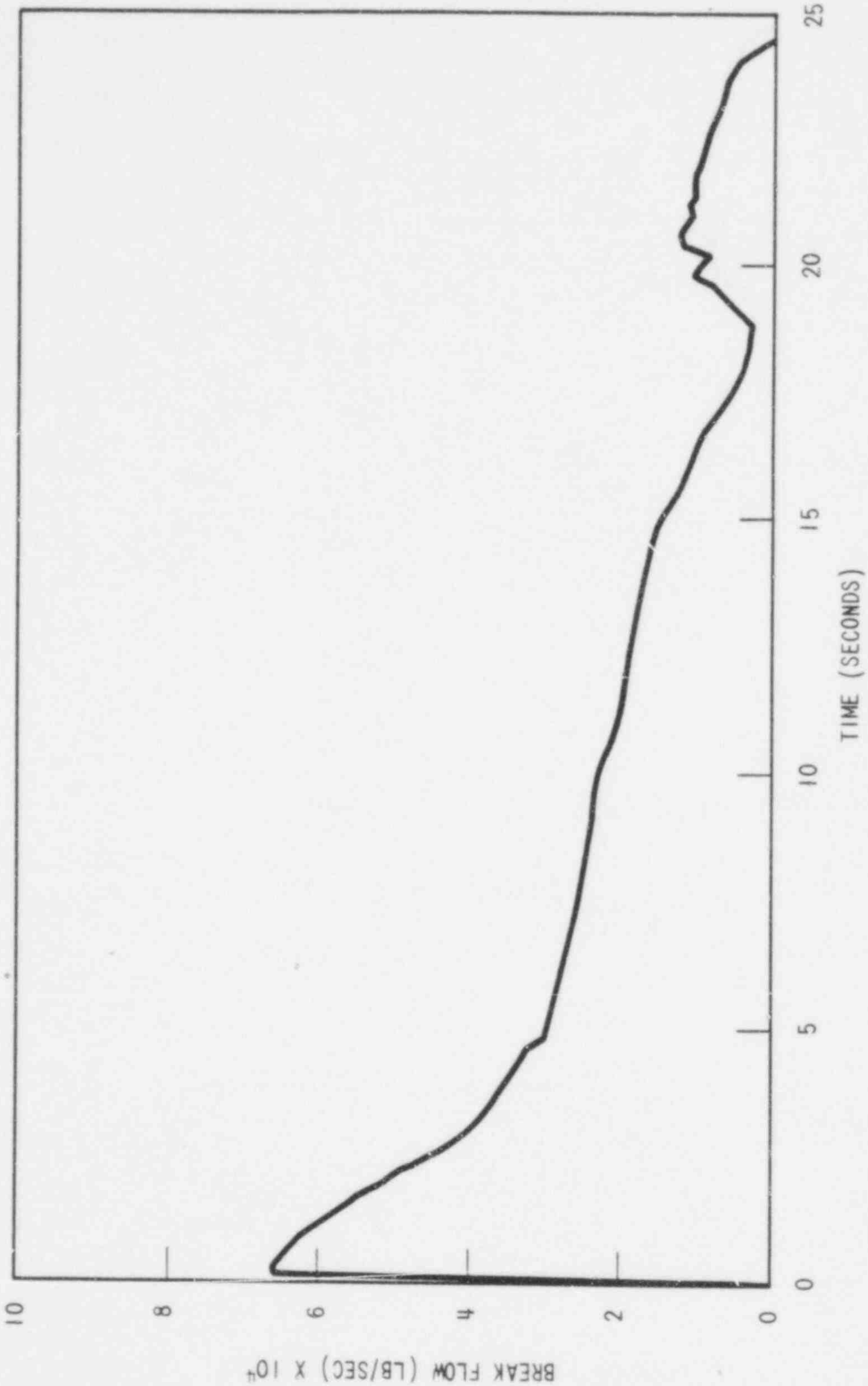


614 094

WCAP – 9500

Figure 15.6-15.

Fluid Temperature DECLG (C_D = 0.6)



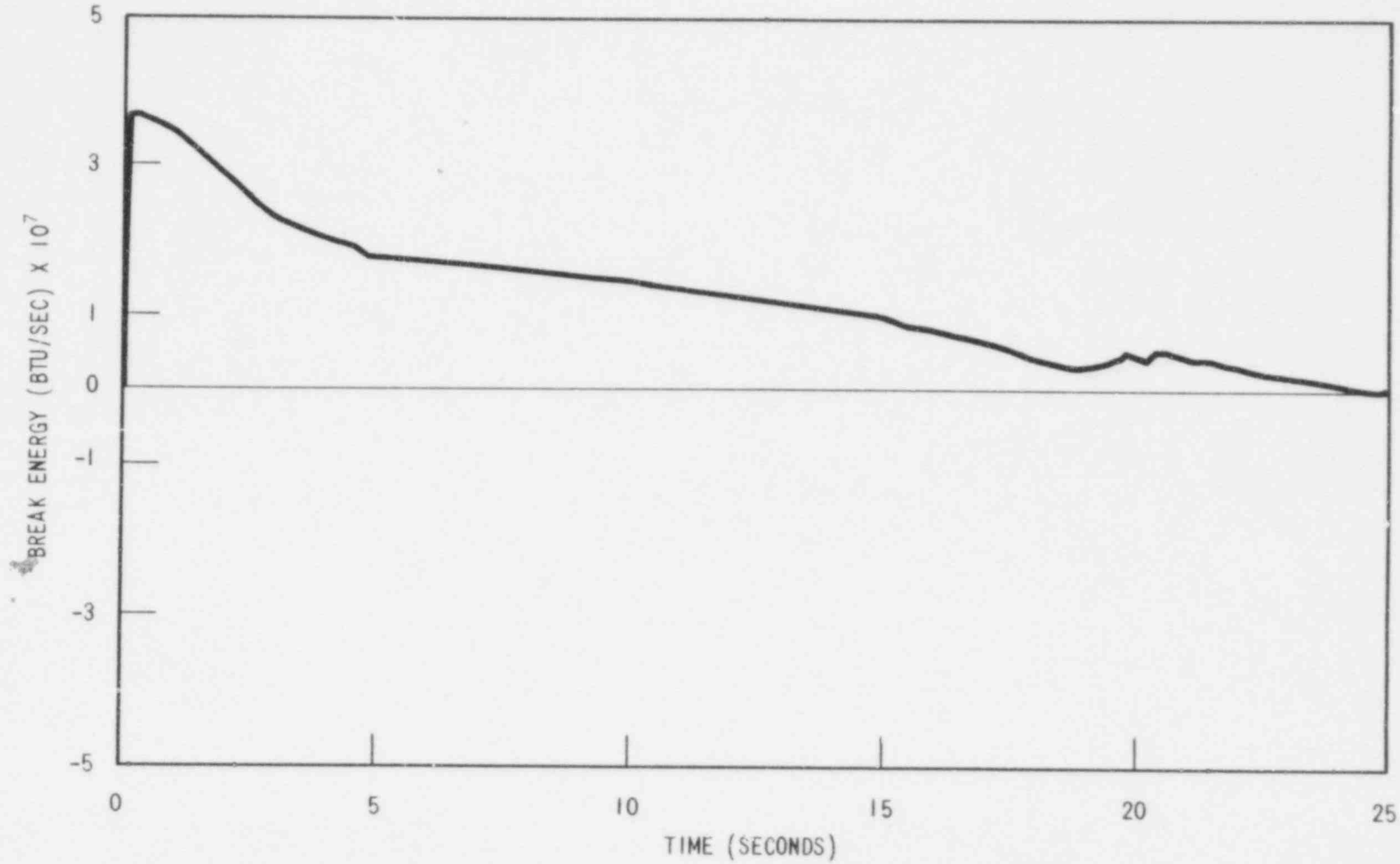
614 095

WCAP – 9500

Figure 15.6-16.

Break Mass Flowrate
DECLG (C_D = 0.6)

614 096

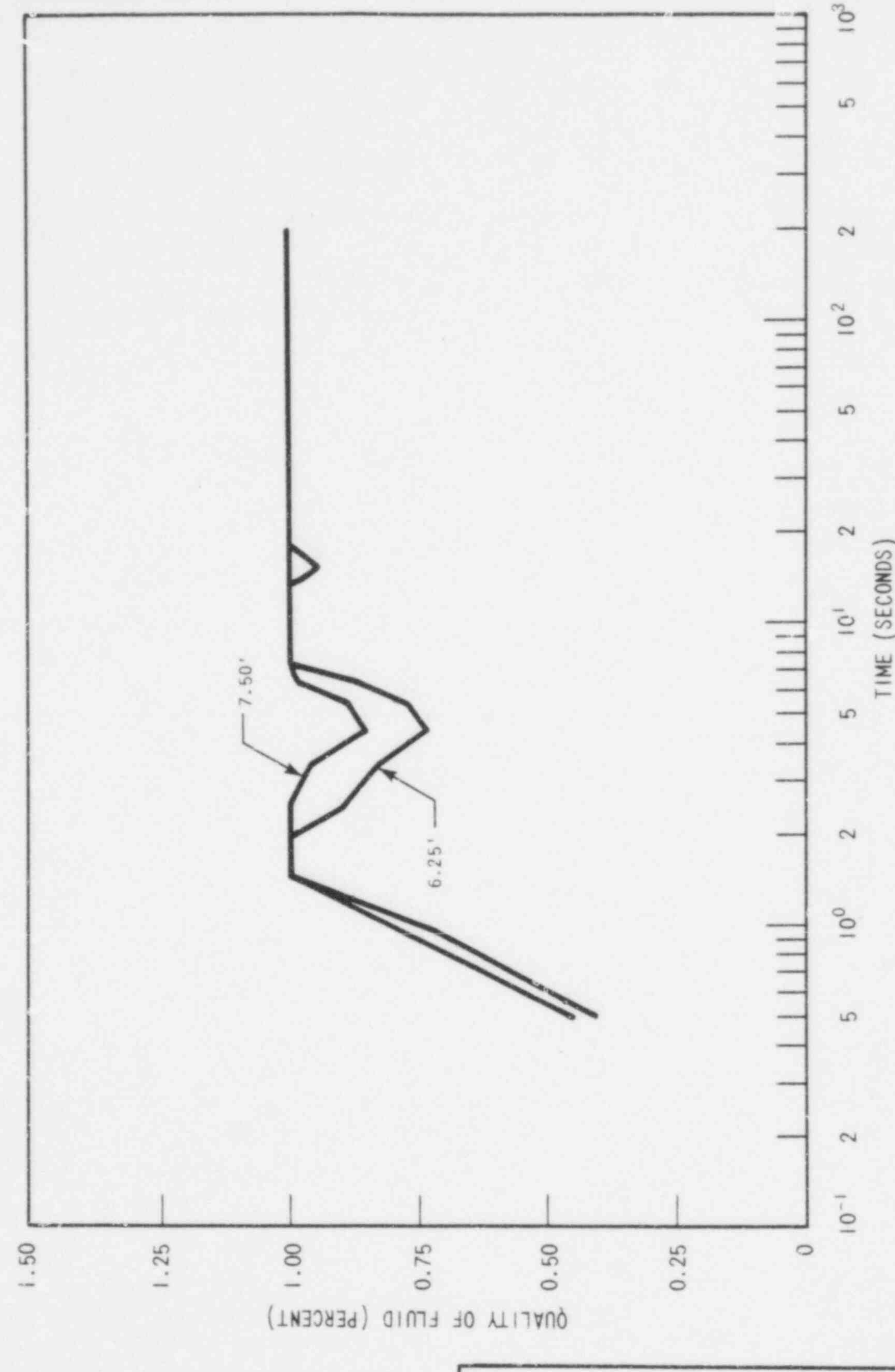


14,395-321

WCAP - 9500

Figure 15.6-17.
Break Energy Release Rate
DECLG (C_D = 0.6)

14,395-322

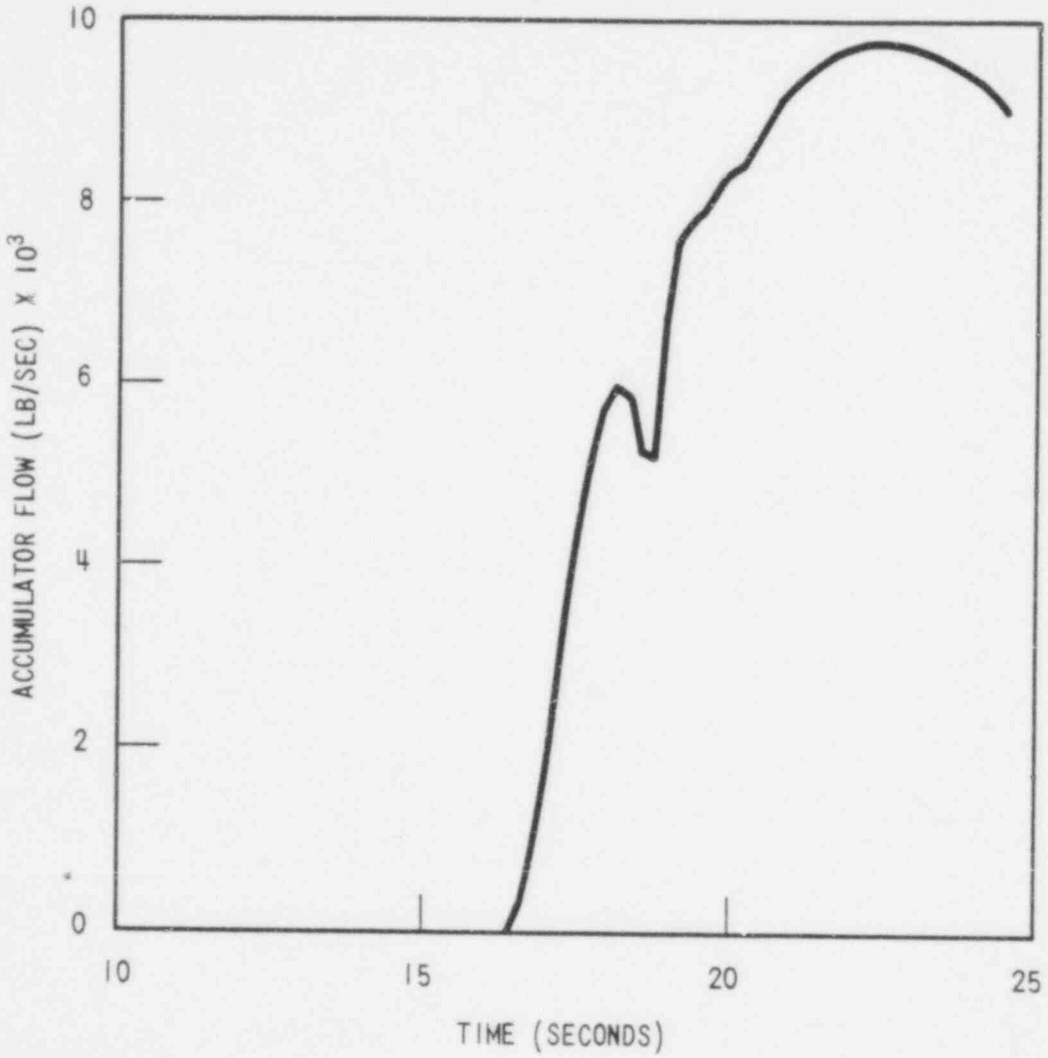


614 097

WCAP – 9500

Figure 15.6-18.

