Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

AW-95-838

June 2, 1995

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: MR. T. R. QUAY

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: PRELIMINARY MARKUPS OF AP600 SSAR CHAPTER 15 (ACCIDENT ANALYSES)

Dear Mr. Quay:

The application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10CFR Section 2.790, Affidavit AW-95-838 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-95-838 and should be addressed to the undersigned.

Very truly yours,

N. J. Liparulo, Manager
Nuclear Safety Regulatory And Licensing Activities

/nja

cc: Kevin Bohrer NRC 12H5

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AW-95-838

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Brian A. McIntyre, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

Sworn to and subscribed
before me this 5 day
of June, 1995

Rose Marie Payne

Notary Public

Notarial Seal
Rose Marie Payne, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Nov. 4, 1996
Member, Pennsylvania Association of Notaries

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(1) I am Manager, Advanced Plant Safety and Licensing, in the Advanced Technology Business Area, of the Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems Business Unit in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) Enclosed is Letter NTD-NRC-95-4480, June 2, 1995 being transmitted by Westinghouse Electric Corporation (W) letter and Application for Withholding Proprietary Information from Public Disclosure, N. J. Liparulo (W), to Mr. T. R. Quay, Office of NRR. The proprietary information as submitted for use by Westinghouse Electric Corporation is in response to questions concerning the AP600 plant and the associated design certification application and is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of licensing advanced nuclear power plant designs.

This information is part of that which will enable Westinghouse to:

(a) Demonstrate the design and safety of the AP600 Passive Safety Systems.

(b) Establish applicable verification testing methods.

(c) Design Advanced Nuclear Power Plants that meet NRC requirements.

(d) Establish technical and licensing approaches for the AP600 that will ultimately result in a certified design.

(e) Assist customers in obtaining NRC approval for future plants.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for advanced plant licenses.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar advanced nuclear power designs and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

CHAPTER 15.0

ACCIDENT ANALYSES

15.0.1 Classification of Plant Conditions

The ANSI 18-2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk and those extreme situations having the potential for the greatest risk should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning are assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle. ~~That is, only seismic Category 1, Class 1E, and qualified equipment, instrumentation, and components are used in the mitigation of the consequences of faulted Conditions II, III, and IV events.~~ The evaluation models and parameters for the accident analysis radiological consequences are discussed in Appendix 15A.

15.0.1.1 Condition I: Normal Operation and Operational Transients

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

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

A typical list of Condition I events follows.

Steady-State and Shutdown Operations

See Table 1.1-1 of Chapter 16.

Global

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant technical specifications are considered in conjunction with other operational modes. These deviations include the following:

- Operation with components or systems out of service [such as an inoperable rod cluster control assembly (RCCA)]

- Leakage from fuel with limited clad defects

- Excessive radioactivity in the reactor coolant:
  - Fission products
  - Corrosion products
  - Tritium

- Operation with steam generator tube leaks

- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to ±10 percent)
- Ramp load changes (up to 5 percent/min)
- Load rejection up to and including design full load rejection transient.

15.0.1.2 Condition II: Faults of Moderate Frequency

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

- Feedwater system malfunctions that result in a decrease in feedwater temperature (See Subsection 15.1.1)

- Feedwater system malfunctions that result in an increase in feedwater flow (See Subsection 15.1.2)

- Excessive increase in secondary steam flow (See Subsection 15.1.3)

- Inadvertent opening of a steam generator relief or safety valve (See Subsection 15.1.4)

- Inadvertent operation of the passive residual heat removal heat exchanger ~~system~~ (See Subsection 15.1.6)

Global

- Loss of external electrical load (See Subsection 15.2.2)
- Turbine trip (See Subsection 15.2.3)
- Inadvertent closure of main steam isolation valves (See Subsection 15.2.4)
- Loss of condenser vacuum and other events resulting in turbine trip (See Subsection 15.2.5)
- Loss of ac power to the station auxiliaries (See Subsection 15.2.6)
- Loss of normal feedwater flow (See Subsection 15.2.7)
- Partial loss of forced reactor coolant flow (See Subsection 15.3.1)
- Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (See Subsection 15.4.1)
- Uncontrolled rod cluster control assembly bank withdrawal at power (See Subsection 15.4.2)
- Rod cluster control assembly misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (See Subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (See Subsection 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (See Subsection 15.4.6)
- Inadvertent operation of the passive ~~emergency~~ core cooling system during power operation (See Subsection 15.5.1).
- Chemical and volume control system malfunction that increased reactor coolant inventory (See Subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (See Subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (See Subsection 15.6.2)~~.~~

15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults which may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the

exclusion area boundary, in accordance with the guidelines of 10 CFR 100. By definition, Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (See Subsection 15.1.5)
- Complete loss of forced reactor coolant flow (See Subsection 15.3.2)
- Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (See Subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (See Subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (See Subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (See Subsection 15.6.5)
- Gas waste management system leak or failure (See Subsection 15.7.1)
- Liquid waste management system leak or failure (Subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (See Subsection 15.7.3)
- Spent fuel cask drop accidents (See Subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (See Subsection 15.1.5)
- Feedwater system pipe break (See Subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (See Subsection 15.3.3)

- Reactor coolant pump shaft break (See Subsection 15.3.4)
- Spectrum of rod cluster control assembly ejection accidents (See Subsection 15.4.8)
- Steam generator tube rupture (See Subsection 15.6.3)
- Loss-of-coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (See Subsection 15.6.5)
- Design basis fuel handling accidents (See Subsection 15.7.4)

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15.0.2 Optimization of Control Systems

A control system setpoint study is performed prior to plant operation to simulate performance of the primary plant control systems and overall plant performance. In this study, emphasis is placed on the development of the overall plant control systems that automatically maintains conditions in the plant within the allowed operating window and with optimum control system response and stability over the entire range of anticipated plant operating conditions. The control system setpoints are developed using the nominal protection system setpoints which are implemented in the plant. Where appropriate (such as in margin to reactor trip analyses), instrumentation errors are considered and are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and control system setpoint study in combination show that the plant can be operated and meet both safety and operability requirements throughout the core life and for various levels of power operation.

The control system setpoint study is comprised of analyses of the following control systems: power control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator level, pressurizer pressure, and pressurizer level.

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15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table ~~15-1~~ 15.0-1 lists the principal power rating values assumed in the analyses performed. The thermal power output includes the effective thermal power generated by the reactor coolant pumps.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table ~~15-3~~ 15.0-3.

15.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are

determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) safety analysis limit values (See Subsection 4.4.1.1.2), as described in Reference 2. This procedure is known as the Revised Thermal Design Procedure (RTDP), and is discussed more fully in Section 4.4.

For accidents that are not departure from nucleate boiling limited, or for which the revised thermal design procedure is not employed, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following conservative steady-state errors are assumed in the analysis:

Core power	±2 percent allowance for calorimetric error
Average reactor coolant system (RCS) temperature	±~~4.5~~ 6.5°F allowance for ~~controller deadband and~~ controller deadband and measurement errors
Pressurizer pressure	±50 psi allowance for steady-state fluctuations and measurement errors

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial departure from nucleate boiling ratio unless otherwise stated in the sections describing the specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.

For transients that may be departure from nucleate boiling (DNB) limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure ~~15.3-1~~ 15.0.3-1. Transients that may be departure from nucleate boiling limited are assumed to begin with an $F_{\Delta H}$ consistent with the initial power level defined in the technical specifications.

The axial power shape used in the departure from nucleate boiling calculation is the 1.55 chopped cosine, as discussed in Subsection 4.4.4.3, for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric rod cluster control assembly (RCCA) conditions.

The radial and axial power distributions just described are input to the THINC code as described in Subsection 4.4.4.5.

For transients which may be overpower limited, the total peaking factor (F_q) is important. Transients that may be overpower limited are assumed to begin with plant conditions including power distributions, which are consistent with reactor operation as defined in the technical specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Subsection 4.4.4.

For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly bank withdrawal from subcritical or lower power startup and rod cluster control assembly ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

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15.0.4 Reactivity Coefficients Assumed in the Accident Analysis

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

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values. The values used are given in Figure ~~15.4-1~~ 15.0.4-1, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

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15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies (RCCAs) as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to or simultaneous with RCCA insertion, a time of 1.8 seconds is used for insertion time to dashpot entry.

In Figure ~~15.5-1~~ 15.0.5-1, the curve labeled "complete loss of flow transients" shows the RCCA position versus time normalized to 1.8 seconds assumed in accident analyses where all reactor coolant pumps are coasting down. In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.4 seconds. The RCCA position versus time normalized to 2.4 seconds is also shown in Figure ~~15.5-1~~ 15.0.5-1.

The use of such a long insertion time provides conservative results for accidents and is intended to apply to all types of rod cluster control assemblies which may be used throughout plant life. Drop time testing requirements are specified in the technical specifications.

Figure ~~15.5-2~~ 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the 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 ~~15.5-2~~ 15.0.5-2 in that it is based on a skewed flux distribution, which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than that shown in the curve, 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 ~~15.5-3~~ 15.0.5-3. The curves shown in this figure were obtained from Figures ~~15.5-1~~ 15.0.5-1 and ~~15.5-2~~ 15.0.5-2. A total negative reactivity insertion following a trip of four percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure ~~15.5-3~~ 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure ~~15.5-1~~ 15.0.5-1 is used as code input.

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

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table ~~15-4~~ 15.0-4. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure ~~15.3-1~~ 15.0.3-1.

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. 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 are determined periodically in accordance with the plant technical specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

The instrumentation uncertainties and calorimetric uncertainties used in establishing the power range high neutron flux setpoint are presented in Table ~~15-5~~ 15.0-5

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

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. Installed plant instrumentation is used for these measurements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features that minimize the probability and effects of fires and explosions.

Chapter 17 discusses the quality assurance program that is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, provide confidence that the normally operating systems and components listed in Table ~~15-6~~ 15.0-6 are 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 (Reference 1) is utilized. The design of safety related systems (including protection systems) is consistent with IEEE Standard 379-1988 and Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in Section 1.9.1.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

15.0.9 Fission Product Inventories

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the reactor core if the accident involves fuel damage. The radiological consequences analyses utilize the conservative design basis source terms identified in Appendix 15A.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-of-coolant accident (LOCA) according to the requirements of 10 CFR 50.46, as described in References 3 and 4. The small break LOCA events utilize 10 CFR 50 Appendix K which assumes 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 an equilibrium cycle.

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

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

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; one half second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect on the hot rod is a reduction of 10 percent of the gamma ray contribution or three percent of the total heat. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining two percent is absorbed by water, thimbles, sleeves, and grids. Combining the three percent total heat reduction from gamma redistribution with this two percent absorption produce as the net effect a factor of 0.95, which exceeds the actual heat production in the hot rod. The actual hot rod heat generation is computed during the AP600 large break LOCA transient as a function of core fluid conditions.

15.0.11 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes--in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (See Section 15.0.6) are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-2.

15.0.11.1 FACTRAN Computer Code

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.
- Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.
- The necessary calculations to handle post-departure from nucleate boiling transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 5.

15.0.11.2 LOFTRAN Computer Code

The LOFTRAN (Reference 6) 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 generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, 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 and safety monitoring system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

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

LOFTRAN also has the capability of calculating the transient value of departure from nucleate boiling ratio (DNBR) based on the input from the core limits illustrated in Figure 15.0.3-1. The core limits represent the minimum value of departure from the nucleate boiling ratio as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 6.

heat exchanger — and safety monitoring

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) ~~system~~, core makeup tanks (CMT) and associated protection system actuation logic. A discussion of these models and additional validation is presented in Appendix 15B. and Reference 10

LOFTTR2 (Reference 8) is a modified version of LOFTRAN with a more realistic break flow model, a two-region SG secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event. LOFTTR2 is further discussed in Reference 8.

heat exchanger

The LOFTTR2 code is modified to allow the simulation of the passive residual heat removal ~~(PRHR) system~~, core makeup tanks (CMT) and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code. A discussion of these models is presented in Appendix 15B. and Reference 10

15.0.11.3 TWINKLE Computer Code

The TWINKLE (Reference 7) program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.
TWINKLE is further described in Reference 7.

15.0.11.4 THINC Computer Code

The THINC code is described in ~~S~~subsection 4.4.4.5.

15.0.11.5 WESTAR ~~Comput~~ Computer Code

The WESTAR code is described in subsection 4.4.

11P*

15.0.12 Component Failures

15.0.12.1 Active Failures

SECY-77-439 (Reference 9) provides a description of active failures. An active failure results in the inability of a component to perform its intended function.

An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a remotely-operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure. Failure of a manual valve to change position under local operator action is included.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a steam generator or pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment such as pumps, fans, and rotating mechanical components, an active failure is the failure of the component to start or to remain operating.

For electrical equipment, the loss of power, such as the loss of offsite power or the loss of a diesel-generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component is considered as an active failure for active components in safety-related passive systems. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety-related equipment and does not include throught-process errors or similar errors that could potentially lead to common cause or multiple errors.

15.0.12.2 Passive Failures

SECY-77-439 also provides a description of passive failures. A passive failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. A passive failure is applied to fluid systems and consists of a breach in the fluid system boundary. Examples include cracking of pipes, sprung flanges, or valve packing leaks.

Passive failures are not assumed to occur until 24 hours after the start of the event. Consequential effects of a pipe leak such as flooding, jet impingement, and failure of a valve with a packing leak must be considered.

Where piping is significantly overdesigned or installed in a system where the pressure and temperature conditions are relatively low, passive leakage is not considered a credible failure mechanism. Line blockage is also not considered as a passive failure mechanism.

15.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists), as described in Section 3.1, of safety-related equipment, is identified in each analysis description. The consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table ~~15-7~~ 15.0-7

15.0.13 Operator Actions

For events where the PRHR ~~system~~ heat exchanger is actuated, the plant automatically cools down to the safe shutdown conditions. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.0.~~14~~ 15 References

1. American National Standards Institute N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1972.

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

3. Lee, N., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081 (Nonproprietary), August 1985.

4. Bajorek, S. M., "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P (Proprietary), August 1991.

15.0.14 Combined License Information

This section has no requirement for additional information to be provided in support of the combined license application.

5. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1972.

6. Burnett, T. W. T., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

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

8. Lewis, R. N., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Nonproprietary), August 1985.

9. Case, E. G., "Single Failure Criterion," SECY-77-439, August 17, 1977.

10. Carlin, E. L., "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, November 1994.

Table 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Thermal power output (MWt)	1940
Effective thermal power generated by the reactor coolant pumps (MWt)	7
Core thermal power (MWt)	1933

Table 15.0-2 (Sheet 1 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.1	Increase in heat removal from the primary system					
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN	0.374	--	Upper curve of Figure 15.0.4-1	0 and 1940
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.374	--	Upper & lower curves of Figure 15.0.4-1	1940
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN	Function of moderator density (see figure 15.1.4-1)	--	See subsection 15.1.4	0 (subcritical)
	Steam system piping failure	LOFTRAN, THINC	Function of moderator density (see figure 15.1.4-1)	--	See subsection 15.1.5	0 (subcritical)
	Inadvertent operation of the PRHR	LOFTRAN	~~0.374~~ See subsection 15.1.6.2.1	--	Upper curve of Figure 15.0.4-1	1940

Table 15.0-2 (Sheet 2 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Sect-tion	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	Initial Thermal Power Output Assumed (MWt)
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN	0.0 and 0.374	--	Lower and upper curves of Figure 15.0.4-1	1940
	Loss of nonemergency ac power to the station auxiliaries	LOFTRAN	0.0	--	Lower curve of Figure 15.0.4-1	1978.8[a]
	Loss of normal feedwater flow	LOFTRAN	0.0	--	Lower curve of Figure 15.0.4-1	1978.8[a]
	Feedwater system pipe break	LOFTRAN	0.374	--	Lower curve of Figure 15.0.4-1	1978.8[a]
15.3	Decrease in reactor coolant system flow rate					
	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, ~~THINC~~ WESTAR	0.0	--	Lower curve of Figure 15.0.4-1	1940
	Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	0.0	--	Lower curve of Figure 15.0.4-1	1978.8[a]

Table 15.0-2 (Sheet 3 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Utilized**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, THINC	--	0.0	Coefficient is consistent with a Doppler defect of -0.67%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN	0.0 and 0.374	--	Upper & lower curves of Figure 15.0.4-1	10%, 60% & 100% of 1940
	RCCA misalignment	See Section 4.3	NA	--	NA	1940
	Startup of an inactive reactor coolant pump at an incorrect temperature	LOFTRAN, FACTRAN THINC	0.374	--	Upper curve of Figure 15.0.4-1	1358
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	--	NA	0 and 1940
	Inadvertent loading and operation of a fuel assembly in an improper position	See Section 4.3	NA	--	NA	1940

Table 15.0-2 (Sheet 4 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

			Reactivity Coefficients Assumed			
Sect-tion	**Faults**	**Computer Codes Utilized**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.4	Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN	Refer to Subsection 15.4.8	Refer to subsection 15.4.8	Coefficient consistent with a Doppler defect of -0.67% ΔK at BOC and -0.63% ΔK at EOC	0 and 1978.8[a]
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN ~~and~~	0 0.374	-	Upper & lower curves of Figure 15.0.4-1	1940
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN	0.0	-	Lower curve of Figure 15.0.4-1	1940
	Steam generator tube failure	LOFTTR2	0.0	-	Lower curve of Figure 15.0.4-1	1978.8[a]
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/TRAC	See Subsection 15.6.5 references	-	See Subsection 15.6.5 references	1971.7

[a] 102% of rated thermal power.
BOC - Beginning of Core Life
EOC - End of Core Life

Table 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS UTILIZED IN ACCIDENT ANALYSES

	RTDP With 10% Steam Generator Tube Plugging	Without RTDP (a)	
		Without Steam Generator Tube Plugging	With 10% Steam Generator Tube Plugging
Thermal Output of NSSS (MWt)	1940.0	1940.0	1940.0
Core Inlet Temperature (°F)	~~530.98~~ 533.40	~~528.6~~ 531.9	~~530.4~~ 532.8
Vessel Average Temperature (°F)	~~565.2~~ 567.6	~~562.8~~ 565.9	~~565.2~~ 567.6
Reactor Coolant System Pressure (psia)	2250.0	2250.0	2250.0
Reactor Coolant Flow Per Loop (GPM)	9.66 ~~9.68~~ E+04	9.71 E+04	9.48 ~~9.50~~ E+04
Steam Flow From NSSS (lbm/hr)	8.43 ~~8.44~~ E+06	8.43 ~~8.44~~ E+06	8.43 ~~8.44~~ E+06
Steam Pressure at Steam Generator Outlet (psia)	~~777.0~~ 794.0	~~777.0~~ 801.0	~~777.0~~ 794.0
Maximum Steam Moisture Content (%)	~~0.25~~ 0.10	~~0.25~~ 0.10	~~0.25~~ 0.10
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	435.0	435.0	435.0
Average Core Heat Flux (BTU/-hr-ft^2)	1.43 E+05	1.43 E+05	1.43 E+05

a. Steady-state errors discussed in ~~S~~ubsection 15.0.3 are added to these values to obtain initial conditions for transient analyses.

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 (s)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	84%	0.5
Source range neutron flux	NA	0.5
Overtemperature ΔT	Variable (see Figure 15.0.3-1)	2.0
Overpower ΔT	Variable (see Figure 15.0.3-1)	2.0
High pressurizer pressure	2460 psia	2.0
Low pressurizer pressure	Insert A ~~1800 psia~~	~~2.0~~
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.45
Reactor coolant pump under speed	90% nominal	0.767
Low steam generator level	0.0% of narrow range level span	2.0
High-2 steam generator level reactor trip	100% of narrow range level span	2.0

A

1800 psia (lower bound) 2.0
2030 psia (upper bound) 0.0

Table 15.0-5 (Sheet 1 of 2)

DETERMINATION OF MAXIMUM POWER RANGE NEUTRON FLUX CHANNEL TRIP SETPOINT, BASED ON NOMINAL SETPOINT AND INHERENT INSTRUMENTATION UNCERTAINTIES

Nominal setpoint (% of rated power)	109

Calorimetric errors in the measurement of secondary system thermal power:

Variable	Accuracy of Measurement of Variable	Effect on Thermal Power Determination (% of Rated Power)
Feedwater temperature	±3.°F	
Steam pressure (small correction on enthalpy)	±6 psi	
Feedwater flow	±0.5% Delta-P instrument span (two channels per steam generator)	
Assumed calorimetric error		2.0 (a)*
Radial power distribution effects on total ion chamber current		7.8 (b)*
Allowed mismatch between power range neutron flux channel and calorimetric measurement		2.0 (c)*

Table 15.0-5 (Sheet 2 of 2)

DETERMINATION OF MAXIMUM POWER RANGE NEUTRON FLUX CHANNEL TRIP SETPOINT, BASED ON NOMINAL SETPOINT AND INHERENT INSTRUMENTATION UNCERTAINTIES

Calorimetric errors in the measurement of secondary system thermal power:

Variable	Accuracy of Measurement of Variable	Effect on Thermal Power Determination (% of rated power)
Instrumentation channel drift and setpoint reproducibility	0.4% of instrument span (120% power span)	0.84(d)*
Instrumentation channel temperature effects		0.48(e)*
*Total assumed error in setpoint (% of rated power): $[(a)^2 + (b)^2 + (c)^2 + (d)^2 + (e)^2]^{1/2}$		±8.4
Maximum power range neutron flux trip setpoint assuming a statistical combination of individual uncertainties (% of rated power)		118

Table 15.0-6 (Sheet 1 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF & Other Equipment
Section ~~15.0.1~~ 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	High-2 steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	--	--
Inadvertent opening of a steam generator safety valve	Low pressurizer pressure, manual "S"	Low pressurizer pressure, low T_{cold}, Low-2 pressurizer level	CMT, feedwater isolation valves, steam line stop valves
Steam system piping failure	"S", low pressurizer pressure, manual	Low pressurizer pressure, low compensated steam line pressure, High-1 containment pressure, low T_{cold}, manual	CMT, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators
Inadvertent operation of the PRHR	Overpower ΔT, power range high neutron flux, low pressurizer pressure, "S", manual	Low pressurizer pressure, low T_{cold}, Low-2 pressurizer level	CMT

Table 15.0-6 (Sheet 2 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF & Other Equipment
Section ~~15.0.2~~ 15.2			
Decrease in heat removal by the secondary system			
Loss of external load/turbine trip	High pressurizer pressure overtemperature ΔT, overpower ΔT, manual	--	Pressurizer safety valves, steam generator safety valves
Loss of non-emergency ac power to the station auxiliaries	Steam generator low-narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low start-up water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Steam generator low-narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low start-up water flow, steam generator low wide range level	PRHR, Steam generator safety valves, pressurizer safety valves
Feedwater system pipe break	Steam generator low narrow range level, high pressurizer pressure, manual	Steam generator low wide range level, low steam line pressure, High-1 containment pressure	PRHR, CMT, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves

Table 15.0-6 (Sheet 3 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF & Other Equipment
Section ~~15.0.3~~ 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low flow, underspeed, manual	--	Steam generator safety valves, pressurizer safety valves
Reactor coolant pump (RCP) shaft seizure (locked rotor)	Low flow, manual, high pressurizer pressure	--	Pressurizer safety valves, steam generator safety valves
Section ~~15.0.4~~ 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Power range high flux (low setpoint), source range high flux, intermediate range high flux, manual	--	--
Uncontrolled RCCA bank withdrawal at power	Power range high flux, overtemperature ΔT, high pressurizer pressure, manual	--	Pressurizer safety valves, steam generator safety valves
RCCA misalignment	Overtemperature ΔT, manual	--	--
Startup of an in-active reactor coolant pump at an incorrect temperature	Power range high flux, low flow (P-8 interlock), manual	--	--

Table 15.0-6 (Sheet 4 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF & Other Equipment
Section ~~15.0.4~~ 15.4 (continued)			
CVS malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, overtemperature ΔT, manual	Source range flux doubling	Low insertion limit annunciators
Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	--	Pressurizer safety valves
Section ~~15.0.5~~ 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the ECCS during power operation	High pressurizer pressure, manual, "safeguards" trip, high pressurizer level	High pressurizer level, low T_{cold}	CMT, Pressurizer safety valves, CVS isolation, PRHR
Section ~~15.0.6~~ 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure	CMT, ADS, accumulator

Table 15.0-6 (Sheet 5 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF & Other Equipment
Section ~~15.0.6~~ 15.6 (continued)			
Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, safeguards ("S"), manual	Low pressurizer pressure, high steam generator level, low steam line pressure	CMT, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steamline and SG blowdown), startup feedwater isolation, CVS pump isolation, pressurizer heater isolation, steam generator PORV isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, safeguards ("S"), manual	High-1 containment pressure, low pressurizer pressure	CMT, accumulator, ADS, Steam generator safety and/or relief valves, PRHR

Table 15.0-7 (Sheet 1 of 2)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Failure
Feedwater temperature reduction[a]	---
Excessive feedwater flow	One protection division
Excessive steam flow	One protection division
Inadvertent secondary depressurization	One CMT discharge valve
Steam system piping failure	One CMT discharge valve
Inadvertent Operation of the PRHR	One protection division
Steam pressure regulator malfunction[b]	---
Loss of external load	One protection division
Turbine trip	One protection division
Inadvertent closure of main steam isolation valve	One protection division
Loss of condenser vacuum	One protection division
Loss of ac power	One PRHR discharge valve
Loss of normal feedwater	One PRHR discharge valve
Feedwater system pipe break	One PRHR discharge valve
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
Reactor coolant pump locked rotor	One protection division
Reactor coolant pump shaft break	One protection division
Rod cluster control assembly (RCCA) bank withdrawal from subcritical	One protection division
RCCA bank withdrawal at power	One protection division
Dropped RCCA, dropped RCCA bank	One protection division
Statically misaligned RCCA[c]	---
Single RCCA withdrawal	One protection division

Table 15.0-7 (Sheet 2 of 2)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Failure
Inactive reactor coolant pump startup	One protection division
Flow controller malfunction(b)	---
Uncontrolled boron dilution	One protection division
Improper fuel loading(c)	---
RCCA ejection	One protection division
Inadvertent emergency core cooling system operation at power	One protection division
Increase in reactor coolant system inventory	One protection division
Inadvertent reactor coolant system depressurization	One protection division
Failure of small lines carrying primary coolant outside containment(c)	---
Steam generator tube rupture	Faulted steam generator power operated relief valve fails open
Spectrum of loss-of-coolant accident	
Small breaks	One 4th stage ADS valve
Large breaks	One CMT discharge valve
Double-ended CMT piping breaks	One protection division (eliminates one 1st stage and one 3rd stage ADS valve)

(a) No protection action required.
(b) Not applicable to AP600.
(c) No transient analysis.

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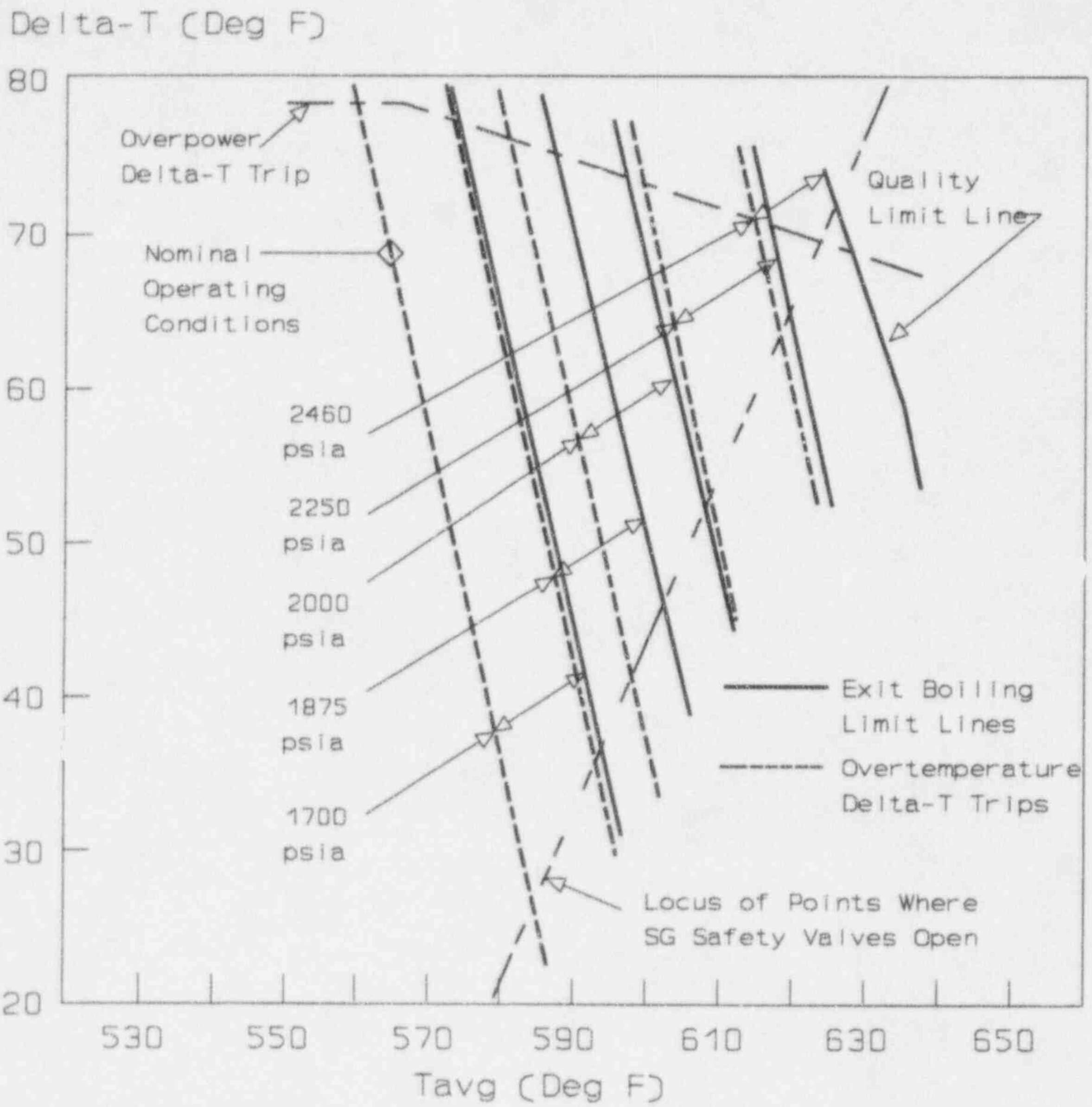

Figure 15.0.3-1

Illustration of Overpower and Overtemperature ΔT Protection

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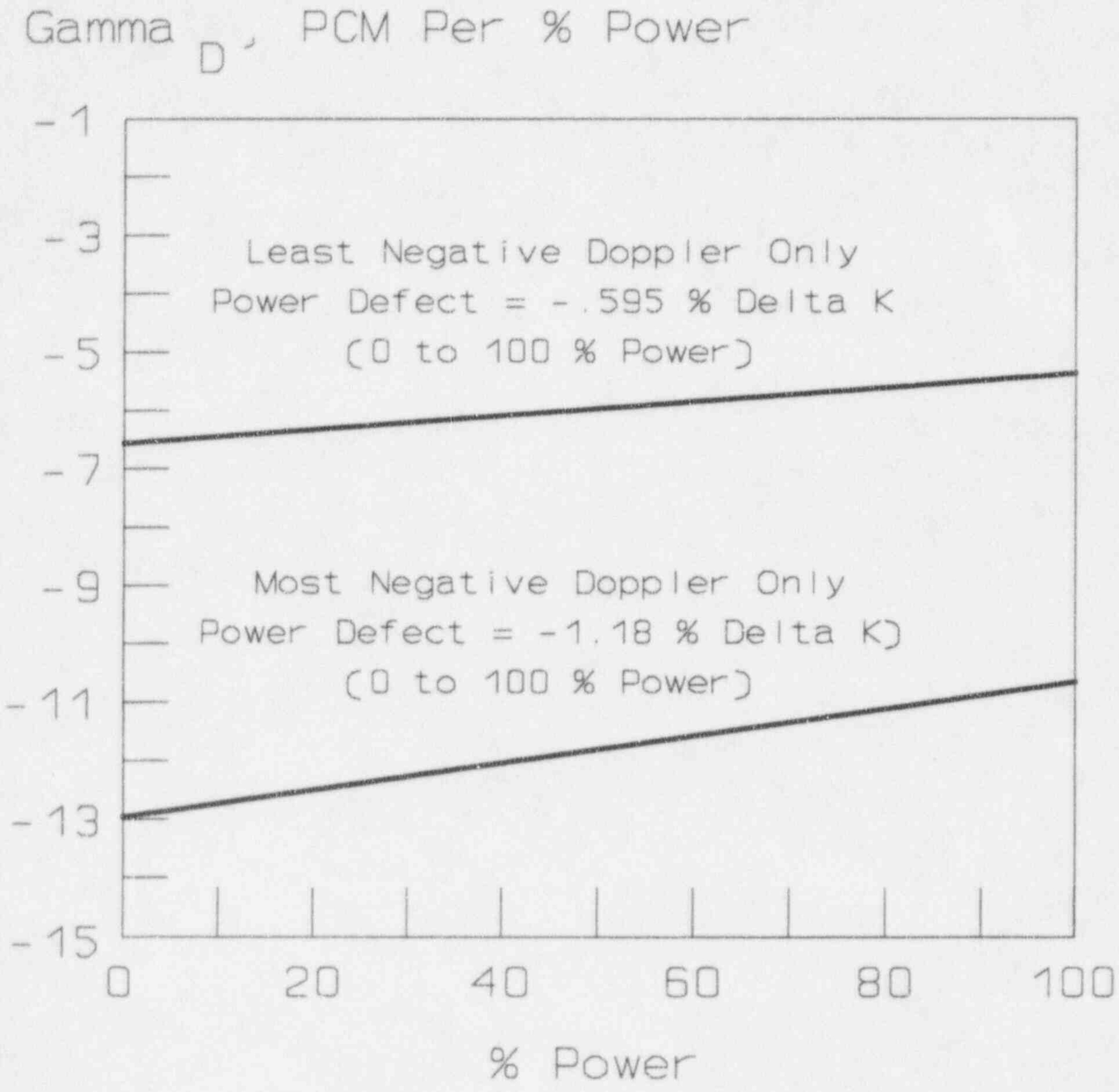

Figure 15.0.4-1

Doppler Power Coefficient used in Accident Analysis

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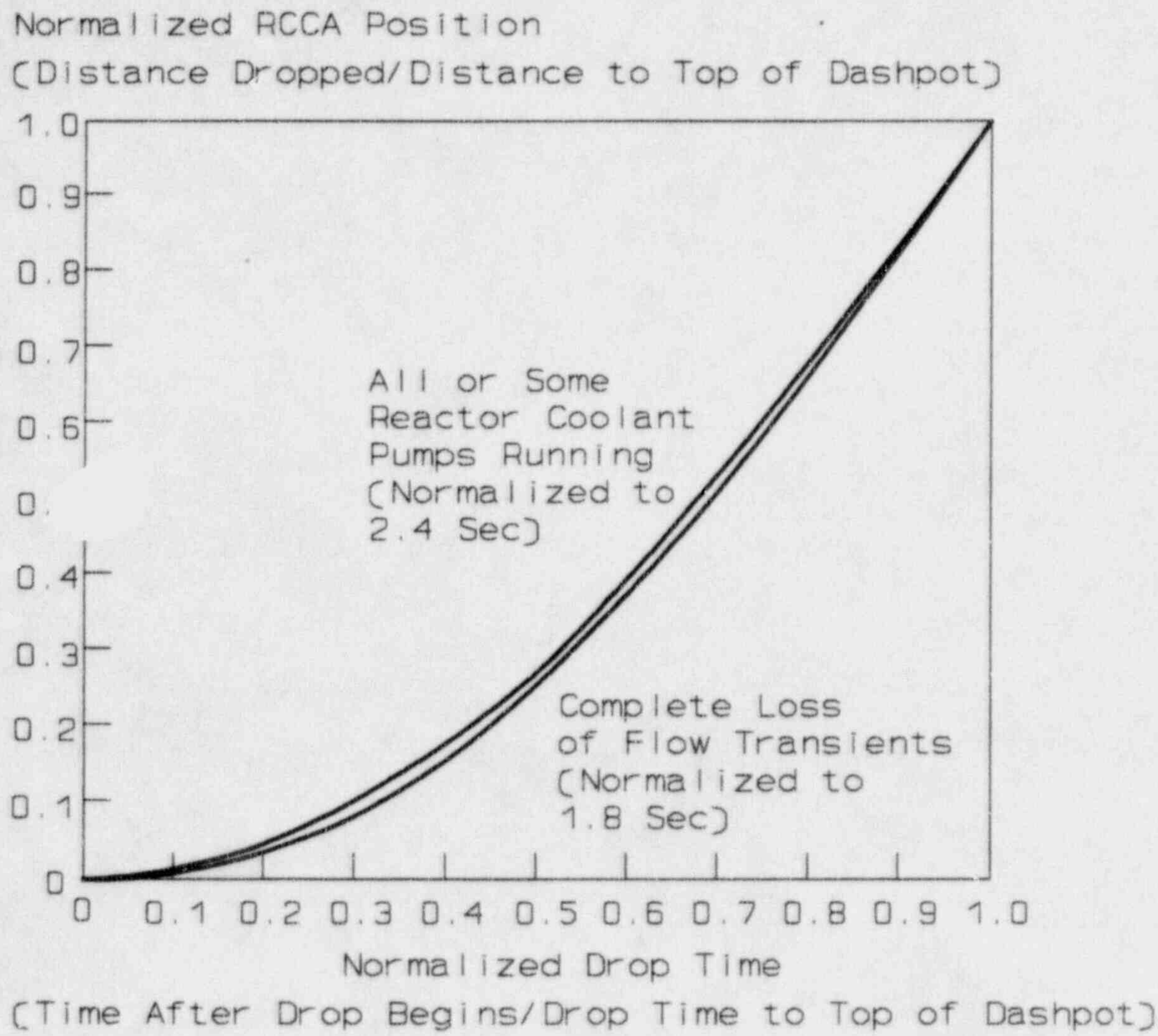

Figure 15.0.5-3

Normalized RCCA Bank Reactivity Worth ~~Versus~~ vs. Drop Time

15.1 Increase in Heat Removal From the Primary System

A number of events are postulated which could result in an increase in heat removal from the reactor coolant system (RCS). Detailed analyses are presented for the events that have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal ~~system~~ heat exchanger

The preceding events are Condition II events, with the exception of small steam system piping failures, which are considered to be Condition III and large steam system piping failure Condition IV events. Subsection 15.1 contains a discussion of classifications and applicable criteria.

The accidents in this section are analyzed. The most severe radiological consequences result from the main steam line break accident discussed in Subsection 15.1.5. Therefore, the radiological consequences are reported only for that limiting case.

15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident

Reductions in feedwater temperature causes 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 overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents any power increase that could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit valves.

A reduction in feedwater temperature may be caused by a low-pressure heater train or a high-pressure heater out of service. At power, this increased subcooling creates 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 a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flows decrease, so the no-load transient is less severe than the full-power case.

The net effec on the reactor coolant system due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow: That is, the reactor reached a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as a Condition II event, fault of moderate frequency.

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 15.8 and listed in Table 15-6.

15.1.1.2 Analysis of Effects and Consequences

15.1.1.2.1 Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following the removal of a low-pressure feedwater heater train or a high-pressure heater from service. These feedwater conditions are then used to recalculate a heat balance through the high-pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

- Plant initial power level corresponding to 100 percent nuclear steam supply system thermal output
- Isolation of one string of low-pressure feedwater heaters

Plant characteristics and initial conditions are further discussed in Subsection 15.3.

15.1.1.2.2 Results

45.4

Isolation of a string of low-pressure feedwater heaters causes a reduction in feedwater temperature that increases the thermal load on the primary system. The calculated reduction in feedwater temperature is ~~12.8~~°F, resulting in an increase in heat load on the primary system of less than 10 percent full power. Therefore, the transient results of this analysis are not presented.

15.1.1.3 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event and the increase in secondary steam flow event. (See Subsections 15.1.2 and 15.1.3.) Based on the results presented in Subsections 15.1.2 and 15.1.3, the applicable SRP Section 15.1.1 evaluation criteria for the decrease in feedwater temperature event are met.

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

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents a power increase that leads to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit value.

An example of excessive feedwater flow is 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 cold feedwater may cause a decrease in reactor coolant system temperature and 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-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and ~~reactor trip on turbine trip~~ trips the reactor.

An increase in normal feedwater flow is classified as a Condition II event, fault of moderate frequency.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.8 and listed in Table 15-6.

15.1.2.2 Analysis of Effects and Consequences

15.1.2.2.1 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 multiloop system, neutron kinetics, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior if excessive feedwater addition occurs because of system malfunction or operator error which allows a feedwater control valve to open fully. The following two cases are analyzed assuming a conservatively large negative moderator temperature coefficient:

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

- Accidental opening of one feedwater control valve with the reactor in automatic control at full power

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

- For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 115 percent of nominal feedwater flow to one steam generator

- For the feedwater control valve accident at zero-load condition, a feedwater control valve malfunction occurs, which results in a step increase in flow to one steam generator from 0 to 115 percent of the nominal full-load value for one steam generator

- For the zero-load condition, feedwater temperature is at a conservatively low value of 40°F

- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown

- The feedwater flow resulting from a fully open control valve is terminated by a steam generator High-2 level trip signal, which closes feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine

Plant characteristics and initial conditions are further discussed in Subsection 15.3. and safety monitoring

Normal reactor control systems are not required to function. The reactor protection system may function to trip the reactor because of overpower or High-2 steam generator water level conditions. No single active failure prevents operation of the ~~reactor~~ protection system. A discussion of anticipated transients without trip considerations is presented in Section 15.8.

15.1.2.2.2 Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the preceding assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.4.1 for an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition. Therefore, the results of the analysis are not presented here. If the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent nominal full power.

The full-power case (maximum reactivity feedback coefficients, automatic rod control) results in the greatest power increase. Assuming the rod control system to be in the manual control mode results in a slightly less severe transient.

When the steam generator water level in the faulted loop reaches the High-2 level setpoint, the feedwater control valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps are tripped. This prevents continuous addition of the feedwater. In addition, a turbine trip and a reactor trip are initiated.

Transient results show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. (See Figures 15.1.2-1 and 15.1.2-2.) The departure from nucleate boiling ratio does not drop below the limit value. Following the reactor trip, the plant approaches a stabilized and safe condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Since the power level rises to a maximum of about 4 percent during the excessive feedwater flow incident, the fuel temperatures also rises until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. Therefore, the peak value does not exceed 118 percent of its nominal value (the assumed high neutron flux trip setpoint). The peak fuel temperature thus remains well below the fuel melting temperature.

The transient results show that departure from nucleate boiling 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 rods are not reduced. The fuel cladding temperature therefore does not rise significantly above its initial value during the transient.

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

15.1.2.3 Conclusions

The results of the analysis show that the departure from nucleate boiling ratios encountered for an excessive feedwater addition at power are above the limit value. The departure from nucleate boiling ratio design basis is described in Section 4.4.

Additionally, the reactivity insertion rate that occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from subcritical condition analysis. See Subsection 15.4.1.

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 a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The ~~reactor~~ plant 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 25 to 100 percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the ~~reactor~~ protection and safety monitoring system. Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5.

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

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

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

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux

An excessive load increase incident is considered to be a Condition II event, as described in Subsection 15.1.

15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Method of Analysis

This accident is analyzed using the LOFTRAN code (Reference 1). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer 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 plant behavior following a 10 percent step load increase from rated load. These cases are as follows:

- Reactor control in manual with minimum moderator reactivity feedback
- Reactor control in manual with maximum moderator reactivity feedback
- Reactor control in automatic with minimum moderator reactivity feedback
- Reactor control in automatic with maximum moderator reactivity feedback

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity; therefore, reductions in coolant temperature have the least impact on core power. 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 the cases with automatic rod control, no credit is taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant quickly stabilizes.

A 10 percent step increase in steam demand is assumed, and each case is studied without credit being taken for pressurizer heaters. Initial reactor power, RCS pressure and temperature are assumed to be at their full power values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 2. Plant characteristics and initial conditions are further discussed in Subsection 15.3

Normal reactor control systems and engineered safety systems are not required to function. The ~~reactor~~ protection and safety monitoring 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 prevents the ~~reactor~~ protection and safety monitoring system from performing its intended function.

15.1.3.2.2 Results

Figures 15.1.3-1 through 15.1.3-10 show the transient with the reactor in the manual control mode. 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 departure from nucleate boiling ratio which increases above its initial value. For the maximum moderator feedback, manually controlled case there is a much faster increase in reactor power due to the moderator feedback. A reduction in the departure from nucleate boiling ratio is experienced, but the departure from nucleate boiling ratio remains above the safety analysis limit.

Figures 15.1.3-11 through 15.1.3-20 show the transient assuming the reactor is in the automatic control mode and no reactor trip signals occur. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum departure from nucleate boiling ratio remains above the safety limit.

For the cases analyzed the plant power stabilizes at an increased power level ~~about 110 percent of its nominal value, but the case with minimum feedback and manual rod control is much slower in reaching its equilibrium condition~~. Normal plant operating procedures are followed to reduce power. Because of the measurement errors assumed in the setpoints, it is possible that reactor trip could actually occur for the automatic control cases. The plant then reaches a stabilized condition following the trip.

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

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

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

15.1.3.3 Conclusions

The analysis just presented above shows that for a 10 percent step load increase the departure from nucleate boiling ratio remains above the safety analysis limit. The design basis for departure from nucleate boiling ratio is described in Section 4.4. The plant rapidly reaches a stabilized condition following the load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line 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 of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following SRP Section 15.1.4 evaluation criterion is satisfied:

Assuming the most reactive stuck rod cluster control assembly (RCCA), with offsite power available, and assuming a single failure in the engineered safety features (ESF) system, there will be no consequential damage to the fuel or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, or the largest of any single steam dump, relief, or safety valve. This criterion is met by showing the departure from nucleate boiling (DNB) design basis is not exceeded.

Accidental depressurization of the secondary system is classified as a Condition II event as described in Subsection 15.1.

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

- Core makeup tank actuation on a safeguard signal from one of the following four signals:
  - Two out of four low pressurizer pressure signals
  - Two out of four low pressurizer level signals
  - Two out of four low T_{cold} signals in any one loop
  - Two out of four low steam line pressure signals in any one loop

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safeguards signal

- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater valves following reactor trip, a safeguards signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (designed to close in less than 10 seconds) on one of the following signals:

  - Two out of four low steam line pressure signals in any one loop (above permissive P-11)

  - Two out of four high negative steam pressure rates in any loop (below permissive P-11)

  Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in Subsection ~~15.8~~ 15.0.8 and listed in Table ~~15-6~~. 15.0.6

15.1.4.2 Analysis of Effects and Consequences

15.1.4.2.1 Method of Analysis

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

- 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 core makeup tank injection

- Analyses to determine that there is no damage to the fuel or reactor coolant system

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

- 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 mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied

- The most 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} (considering moderator temperature and density effects) versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power reactivity feedback is modeled as a function of core thermal power and core mass flow. The feedback calculations performed in LOFTRAN are discussed further in Subsection 15.1.5.2.1

- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (~~3200~~ 3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The case studied is a steam flow of ~~598~~ 520 pounds per second at 1200 psia with offsite power available. This conservatively models 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 conditions

Should the reactor be just critical or operating at power at the time of a steam release, the reactor is tripped by the normal overpower protection 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 steam release before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached.

After the additional stored energy is removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis, which assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for steam line release occurring at power.

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

- Perfect moisture separation in the steam generator

- Offsite power is available, since this maximizes the cooldown

- Maximum cold startup feedwater flow

- Four reactor coolant pumps are initially operating

- Manual actuation of the passive residual heat removal system at time zero is conservatively assumed to maximize the cooldown

15.1.4.2.2 Results

The results presented conservatively indicate the events that would occur assuming a secondary system steam release since it is postulated that the conditions just described occur simultaneously.

Figures 15.1.4-2 through 15.1.4-13 show the transient results for a steam flow of ~~590~~ 520 pounds per second at 1200 psia.

The assumed steam release is typical of the capacity of any single steam dump, relief, or safety valve. Core makeup tanks injection and the associated tripping of the reactor coolant pumps are initiated automatically by low pressurizer pressure safeguard signal. Boron solution at ~~3200~~ 3400 ppm enters the reactor coolant system, providing enough negative reactivity to prevent core damage. Later in the transient, as the reactor coolant pressure continues to fall, the accumulators actuate and inject boron solution at ~~1900~~ 2600 ppm.

The transient is conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes, and the passive residual heat removal system is assumed to be actuated at time zero. Since the limiting portion of the transient occurs over a period of about five minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

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

15.1.4.3 Margin to Critical Heat Flux

~~The~~ departure from nucleate boiling analysis is performed for the ~~steam system piping failure event (Subsection 15.1.5.2.4) bounds the results for~~ inadvertent opening of a steam generator relief or safety valve. That ~~bounding~~ analysis demonstrates that the departure from nucleate boiling design basis, as described in Section 4.4, is met for the inadvertent opening of a steam generator relief or safety valve.

15.1.4.4 Conclusions

The analysis shows that the criterion stated earlier in this subsection is satisfied. For an inadvertent opening of any single steam dump or a steam generator relief or safety valve, the departure from nucleate boiling design basis is met.

15.1.5 Steam System Piping Failure

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line 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 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 is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core becomes critical and returns to power. A return to power following a steam line rupture is a potential problem mainly because of the existing high-power peaking factors, assuming the most reactive rod cluster control assembly to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the passive core cooling system.

The analysis of a main steam line rupture is performed to demonstrate that the following SRP Section 15.1.5 evaluation criterion is satisfied:

Assuming the most reactive stuck rod cluster control assembly with or without offsite power and assuming a single failure in the engineered safety features (ESFs), the core cooling capability is maintained. Radiation doses do not exceed the guidelines of 10 CFR 100.

Departure from nucleate boiling and possible clad perforation following a steam pipe rupture are not necessarily unacceptable. But, the following analysis, in fact, shows that the departure from nucleate boiling design basis is not exceeded for any rupture, assuming the most reactive assembly stuck in its fully withdrawn position. The departure from nucleate boiling ratio design basis is discussed in Section 4.4.

A major steam line rupture is classified as a Condition IV event.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Subsection ~~15.1.3.~~ 15.0.1.3

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat retards the cooldown, thereby reducing the likelihood that the reactor returns to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here.

Certain assumptions used in this analysis are discussed in Reference 4. Reference 4 also contains a discussion of the spectrum of break sizes and power levels analyzed.

The following functions provide the protection for a steam line rupture: (See Subsection 7.2.1.1.2.)

- Core makeup tank actuation from any of the following:
  - Two out of four low pressurizer pressure signals
  - Two out of four High-1 containment pressure signals
  - Two out of four low steam line pressure signals in any loop
  - Two out of four low T_{cold} signals in any one loop

- Two out of four low pressurizer level signals

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safeguards signal

- Redundant isolation of the main feedwater lines

Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves, the safeguards signal rapidly closes all feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Fast-acting main steam line isolation valves (MSIVs) (designed to close in less than 10 seconds) on any of the following:
  - Two out of four High-1 containment pressure
  - Two out of the four low steam line pressure signals in any one loop (above permissive P-11)
  - Two out of four high negative steam pressure rates in any one loop (below permissive P-11)

A fast-acting main steam isolation valve is provided in each steam line. These valves fully close within 10 seconds of actuation following a large break in the steam line. For breaks downstream of the main steam line isolation valves, closure of at least one valve in each line completely terminates the blowdown.

For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the main steam line isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

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

Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

15.1.5.2.1 Method of Analysis

The analysis of the steam pipe rupture is performed to determine the following:

- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code (Reference 1) is used to determine the system transient.

- The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, THINC, is used to determine if departure from nucleate boiling occurs for the core transient conditions computed by the LOFTRAN code.

The following conditions are assumed to exist at the time of a main steam line break accident:

- End-of-cycle shutdown margin at no-load, equilibrium xenon conditions, and the most reactive rod control assembly stuck in its fully withdrawn position. Operation of the control rod mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- 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 has been included. The k_{eff} (considering moderator temperature and density effects) versus temperature at 1000 psia corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power reactivity feedback is modeled as a function of thermal power and core mass flow.

The core properties used in the LOFTRAN mode for feedback calculations are generated by combining those in the sector nearest the affected steam generator with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighing of the fluid properties from the coldest core sector.

To verify the conservatism of this method, the power predictions of the LOFTRAN point kinetics modeling are confirmed by comparison with detailed core analysis for the limiting conditions of the cases considered. This core analysis explicitly models the hypothetical core configuration (i.e., stuck rod cluster control assembly, non-uniform inlet temperatures, pressure, flow, and boron concentration) and directly evaluates the total reactivity feedback including power, boron, and density redistribution in an integral fashion. The effect of void formation is also included.

Comparison of the results from the detailed core analysis with the LOFTRAN predictions verify the overall conservatism of the methodology. That is, the specific power, temperature,

and flow conditions used to perform the departure from nucleate boiling analysis are conservative.

- The portions of the passive core cooling system used in mitigating a steam line rupture are the core makeup tanks and the accumulators. There are no single failures that prevent core makeup tank injection. In modeling the core makeup tanks and the accumulators, conservative assumptions are used that minimize the capability to add borated water. Specifically, the core makeup tank injection line characteristics modeled reflect the failure of one core makeup tank discharge valve.

- Maximum overall fuel-to-coolant heat transfer coefficient, to maximize rate of cooldown

- Since the steam generators are provided with integral flow restrictors with a 1.4-square-foot throat area, any rupture in a steam line with a break area greater than 1.4 square feet, regardless of location, has the same effect on the primary plant as the 1.4-square-foot-double-ended rupture. The limiting case considered in determining the core power and reactor coolant system transient is the complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available. The results of this case clearly bound the loss of offsite power for the following four reasons:

  - Loss of offsite power results in an immediate reactor coolant pump coastdown at the initiation of the transient. This reduces the severity of the reactor coolant system cooldown by reducing primary-to-secondary heat transfer. The lessening of the cooldown, in turn, reduces the magnitude of the return to power.

  - Following actuation, the core makeup tank provides borated water that injects into the reactor coolant system. Flow from the core makeup tank increases if the reactor coolant pumps have coasted down. Therefore, the analysis performed with offsite power and continued reactor coolant pump operation actually minimizes the rate of boron injection into the core and is conservative.

  - In recognition of the preceding item, the protection system automatically provides a safety-related signal that initiates the coastdown of the reactor coolant pumps in parallel with core makeup tank actuation. Since this reactor coolant pump function is actuated early during the steam line break event (right after core makeup tank actuation), there is very little difference in the predicted departure from nucleate boiling ratio between cases with and without offsite power.

  - Because of the passive nature of the safety injection system, the loss of offsite power does not delay the actuation of the safety injection system.

- Power peaking factors corresponding to one stuck rod cluster control assembly and nonuniform core inlet coolant temperatures are determined at the 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 rod cluster control assembly during the return to power phase following the steam

line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and therefore may differ for each case studied.

The analysis assumes initial hot standby conditions at time zero since this represents the most pessimistic initial condition. If the reactor is just critical or operating at power at the time of a steam line break, the reactor is 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 reduces the cooldown caused by the steam line break before the no-load conditions of 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.

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

- Perfect moisture separation in the steam generator.

- Maximum cold startup feedwater flow plus nominal 100 percent main feedwater flow.

- Four reactor coolant pumps are initially operating.

- Manual actuation of the passive residual heat removal system at time zero is conservatively assumed to maximize the cooldown.

15.1.5.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a steam line rupture, since it is postulated that the conditions ~~just~~ described occur simultaneously.

15.1.5.2.3 Core Power and Reactor Coolant System Transient

Figures 15.1.5-1 through 15.1.5-14 show the reactor coolant system transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition.

Offsite power is assumed available so that, initially, full reactor coolant flow exists. During the course of the event, the reactor protection system initiates a coastdown of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. The transient shown

assumes an uncontrolled steam release from only one steam generator. Steam release from more than one steam generator is prevented by automatic trip of the fast-acting main steam isolation valves in the steam lines by high containment pressure signals or by low steam line pressure signals. Even with the failure of one valve, release is limited to ~~no more than~~ approximately 10 seconds for the other steam generator while the one generator blows down. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

As shown in Figure 15.1.5-3, the core attains criticality with the rod cluster control assemblies inserted (with the design shutdown assuming the most reactive rod cluster control assembly stuck) before boron solution at ~~3200~~ 3400 ppm (from core makeup tanks) or ~~1900~~ 2600 ppm (from accumulators) enters the reactor coolant system. A peak core power significantly lower than the nominal full-power value is attained.

The calculation assumes that the boric acid is mixed with and diluted by the water flowing in the reactor coolant system before entering the reactor core. The concentration after mixing depends upon the relative flow rates in the reactor coolant system and from the core makeup tanks or accumulators (or both). The variation of mass flow rate in the reactor coolant system due to water density changes is included in the calculation. So is the variation of flow rate from the core makeup tanks or accumulators (or both) due to changes in the reactor coolant system pressure and temperature and the pressurizer level. The reactor coolant system and passive injection flow calculations include the line losses.

Strikeout ¶

At no time during the analyzed steam line break event does the core makeup tank level ~~even~~ approach the setpoint for actuation of the automatic depressurization system. The combination of injection flow from the accumulators and recirculation flow from the reactor coolant system cold legs into the core makeup tanks maintains a relatively large core makeup tank water inventory throughout the event.

Replacement 1 (next page)

The passive residual heat removal system provides a passive, long-term mean of removing the core decay and stored heat by transferring the energy via the passive residual heat removal heat exchanger~~s~~ to the in-containment refueling water storage tank (IRWST). Normally the passive residual heat removal ~~system~~ heat exchanger is actuated automatically when the steam generator level falls below the low wide range level. For the main steam line rupture case analyzed, the passive residual heat removal ~~system~~ heat exchanger is conservatively actuated at time zero to maximize the cooldown.

15.1.5.2.4 Margin to Critical Heat Flux

The case presented in Subsection 15.1.5.2.2 conservatively models the expected behavior of the plant during a steam system piping failure. This includes the tripping of the reactor coolant pumps coincident with core makeup tank actuation.

A departure from nucleate boiling analysis is performed using limiting assumptions that bound those of Subsection 15.1.5.2.2. ~~Specifically, the departure from nucleate boiling analysis conservatively assumes full reactor coolant flow through the transient. A substantial return to power is predicted for this case.~~

analysis

~~In contrast,~~ under the low flow (natural circulation) conditions actually present in the AP600 transient, the return to power is severely limited by the large negative feedback due to flow and power. The results of the bounding, ~~full flow case~~ demonstrate that the departure from nucleate boiling design basis, as described in Section 4.4, is met for the steam system piping failure event.

15.1.5.3 Conclusions

The analysis shows that the departure from nucleate boiling design basis is met for the steam system piping failure event. Departure from nucleate boiling and possible clad perforation following a steam pipe rupture are not precluded by the criteria. The preceding analysis shows that no departure from nucleate boiling occurs for the rupture assuming the most reactive rod cluster control assembly stuck in its fully withdrawn position. The radiological consequences of this limiting event are within the dose criteria of 10 CFR 100.

15.1.5.4 Radiological Consequences

Subsection 15.1.5.4 to follow

Replacement 1 (for Section 15.4.5)

At no time during the analyzed steam line break event does the core makeup tank level even approach the setpoint for actuation of the automatic depressurization. During all non-LOCA events, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tank is exactly offset by an equal volume of recirculation flow that enters the core makeup tanks via the reactor coolant system cold leg balance lines.

15.1.6 Inadvertent Operation of the Passive Residual Heat Removal ~~System~~ Heat Exchanger

15.1.6.1 Identification of Causes and Accident Description

Strikeout 1¶s

The inadvertent actuation of the passive residual heat removal system causes an increase in the core reactivity by decreasing reactor coolant temperature. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents any power increase that could lead to a departure from nucleate boiling ratio less than the safety analysis limit. In addition, since the reactor coolant system depressurizes during this event, the low pressurizer pressure function could generate a reactor trip.

These protection functions do not terminate operation of the passive residual heat removal system, however. So the RCS continues to cool down and depressurize. The safety injection low pressurizer pressure setpoint actuates the core makeup tank and brings the plant to a stable condition.

Replacement A (next page)

The inadvertent actuation of the passive residual heat removal system could be caused by operator error or a false actuating signal. Actuation of the passive residual heat removal ~~system~~ heat exchanger involves opening the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the passive residual heat removal ~~system~~ heat exchanger~~s~~, and back into the associated steam generator cold leg plenum.

The passive residual heat removal ~~system~~ heat exchanger~~s~~ ~~are~~ is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the passive residual heat removal ~~system~~ heat exchanger is enhanced. The heat sink for the passive residual heat ~~system~~ removal heat exchanger is provided by the in-containment residual water storage tank, in which the passive residual heat removal ~~system~~ heat exchanger~~s~~ ~~are~~ is submerged.* Since the passive residual heat removal ~~system~~ heat exchanger is connected to only one reactor coolant system loop, the cooldown resulting from its actuation is asymmetric with respect to the core.

Insert 1 (next page)

15.1.6 Inadvertent Operation of the Passive Residual Heat Removal Heat Exchanger

Replacement A

The inadvertent actuation of the passive residual heat removal heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) is intended to prevent a power increase that could lead to a departure from nucleate boiling ratio less than the safety analysis limit. In addition, since the cold leg temperature is reduced and the reactor coolant system depressurizes during this event, the low cold leg temperature or low pressurizer pressure functions could generate a reactor trip. These protection functions do not terminate operation of the passive residual heat removal heat exchanger.

Insert 1

Since the fluid in the heat exchanger is in thermal equilibrium with water in the tank, the initial flow out of the passive residual heat removal heat exchanger is significantly colder than the reactor coolant system fluid. Following this initial insurge, the reduction in cold leg temperature is limited by the cooling capability of the passive residual heat removal heat exchanger.

heat exchanger

~~The injection of relatively cold water into the reactor coolant system produces a reactivity insertion in the presence of a negative moderator temperature coefficient.~~ The response of the plant to an inadvertent passive residual heat removal ~~system~~ actuation with the plant at no-load conditions is bounded by the analyses performed for the inadvertent opening of a steam generator relief or safety valve event (Subsection 15.1.4) and the steam system piping failure event (Subsection 15.1.5). Both of these events are conservatively analyzed assuming passive residual heat removal ~~system~~ heat exchanger actuation coincident with the steam line depressurization. Therefore, only the response of the plant to an inadvertent passive residual heat removal ~~system~~ initiation with the core at power is considered here.

15.0.8 heat exchanger

The inadvertent actuation of the passive residual heat removal ~~system~~ event is a Condition II event, a fault of moderate frequency. Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection ~~15.8~~ and listed in Table ~~15-6~~. The following reactor protection system functions ~~provide protection in the event of an inadvertent passive residual heat removal~~ ~~system~~ heat exchanger actuation. are available to

15.0-6

- The overpower reactor trips (neutron flux and ΔT)
- Two out of four low pressurizer pressure signals
- Two out of four low T_{cold} signals in any one loop
- Two out of four low pressurizer level signals

15.1.6.2 Analysis of Effects and Consequences

15.1.6.2.1 Method of Analysis

The excessive heat removal due to an inadvertent passive residual heat removal ~~system~~ heat exchanger actuation transient is analyzed by using the ~~detailed digital~~ computer code LOFTRAN (Reference 1). This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate plant behavior in the event of an inadvertent passive residual heat removal ~~system~~ heat exchanger actuation due to an operator error or a false actuation signal that opens the valves that normally isolate the passive residual heat removal ~~system~~ heat exchanger from the remainder of the reactor coolant system. Both full power and zero-load conditions ~~need~~ are to be considered. ~~However, as indicated earlier,~~ the analyses for the inadvertent opening of a steam generator relief or safety valve event (Subsection 15.1.4) and the steam system piping failure event (Subsection 15.1.5) bound the results for the zero-power inadvertent passive residual heat removal ~~system~~ heat exchanger actuation transient.

The case considered here is the response of the plant to an inadvertent passive residual heat removal ~~system~~ heat exchanger initiation with the core initially operating at full power. The reactivity insertion transient arising from the inadvertent actuation of the passive residual heat removal ~~system~~ heat exchanger is calculated including the following ~~six~~ assumptions:

- With the core at full power, the inadvertent passive residual heat removal ~~system~~ heat exchanger actuation occurs at 10 seconds. The LOFTRAN code explicitly models the performance of the passive residual heat removal ~~system~~ heat exchanger and the resulting cooldown transient experienced by the reactor coolant system.

Strikeout #5

- A conservative model for predicting the power excursion experienced by the core.

This includes the use of a moderator density coefficient reflecting a most negative moderator temperature coefficient of reactivity in conjunction with a low level of power feedback. Additionally, the temperature of the colder loop is weighted more heavily in the calculation of core reactivity, thereby maximizing the predicted peak power.

- The reactor trip on high neutron flux is conservatively ignored. Instead, the overtemperature and overpower ΔT trips are the only overpower/overtemperature protection functions which are credited in the analysis.

Replacement B (next page)

- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown.

- ~~Core makeup tank actuation on two out of four low pressurizer level and low pressurizer pressure safeguards signals is modeled.~~

- Control systems are assumed to function only if their operation results in more severe accident results. For the inadvertent passive residual heat removal ~~system~~ heat exchanger actuation event, both cases with and without automatic rod control are ~~analyzed.~~ considered

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. No single active failure prevents operation of the reactor protection system functions assumed in the analysis. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.1.6.2.2 Results

The system responses to an inadvertent passive residual heat removal ~~system~~ heat exchanger actuation at power event, with manual rod control, are shown in Figures 15.1.6-1 through 15.1.6-~~12~~ 10. The full-power case with manual rod control results in the greatest power increase.

Replacement C (next page)

Assuming the rod control system to be in automatic results in a slightly less limiting transient, as the control rods are inserted in response to a primary-to-secondary power mismatch. The results show the increase in nuclear power and ΔT associated with the inadvertent passive residual heat removal system actuation at full power. The departure from nucleate boiling ratio does not drop below the limit value.

~~Replacement D (next page)~~

Strikeout #

Following the reactor trip, core makeup tanks are actuated and reactor coolant pumps are tripped on a low pressurizer level signal. The plant reaches a safe and stable condition. Standard plant shutdown procedures may then be followed to further cool down the plant or return to power operation.

15.1.6 Inadvertent Operation of the Passive Residual Heat Removal Heat Exchanger

Replacement B

- A conservative model for predicting the power excursion experienced by the core.

  This includes the use of a negative moderator coefficient corresponding to the end-of-life rodded core. The variation of the coefficient with temperature and pressure has been included in conjunction with a low level of power feedback.

  The core properties used in the LOFTRAN Code for feedback calculations are generated by combining those in the sector nearest the loop with the passive residual heat removal system with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighing of the fluid properties from the coldest core sector.

  To verify the conservatism of this method, the power predictions of the LOFTRAN point kinetics modeling are confirmed by comparison with detailed core analysis for the limiting conditions of the cases considered. This core analysis explicitly models the hypothetical core configuration (i.e., non-uniform inlet temperatures, pressure, flow and boron) and directly evaluates the total reactivity feedback including power, boron, and density redistribution in an integral fashion.

  Comparison of the results from the detailed core analysis with the LOFTRAN predictions verify the overall conservatism of the methodology. That is, the specific power, temperature, and flow conditions used to perform the departure from nucleate boiling analysis are conservative.

- The reactor trips on high neutron flux, overtemperature and overpower ΔT trips are conservatively ignored. The analysis demonstrates that the applicable safety analysis limits are met without a reactor trip being generated.

Replacement C

The inadvertent operation of the passive residual heat removal heat exchanger incident is an overpower transient for which the fuel temperature rises. Assuming a reactor trip does not occur the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in power demanded by the system. In the limiting case analyzed, the plant power stabilizes at about 111 percent of its nominal value.

Since the power level rises during the inadvertent passive residual heat removal ~~system~~ heat exchanger initiation, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. The peak fuel temperature remains below the fuel melting temperature.

The transient results show that departure from nucleate boiling does not occur at any time during the inadvertent passive residual heat removal ~~system~~ heat exchanger actuation event. So the ability of the primary coolant to remove heat from the fuel rods is not reduced. The calculated sequence of events for this accident is shown in Table 15.1.2-1.

Insert 2 (next page)

15.1.6.3 Conclusions

The results of the analysis show that the departure from nucleate boiling ratios encountered for an inadvertent actuation of the passive residual heat removal ~~system~~ heat exchanger at power are above the safety analysis limit values. (The departure from nucleate boiling ratio design basis is described in Section 4.4.) The results for an inadvertent passive residual heat removal ~~system~~ heat exchanger actuation initiated from zero load conditions are bounded by the inadvertent opening of a steam generator relief or safety valve event (Subsection 15.1.4) and the steam system piping failure event (Subsection 15.1.5).

15.1.~~7~~8 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), and WCAP-7907-A (Nonproprietary), April 1984.

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

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

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

15.1.7 Combined License Information

This section has no requirement for additional information to be provided in support of the combined license application.

Table 15.1.2-1 (Sheet 1 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

Accident	Event	Time(s)
Feedwater system malfunctions that result in an increase in feedwater flow	One main feedwater control valve fails fully open	0.0
	Minimum DNBR occurs	~~58.5~~ 55.4
	Turbine trip/feedwater isolation on high steam generator level	~~287.7~~ 486.0
	Reactor trip on ~~turbine trip~~ high steam generator level	~~291.7~~ 490.0
	Feedwater isolation	~~299.7~~ 498.0
Excessive increase in secondary steam		
1. Manual reactor control (minimum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	~~860.~~ 220.
2. Manual reactor control (maximum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	~~76.~~ 70.
3. Automatic reactor control (minimum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	~~89.~~ 140.
4. Automatic reactor control (maximum moderator feedback)	10 percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	~~47.~~ 50.

15.1.6 Inadvertent Operation of the Passive Residual Heat Removal Heat Exchanger

Insert 2

The inadvertent operation of the passive residual heat removal heat exchanger is not included among the design overpower transients considered in Subsection 4.3. The conservative safety analysis assumptions applied to this event do not credit a reactor trip to preclude the core power from rising above 118 percent of rated thermal power. The nature of this excessive cooldown transient dictates that the predicted power excursion is associated with very low core inlet temperatures which can partially offset the penalties associated with the high power.

Table 15.1.2-1 (Sheet 2 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

Accident	Event	Time(s)
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0
	Criticality attained	~~81.4~~ 96.4
	Safeguards actuation signal on safeguards low T_{cold} ~~steam line pressure~~	~~93.8~~ 139.0
	Core makeup tank actuation	~~115.8~~ 161.0
	Boron reaches core	~~162.8~~ 176.7
Steam system piping failure	Steam line ruptures	0
	Safeguards actuation signal on safeguards low steam line pressure	0.8
	Pressurizer empty	~~13.6~~ 15.6
	Criticality attained	~~22.0~~ 15.6
	Boron reaches core	~~47.2~~ 32.0

Table 15.1.2-1 (Sheet 3 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

Accident	Event	Time(s)
Inadvertent operation of the PRHR	Inadvertent actuation of the PRHR	10.0
	~~High neutron flux reactor trip setpoint reached (conservatively ignored)~~	~~16.0~~
	~~Overpower ΔT trip setpoint reached~~	~~16.9~~
	~~Reactor trip (rod motion) occurs~~	~~18.9~~
	Minimum DNBR occurs	~~18.9~~ 24.2
	~~Low-2 pressurizer level setpoint reached~~	~~170.0~~
	~~Core makeup tank actuation~~	~~192.2~~
	Equilibrium condition reached (approximate time only)	100

Table 15.1.5-1

To follow

Table 15.1.5-2

to follow

Table 15.1.5-3

To follow

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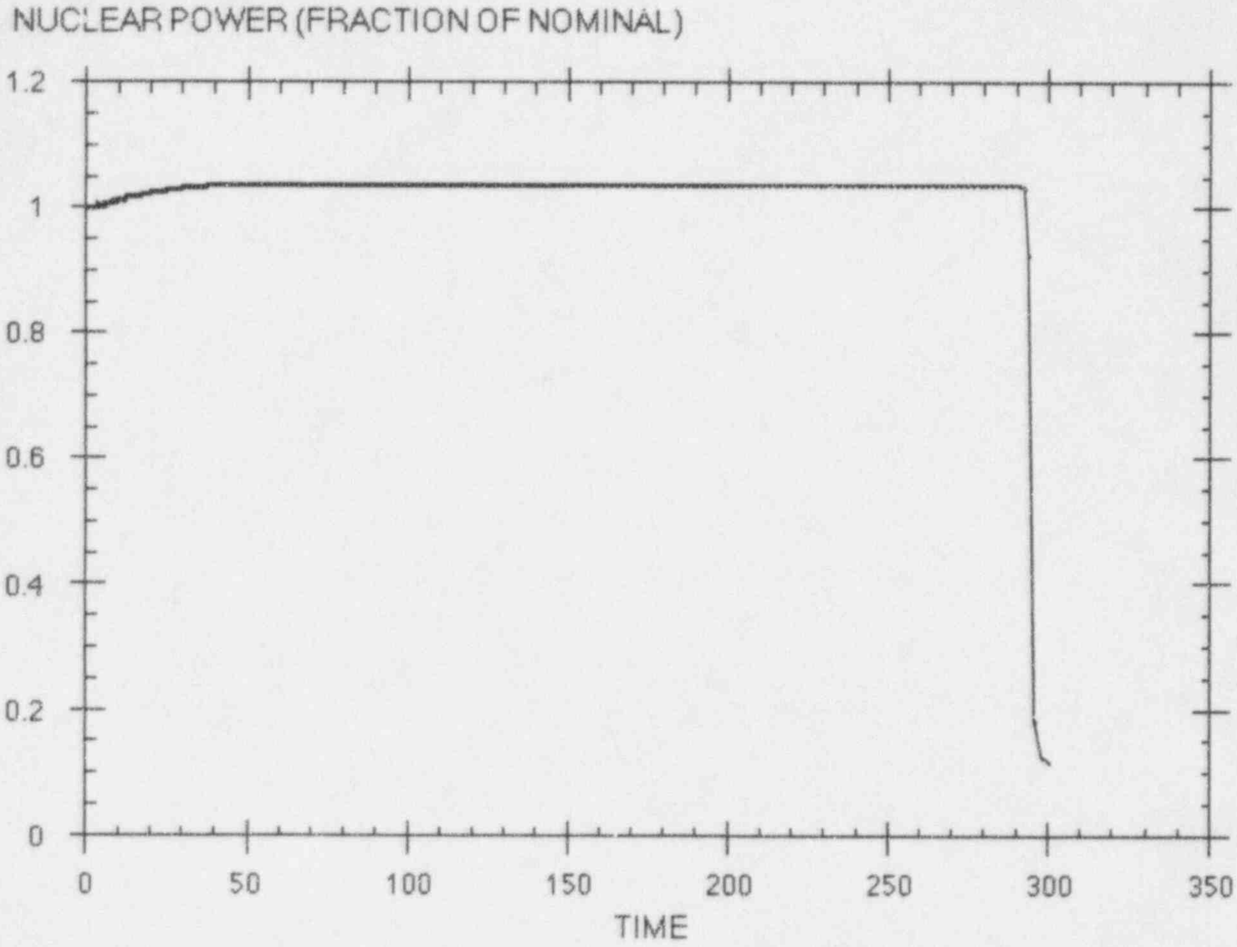
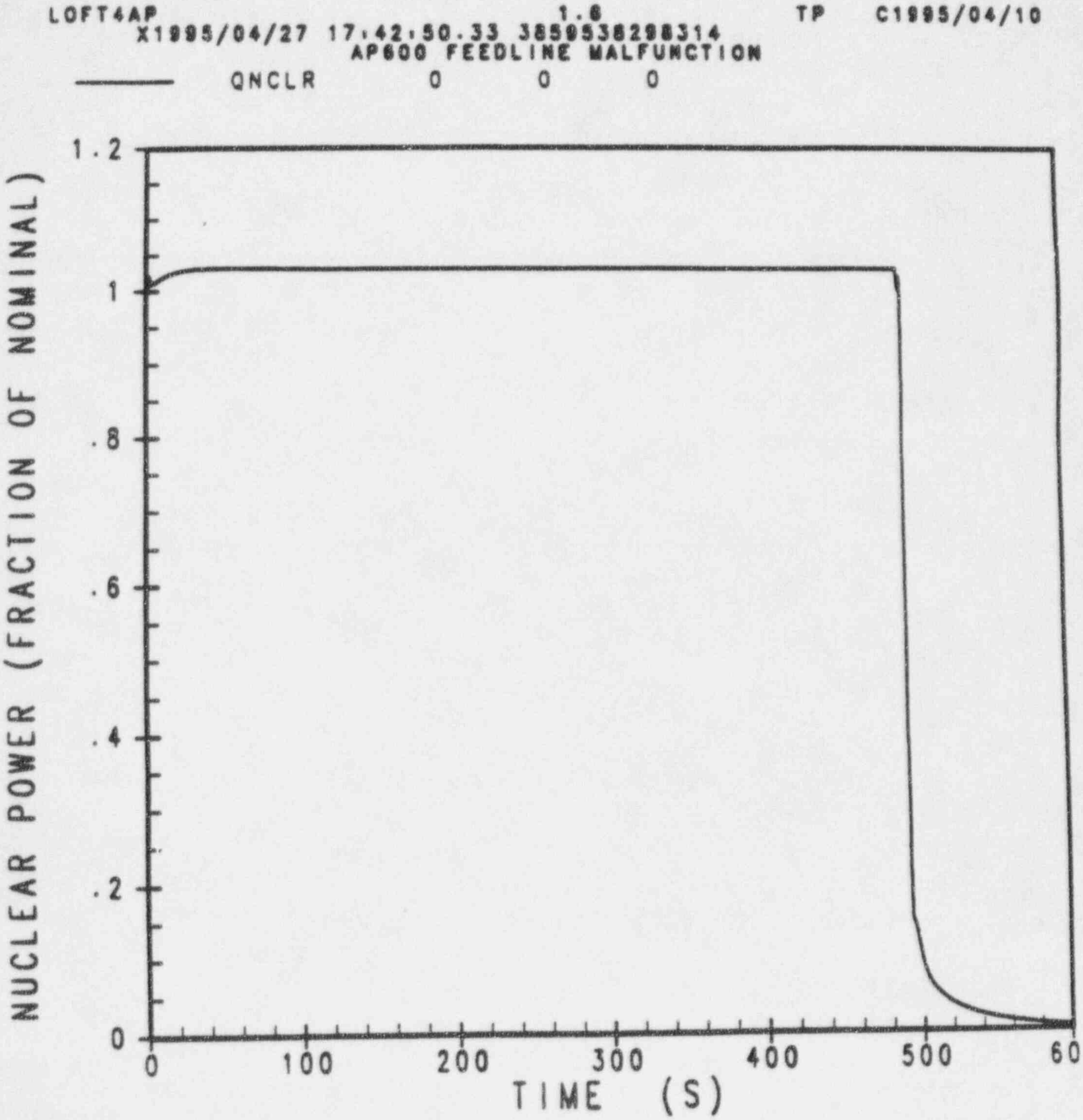


Figure 15.1.2-1

Feedwater Control Valve Malfunction Nuclear Power

even pg

15.1.2-1



LOFT4AP 1.6 TP C1995/04/10
X1995/04/27 17:42:50.33 3859538298314
AP600 FEEDLINE MALFUNCTION
QNCLR 0 0 0
NUCLEAR POWER (FRACTION OF NOMINAL)
1.2
1
.8
.6
.4
.2
0
0
100
200
300
400
500
60
TIME (S)

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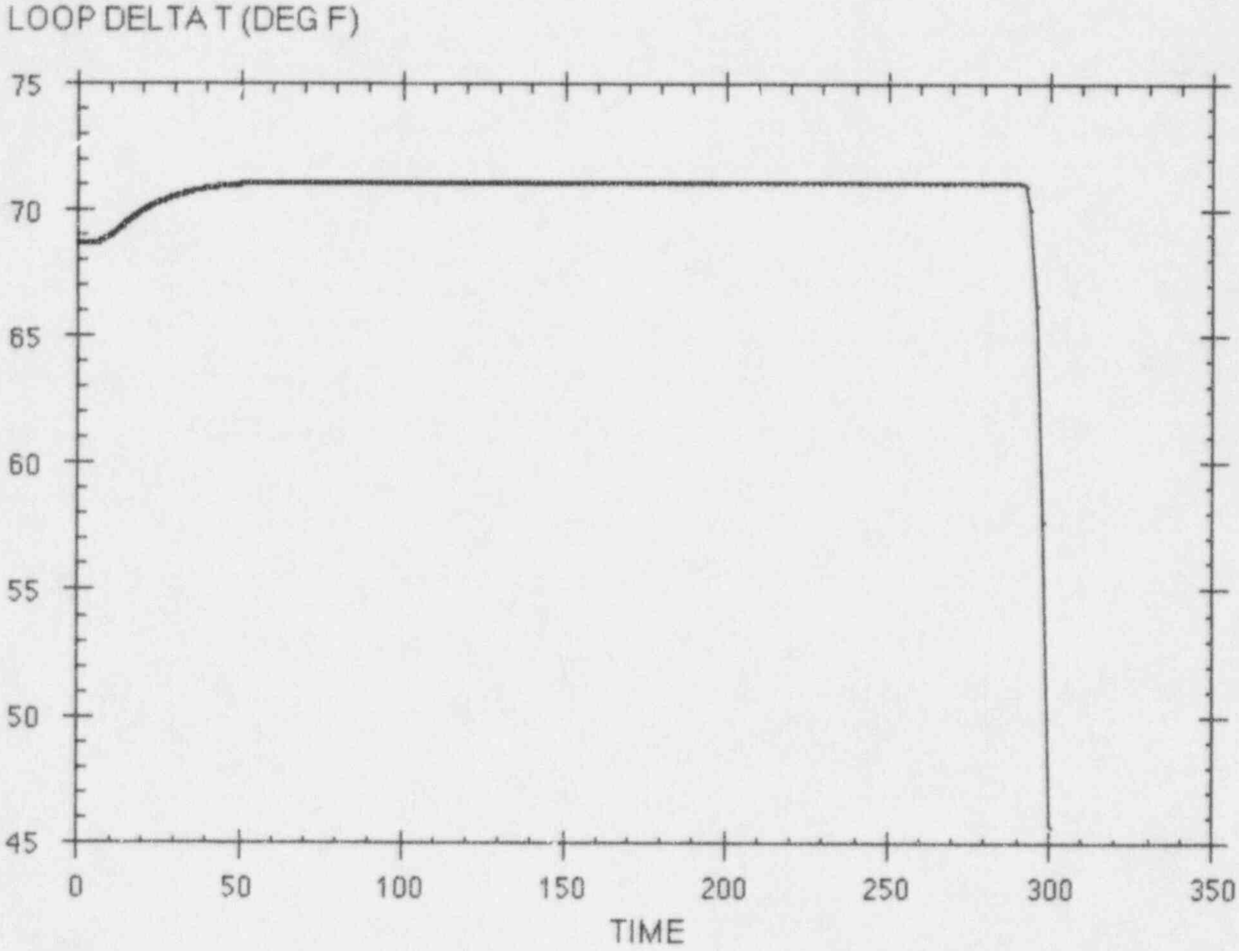
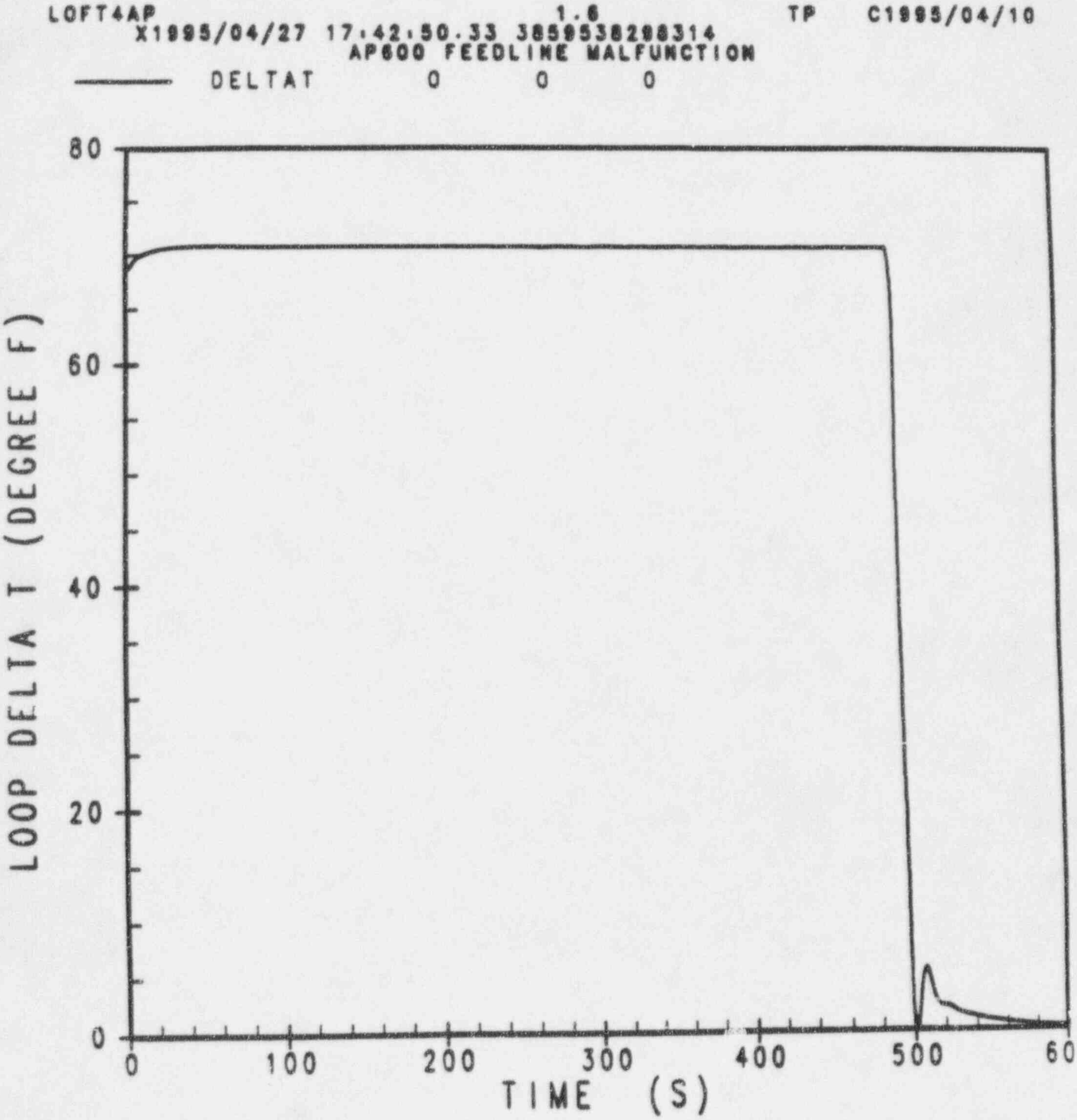


Figure 15.1.2-2

Feedwater Control Valve Malfunction Loop Delta T

even pg

15.1.2-2



LOFT4AP 1.6 TP C1995/04/10
X1995/04/27 17:42:50.33 3859538298314
AP600 FEEDLINE MALFUNCTION
DELTAT 0 0 0
LOOP DELTA T (DEGREE F)
80
60
40
20
0
0
100
200
300
400
500
60
TIME (S)

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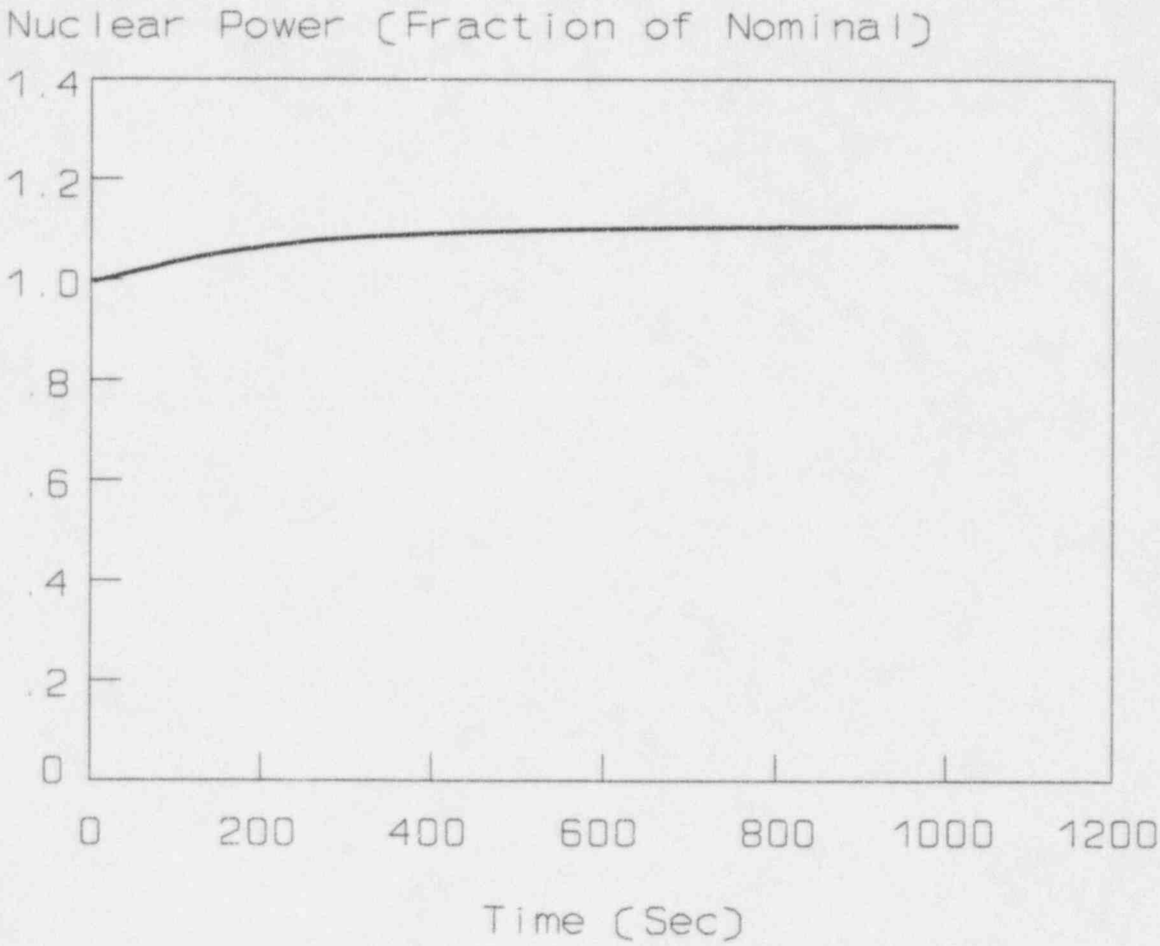

Figure 15.1.3-1

Nuclear Power (Fraction of Nominal) vs. Time for 10 Percent Step Load Increase, Manual Control and Minimum Moderator Feedback

15.1.3-)

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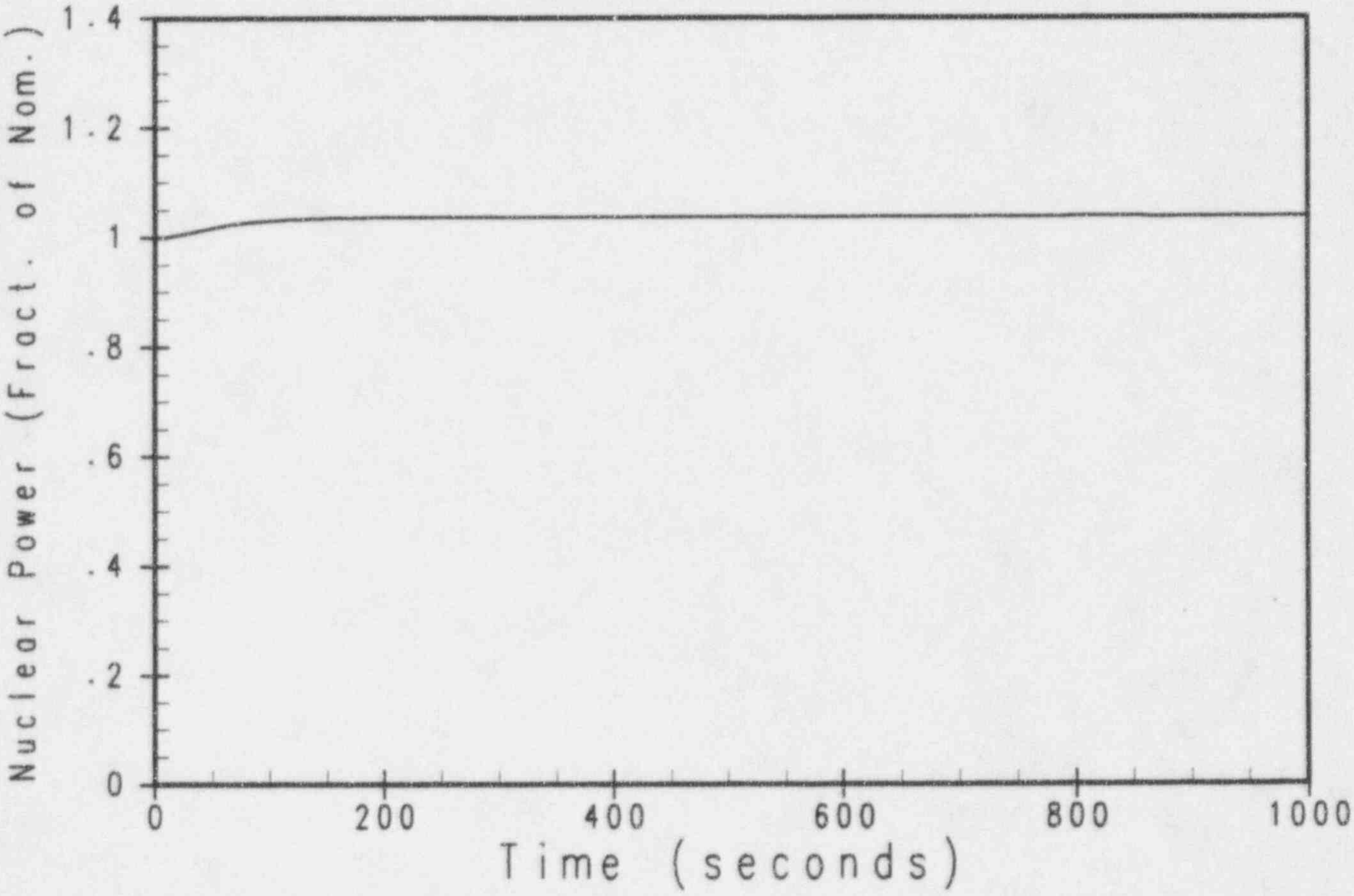
X1995/04/04 08:50:36.05 2983823717538
Case 1: Minimum Feedback (BOL) with Manual Rod Control



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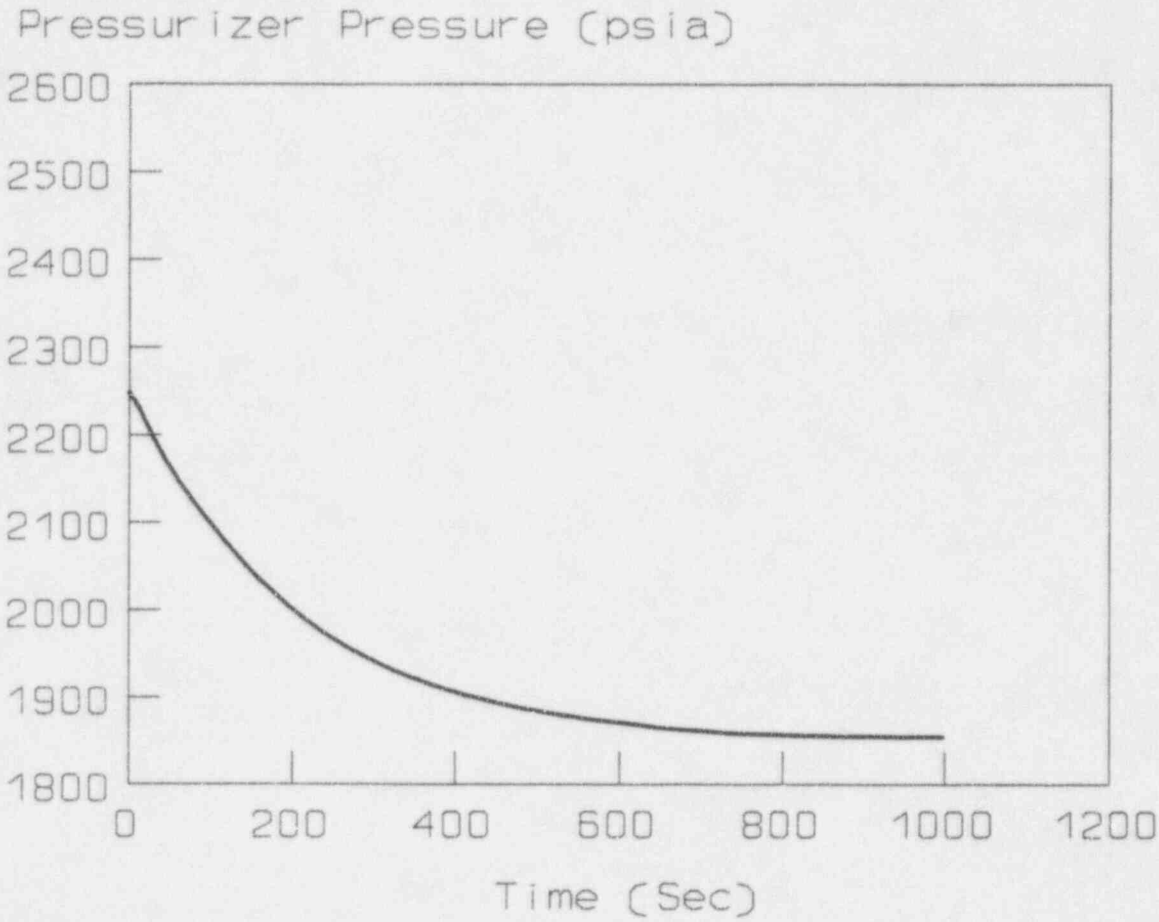
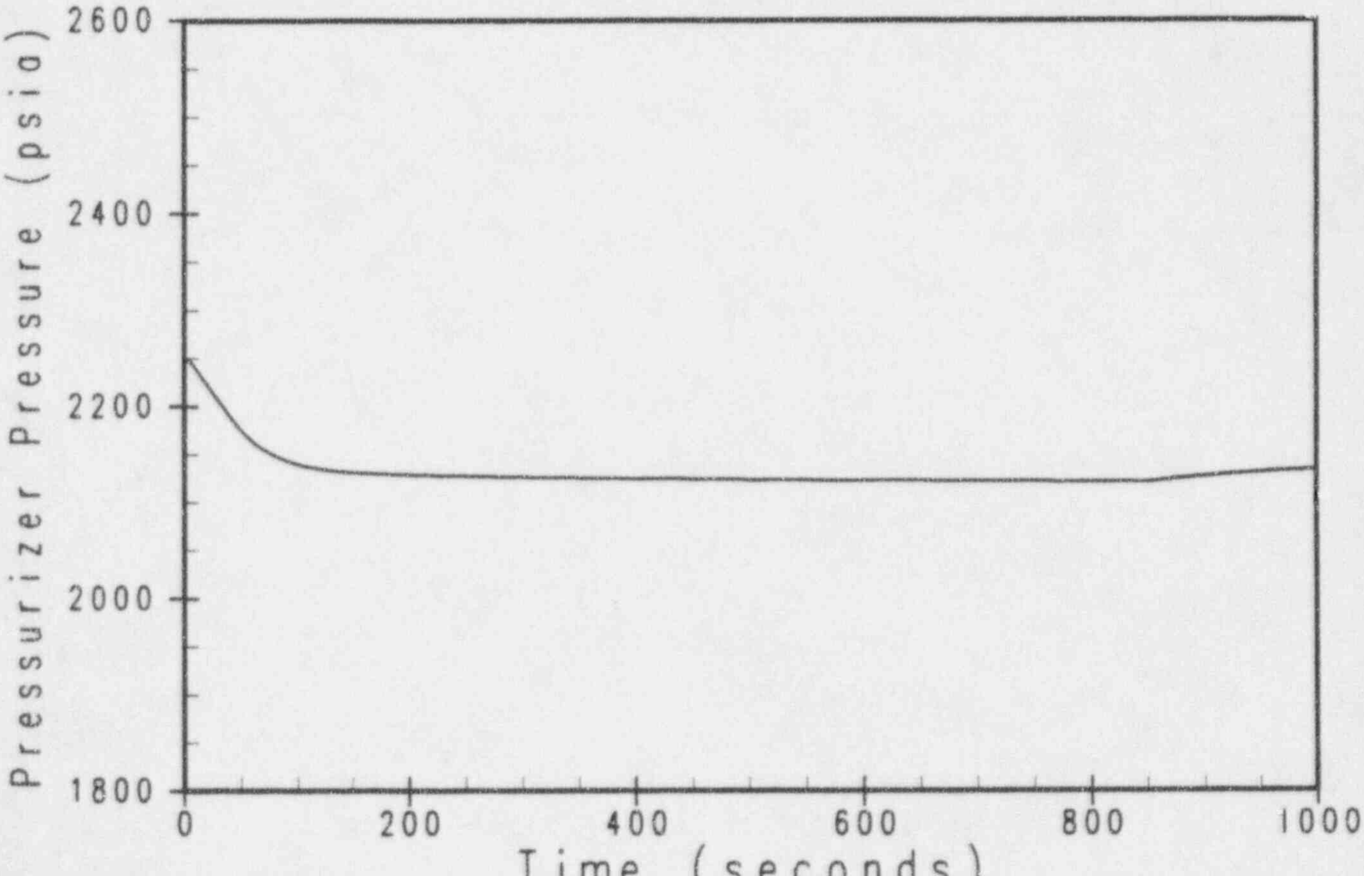


Figure 15.1.3-2

Pressurizer Pressure (psia) vs. Time for 10 Percent Step Load Increase, Manual Control and Minimum Moderator Feedback

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:50:36.05 2983823717538
Case 1: Minimum Feedback (BOL) with Manual Rod Control



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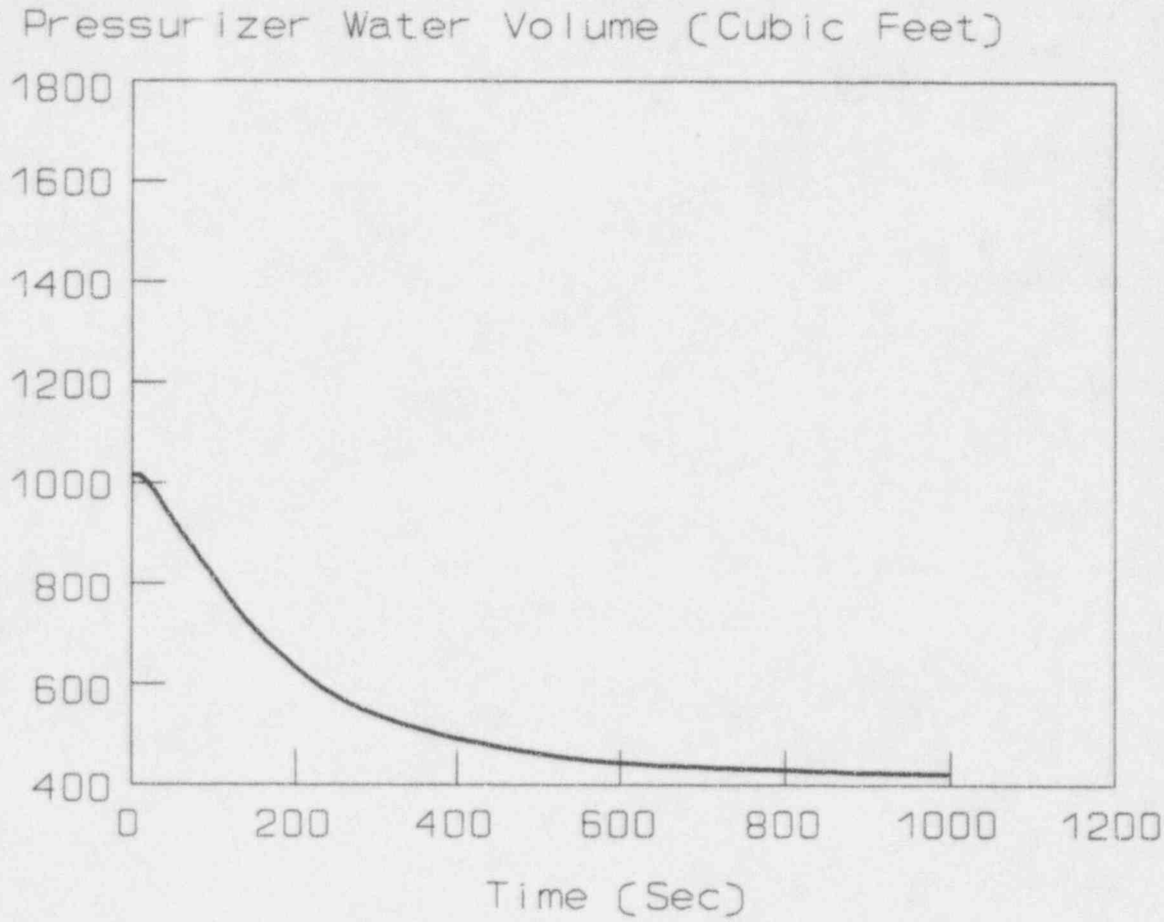
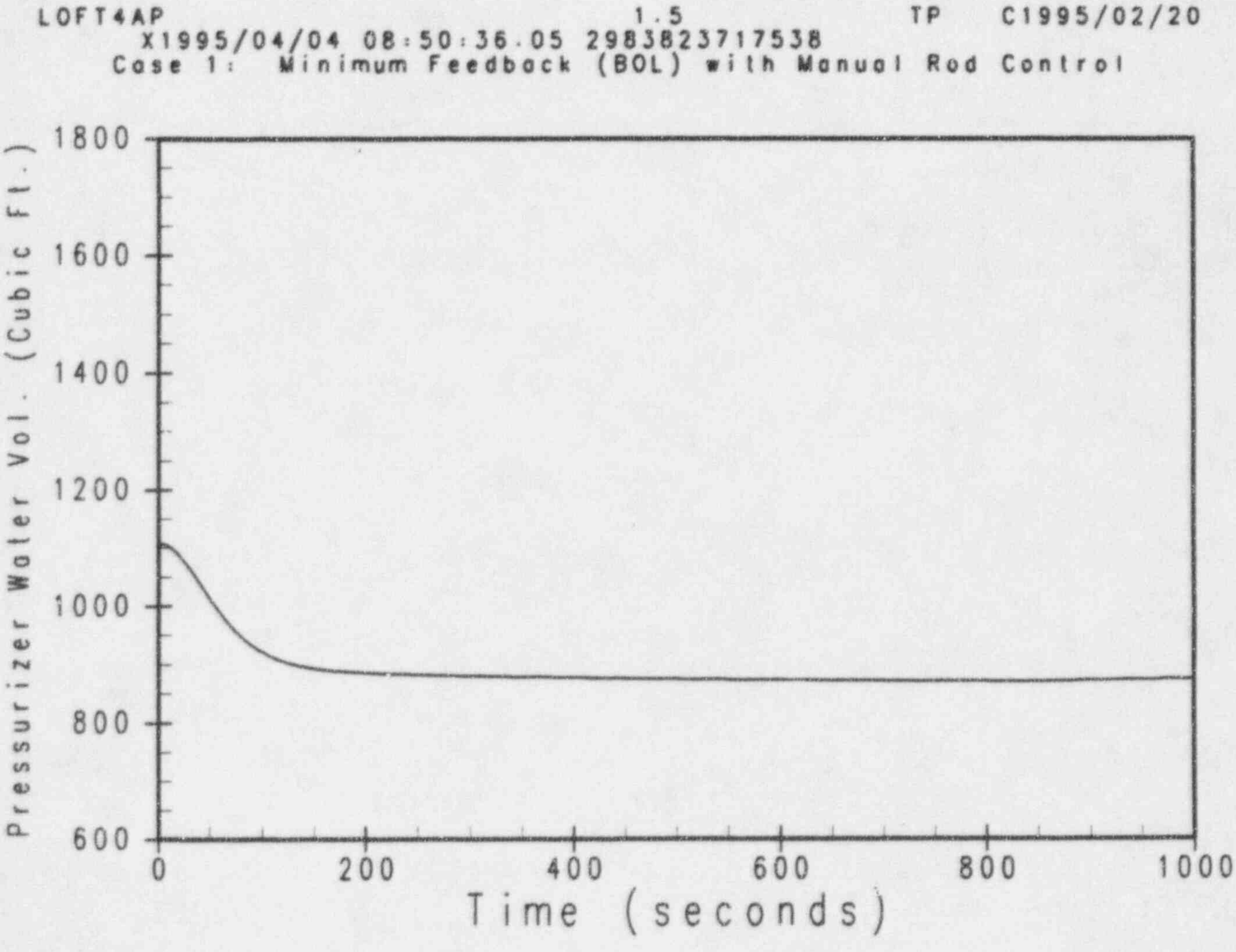


Figure 15.1.3-3

Pressurizer Water Volume (ft^3) vs. Time for 10 Percent Step Load Increase, Manual Control and Minimum Moderator Feedback

even pg


LOFT4AP
1.5
TP
C1995/02/20
X1995/04/04 08:50:36.05 2983823717538
Case 1: Minimum Feedback (BOL) with Manual Rod Control
Pressurizer Water Vol. (Cubic Ft.)
1800
1600
1400
1200
1000
800
600
0
200
400
600
800
1000
Time (seconds)

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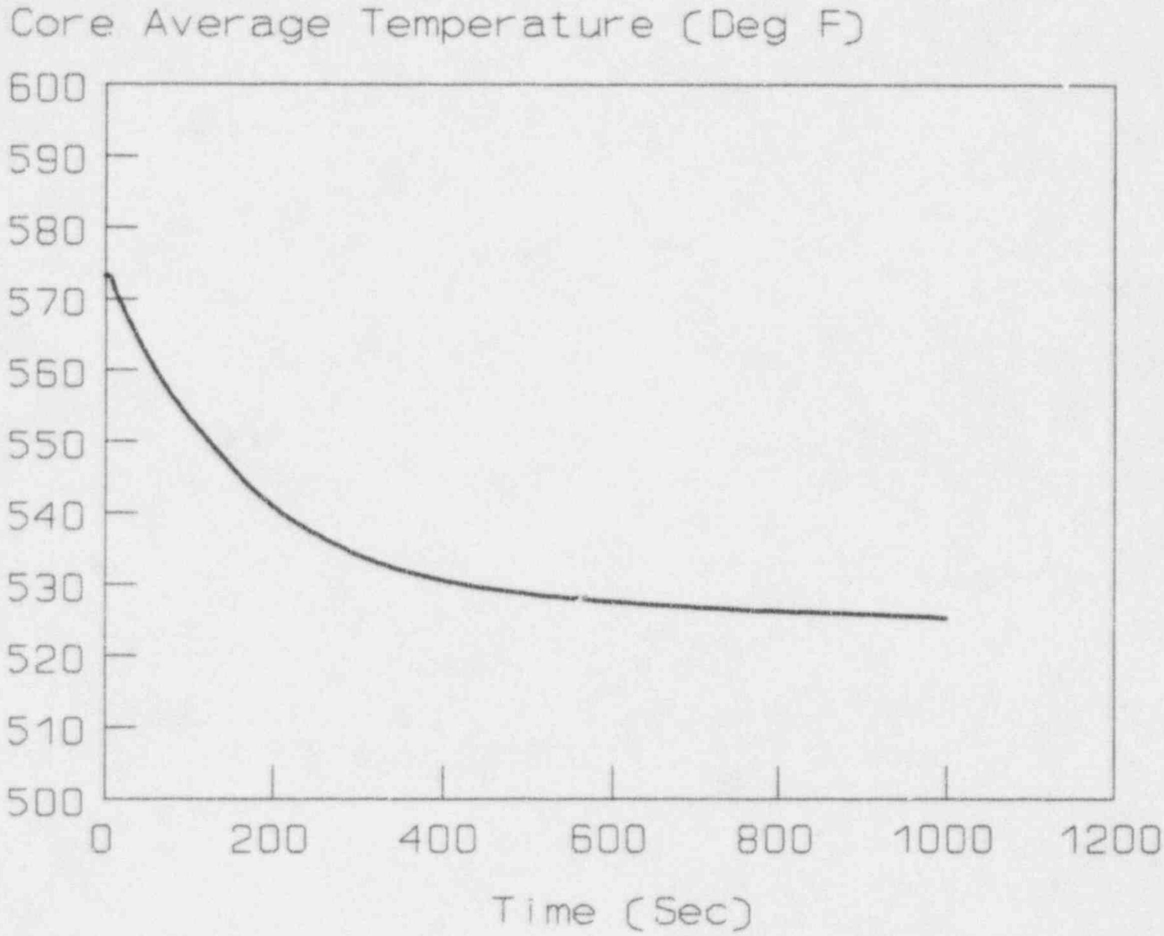


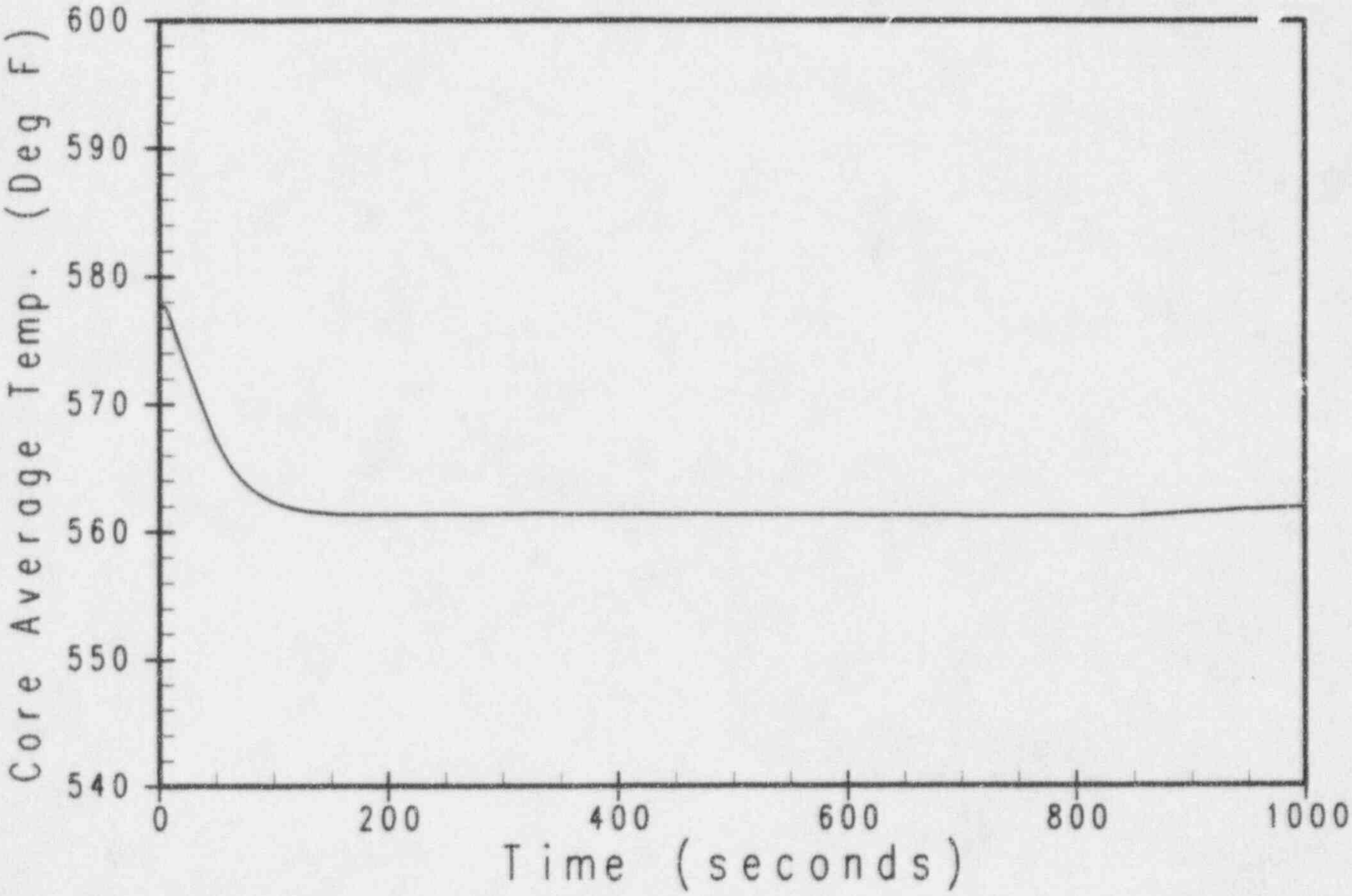
Figure 15.1.3-4

Core Average Temperature (°F) vs. Time for 10 Percent Step Load Increase, Manual Control and Minimum Moderator Feedback

even pg

15.1.3-4

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:50:36.05 2983823717538
Case 1: Minimum Feedback (BOL) with Manual Rod Control



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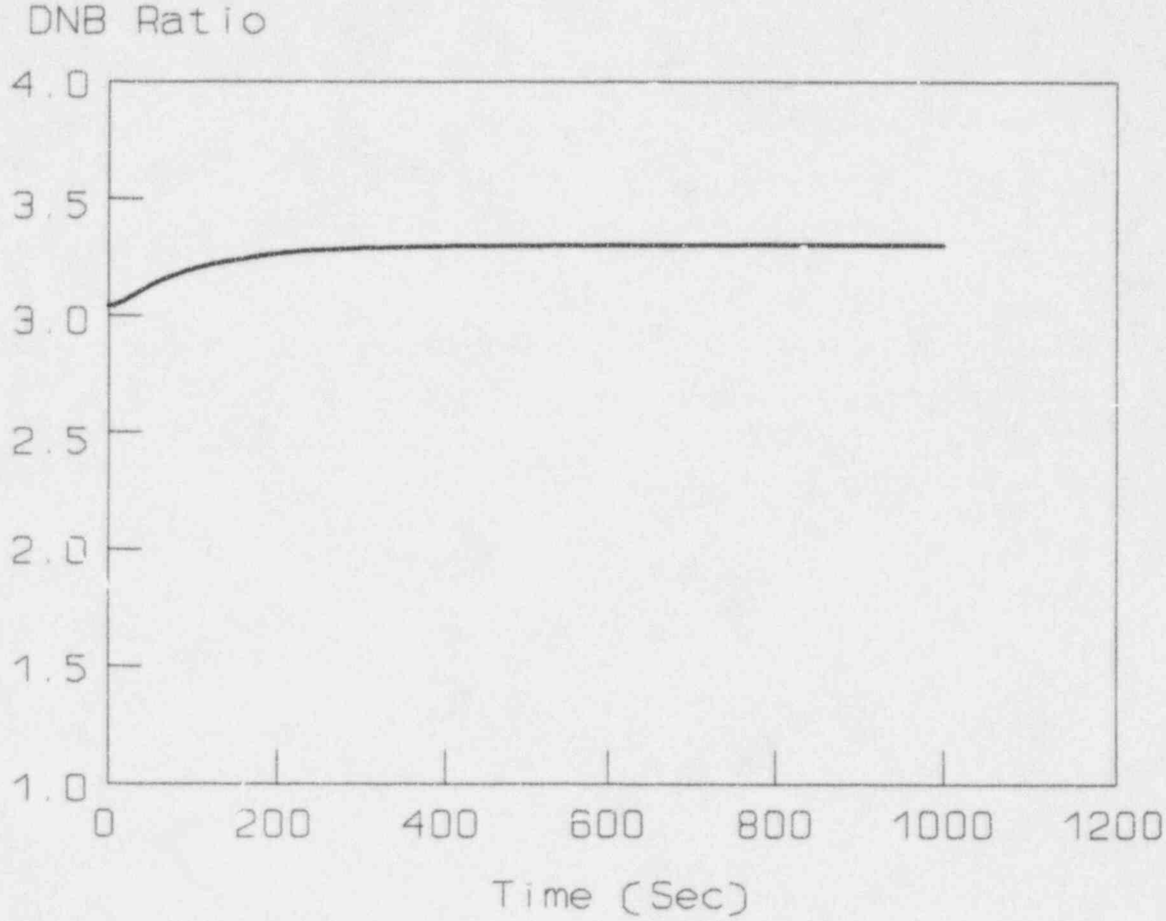
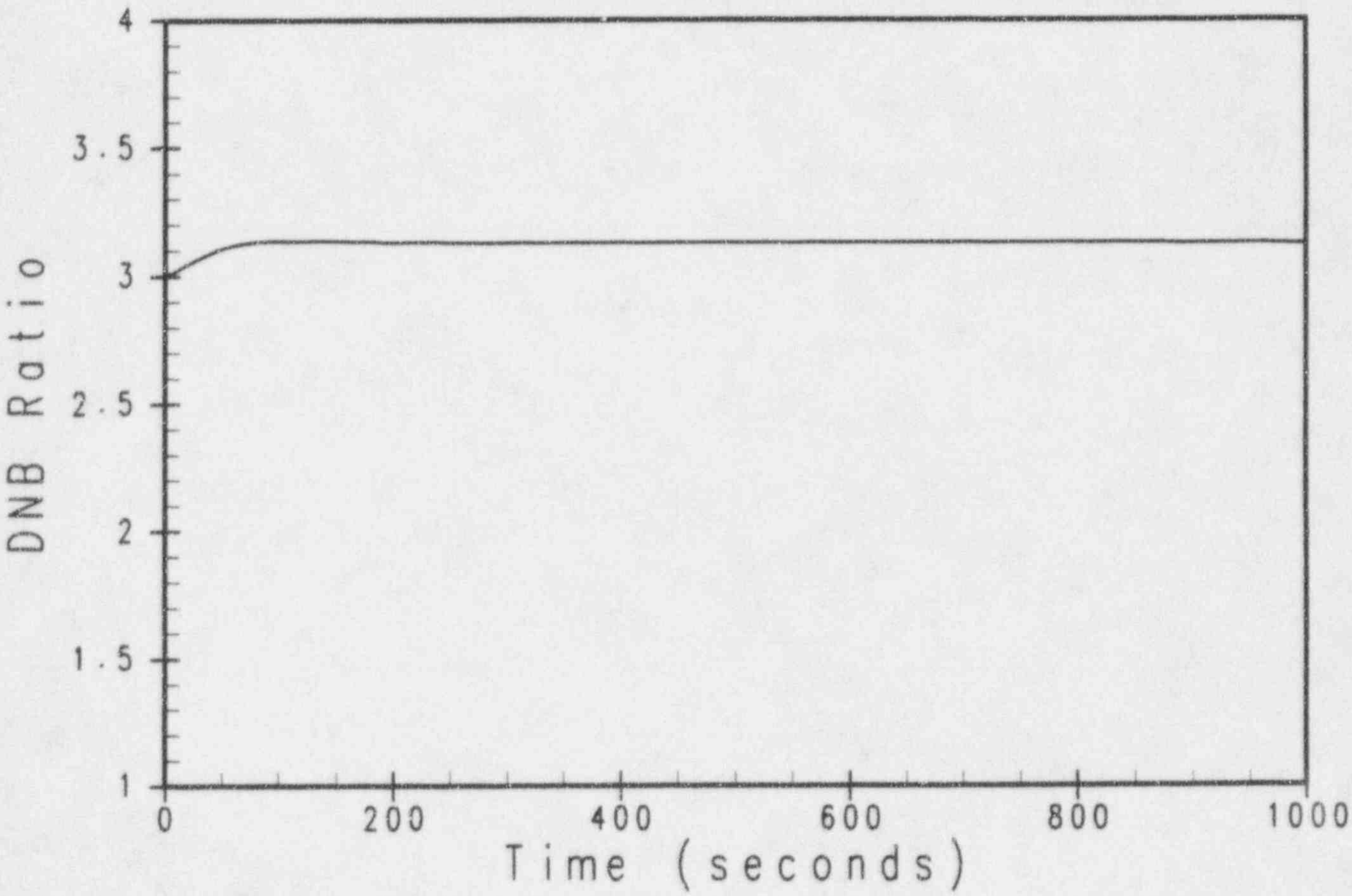


Figure 15.1.3-5

DNB Ratio vs. Time for 10 Percent Step Load Increase, Manual Control and Minimum Moderator Feedback

even pg

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:50:36.05 2983823717538
Case 1: Minimum Feedback (BOL) with Manual Rod Control



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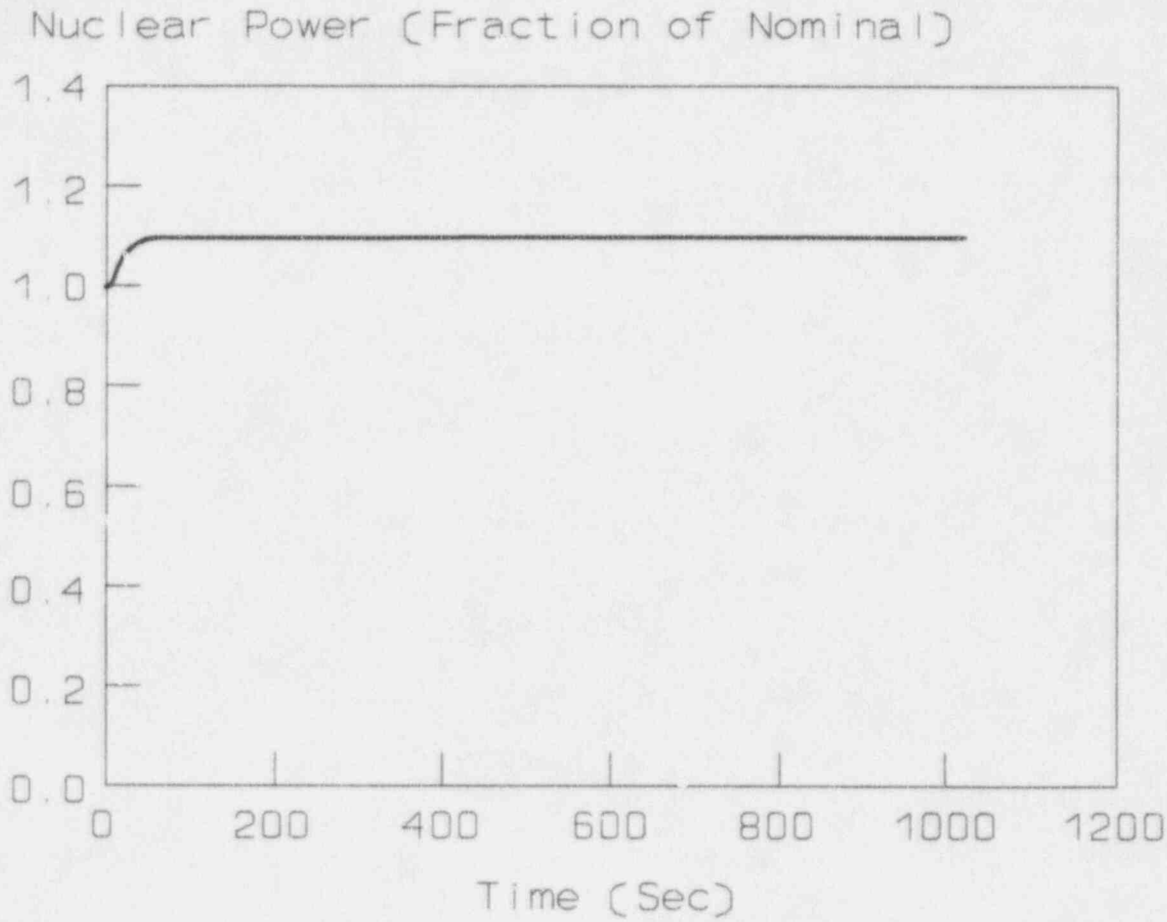
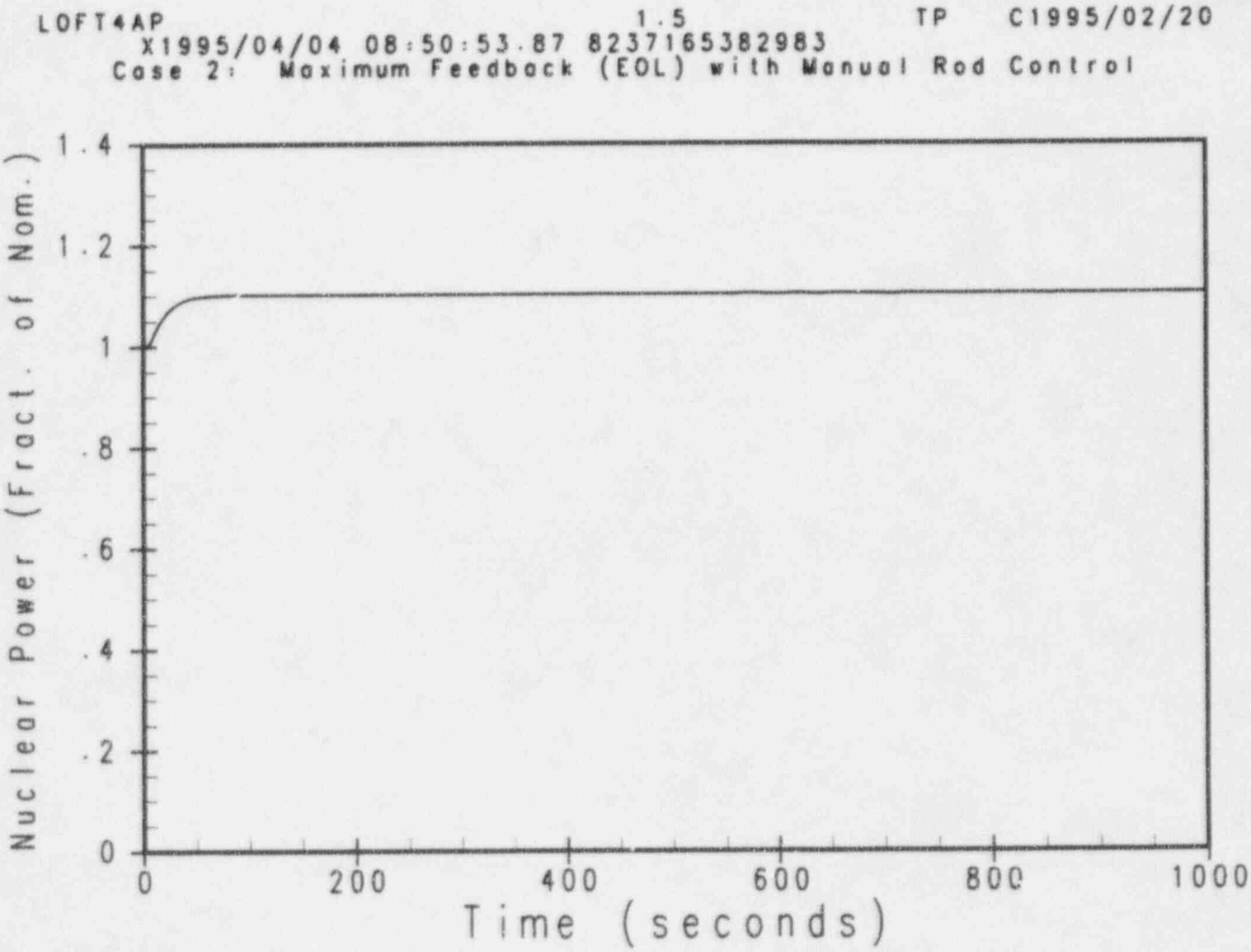


Figure 15.1.3-6

Nuclear Power (Fraction of Nominal) vs. Time for 10 Percent Step Load Increase, Manual Control and Maximum Moderator Feedback

even pg


LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:50:53.87 8237165382983
Case 2: Maximum Feedback (EOL) with Manual Rod Control
Nuclear Power (Fract. of Nom.)
1.4
1.2
1
.8
.6
.4
.2
0
0
200
400
600
800
1000
Time (seconds)

Replace (next page)

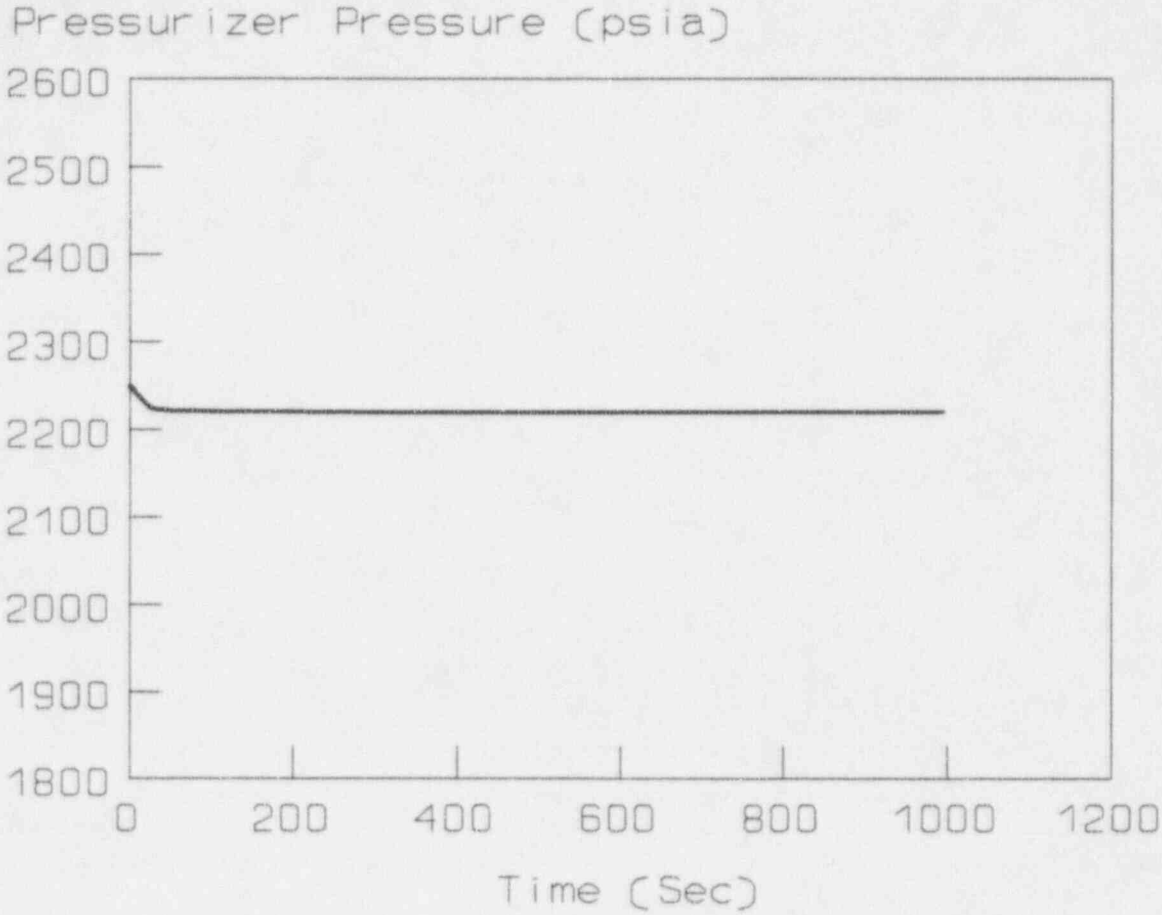


Figure 15.1.3-7

Pressurizer Pressure (psia) vs. Time for 10 Percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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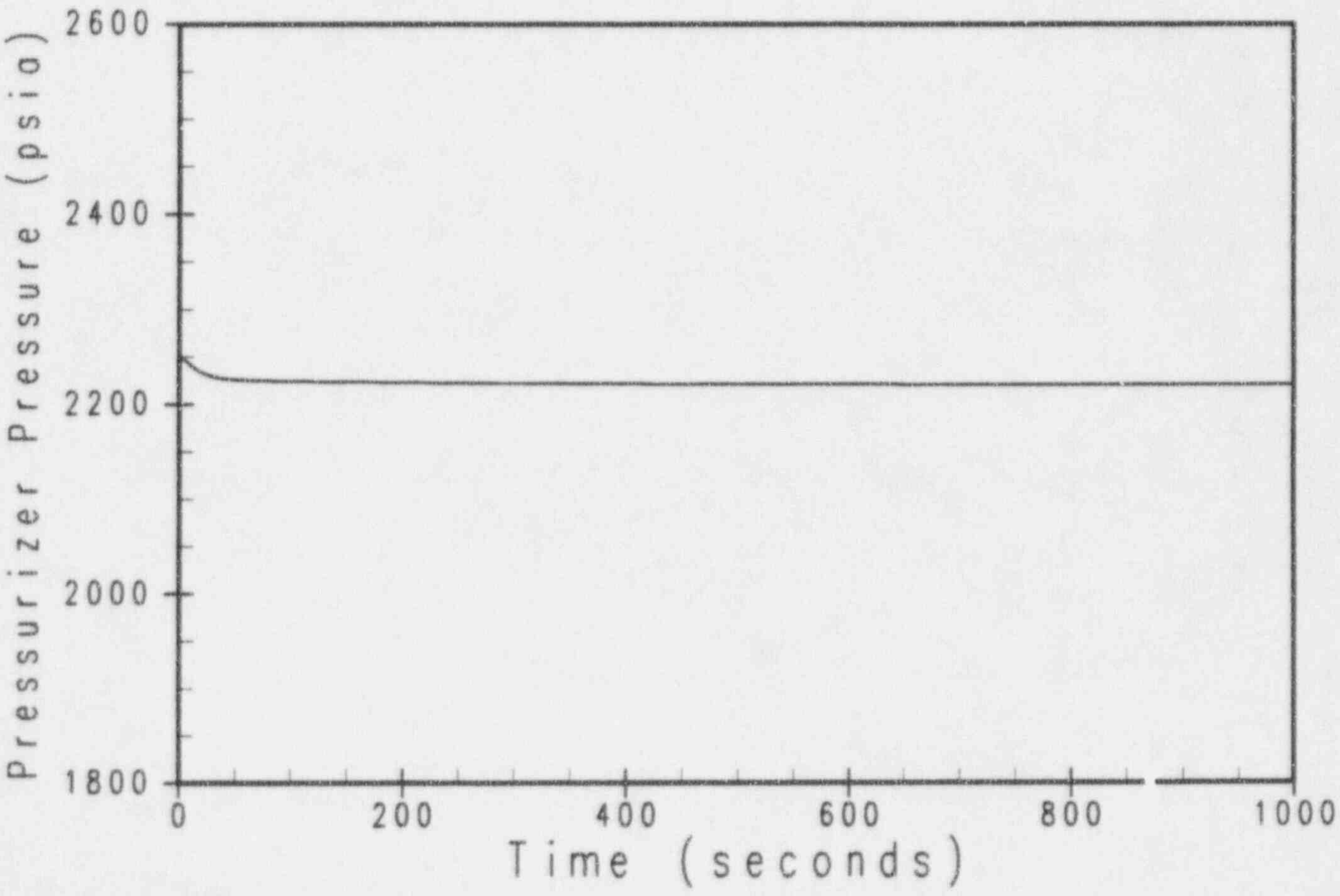
X1995/04/04 08:50:53.87 8237165382983
Case 2: Maximum Feedback (EOL) with Manual Rod Control



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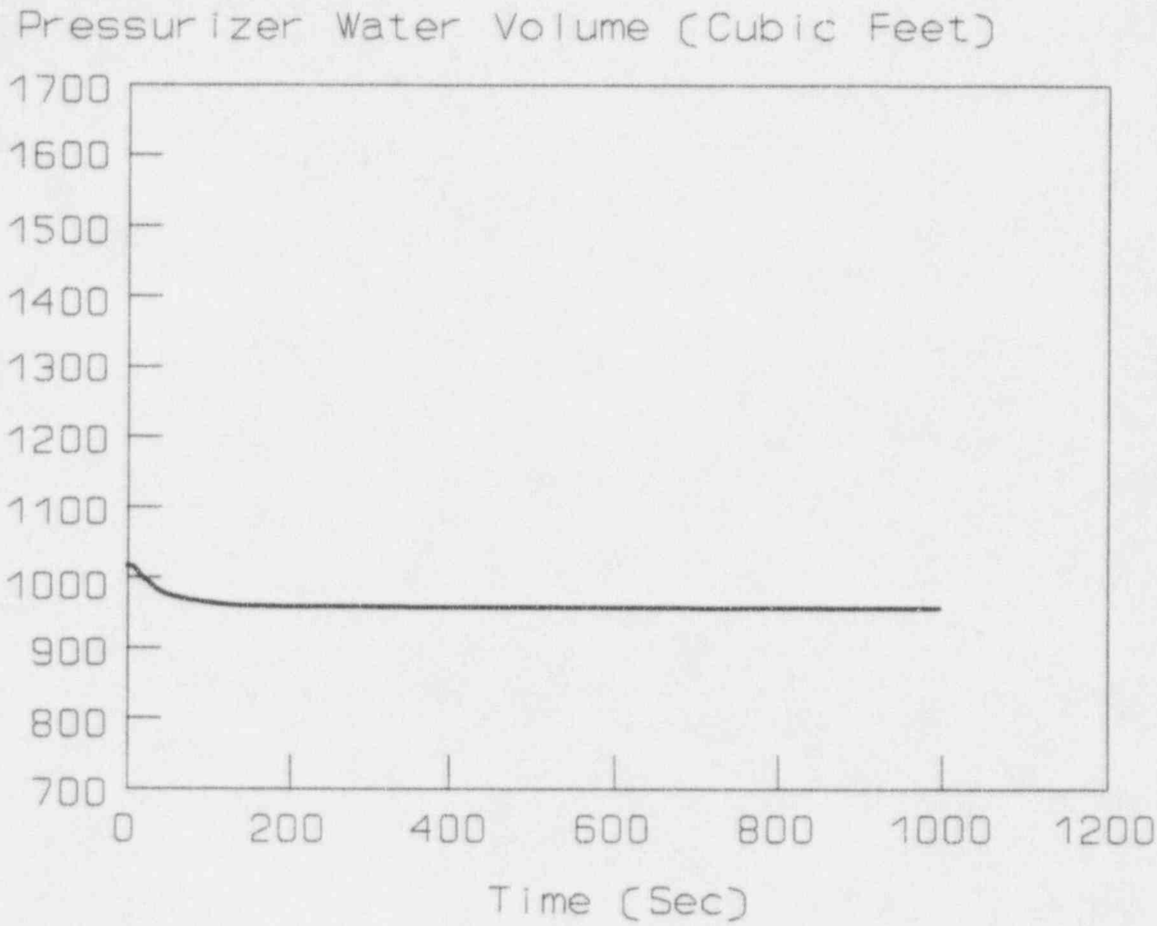
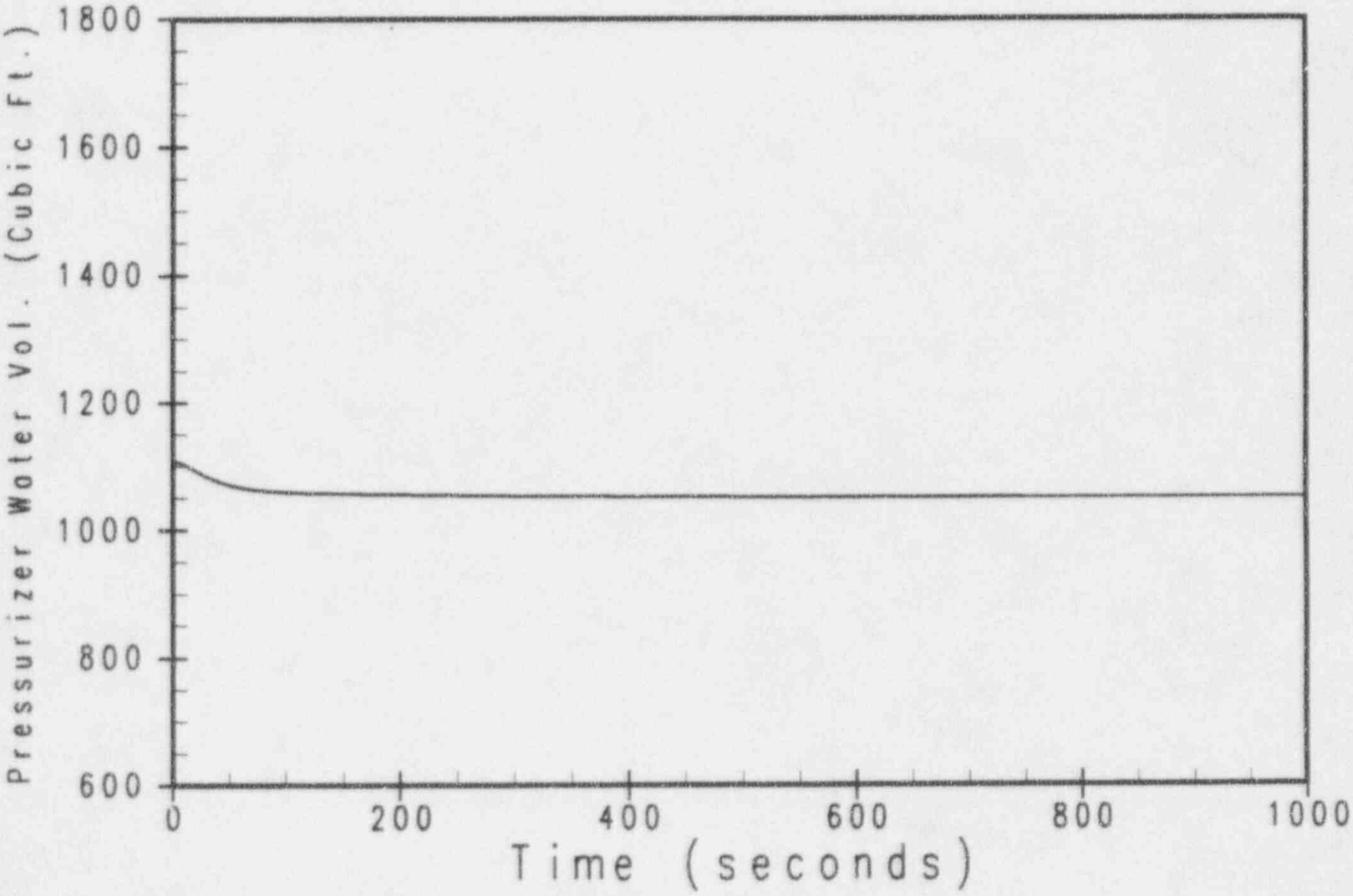


Figure 15.1.3-8

Pressurizer Water Volume (ft^3) vs. Time for 10 Percent Step Load Increase, Manual Control and Maximum Moderator Feedback

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:50:53.87 8237165382983
Case 2: Maximum Feedback (EOL) with Manual Rod Control



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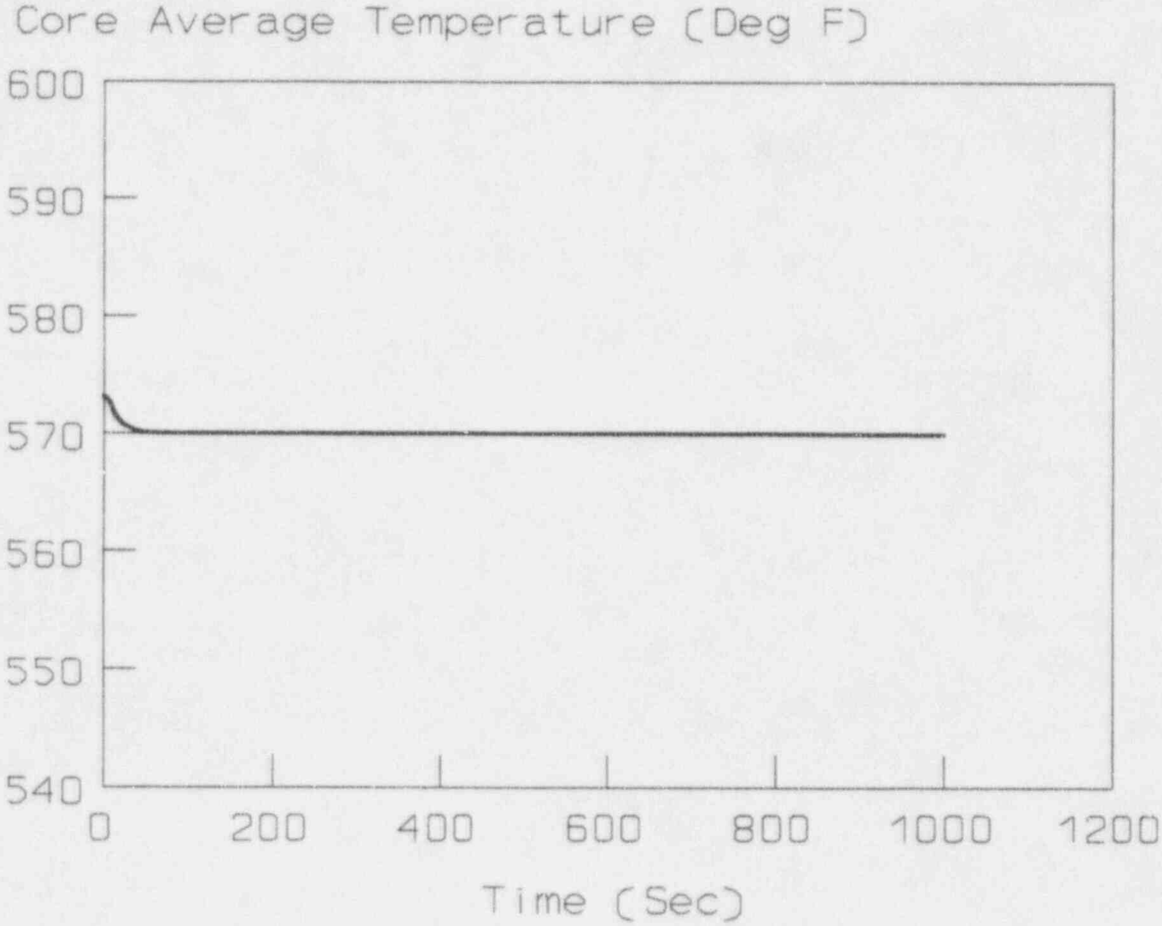

Figure 15.1.3-9

Core Average Temperature (°F) vs. Time for 10 Percent Step Load Increase, Manual Control and Maximum Moderator Feedback

even pg

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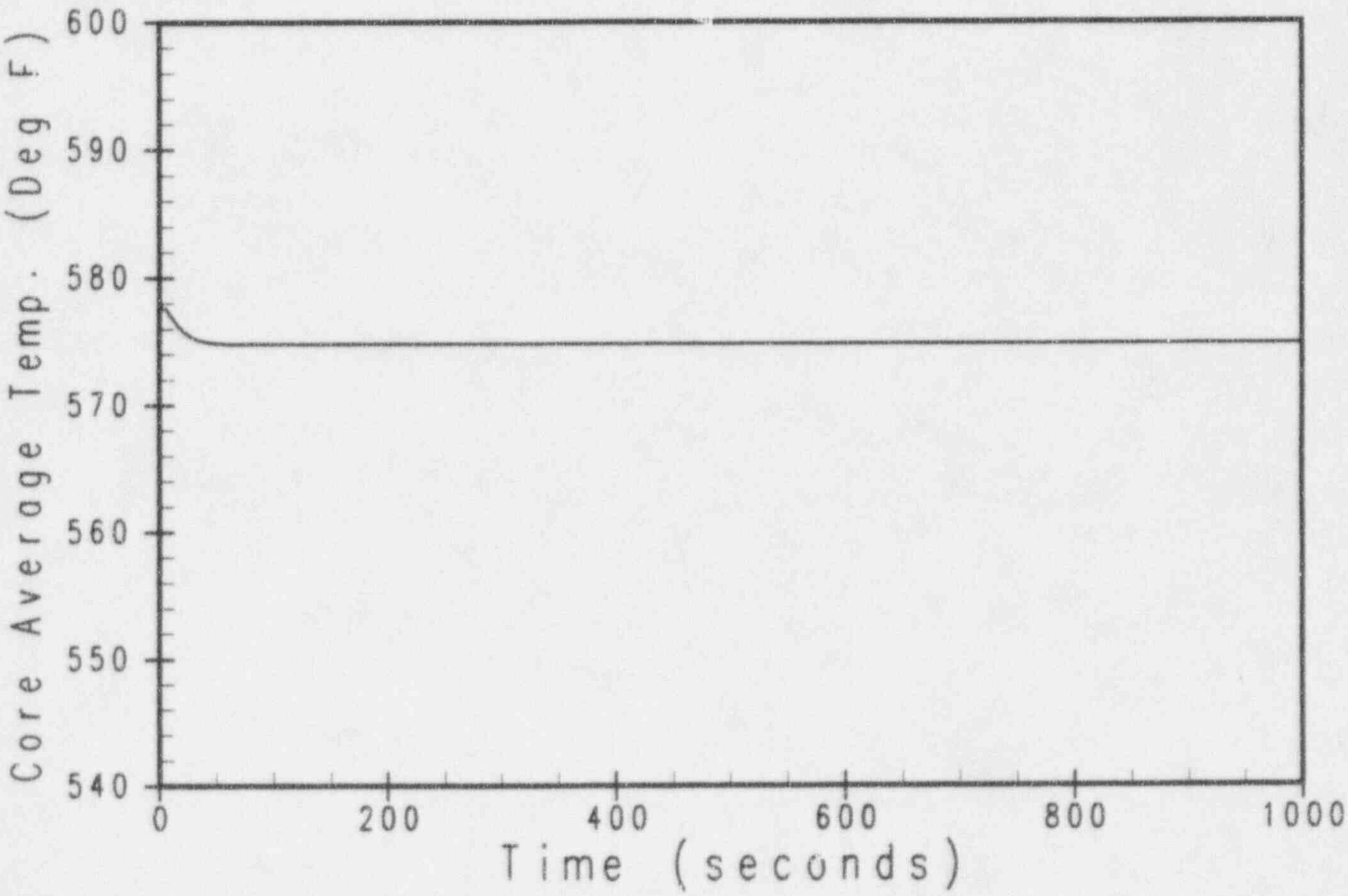
X1995/04/04 08:50:53.87 8237165382983
Case 2: Maximum Feedback (EOL) with Manual Rod Control



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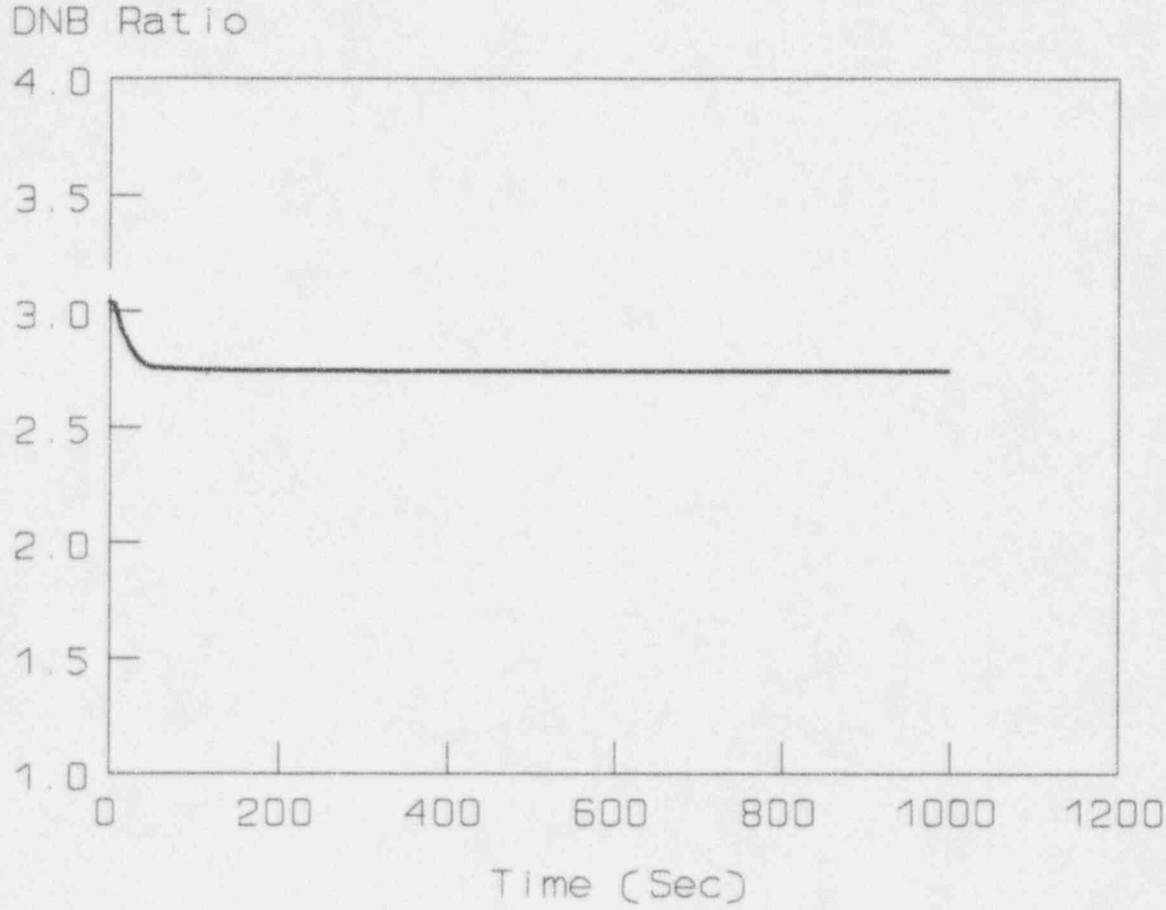
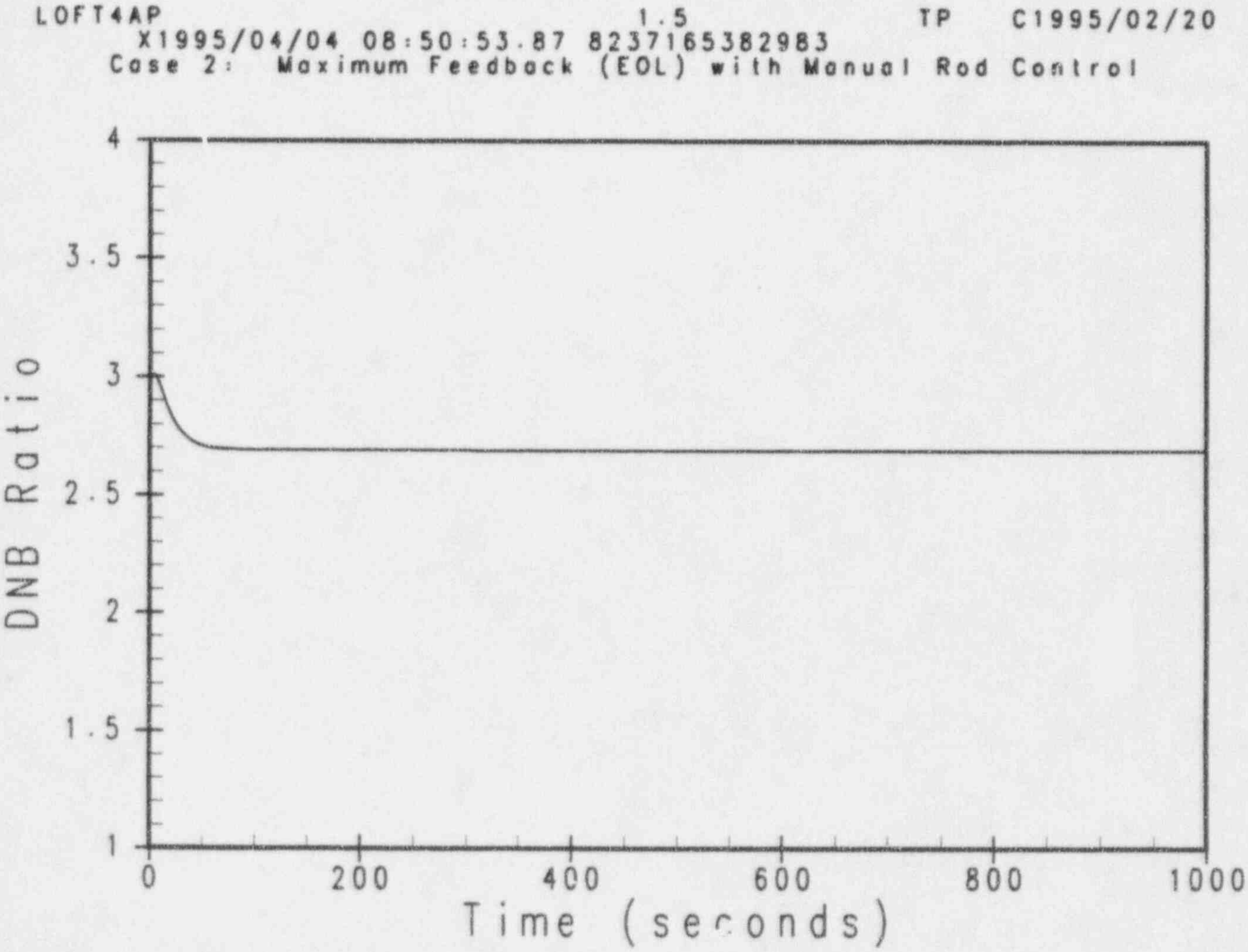


Figure 15.1.3-10

DNB Ratio vs. Time for 10 Percent Step Load Increase, Manual Control and Maximum Moderator Feedback

even pg


LOFT4AP
1.5
TP C1995/02/20
X1995/04/04 08:50:53.87 8237165382983
Case 2: Maximum Feedback (EOL) with Manual Rod Control
DNB Ratio
4
3.5
3
2.5
2
1.5
1
0
200
400
600
800
1000
Time (seconds)

Replace
(next page)

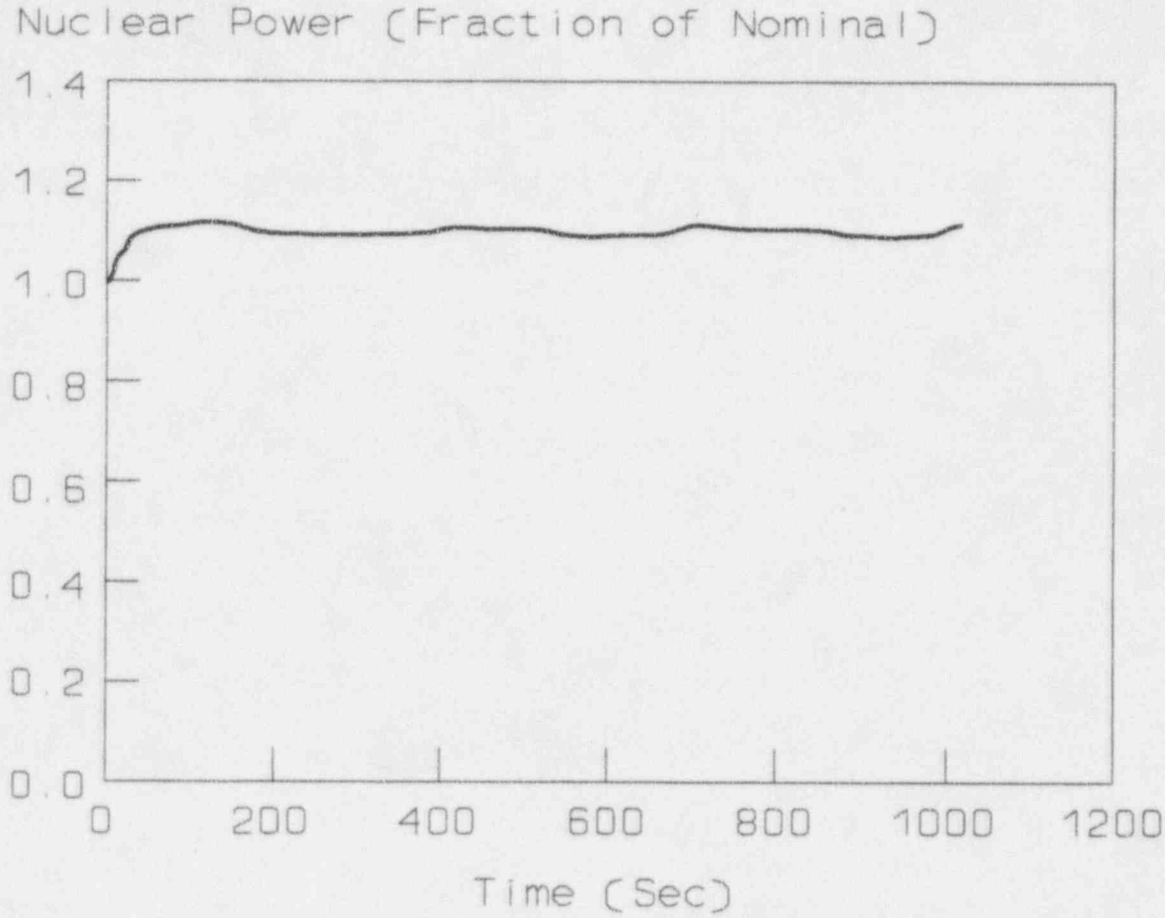


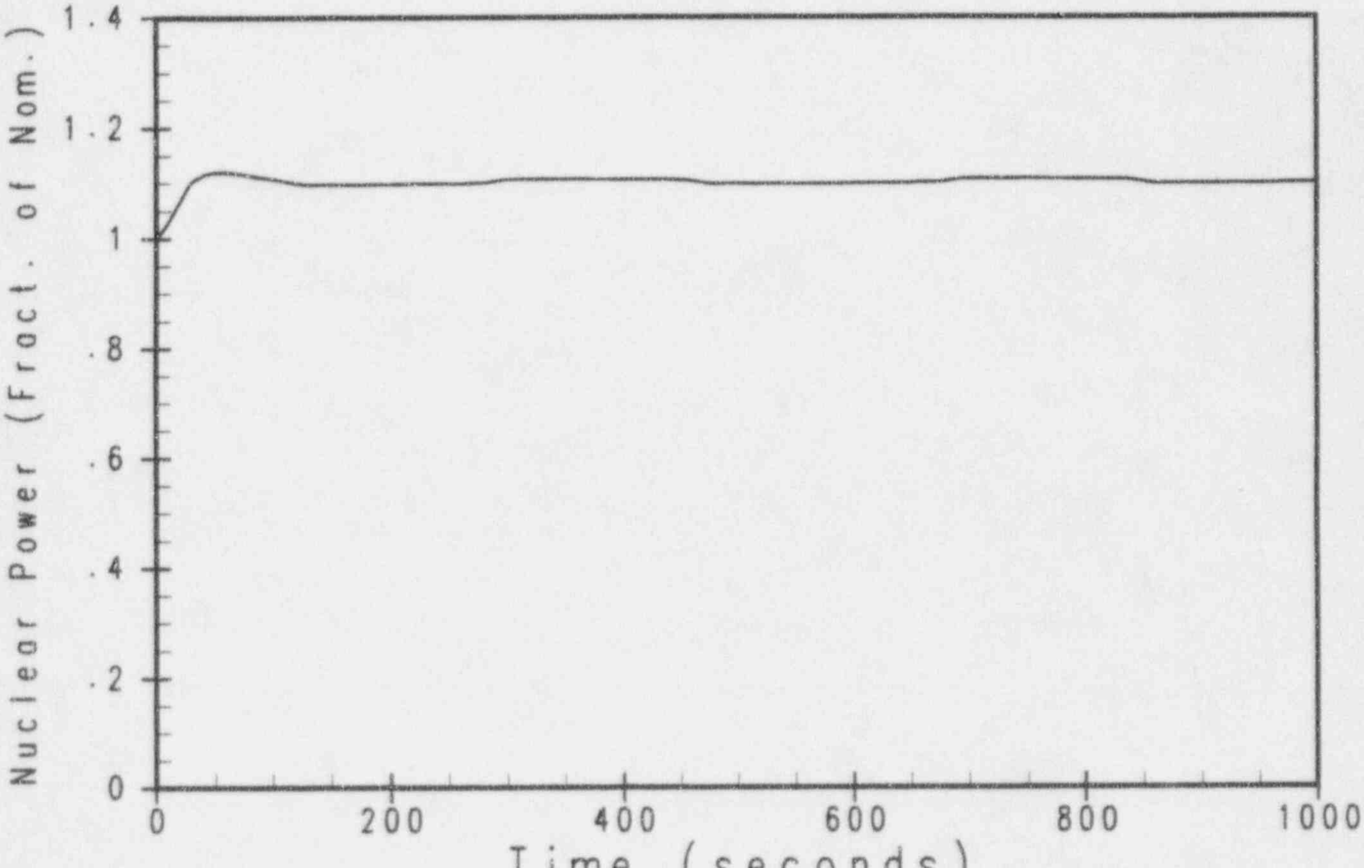
Figure 15.1.3-11

Nuclear Power (Fraction of Nominal) vs. Time for 10 Percent Step Load Increase, Automatic Contro. and Minimum Moderator Feedback

even pg

15.1.3-11

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:51:08.63 2983823847538
Case 3: Minimum Feedback (BOL) with Automatic Rod Control



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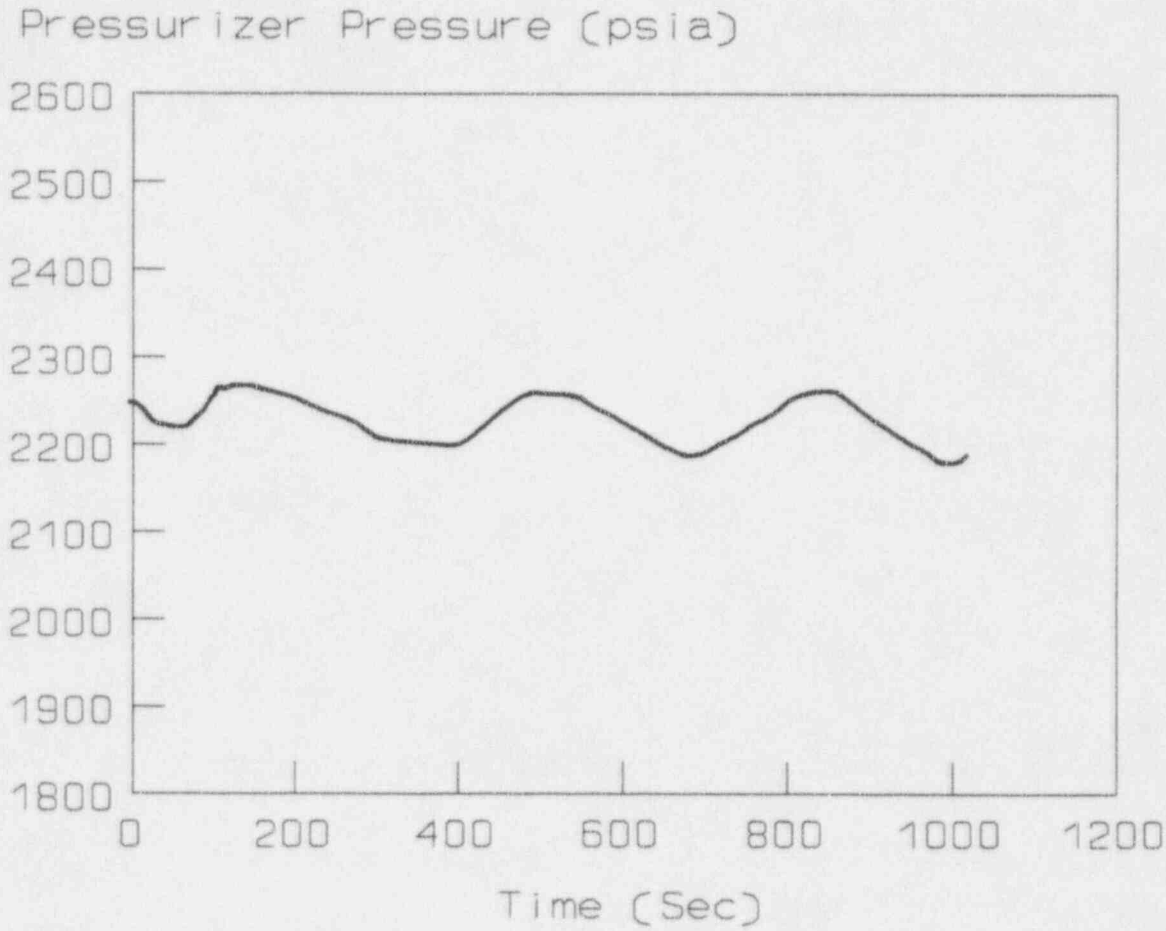
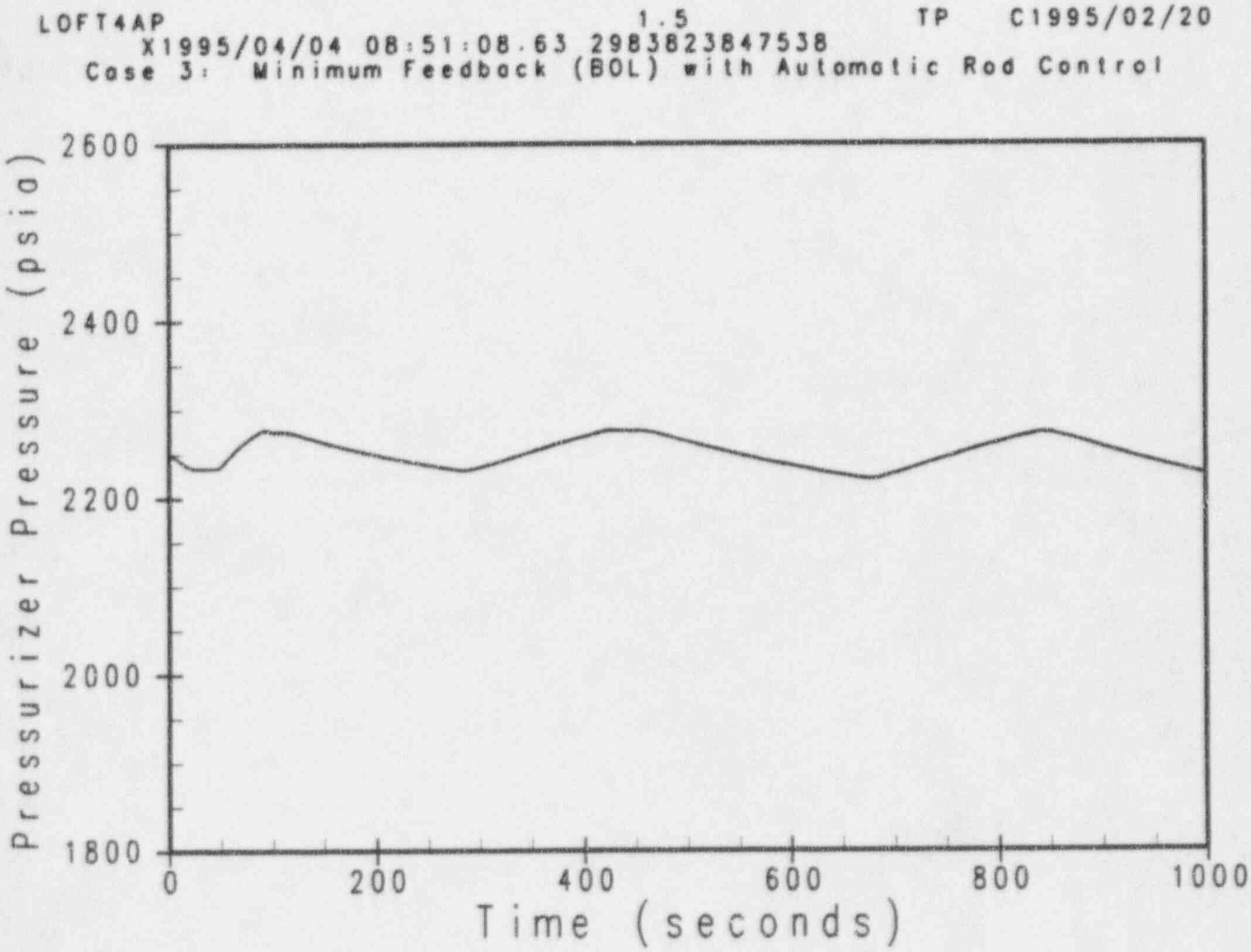