Fluid Quality DECLG (C_D = 0.6)



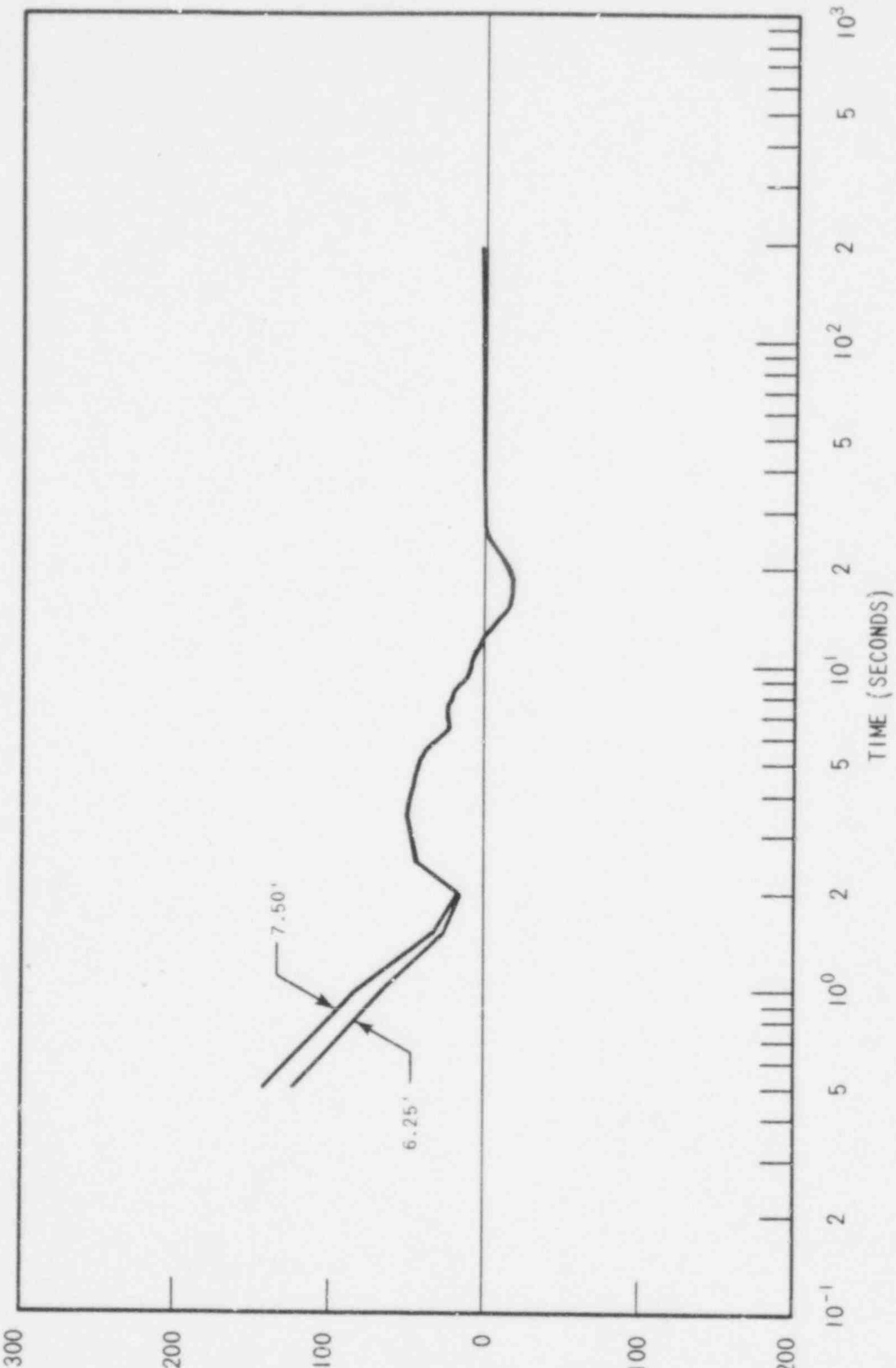
WCAP - 9500

Figure 15.6-19.

Accumulator Flow During Blowdown DECLG (C_D = 0.6)

614 098

14,395-324

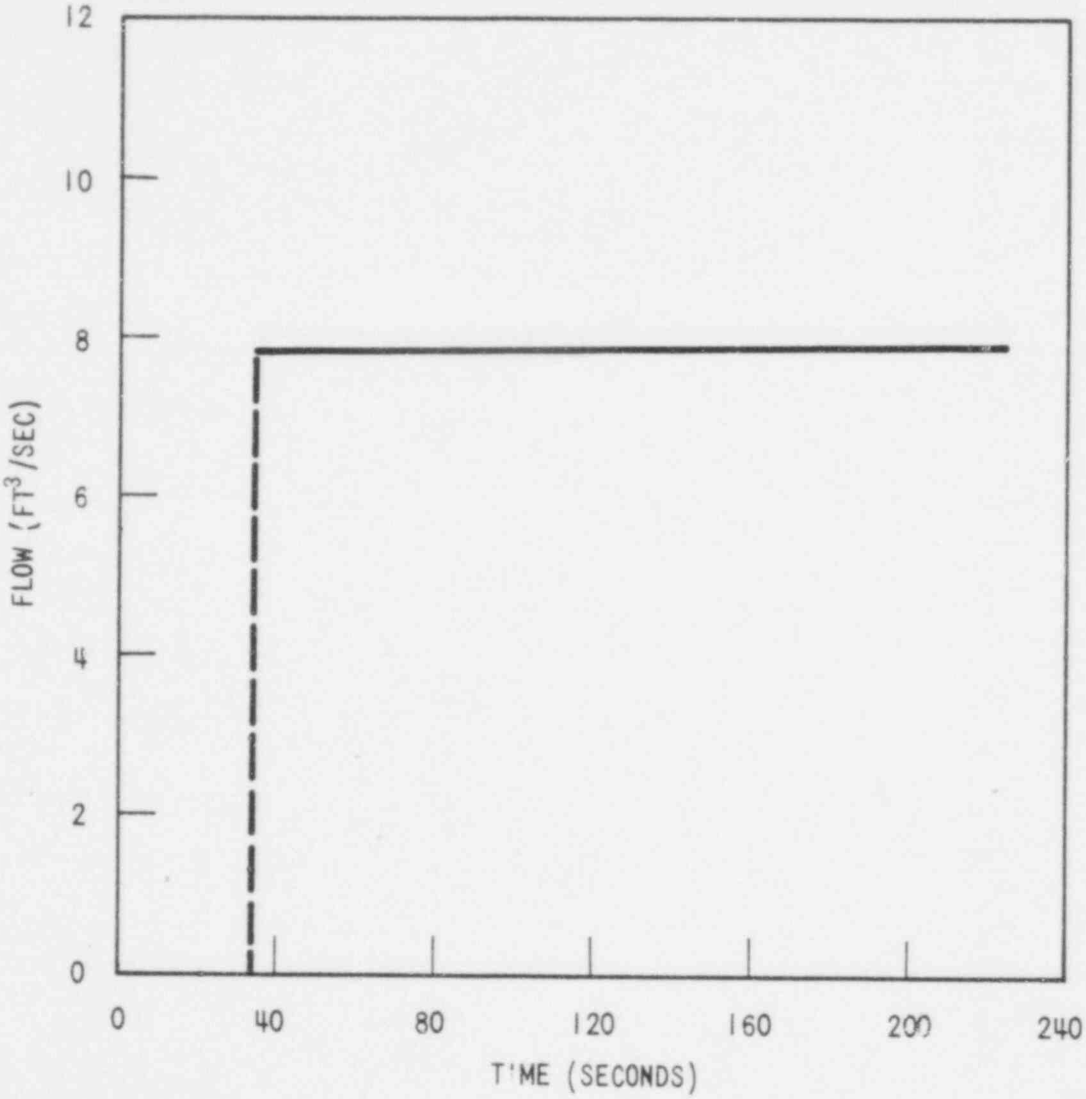


WCAP – 9500

Figure 15.6-20.

Mass Velocity DECLG (C_D = 0.6)

614 099



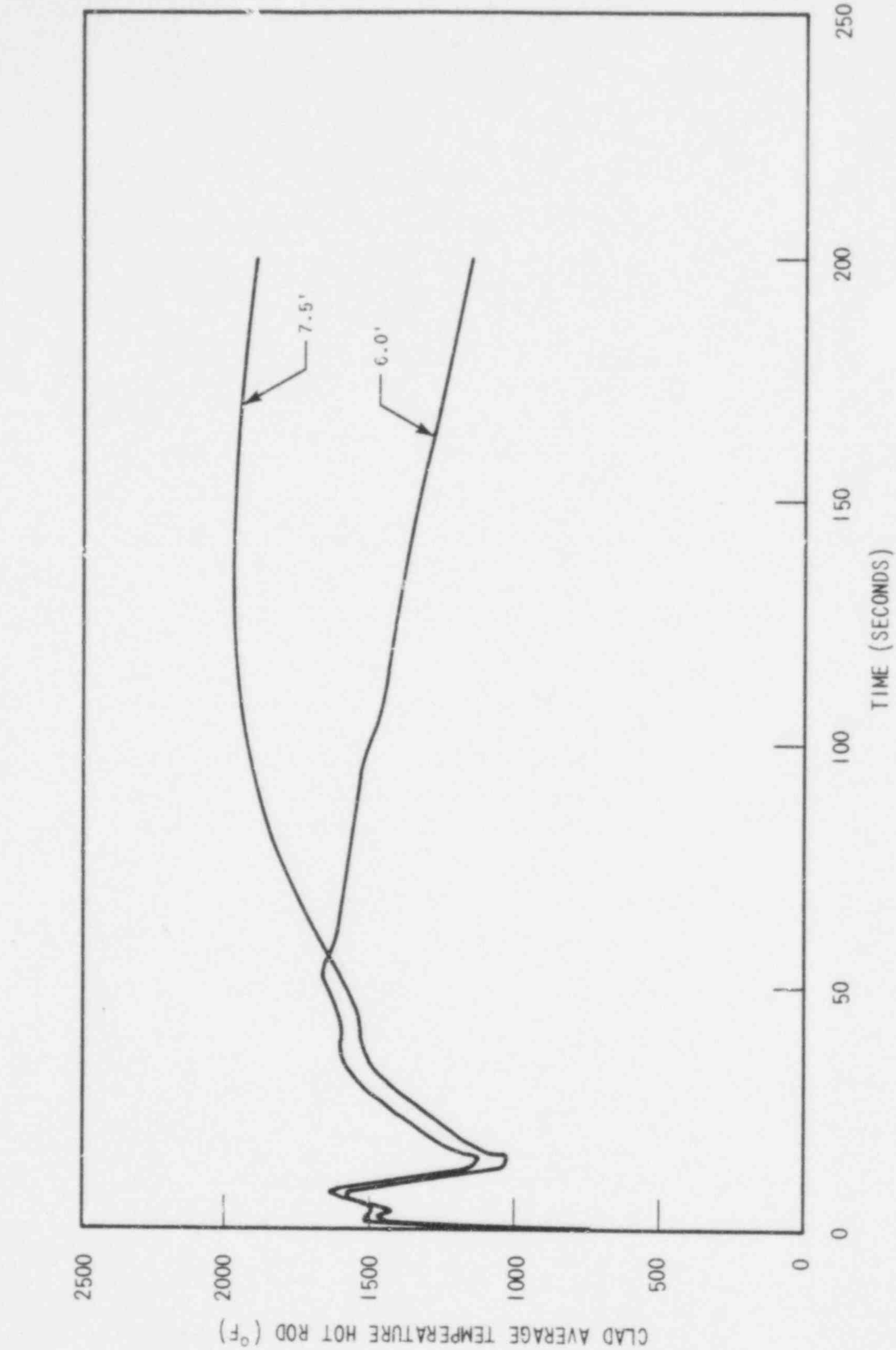
614 100

WCAP - 9500

Figure 15.6-21.

Pumped ECCS Flow During Reflood - DECLG (C_D = 0.6)

14,395-326

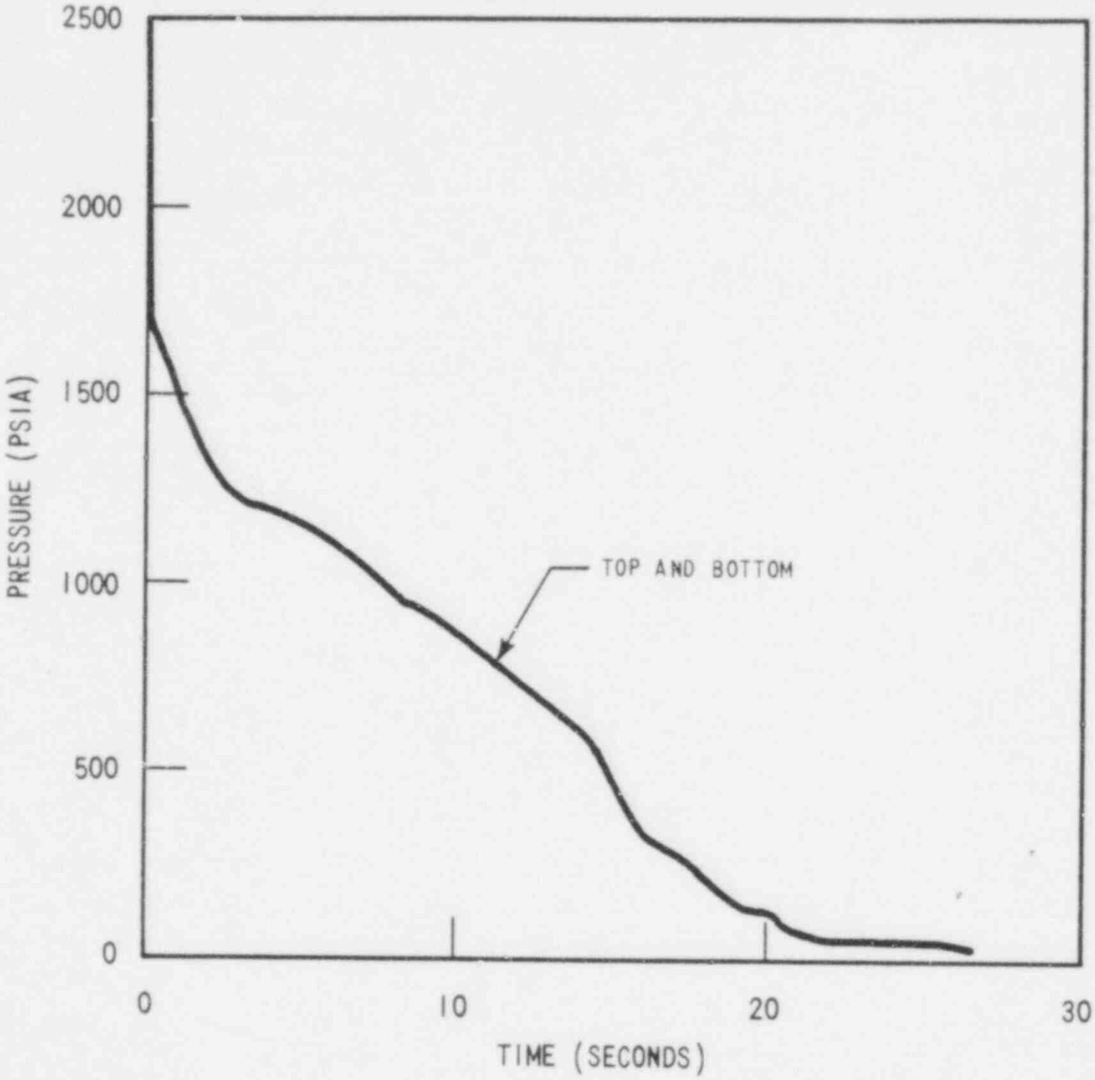


614 101

WCAP - 9500

Figure 15.6-22.

Peak Clad Temperature
DECLG (C_D = 0.8)



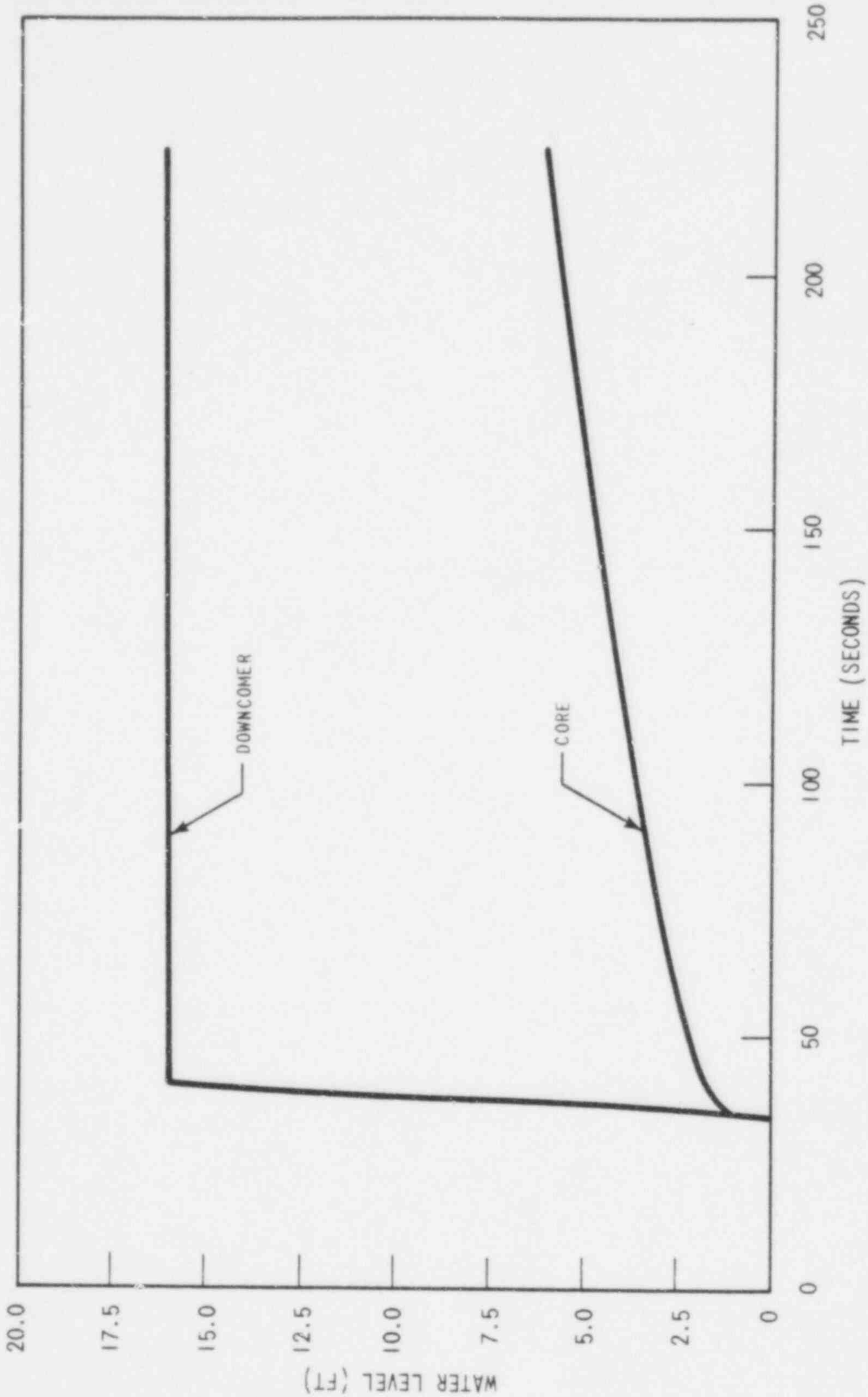
WCAP-9500

Figure 15.6-23.

Core Pressure DECLG (C_D = 0.8)

614 102

14,395-328



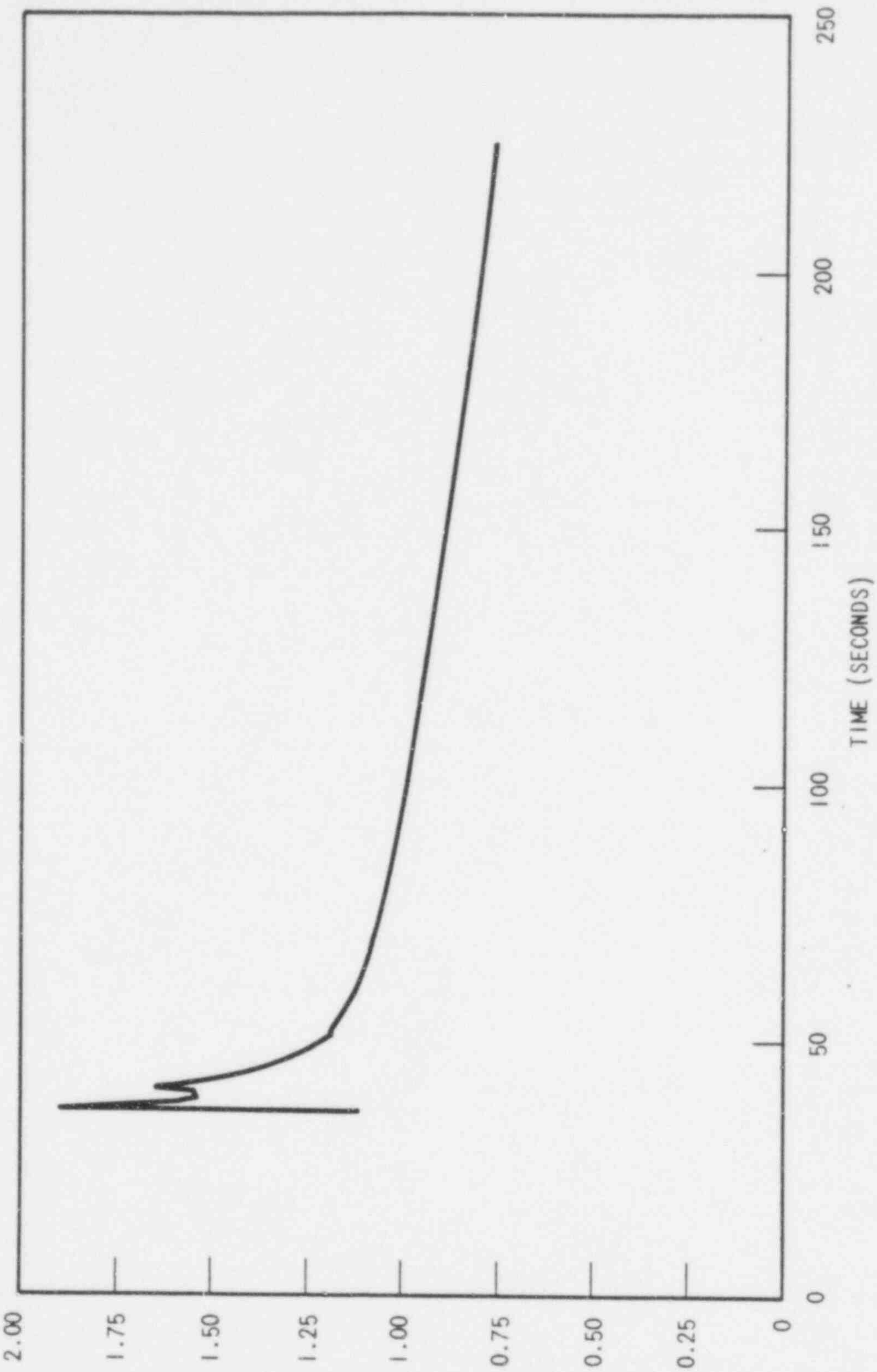
WCAP - 9500

Figure 15.6-24.

Downcomer and Core Water Levels During Reflood (C_D = 0.8)

614 103

14,395-329

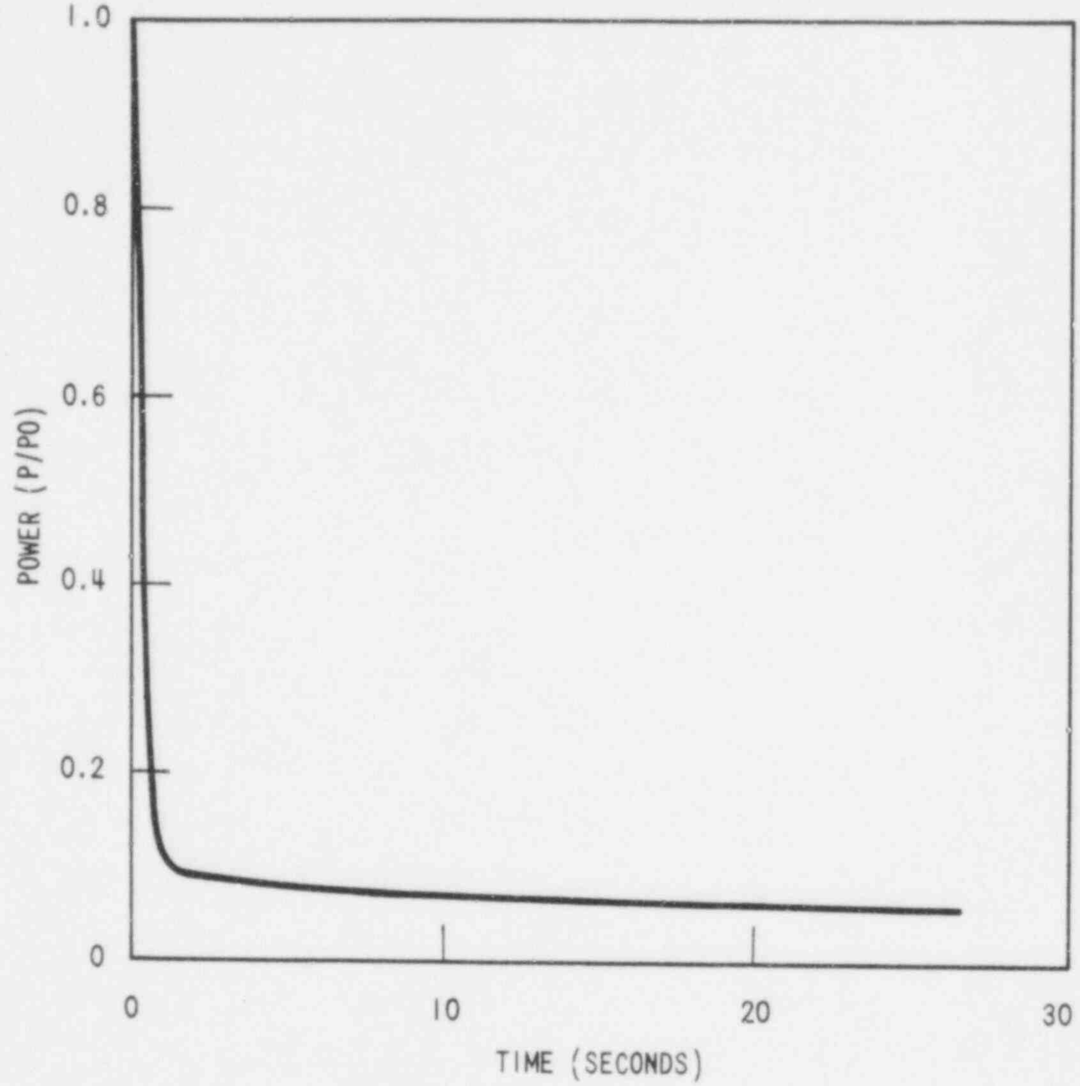


WCAP - 9500

Figure 15.6-25.

Core Inlet Velocity During Reflood
(C_D = 0.8)

614 104



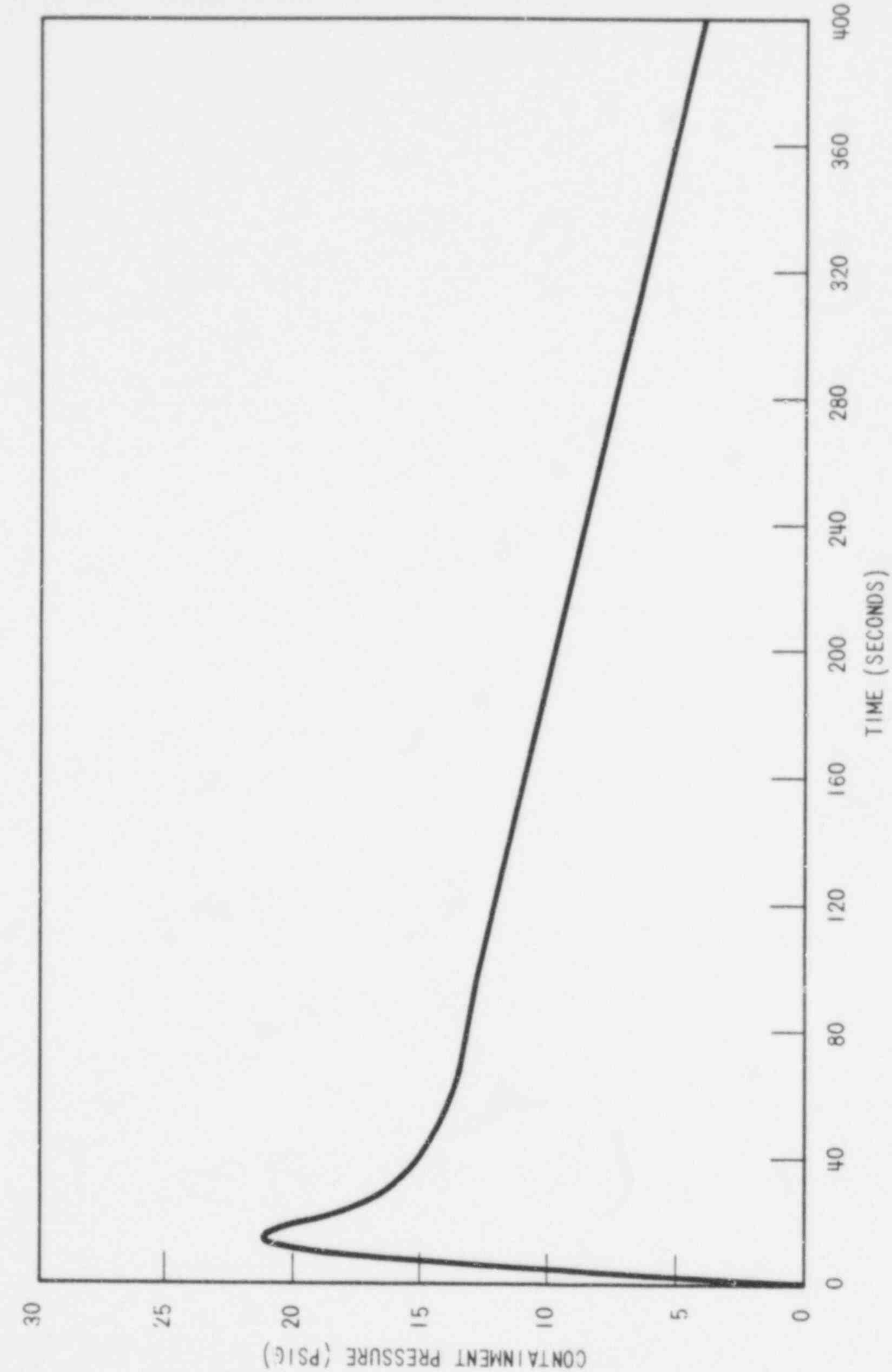
614 105

WCAP – 9500

Figure 15.6-26.