Figure 15.1.3-12

Pressurizer Pressure (psia) vs. Time for 10 Percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

even pg


LOFT4AP
1.5
TP
C1995/02/20
X1995/04/04 08:51:08.63 2983823847538
Case 3: Minimum Feedback (BOL) with Automatic Rod Control
Pressurizer Pressure (psia)
2600
2400
2200
2000
1800
0
200
400
600
800
1000
Time (seconds)

Replace (next page)

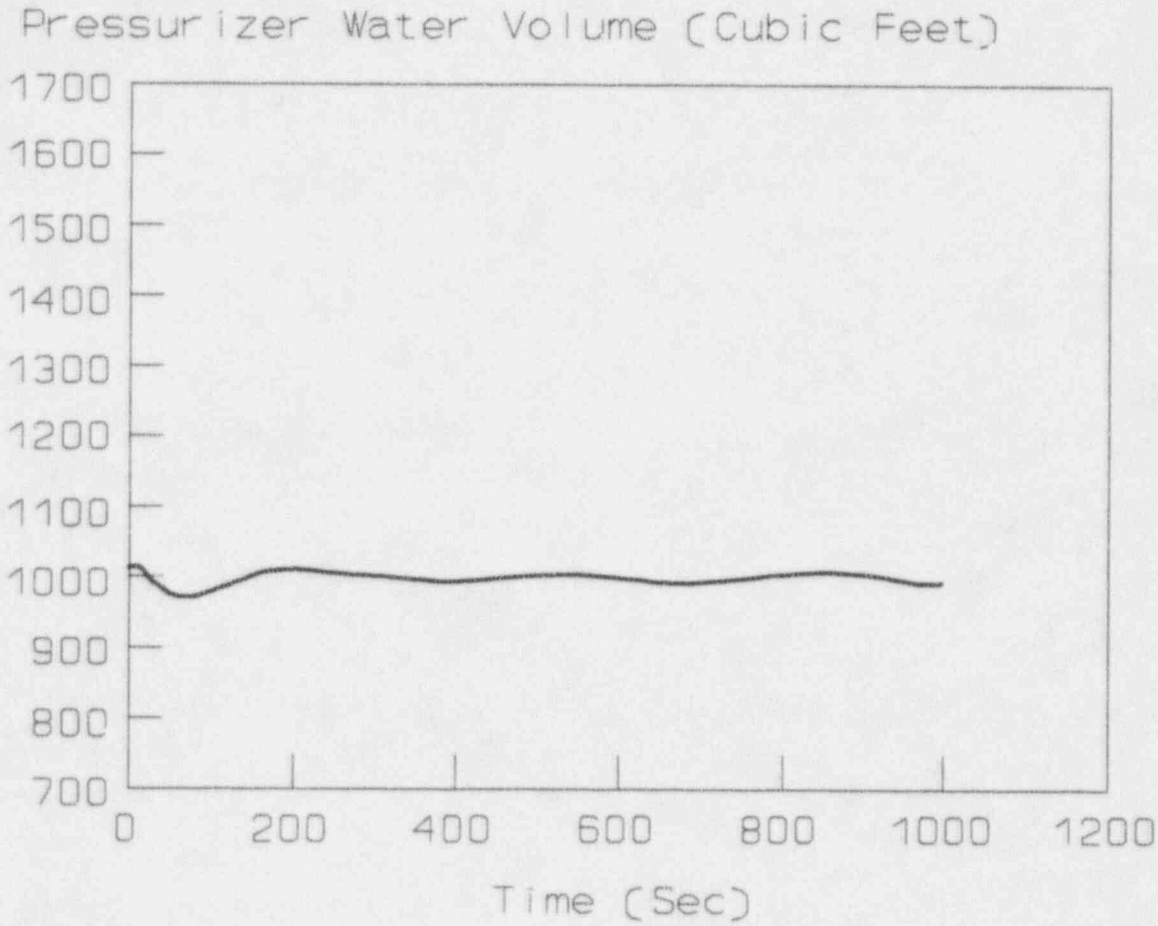


Figure 15.1.3-13

Pressurizer Water Volume (ft^3) vs. Time for 10 Percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

even pg

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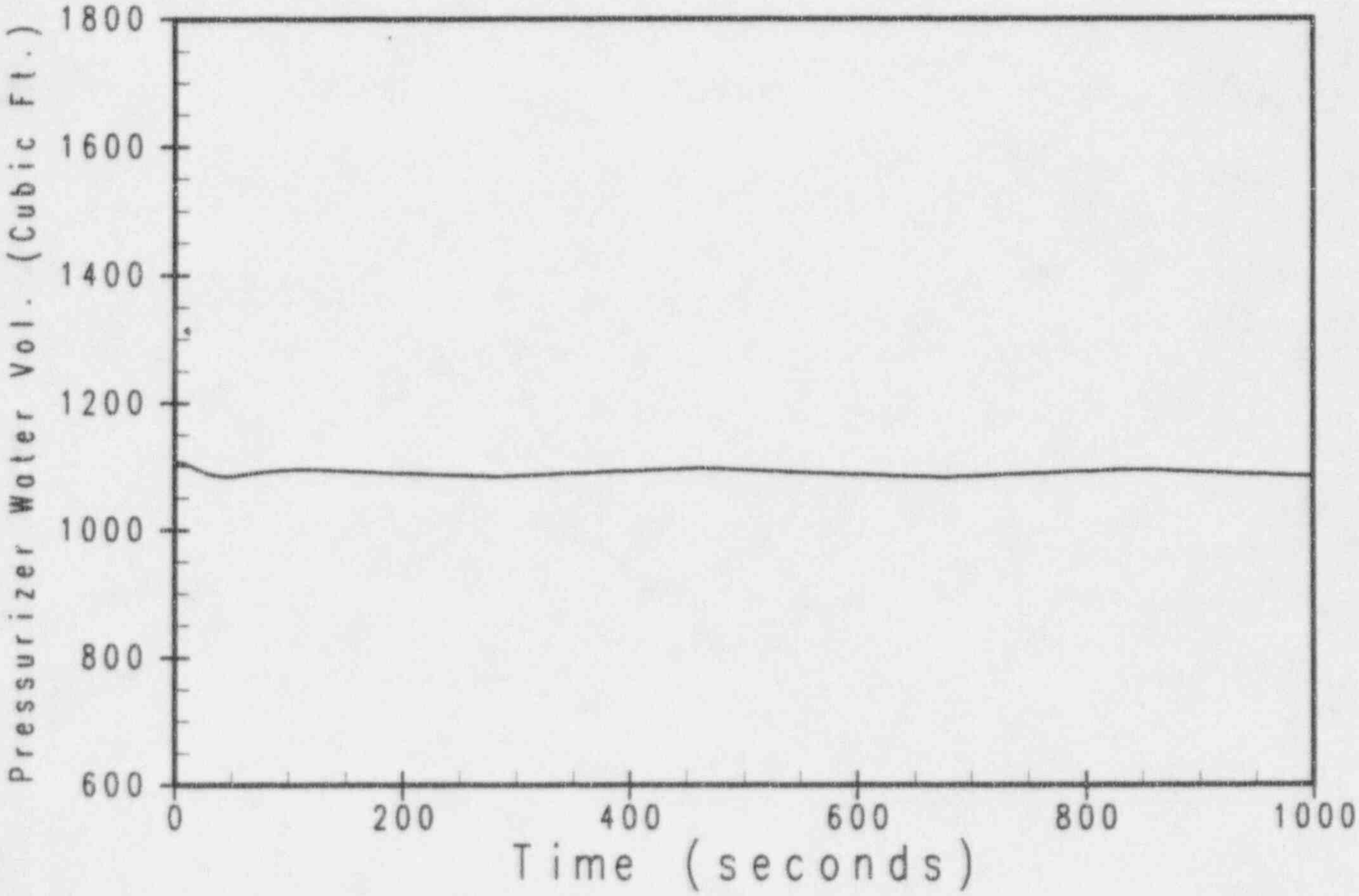
X1995/04/04 08:51:08.63 2983823847538
Case 3: Minimum Feedback (BOL) with Automatic Rod Control



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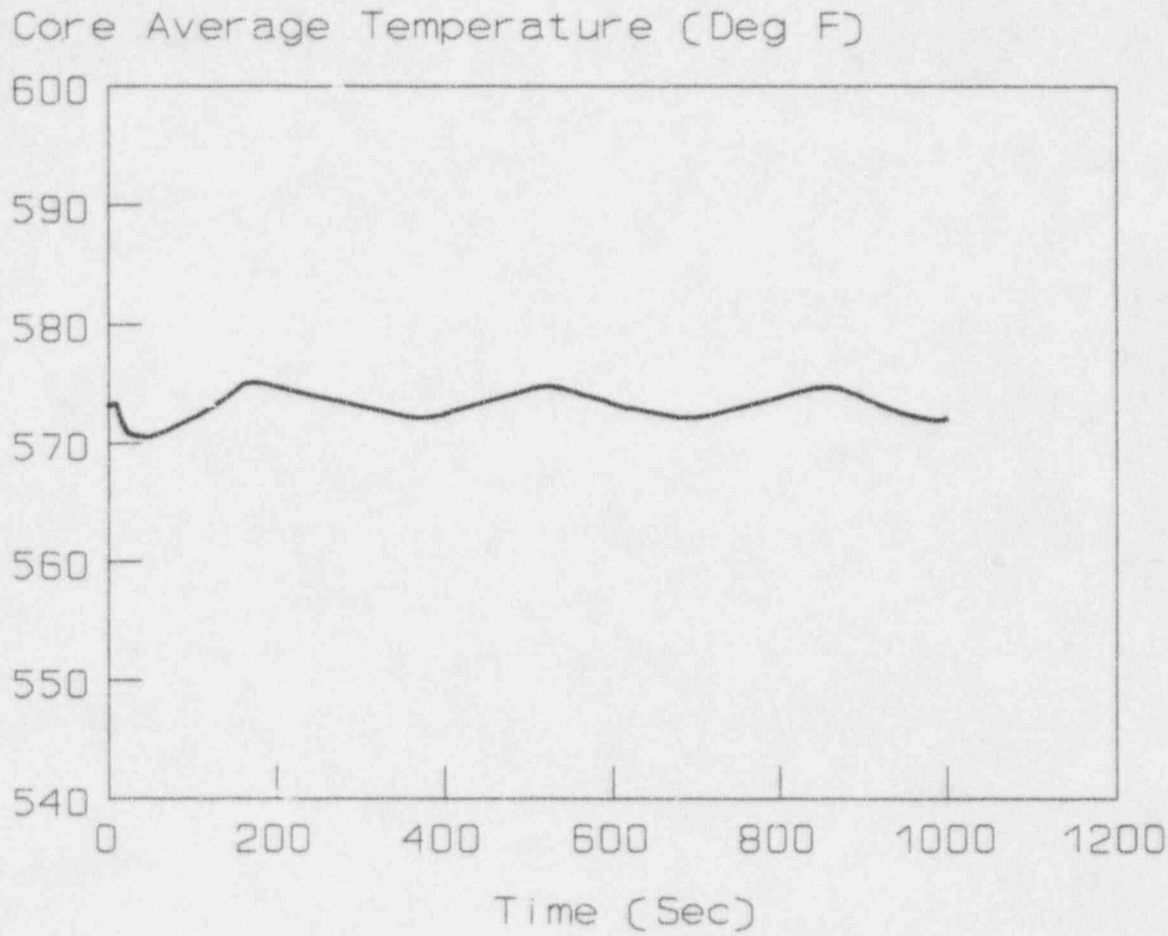

Figure 15.1.3-14

Core Average Temperature (°F) vs. Time for 10 Percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

even pg

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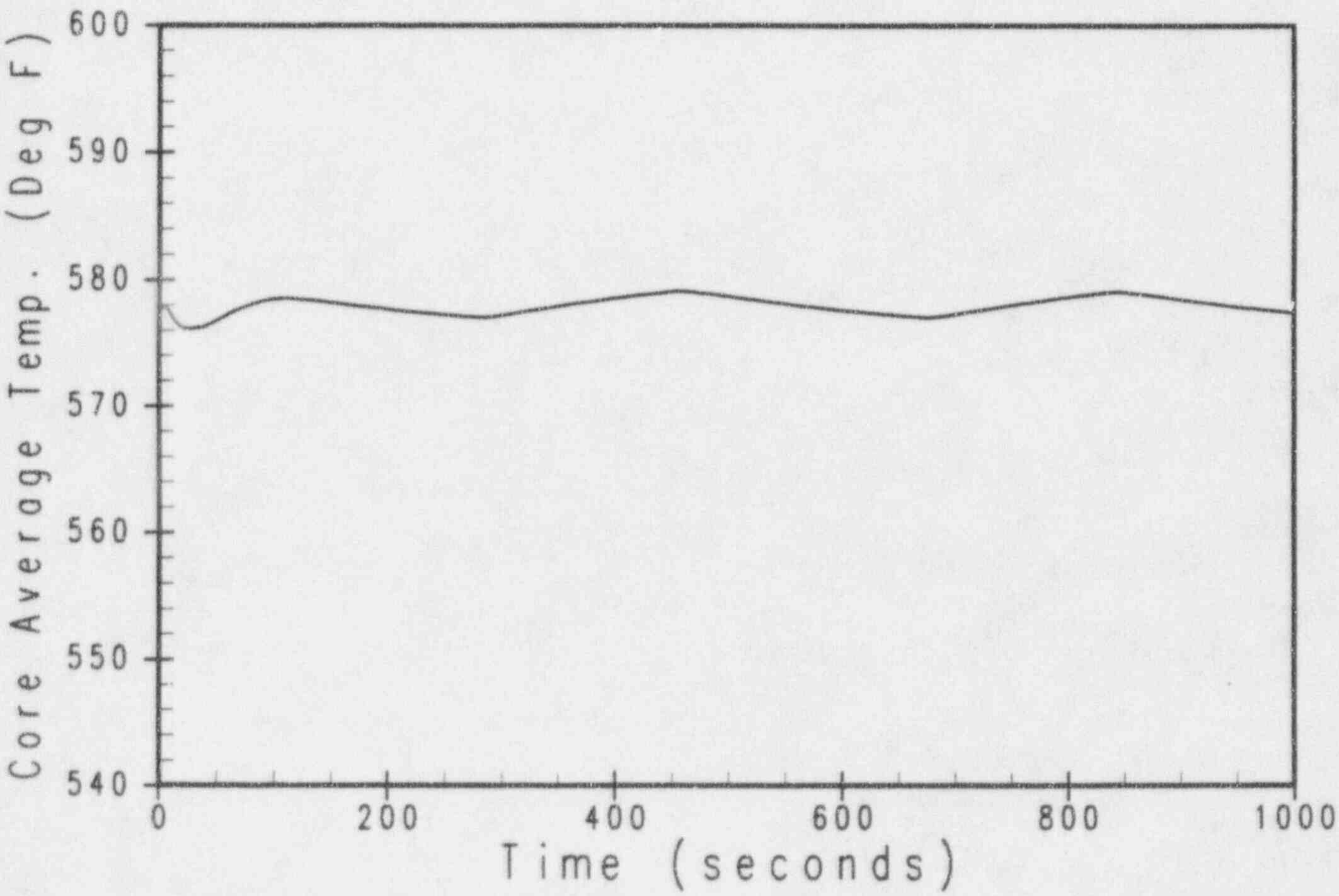
X1995/04/04 08:51:08.63 2983823847538
Case 3: Minimum Feedback (BOL) with Automatic Rod Control



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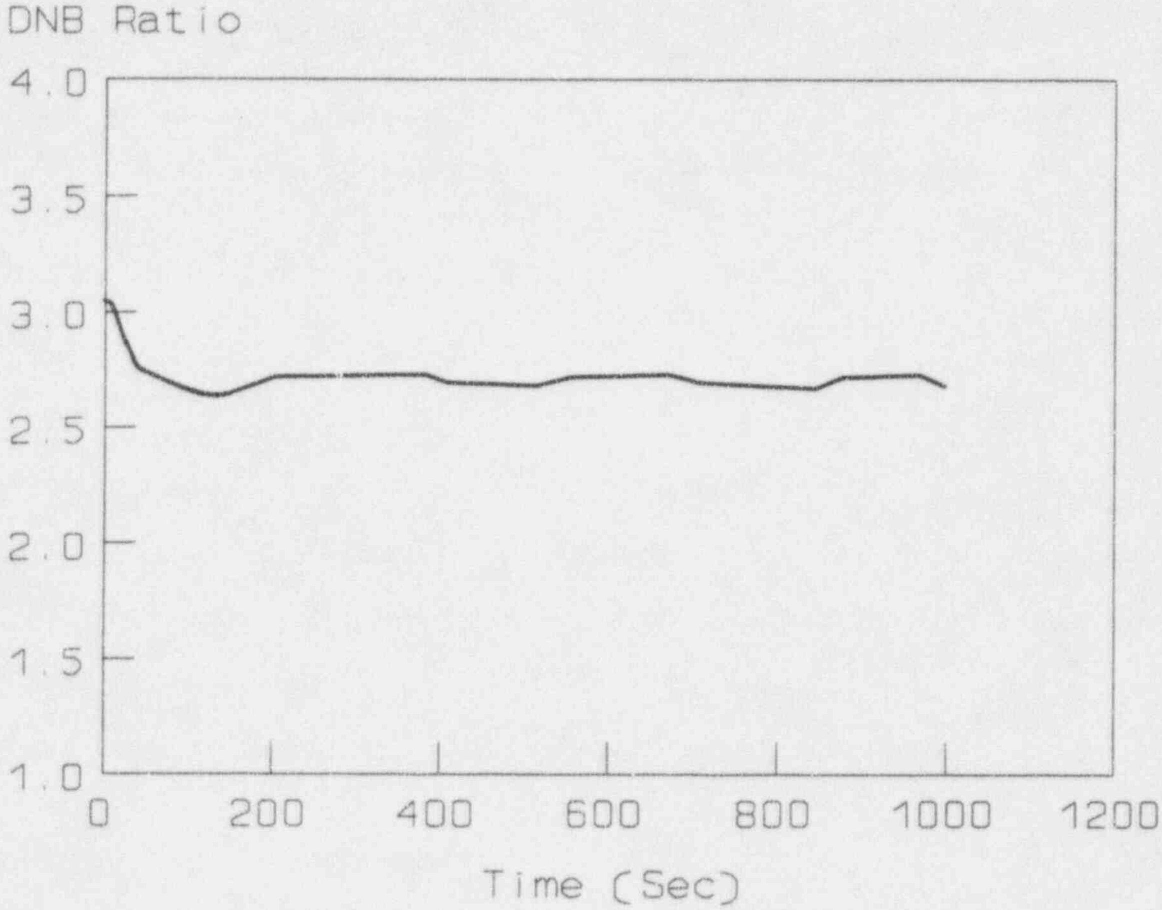

Figure 15.1.3-15

DNB Ratio vs. Time for 10 Percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

even pg

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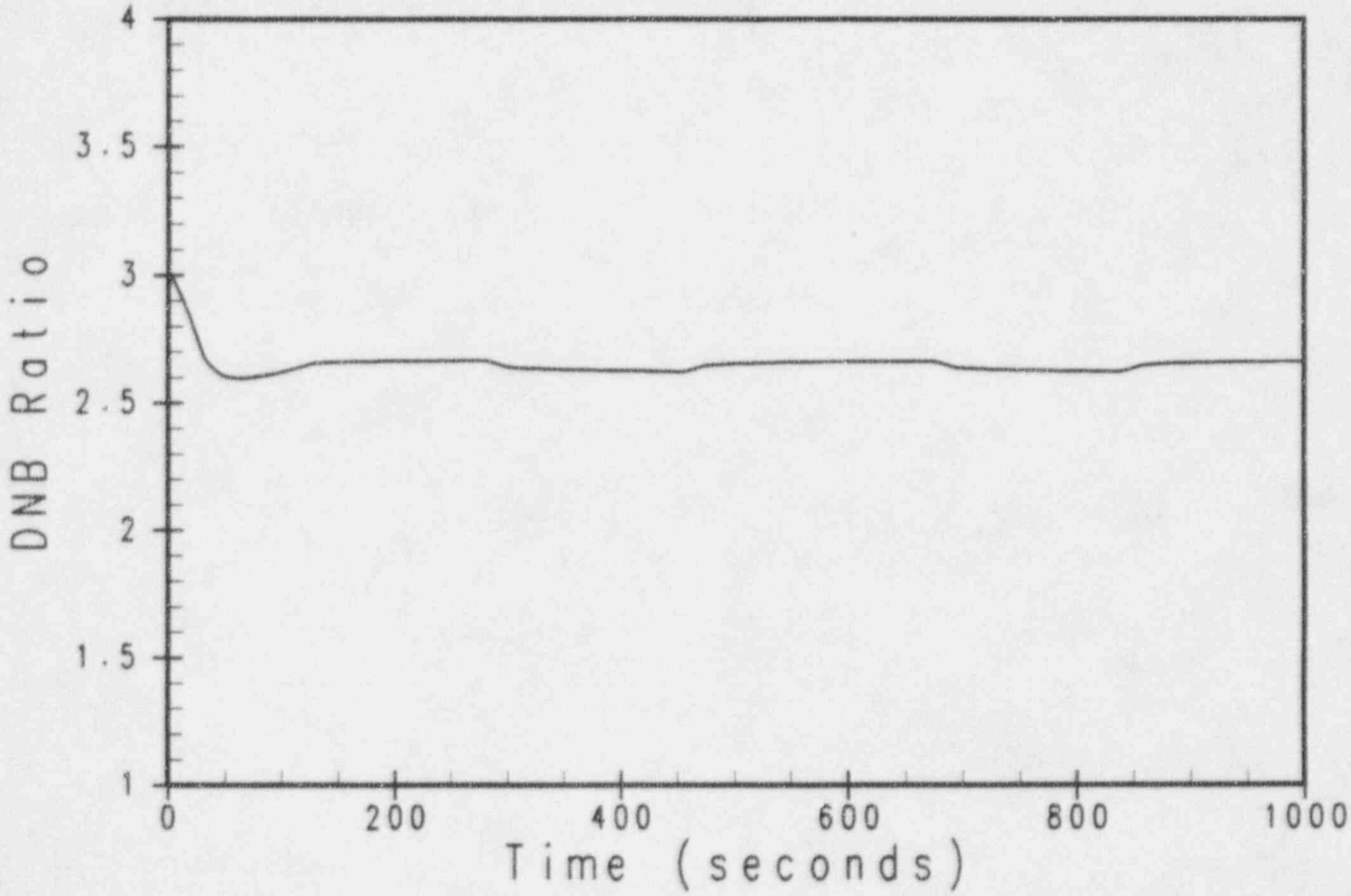
X1995/04/04 08:51:08.63 2983823847538
Case 3: Minimum Feedback (BOL) with Automatic Rod Control



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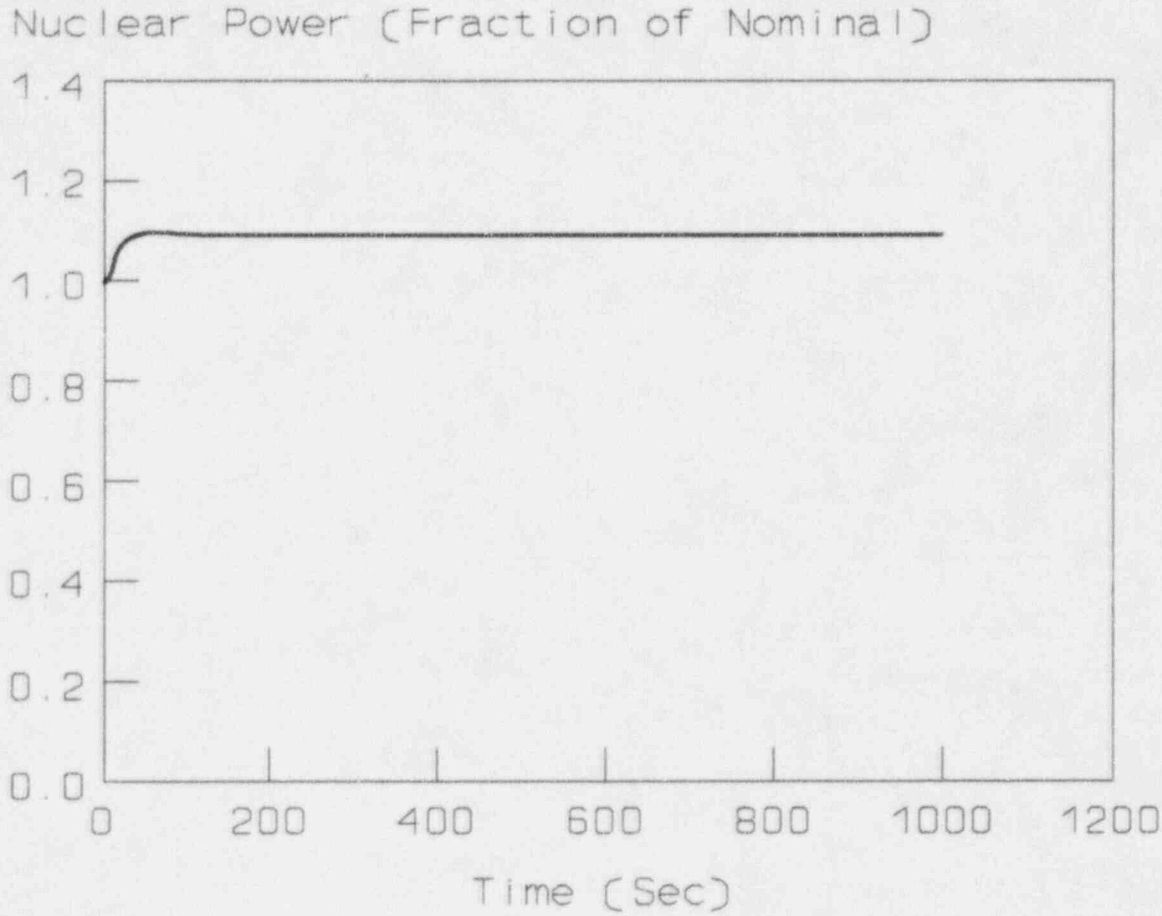
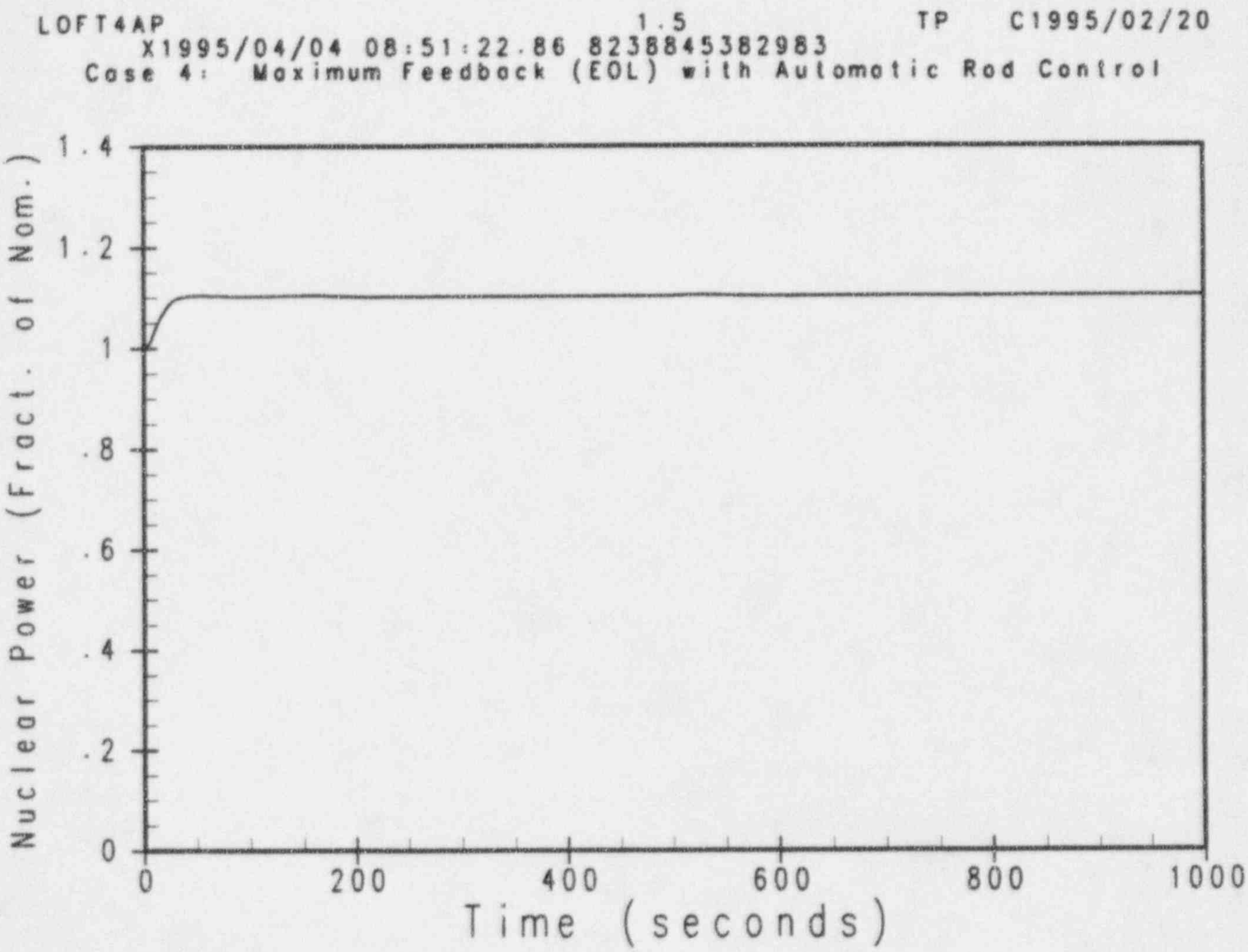


Figure 15.1.3-16

Nuclear Power (Fraction of Nominal) vs. Time for 10 Percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

even pg


LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:51:22.86 8238845382983
Case 4: Maximum Feedback (EOL) with Automatic Rod Control
Nuclear Power (Fract. of Nom.)
1.4
1.2
1
.8
.6
.4
.2
0
0
200
400
600
800
1000
Time (seconds)

Replace (next page)

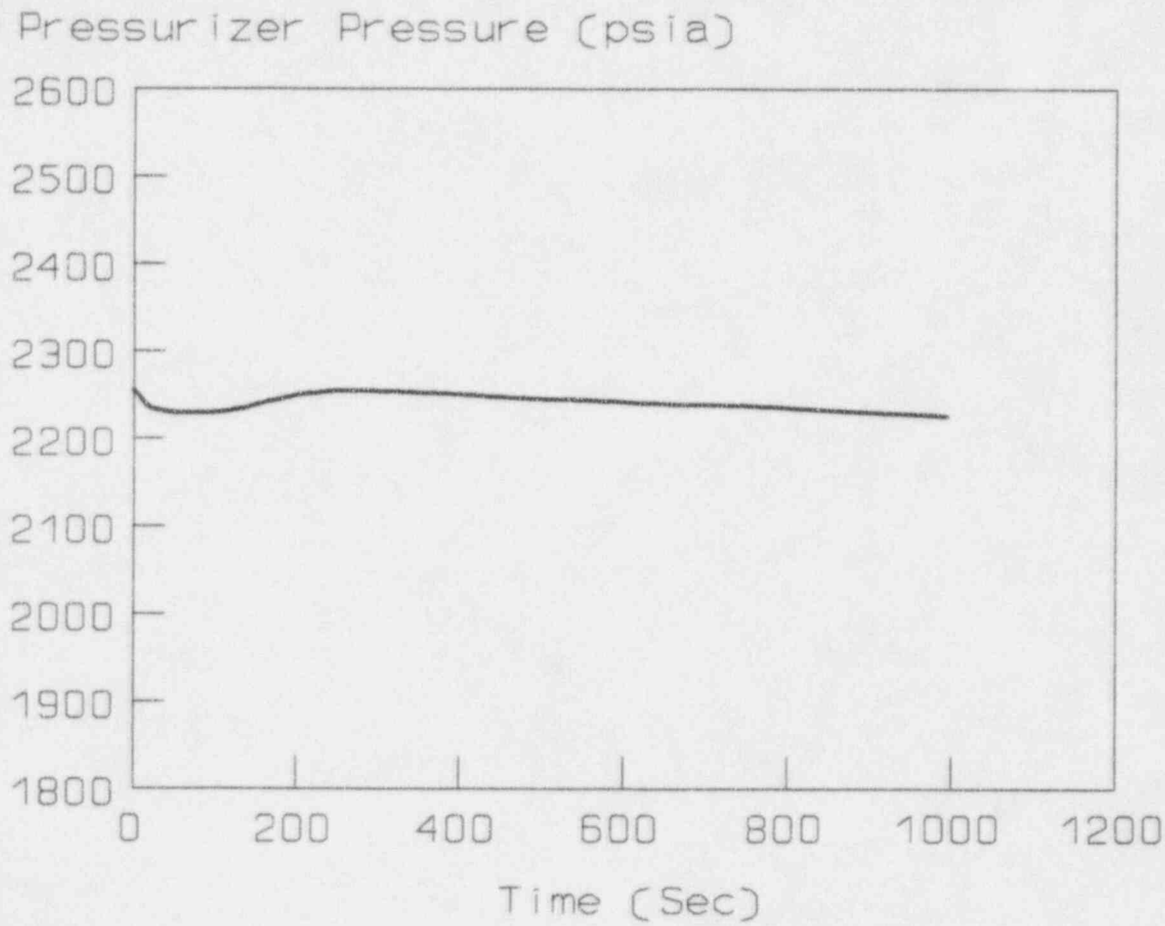
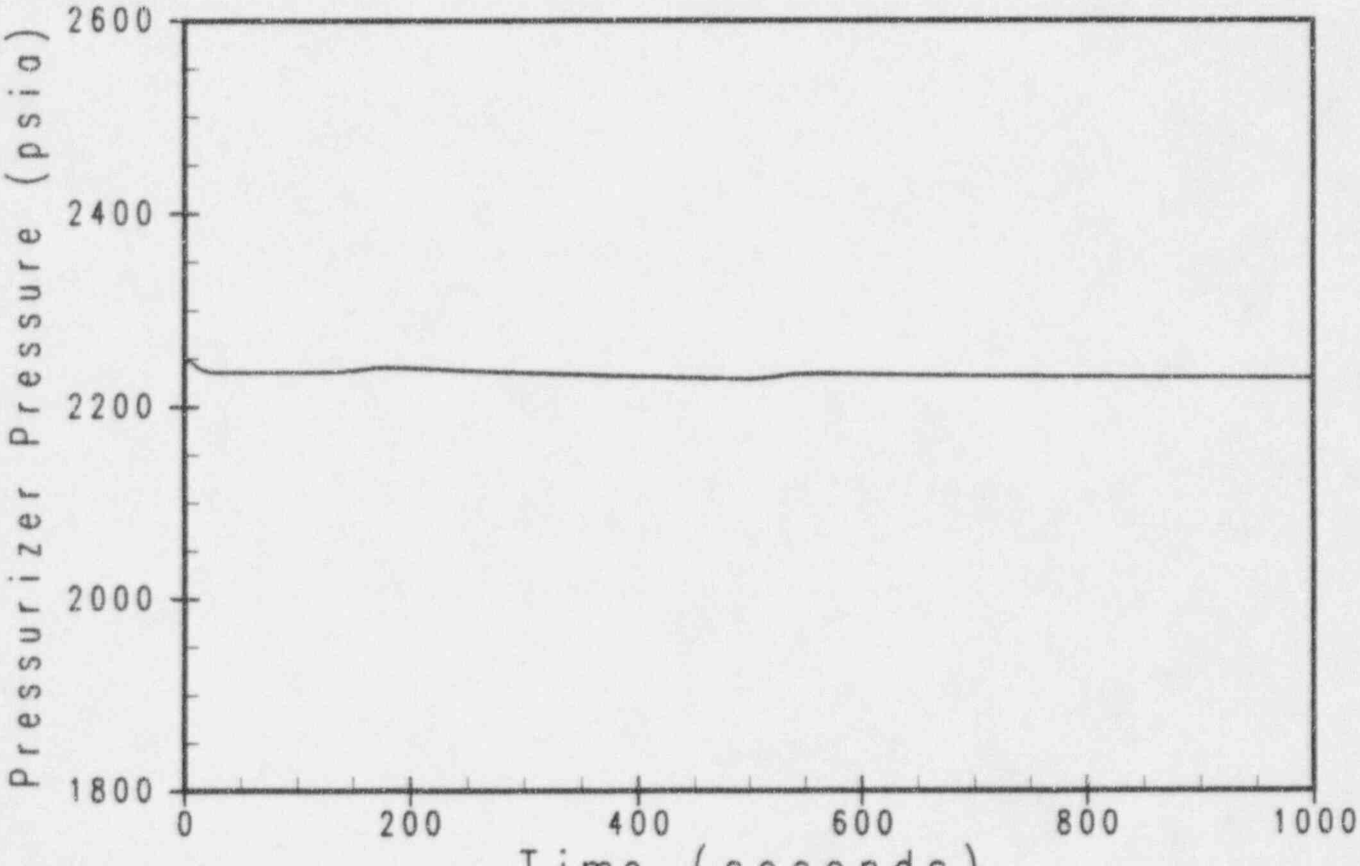


Figure 15.1.3-17

Pressurizer Pressure (psia) vs. Time for 10 Percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

even pg

LOFT4AP 1.5 TP C1995/02/20
X1995/04/04 08:51:22.86 8238845382983
Case 4: Maximum Feedback (EOL) with Automatic Rod Control



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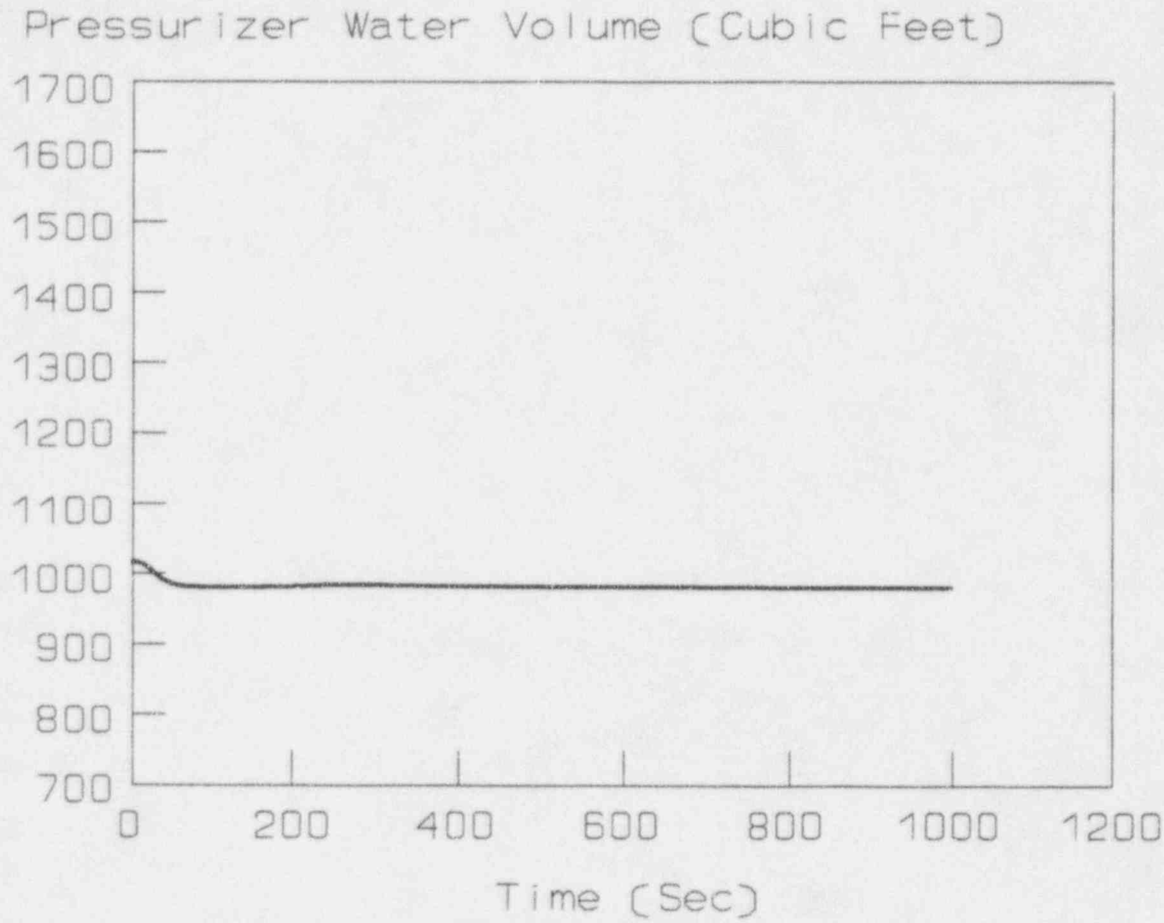

Figure 15.1.3-18

Pressurizer Water Volume (ft^3) vs. Time for 10 Percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

even pg

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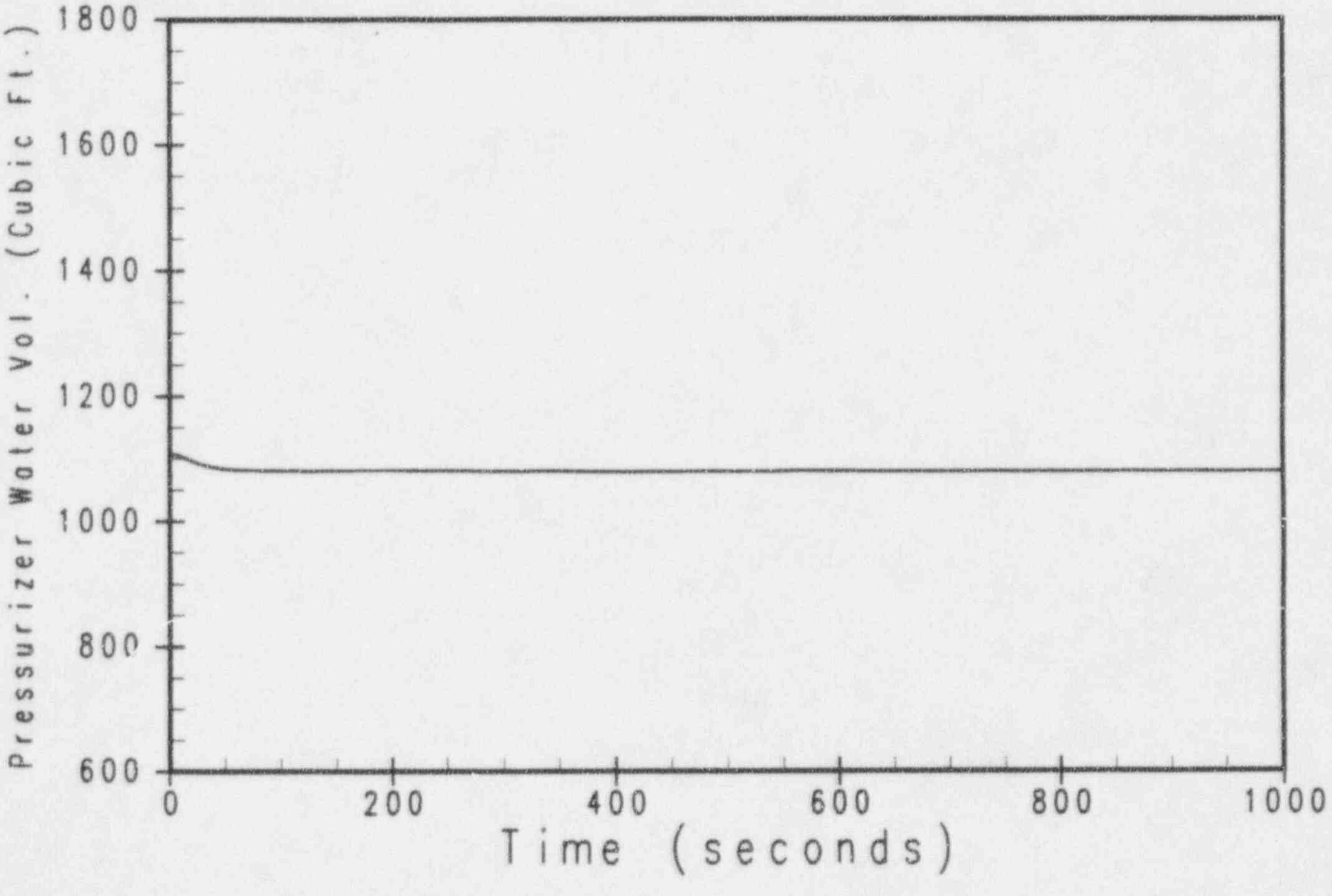
X1995/04/04 08:51:22.86 8238845382983
Case 4: Maximum Feedback (EOL) with Automatic Rod Control



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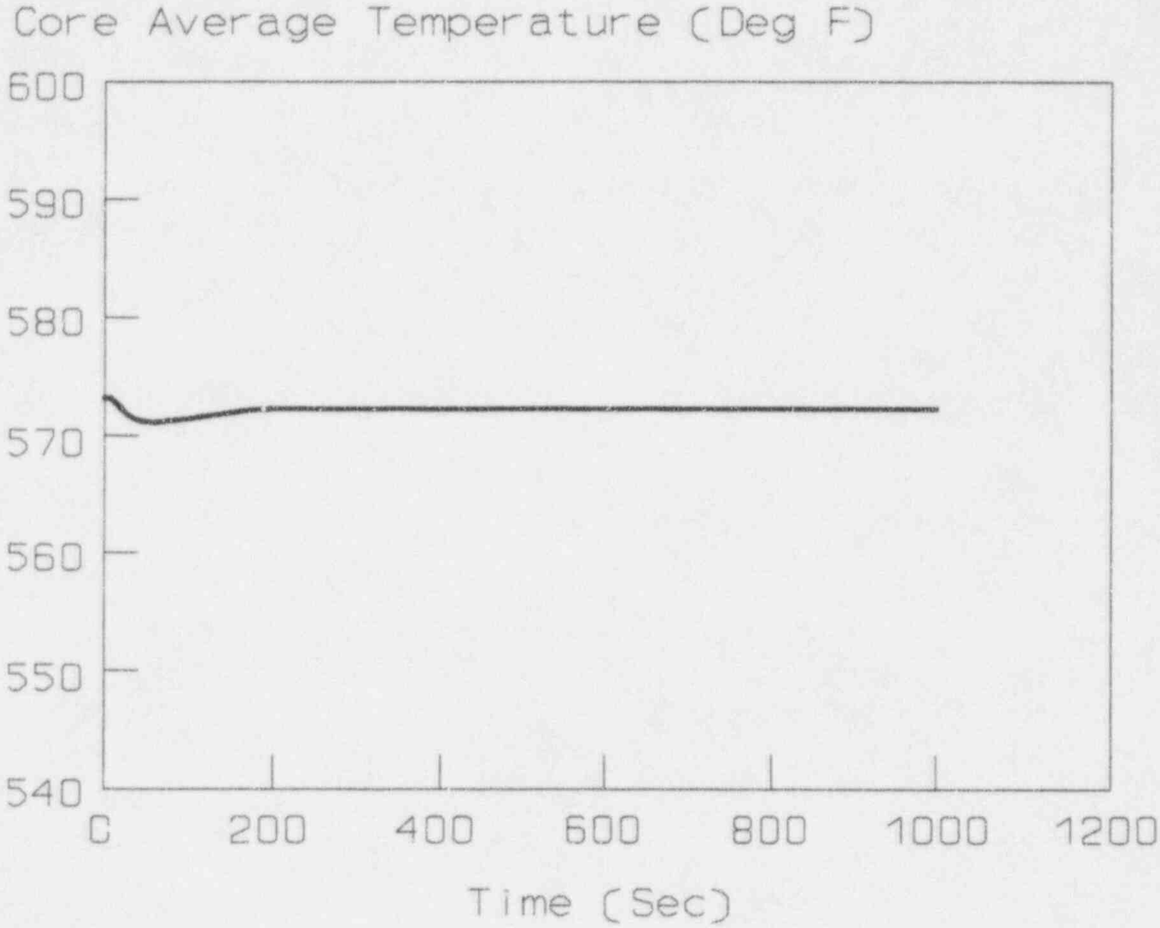

Figure 15.1.3-19

Core Average Temperature (°F) vs. Time for 10 Percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

even pg

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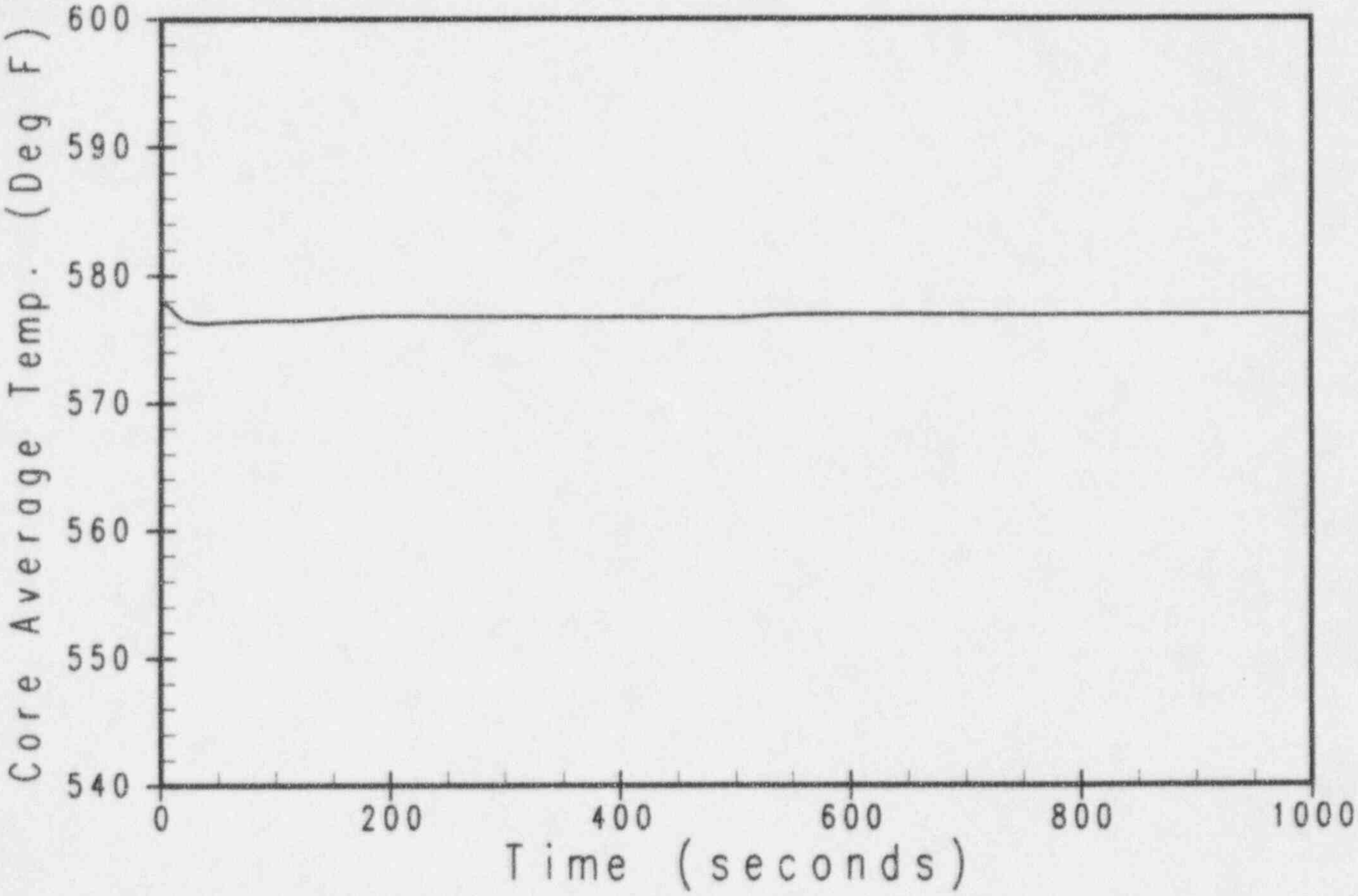
X1995/04/04 08:51:22.86 8238845382983
Case 4: Maximum Feedback (EOL) with Automatic Rod Control



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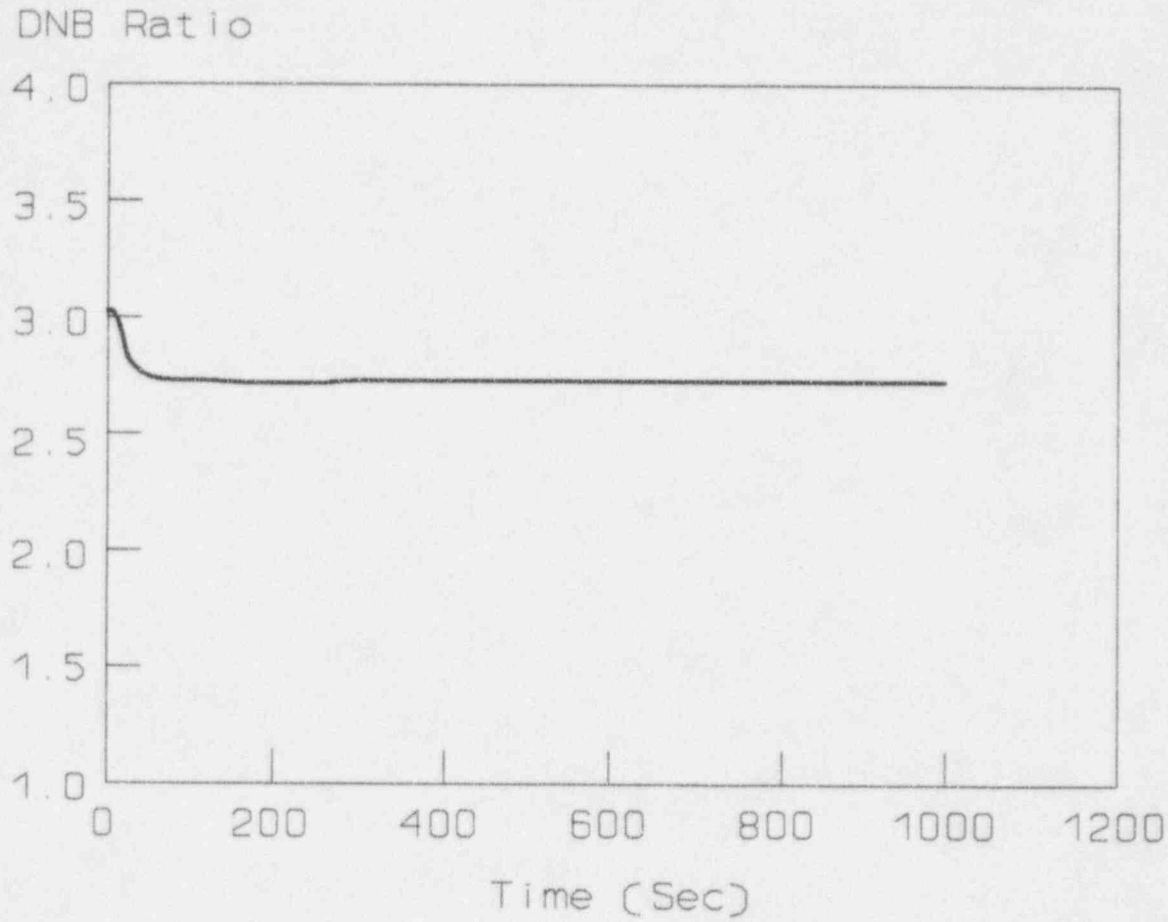

Figure 15.1.3-20

DNB Ratio vs. Time for 10 Percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

even pg

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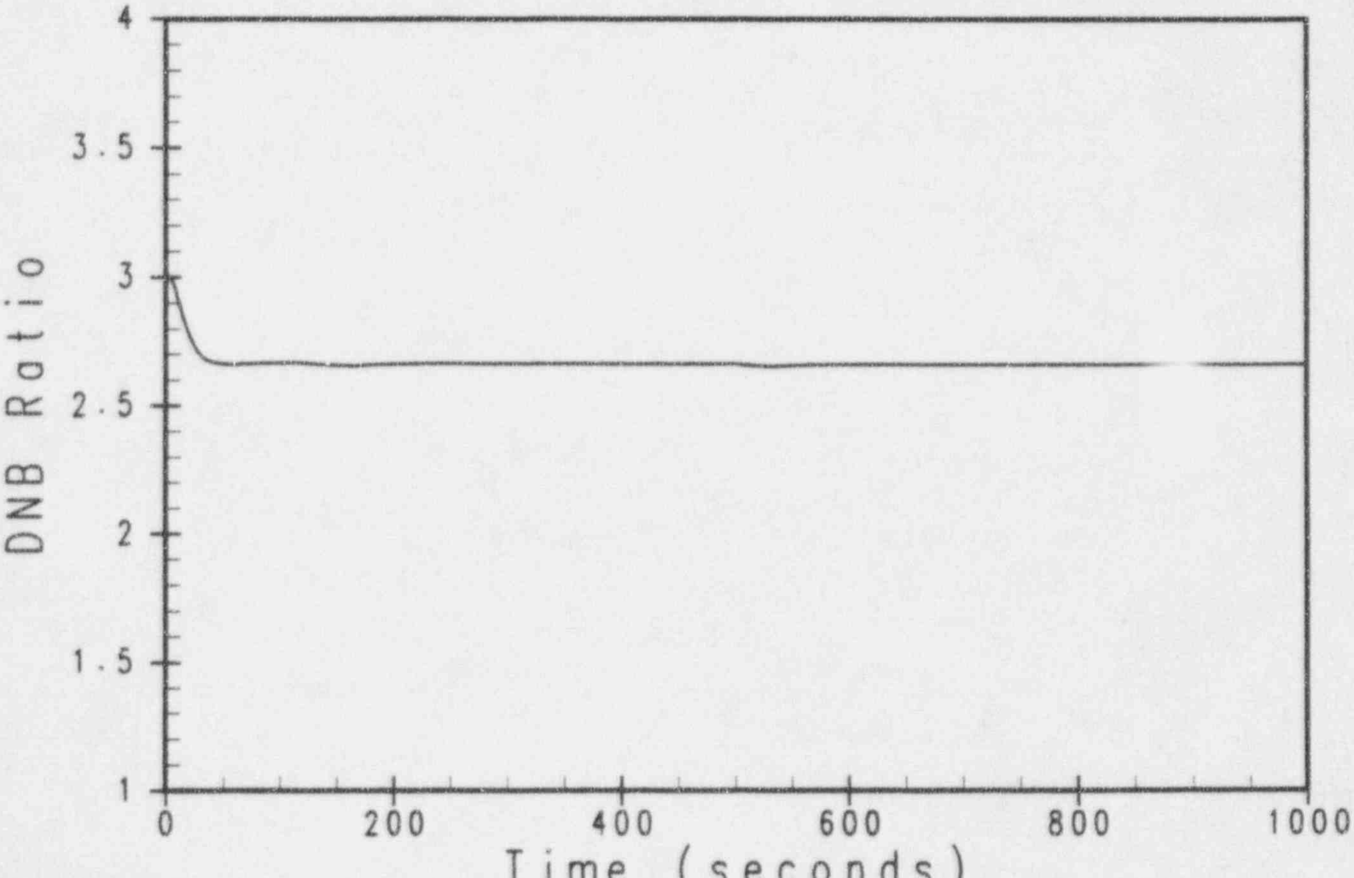
Case 4: Maximum Feedback (EOL) with Automatic Rod Control



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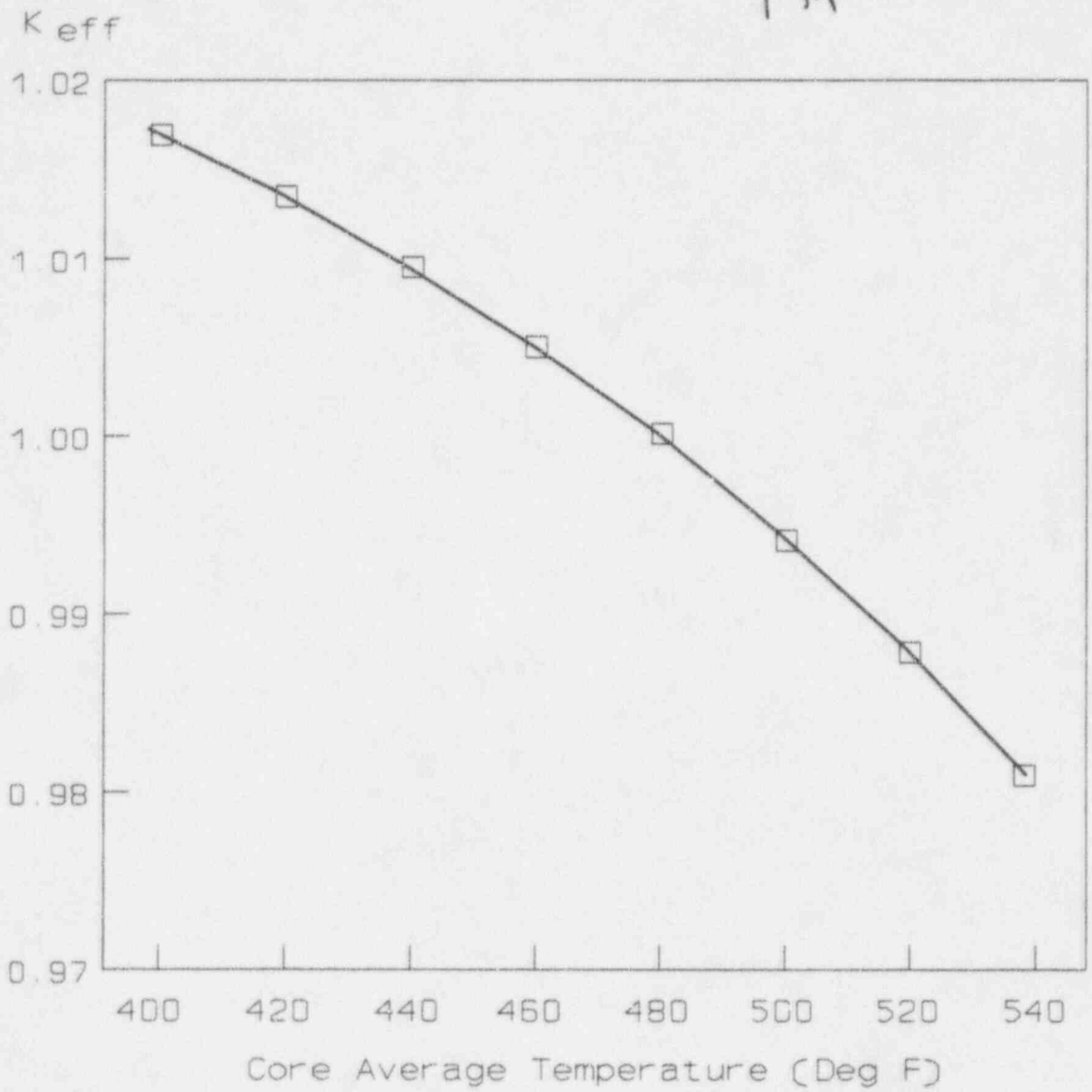


Figure 15.1.4-1

K_{eff} vs. Core Average Temperature
Steam Line Break Events

even pg

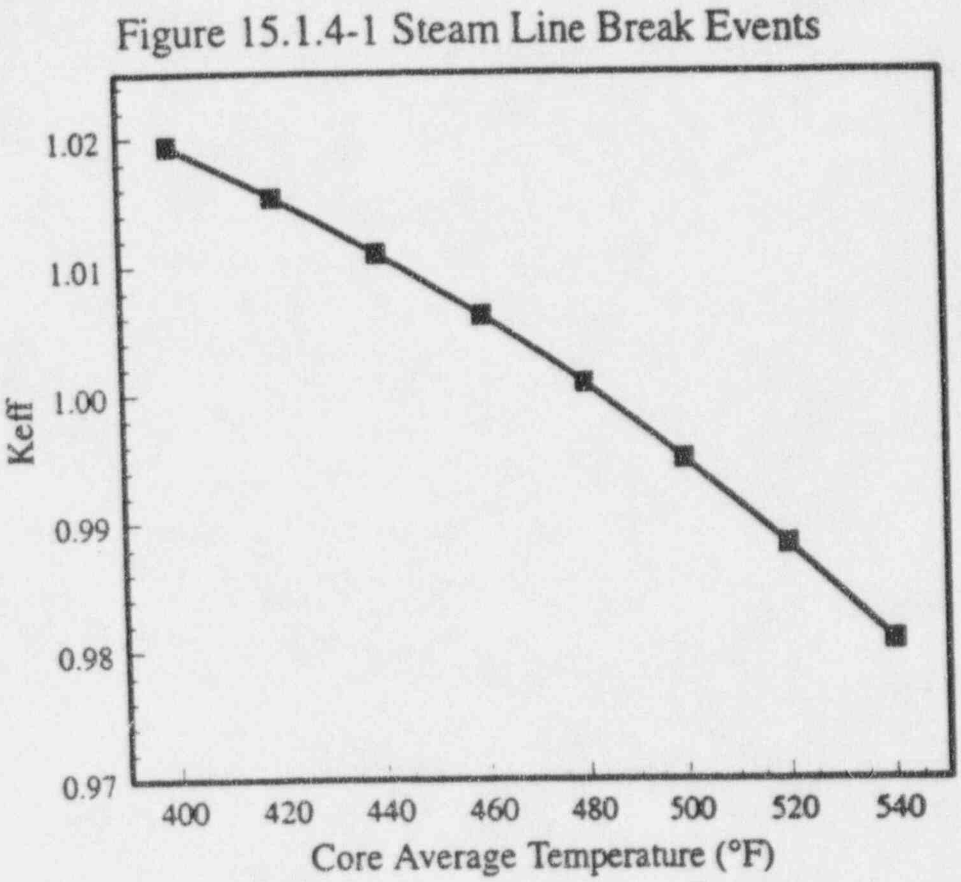


Figure 15.1.4-1 Steam Line Break Events
Keff
1.02
1.01
1.00
0.99
0.98
0.97
400
420
440
460
480
500
520
540
Core Average Temperature (°F)

Replace

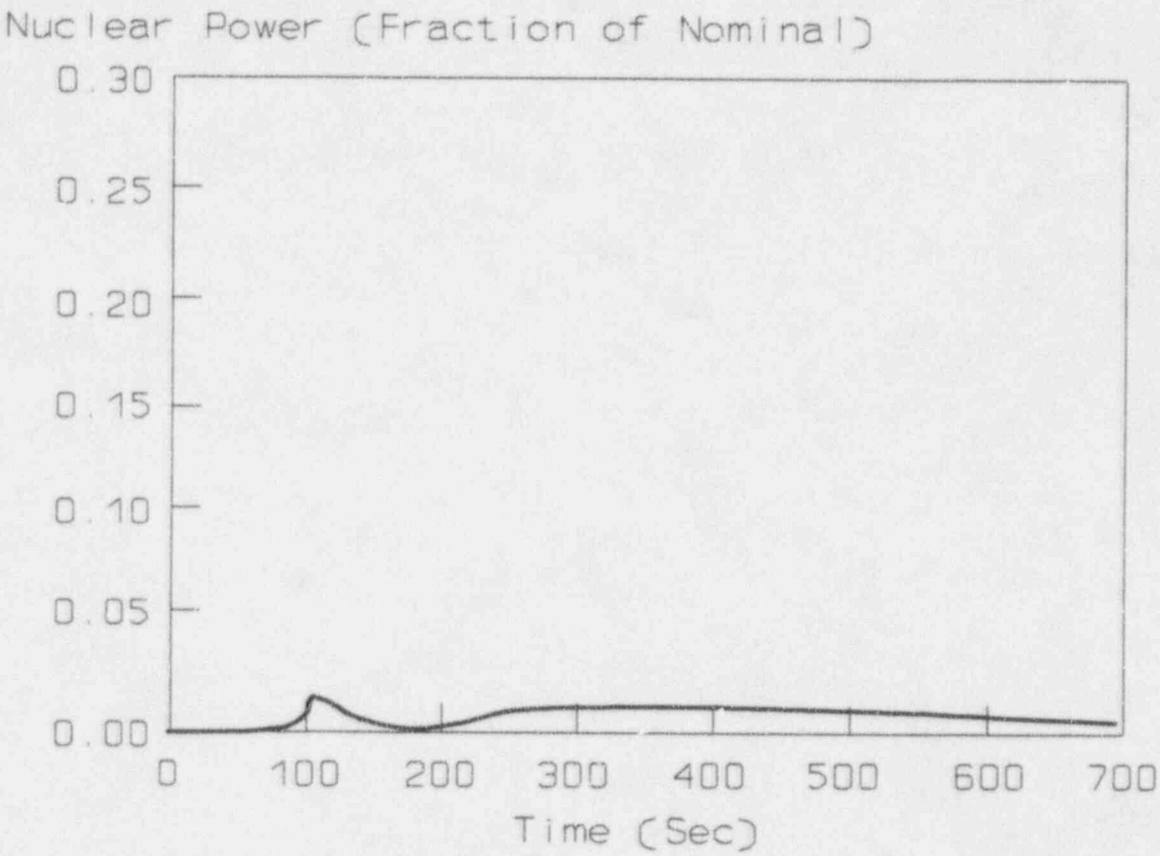
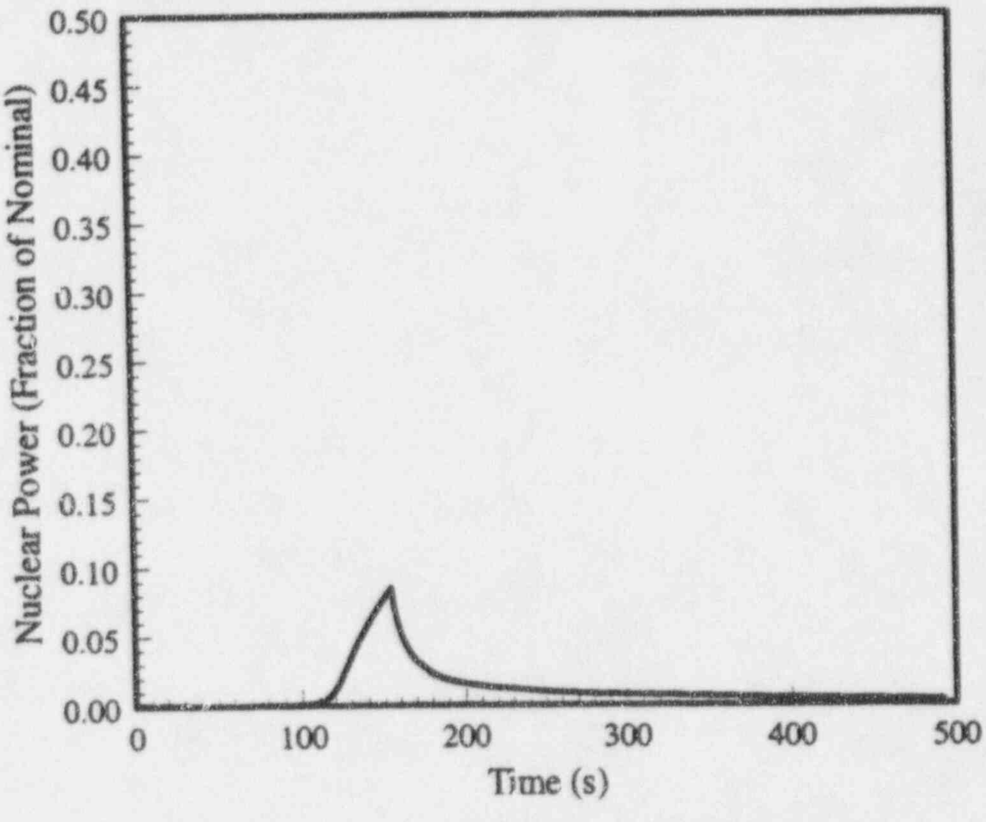


Figure 15.1.4-2

Nuclear Power Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-2 Steam Line Break Events



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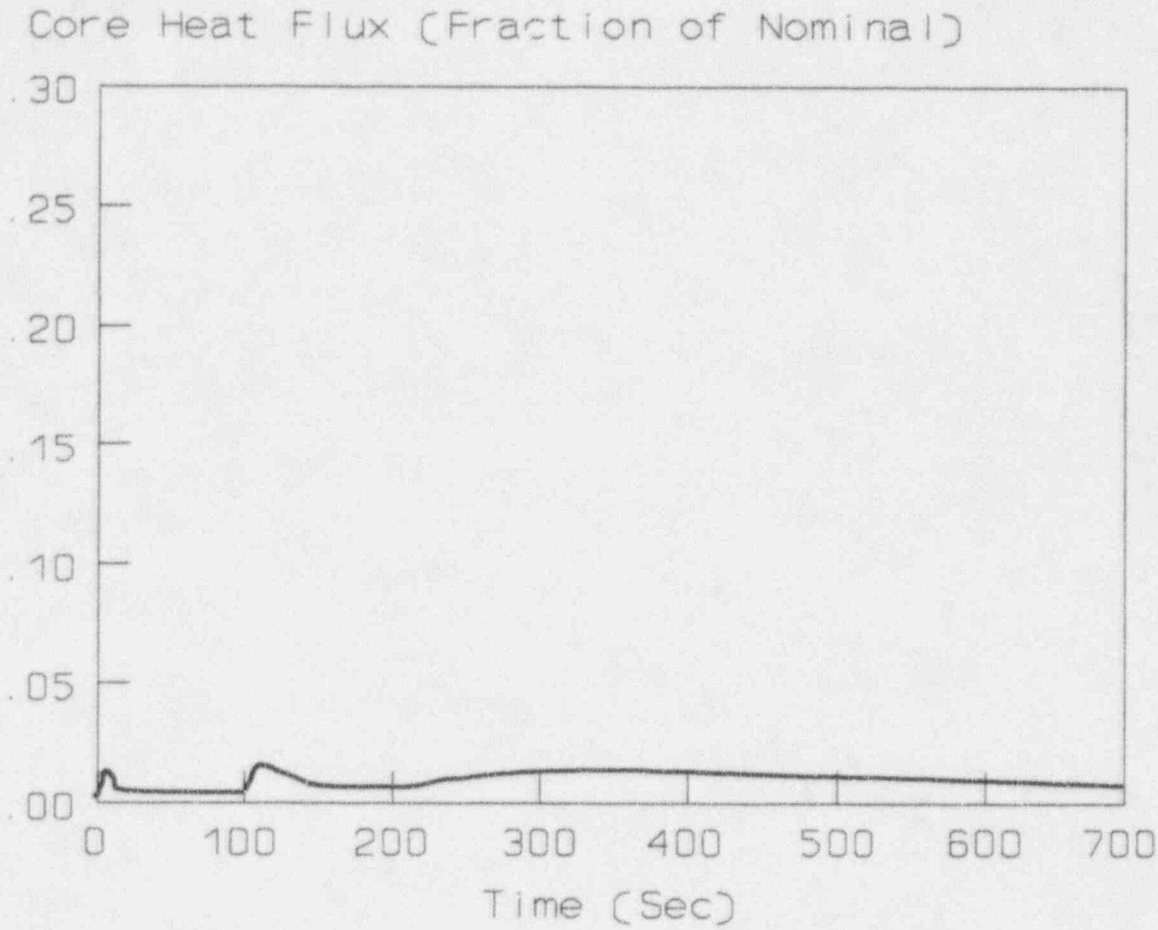
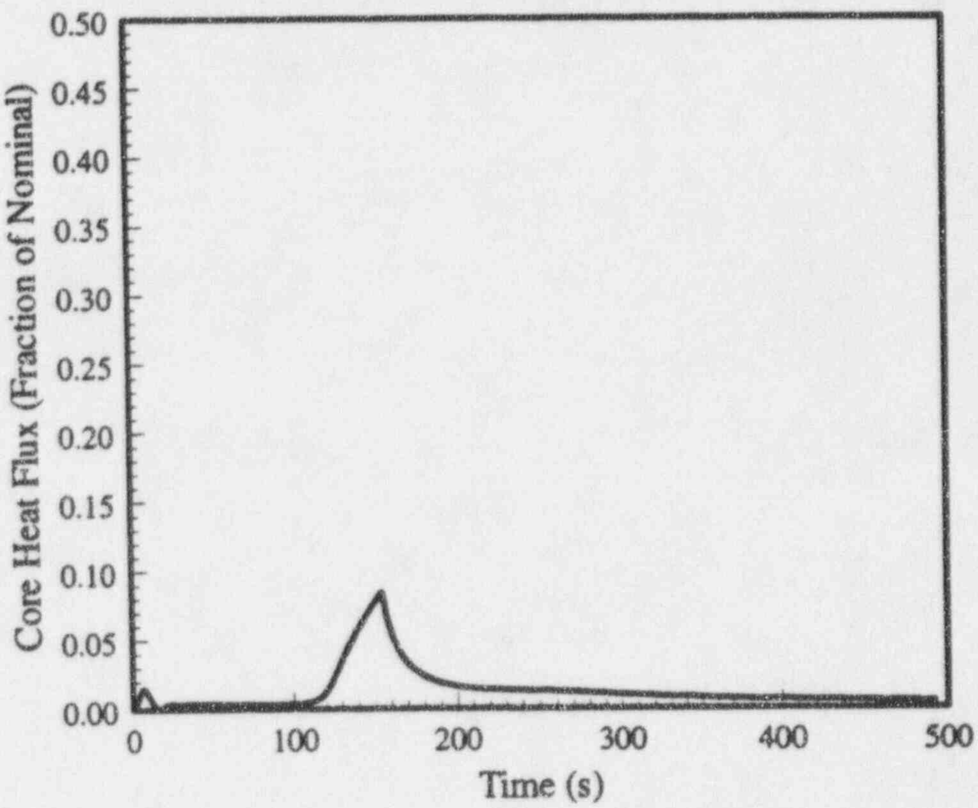


Figure 15.1.4-3

Core Heat Flux Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-3 Steam Line Break Events



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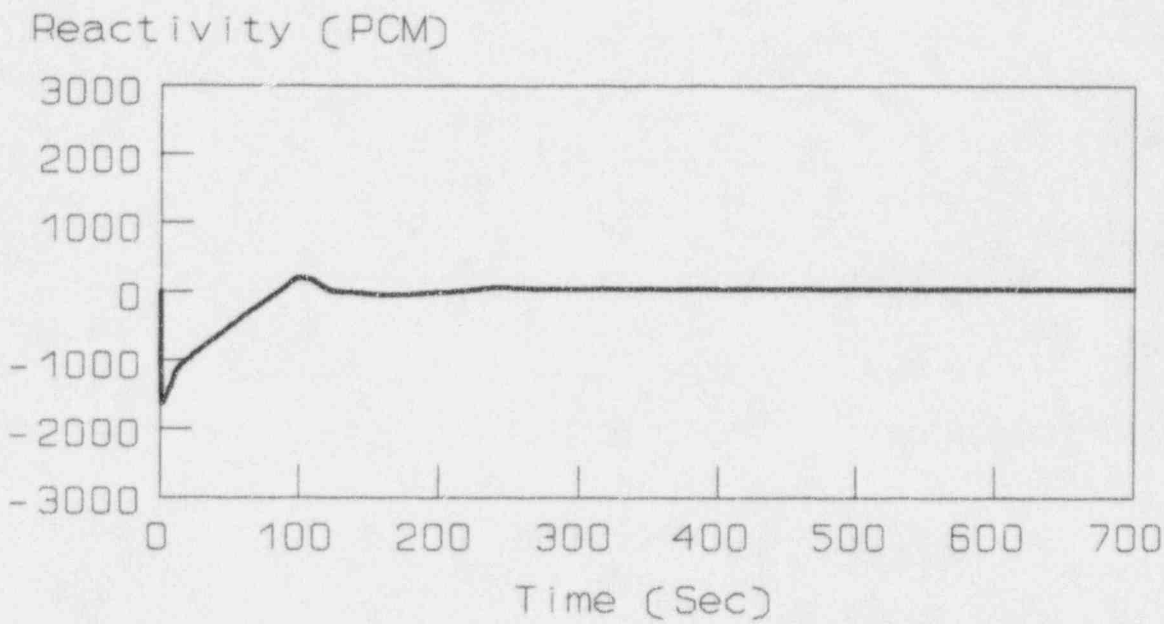
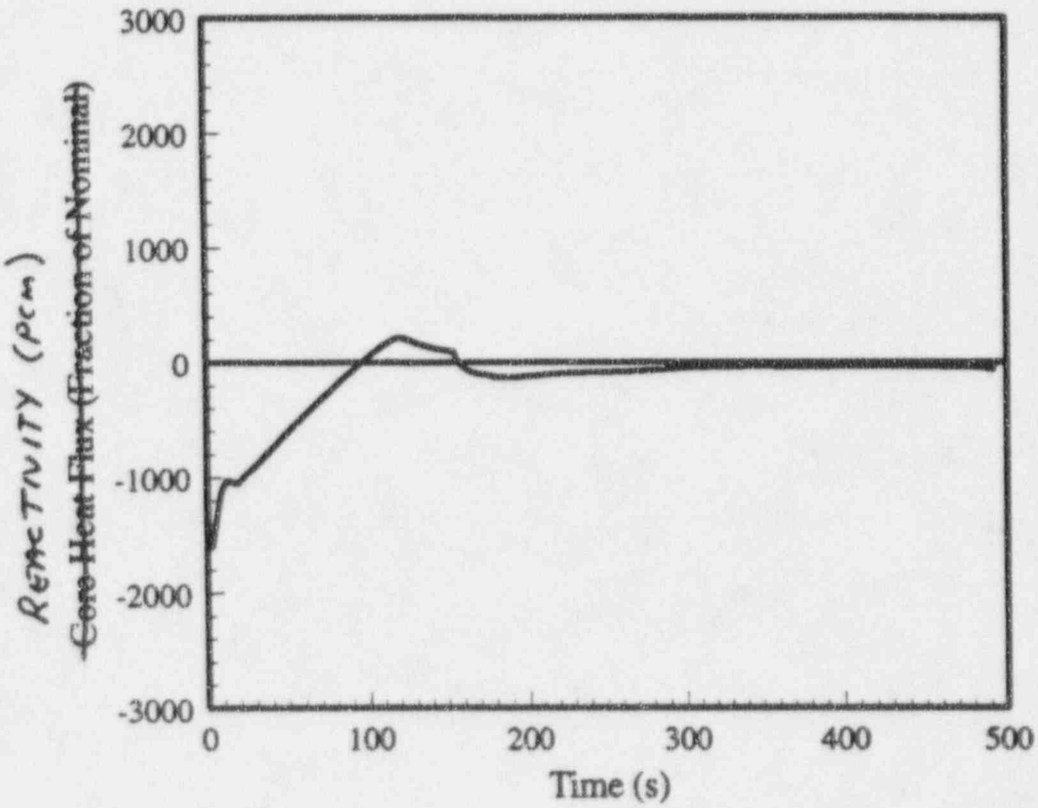


Figure 15.1.4-4

Reactivity Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-4 Steam Line Break Events



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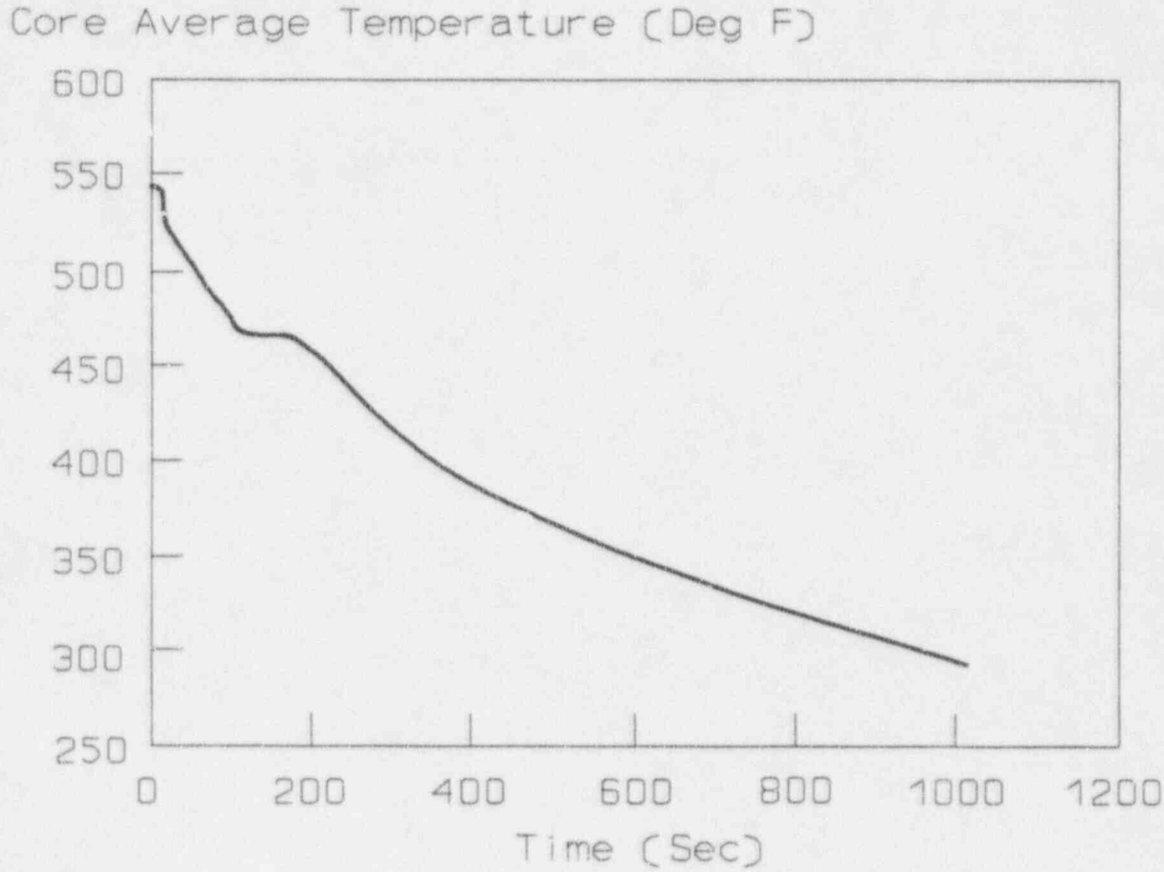
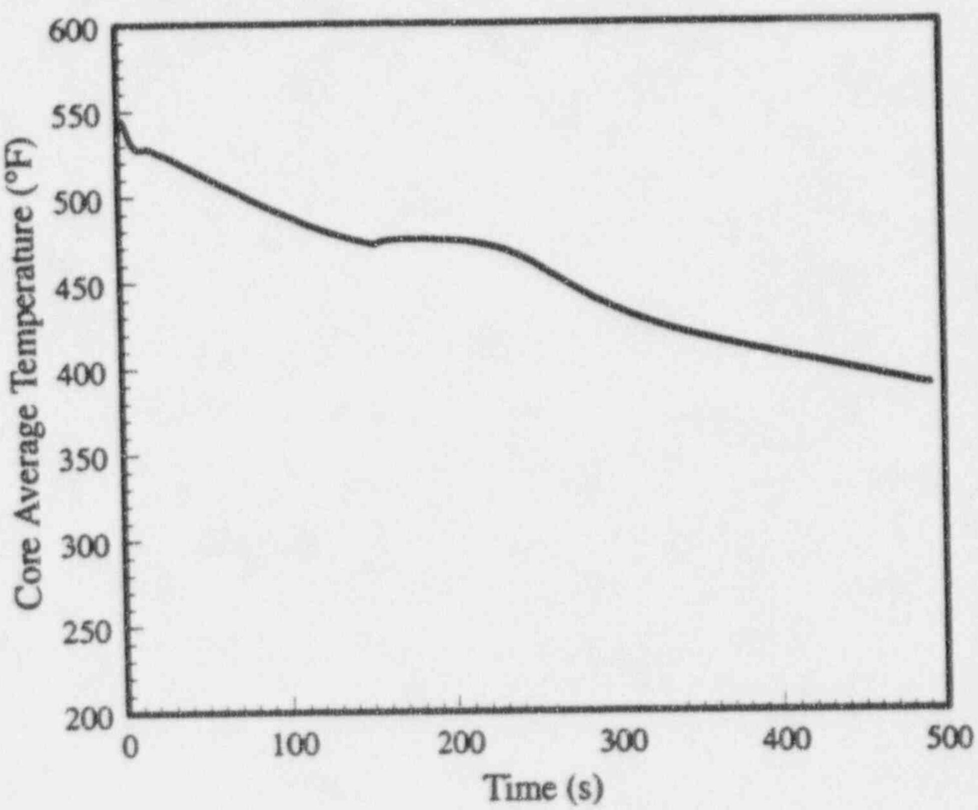


Figure 15.1.4-5

Core Average Temperature Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Figure 15.1.4-5 Steam Line Break Events



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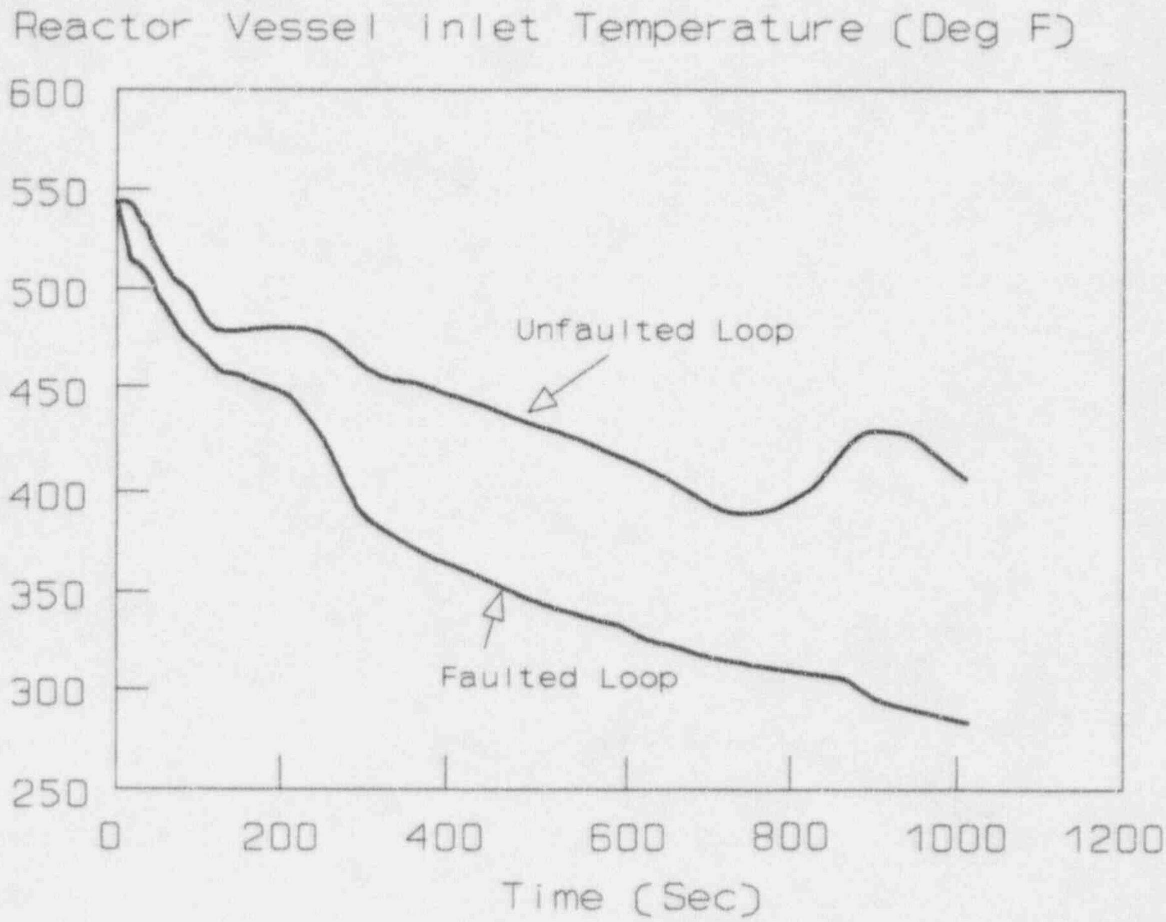
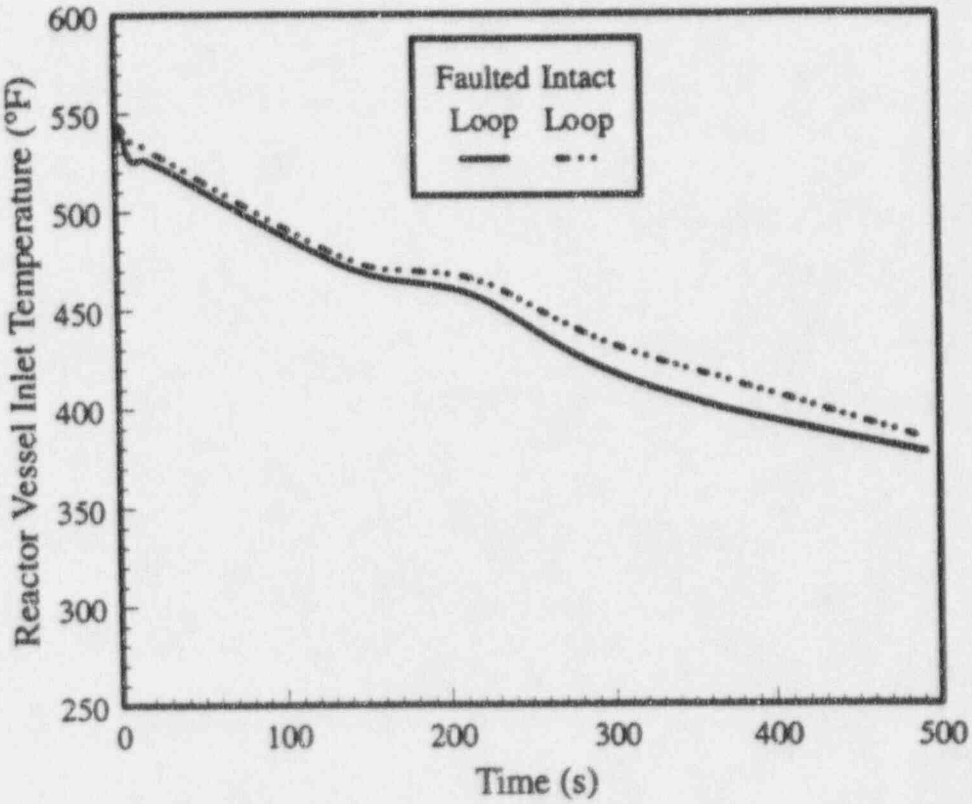


Figure 15.1.4-6

Reactor Vessel Inlet Temperature Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-6 Steam Line Break Events



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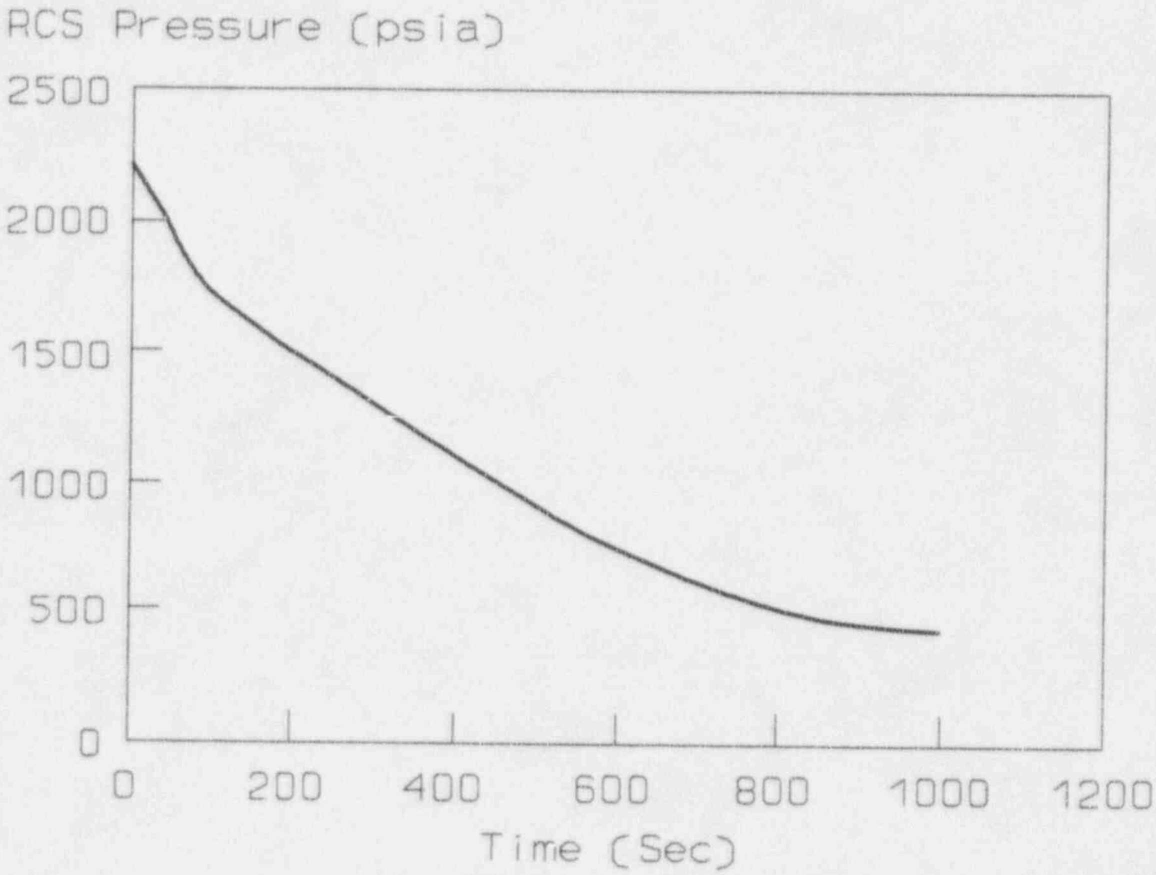
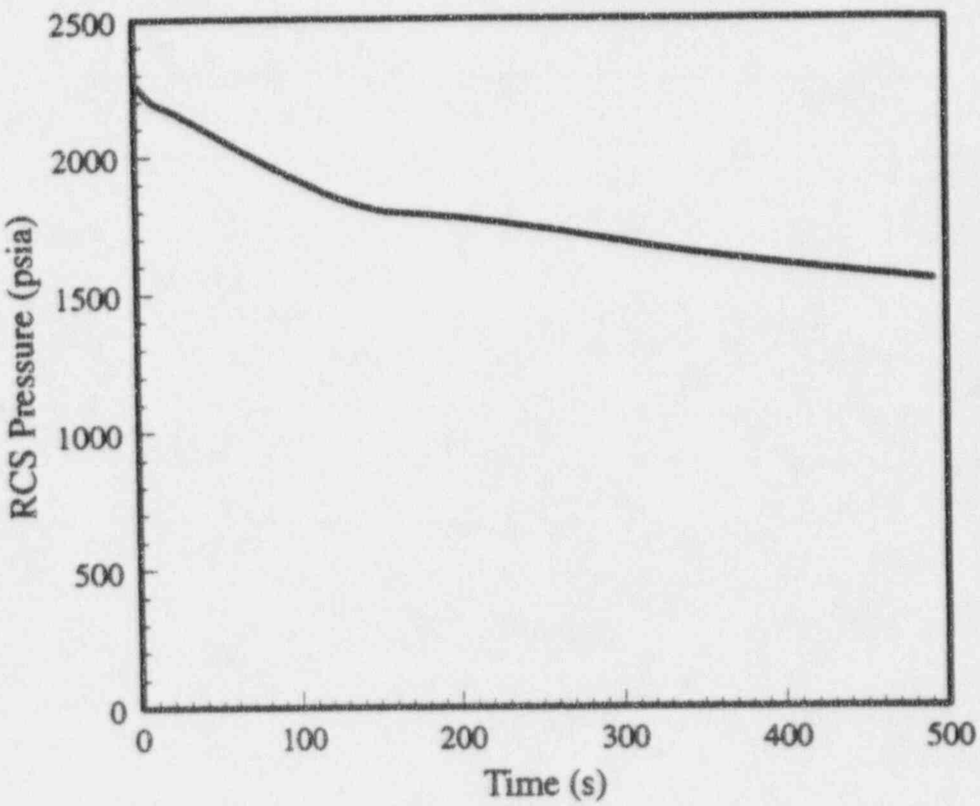


Figure 15.1.4-7

RCS Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Figure 15.1.4-7 Steam Line Break Events



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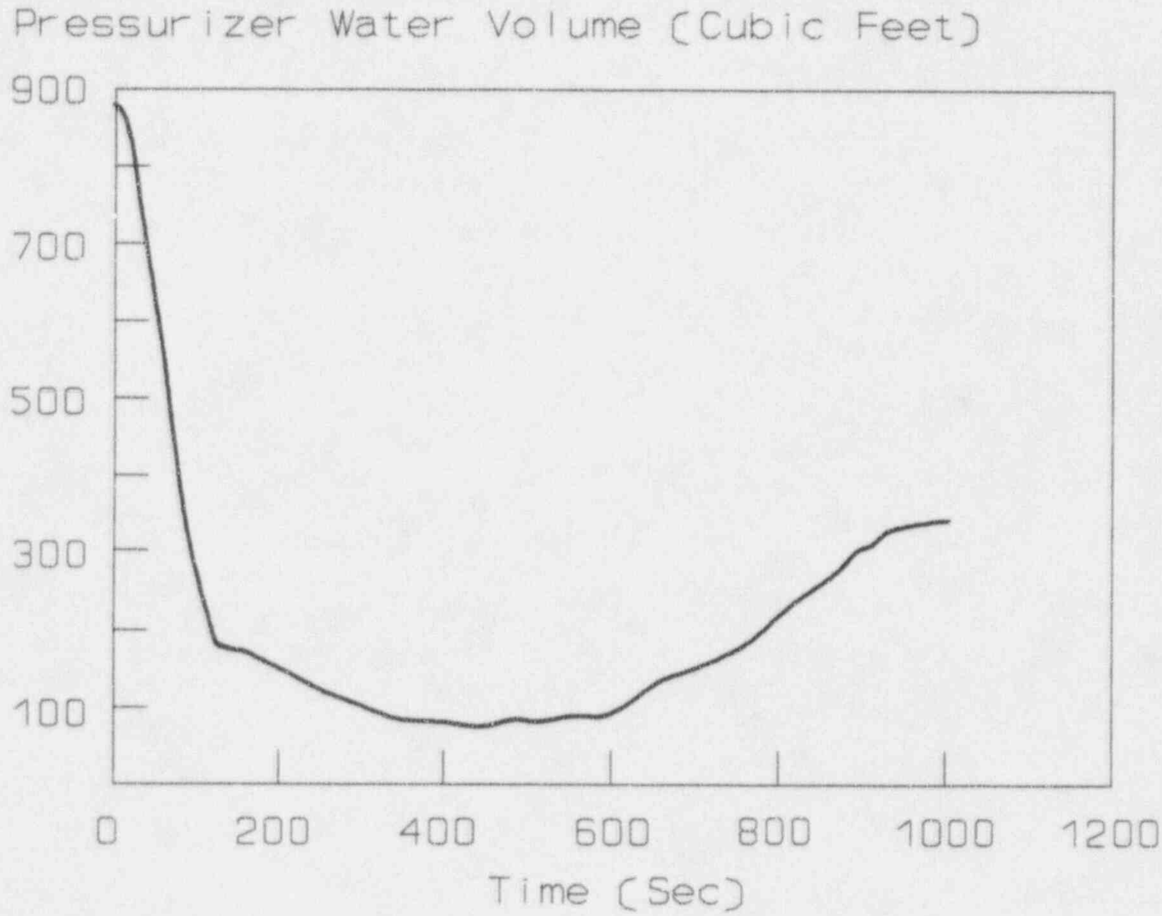
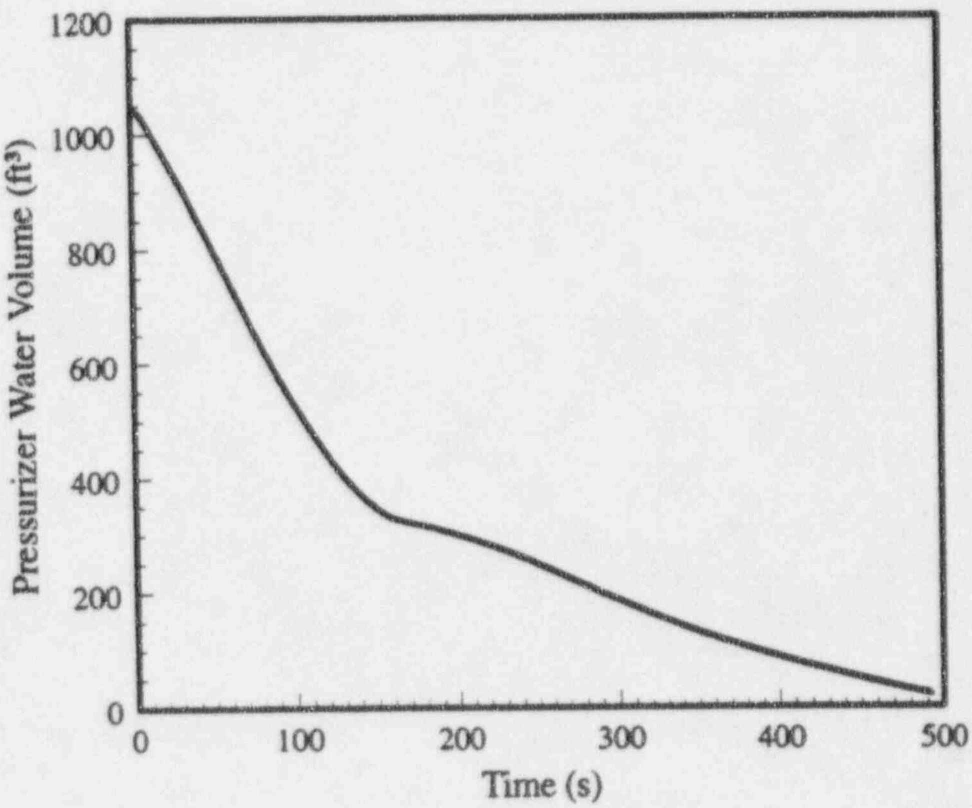


Figure 15.1.4-8

Pressurizer Water Volume Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-8 Steam Line Break Events



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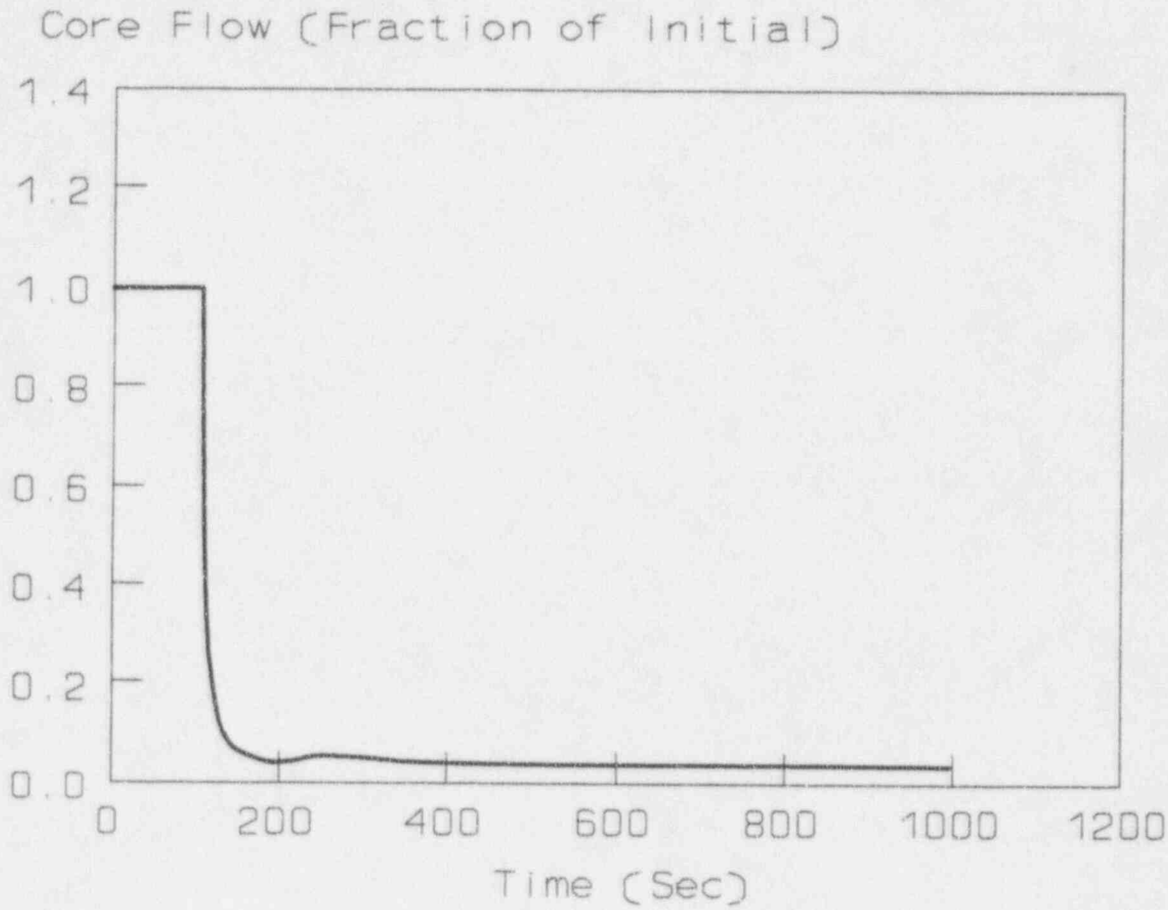
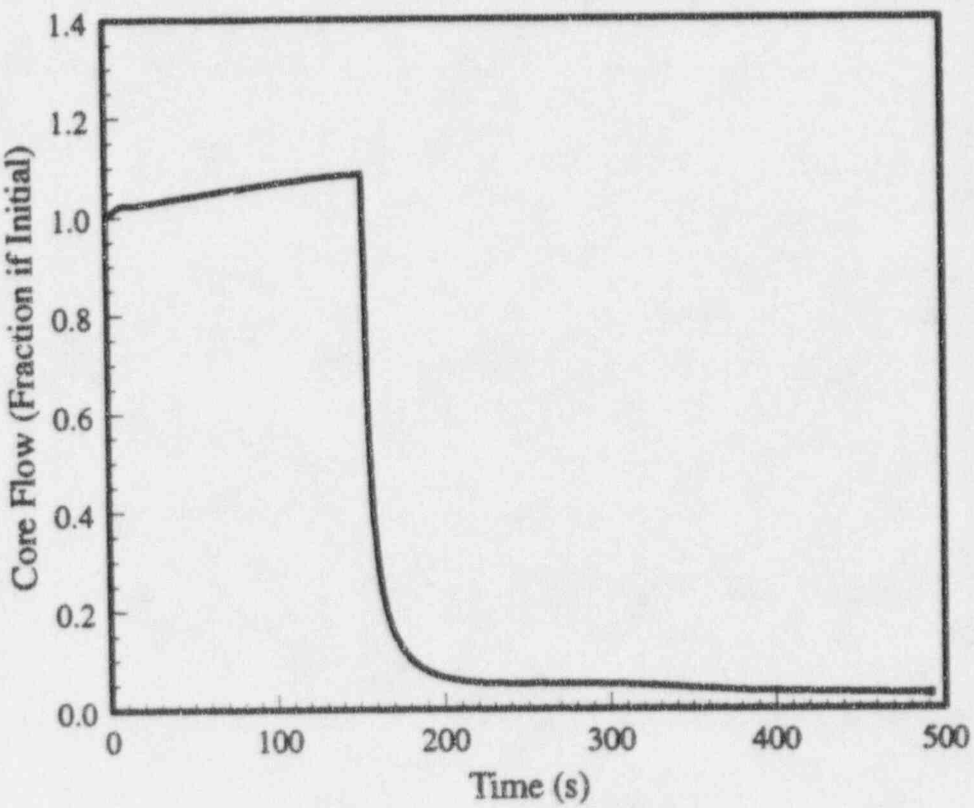


Figure 15.1.4-9

Core Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-9 Steam Line Break Events



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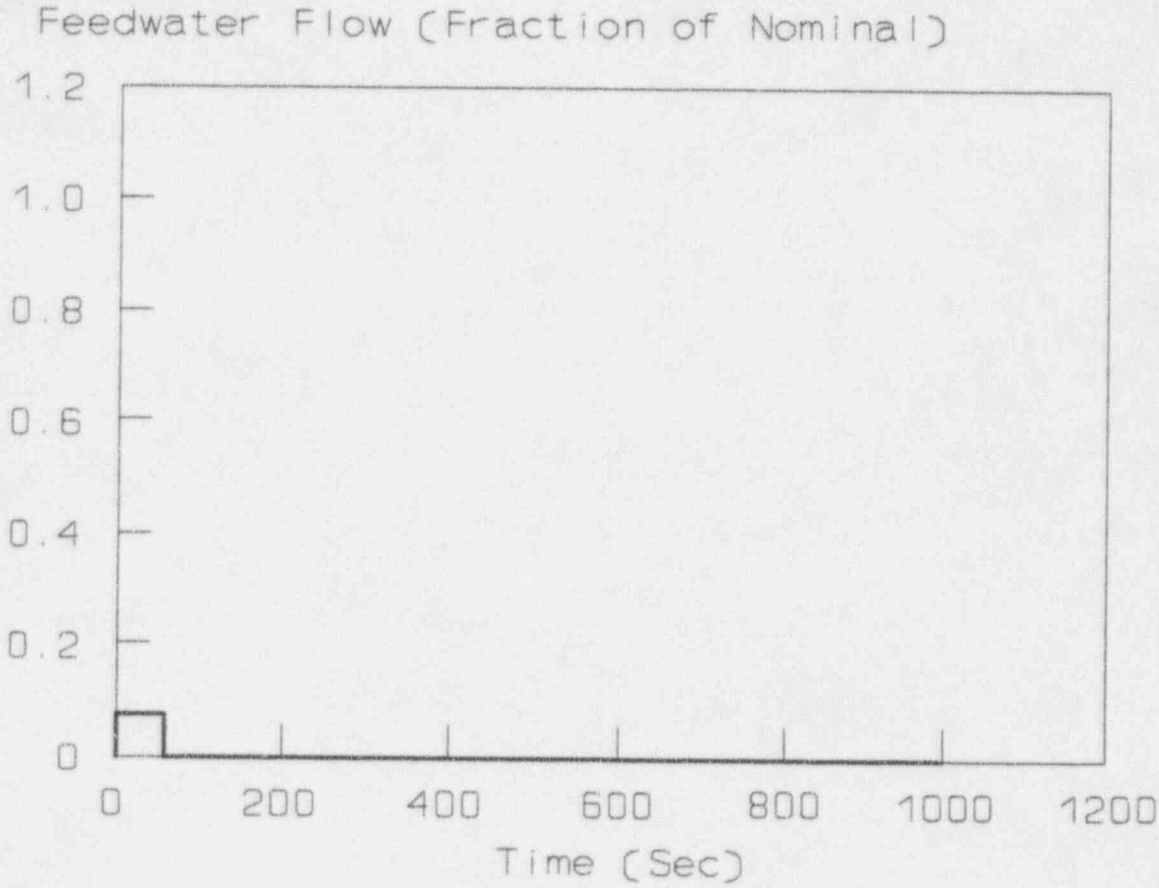
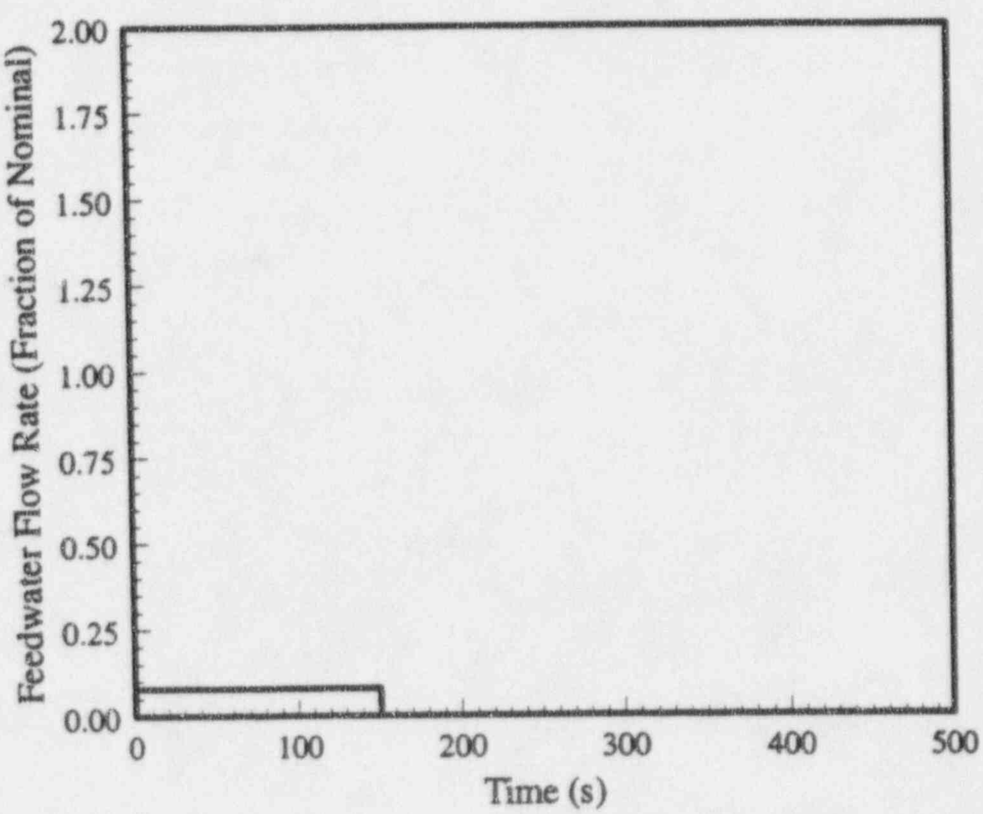


Figure 15.1.4-10

Feedwater Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

even pg

Figure 15.1.4-10 Steam Line Break Events



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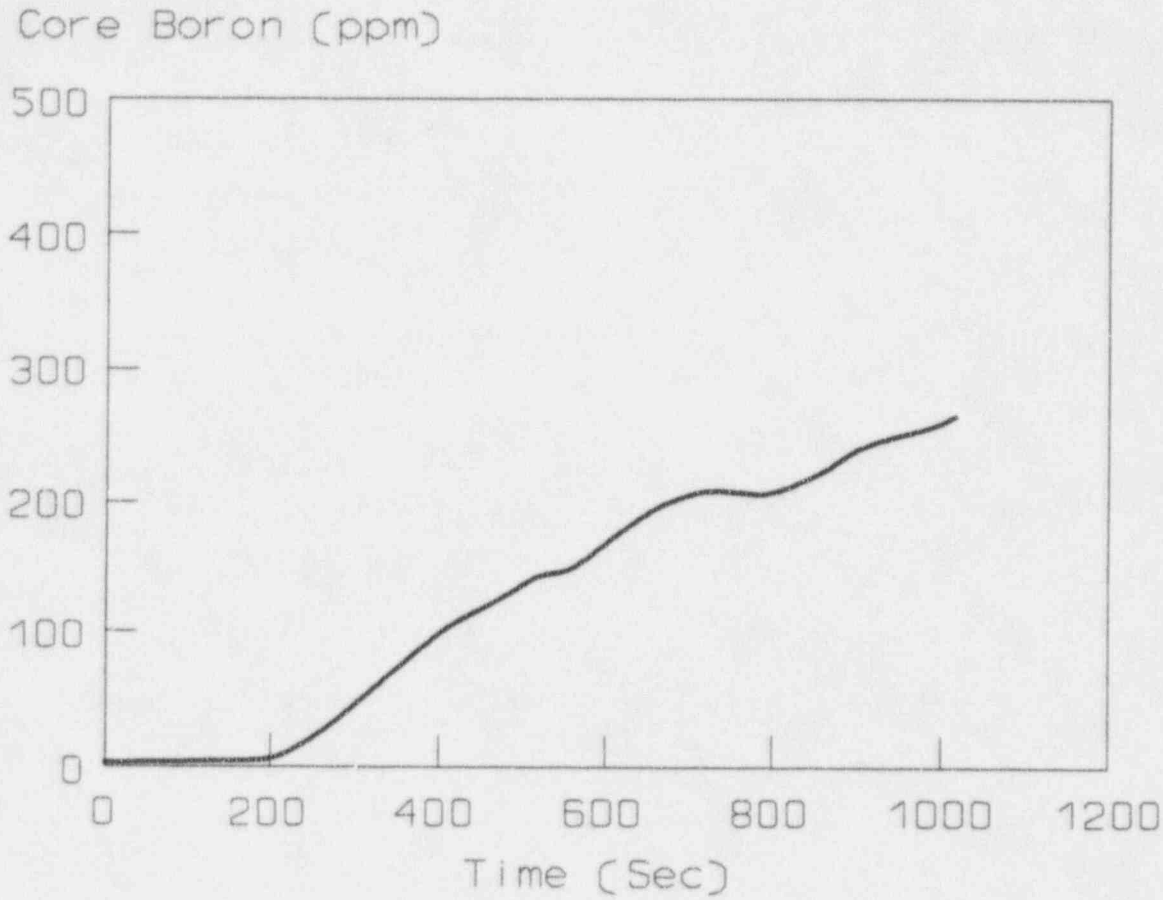
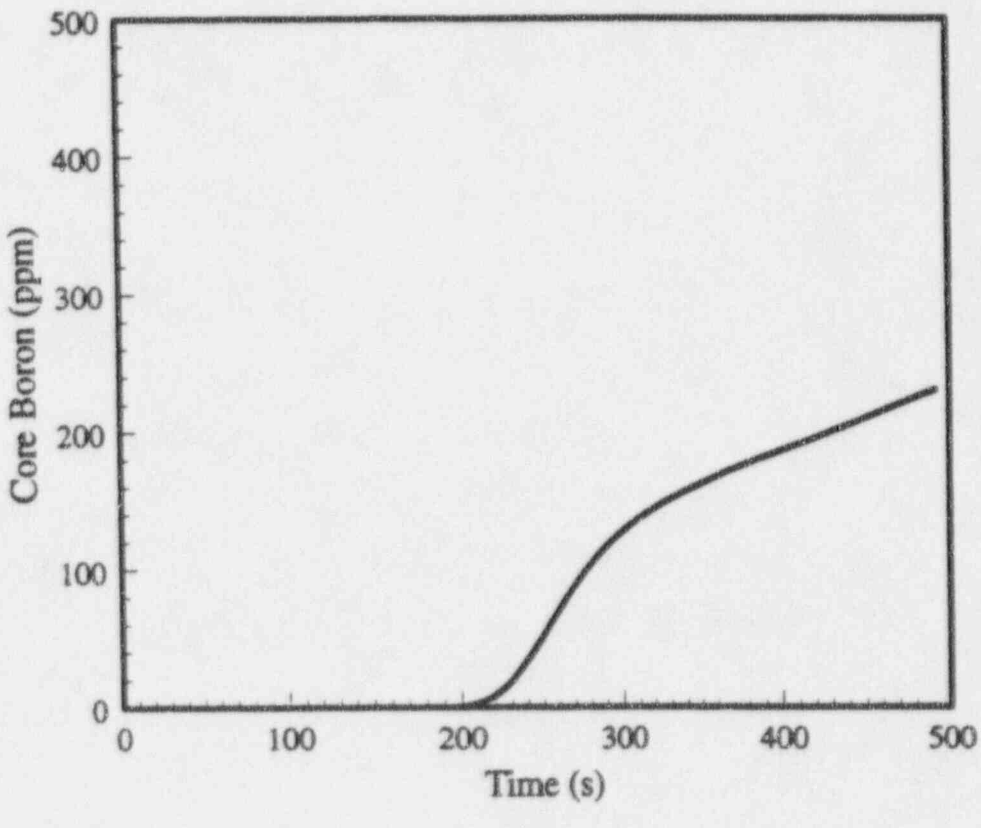


Figure 15.1.4-11

Core Boron Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Figure 15.1.4-11 Steam Line Break Events



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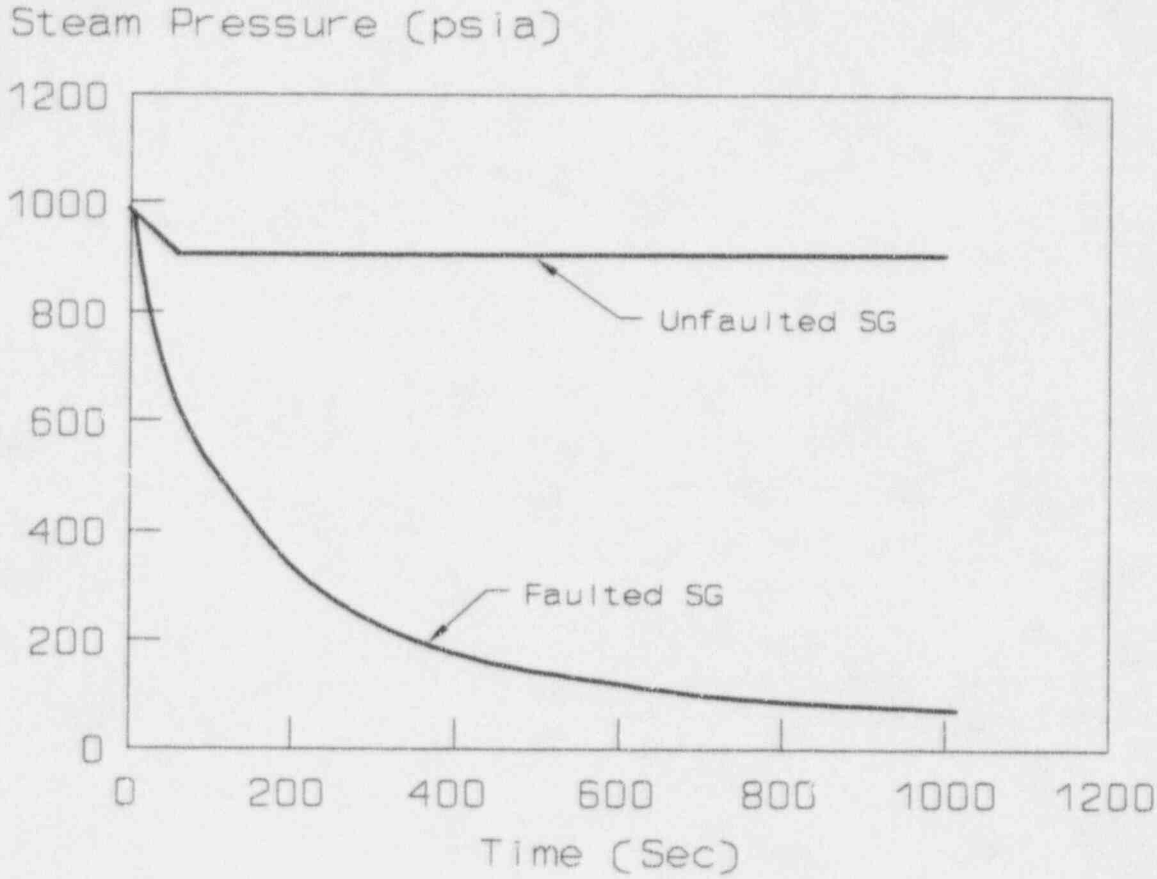
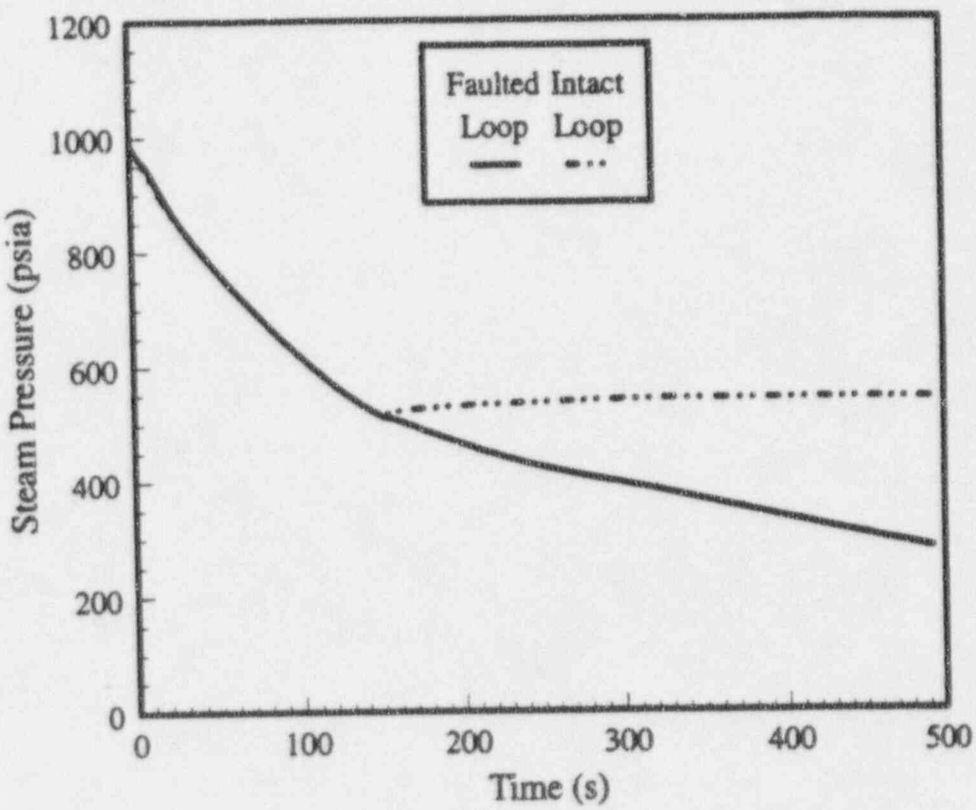


Figure 15.1.4-12

Steam Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Figure 15.1.4-12 Steam Line Break Events



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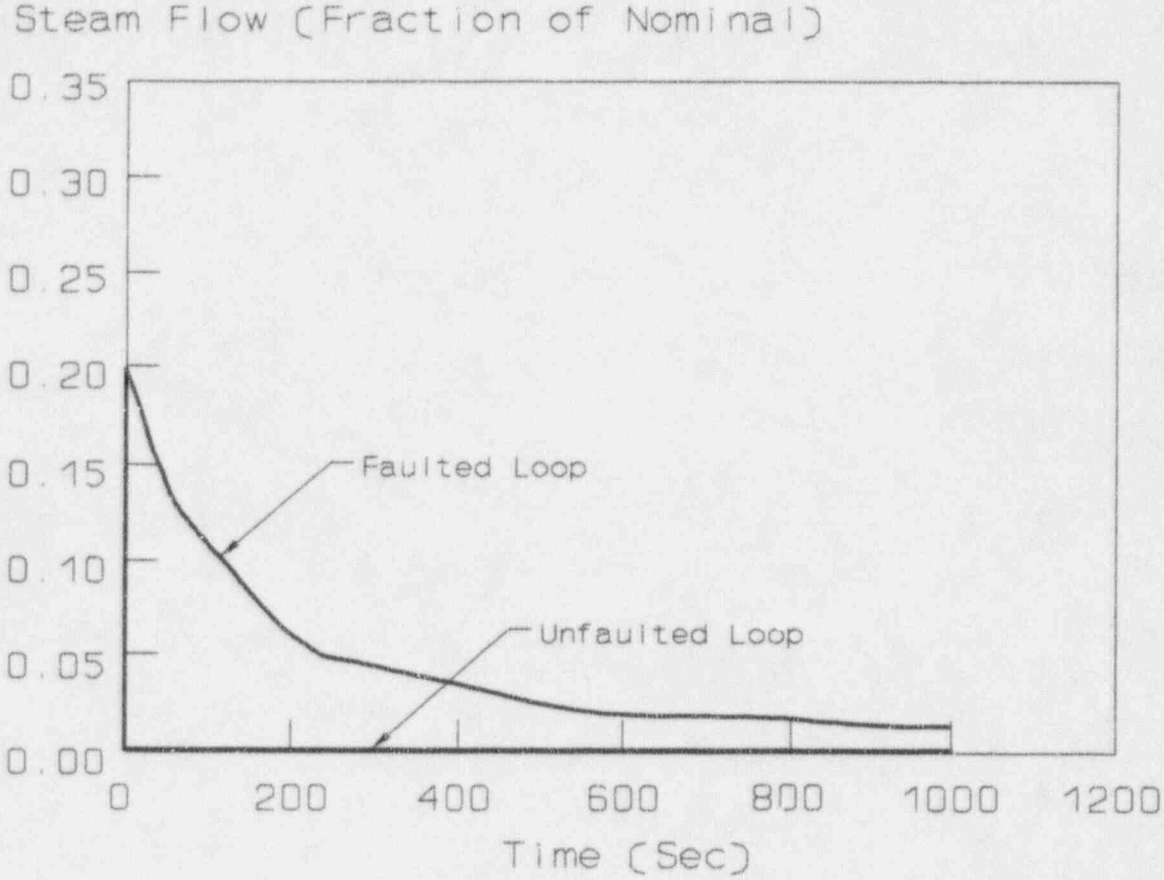
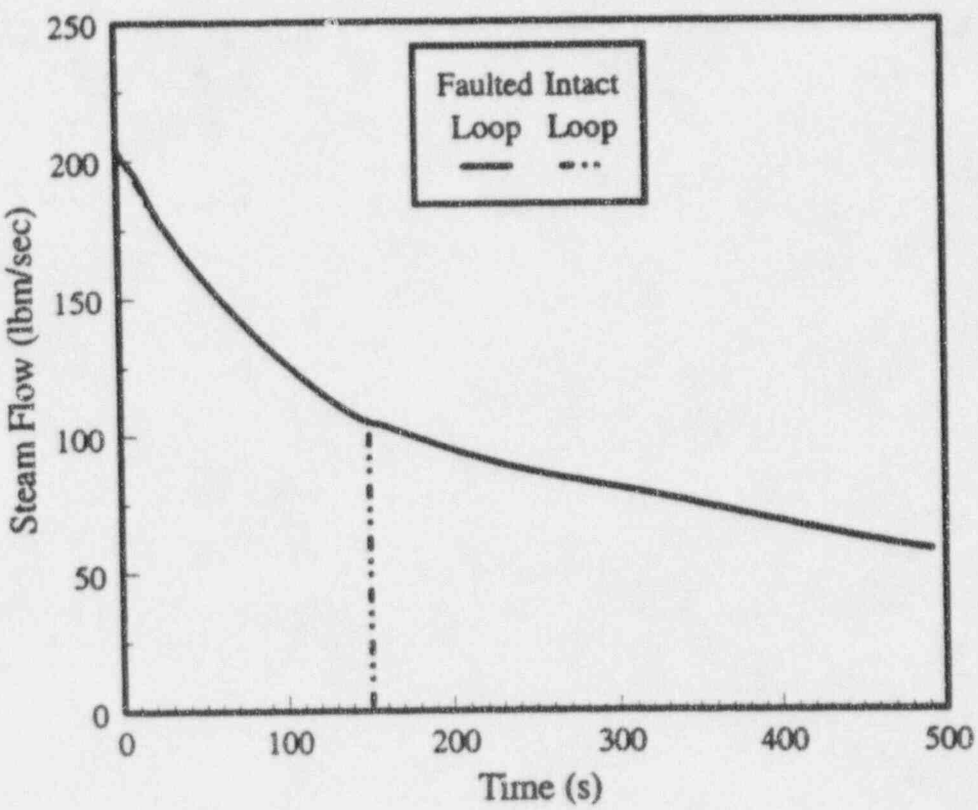


Figure 15.1.4-13

Steam Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Figure 15.1.4-13 Steam Line Break Events



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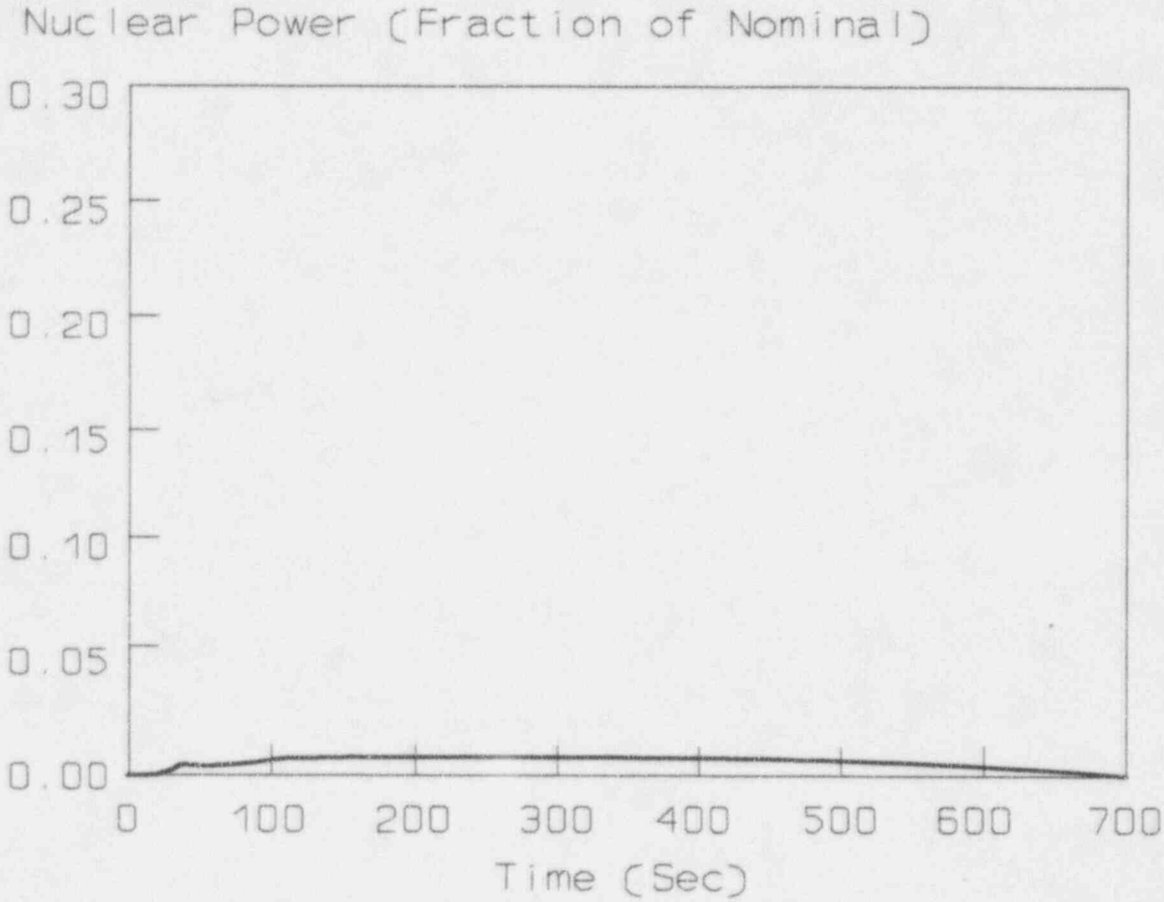
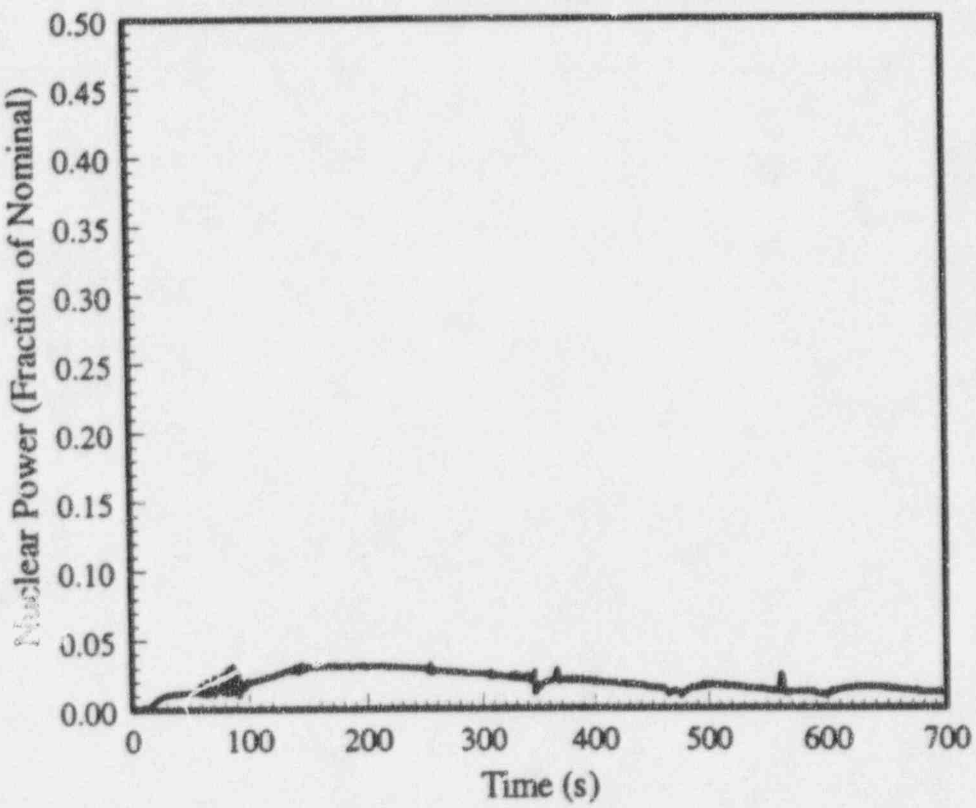


Figure 15.1.5-1

Nuclear Power Transient Steam System Piping Failure

even pg

Figure 15.1.5-1 Steam System Piping Failure



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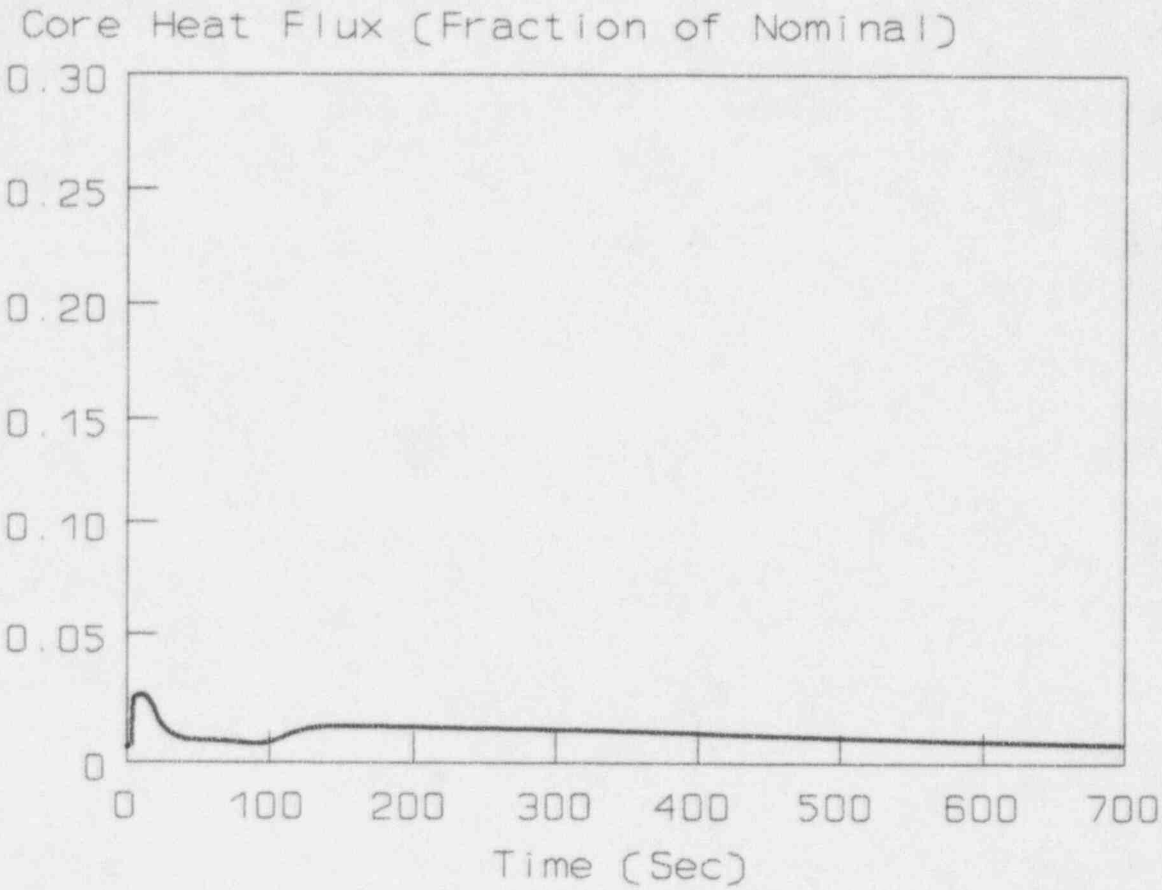
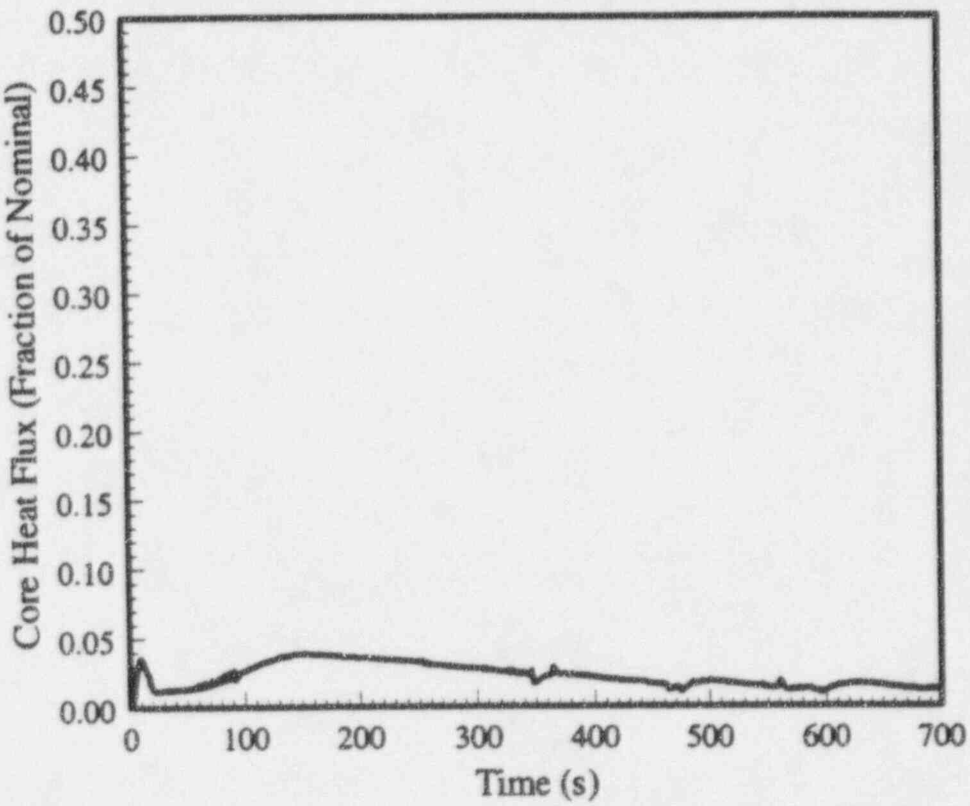


Figure 15.1.5-2

Core Heat Flux Transient Steam System Piping Failure

even pg

Figure 15.1.5-2 Steam System Piping Failure



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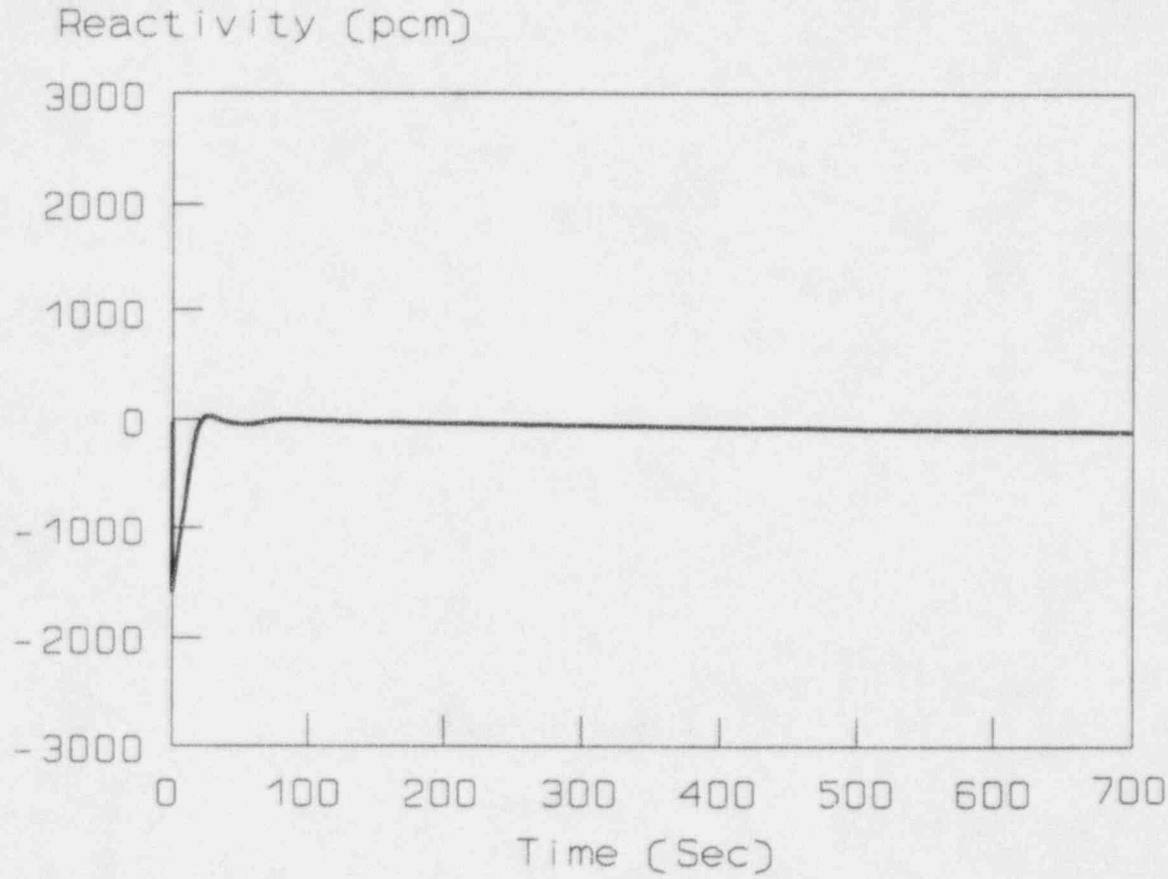
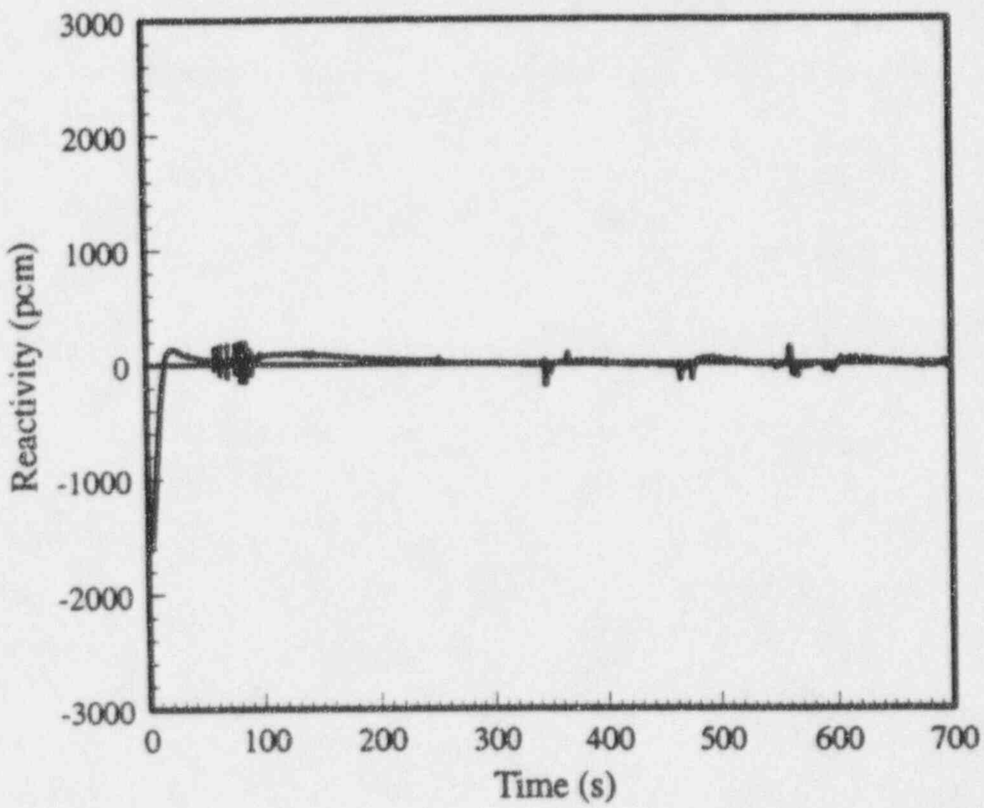


Figure 15.1.5-3

Reactivity Transient Steam System Piping Failure

even pg

Figure 15.1.5-3 Steam System Piping Failure



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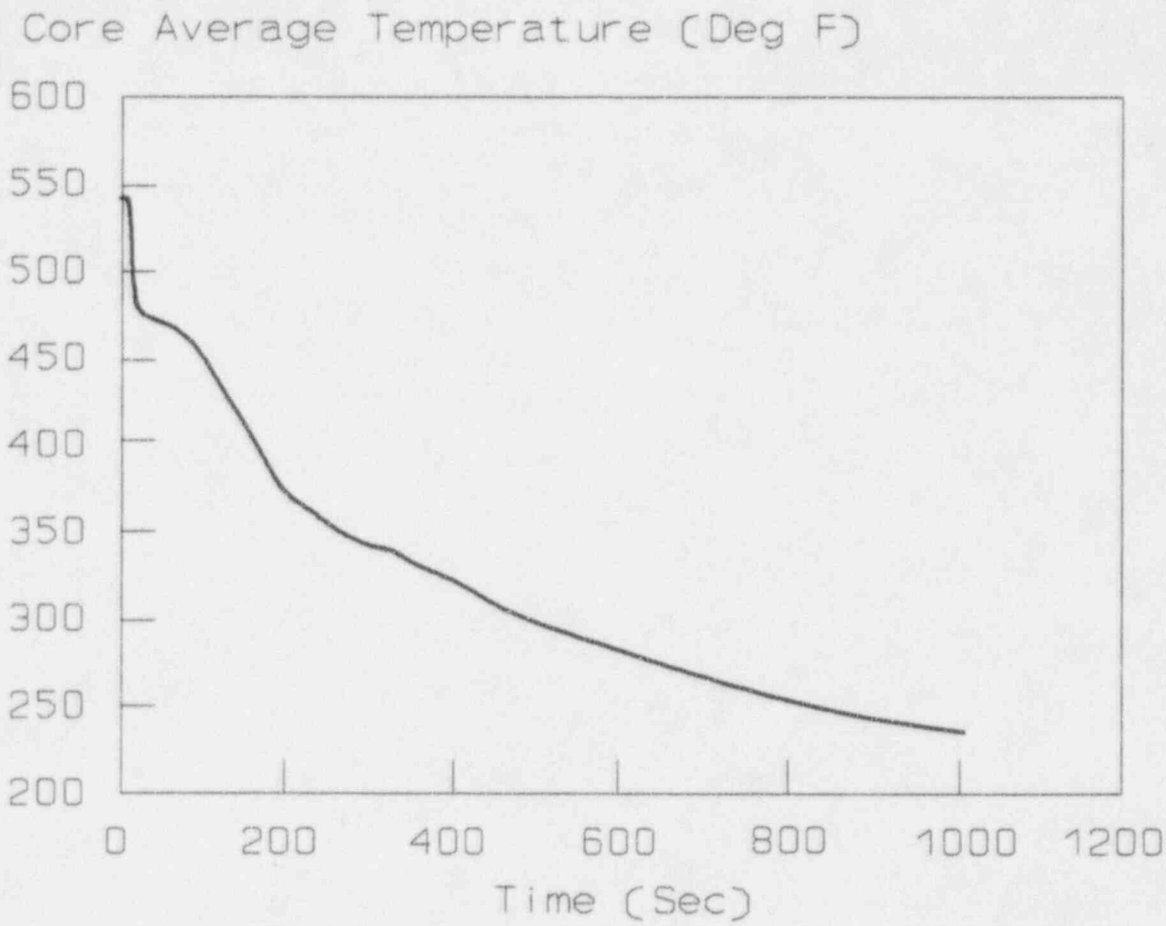

Figure 15.1.5-4

**Core Average Temperature Transient
Steam System Piping Failure**

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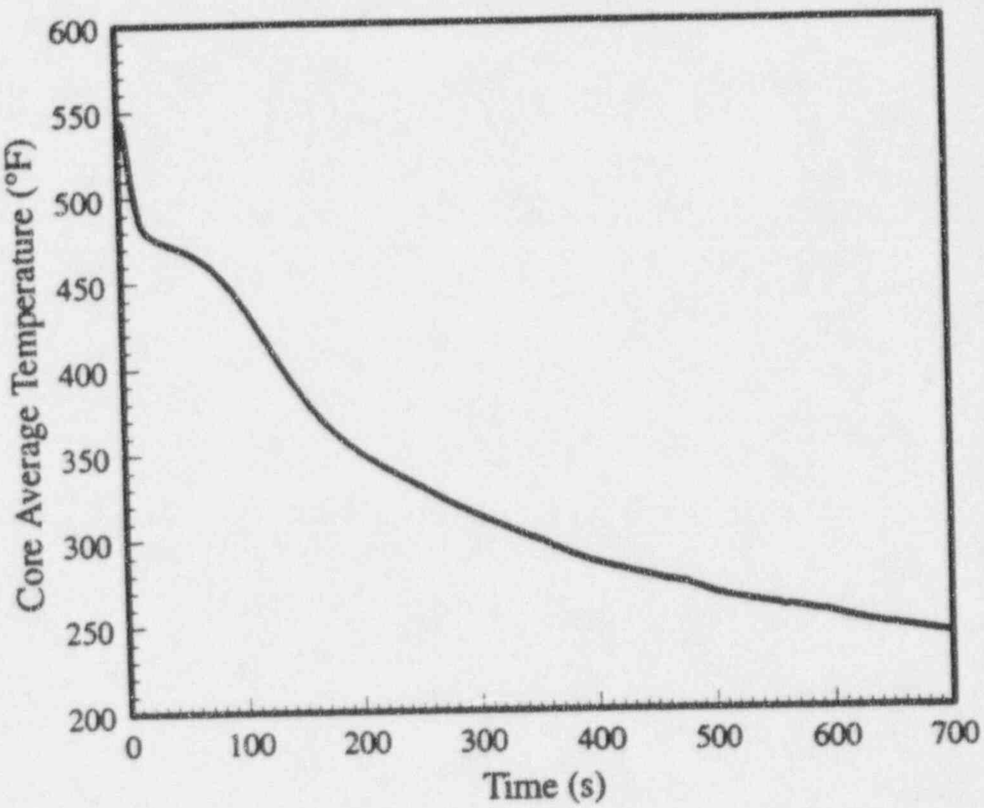
Figure 15.1.5-4 Steam System Piping Failure



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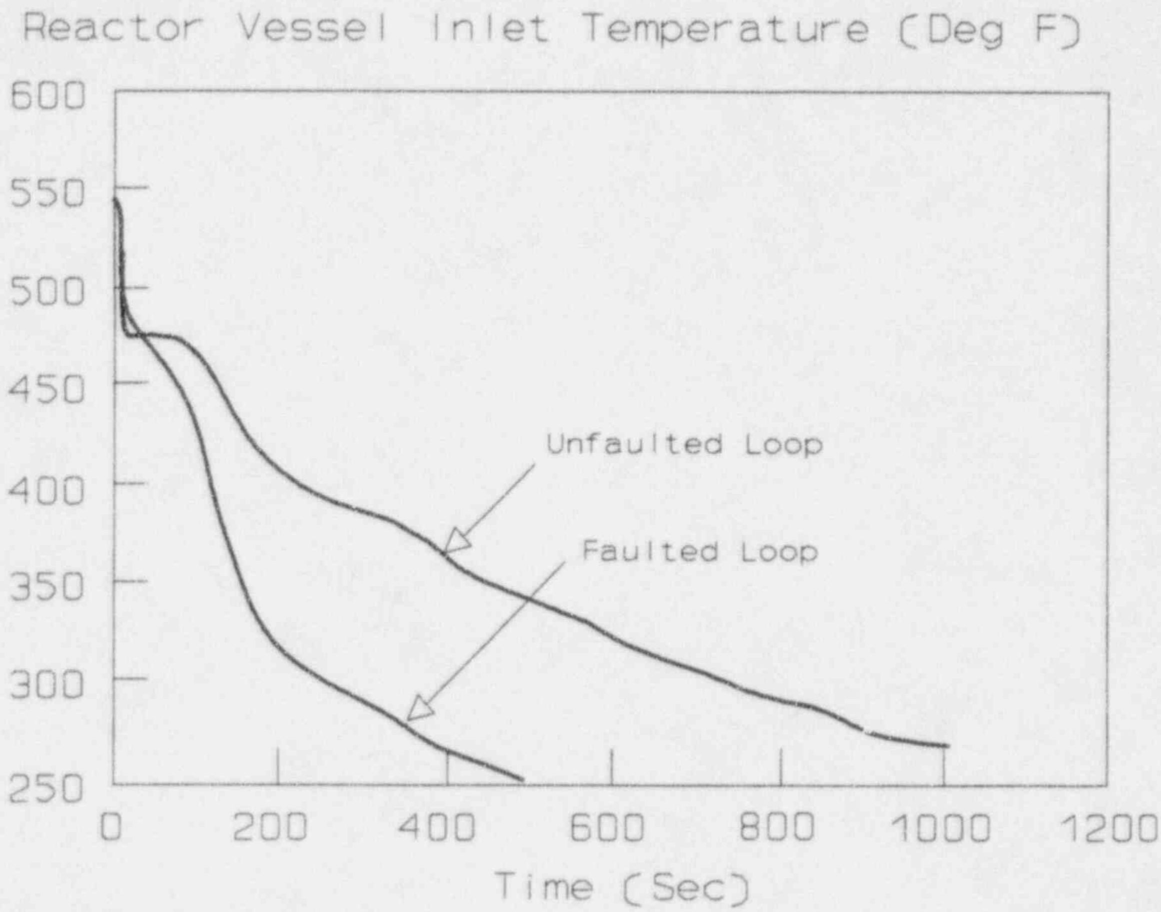
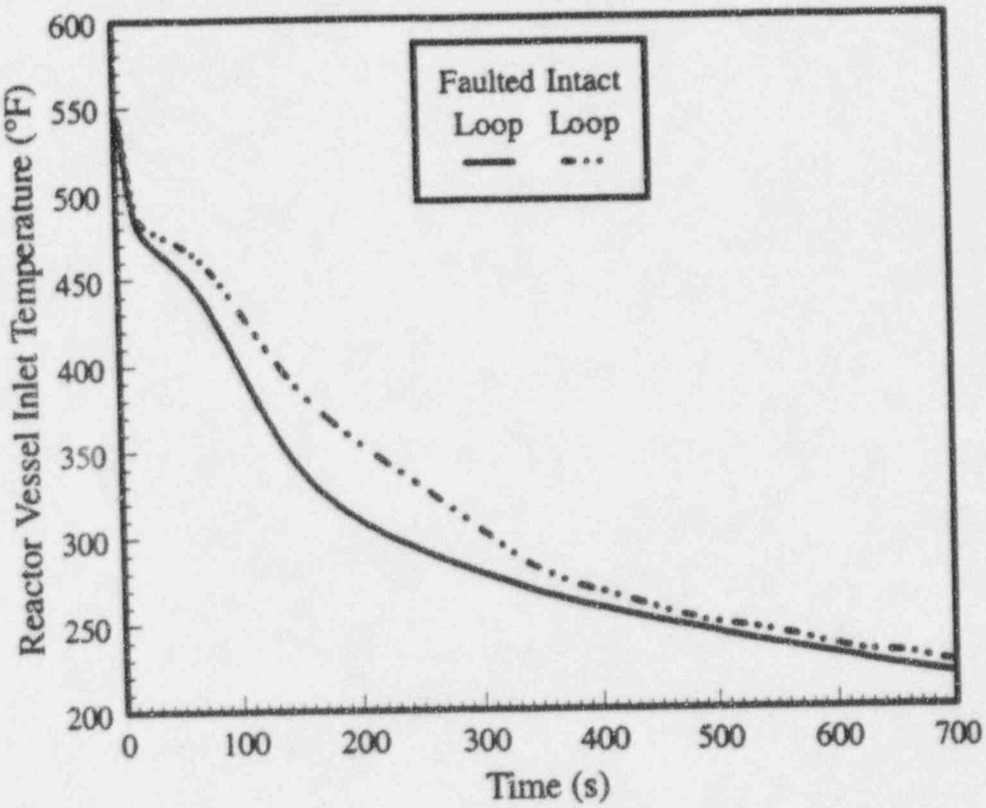


Figure 15.1.5-5

Reactor Vessel Inlet Temperature Transient
Steam System Piping Failure

Figure 15.1.5-5 Steam System Piping Failure



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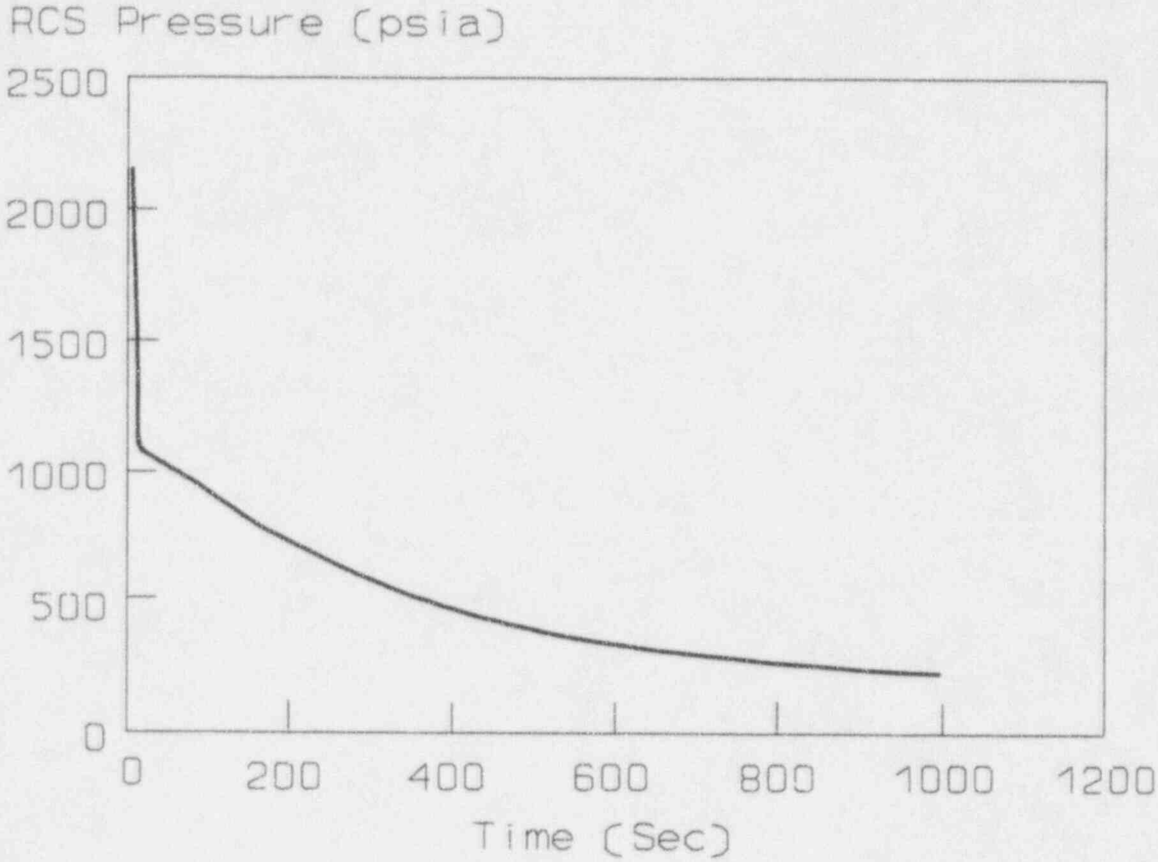

Figure 15.1.5-6

Reactor Coolant System Pressure Transient
Steam System Piping Failure

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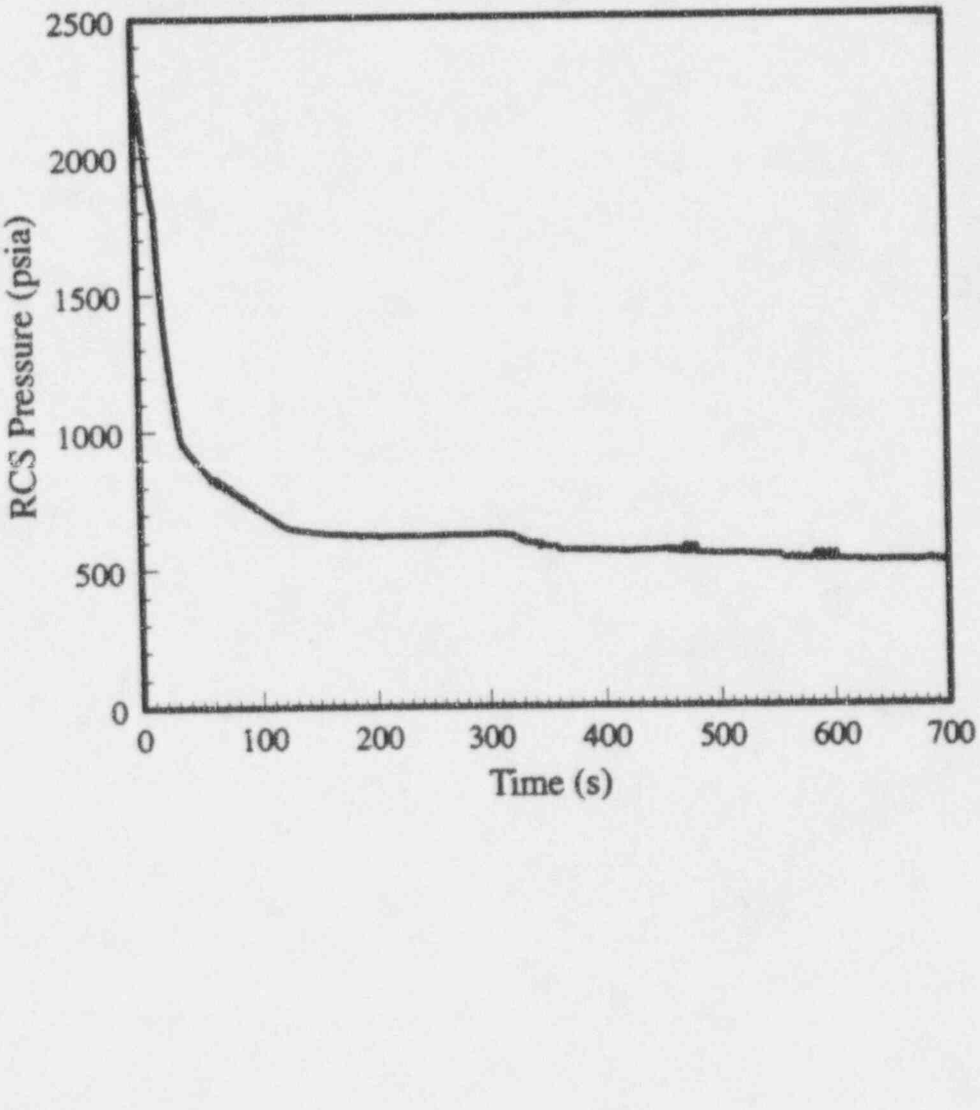
Figure 15.1.5-6 Steam System Piping Failure