Core Power Transient
DECLG (C_D = 0.8)

14,395-331

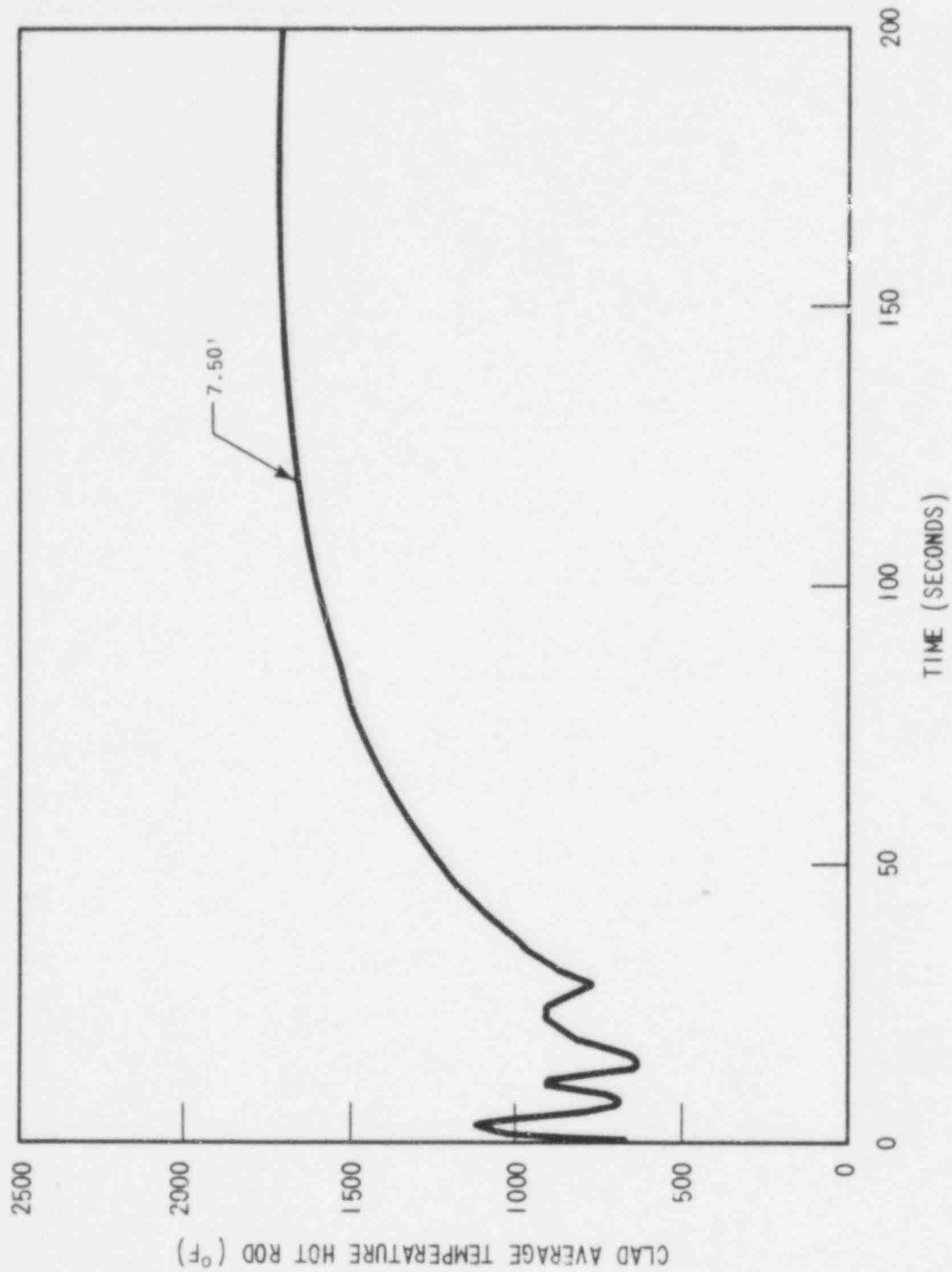


614 106

WCAP – 9500

Figure 15.6-27.

Containment Pressure DECLG (C_D = 0.8)

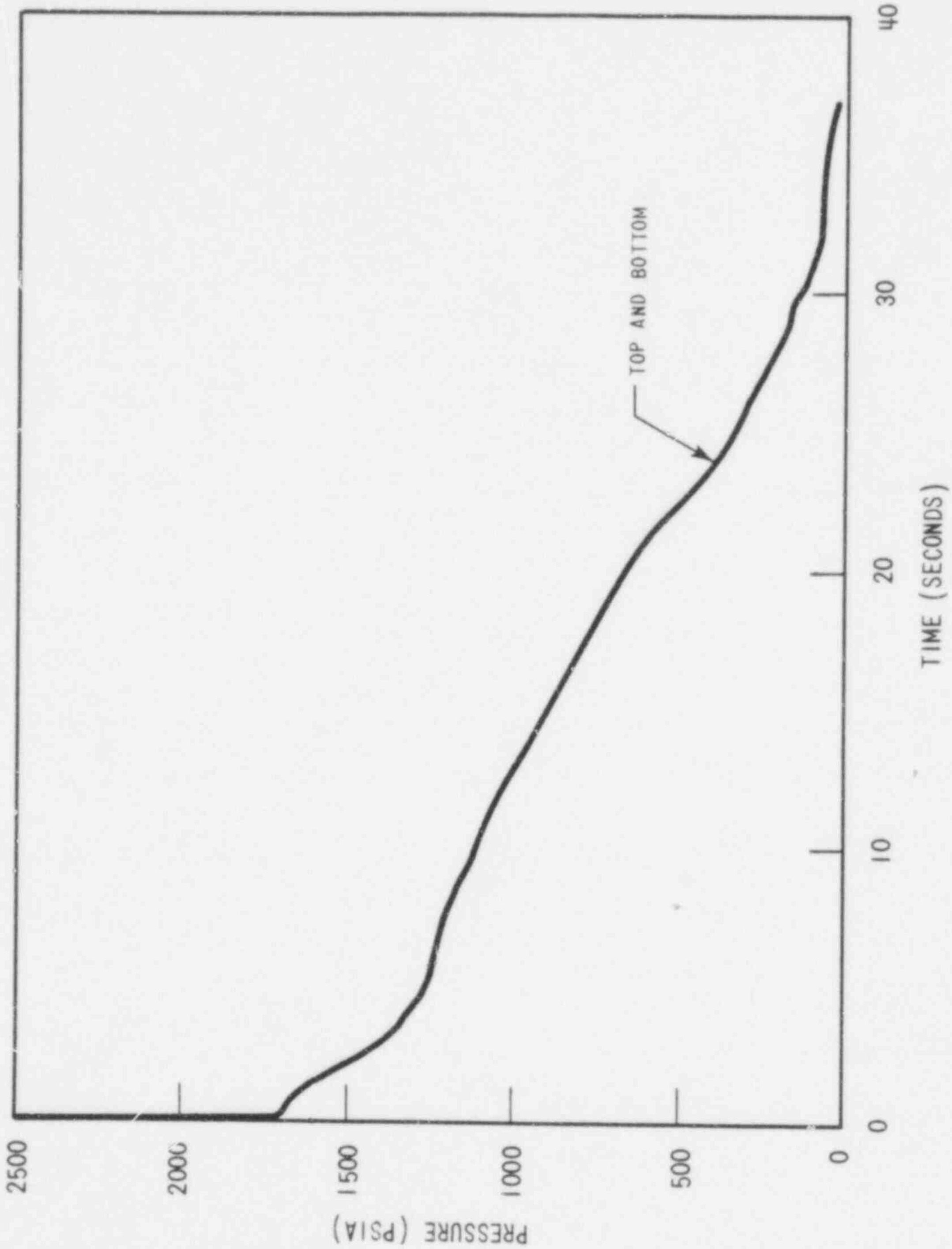


WCAP – 9500

Figure 15.6-28.

Peak Clad Temperature
DECLG (C_D = 0.4)

614 107



614 108

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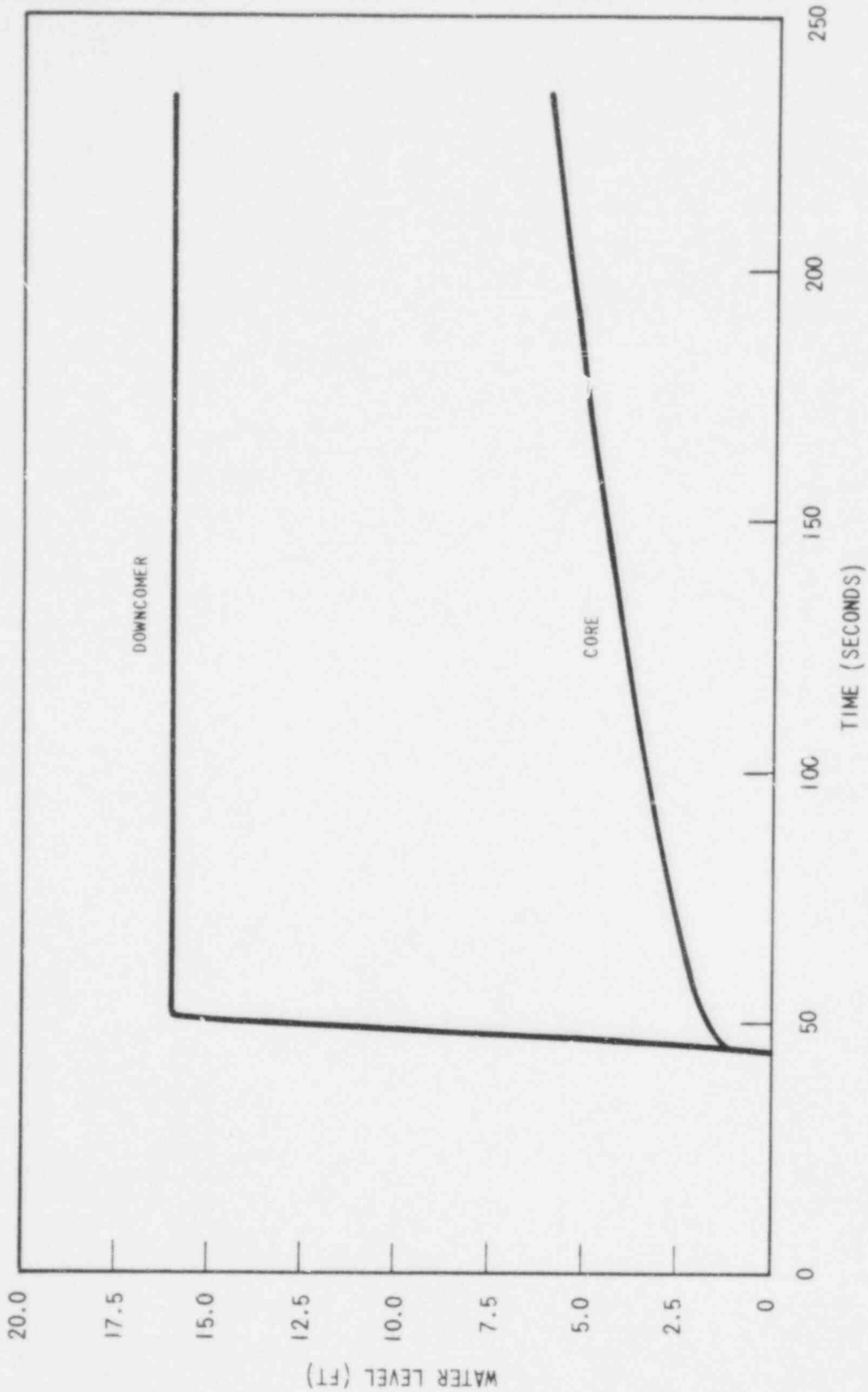
Figure 15.6-29.

Core Pressure DECLG (C_D = 0.4)

14,395-334



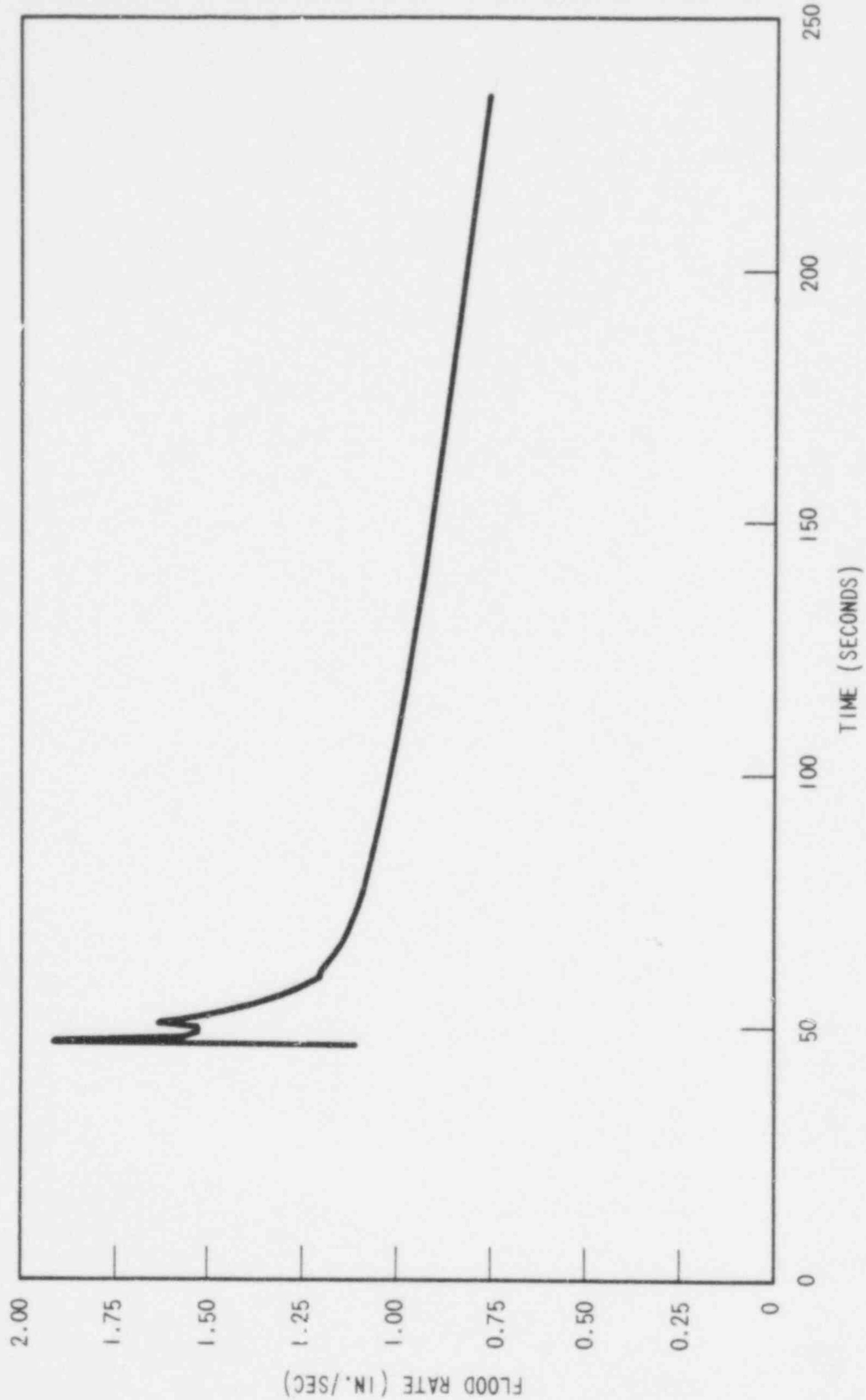
614 10

WCAP – 9500

Figure 15.6-30.

Downcomer and Core Water Level During Reflood DECLG (C_D = 0.4)

14,395-335

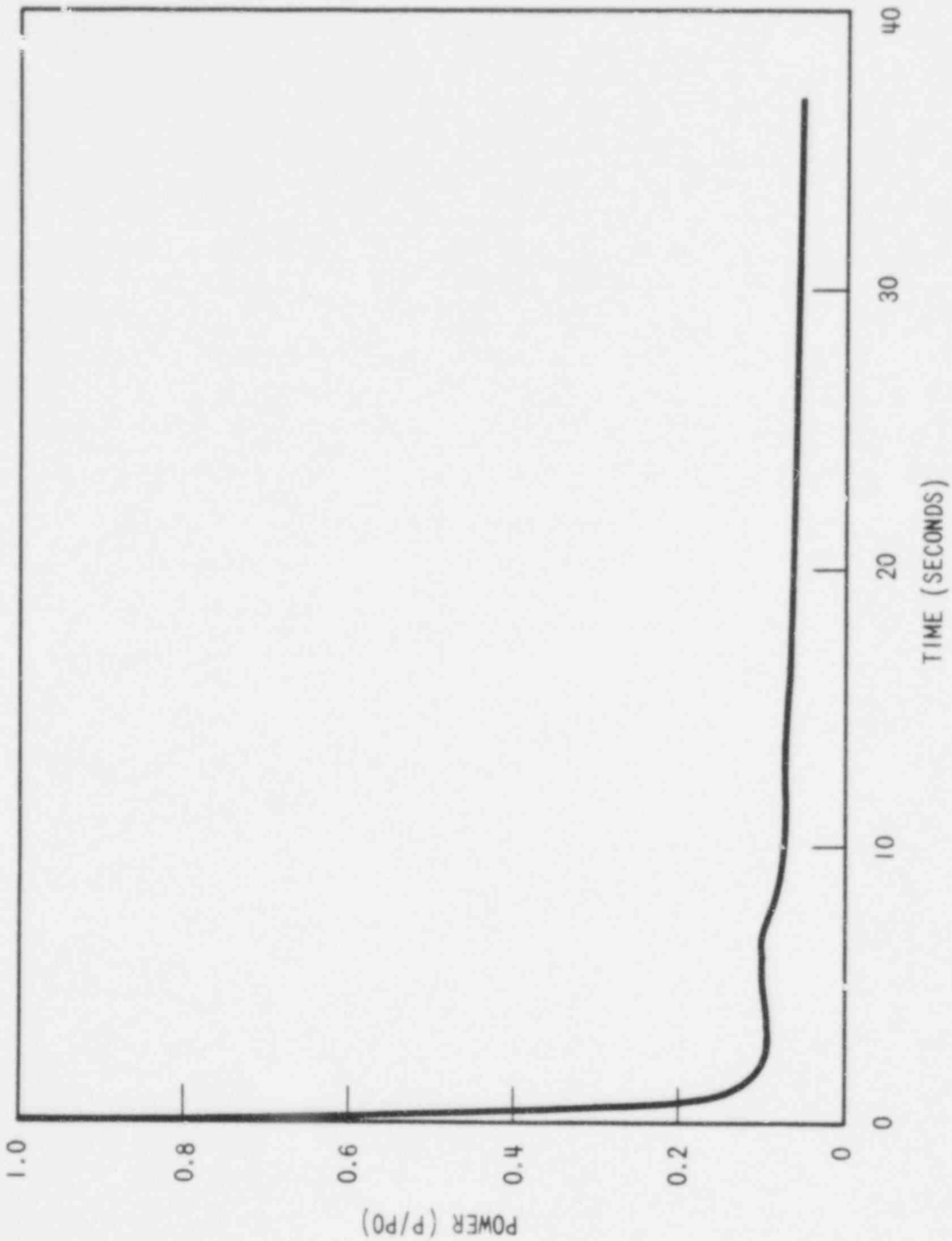


WCAP - 9500

Figure 15.6-31.

Core Inlet Velocity During Reflood
DECLG (C_D = 0.4)

614 110



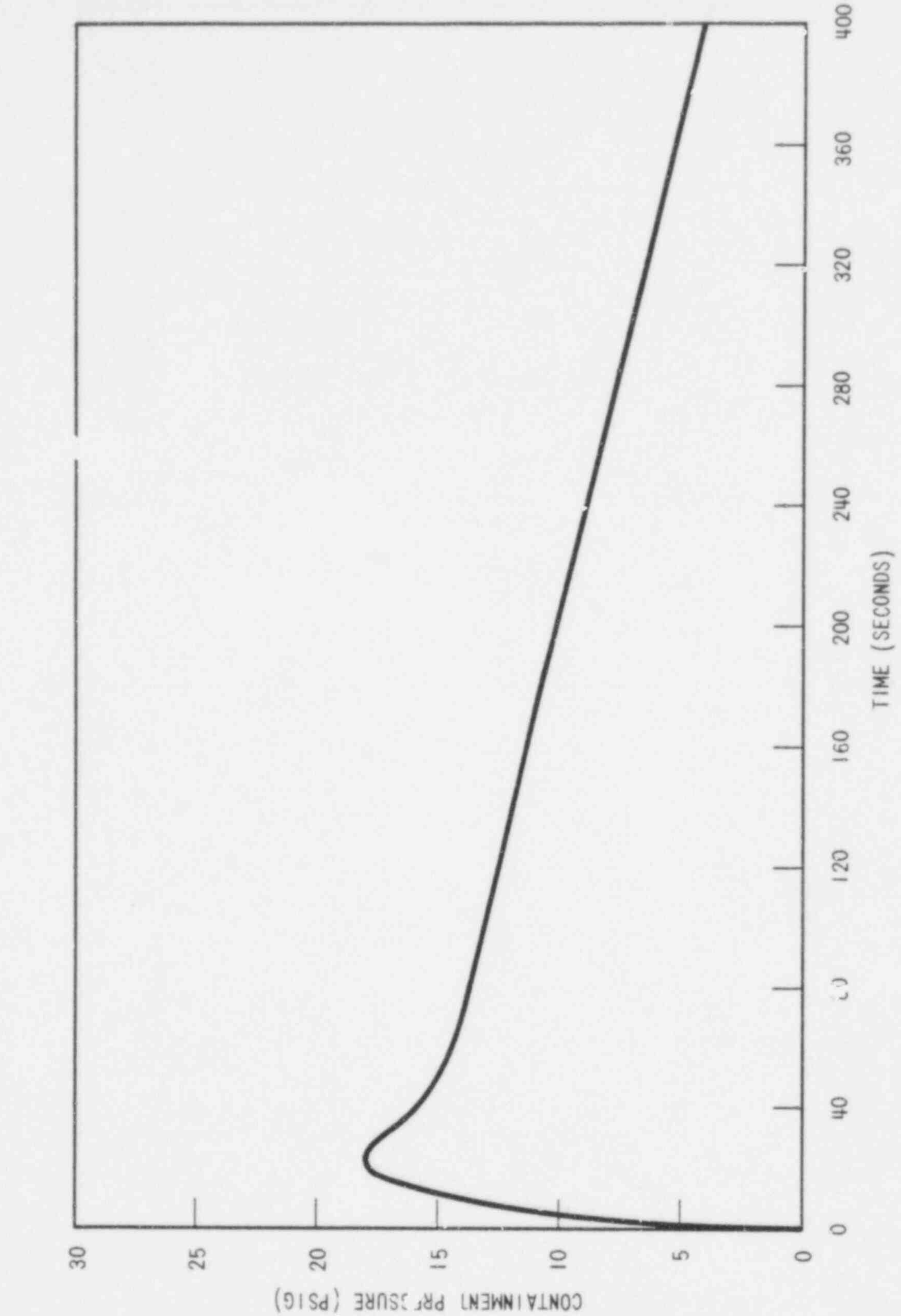
614 111

WCAP - 9500

Figure 15.6-32.

Core Power Transient
DECLG (C_D = 0.4)

14,395-337

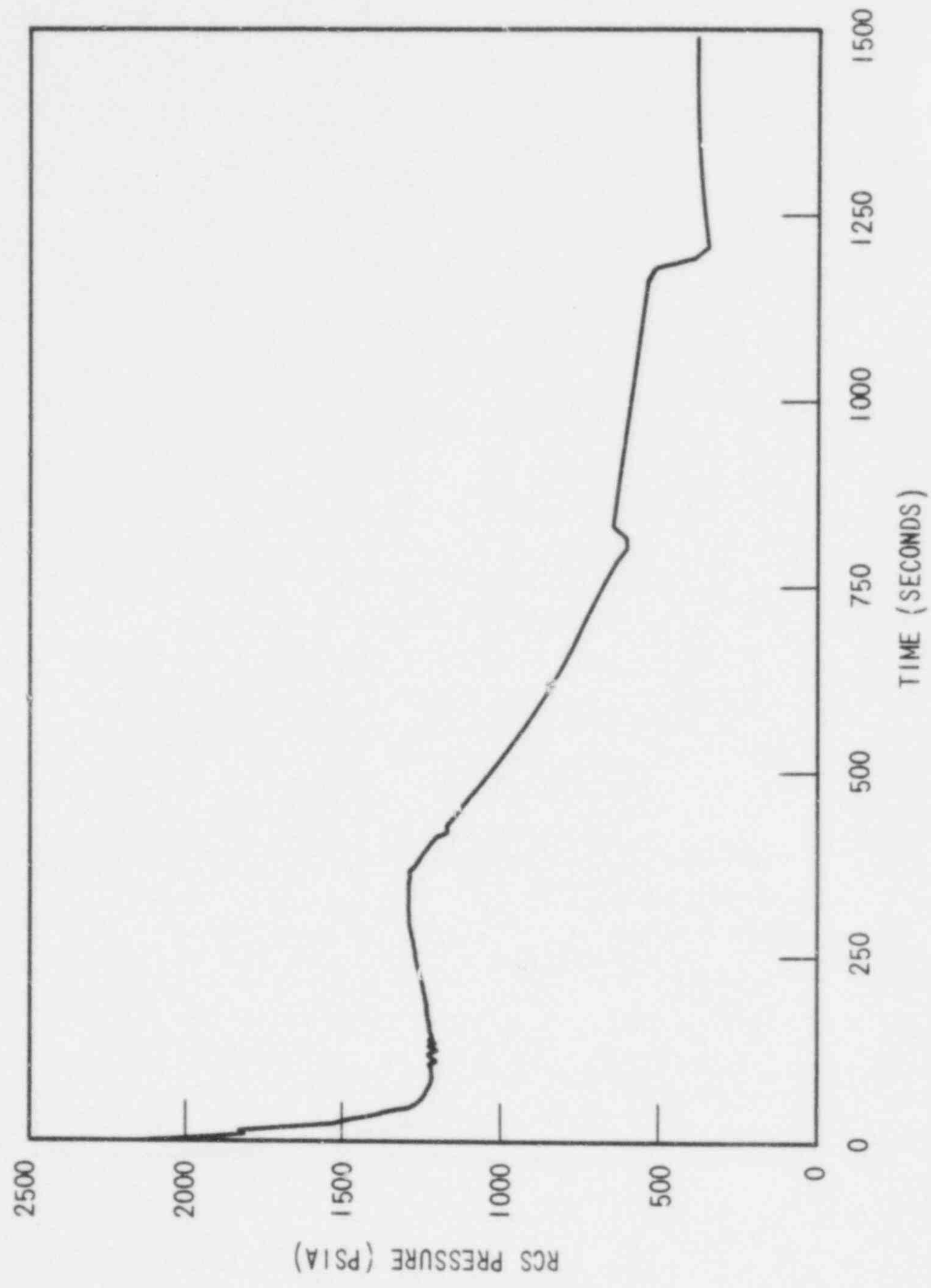


WCAP - 9500

Figure 15.6-33.

Containment Pressure DECLG (C_D = 0.4)

614 112

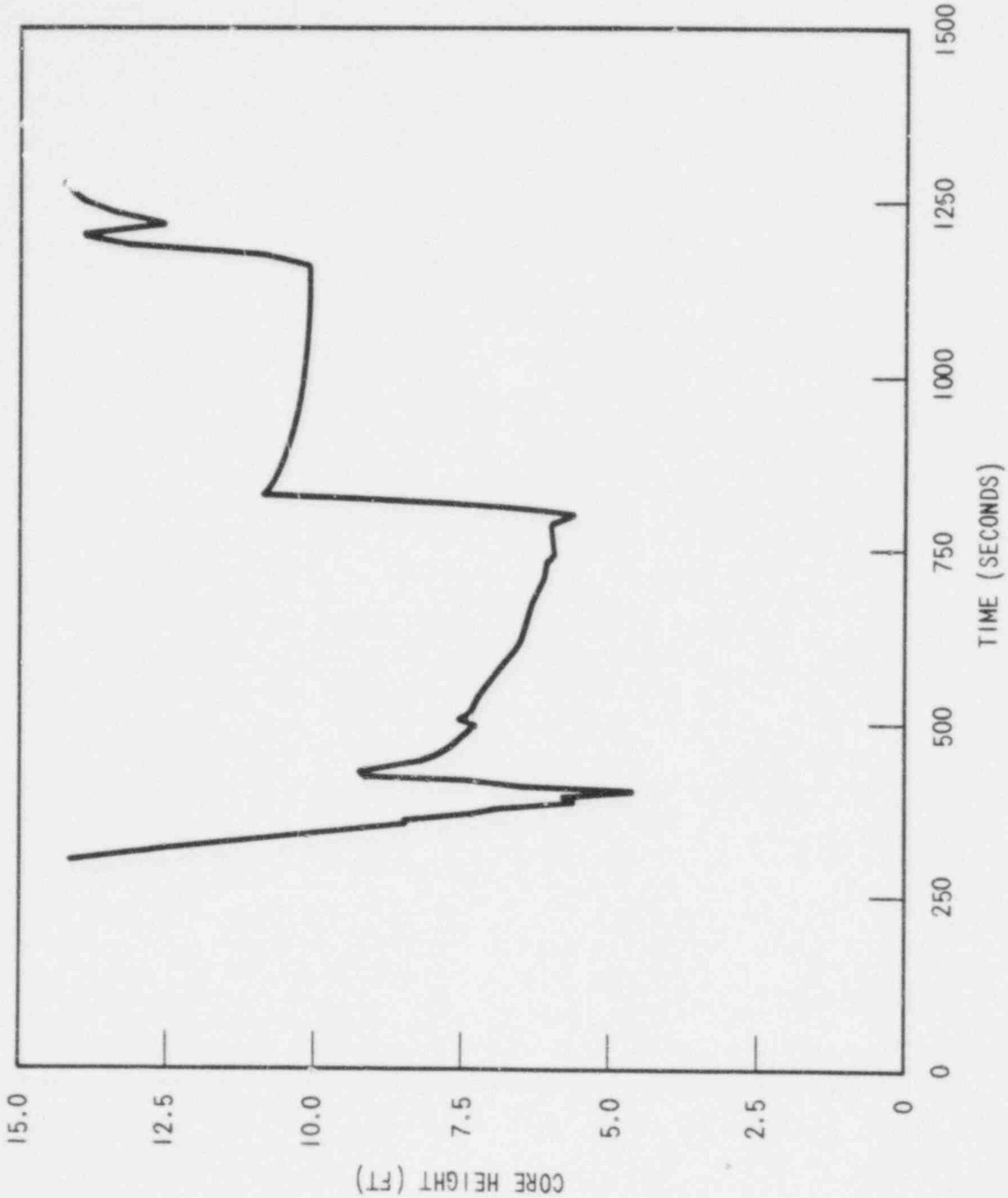


4 113

WCAP-9500

Figure 15.6-34.

RCS Depressurization Transient
(4 Inch Break)

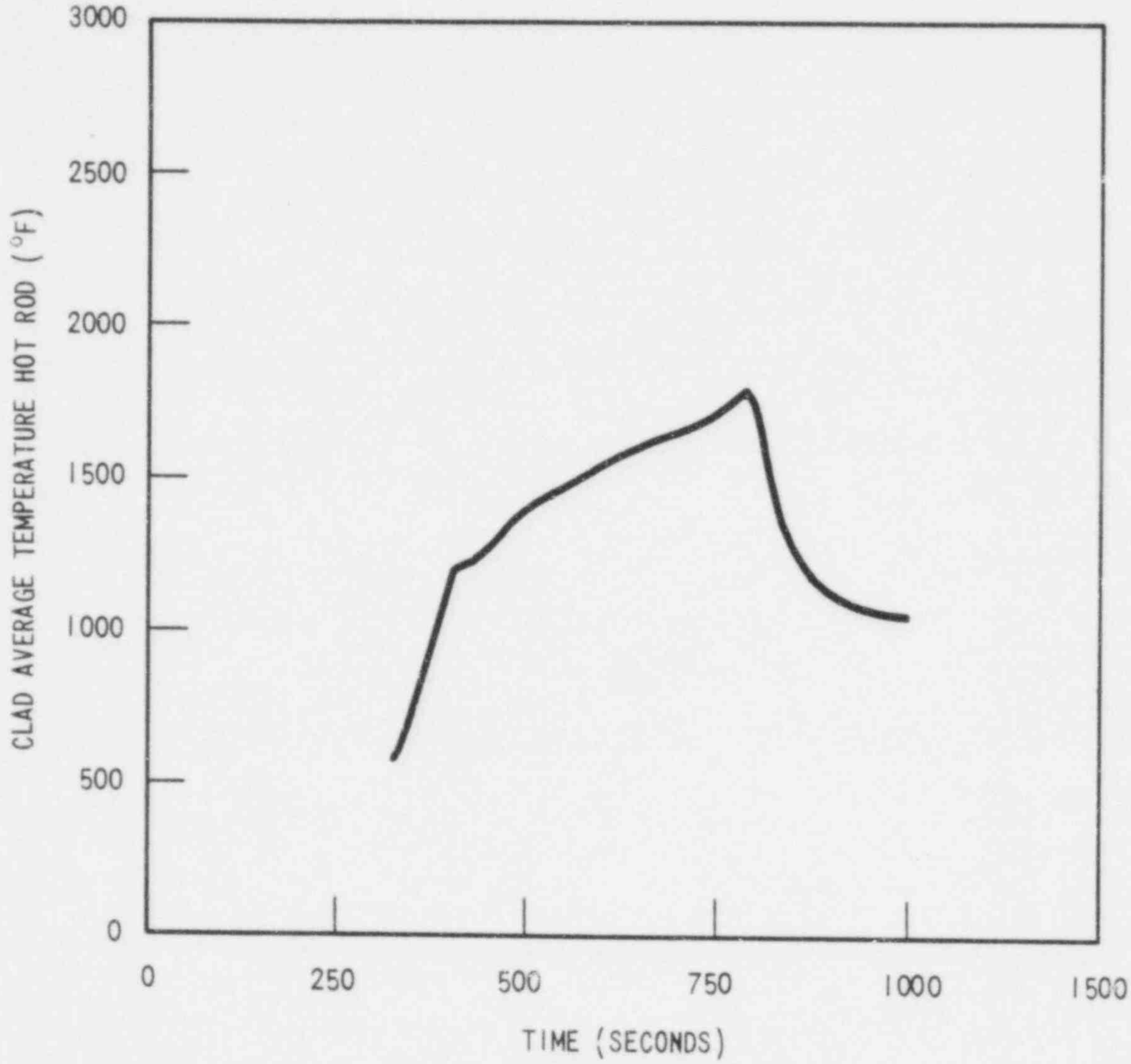


614 114

WCAP - 9500

Figure 15.6-35.