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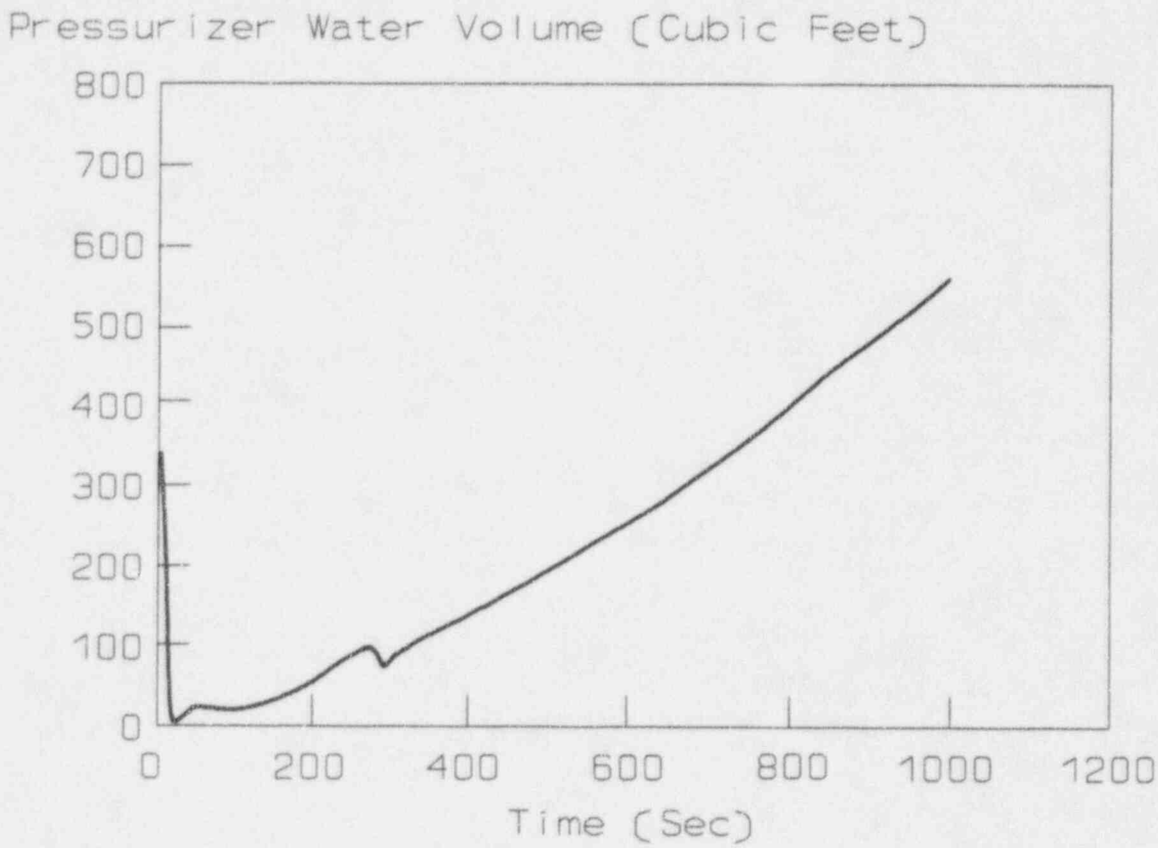

Figure 15.1.5-7

Pressurizer Water Volume Transient
Steam System Piping Failure

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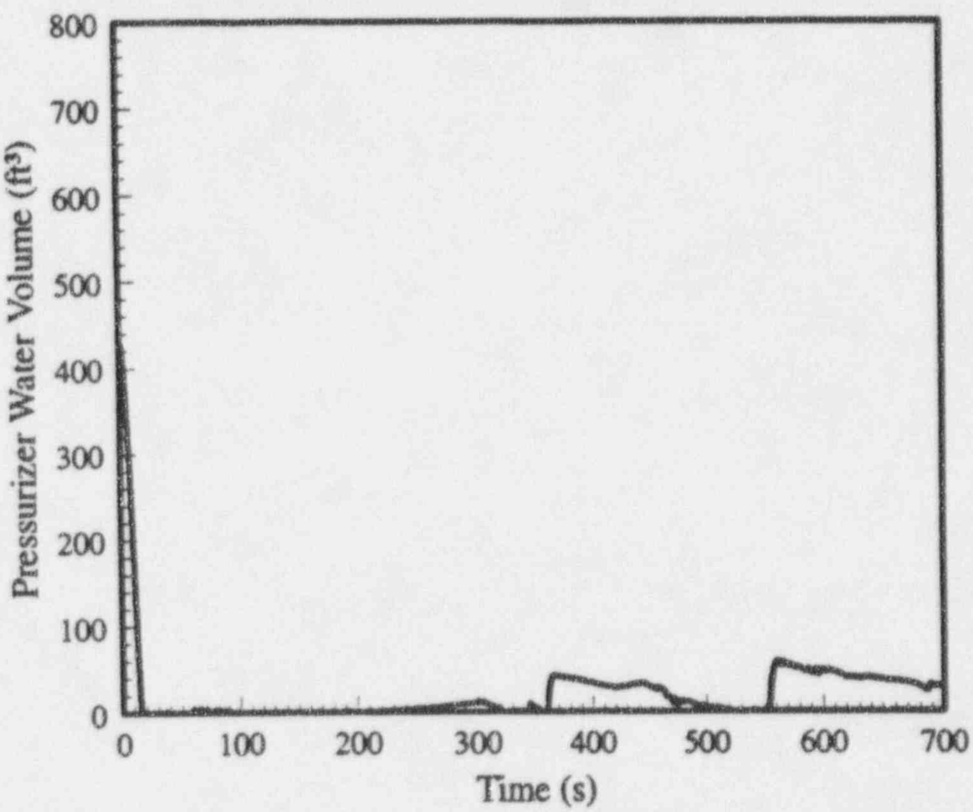
Figure 15.1.5-7 Steam System Piping Failure



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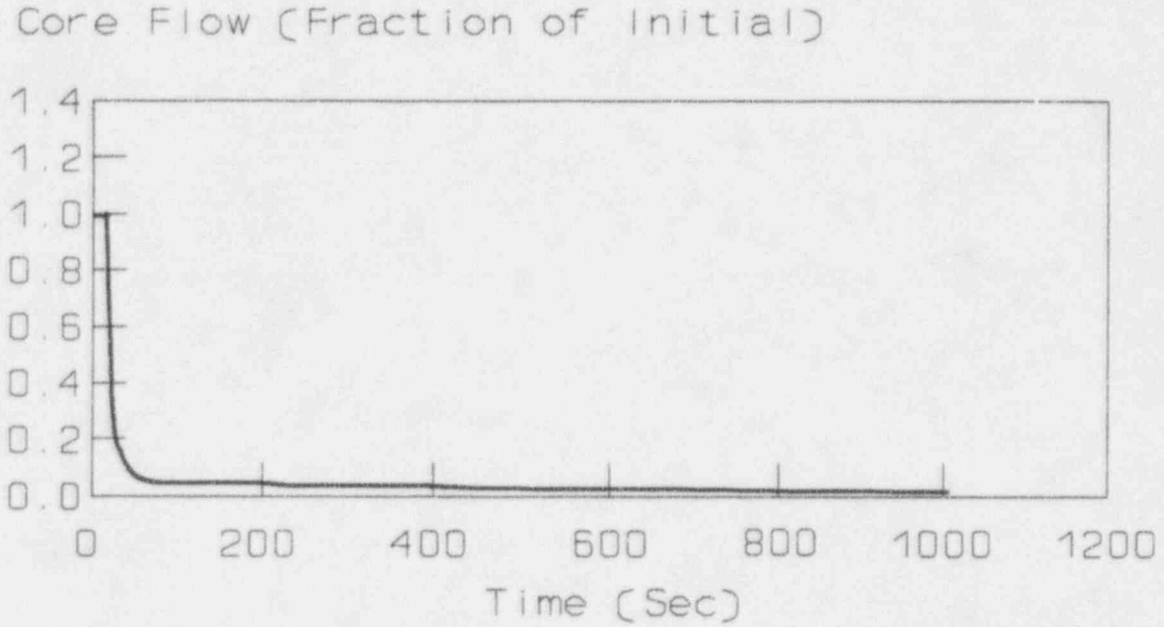

Figure 15.1.5-8

Core Flow Transient Steam System Piping Failure

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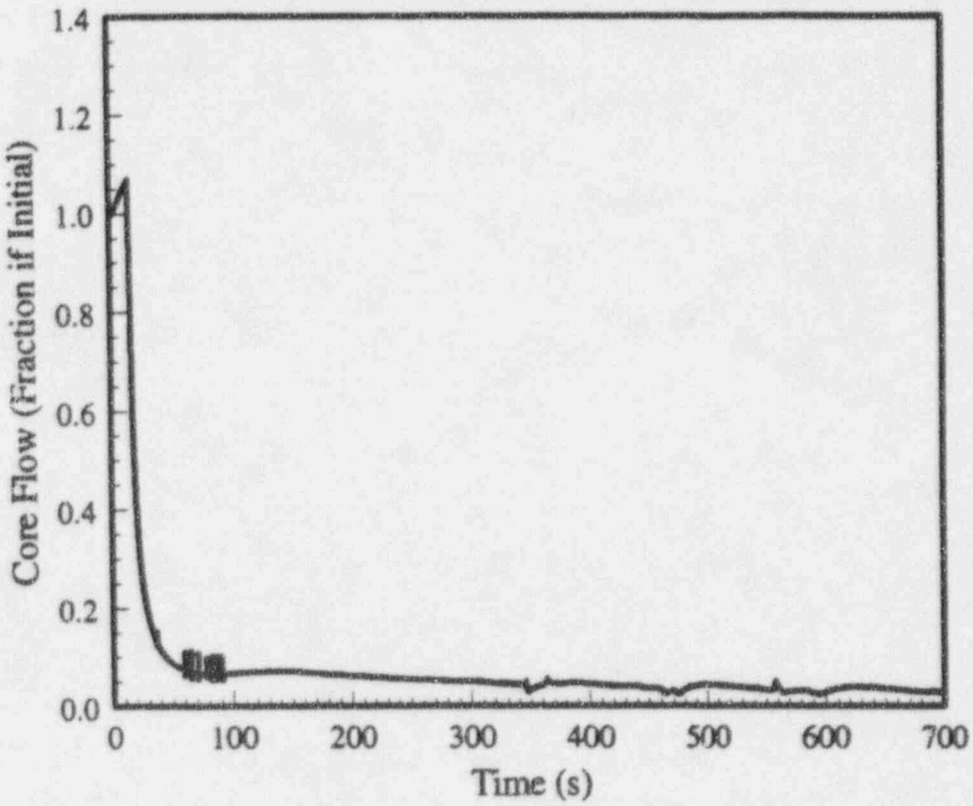
Figure 15.1.5-8 Steam System Piping Failure



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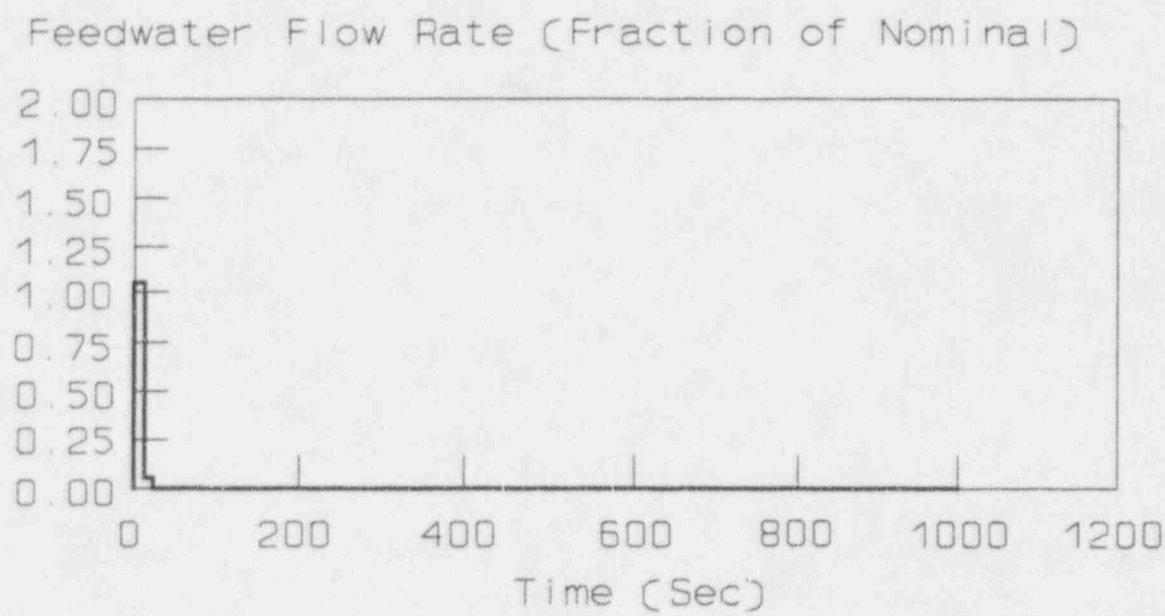
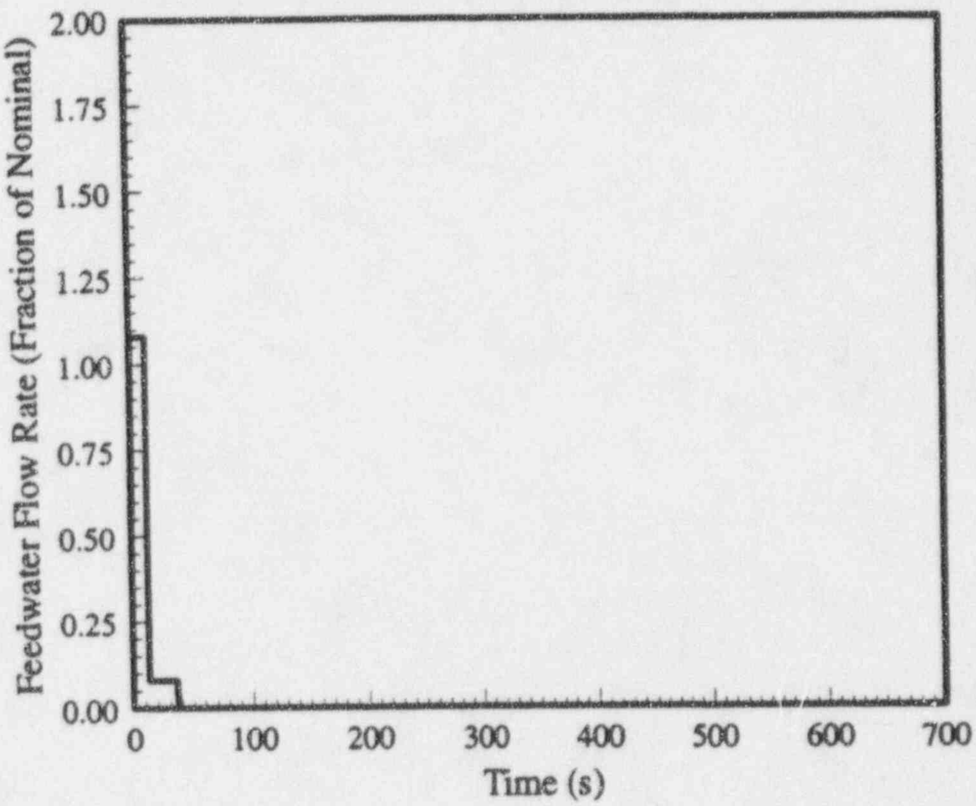


Figure 15.1.5-9

Feedwater Flow Transient Steam System Piping Failure

Figure 15.1.5-9 Steam System Piping Failure



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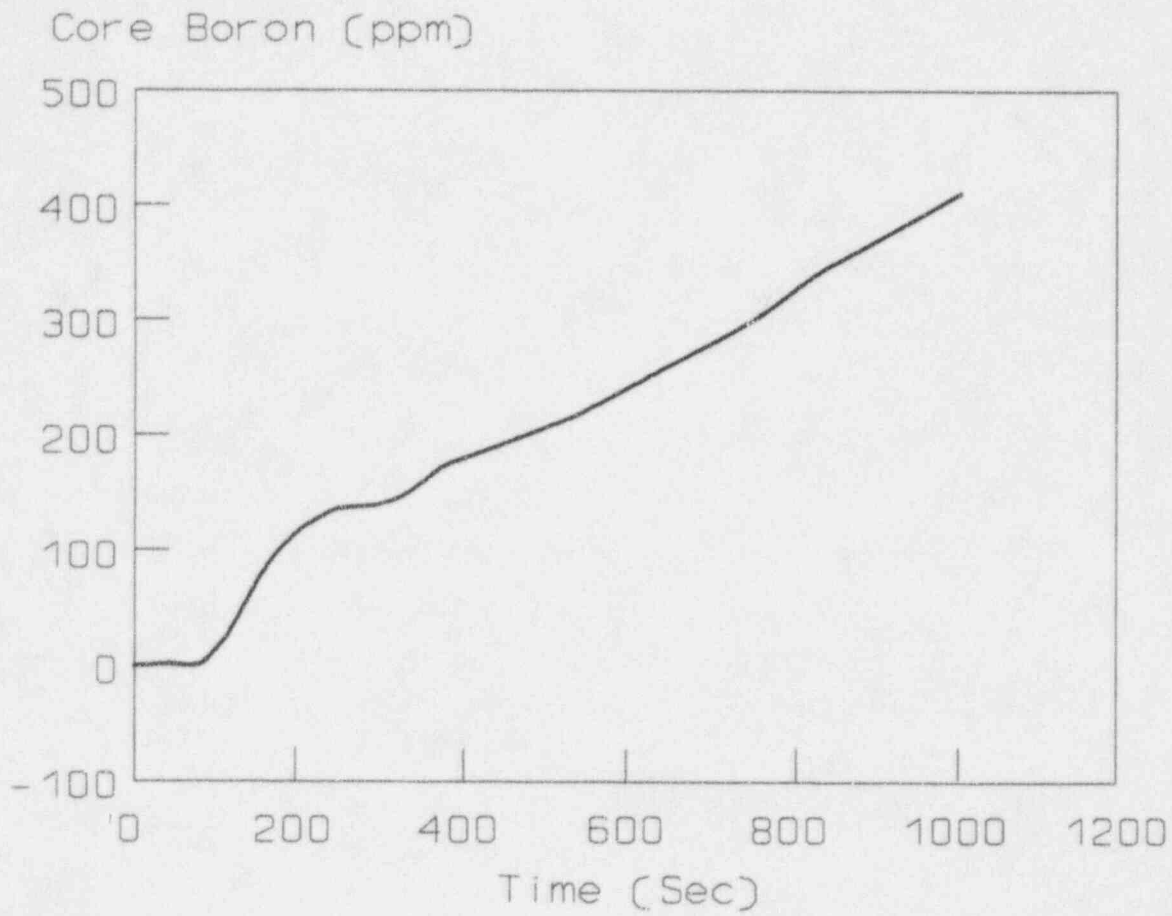

Figure 15.1.5-10

Core Boron Transient Steam System Piping Failure

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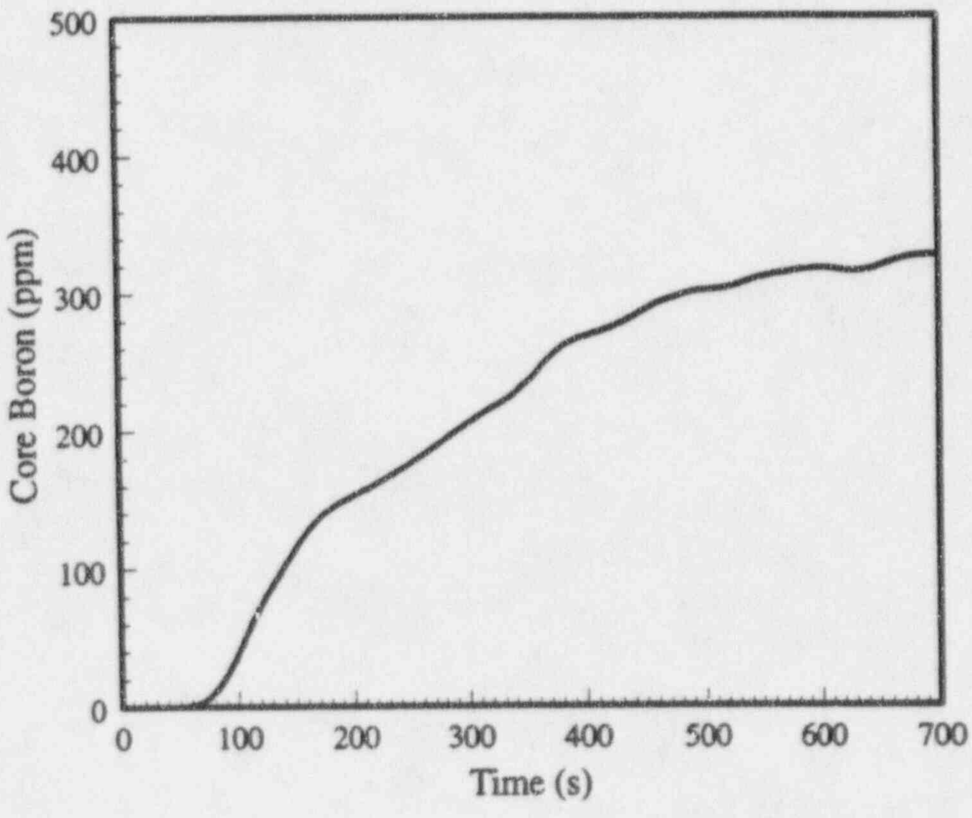
Figure 15.1.5-10 Steam System Piping Failure



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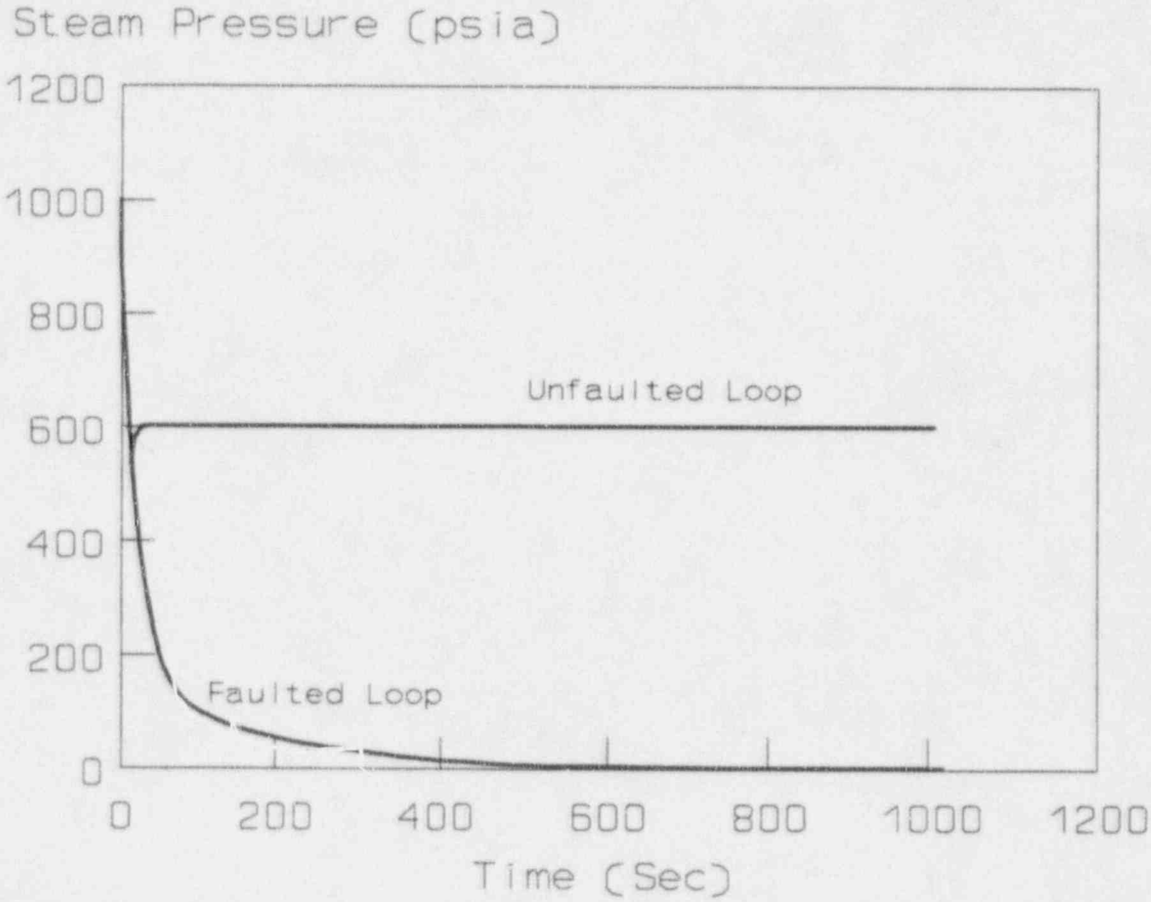

Figure 15.1.5-11

Steam Pressure Transient Steam System Piping Failure

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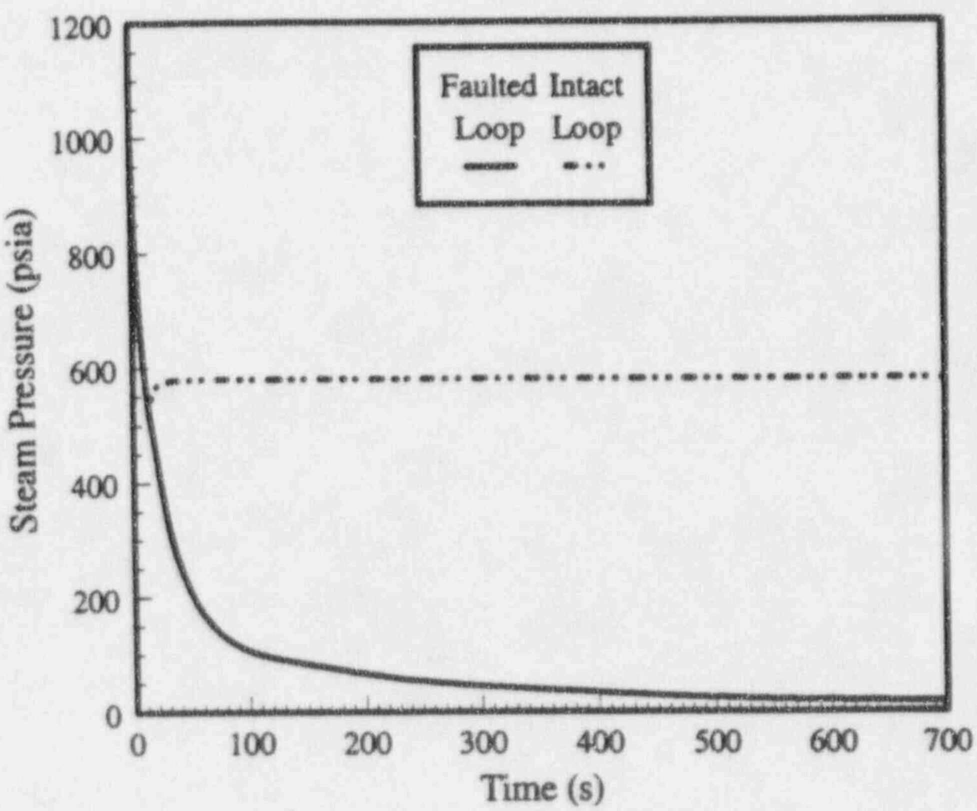
Figure 15.1.5-11 Steam System Piping Failure



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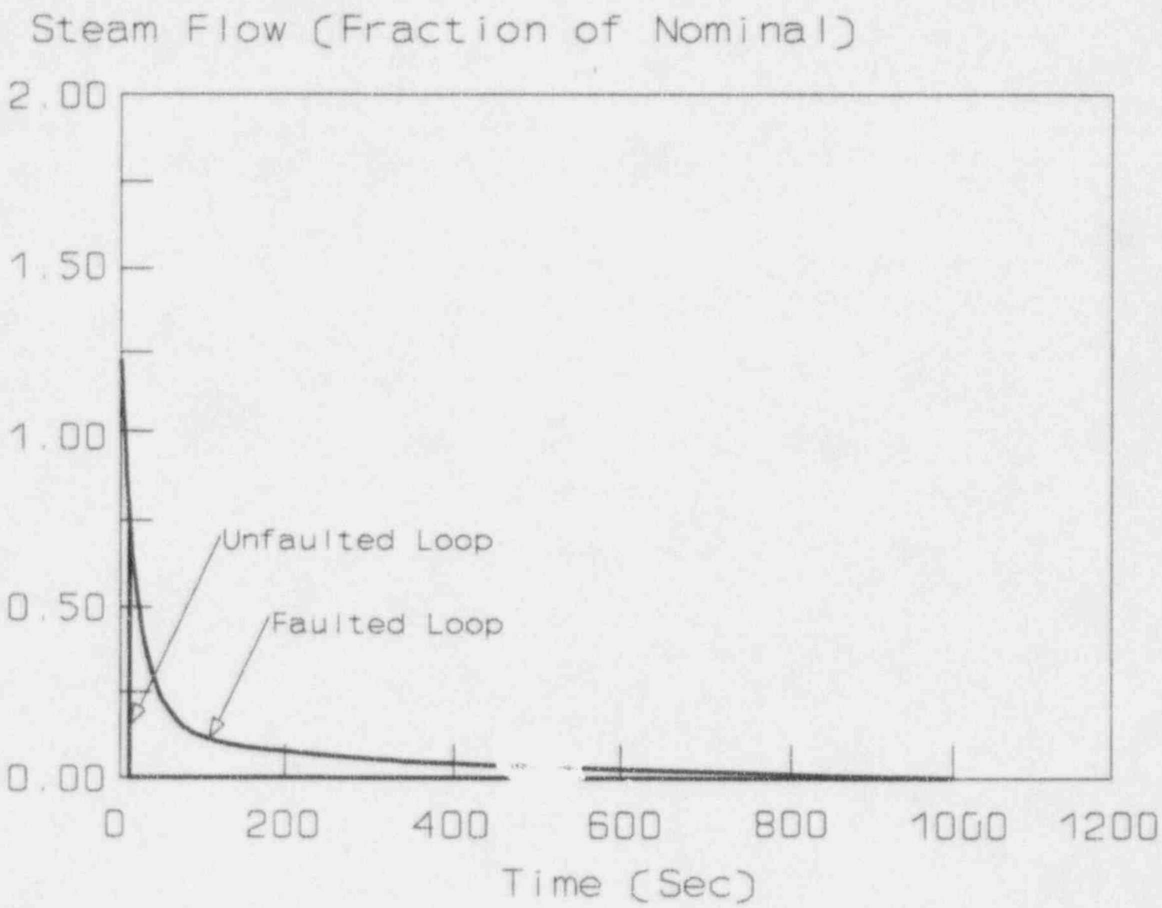

Figure 15.1.5-12

Steam Flow Transient Steam System Piping Failure

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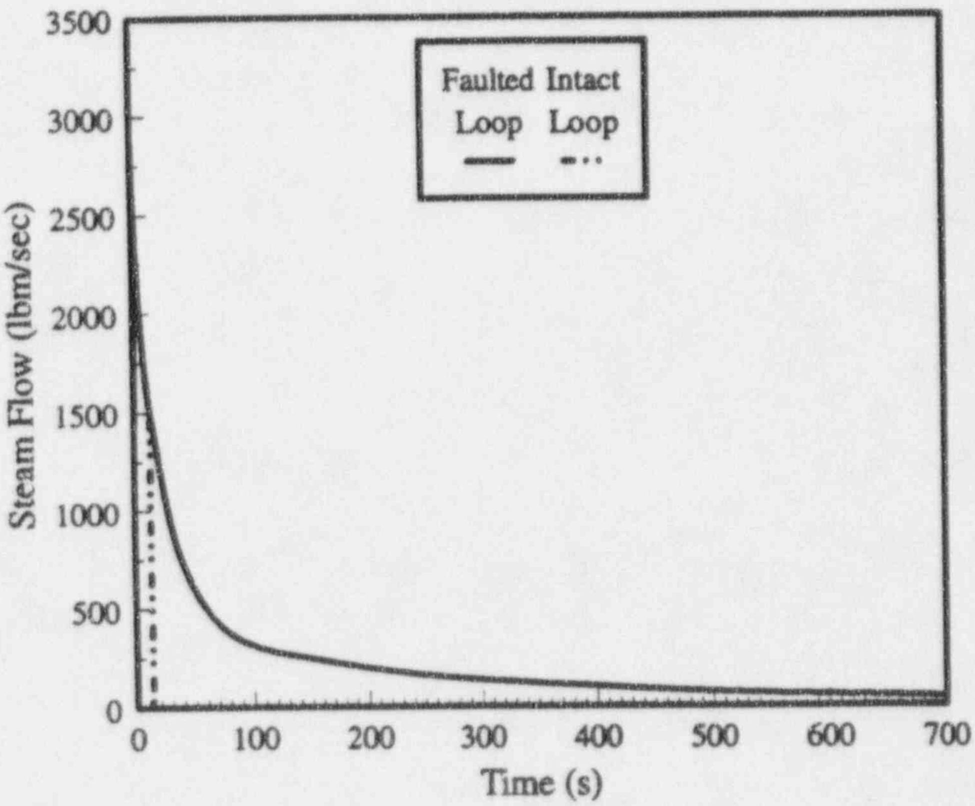
Figure 15.1.5-12 Steam System Piping Failure



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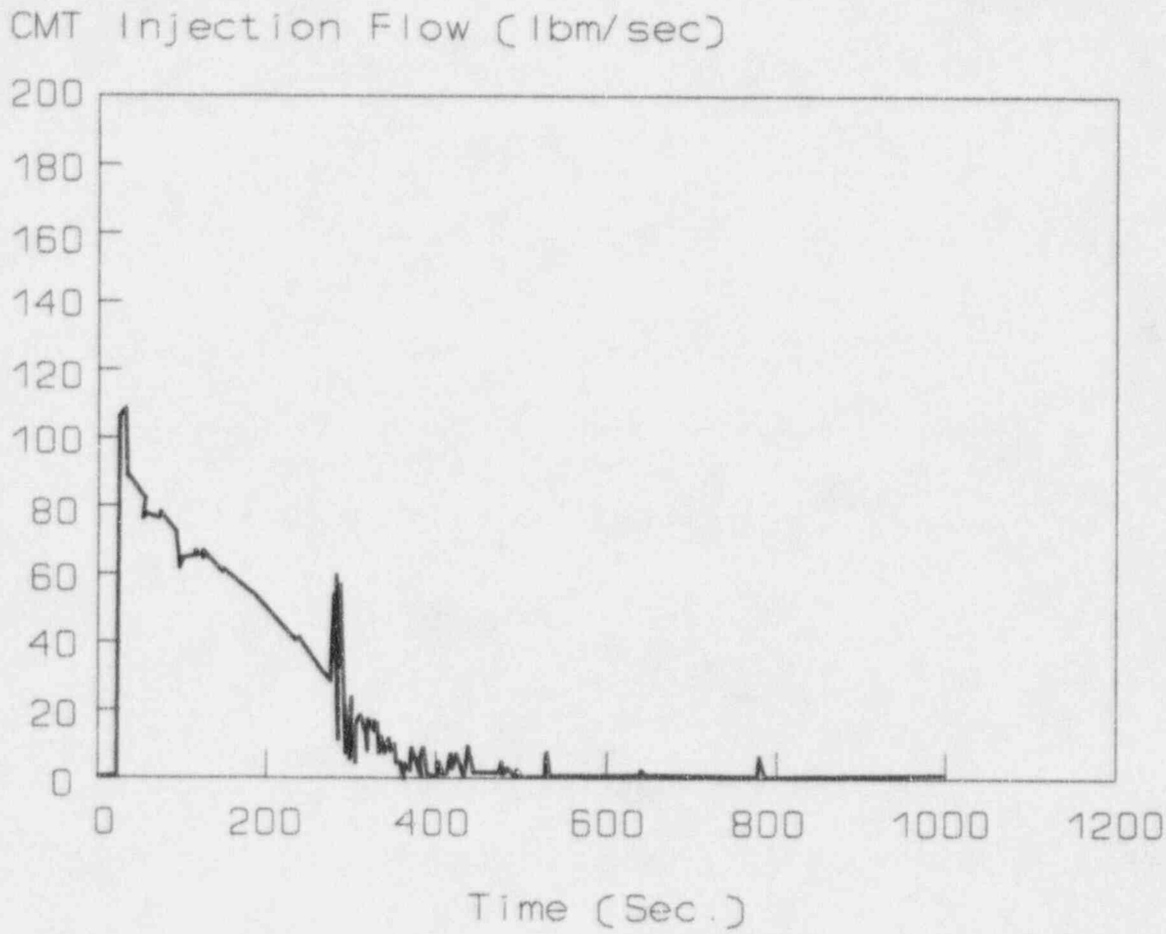

Figure 15.1.5-13

Core Makeup Tank Injection Flow
Steam System Piping Failure

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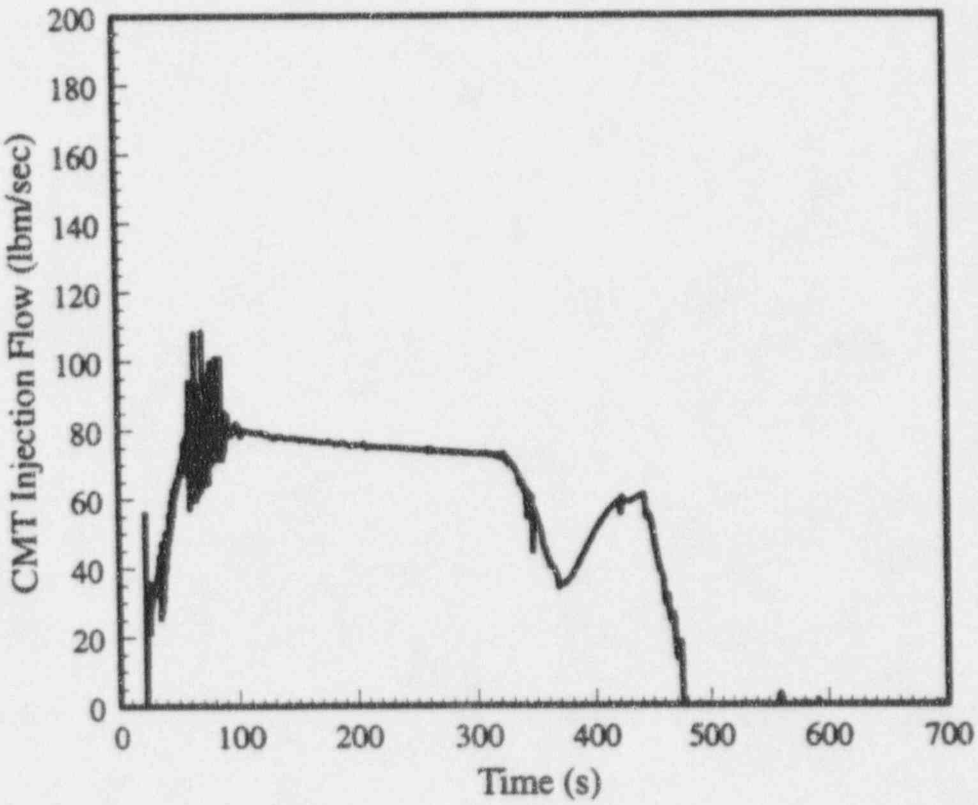
Figure 15.1.5-13 Steam System Piping Failure



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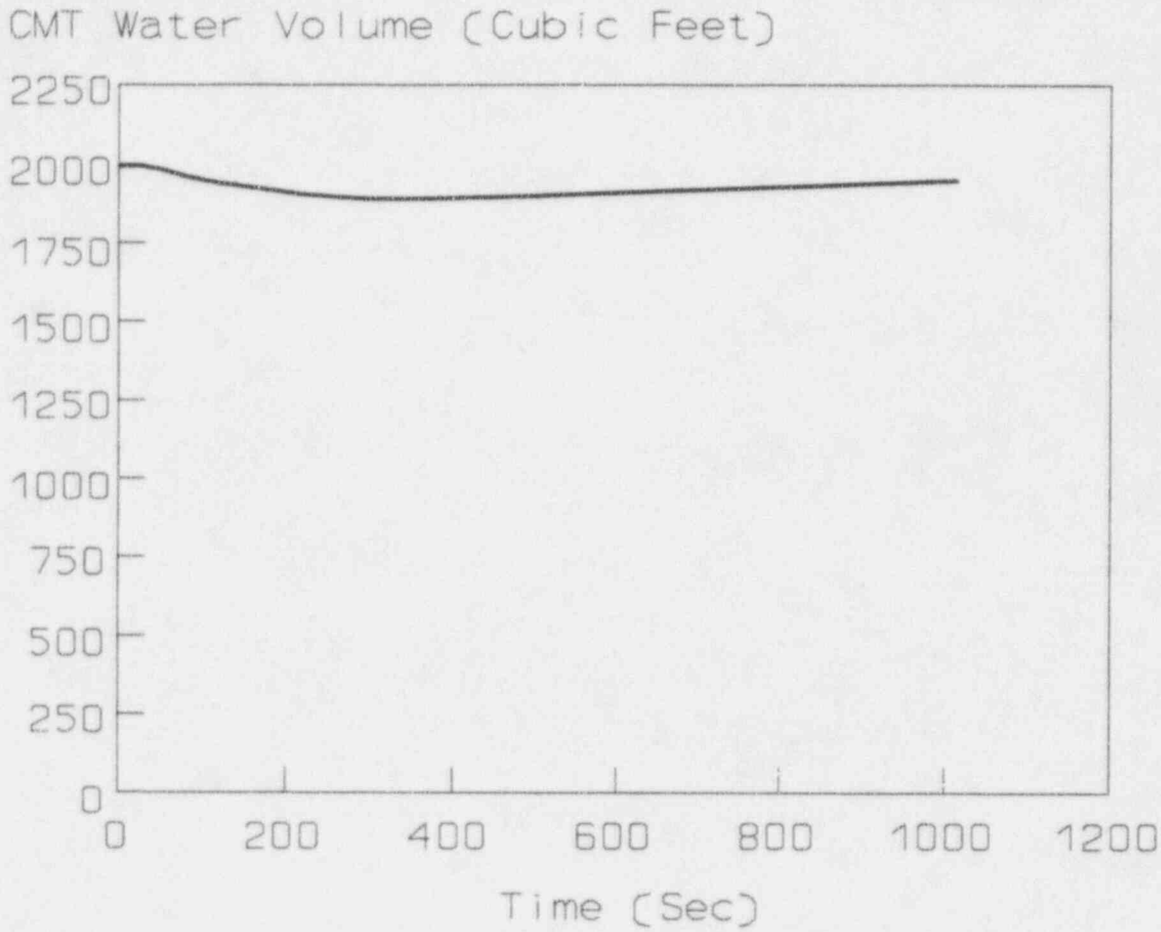

Figure 15.1.5-14

Core Makeup Tank Water Volume Steam System Piping Failure

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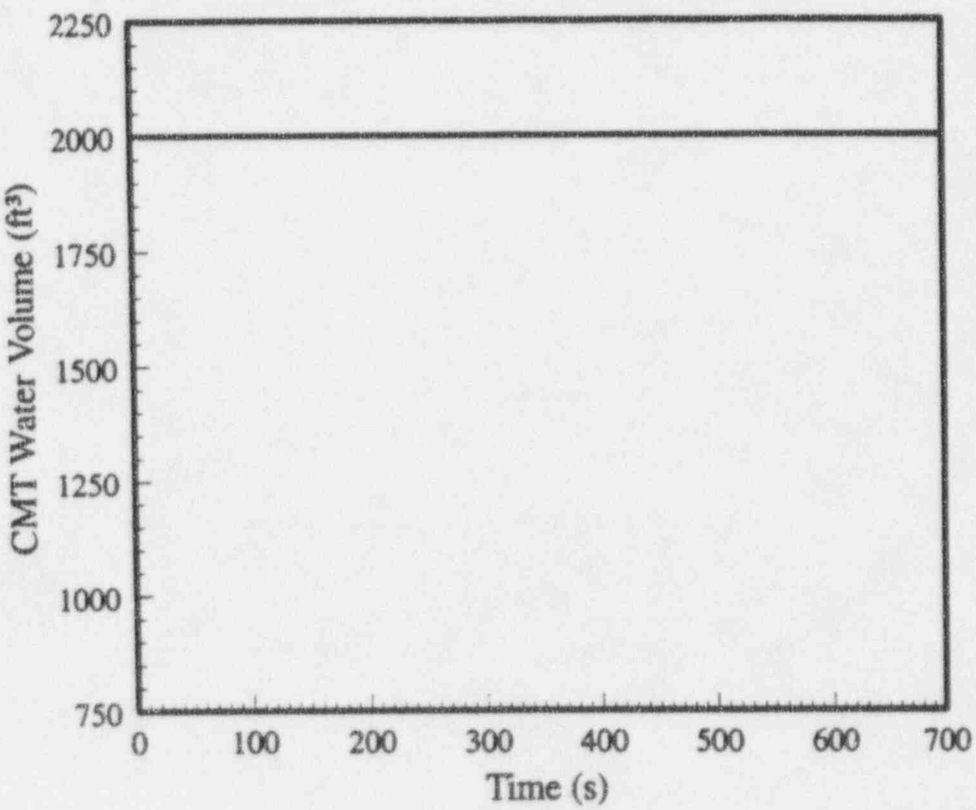
Figure 15.1.5-14 Steam System Piping Failure



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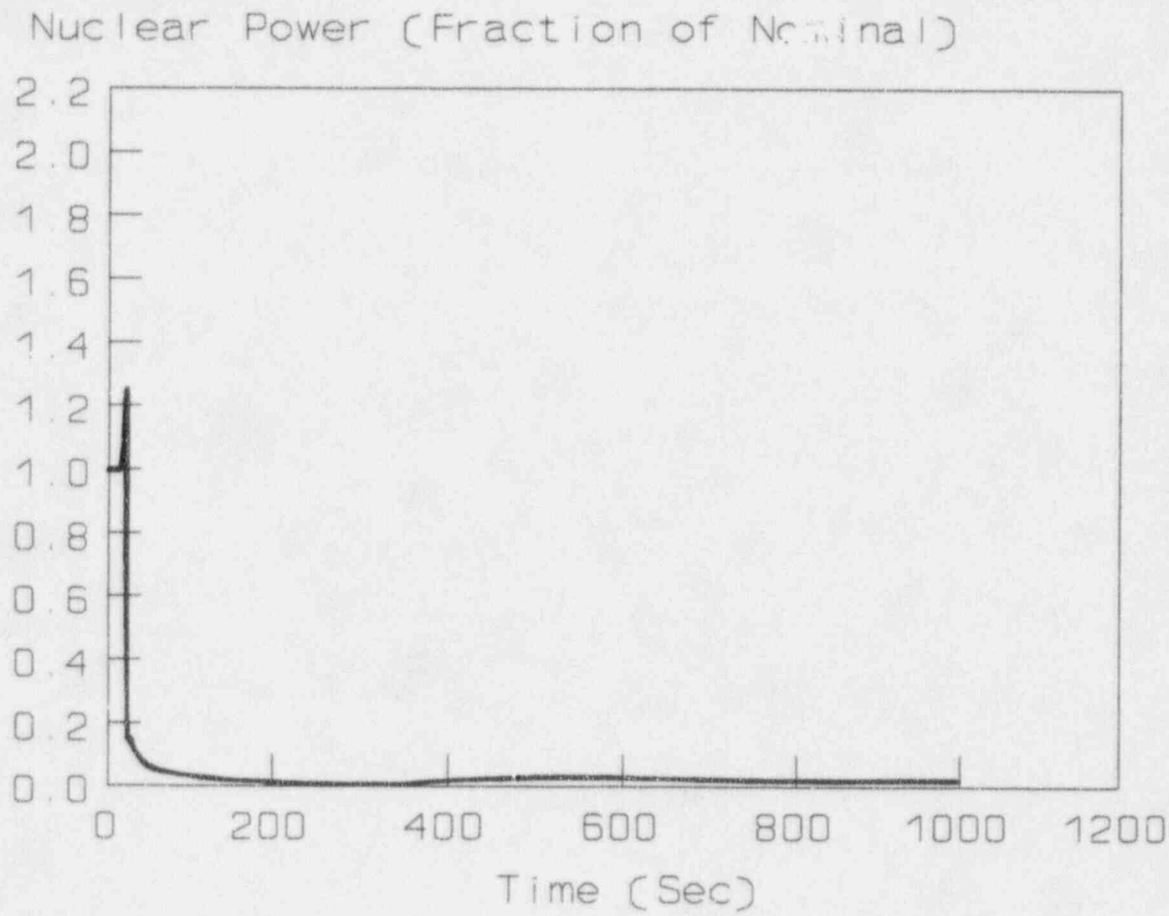

Figure 15.1.6-1

Nuclear Power Transient Inadvertent Operation of the PRHR

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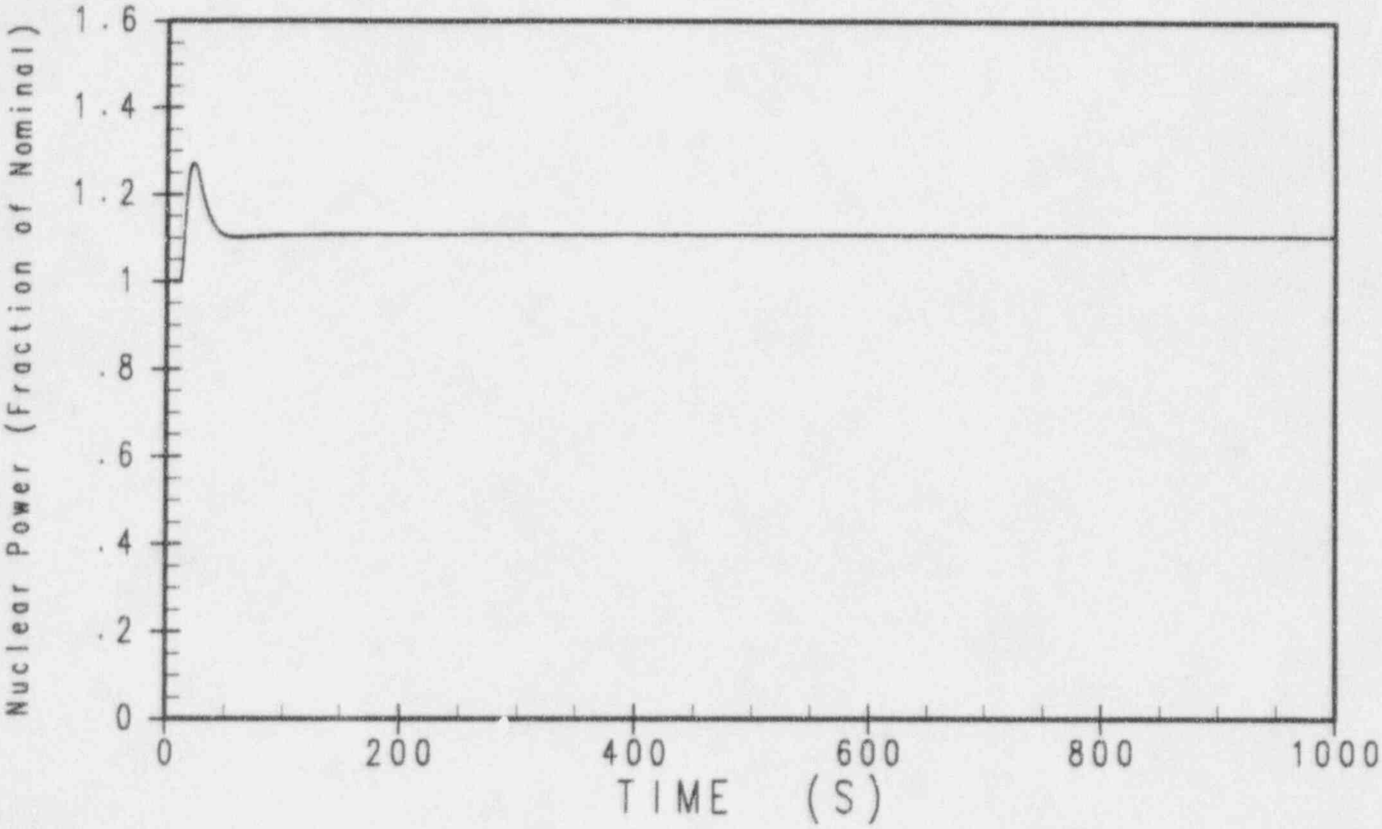
Figure 15.1.6-1



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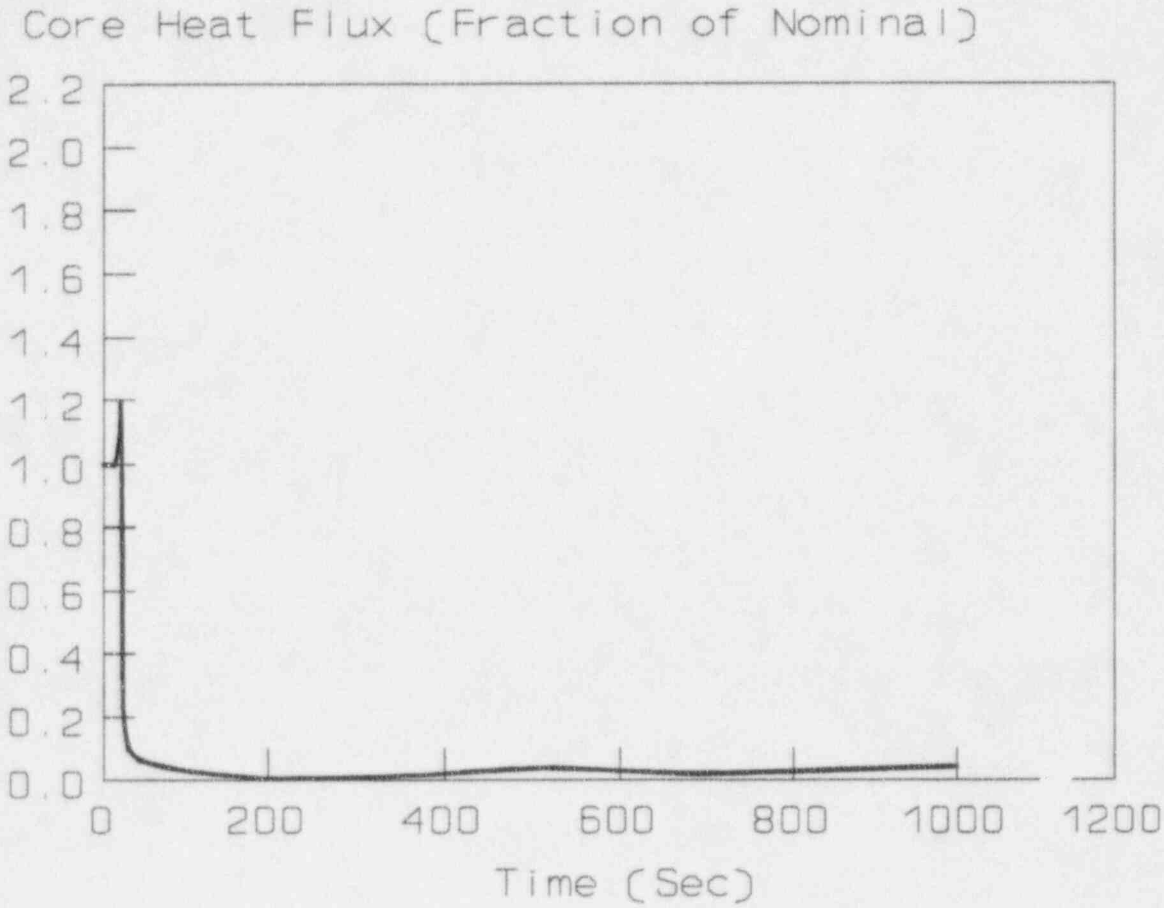
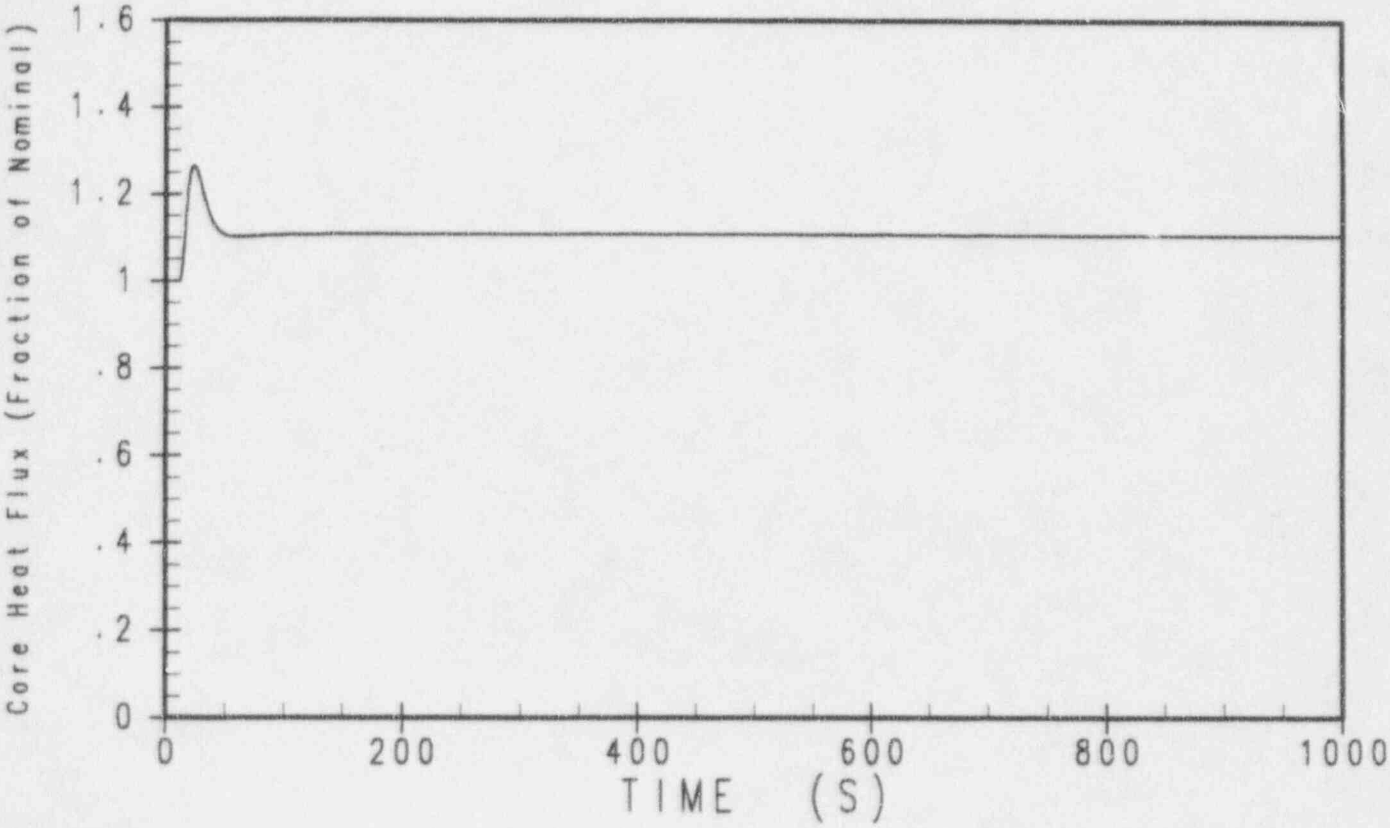


Figure 15.1.6-2

Core Heat Flux Transient Inadvertent Operation of the PRHR

Figure 15.1.6-2



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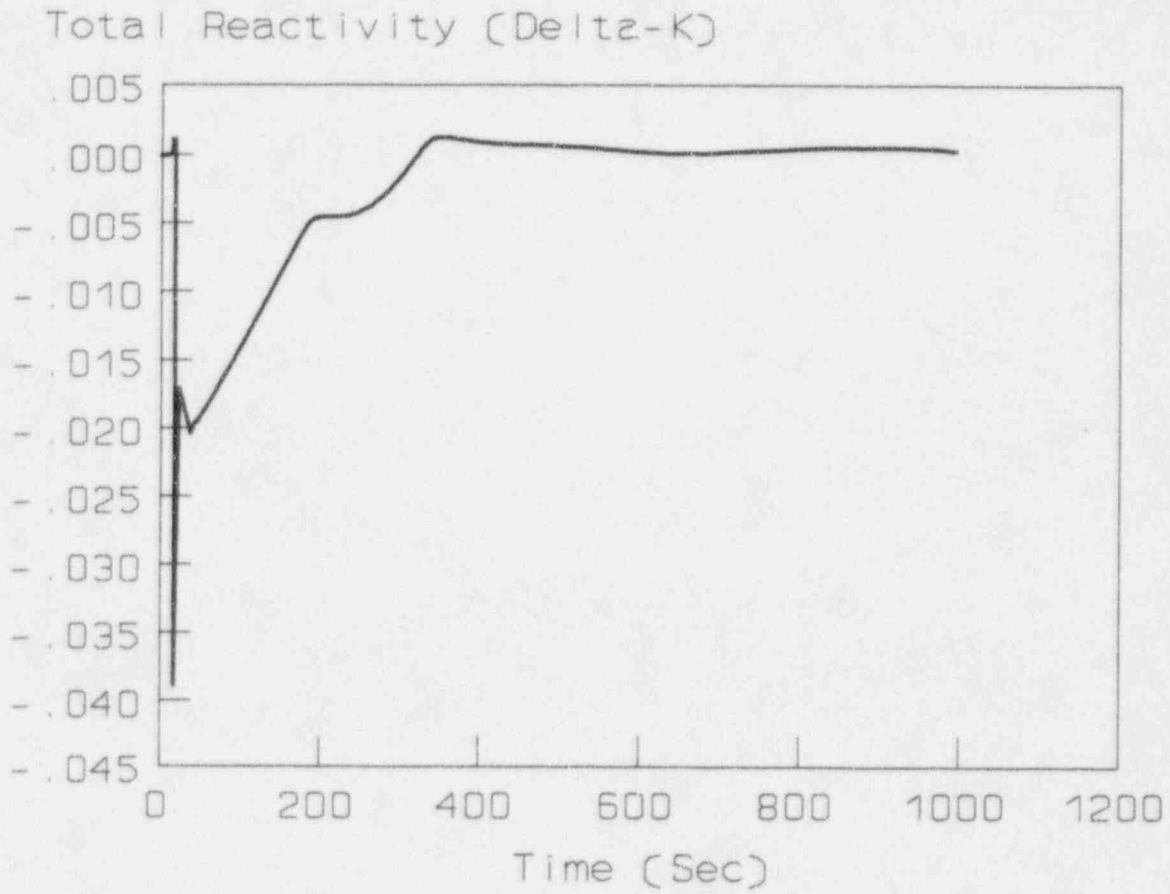
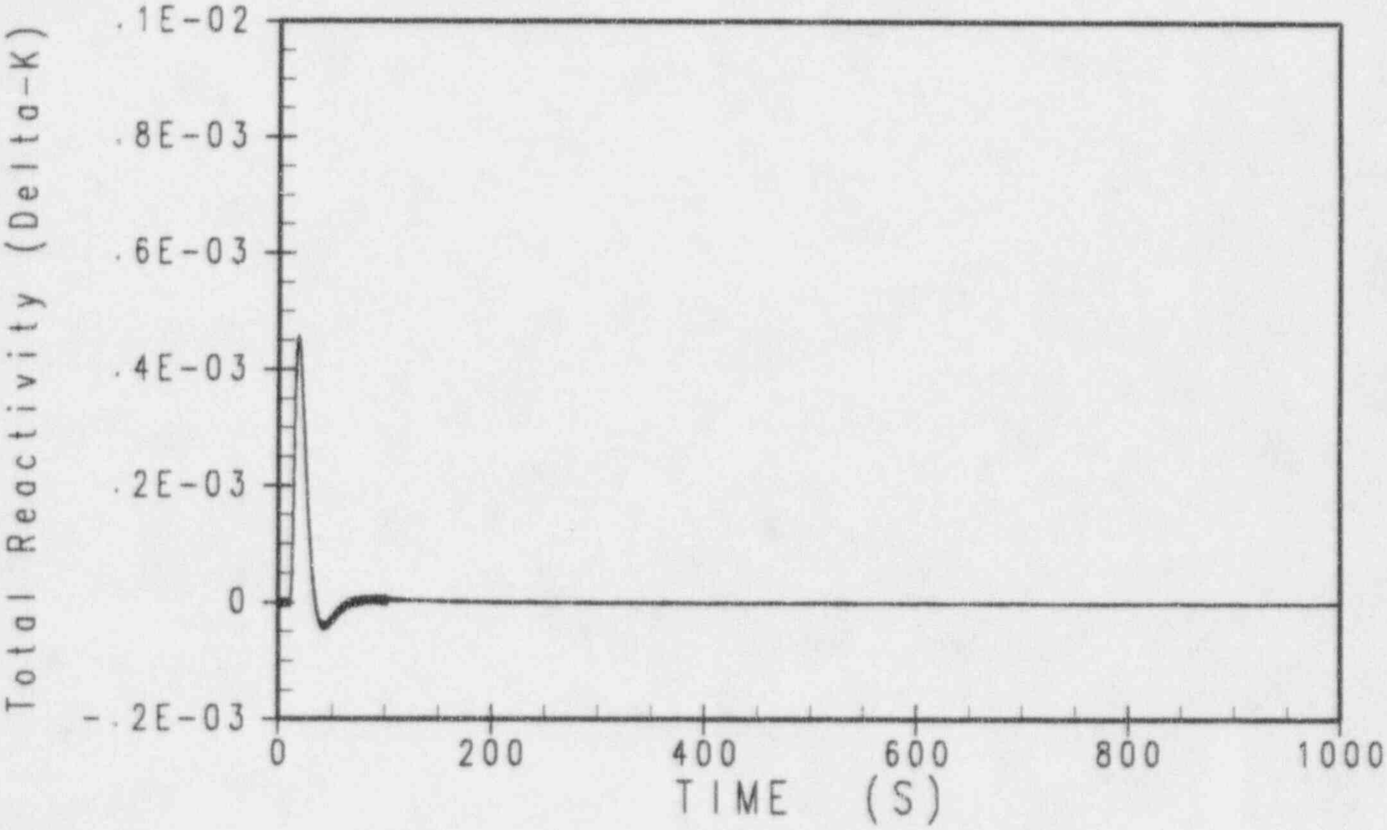


Figure 15.1.6-3

Reactivity Transient Inadvertent Operation of the PRHR

Figure 15.1.6-3



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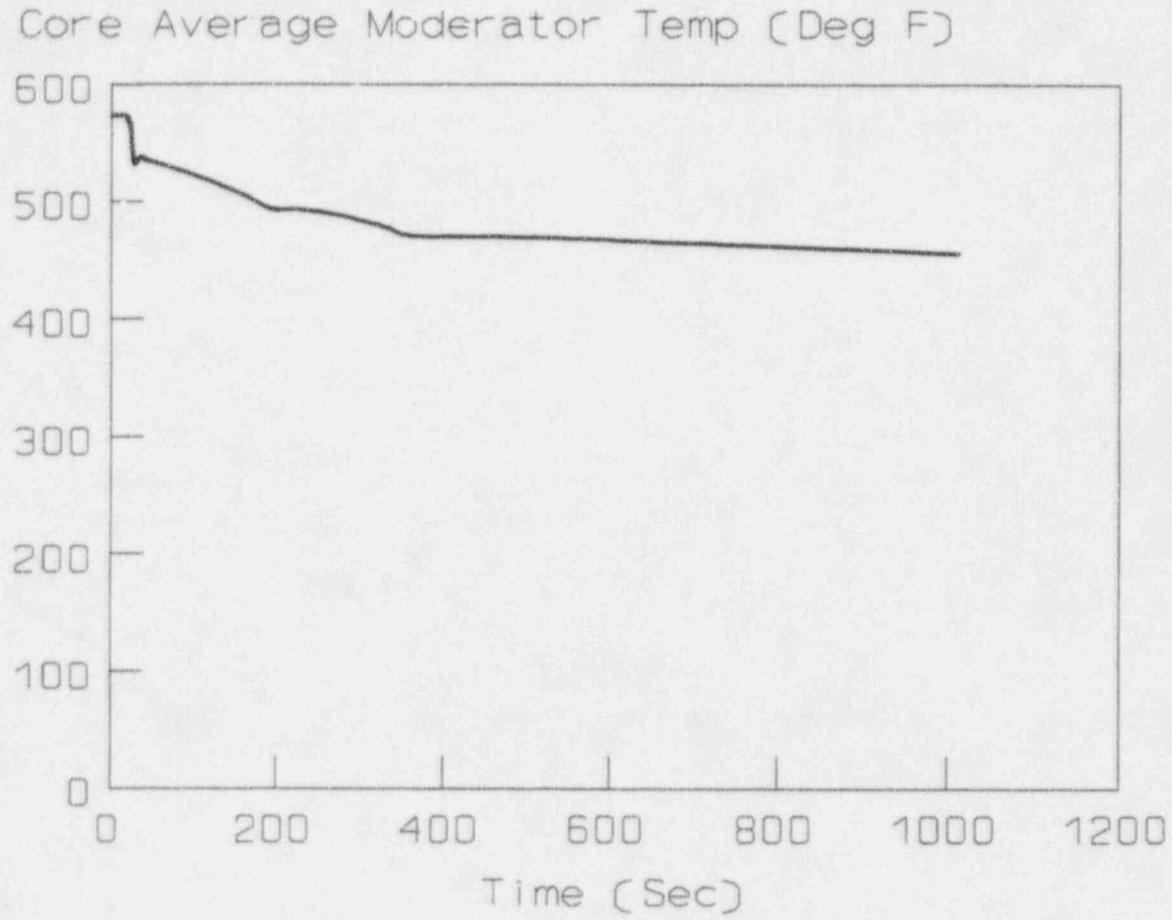

Figure 15.1.6-4

Core Average Moderator Temperature Transient
Inadvertent Operation of the PRHR

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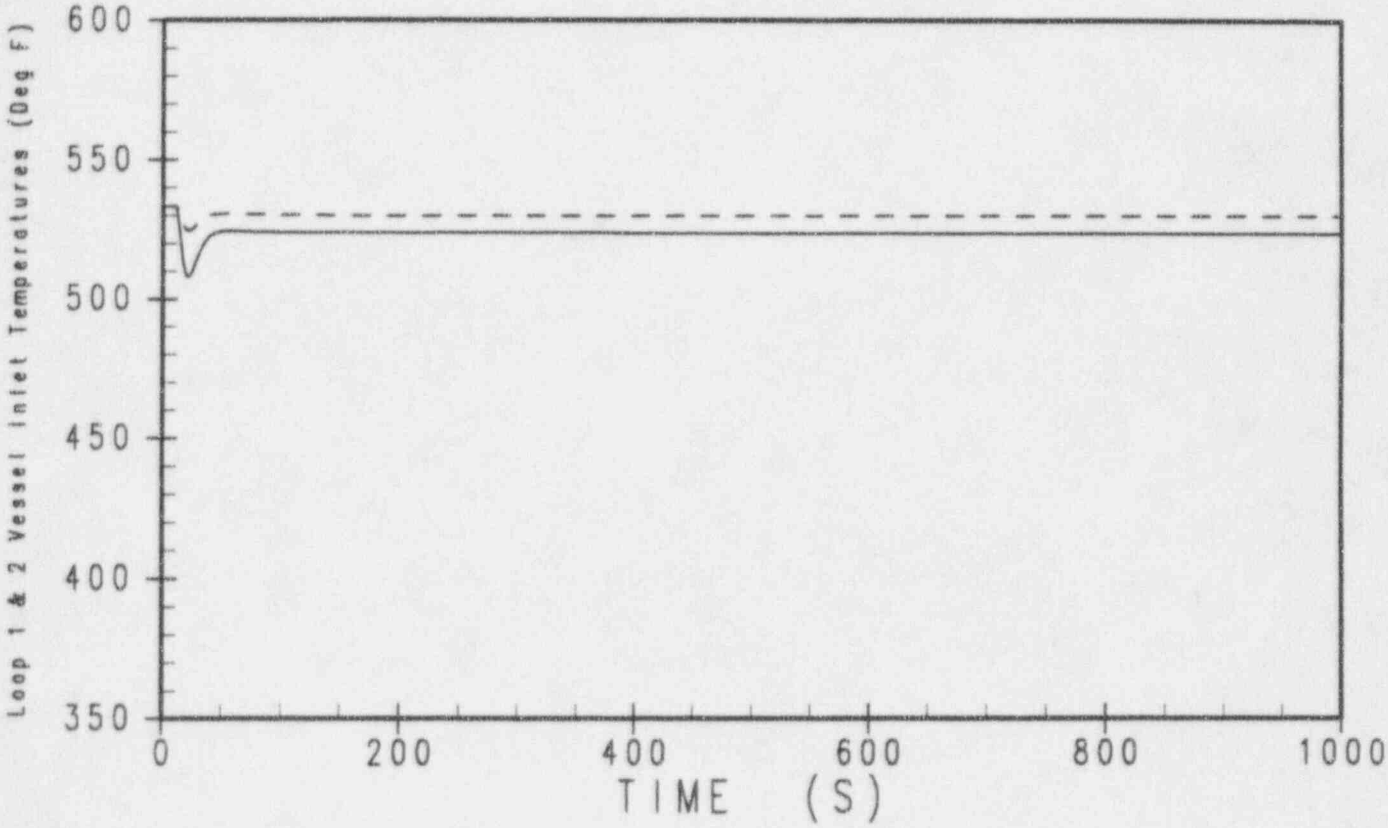
Figure 15.1.6-4



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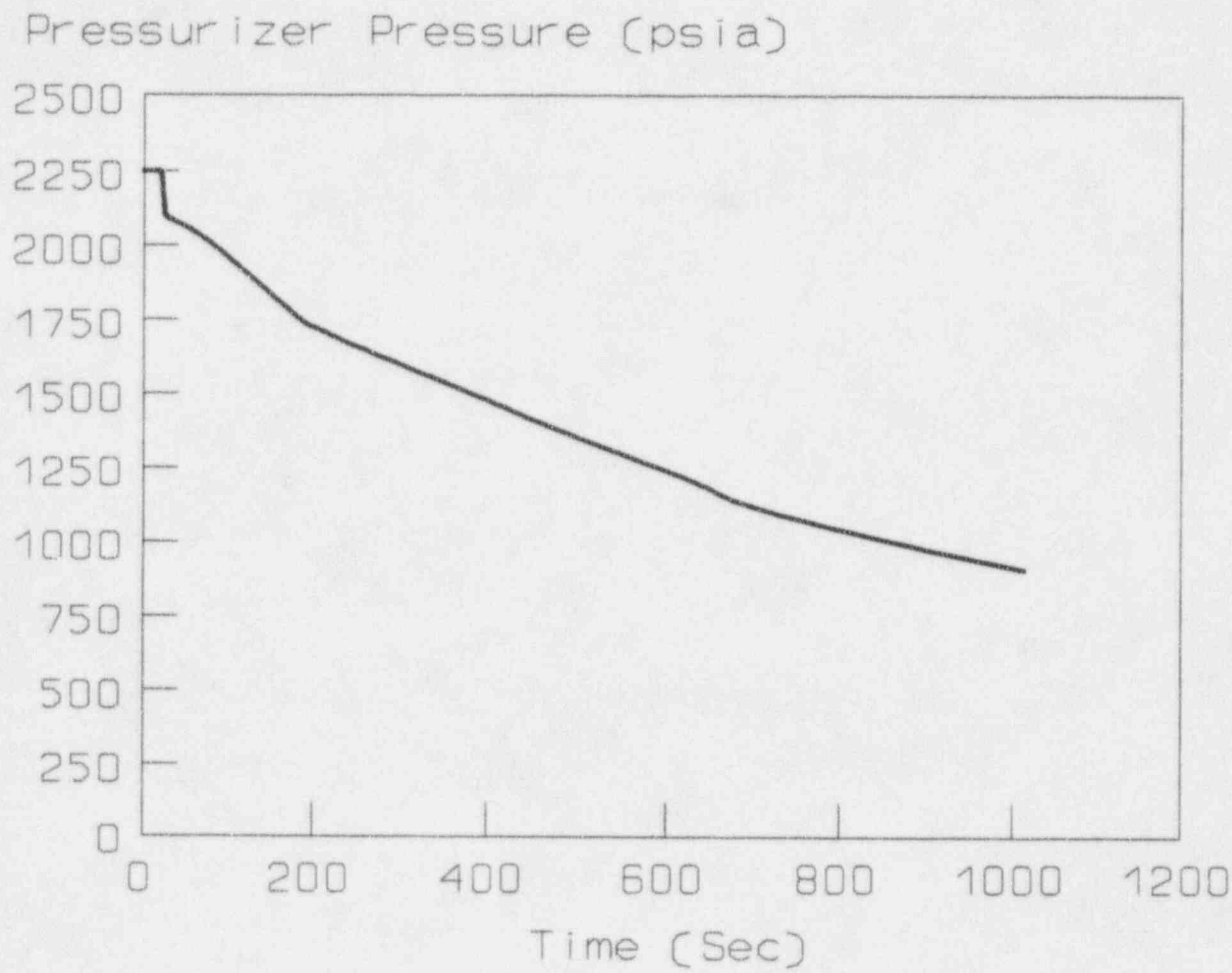

Figure 15.1.6-5

Pressurizer Pressure Transient Inadvertent Operation of the PRHR

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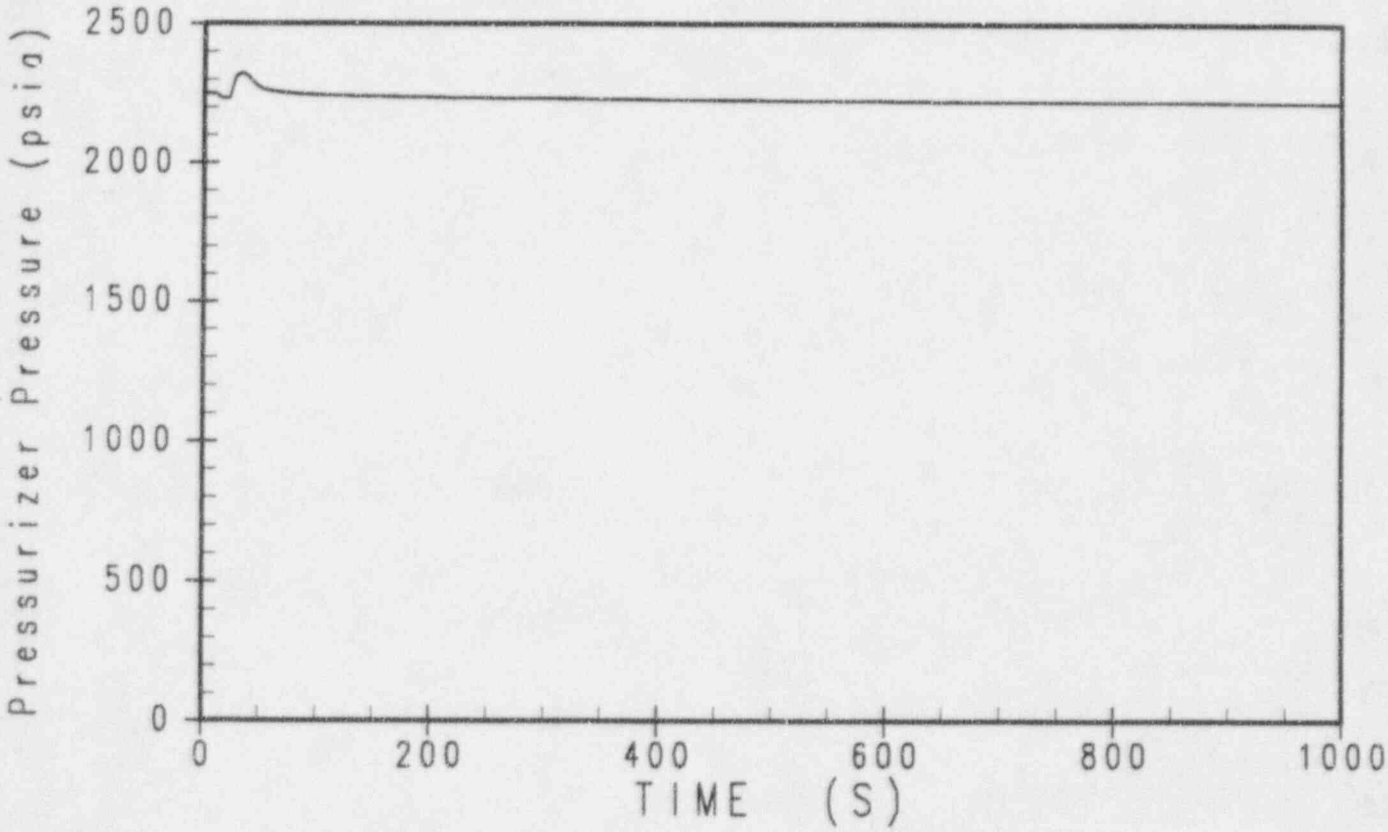
Figure 15.1.6-5



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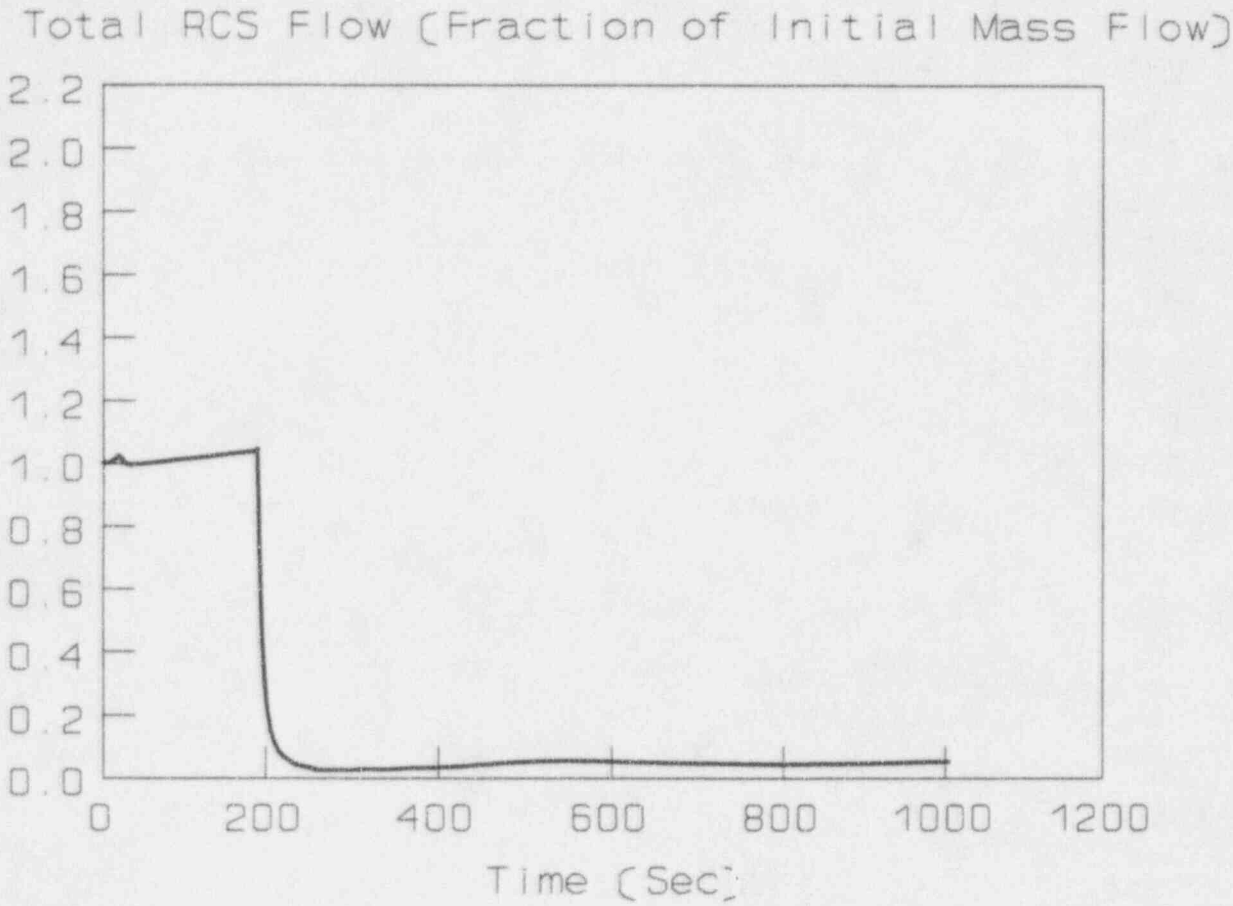

Figure 15.1.6-6

RCS Flow Transient Inadvertent Operation of the PRHR

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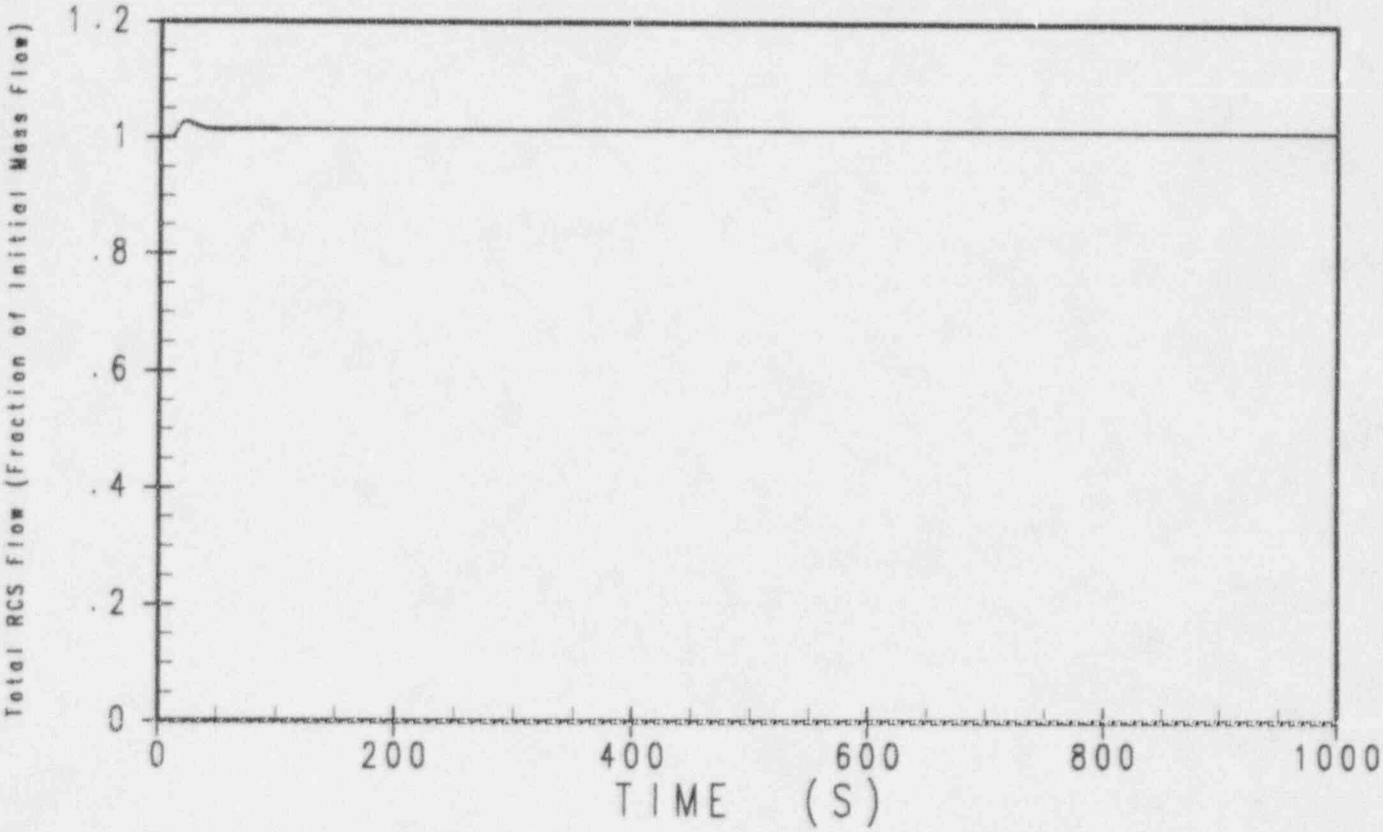
Figure 15.1.6-6



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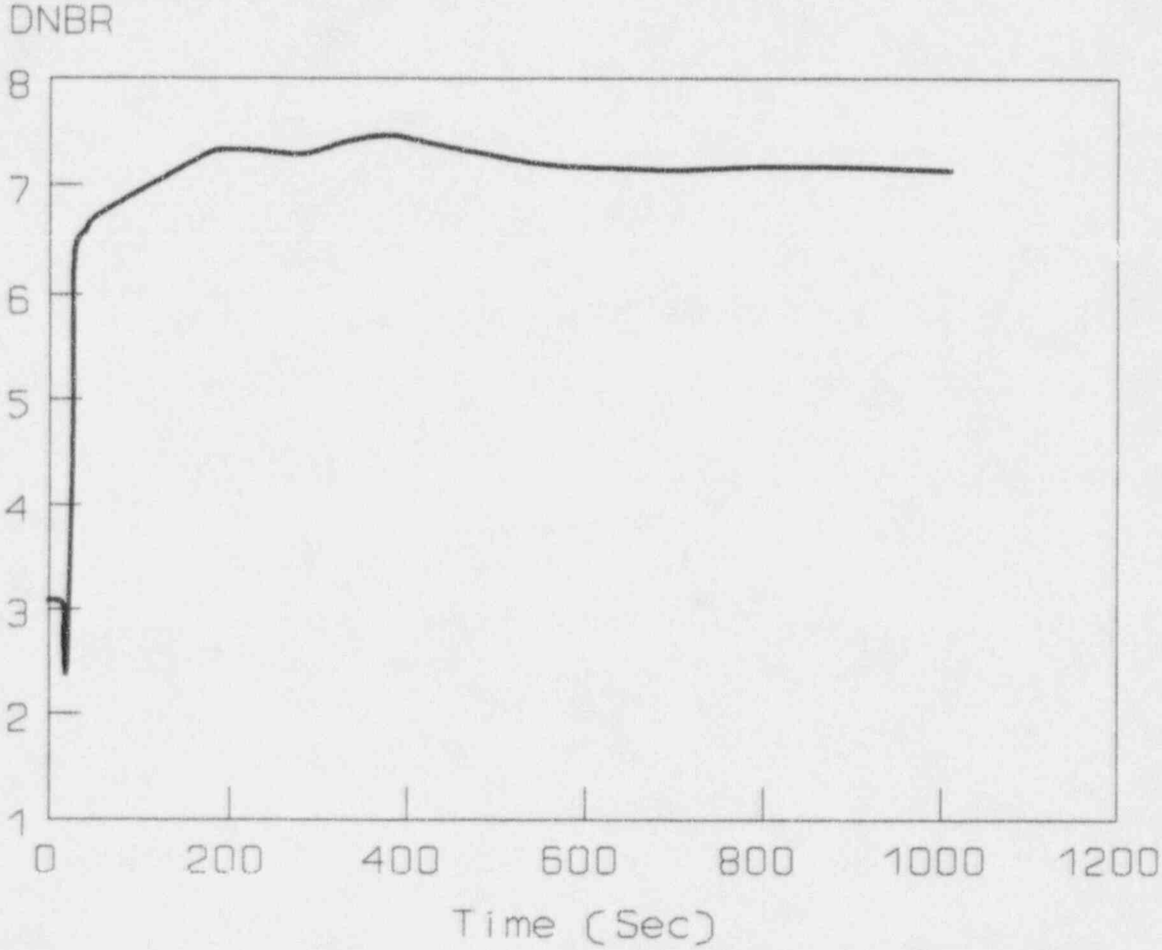

Figure 15.1.6-7

DNBR Transient Inadvertent Operation of the PRHR

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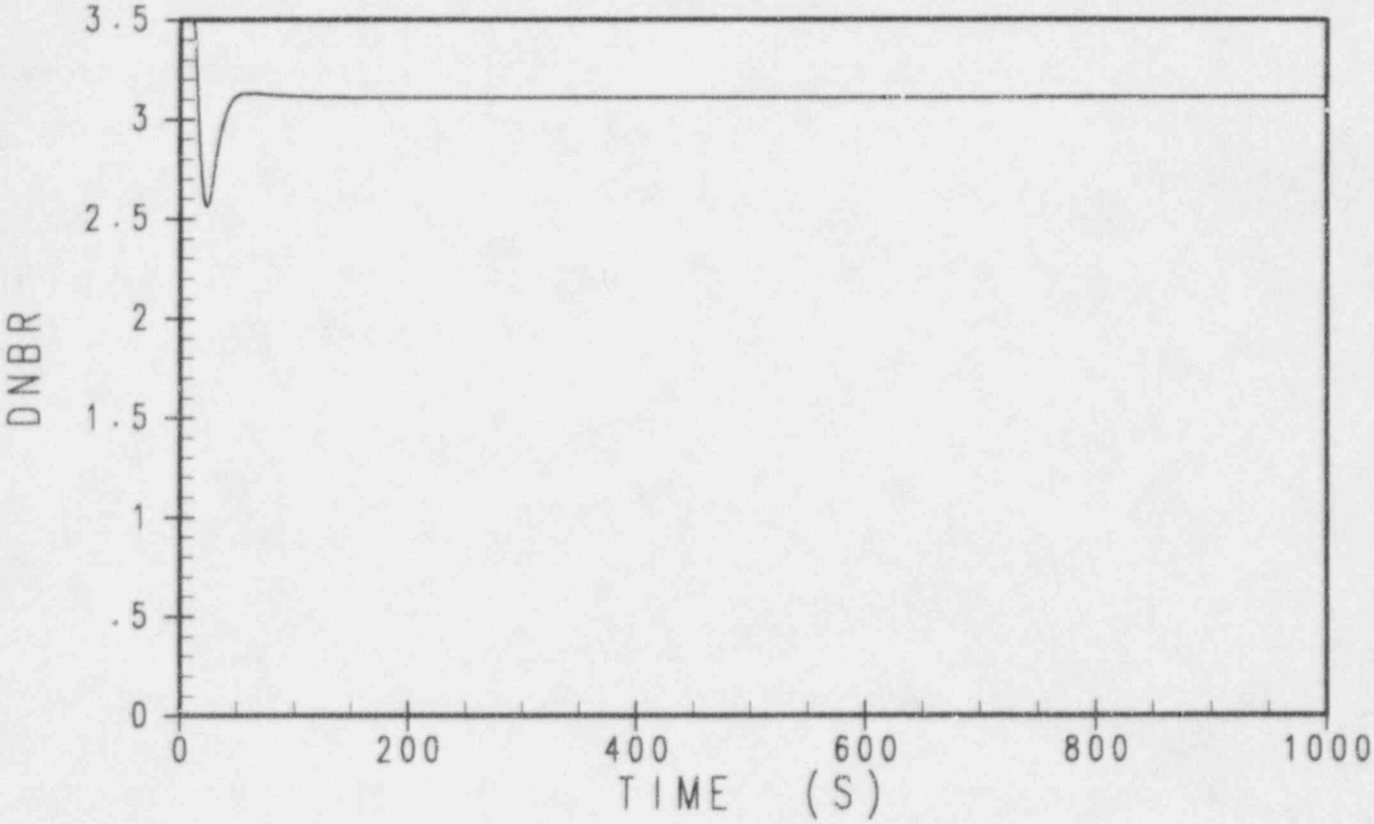
Figure 15.1.6-7



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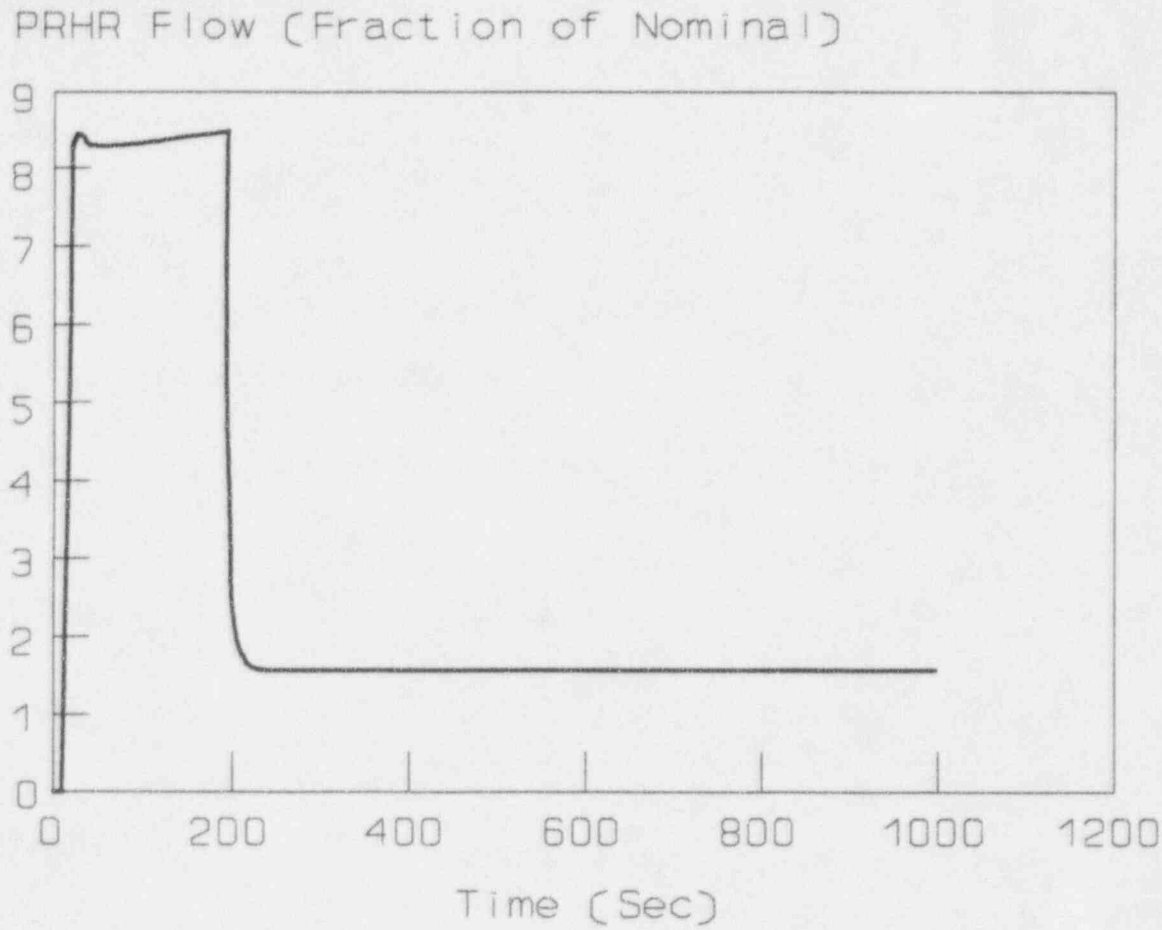

Figure 15.1.6-8

PRHR Flow Transient Inadvertent Operation of the PRHR

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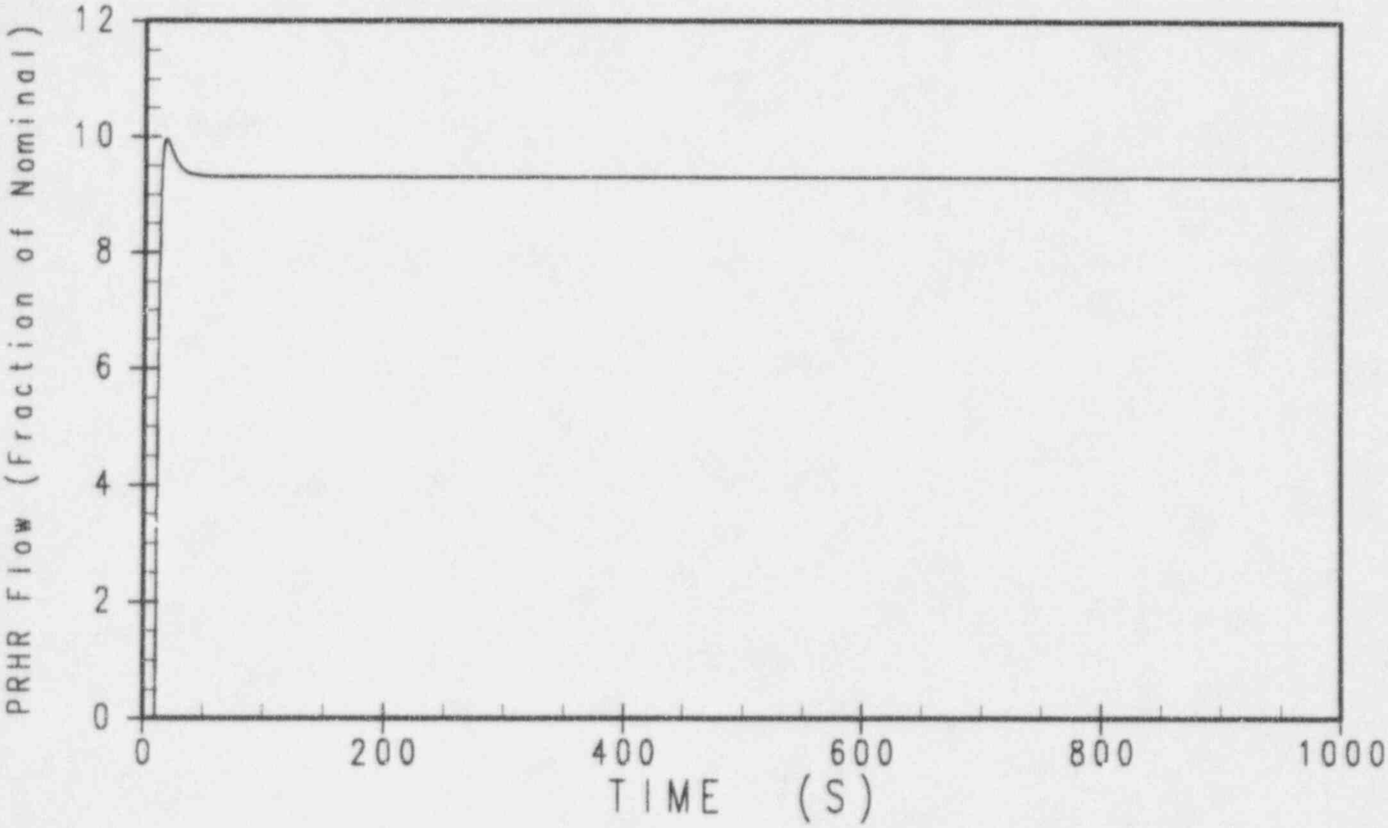
Figure 15.1.6-8



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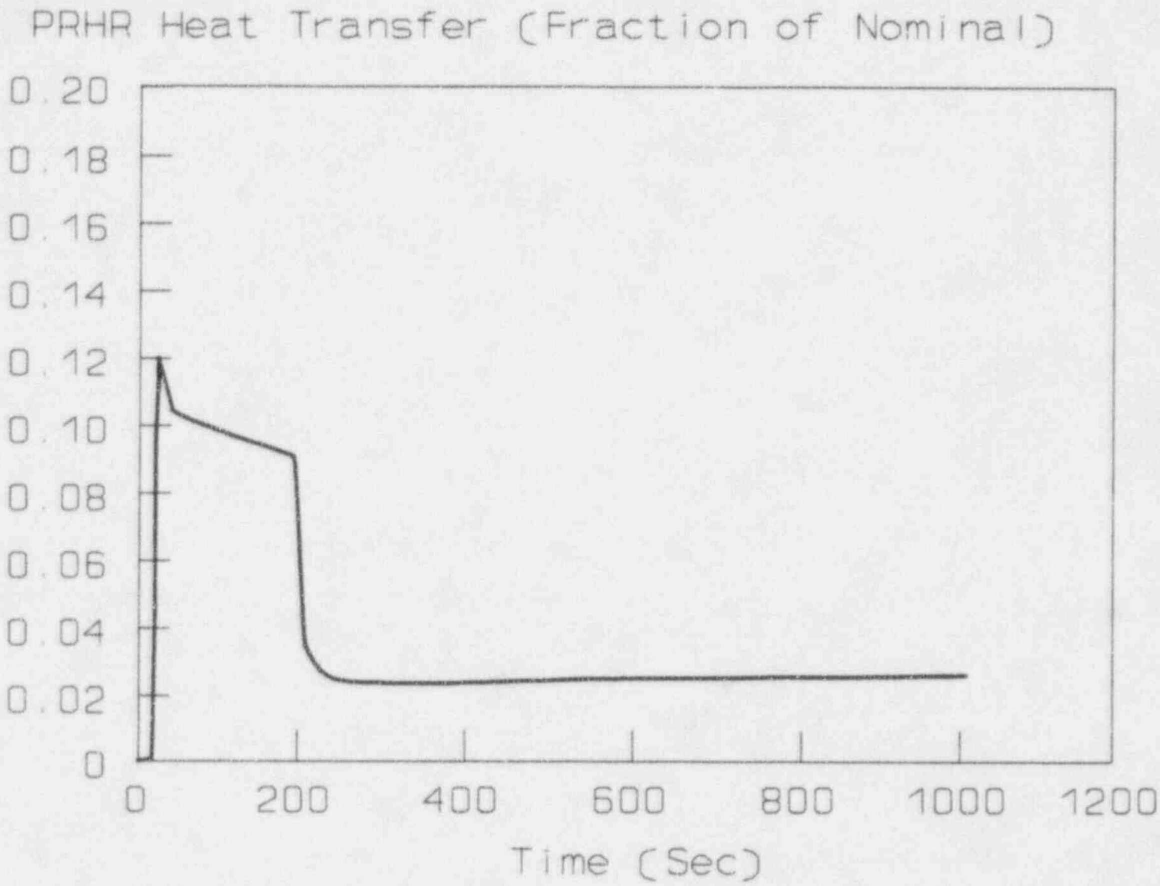

Figure 15.1.6-9

PRHR Heat Transfer Transient Inadvertent Operation of the PRHR

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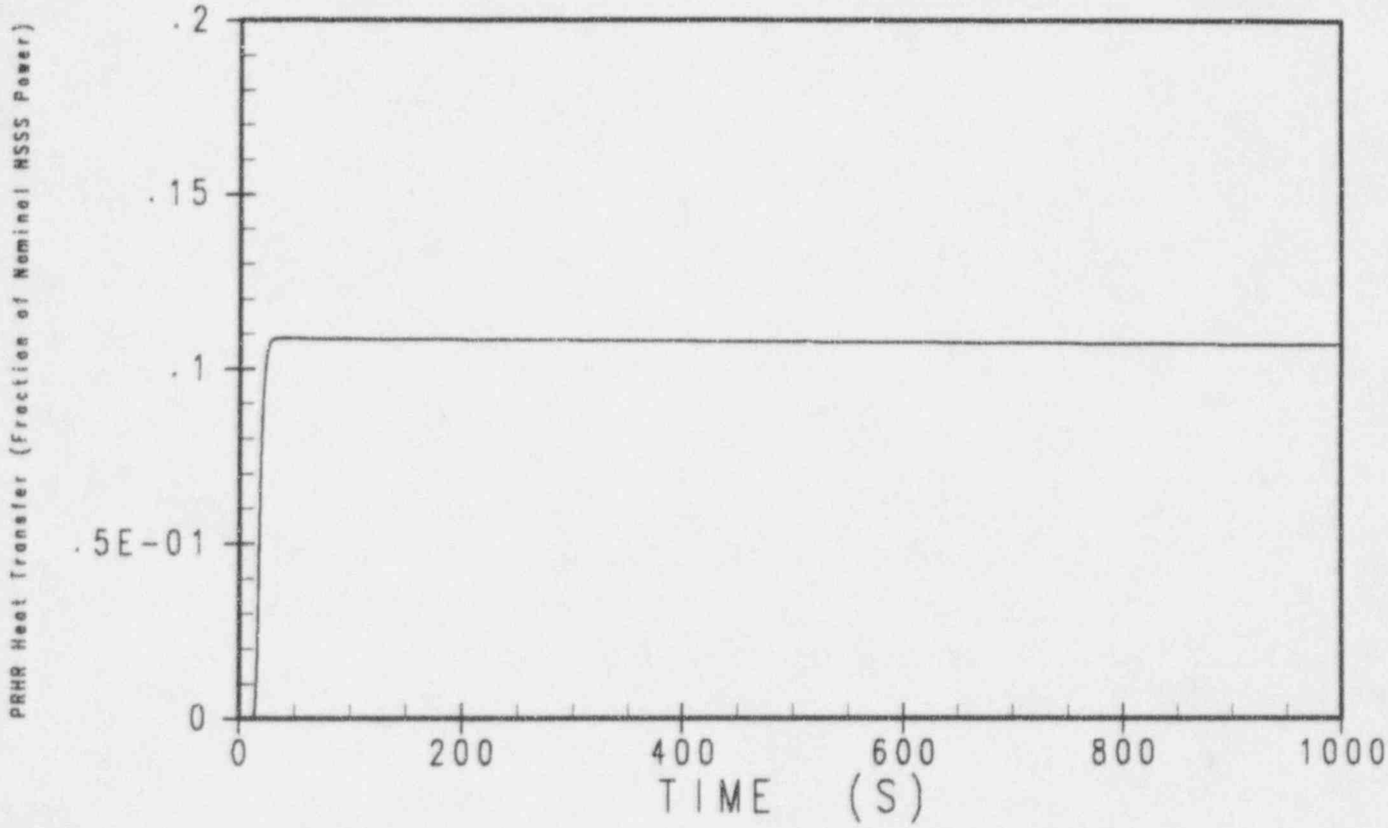
Figure 15.1.6-9



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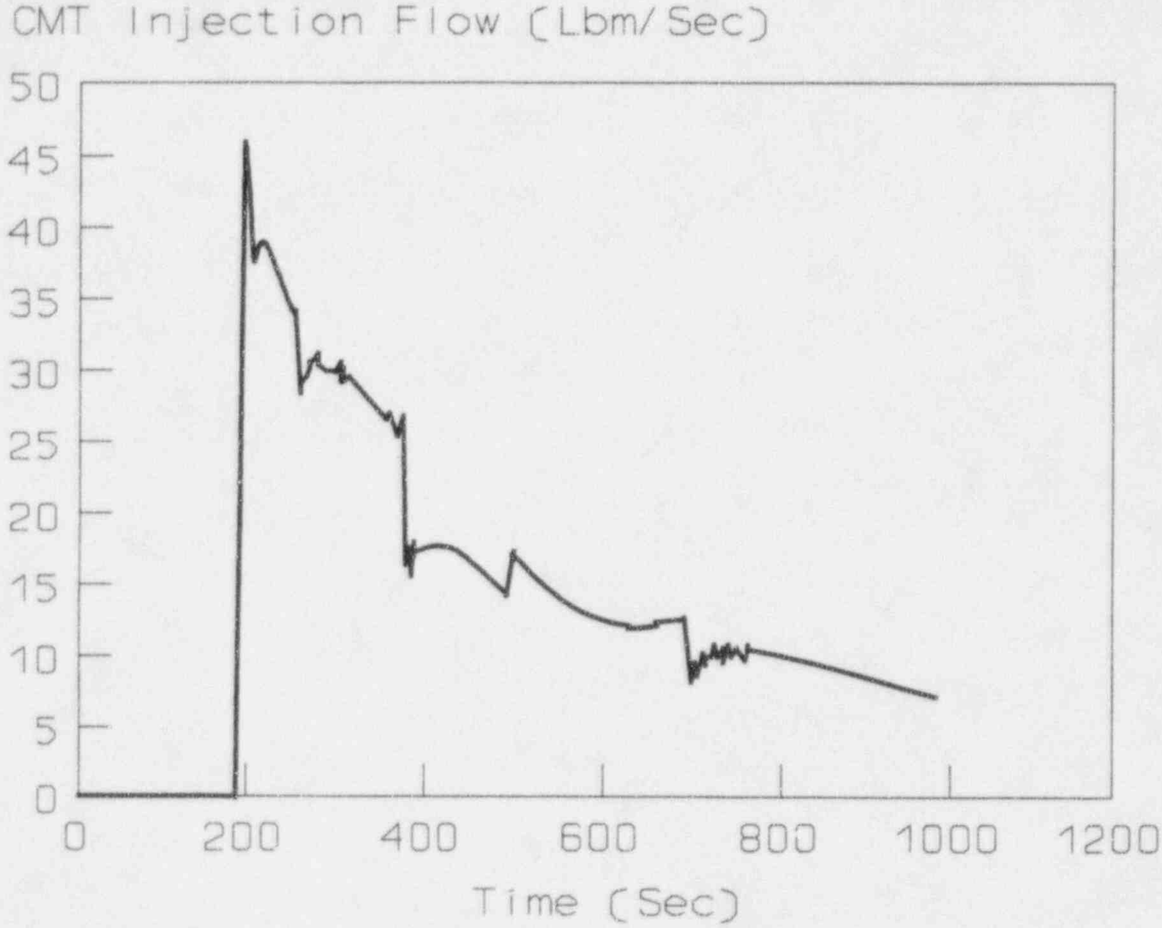

Figure 15.1.6-10

Core Make Up Tank Injection Flow
Inadvertent Operation of the PRHR

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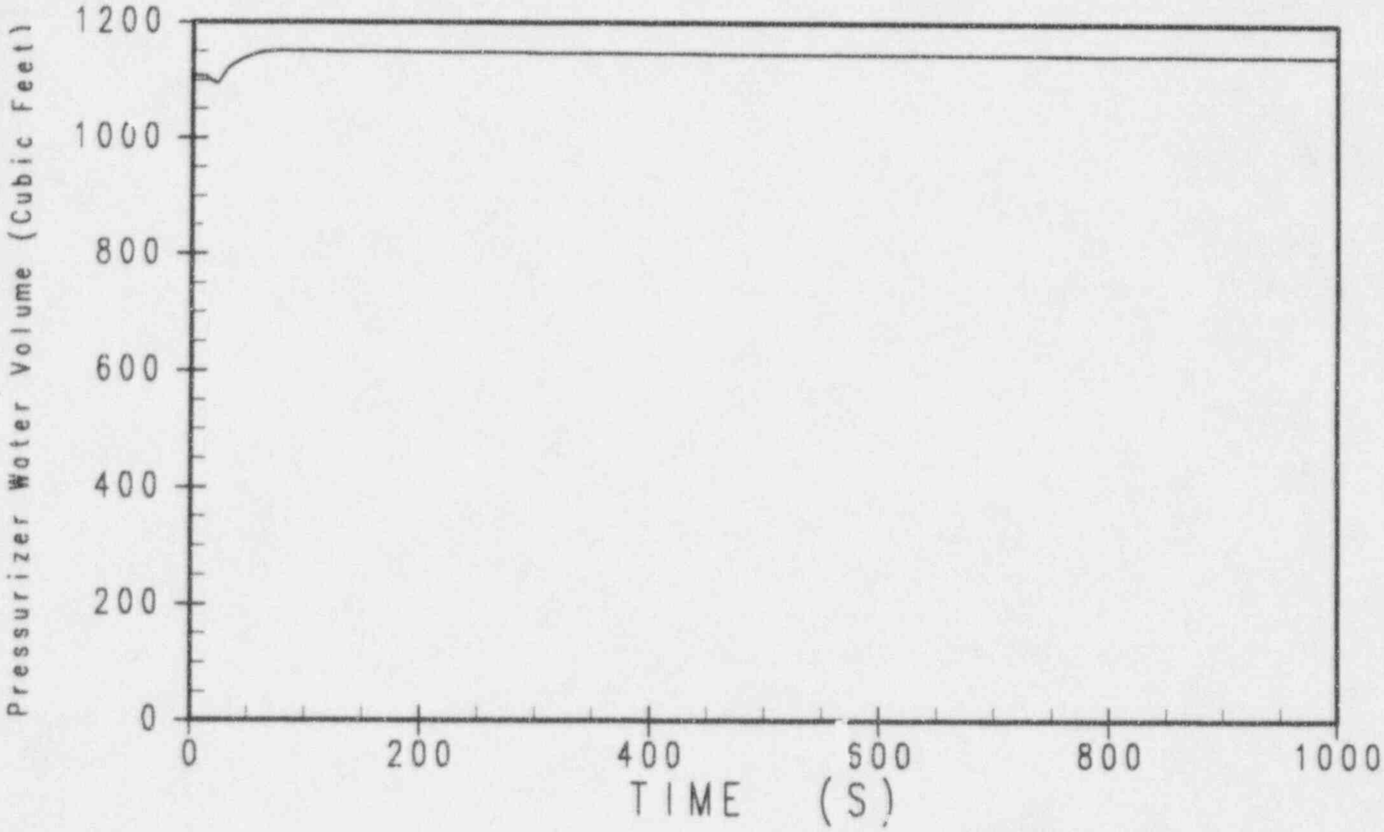
Figure 15.1.6-10



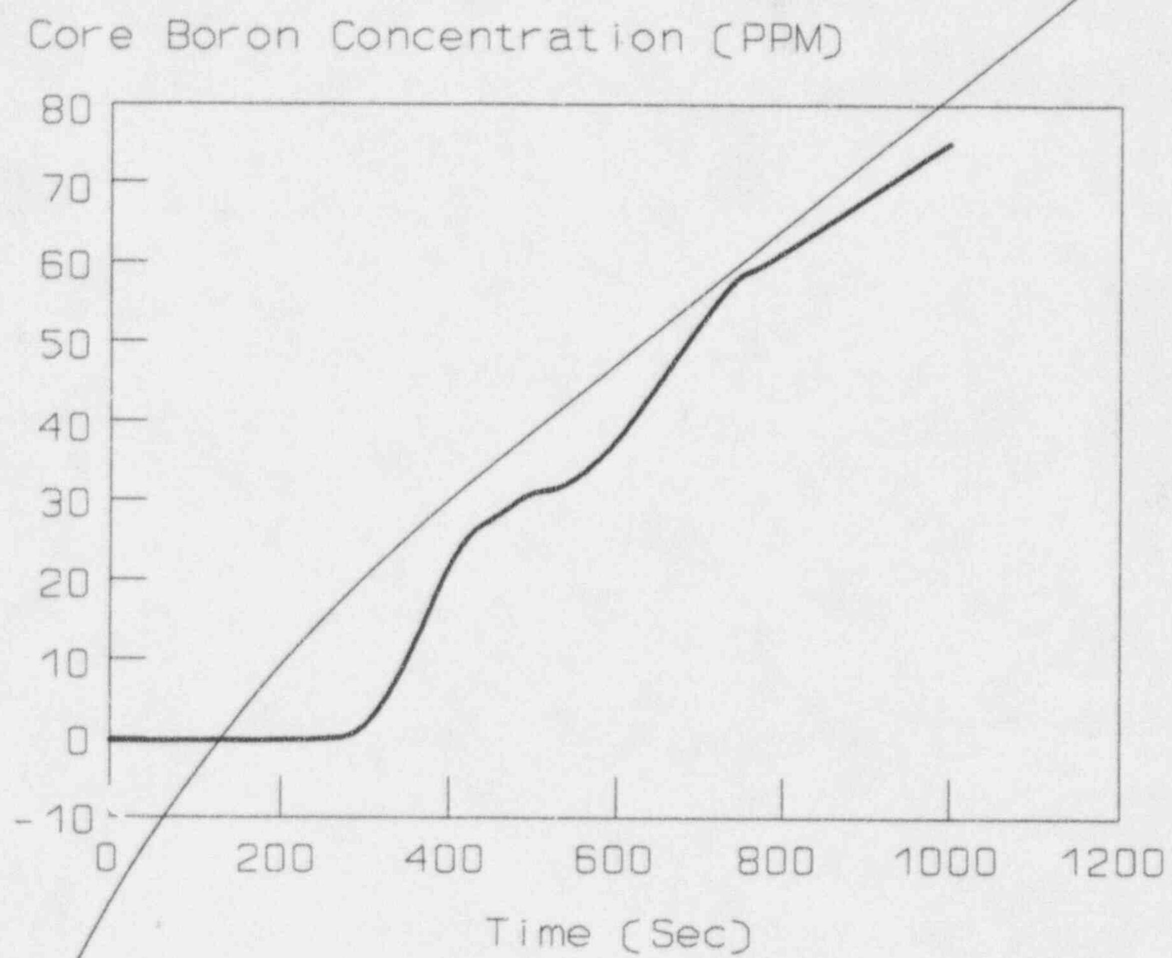


Figure 15.1.6-11

Core Boron Transient
Inadvertent Operation of the PRHR

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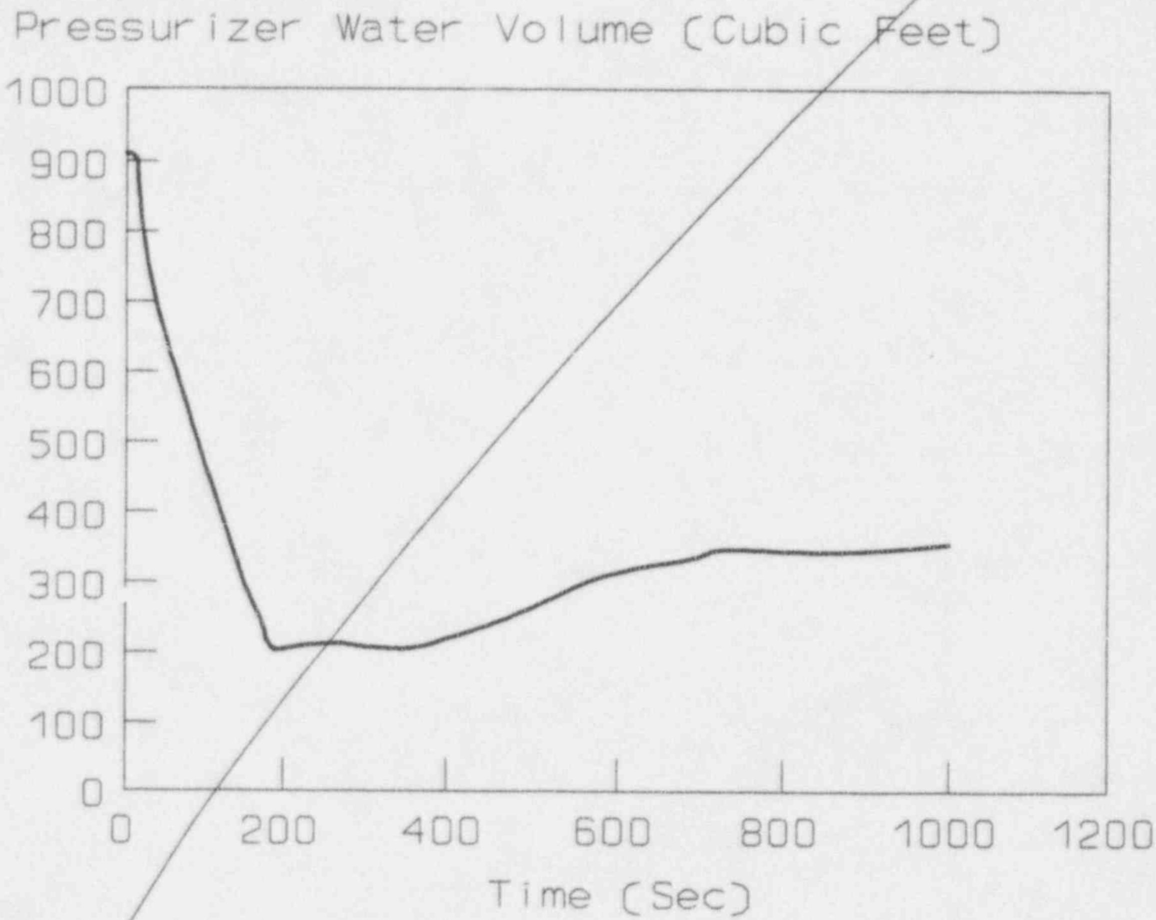

Figure 15.1.6-12

Pressurizer Water Volume Transient
Inadvertent Operation of the PRHR

15.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents are postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system. Detailed analyses are presented in this section for the following events which are identified as more limiting than the others:

- Steam pressure regulator malfunction or failure that results in decreasing steam flow
- Loss of external electrical load
- Turbine trip
- Inadvertent closure of main steam isolation valves
- Loss of condenser vacuum and other events resulting in turbine trip
- Loss of ac power to the station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break

The above items are considered to be Condition II events, with the exception of a feedwater system pipe break, which is considered to be a Condition IV event.

The radiological consequences of the accidents in this section are bounded by the radiological consequences of a main steam line break (See Subsection 15.1.5).

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

There are no steam pressure regulators in the AP600 whose failure or malfunction cause a steamflow transient.

15.2.2 Loss of External Electrical Load

15.2.2.1 Identification of Causes and Accident Description

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

For a loss of electrical load without subsequent turbine trip, no direct reactor trip signal is generated, and the plant is expected to trip from the reactor protection system if a safety limit

is approached. A continued steam load of approximately five percent exists 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 is provided by high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trip. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external electrical load, the maximum turbine overspeed ~~would be approximately 111 percent, resulting in an overfrequency of less than 67 Hz (111% x 60 Hz). This resulting overfrequency~~ is not expected to damage the voltage and frequency sensors. Any degradation in their performance is ascertained at that time. Any increased frequency to the reactor coolant pump motors results in a slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, the overfrequency condition is not seen by the reactor protection system equipment, or other safety-related loads. Safety-related loads and the reactor protection system equipment are supplied from the 120-volt ac instrument power supply system, which, in turn, is supplied from the inverters. The inverters are supplied from a dc bus energized from batteries or by a regulated ac voltage.

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

The steam generator safety valve capacity is sized to remove the steam flow at the guaranteed nuclear steam supply system thermal rating from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink with the plant initially operating at the maximum turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

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

A loss of external load is classified as a Condition II event, fault of moderate frequency.

A loss-of-external-load event results in a plant transient that is bounded by the turbine trip event analyzed in Subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss-of-external-load event.

The primary side transient is caused by a decrease in heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow. (Should feedwater flow not be reduced, a larger heat

sink is available and the transient is less severe.) Reduction 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, which occurs in approximately 0.15 seconds. The transient in primary pressure, temperature, and water volume is less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

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

15.2.2.2 Analysis of Effects and Consequences

Refer to 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 bound those expected for the loss of external load, as discussed in Subsection 15.2.2.1.

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

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

15.2.2.3 Conclusions

Based on results obtained for the turbine trip event and considerations described in Subsection 15.2.2.1, the applicable SRP Section 15.2.1 evaluation criteria for a loss-of-external-load event are met. (See Sub-section 15.2.3)

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

The turbine stop valves close rapidly (about ~~0.15~~ 0.3 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

- Generator trip
- Low condenser vacuum
- Loss of lubricating oil
- Turbine thrust bearing failure
- Turbine overspeed

- Manual trip
- Reactor trip

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

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

A turbine trip is classified as a Condition II event, fault of moderate frequency.

A turbine trip is more limiting than loss of external load, loss of condenser vacuum, and other events which result in a turbine trip. As such, this event is analyzed in detail. Results and discussion of the analysis are presented in Subsection 15.2.3.2.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent of full power, without rapid power reduction, primarily to show the adequacy of the pressure-relieving devices, and also to demonstrate core protection margins. The turbine is assumed to trip without actuating the rapid power reduction system. This assumption delays reactor trip until conditions in the reactor coolant system result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater or the passive residual heat removal system (except for long-term recovery) to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the ~~detailed digital~~ computer program LOFTRAN (Reference 2). The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The program computes pertinent plant variables, including temperatures, pressures and power level. The LOFTRAN code is modified to incorporate the specific passive safeguards system features for the AP600. A description of these modifications are presented in Appendix 15B.

The major assumptions used in the analysis are summarized below:

Initial Operating Conditions

The accident is analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full power operation. Uncertainties in initial conditions are included in the departure from nucleate boiling ratio (DNBR) limit as described in WCAP-11397 (Reference 3).

Reactivity Coefficients

Two cases are analyzed:

- **Minimum Reactivity Feedback** - A least negative moderator temperature coefficient and a least negative Doppler-only power coefficient are assumed. (See Figure 15.0.4-1)
- **Maximum Reactivity Feedback** - A conservatively large negative moderator temperature coefficient and a most negative Doppler-only power coefficient are assumed. (See Figure 15.0.4-1)

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 is in automatic control, the control rod banks move prior to trip and reduce the severity of the transient.

Steam Release

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

Pressurizer Spray

Two cases for both the minimum and maximum reactivity feedback cases are analyzed:

- Full credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are also available with maximum capacity.
- No credit is taken for the effect of pressurizer spray in reducing or limiting the coolant pressure. Safety valves are operable with maximum capacity

heat exchanger

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 startup feedwater flow or the passive residual heat removal, since a stabilized plant condition is reached before startup feedwater initiation or passive residual heat removal is normally assumed to occur. The startup feedwater flow or passive residual heat removal removes core decay heat following plant stabilization.

Reactor Trip

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

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

Plant systems and equipment which may be required to function to mitigate the effects of a turbine trip event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

The ~~reactor~~ protection and safety monitoring 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 prevents operation of any system required to function. Normal reactor coolant system and engineered safety systems are not required to function. However, cases are analyzed both with and without the operation of pressurizer spray to determine the worst case for presentation.

15.2.3.2.2 Results

The transient responses for a turbine trip from 100 percent of full-power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 15.2.3-1 through 15.2.3-24). The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2.3-1 through 15.2.3-6 show the transient responses for the total loss of steam load with minimum reactivity feedback, assuming full credit for the pressurizer spray and pressurizer safety valves. No credit is taken for the steam bypass. The reactor is tripped by the high pressurizer pressure trip channel. The minimum DNBR remains well above the safety analysis limit values. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint.

Figures 15.2.3-7 through 15.2.3-12 show the responses for the total loss of steam load with maximum reactivity feedback. All other plant parameters are the same as the above. The ~~departure from nucleate boiling ratio increases throughout the transient and does not drop below its initial value.~~ minimum departure from nucleate boiling ratio remains well above the safety analysis limit values. The pressurizer safety valves and steam generator safety valves prevent overpressurization in primary and secondary systems. The rise in the reactor coolant

system average temperature causes a large reduction in neutron flux due to reactivity feedback effects, resulting in a decrease in pressurizer pressure.

The turbine trip accident is also studied assuming the plant to be initially operating at 100 percent of full power with no credit taken for the pressurizer spray or the turbine bypass system. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2.3-13 through 15.2.3-18 show the transients with minimum reactivity feedback. The neutron flux remains essentially constant at 100 percent of full power until the reactor is tripped. The * Insert A ~~departure from nucleate boiling ratio increases throughout the transient.~~ In this case, the pressurizer safety valves are actuated and maintain reactor coolant system pressure below 110 percent of the design value.

Figures 15.2.3-19 through 15.2.3-24 show the transients with maximum reactivity feedback, with the other assumptions being the same as in the preceding case. Again, the * Insert A ~~departure from nucleate boiling ratio increases throughout the transient~~ and the pressurizer safety valves are actuated to limit primary pressure.

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

15.2.3.3 Conclusions

Results of the analyses, including those in Reference ~~2~~ 1, show that the plant design is such that a turbine trip presents no challenge to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The analyses show that the departure from nucleate boiling ratio does not decrease below the safety analysis limit at any time during the transient. Thus, the departure from nucleate boiling design basis, as described in Section 4.4, is met.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

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

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip-initiating events are described in Subsection 15.2.3. A loss of condenser vacuum prevents the use of steam dump to the condenser; however, since steam dump is assumed to be unavailable in the turbine trip analysis, no additional adverse effects result if the turbine trip is caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Subsection 15.2.3 apply to the loss of the condenser vacuum. In addition, analyses for the other possible causes of a turbine trip, listed in Subsection 15.2.3.1, are covered by

Insert A minimum ~~DNBR~~ departure from nucleate boiling ratio remains well above the safety analysis limit values

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 ac Power to the Plant Auxiliaries

15.2.6.1 Identification of Causes and Accident Description

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The on-site standby ac power system remains available but is not credited to mitigate the accident.

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 system is accompanied by a reactor coolant flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip:

- 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
- Due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

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

- Plant vital instruments are supplied from the Class 1E and UPS.
- As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam flow rate through the power-operated 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.
- As the no-load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if the startup feedwater is available to supply water to the steam generators.
- The onsite standby power system, if available, supplies ac power to the selected plant permanent nonsafety loads.
- If startup feedwater is not available, the PRHR is actuated. The PRHR ~~system~~ heat exchanger transfers the core decay heat and sensible heat to the IRWST and provides an uninterrupted core heat removal capability following any loss of normal and startup feedwater.

The startup feedwater system, if available, is started automatically when low level occurs in either steam generator.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system. If that system is not available then emergency core decay heat removal is provided by the passive residual heat removal heat exchanger. The passive residual heat removal heat exchangers ~~consist of two~~ C-tube heat exchangers (HXs), connected through inlet and outlet headers to the reactor coolant system. The inlet to the heat exchangers is from the reactor coolant system hot leg and the return is to the SG outlet plenum. The heat exchangers are located above the core to provide natural circulation flow when the RCPs are not operating. The in-containment refueling water storage tank provides the heat sink for the HXs. The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system (PCS), keeps the reactor coolant subcooled indefinitely. After the in-containment refueling water storage tank water reaches saturation (in about two hours), steam starts to vent to the containment atmosphere and the condensation which collects on the containment steel shell (cooled by passive containment cooling system) returns to the in-containment refueling water storage tank, maintaining fluid level for the passive residual heat removal heat exchanger heat sink. Without any recovery of condensate, the in-containment refueling water storage tank inventory is sufficient to provide the passive residual heat removal heat exchanger operation for 72 hours.

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 and PRHR loops.

A loss of ac power to the plant auxiliaries is a Condition II event, a fault of moderate frequency. This event is more limiting with respect to long-term heat removal than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power, which is discussed in Subsection 15.2.3. A loss of offsite power to the plant auxiliaries, can also result in a loss of normal feedwater if the condensate pumps lose their power supply.

Following the reactor coolant pump coastdown caused by the loss of ac power, the natural circulation capability of the reactor coolant system removes residual and decay heat from the core, aided by the passive residual heat removal system. An analysis is presented here to show that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Subsection 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences

15.2.6.2.1 Method of Analysis

A detailed analysis using a modified version of the LOFTRAN code (Reference 2) described in Appendix 15B is performed to simulate the system transient following a plant loss of

offsite power. The simulation describes the plant neutron kinetics and reactor coolant system, including the natural circulation, pressurizer, and steam generator system responses. The digital program computes pertinent variables, including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The loss of ac power to the station auxiliaries is evaluated to demonstrate the adequacy of the reactor protection, ~~engineered safeguards systems~~ (the PRHR heat exchanger ~~system~~) and RCS natural circulation capability in removing long term decay heat and preventing excessive heatup of the RCS with possible RCS overpressurization or loss of RCS water.

The assumptions used in this analysis minimize the energy removal capability of the system and maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion.

The assumptions used in the analysis are similar to the loss of normal feedwater flow accident (see Subsection 15.2.7) except that power is assumed to be lost to the reactor coolant pumps at the time of the reactor trip.

The assumptions used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design power rating wiih initial reactor coolant temperature ~~4.5~~ 6.5°F above the nominal value and the pressurizer pressure 50 psi above the nominal value.

- Core residual heat generation is based on ANSI 5.1 (Reference 3). ANSI 5.1 is a conservative representation of the decay energy release rates.

- Reactor trip occurs on steam generator low level (narrow range). Offsite power is assumed to be lost at the time of reactor trip. This is more conservative than the case in which offsite power is lost at time zero, because of the lower steam generator water mass at the time of the reactor trip.

- A heat transfer coefficient is assumed in the steam generator associated with reactor coolant system natural circulation flow conditions following the reactor coolant pump coastdown.

- The passive residual heat removal ~~system~~ heat exchanger is actuated by the low steam generator water level (narrow range) coincident with a low startup feedwater flow rate (startup feedwater is assumed unavailable).

- Conservative PRHR heat transfer coefficients (low) associated with the low PRHR flow rate caused by the RCP trip are assumed.

- For the loss of ac power to the station auxiliaries, the only safety function required is core decay heat removal. That is accomplished by the PRHR heat exchanger ~~system~~; the worst single failure is assumed to occur in the PRHR heat exchanger ~~system~~.

heat exchanger

The actuation of the PRHR ~~system~~ requires the opening of one of the two fail open valves arranged in parallel at the PRHR discharge. Since no single failure can be assumed that impairs the opening of both valves, the failure of a single valve is assumed. ~~Moreover, only one out of the two PRHR HXs is assumed to be available.~~

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

- The pressurizer safety valves are assumed to function.

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

Plant systems and equipment which are necessary to mitigate the effects of a loss of ac power to the station auxiliaries are discussed in Subsection 15.0.8 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 ac power, as analyzed here. The PRHR ~~system~~ heat exchanger is required to function with a minimum heat transfer capability. No single active failure prevents operation of any system required to function.

15.2.6.2.2 Results

The transient response of the reactor coolant system following a loss of ac power to the plant auxiliaries is shown in Figures 15.2.6-1 through 15.2.6-11. The calculated sequence of events for this event is listed in Table 15.2-1.

The LOFTRAN code results show that the natural circulation flow and the passive residual heat removal system are sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

Immediately following the reactor trip, the PRHR heat transfer capability and the steam generator heat extraction rate are sufficient to slowly cool down the plant. At about ~~2000~~ 200 seconds following reactor trip, the decrease in the steam generator water inventory results in a decrease in steam generator heat transfer rate and consequently in a slow heatup of the RCS.

At about ~~3000~~ 1600 seconds following reactor trip, the PRHR heat transfer rate overcomes the core decay heat and the plant starts a slow, steady cooldown.

Insert ①

15.2.6.3 Conclusions

Results of the analysis show that for the loss of ac power to plant auxiliaries event all safety criteria are met. Since DNBR remains above the safety analysis limit values, the core is not adversely affected. PRHR heat removal capacity is sufficient to prevent water relief through the pressurizer safety valves.

The analysis demonstrates that sufficient long-term reactor coolant system heat removal capability exists via natural circulation and the passive residual heat removal system following

reactor coolant pump coastdown to prevent fuel or clad damage and so that the reactor coolant system is not overpressurized.

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 ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If an alternative heat sink such as startup feedwater or the PRHR is not supplied to the plant, core residual heat following reactor trip heats the primary system water to the point where water relief from the pressurizer occurs, resulting in a substantial loss of water from the reactor coolant system. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables do not approach a departure from nucleate boiling condition.

A small secondary system break can affect normal feedwater flow control causing low steam generator levels prior to protective actions for the break. This scenario is addressed by the assumptions made for the feedwater system pipe break (see Subsection 15.2.8).

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

- The steam generator water inventory decreases as a consequence of the continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater flow eventually leads to the reactor trip on a low steam generator water level signal. The same signal also actuates startup feedwater system.

- As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. The condenser is assumed to be unavailable for turbine bypass. If the steam flow path through the power-operated 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.

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

- If startup feedwater is not available, the PRHR is actuated on either a low steam generator water level (narrow range) coincident with low startup feedwater flow rate signal insert A. The PRHR heat exchanger ~~system~~ transfers the core decay heat and sensible heat to the IRWST so that core heat removal is uninterrupted following a loss of normal and startup feedwater.

A loss of normal feedwater is classified as a Condition II event, a fault of moderate frequency.

Insert A -> or a low-low steam generator water level (wide range) signal.

Insert 1
A low Tcold "S" signal is eventually reached: the RCPs are tripped and the CMTs start injecting cold borated water in the RCS. PRHR capacity is then lowered and the RCS starts to heat up.

Pressurizer safety valves open to discharge steam to containment and reclose later in the transient when PRHR capacity exceeds the decay heat production rate.

The capacity of the PRHR is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.6-5 & 6, in the long term, the plant starts a slow cooldown driven by the PRHR system. Plant procedures may be followed to further cool down the plant.

The reactor trip on low narrow range water level in either steam generator provides the necessary protection against a loss of normal feedwater.

The startup feedwater system is started automatically, as discussed in Subsection 15.2.6.1. If startup feed is unavailable then the PRHR ~~system~~ heat exchanger is started as discussed in Subsection 15.2.6.

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

15.2.7.2 Analysis of Effects and Consequences

15.2.7.2.1 Method of Analysis

A detailed analysis using a modified version of the LOFTRAN code (Reference 2), described in Appendix 15B, is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant neutron kinetics, reactor coolant system (including the natural circulation), pressurizer, and steam generators. The 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:

- The plant is initially operating at 102 percent of the design power rating.

- Reactor trip occurs on steam generator low (narrow range) level.

Strikeout

- The PRHR system is actuated by the low steam generator water level signal in coincidence with the low startup feedwater flow rate signal (startup feedwater system is conservatively assumed unavailable).

- Since for the loss of normal feedwater mitigation, the only safety function required is the core decay heat removal, that is carried by the PRHR ~~system~~ heat exchanger, the worst single failure is assumed to occur in the PRHR ~~system~~ heat exchanger. The actuation of the PRHR ~~system~~ heat exchanger requires the opening of one of the two fail open valves arranged in parallel at the PRHR discharge. Since no single failure can be assumed that impairs the opening of both valves, the failure of a single valve is assumed. ~~Only one out of the two PRHR heat exchangers is assumed to be available.~~

- The passive residual heat removal ~~system~~ heat exchanger is actuated by the Low-low steam generator water level wide range signal.

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

- The initial reactor coolant average temperature is ~~4.5~~ 6.5°F higher than the nominal value, and initial pressurizer pressure is 50 psi higher than nominal.

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

As such, the assumptions used in this analysis minimize the energy removal capability of the system and maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion.

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 narrow range level trip. The reactor coolant pumps continue to run until they are automatically tripped when the CMTs are actuated.

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

Plant systems and equipment which are necessary to mitigate the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 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 PRHR heat exchanger ~~system~~ is required to function with a minimum heat transfer rate capability. No single active failure prevents operation of any system to perform its required function. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.2.7.2.2 Results

Figures 15.2.7-1 through 15.2.7-10 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction. Steam flow through the safety valves continues to dissipate the stored and core decay heat.

The capacity of the PRHR, when the reactor coolant pumps are operating, is much larger than the decay heat and in the first part of the transient the RCS is cooled down and the pressure decreases.

~~A safety injection~~ A low Tcold "S" signal is eventually reached: the RCPs are tripped and the CMTs start injecting cold borated water in the RCS. PRHR capacity is then lowered and RCS starts to heat up.

Pressurizer safety valves open to discharge steam to the containment and reclose later in the transient when PRHR capacity exceeds the decay heat production rate.

The capacity of the PRHR is sufficient to avoid water relief through the pressurizer safety valves.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figure 15.2.7-3 & 4, in the long term, the plant starts a slow cooldown driven by the PRHR system. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the safety analysis limit values and RCS and steam generator pressures remain below 110% of their design values.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

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

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

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- 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.
- Fluid in the steam generator may be discharged through the break and would then not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main ~~or startup~~ feedwater after trip.

The passive residual heat removal ~~system~~ heat exchanger functions to:

- Prevent substantial overpressurization of the reactor coolant system (less than 110 percent of design pressures).

- Maintain sufficient liquid in the reactor coolant system so that the core remains in place and geometrically intact with no loss of core cooling capability.

A major feedwater line rupture is classified as a Condition IV event.

The severity of the feedwater line rupture transient depends on a number of system parameters, including break size, initial reactor power, and the functioning of various control and safety related systems. Sensitivity studies presented in Reference 4 illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. The main feedwater control system is assumed to malfunction due to an adverse environment. The water levels in both steam generators are assumed to decrease equally until the Low-low steam generator level reactor trip setpoint is reached. After reactor trip, a double-ended rupture of the largest feedwater line is assumed. These assumptions conservatively bound the most limiting feedwater line rupture that can occur. Analysis is performed at full power assuming the loss of offsite power at the time of the reactor trip. This is more conservative than the case where power is lost at the initiation of the event. The case with offsite power available is not presented since, due to the fast CMT actuation (on a "S" signal generated by the low steam line pressure), the RCPs are tripped by the protection system a few seconds after the reactor trip. The only difference between the cases with and without offsite power available is the RCPs operating status.

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

- A reactor trip on any of the following four conditions:

  - High pressurizer pressure
  - Overtemperature ΔT
  - Low steam generator water level in either steam generator
  - Safeguards signals from either of the following:
    - Two out of four low steam line pressure in either steam generator.
    - Two out of four high containment pressure (High-1).

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

- The PRHR ~~system~~ heat exchanger (HX) provides a passive type, method for decay heat removal. The ~~system~~ heat exchanger is a ~~consists of two~~ C-tube ~~heat exchangers (HXs)~~ located inside the IRWST ~~tank~~. The HX~~s~~ ~~are~~ is above the RCS ~~in order~~ to provide natural circulation of the reactor coolant. Operation of the PRHR ~~system~~ heat exchanger is initiated by the opening of one of the two parallel power operated valves at the PRHR cold leg.

  Refer to Subsection 6.3.2.2.5 for a description of the PRHR.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

A detailed analysis using a modified version, described in Appendix 15B, of the LOFTRAN code (Reference 2) is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, reactor coolant system (including natural circulation), pressurizer, steam generators, and feedwater system responses 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 used in the analysis are as follows:

- The plant is initially operating at 102 percent of the design plant rating.
- Initial reactor coolant average temperature is ~~4.5~~ 6.5°F above the nominal value, and the initial pressurizer pressure is 50 psi above its nominal value.
- No credit is taken for pressurizer spray.
- Initial pressurizer level is at a conservative maximum value and a conservative initial steam generator water level is assumed in both steam generators.
- No credit is taken for the high pressurizer pressure reactor trip.
- Main feedwater to both steam generators is assumed to stop at the time the break occurs. (all main feedwater spills out through the break.)
- A double-ended break area of 1.12 square feet is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.
- A conservative feedwater line break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time in which 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.
- Reactor trip is assumed to be initiated when the low steam generator narrow range level setpoint is reached on the ruptured steam generator.
- The passive residual heat removal ~~system~~ heat exchanger is actuated by the low steam generator water level (wide range) signal. A 17-second delay is assumed following the low level signal to allow time for the alignment of passive residual heat removal valves.

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

- No credit is taken for charging or letdown.

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

- Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip (Reference 3).

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

  - High pressurizer pressure
  - Overtemperature ΔT
  - High pressurizer level
  - High containment pressure

Receipt of a low steam generator water level narrow range signal in at least one steam generator starts the motor driven startup feedwater pumps, which in turn initiate the startup feedwater flow to the steam generators. The PRHR is initiated if the steam generator water level drops to the low steam generator level (wide range) or if a low startup feedwater flow is concomitant to a low steam generator water level (narrow range) signal. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes all main steam line and feed line isolation valves. This signal also gives a safeguard "S" signal which initiates flow of cold borated water from the CMTs to the RCS.

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

The plant control system is not ~~No reactor control systems are~~ assumed to function to mitigate the consequences of the event. The ~~reactor~~ protection and safety monitoring system is required to function following a feedwater line rupture as analyzed here. No single active failure prevents operation of this system.

The engineered safety features assumed to function are the passive residual heat removal system, core makeup tank and steam line isolation valves.

For the case without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the reactor coolant system is shown (see Subsection 15.2.6) to be sufficient to remove core decay heat following reactor trip for the loss of ac power transient. Pump coastdown characteristics are demonstrated in Subsections 15.3.1 and 15.3.2 for single and multiple reactor coolant pump trips, respectively.

A detailed description and analysis of the core makeup tank is described in Subsection 6.3.2.2.1. The passive residual heat removal ~~system~~ heat exchanger is described in Subsection 6.3.2.2.5.

15.2.8.2.2 Results

Calculated plant parameters following a major feedwater line rupture are shown in Figures 15.2.8-1 through 15.2.8-10. The calculated sequence of events for the case analyzed is listed in Table 15.2-1.

The results presented in Figures 15.2.8-5 and 15.2.8-7 show that pressures in the RCS and main steam system remain below 110 percent of the respective design pressure. Pressurizer pressure decreases after reactor trip on the low steam generator water level (~~58.6~~ 183.0 seconds) due to the loss of heat input.

In the first part of the transient, due to the conservative analysis assumptions, the system response following the feedwater line rupture is similar to the loss of ac power to the station auxiliaries (Subsection 15.2.6).

A few seconds after the trip, the CMTs are actuated (~~69.7~~ 111.2 seconds) on low steam line pressure in the ruptured loop while the PRHR is actuated on a low steam generator water level wide range (~~86.7~~ 107.1 seconds).

The addition of the PRHR, and the CMT flow rates aids in cooling down the primary system and helps to provide sufficient fluid to keep the core covered with water.

In the long term, pressurizer safety valves open again due to the mismatch between decay heat and PRHR heat transfer capability. In the first part of the transient, there is a strong cooling effect due to the CMTs that inject cold water into the RCS and receive hot water from the cold leg. In the long term, this effect is much lower due to the heatup of the CMTs. Also, the injection driving head is lowered.

RCS temperatures are low (below 500°F at 7000 seconds.) and in this condition, the PRHR is not able to remove the entire decay heat. RCS temperatures tend to increase until an equilibrium between decay heat power and heat absorbed by the PRHR is reached. Finally, after about five hours the PRHR heat transfer capability exceeds the decay heat power and the RCS temperatures, pressure and pressurizer water volumes start to steadily decrease. As previously stated, core cooling capability is maintained throughout the transient since RCS inventory is increasing due to CMT injection.

15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the passive residual heat removal system capacity is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to maintain the core cooling capability. Radioactivity doses from the postulated feedwater lines rupture are less than those presented for the postulated main steam line break. The SRP Section 15.2.8 evaluation criteria are therefore met.

15.2.~~9~~10 References

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

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

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

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

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), April 1989.

15.2.9 Combined License information

This section has no requirement for additional information to be provided in support of the combined license application.

Table 15.2-1 (Sheet 1 of 5)

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

Accident	Event	Time(s)
I. Turbine Trip		
A. With pressurizer control (minimum reactivity feedback)	Turbine trip; loss of main feedwater	0.0
	High pressurizer pressure reactor trip point reached	~~6.4~~ 7.1
	Rods begin to drop	~~8.4~~ 9.1
	Minimum DNBR occurs	~~9.5~~ 10.5
	Peak pressurizer pressure occurs	~~10.5~~ 11.0
	Initiation of steam release from steam generator safety valves	~~12.0~~ 13.0
B. With pressurizer control (maximum reactivity feedback)	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR occurs	~~4.0~~ 3.5
	High pressurizer pressure reactor trip setpoint reached	~~6.4~~ 7.3
	Rods begin to drop	~~8.4~~ 9.3
	Peak pressurizer pressure occurs	~~10.0~~ 10.5
	Initiation of steam release from steam generator safety valves	~~12.0~~ 13.0

Table 15.2-1 (Sheet 2 of 5)

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

Accident	Event	Time (s)
C. Without pressurizer control (minimum reactivity feedback)	Turbine trip; loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip point reached	~~6.7~~ 5.8
	Rods begin to drop	~~7.7~~ 7.8
	Minimum DNBR occurs	9.0
	Peak pressurizer pressure occurs	9.5
	Initiation of steam release from steam generator safety valves	~~12.0~~ 13.0
D. Without pressurizer control (maximum reactivity feedback)	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR occurs	4.0
	High pressurizer pressure reactor trip	5.7
	Rods begin to drop	7.7
	Peak pressurizer pressure occurs	9.0
	Initiation of steam release from steam generator safety valves	~~12.0~~ 13.0

Table 15.2-1 (Sheet 3 of 5)

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

Accident	Event	Time (s)
II. Loss of ac power to the plant auxiliaries	Feedwater lost	10.0
	Low steam generator water level reactor trip reached	58.6
	Rods begin to drop, ac power is lost, reactor coolant pumps start to coastdown	60.6
	Pressurizer safety valves open	64.0
	Steam generator safety valves open	65.0
	Maximum pressurizer pressure reached	68.0
	Maximum pressurizer water volume reached	72.5
	Pressurizer safety valves reclose	74.0
	PRHR actuation on low SG water level (narrow range)	123.6
	PRHR extracted heat matches decay heat	~3000.0 (50 min.)
	SG safety valves reclose	~6000.0

Replace with following page

Table 15.2-1 (Sheet 4 of 5)

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

Accident	Event	Time (s)
III. Loss of Normal Feedwater Flow	Feedwater lost	10.0
	Low steam generator water level (narrow range) reactor trip reached	58.6
	Rods begin to drop	60.6
	Maximum pressurizer pressure reached	63.5
	Maximum pressurizer water volume reached	64.0
	SG safety valves open	68.0
	PRHR actuation on low SG water level (narrow range)	123.6
	SG safety valves reclose	146.0
	CMTs actuation and RCP trip on low T-cold ("S" signal)	770.0
	Pressurizer safety valves first open	~3050.0
	Pressurizer safety valves final reclose	~15000.0
	PRHR extracted heat matches decay heat	~16500.0

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Table 15.2-1
(Sheet 3 of 5)

Time Sequence of Events for Incidents which Result in a Decrease in Heat Removal by the Secondary System

Accident	Event	Time (s)
II. Loss of ac power to the plant auxiliaries	Feedwater lost	10.0
	Low steam generator water level reactor trip reached	83.8
	Rods begin to drop, ac power is lost, reactor coolant pumps start to coastdown	85.8
	Pressurizer safety valves open	86.6
	Steam generator safety valves open	87.3
	Maximum pressurizer pressure reached	87.8
	Pressurizer safety valves reclose	90.9
	Maximum pressurizer water volume reached	97.2
	PRHR actuation on low SG water level (narrow range) coincident with low SFW	151.
	Pressurizer safety valves open	806.
	Pressurizer safety valves reclose	1,455.
	SG safety valves reclose	7,902.
	CMT actuation on low T-cold "S" signal	8,002.
	Steamline isolation on low T-cold "S" signal	8,002.
	Pressurizer safety valves open	18,180.
	Pressurizer safety valves reclose	22,082.
	PRHR extracted heat matches decay heat	~24,000.

Table 15.2-1
(Sheet 4 of 5)

Time Sequence of Events for Incidents which Result in a Decrease in Heat Removal by the Secondary System

Accident	Event	Time (s)
III. Loss of Normal Feedwater Flow	Feedwater lost	10.0
	Low steam generator water level (narrow range) reactor trip reached	83.8
	Rods begin to drop	85.8
	Pressurizer safety valves open	86.6
	SG safety valves open	87.3
	Maximum pressurizer pressure reached	87.8
	Pressurizer safety valves reclose	89.1
	Maximum pressurizer water volume reached	89.2
	PRHR actuation on low SG water level (wide range)	143.
	SG safety valves reclose	188.
	CMTs actuation on low T-cold "S" signal	568.
	Steamline isolation on low T-cold "S" signal	568.
	RCP trip on low T-cold "S" signal	583.
	Pressurizer safety valves open	8,670.
	Pressurizer safety valves reclose	18,574.
	PRHR extracted heat matches decay heat	~25,000.

Table 15.2-1 (Sheet 5 of 5)

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

Accident	Event	Time (s)
IV. Feedwater System Pipe Break	Main feedline break occurs	10.0
	Pressurizer safety valves open	~~51.0~~ 40.5
	Low Steam Generator water level (narrow range) reactor trip reached	~~58.6~~ 83.0
	Rods begin to drop	~~60.6~~ 85.0
	Loss of offsite power occurs	~~60.6~~ 85.0
	~~Low RCS flow rate~~	~~61.2~~
	Low steamline pressure setpoint	~~69.7~~ 89.2
	Pressurizer safety valves close	~~70.5~~ 96.0
	All steam and feedline isolation valves close	~~81.7~~ 101.2
	PRHR actuation on low SG water level (wide range)	~~86.7~~ 107.1
	CMT valves fully opened	~~91.7~~ 111.24
	Faulted Steam Generator empties	~~95.0~~ 111.0
	Intact Steam Generator safety valves open	~~187.0~~ 144
	Intact Steam Generator safety valves reclose	~~358.0~~ 558.
	Pressurizer safety valves open	~~>2500.0~~ ~3640
	PRHR extracted heat matches decay heat	~~~18100.0 (about 5 hr)~~ ~11550
	~~Pressurizer safety valves reclose~~	~~~18200.0~~

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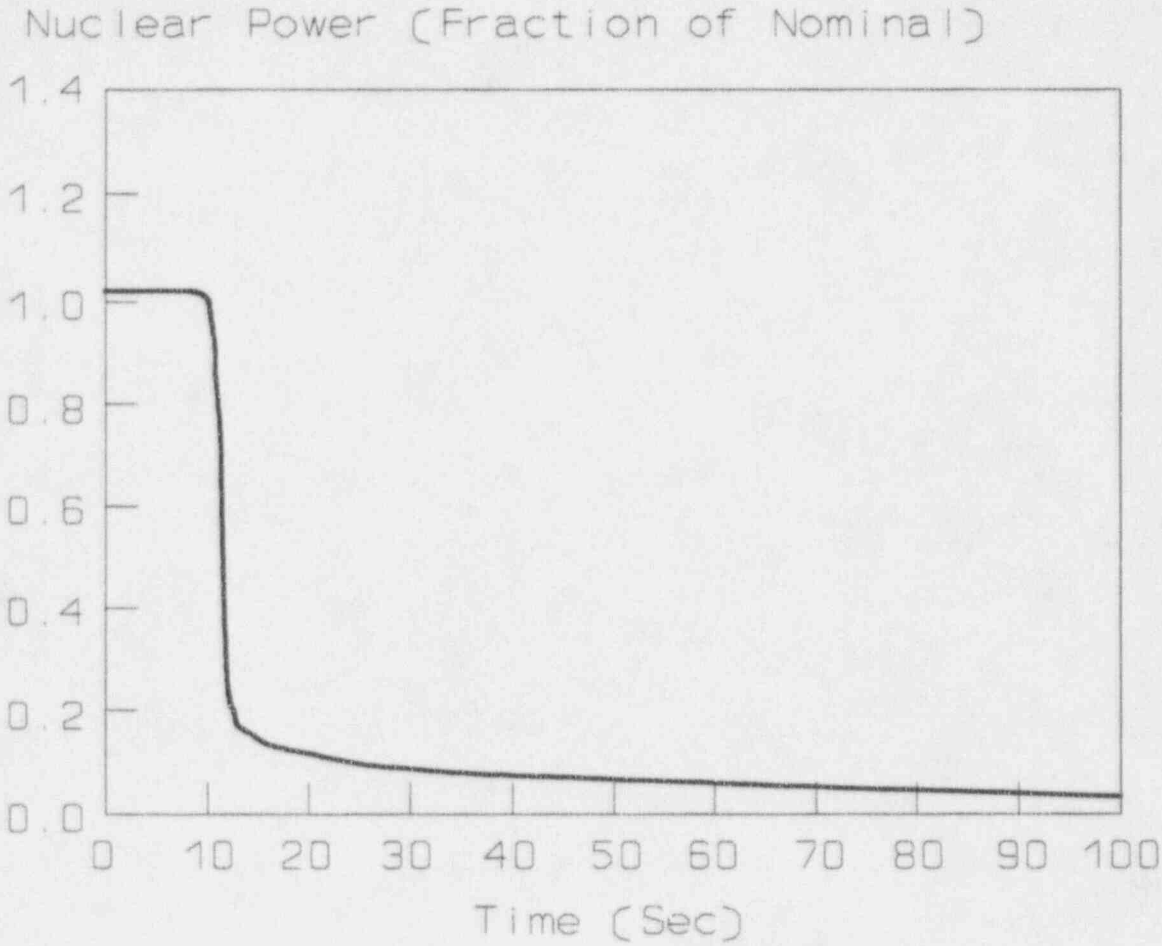

Figure 15.2.3-1

Nuclear Power (Fraction of Nominal) vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

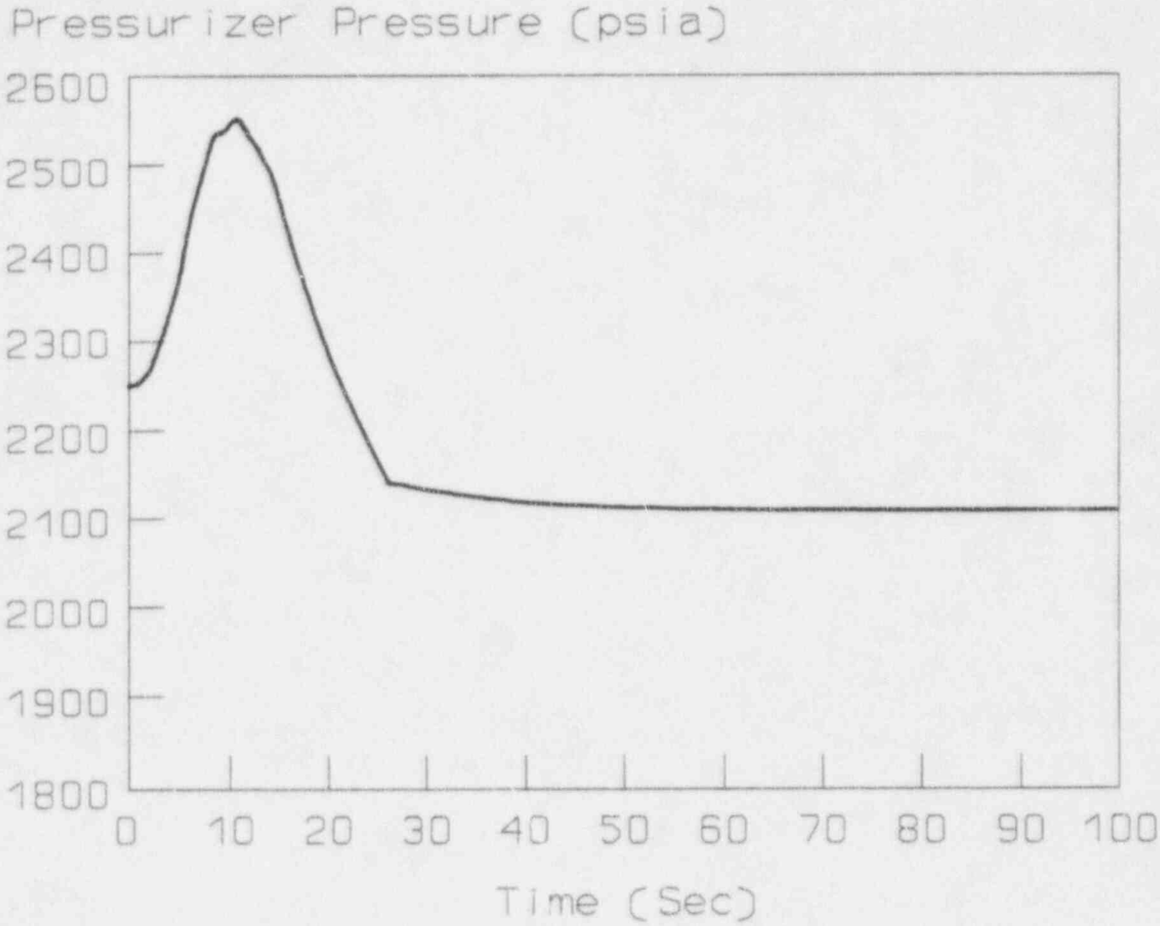

Figure 15.2.3-2

Pressurizer Pressure (psia) vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

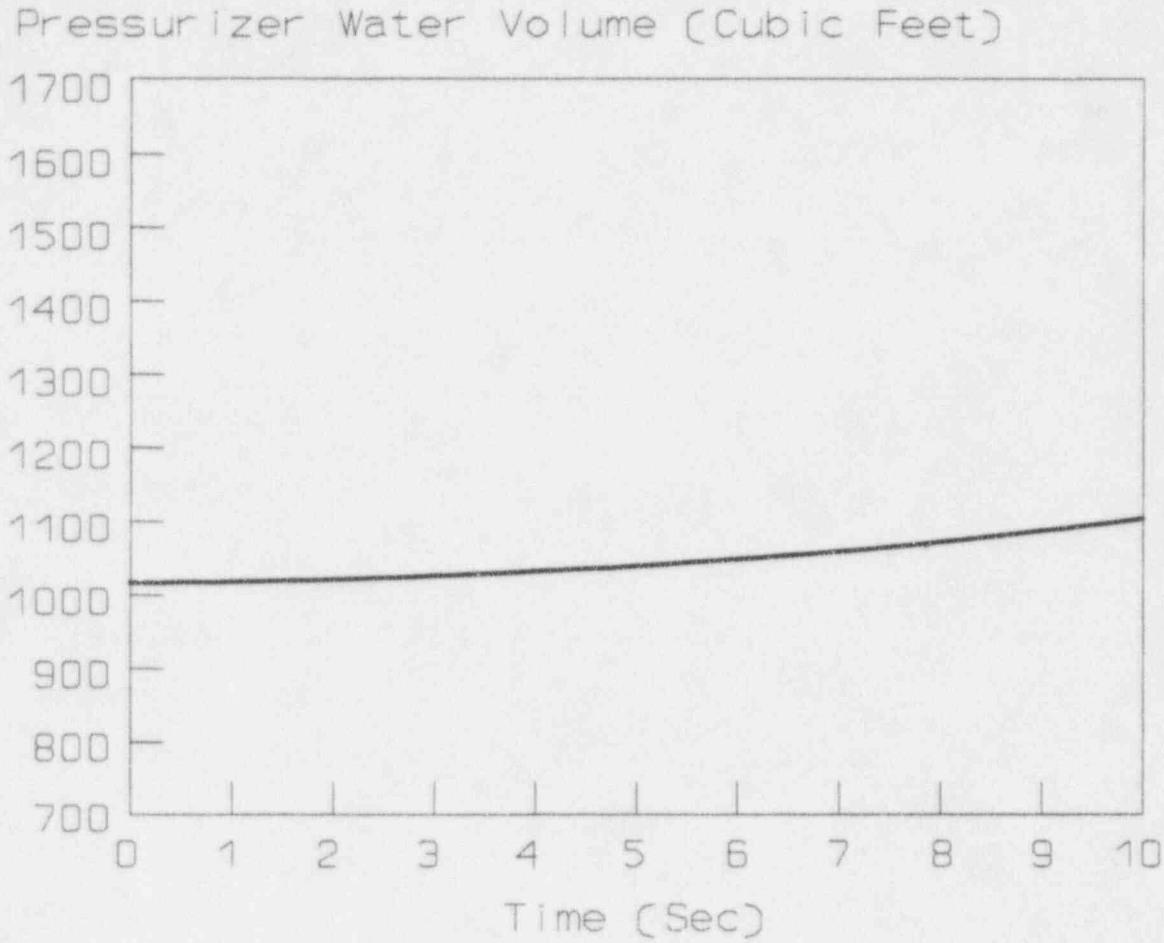

Figure 15.2.3-3

Pressurizer Water Volume (ft^3) vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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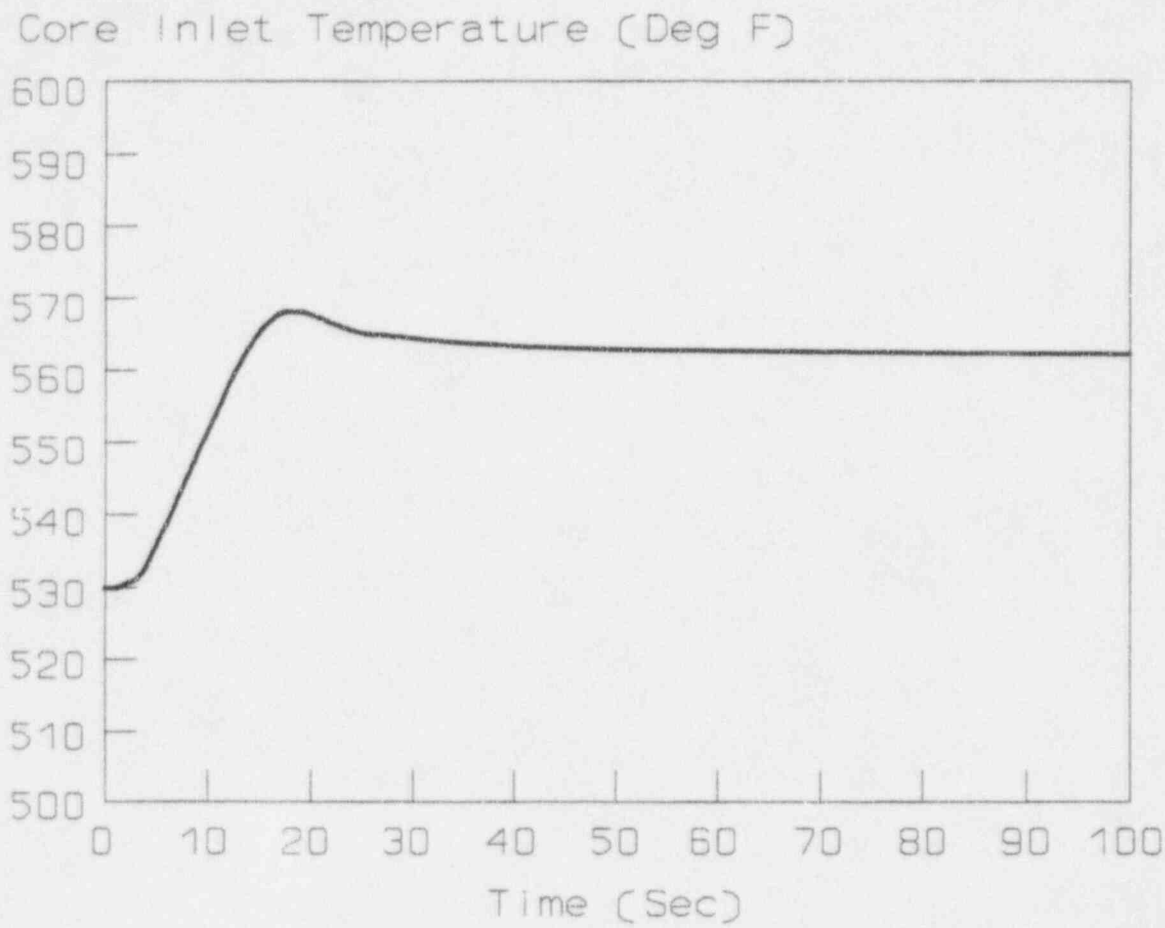

Figure 15.2.3-4

Core Inlet Temperature (°F) vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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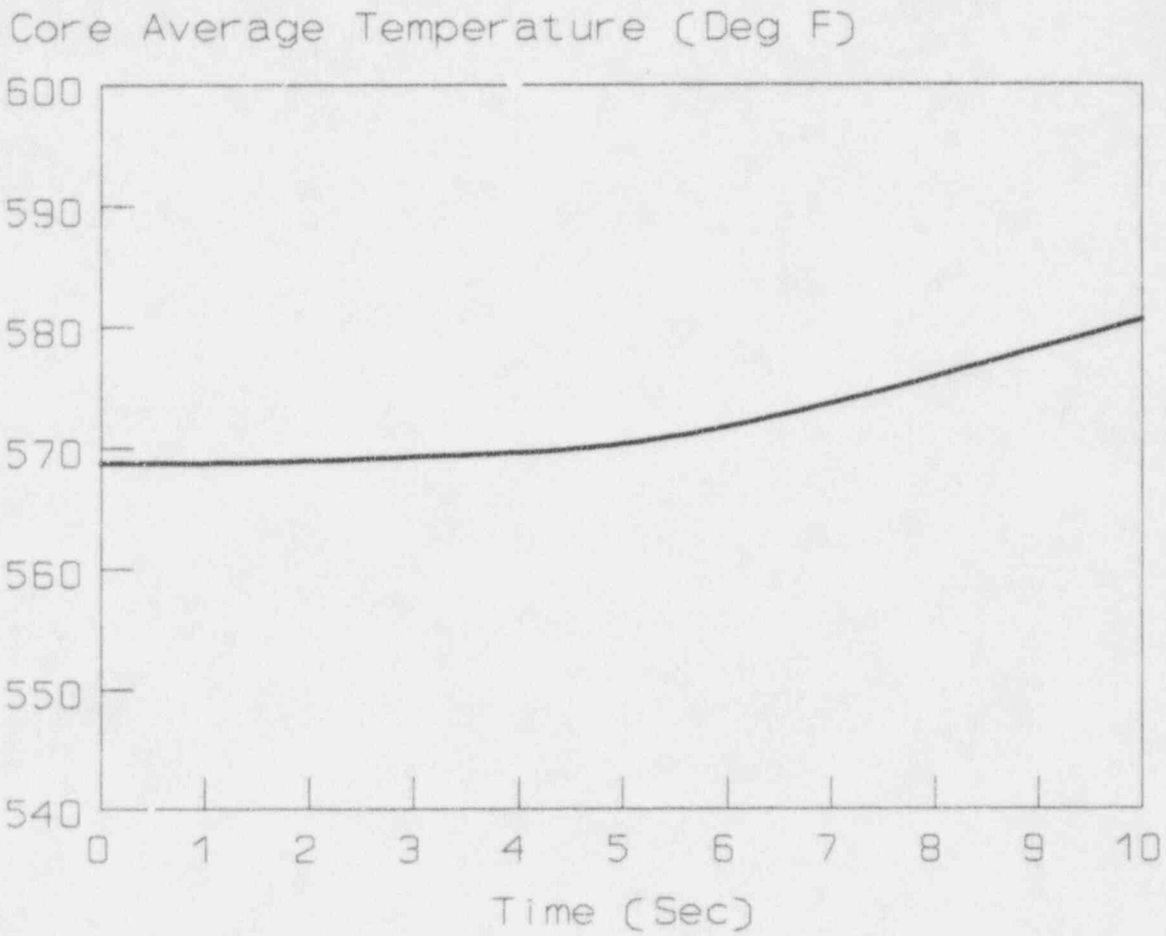

Figure 15.2.3-5

Core Average Temperature (°F) vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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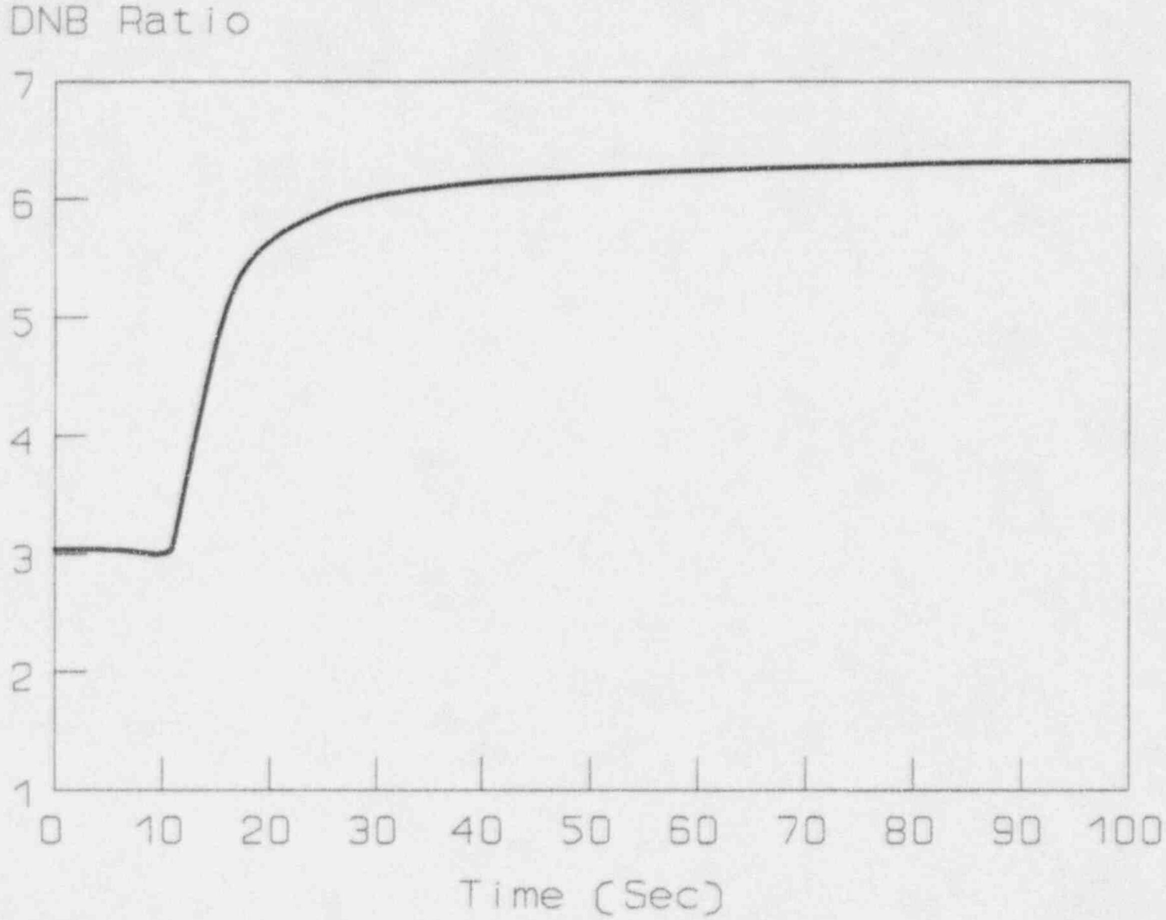

Figure 15.2.3-6

DNB Ratio vs. Time for Turbine Trip Accident with Pressurizer Spray and Minimum Moderator Feedback

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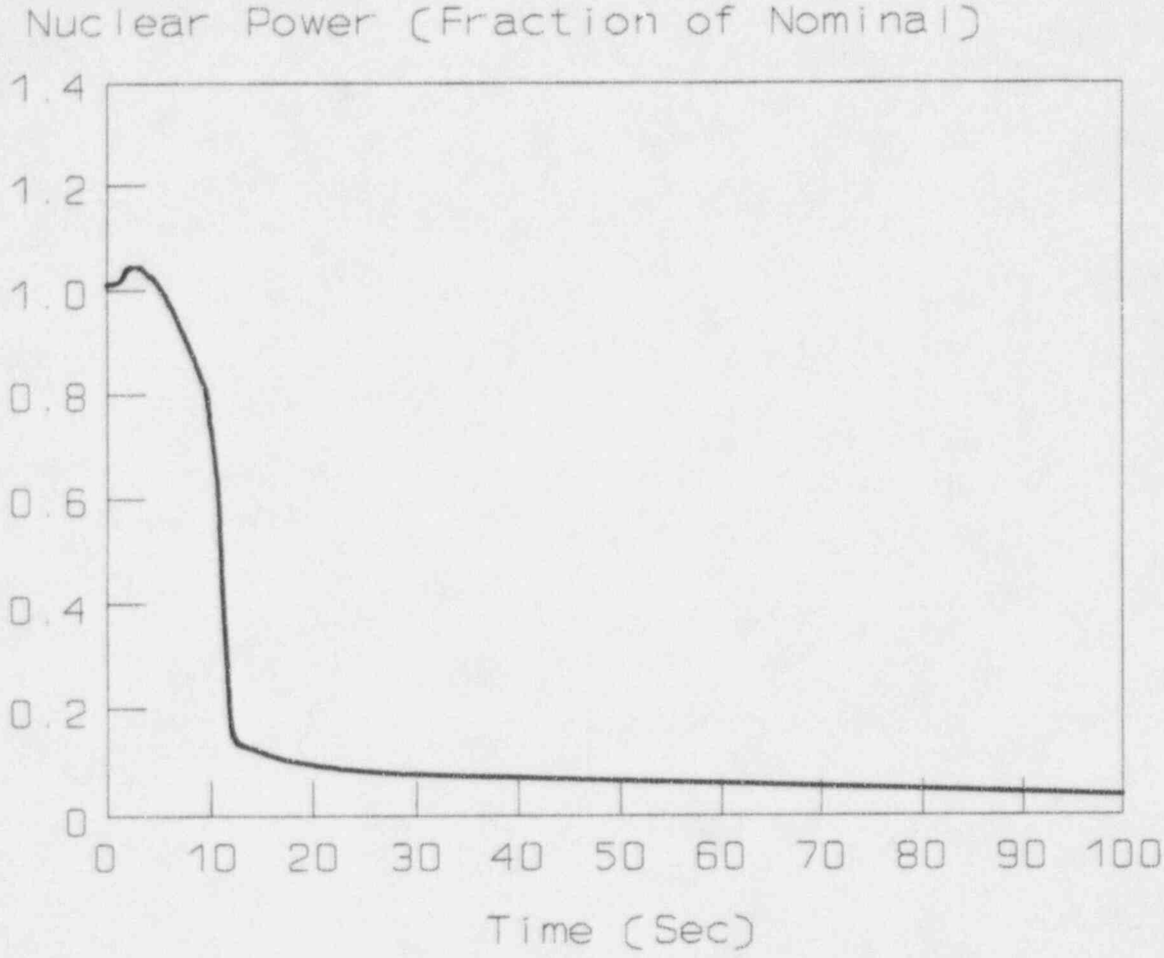

Figure 15.2.3-7

Nuclear Power (Fraction of Nominal) vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

even pg

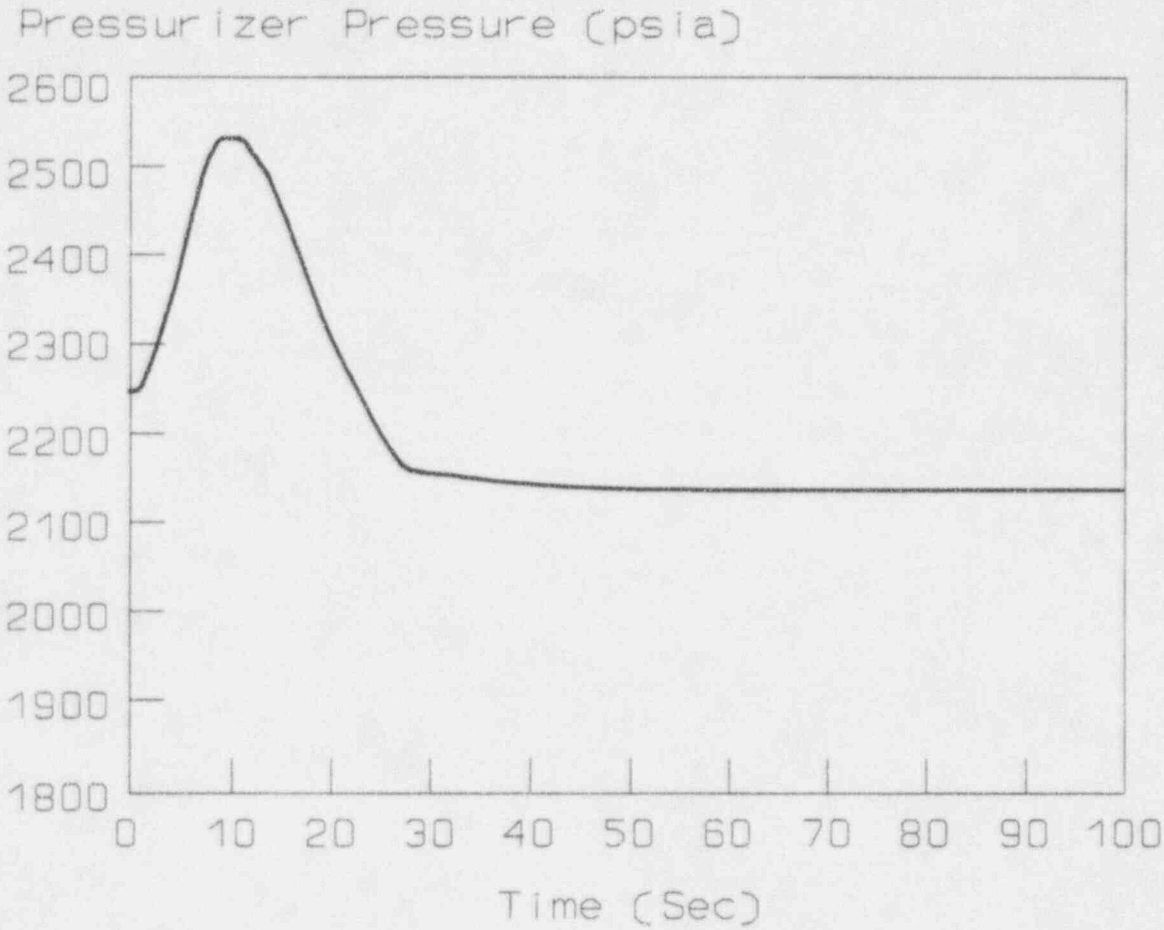

Figure 15.2.3-8

Pressurizer Pressure (psia) vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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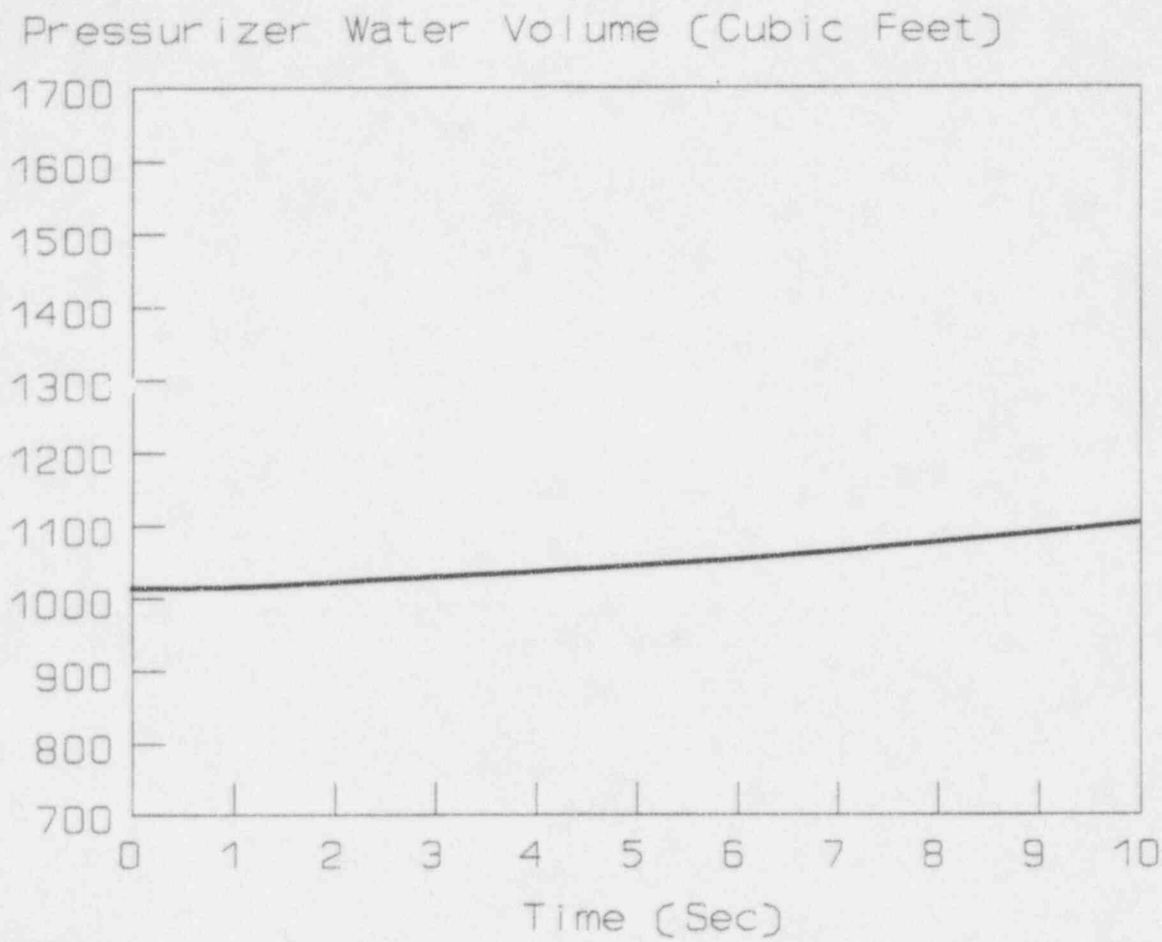

Figure 15.2.3-9

Pressurizer Water Volume (ft^3) vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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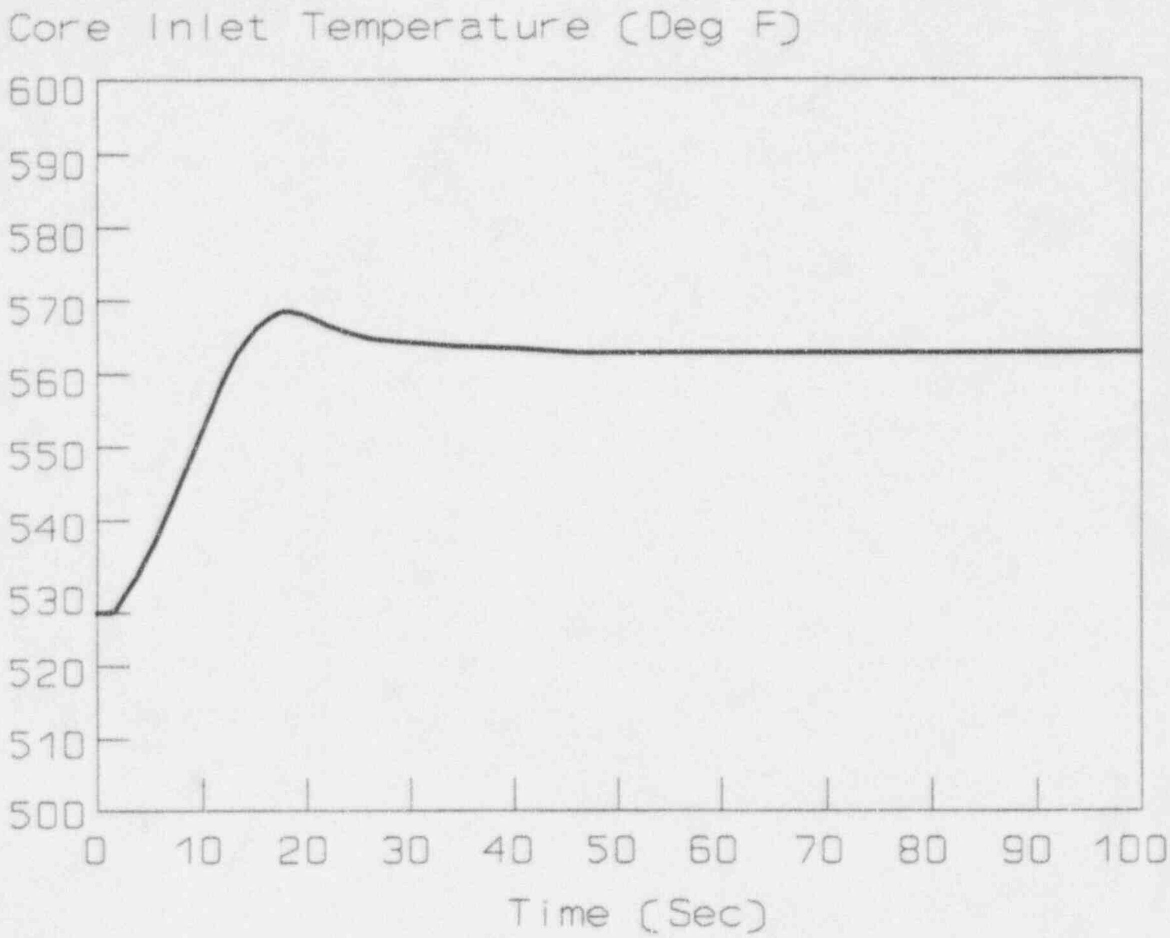

Figure 15.2.3-10

Core Inlet Temperature (°F) vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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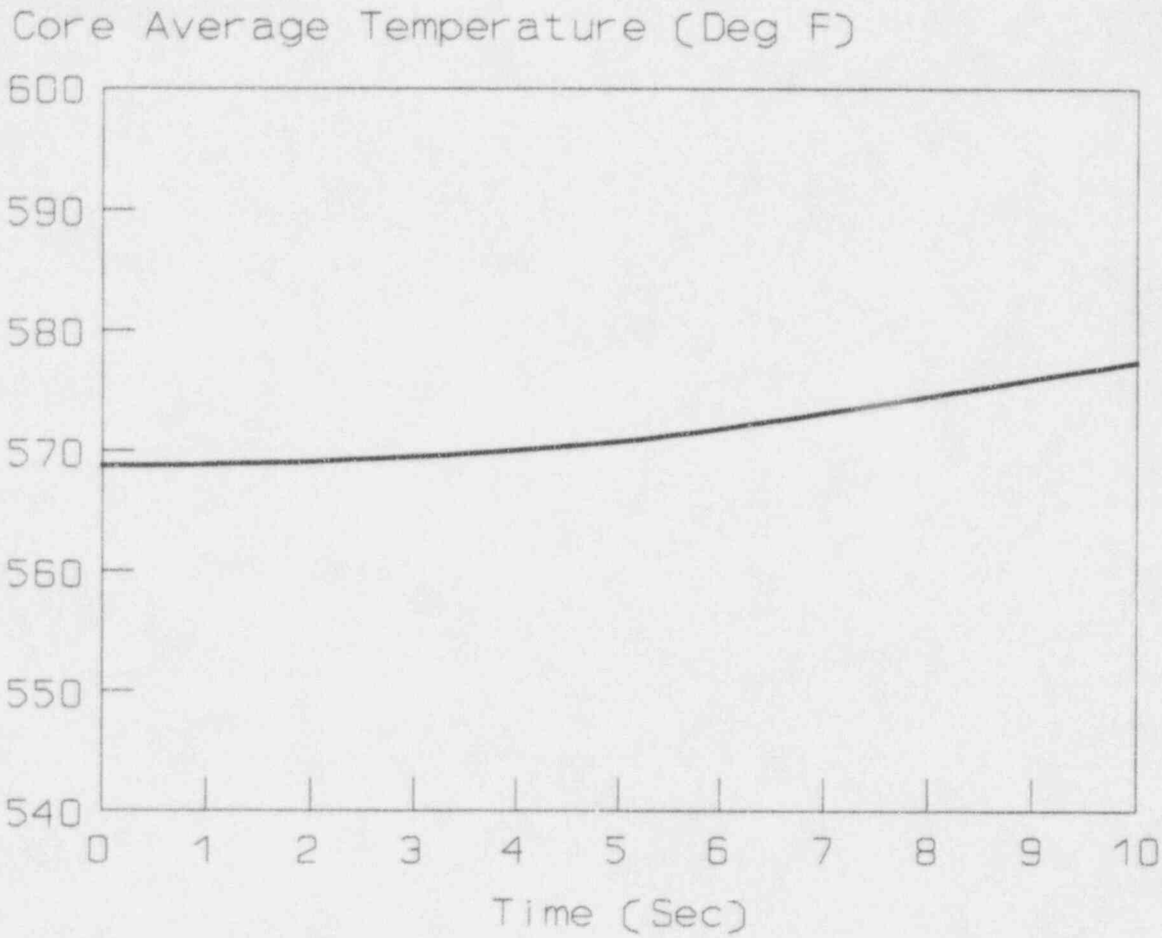

Figure 15.2.3-11

Core Average Temperature (°F) vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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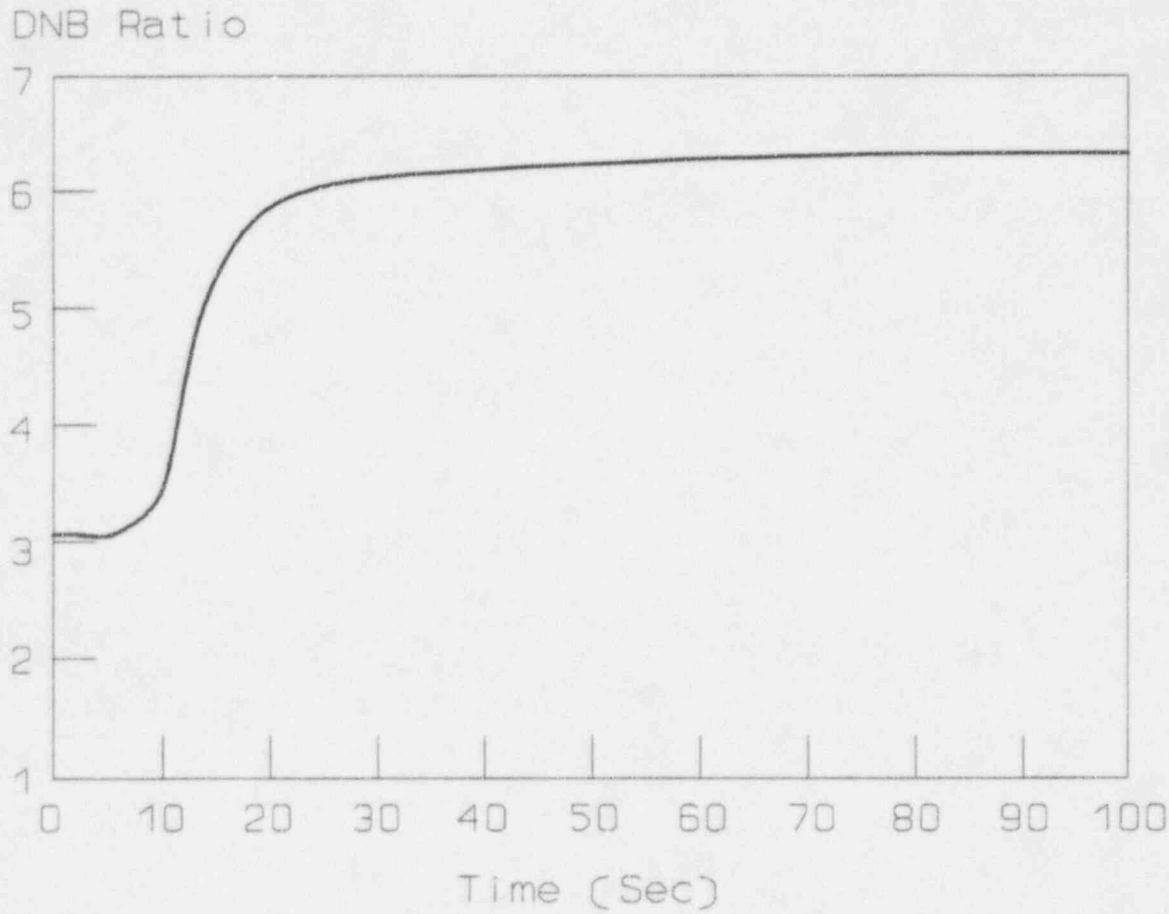

Figure 15.2.3-12

DNB Ratio vs. Time for Turbine Trip Accident with Pressurizer Spray and Maximum Moderator Feedback

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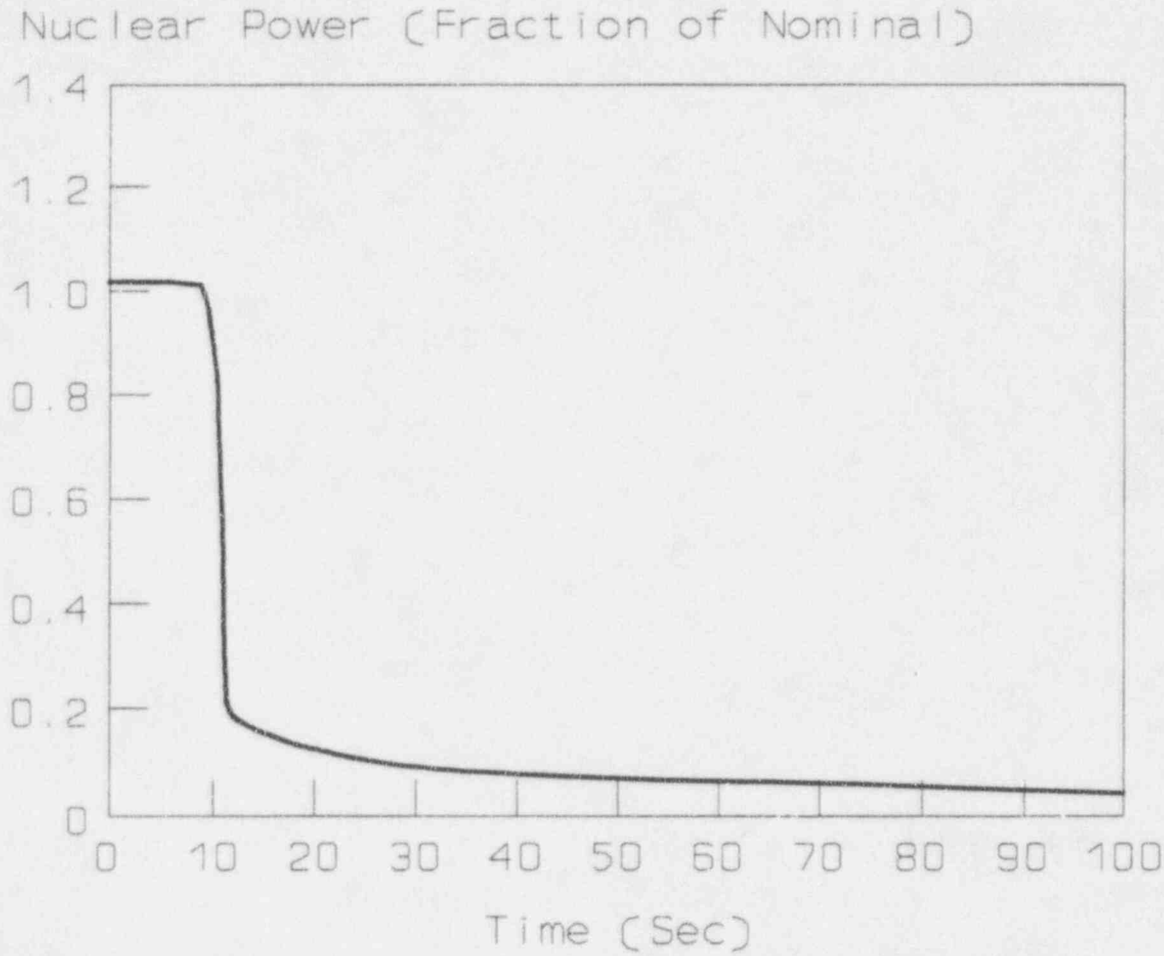

Figure 15.2.3-13

Nuclear Power (Fraction of Nominal) vs. Time for Turbine Trip Accident without Pressurizer Spray and Minimum Moderator Feedback

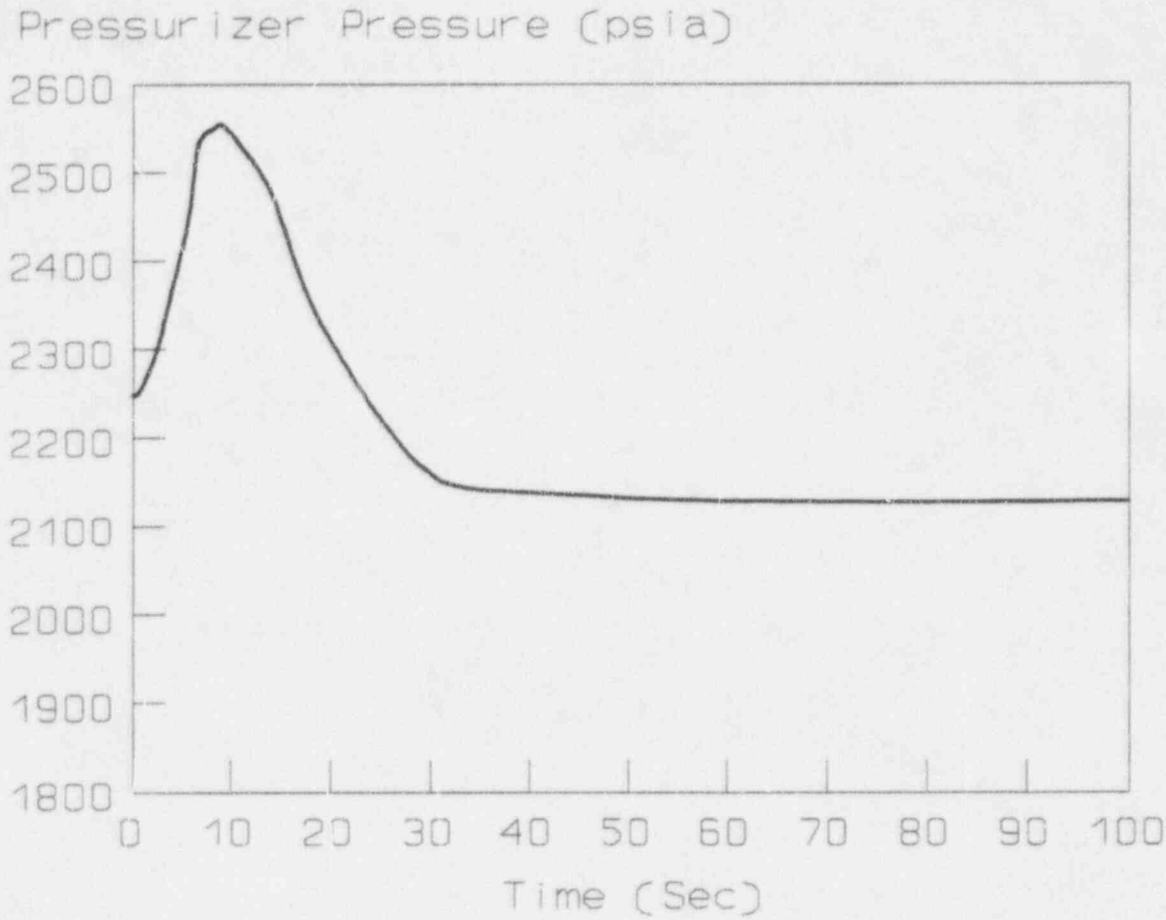

Figure 15.2.3-14

Pressurizer Pressure (psia) vs. Time for Turbine Trip Accident without Pressurizer Spray and Minimum Moderator Feedback

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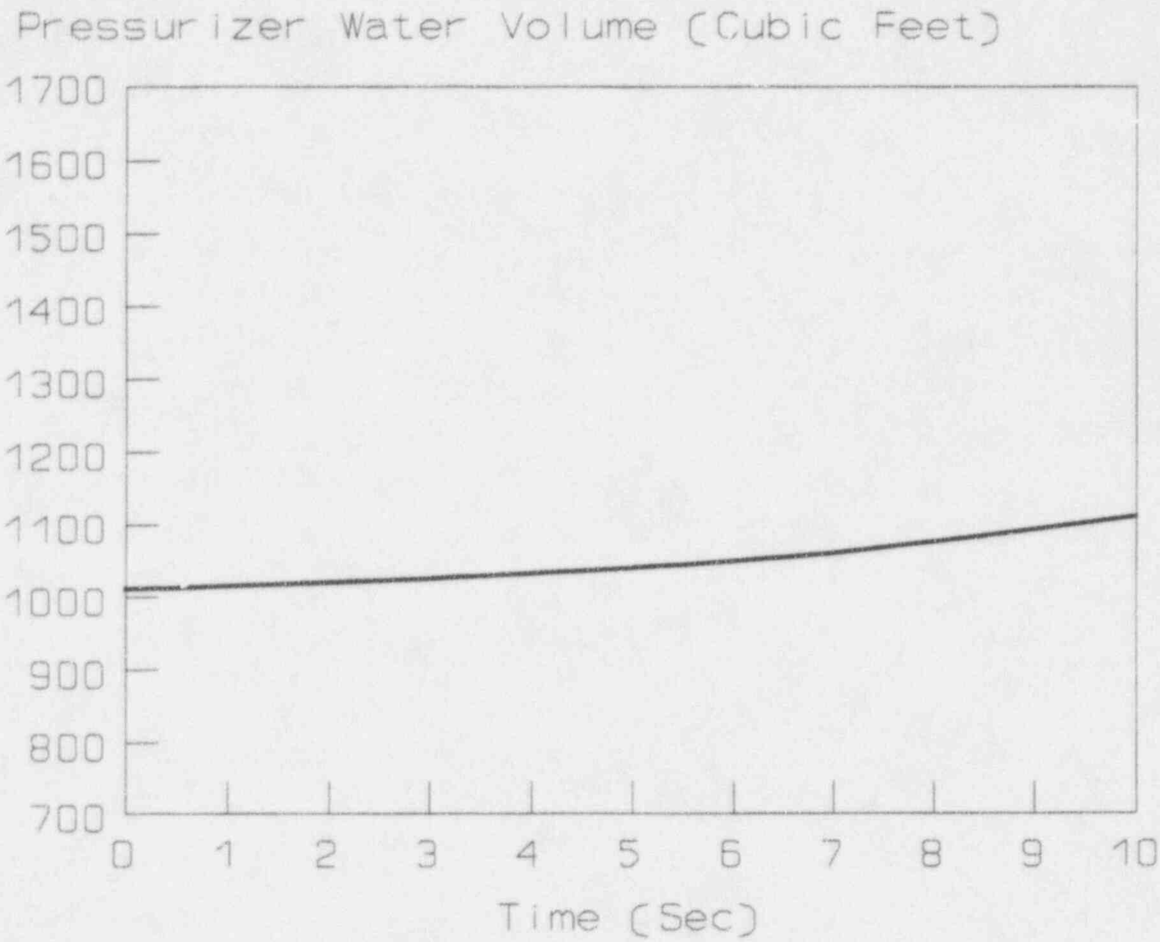

Figure 15.2.3-15

Pressurizer Water Volume (ft^3) vs. Time for Turbine Trip Accident without Pressurizer Spray and Minimum Moderator Feedback

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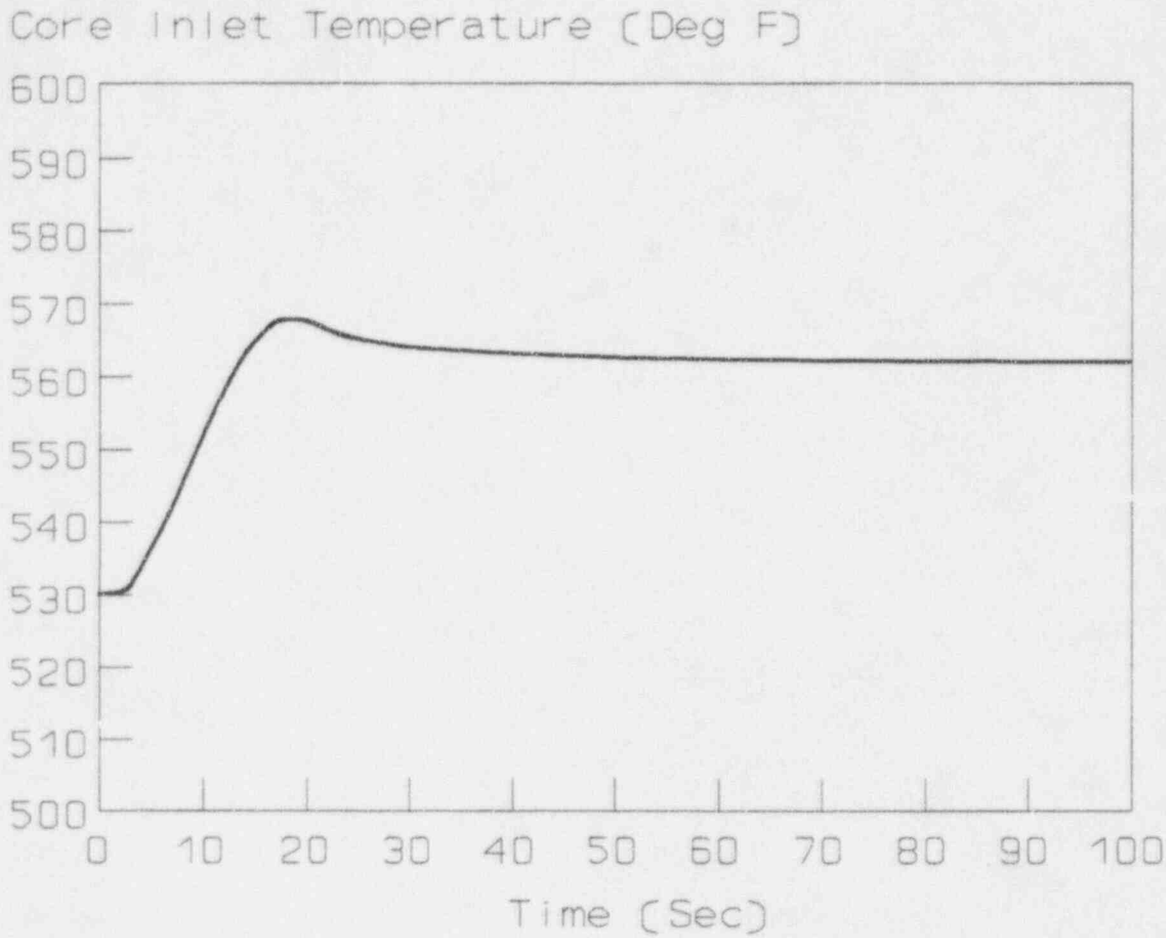

Figure 15.2.3-16

Core Inlet Temperature (°F) vs. Time for Turbine Trip Accident without Pressurizer Spray and Minimum Moderator Feedback

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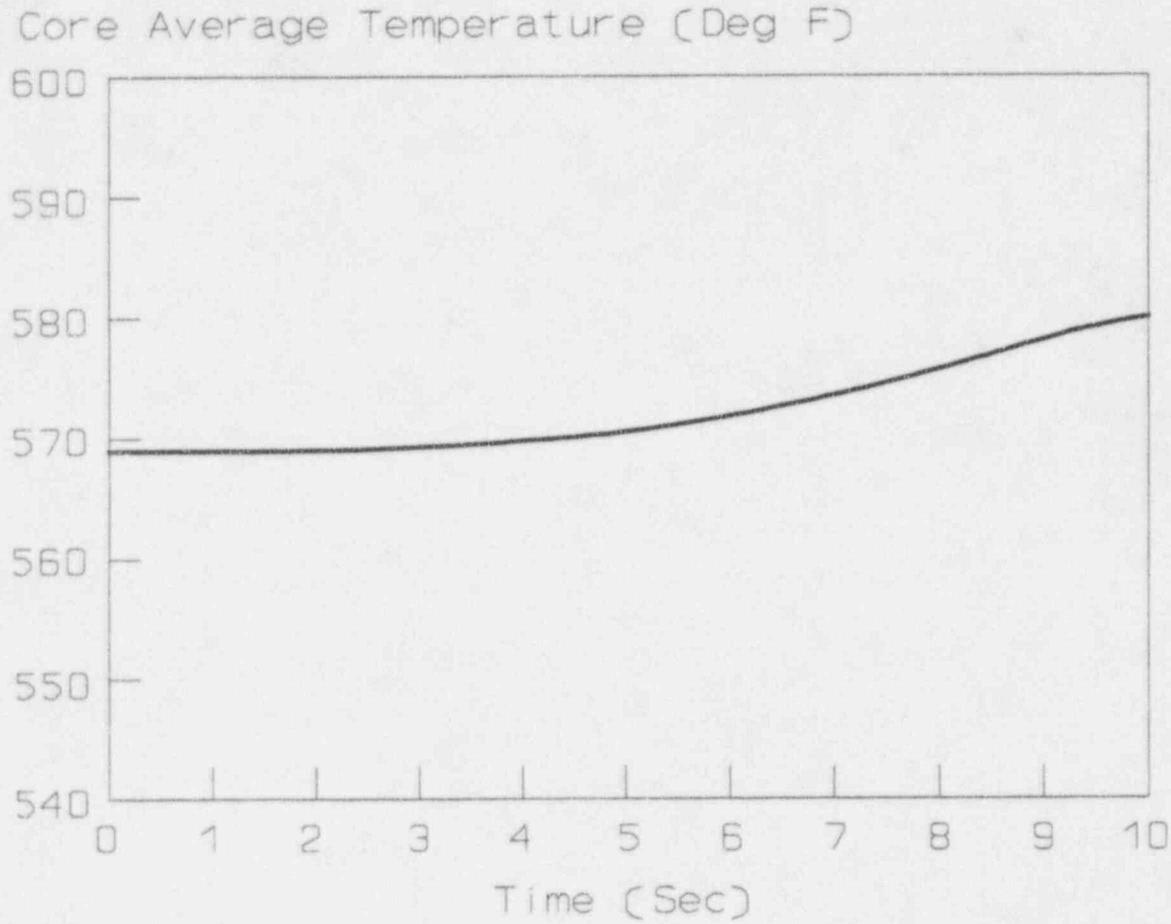

Figure 15.2.3-17

Core Average Temperature (°F) vs. Time for Turbine Trip Accident without Pressurizer Spray and Minimum Moderator Feedback

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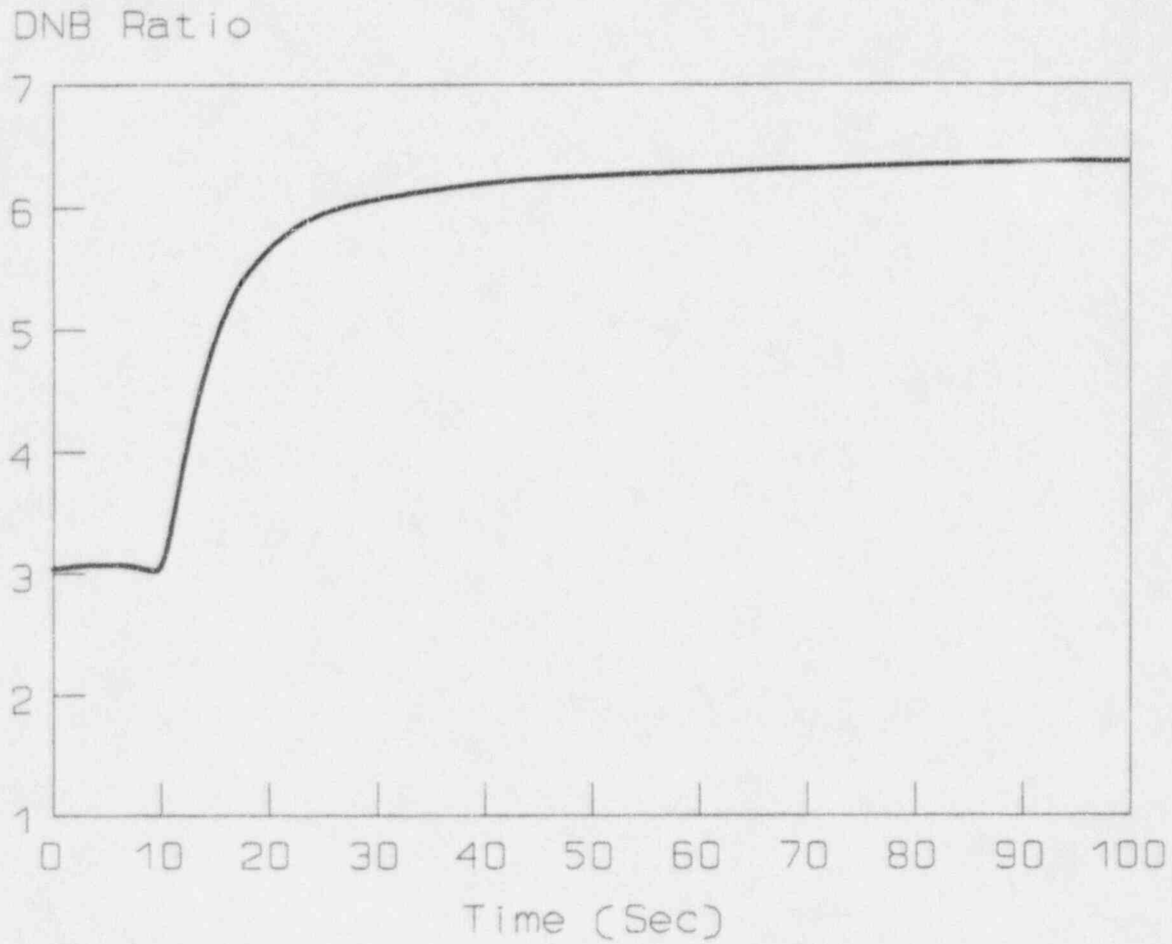

Figure 15.2.3-18

DNB Ratio vs. Time for Turbine Trip Accident Without Pressurizer Spray and Minimum Moderator Feedback

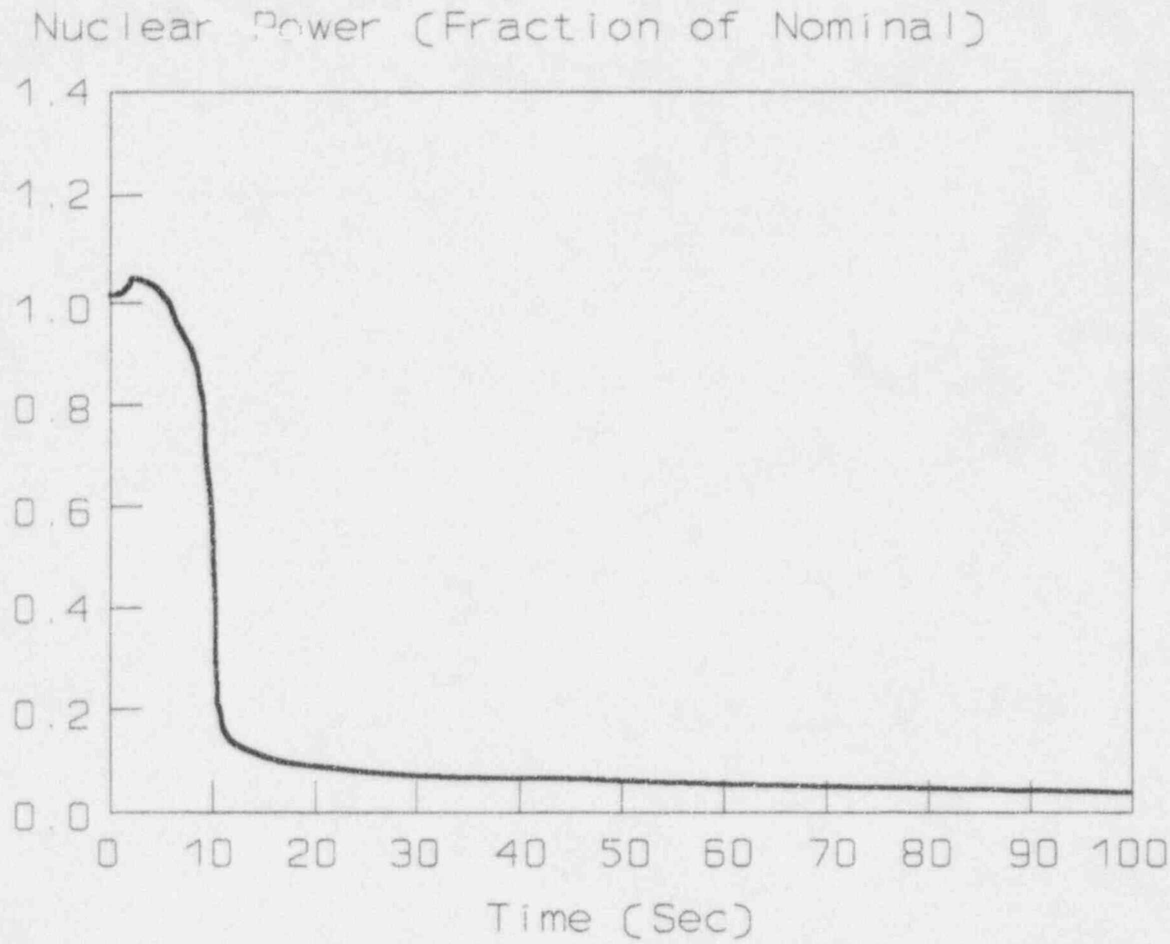

Figure 15.2.3-19

Nuclear Power (Fraction of Nominal) vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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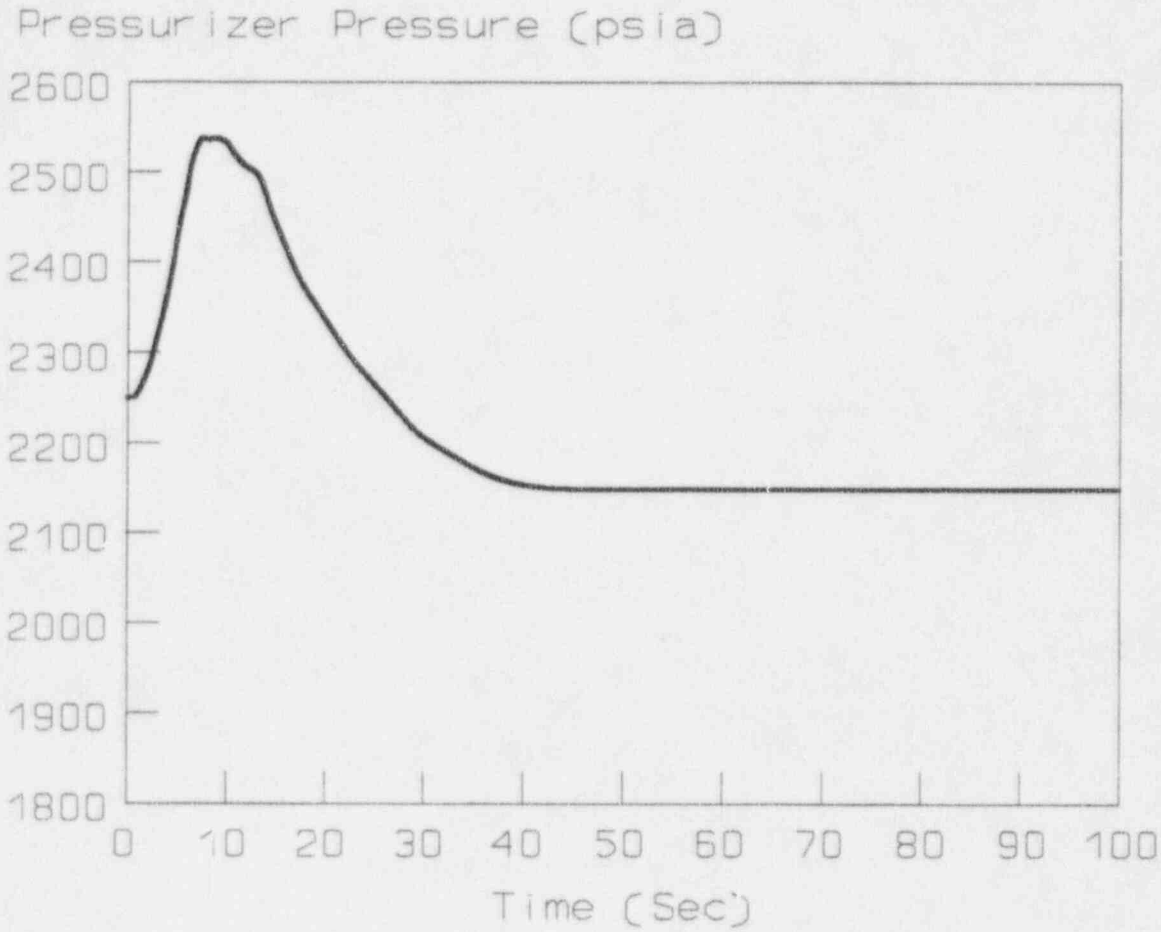

Figure 15.2.3-20

Pressurizer Pressure (psia) vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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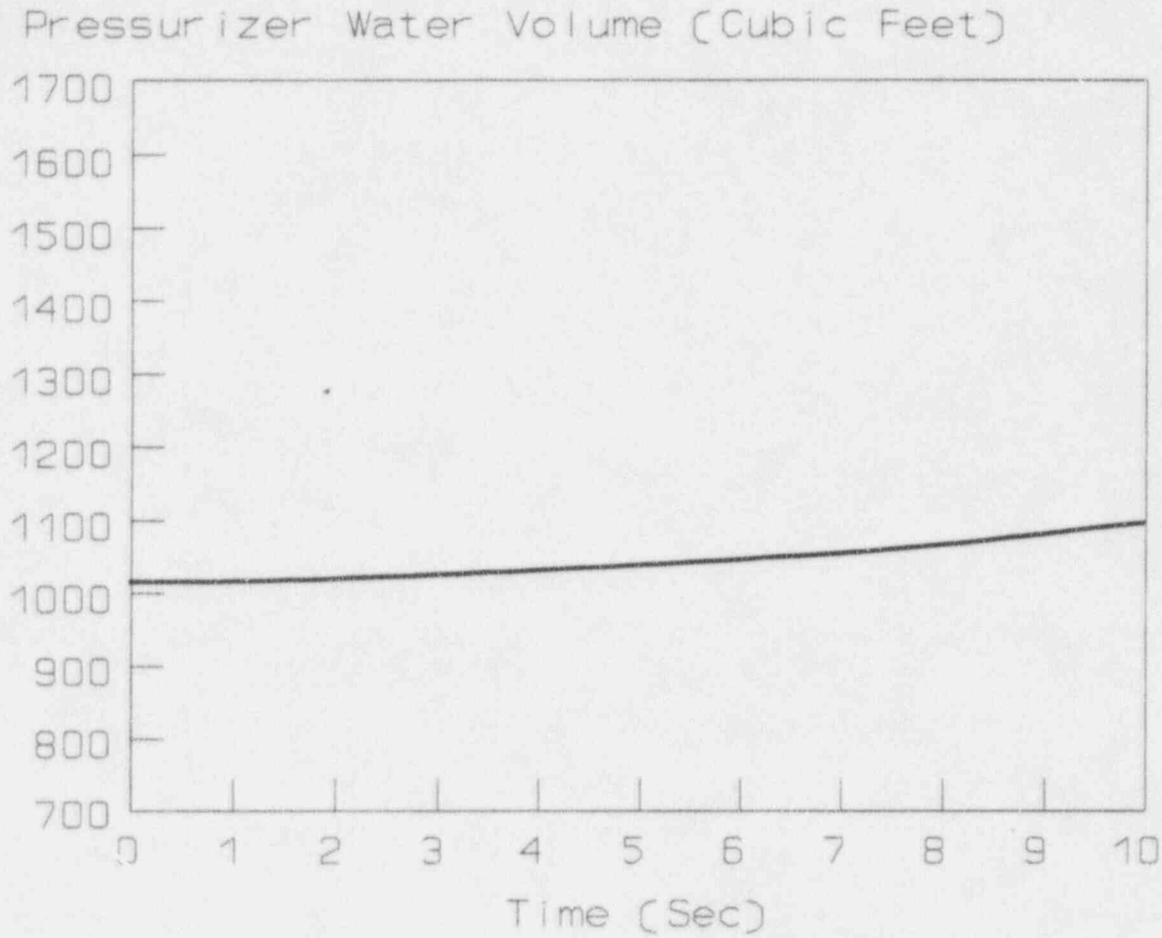

Figure 15.2.3-21

Pressurizer Water Volume (ft^3) vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

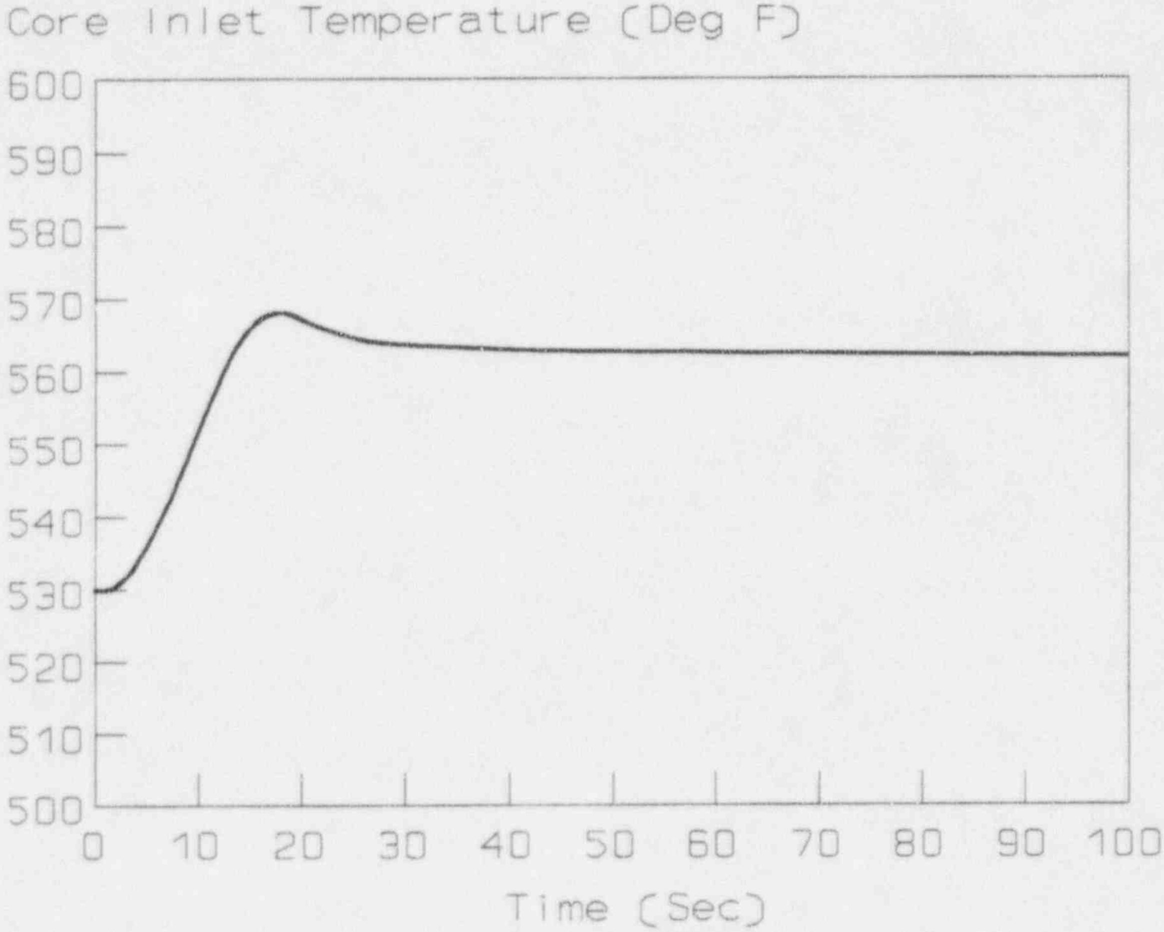

Figure 15.2.3-22

Core Inlet Temperature (°F) vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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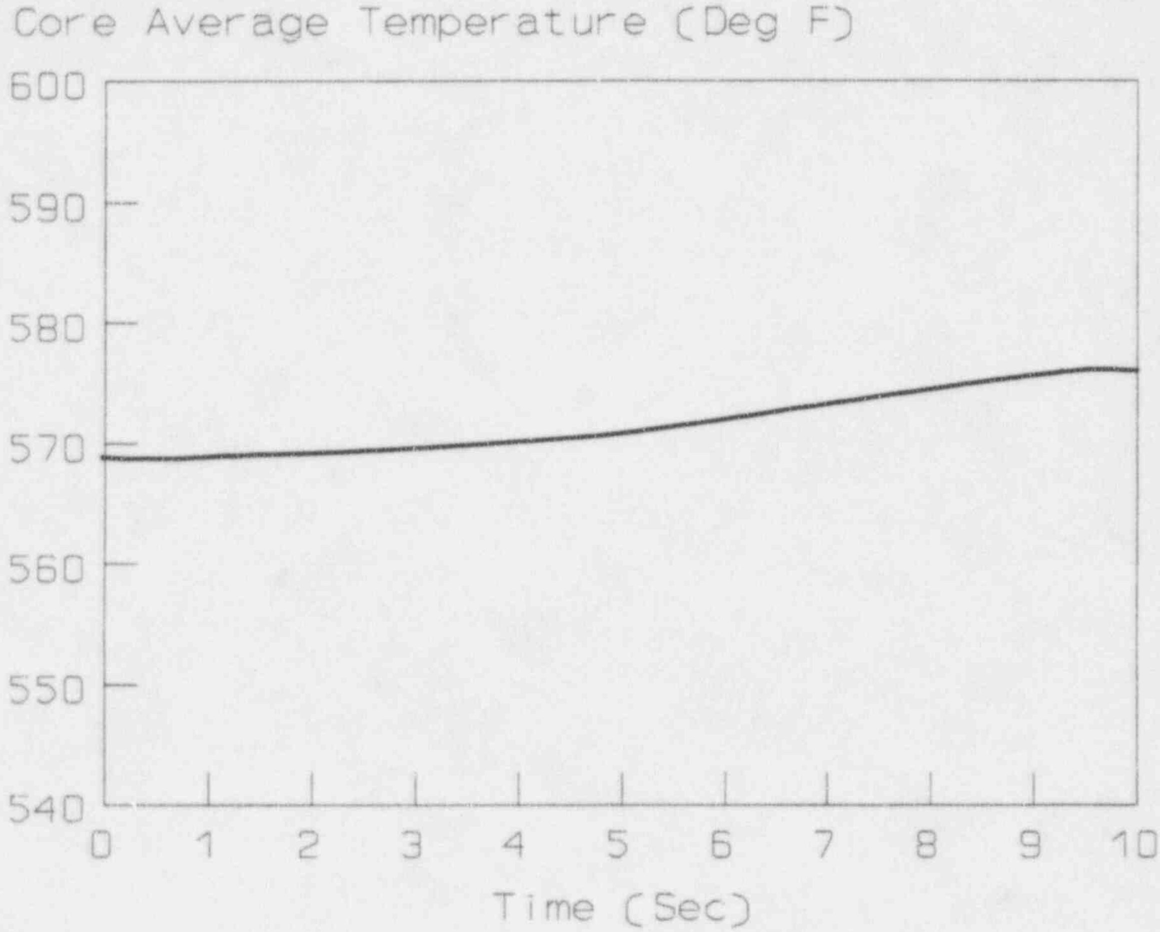

Figure 15.2.3-23

Core Average Temperature (°F) vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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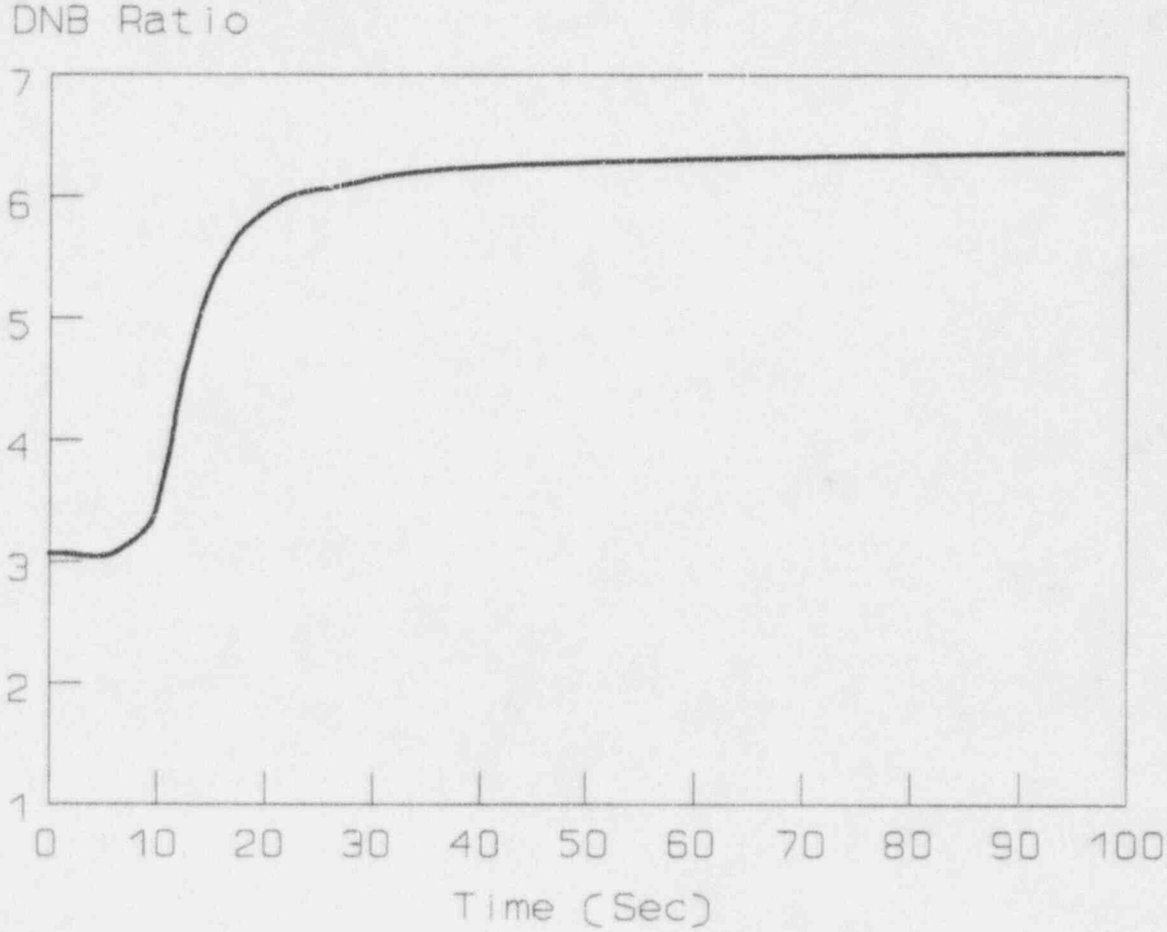

Figure 15.2.3-24

DNB Ratio vs. Time for Turbine Trip Accident Without Pressurizer Spray and Maximum Moderator Feedback

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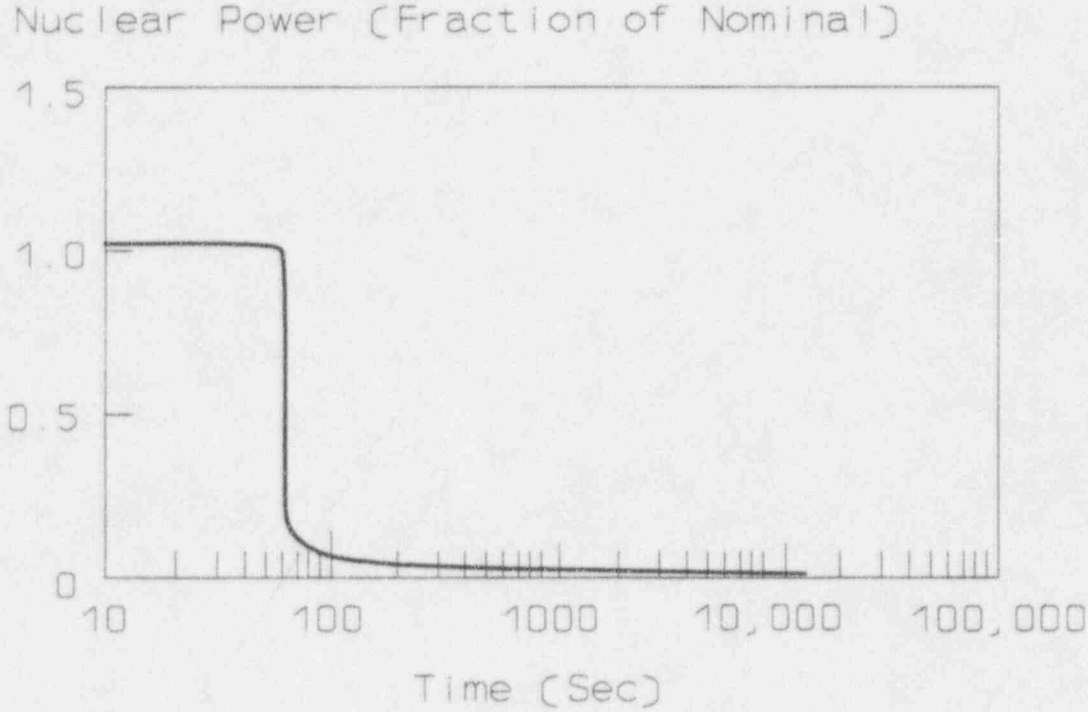

Figure 15.2.6-1

Nuclear Power Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

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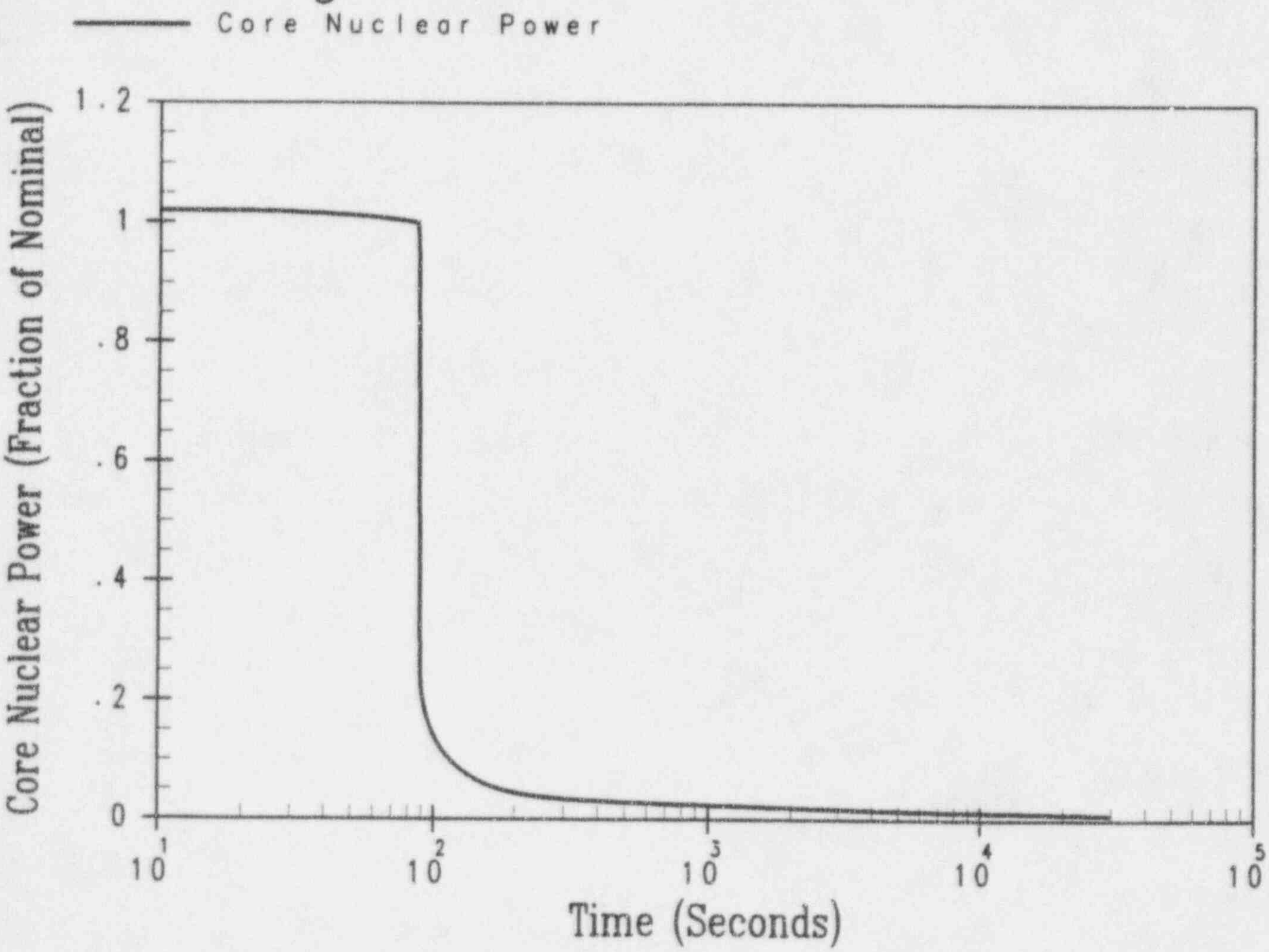
Figure 15.2.6-1



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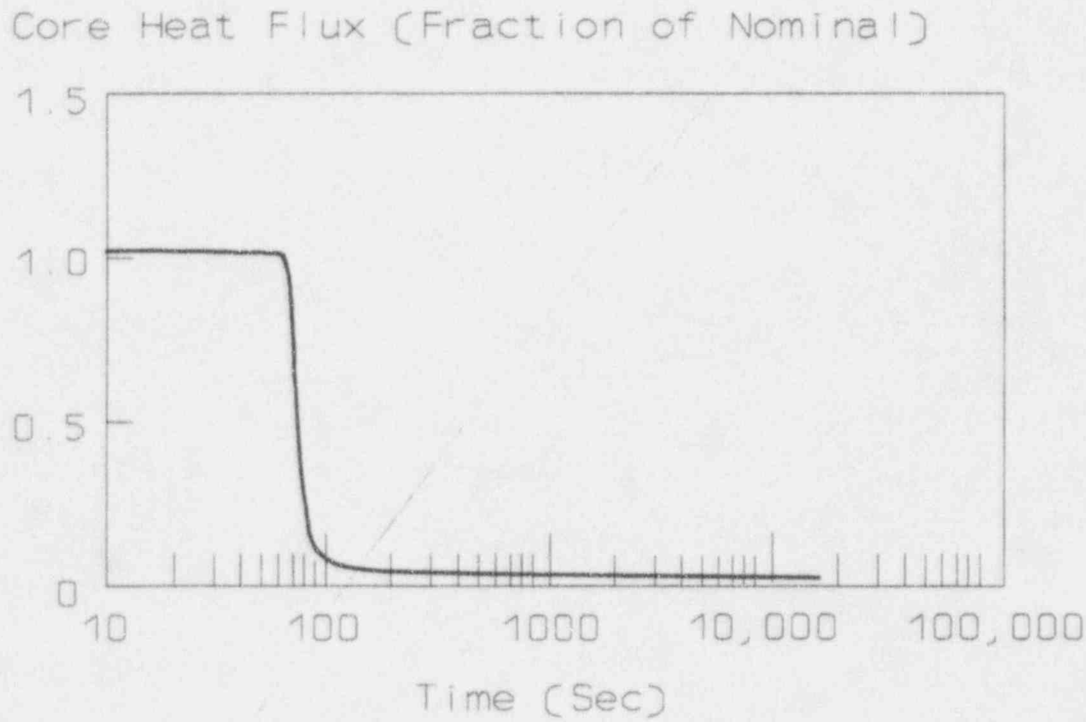

Figure 15.2.6-2

Core Heat Flux Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

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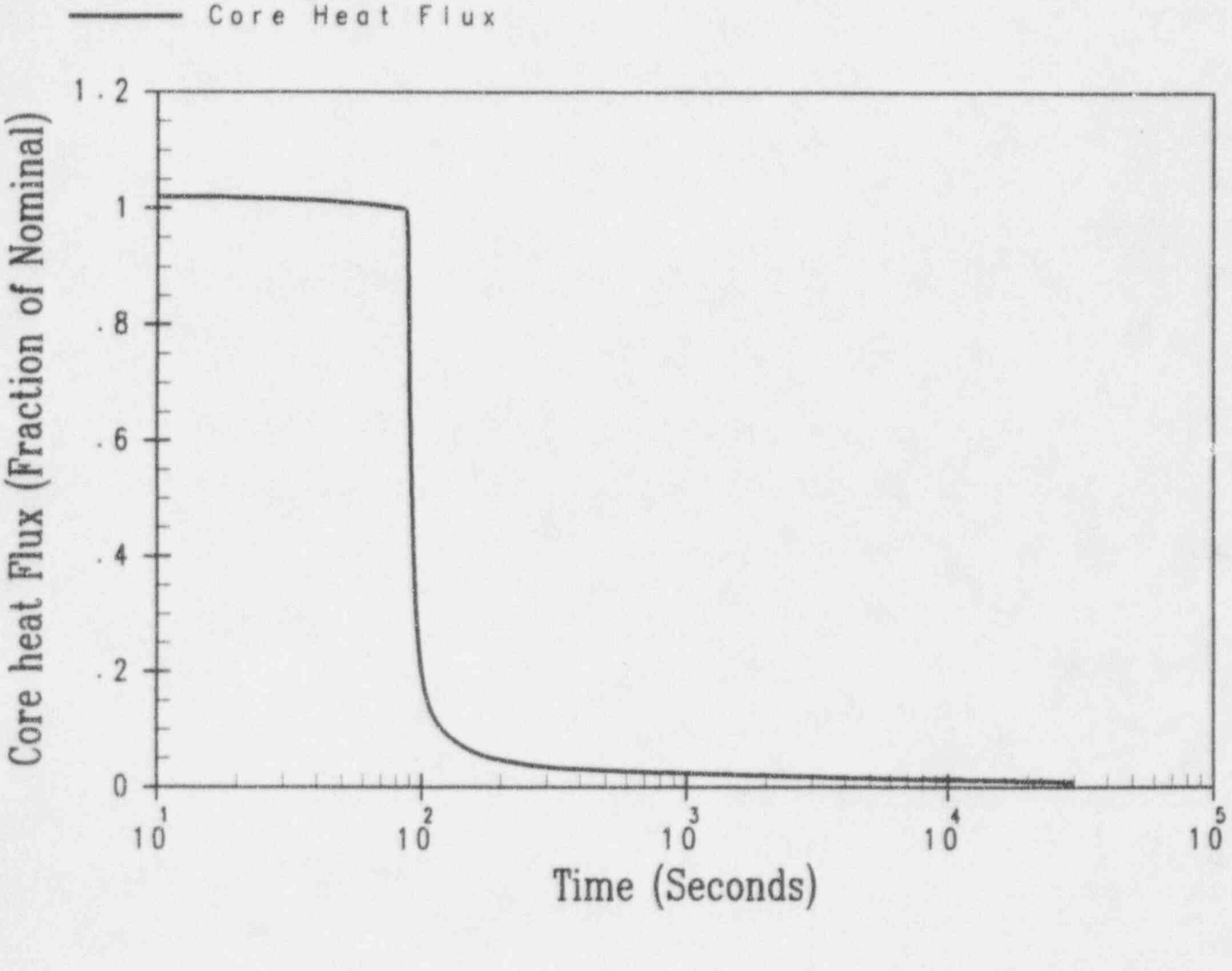
Figure 15.2.6-2



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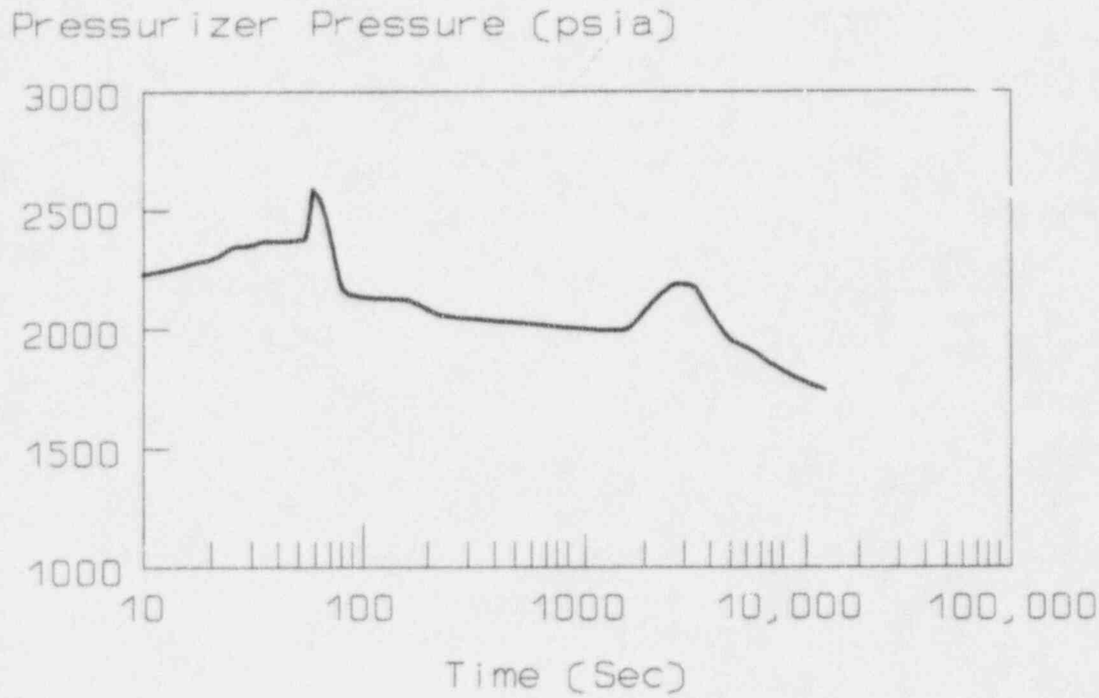

Figure 15.2.6-3

Pressurizer Pressure Transient for Loss of Nonemergency AC Power to the Plant Auxiliaries

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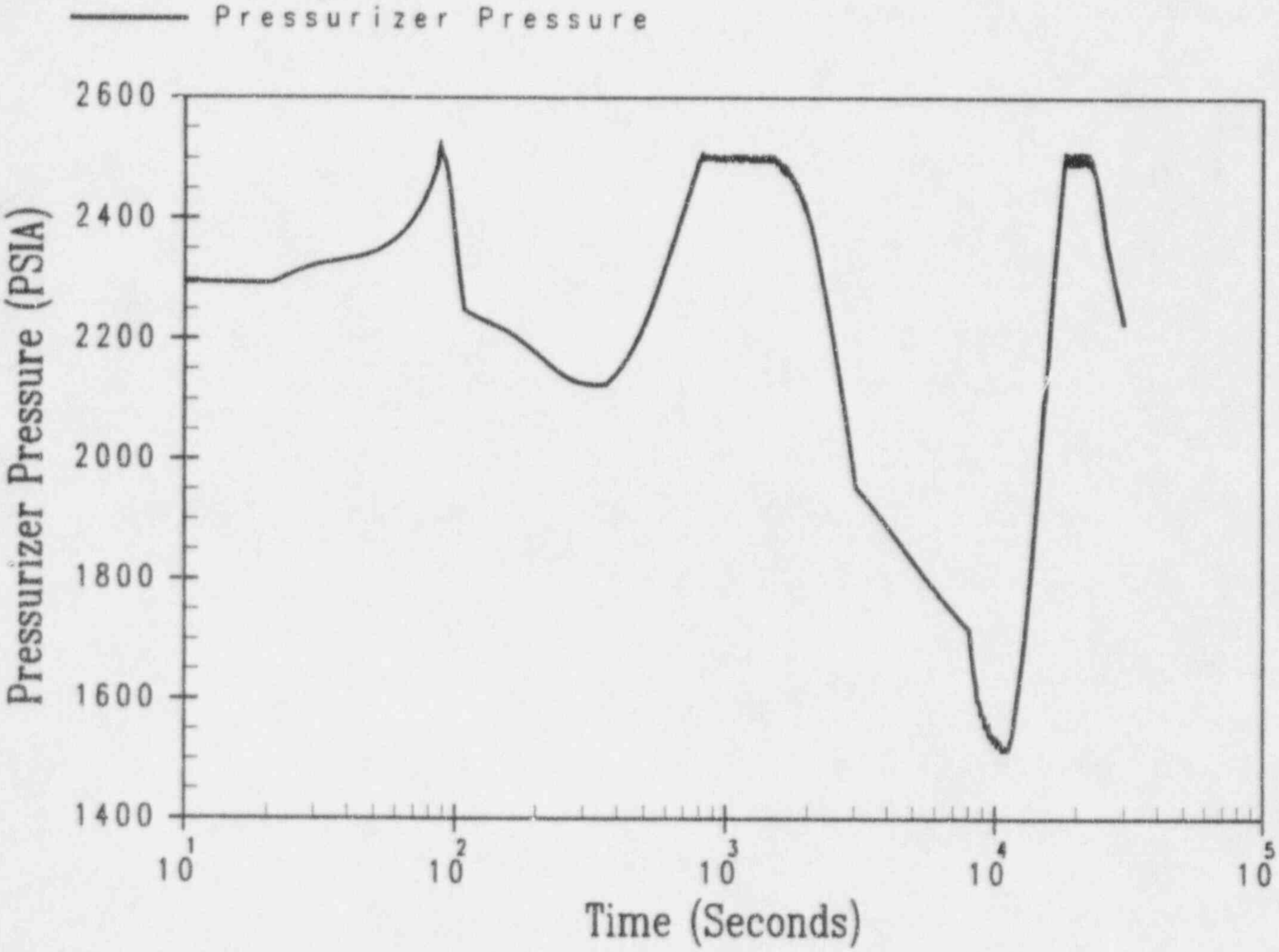
Figure 15.2.6-3



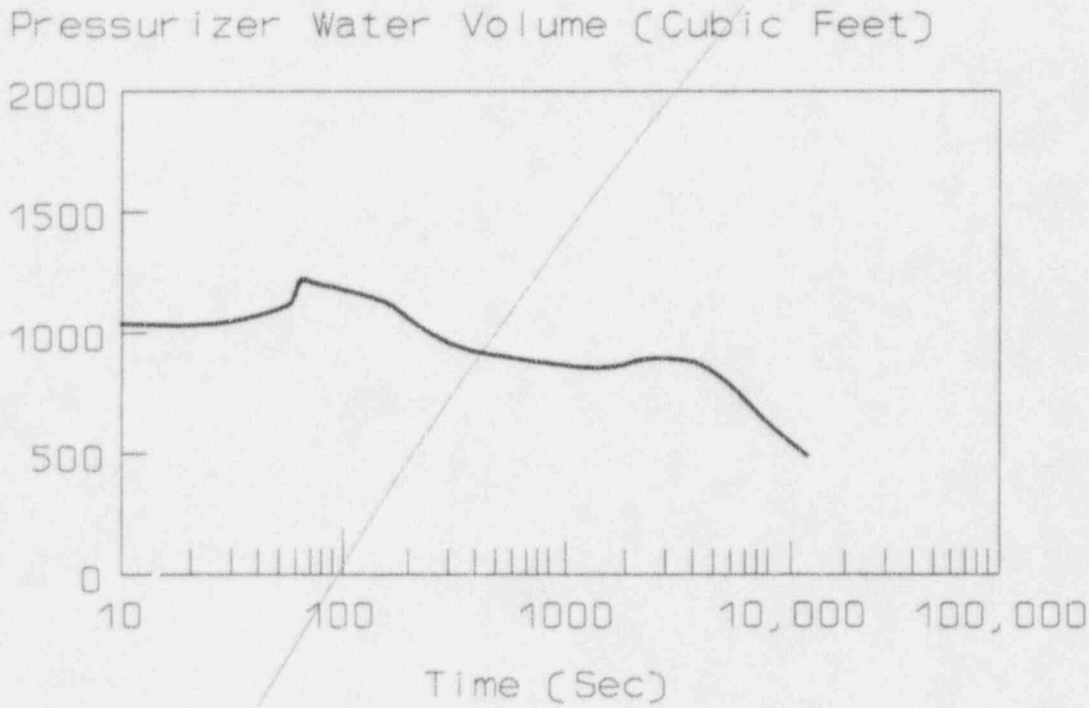


Figure 15.2.6-4

Pressurizer Water Volume Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

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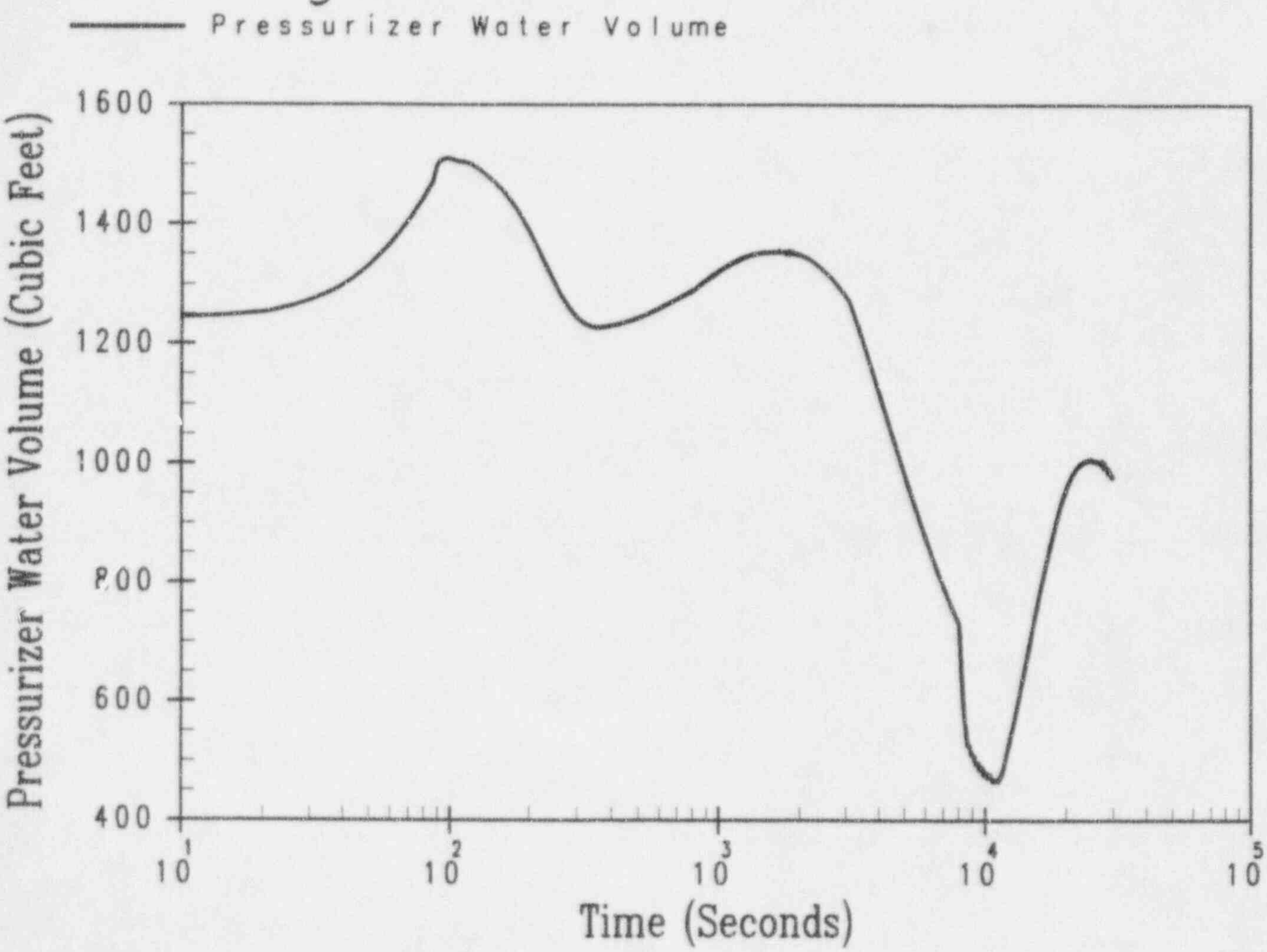
Figure 15.2.6-4



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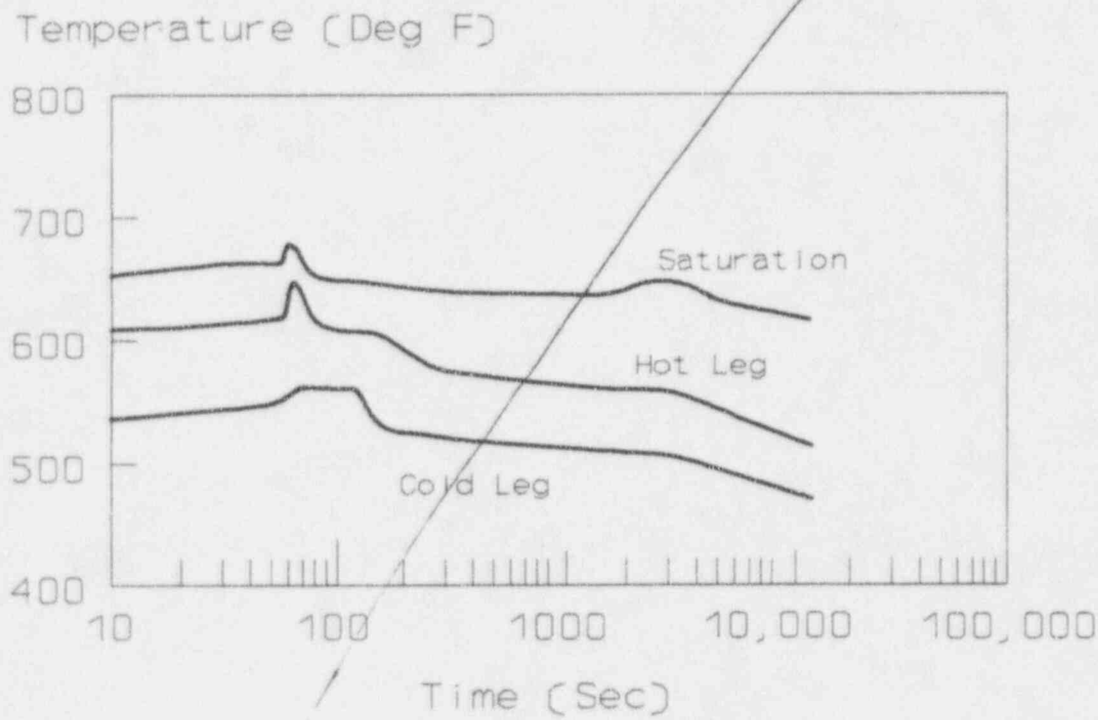

Figure 15.2.6-5

RCS Temperature Transients in Loop Containing the PRHR For Loss of Nonemergency AC Power to the Plant Auxiliaries

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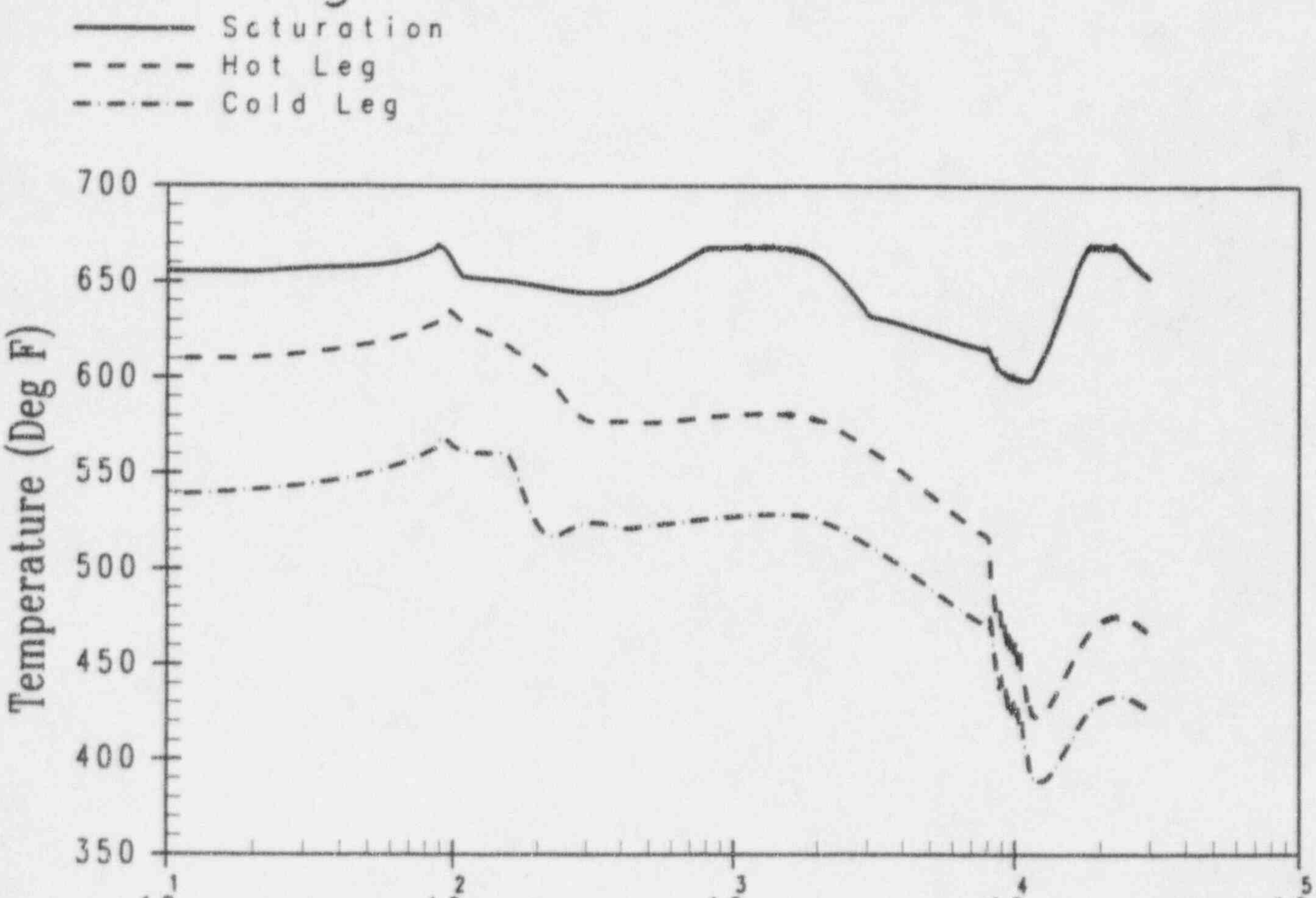
Figure 15.2.6-5



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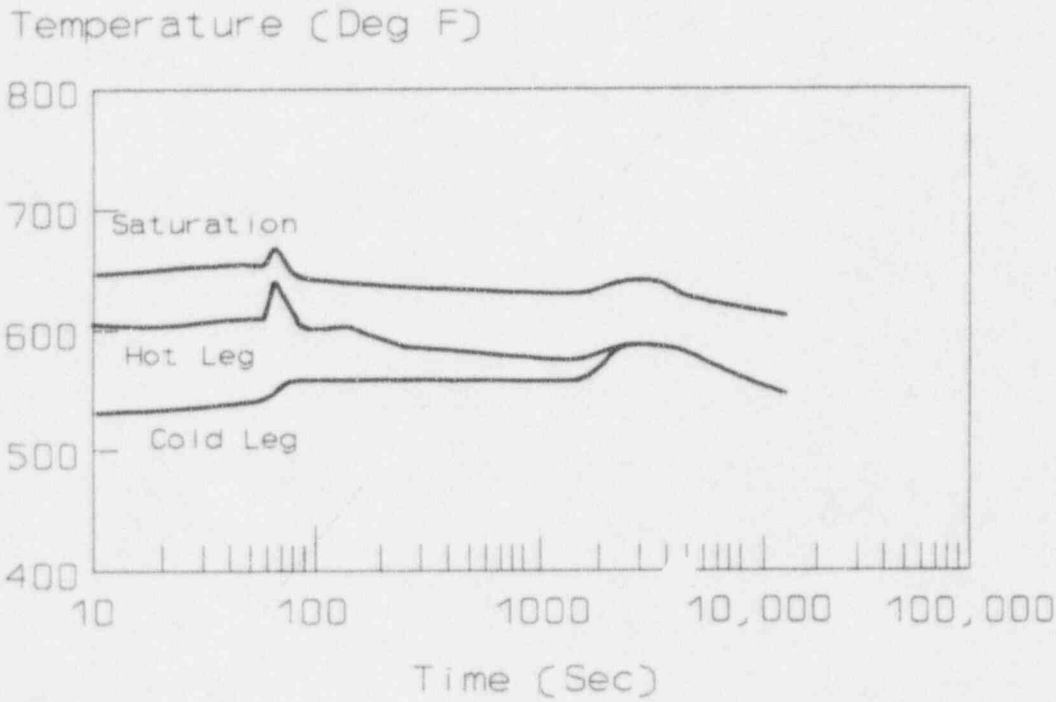
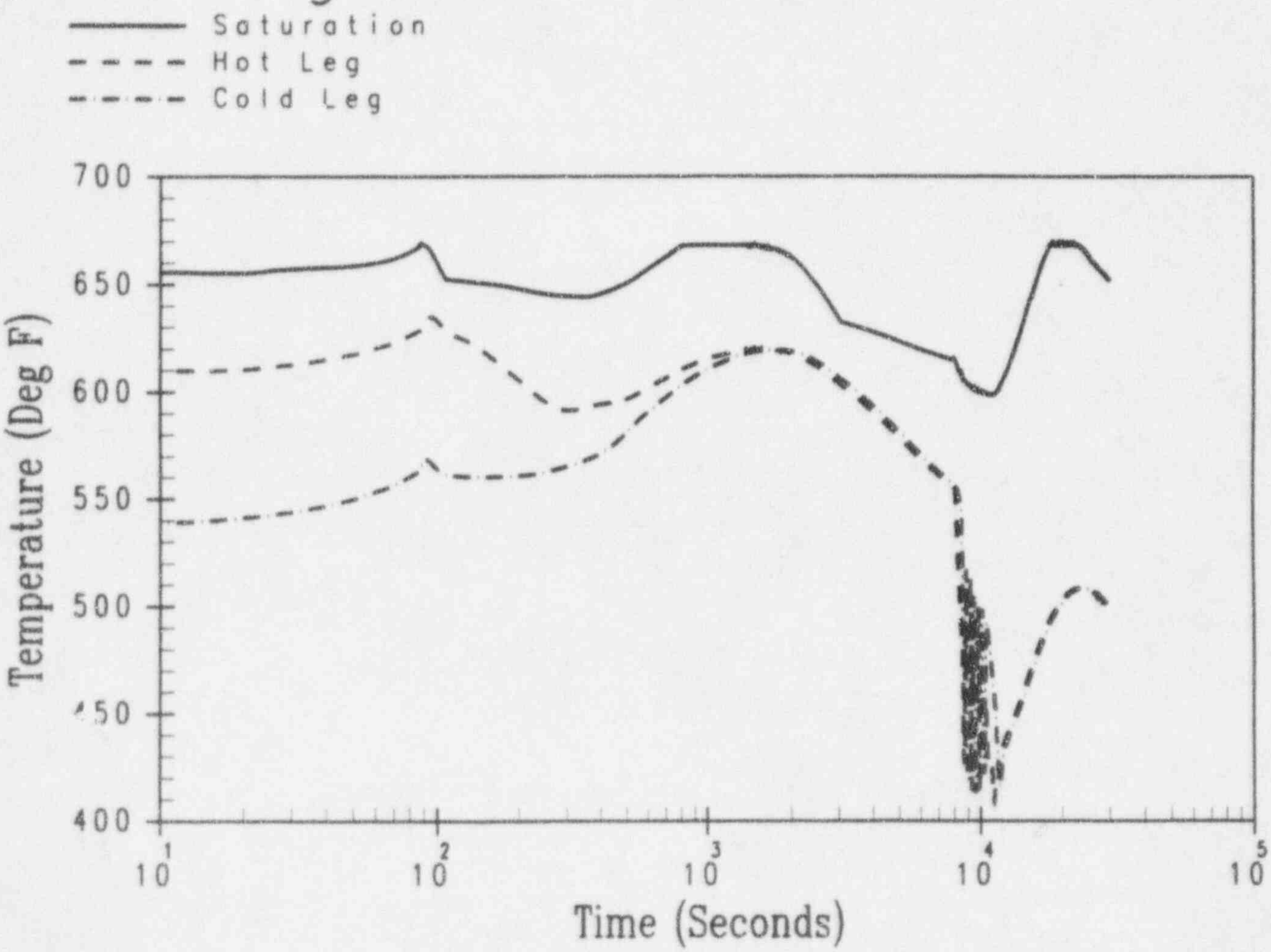


Figure 15.2.6-6

RCS Temperature Transients in Loop Not Containing the PRHR For Loss of Nonemergency AC Power to the Plant Auxiliaries

Figure 15.2.6-6



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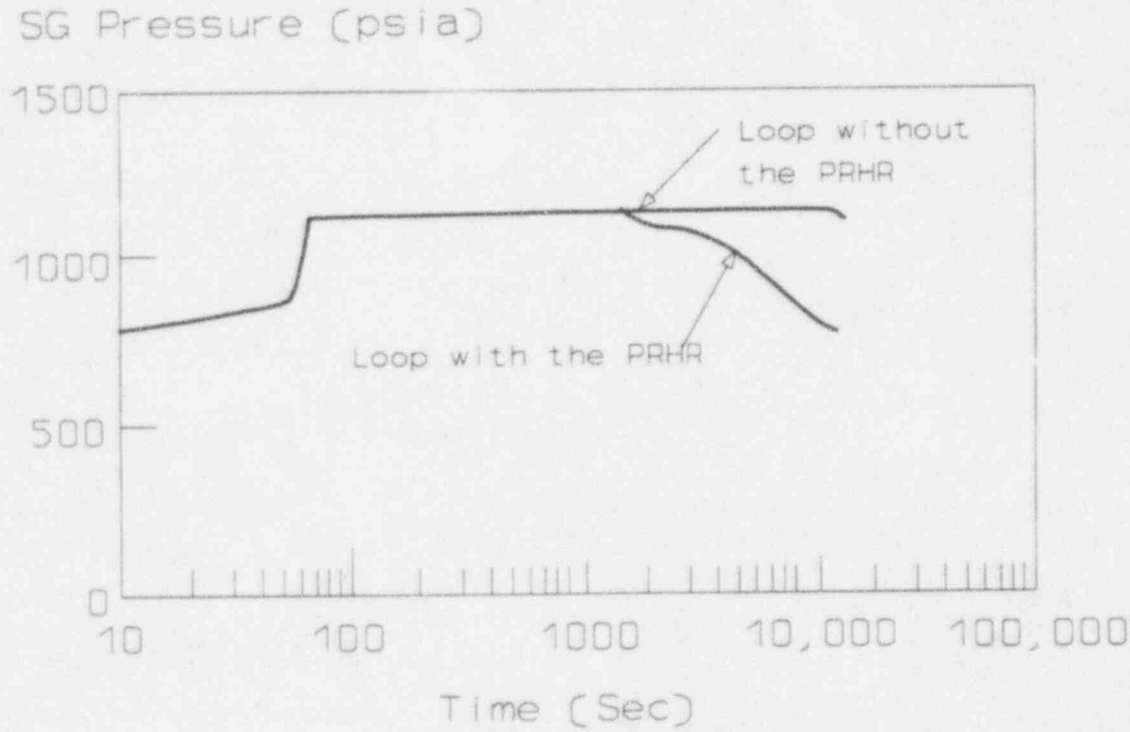
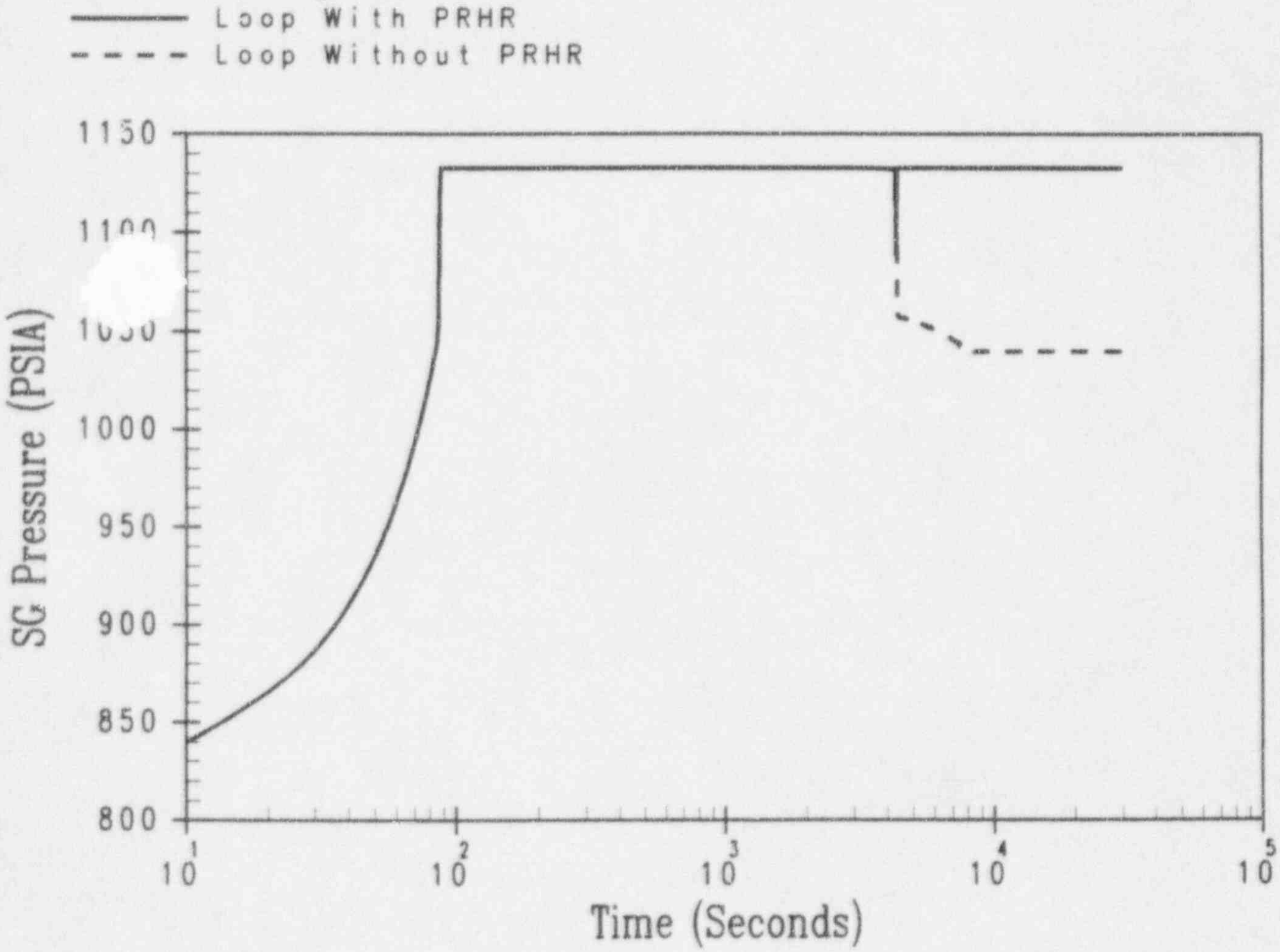


Figure 15.2.6-7

Steam Generator Pressure Transients For Loss of Nonemergency AC Power to the Plant Auxiliaries

Figure 15.2.6-7



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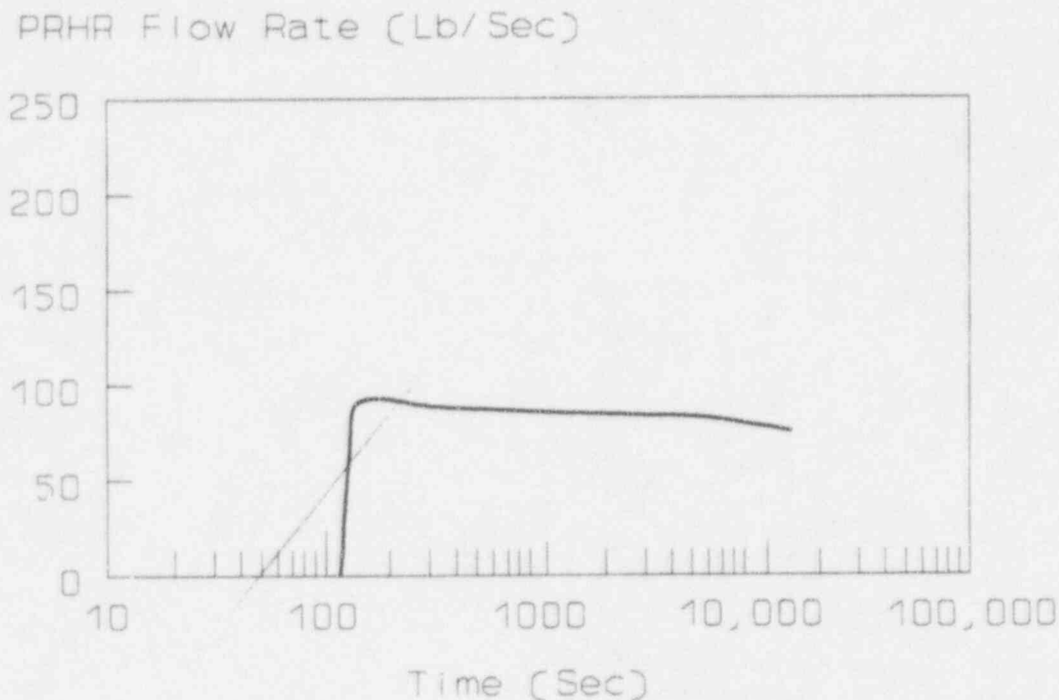
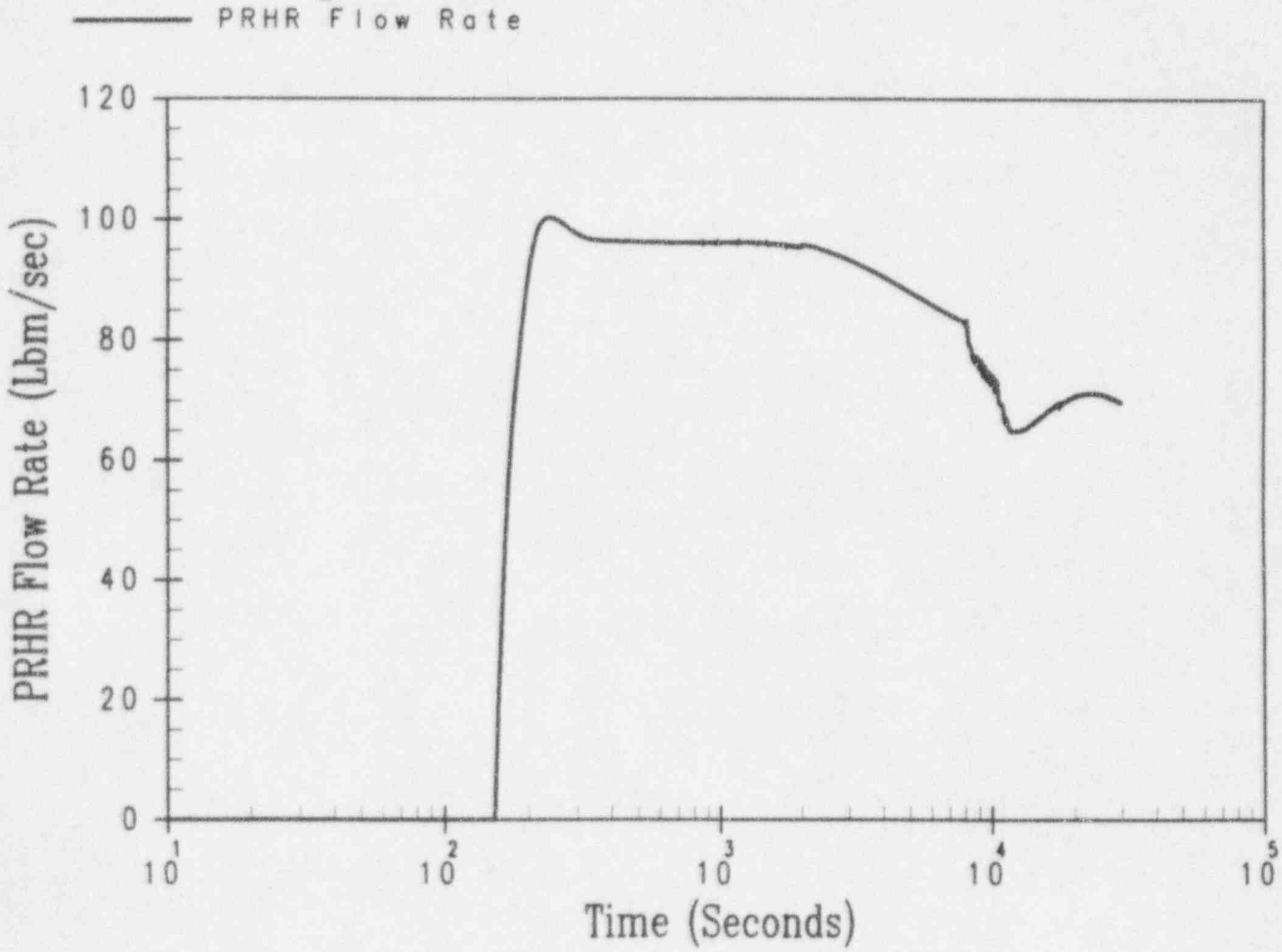


Figure 15.2.6-8

PRHR Flow Rate Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

Figure 15.2.6-8



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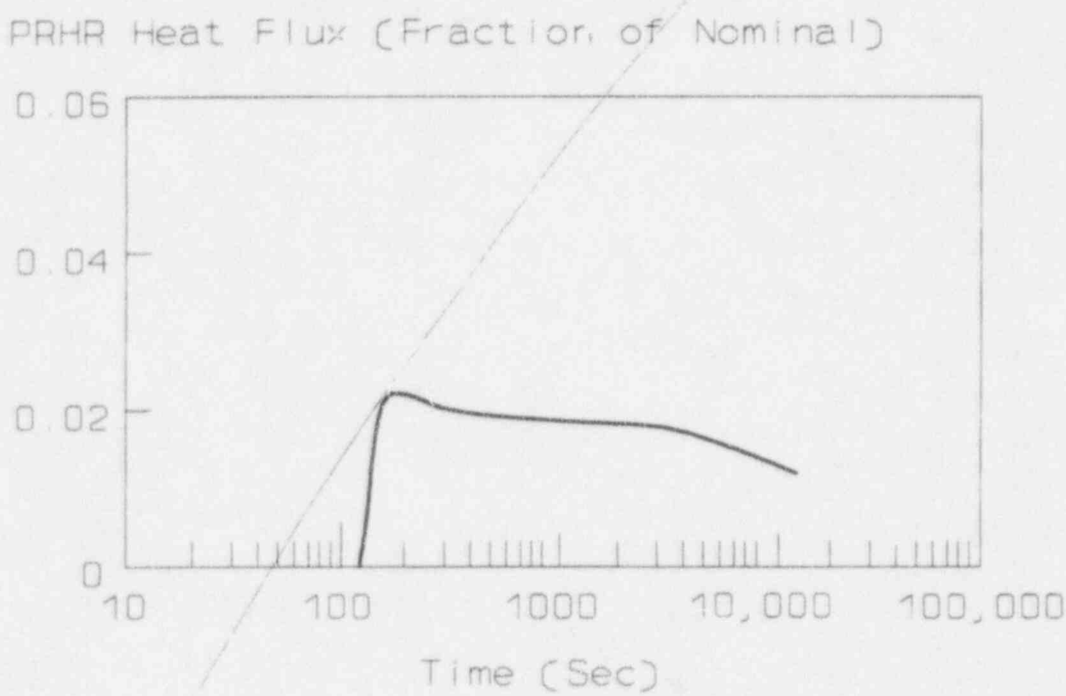
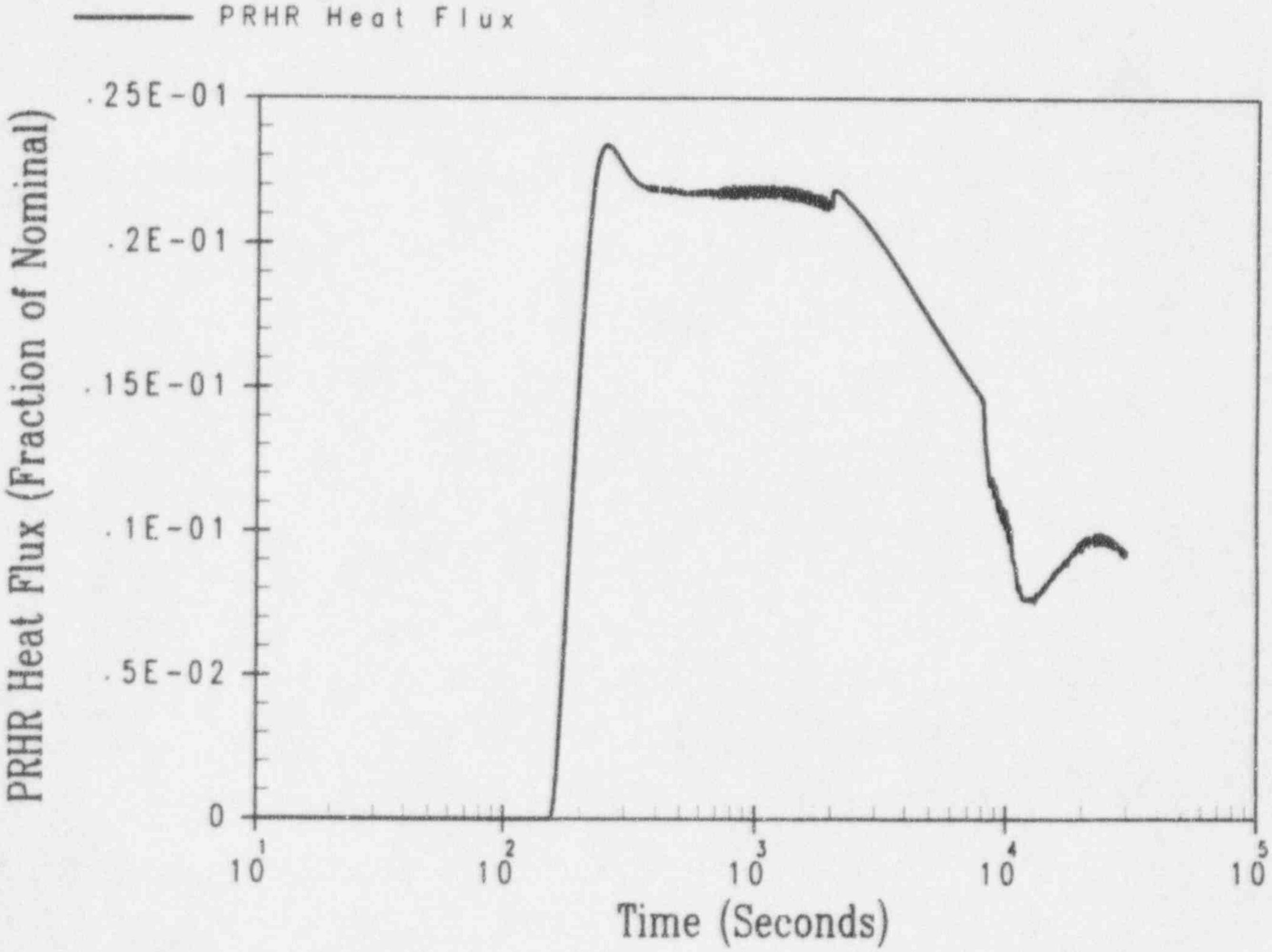


Figure 15.2.6-9

PRHR Heat Flux Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

Figure 15.2.6-9



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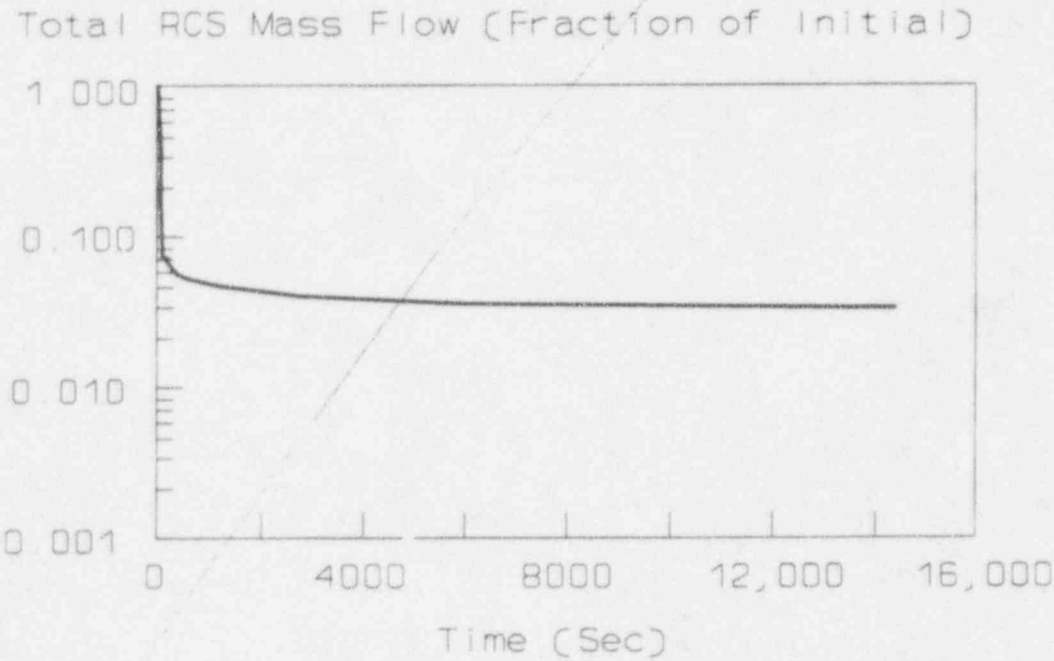
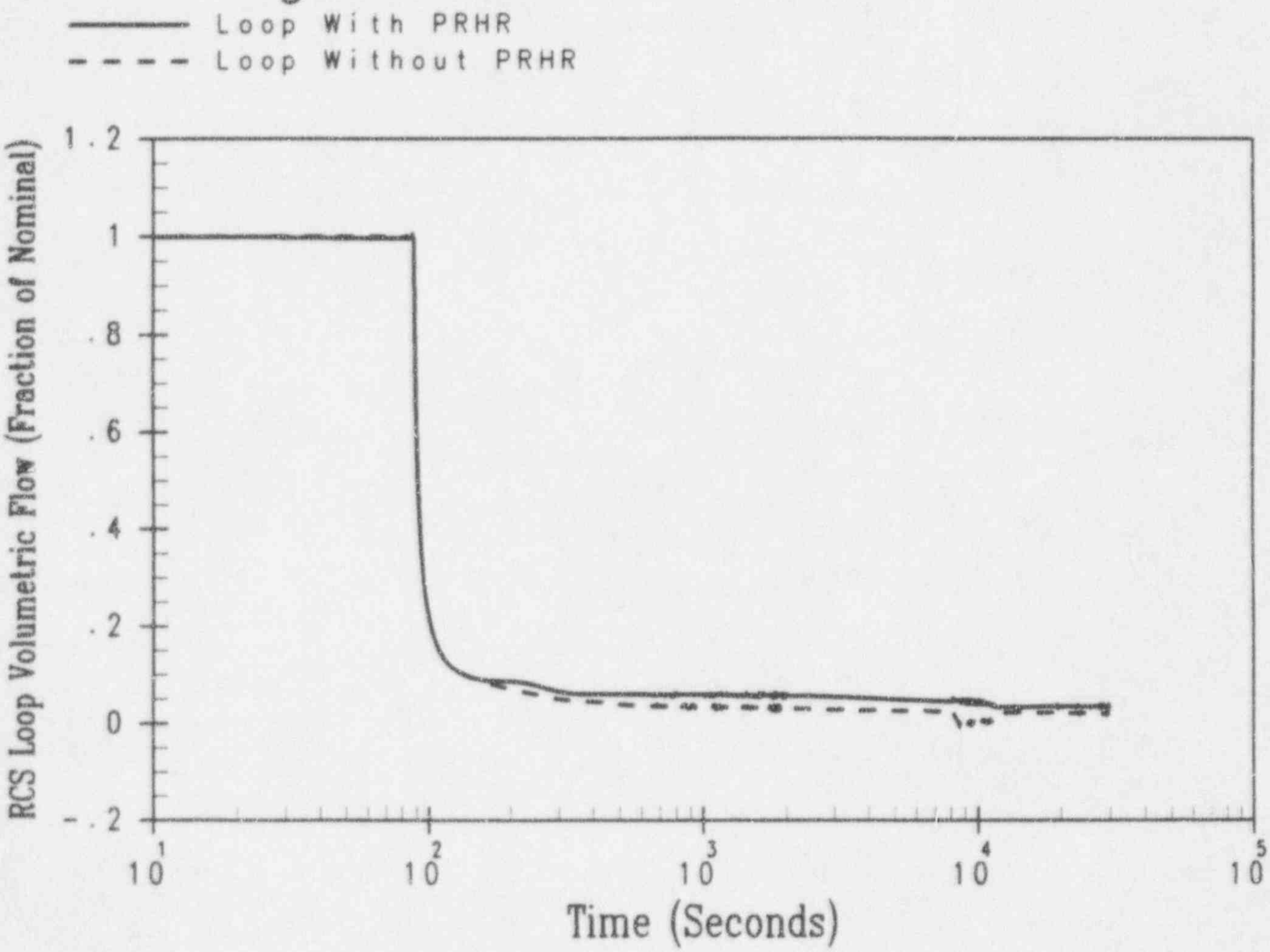


Figure 15.2.6-10

Reactor Coolant Mass Flow Rate Transient For Loss of Nonemergency AC Power to the Plant Auxiliaries

Figure 15.2.6-10



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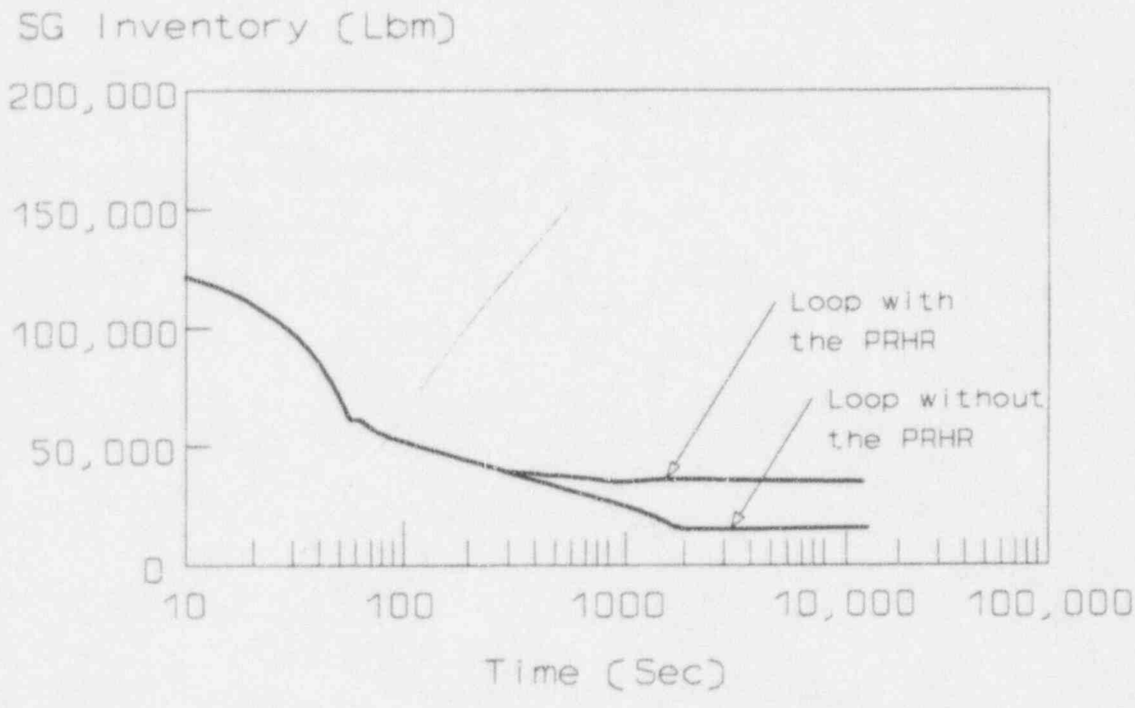
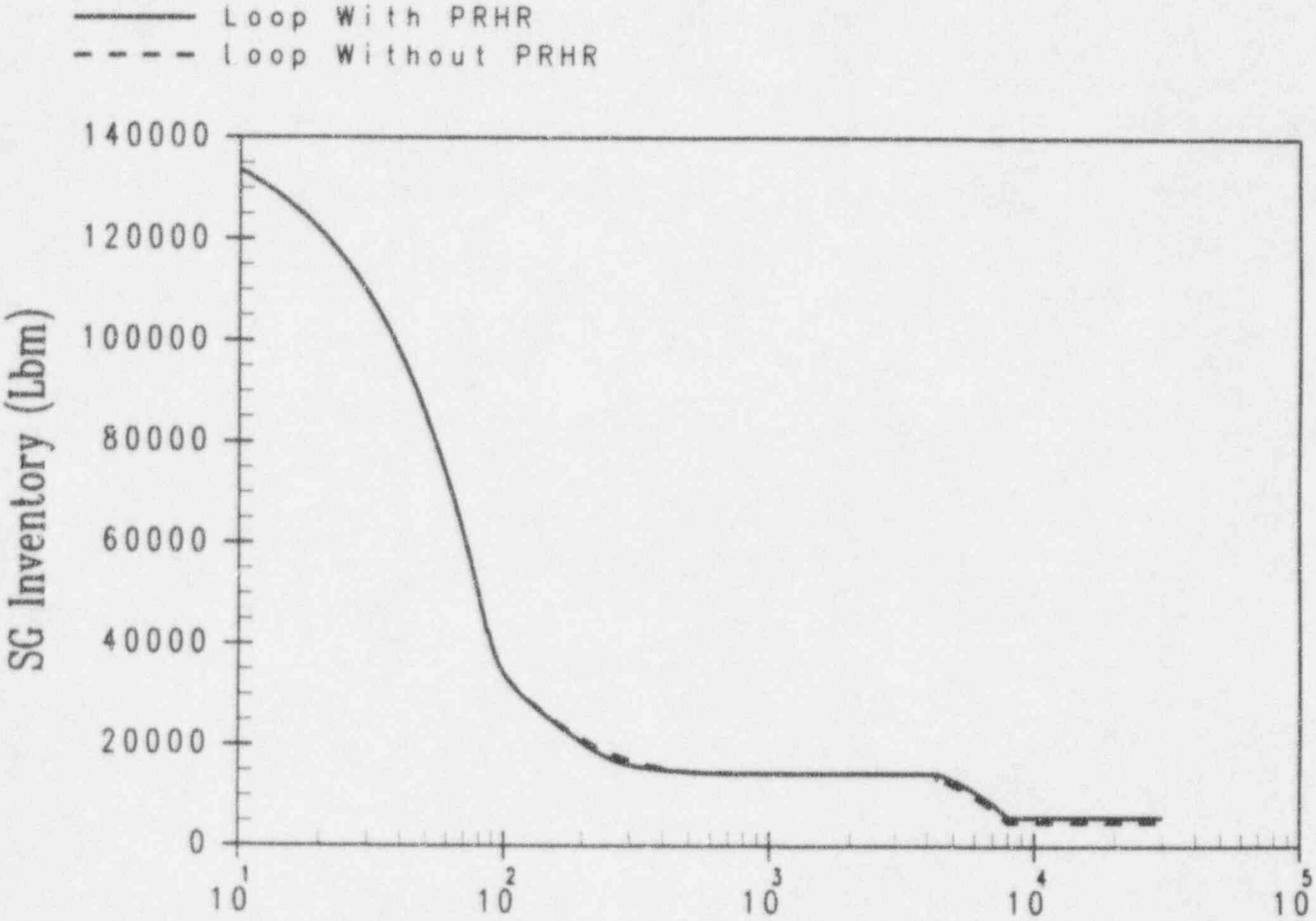


Figure 15.2.6-11

Steam Generator Inventory Transients

Figure 15.2.6-11



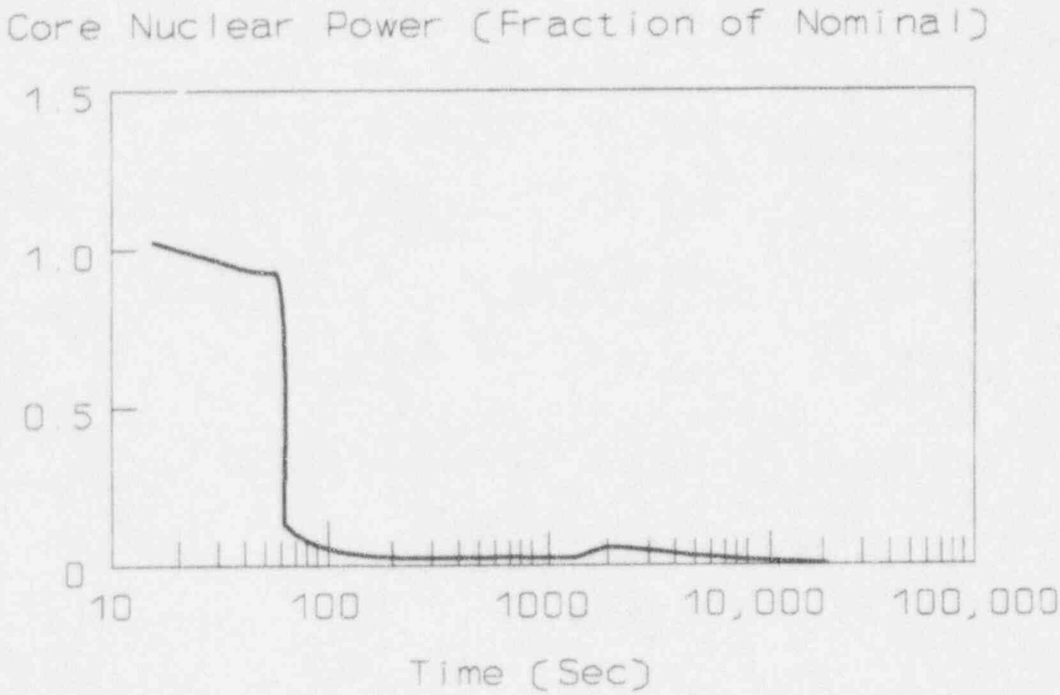
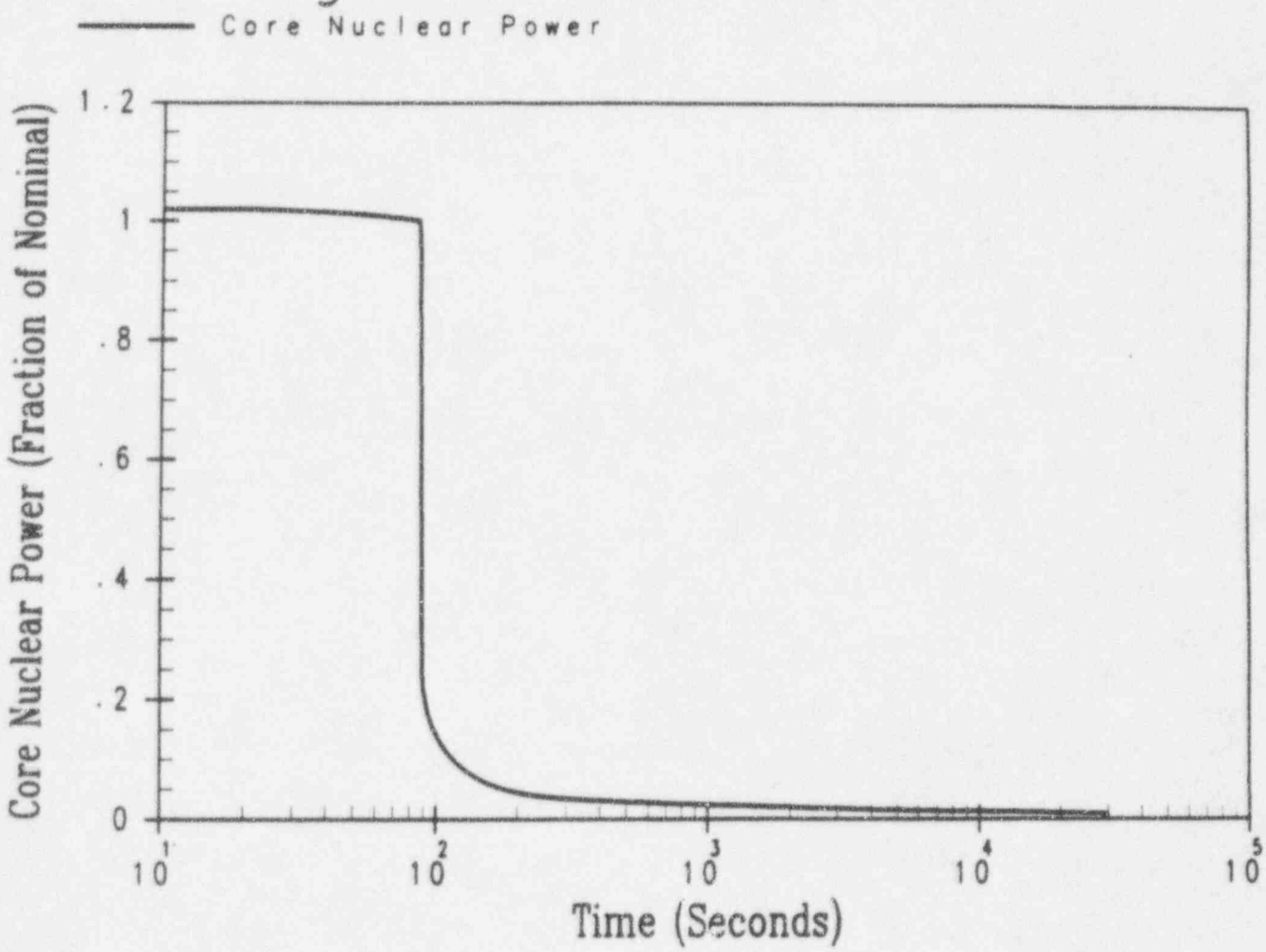


Figure 15.2.7-1

Nuclear Power Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-1



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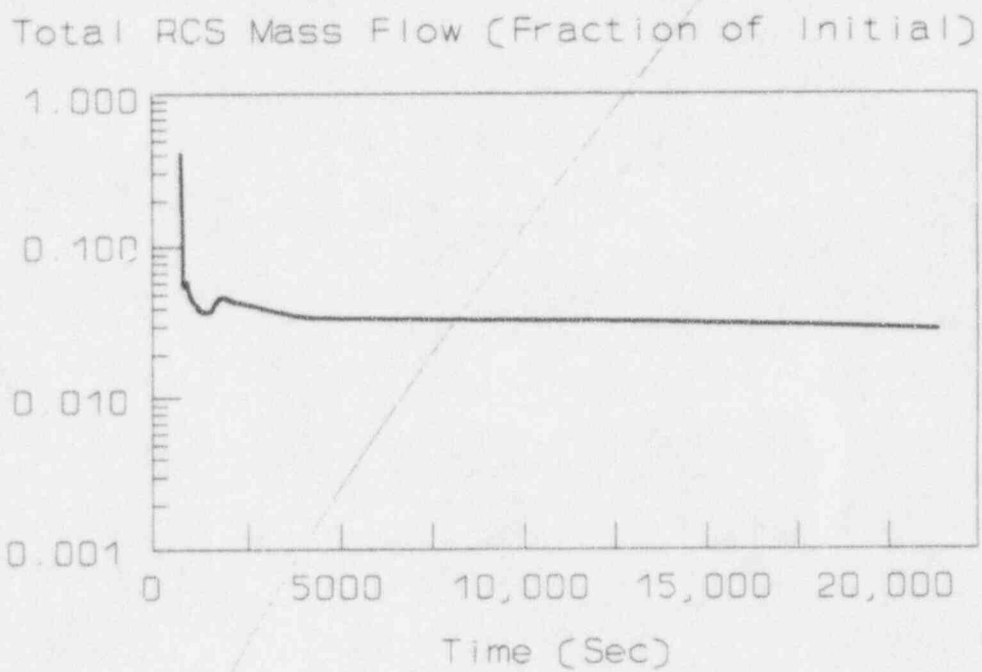
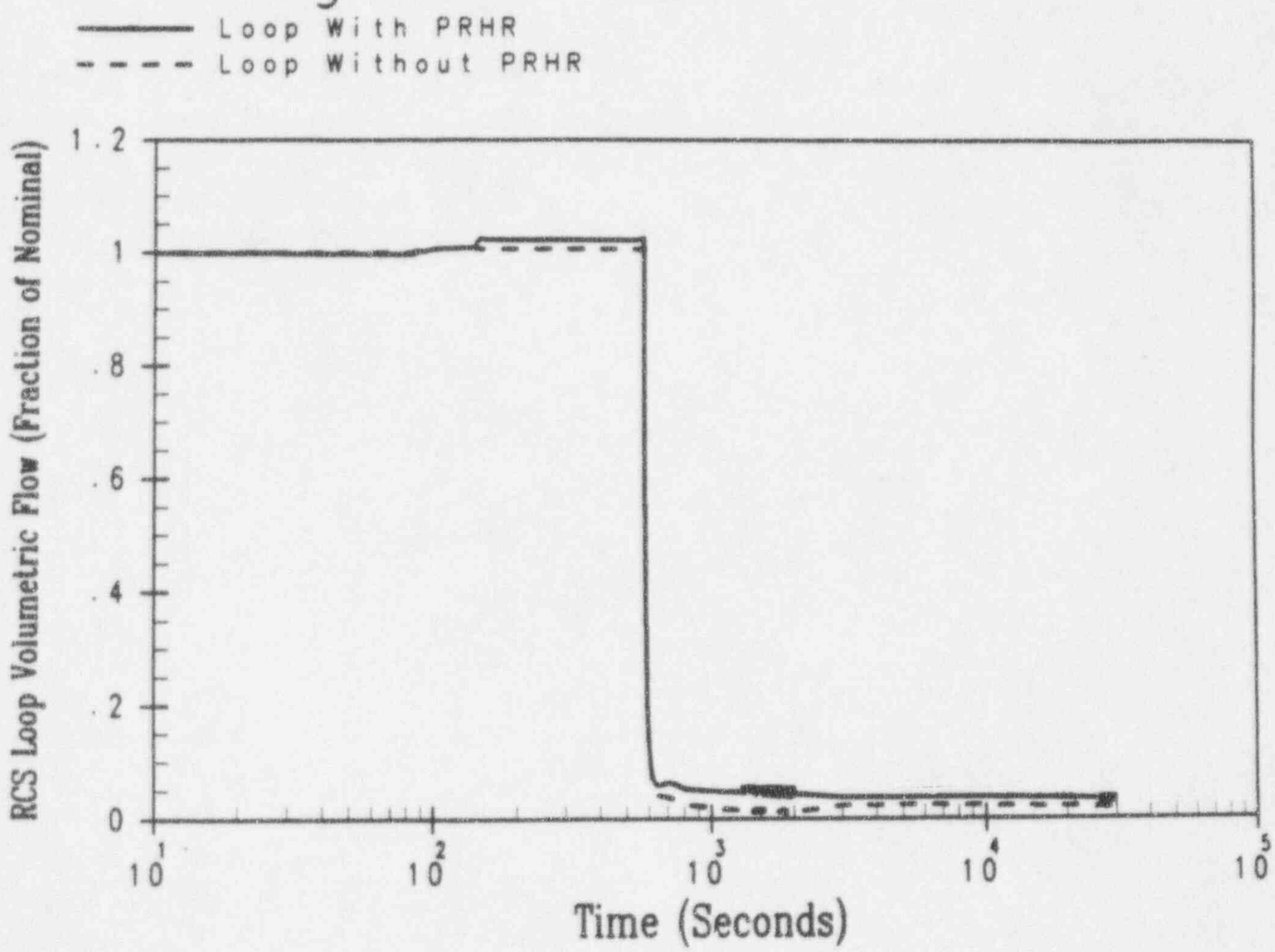


Figure 15.2.7-2

RCS Mass Flow Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-2



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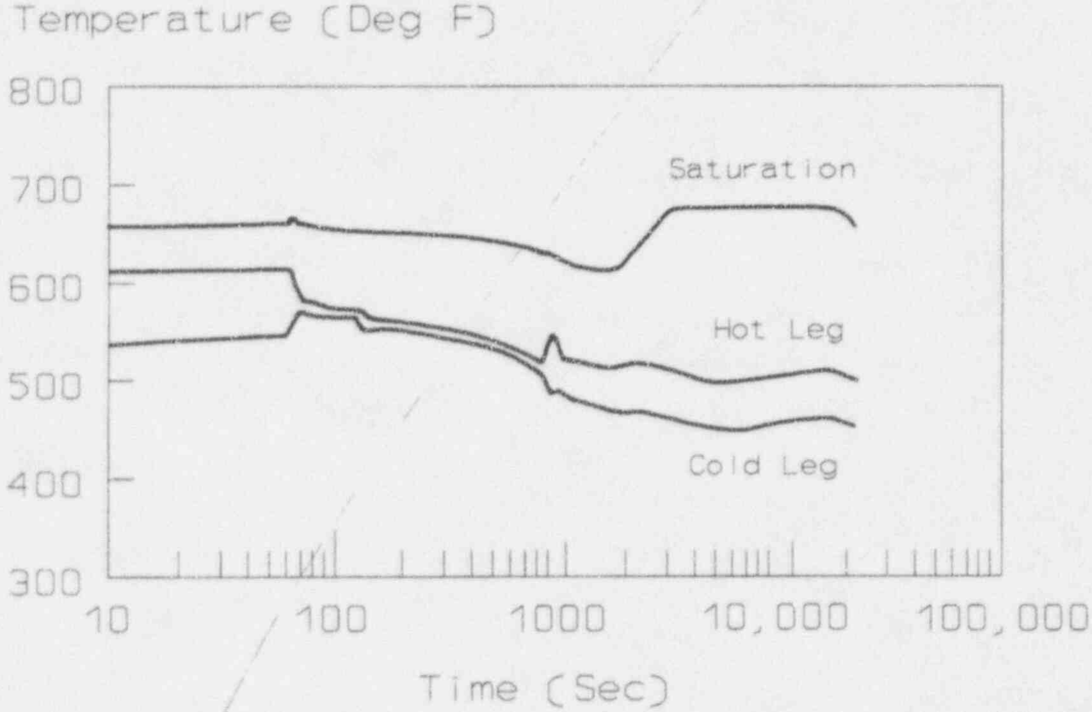
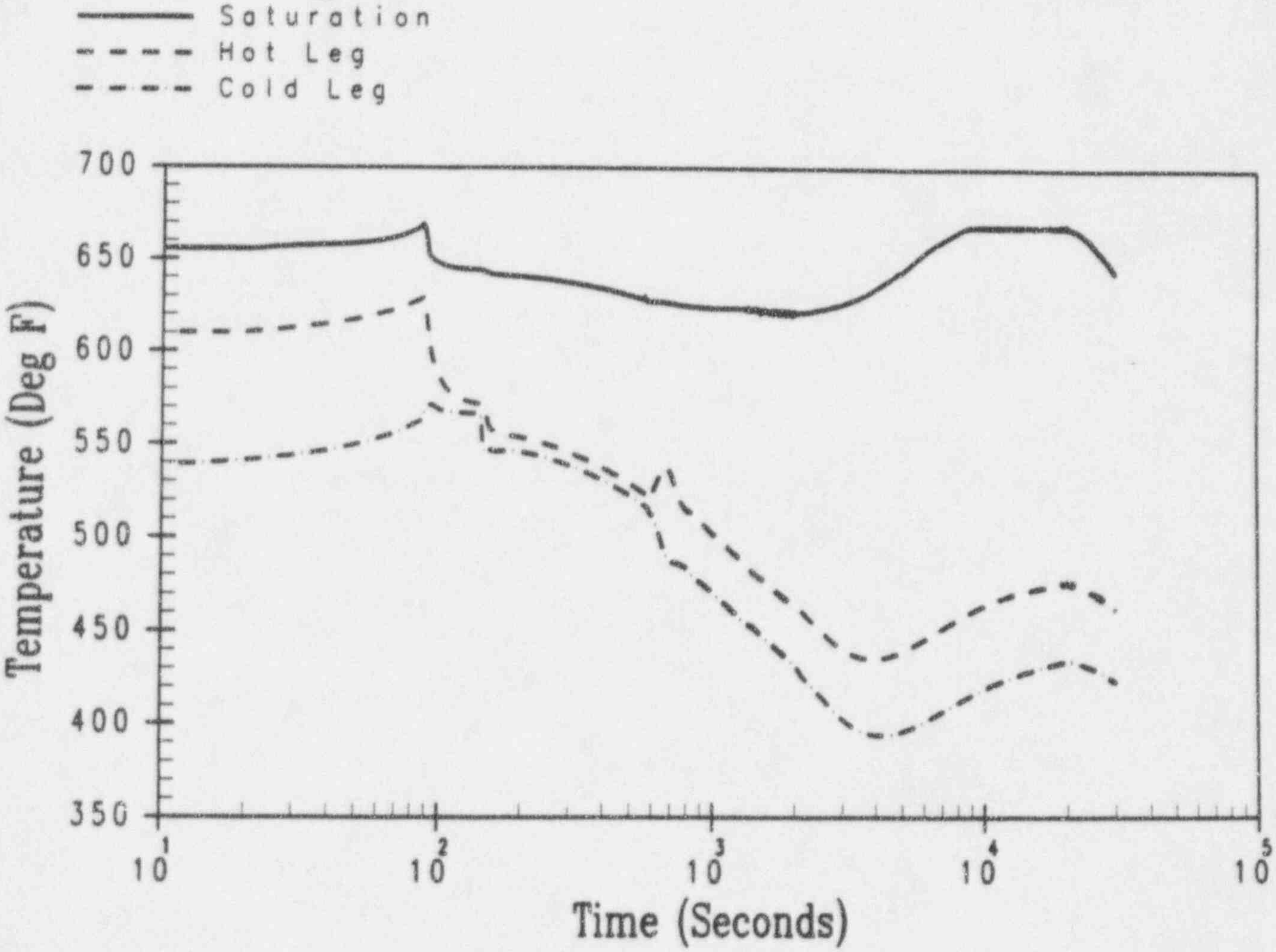


Figure 15.2.7-3

RCS Temperature Transients in Loop Containing the PRHR For Loss of Normal Feedwater Flow

Figure 15.2.7-3



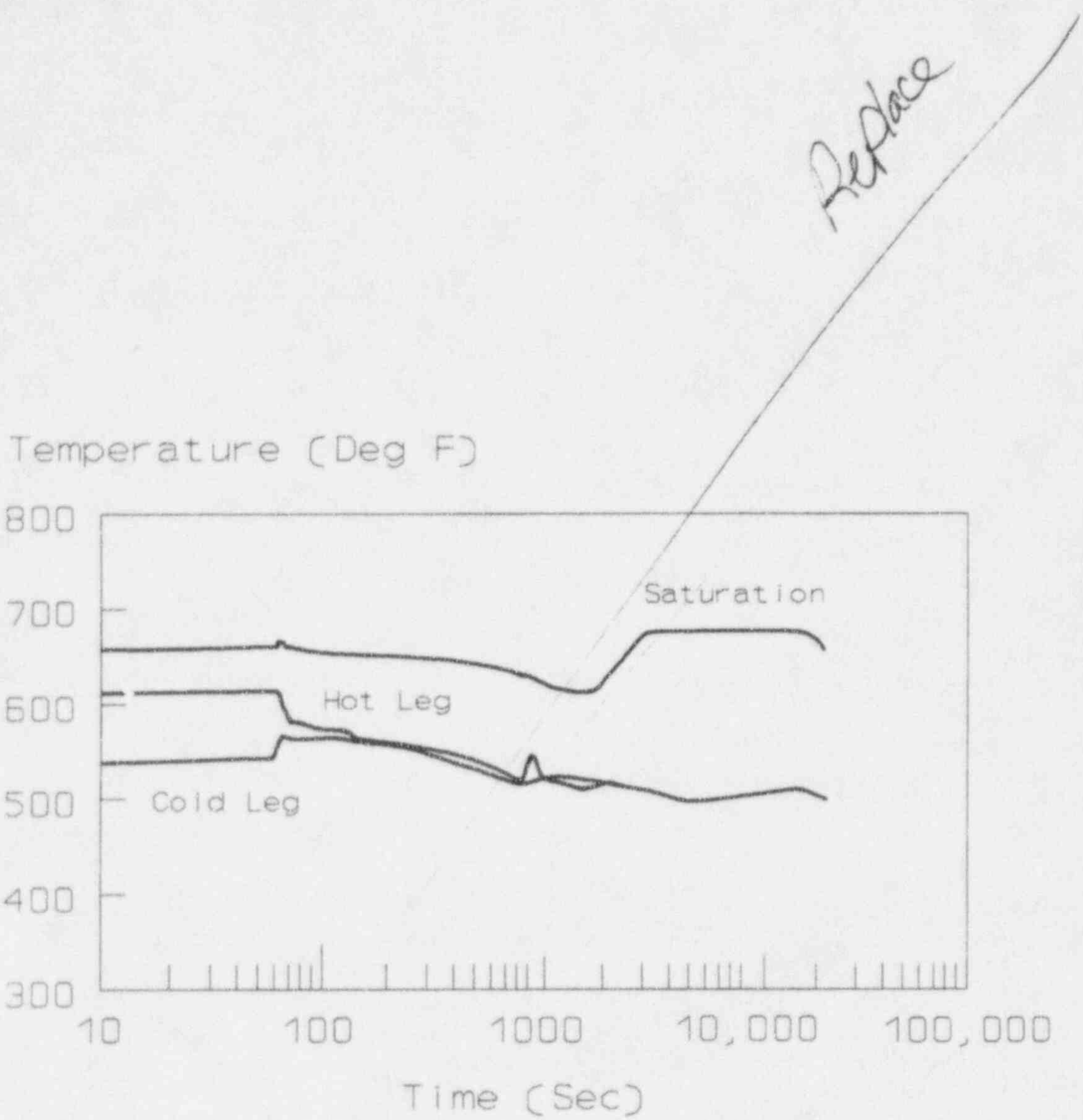
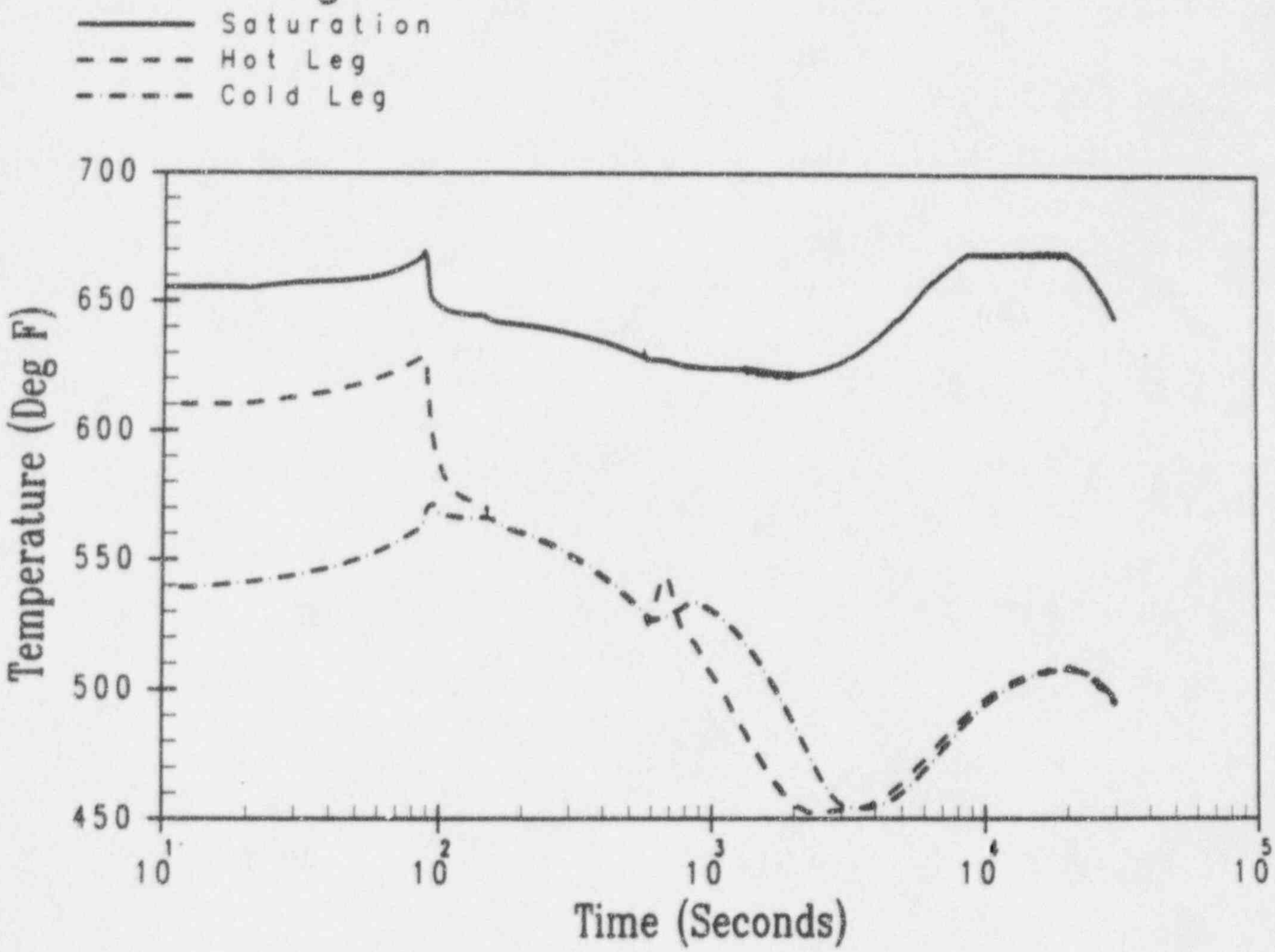


Figure 15.2.7-4

RCS Temperature Transients in Loop Not Containing the PRHR For Loss of Normal Feedwater Flow

Figure 15.2.7-4



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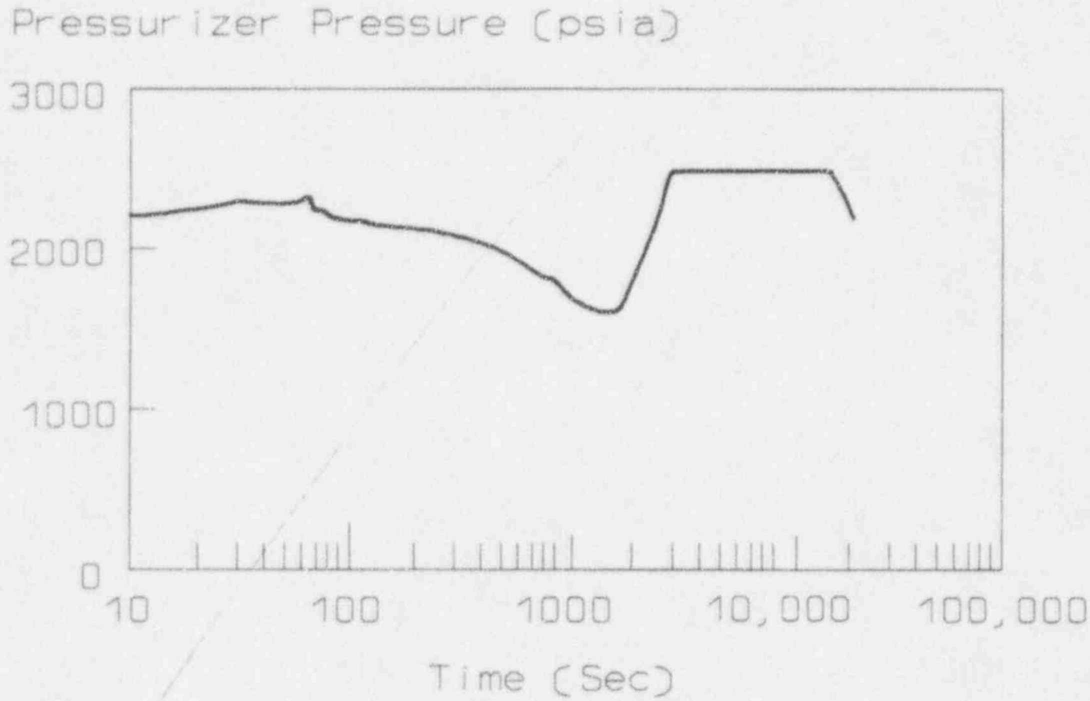
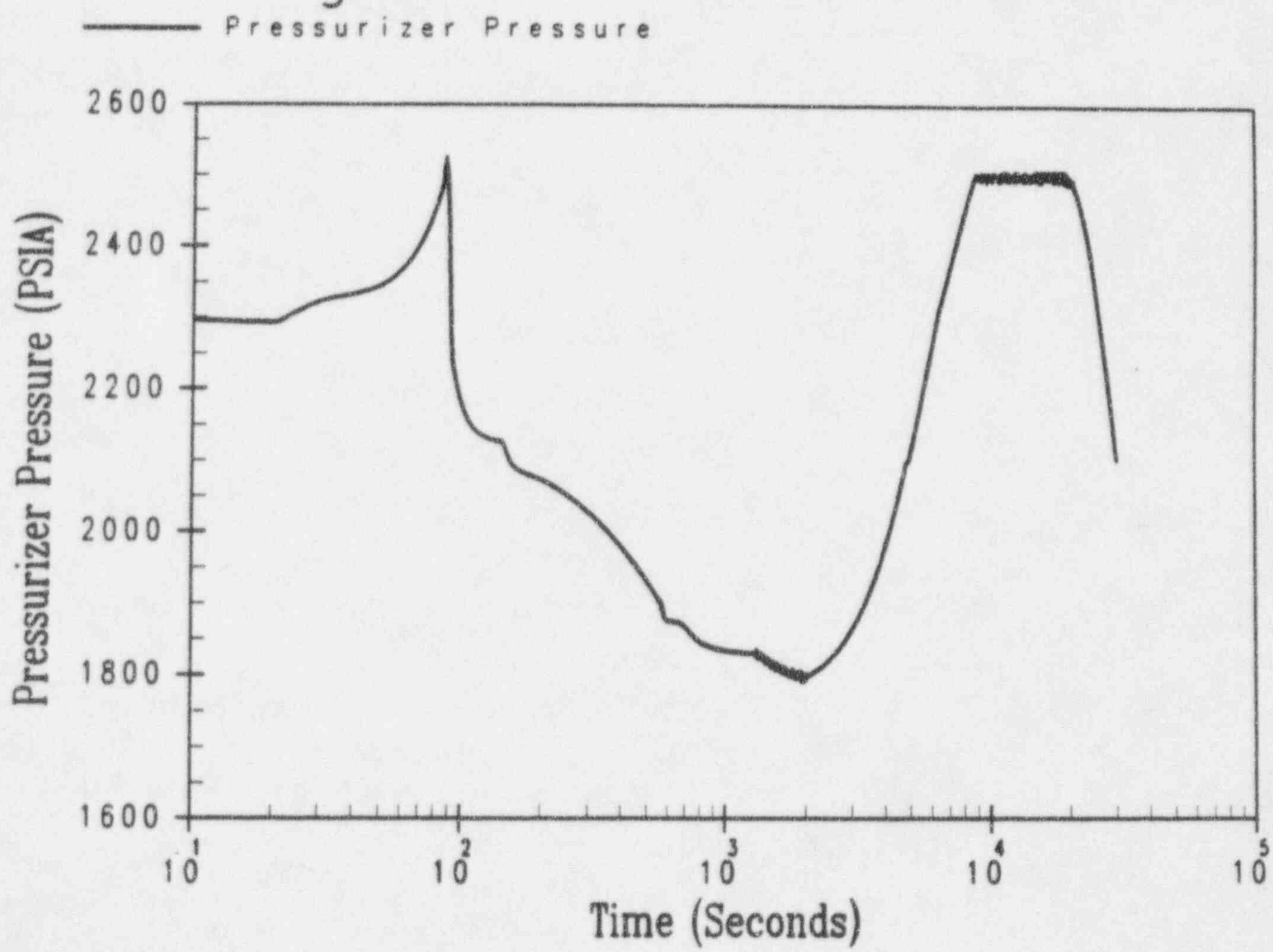