Core Mixture Height (4 Inch Break)

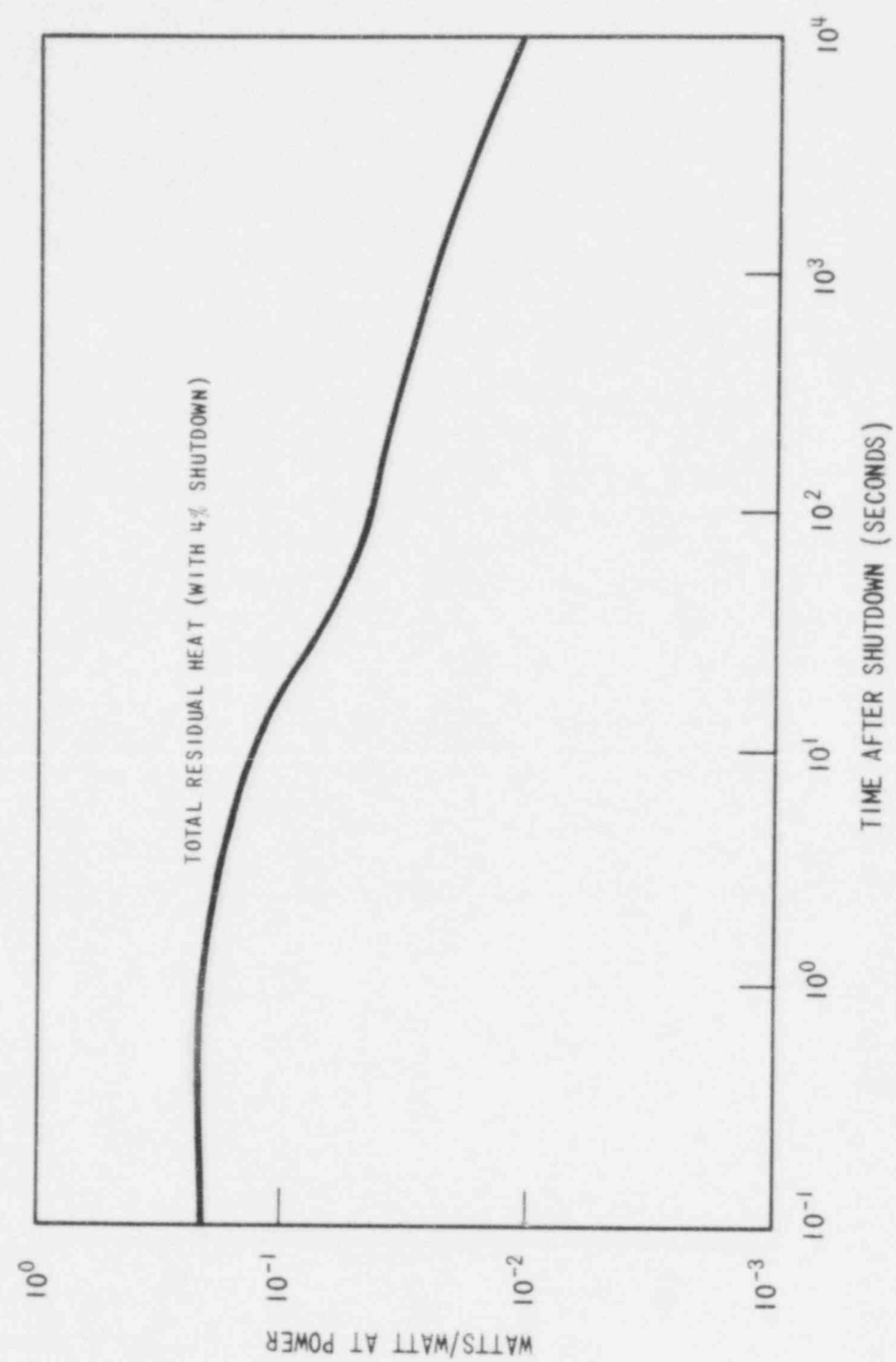


614 115

WCAP-9500

Figure 15.6-36.

Clad Temperature Transient
(4 Inch Break)



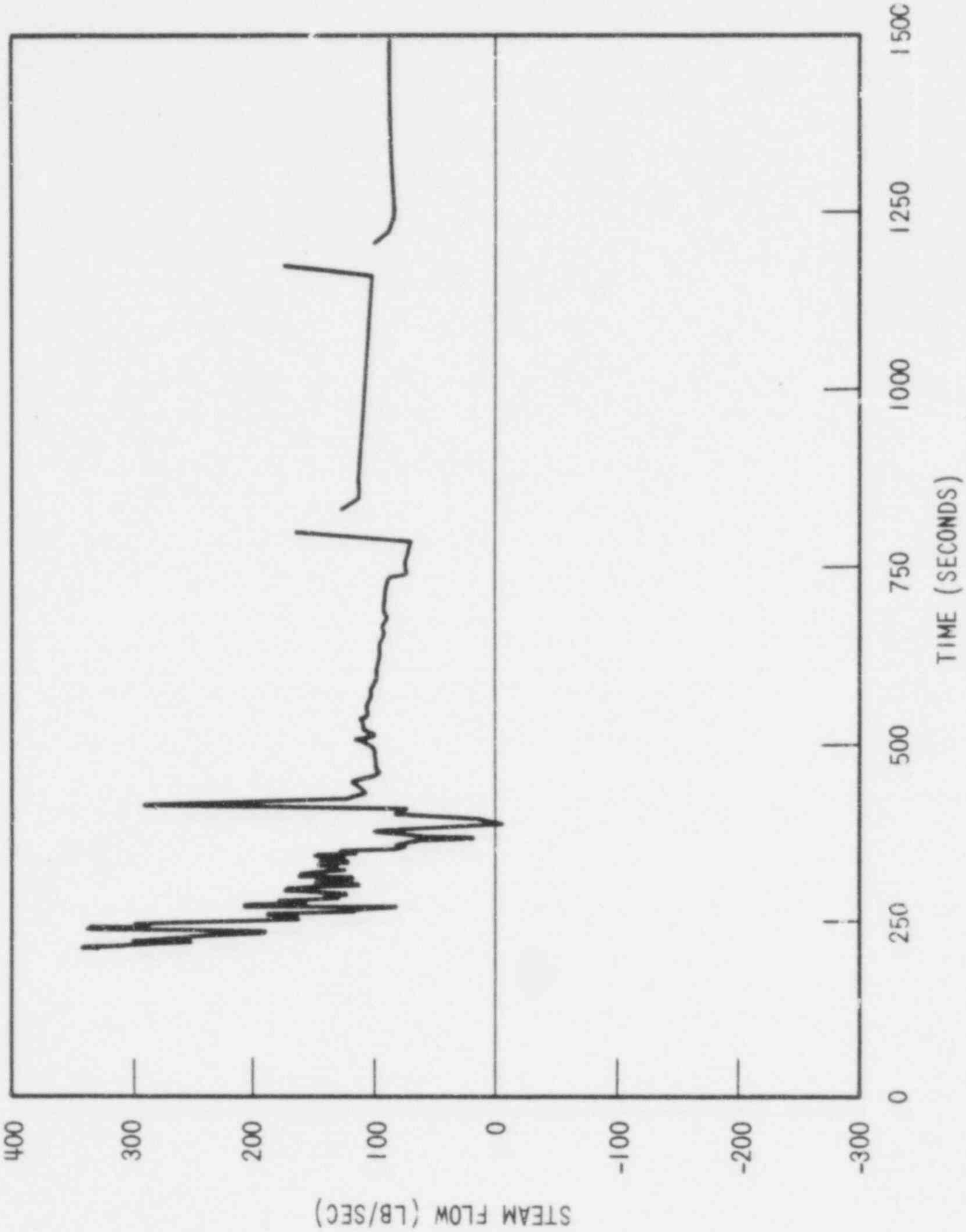
WCAP – 9500

Figure 15.6-37.

Core Power After Reactor Trip

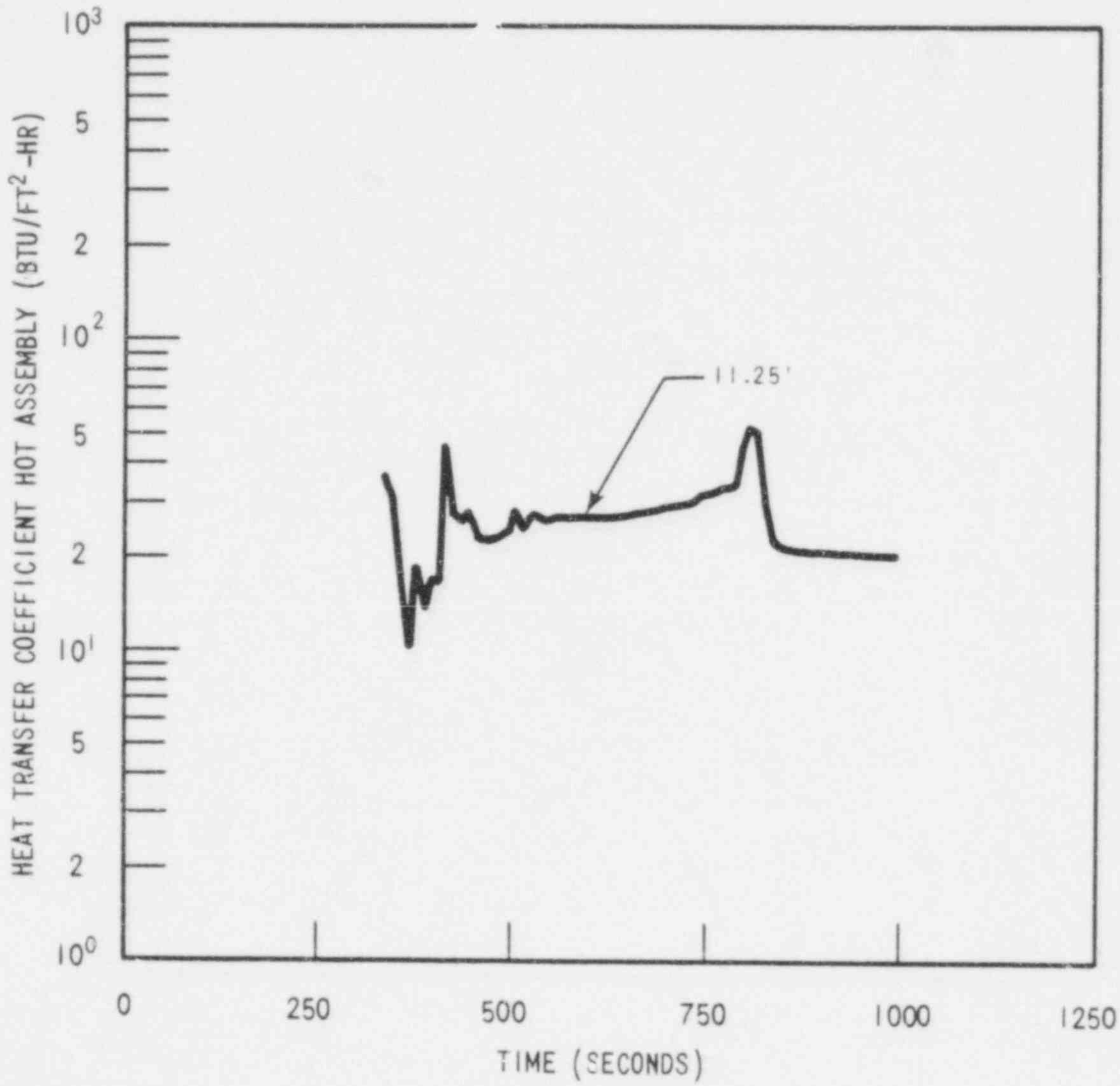
614 116

14,395-342



614 117

WCAP – 9500
Figure 15.6-38. Steam Flow (4 Inch Break)

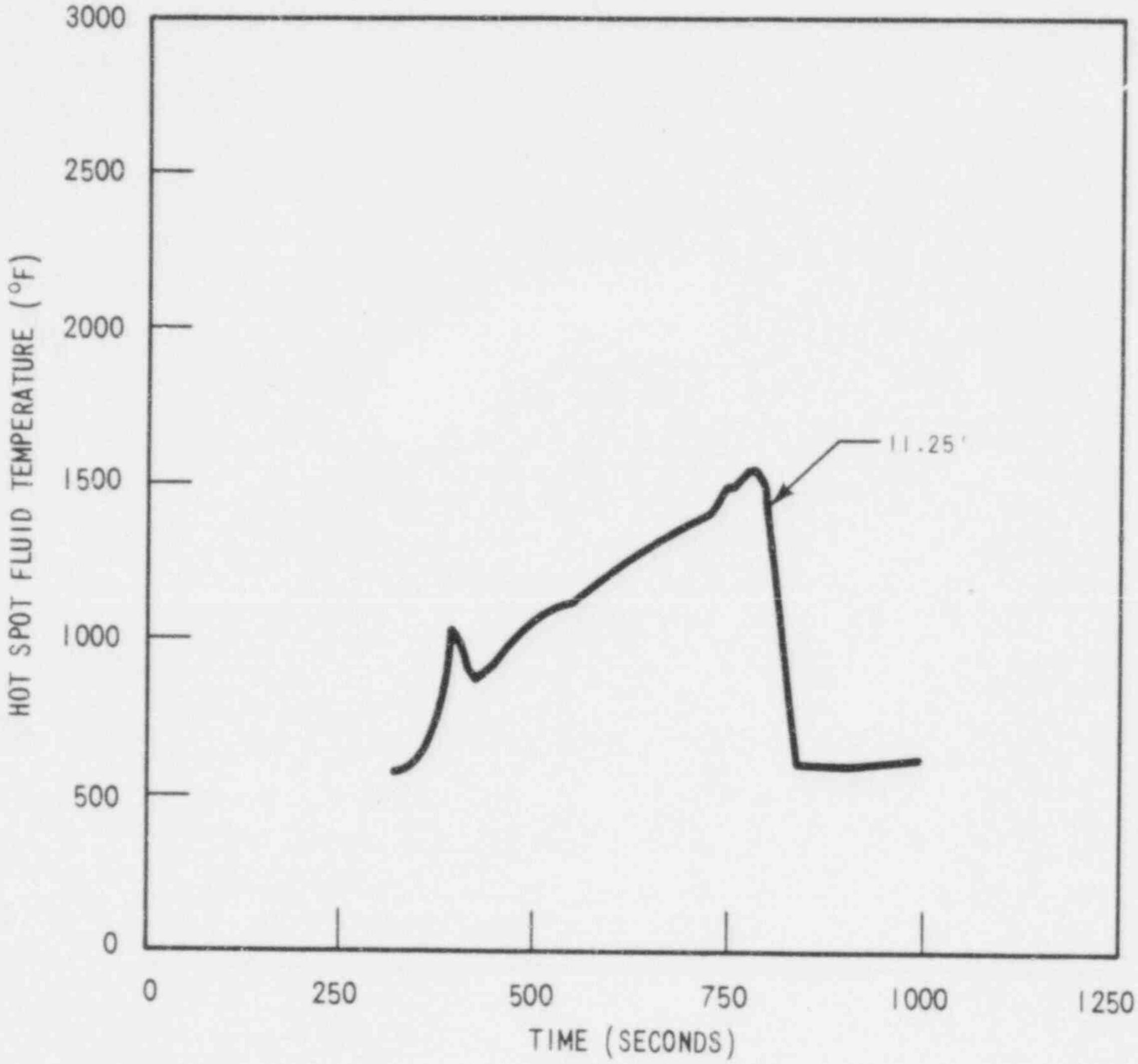


614 1/8

WCAP – 9500

Figure 15.6-39.