Figure 15.2.7-5

Pressurizer Pressure Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-5



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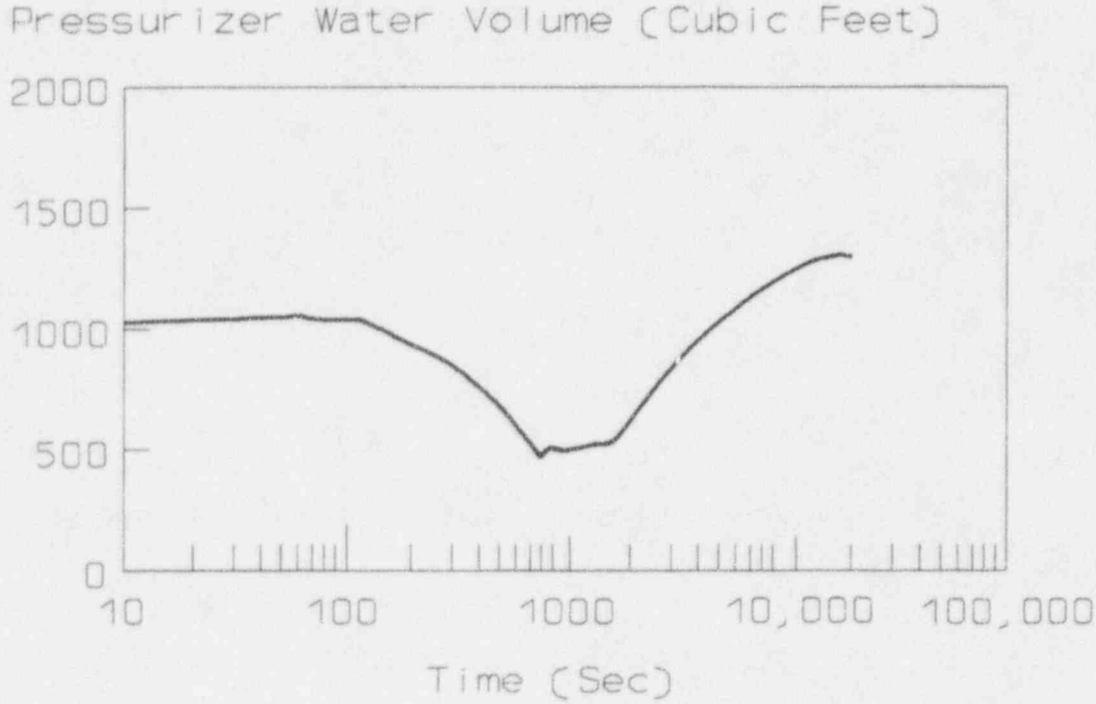
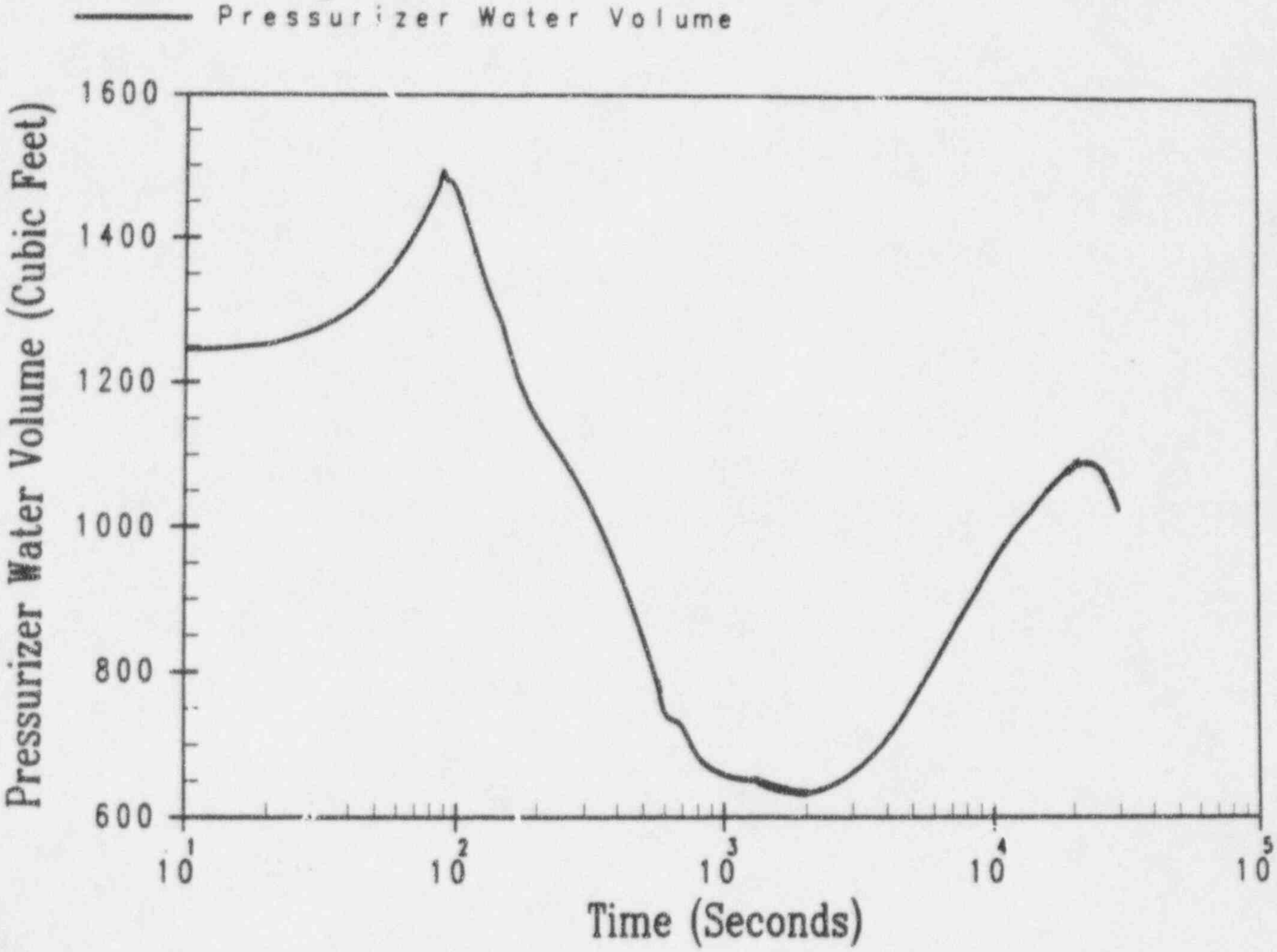


Figure 15.2.7-6

Pressurizer Water Volume Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-6



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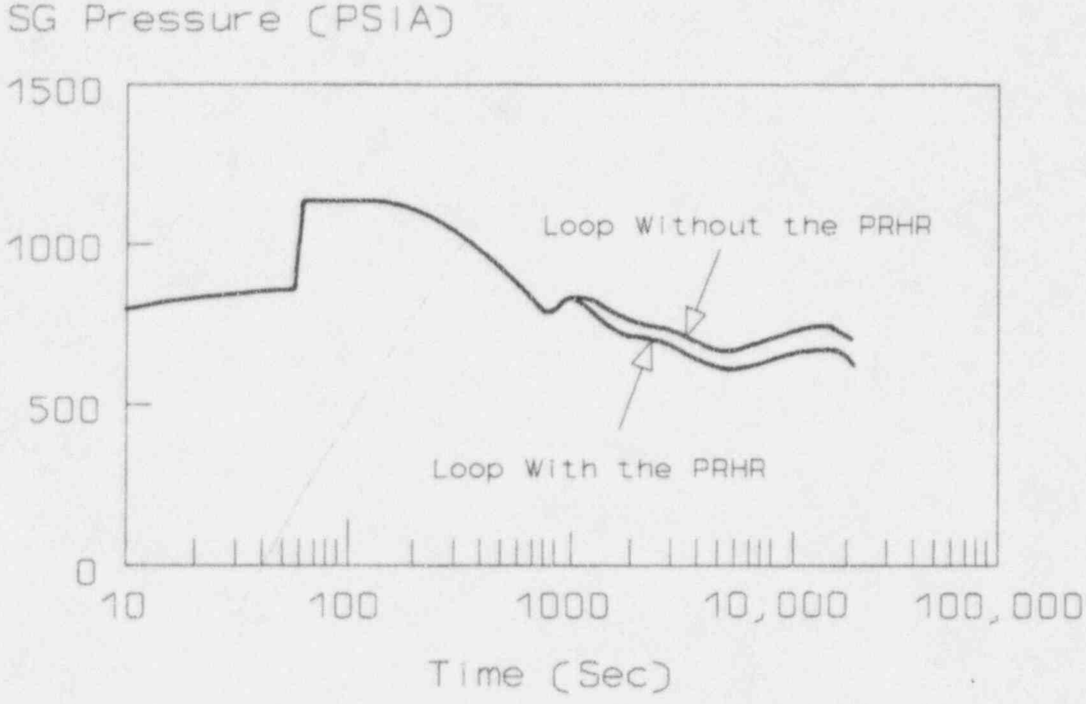
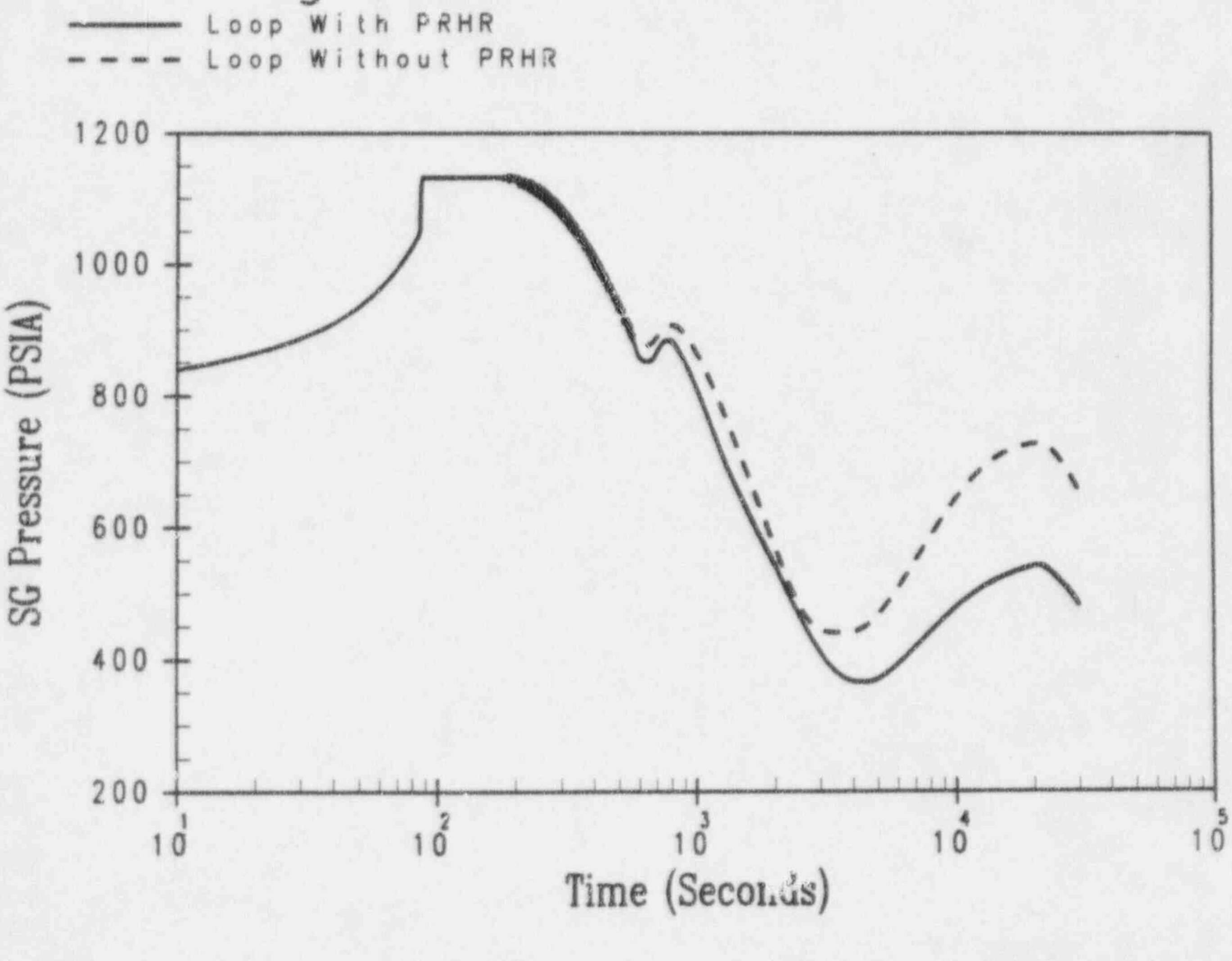


Figure 15.2.7-7

Steam Generator Pressure Transients For Loss of Normal Feedwater Flow

Figure 15.2.7-7



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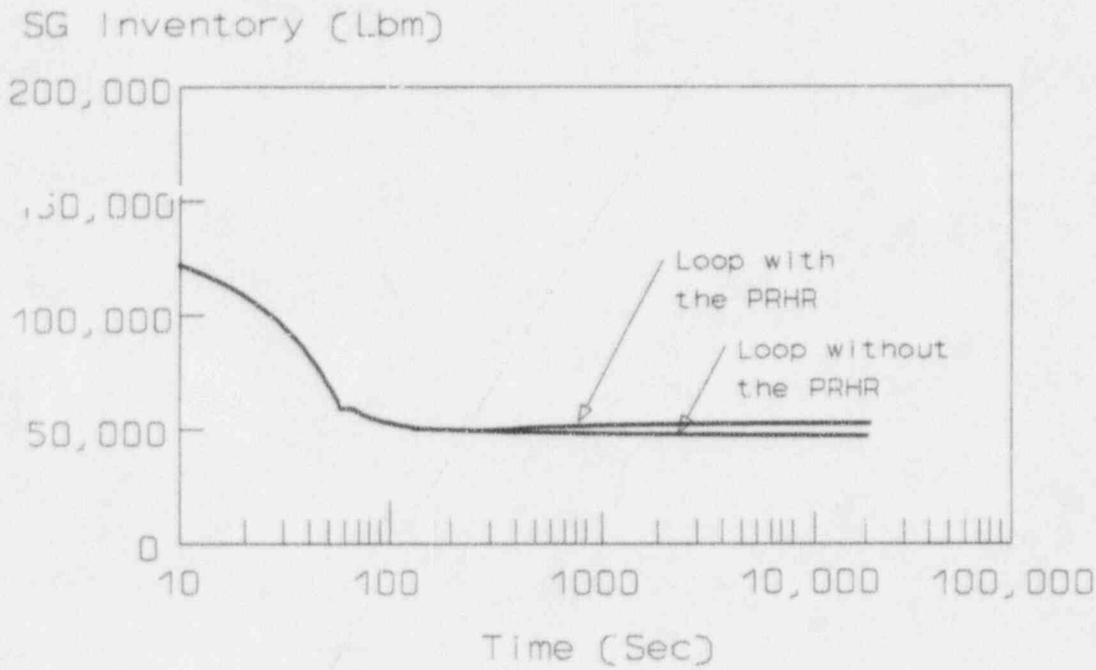
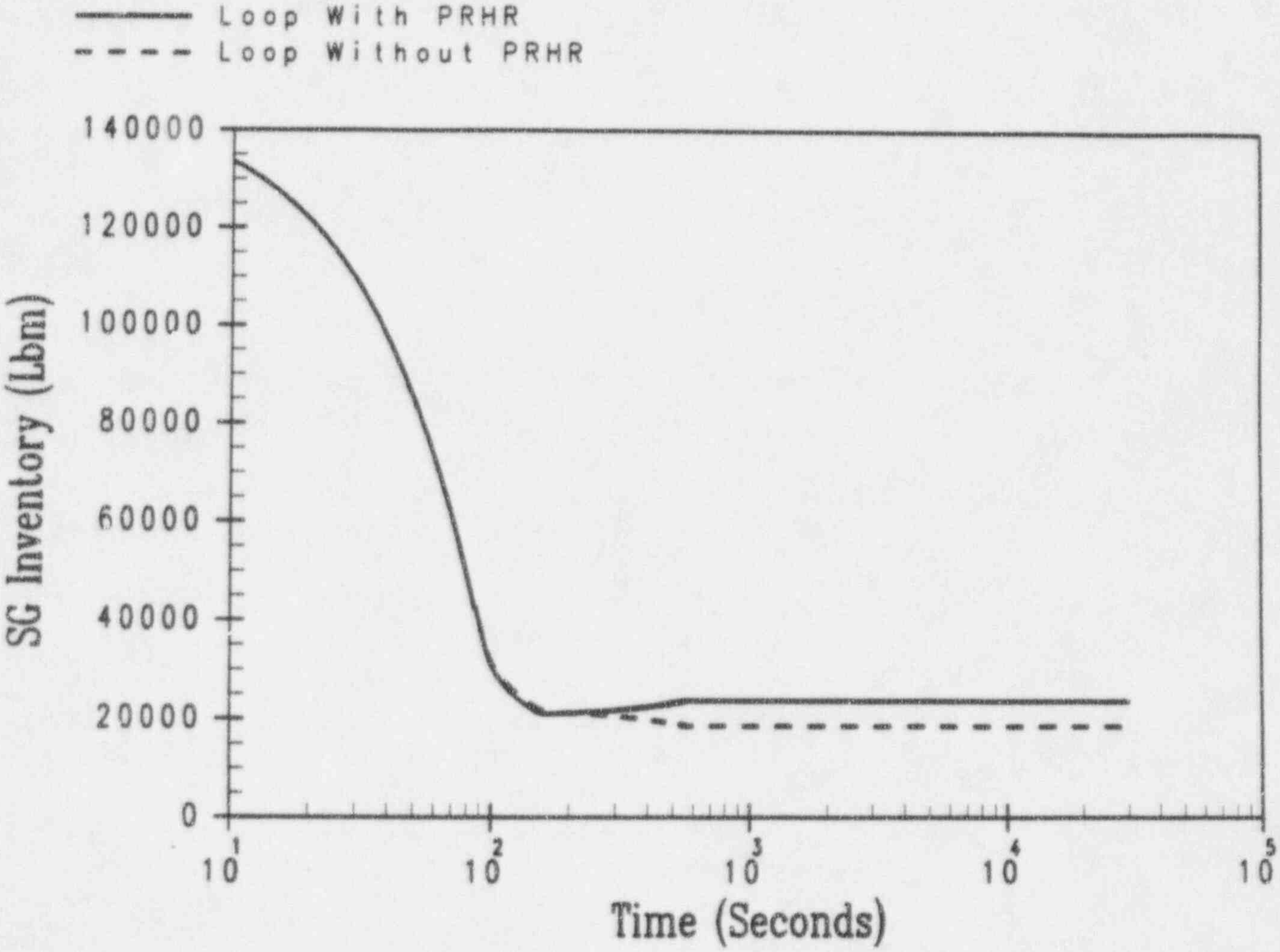


Figure 15.2.7-8

Steam Generator Inventory Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-8



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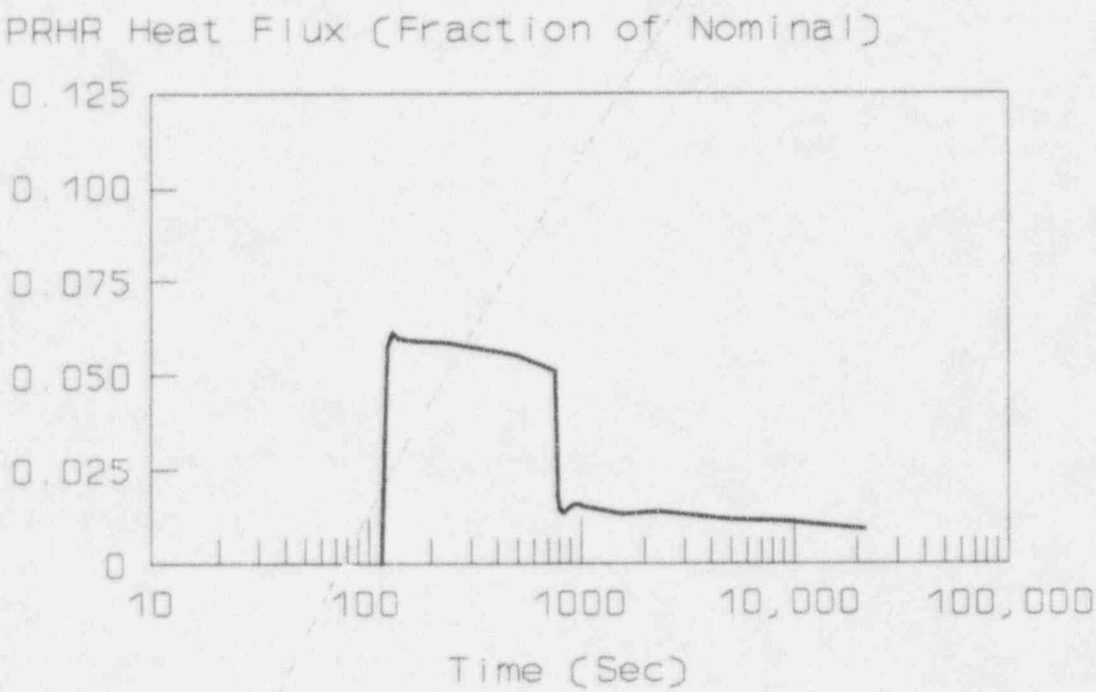
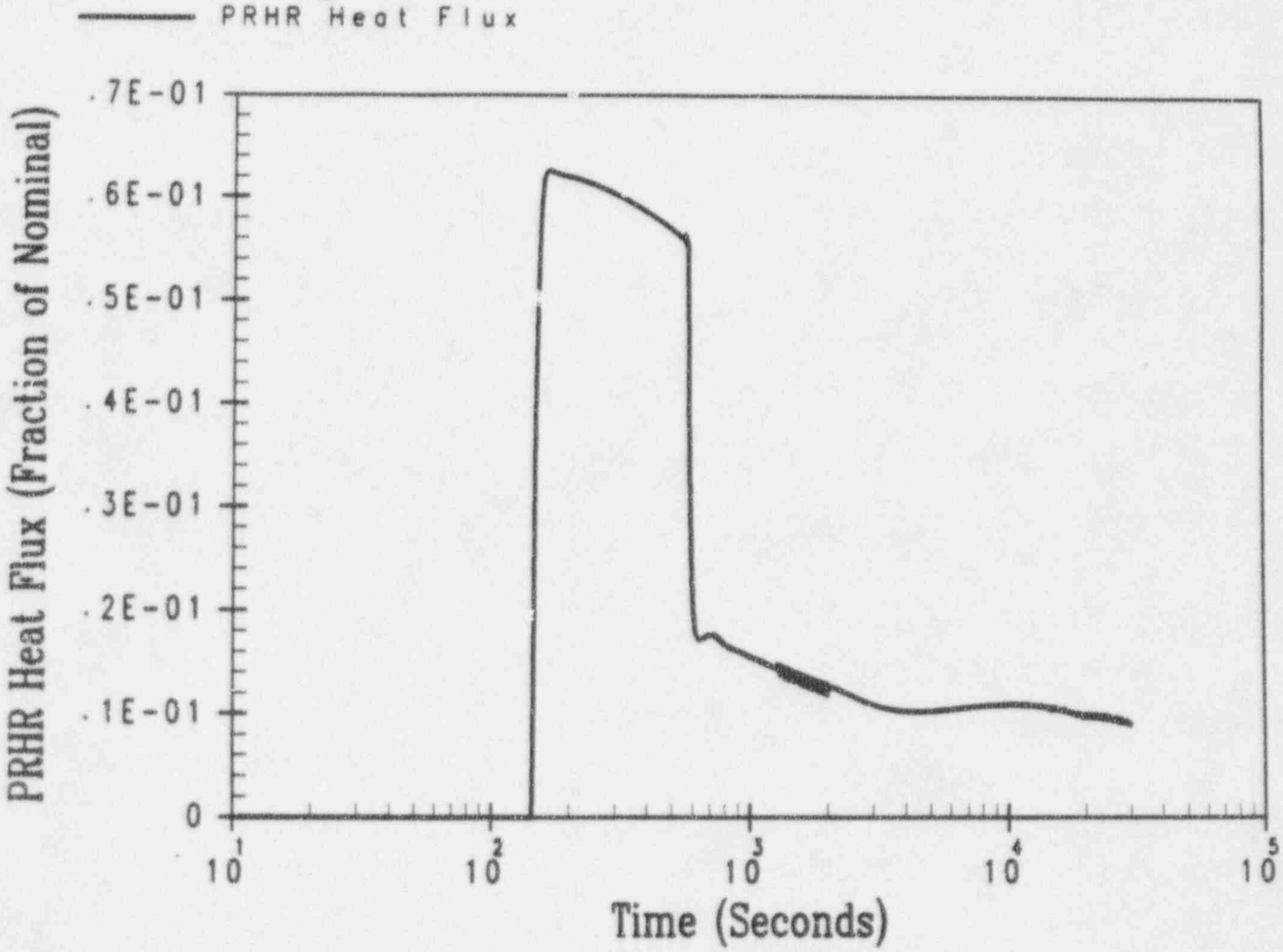


Figure 15.2.7-9

PRHR Heat Flux Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-9



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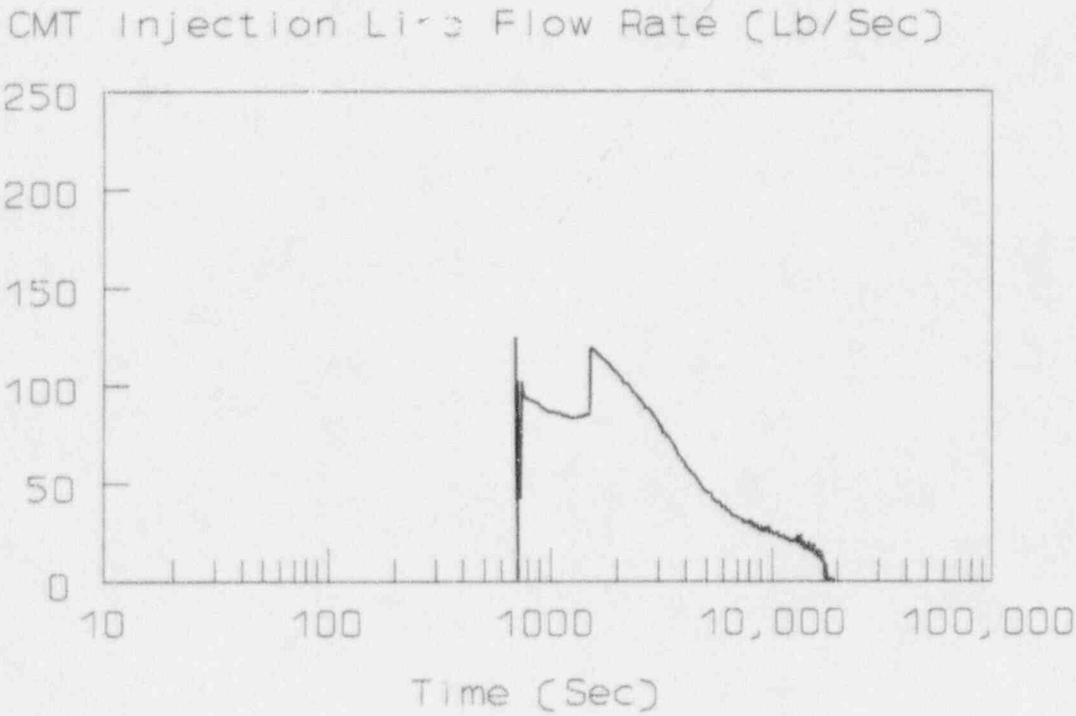
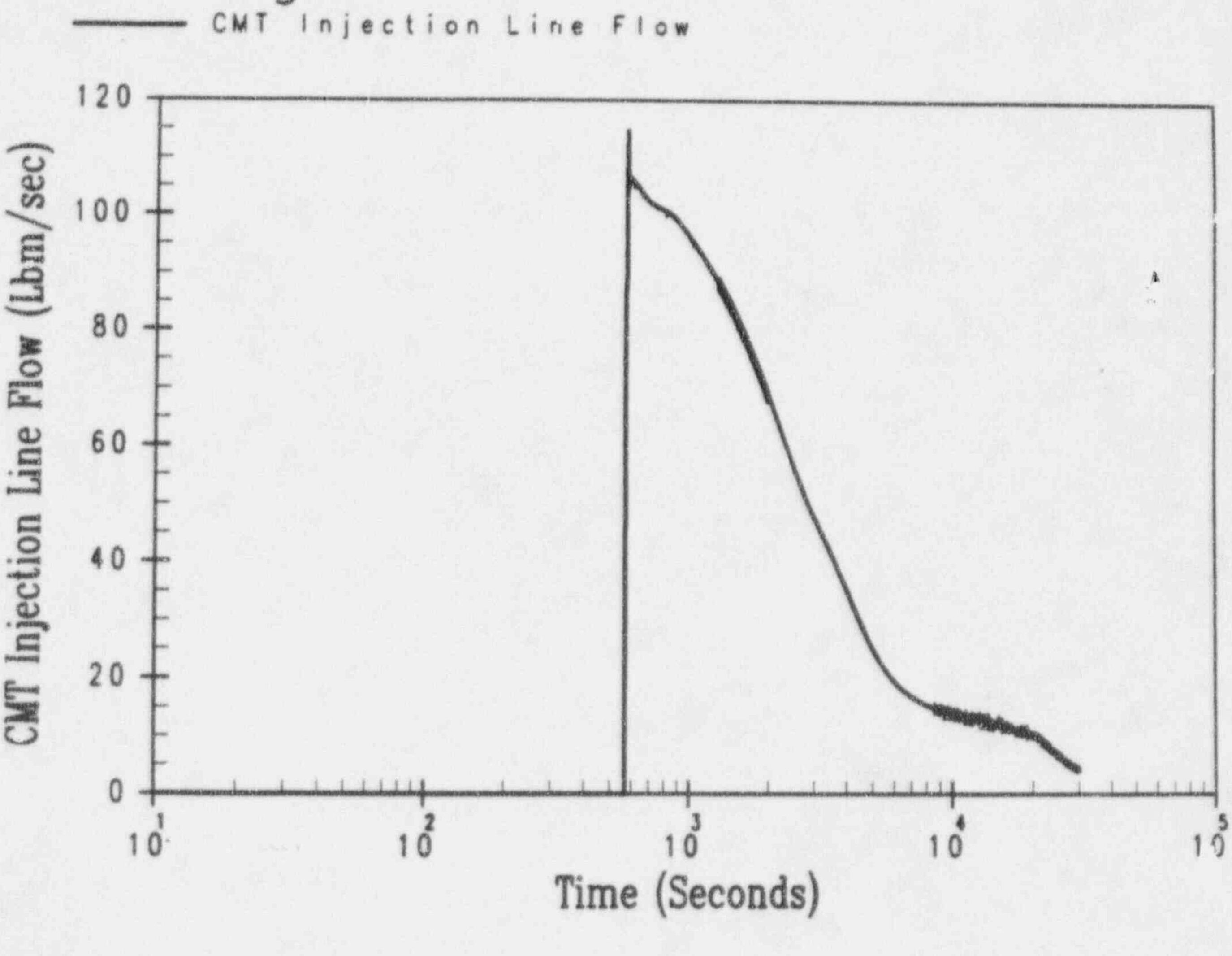


Figure 15.2.7-10

CMT Injection Flow Transient For Loss of Normal Feedwater Flow

Figure 15.2.7-10



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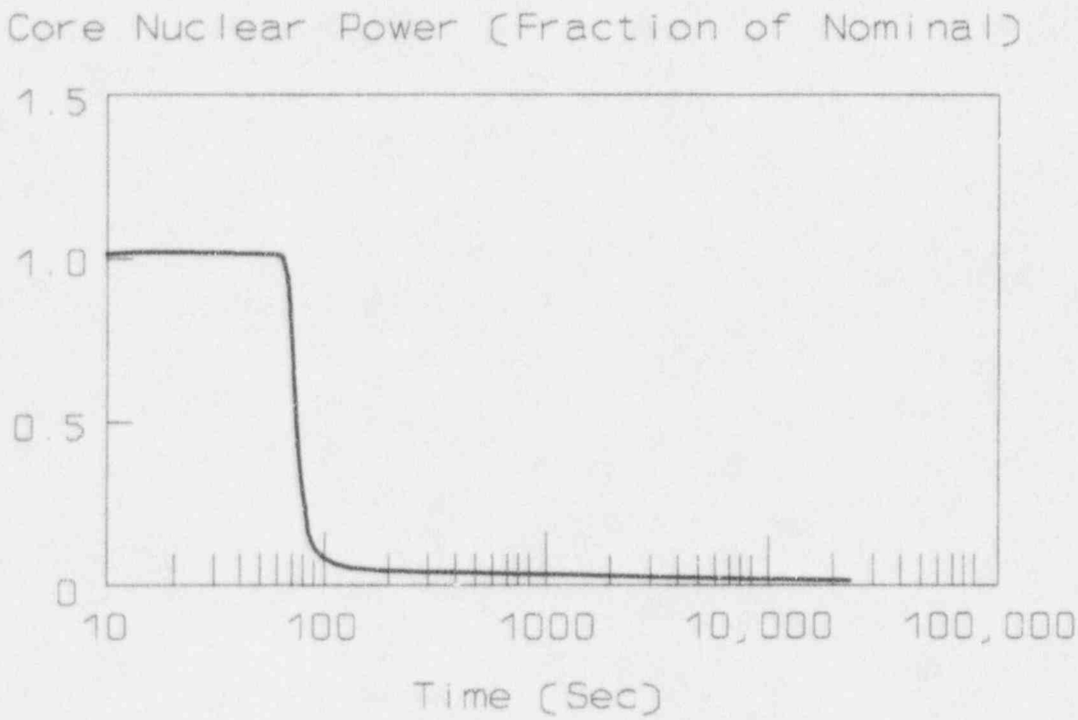
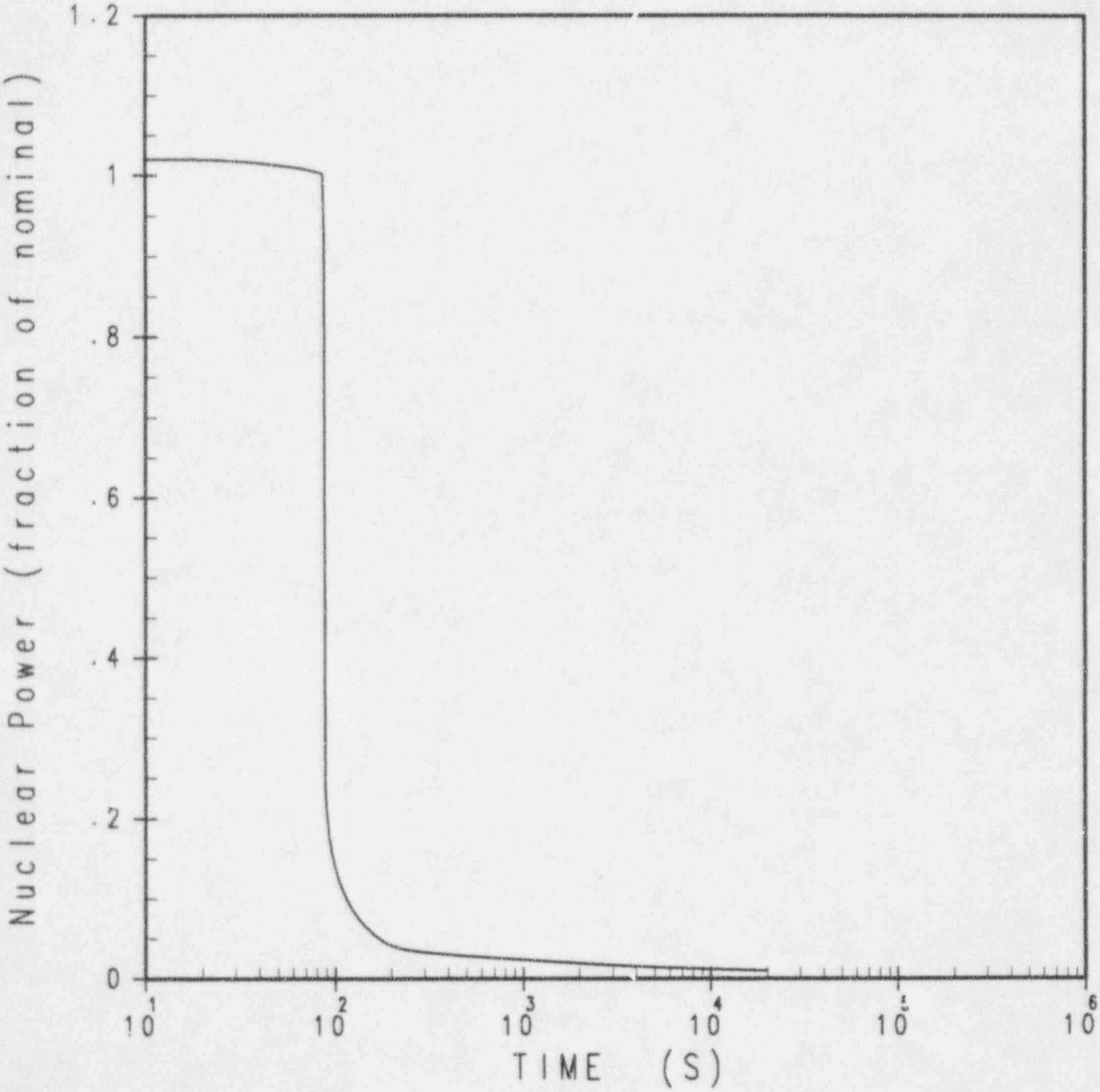


Figure 15.2.8-1

Nuclear Power Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-1



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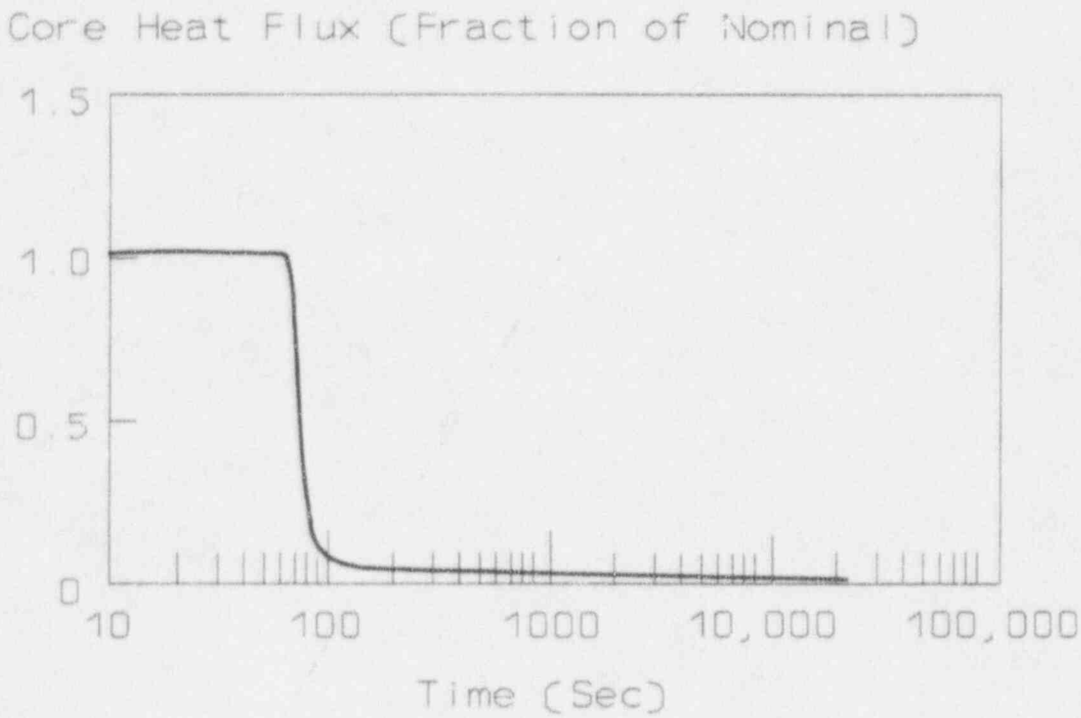
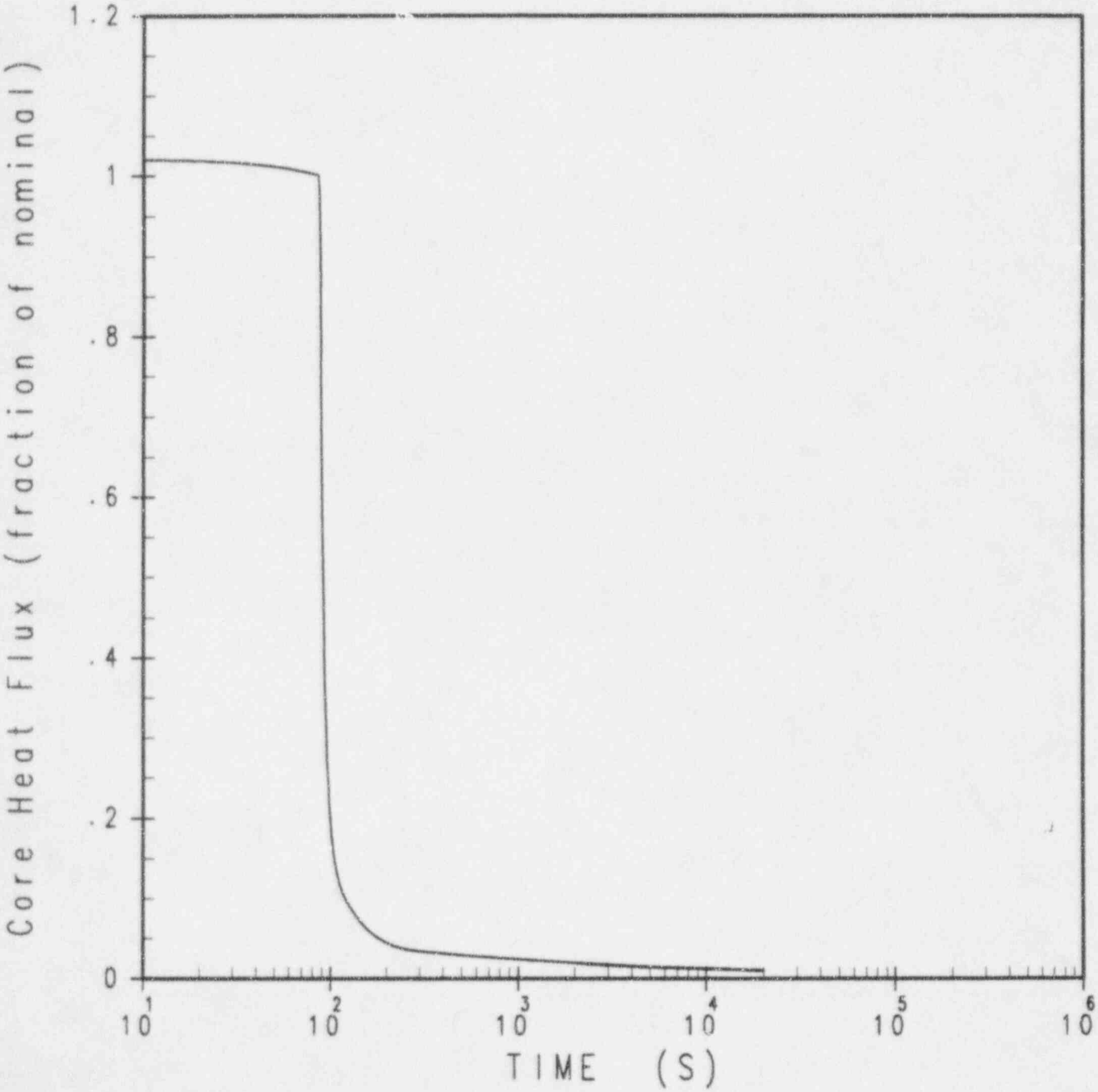


Figure 15.2.8-2

Core Heat Flux Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-2



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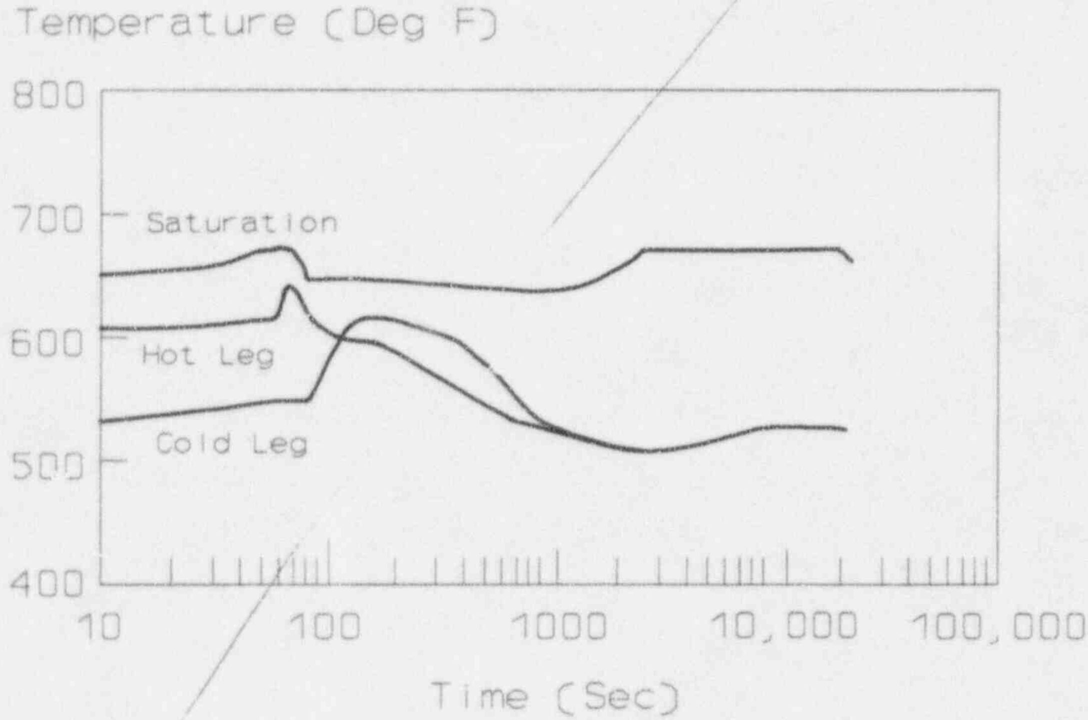
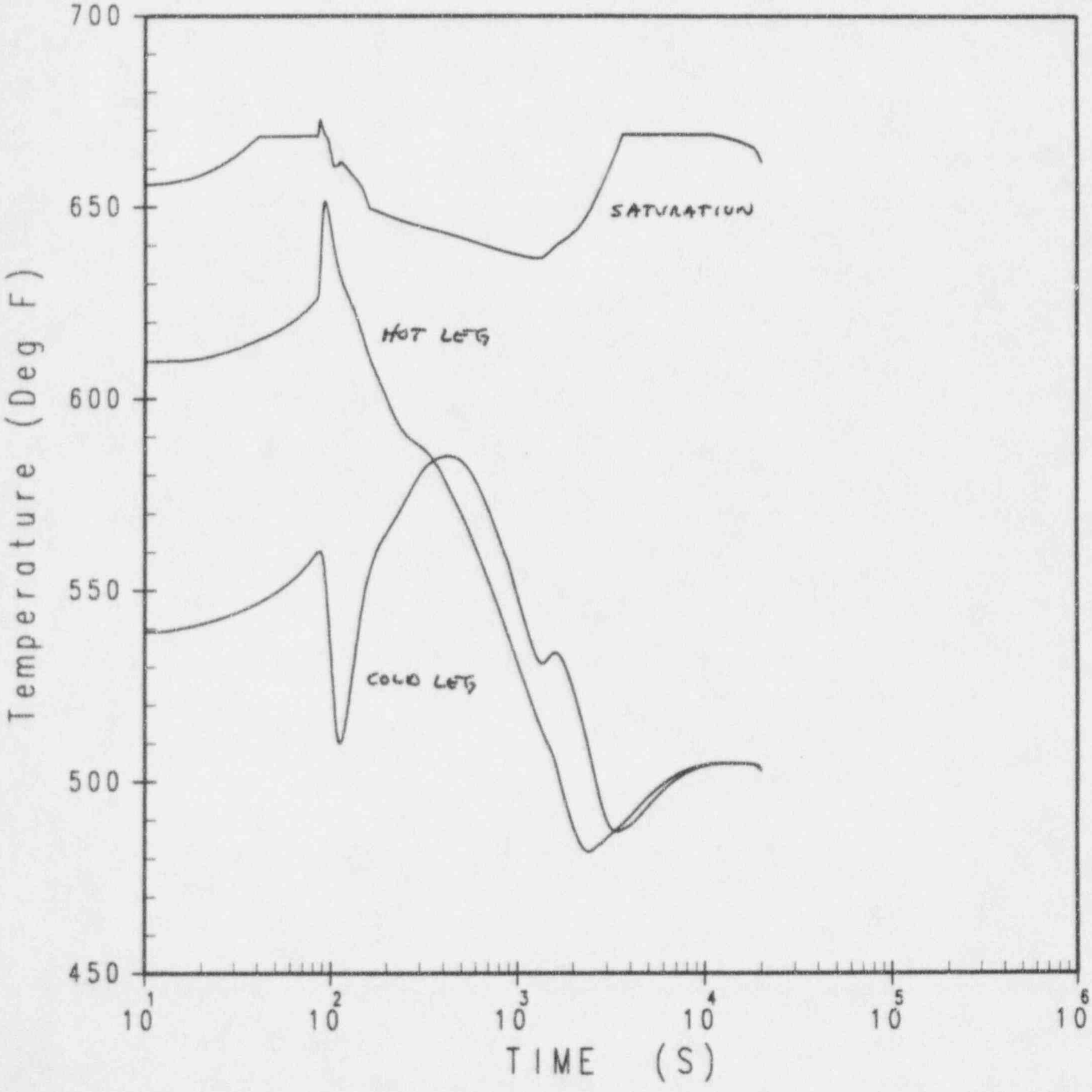


Figure 15.2.8-3

Faulted Loop RCS Temperature Transients For Main Feedwater Line Rupture

FIGURE 15.2.8-3



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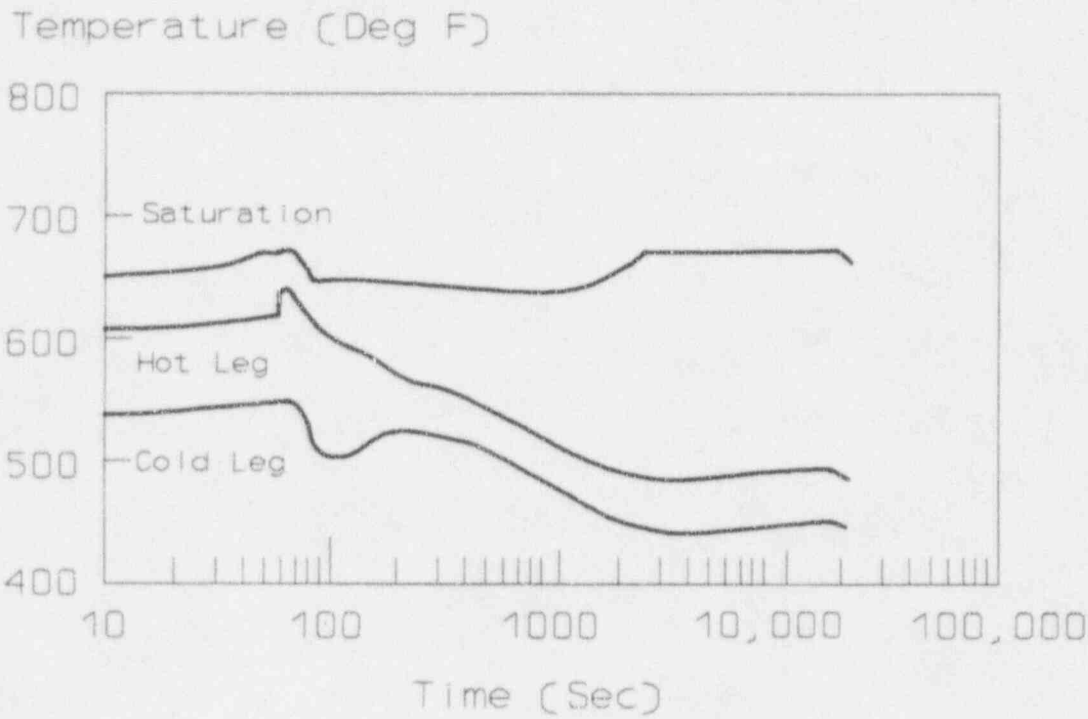
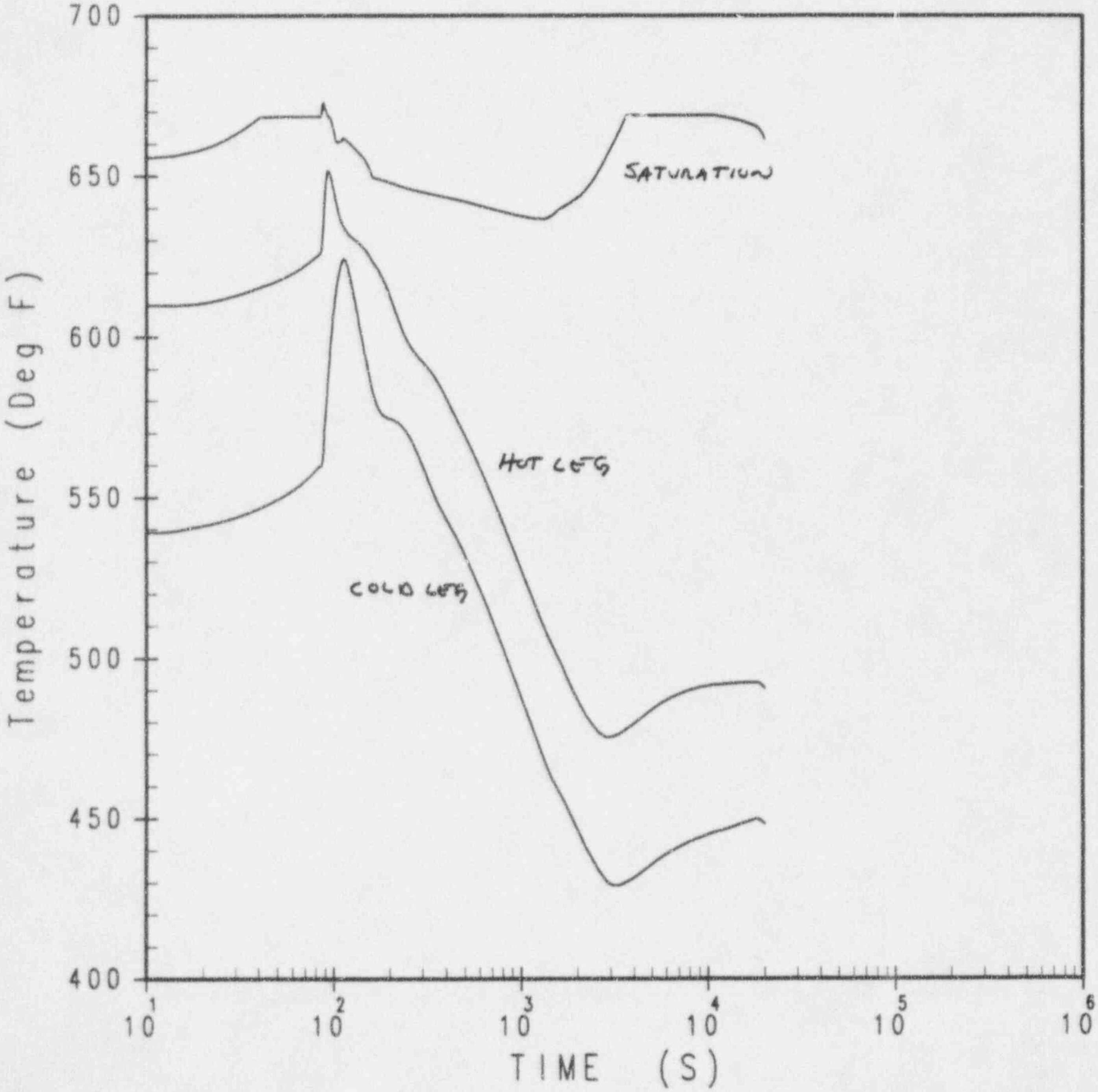


Figure 15.2.8-4

Intact Loop RCS Temperature Transients For Main Feedwater Line Rupture

FIGURE 15.2.8-4



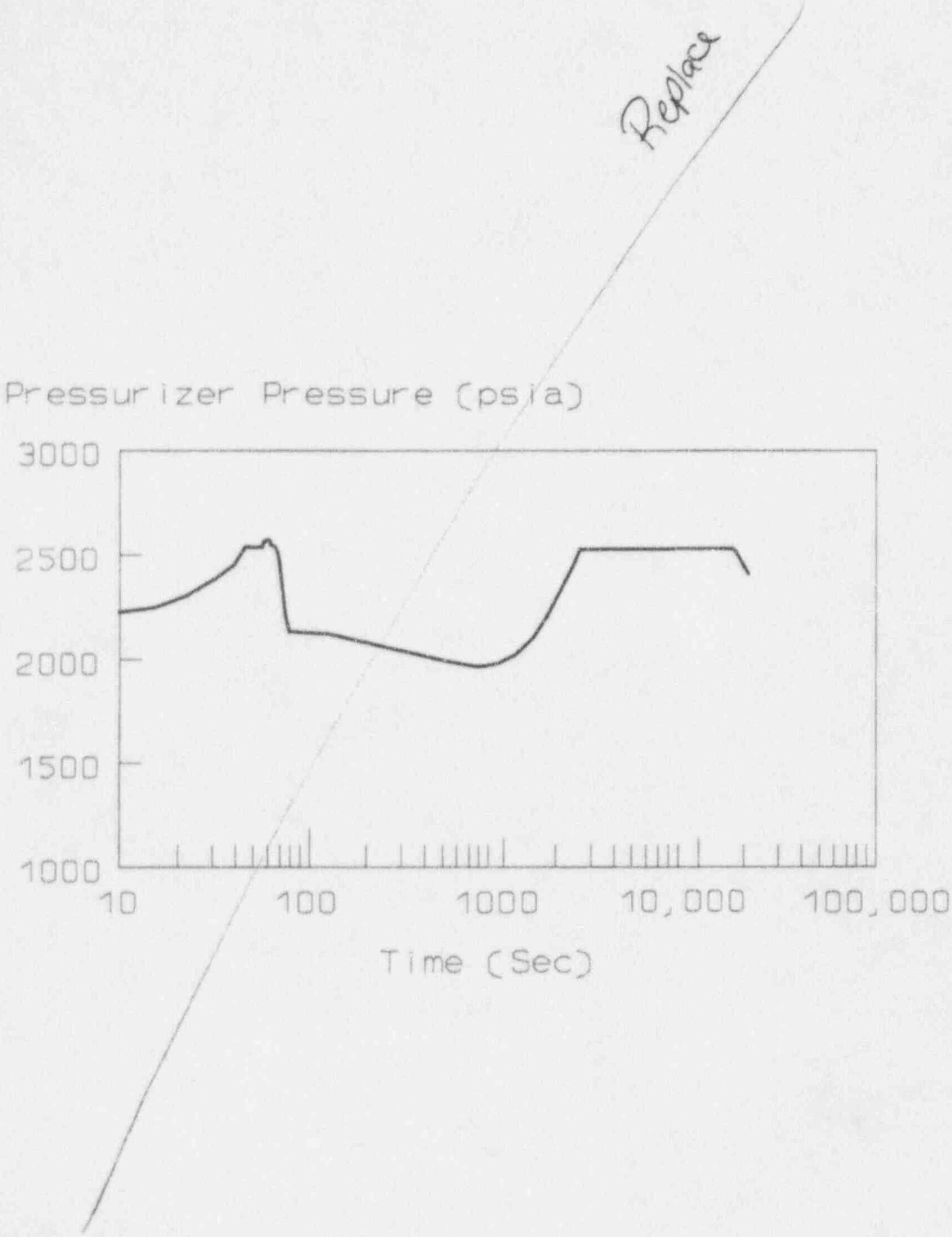
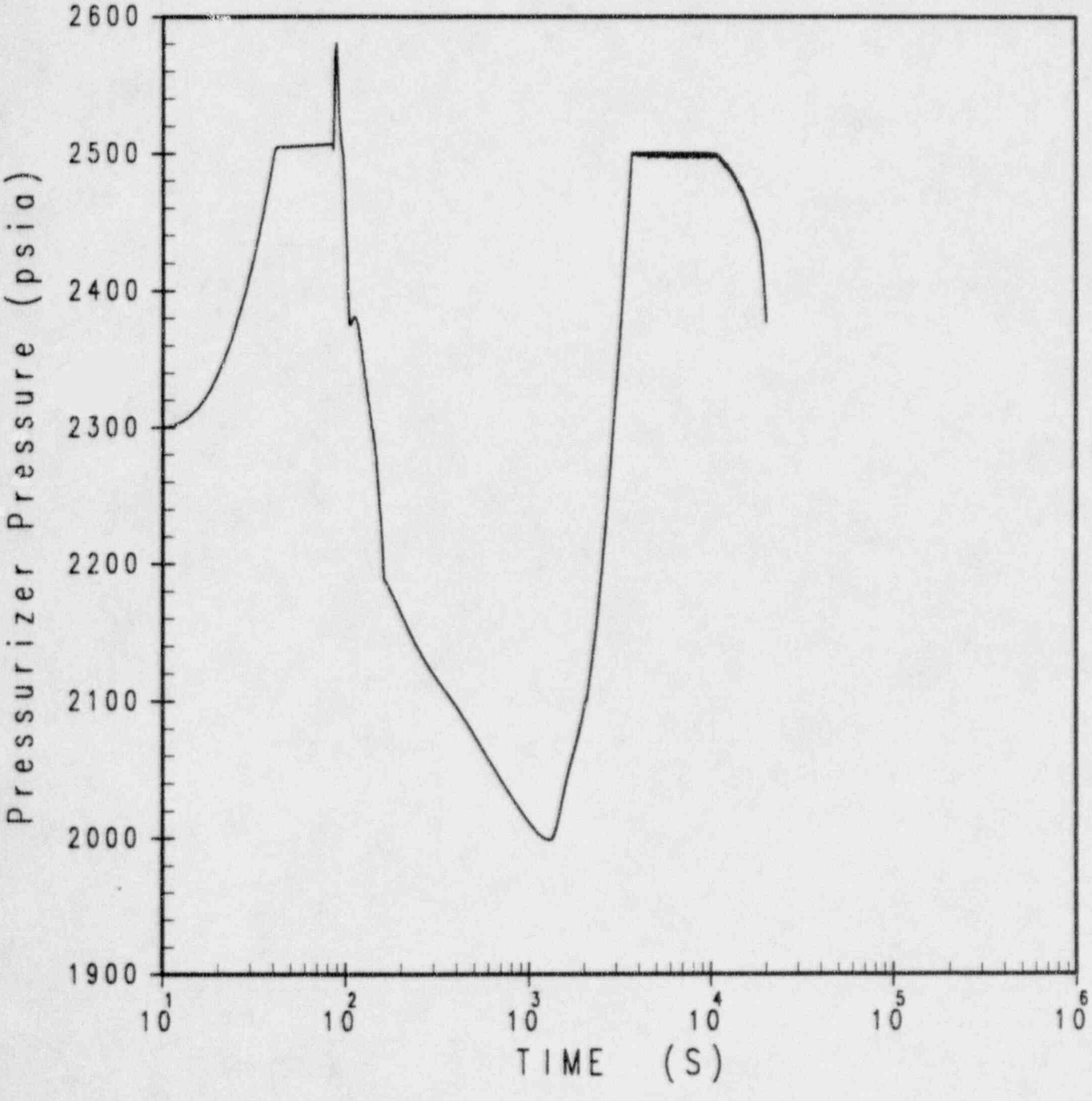


Figure 15.2.8-5

Pressurizer Pressure Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-5



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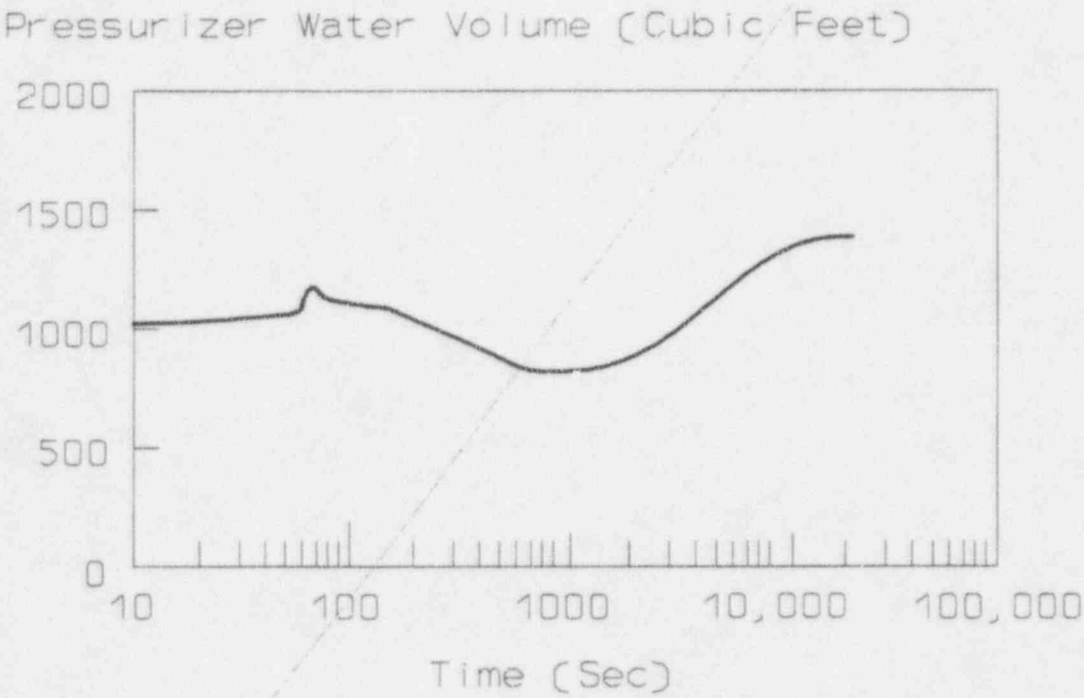
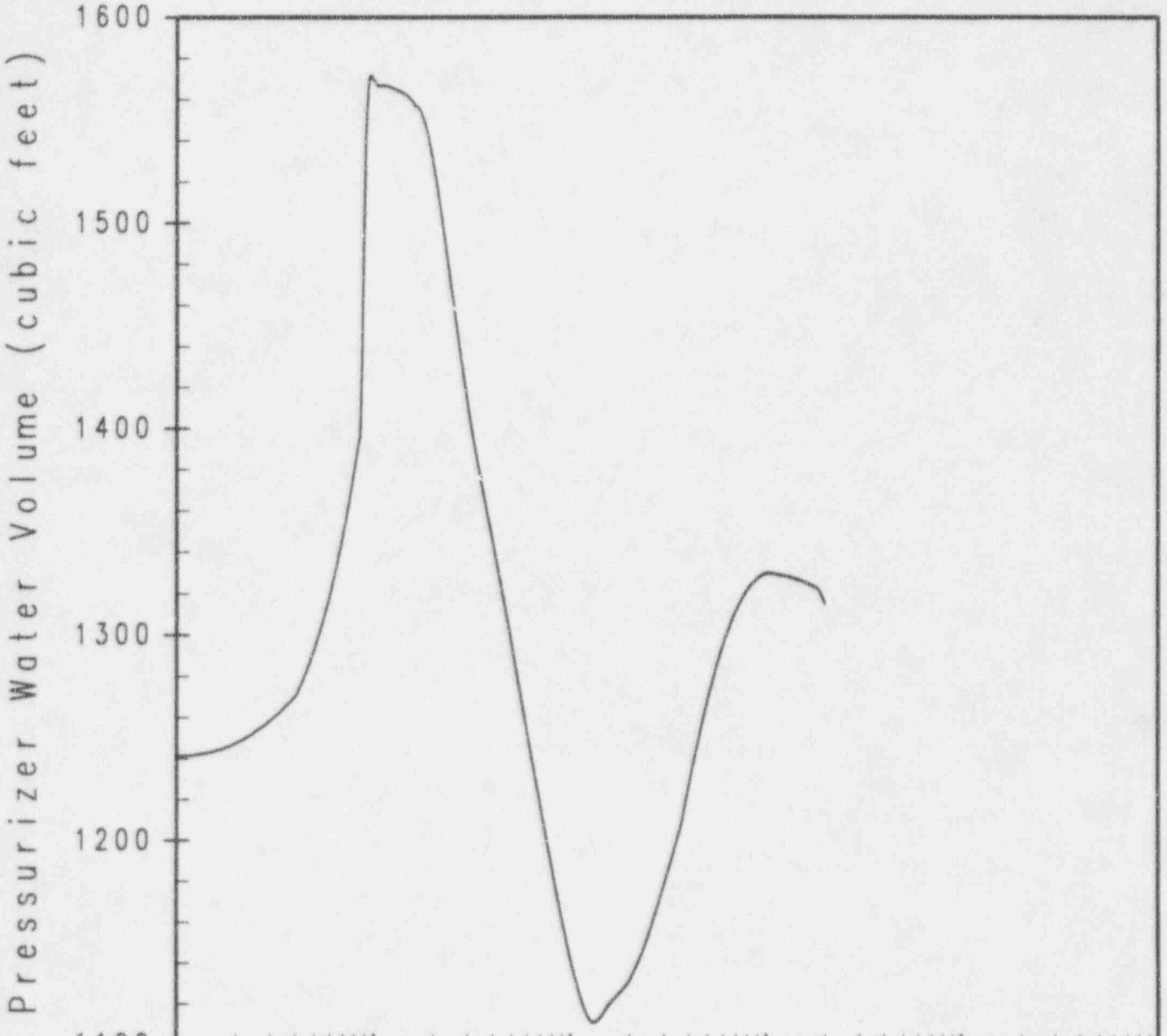


Figure 15.2.8-6

Pressurizer Water Volume Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-6



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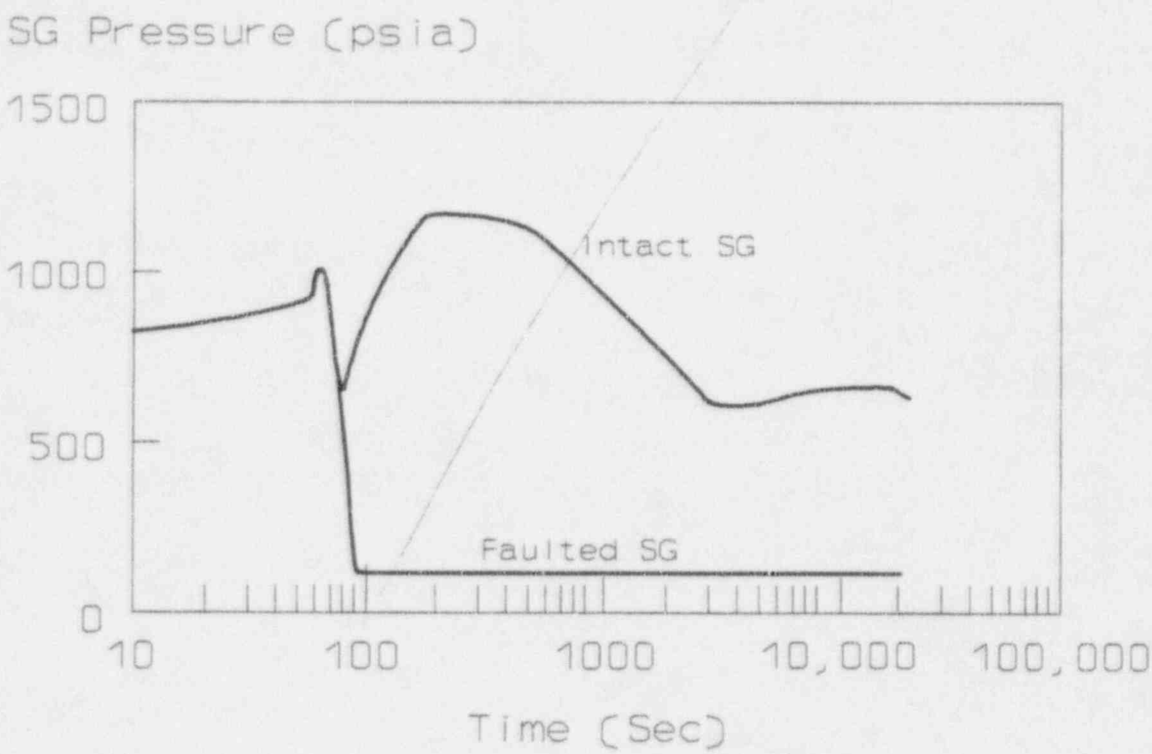
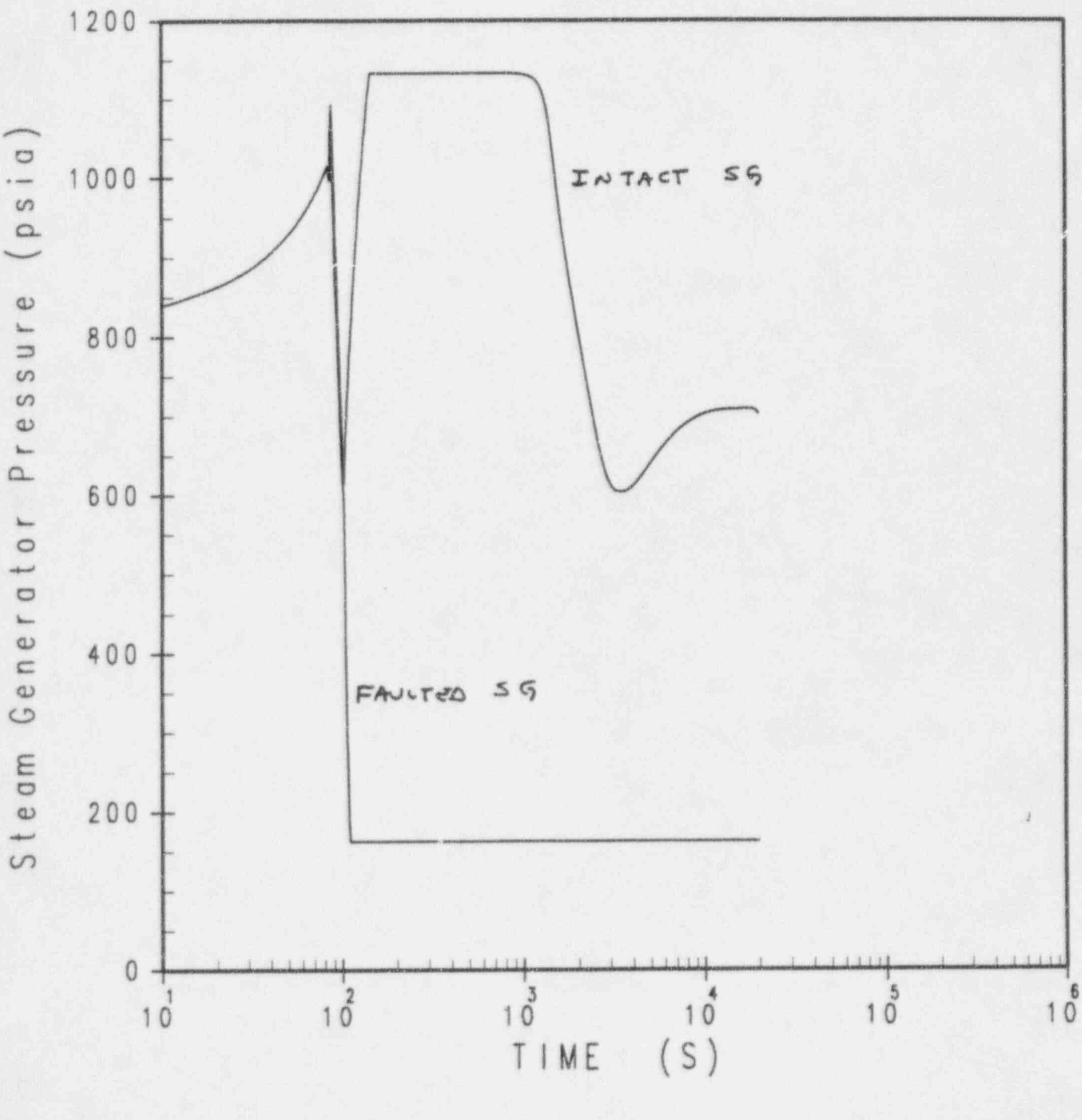


Figure 15.2.8-7

Steam Generator Pressure Transients For Main Feedwater Line Rupture

FIG. 15.2.8-7



Steam Generator Pressure (psia)
1200
1000
800
600
400
200
0
INTACT SG
FAULTED SG
10^1
10^2
10^3
10^4
10^5
10^6
TIME (S)

Replace

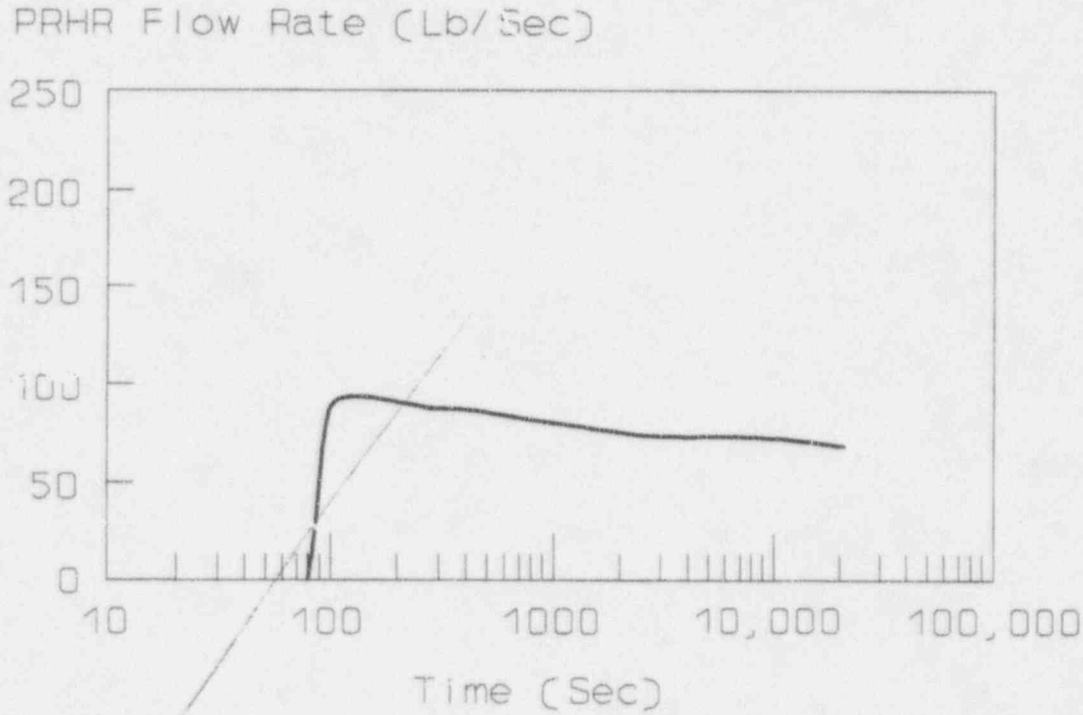
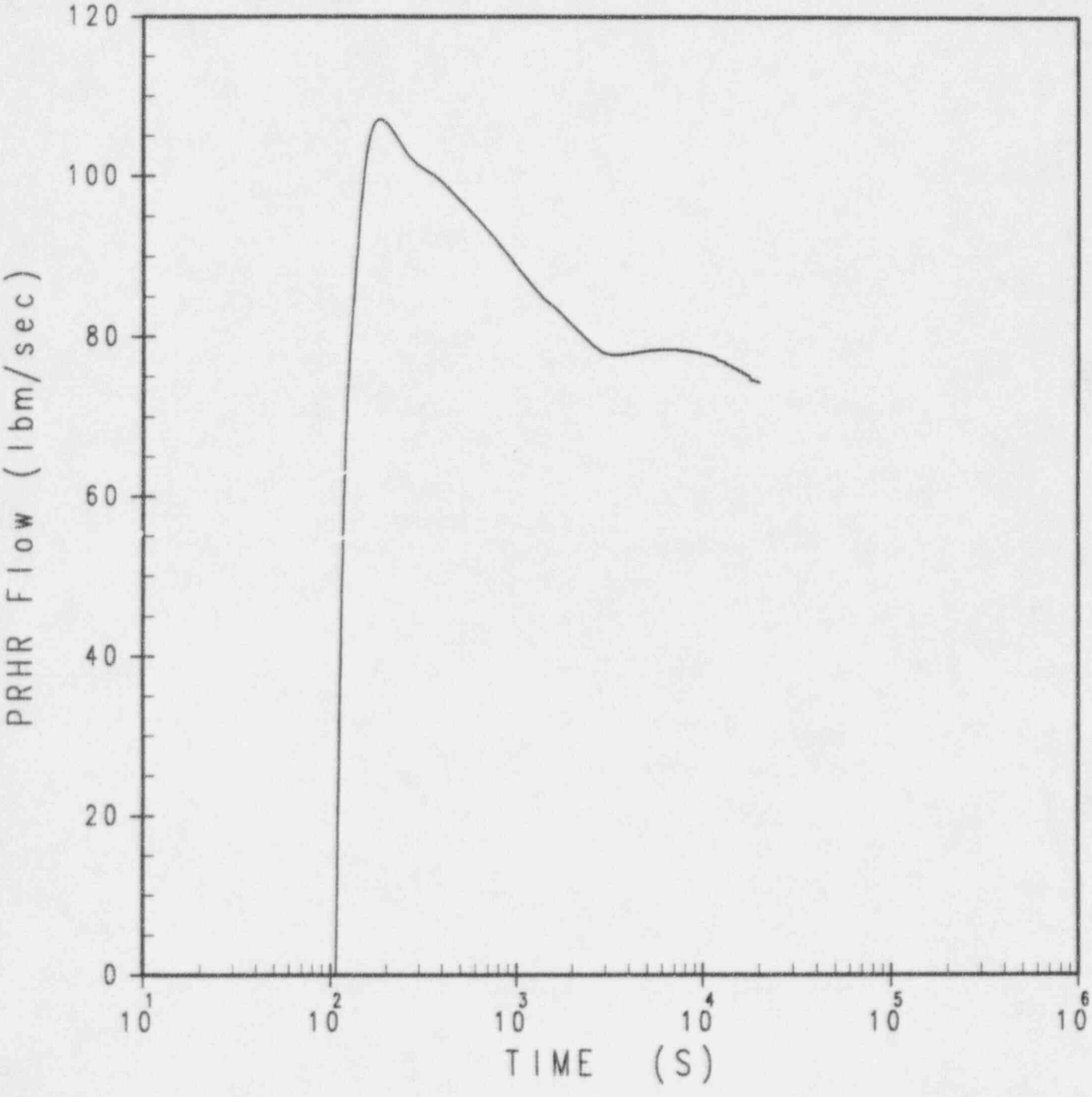


Figure 15.2.8-8

PRHR Flow Rate Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-8



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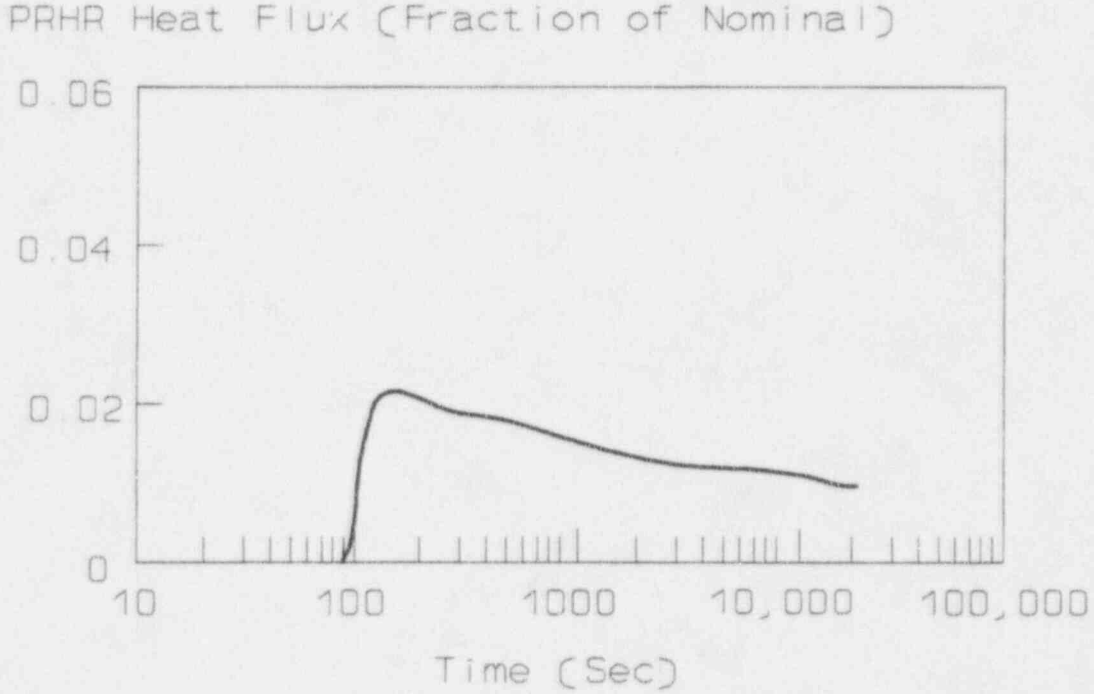
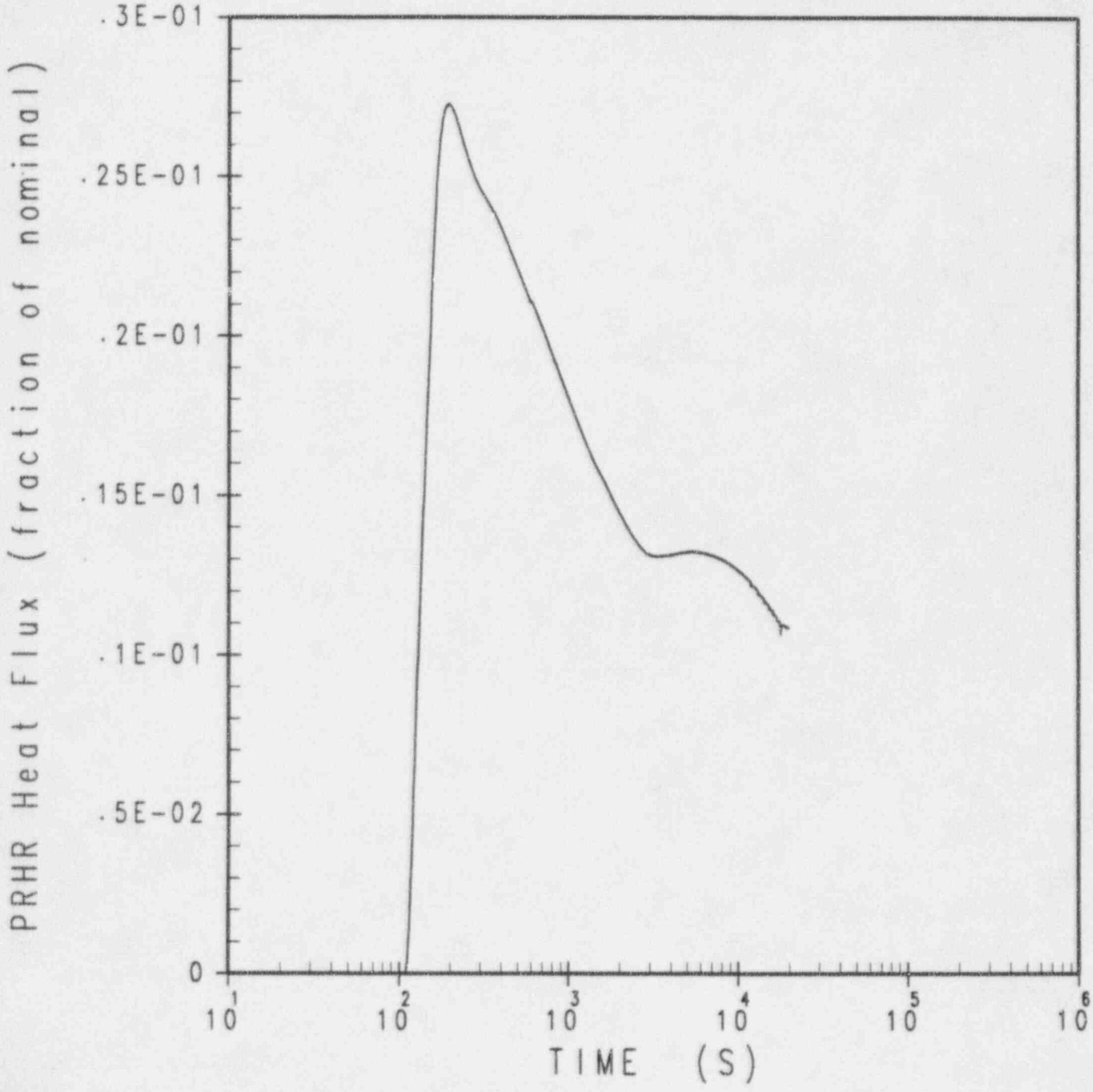


Figure 15.2.8-9

PRHR Heat Flux Transient Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-9



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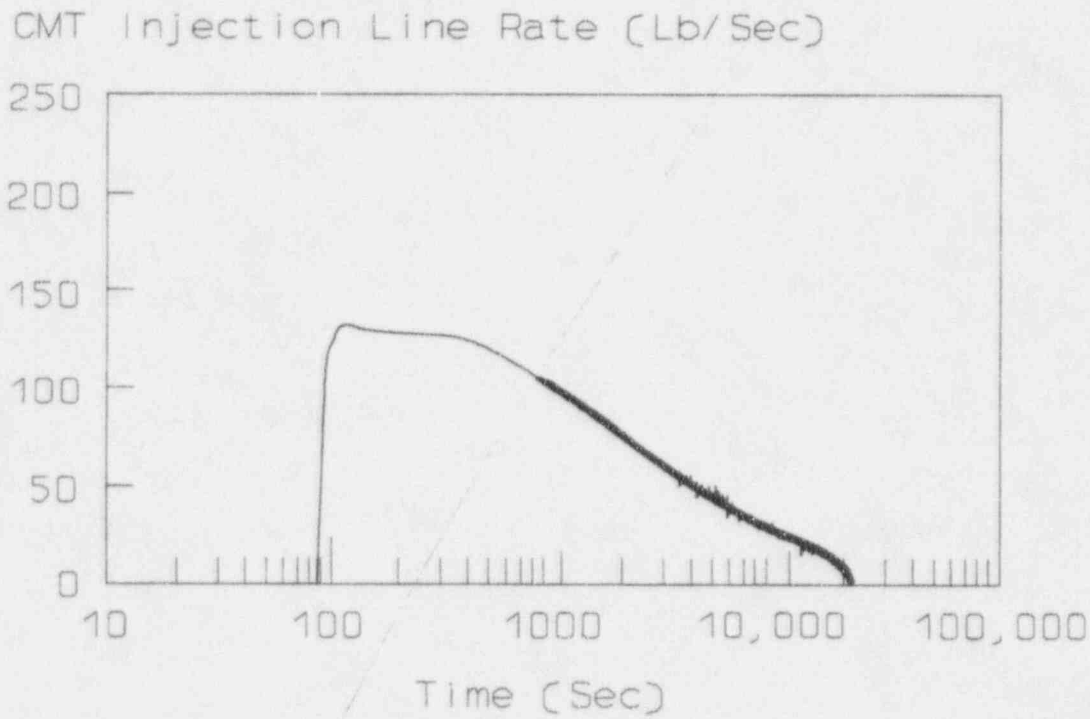
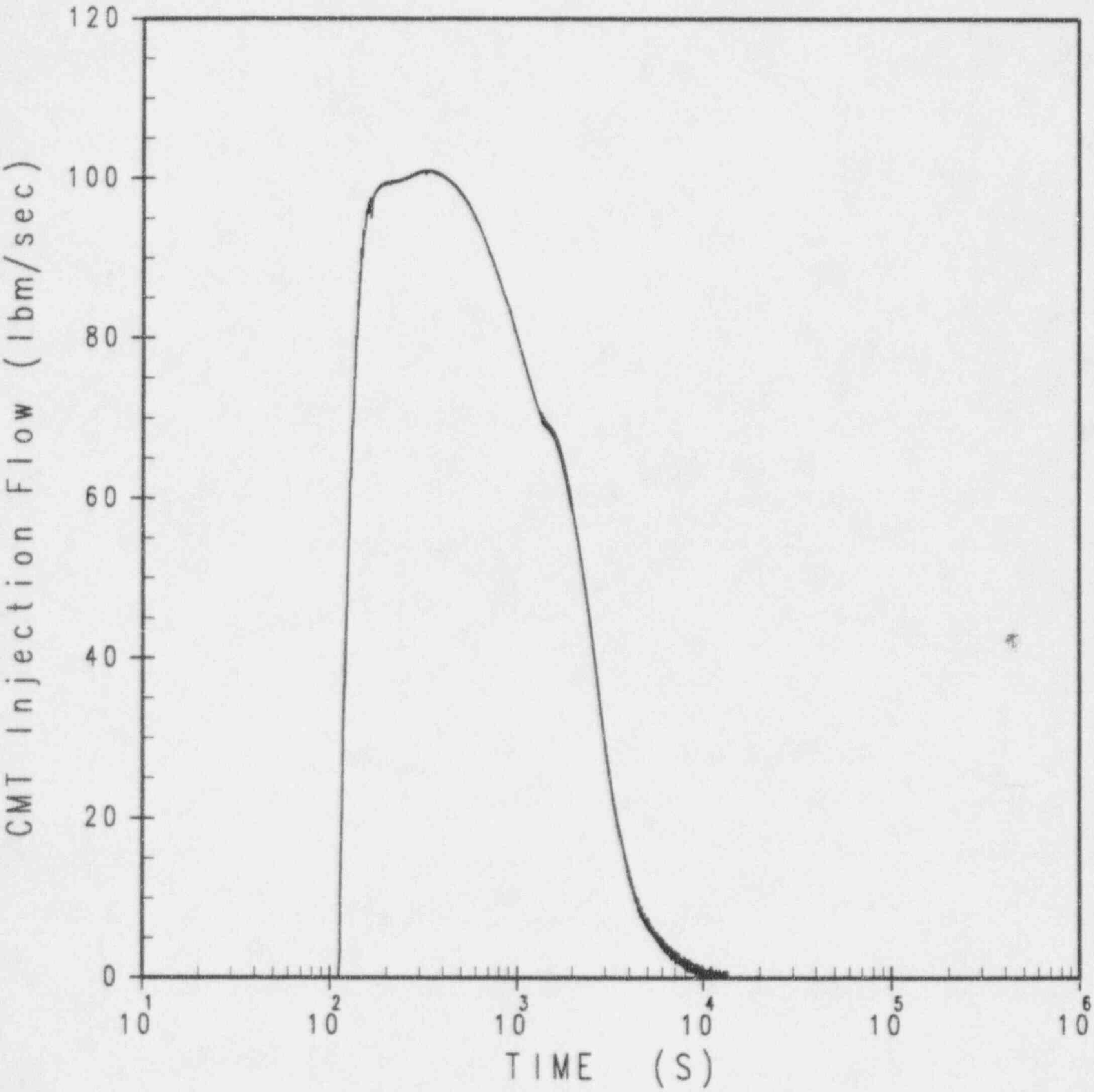


Figure 15.2.8-10

CMT Injection Flow Rate Transient For Main Feedwater Line Rupture

FIGURE 15.2.8-10



15.3 Decrease in Reactor Coolant System Flow Rate

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

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

The first event is a Condition II event; the second event is a Condition III event, and the last two are Condition IV events.

The four limiting flow decrease events described above are analyzed in this section. It has been determined that the most severe radiological consequences result from the reactor coolant pump (RCP) shaft seizure accident discussed in Subsection 15.3.3. Doses are reported only for that ~~limiting~~ case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

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

Normal power for the pumps is supplied through two buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power. The pumps continue to operate.

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

Protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two out of four Low-flow signals. Above permissive P8, low flow in any one cold leg actuates a reactor trip. (See Section 7.2.) Between approximately 10 percent power (permissive P10) and the power level corresponding to permissive P8, low flow in any two cold legs actuates a reactor trip.

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

Strikeout ¶

A partial loss of flow involving the loss of either one or two pumps is analyzed. The transient response for the partial loss of two reactor coolant pumps is found to be more limiting. Therefore, the transient analysis is reported only for the limiting partial loss of two pumps transient.

This transient is analyzed by three computer codes. First, the LOFTRAN code (Reference 1) is used to calculate the core flow during the transient, the time of reactor trip based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients as predicted from the loss of two reactor coolant pumps. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the ~~THINC~~ WESTAR code (Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical cell or the thimble cell.

15.3.1.2.2 Initial Conditions

Initial reactor power, ~~and~~ pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit, as described in Reference 5.

Plant characteristics and initial conditions assumed in this analysis are further discussed in Subsection 15.0.3.

15.3.1.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. (See Figure ~~15.0-41~~ 15.0.4-1.) This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent power of 0.0118 Δk.

The least negative moderator temperature coefficient is assumed, since this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached.

For these analyses, a curve of trip reactivity versus time based on a 2.4 second rod cluster control assembly (RCCA) insertion time to the dashpot is used. (See Subsection 15.0.5.)

15.3.1.2.4 Flow Coastdown

A conservatively calculated flow coastdown is used to simulate the transient. The flow coastdown is calculated based on reactor coolant system (RCS) pressure losses and reactor coolant pump characteristics. Reactor coolant fluid momentum is ~~conservatively~~ neglected in order to obtain a conservative calculation.

Plant systems and equipment necessary to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.1.2.5 Results

Figures 15.3.1-~~1A~~ 2 through 15.3.1-~~4~~ 6 show the transient response for the loss of two reactor coolant pumps. Figure 15.3.1-~~4~~ 6 shows the DNBR to be always greater than the ~~limiting~~ design value.

The plant is tripped by the Low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. The affected reactor coolant pumps continue to coast down, and the core flow reaches a new equilibrium value.

With the reactor tripped, a stable plant condition is attained. Normal plant shutdown may then proceed.

15.3.1.3 Conclusions

The analysis shows that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the ~~limiting~~ design basis value at any time during the transient. The DNBR design basis is described in Section 4.4. The applicable SRP Section 15.3.1 evaluation criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

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

Electric power for the reactor coolant pumps is supplied through buses from unit auxiliary transformers connected to the generator. When a generator trip occurs, the unit auxiliary transformers receive power from external power lines, and the pumps continue to supply coolant flow to the core.

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

- Reactor coolant pump underspeed

- Low reactor coolant loop flow

The reactor trip on reactor coolant pump underspeed protects against conditions that can cause a loss of voltage to the reactor coolant pumps. This function is blocked below approximately 10 percent power (permissive P10).

The reactor trip on reactor coolant pump underspeed 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 five Hertz per second, this trip protects the core from underfrequency events. Reference 3 provides analyses of grid frequency disturbances and the resulting protection requirements which are applicable to the AP600 plant

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions that affect only one or two reactor coolant loop cold legs. This function is generated by two out of four low flow signals per reactor coolant loop cold leg. Above permissive P8, low flow in any loop actuates a reactor trip. Between approximately 10 percent power (permissive P10) and the power level corresponding to permissive P8, low flow in any two reactor coolant loop cold legs actuates a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hertz per second, this trip function also protects the core from this underfrequency events. This effect is fully described in Reference 3.

15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of all four reactor coolant pumps.

For the case analyzed with a complete loss of voltage, followed by the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1, with one exception. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the reactor coolant pump underspeed trip.

A loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. The results of the complete loss of voltage, followed by the reactor coolant pump coasting down, bound the complete loss of flow initiated by a frequency decay of up to 5 Hertz per second. Therefore, only the results of the complete loss of voltage case are presented in Subsection 15.3.2.2.2.

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-~~4~~ 6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is assumed to trip on the reactor coolant pump

underspeed signal. Figure 15.3.2-~~4~~ 6 shows that the departure from nucleate boiling ratio (DNBR) is always greater than the ~~limiting~~ design limit value.

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

15.3.2.3 Conclusions

The analysis performed demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the ~~safety analysis~~ design basis limit value at any time during the transient. The design basis for the DNBR is described in Section 4.4. The applicable SRP Section 15.3.1 evaluation criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

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

Following 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 second, 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 (RCS). The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

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

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 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, and the nuclear power following reactor trip. ~~It~~ This code is also used to determine the peak pressure. The thermal behavior

of the fuel located at the core hot spot is investigated by using the FACTRAN code (Reference 2). This code uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (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, that is, maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at reactor trip.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia). This allows 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.

15.3.3.2.2 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1.45 seconds 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 spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to result in a lower peak pressure, an additional conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.

15.3.3.2.3 Evaluation of Departure from Nucleate Boiling in the Core During the Accident

For this accident, departure from nucleate boiling (DNB) is calculated to occur in the core. Therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be 2.6 times the average rod power (that is, $F_Q = 2.6$) at the initial core power level.

15.3.3.2.4 Film-Boiling Coefficient

The film-boiling coefficient is calculated in the FACTRAN code (Reference 2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The

nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB is assumed to start at the beginning of the accident.

15.3.3.2.5 Fuel Clad Gap Coefficient

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

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation is used to define the rate of the zirconium-steam reaction:

$$\frac{d\,(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45{,}500}{1.9867}\right)$$

where:

w = amount reacted (mg/cm^2)
t = time (s)
T = temperature (°K) T Kelvin

The reaction heat is 1510 cal/g.

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

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3.3-1A through 15.3.3-~~4D~~ 7 show the transient results for one locked rotor with four loops in operation without offsite power available. ~~This~~ The without-offsite power case bounds the results for the case with offsite power. 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 causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak clad surface temperature is considerably less than 2700°F. The clad temperature is conservatively calculated, assuming that departure from nucleate boiling occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. With the reactor tripped, a stable plant condition eventually is attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

Subsection 15.3.3.3 to follow

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

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

Following 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 second, 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 and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

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

15.3.4.2 Conclusion

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 departure from nucleate boiling (DNB) occurs at the beginning of the transient. The calculated results presented for the locked rotor analysis bound the reactor coolant pump shaft break event.

Insert A

15.3.~~5~~ 6 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

A 15.3.5 Combined License Information
This section has no requirement for additional information to be provided in support of the combined license application.

2. Hargove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1975.

3. Baldwin, M. S., et al., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, May 1975.

4. Van Houten, R., "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout," NUREG-0562, June 1979.

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), April 1989.

Table 15.3-1

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

Accident	Event	Time(s)
Partial loss of forced reactor coolant flow		
Loss of two pumps with four pumps running	Coastdown begins	0.00
	Low-flow reactor trip	0.56
	Rods begin to drop	~~2.02~~ 2.01
	Minimum DNBR occurs	3.90
Complete loss of forced reactor coolant		
Loss of four pumps with four pumps running	Operating pumps lose power and begin coasting down	0.00
	Reactor coolant pump under speed trip point reached	0.32
	Rods begin to drop	~~1.10~~ 1.09
	Minimum DNBR occurs	~~[illegible]~~ 2.80
Reactor coolant pump shaft seizure (locked rotor)		
One locked rotor with four pumps running with offsite power available.	Rotor on one pump locks	0.00
	Low-flow trip point reached	0.01
	Rods begin to drop	1.46
	Maximum clad temperature occurs	~~2.00~~ 2.20
	Maximum reactor coolant system pressure occurs	~~3.70~~ 4.20
One locked rotor with four pumps running without offsite power available.	Rotor on one pump locks	0.00
	Low-flow trip point reached	0.01
	Rods begin to drop, loss of offsite power occurs	1.46
	Maximum clad temperature occurs	3.10
	Maximum reactor coolant system pressure occurs	~~4.20~~ 4.[illegible]0

Table 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS (FOUR REACTOR COOLANT PUMPS OPERATING INITIALLY)

	Without Offsite Power Available
Maximum RCS pressure (psia)	~~2714~~ 2657
Maximum clad temperature, core hot spot (°F)	~~1823~~ 1817
$Zr-H_2O$ reaction, core hot spot (percent by weight)	~~0.47~~ 0.46

Table 15.3-3

Table 15.3-3 to follow

Table 15.3-4

Table 15.3-4 to follow

Table 15.3-5 to follow

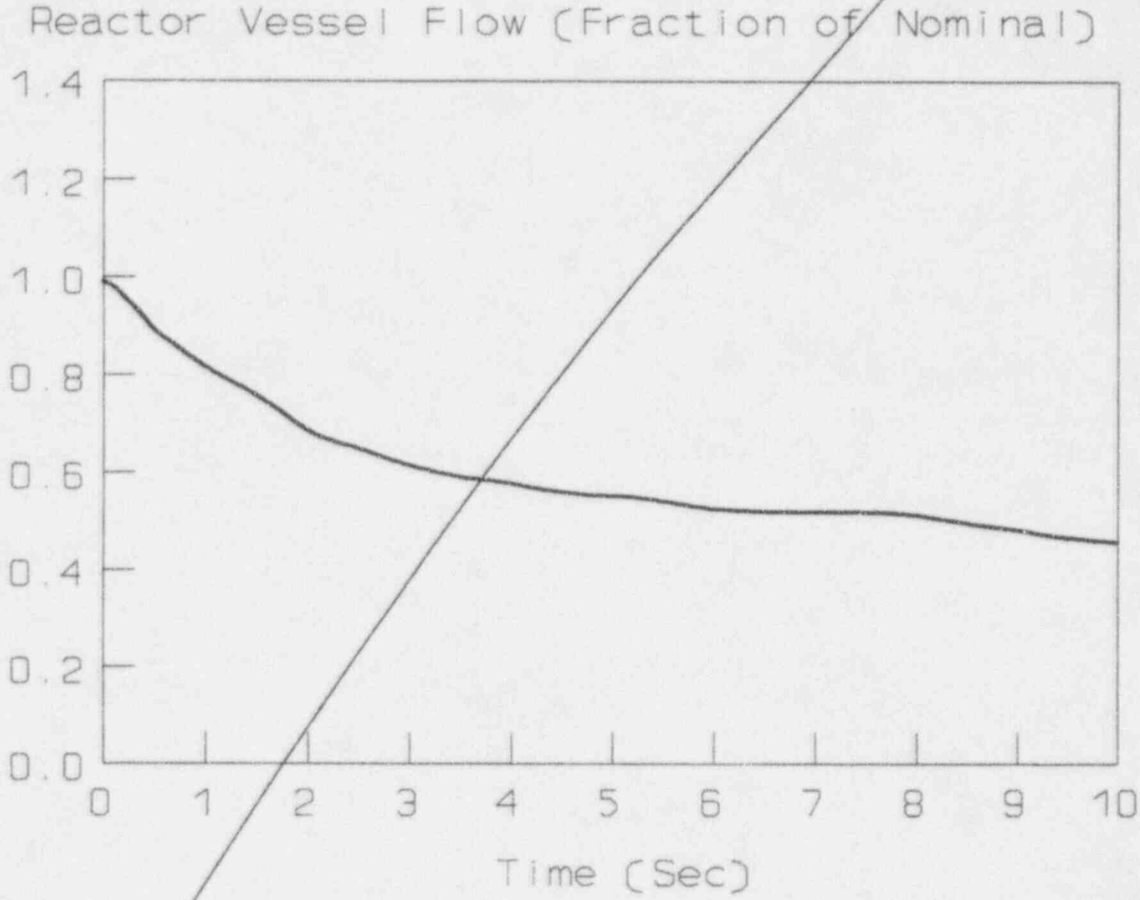


Figure 15.3.1-1 A

Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

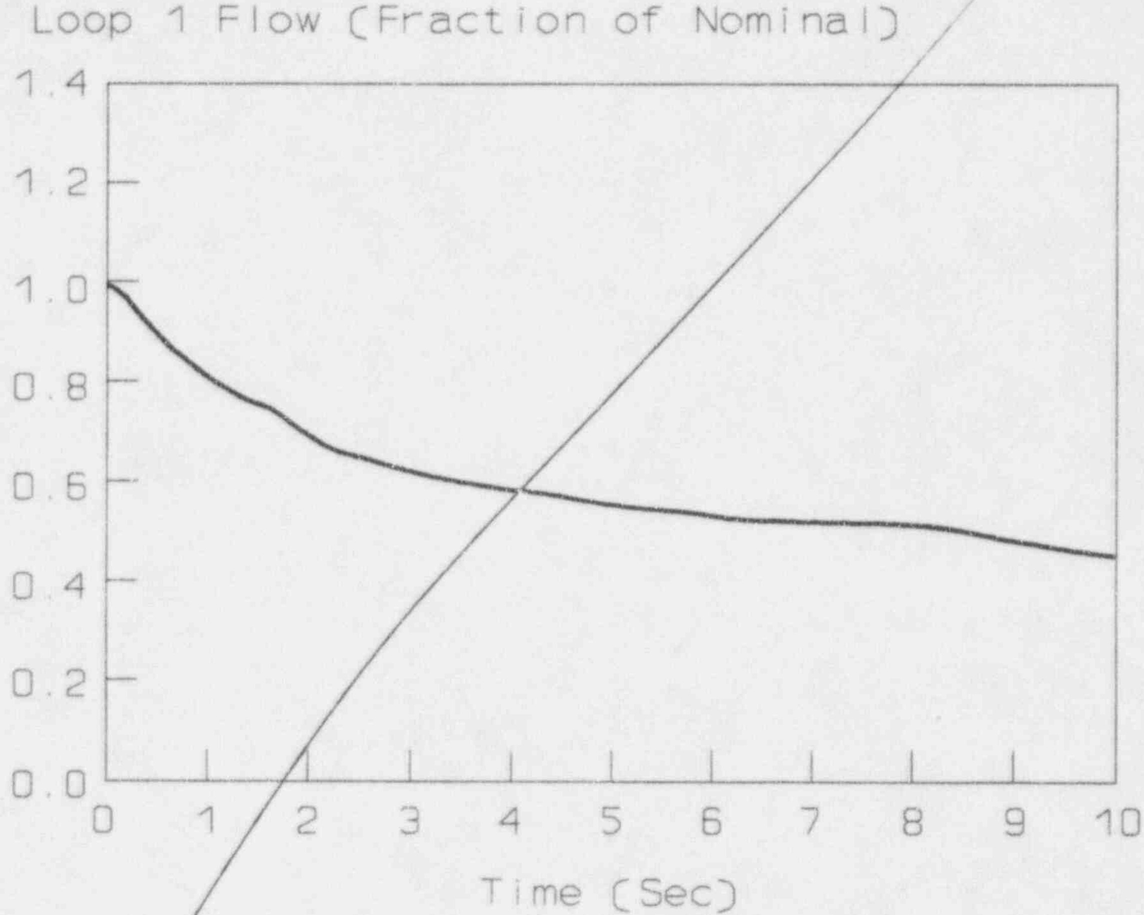


Figure 15.3.1-1 B

Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

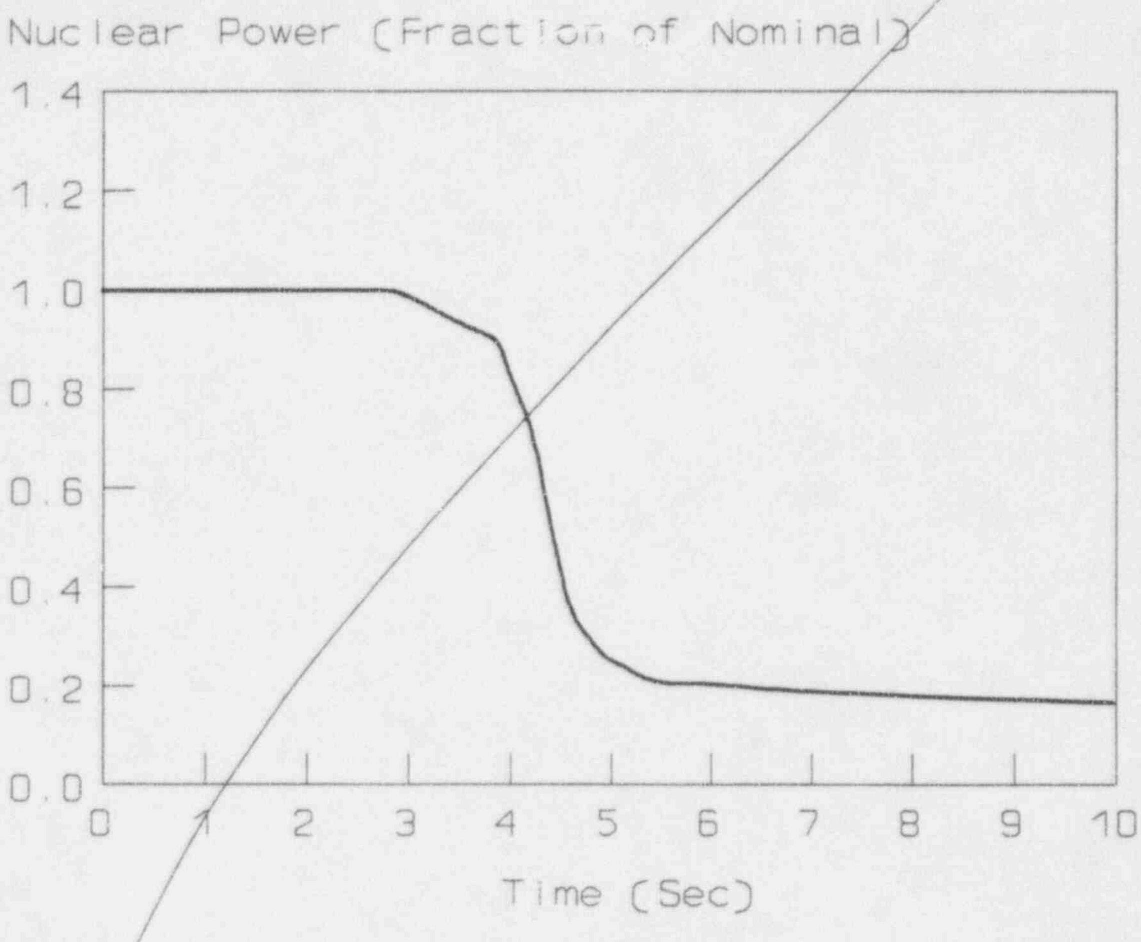


Figure 15.3.1-2 A

Nuclear Power Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

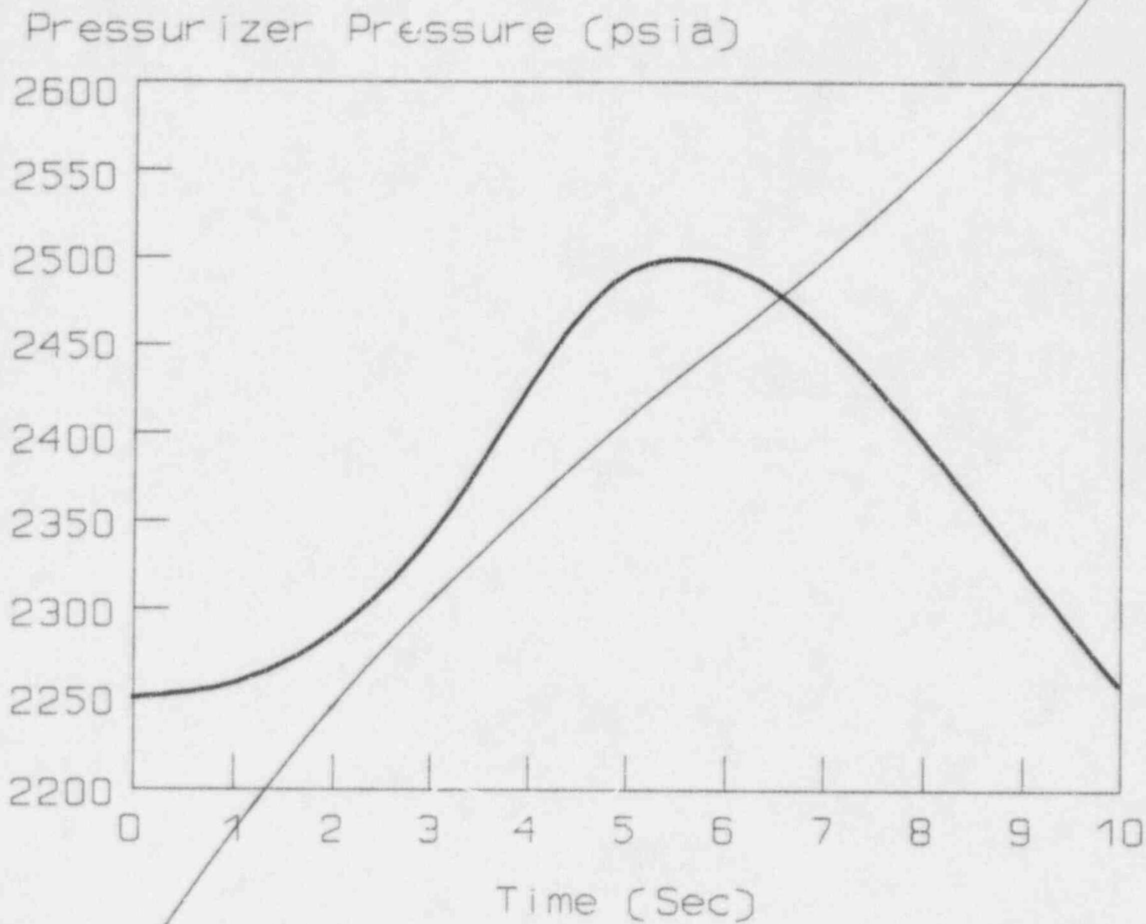


Figure 15.3.1-2 B

Pressurizer Pressure Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

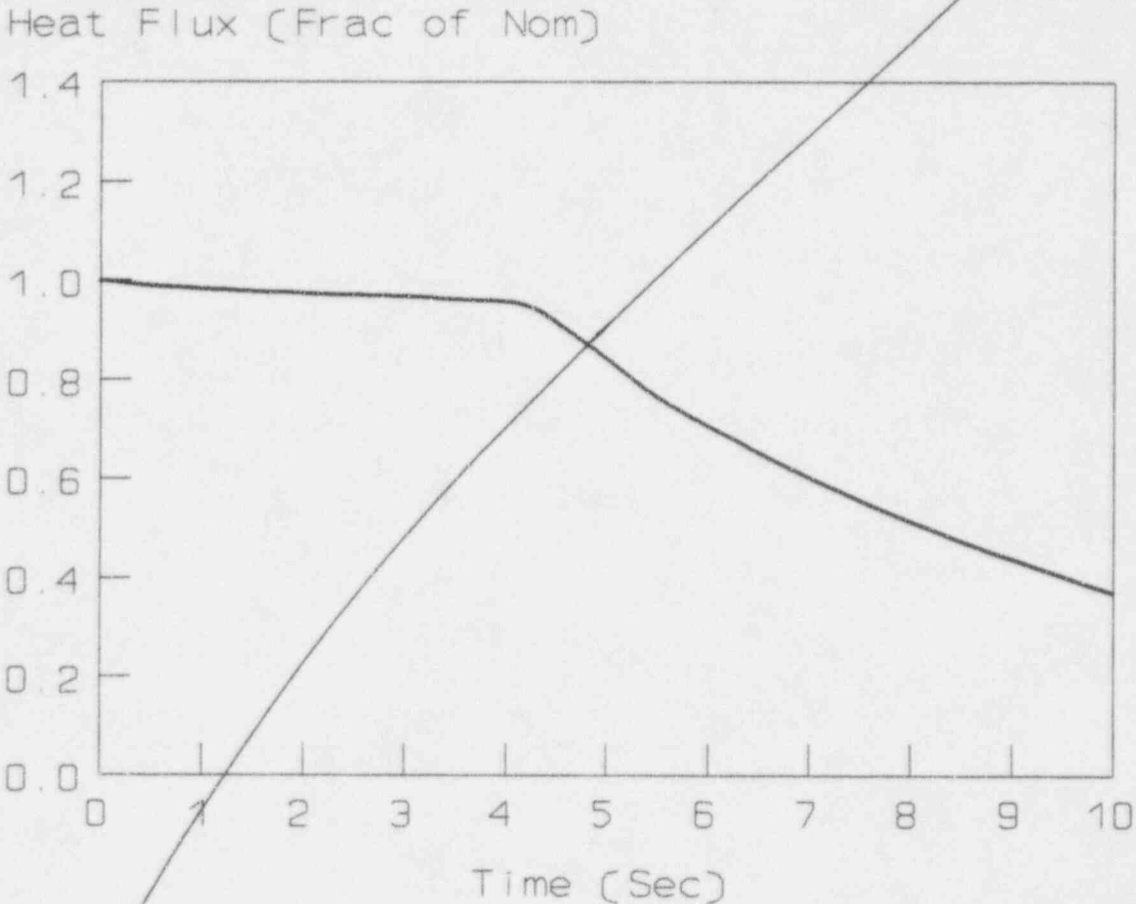


Figure 15.3.1-3 A

Average Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

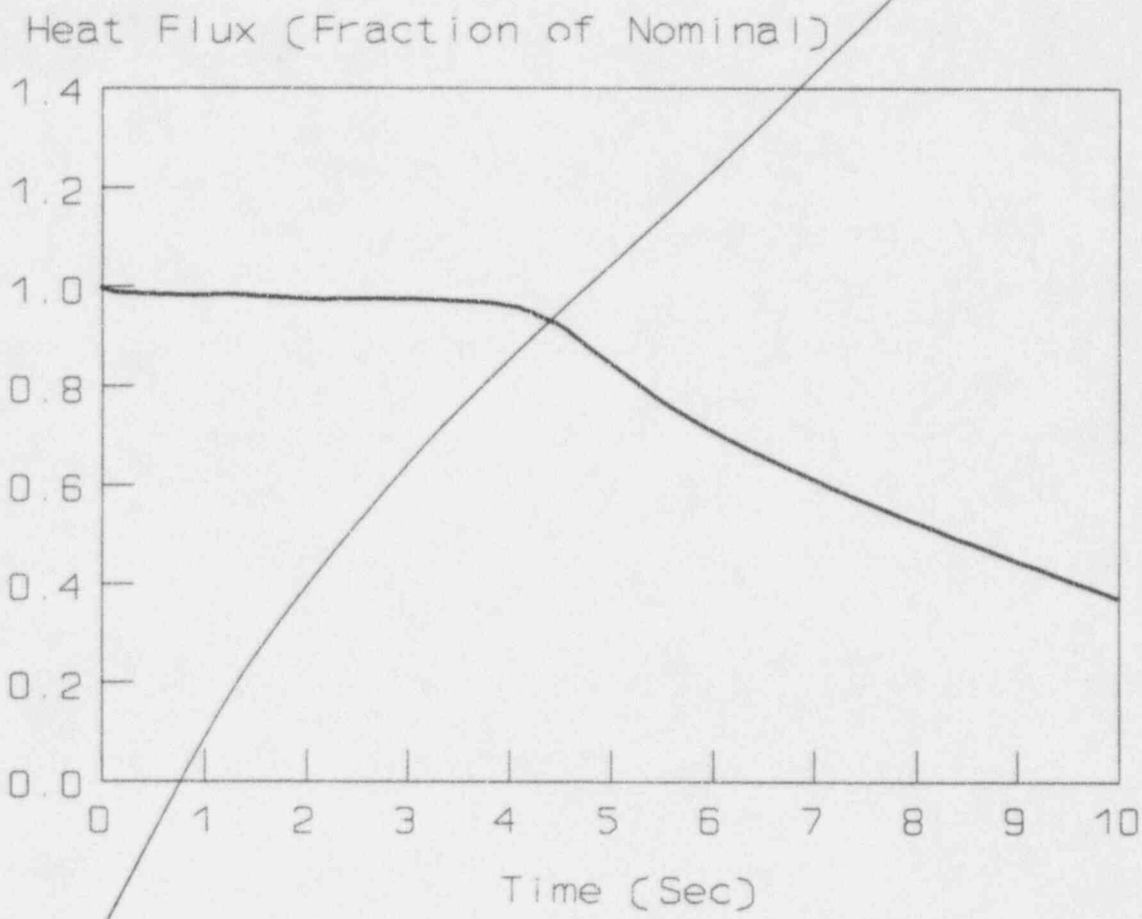


Figure 15.3.1-3 B

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

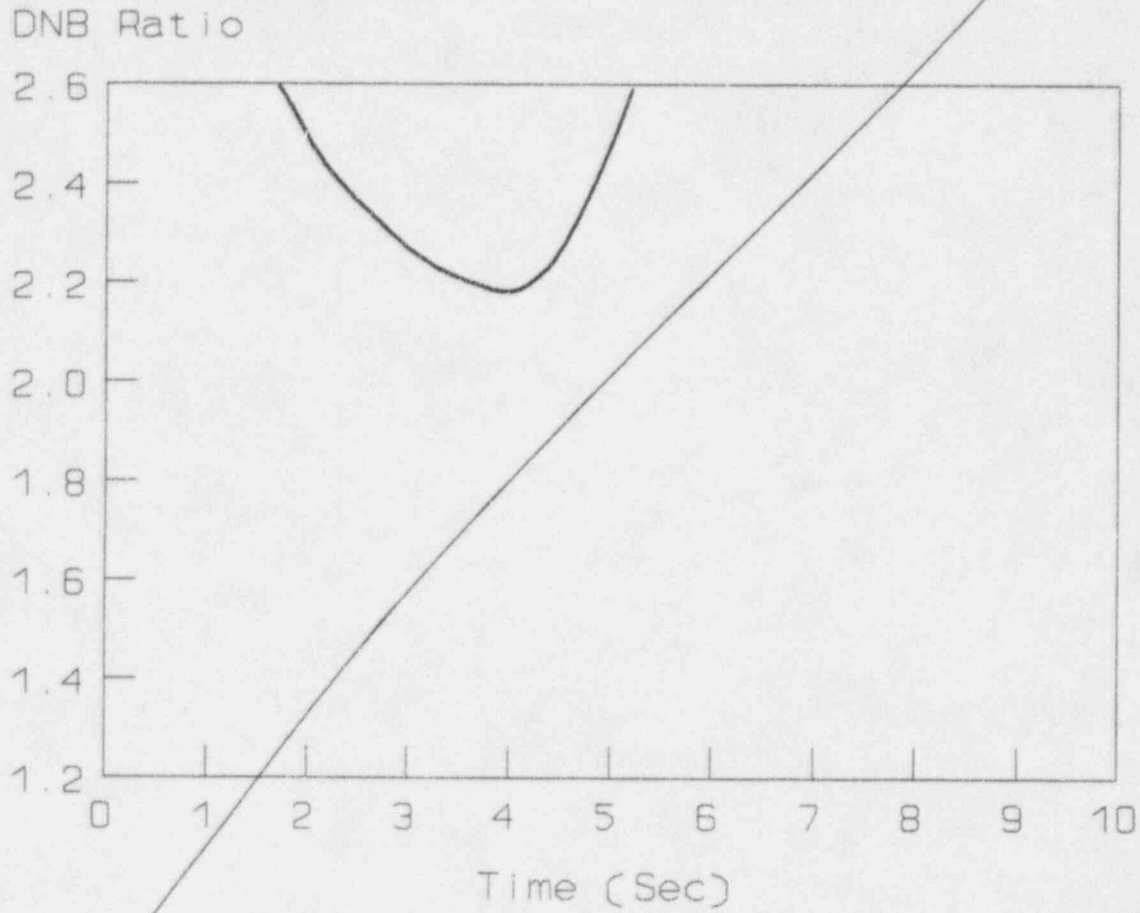


Figure 15.3.1-4

DNBR vs. Time for Four Cold Legs in Operation, Two Pumps Coasting Down

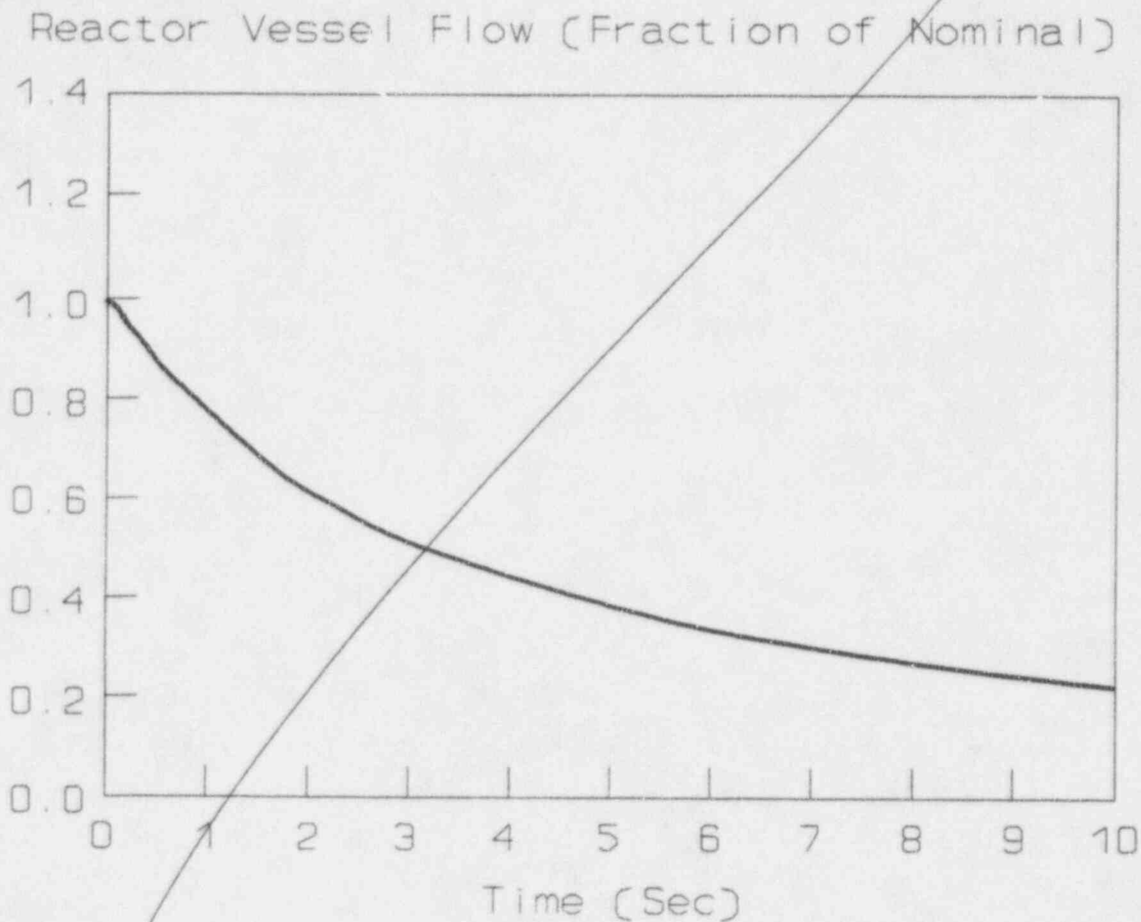
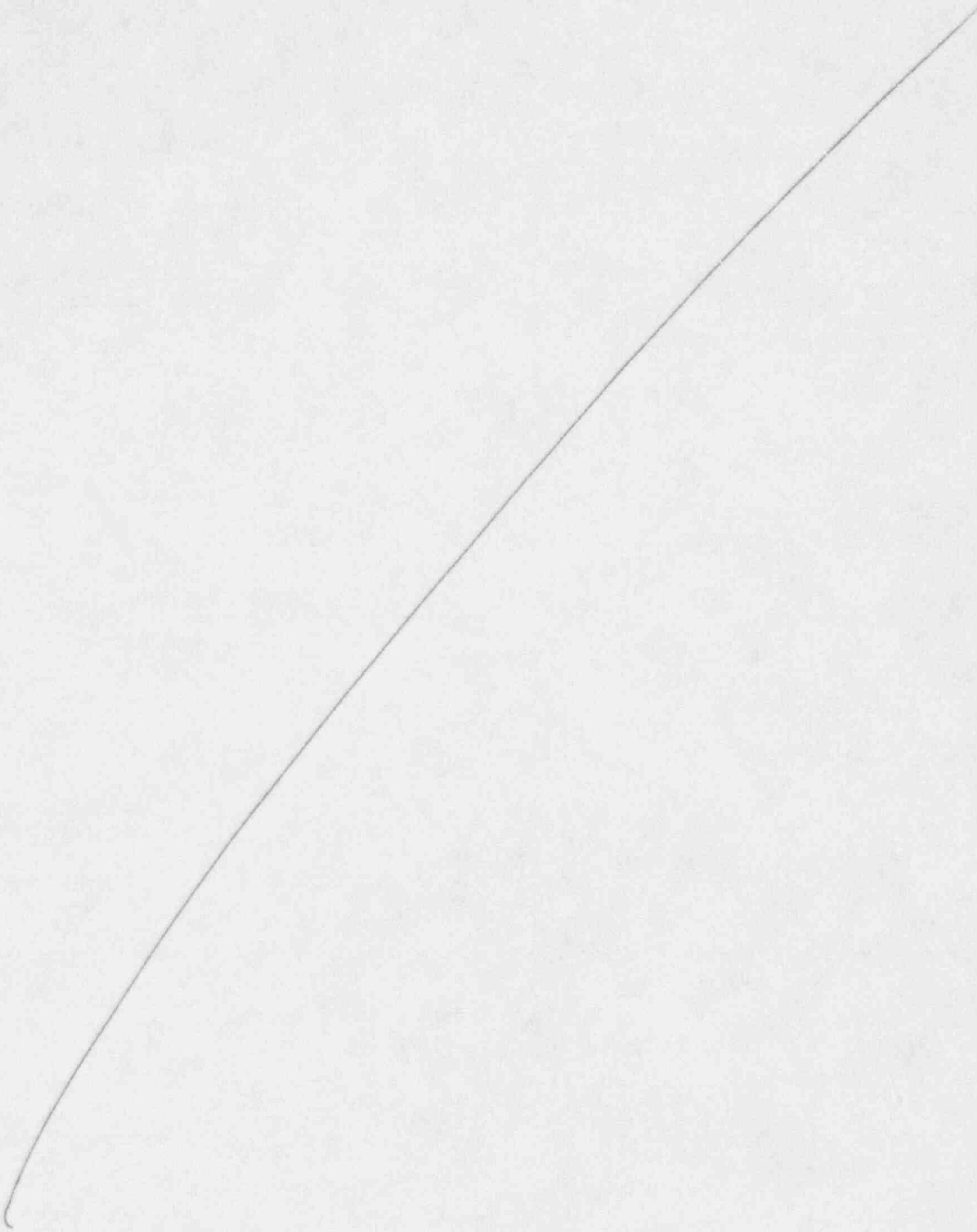


Figure 15.3.2-1

Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



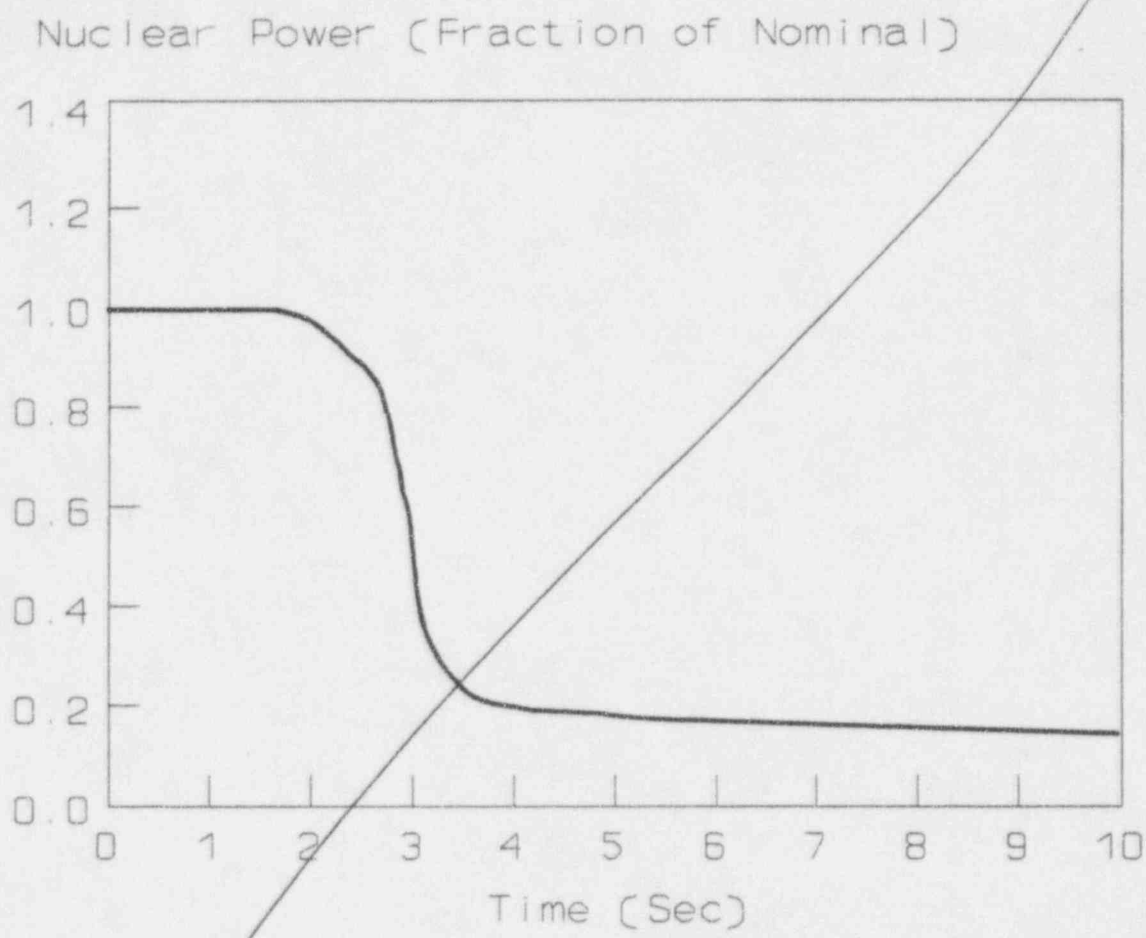
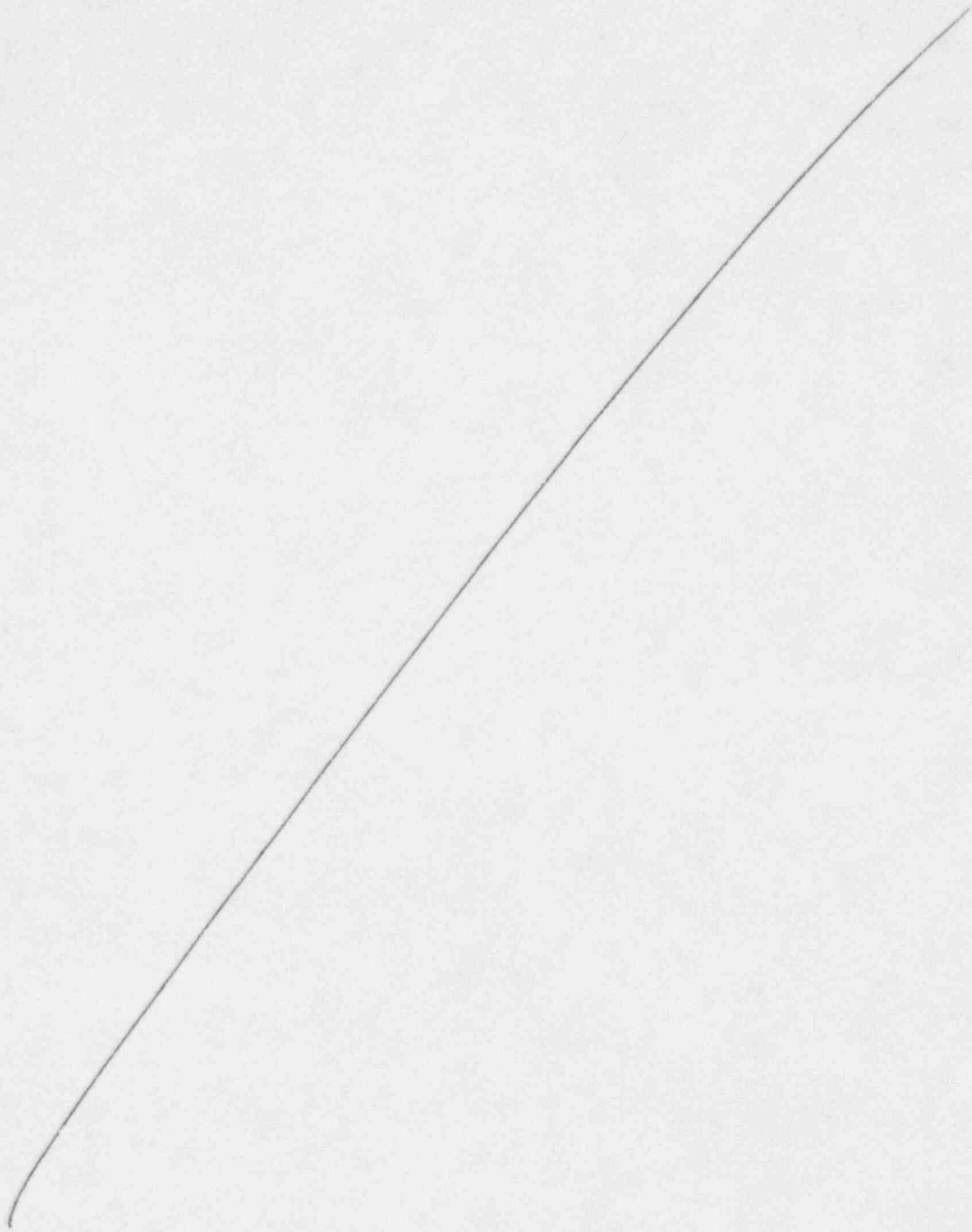


Figure 15.3.2-2 A

Nuclear Power Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



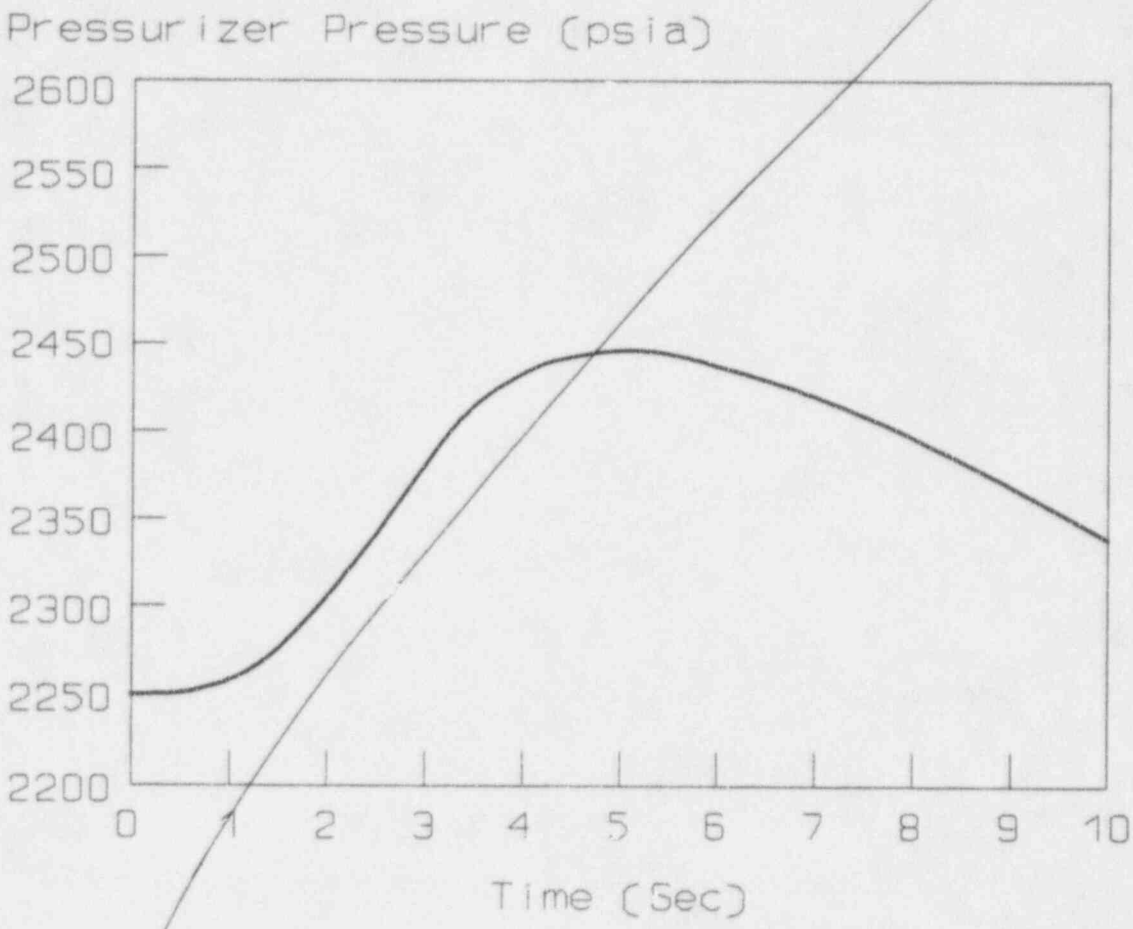


Figure 15.3.2-2 B

Pressurizer Pressure Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

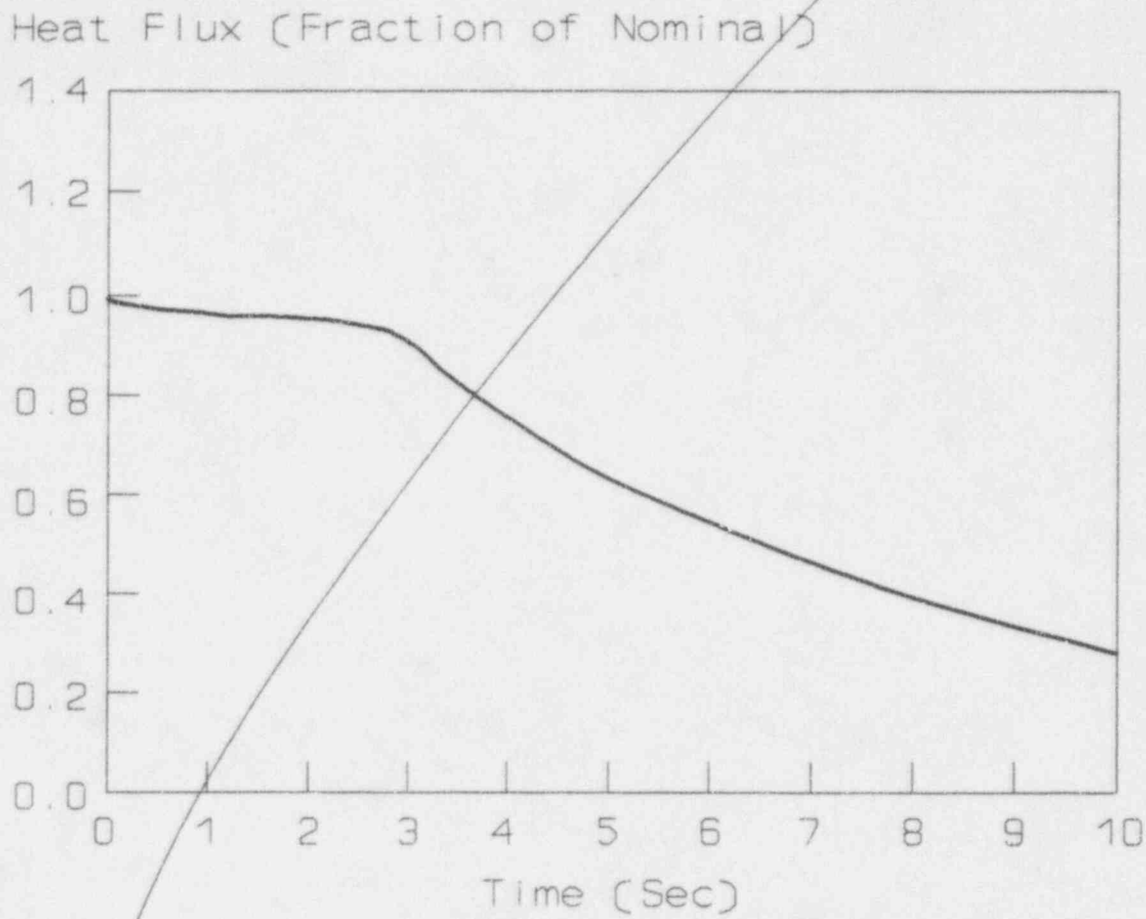
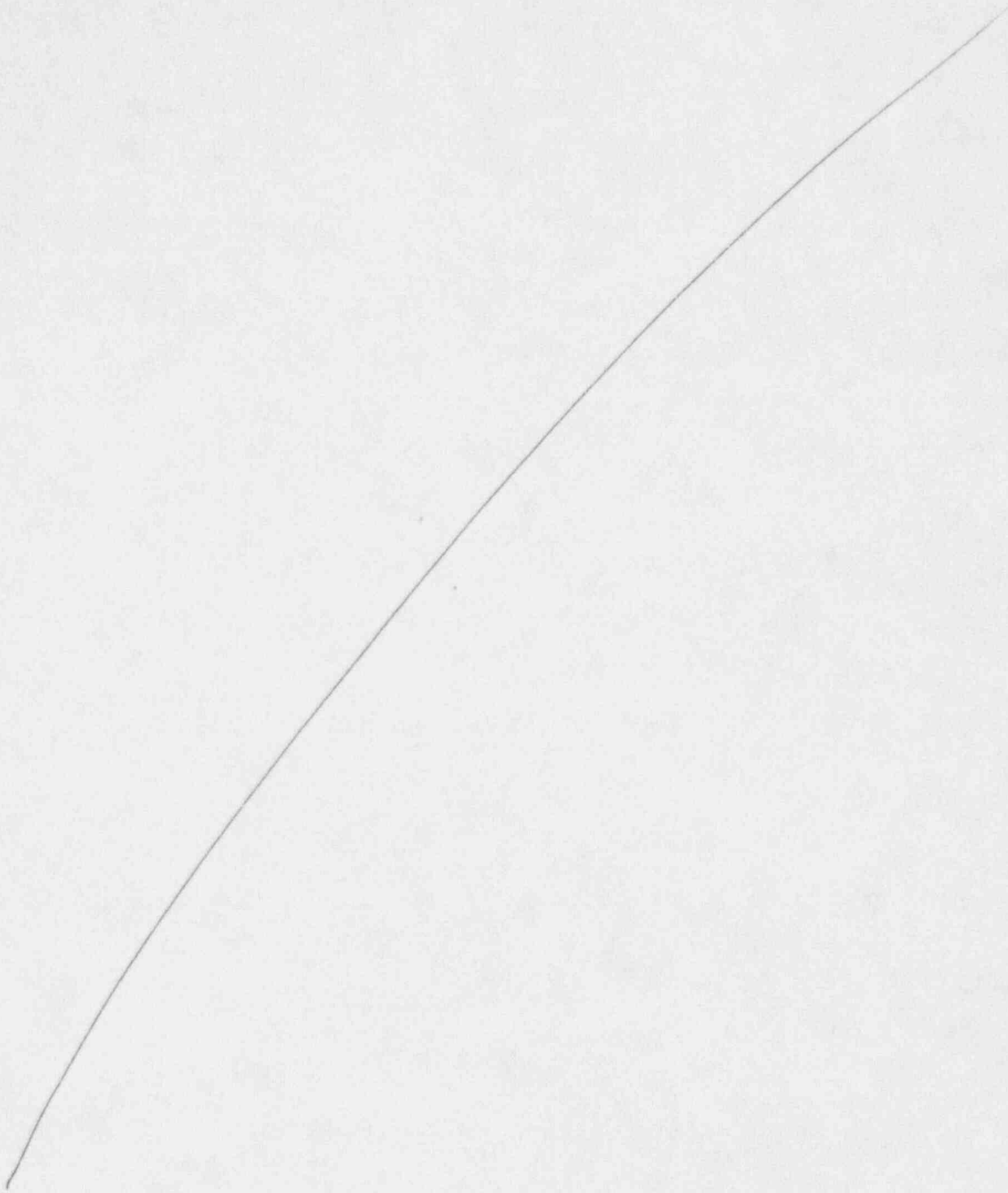


Figure 15.3.2-3 A

Average Channel Heat Flux Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



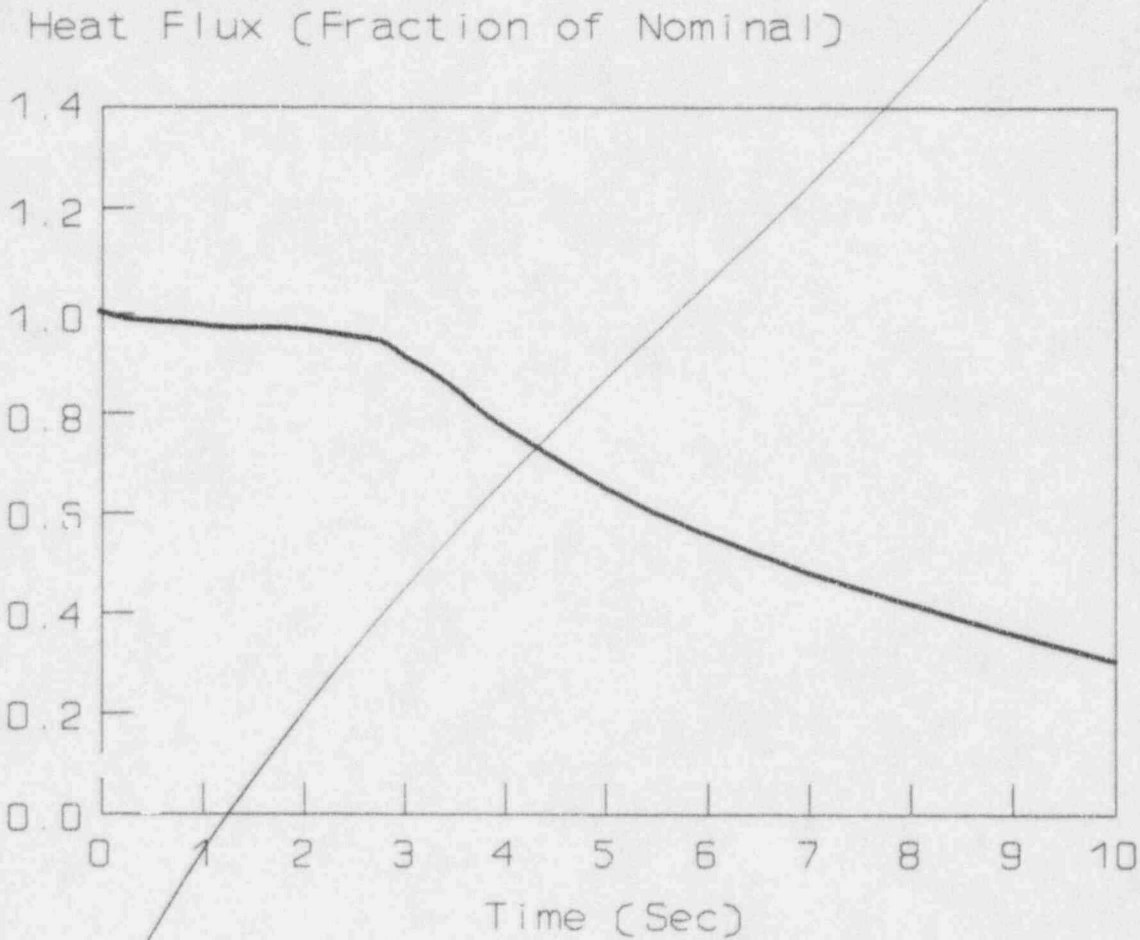
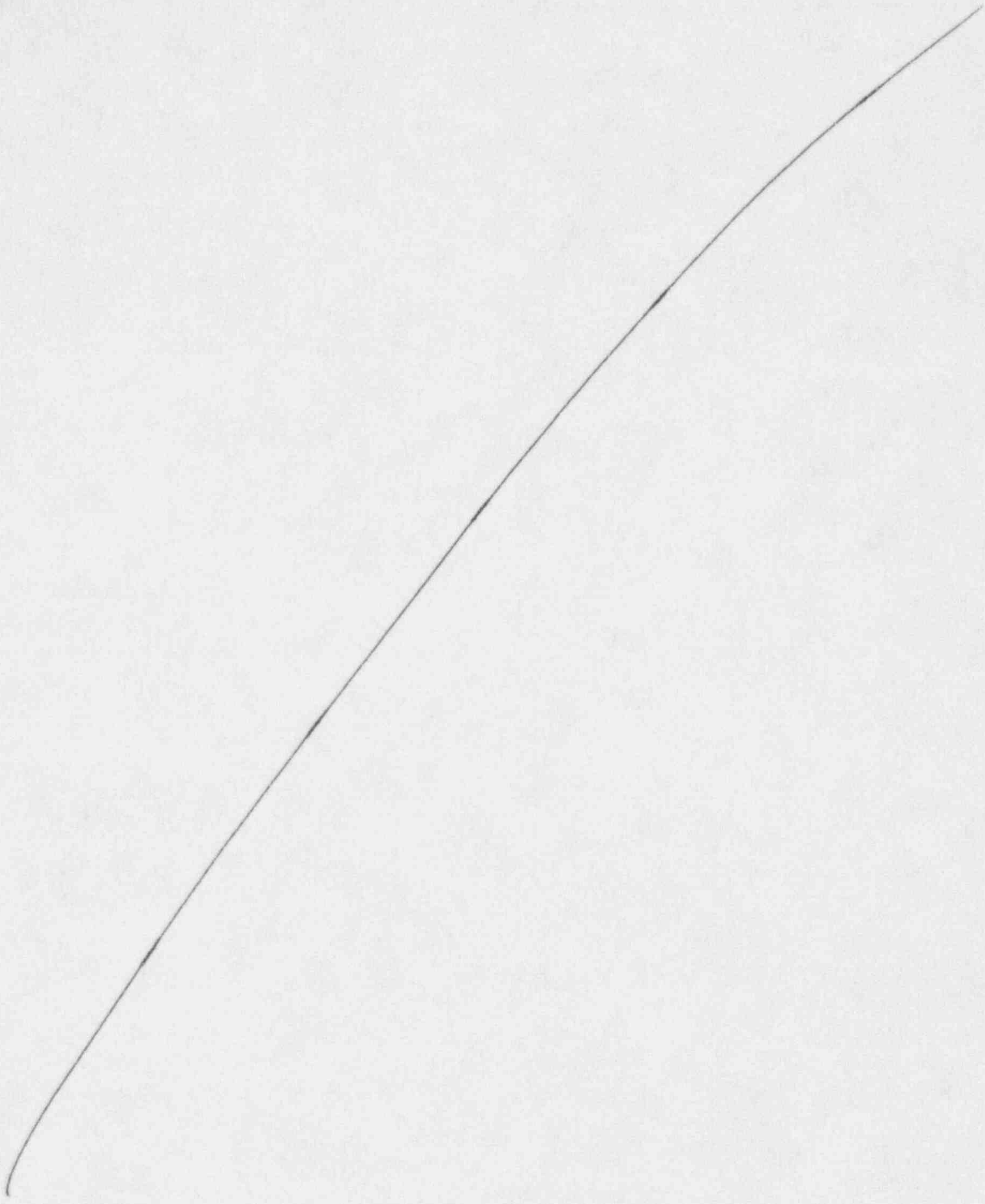


Figure 15.3.2-3 B

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Four Pumps Coasting Down



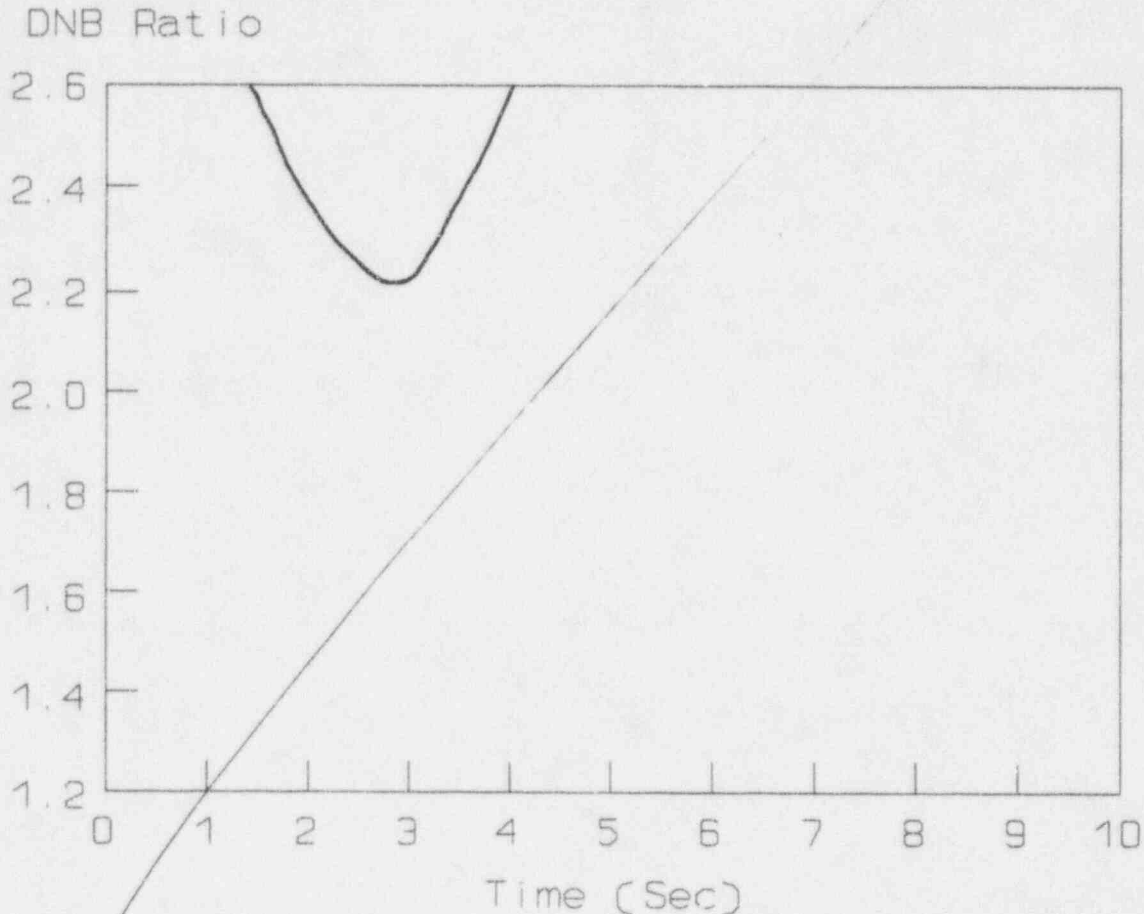
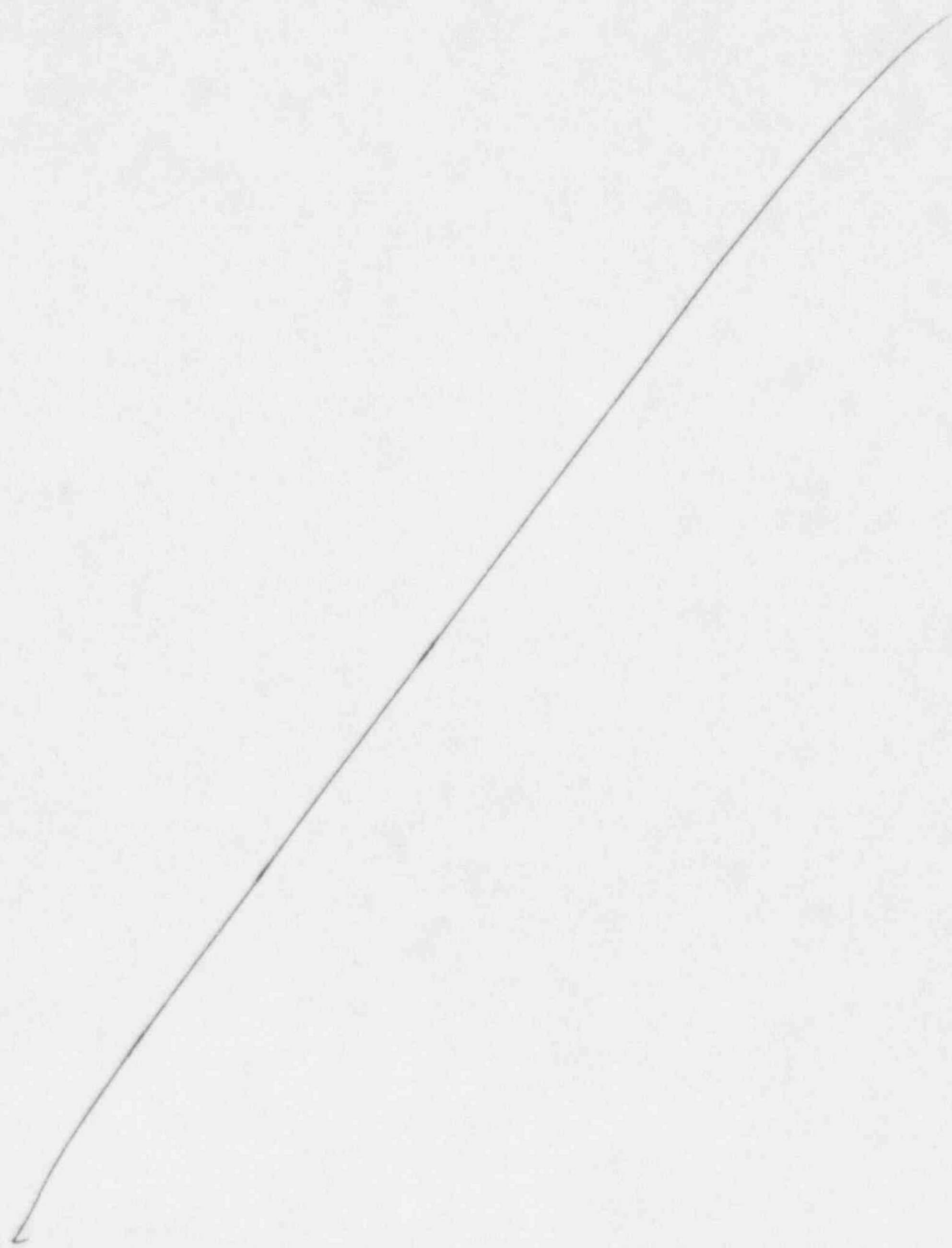


Figure 15.3.2-4

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



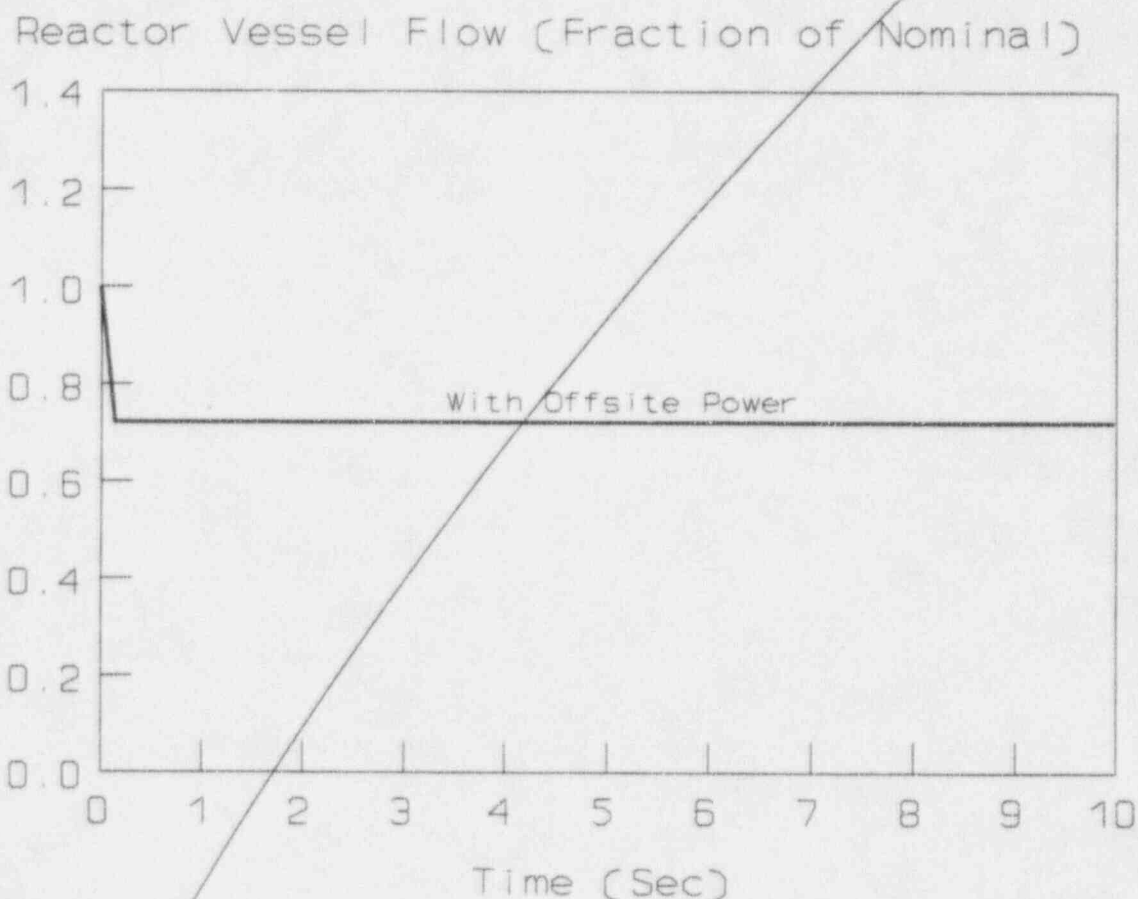
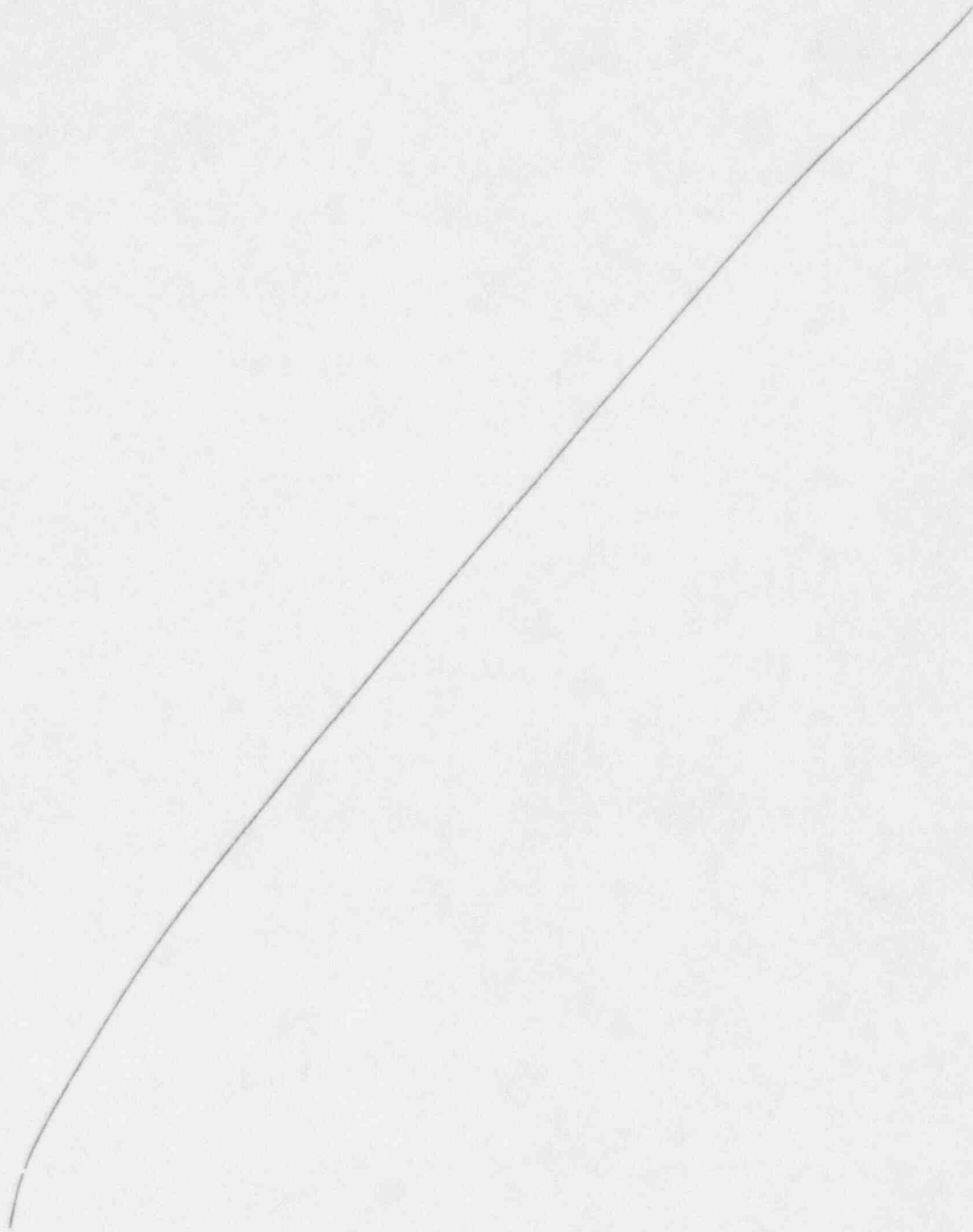


Figure 15.3.3-1 A

Flow Transient for Four Cold Legs in Operation, One Locked Rotor



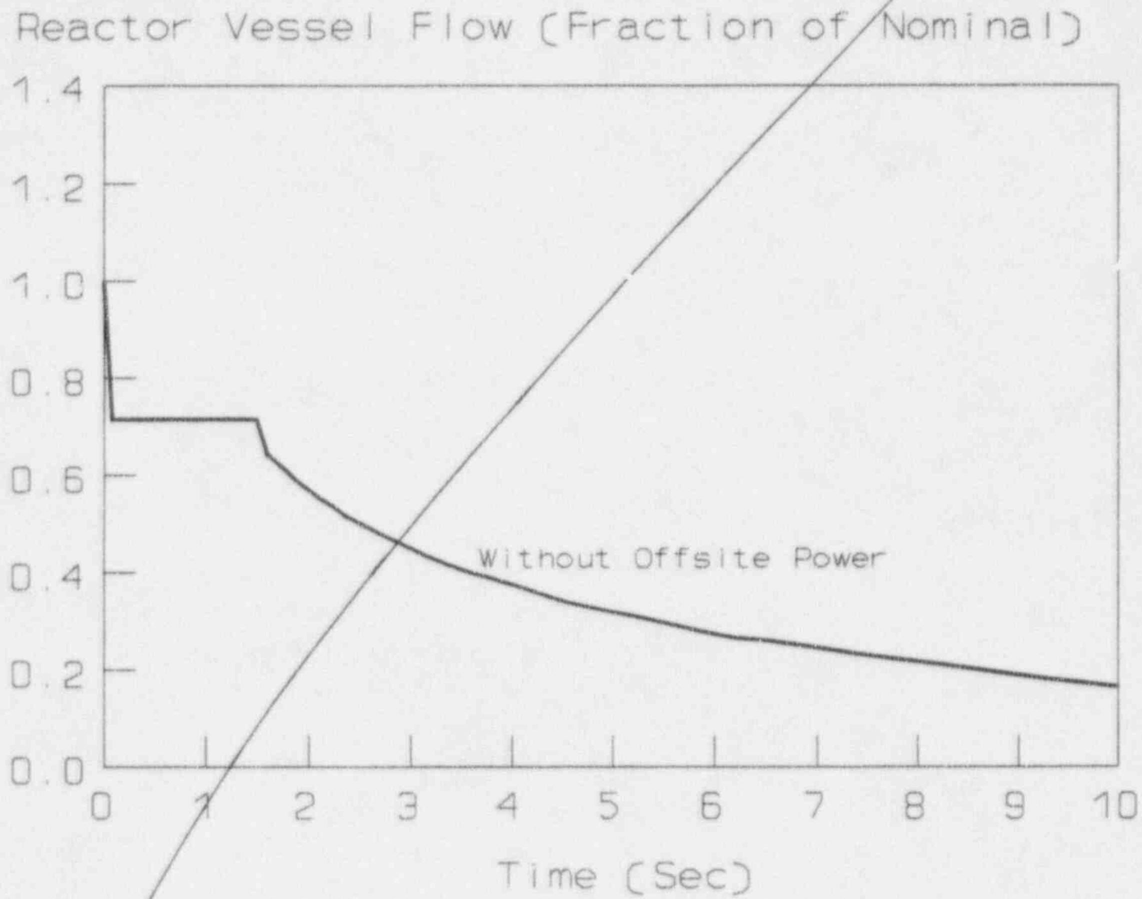
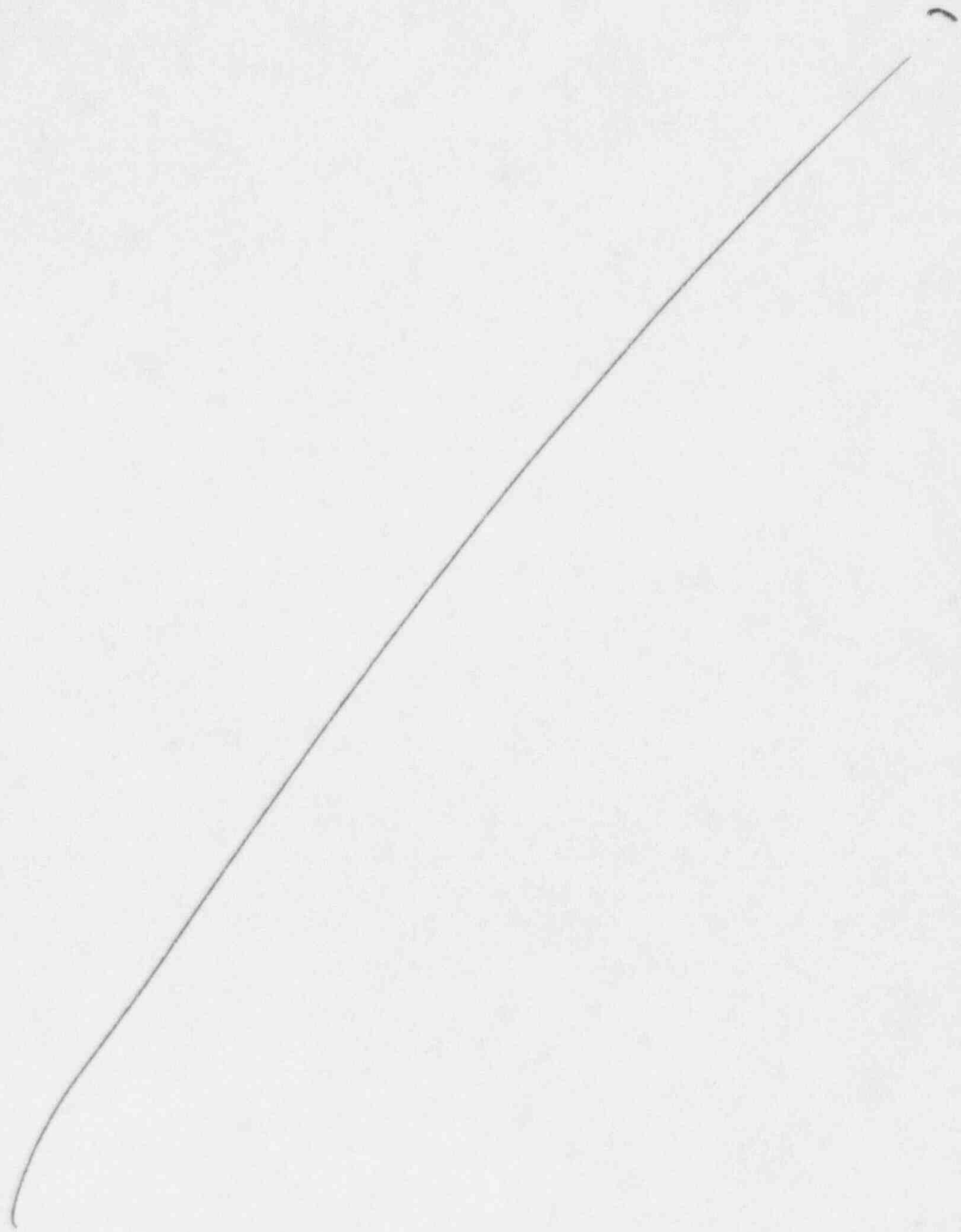


Figure 15.3.3-1 B

Flow Transient for Four Cold Legs in Operation, One Locked Rotor



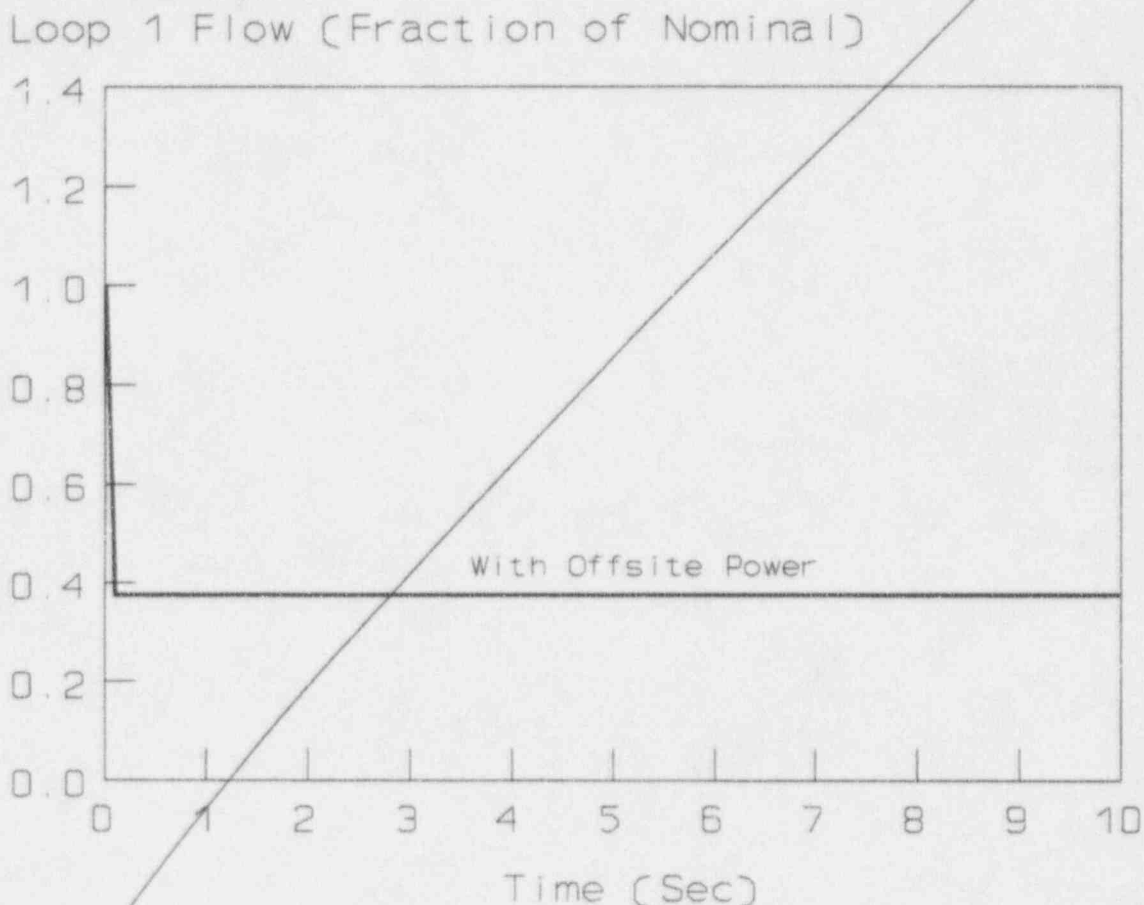


Figure 15.3.3-1 C

Flow Transient for
Four Cold Legs in Operation, One Locked Rotor

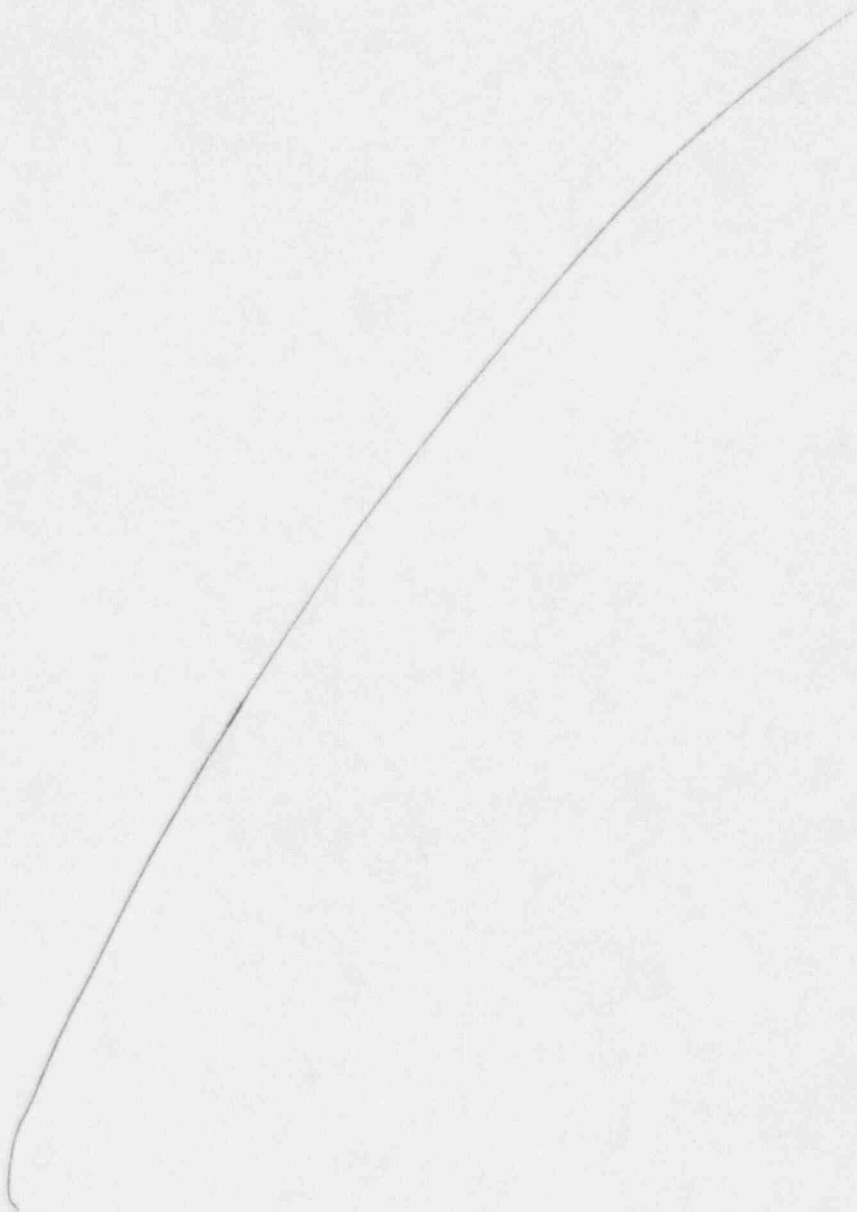


figure 15.3.3-1 was messed up

Figure 15.3.1-1
Flow Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

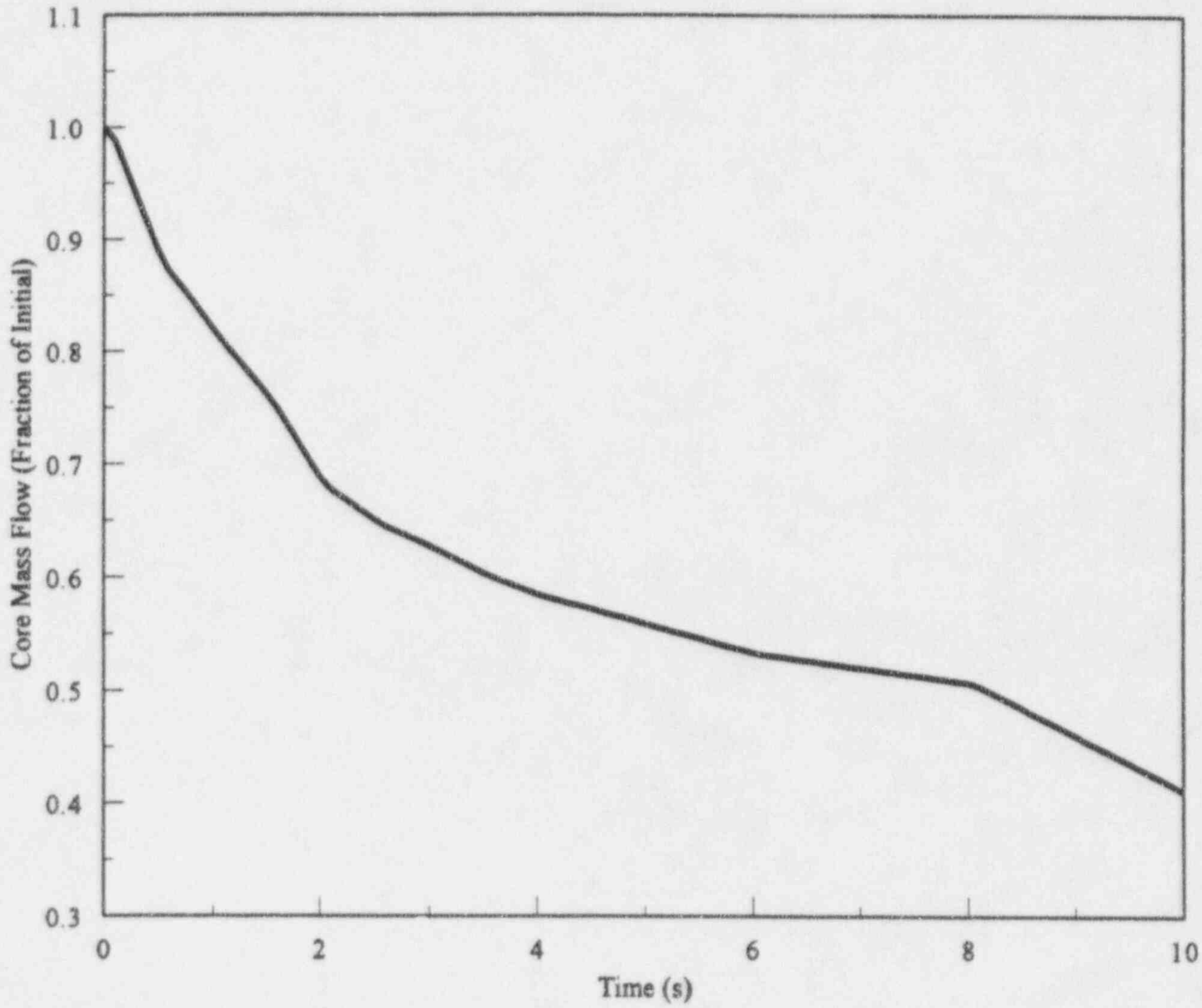


Figure 15.3.1-2
Nuclear Power Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

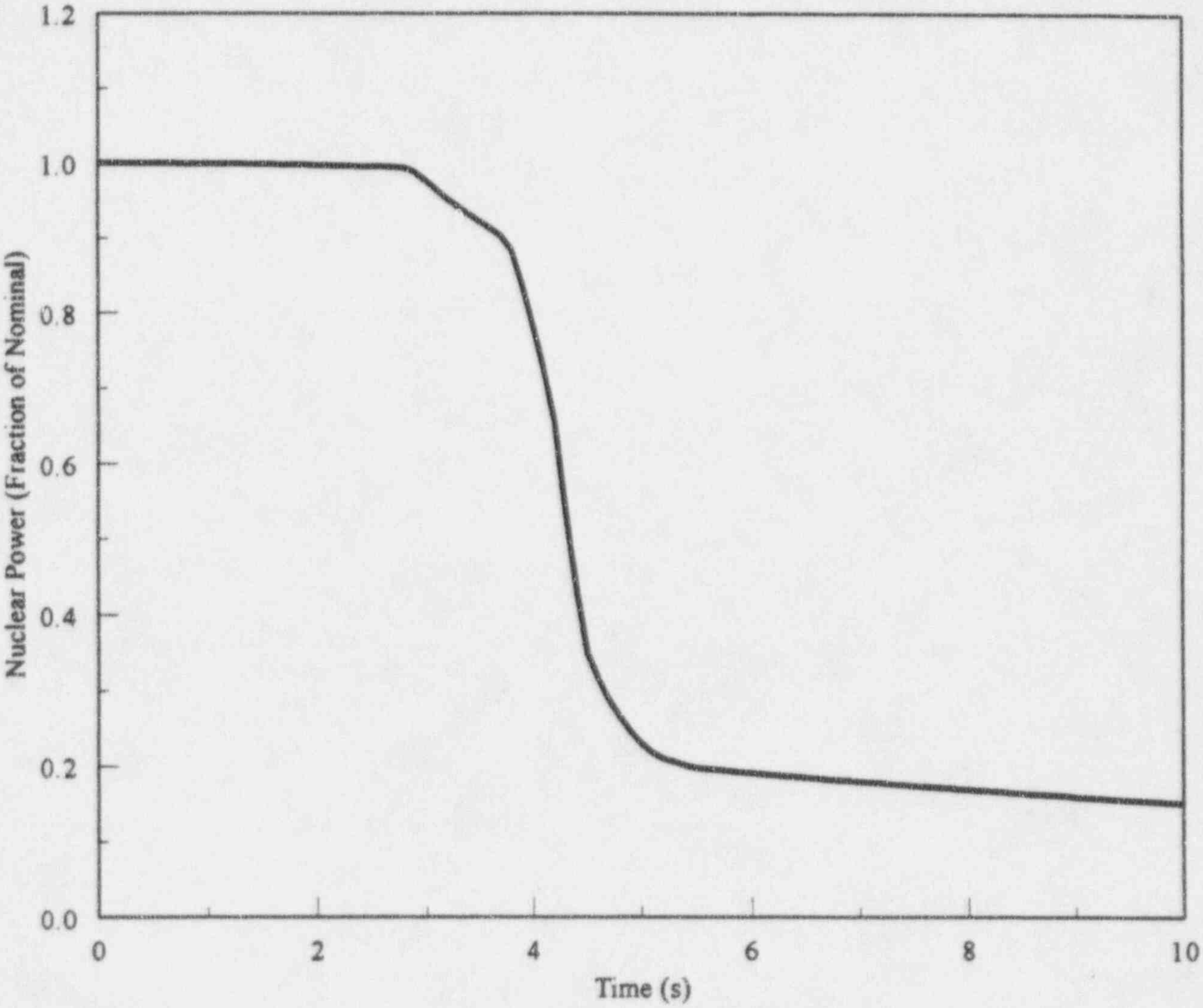


Figure 15.3.1-3
Pressurizer Pressure Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

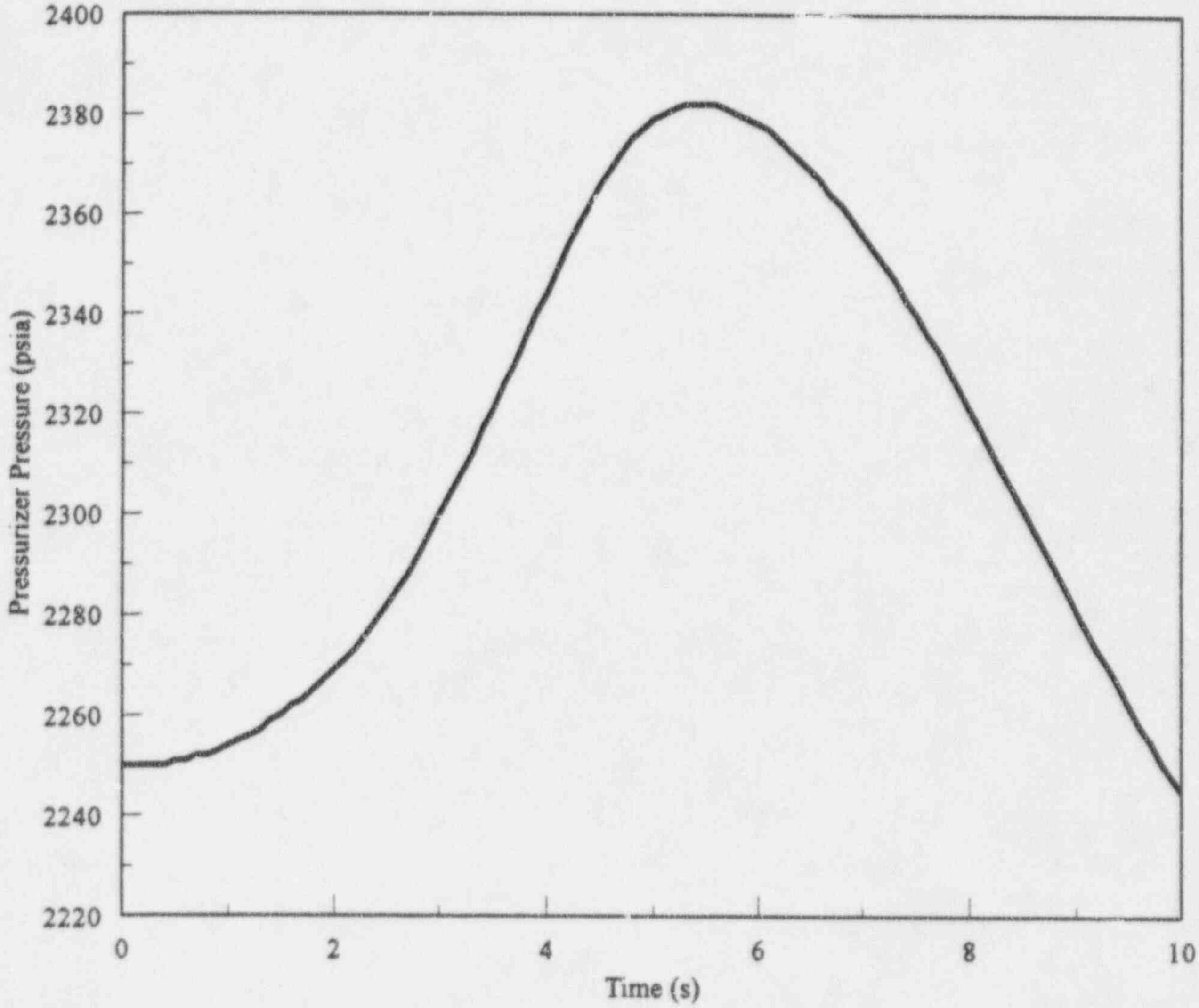


Figure 15.3.1-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

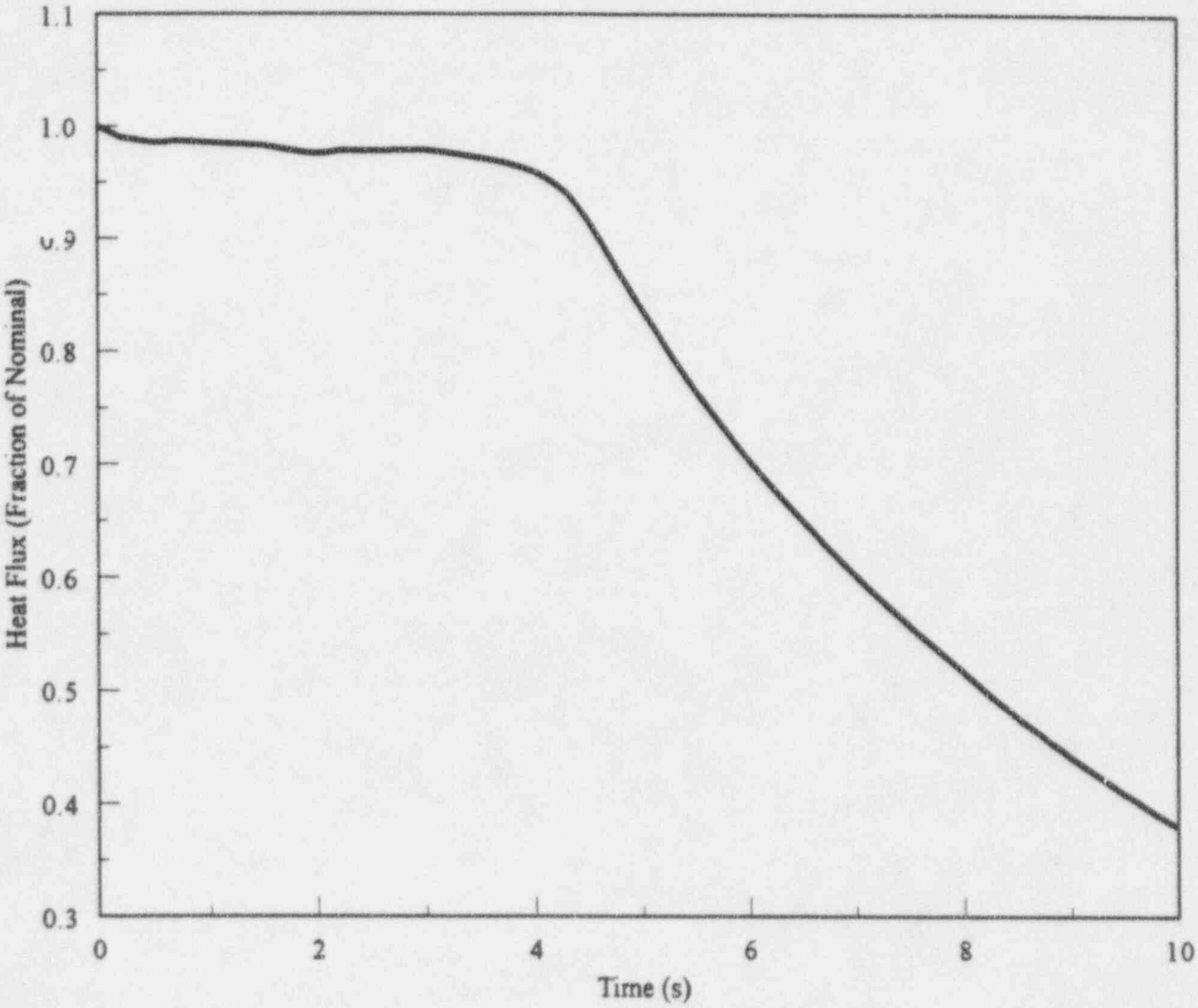


Figure 15.3.1-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

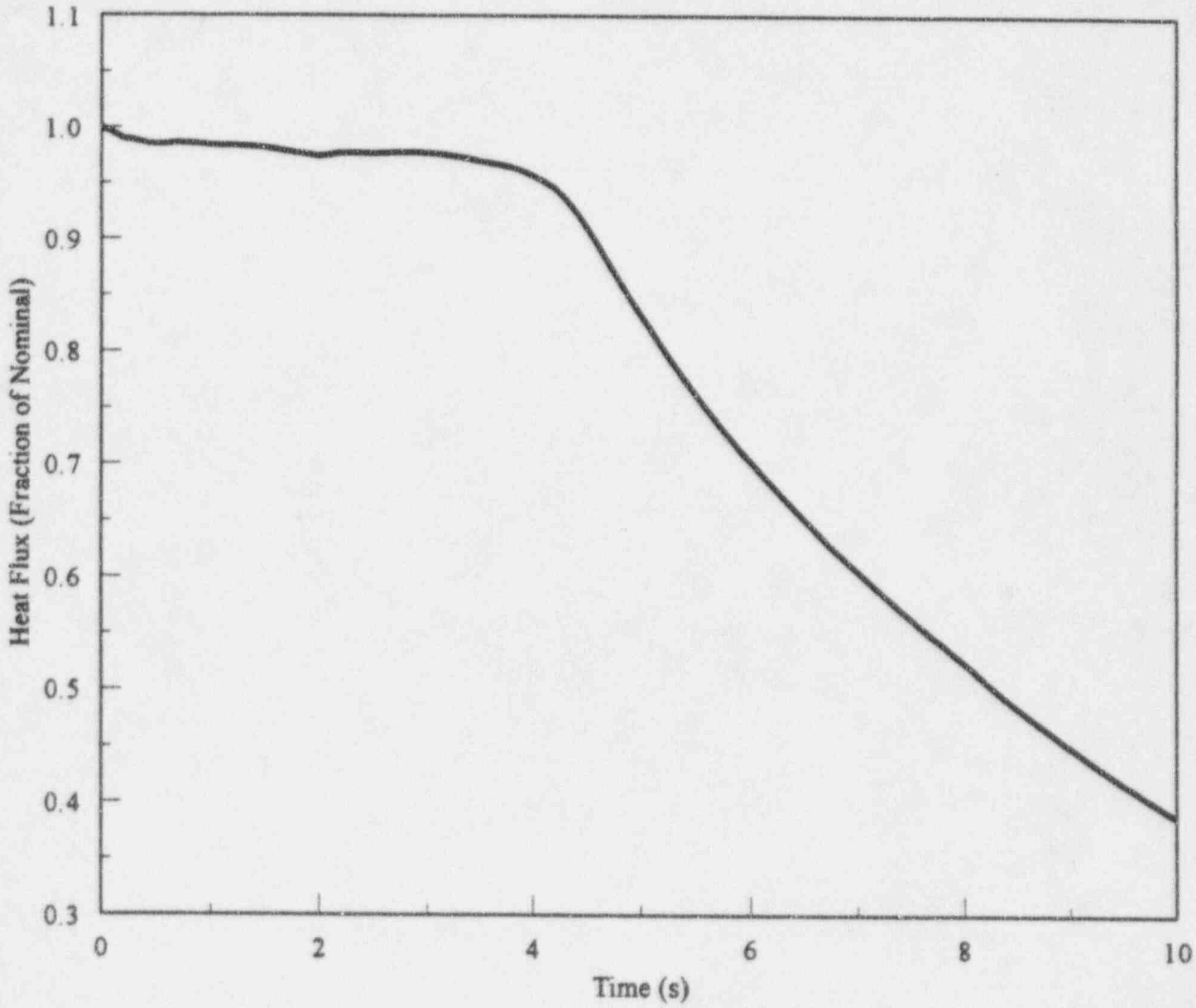


Figure 15.3.1-6
DNBR Transient for
Four Cold Legs in Operation,
Two Pumps Coasting Down

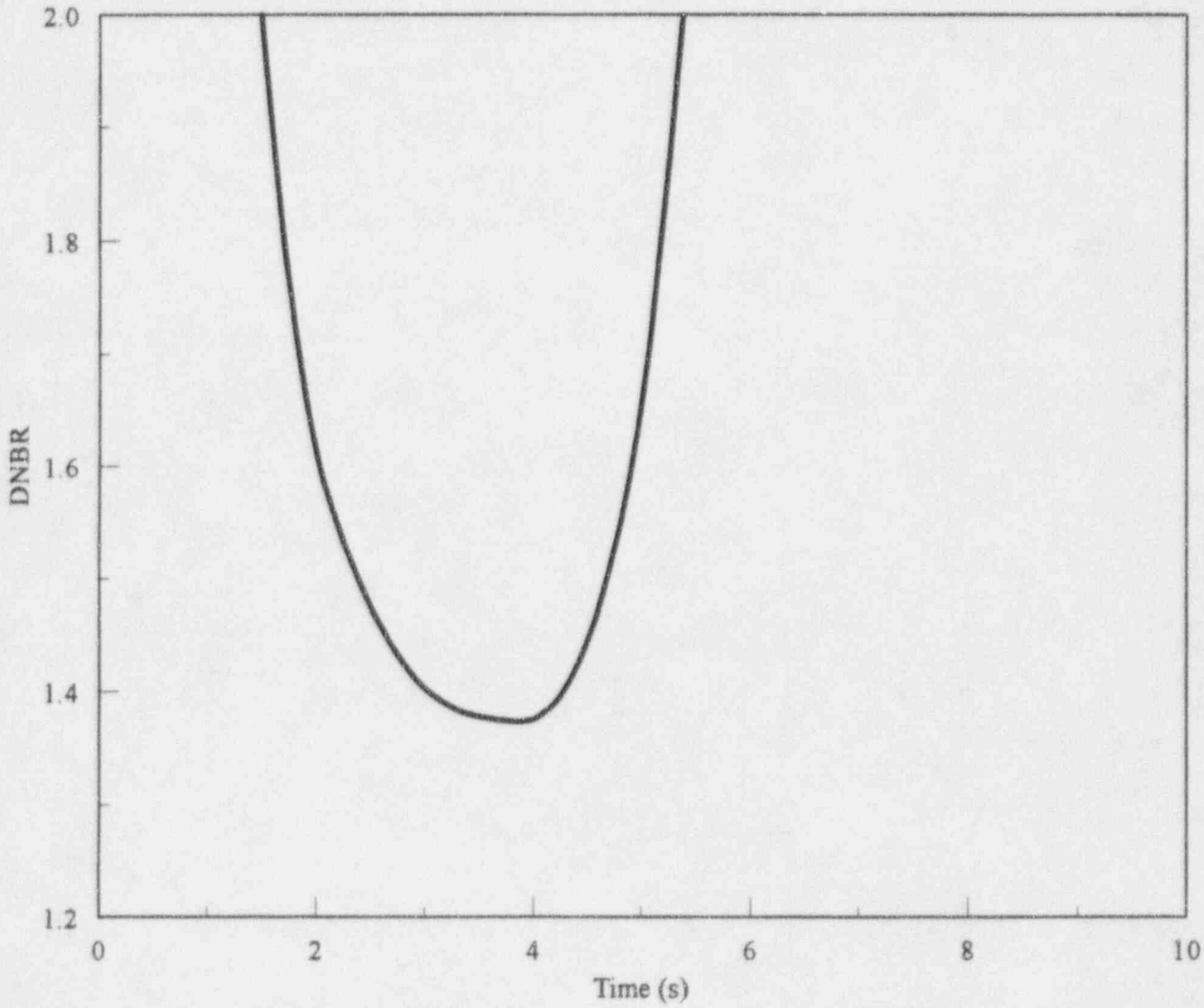


Figure 15.3.2-1
Flow Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

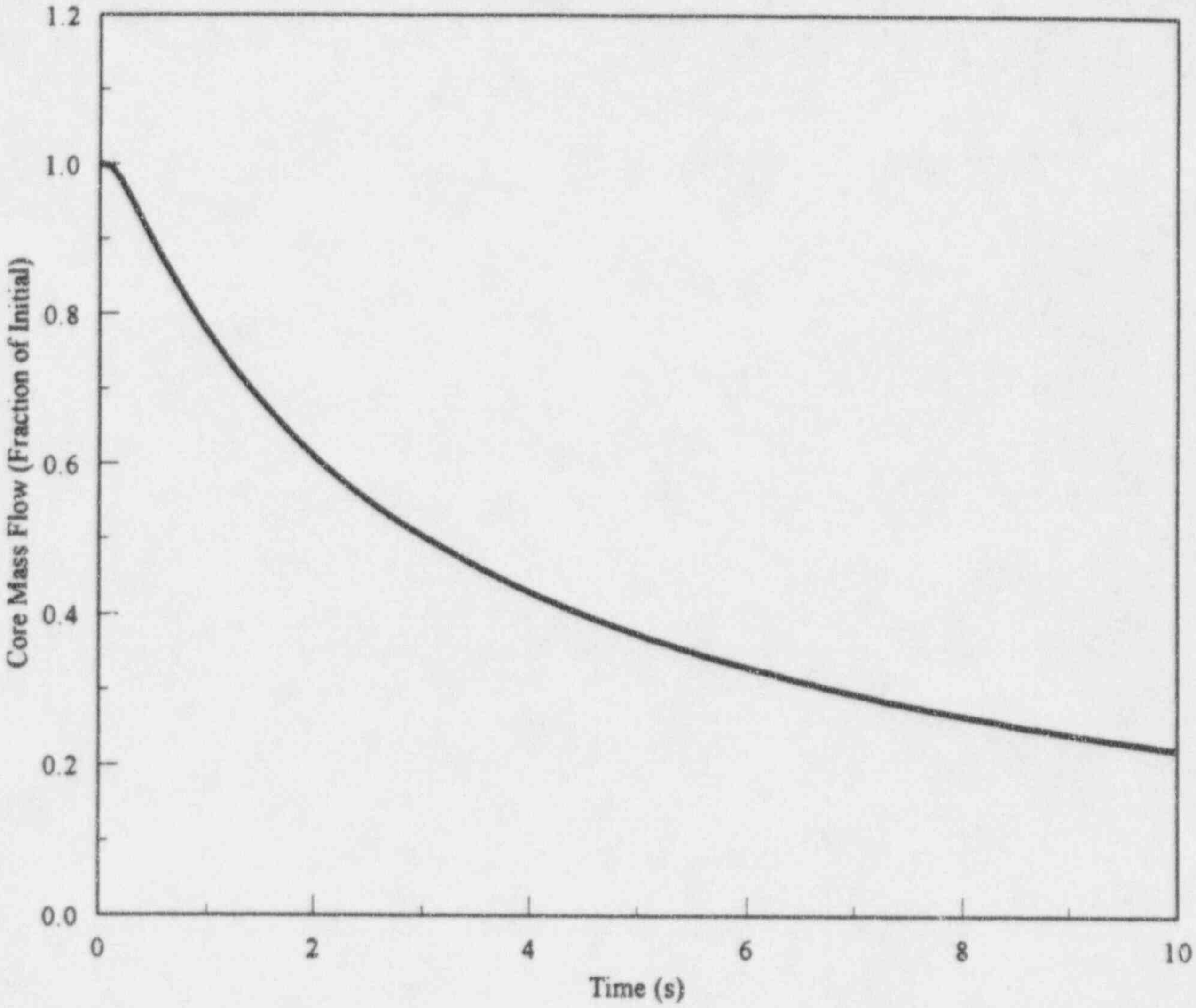


Figure 15.3.2-2
Nuclear Power Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

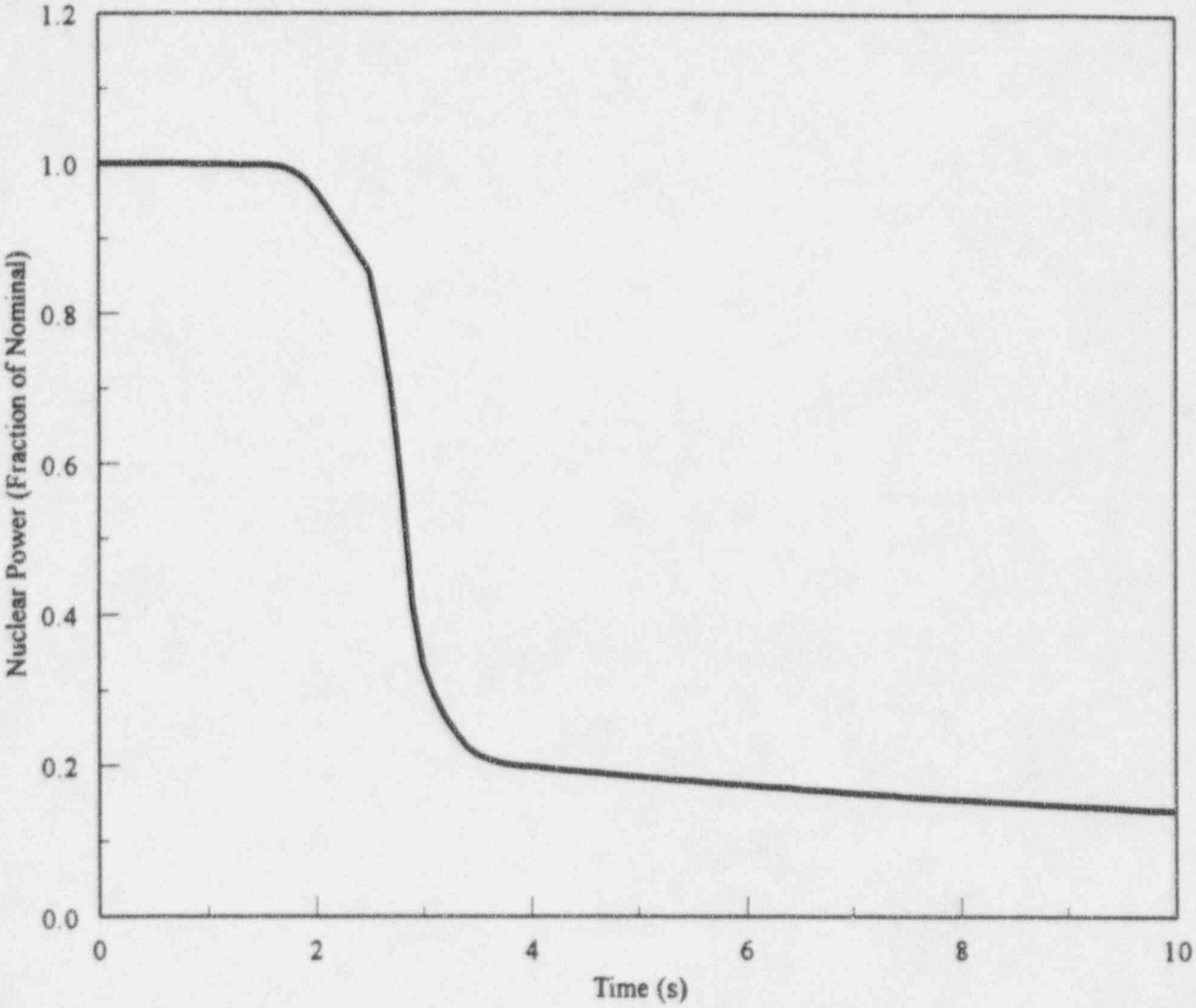


Figure 15.3.2-3
Pressurizer Pressure Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

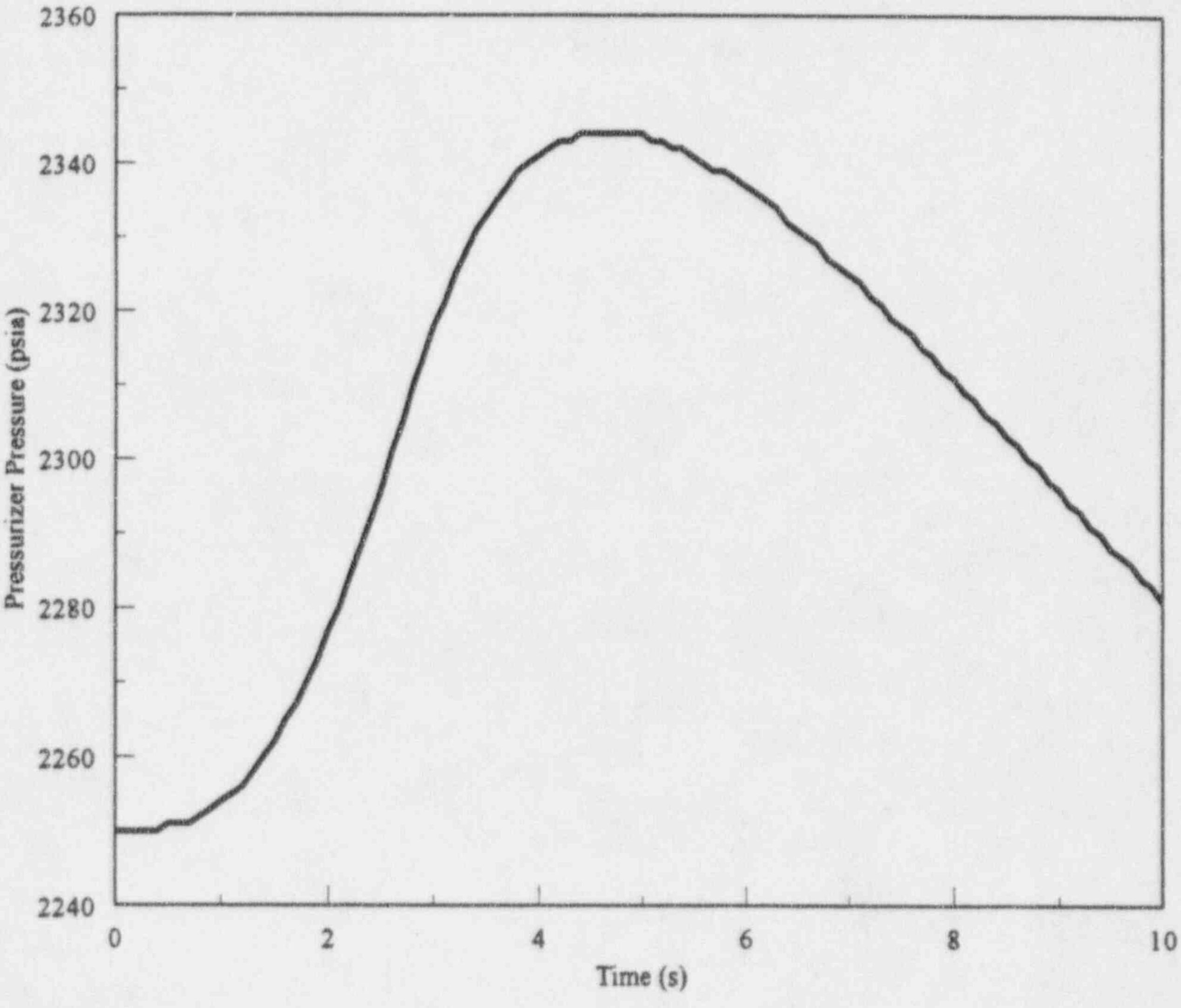


Figure 15.3.2-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

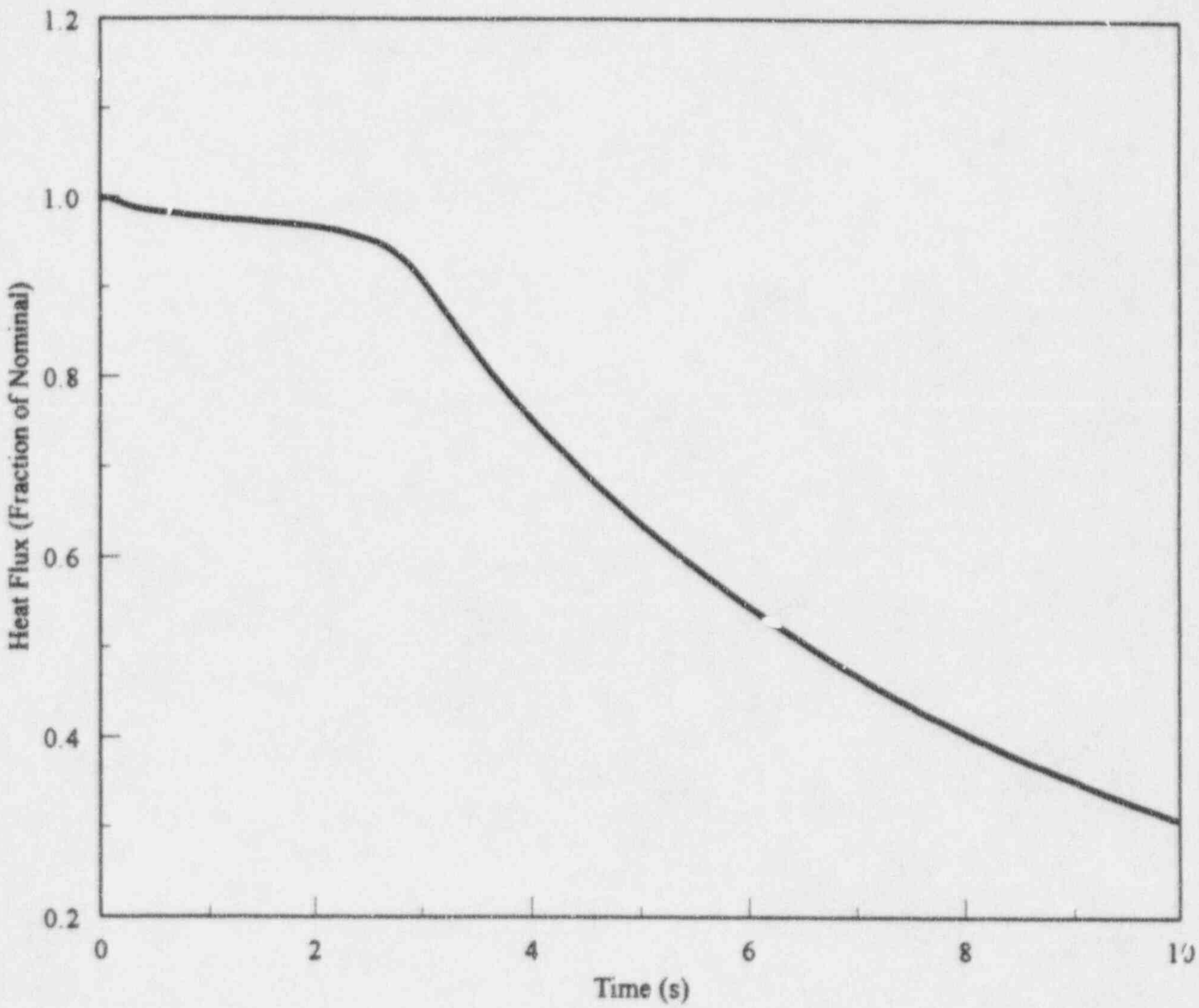


Figure 15.3.2-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

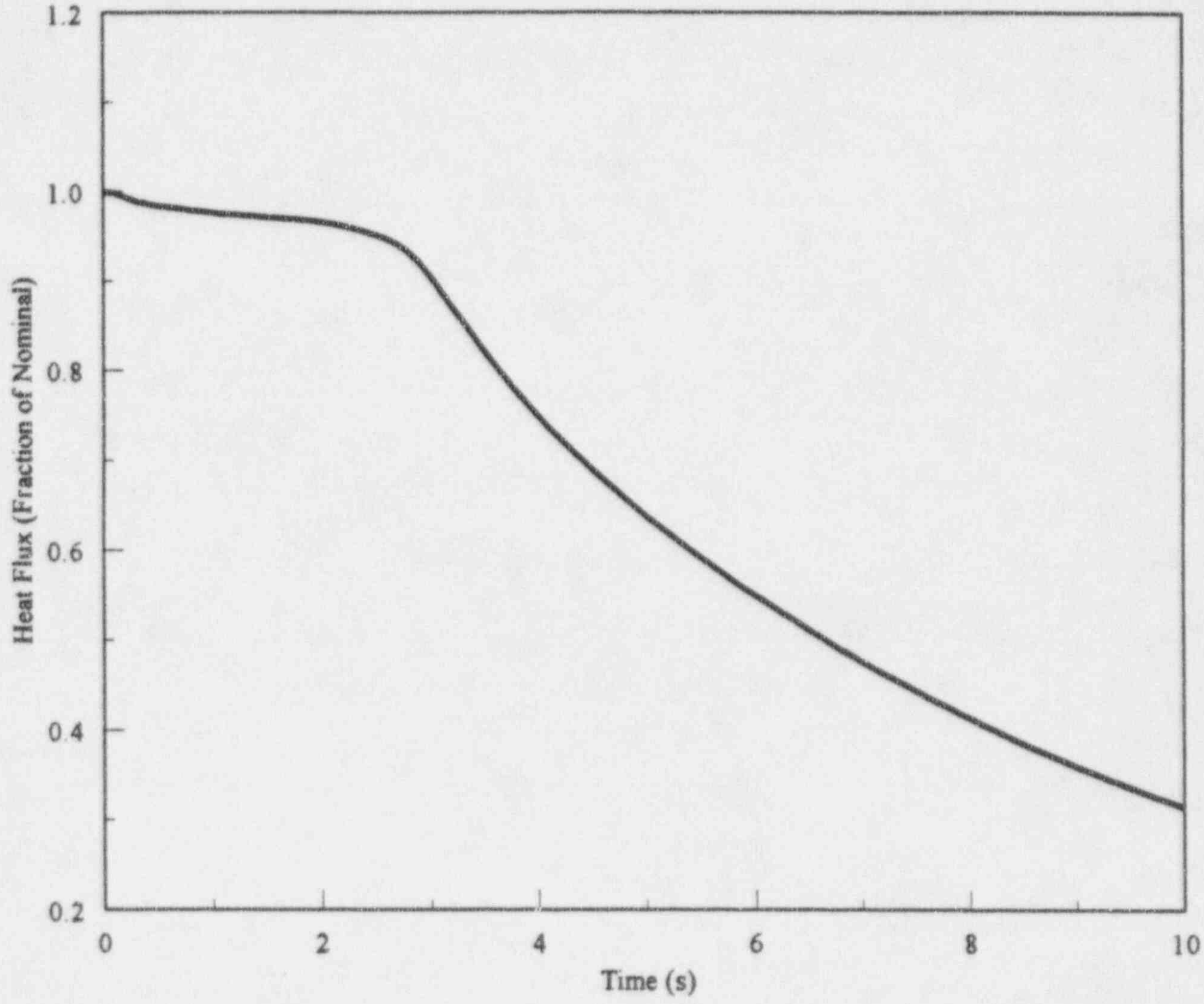


Figure 15.3.2-6
DNBR Transient for
Four Cold Legs in Operation,
Four Pumps Coasting Down

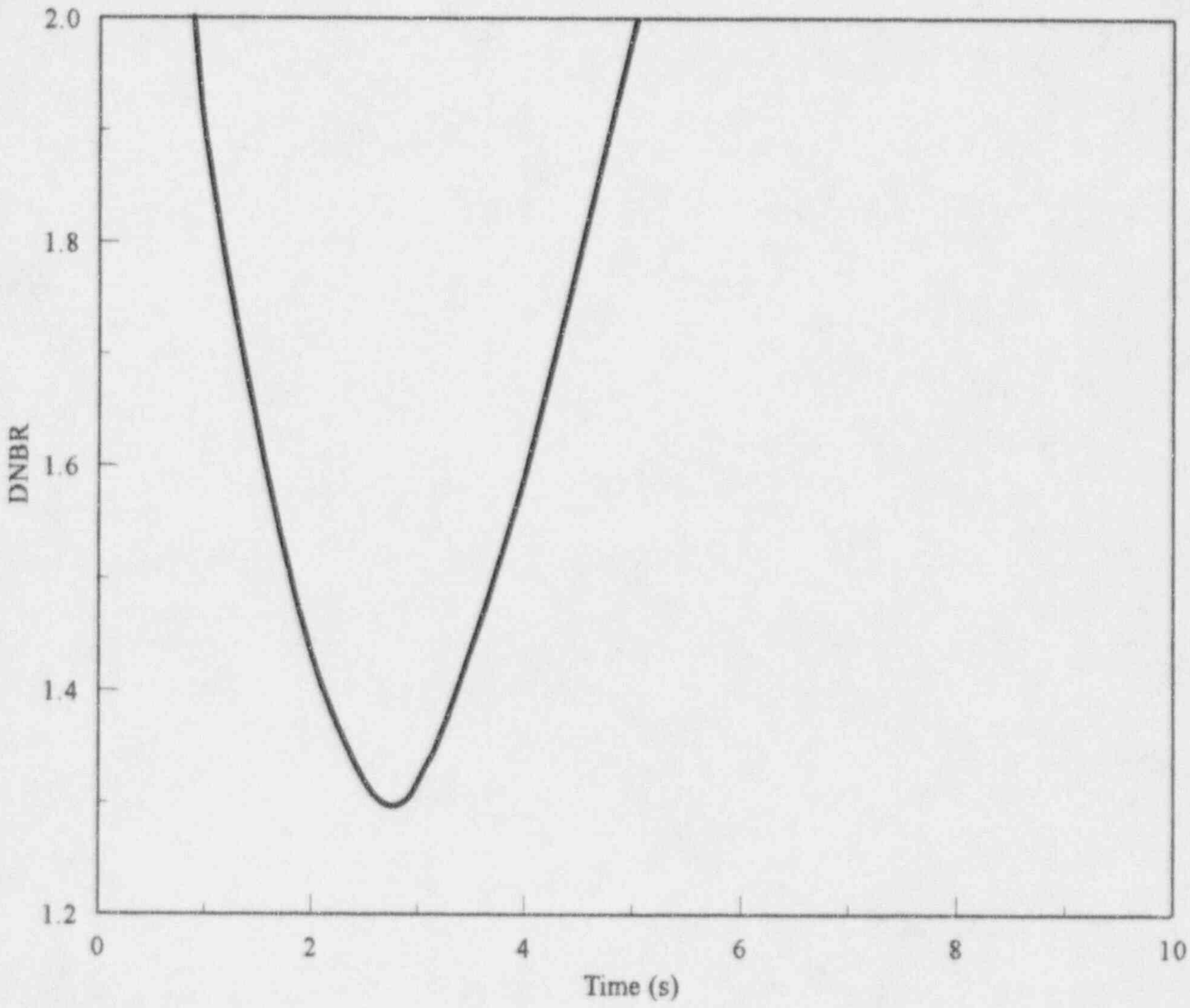


Figure 15.3.3-1
Flow Transient for
Four Cold Legs in Operation,
One Locked Rotor

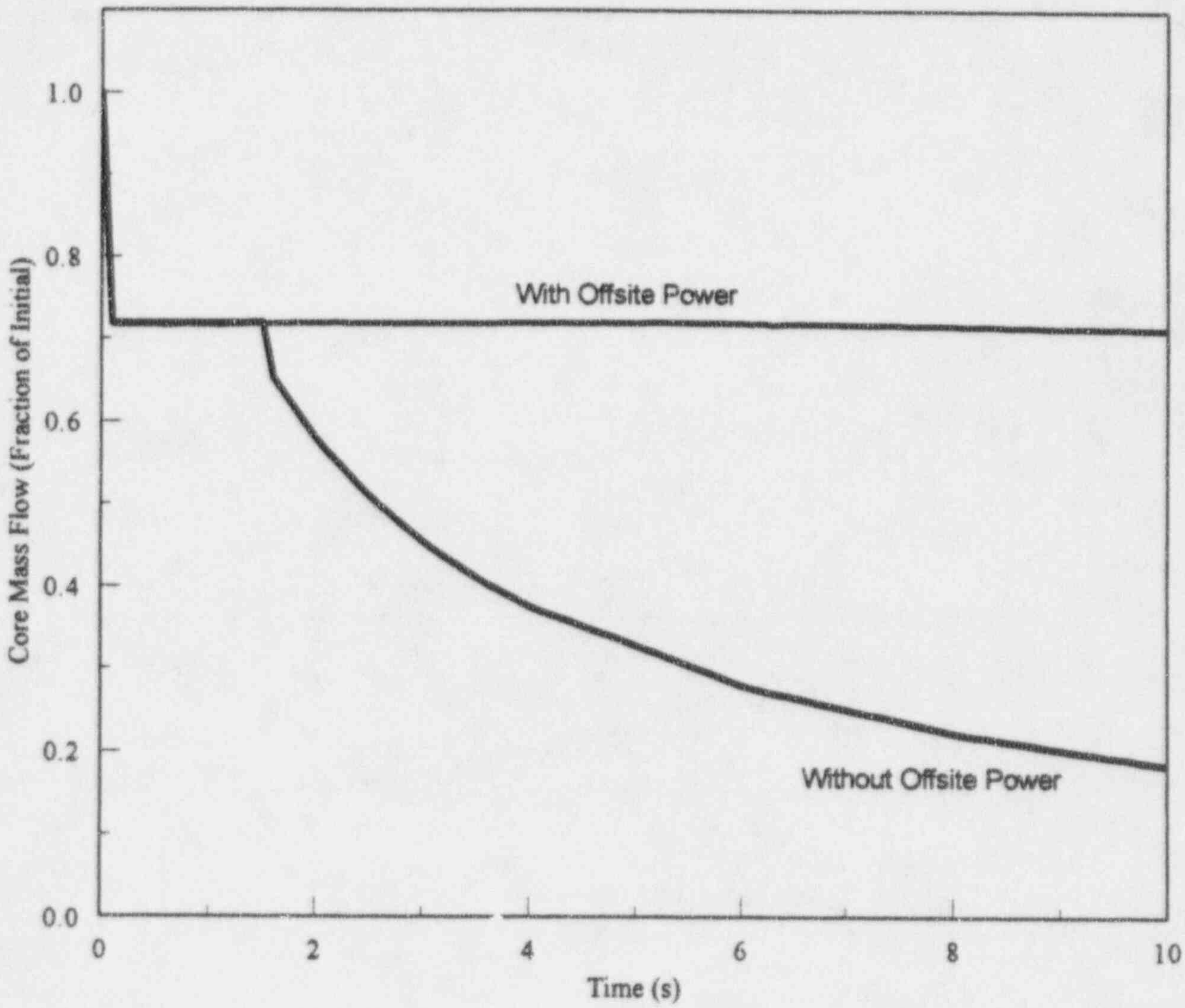


Figure 15.3.3-2
Flow Transient for
Four Cold Legs in Operation,
One Locked Rotor

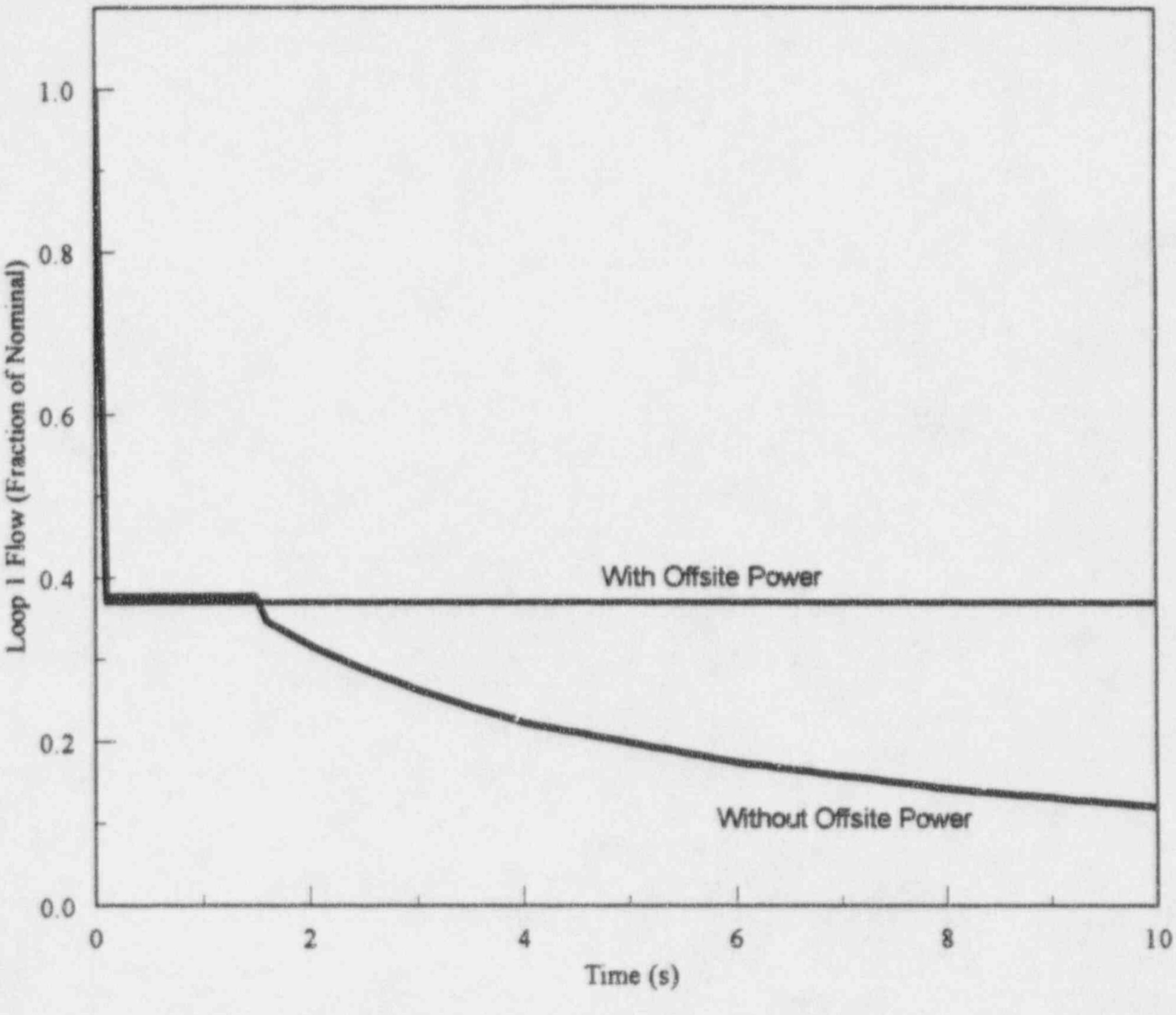


Figure 15.3.3-3
Peak Reactor Coolant Pressure Transient for
Four Cold Legs in Operation,
One Locked Rotor

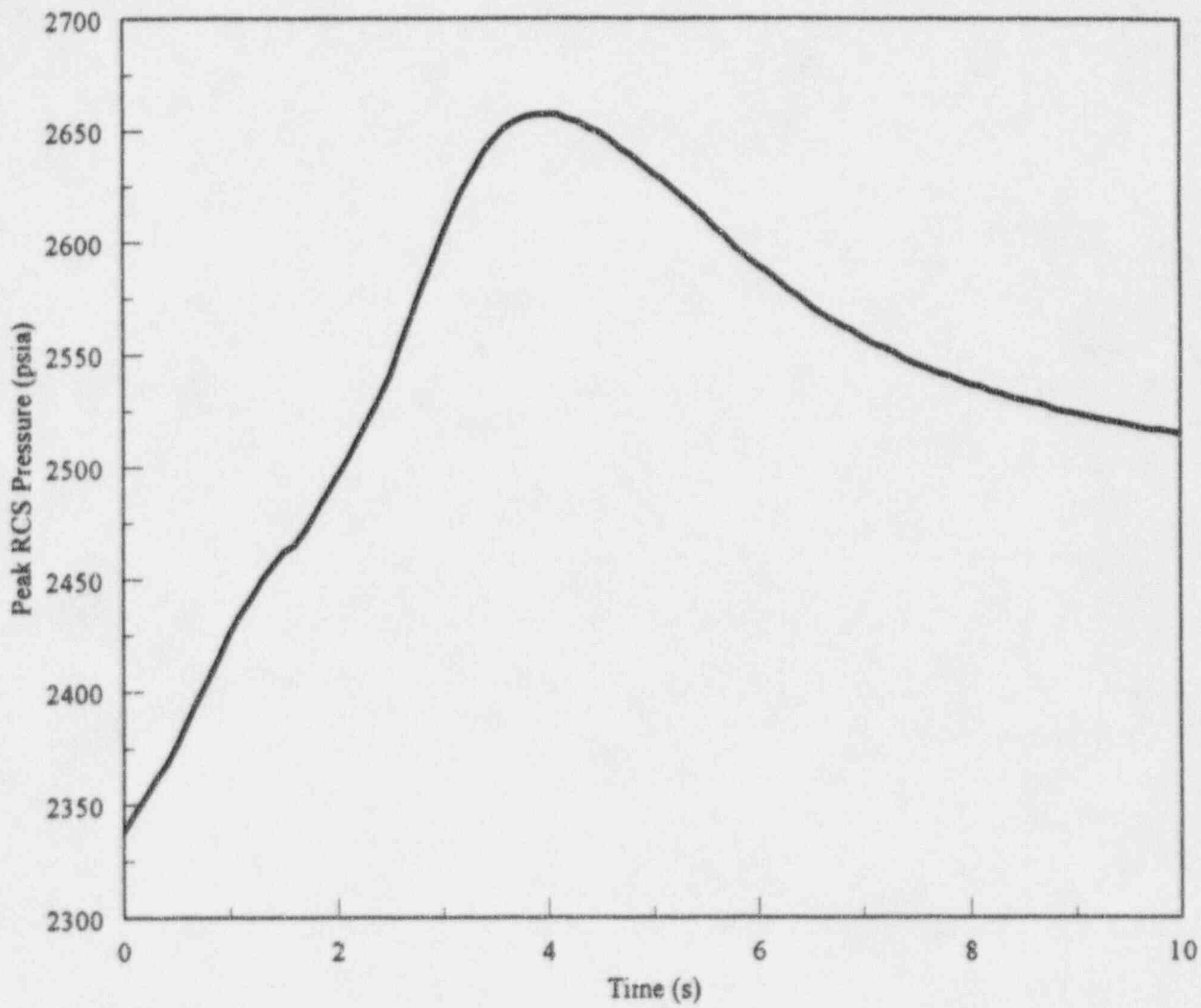


Figure 15.3.3-4
Average Channel Heat Flux Transient for
Four Cold Legs in Operation,
One Locked Rotor

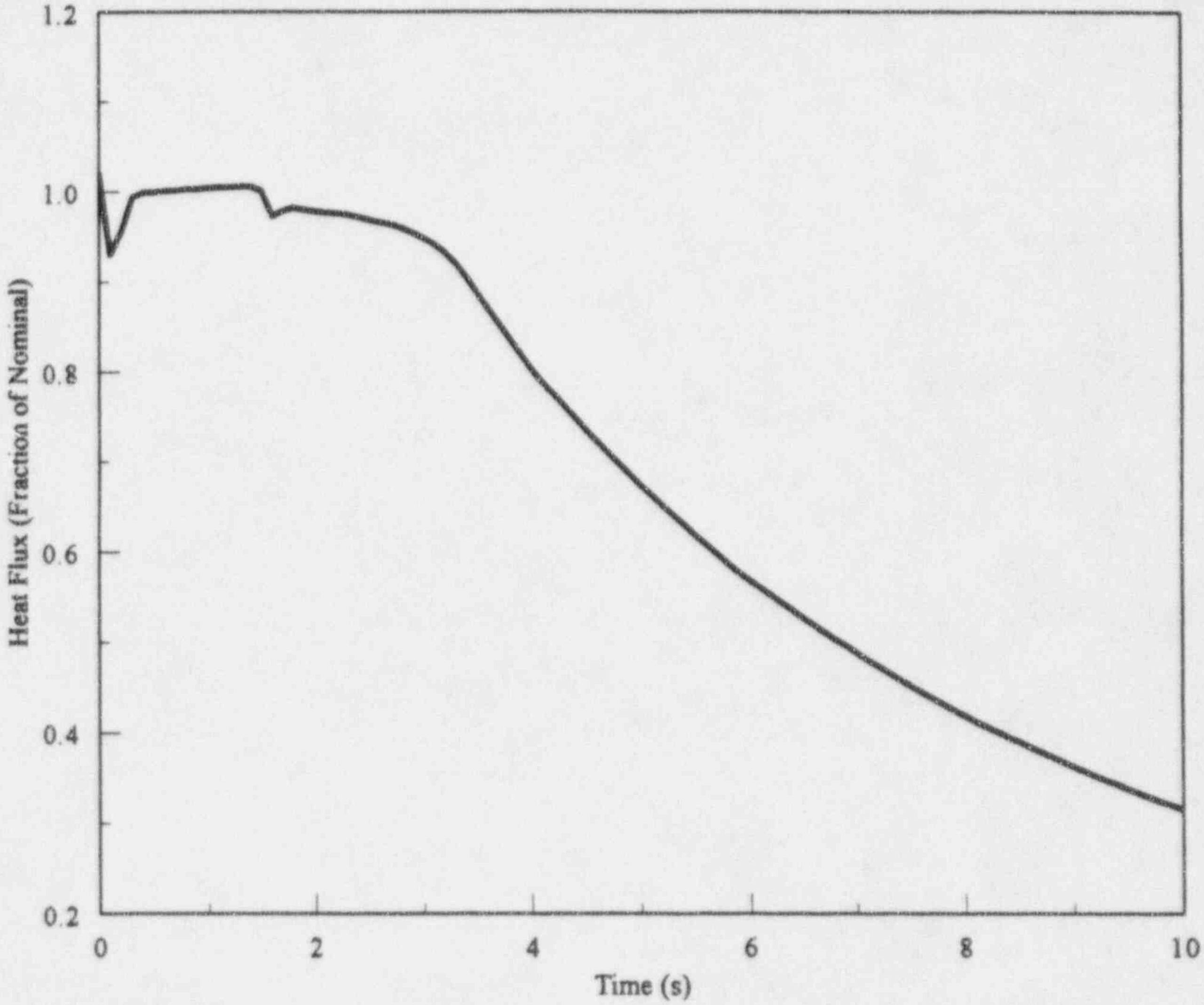


Figure 15.3.3-5
Hot Channel Heat Flux Transient for
Four Cold Legs in Operation,
One Locked Rotor

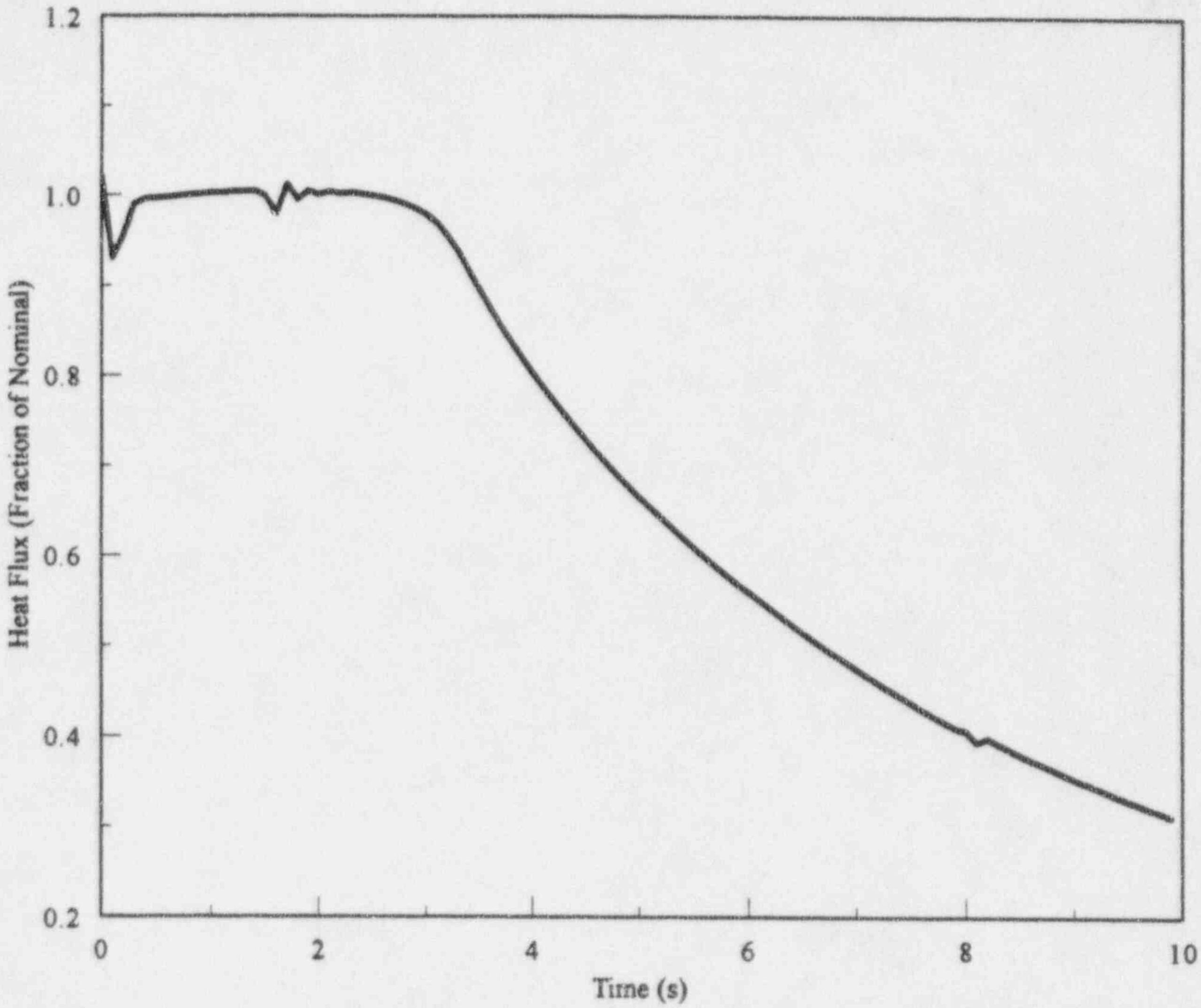


Figure 15.3.3-6
Nuclear Power Transient for
Four Cold Legs in Operation,
One Locked Rotor

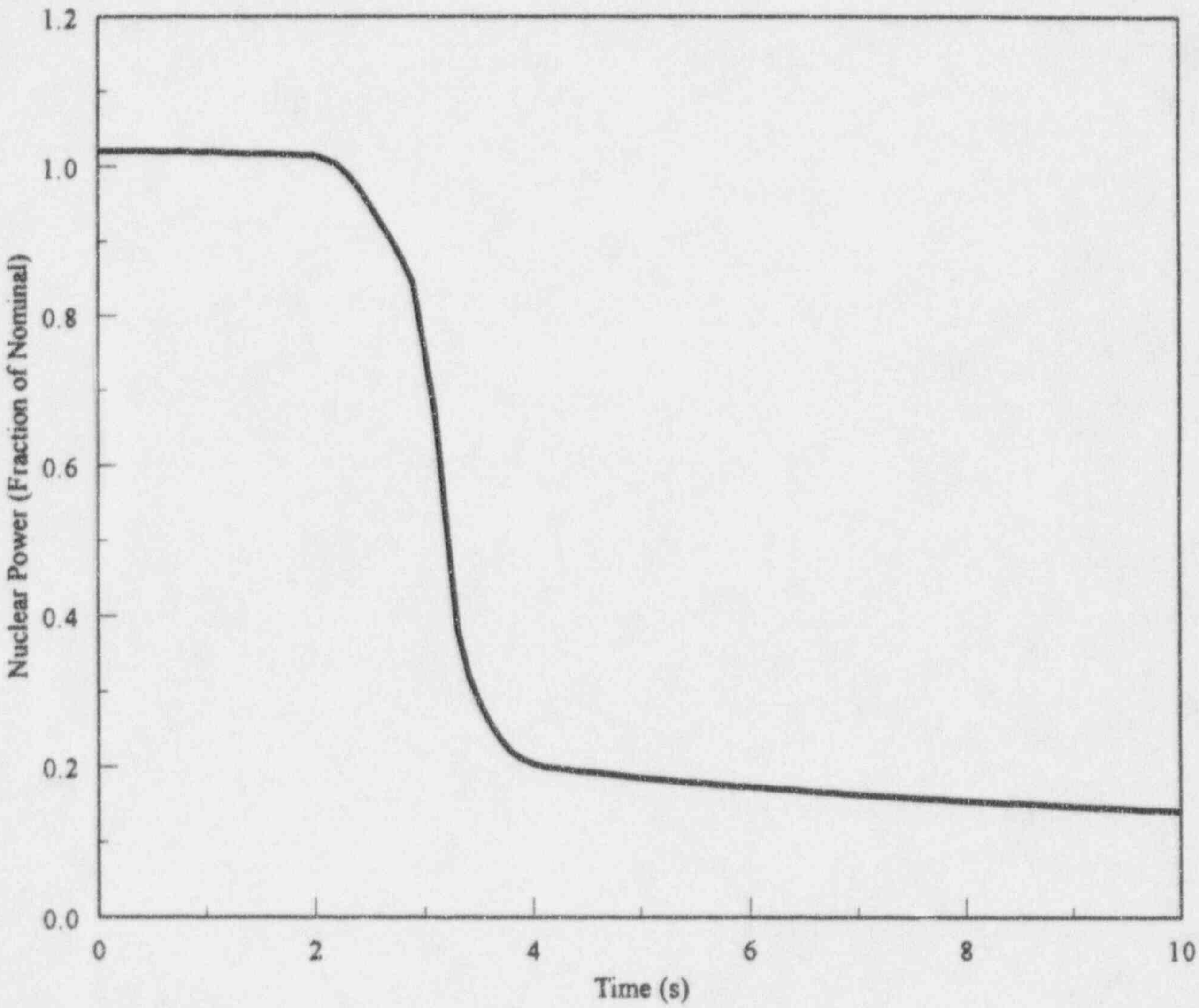
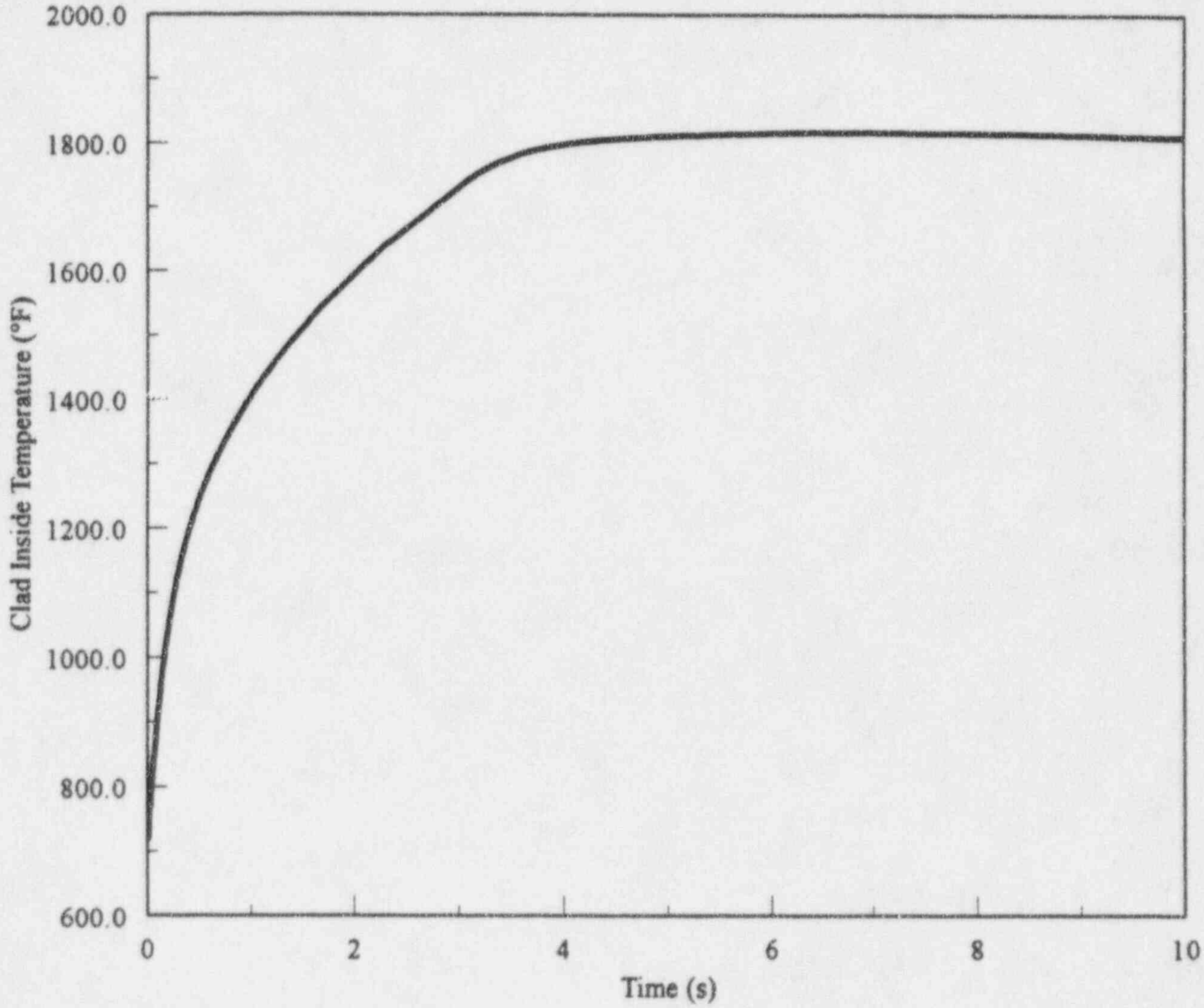


Figure 15.3.3-7
Clad Inside Temperature Transient for
Four Cold Legs in Operation,
One Locked Rotor



15.4 Reactivity and Power Distribution Anomalies

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

Discussions of the following incidents are presented in this section:

A. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or Low-power startup condition.

B. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power.

C. Rod cluster control assembly misalignment.

D. Startup of an inactive reactor coolant pump at an incorrect temperature.

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that results in an increased reactor coolant flow rate (not applicable to AP600).

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant.

G. Inadvertent loading and operation of a fuel assembly in an improper position.

H. Spectrum of rod cluster control assembly ejection accidents.

Items A, B, D, and F above are Condition II events, item G a Condition III event, and item H a Condition IV event. Item C entails both Conditions II and III events.

The applicable accidents in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing provided in Subsection 15.4.8.

Therefore, radiological consequences are reported only for that limiting case.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

A rod cluster control assembly (RCCA) withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of rod cluster control assemblies which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor 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 grouped into preselected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks are withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event (a fault of moderate frequency) as defined in Subsection 15.0.1.

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

- Source Range High Neutron Flux Reactor Trip

  This trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

- Intermediate Range High Neutron Flux Reactor Trip

  This trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above

approximately 10 percent of full power. It is automatically reinstated when the coincident two out of the four channels indicate a power level below this value.

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

  This trip function is actuated when two out of four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

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

  This trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.

- High Nuclear Flux Rate Reactor Trip

  This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the a preset setpoint. The trip may be manually bypassed after the coincident two out of four nuclear power range channels are manually reset.

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

15.4.1.2 Analysis of Effects and Consequences

15.4.1.2.1 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. In the first stage, the average core nuclear calculation is performed using spatial neutron kinetics methods, TWINKLE (Reference 1), to determine the average power generation with time, including the various total core feedback effects, (doppler reactivity and moderator reactivity).

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in THINC (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. To give conservative results for a startup accident, the following assumptions are made:

- Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values, as a function of power, are used (See Table 15.0-2).

- 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. After the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux. (See Table 15.0-2.)

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

- Reactor trip is assumed to be initiated by the power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent uncertainty increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 to 35 percent.

  Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position. See Subsection 15.0.5 for rod cluster control assembly insertion characteristics.

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

- The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their High-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

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

- Three reactor coolant pumps are assumed to be in operation.

- Pressurizer pressure is assumed to be 50 psi below nominal for steady-state fluctuations and measurement uncertainties.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or components adversely affect the consequences of the accident.

15.4.1.2.2 Results

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

Figure 15.4.1-1 shows the average neutron flux transient. The energy release and the fuel temperature increases are relatively small. The heat flux response (of interest for DNB considerations) is also shown in Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is margin to DNB during the transient since the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. Figure 15.4.1-3 shows the response of the average fuel and cladding temperatures. The minimum DNBR at all times remains above the safety analysis limit value.

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

15.4.1.3 Conclusions

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

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

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

and safety monitoring

eventually result in departure from nucleate boiling (DNB). Therefore, to avert damage to the fuel clad, the ~~reactor~~ protection system is designed to terminate any such transient before the departure from nucleate boiling ratio (DNBR) falls below the safety analysis limit.

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

and safety monitoring

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

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

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

- Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable heat generation rate (kW/ft) from being exceeded.

- A high pressurizer pressure reactor trip is actuated from any two out of four pressure channels when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A high pressurizer water level reactor trip is actuated from any two out of four level channels that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).

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

- High neutron flux (two out of four power range)
- Overpower ΔT (two out of four)
- Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7. Figure 15.0.3-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include adverse instrumentation and setpoint uncertainties so that under nominal conditions a trip occurs well within the area bounded by these lines. The utility of this diagram is that the limit imposed by any given DNBR is represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. Points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit value.

The diagram shows that DNB is prevented for cases where the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- High neutron flux (fixed setpoint)
- High pressurizer pressure (fixed setpoint)
- Low pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed by the LOFTRAN code. (Reference 3) This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0.3-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. To perform a conservative analysis for an uncontrolled RCCA bank withdrawal at power accident, the following assumptions are made:

- The nominal initial conditions are assumed in accordance with the revised thermal design procedure (RTDP). Uncertainties in the initial conditions are included in the DNBR limit as described in Reference 11.

- Reactivity coefficients--two cases are analyzed:

  Minimum Reactivity Feedback - A least negative moderator temperature coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed (See Figure 15.0.4-1.)

  Maximum Reactivity Feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed (See Figure 15.0.4-1.)

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

- The rod cluster control assembly trip insertion characteristic is based on the assumption that the highest-worth assembly is stuck in its fully withdrawn position.
- A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks, having the maximum combined worth at maximum speed.

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

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.4.2.2.2 Results

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative 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 temperature and pressure result, and margin to DNB is maintained. The design basis for DNBR is described in Section 4.4.

The transient response for a representative slow rod cluster control assembly withdrawal from full power is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The minimum DNBR is greater than the safety analysis limit value.

Figure 15.4.2-13 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback. Two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the safety analysis limit value.

Figures 15.4.2-14 and 15.4.2-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting at 60 and 10 percent power, respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the safety analysis limit.

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

Referring to Figure 15.4.2-14, for example, it is noted that:

A. For high reactivity insertion rates (between 10 pcm/s and 110 pcm/s) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. Reactor trip is initiated by high neutron flux for reactivity insertion rates between approximately ~~75~~ 85 pcm/s and 110 pcm/s for the maximum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature with resultant high minimum DNBRs during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures remain more nearly in equilibrium with the neutron flux. Thus, minimum DNBR during the transient decreases with decreasing insertion rate.

B. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. This trip circuit is described in Chapter 7. The average temperature contribution to the circuit is lead-lag compensated to decrease the effect of the thermal capacity of the reactor coolant system in response to power increases.

C. For reactivity insertion rates less than ~~8~~ 30 pcm/s for the minimum feedback cases, the rise in reactor coolant system pressure is sufficiently high that the pressurizer safety valve setpoint is reached prior to reactor trip. Opening of this valve limits the rise in reactor coolant pressure as the temperature continues to rise. Since the overtemperature ΔT reactor trip setpoint is based on both temperature and pressure, limiting the reactor coolant pressure by opening the pressurizer safety valve brings about the overtemperature ΔT earlier than if the valve remains closed. For this reason, the overtemperature ΔT setpoint initiates reactor trip at reactivity insertion rates of approximately 7 pcm/s for the minimum feedback cases and as high as ~~60~~ 75 pcm/s for the maximum feedback cases.

D. For reactivity insertion rates less than approximately ~~4~~ 5 pcm/s for the minimum feedback cases and less than approximately 50 pcm/s for maximum feedback cases, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load of the reactor coolant system, sharply decreases the rate of increase of reactor coolant system average temperature. This decrease in the rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation. This causes the overtemperature ΔT setpoint to be reached later, with resulting lower minimum (DNBRs).

The delay in overtemperature ΔT reactor trip due to the opening of the steam generator safety valves outweighs the effect of the opening of the pressurizer safety valve (as described in item C) and causes the high neutron flux setpoint to initiate reactor trip for reactivity insertion rates less than approximately ~~3~~ 4 pcm/s for cases of minimum feedback and less than approximately ~~30~~ 40 pcm/s for cases of maximum feedback. At these slow insertion rates, a sharp decrease in minimum DNBR occurs.

E. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient. For reactivity insertion rates less than approximately 1 pcm/s for minimum feedback cases and insertion rates less than approximately ~~10~~ 15 pcm/s for maximum feedback cases, the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNB) due to the fact that for these lower reactivity insertion rates the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

For transients initiated from full power (See Figure 15.4.2-13), the competing effects due to the opening of the pressurizer safety valve and steam generator safety valves described in items C and D are demonstrated only for the maximum feedback cases. Both the overtemperature ΔT and high neutron flux trips are equally effective in terminating the transient for insertion rates between approximately 10 pcm/s and 20 pcm/s. The effect of the opening of the steam generator safety valves is demonstrated for the reactivity insertion rate of ~~15~~ 13 pcm/s where the sharp peak in minimum DNBR occurs.

The transients initiated from 10% power (See Figure 15.4.2-15) exhibit the same trends in minimum DNBR, but the results are bounded by those from the transients initiated from higher power levels.

Figures 15.4.2-13, 15.4.2-14, and 15.4.2-15 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature still remains below the fuel melting temperature.

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

The reactor is tripped fast enough during the RCCA bank withdrawal at power transient that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

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

15.4.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip functions provide adequate protection over the entire range of possible reactivity insertion rates. (The minimum value of DNBR is always larger than the limiting value.)

15.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misoperation accidents include:

- One or more dropped RCCAs within the same group
- ~~A dropped RCCA bank~~
- Statically misaligned RCCA
- Withdrawal of a single RCCA

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

RCCAs are moved in preselected banks, and the banks are moved in a preselected sequence. Each bank of RCCAs is divided into one or two groups of four or five RCCAs each. The rods comprising a group operate in parallel through a multiplexing system, using 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 rod cluster control assembly attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the rod cluster control assemblies of a rod group are driven in parallel, any single failure which causes rod withdrawal affects the entire group.* Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. The single RCCA withdrawal event is a Condition III incident, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could withdraw a single RCCA in the control bank, since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage.

A single electrical or mechanical failure in the ~~power~~ plant control system could, at most, result in dropping one or more RCCAs within the same group.

Thus, consistent with the philosophy and format of American National Standards Institute ANSI N18.2, the event is classified as a Condition III incident. 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 . . ." (Reference 12).

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

A dropped RCCA or RCCA bank ~~is~~ may be detected by~~:~~ one or more of the following

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen by the in-core or ex-core neutron detectors or core exit thermocouples, through ~~the~~ on-line core monitoring ~~system~~
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication
- ~~The dropped RCCA protection system, which activates an alarm and a control rod withdrawal block. (This system is part of the ESF protection system. See Subsection 7.2.1.1.10.)~~

Misaligned RCCAs are detected by~~:~~ one or more of the following

- Asymmetric power distribution as seen by the in-core or ex-core neutron detectors or core exit thermocouples, through ~~the~~ on-line core monitoring ~~system~~
- Rod deviation alarm
- Rod position indicators

The resolution of the rod position indicator channel is ±5 percent span (±7.5 inches). A deviation of any rod cluster control assembly from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess

of five percent of span. If the rod deviation alarm is not operable, the operator takes action as required by the technical specifications.

If one or more of the rod position indicator channels is out of service, detailed operating instructions are followed to verify the alignment of the nonindicated RCCAs. The operator also takes action as required by the technical specifications.

In the extremely unlikely event of multiple electrical failures which result in single RCCA withdrawal, rod deviation and rod control urgent failure are both displayed on the plant annunciator, and the rod position indicators indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indication. 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 rod cluster control assembly. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III SRP Section 15.4.3 evaluation criteria are met; however, due to the increase in local power density, the limits in Figure 15.0.3-1 may be exceeded.

Plant systems and equipment which are available to mitigate the effects of the various control rod misoperations are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

15.4.3.2.1.1 Method of Analysis

- One or More Dropped Rod Cluster Control Assemblies from the Same Group

  A drop of one or more rod cluster control assemblies (RCCAs) from the same group~~, or a dropped RCCA bank,~~ results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. ~~For this situation, the rod drop is detected by the dropped RCCA protection system, which actuates an alarm and an automatic rod withdrawal block. The rod withdrawal block eliminates the potential for a reactor power overshoot, which could result from RCCA bank outward motion in the automatic rod control mode; thus the reactor behavior is the same as in the manual control mode.~~ Following the drop, the reduced core power and continued steam demand to the turbine causes the reactor coolant temperature ~~to decrease.~~ In the presence of a negative moderator temperature coefficient, the reactor power rises monotonically back to the initial power level at a reduced inlet temperature with no power overshoot. ~~The most limiting conditions occur at the endpoint of the transient, where the reactor power is the highest.~~ If the RCS temperature

In the manual control mode, the plant will establish a new equilibrium condition. The new equilibrium condition is reached through reactivity feedback

The absence of any power overshoot establishes the automatic operating mode as a limiting case.

reduction is very large, the turbine power may not be able to be maintained due to the reduction in the secondary-side steam pressure and the volumetric flow limit of the turbine system. In this case, the equilibrium power level is less than the initial power. INSERT A

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (Reference 3). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures and power level.

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system ~~endpoint~~ transient conditions and reactor power. By ~~incorporating the primary conditions from the transient endpoint and~~ combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling (DNB) design basis is shown to be met using the THINC code.

- ~~Dropped Rod Cluster Control Assembly Bank~~

~~A RCCA bank drop is detected by the dropped RCCA protection system, which actuates an alarm and an automatic control rod withdrawal block. This event is similar to the single or multiple rod drop from the same group event addressed above except that there is a potential for a greater reactivity insertion and, due to the symmetric arrangement of the RCCAs in a bank, the radial power distribution is symmetric. The method of analysis for this event is the same as for the drop of one or more RCCAs from the same group. Due to the radial symmetry of the power distribution, this event is typically less limiting than the drop of one or more RCCAs from the same group.~~

- Statically Misaligned Rod Cluster Control Assembly

Steady-state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC code to calculate the departure from nucleate boiling ratio (DNBR).

15.4.3.2.1.2 Results

- One or More Dropped Rod Cluster Control Assemblies

Figures 15.4.3-1 through 15.4.3-4 show the typical transient response of the reactor to a dropped ~~RCCA~~ rod (or ~~RCCAs~~ rods) ~~in manual control, or~~ in automatic control. ~~with the rod withdrawal block engaged. With no control rod motion, the drop of single or multiple RCCAs results in an initial decrease in core power, a drop in the reactor coolant inlet temperature, and then a monotonic increase in reactor power to the initial level caused by the reactivity effect of the cooldown. Figure 15.4.3-5 shows the DNBR versus time during the transient. A constant radial power distribution is assumed throughout the transient to show the DNBR variation due only to changes in average core power and coolant conditions. This result demonstrates that the most limiting conditions for the~~ INSERT B

INSERT A

In the automatic control mode, the Plant Control System detects the drop in core power and initiates withdrawal of a control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to the initial power level. The magnitude of the power overshoot is a function of: the Plant Control System characteristics, core reactivity coefficients, the dropped rod worth, and the available control bank worth.

INSERT B

The nuclear power and heat flux drop to a minimum value and recover under the influence of both rod withdrawal and thermal feedback. The prompt decrease in power is governed by the dropped rod worth since the Plant Control System does not respond during the short rod drop time period. The Plant Control System detects the reduction in core power and initiates control bank withdrawal in order to restore the primary side power. Power overshoot occurs, after which the core power is restored to the initial power level.

The primary system conditions are combined with the hot channel factors from the nuclear analysis for the DNB evaluation. Uncertainties in the initial conditions are included in the DNB evaluation as discussed in Subsection 15.0.3.2. The calculated minimum DNBR was found to be greater than the safety analysis limit value for this plant, for any single or multiple rod drop from the same group.

~~evaluation of the DNBR using the dropped rod power distribution are at the transient endpoint.~~

~~For this analysis, the effect of the single or multiple rod drop was conservatively analyzed by assuming the reactor power returns to its initial power level with no credit for the reduction in core inlet temperature. Uncertainties in the initial conditions are included in the DNB evaluation as described in Subsection 15.0.3.2. The resulting radial hot channel factor was found to be less than the value which causes the DNBR to drop below the safety analysis limit value for this plant, for any single or multiple rod drop from the same group.~~

- ~~Dropped Rod Cluster Control Assembly Bank~~

~~A dropped bank typically results in a negative reactivity insertion greater than 300 pcm. This event is detected by the dropped RCCA protection system, which acts to prevent automatic control rod withdrawal. The transient therefore proceeds in an identical fashion as described above for the drop of one or more RCCAs from the same group. If the bank worth is very large, or the moderator temperature coefficient is very small as at the beginning of a cycle, this may result in a severe enough cooldown to actuate a reactor trip on low pressurizer pressure. Since the reactor power is also very low under these conditions, these cases are not limiting.~~

~~For this analysis, the effect of the RCCA bank drop was conservatively analyzed by assuming the reactor power returns to its initial power level with no credit for the reduction in core inlet temperature. Uncertainties in the initial conditions are included in the DNB evaluation as described in Subsection 15.0.3.2. The resulting radial hot channel factor was found to be less than the value which causes the DNBR to drop below the safety analysis limit value for any RCCA bank drop.~~

The analysis described above includes consideration of drops of the RCCA groups ~~or banks~~ which can be selected for insertion as part of the rapid power reduction system. This system is provided to allow the reactor to ride out a complete loss of load from full power without a reactor trip, and is described in Subsection 7.7.1.10. If these RCCAs are inadvertently dropped (in the absence of a loss of load signal), the transient behavior is the same as for the RCCA ~~bank~~ drop event described above. The evaluation showed that the DNBR does not drop below the safety analysis limit value as a result of the inadvertent actuation of the rapid power reduction system.

- Statically Misaligned Rod Cluster Control Assembly

The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim (MSHIM) or axial offset (AO) rod banks are inserted up to their insertion limit with one RCCA fully withdrawn, while the reactor is at full power. Multiple independent alarms, including a bank insertion limit or rod deviation alarm, alert the operator well before the postulated conditions are approached.

For RCCA misalignments in which the MSHIM or AO banks are inserted to their respective insertion limits with any one RCCA fully withdrawn, the DNBR does not fall below the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperature 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 DNB evaluation as described in Subsection 15.0.3.2.

Strikeout
R

For RCCA misalignments in which one RCCA is fully inserted, with the rest of the RCCAs at or above their insertion limits, this event is covered by the analysis of the consequences of the drop of a single RCCA, which is presented above.

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 is that corresponding to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which causes fuel melting.

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

15.4.3.2.2 Single Rod Cluster Control Assembly Withdrawal

15.4.3.2.2.1 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 DNBR for the event. The case of the worst rod withdrawn from the MSHIM or AO bank inserted at the insertion limit, with the reactor initially at full power, is 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.

15.4.3.2.2.2 Results

For the single rod withdrawal event, two cases are 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 DNBRs 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 DNBR from falling below the safety analysis

limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip is expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5 percent. The limiting dose evaluation is bounded by the locked rotor results presented in Subsection 15.3.3.

2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA result in the immobility of the other RCCAs in the controlling bank. The transient then proceeds in the same manner as case 1 described above.

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

15.4.3.3 Conclusions

in those

For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs groups ~~or banks~~ selected to be inserted as part of the rapid power reduction system, it is shown that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For cases of any one RCCA fully inserted, or the MSHIM or AO banks inserted to their rod insertion limits with any single RCCA in one of those banks fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the MSHIM or AO banks at their insertion limits, an upper bound of the number of fuel rods experiencing DNB is five percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one reactor coolant pump out of service, the affected reactor coolant loop flow rate is less than half of its nominal value. If the reactor is operating at power, the steam generator in the inactive pump loop removes less than half the total power. ~~The cold leg temperature in the inactive pump loop is lower than the average reactor core inlet temperature, while the hot leg temperatures remain the same in both loops.~~

in an ~~significant~~ increase

Starting an idle reactor coolant pump ~~without bringing the inactive pump loop cold leg temperature closer to the core inlet temperature~~ results in the injection of cold water into the core, which causes a reactivity insertion and subsequent power increase.

The incident is a Condition II event (a fault of moderate frequency), as defined in Subsection 15.0.1.

If the startup of an inactive reactor coolant pump accident occurs, the transient is terminated automatically by a reactor trip on power range high neutron flux setpoint or low flow (P-8 interlock).

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis

This transient is analyzed using 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 (Section 4.4) is then used to calculate the departure from nucleate boiling ratio (DNBR) during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and on heat flux as calculated by FACTRAN.

Plant characteristics and initial conditions are discussed in Subsection 15.0.3. To obtain conservative bounding results for the startup of an inactive pump accident, the following assumptions are made:

- Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to departure from nucleate boiling (DNB). For this analysis a conservative value of 70 percent nominal power is assumed. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive pump cold leg temperature.

- Following startup of the idle pump, the inactive pump loop flow ~~reverses and~~ accelerates to its nominal full-flow value. For this analysis it is conservatively assumed that the flow rate acceleration occurs in 10 seconds (a linear ramp).

- A conservatively large negative moderator temperature coefficient.

- A least negative Doppler-only power coefficient. (See Figure 15.0.4-1.)

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

- The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power.

Plant systems and equipment available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.4.2.2 Results

The results following the startup of an idle pump with the preceding assumptions are shown in Figures 15.4.4-1 through 15.4.4-~~5~~6.

As shown in these curves, during the first part of the transient the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit. (See Section 4.4 for a description of the DNBR design basis.)

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

The calculated sequence of events for this accident is shown in Table 15.4-1. The transient results illustrated in Figures 15.4.4-1 through 15.4.4-~~5~~6 indicate that a stabilized plant condition, with the reactor tripped, is rapidly approached. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.4.4.3 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the limiting DNBR, so the DNB design basis as described in Section 4.4 is met.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop That Results in an Increased Reactor Coolant Flow Rate

This subsection is not applicable to the AP600.

15.4.6 Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

One of the two principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system (DWS) into the reactor coolant system through the reactor makeup portion of the chemical and volume control system (CVS). Boron dilution with these systems is manually initiated

under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup water boron concentration to that of the RCS during normal charging.

An inadvertent boron dilution is caused by the failure of the DWS or CVS ~~makeup control system~~, either by controller operator or mechanical failure. The CVS and DWS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, allow sufficient time for automatic or operator response to terminate the dilution.

An inadvertent dilution from the DWS through the CVS may be terminated by isolating the makeup pump suction line to the DWS storage tank. Flow from the DWS, which is the source of unborated water, may be terminated by closing isolation valves in the CVS. ~~Any~~ Lost shutdown margin may be regained by opening the isolation valve to the boric acid tank (BAT), thus allowing the addition of borated water (greater than 4000 ppm) to the RCS.

Generally, to dilute, the operator performs two distinct actions:

- Switch control of the makeup from the automatic makeup mode to the dilute mode.
- Depress the start button (start CVS pumps).

Failure to carry out either of those actions prevents initiation of dilution. Because the AP600 CVS makeup pumps do not run continuously (they are expected to be operated once per day to make up for RCS leakage), a makeup pump is started when the ~~makeup~~ volume control system is placed into dilute mode.

The status of the RCS makeup is continuously available to the operator by the following:

- Indication of the boric acid and blended flow rates
- CVS makeup pumps status lights
- Deviation alarms, if the boric acid or blended flow rates deviate by more than the specified tolerance from the preset values
- Source range neutron flux - when reactor is subcritical
  - High flux at shutdown alarm
  - Indicated source range neutron flux count rates
  - Audible source range neutron flux count rate
  - Source range neutron ~~flux doubling~~ multiplication alarm
    step
- When the reactor is critical
  - Axial flux difference alarm (reactor power $\geq$50 percent RTP)

- Control rod insertion limit low and low-low alarms
- Overtemperature ΔT alarm (at power)
- Overtemperature ΔT reactor trip
- Power range neutron flux - high, both high and low setpoint reactor trips.

This event is a Condition II incident (a fault of moderate frequency), as defined in Subsection 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

To cover ~~all~~ the phases of plant operation, boron dilutions during ~~r~~efueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation ~~is~~ are considered in this analysis. Conservative values for necessary parameters are used, (high RCS critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual RCS volumes). These assumptions result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

15.4.6.2.1 Dilution During Refueling (Mode 6)

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water by locking closed specified valves in the CVS during refueling operations. These valves block the flow paths that can allow unborated makeup water to reach the RCS. ~~Any~~ makeup which is required during refueling uses water supplied from the BAT (which contains borated water) by the CVS makeup pumps.

15.4.6.2.2 Dilution During Cold Shutdown (Mode 5)

The following conditions are assumed for inadvertent boron dilution while in this operating mode:

- A dilution flow of 150 gpm of unborated water.

- ~~Two RCS water volumes are considered for this mode of operation. The first is 2235 ft³~~ A volume of 2245 ft³ ~~and is~~ Stet a conservative estimate of the minimum active RCS volume corresponding to the water level drained to mid-loop in the vessel while on normal residual heat removal. ~~The second volume is 2590 ft³, which is a conservative estimate of the active volume with the RCS filled while on normal residual heat removal. Even with the RCS filled,~~ the assumed active volume does not include the volume of the reactor vessel upper head region.

- Control rods (RCCAs) fully inserted, which is the normal condition in cold shutdown and a critical boron concentration of ~~1359~~ 1315 ppm. This is a conservative boron concentration with control rods inserted and even allows for the most reactive rod to be stuck in the fully withdrawn position.

- The shutdown margin equal to 1.6 percent Δk/k, the minimum value required by technical specifications for the cold shutdown mode. Combined with the preceding, this gives a shutdown boron concentration of ~~1501~~ 1489 ppm.

In the event of an inadvertent boron dilution transient while in this mode of operation, the source range nuclear instrumentation detects ~~a doubling~~ an increase of 60% of the neutron flux by comparing the current source range flux to that of about 10 minutes earlier. Upon detection of the flux ~~doubling~~ increase, an alarm is sounded for the operator, and valve movement to terminate the dilution is automatically initiated.

~~In fact,~~ upon any reactor trip signal, source range flux ~~doubling~~ multiplication signal, loss of offsite power, or safety injection signal, a safety function automatically isolates the potentially unborated water from the DWS, thereby terminating the dilution. Additionally, the suction lines for the CVS pumps are automatically realigned to draw borated (greater than 4000 ppm) water from the CVS boric acid tank. The realignment of the CVS valves to terminate the dilution is a safety-related function. The realignment of pump suction to the boric acid tank is a non-safety related operation. The CVS pumps themselves are non-safety related, so their operation is not credited in the analysis as a protection function. However, the analysis does consider the initial portion of this boration phase by treating it as a continuing dilution until any unborated water in the CVS lines is purged.

Under the conditions just defined, the automatic protective actions initiate about ~~8~~ 8.9 minutes after the start of dilution. These automatic actions are carried out to minimize the approach to criticality and to maintain the plant in a subcritical condition. After the automatic protection functions take place, the operator may take action to restore the technical specification shutdown margin.

15.4.6.2.3 Dilution During Hot Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 200 gpm of unborated water.
- An RCS water volume of ~~2590~~ 2601 ft^3. This is a conservative estimate of the minimum volume of the RCS, while on normal RHR and with the RCS filled and vented.
- All control rods fully inserted, except the most reactive rod which is assumed stuck in the fully withdrawn position, and a conservative critical boron concentration of ~~1331~~ 1308 ppm.
- The shutdown margin equal to 1.6 percent Δk/k, the minimum value required by technical specifications for the hot shutdown mode. Combined with the preceding assumption, this gives a shutdown boron concentration of ~~1479~~ 1482 ppm.

In the event of an inadvertent boron dilution transient while in this mode of operation, the source range nuclear instrumentation detects ~~a doubling~~ an increase of 60% of the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm for the operator.

As in Mode 5, the safety analysis considers the potential penalty of the subsequent non-safety related boration function by accounting for the purge volume associated with the CVS piping. Under the conditions just defined, these protective actions initiate about ~~10~~ 8.9 minutes after start of dilution. No operator action is required to terminate this transient, though operator action can be taken to restore the technical specification shutdown margin.

15.4.6.2.4 Dilution During Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 200 gpm of unborated water.
- The RCS volume is ~~5800~~ 5737 ft^3. This is a conservative estimate of the minimum active volume of the RCS with the RCS filled and vented and one RCP running.
- A critical boron concentration of ~~272~~ 244 ppm. This is a conservative boron concentration assuming control rods are fully inserted minus the most reactive rod which is assumed stuck in the fully withdrawn position.
- The shutdown margin equal to 1.6 percent $\Delta k/k$, the minimum value required by technical specifications for the hot standby mode. Combined with the preceding, this gives a shutdown boron concentration of 426 ppm.

In the event of an inadvertent boron dilution transient while in this mode of operation, the source range nuclear instrumentation detects ~~a doubling~~ an increase of 60% of the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm for the operator.

As in the analyses for Modes 4 and 5, the only consideration of the boration function in safety-analysis is to account for the additional dilution effect due to the purge volume associated with the CVS piping. Under the conditions just defined, these protective actions initiate about ~~70~~ 116 minutes after start of dilution. No operator action is required to terminate this transient, though operator action could be taken to restore the technical specification shutdown margin.

15.4.6.2.5 Dilution During Startup (Mode 2)

In this mode, the plant is taken from one long-term mode of operation (hot standby), to another (power). The plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. Normal actions taken to change power level, either up or down, require operator initiation. The technical specifications require an available shutdown margin of 1.6 percent $\Delta k/k$ and four reactor coolant pumps operating. Other conditions assumed are the following:

- A dilution flow of 200 gpm of unborated water.

- A minimum RCS water volume of ~~6566~~ 6579 ft^3. This is a very conservative estimate of the active RCS volume, minus the pressurizer volume.

- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is ~~1003~~ 784 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is ~~513~~ 517 ppm. Full rod insertion, minus the most reactive stuck rod, occurs because of reactor trip.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control, with the operator required to maintain a ~~very~~ high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and then manually withdraw the control rods, a process that takes several hours. The technical specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus providing confidence that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation is slow enough to allow the operator to manually block the source range reactor trip after receiving the P-6 permissive signal from the intermediate range detectors (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

Upon any reactor trip signal, source range flux ~~doubling~~ multiplication signal, loss of offsite power, or a safety injection signal, a safety related function automatically isolates the potentially unborated water from the DWS, thereby terminating the dilution. Additionally, the suction lines for the CVS pumps are automatically realigned to draw borated water from the CVS boric acid tank.

However, since the realignment of the suction for the CVS pumps to the boric acid tank is a non-safety related operation, the only consideration given to the reboration phase of the event in the safety analysis is with respect to the unborated CVS purge volume.

After reactor trip, the dilution would have to continue for approximately ~~76~~ 93.2 minutes to overcome the available shutdown margin. Even assuming that the nonsafety related boration operation does not occur, the unborated water that may remain in the purge volume of the CVS is not sufficient to return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the DWS prevents a post-trip return to criticality.

15.4.6.2.6 Dilution During Full Power Operation (Mode 1)

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The technical specifications require an available shutdown margin of 1.6 percent Δk/k and four reactor coolant pumps operating. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the CVS makeup pumps. The

analysis is performed assuming two CVS pumps are in operation, even though normal operation is with one pump. Conditions assumed for a dilution in this mode are the following:

- A dilution flow of 200 gpm of unborated water.
- A minimum RCS water volume of ~~6566~~ 6579 ft^3. This is a very conservative estimate of the active RCS volume, minus the pressurizer volume.
- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is ~~1003~~ 784 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is ~~408~~ 324 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise cause the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. Upon any reactor trip signal a safety-related function automatically isolates the unborated water from the DWS, thereby terminating the dilution. Additionally, the suction lines for the CVS pumps are automatically realigned to draw borated water from the CVS boric acid tank.

However, since the realignment of the suction for the CVS pumps to the boric acid tank is a nonsafety related operation, the only consideration given to the reboration phase of the event in the safety analysis is with respect to the unborated purge volume.

After reactor trip, the dilution would have to continue for at least ~~78.3~~ 57.2 minutes to overcome the available shutdown margin. The unborated water that may remain in the purge volume of the CVS does not return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the DWS precludes a post-trip return to criticality.

The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be ~~2.0~~ 1.6 pcm per second and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Before reaching the overtemperature ΔT reactor trip, the operator will have received an alarm on overtemperature ΔT and an overtemperature ΔT turbine runback.

With the reactor in automatic rod control, the pressurizer level controller limits the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller throttles charging flow down to match letdown rate. For the safety analysis, a conservative dilution flow rate of 200 gpm is assumed. With the reactor in automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. This action by the controller results in at least three alarms to the operator:

A. Rod insertion limit - low level alarm

B. Rod insertion limit - low-low level alarm if insertion continued after item A

C. Axial flux difference alarm (ΔI outside of the target band).

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. The operator has at least ~~81.7~~ 61.64 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at beginning of cycle. The time is significantly longer at end of cycle, because of the low initial boron concentration.

The preceding results demonstrate that in all modes of operation, an inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition.

15.4.6.3 Conclusions

Inadvertent boron dilution events are prevented during refueling and automatically terminated during cold shutdown, hot shutdown, and hot standby modes. Inadvertent boron dilution events during start-up or power operation, if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and any post-trip return to criticality is prevented.

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 those which can arise from the inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly with pellets of the wrong enrichment. This leads 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 five percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing fuel assembly enrichment errors or loading errors which cause power shapes to be peaked in excess of the design value. Power distribution related measurements are incorporated into the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distributions measurements in Westinghouse PWRs.

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. During core loading, the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to a combination of misplaced fuel assemblies would significantly increase peaking factors and is readily observable with the online core monitoring system. The fixed incore instrumentation within the instrumented fuel assembly locations is augmented with core exit thermocouples. There is a high probability that these thermocouples would also indicate any abnormally high coolant temperature rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

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

15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady-state power distributions in the x-y plane of the core are calculated at 30 percent rated thermal power using a two-dimensional few group diffusion code TORTISE, which is an updated version of TURTLE (Reference 13). A discrete representation is used wherein each individual fuel rod is described by a mesh interval. Representative power distributions in the x-y plane for a correctly loaded core are given in chapter 4.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown in the in-core detector locations. (See Figures 15.4.7-1 through 15.4.7-4).

15.4.7.2.2 Results

The following core loading error cases are analyzed:

Case A:

Case in which a region 1 assembly is interchanged with a region 3 assembly. The particular case considered is the interchange of two ~~adjacent~~ assemblies near the periphery of the core. (See Figure 15.4.7-1.)

Case B:

Case in which a region 1 assembly is interchanged with a neighboring region 2 fuel assembly. For the particular case considered the interchange is assumed to take place closer to core the 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. (See Figure 15.4.7-2.)

Case C:

Enrichment error: Case in which a region 2 fuel assembly is loaded in the core central position. (See Figure 15.4.7-3.)

Case D:

Case in which a region 2 fuel assembly instead of a region 1 assembly is loaded near the core periphery. (See Figure 15.4.7-4.)

15.4.7.3 Conclusions

Fuel assembly enrichment errors are 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 departure from nucleate boiling ratio and increased fuel and clad temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or 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 and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.8.1.1 Design Precautions and Protection

15.4.8.1.1.1 Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to prevent the possibility of a rod cluster control assembly drive mechanism housing failure are listed below:

- Each control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head. The housings are checked during the hydrotest of the completed reactor coolant system.

- 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 SSE can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

- The latch mechanism housing and rod travel housing are each a single length of forged stainless steel. This material exhibits excellent notch toughness at temperatures that are encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional confidence that gross failure of the housing does not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds which are subject to periodic inspections.

15.4.8.1.1.2 Nuclear Design

If a rupture of an RCCA drive mechanism housing is postulated, the operation using chemical shim is such that the severity of an ejected rod cluster control assembly is inherently limited. In general, the reactor is operated with the power control (or MSHIM) RCCAs inserted only far enough to permit load follow. The axial offset (AO) RCCAs are positioned so that the targeted AO can be met throughout core life. Reactivity changes caused by core depletion and xenon transients are normally compensated for by boron changes and the MSHIM banks, respectively. Further, the location and grouping of the power control and axial offset RCCAs are selected with consideration for a rod cluster control assembly ejection accident. Therefore, should a rod cluster control assembly be ejected from its normal position during full-power operation, a less severe reactivity excursion than analyzed is expected.

However, it may occasionally be desirable to operate with larger than normal insertions. For this reason, a power control and axial offset rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit provides adequate shutdown capability and an acceptable power distribution. The position of the RCCAs is continuously indicated in the main control room. An alarm occurs if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low level alarm and emergency boration at the low-low level alarm.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in Reference 5. The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.1.4 Effects on Adjacent Housings

Investigations have shown that failure of a rod cluster control assembly mechanism housing, due to either longitudinal or circumferential cracking, does not cause damage to adjacent housings. The control rod drive mechanism is described in Subsection 3.9.4.1.1.

15.4.8.1.1.5 Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing occurs, the region of the position indicator assembly opposite the break is stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path is 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 occurs, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings are on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking is expected. Housings adjacent to a failed housing in locations other than the periphery would not bend due to the rigidity of multiple adjacent housings.

15.4.8.1.1.6 Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing occurs, the broken-off section of the housing is ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft tends 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 reaches the missile shield, it partially penetrates the shield and dissipates its kinetic energy. The water jet from the break continues to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing is short enough to clear the break when fully ejected, it rebounds after impact with the missile shield. The top end plates of the position indicator coil assemblies 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 occurs, the low kinetic energy of the rebounding projectile is not expected to cause significant damage (sufficient to cause failure of an adjacent housing).

15.4.8.1.1.7 Consequences

The probability of damage to an adjacent housing is considered remote. If damage is postulated, it is not expected to lead to a more severe transient since rod cluster control assemblies 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 rod cluster control assembly not to fall on

receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.1.8 Summary

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

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See Subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of a rod cluster control assembly ejection accident, some fuel damage is considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project (Reference 6). Extensive tests of uranium dioxide (UO_2) zirconium-clad fuel rods representative of those in pressurized water reactor cores such as AP600 have demonstrated failure thresholds in the range of 240 to 257 cal/g. Other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT (Reference 7) results, which indicated a failure threshold of 280 cal/g. Limited results indicate 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/g for unirradiated rods and 200 cal/g for irradiated rods. Catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/g.

In view of the preceding experimental results and conformance with Regulatory Guide 1.77, criteria are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- Average fuel pellet enthalpy at the hot spot is below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.
- Peak reactor coolant pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.
- Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of the first criterion.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the rod cluster control assembly (RCCA) ejection transients 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 (Doppler reactivity and moderator reactivity). Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

A detailed discussion of the method of analysis appears in Reference 5.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 1), is used for the average core transient analysis. This code uses cross sections generated by LEOPARD (Reference 8) to solve the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for the 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 rod cluster control assembly movement. Since the radial dimension is missing, it is still necessary to employ conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Subsection 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal value multiplied by the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. The assumption is made that the hot spots before and after ejection are coincident. This is conservative, since the peak after ejection occurs in or adjacent to the assembly with the ejected rod, and before ejection, the power in this region is depressed.

The hot spot analysis is performed using the detailed fuel and clad 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 material properties are represented as functions of temperature. A parabolic 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 departure from nucleate boiling (DNB) and the Bishop-Sandburg-Tong correlation (Reference 9) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero-bulk fluid quality. The departure from nucleate boiling ratio is not calculated. Instead, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient is 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 15.0.11.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat absorption by 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 (Section 4.4) calculation is performed to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit for the possible pressure reduction caused by the assumed failure of the control rod pressure housing is made.

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 next. Table 15.4-3 presents the parameters used in this analysis.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

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

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

Power distributions before and after ejection for a worst case can be found in Reference 5. During plant startup physics testing, rod worths and power distributions have been measured

in the zero-power configuration and compared to values used in the analysis. The ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

15.4.8.2.1.2 Reactivity Feedback Weighting Factors

The largest temperature rises and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis.

Physics calculations are carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes are compared and effective weighting factors are shown to be conservative. These weighting factors take the form of multipliers that, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape.

In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition matches 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 are shown to be conservative compared to three-dimensional analysis (Reference 5).

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentrations at the beginning of cycle and end of cycle 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 just discussed, 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 one. The Doppler defect used is given in Subsection 15.0.4. The Doppler weighting factor increases under accident conditions, as previously discussed.

15.4.8.2.1.4 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 cycle (BOC) and 0.50 percent at end of cycle (EOC) 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. To allow for future cycles, pessimistic estimates of β_{eff} of 0.55 percent at BOC and 0.44 percent at EOC are used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-3 and includes the effect of one stuck rod cluster control assembly. These values are reduced by the ejected rod reactivity. The shutdown reactivity is simulated by dropping a rod of the required worth into the core. The start of rod motion occurs 0.5 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power (HFP) 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, adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Calculations show that the effect of two stuck rod cluster control assemblies (one of which is the worst ejected rod) is to reduce the shutdown by about an additional one percent Δk. Therefore, following a reactor trip resulting from a rod cluster control assembly ejection accident, the reactor is subcritical when the core returns to hot zero power.

15.4.8.2.1.6 Reactor Protection

As discussed in Subsection 15.4.8.1.1.3, reactor protection for a rod ejection is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are part of the reactor trip system (RTS). No single failure of the RTS negates the protection functions required for the rod ejection accident or adversely affects the consequences of the accident.

15.4.8.2.1.7 Results

Since the control rod insertion limits for the AP600 are multidimensional, a significant number of rodded configurations are evaluated in order to determine the most limiting cases, (that is those cases that produced the least amount of margin to the SRP section 15.4.8 evaluation acceptance criteria). The hot zero power cases and hot full power cases assume that the MSHIM and axial offset control RCCAs are inserted to their respective insertion limits before the event. The limiting RCCA ejection cases, for both the beginning and end of cycle at zero and full power, are presented next.

- Beginning of Cycle, Full Power

  The limiting ejected rod worth and hot channel factor are conservatively ~~calculated~~ assumed to be 0.38 percent Δk and 7.0, respectively. The peak hot spot clad average temperature is

2597°F. The peak hot spot fuel temperature reaches melting at 4900°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.

- Beginning of Cycle, Zero Power

  For this condition, the limiting ejected rod ~~has a~~ worth ~~of 0.744 percent Δk~~ and ~~a~~ hot channel factor ~~of 13.0.~~ are conservatively assumed to be 0.744 percent ΔK and 13.0, respectively. The peak hot spot clad average temperature is 2795°F, and the peak hot spot fuel temperature is 3881°F.

- End of Cycle, Full Power

  The ejected rod worth and hot channel factor are conservatively ~~calculated~~ assumed to be 0.33 percent Δk and 9.0, respectively. The peak hot spot clad average temperature is 2523°F. The peak hot spot fuel temperature reaches melting at 4800°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.

- End of Cycle, Zero Power

  The ejected rod worth and hot channel factor for this case are conservatively assumed to be 0.83 percent Δk and 16.0, respectively. The peak hot spot clad average temperature is 2901°F and the peak hot spot fuel temperature is 3862°F.

A summary of the preceding cases is given in Table 15.4-3. The nuclear power and fuel and clad temperature transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-4.

The calculated sequence of events for the limiting case rod ejection accidents, as shown in Figures 15.4.8-1 through 15.4.8-4, is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of a rod cluster control assembly constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents (LOCAs) are discussed in Subsection 15.6.5. Following the rod cluster control assembly ejection, the plant response is the same as any other LOCA event.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 15 percent of the rods are assumed to enter DNB based on a detailed three-dimensional THINC analysis (Reference 5). Although limited (less than 10 percent) fuel melting at the hot spot is allowed for the full-power cases, in practice, melting is not expected since the analysis conservatively assumes that the hot spots before and after ejection are coincident.

15.4.8.2.1.9 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of about one dollar at beginning of cycle, hot full power, will demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Limit C as described in the ASME Code, Section III. Since the severity of the present analysis does not exceed the worst-case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the reactor coolant system.

15.4.8.2.1.10 Lattice Deformations

A large temperature gradient exists 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 indicate that this bowing results in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is sufficient to withstand the forces produced.

Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow will be sufficient to produce sufficient lattice forces. Even if massive and rapid boiling, sufficient to distort the lattices, is hypothetically postulated, the large void fraction in the hot spot region produces 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 is concluded that no mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

15.4.8.3 to follow

Insert A

15.4.~~9~~ 10 References

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A 15.4.9 Combined License Information

This section has no requirement for additional information to be provided in support of the combined license application.

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6. Taxelius, T. G., ed, "Annual Report-SPERT Project, October 1968, September 1969," Idaho Nuclear Corporation, IN-1370, June 1970.

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13. Barry, R. F., and Altomore, A., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A (Proprietary) and WCAP-7758-A (Nonproprietary), February 1975.

Table 15.4-1 (Sheet 1 of 5)

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

Accident	Event	Time(s)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux (low setting) setpoint reached	10.3
	Peak nuclear power occurs	10.5
	Rods begin to fall into core	10.8
	Minimum DNBR occurs	12.6
	Peak average clad temperature occurs	12.8
	Peak heat flux occurs	12.8
	Peak average fuel temperature occurs	13.0
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high-reactivity insertion rate (75 pcm/s)	0.0
	Power range high neutron flux high trip point reached	~~4.7~~ 4.9
	Rods begin to fall into core	~~5.6~~ 5.8
	Minimum DNBR occurs	~~6.2~~ 6.3

Table 15.4-1 (Sheet 2 of 5)

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

Accident		Event	Time(s)
2.	Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/s)	0.0
		Overtemperature ΔT setpoint reached	~~700.5~~ 718.7
		Minimum DNBR occurs	~~702.2~~ 721.1
		Rods begin to fall into core	~~702.5~~ 720.7
Startup of inactive reactor coolant loop at an incorrect temperature		Initiation of pump startup	0.0
		Power reached P8 trip setpoint	~~2.3~~ 1.9
		Rods begin to drop	~~3.8~~ 2.8
		Minimum DNBR occurs	~~4.0~~ 3.8
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the rector coolant			
1.	Dilution during startup	Power range - low setpoint reactor trip due to dilution	0.0
		Dilution automatically terminated by DWS isolation	~~210~~ 170

Table 15.4-1 (Sheet 3 of 5)

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

	Accident	Event	Time(s)
2.	Dilution during Full-power operation		
	a. Automatic reactor control	Operation receives low-low rod insertion limit alarm due to dilution	0
		Shutdown margin lost	~~4903~~ 3698
	b. Manual reactor control	Initiate dilution	0
		Reactor trip on overtemperature ΔT due to dilution	~~304~~ 265
		Dilution automatically terminated by DWS isolation	~~414~~ 435
RCCA ejection accident			
1.	End of cycle, full power	Initiation of rod ejection	0.00
		Power range high neutron flux (high setting) setpoint reached	0.03
		Peak nuclear power occurs	0.14
		Rods begin to fall into core	0.53
		Peak fuel average temperature occurs	2.32
		Peak clad temperature occurs	2.39
		Peak heat flux occurs	2.40

Table 15.4-1 (Sheet 4 of 5)

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

	Accident	Event	Time(s)
2.	Beginning of cycle, zero power	Initiation of rod ejection	0.00
		Power range high neutron flux (low setting) setpoint reached	0.32
		Peak nuclear power occurs	0.39
		Rods begin to fall into core	0.82
		Peak clad temperature occurs	2.58
		Peak heat flux occurs	2.61
		Peak fuel average temperature occurs	2.61
3.	Beginning of cycle, full power	Initiation of rod ejection	0.00
		Power range high neutron flux (high setting) setpoint reached	0.03
		Peak nuclear power occurs	0.14
		Rods begin to fall into core	0.53
		Peak fuel average temperature occurs	2.22
		Peak clad temperature occurs	2.31
		Peak heat flux occurs	2.31

Table 15.4-1 (Sheet 5 of 5)

TIME SEQUENCE OF EVEN'S FOR INCIDENTS WHICH RESULT IN REACTIVITY AND PO' 'R DISTRIBUTION ANOMALIES

	Accident	Event	Time(s)
4.	End of cycle, zero power	Initiation of rod ejection	0.00
		Power range high neutron flux (low setting) setpoint reached	0.20
		Peak nuclear power occurs	0.24
		Rods begin to fall into core	0.70
		Peak clad temperature occurs	1.81
		Peak heat flux occurs	1.81
		Peak fuel average temperature occurs	2.00

Table 15.4-2

PARAMETERS

Assumed Dilution Flowrates:

Mode	Flowrate (gal/min)
~~1, 2, 3 & 4~~	200
~~5~~ 1 through 5	~~150~~

Volumes:

Mode	Volume (ft^3)	Volume (gal)
~~1,2~~ 1&2	~~6566~~ 6579	~~49,114~~ 49,211
3~~, 4a (Note a)~~	~~5800~~ 5737	~~43,384~~ 42,913
~~4b (Note b)~~	~~2590~~ 2601	~~19,373~~ 19,455
~~5a (filled) (Note c)~~	~~2590~~	~~19,373~~
~~5b (drained) (Note c)~~	~~2235~~ 2245	~~16,718~~ 16793

~~Notes~~

~~a. Volume with at least one reactor coolant pump operating.~~
~~b. Volume with the plant on normal RHR.~~
~~c. Drained refers to the reactor vessel coolant level at the mid-plane of the nozzles.~~

Table 15.4-3

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

Time in Life	HZP Beginning	HFP Beginning	HZP End	HFP End
Power level (%)	0	102	0	102
Ejected rod worth (%Δk)	0.744	0.38	0.83	0.33
Delayed neutron fraction (%)	0.55	0.55	0.44	0.44
Feedback reactivity weighting	2.071	1.30	2.32	1.60
Trip reactivity (%Δk)	2.0	4.0	2.0	4.0
F_q before rod ejection	---	2.756	---	2.756
F_q after rod ejection	13.0	7.0	16.0	9.0
Number of operational pumps	2	4	2	4
Maximum fuel pellet average temperature (°F)	3470	4138	3508	4050
Maximum fuel center temperature (°F)	3881	4956	3862	4861
Maximum clad average temperature (°F)	2795	2597	2901	2523
Maximum fuel stored energy (cal/g)	147	182	149	177
Percent of fuel melted at hot spot	0	<10	0	<10

HZP – Hot Zero Power
HFP – Hot Full Power

Table 15.4-4 (Sheet 1 of 2)

To follow

Table 15.4-4 (Sheet 2 of 2)

To follow

Table 15.4-5

To follow

Table 15.4-6

To follow

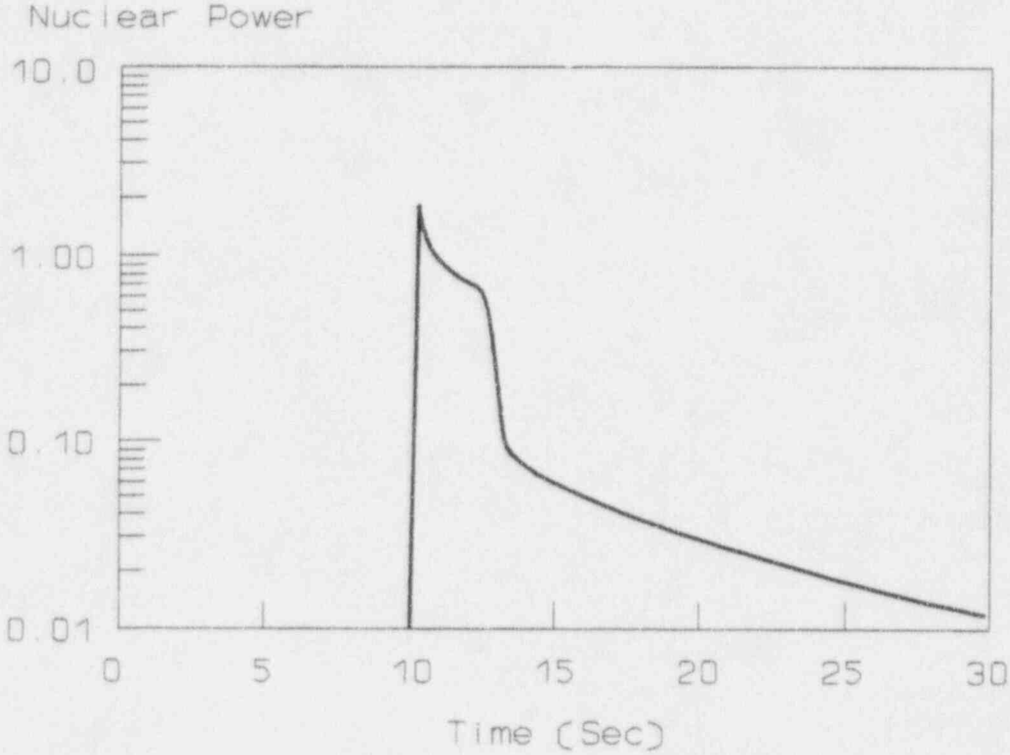

Figure 15.4.1-1

RCCA Withdrawal from Subcritical Nuclear Power

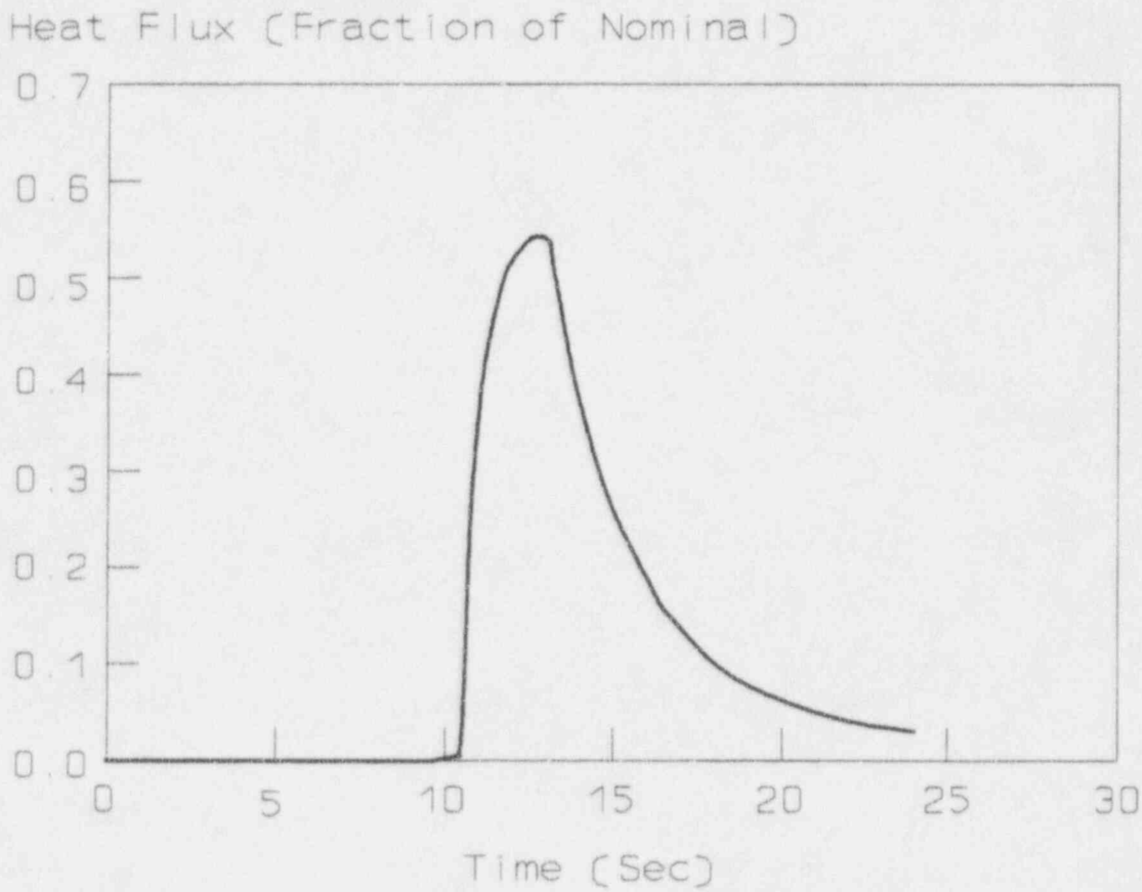

Figure 15.4.1-2

RCCA Withdrawal from Subcritical Thermal Flux

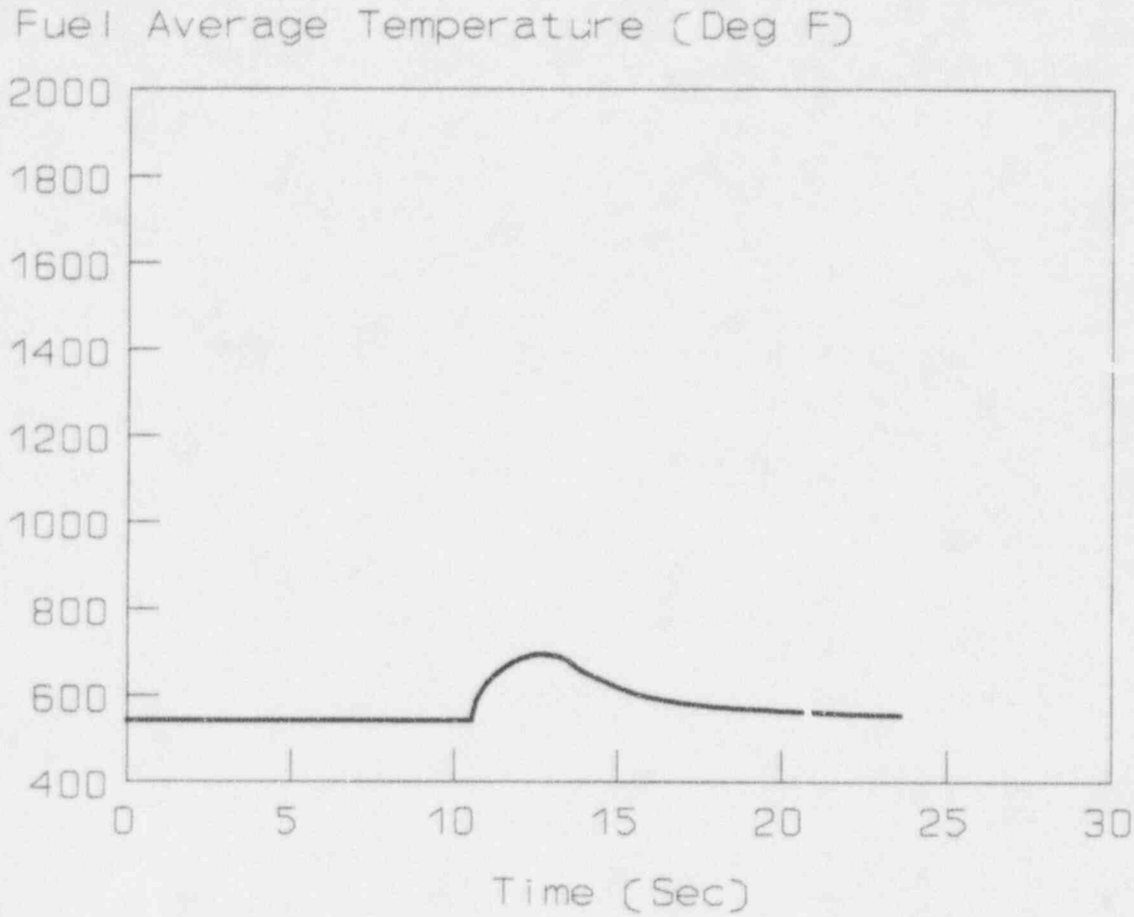

Figure 15.4.1-3 A

RCCA Withdrawal from Subcritical
Fuel Average Temperature

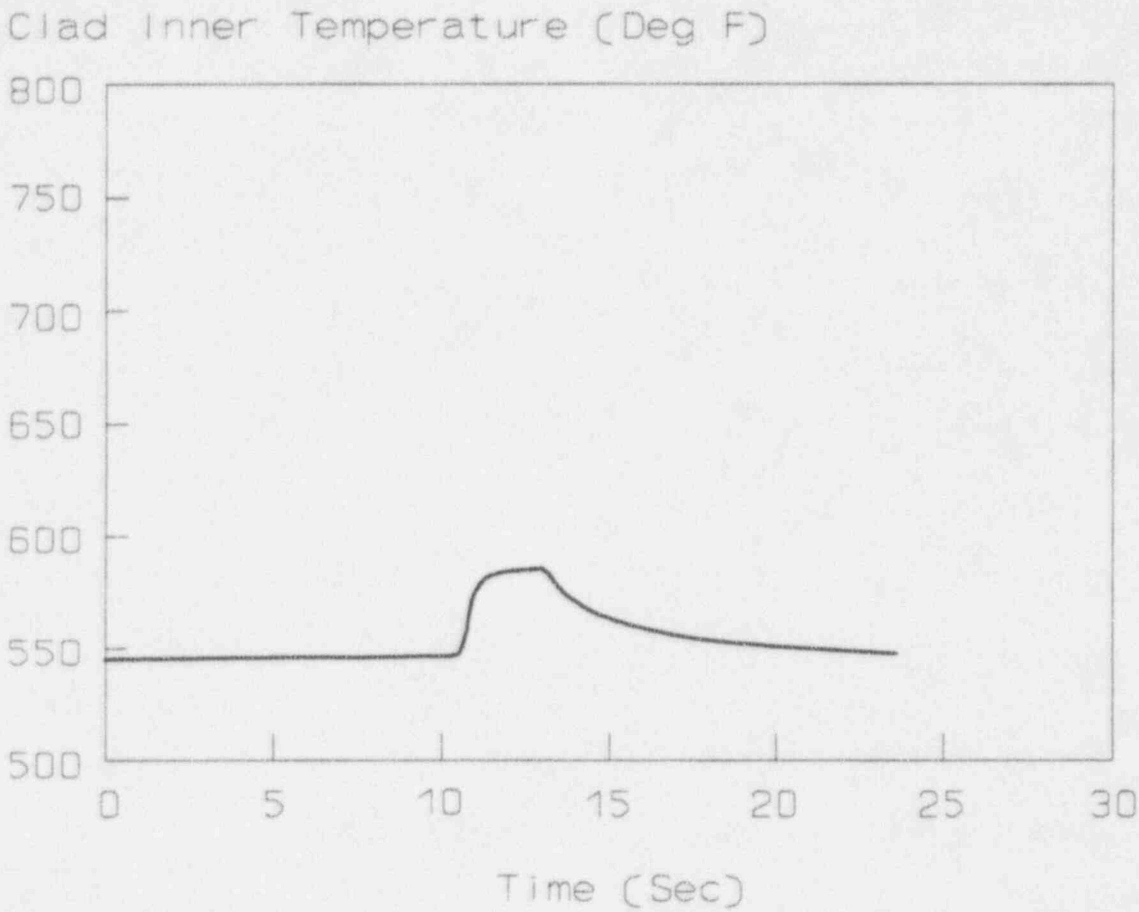

Figure 15.4.1-3 B

RCCA Withdrawal from Subcritical
Clad Inner Temperature

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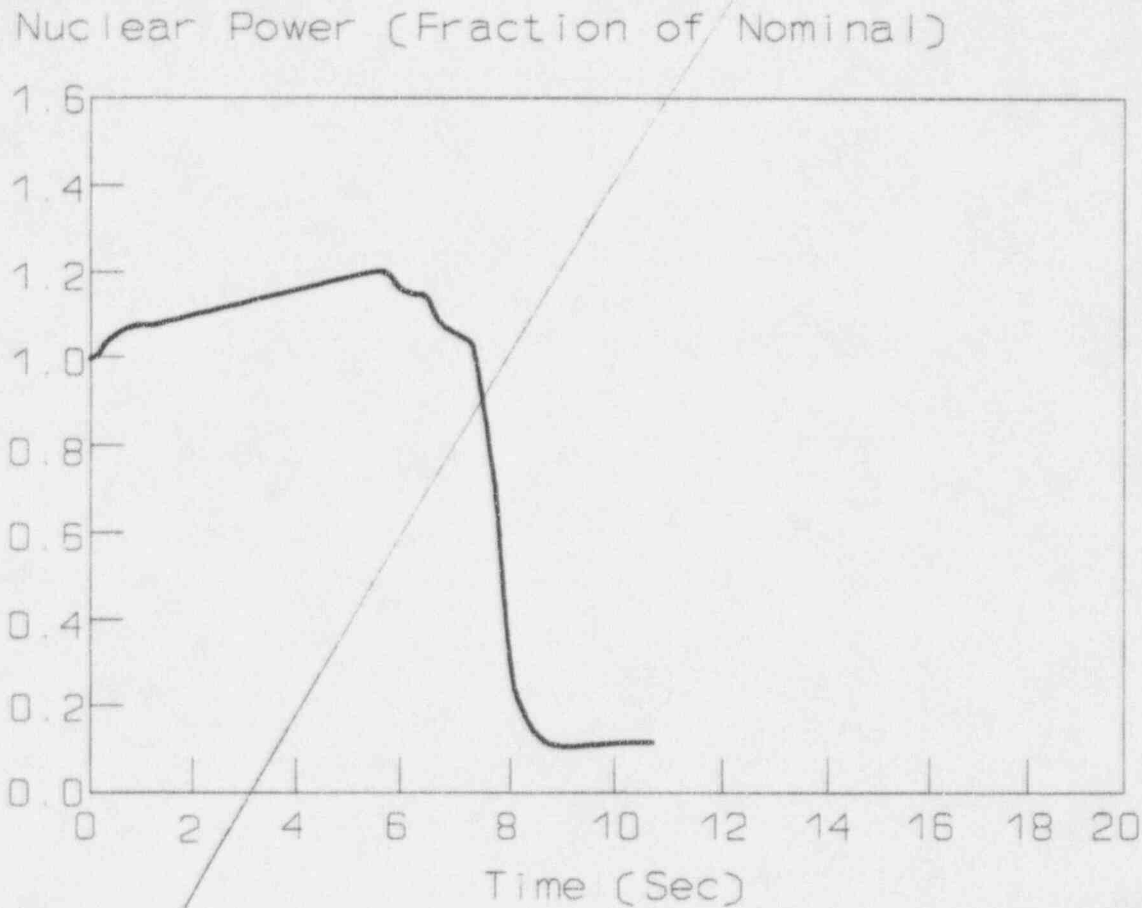
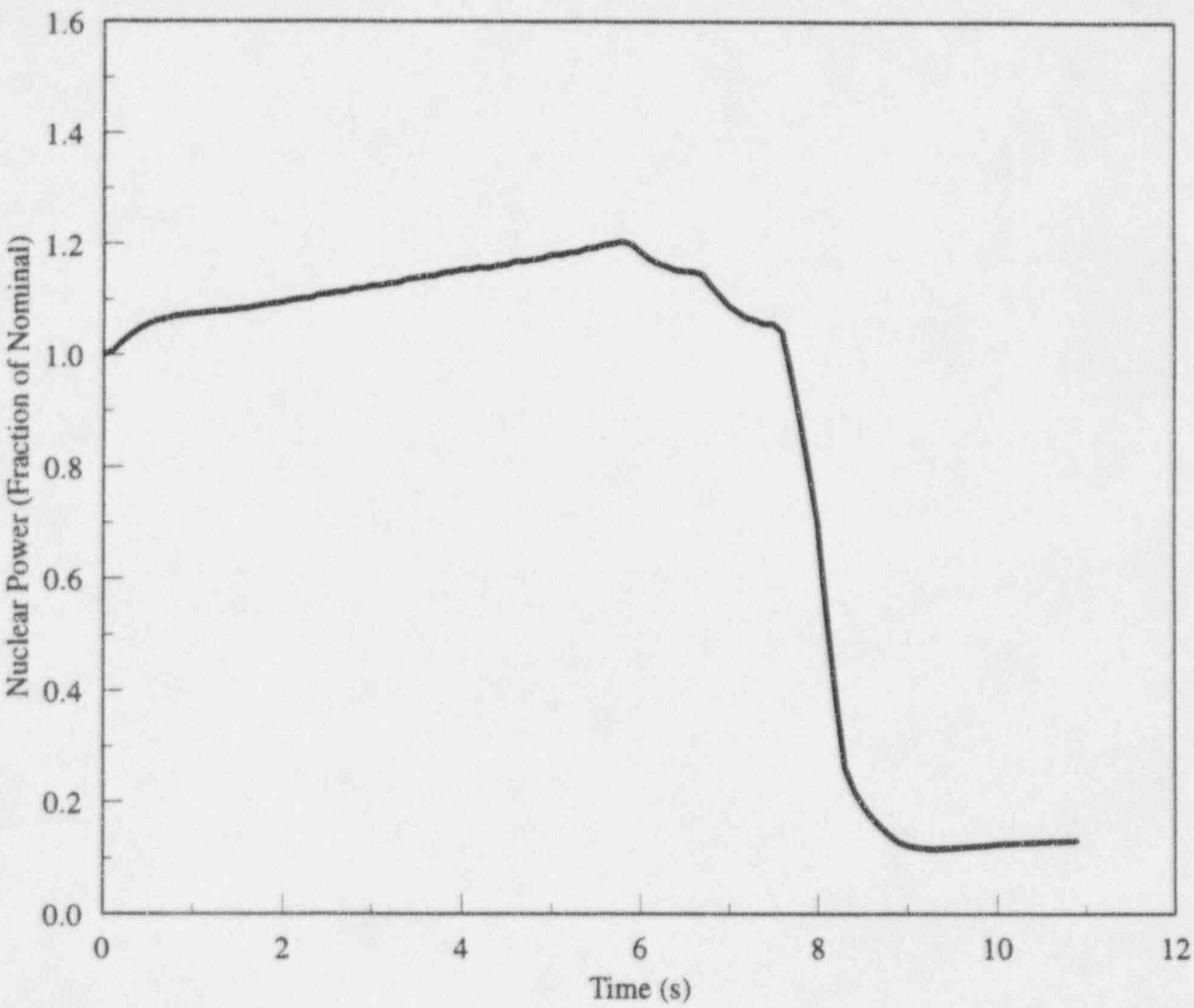


Figure 15.4.2-1

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-1 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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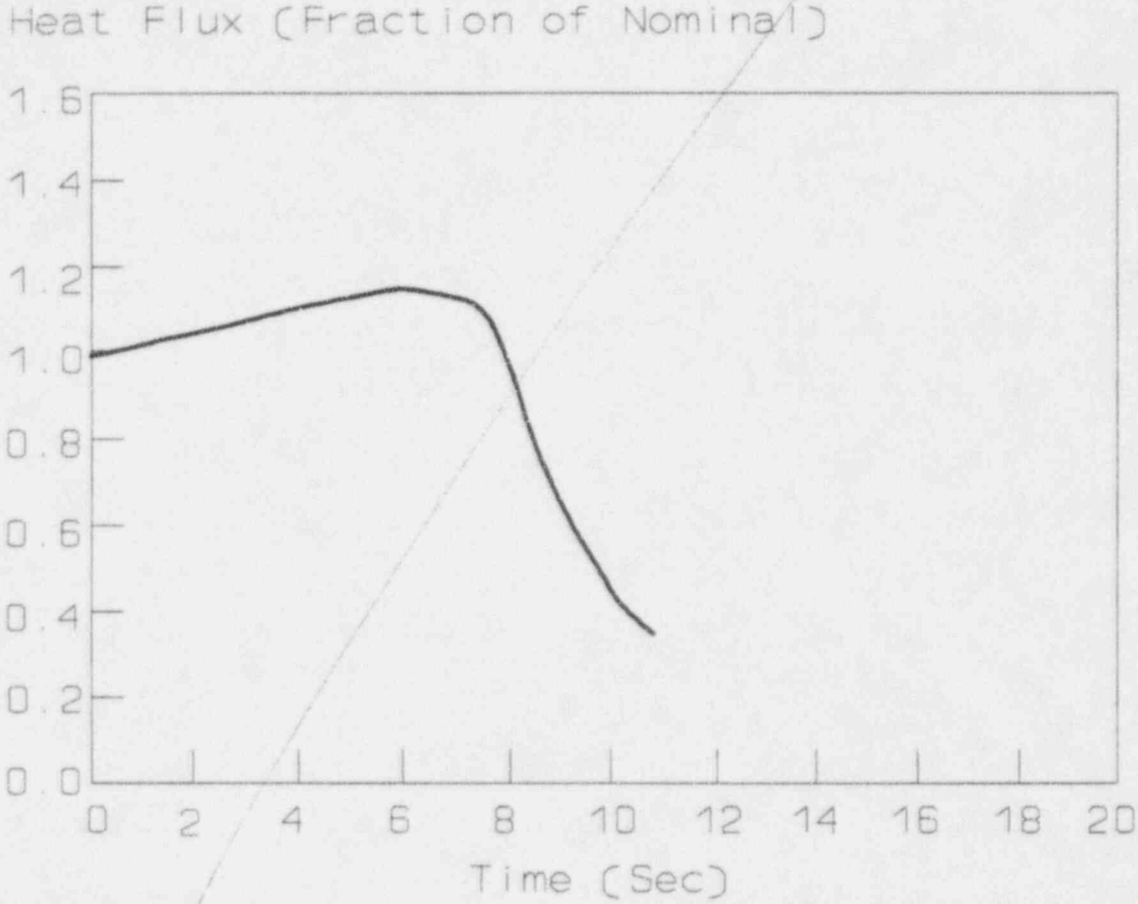
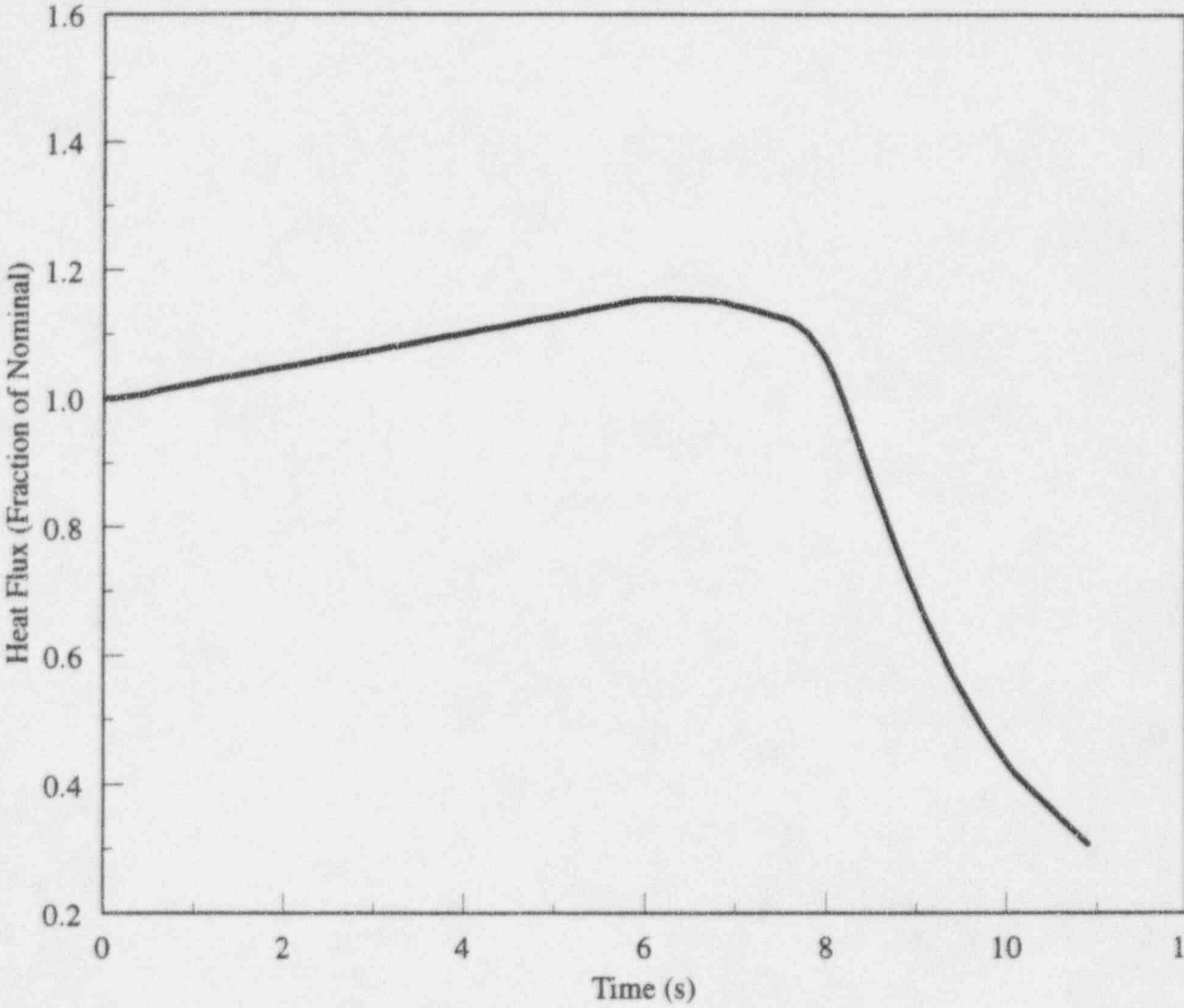


Figure 15.4.2-2

Thermal Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-2 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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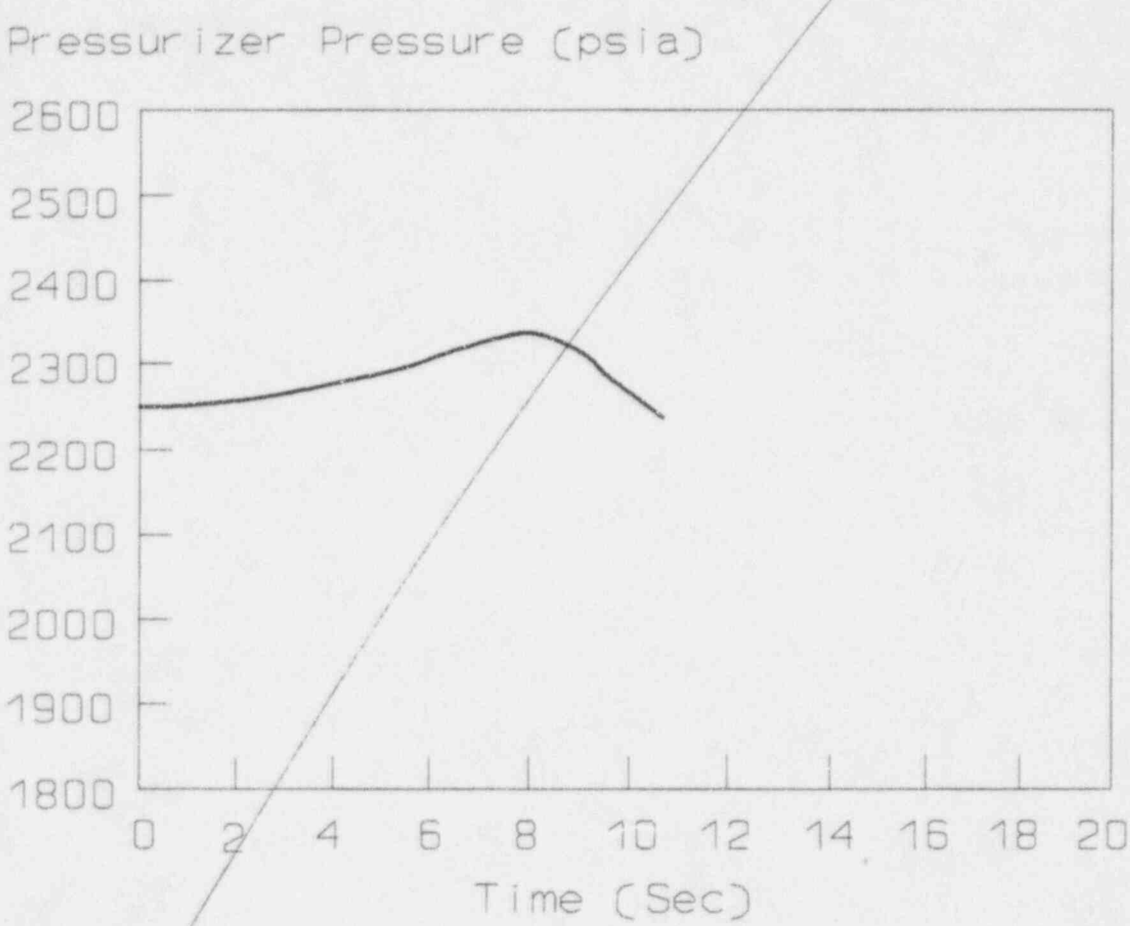
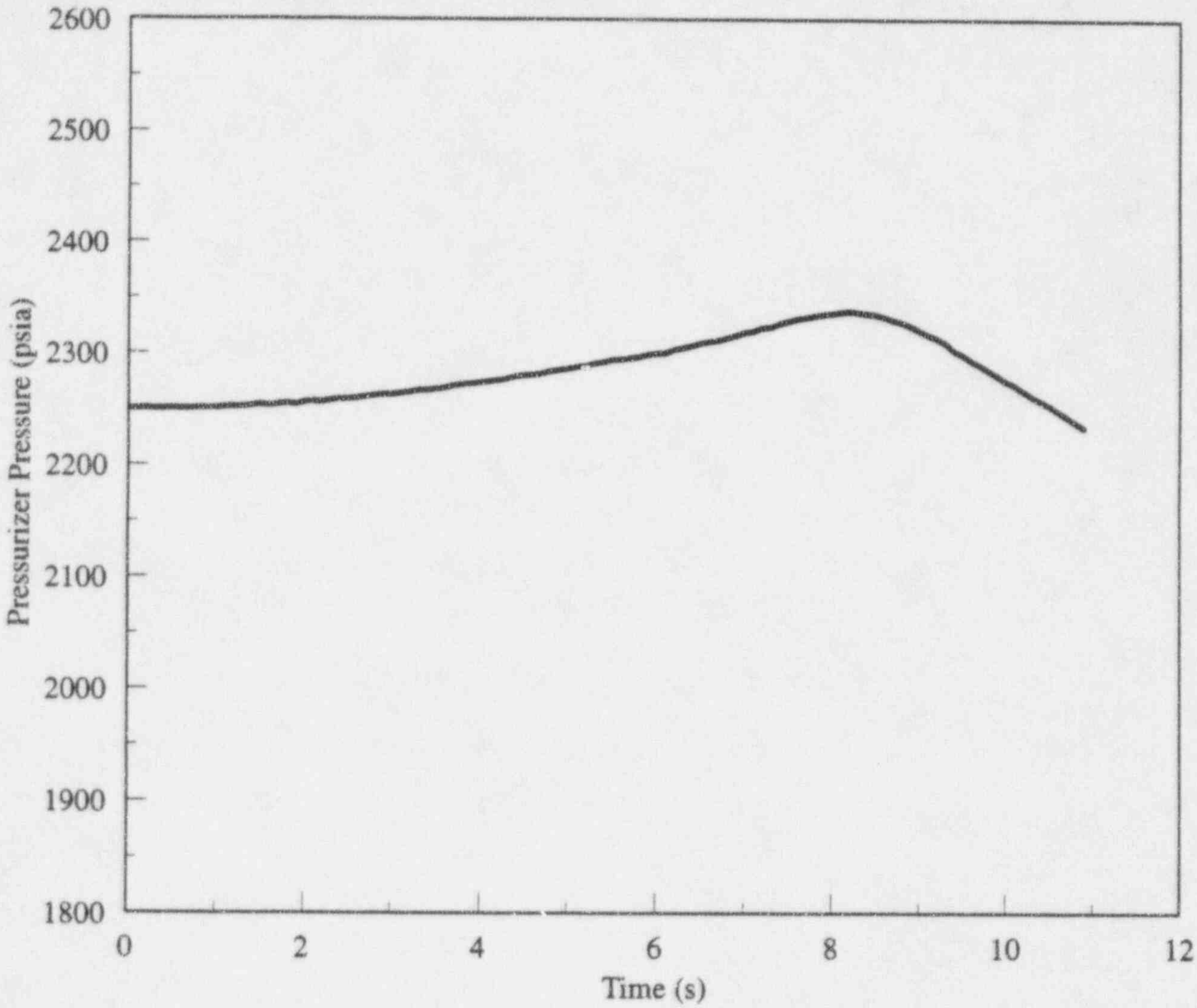


Figure 15.4.2-3

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-3 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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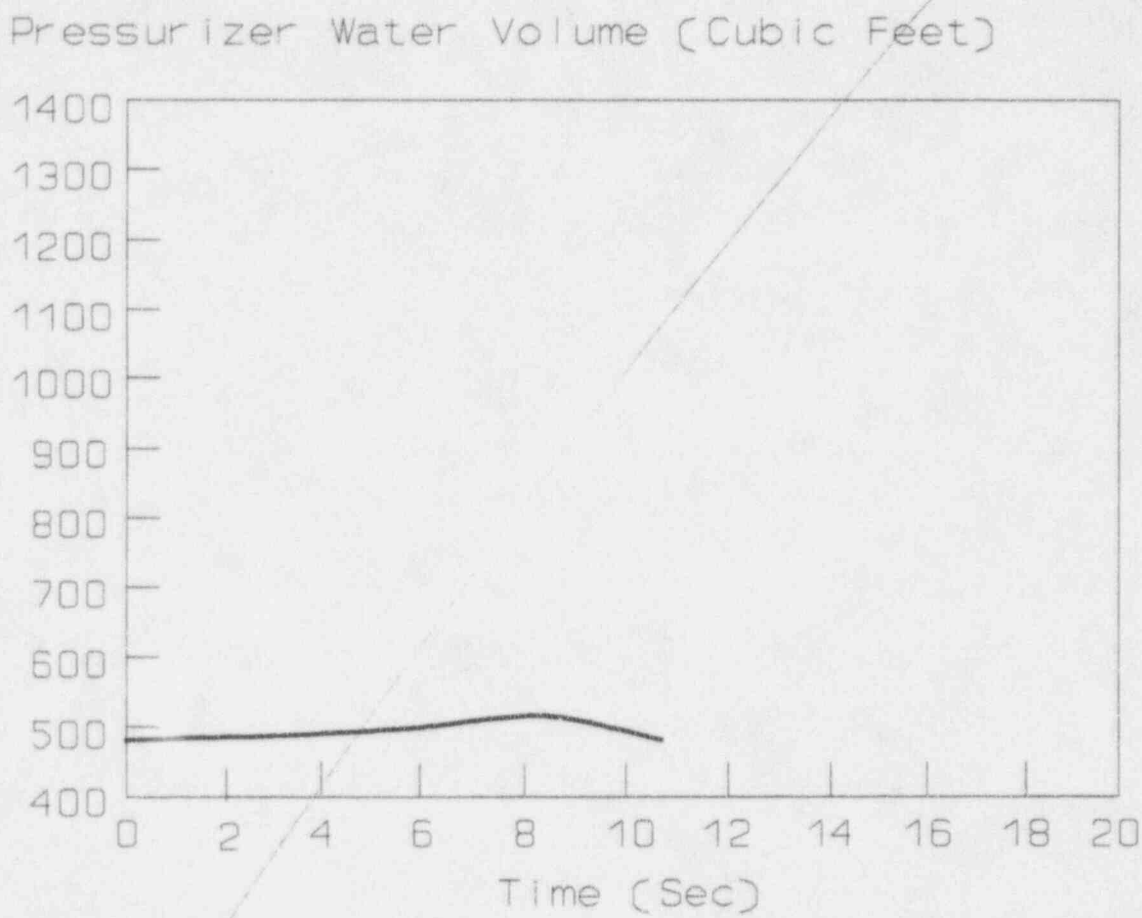
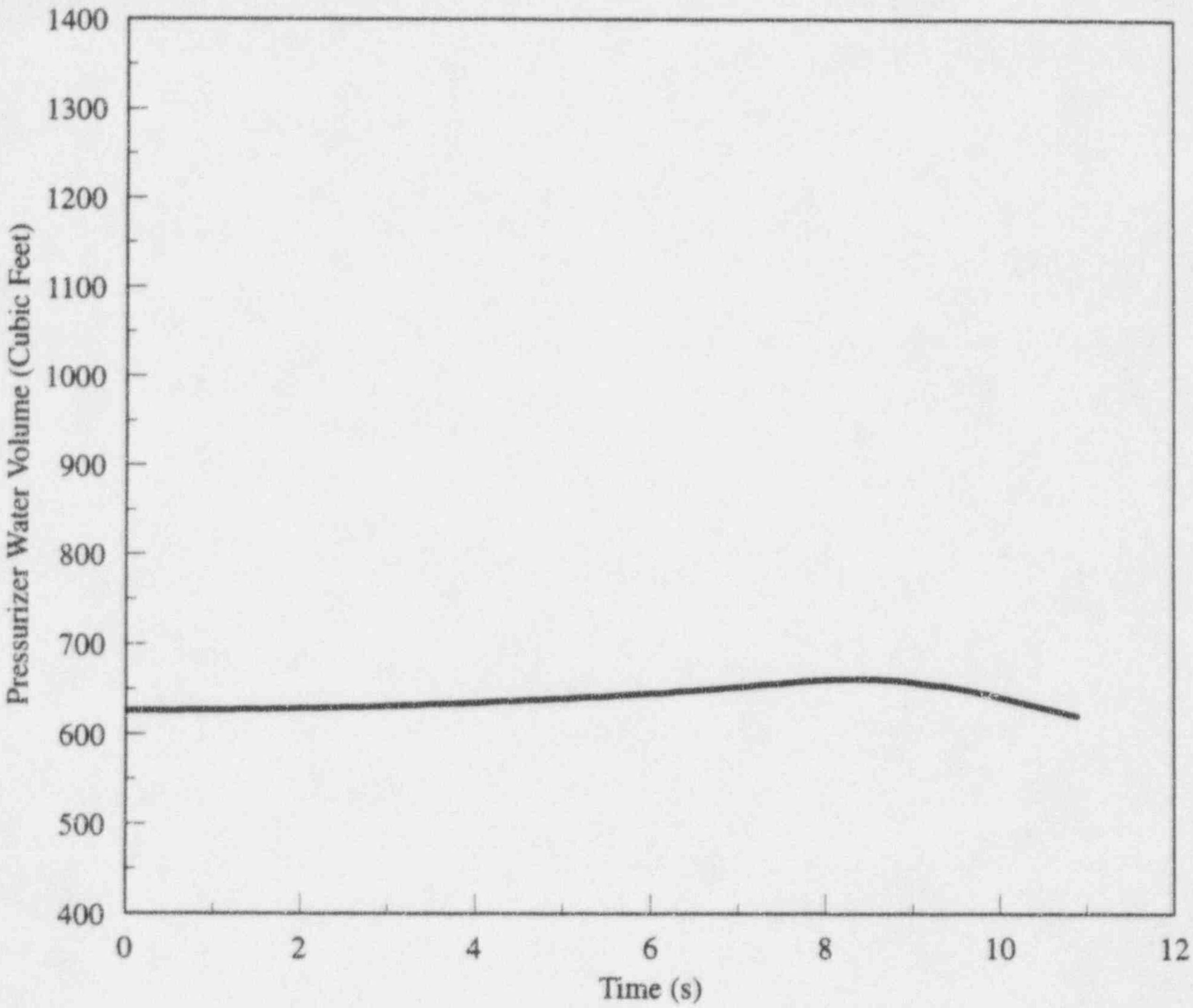