Rod Film Heat Transfer Coefficient (4 Inch Break)



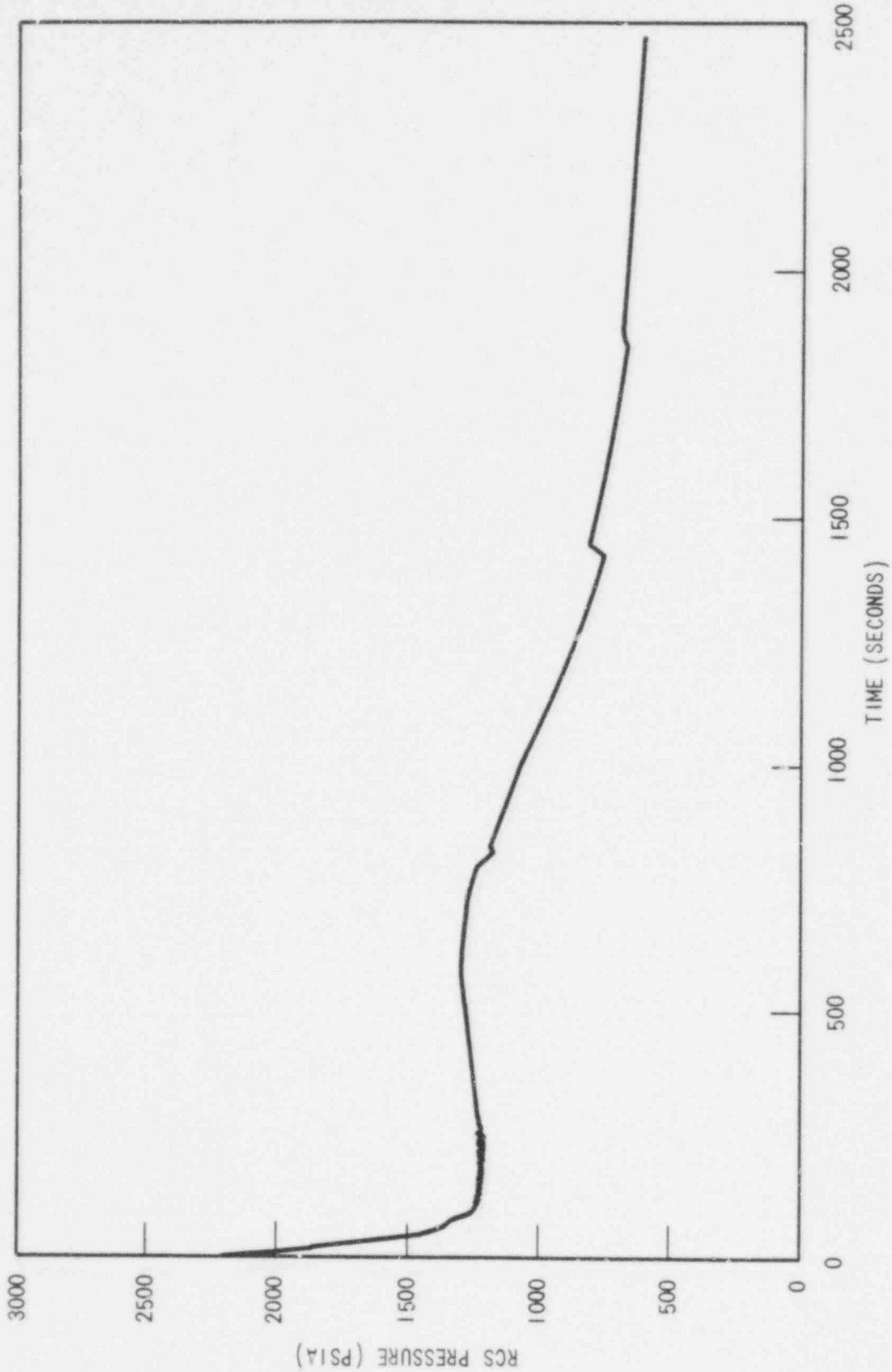
WCAP – 9500

Figure 15.6-40.

Hot Spot Fluid Temperature
(4 Inch Break)

614 .19

14,395-345



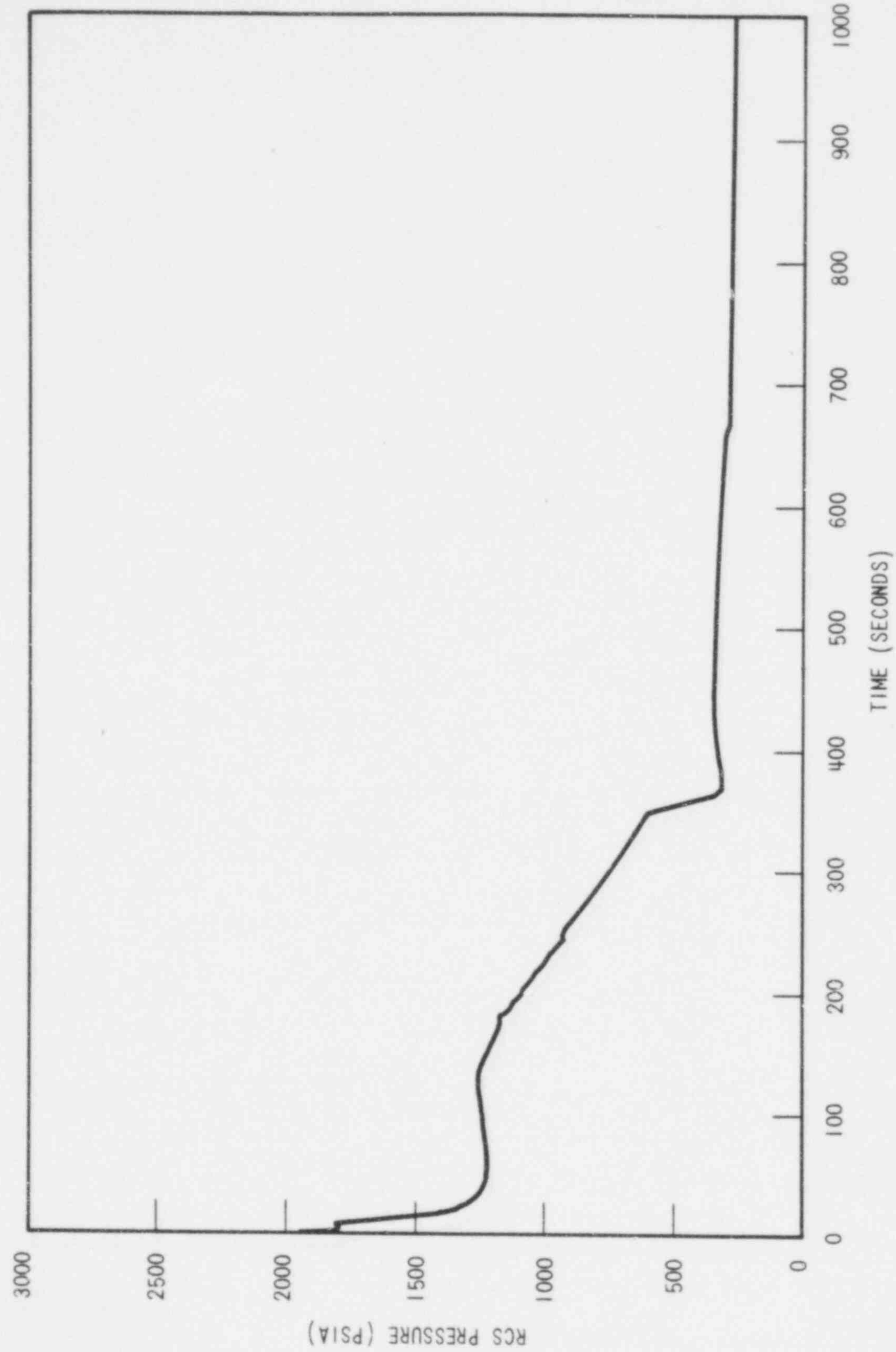
WCAP – 9500

Fic ure 15.6-4l.

RCS Depressurization Transient
(3 Inch Break)

614 120

14,395-346



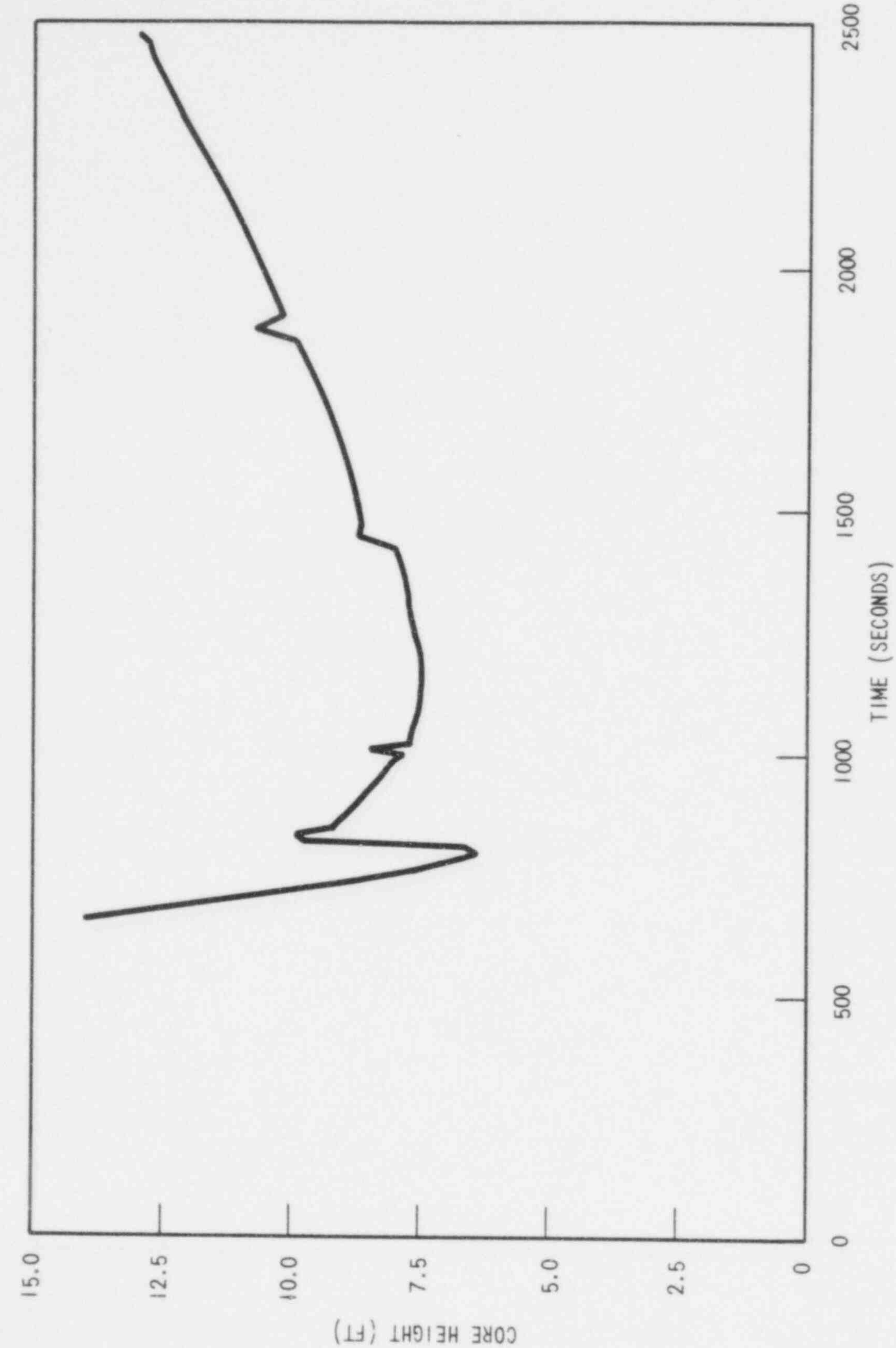
614 121

WCAP - 9500

Figure 15.6-42.

RCS Depressurization Transient
(6 Inch Break)

14,395-347



WCAP – 9500

Figure 15.6-43.

Core Mixture Height (3 Inch Break)

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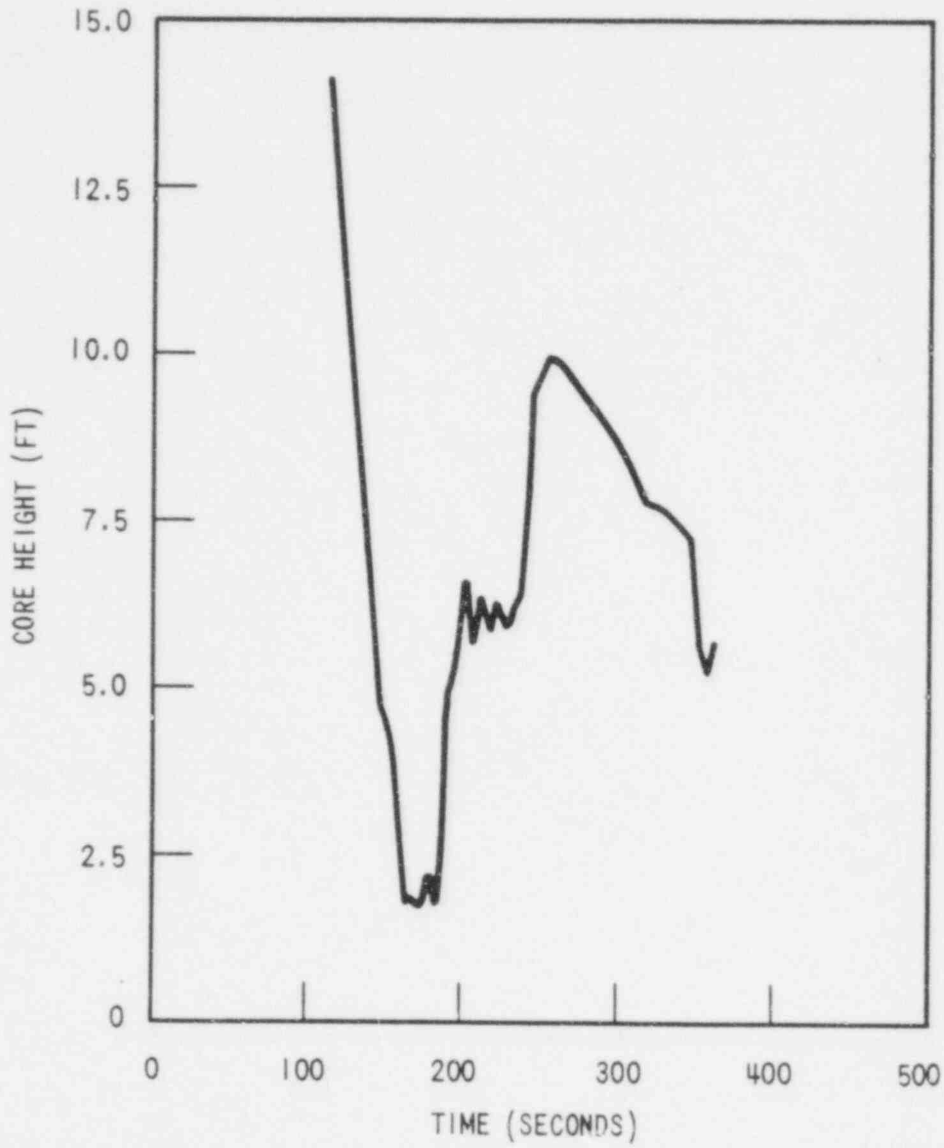

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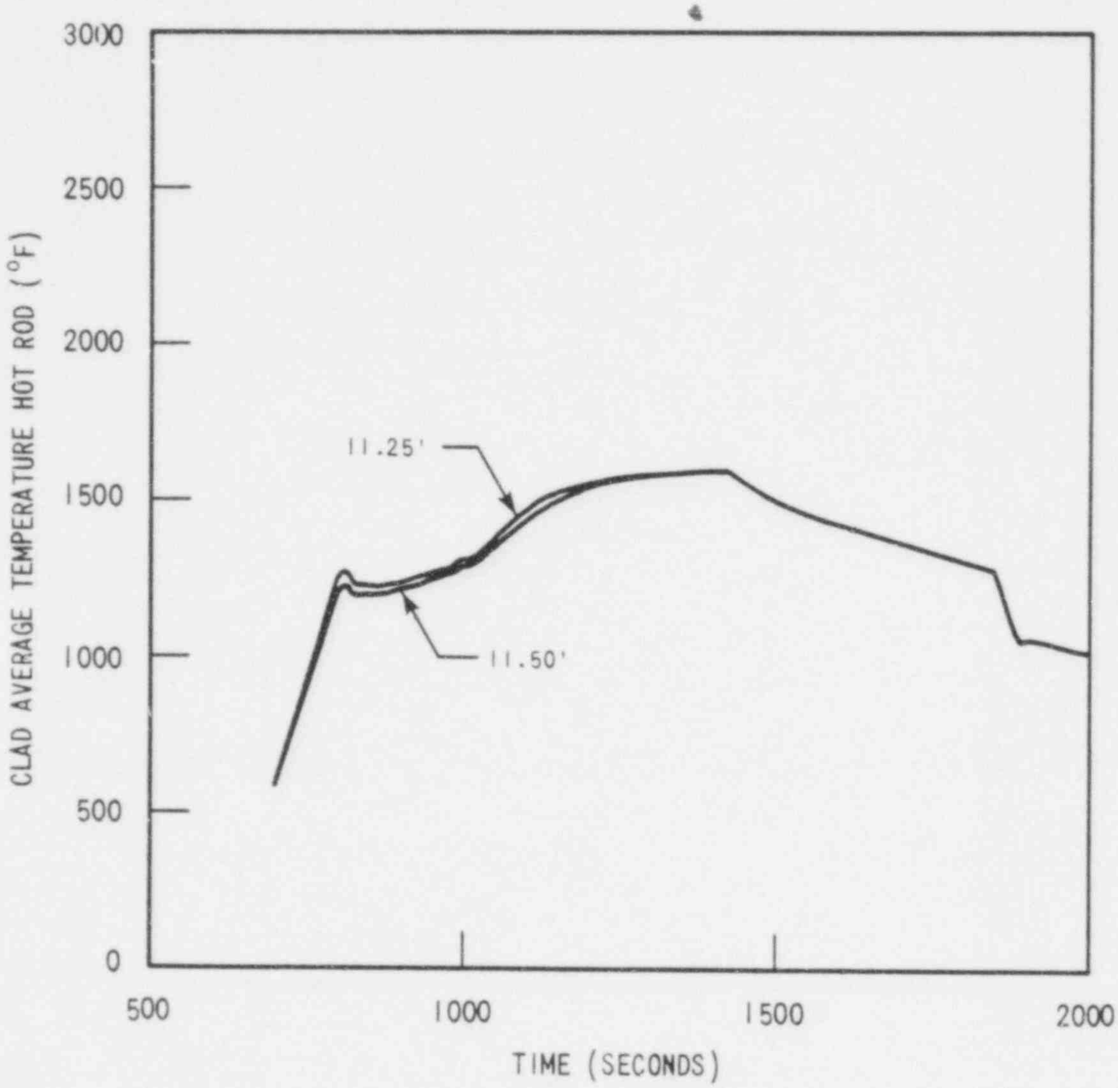
WCAP - 9500

Figure 15.6-44.

Core Mixture Height (6 Inch Break)



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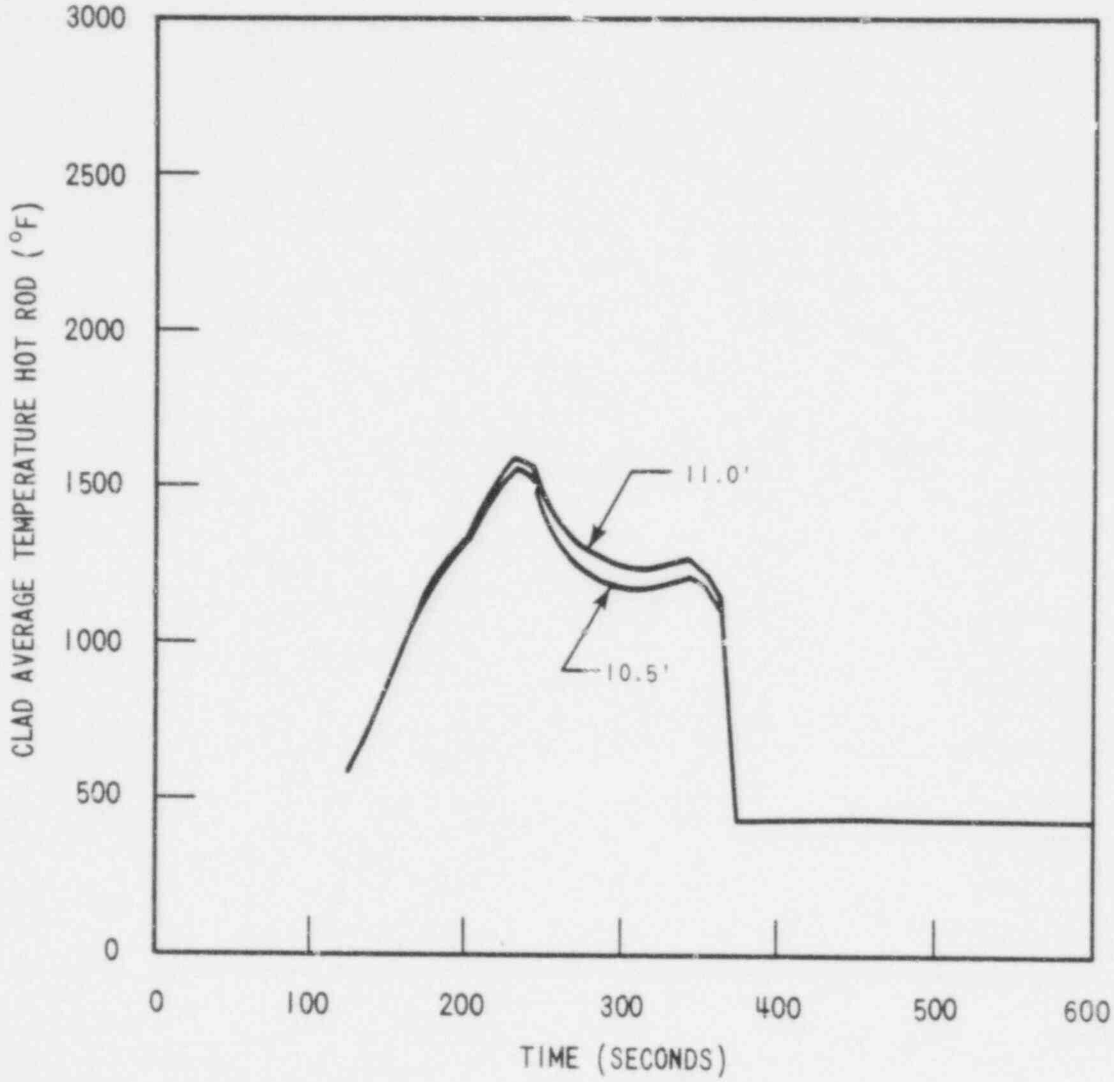
WCAP - 9500

Figure 15.6-45.

Clad Temperature Transient
(3 Inch Break)

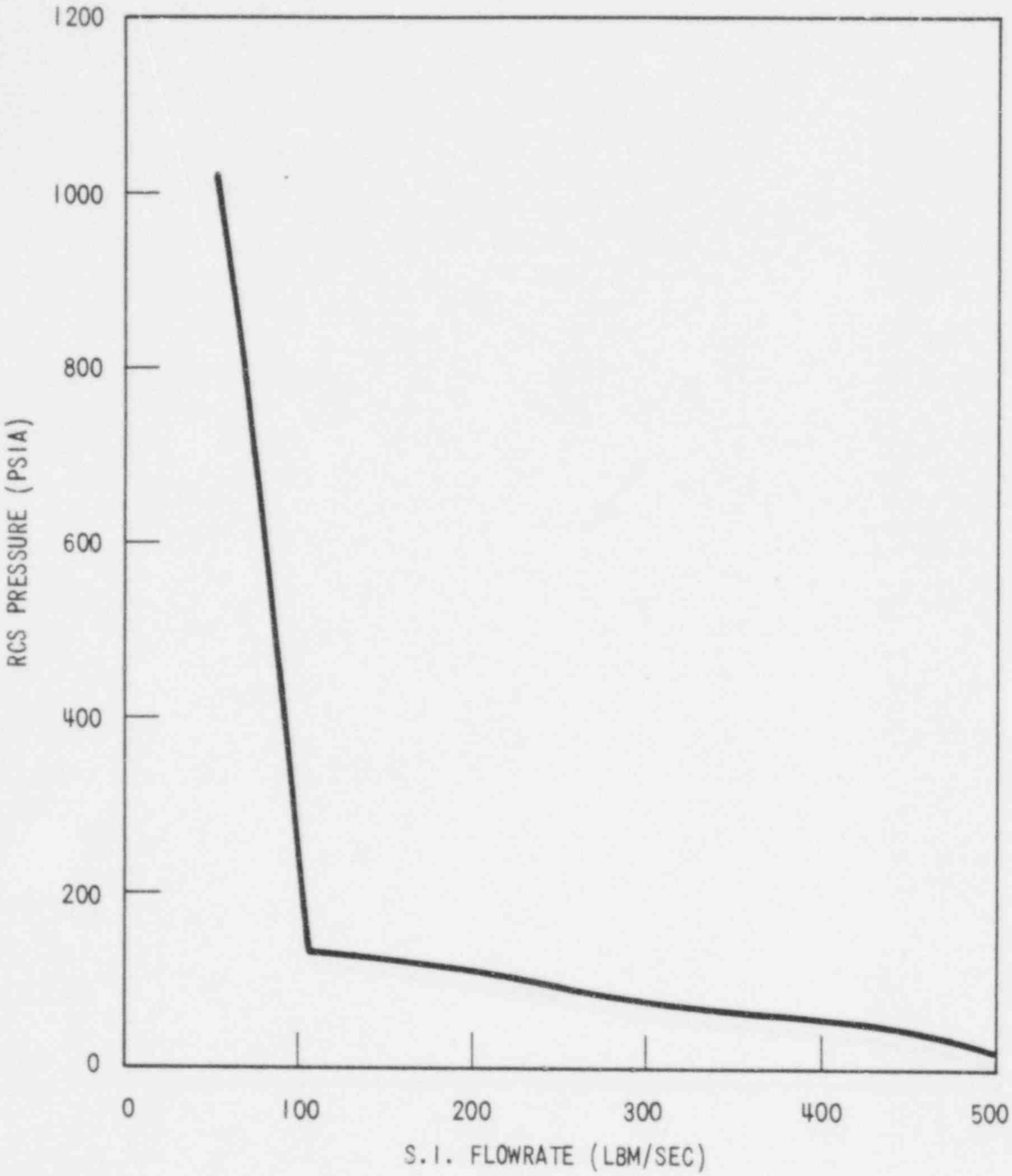


614 125

WCAP - 9500

Figure 15.6-46.

Clad Temperature Transient
(6 Inch Break)



WCAP - 9500

Figure 15.6-47.

Safety Injection Flowrate

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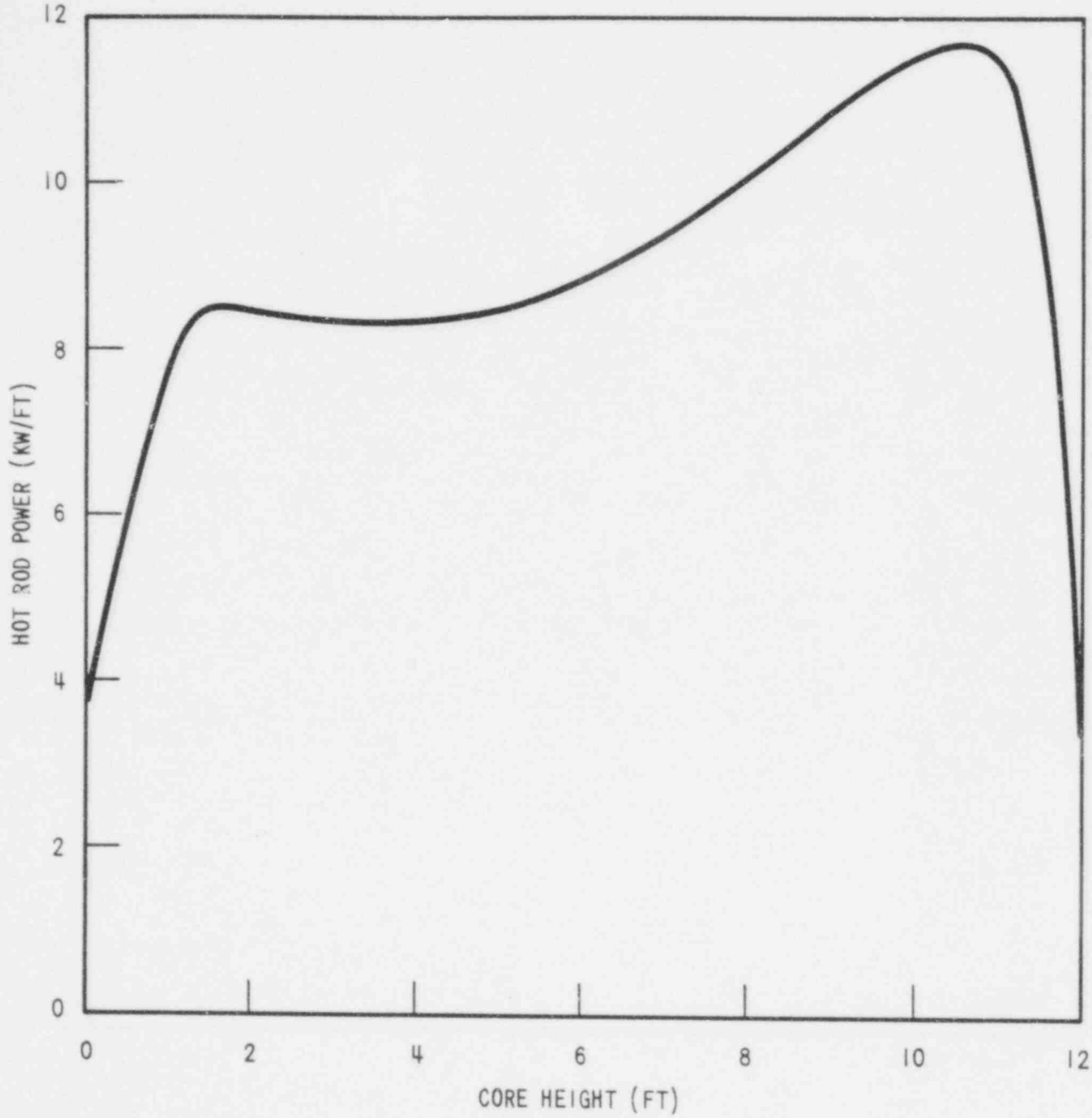

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WCAP - 9500

Figure 15.6-48.

Small Break Power Distribution

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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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 TO BE USED IN RADIOLOGICAL CONSEQUENCES OF THE 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	≤ 1200 psia	≤ 1200 psia
Minimum water depth between top of damaged fuel rods and pool surface	≥ 23 feet	≥ 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		

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TABLE 15.7-3 (Continued)

PARAMETERS TO BE USED IN RADIOLOGICAL CONSEQUENCES OF THE 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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