Figure 15.4.2-4

Pressurizer Water Volume for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-4 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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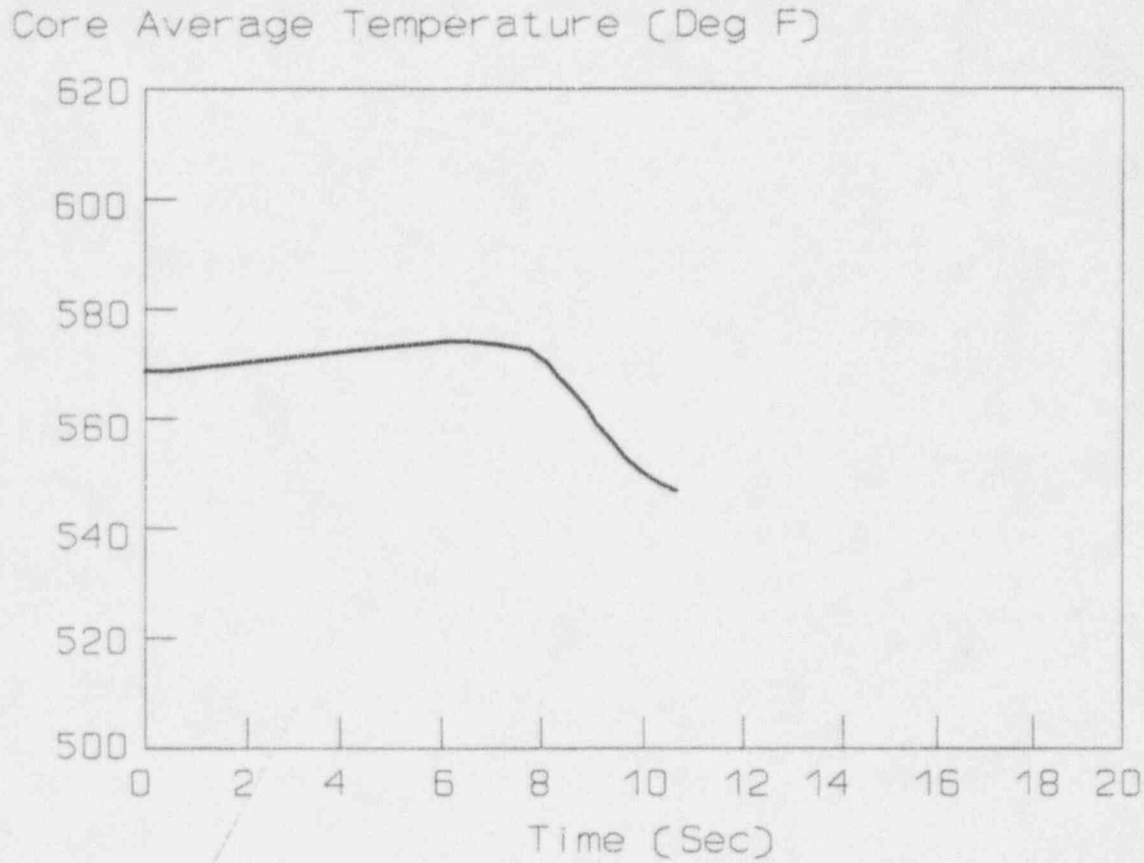
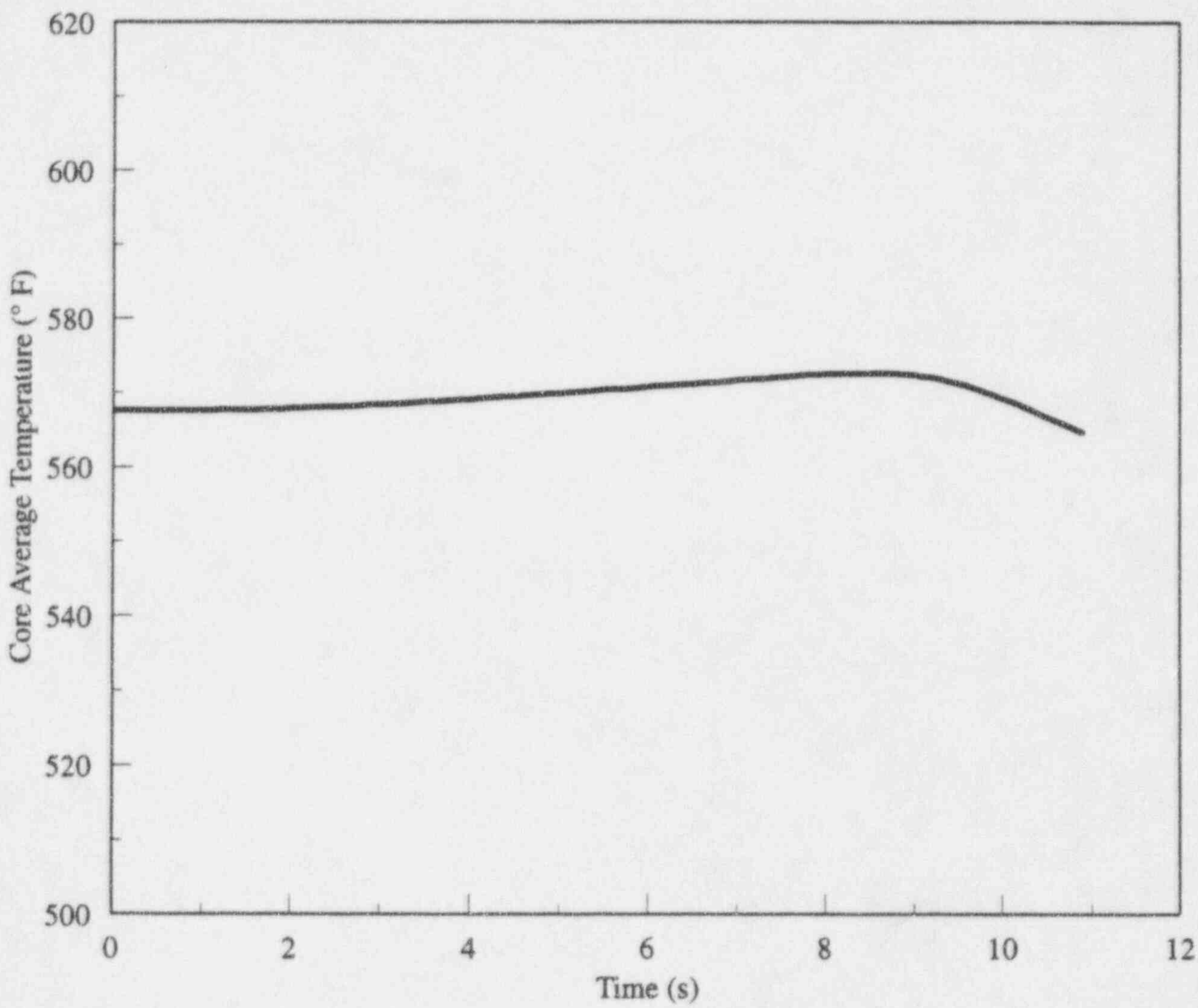


Figure 15.4.2-5

Core Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-5 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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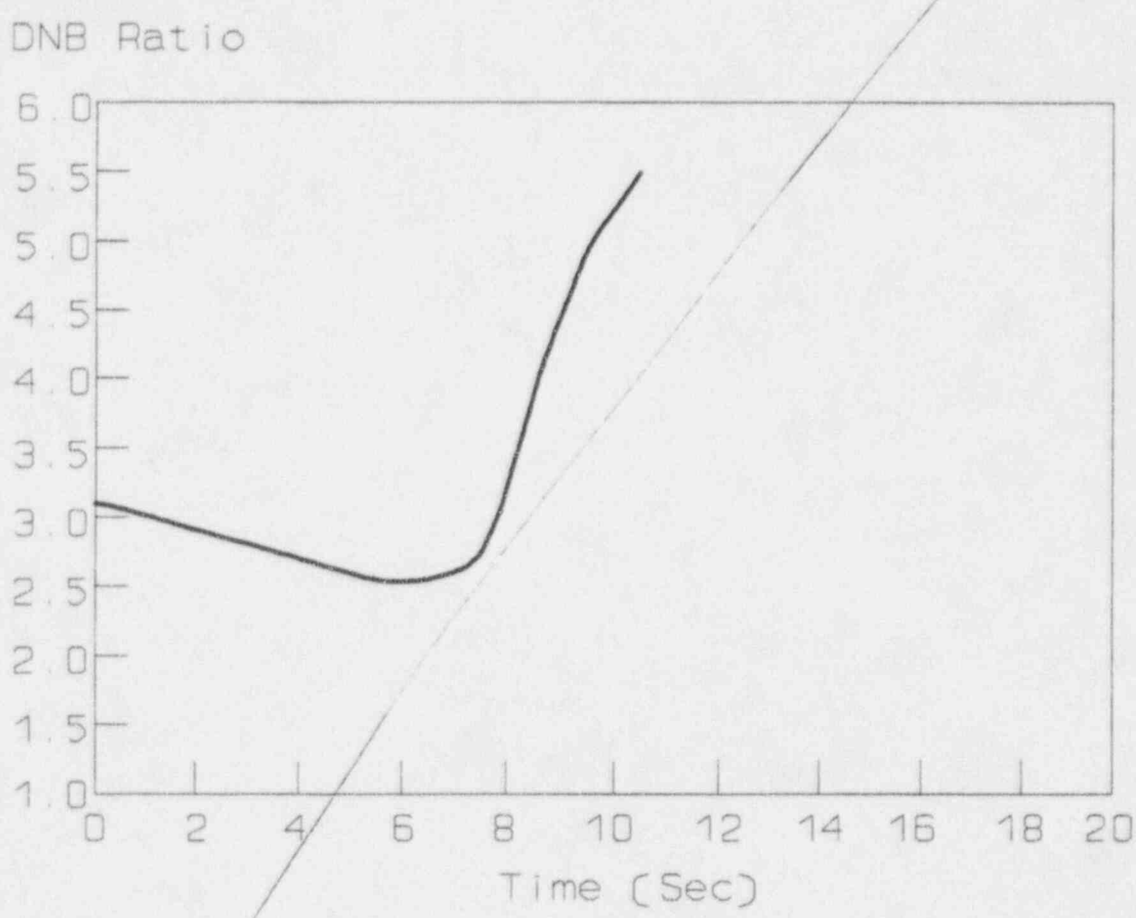
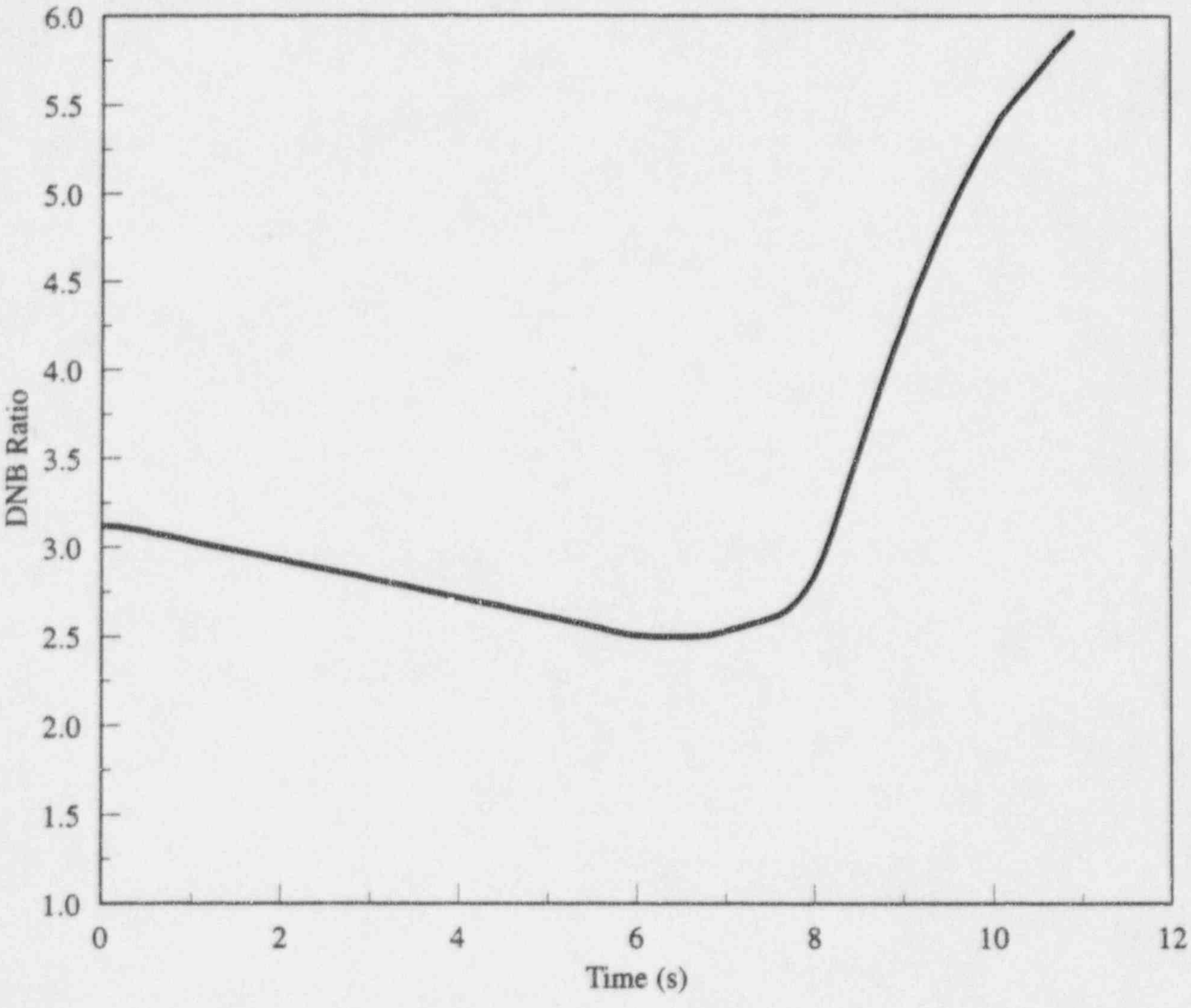


Figure 15.4.2-6

DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Withdrawal Rate)

Figure 15.4.2-6 RWAP 100% Maximum Feedback 75 pcm/s Insertion Rate



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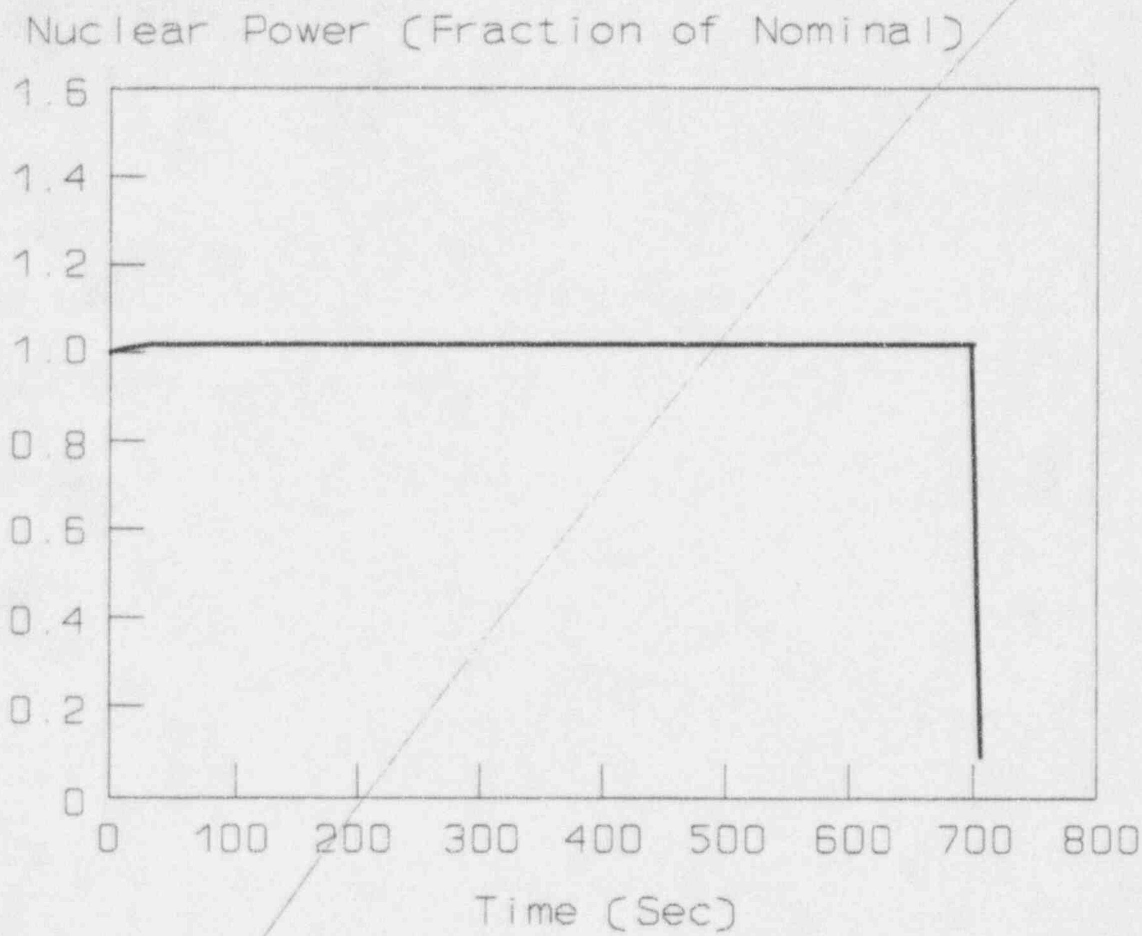
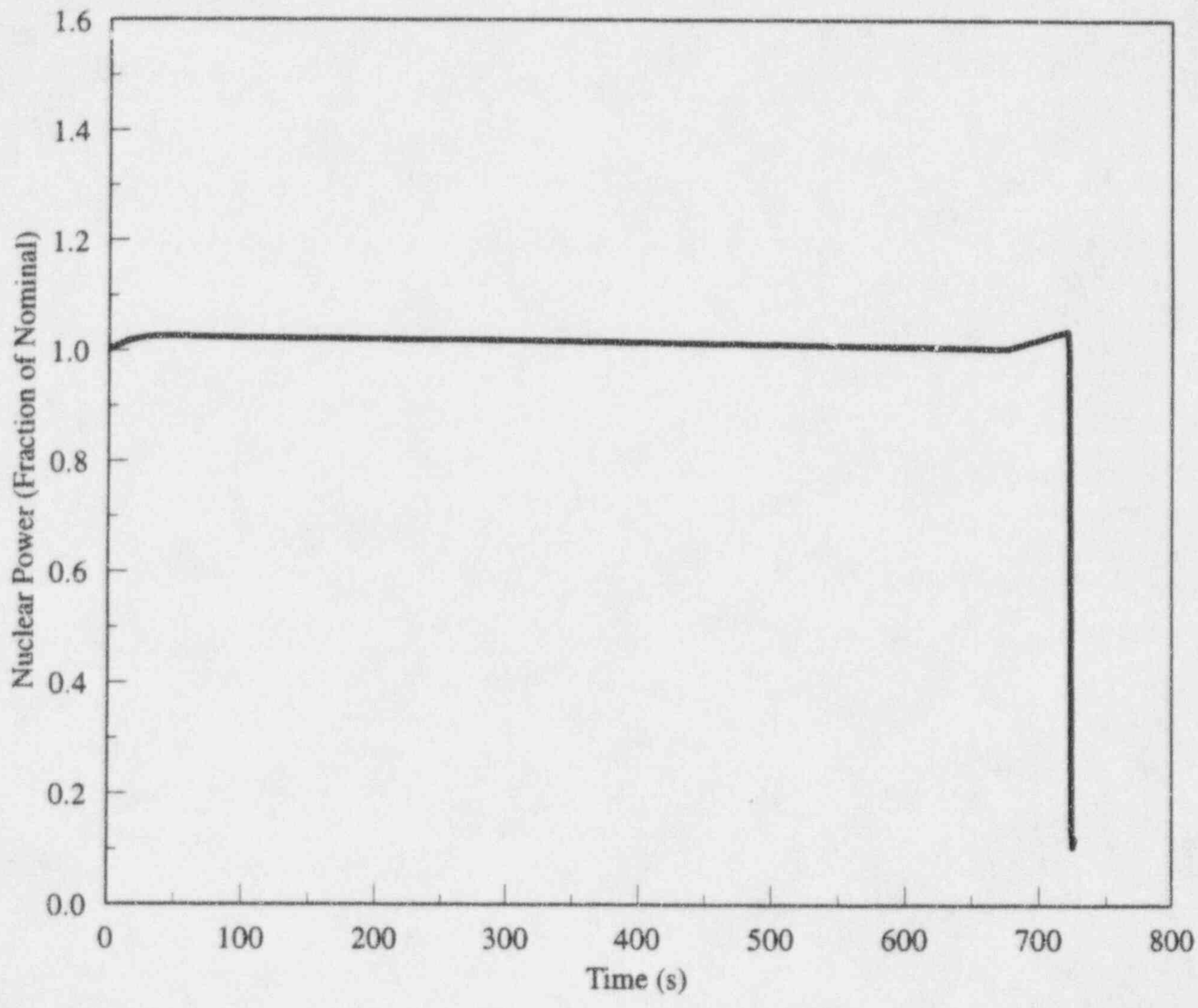


Figure 15.4.2-7

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-7 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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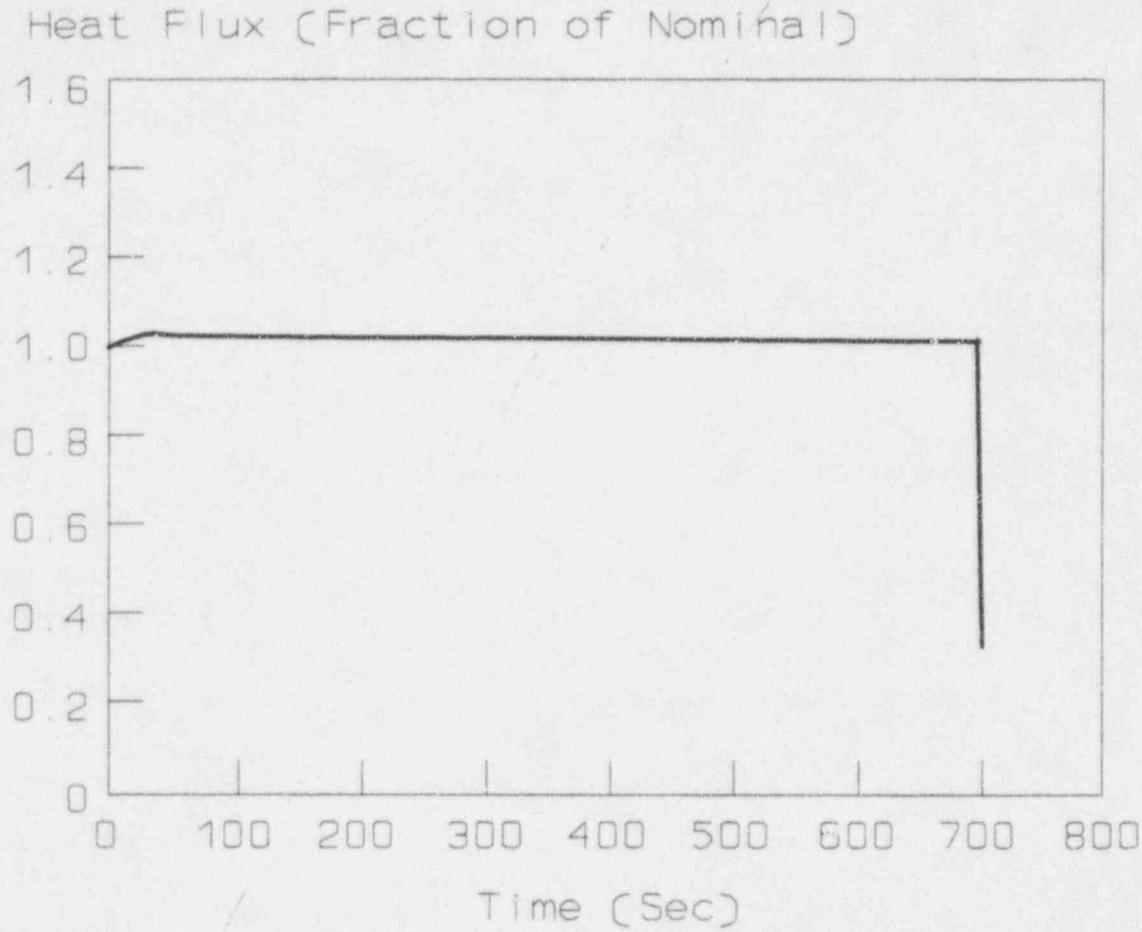
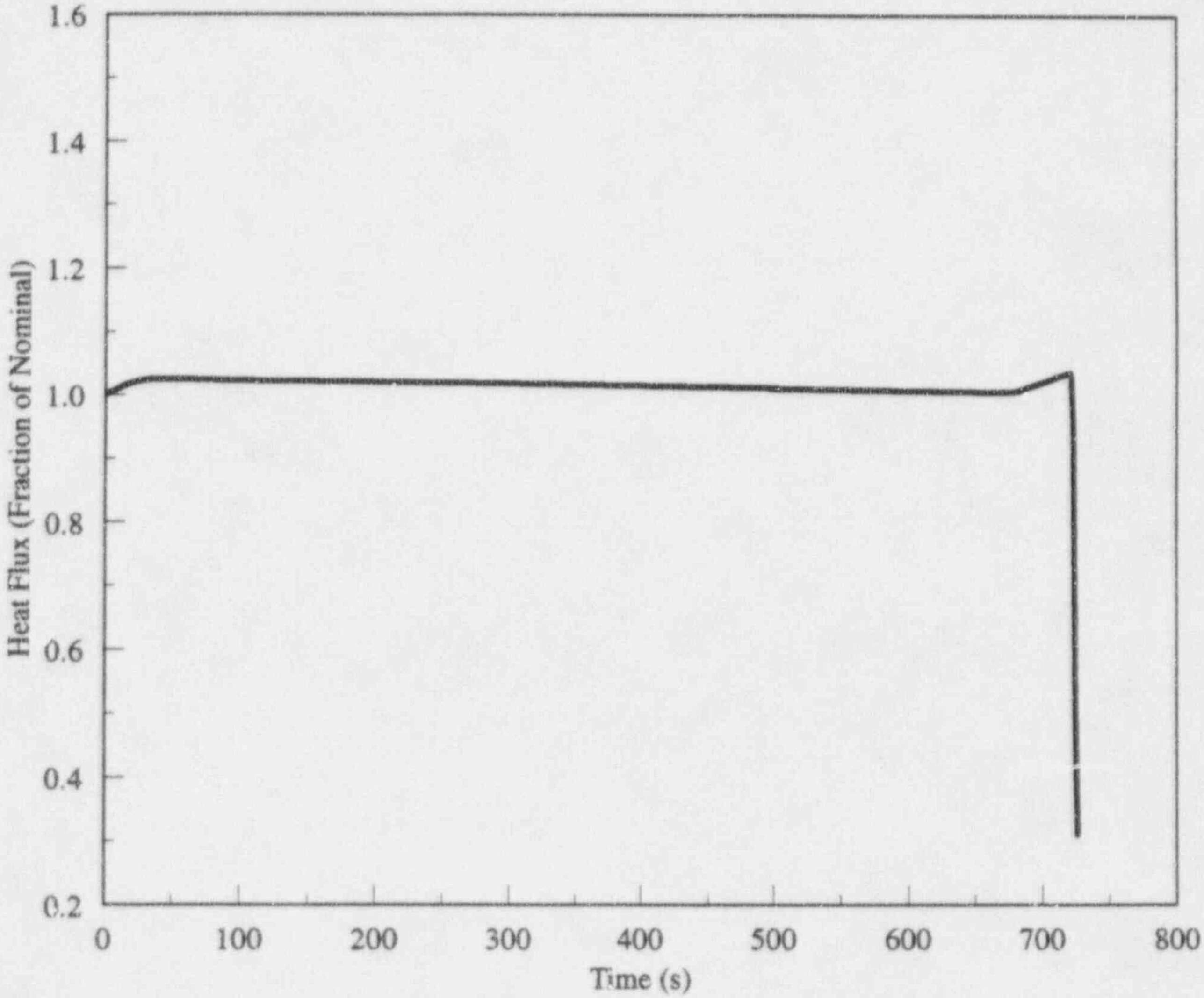


Figure 15.4.2-8

Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-8 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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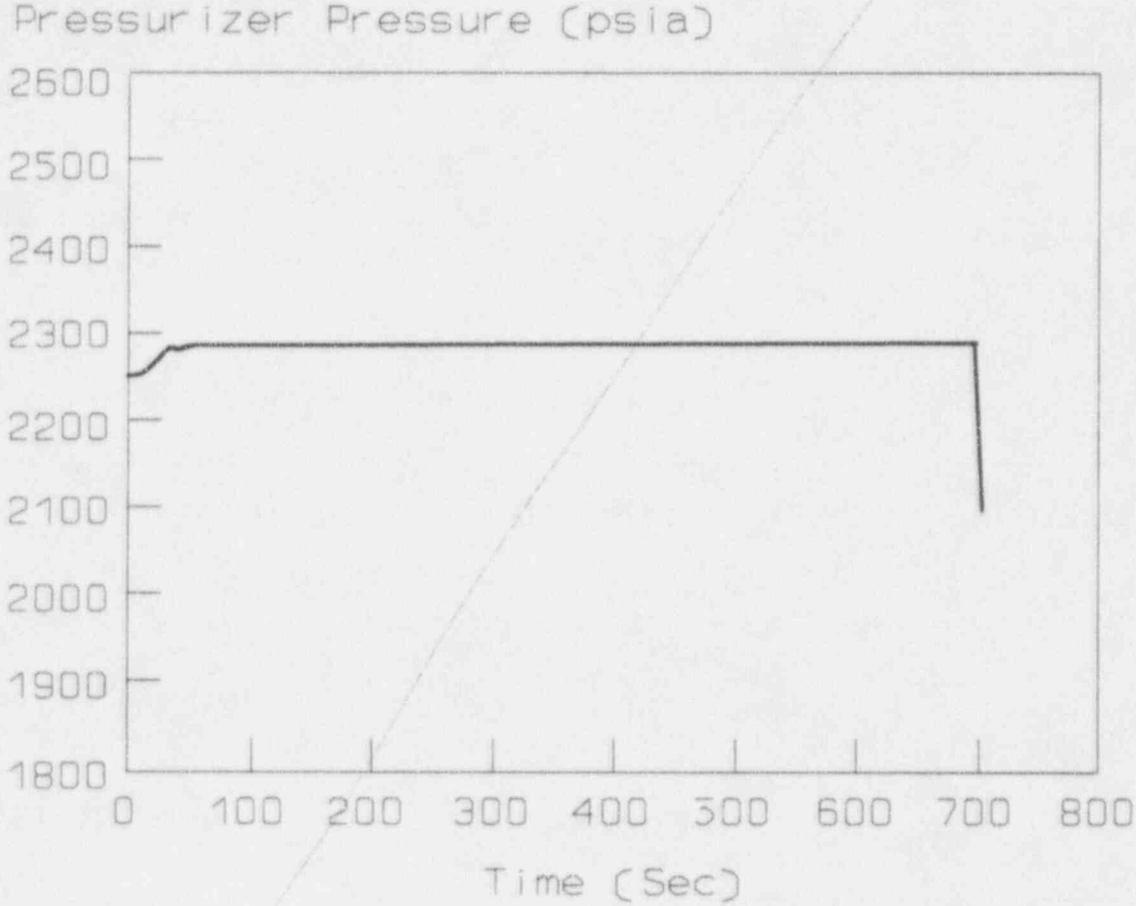
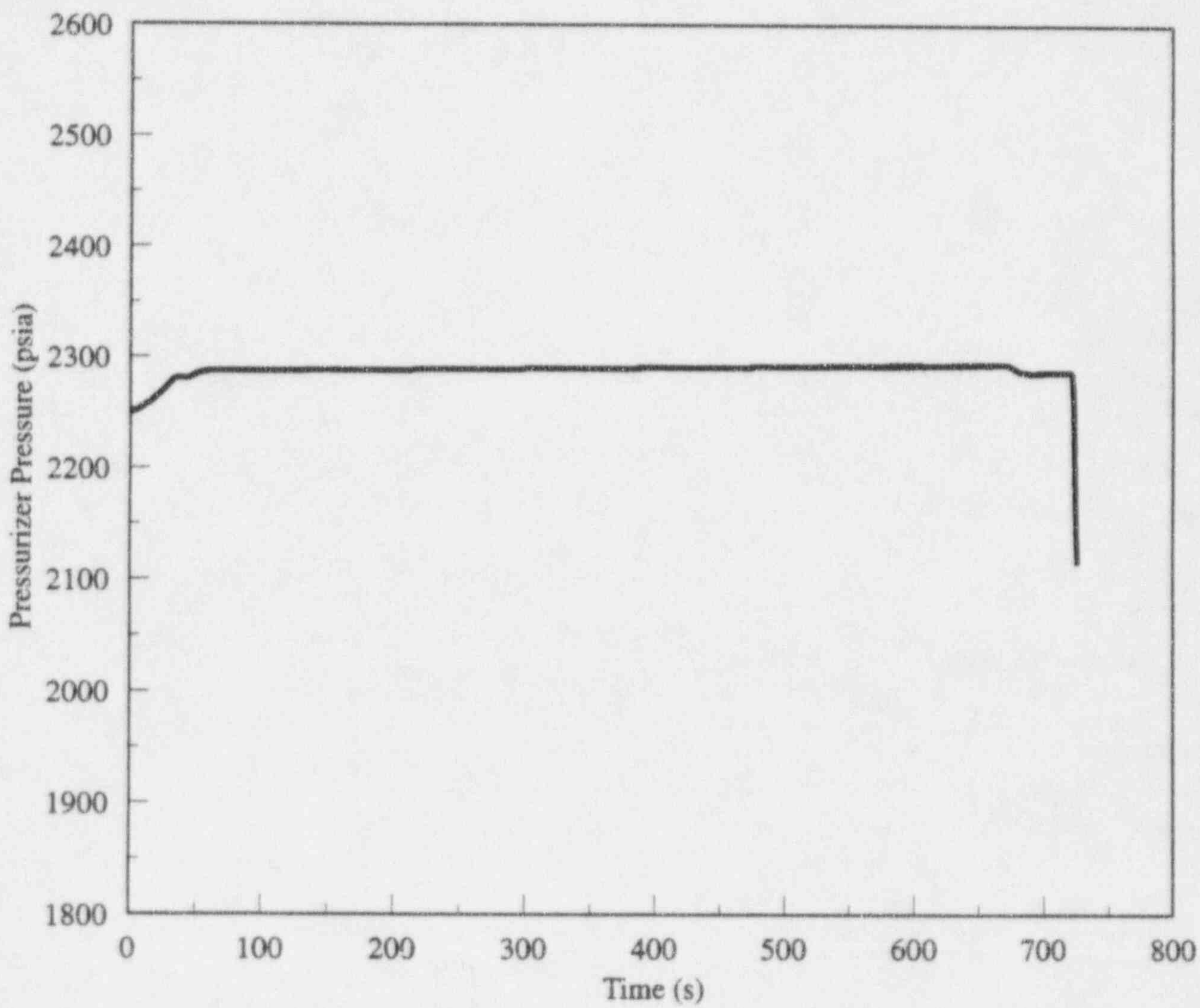


Figure 15.4.2-9

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-9 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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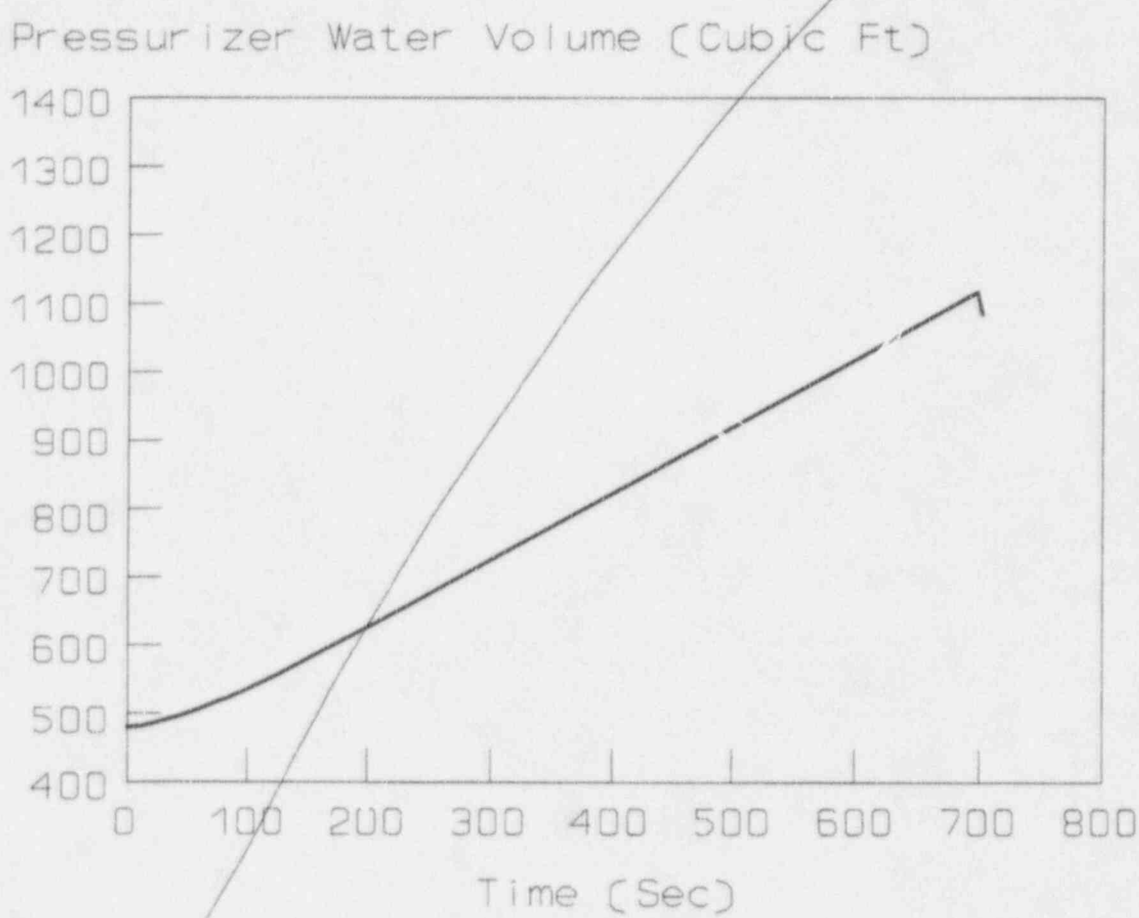
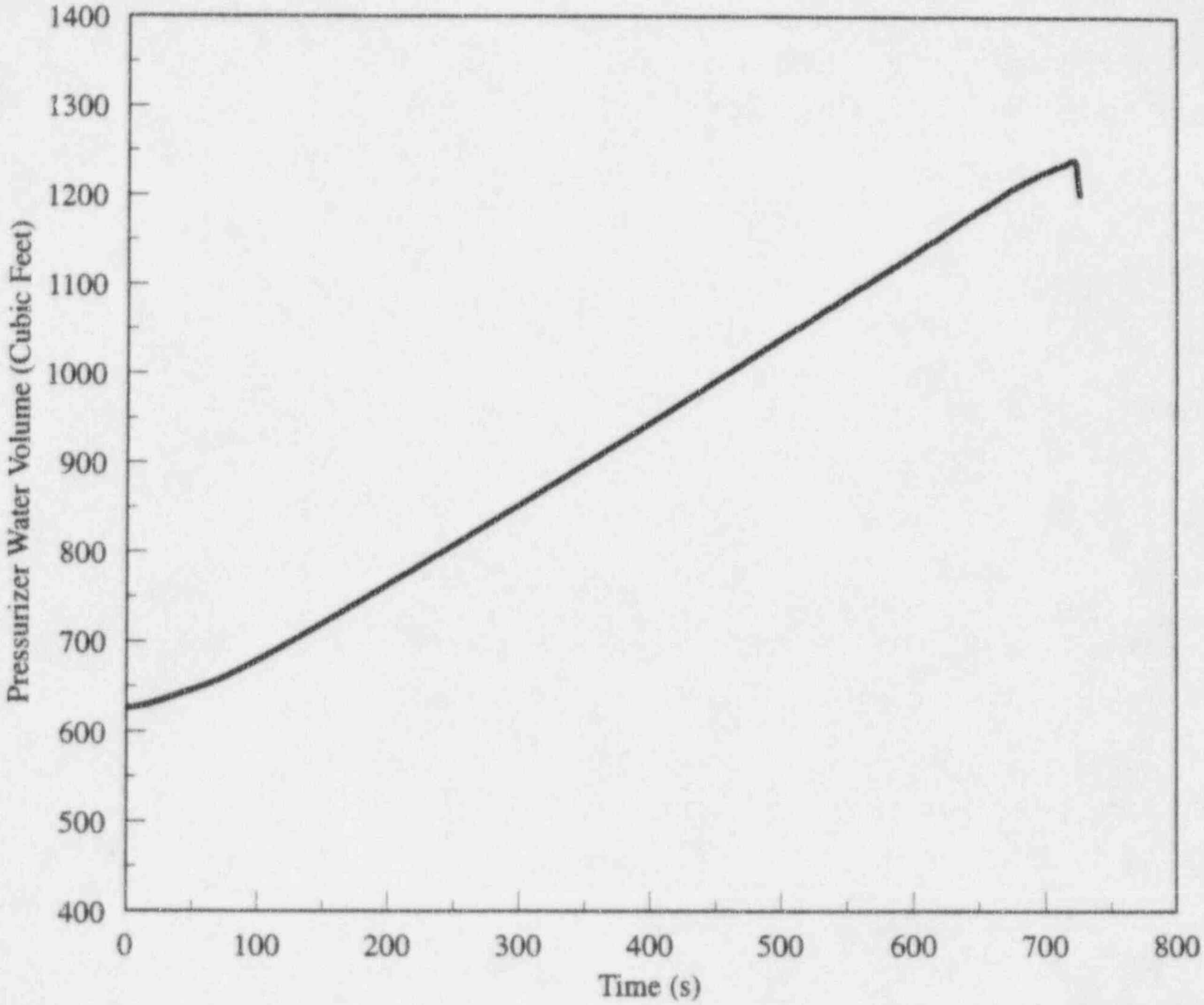


Figure 15.4.2-10

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-10 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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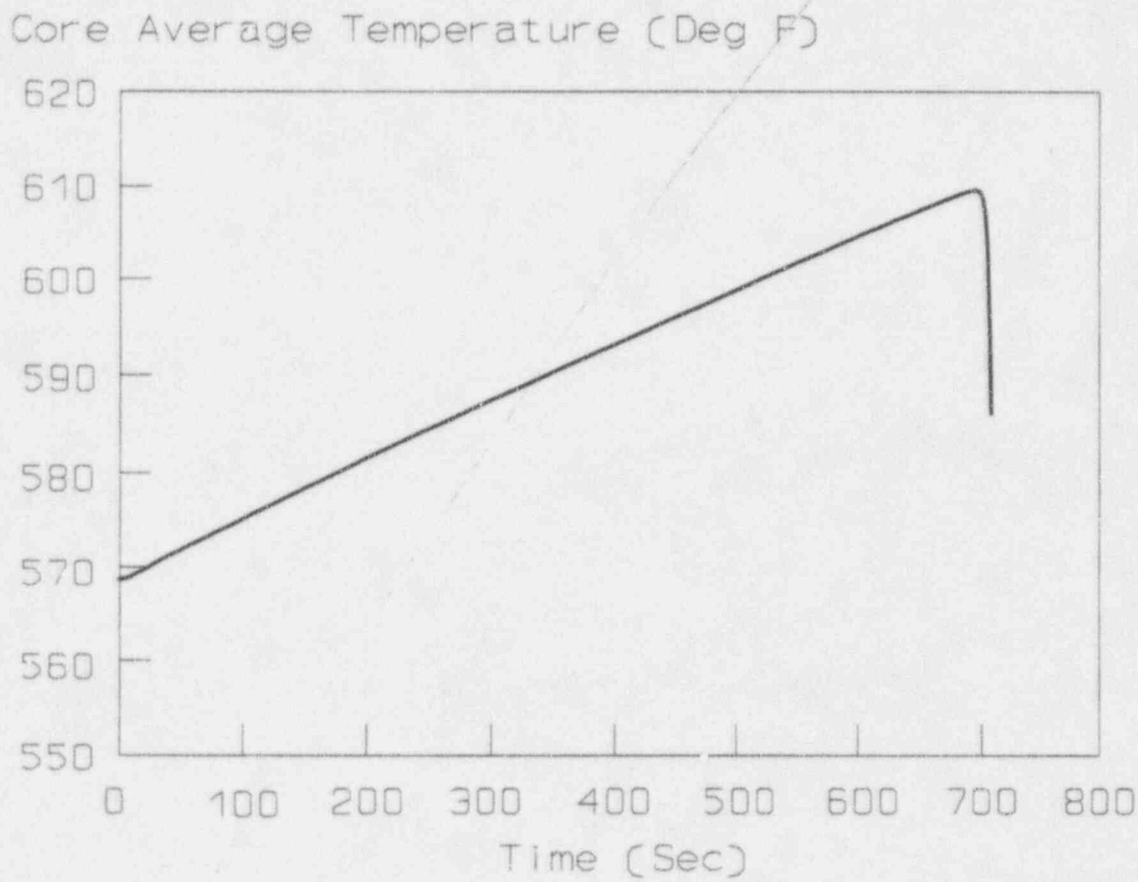
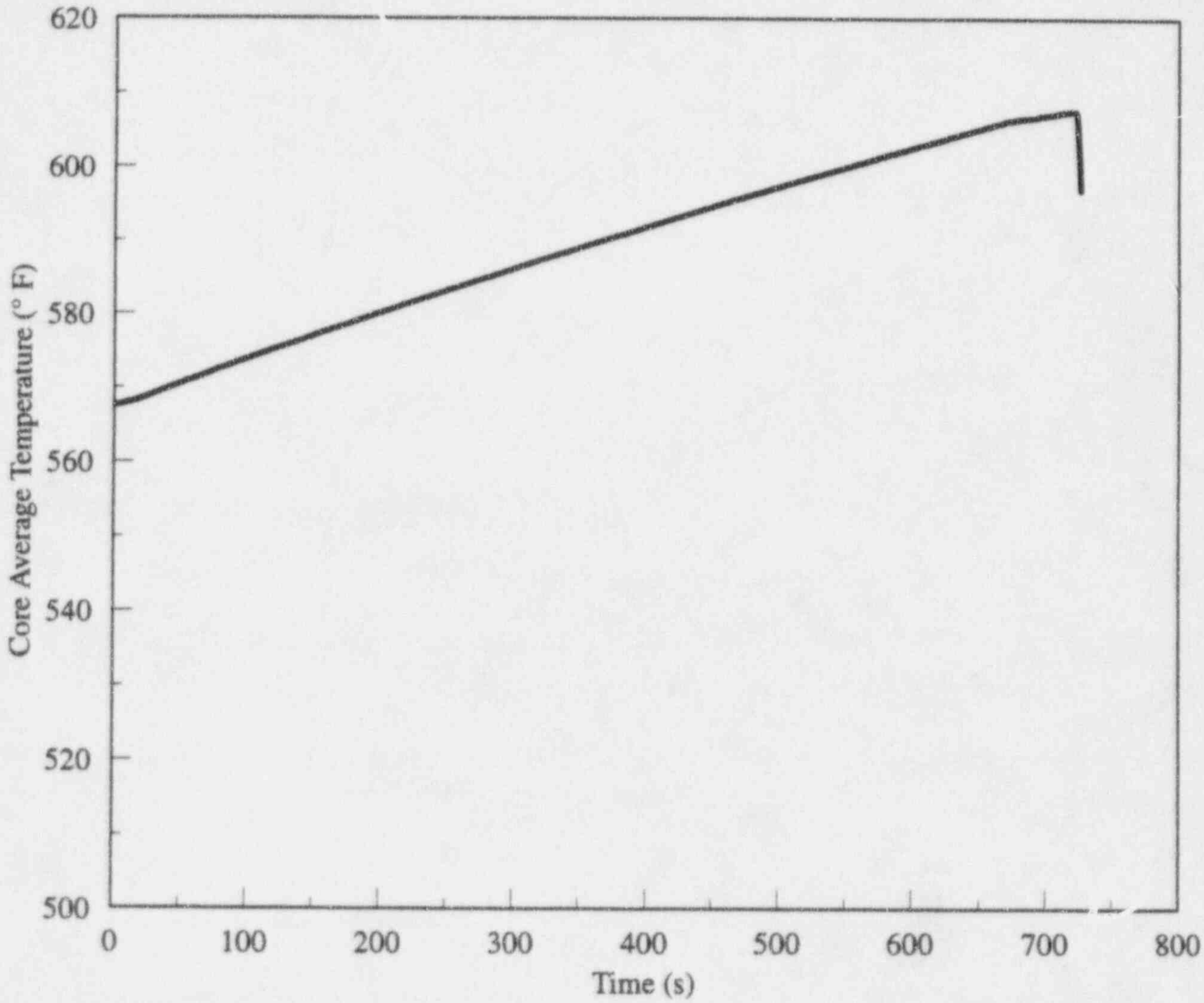


Figure 15.4.2-11

Core Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-11 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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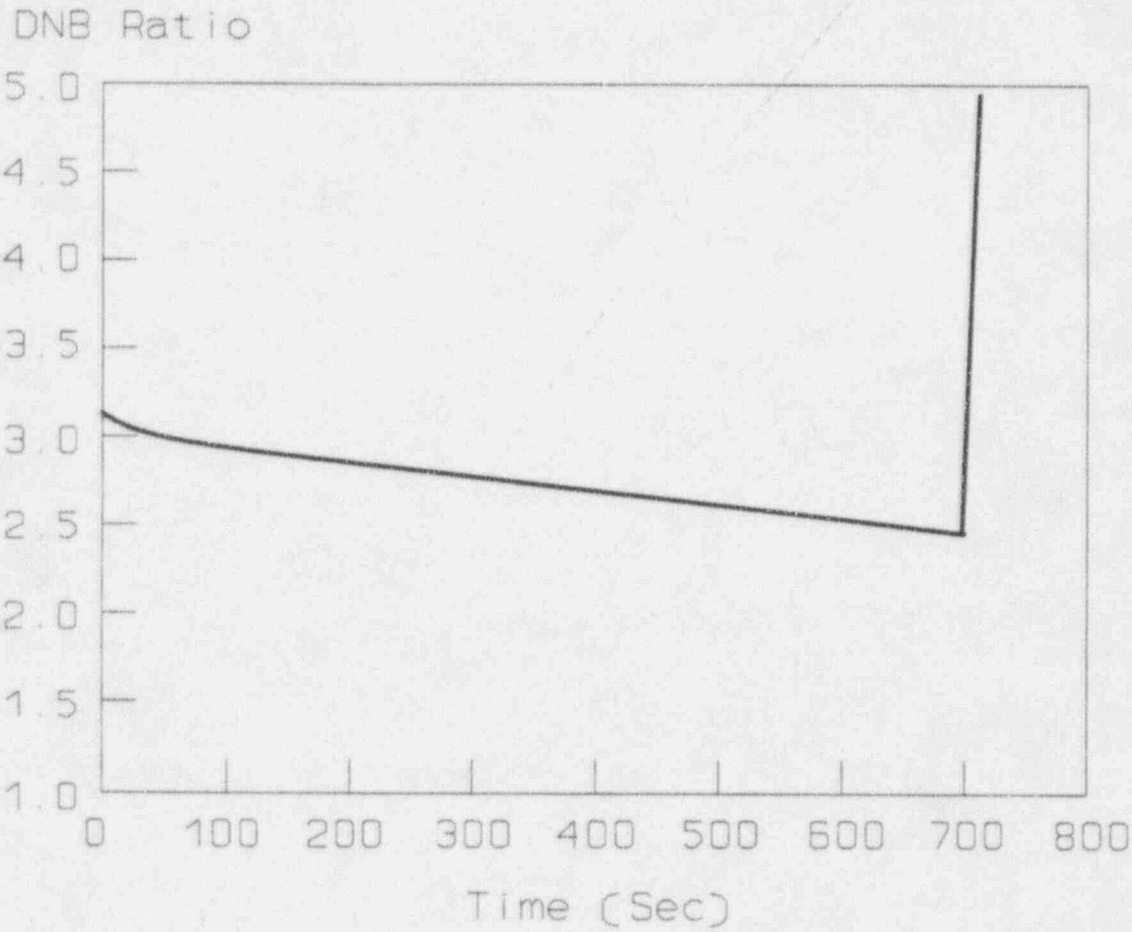
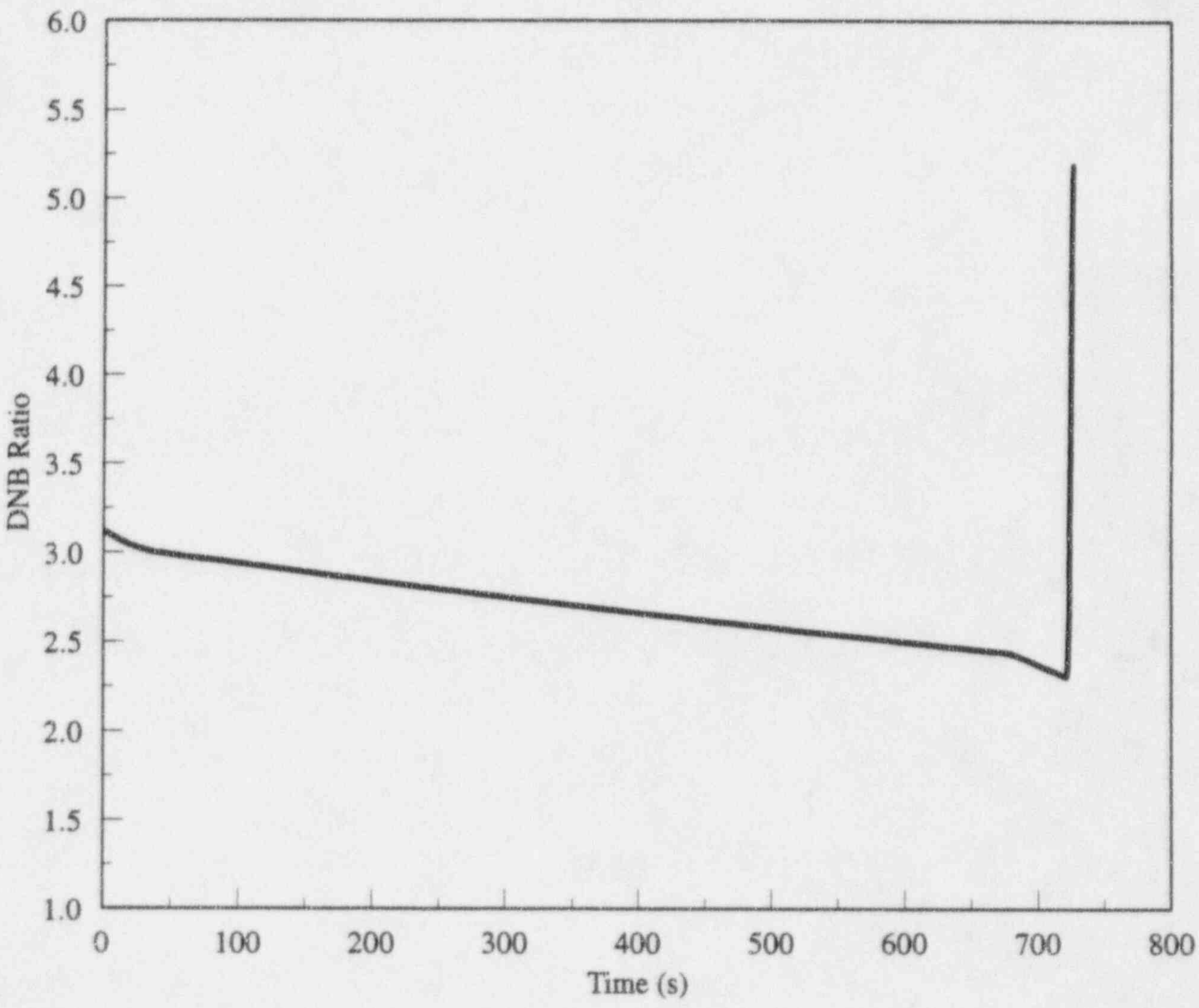


Figure 15.4.2-12

DNBR Ratio Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Withdrawal Rate)

Figure 15.4.2-12 RWAP 100% Maximum Feedback 3 pcm/s Insertion Rate



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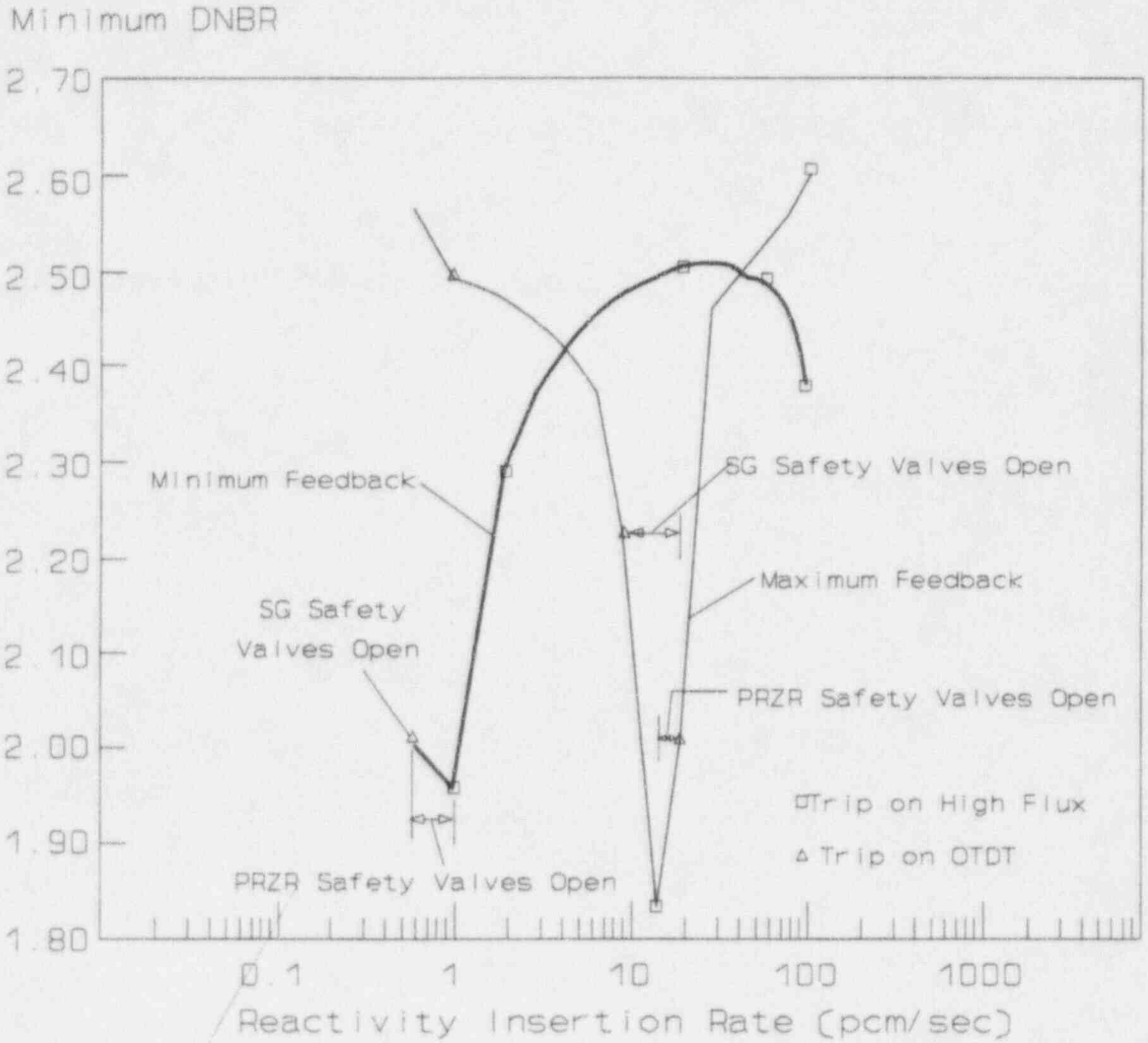


Figure 15.4.2-13

Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal at 100 Percent Power

Figure 15.4.2-13 RWAP 100 Percent Power Minimum DNBR vs. Reactivity Insertion Rate

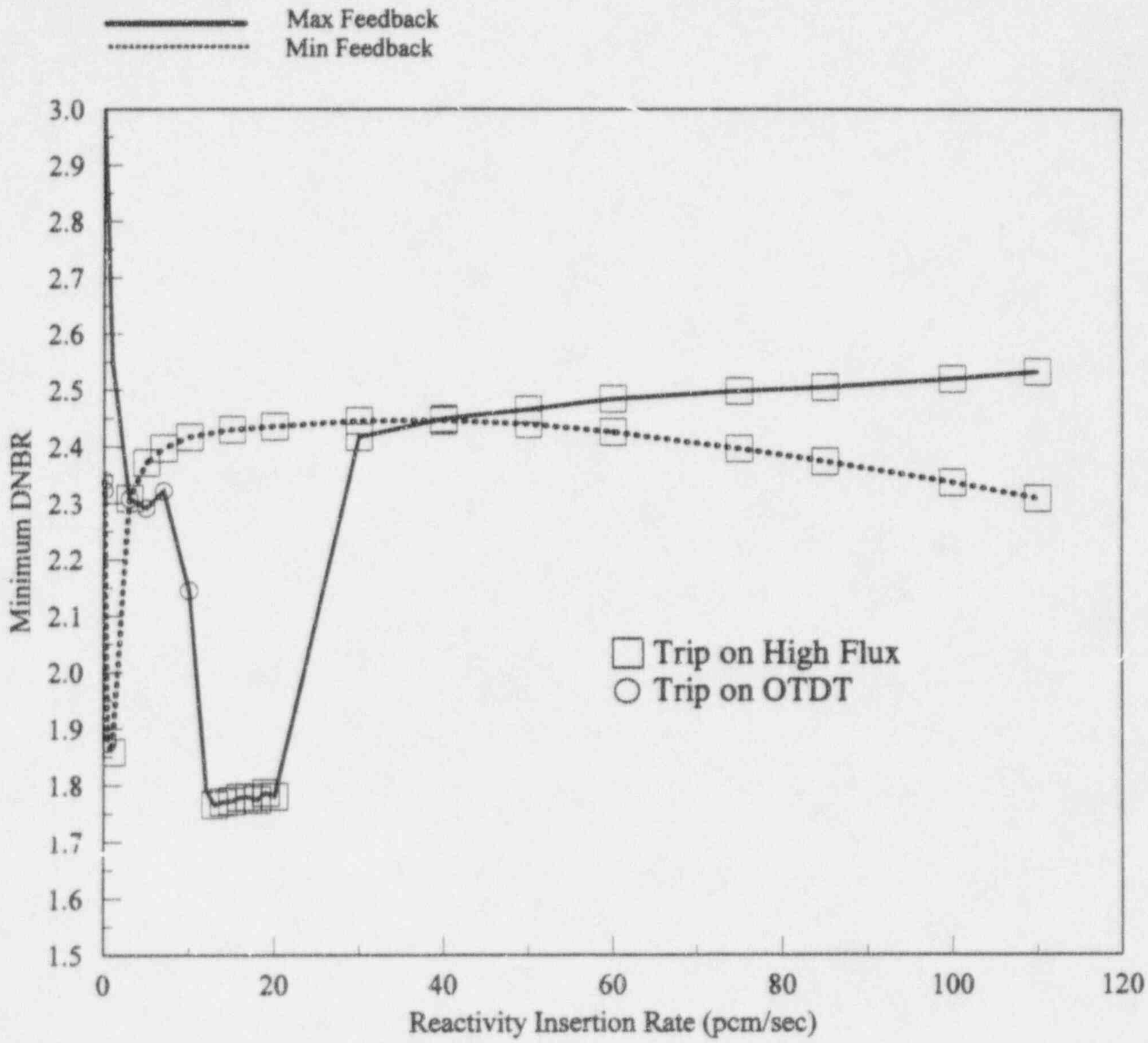
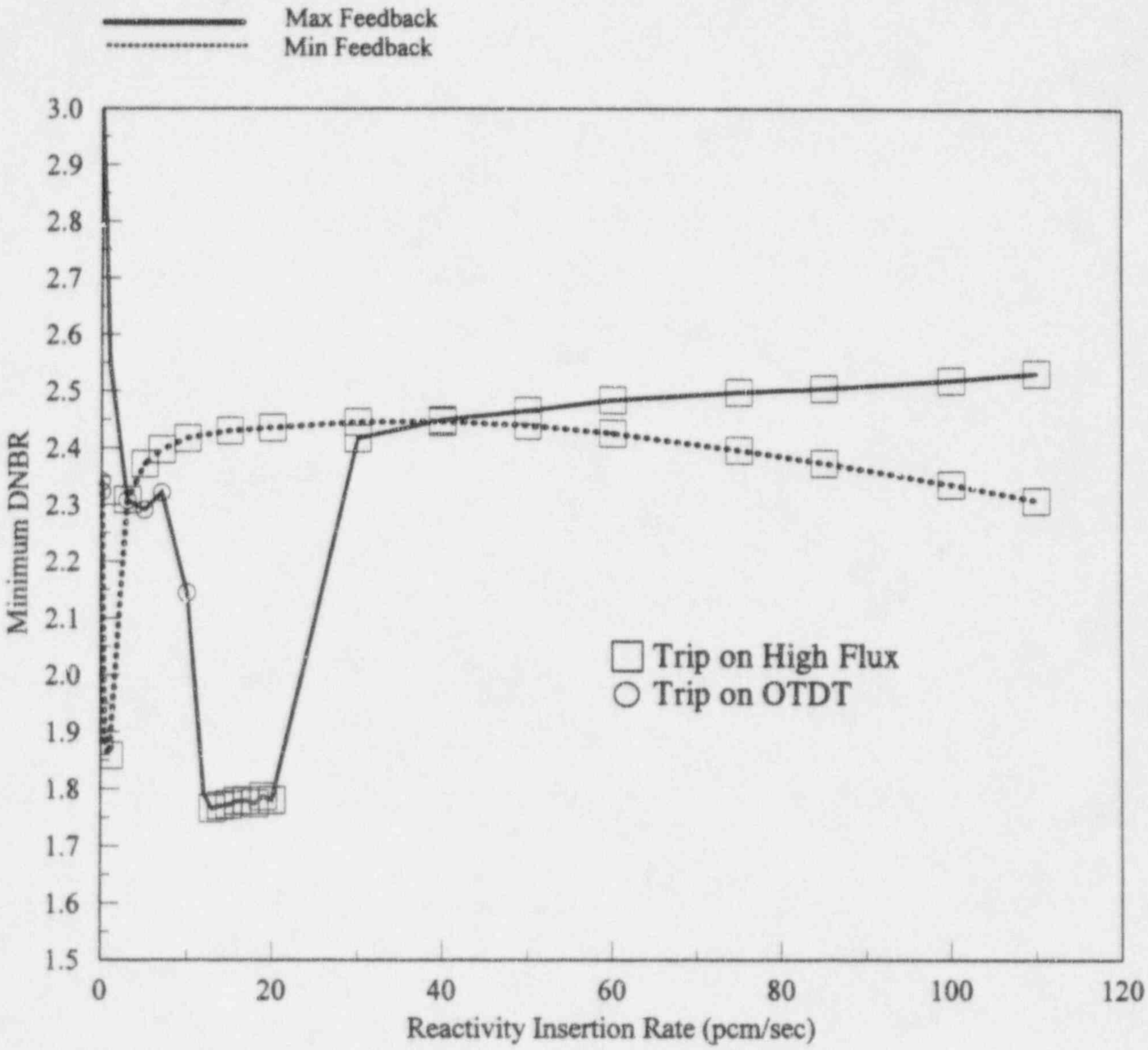


Figure 15.4.2-13 RWAP 100 Percent Power Minimum DNBR vs. Reactivity Insertion Rate



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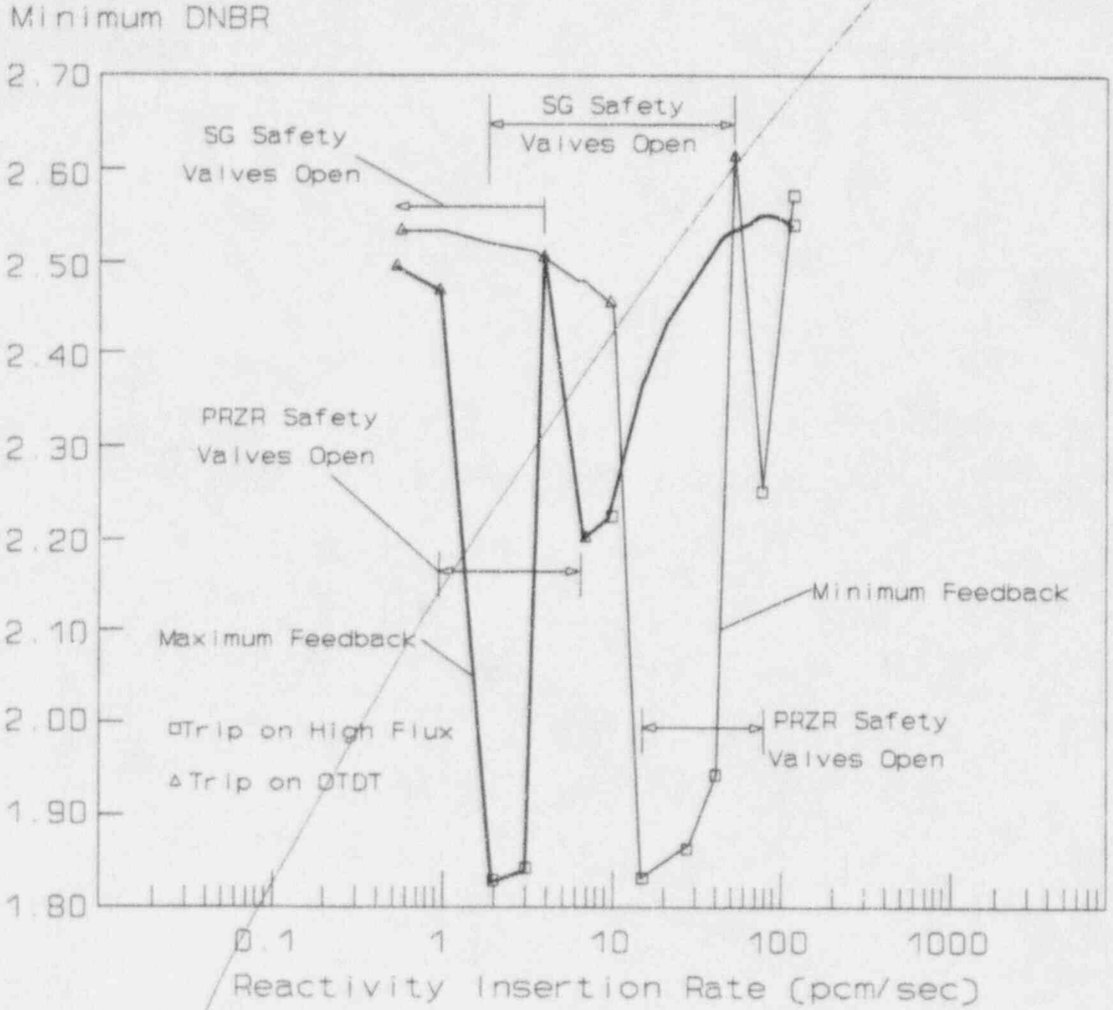
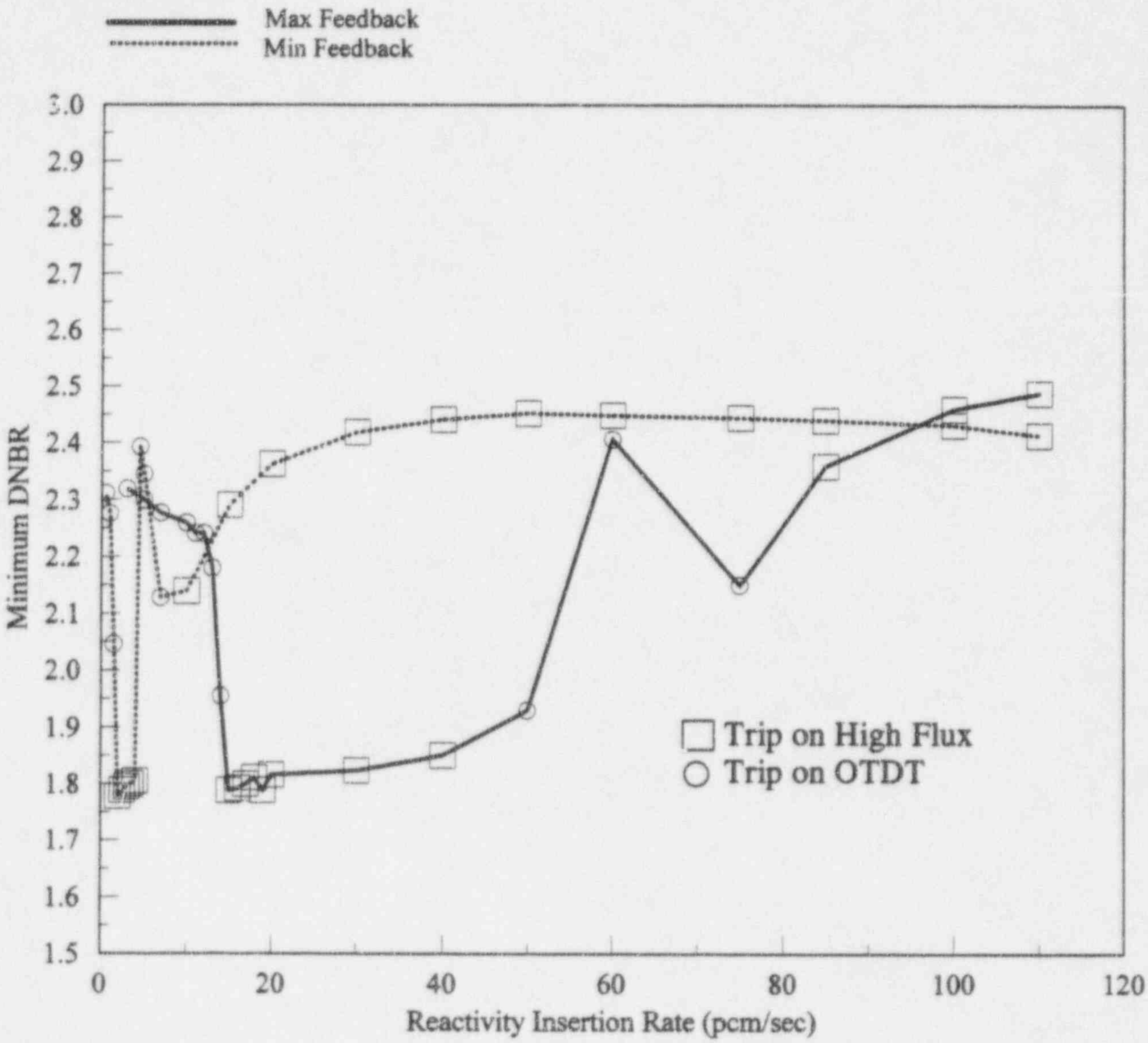


Figure 15.4.2-14

Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal at 60 Percent Power

Figure 15.4.2-14 RWAP 60 Percent Power Minimum DNBR vs. Reactivity Insertion Rate



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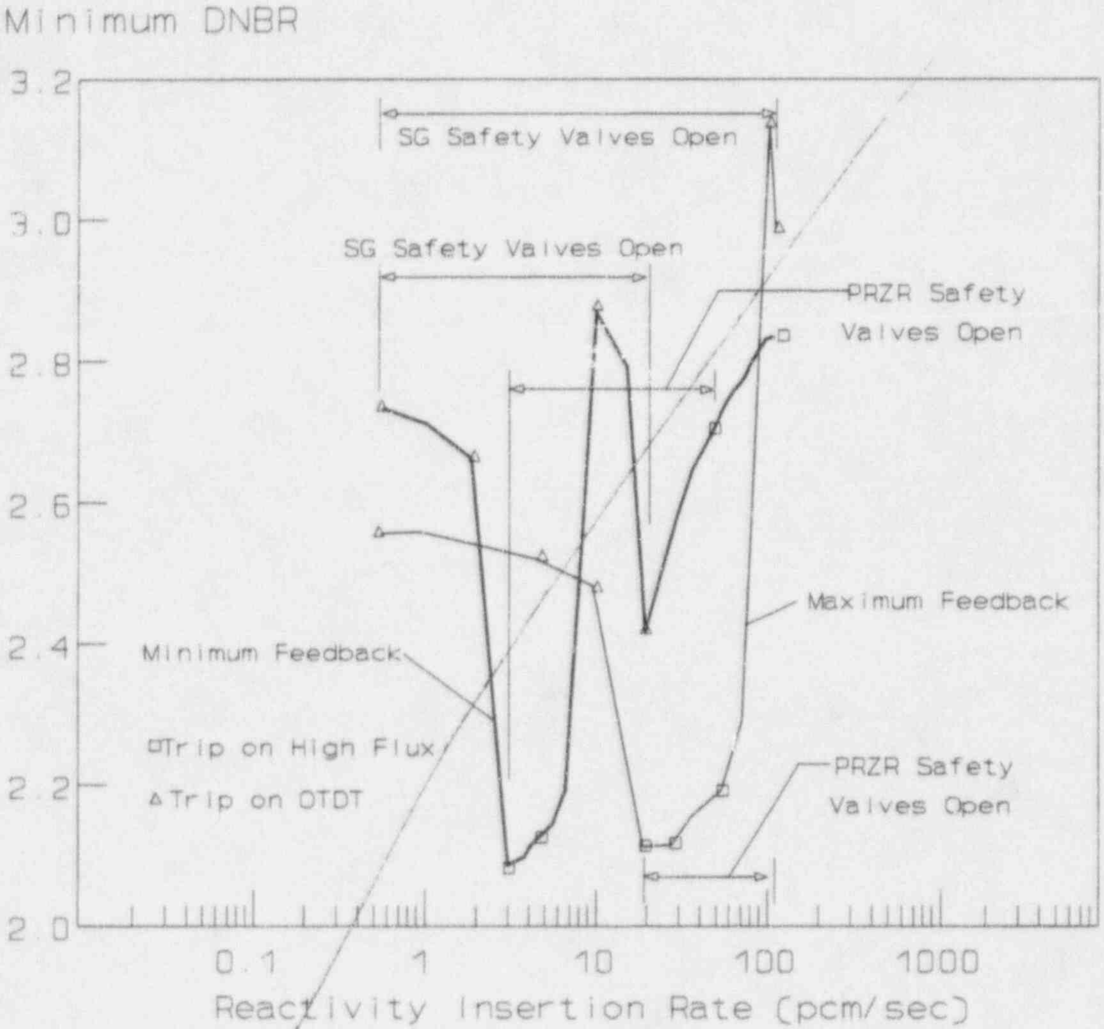
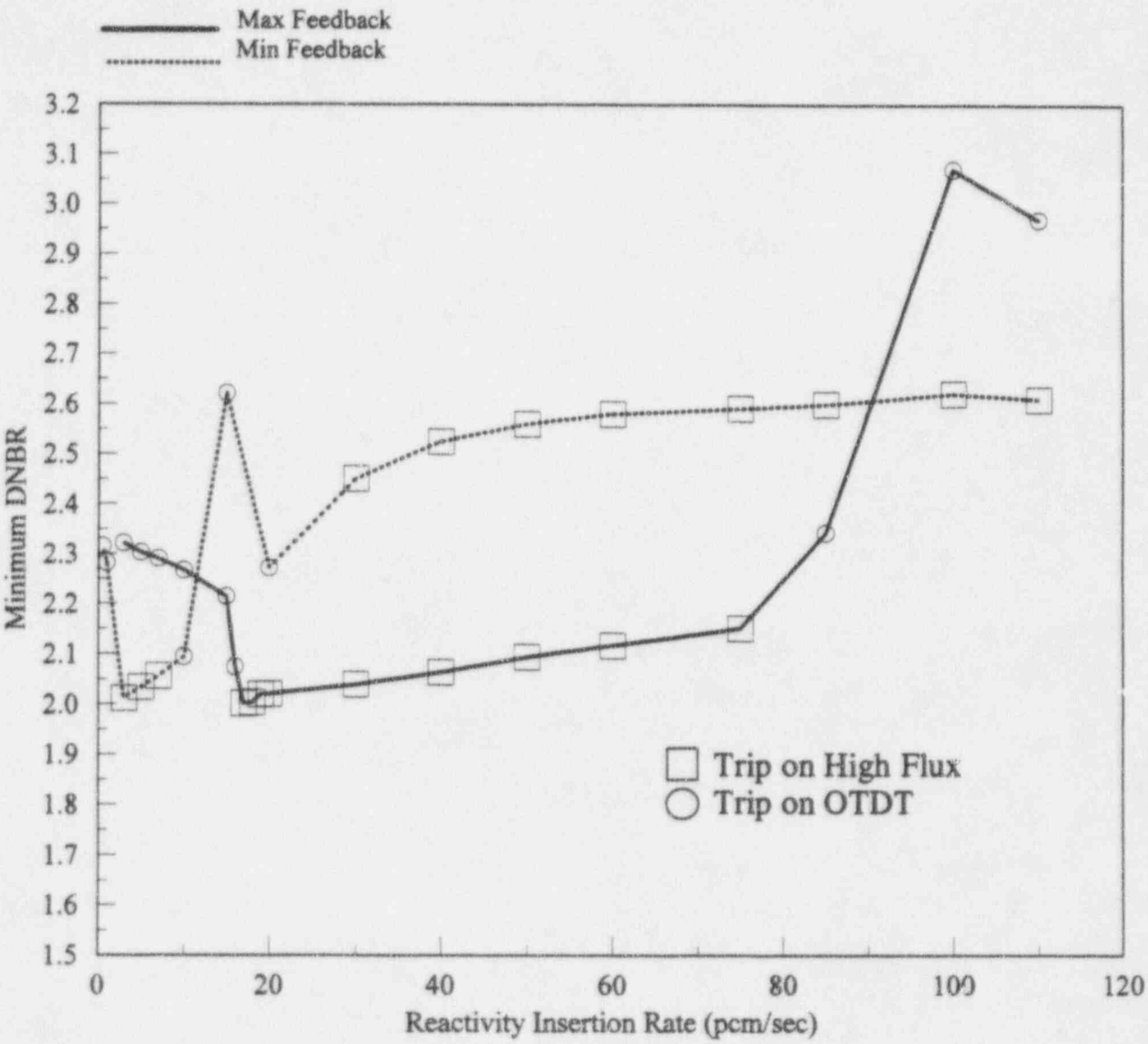


Figure 15.4.2-15

Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal at 10 Percent Power

Figure 15.4.2-15 RWAP 10 Percent Power Minimum DNBR vs. Reactivity Insertion Rate



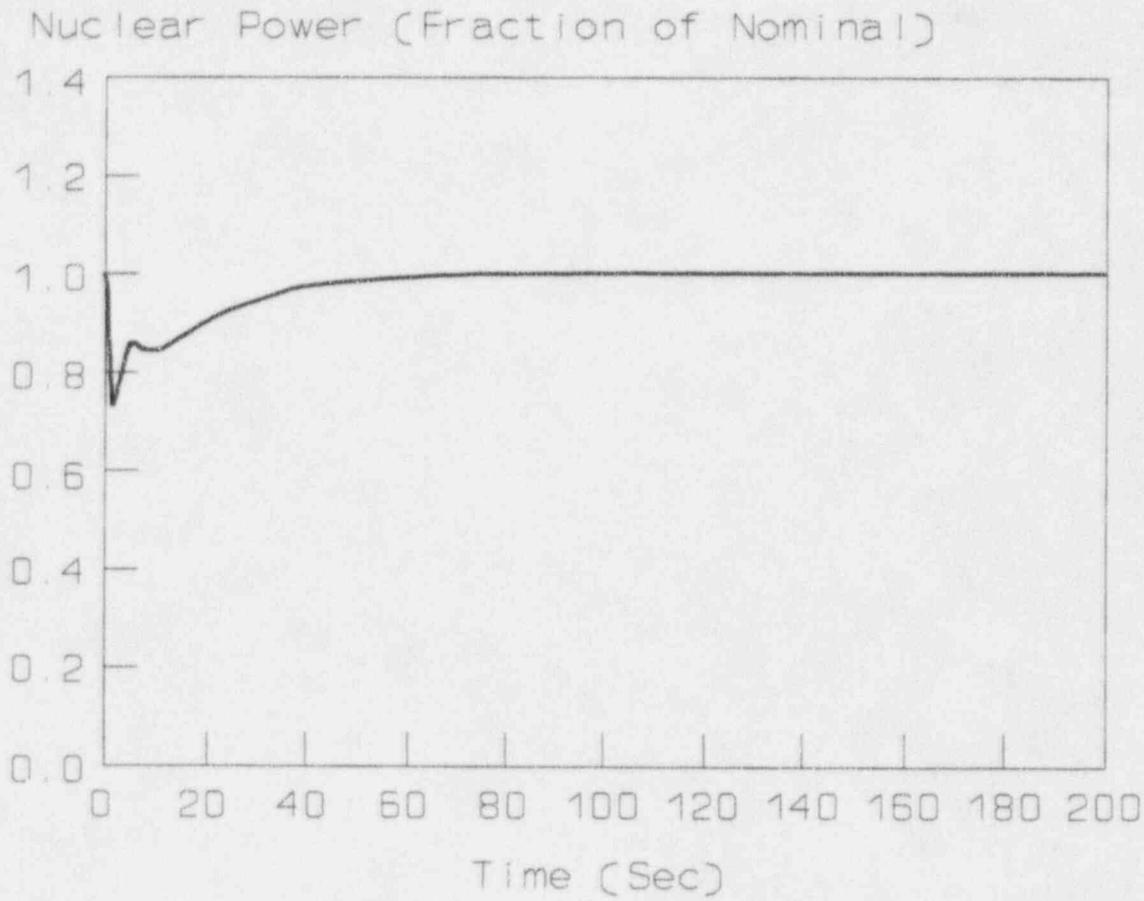


Figure 15.4.3-1

Nuclear Power Transients for Dropped RCCA

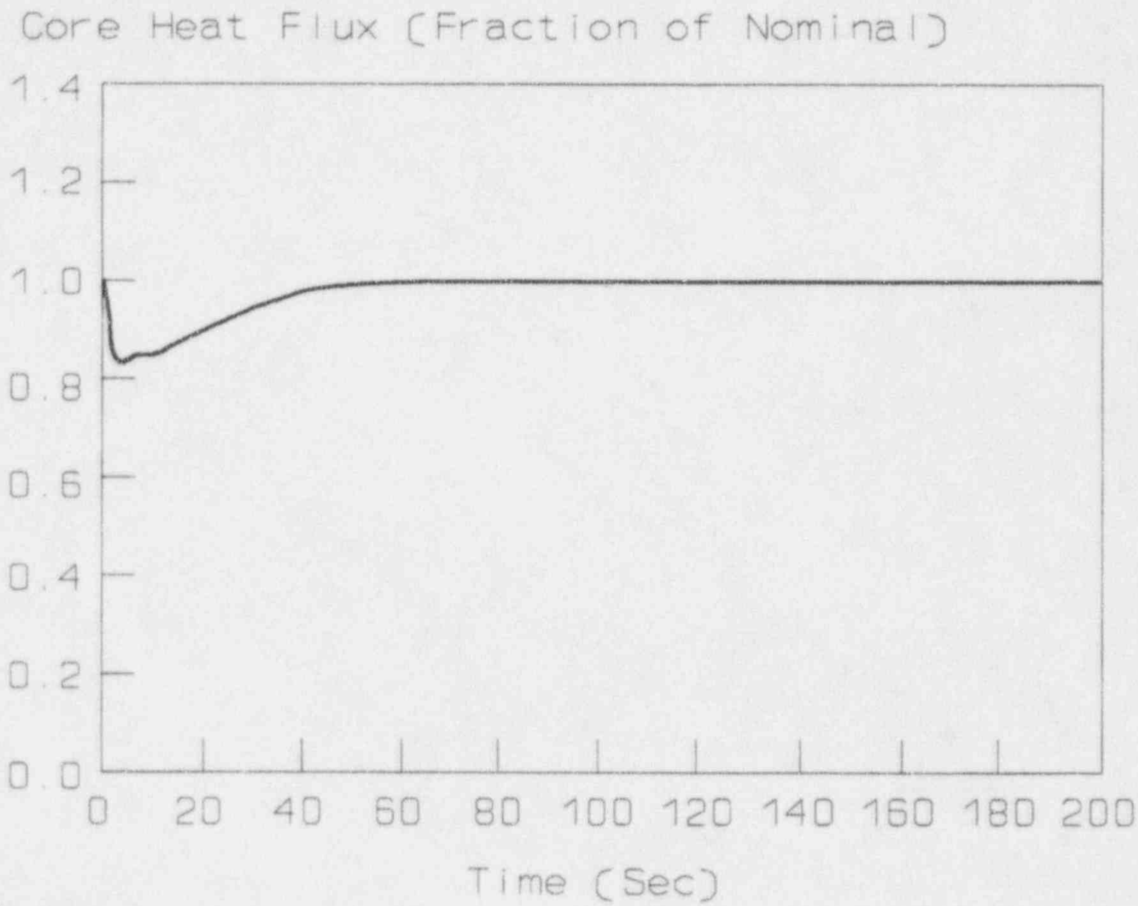


Figure 15.4.3-2

Core Heat Flux Transients for Dropped RCCA

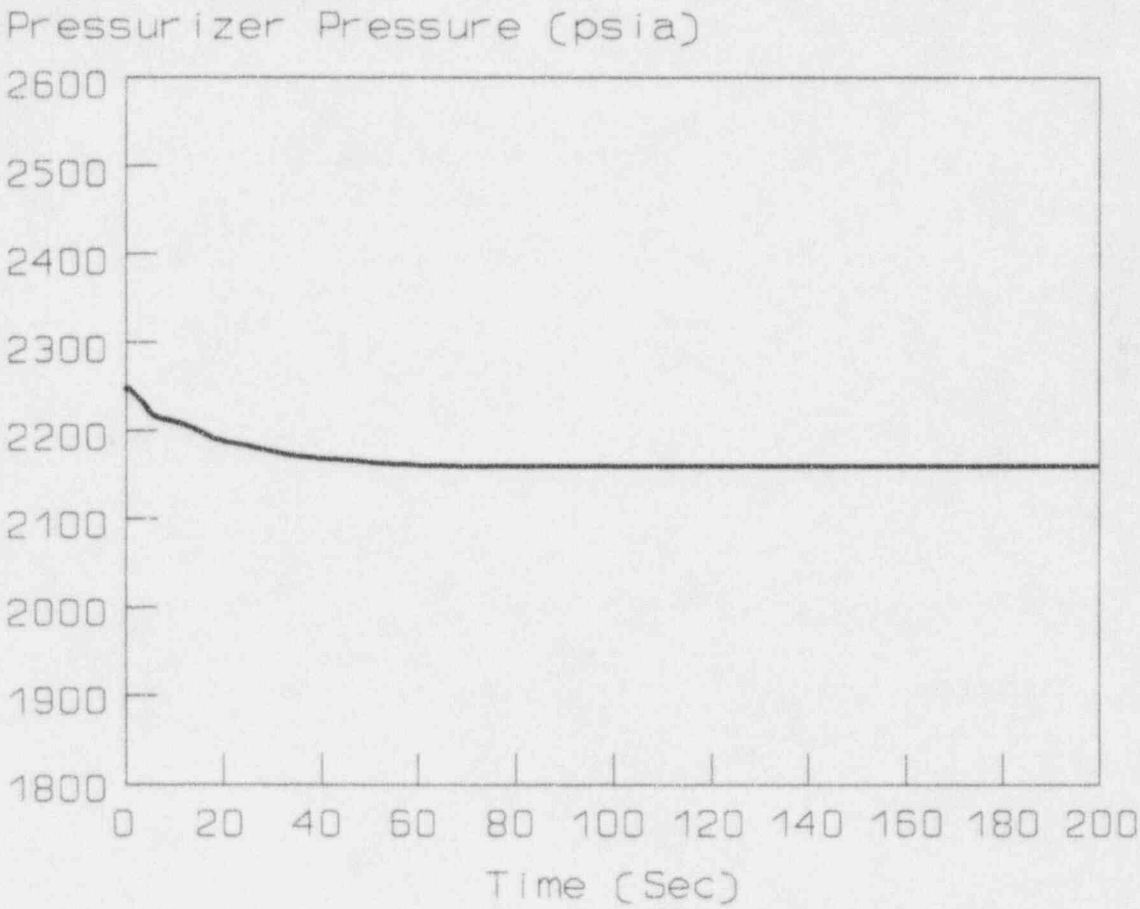


Figure 15.4.3-3

Pressurizer Pressure Transient for Dropped RCCA

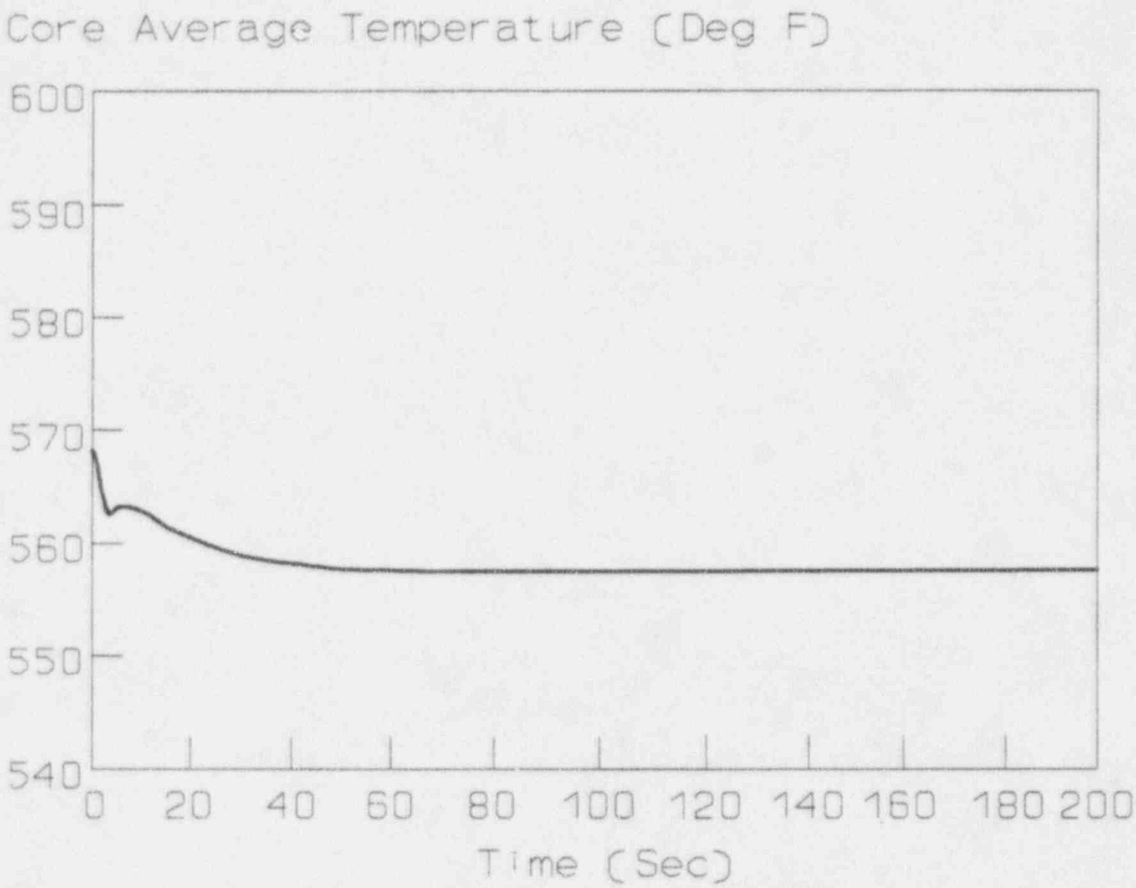


Figure 15.4.3-4

Cor: Average Temperature Transient for Dropped RCCA

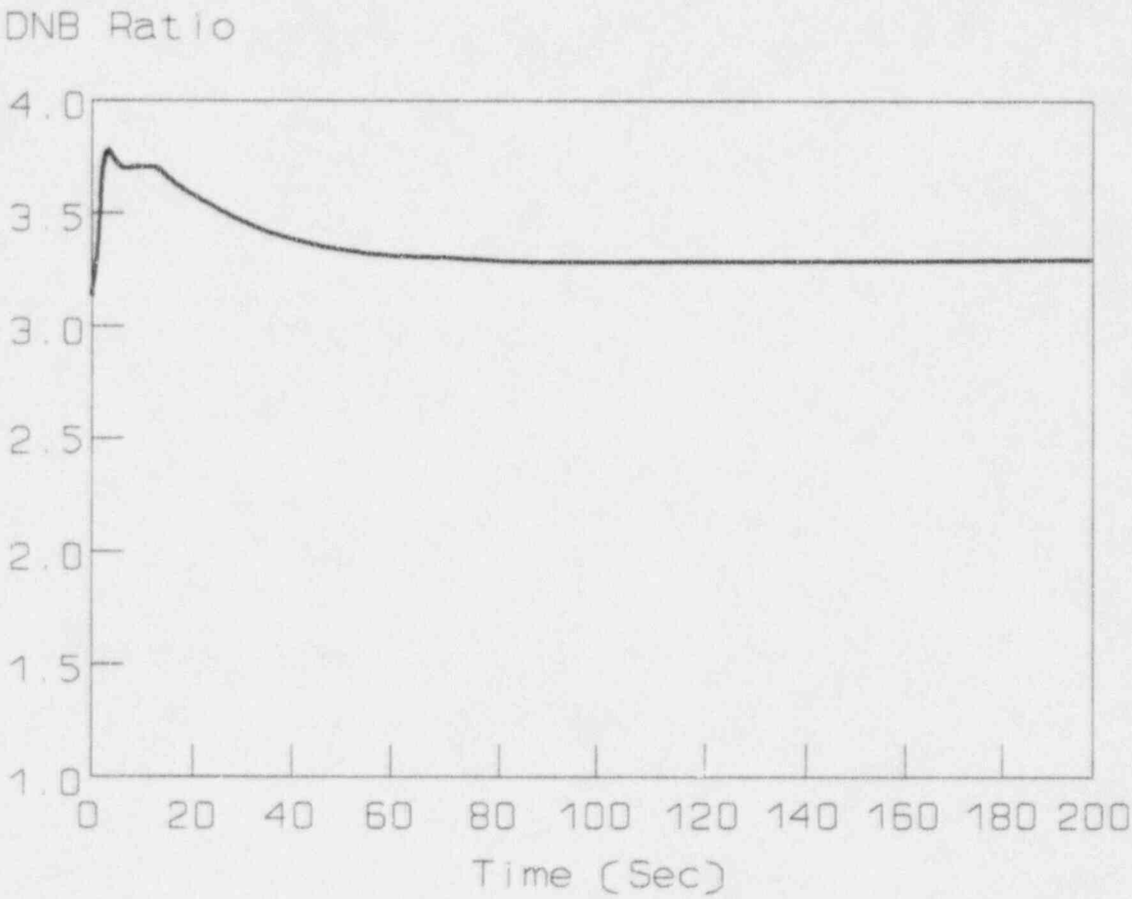


Figure 15.4.3-5

DNBR Transient for Dropped RCCA

REPLACE WITH
FIGURE ON FOLLOWING PAGE

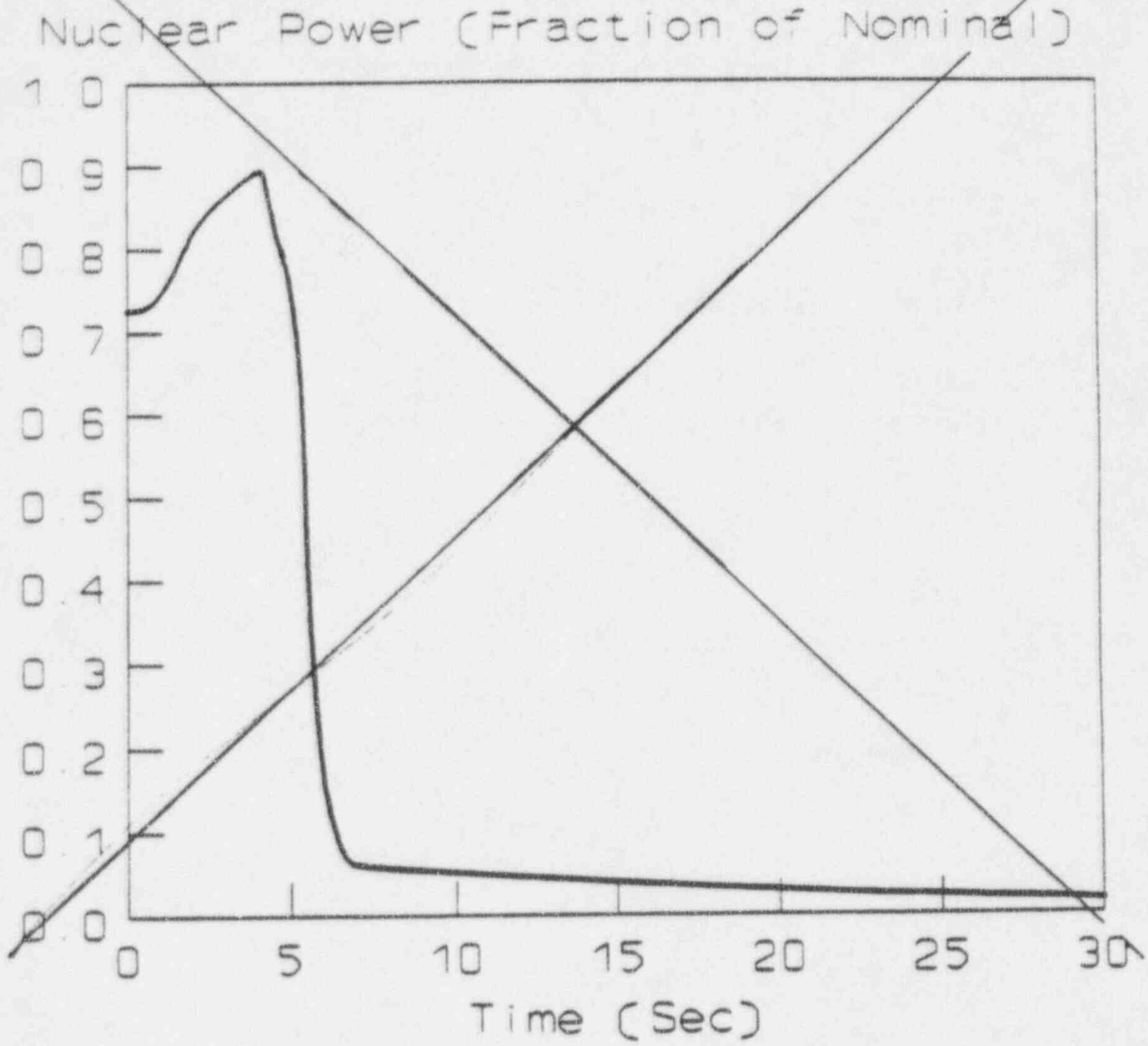
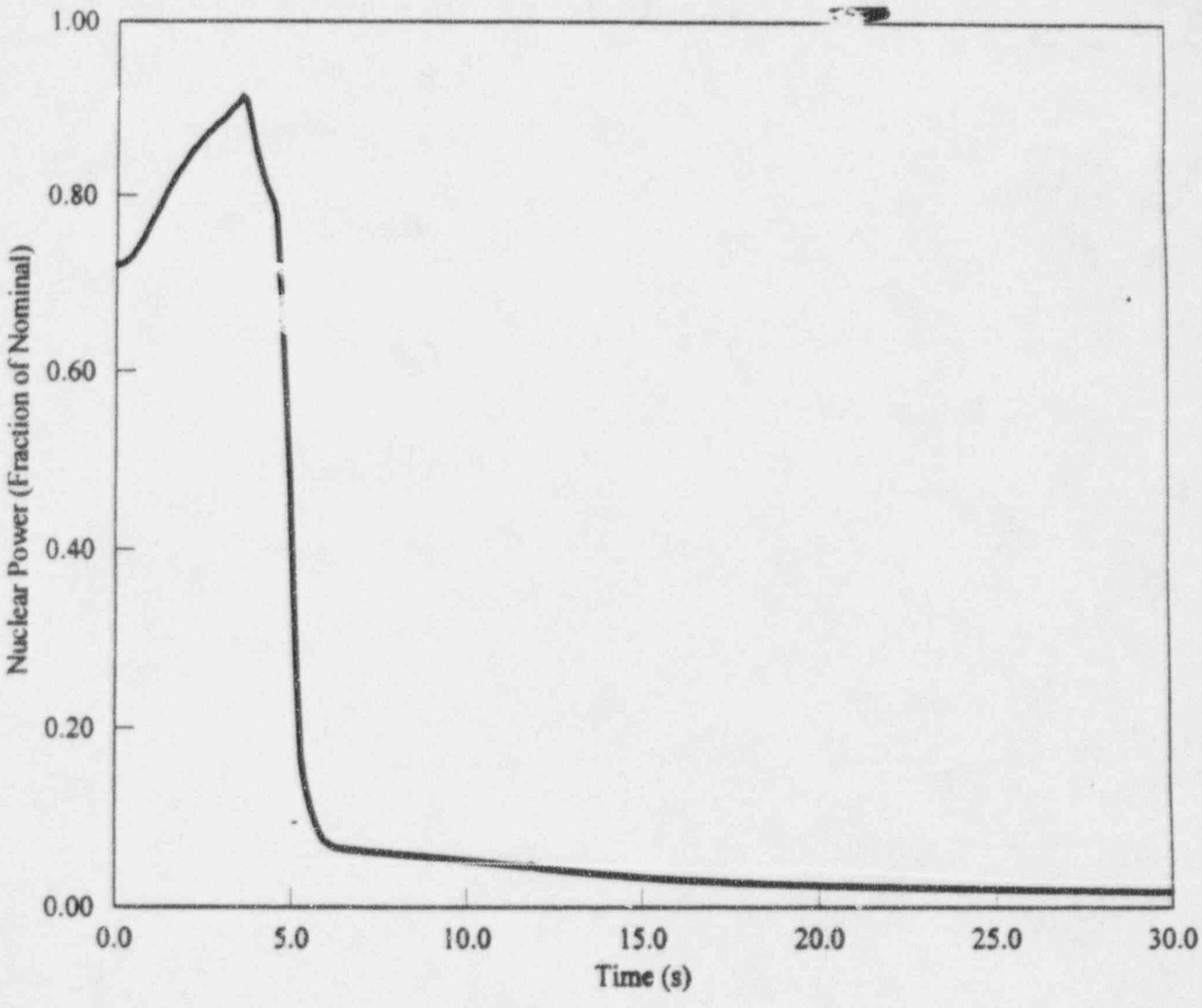


Figure 15.4.4-1

Nuclear Power Transient for an Improper Startup of an Inactive Reactor Coolant Pump

Figure 15.4.4-1 Improper Startup of an Inactive Reactor Coolant Pump



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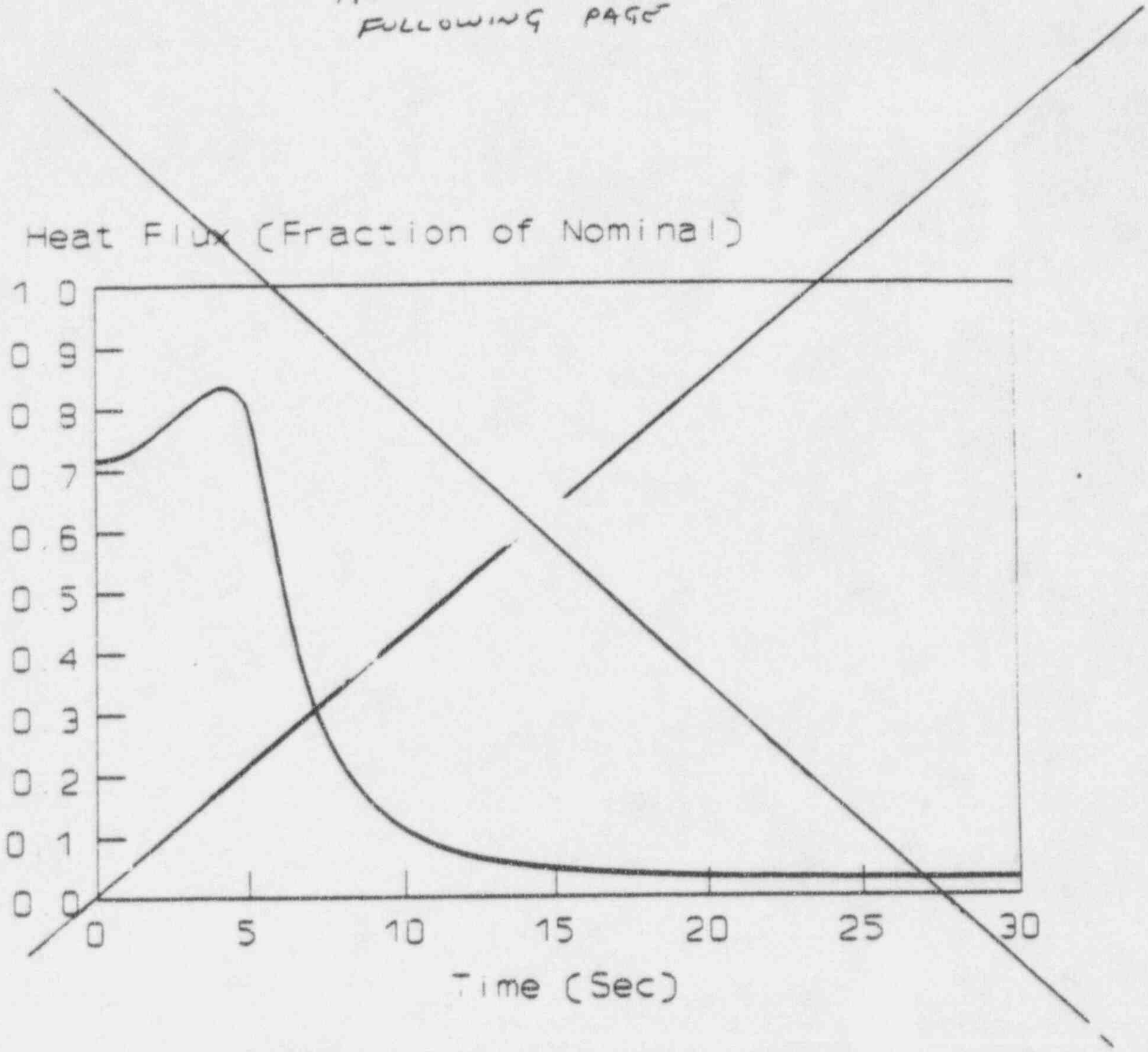
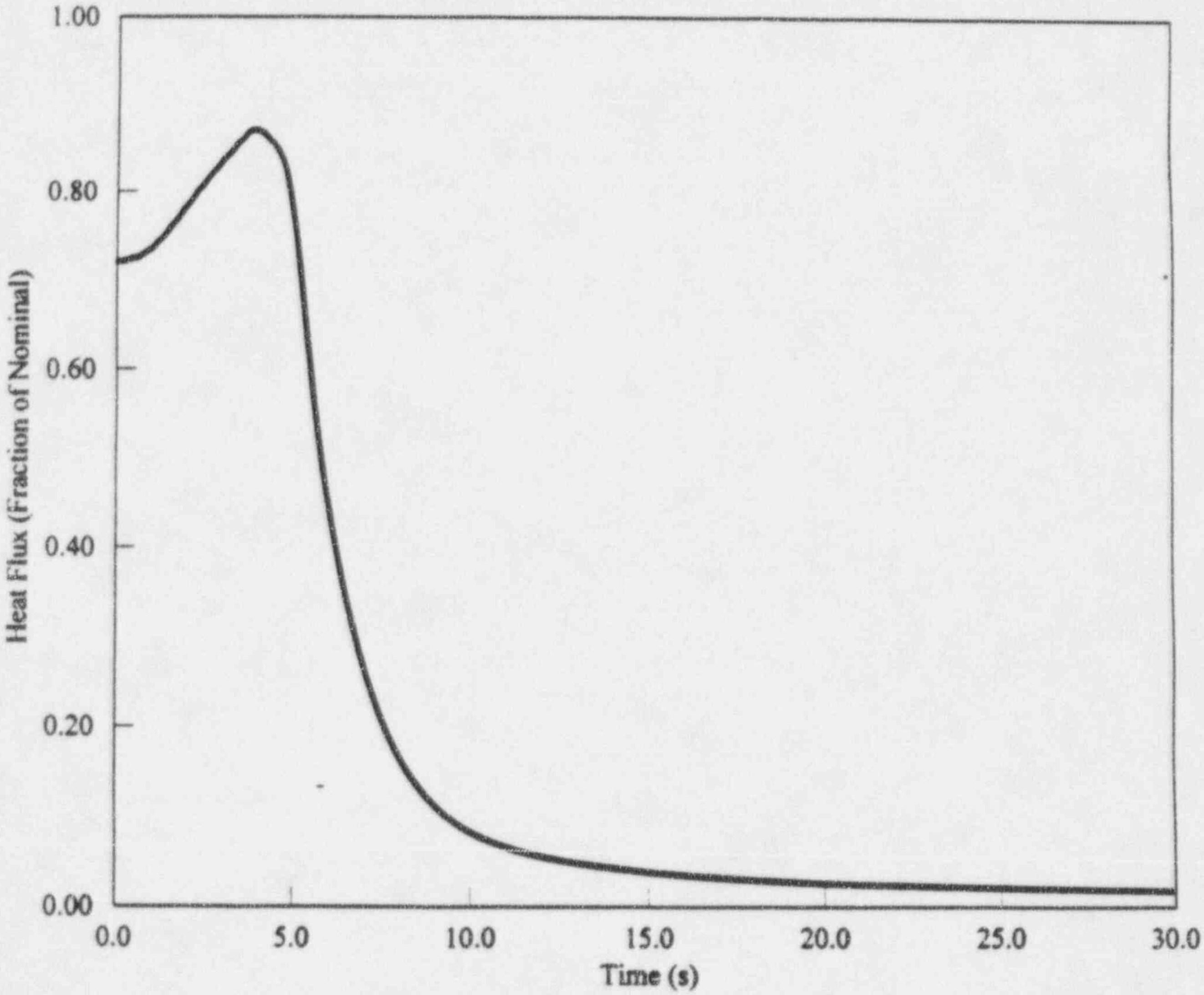


Figure 15.4.4-2

Heat Flux Transient for an Improper Startup of an Inactive Reactor Coolant Pump

Westinghouse

Figure 15.4.4-2 Improper Startup of an Inactive Reactor Coolant PUMP ~~Loop~~



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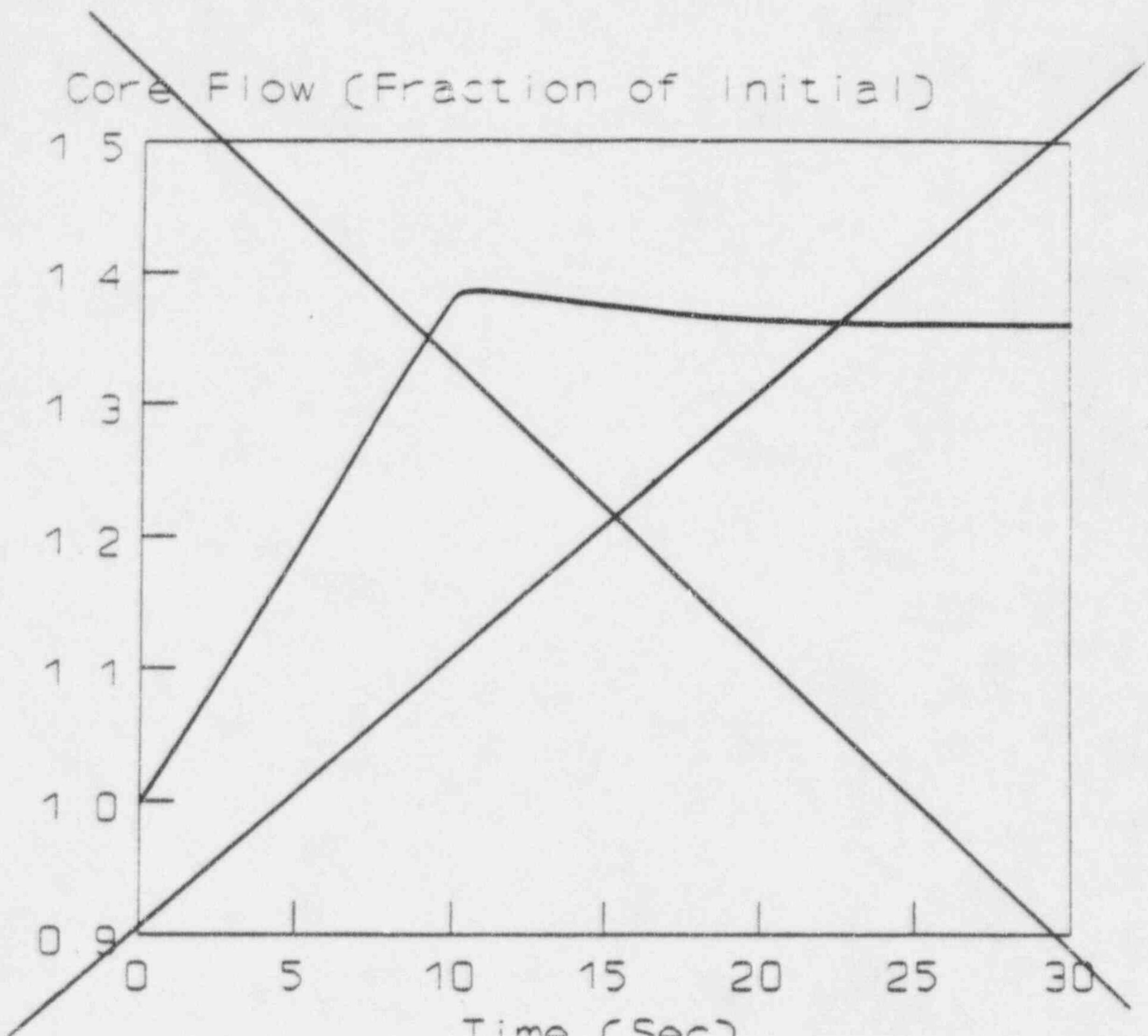

Figure 15.4.4-3

Core Flow Transient for an Improper Startup of an Inactive Reactor Coolant Pump

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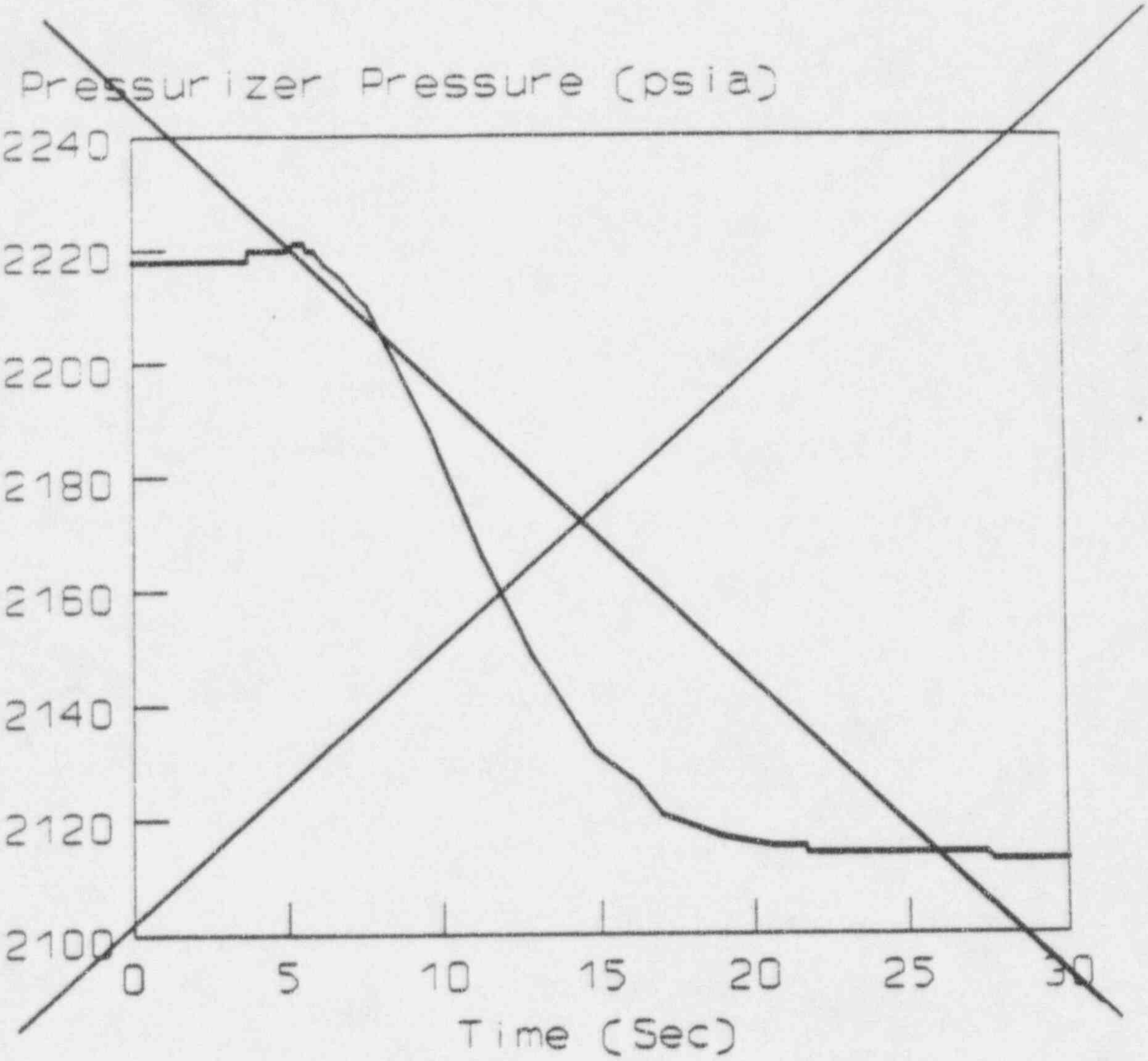


Figure 15.4.4-4 (A) 2

Pressurizer Pressure Transient for an Improper Startup of an Inactive Reactor Coolant Pump

Figure 15.4.4-3 Improper Startup of an Inactive Reactor Coolant Pump

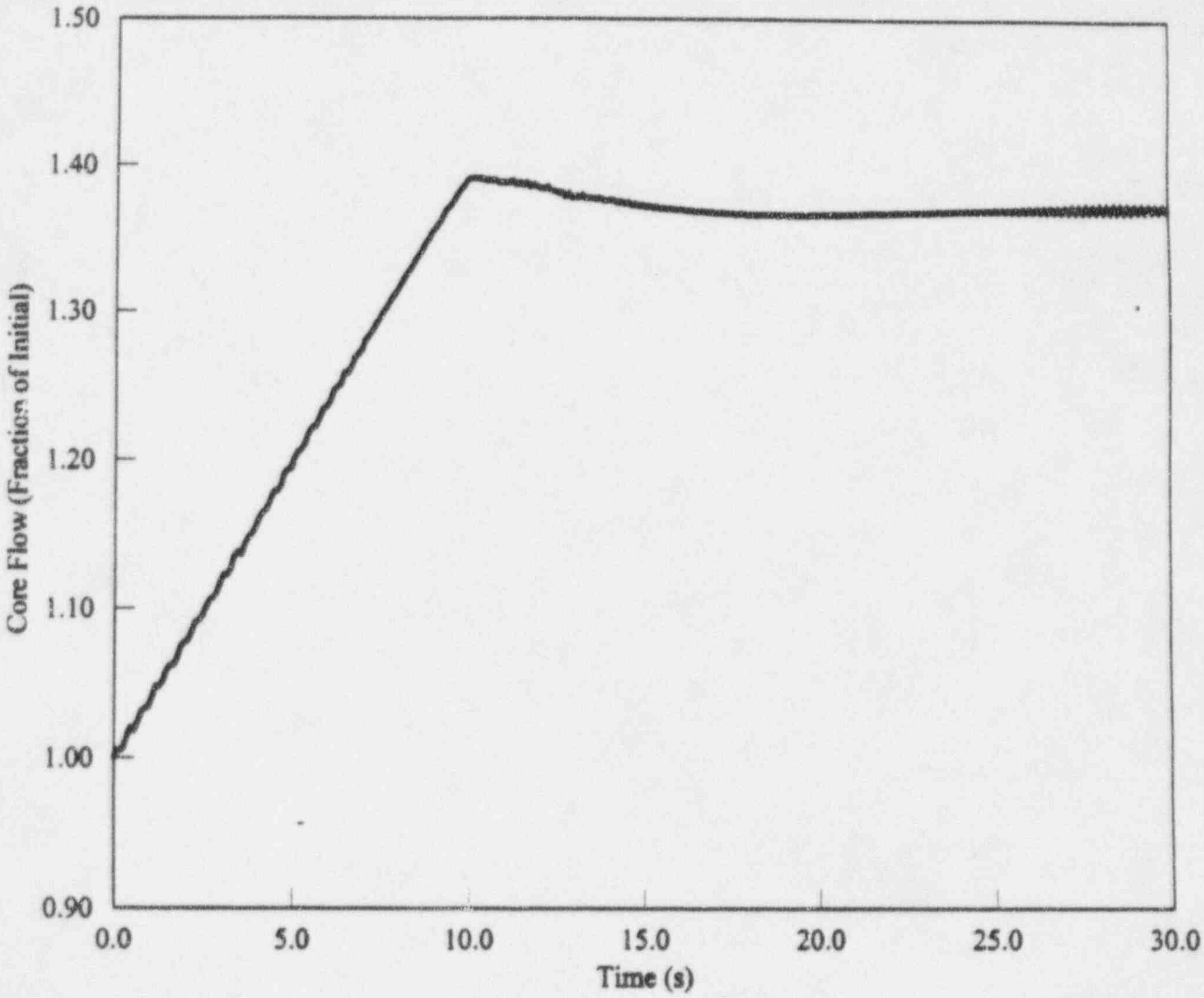
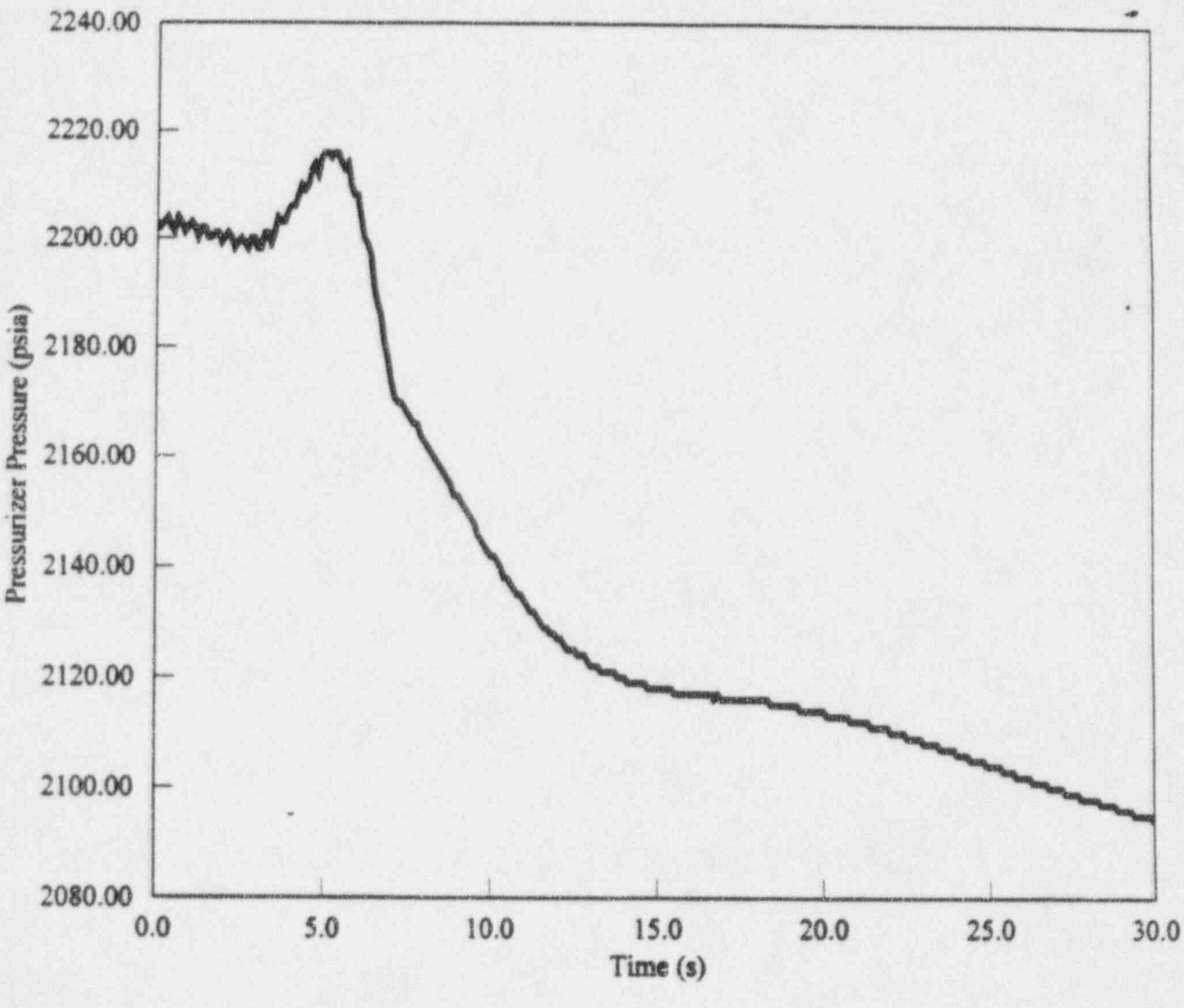


Figure 15.4.4-4 Improper Startup of an Inactive Reactor Coolant Pump



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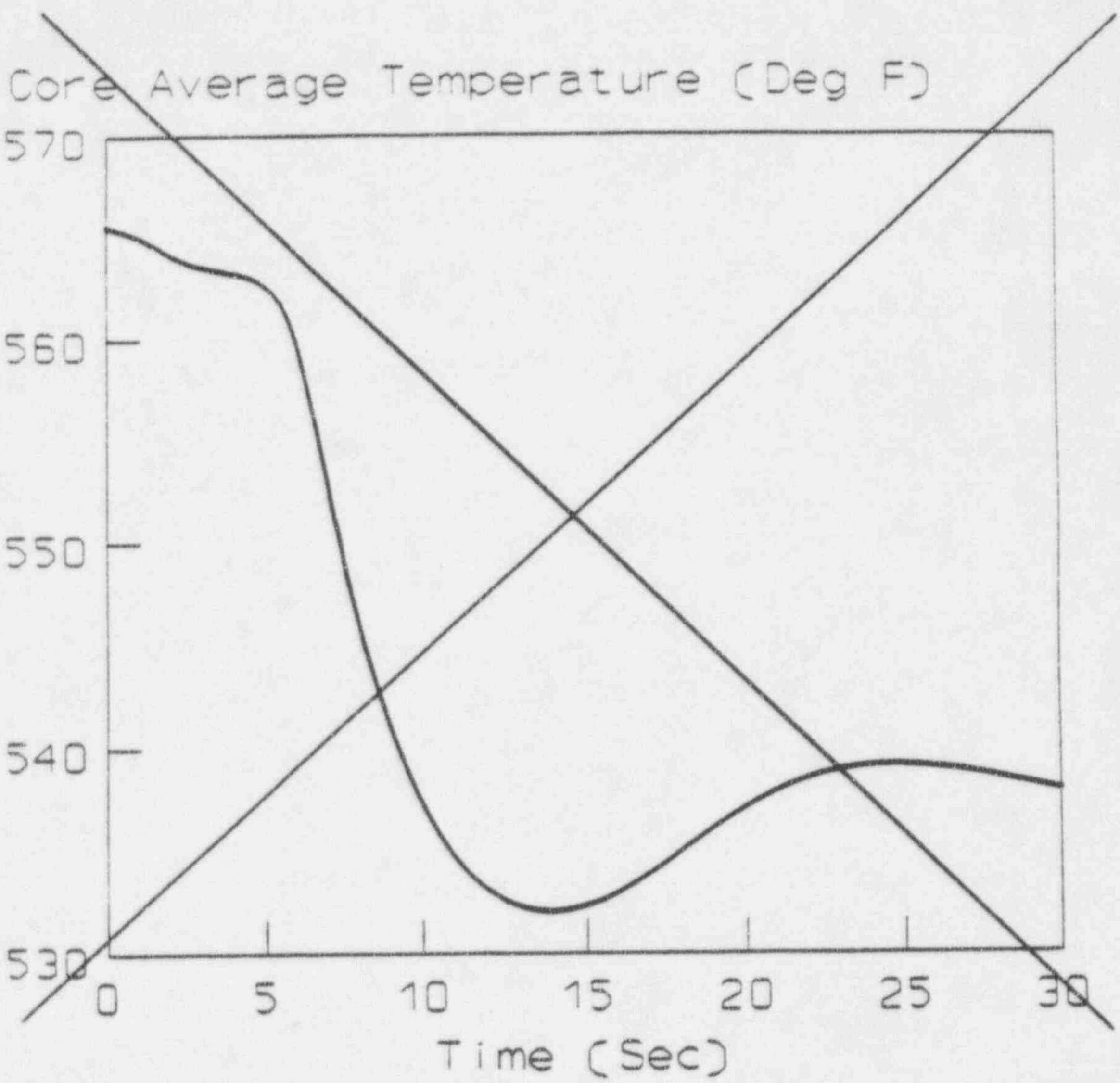
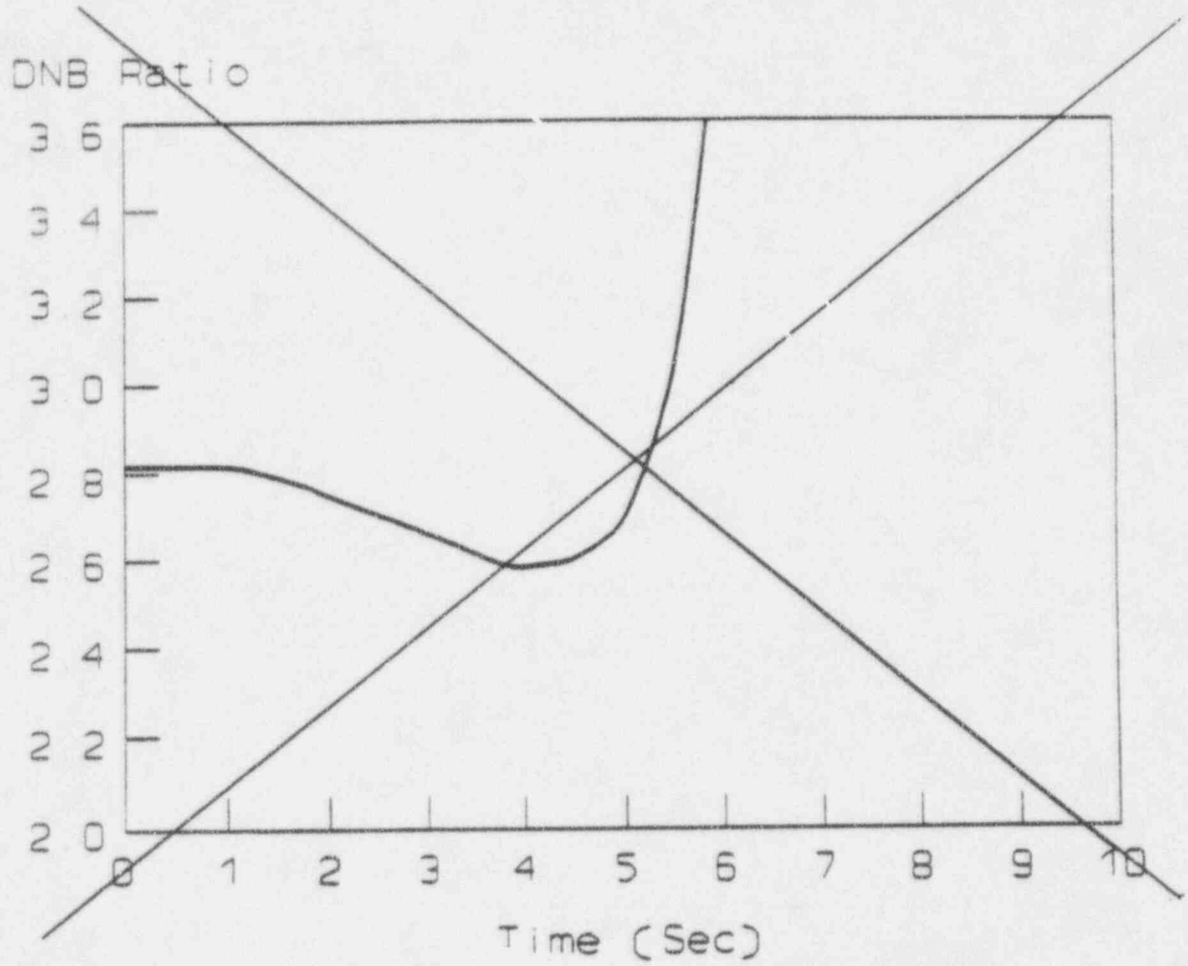


15.4.4-5
Figure ~~5.4.4-4-B~~

Core Average Transient for an Improper Startup of an Inactive Reactor Coolant Pump

AP600

REPLACE WITH FIGURE ON FOLLOWING PAGE



15.4.4-6

Figure ~~15.4.4-5~~

DNBR Transient for an Improper Startup of an Inactive Reactor Coolant Pump

Westinghouse

Figure 15.4.4-5 Improper startup of an Inactive Reactor Coolant Pump

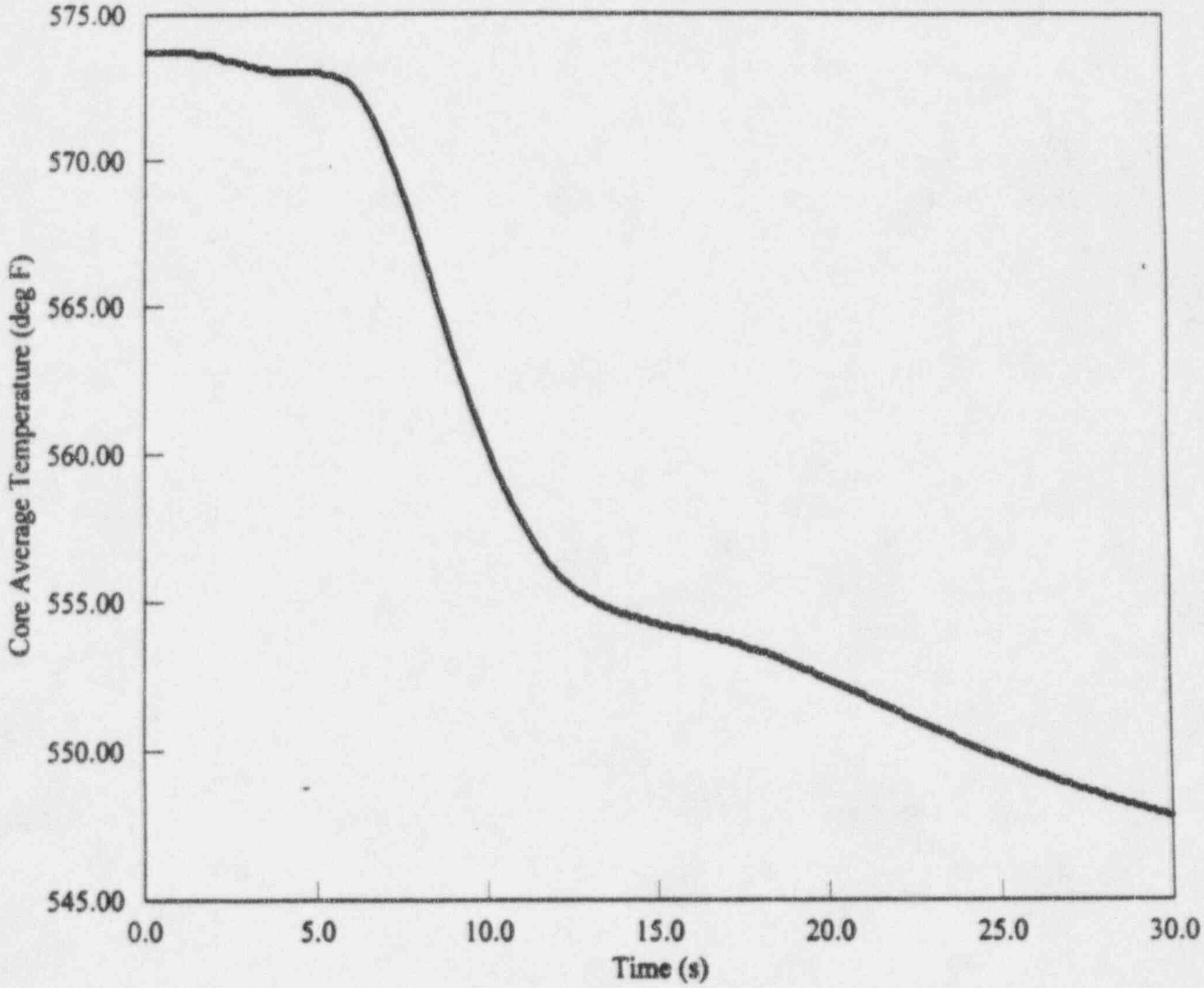
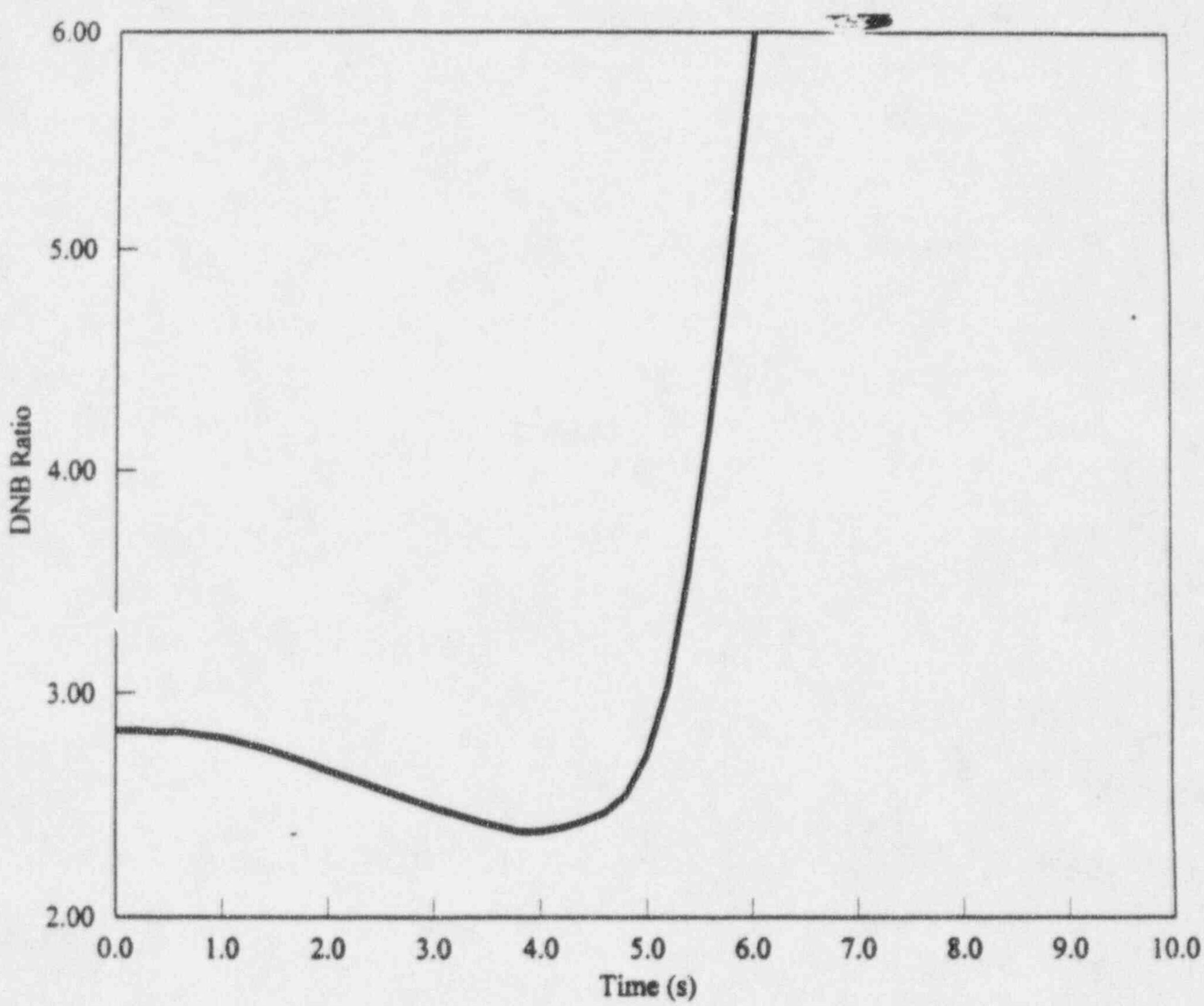


Figure 15.4.4-6 Improper Startup of an Inactive Reactor Coolant Pump



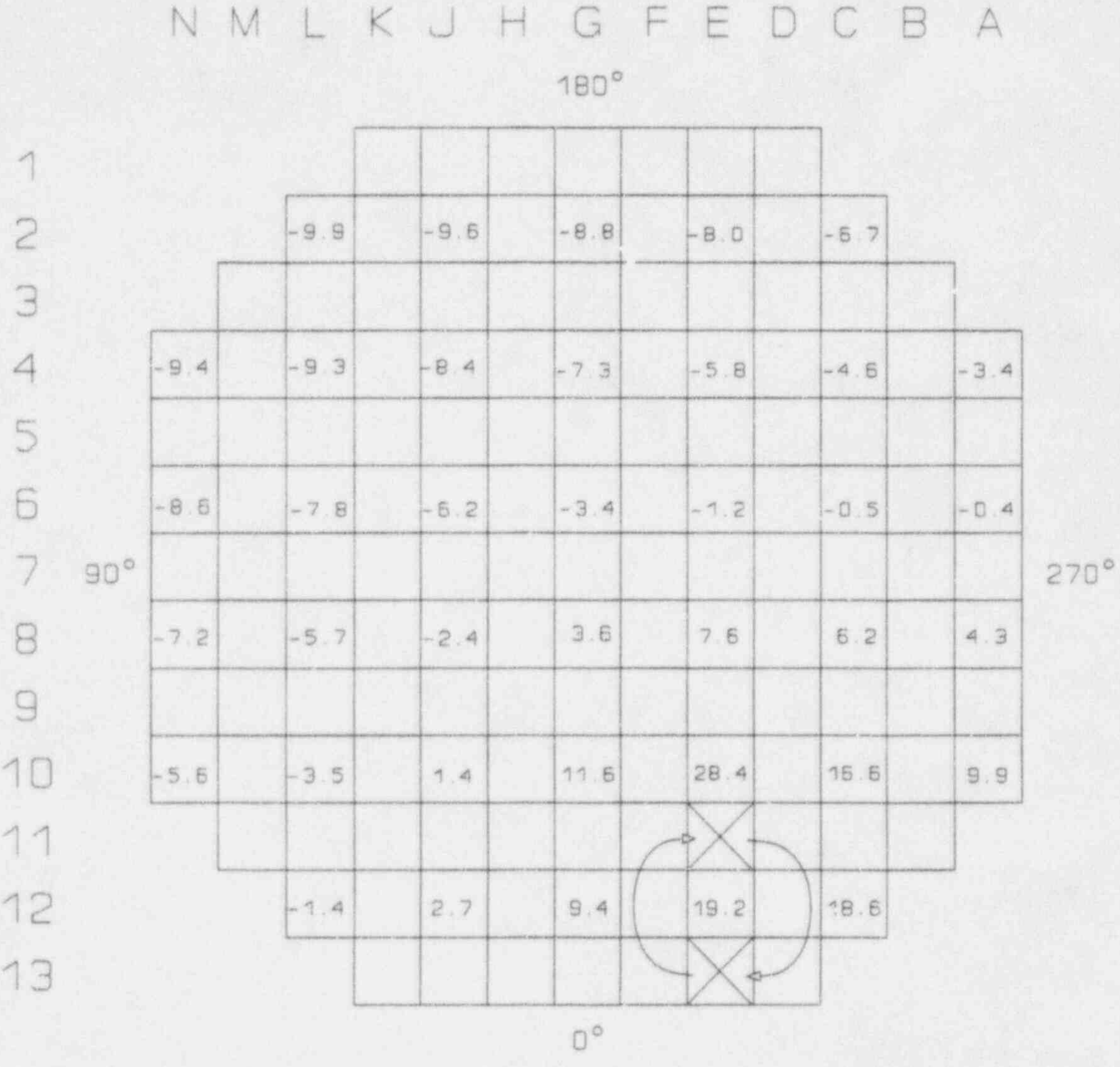


Figure 15.4.7-1

Representative % Cnange in Local Assembly Average Power for Interchange between Region 1 and Region 3 Assembly

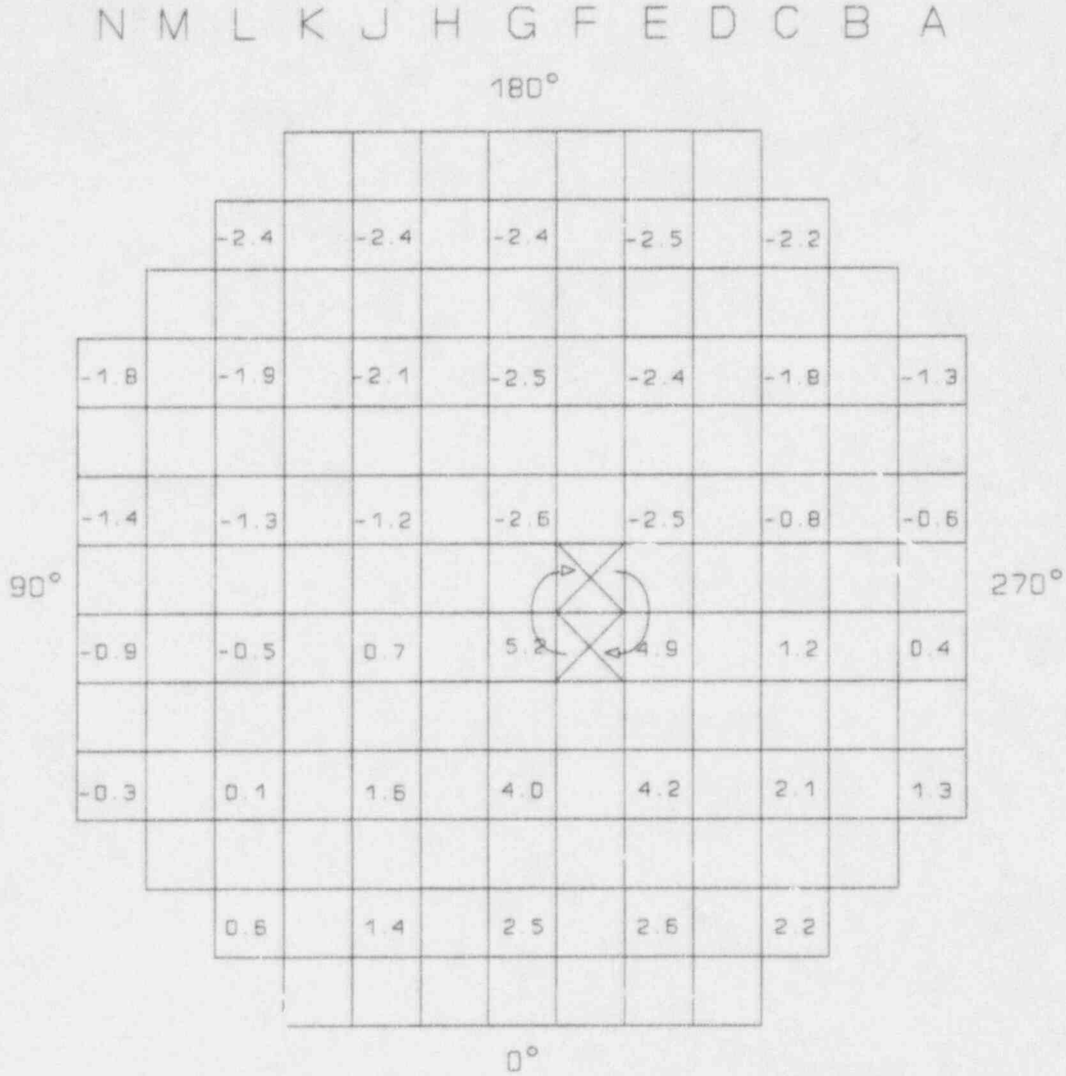


Figure 15.4.7-2

Representative % Change in Local Assembly Average Power for Interchange between Region 1 and Region 2 Assembly with the BP Rods Transferred to Region 1 Assembly

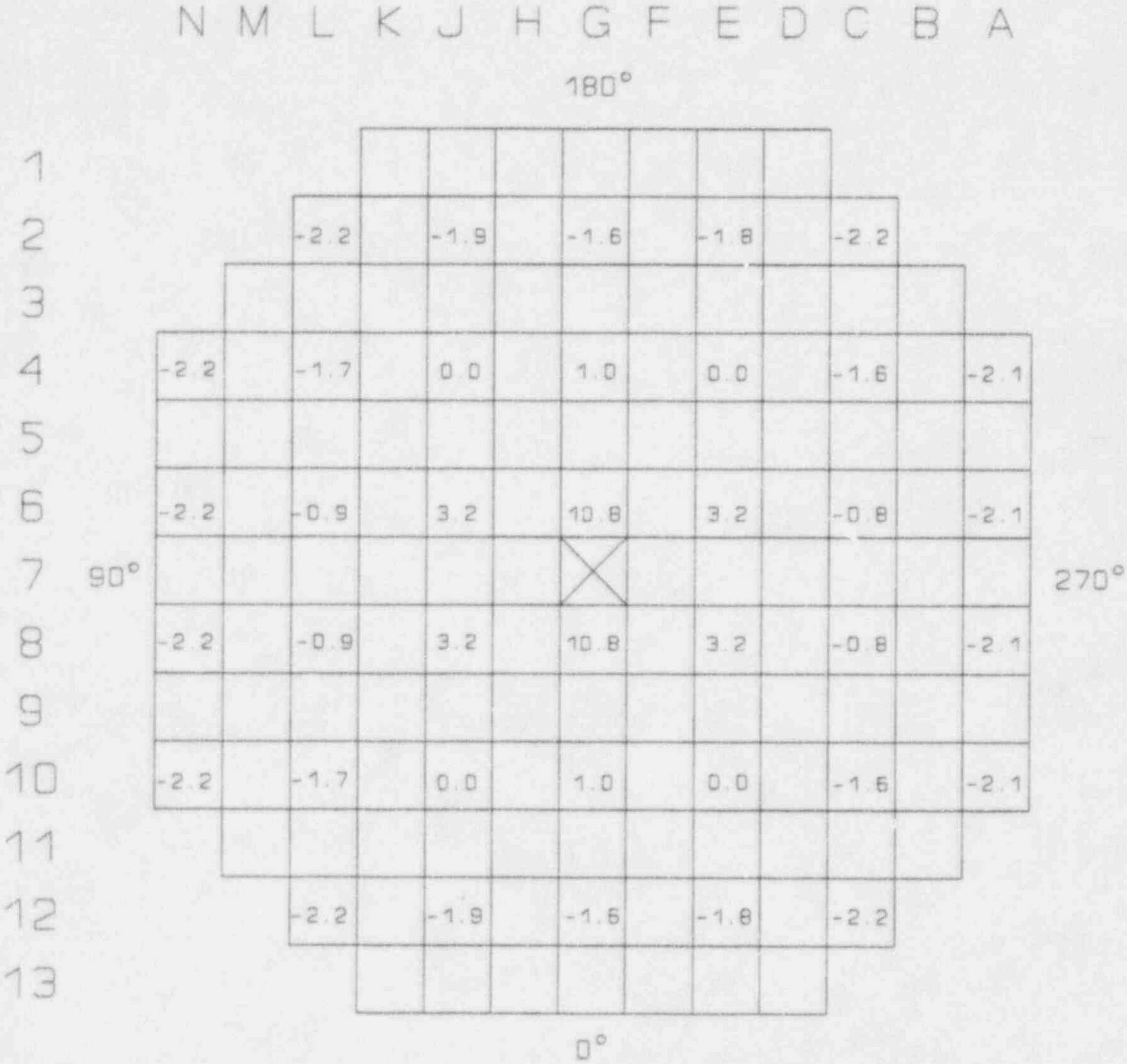


Figure 15.4.7-3

Representative % Change in Local Assembly Average Power for Enrichment Error (Region 2 Assembly Loaded into Core Central Position)

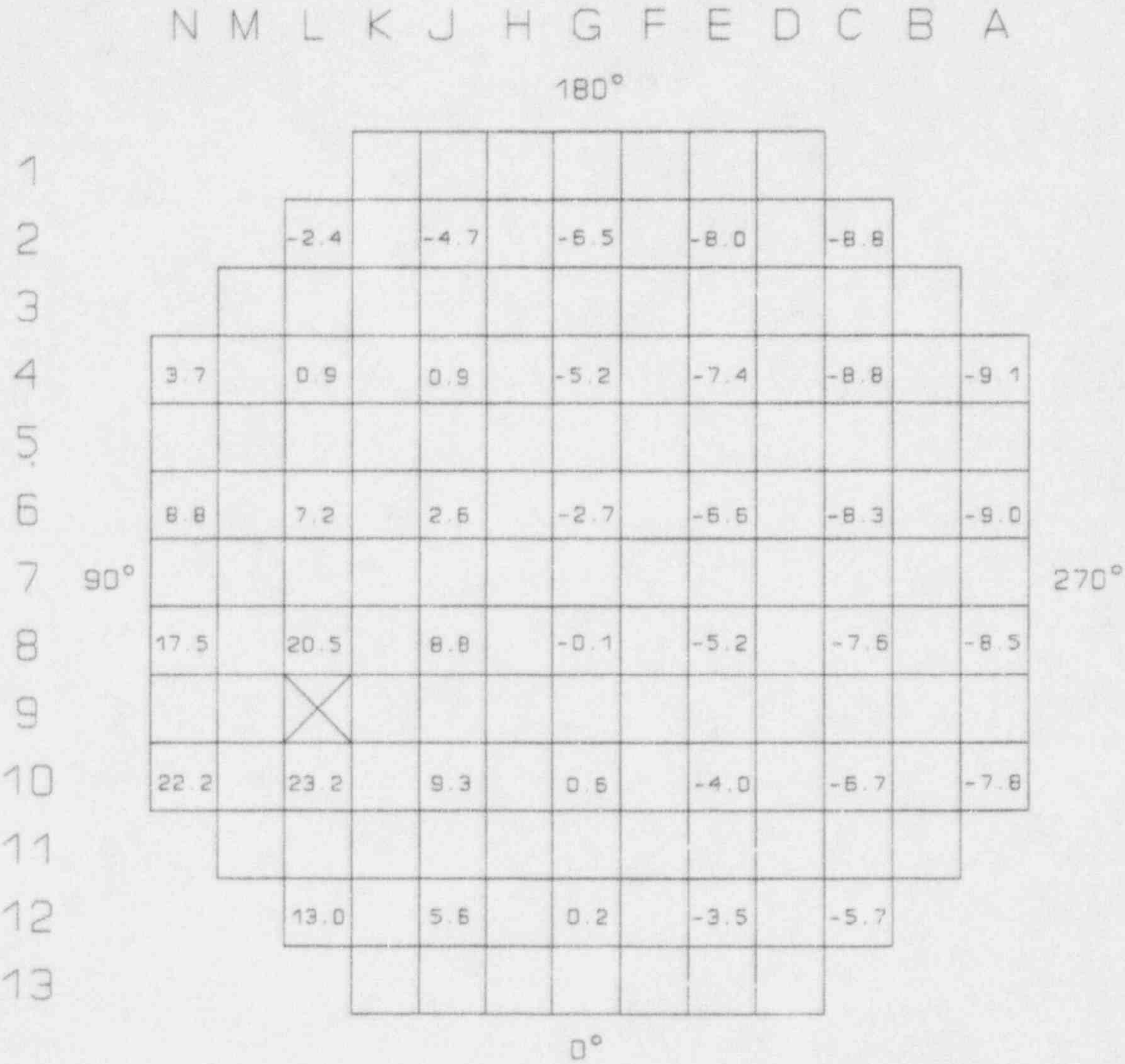


Figure 15.4.7-4

Representative % Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

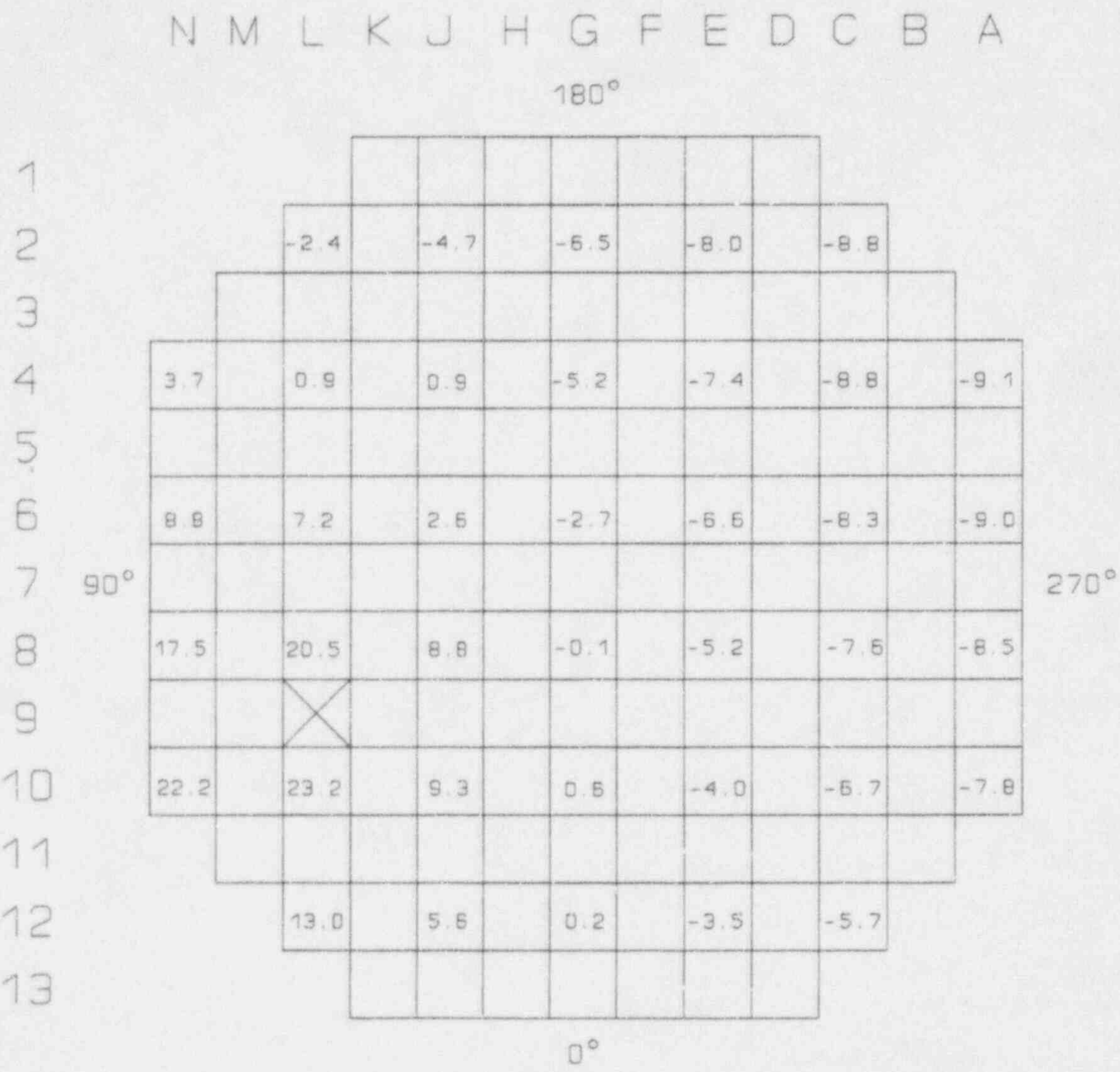


Figure 15.4.7-4

Representative % Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

Figures 15.4.8-1 through 15.4.8-4
to follow

15.5 Increase in Reactor Coolant Inventory

Discussion and analysis of the following events are presented in this section:

- Inadvertent operation of the core makeup tanks (CMT) during power operation
- Chemical and volume control system (CVS) malfunction that increases reactor coolant inventory

These Condition II events cause an increase in reactor coolant inventory.

15.5.1 Inadvertent Operation of the Core Makeup Tanks (CMT) During Power Operation

15.5.1.1 Identification of the Causes and Accident Description

Spurious CMT operation at power could be caused by operator error, a false electrical actuation signal or a valve malfunction. A spurious signal may originate from any of the safeguards ("S") actuation channels as described in Section 7.3.

Replace with Insert 1

Following the actuation signal, the reactor is tripped, CMT discharge valves and balance line valves are opened, the reactor coolant pumps are tripped, and the two charging pumps on the chemical and volume control system (CVS) are actuated.

However, it cannot be assumed that a single fault that actuates the CMTs also produces a reactor trip. The opening of the CMT discharge valves, due to operator error or valve failure, does not result in an immediate reactor trip but at the same time no flow is injected into the RCS if the reactor coolant pumps are operating and the balance line valves are closed (RCPs provide a very high counter pressure at the vessel injection).

The operator is alerted to the CMT discharge valve status by the CMT valve indication lights in the main control room. The operator determines that the signal is a spurious one and in this case resets the valve status.

Replace with Insert 2

A spurious "S" signal results in a reactor trip. Following the reactor trip, the reactor power drops and the average RCS temperature decreases with subsequent coolant shrinkage. However, a few seconds after the trip, the two CVS pumps actuate, the CMTs discharge, the RCPs trip and the balance line valves open. The CMTs and CVS pumps then start injecting highly borated water into the RCS. The primary coolant system shrinkage is counteracted by the CMT and CVS injection and the pressurizer volume starts to increase because of the heat up of the cold injected fluid by the decay heat. The CVS pumps are automatically isolated and the PRHR is actuated on a high pressurizer level signal but this transient is terminated by the operator manually isolating the CMTs. The operator determines whether the signal is spurious. If the "S" signal is determined to be spurious, the operator terminates the CMT injection and maintains the plant in the hot standby condition as determined by the appropriate

Replace with Insert 3

~~recovery procedure. If the ECCS actuation instrumentation must be repaired, continued plant operation can follow the technical specifications.~~

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

The plant response to a spurious "S" signal is analyzed by employing a modified version of the computer program LOFTRAN described in Subsection 15.0.10.2. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, and the passive residual heat removal (PRHR) heat exchanger (HX) and the CMT ~~systems~~. The program computes pertinent plant variables including temperatures, pressures and power level.

Reactor power and average temperature drop immediately following the trip and the operating conditions never approach the core limits.

A typical transient is presented representing the minimum reactivity feedback. ~~and the maximum CMT injection capability~~. Control systems are conservatively not assumed to function during the transient. Cases with the turbine bypass (steam dump) and feedwater control systems working result in lower secondary and primary temperatures and in greater margin to pressurizer overfill ~~longer time for operator action~~. CMT and PRHR system performance is conservatively simulated. (CMT flow rates to the RCS have been maximized and the PRHR sizing and flow rates have been minimized).

Replace with Insert 4

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. The assumptions are as follows:

- Initial operating conditions

  The initial reactor power, RCS pressure and average temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

  Replace with Insert 5

- Moderator and Doppler coefficients of reactivity

  A least negative moderator temperature coefficient is used. A low (absolute value) Doppler power coefficient is assumed. Reactivity parameters are of no importance for this analysis since the reactor is immediately tripped and there are no concerns about the core shutdown since there is boron injection from the CMTs ~~and CVS~~.

- Control Systems

  Control systems are not assumed to function since their operation results in a better behaved transient.

- Boron Injection

Insert 1

The inadvertent opening of the CMT discharge valves, due to operator error or valve failure is not expected to result in CMT injection flow if the RCPs are operating, because the RCPs provide a counter pressure at the direct vessel injection nozzle. However, if the counter pressure is not sufficient to prevent CMT injection, a boration would occur similar to a CVS malfunction event. The limiting CVS malfunction scenario presented in Subsection 15.5.2 minimizes RCS average temperature, maximizes RCS pressure, and maximizes pressurizer water volume at the time of "S" signal. This scenario bounds the inadvertent opening of the CMT discharge valves.

Insert 2

The case of a Spurious "S signal is analyzed here. Following a spurious "S" signal, the reactor is tripped, the CMT discharge valves are opened, the RCPs are tripped, the PRHR is actuated, the pressurizer heaters are blocked, and the two charging pumps on the chemical and volume control system (CVS) are isolated. Following reactor trip, the reactor power drops and the average RCS temperature decreases with subsequent coolant shrinkage. The PRHR heat exchanger further contributes to the shrinkage. However, a few seconds after reactor trip, the RCPs trip, and the CMTs start injecting highly borated water into the RCS. The primary coolant system shrinkage is counteracted by the CMT injection and the pressurizer volume starts to increase because of the heat up of the cold injected fluid by the decay heat.

Insert 3

Eventually, the CMTs heat up and the gravity driven recirculation is significantly reduced. The PRHR heat exchanger continues to extract heat from the RCS and the pressurizer water volume starts to decrease. Ultimately, the CMTs stop recirculating and the PRHR heat removal matches decay heat.

Insert 4

CMT enthalpies have been maximized. This is conservative, because it minimizes the cooling provided by the CMTs as flow recirculates, thereby increasing the peak pressurizer water volume during the transient. PRHR heat transfer capability has been minimized.

Insert 5

The initial reactor power is assumed to be 102% of nominal. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial RCS average temperature is assumed to be 6.5°F above nominal.

Insert 6

The transient is initiated by a spurious "S" signal. The CMT discharge valves are assumed to open immediately and begin injecting 3400 ppm water when the RCPs are tripped.

Insert 7

The pressure in the primary and secondary systems increase initially due to the assumed unavailability of the non-safety related control systems. The primary and secondary system pressures eventually decrease as the PRHR system removes decay heat.

REPLACE WITH INSERT 6

The transient is initiated by a spurious "S" signal. The CVS pumps actuate and the CMT discharge and balance line valves open after a seven second delay. The CMTs are assumed to be filled with 3500 ppm borated water and the water from the CVS contains 4375 ppm boron.

- Reactor Trip

  Reactor trip is initiated by the "S" signal.

- Plant systems and equipment which are available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affect the consequences of the accident.

15.5.1.3 Results

Figures 15.5.1-1 through 15.5.1-~~7~~ 8 show the transient response to the inadvertent operation of the CMT during power operation. The spurious "S" signal occurs at 10 seconds accompanied by the ~~actuation~~ opening of the ~~two CVS pumps~~ CMT discharge valves. After a two second delay the neutron flux starts decreasing immediately due to the reactor trip which is immediately followed by the turbine trip. ~~The CMT valves are assumed to open at 17 seconds followed by a trip of~~ The reactor coolant pumps are tripped at ~~19~~ 25 seconds. ~~The CVS pumps are isolated and~~ The PRHR is actuated ~~at the high pressurizer level set point but CMT injection is sustained throughout the transient~~ at 32 seconds, following a conservative time delay. The departure from nucleate boiling ratio increases throughout the transient. The pressure in the primary and secondary systems increase due to the assumed unavailability of the non safety-related control system.

REPLACE WITH INSERT 7

The CMTs work in recirculation mode, meaning they are always filled with water since cold borated water injected through the injection line is replaced by hot water coming from the cold leg (balance line). The ~~RCS mass increases during the transient and the~~ pressurizer level increases~~.~~, until the CMT recirculation is decreased sufficiently and the PRHR system heat removal rate approaches that of the core decay heat generation.

NEW ¶ INSERT 8

The calculated sequence of events is shown in Table 15.5-1. Recovery from the accident is discussed in Subsection 15.5.1.1.

15.5.1.4 Conclusions

Replace with Insert 9

Results of the analysis show that spurious CMT operation presents no challenge to the integrity of the RCS. If the accident is due to a valve failure or to a spurious signal at the component level, no injection is expected from the CMTs, and a reactor trip signal is not enerated. The operator is alerted by the status lights in the main control room and takes action to terminate the event as discussed in Subsection 15.5.1.1.

If the reactor is tripped as an immediate consequence of the spurious "S" signal, the plant operating conditions never approach the core limit. Pressurizer pressure always remains below 110 percent of the design limit. The operator has more than 83 minutes to terminate the accident before primary water is discharged from the pressurizer safety valves.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

15.5.2.1 Identification of Causes and Accident Description

An increase of reactor coolant inventory which results from addition of cold unborated water to the reactor coolant system is analyzed in Subsection 15.4.6, chemical and volume control system malfunction that results in a decrease in boron concentration.

In this Subsection 15.5.2, the increase of RCS inventory due to the addition of borated water is analyzed.

The increase of RCS coolant inventory may be due to the spurious operation of one or both of the CVS pumps or by the closure of the letdown path. If the CVS is injecting highly borated water into the RCS, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage and pressurizer pressure and water level decrease. The load decreases due to the effect of reduced steam pressure after the turbine throttle valve fully opens. ~~If the automatic rod control system is used, these effects are lessened by the rods moving out of the core. More mass accumulates in the RCS. While the reactor is eventually tripped by the reactor protection system (low pressurizer pressure, high pressurizer water level or low steam line pressure "S" signal reactor trip), the RCS pressure and volume transient is not over until the CVS and CMTs are isolated and stop injecting water into the RCS. While the CVS is automatically isolated on a high pressurizer water level signal, the CMTs are manually isolated by the operator.~~

~~The time of the trip is affected by initial operating conditions, including core burnup history, which affects initial boron concentration, rate of change of boron concentration, and doppler and moderator coefficients.~~

Replace with Insert 10

~~CVS pumps are automatically tripped and the PRHR is actuated on the pressurizer high water level trip. If the CMTs are not actuated during the transient, CVS isolation terminates the accident. If the CMTs are actuated during the transient, the transient proceeds like a spurious CMT actuation transient after the CVS is isolated. The only difference is the lower average temperature and pressure associated with the power mismatch during the first part of the transient. The operator then determines if conditions exist to isolate the CMTs and terminate the accident.~~

~~The time at which the reactor trip occurs is of no concern for the transient.~~

This event is a Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

Insert 8
At about 8,700 seconds, the PRHR heat flux approaches core decay heat, and the pressurizer water volume stops increasing. At about 10,000 seconds, the CMTs essentially stop injecting, and the pressurizer safety valves close. At about 15,000 seconds, the PRHR heat flux matches the core decay heat.

Insert 9
Results of the analysis show that a spurious "S" signal and subsequent CMT operation does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the safety analysis limit values and RCS and steam generator pressures remain below 110% of their design values. If the accident is due to the inadvertent opening of the CMT discharge valves, due to operator error or valve failure, CMT injection flow is not expected if the RCPs are operating, because the RCPs provide a counter pressure at the direct vessel injection nozzle. However, if the counter pressure is not sufficient to prevent CMT injection, a boration would occur similar to a CVS malfunction event. The limiting CVS malfunction scenario presented in Subsection 15.5.2 minimizes RCS average temperature, maximizes RCS pressure, and maximizes pressurizer water volume at the time of "S" signal. This scenario bounds the inadvertent opening of the CMT discharge valves.

Insert 10
At high CVS boron concentrations, low reactivity feedback conditions, and the reactor in manual rod control, an "S" signal will be generated by either the low Tcold or low steamline pressure setpoints before the CVS can inject a significant amount of water into the RCS. In this case, the CVS malfunction event proceeds similarly to and is only slightly more limiting than the Spurious "S" signal event analyzed in Subsection 15.5.1. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is terminated by automatic isolation of the CVS on the safety-related high pressurizer level setpoint.

Under typical operating conditions for the AP600, the boron concentration of the injected CVS water is equal to that of the RCS. If the CVS is functioning in this manner and the pressurizer spray system functions properly to prevent a high pressure reactor trip signal, no "S" signals are generated and this specific event is also terminated by automatic isolation of the CVS on the safety related high pressurizer level setpoint.

While the scenarios discussed above are the most probable outcomes of a CVS malfunction, several combinations of boron concentrations, feedback conditions, and plant system interactions have been identified which can result in more limiting scenarios with respect to pressurizer overfill. The key factors which make this event more limiting than the Spurious "S" signal event analyzed in Subsection 15.5.1 are that the RCS is at a lower average temperature, higher pressure, and a higher pressurizer level at the time an "S" signal is generated. These factors produce a greater volume of higher density water, and thus, a larger RCS mass at the time of the "S" signal. In addition, at lower RCS average temperature, the PRHR is less effective in removing decay heat, which results in greater expansion of the cold

water injected by the CMTs.

The limiting analysis scenario minimizes RCS average temperature, maximizes RCS pressure, and maximizes pressurizer water volume at the time of "S" signal. This scenario is as follows:

> Both of the CVS pumps spuriously begin delivering flow at a boron concentration slightly higher than that of the RCS. (Assuming that a CVS malfunction results in both CVS pumps delivering flow is a conservative assumption. One CVS pump is automatically controlled and one is manually controlled.)
>
> The nonsafety-related pressurizer spray is assumed to be unavailable, and a high pressurizer pressure reactor trip signal is generated.
>
> Soon after reactor trip, main feedwater is isolated on a low Tavg coincident with reactor trip signal, and as voids collapse in the steam generator a low steam generator level narrow range signal is assumed to be generated.
>
> The nonsafety-related startup feedwater system is assumed unavailable, such that the PRHR heat exchanger is actuated after a time delay on a low steam generator level coincident with low startup feedwater signal.
>
> The PRHR cools the plant and a low Tcold "S" signal is generated. On an "S" signal, the CVS is automatically isolated, the CMT discharge valves are opened, the RCPs are tripped, and the pressurizer heaters are blocked.

Insert 11
Using an iterative analysis process, a set of reactivity feedback parameters, CVS flow, CVS boron concentration, and initial pressurizer water volume are chosen such that the time of the low Tcold "S" signal is coincident with the time of the high pressurizer level CVS isolation signal. The pressurizer pressure is also at the safety valve setpoint at the time of "S" signal. These conditions result in the most limiting transient with respect to margin to pressurizer overfill.

Insert 12
The initial reactor power is assumed to be 102% of nominal. The initial pressurizer pressure is assumed to be 50 psi above nominal. The initial RCS average temperature is assumed to be 6.5°F above nominal.

Insert 13
A least negative moderator temperature coefficient, a low (absolute value) Doppler power coefficient, and a maximum boron worth are assumed. The most limiting CVS boron concentration is chosen for this set of reactivity feedback parameters. For a different set of reactivity feedback parameters, a different CVS boron concentration can result in an identical transient.

15.5.2.2 Analysis of Effects and Consequences

15.5.2.2.1 Method of Analysis

heat exchanger

The malfunction of the CVS system is analyzed by employing a modified version of the computer program LOFTRAN (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, and the passive residual heat removal ~~(PRHR)~~ and the CMT ~~system~~. The program computes pertinent plant variables including temperatures, pressures and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. ~~Analysis of several cases has shown that the results are relatively independent of the time of the trip.~~

Replace with Insert 11

~~A typical transient is presented representing the minimum reactivity feedback and the rod control system in manual operation.~~

~~The CVS system is conservatively simulated (flow rates to the RCS are maximized). Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.~~

The assumptions are as follows:

- Initial Operating Conditions

  Replace with Insert 12

  The initial reactor power, RCS pressure and average temperature are at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

- Moderator and Doppler coefficients of reactivity

  Replace with Insert 13

  A least negative moderator temperature coefficient is used. A low (absolute value) Doppler power coefficient is assumed.

- Reactor Control

  Rod control is not modeled.

- Pressurizer Heaters

  Replace with Insert 14

  Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure drop.

- Boron Injection

  Replace with Insert 15

  After 10 seconds at steady state, two CVS pumps start injecting 4375 ppm borated water into the RCS.

- Turbine Load

  The turbine load is assumed constant until the governor drives the throttle valve wide open. Then the turbine load drops as steam pressure drops.

- Reactor Trip

  Replace with Insert 16

  Reactor trip is initiated by either a low pressurizer pressure or a low steam line pressure. Plant systems and equipment which are available to mitigate the effect of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely effects the consequences of the accident.

15.5.2.2.2 Results

Figures 15.5.2-1 through 15.5.2-~~8~~ 9 show the transient response to a CVS malfunction that results in an increase of RCS inventory. Neutron flux ~~starts decreasing immediately~~ slowly decreases due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valves are wide open.

Replace with Insert 17

The mismatch between load and nuclear power causes the RCS average temperature, pressurizer water level, pressurizer pressure and steam generator pressure to drop. When the low steam line pressure "S" signal set point is reached, the reactor trips and the control rods start moving into the core. The departure from nucleate boiling ratio increases throughout the transient.

The CMTs are actuated at about 140 seconds and the RCPs are tripped with a delay of two seconds after the generation of the "S" signal. The RCS mass and pressurizer water level increase and at 1446 seconds the high pressurizer water level set point is reached, the CVS is isolated and the PRHR is actuated. The transient continues since the CMTs inject water in the RCS.

At 4700 seconds there is still a significant steam volume in the pressurizer. The operator terminates the accident as discussed in Subsection 15.5.2.1.

The calculated sequence of events is shown in Table 15.5-1.

15.5.2.3 Conclusions

~~Results of the analysis show that CVS malfunction represents no hazard to the integrity of the RCS.~~ Insert 18

Replace with Insert 19

If the reactor does not trip immediately, either the low pressurizer pressure, the high pressurizer water level, or a "S" signal from the low steam pressure is actuated. This signal also trips the turbine preventing excess cooldown.

~~If during the transient, the CMTs are actuated, the operator determines if conditions exist to stop CMT flow. More than 78 minutes are available to the operator to isolate the CMTs and terminate the event before primary water is discharged from the pressurizer safety valves.~~

15.5.3 Boiling Water Reactor Transients

This subsection is not applicable to the AP600.

15.5.~~4~~5 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

15.5.4 Combined License Information

This section has no requirement for additional information to be provided in support of the combined license application.

Replace with table on next 2 pages

Table 15.5-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time(s)
Inadvertent operation of the ECCS (CMT and CVS actuation) due to a spurious "S" signal during power operation	Spurious "S" signal	10.
	Two CVS pumps actuate	10.
	Rod motion begins	12.
	CMTs actuate	17.
	RCPs trip	25.
	Pressurizer safety valves open	264.
	High pressurizer water level setpoint reached, CVS pumps isolated and PRHR actuated	729.
	Operator terminates event	5000.
CVS malfunction that increases reactor coolant inventory	CVS started	10.
	Low steam pressure "S" signal	131.7
	Rods begin to drop	133.7
	CMTs actuated	138.7
	Steam and feed lines isolation	143.7
	RCP tripped	146.7
	CMT upper dome uncovers	168.
	CMT upper dome recovers	345.
	PRZ safety valves open	528.
	High pressurizer water level setpoint reached, CVS pumps isolated and PRHR actuated	1494.
	Operator terminates event	4700.

Table 15.5-1
(Part 1 of 2)

Time Sequence of Events for Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (s)
Inadvertent operation of the ECCS (CMT and CVS actuation) due to a spurious "S" signal during power operation	Spurious "S" signal	10.
	CMT discharge valves open	10.
	Rod motion begins	12.
	RCPs trip	25.
	Steam and feed lines isolated	22.
	PRHR actuated	32.
	Pressurizer safety valves open	2,200.
	Peak pressurizer water volume	8,700.
	CMTs stop recirculating	10.000.
	Pressurizer safety valves close	10,000.
	PRHR matches decay heat	15,000.

Table 15.5-1
(Part 2 of 2)

Time Sequence of Events for Incidents Which Result in an Increase in Reactor Coolant Inventory

Accident	Event	Time (s)
CVS malfunction that increases reactor coolant inventory	CVS charging pumps started	10.
	High pressurizer pressure reactor trip signal	510.
	Rods begin to drop	512.
	Main feedwater isolated on Low-2 Tavg coincident with reactor trip	516.
	Pressurizer safety valves open	550.
	PRHR actuated on low steam generator narrow range level coincident with low SFW	564.
	Low Tcold "S" signal	991.
	Steamlines isolated	991.
	High pressurizer level CVS isolation setpoint	993.
	CVS charging pumps isolated	1,003.
	RCPs tripped	1,006.
	CMT discharge valves opened	1,013.
	Peak pressurizer water volume	18,800.
	Pressurizer safety valves close	18,800.
	CMTs stop recirculating	20,000.
	PRHR Matches Decay Heat	20,000.

Replace

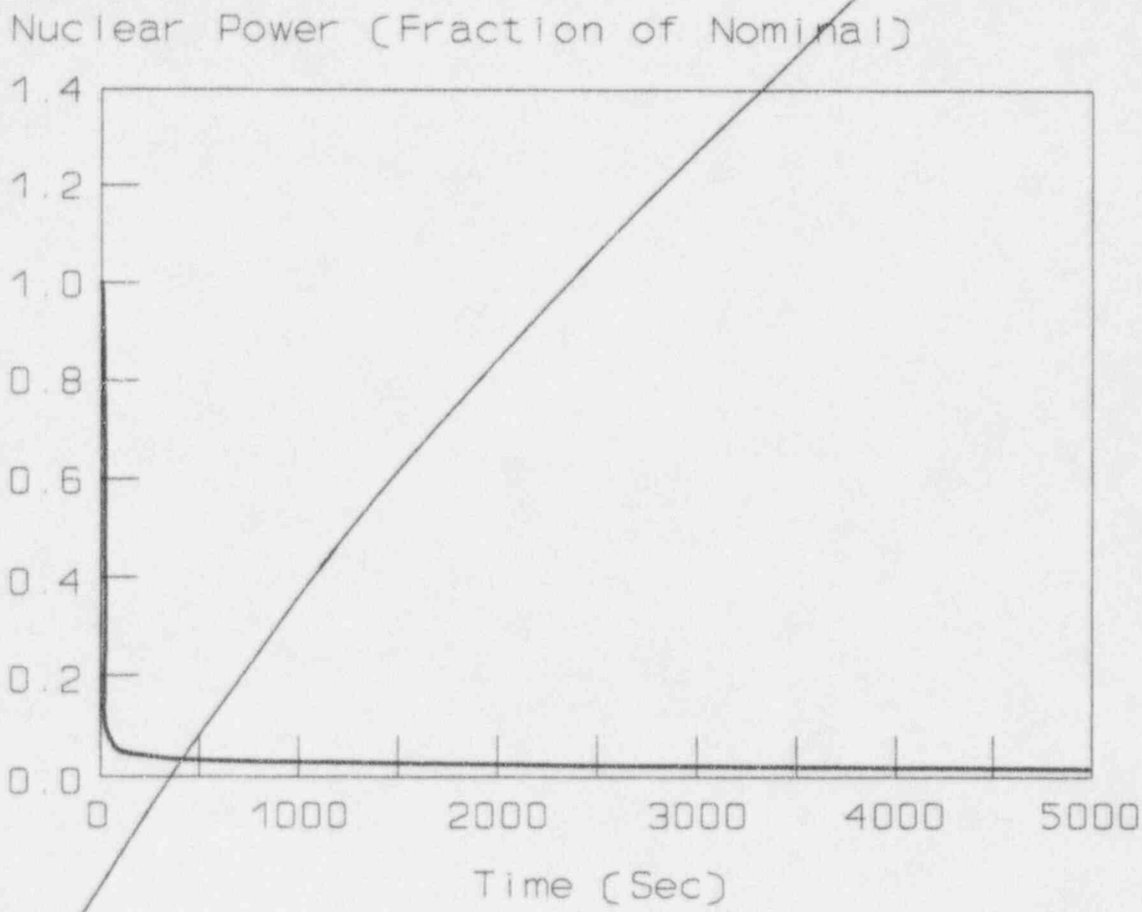
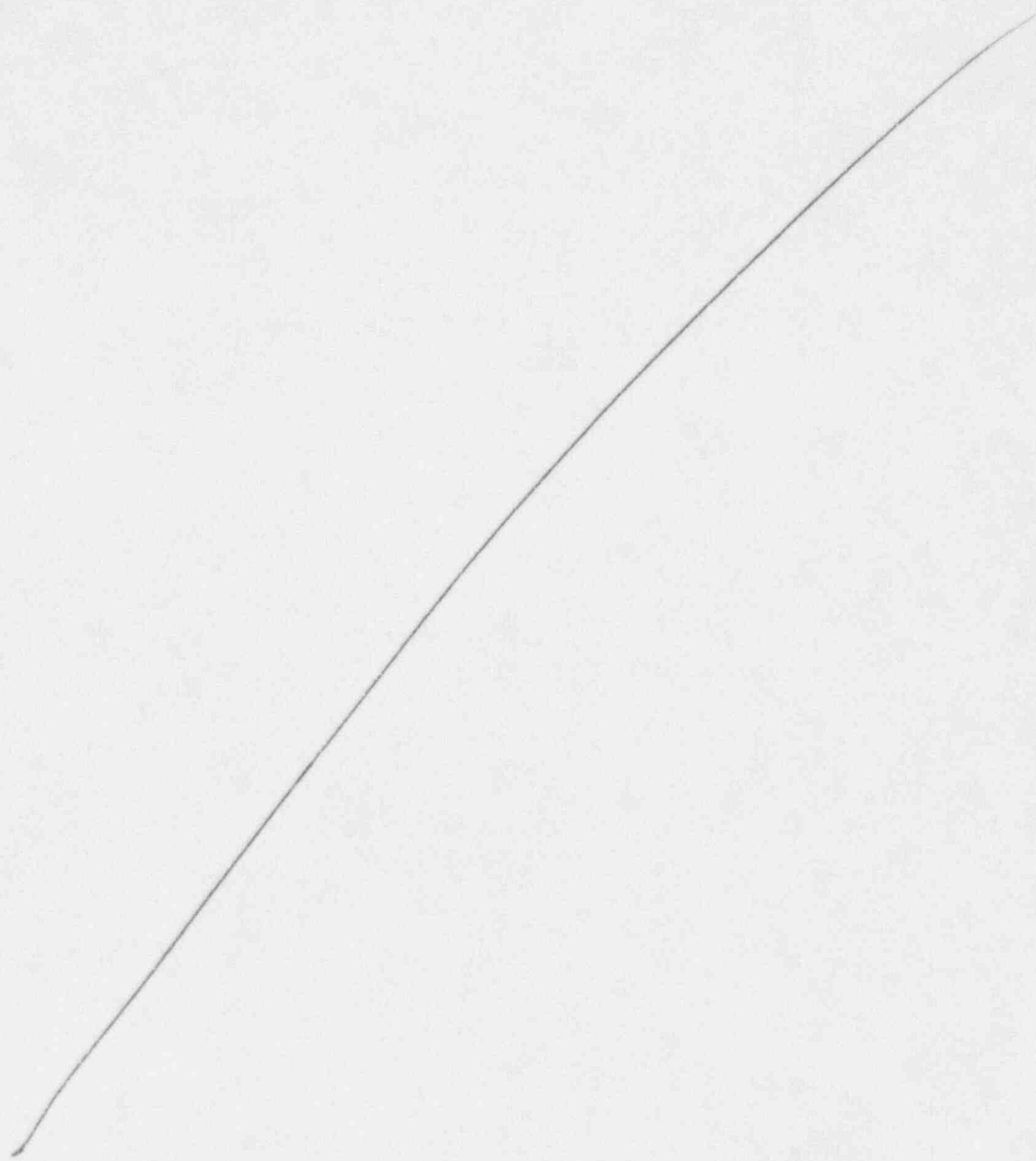
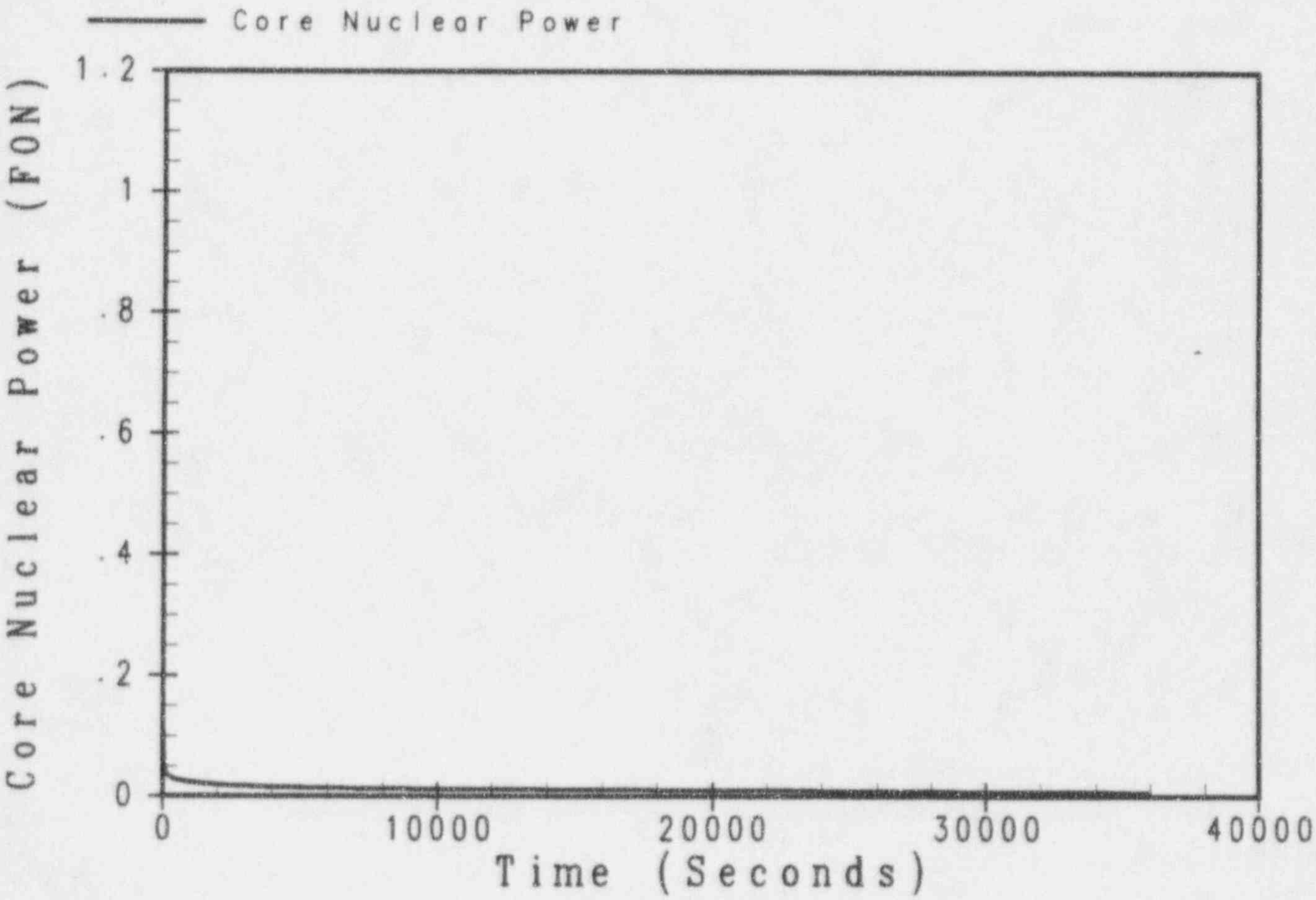


Figure 15.5.1-1

Nuclear Power (Fraction of Nominal) vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal



15.5.1-1


Core Nuclear Power
Core Nuclear Power (FON)
1.2
1
.8
.6
.4
.2
0
0
10000
20000
30000
40000
Time (Seconds)

Replace

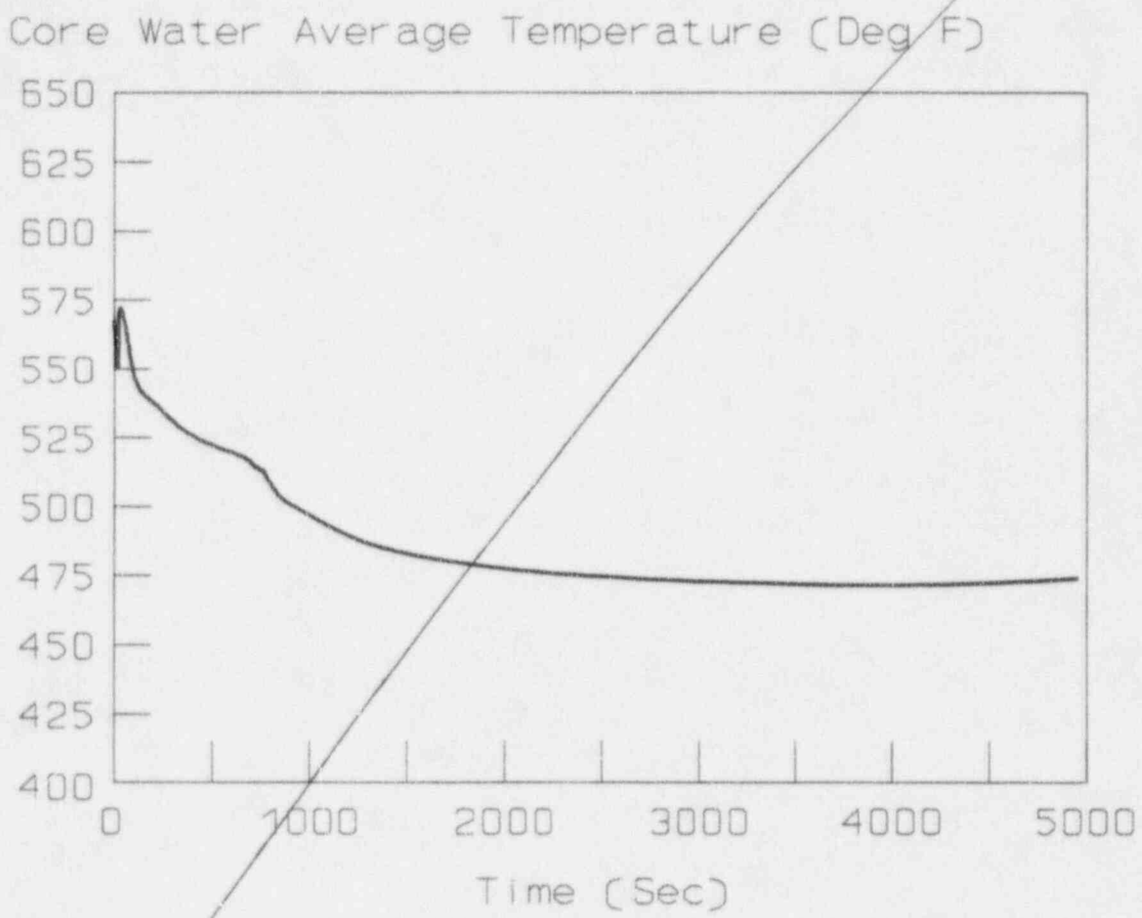
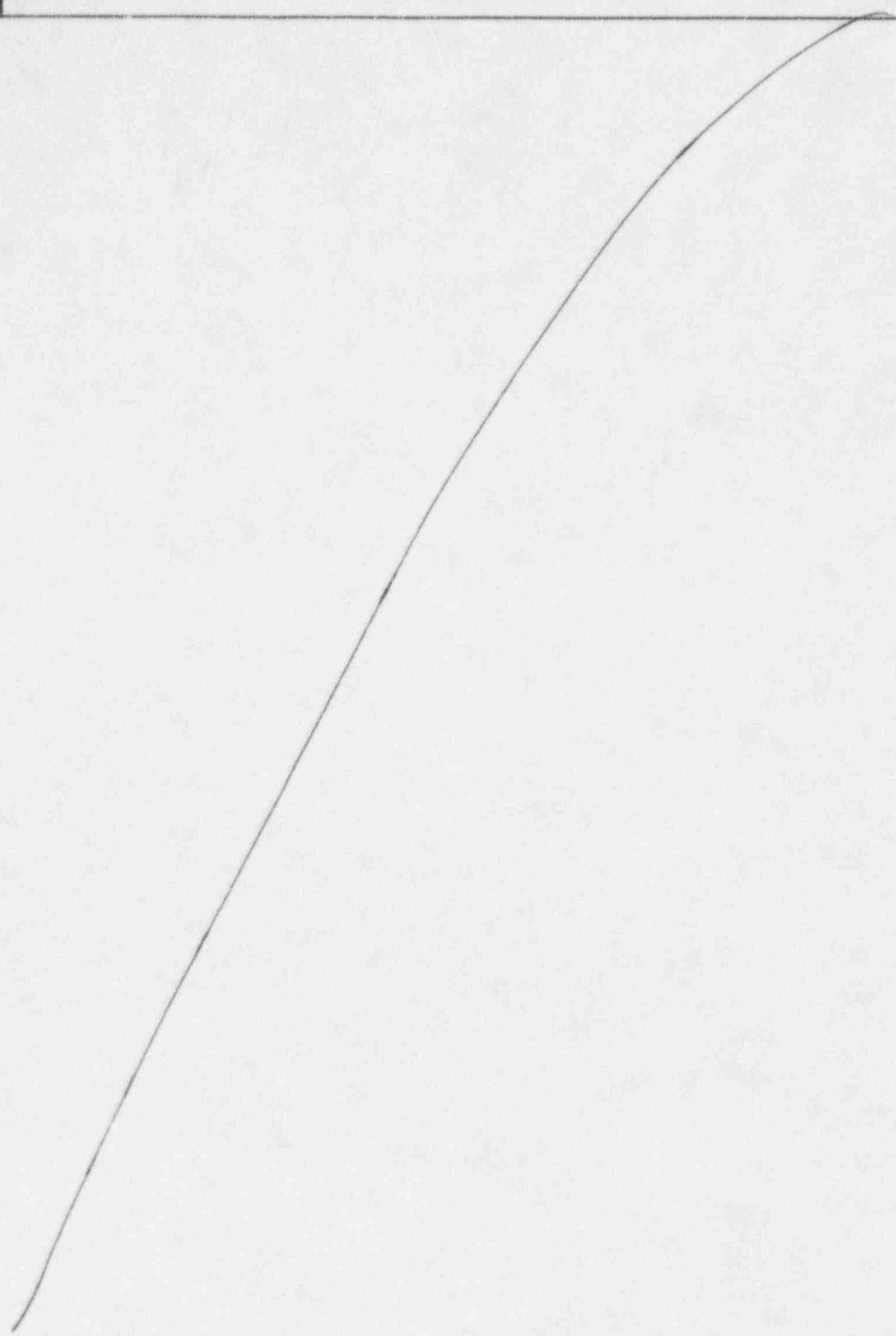
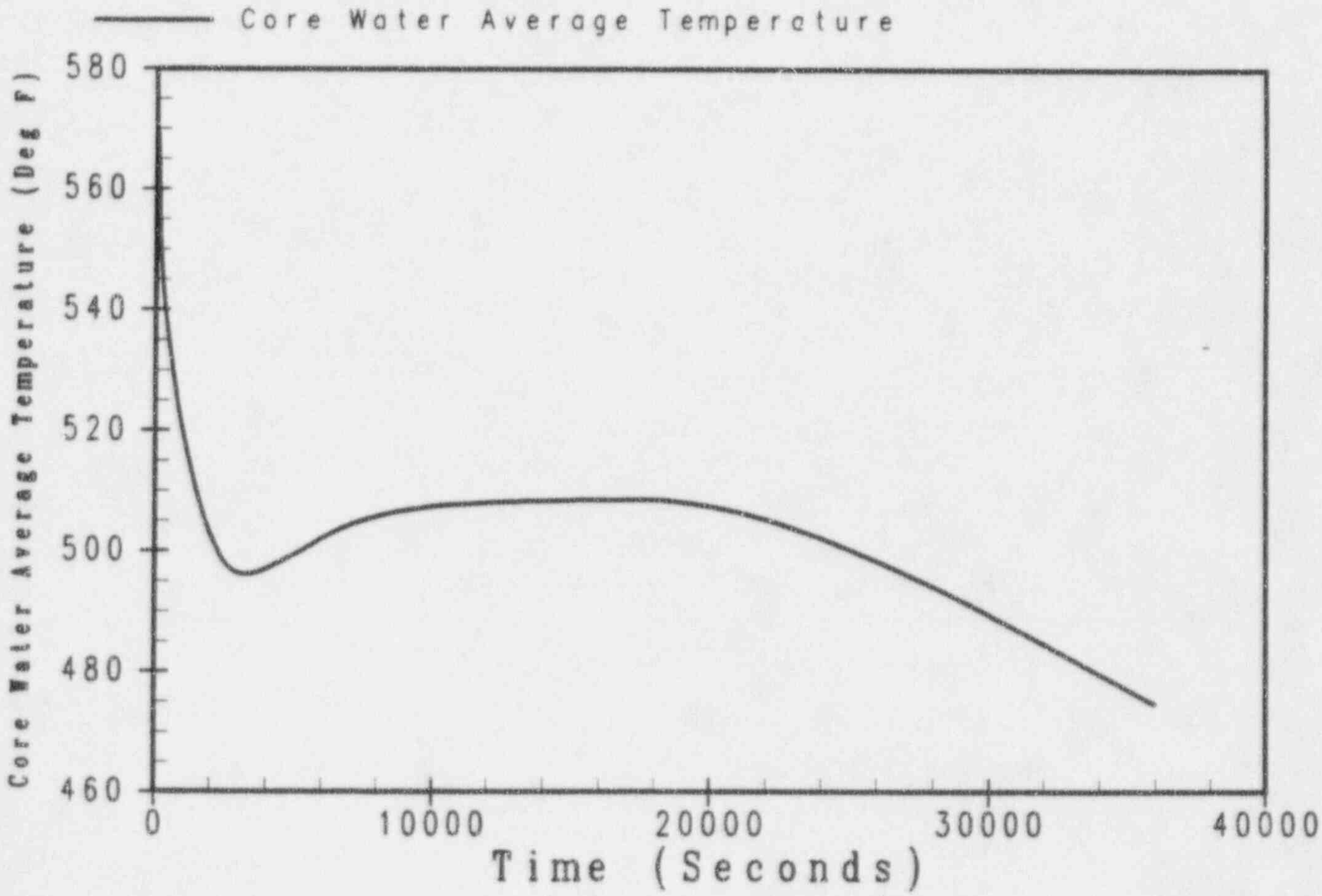


Figure 15.5.1-2

Core Average Water Temperature (°F) vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal



15.5.1-2


Core Water Average Temperature
580
560
540
520
500
480
460
0
10000
20000
30000
40000
Time (Seconds)
Core Water Average Temperature (Deg F)

Replace.

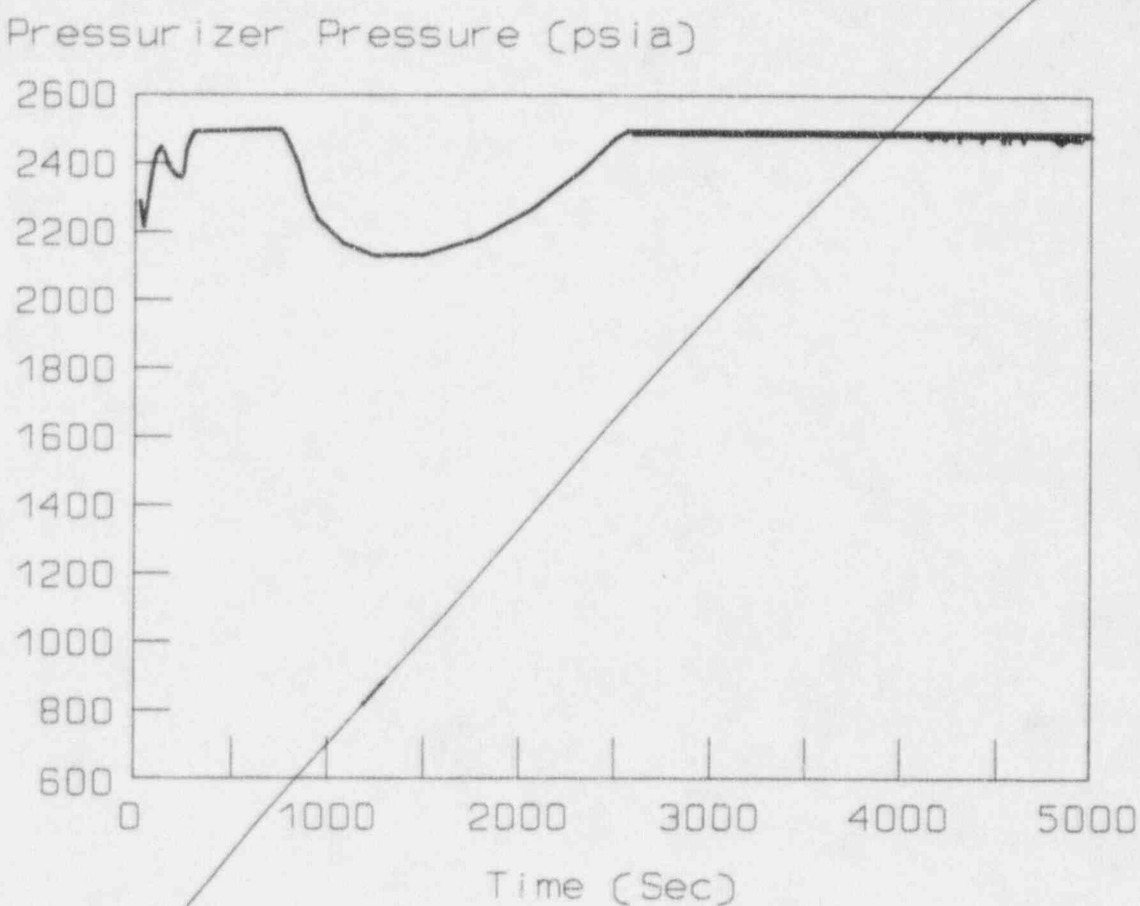
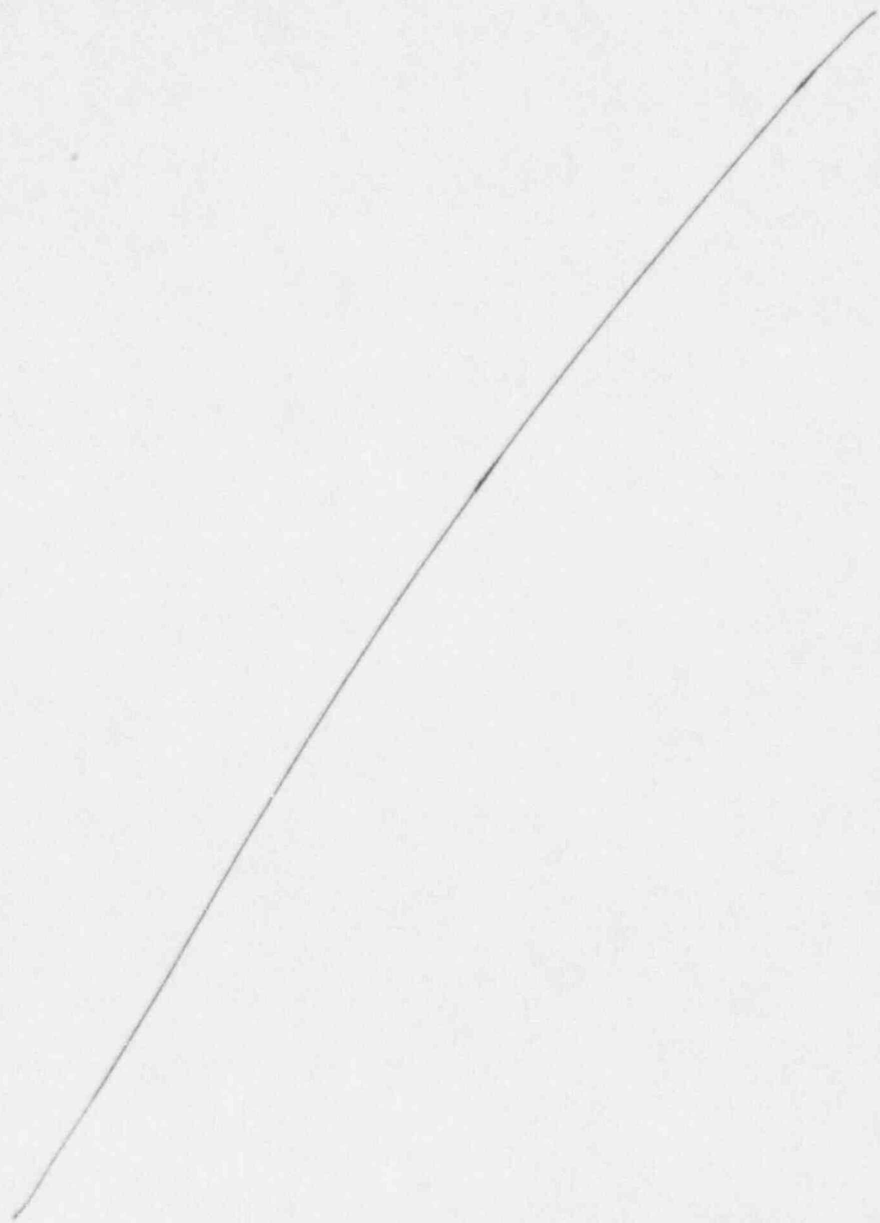
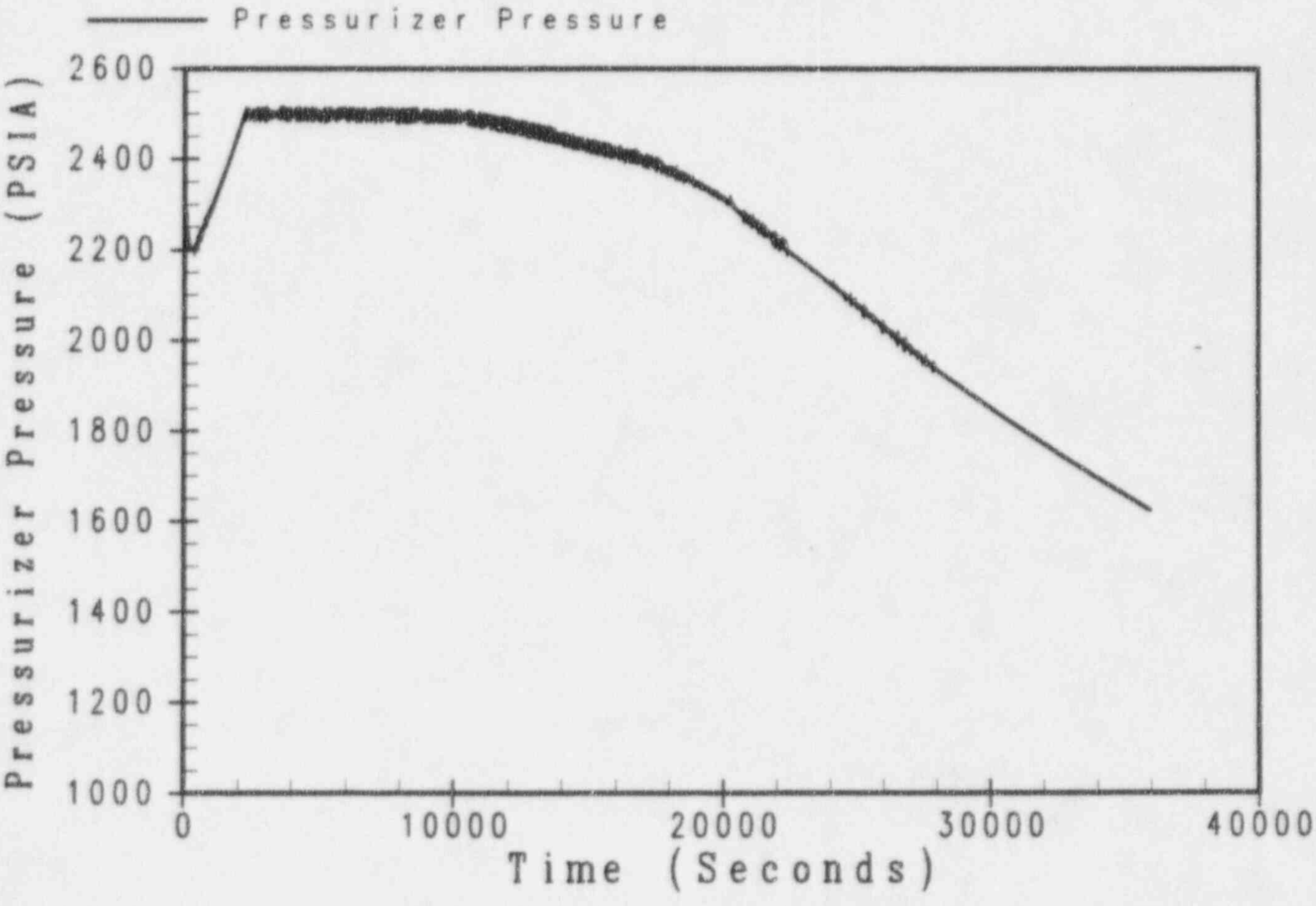


Figure 15.5.1-3

Pressurizer Pressure (psia) vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal



15.5.1-3


Pressurizer Pressure
2600
2400
2200
2000
1800
1600
1400
1200
1000
Pressurizer Pressure (PSIA)
0
10000
20000
30000
40000
Time (Seconds)

Replace

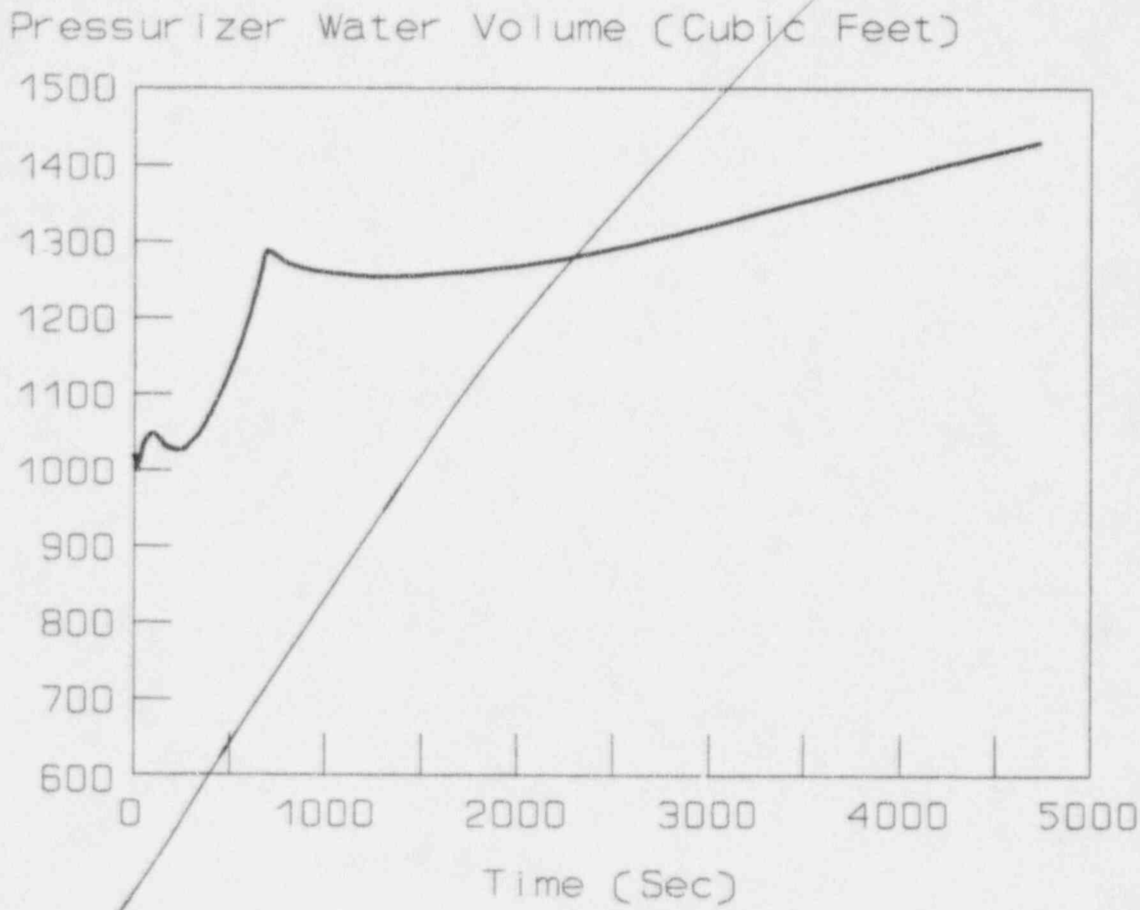
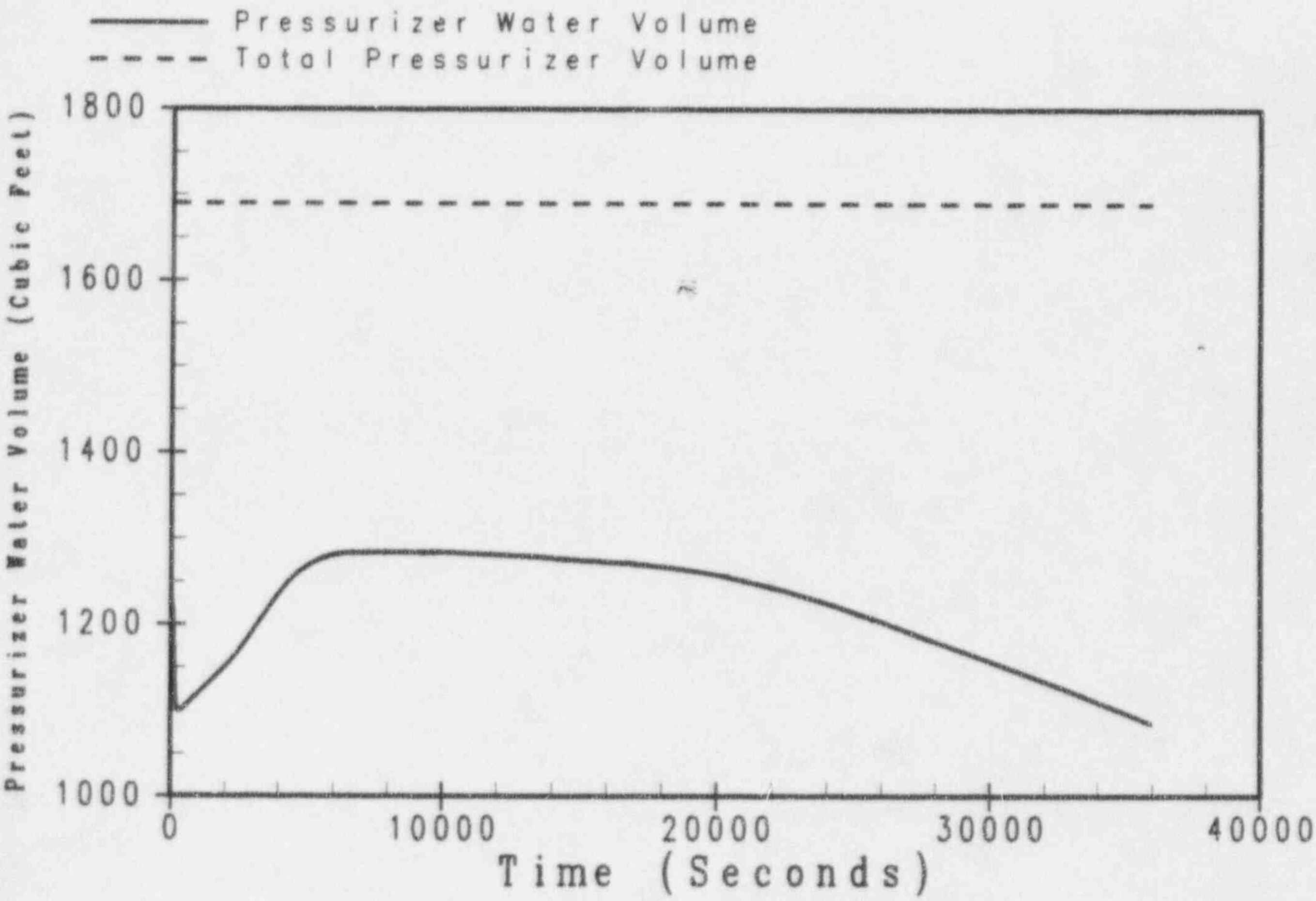


Figure 15.5.1-4

Pressurizer Water Volume (ft^3) vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal

15.5.1-4



Pressurizer Water Volume
Total Pressurizer Volume
1800
1600
1400
1200
1000
Pressurizer Water Volume (Cubic Feet)
0
10000
20000
30000
40000
Time (Seconds)

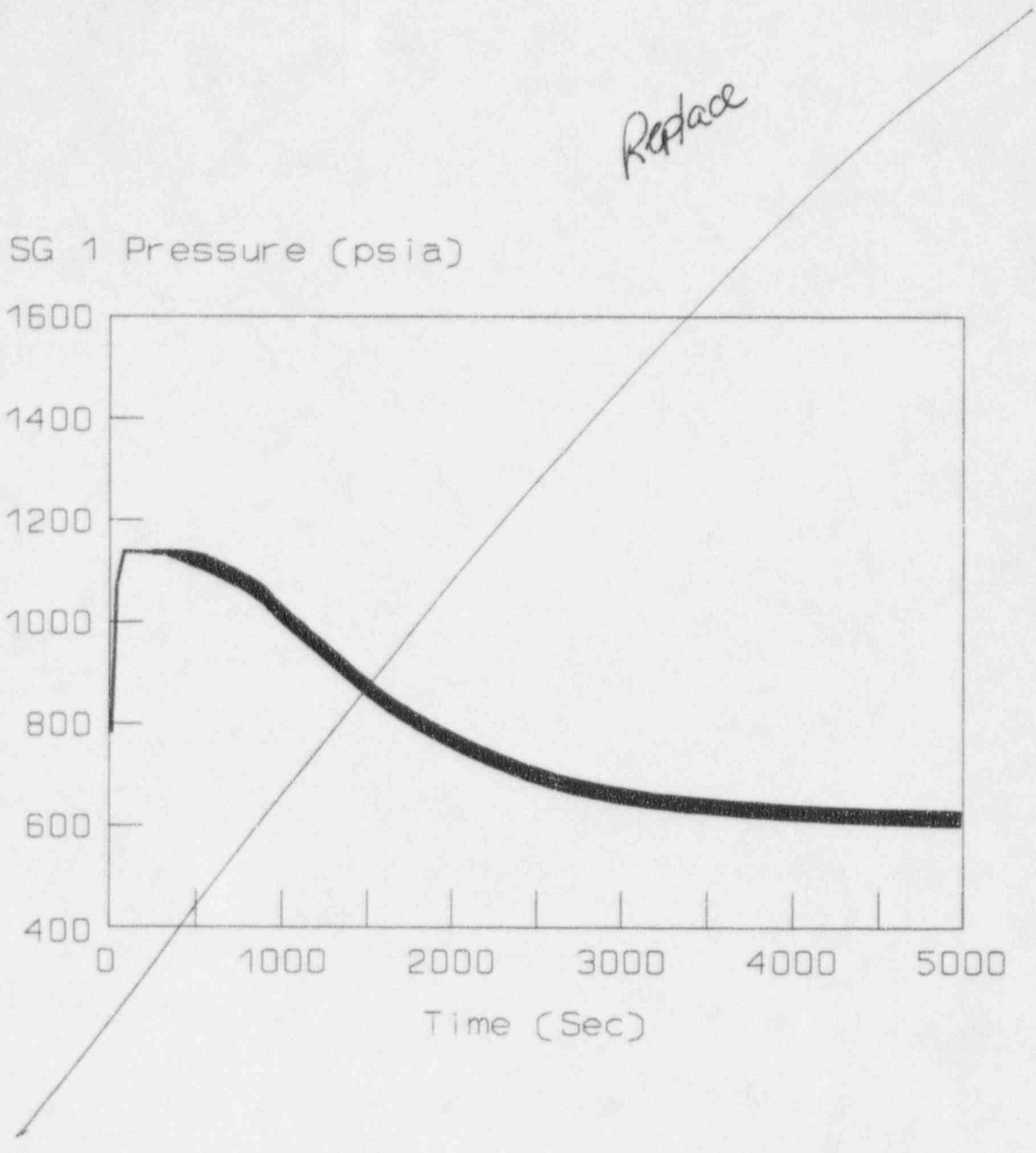
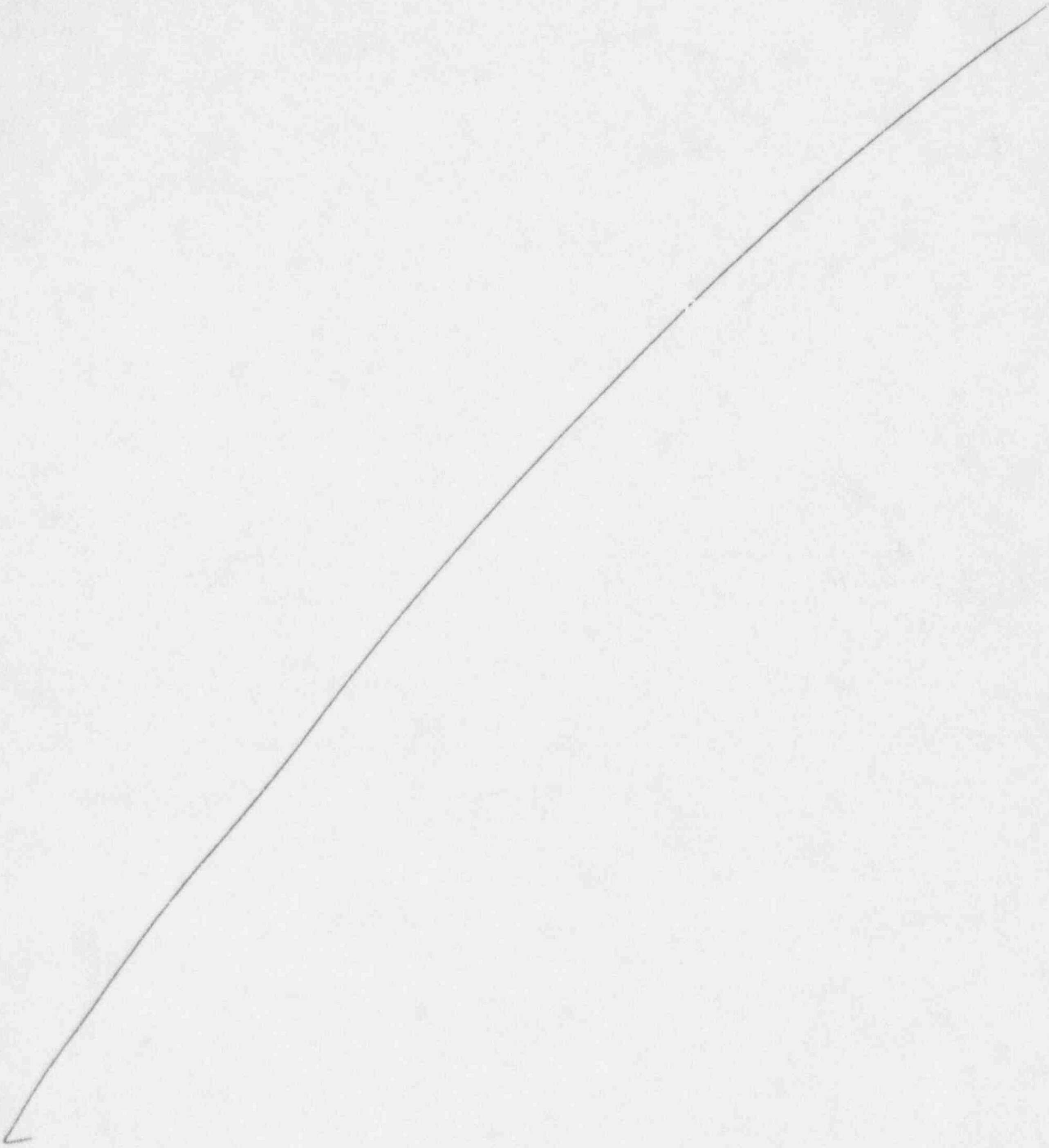
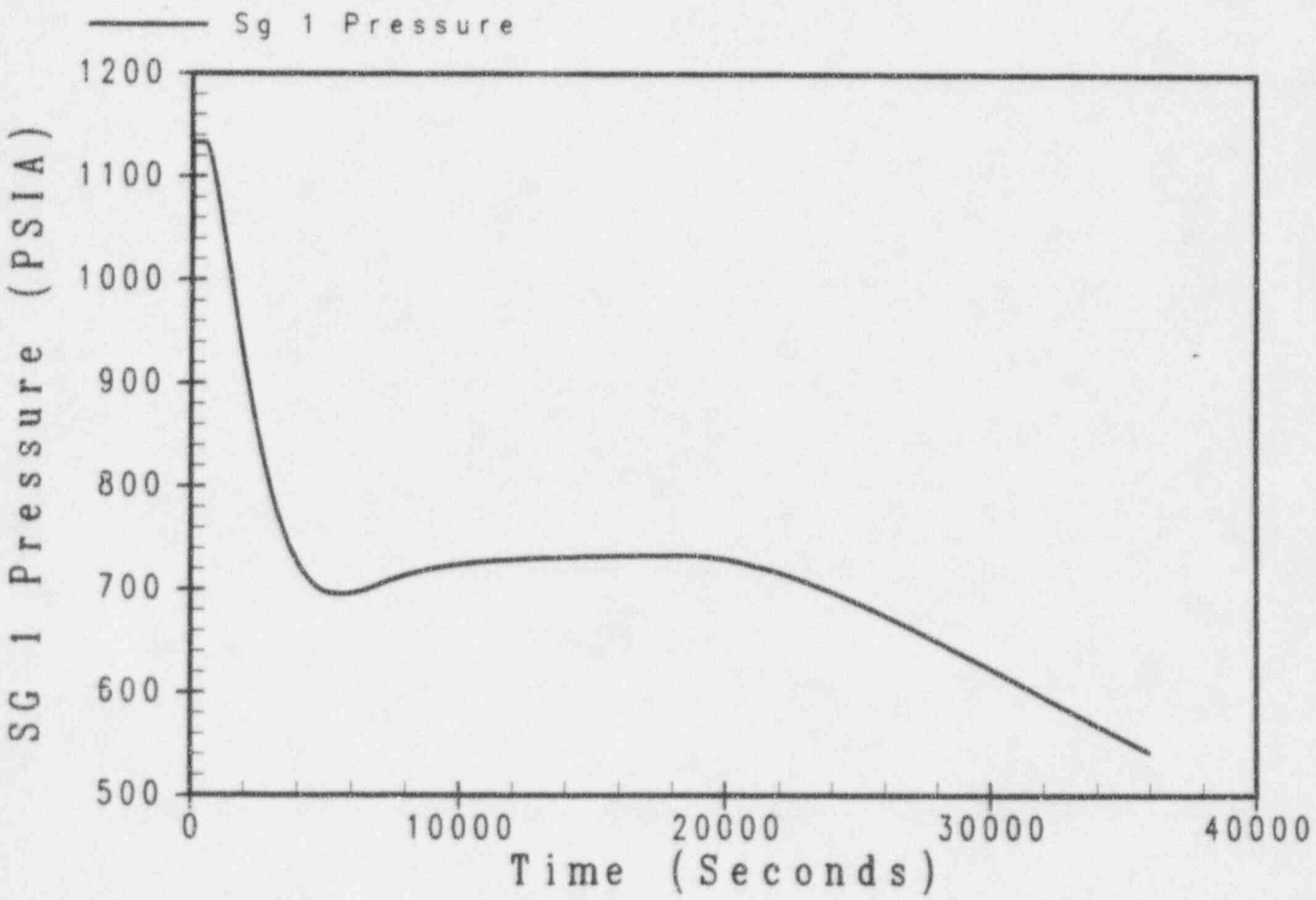


Figure 15.5.1-5

Steam Generator Pressure (psia) vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal



15.5.1-5


Sg 1 Pressure
SG 1 Pressure (PSIA)
1200
1100
1000
900
800
700
600
500
0
10000
20000
30000
40000
Time (Seconds)

Replace

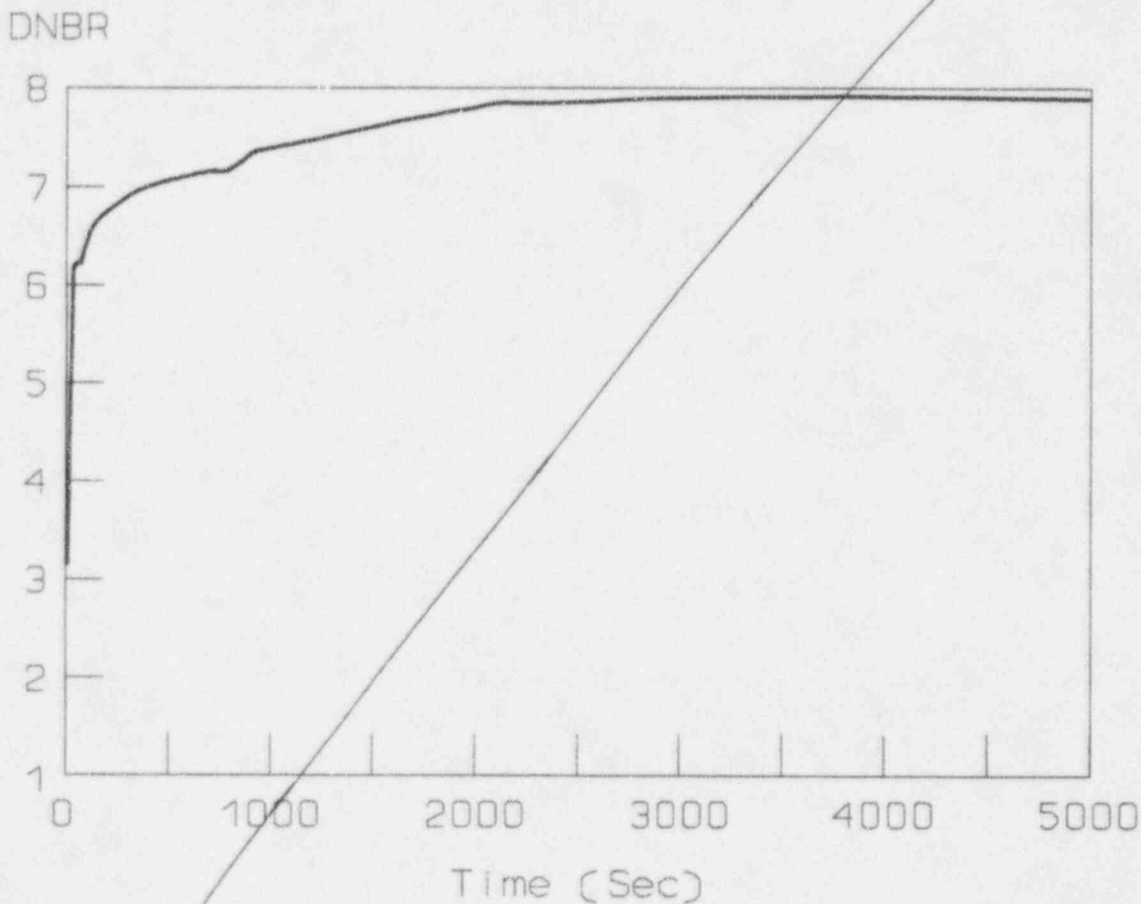
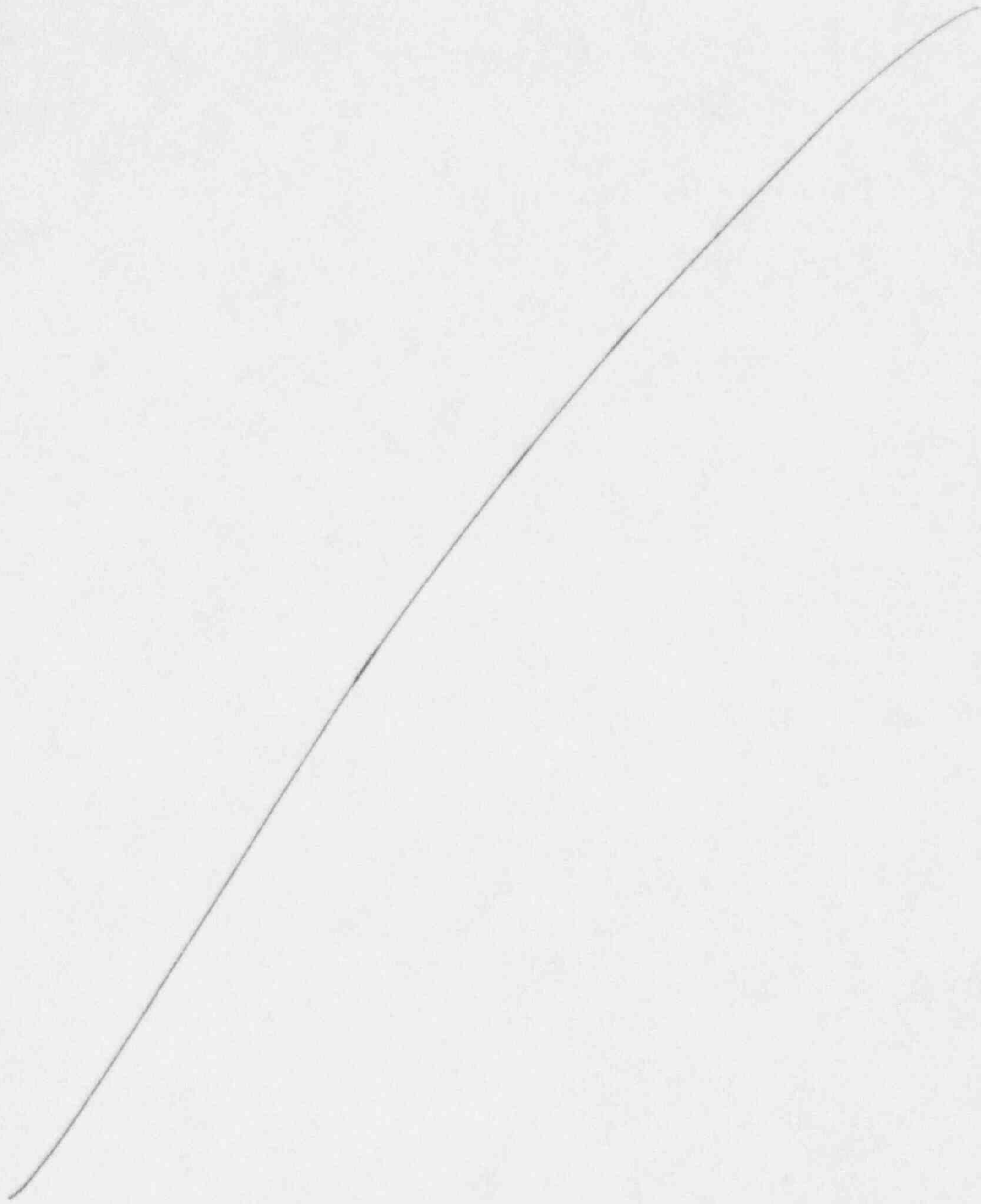
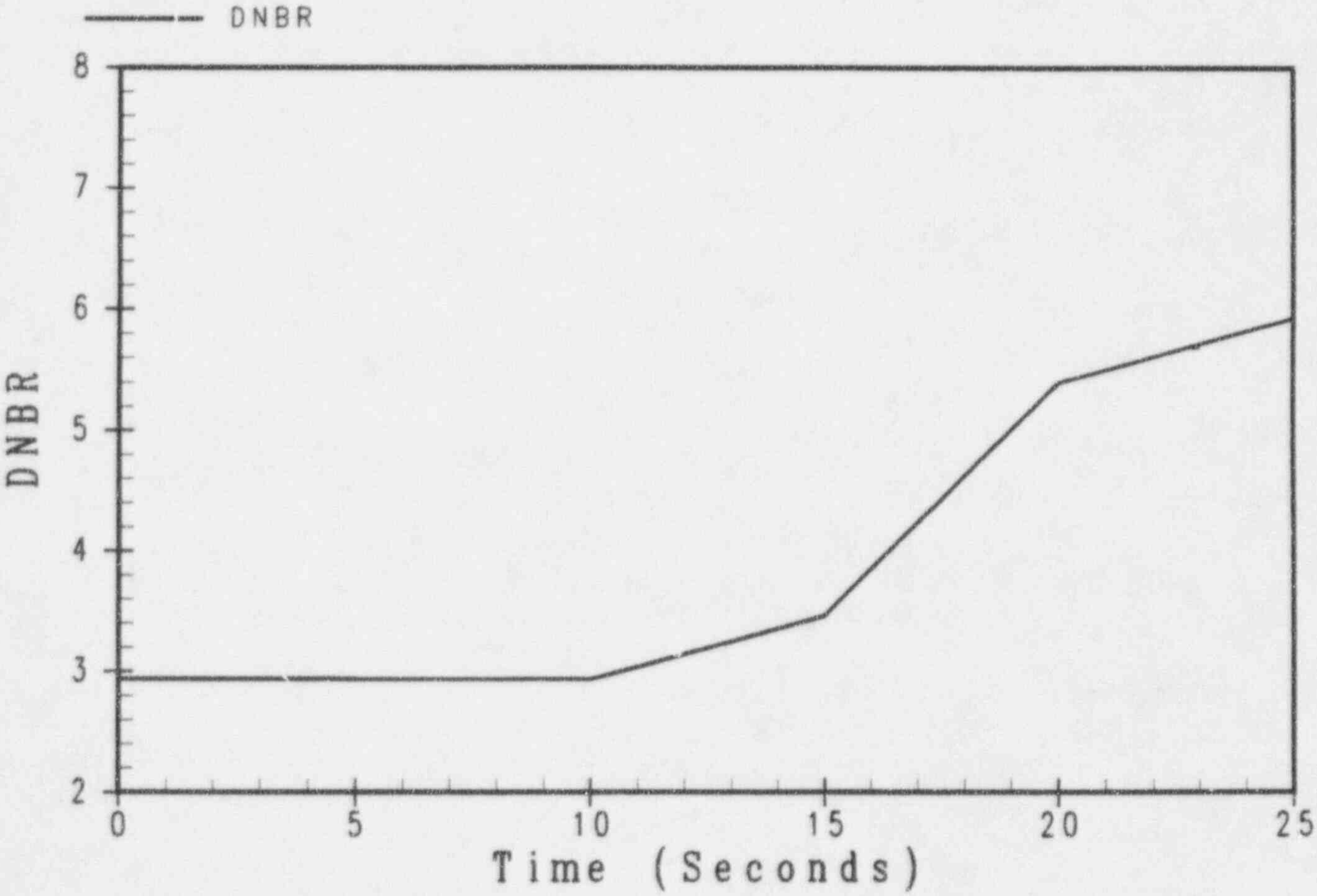


Figure 15.5.1-6

DNB Ratio vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal



15.5.1-6



Replace

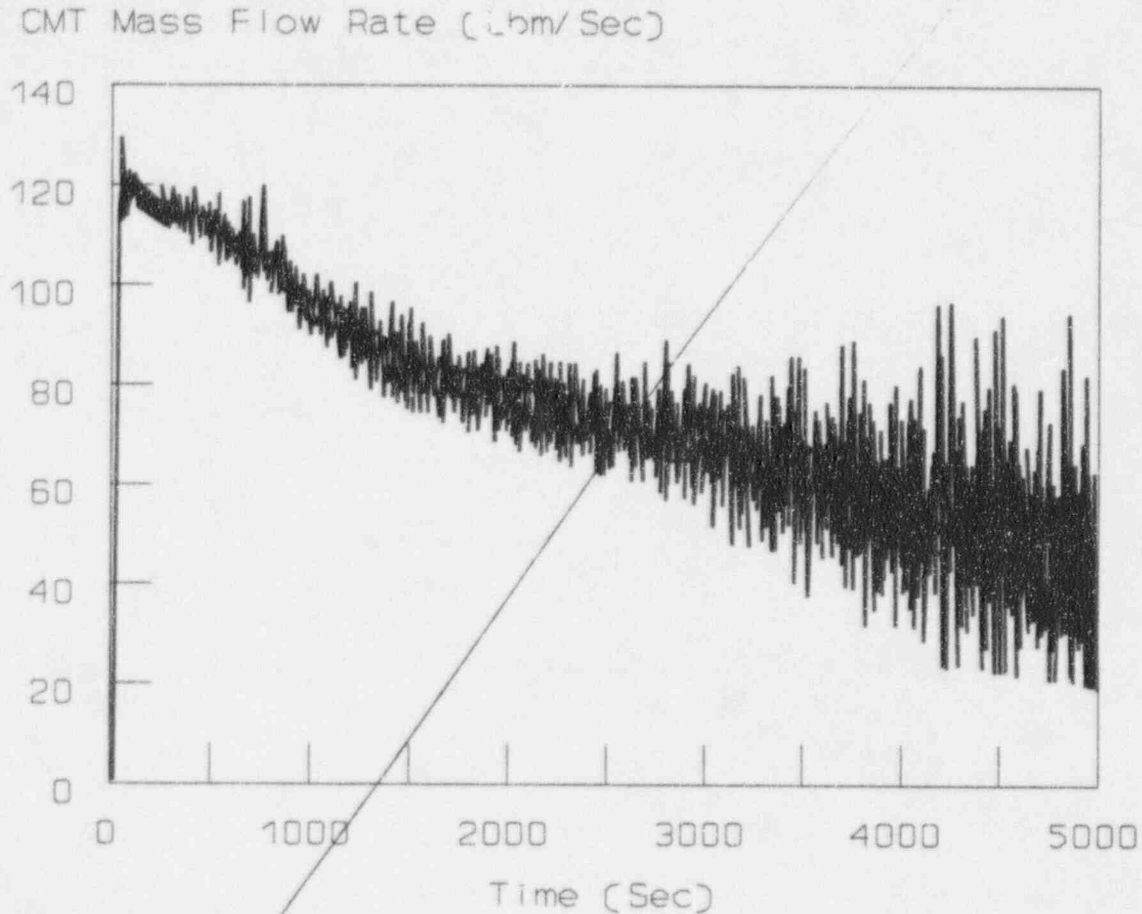
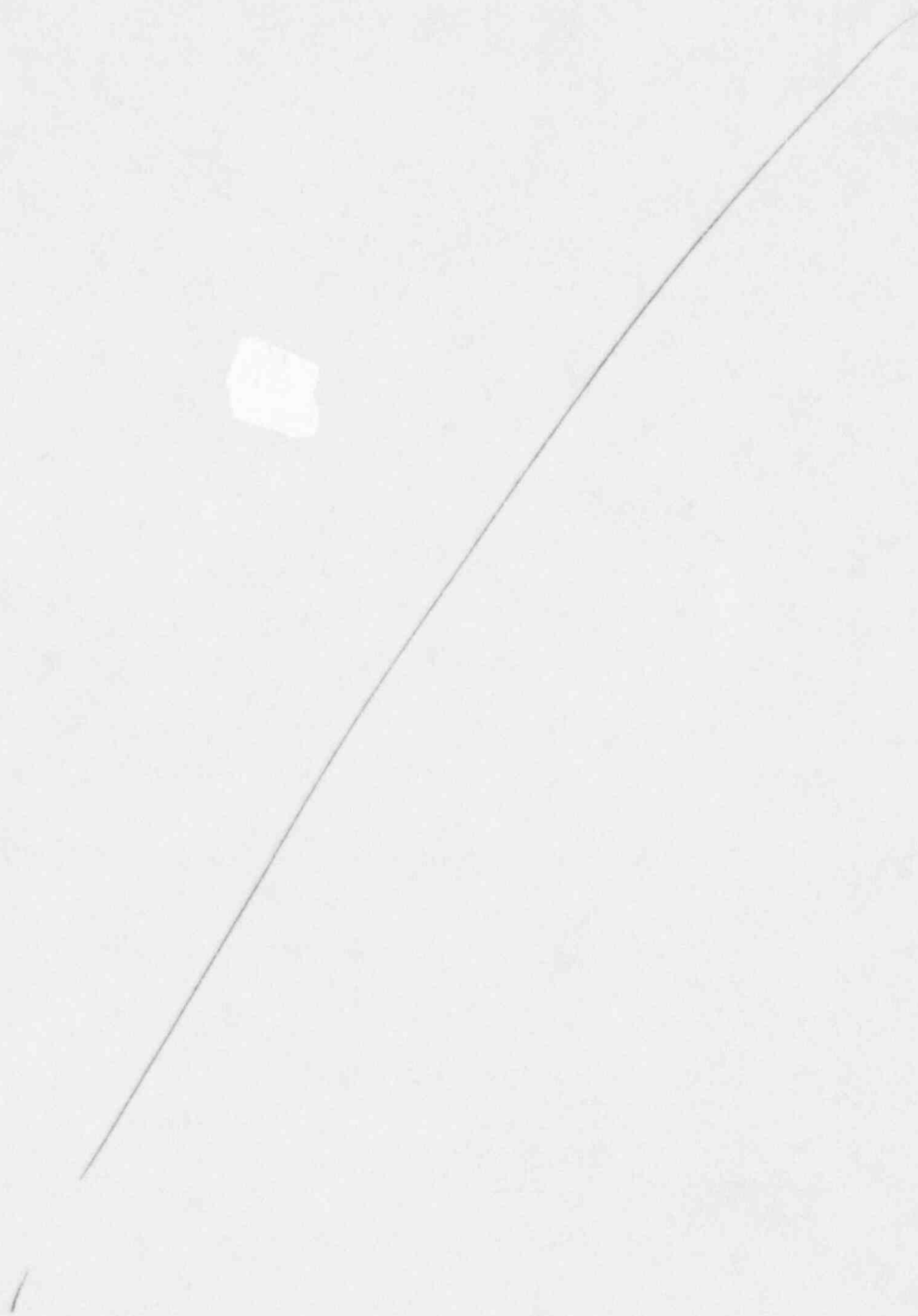
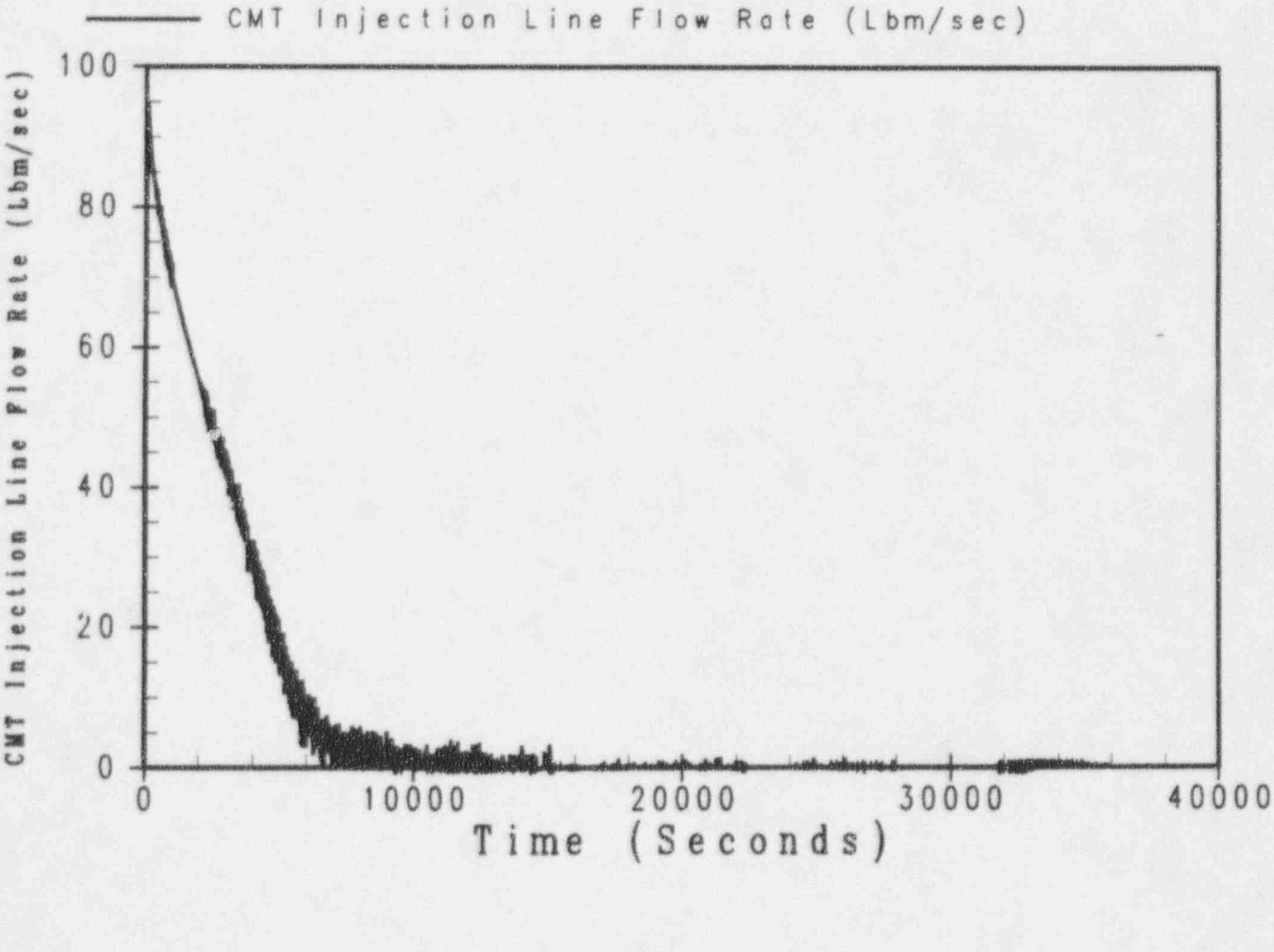


Figure 15.5.1-7

Core Mass Flow Rate vs. Time for Inadvertent Operation of the ECCS due to a Spurious "S" Signal

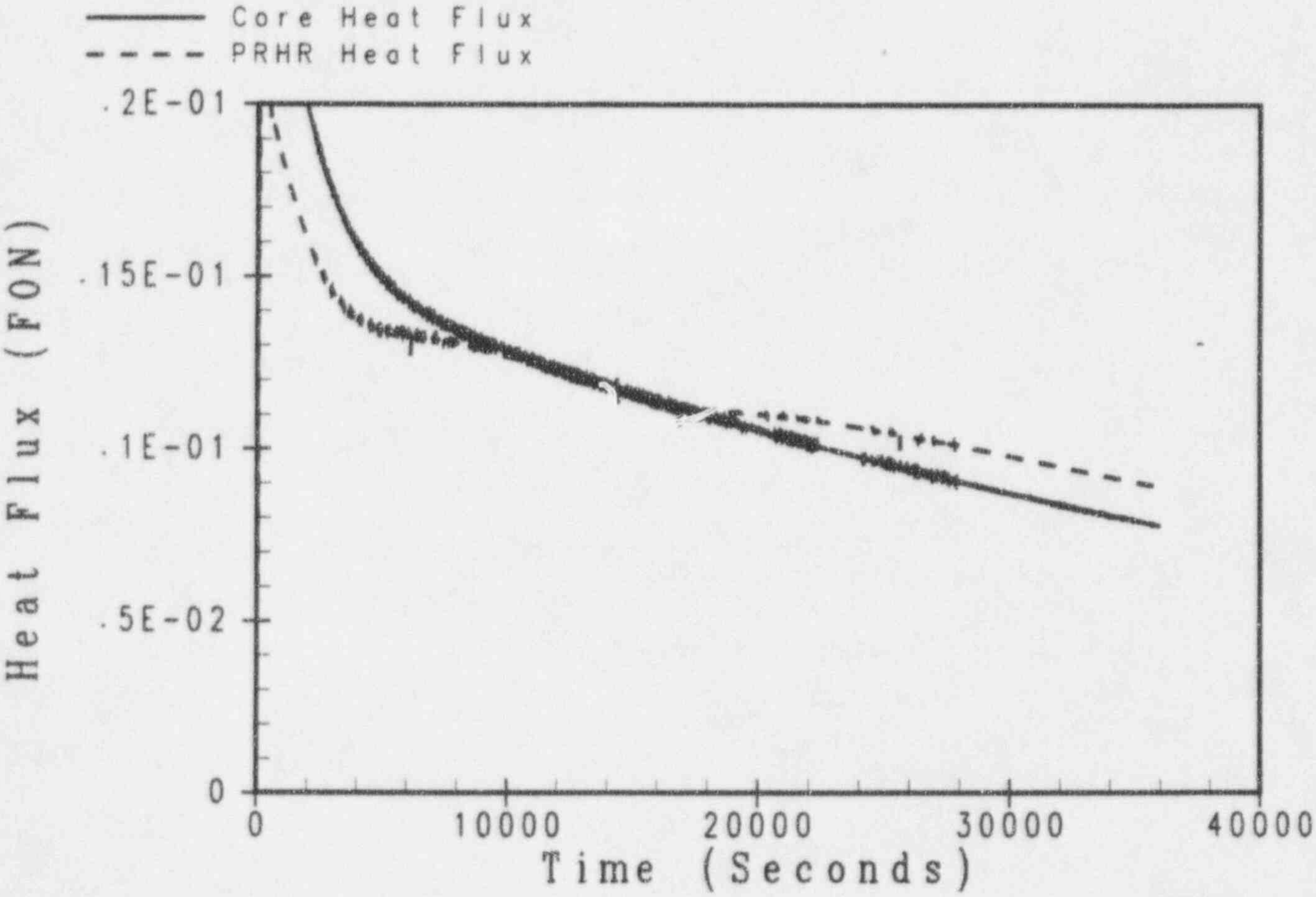


15.5.1-7



CMT Injection Line Flow Rate (Lbm/sec)
CMT Injection Line Flow Rate (Lbm/sec)
100
80
60
40
20
0
0
10000
20000
30000
40000
Time (Seconds)

15.5.1-8



Core Heat Flux
PRHR Heat Flux
Heat Flux (FON)
.2E-01
.15E-01
.1E-01
.5E-02
0
0
10000
20000
30000
40000
Time (Seconds)

Replace

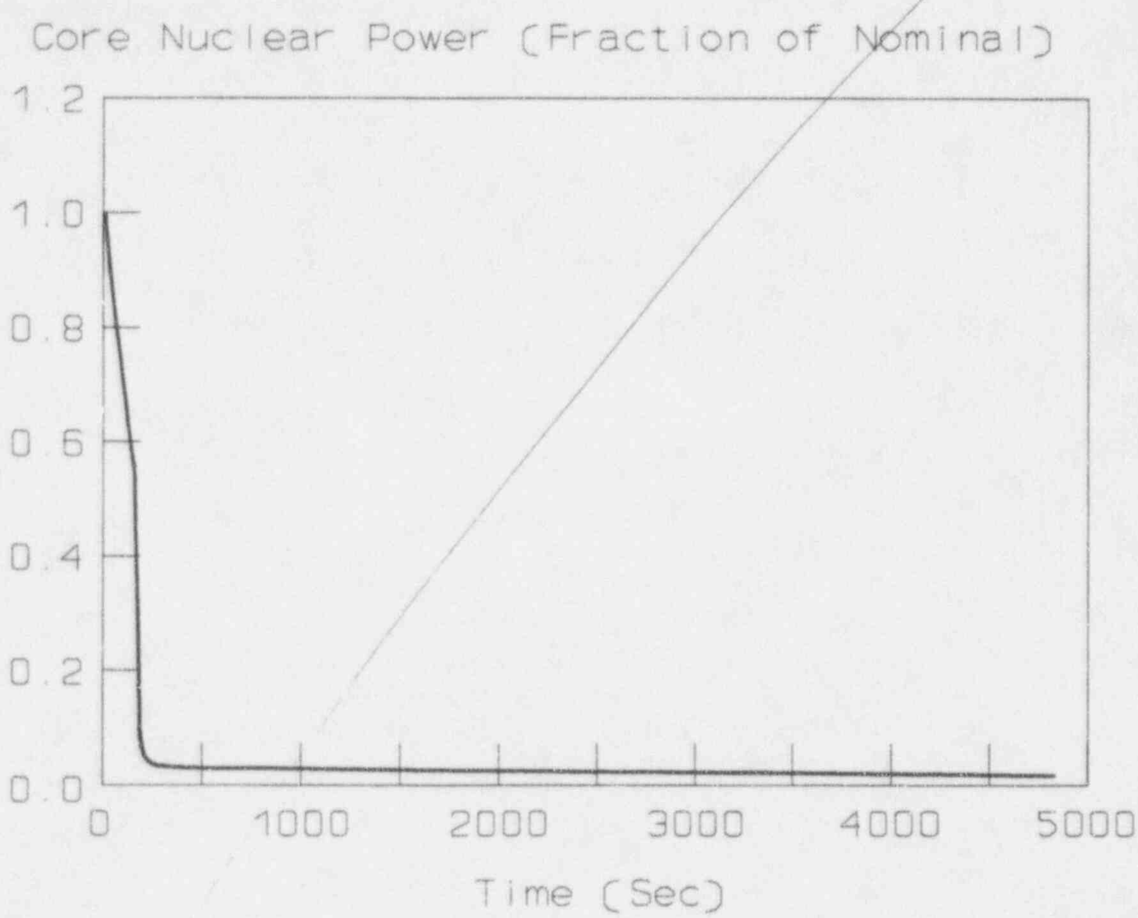
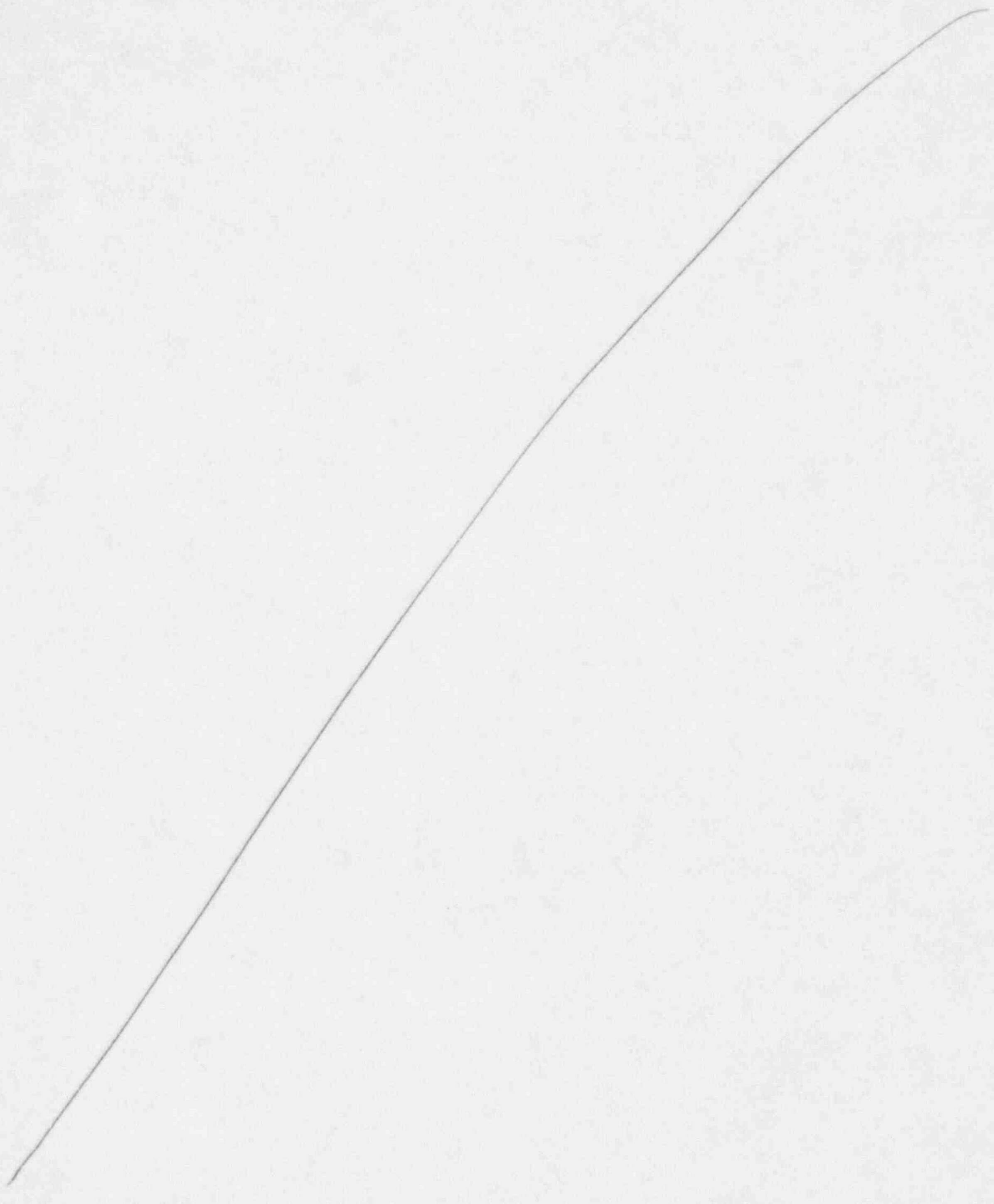
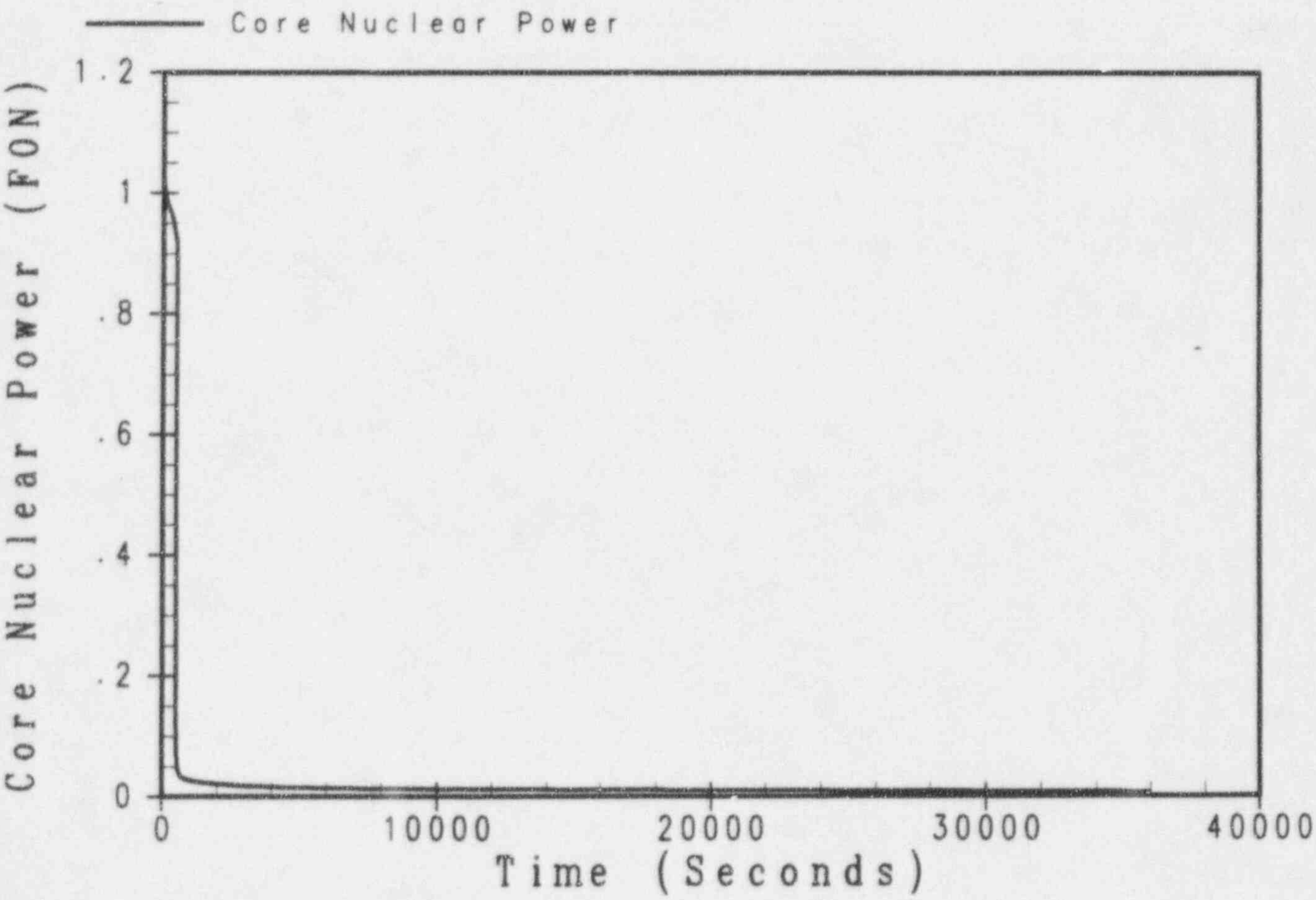


Figure 15.5.2-1

Nuclear Power (Fraction of Nominal) vs. Time for Chemical and Volume Control System Malfunction



15.5.2-1


Core Nuclear Power
Core Nuclear Power (FON)
1.2
1
.8
.6
.4
.2
0
0
10000
20000
30000
40000
Time (Seconds)

Replace

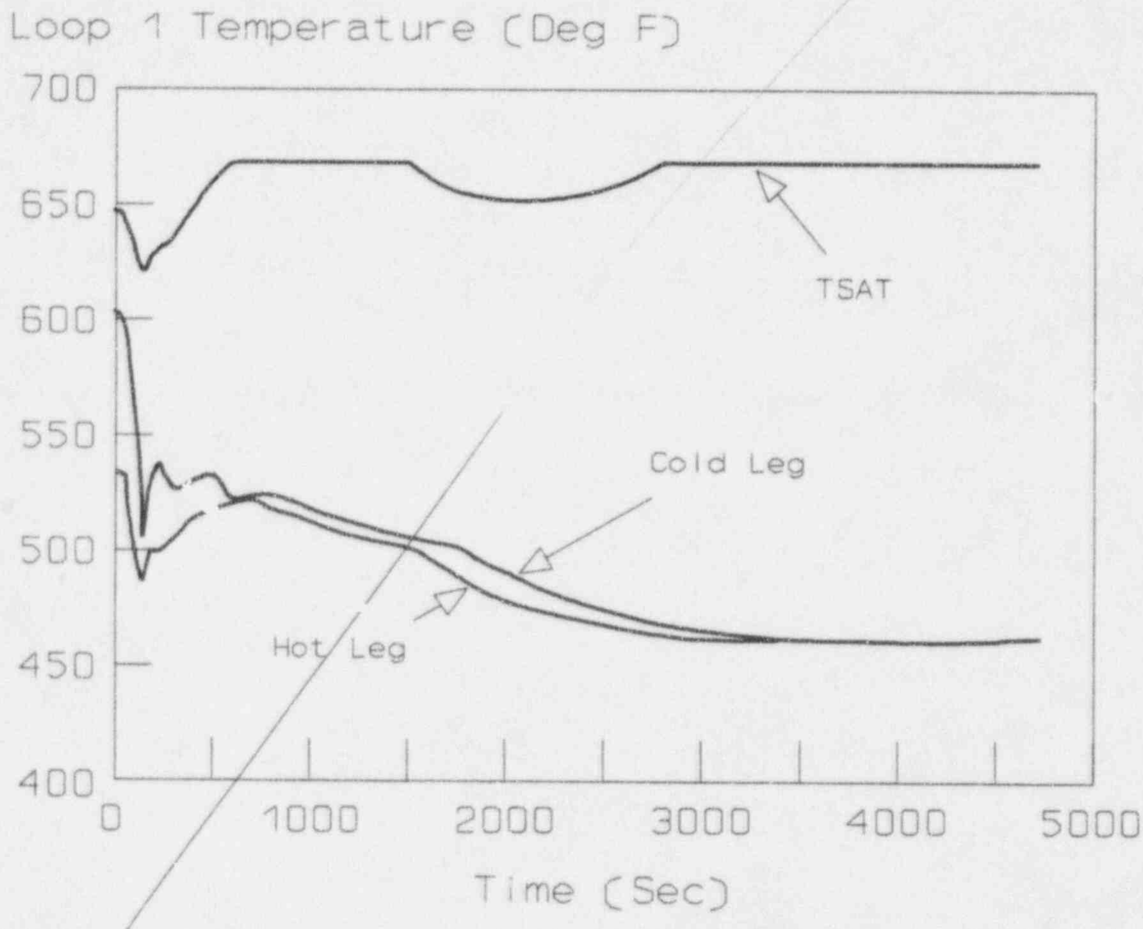
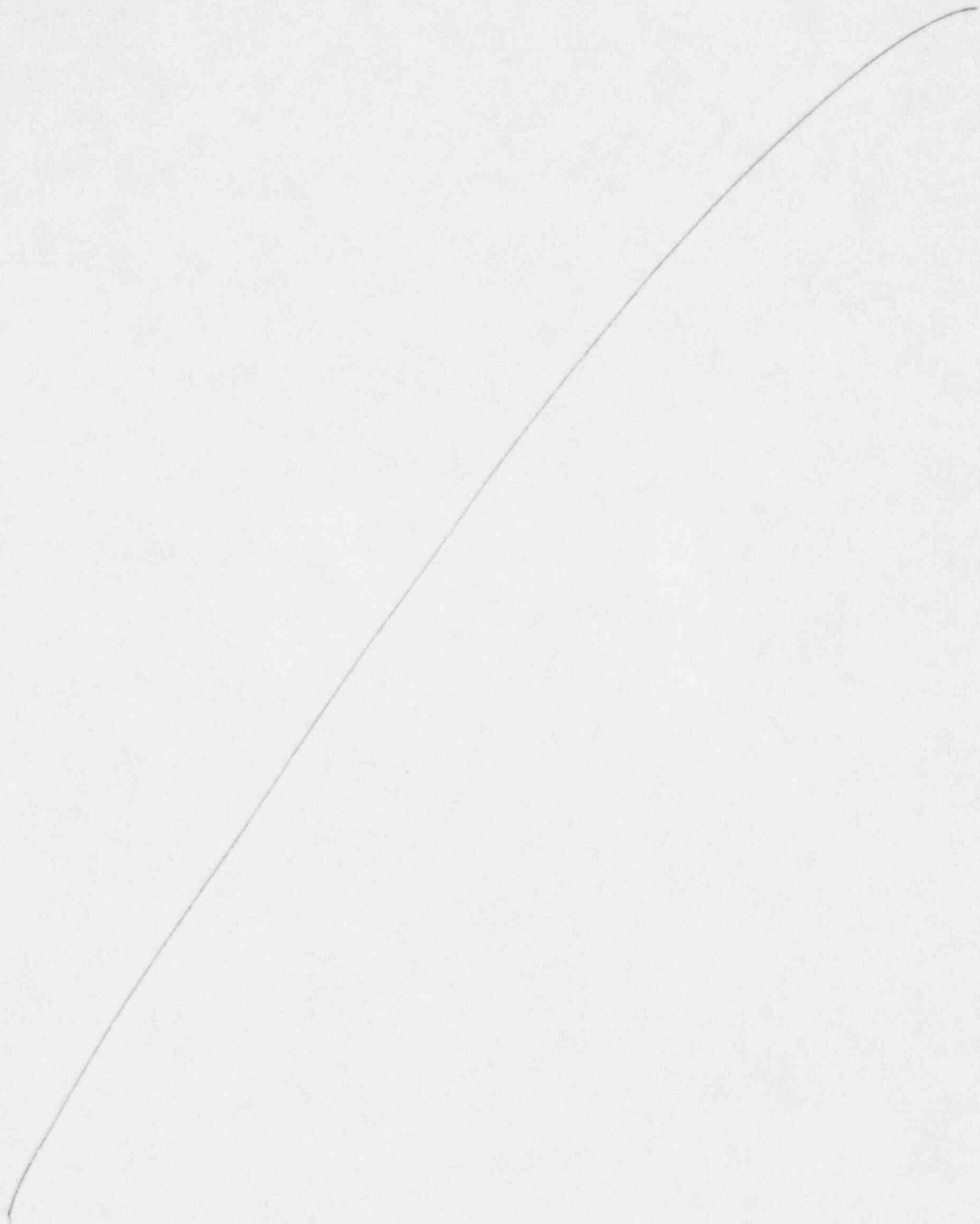
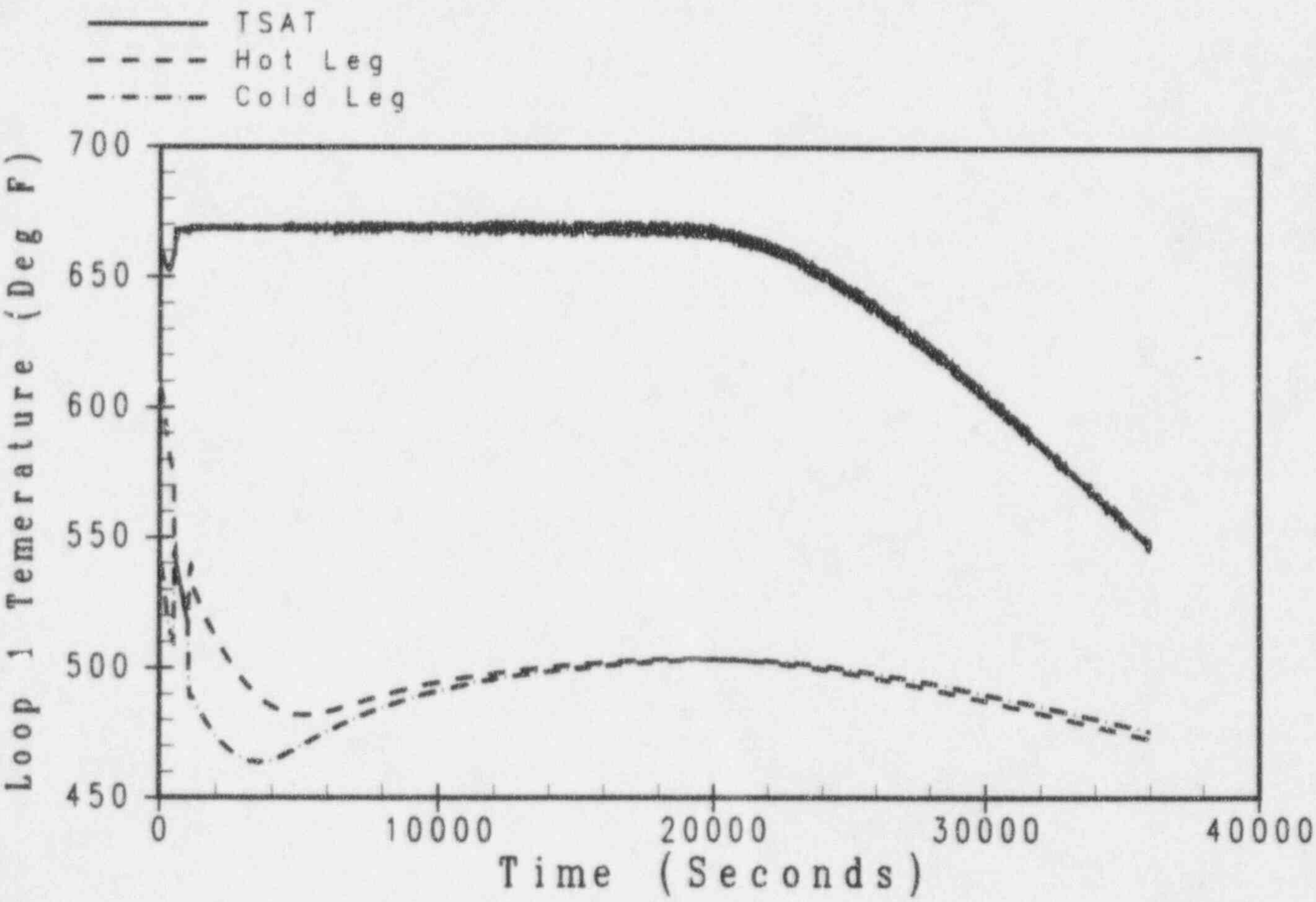


Figure 15.5.2-2

Loop 1 Temperatures (°F) vs. Time for Chemical and Volume Control System Malfunction



15.5.2-2



TSAT
Hot Leg
Cold Leg
Loop 1 Temerature (Deg F)
700
650
600
550
500
450
0
10000
20000
30000
40000
Time (Seconds)

Replace

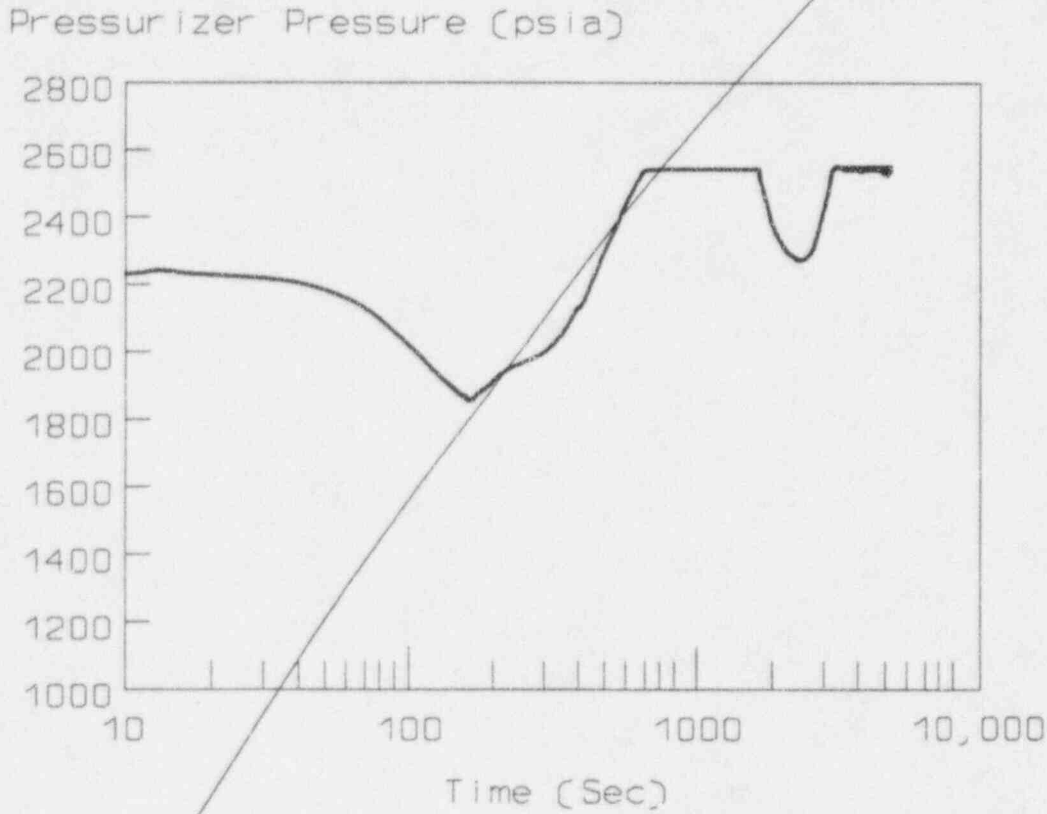
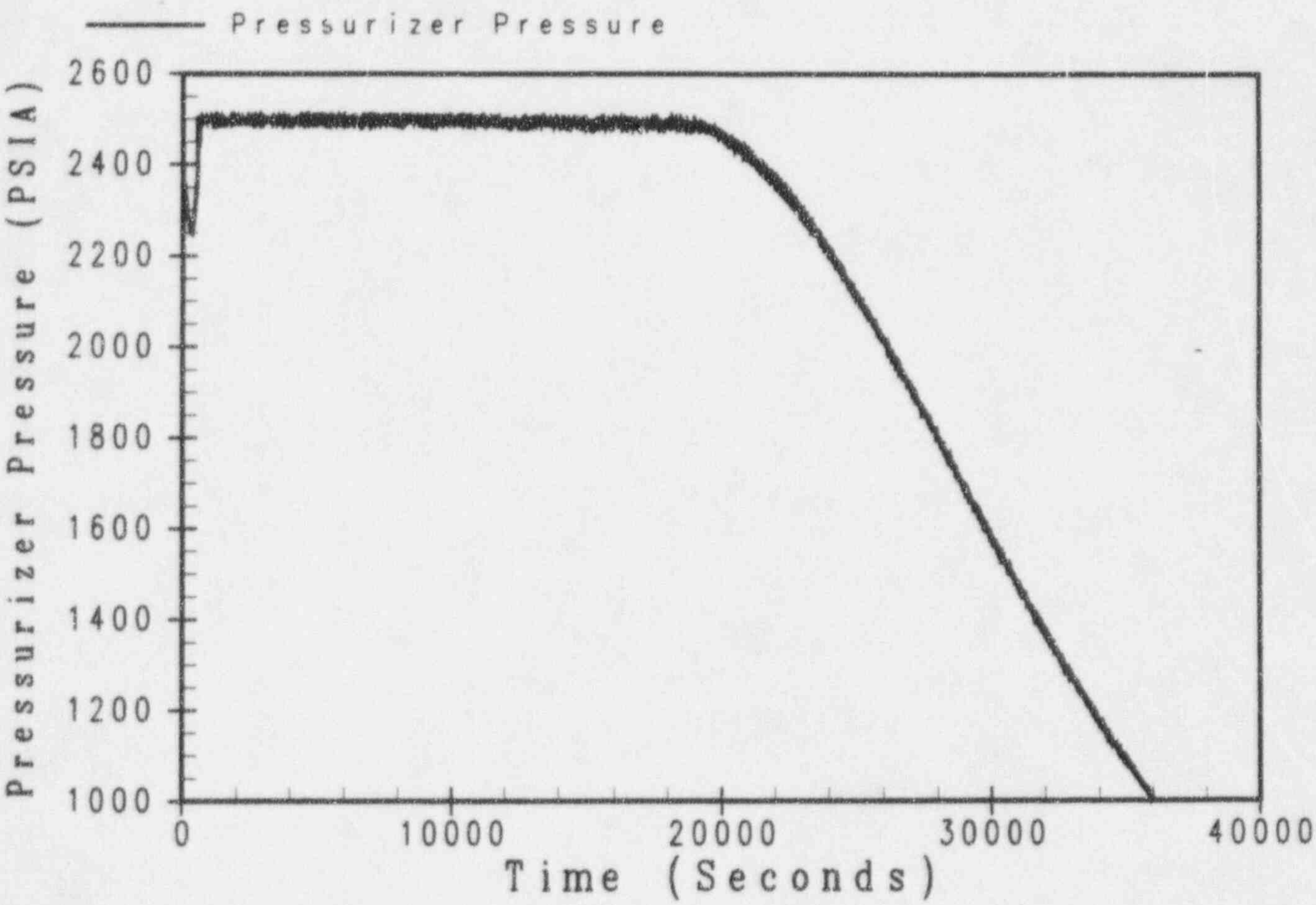


Figure 15.5.2-3

Pressurizer Pressure (psia) vs. Time for Chemical and Volume Control System Malfunction

15.5.2-3


Pressurizer Pressure
Pressurizer Pressure (PSIA)
2600
2400
2200
2000
1800
1600
1400
1200
1000
0
10000
20000
30000
40000
Time (Seconds)

Replace

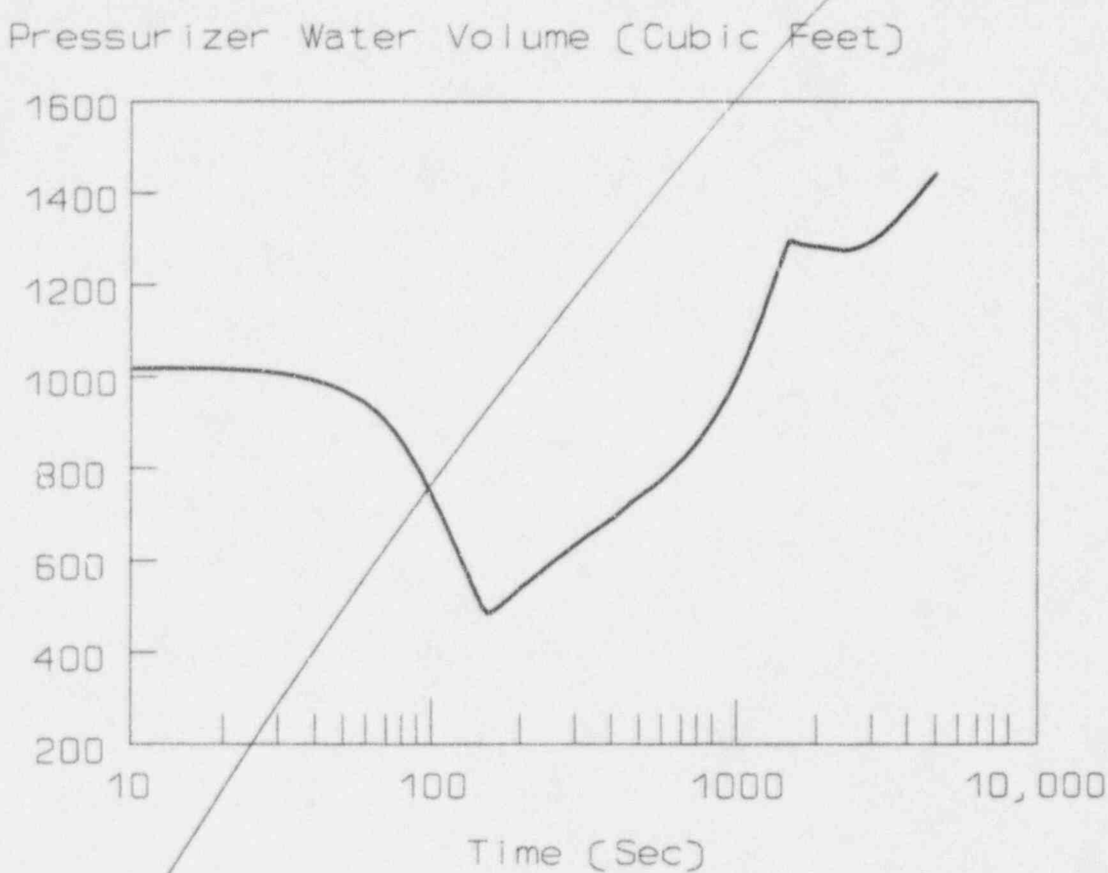
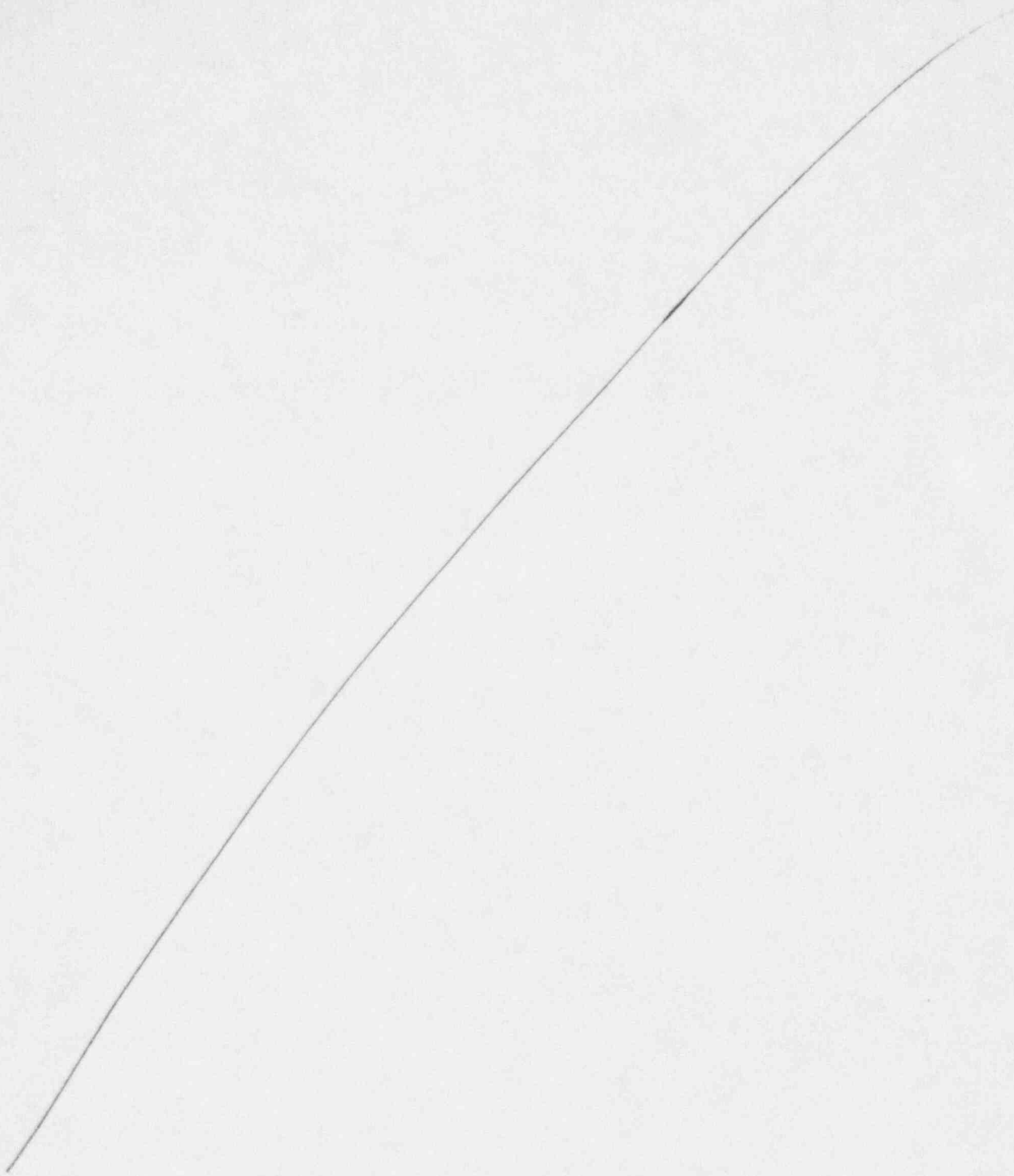
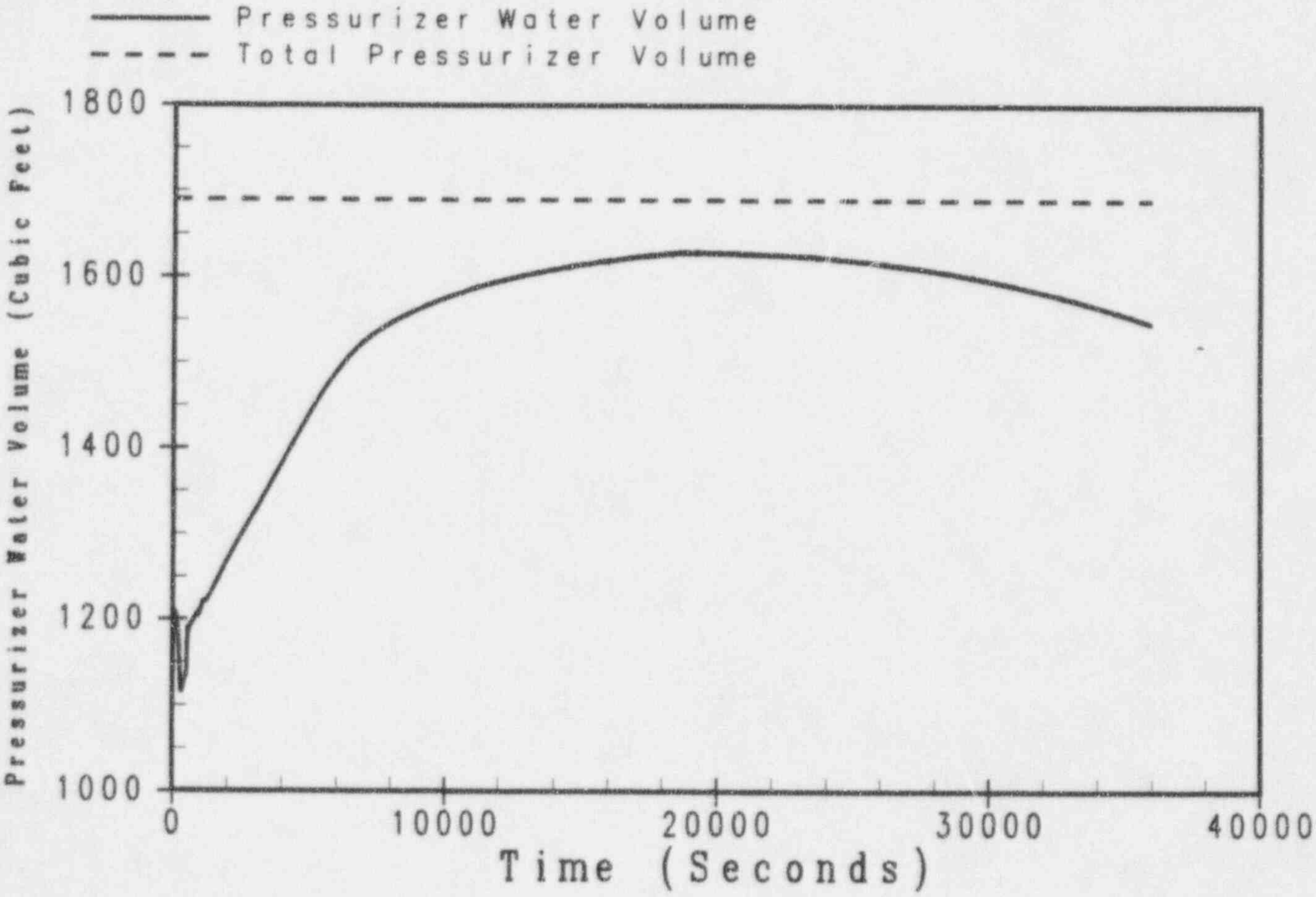


Figure 15.5.2-4

Pressurizer Water Volume (ft^3) vs. Time for Chemical and Volume Control System Malfunction



15.5.2-4


Pressurizer Water Volume
Total Pressurizer Volume
1800
1600
1400
1200
1000
Pressurizer Water Volume (Cubic Feet)
0
10000
20000
30000
40000
Time (Seconds)

Replace

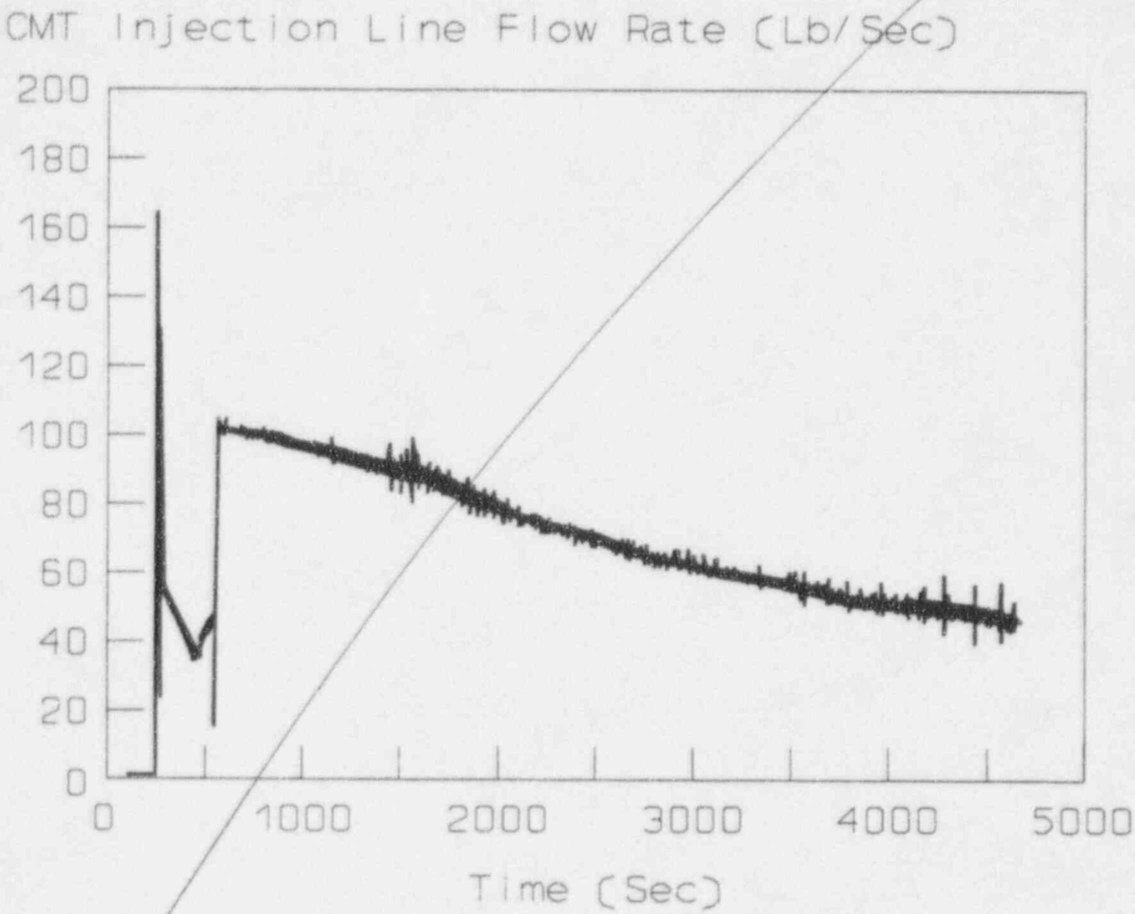
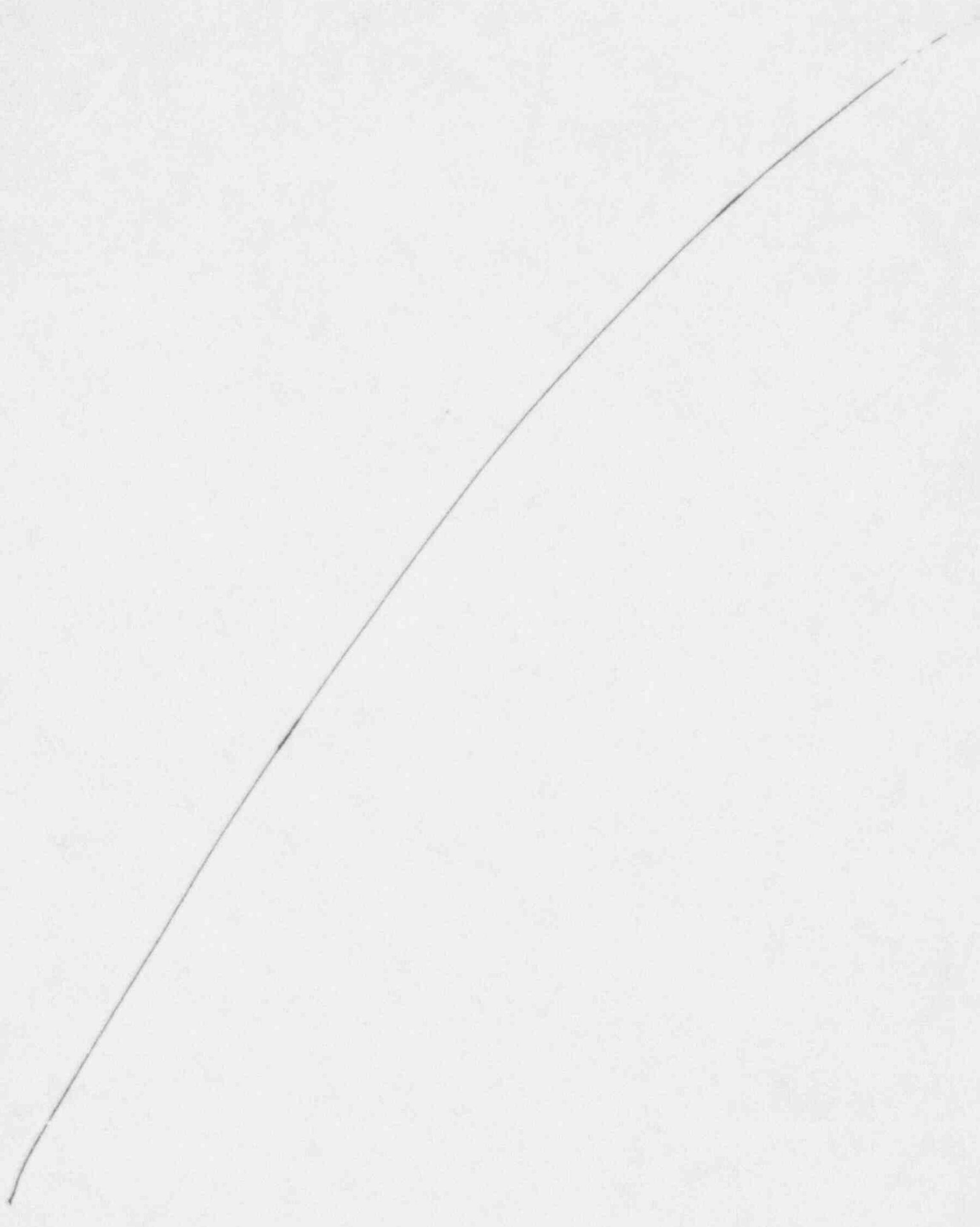
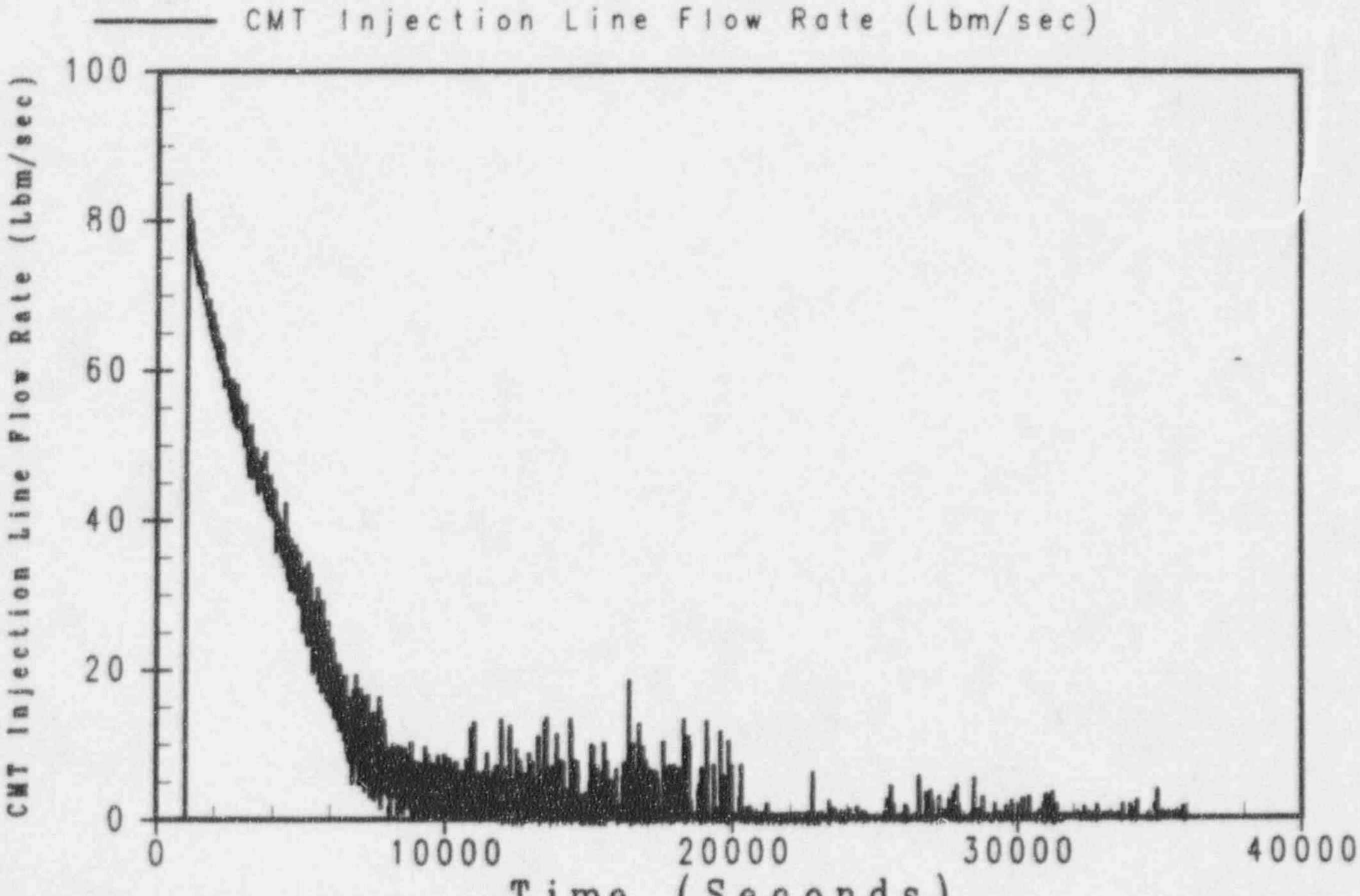


Figure 15.5.2-5

CMT Injection Flow Rate (lbm/sec) vs. Time for Chemical and Volume Control System Malfunction



15.5.2-5


CMT Injection Line Flow Rate (Lbm/sec)
CMT Injection Line Flow Rate (Lbm/sec)
100
80
60
40
20
0
0
10000
20000
30000
40000
Time (Seconds)

Replace

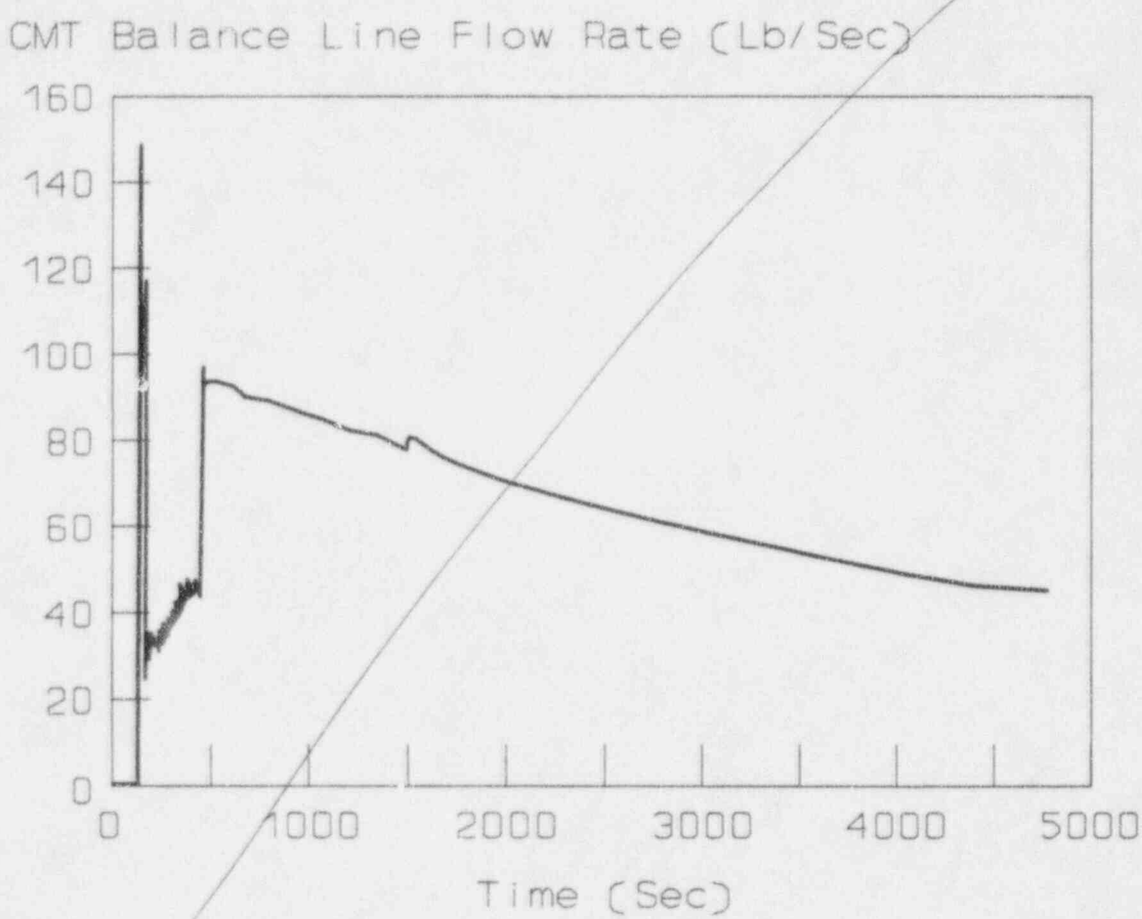
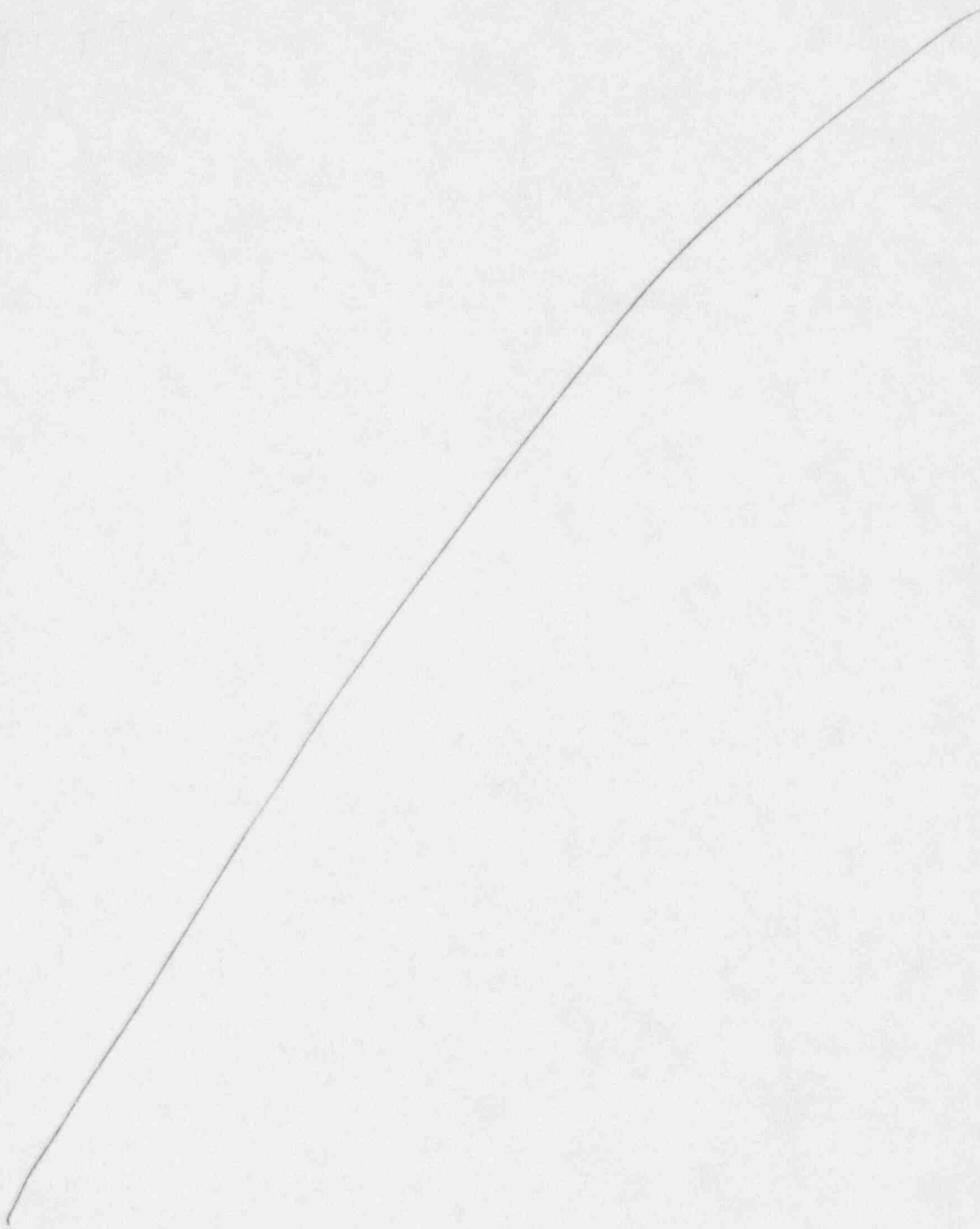
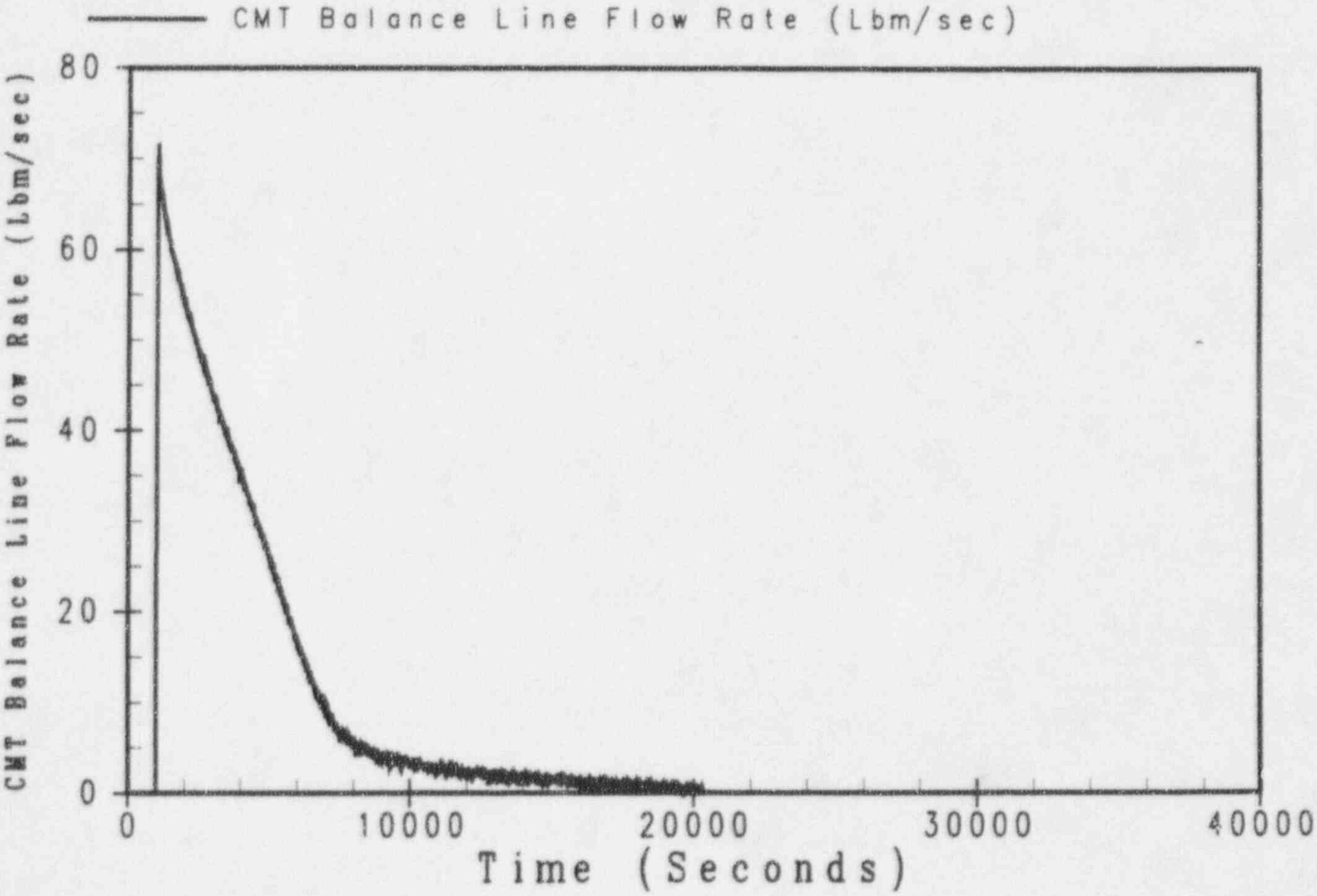


Figure 15.5.2-6

Balance Line Flow Rate (lbm/sec) vs. Time for Chemical and Volume Control System Malfunction



15.5.2-6



CMT Balance Line Flow Rate (Lbm/sec)
CMT Balance Line Flow Rate (Lbm/sec)
80
60
40
20
0
0
10000
20000
30000
40000
Time (Seconds)

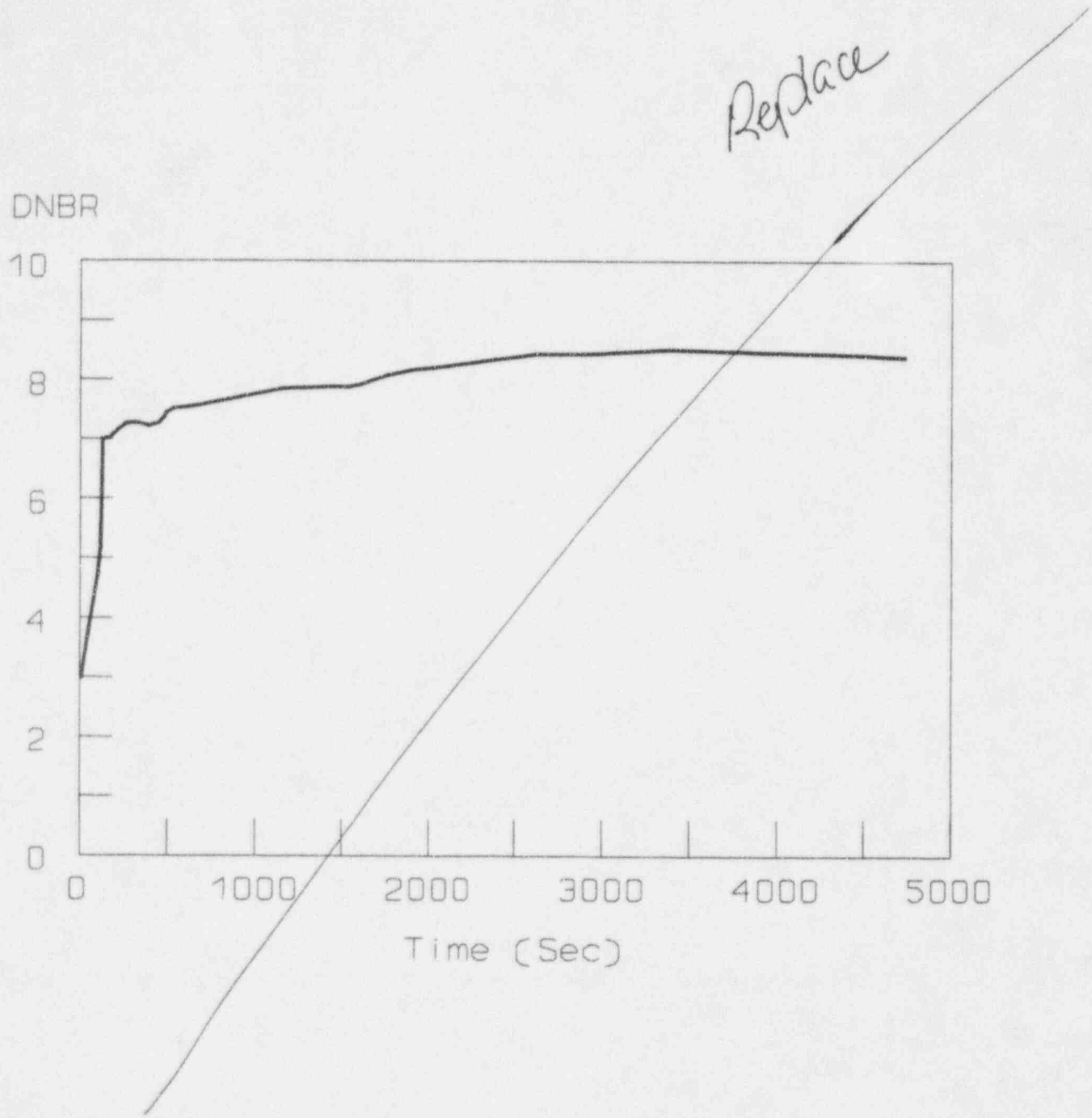
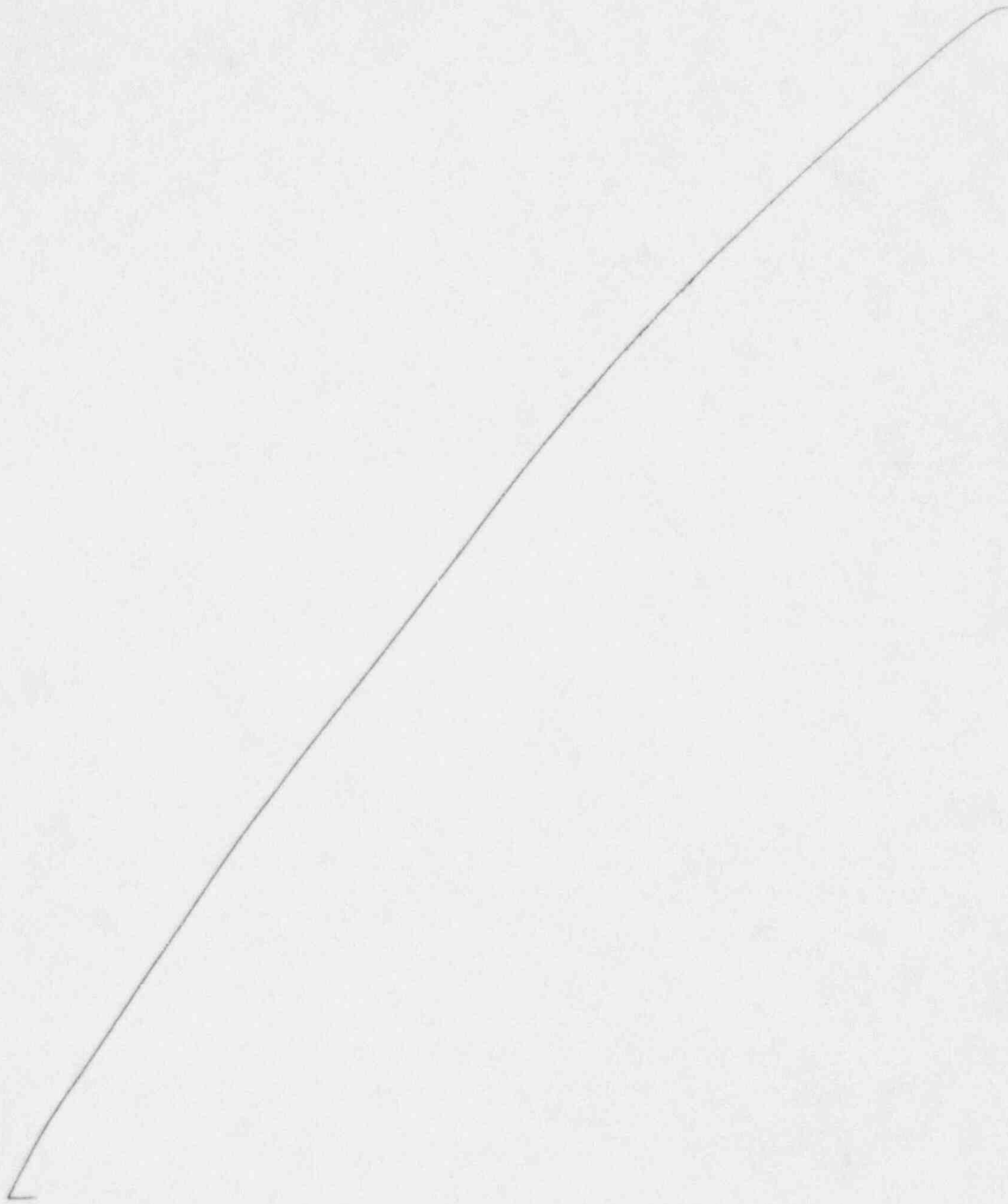
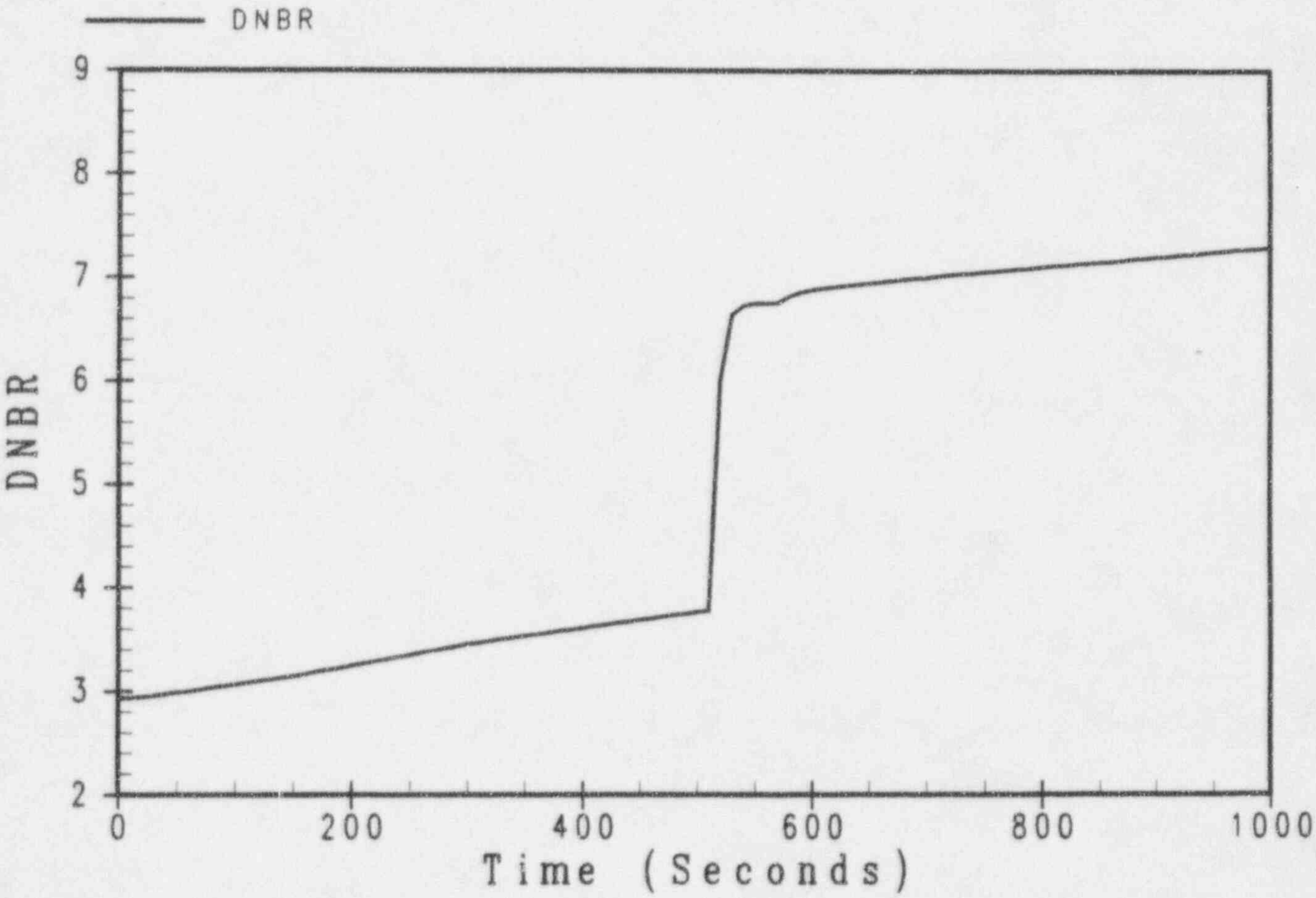


Figure 15.5.2-7

DNB Ratio vs. Time for Chemical and Volume Control System Malfunction



15.5.2-7



Replace

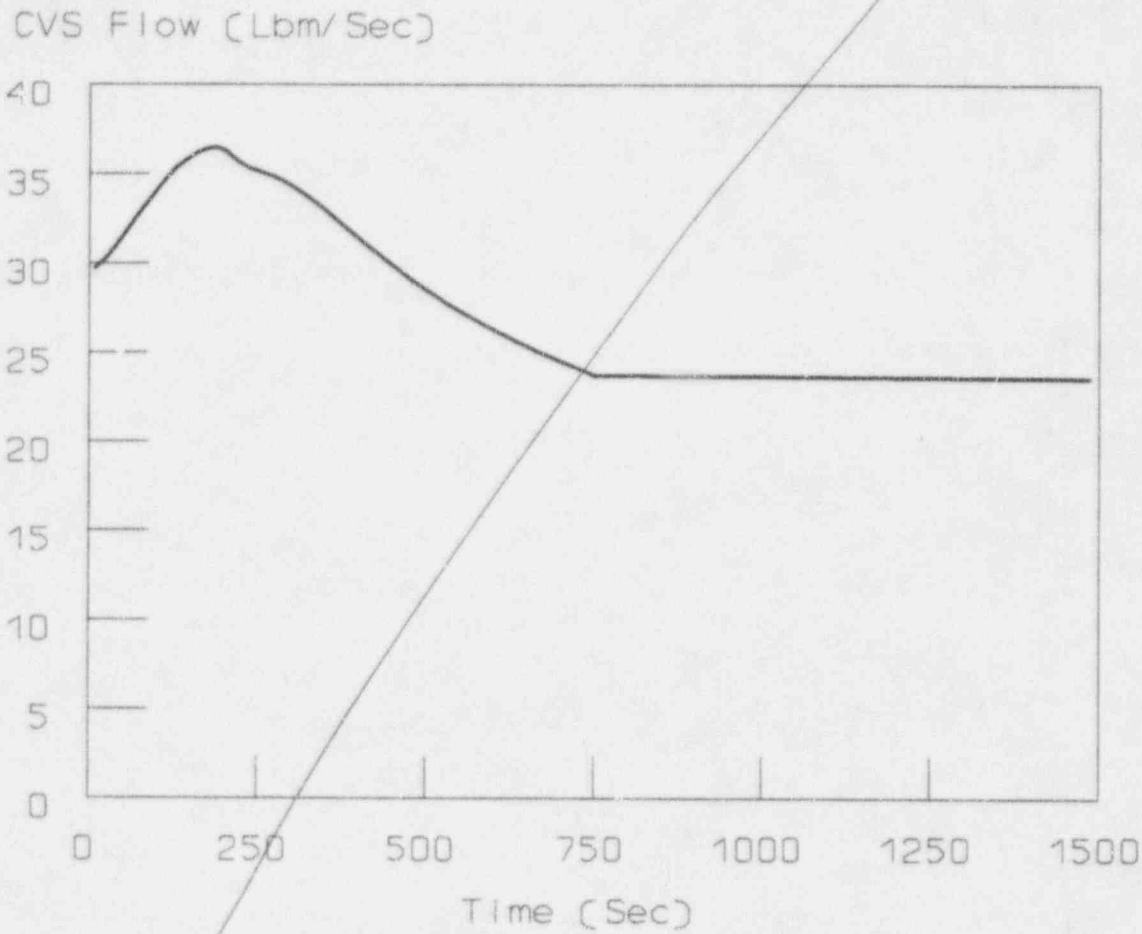
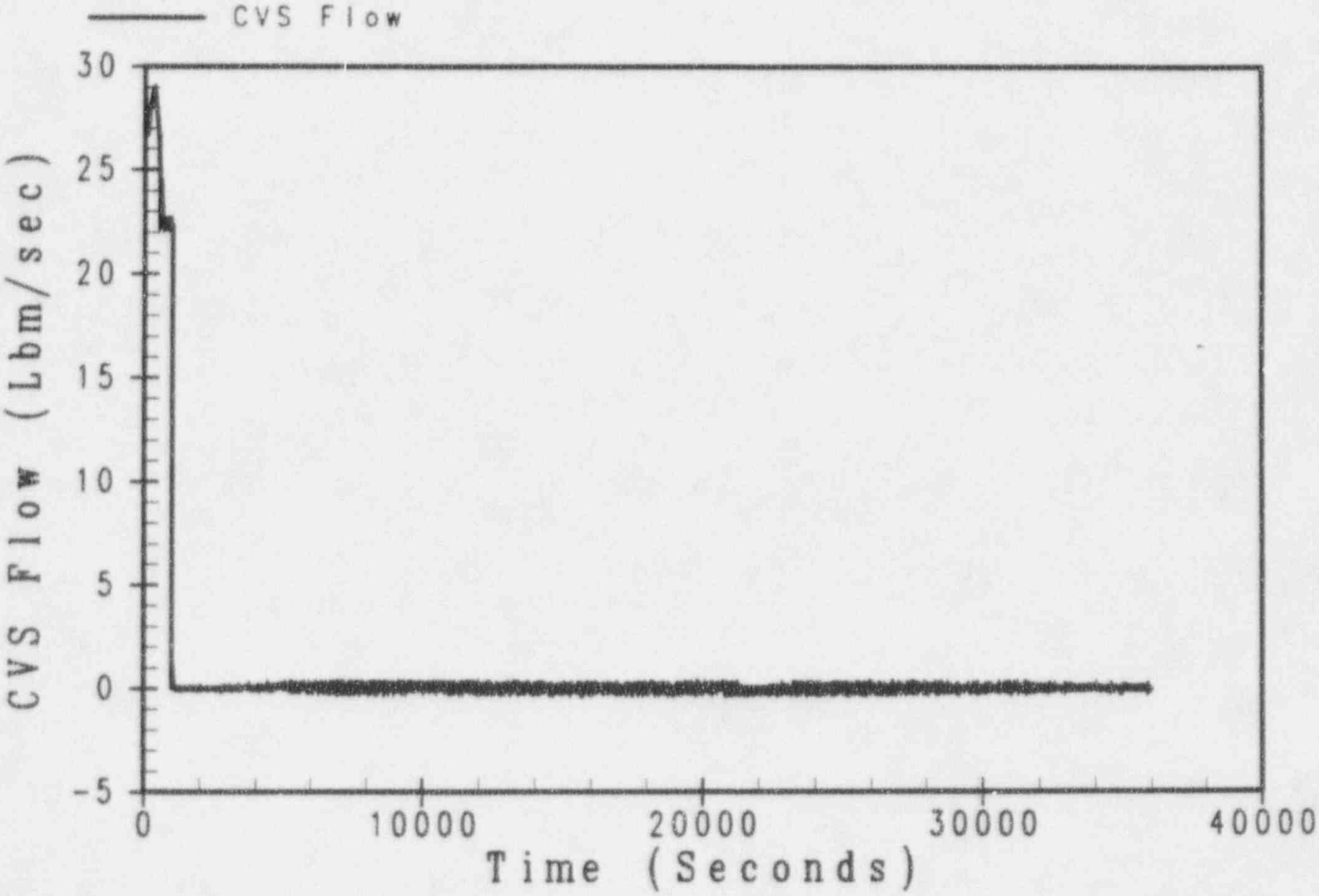


Figure 15.5.2-8

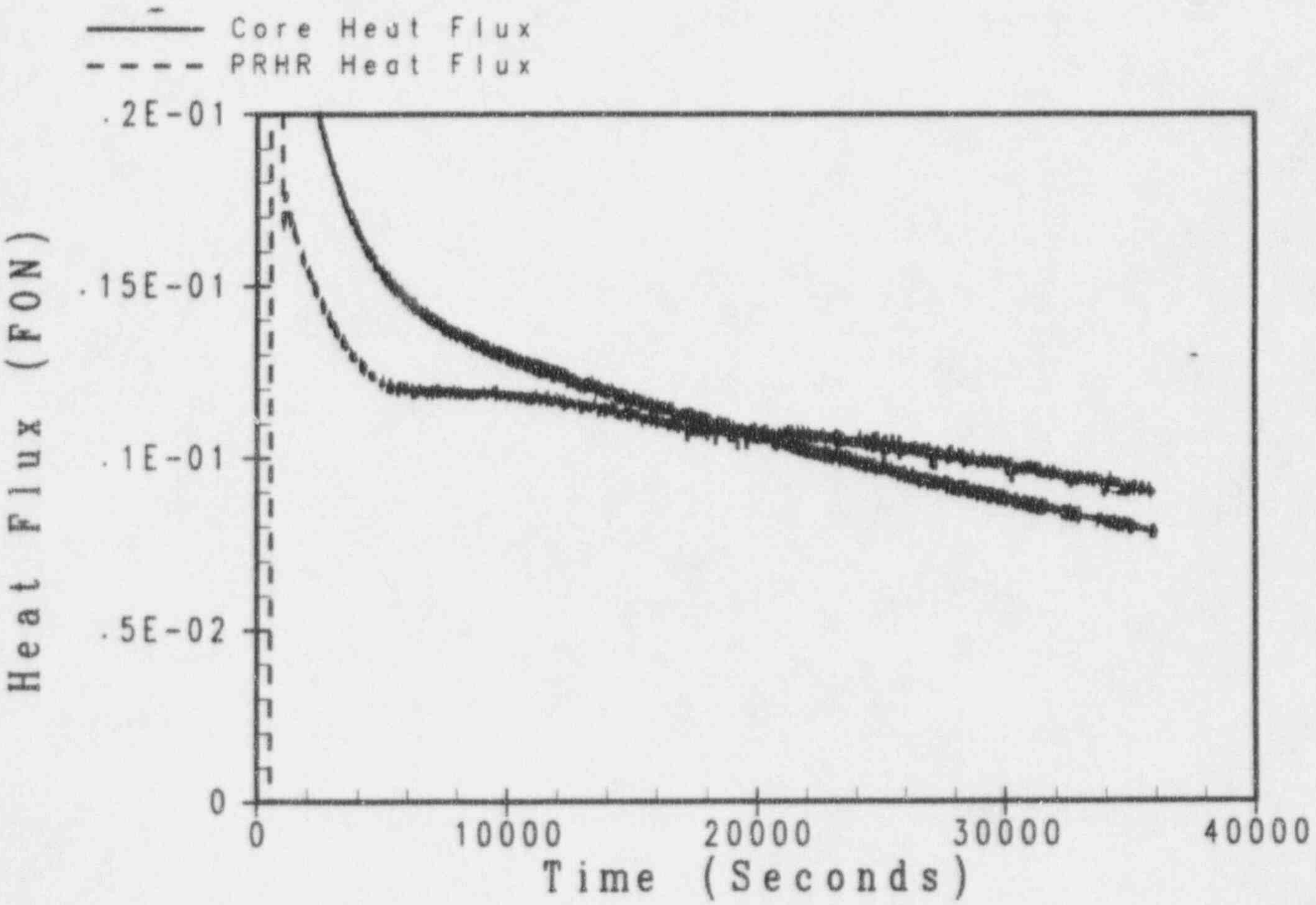
CVS Flow Rate (lbm/sec) vs. Time for Chemical and Volume Control System Malfunction

15.5.2-8



CVS Flow
CVS Flow (Lbm/sec)
Time (Seconds)
30
25
20
15
10
5
0
-5
0
10000
20000
30000
40000

15.5.2-9


Core Heut Flux
PRHR Heat Flux
.2E-01
.15E-01
.1E-01
.5E-02
0
Heat Flux (FON)
0
10000
20000
30000
40000
Time (Seconds)