ATTACHMENT 5

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 187

Revised Pages for Updated Safety Analysis Report (USAR)

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Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

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With Auto Pressure Control (DNB Case)

Without Auto Pressure Control (RCS Overpressure)

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With Offsite Power Available

With a Loss of Offsite Power

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14.0 SAFETY ANALYSIS

No CHANGES

SAFETY ANALYSIS OVERVIEW

In this section the safety aspects of the plant are evaluated to demonstrate that the plant can be operated safely and that radiological consequences from postulated accidents do not exceed the guidelines of 10 CFR 100.

The American Nuclear Society (ANS), Reference 1, has classified plant conditions into four categories in accordance with the 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: Incidents of Moderate Frequency
- Condition III: Infrequent Incidents
- Condition IV: Limiting Faults

A description of each category including design requirements, acceptance criteria, and the applicable design basis transient events is provided below:

Condition I: Normal Operation and Operational Transients

Definition

Condition I occurrences are operations that are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.

Design Requirements

Condition I occurrences shall be accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.

Events

Normal Operation (Base Load and Load Follow)

Acceptance Criteria

- No Clad Damage/Fuel Melting
- Reactor Coolant System Pressure < Design Limits
- Main Steam System Pressure < Design Limits
- Containment Pressure and Temperature < Design Limits

Condition II: Incidents of Moderate Frequency

NO CHANGES

Definition

Condition II occurrences include incidents, any one of which may occur during a calendar year for a particular plant.

Design Requirements

Condition II incidents shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Any release of radioactive materials in effluents to unrestricted areas shall be in conformance with Paragraph 20.1 of 10 CFR Part 20, "Standards for Protection Against Radiation".

By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently. A single Condition II incident shall not cause consequential loss of function of any barrier to the escape of radioactive products. (No fuel rod failure or RCS overpressurization).

Transient Events

- Uncontrolled RCCA Withdrawal From Sub-critical
- Uncontrolled RCCA Withdrawal at Power
- RCCA Misalignment (Dropped/Static)
- Chemical and Volume Control System Malfunction
- Startup of Inactive Reactor Coolant Loop
- Feedwater System Malfunction
- Excessive Load Increase
- Partial Loss of Reactor Coolant Flow
- Loss of External Load
- Loss of Normal Feedwater
- Loss of AC Power to Plant Auxiliaries

Acceptance Criteria

- Reactor Coolant System Pressure < 110% of Design (2750 psia)
- MDNBR > MDNBR Limit
- Fuel Centerline Temp < 4700°F
- Dose Consequences < 10CFR20
- Main Steam System Pressure < 110% of Design (1210 psia)
- Containment Pressure and Temperature < Design Limits

Condition III: Infrequent Incidents

Definition

NO CHANGES?

Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant.

Design Requirements

Condition III incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.

The release of radioactive material due to Condition III incidents may exceed guidelines of 10 CFR Part 20, "Standards for Protection Against Radiation", but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.

A Condition III incident shall not, by itself, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or reactor containment barriers.

Transient Events

- Small LOCA
- Small Steam Line Break
- Complete Loss of Reactor Coolant Flow
- Single RCCA Withdrawal at Power
- Fuel Assembly Misloading
- Volume Control Tank Rupture

Acceptance Criteria

Most incidents use Condition II criteria, which are more limiting than the Condition III criteria. If these are not satisfied, the following criteria ~~are~~ may be applied:

- MDNBR < MDNBR Limit - Small Fraction of Fuel Rods (< 5%)
- Dose Consequences < 10% of 10CFR100
- RCS Pressure < 2900 psia
- Containment Pressure and Temperature < Design Limits

Condition IV: Limiting Faults

Definition

Condition IV occurrences are faults that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV faults are the most drastic, which must be designed against, and thus represent the limiting design cases.

Design Requirements

Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10 CFR 100, "Reactor Site Criteria". A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the Reactor Coolant System and the Reactor Containment System.

Events

- Large LOCA
- Steam Generator Tube Rupture
- Main Steam Line Break (MSLB)
- Locked Rotor
- RCCA Ejection
- Fuel Handling

Acceptance Criteria

- Dose Consequences < 10CFR100
- RCS Pressure < 2900 psia (emergency)
- < 4000 psia (faulted)
- Containment Pressure and Temperature < Design Limits

The following events have event specific limits that are more limiting than the Condition IV criteria:

- Main Steam Line Break
  MDNBR > MDNBR Limit (MSLB)
- Locked Rotor
  Peak Clad Temperature < 2700°F
  Percentage of Fuel Rods Experiencing DNB < ~~40%~~ 50%
- RCCA Ejection
  ~~Peak Clad Temperature < 2700°F~~
  Average Fuel Enthalpy < 200 cal/g
  Fuel Melt < Innermost 10% of the fuel pellet at the hot spot

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, Reactor Protection System and Engineered Safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle.

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

This section is divided into three subsections, dealing with various behavior categories:

- Core and Coolant Boundary Protection Analysis, Section 14.1
  The abnormalities presented in Section 14.1 have no off-site radiation consequences.

- Standby Safety Features Analysis, Section 14.2
  The accidents presented in Section 14.2 are more severe than those discussed in 14.1 and may cause release of radioactive material to the environment.

- Rupture of a Reactor Coolant Pipe, Section 14.3
  The accident presented in Section 14.3, the rupture of a reactor coolant pipe, is the worst-case accident analyzed and is the primary basis for the design of engineered safety features. It is shown that the consequences of even this accident are within the guidelines of 10 CFR100.

SAFETY ANALYSIS ASSUMPTIONS

Parameters and assumptions that are common to the safety analyses are described below to avoid repetition in subsequent sections.

Operating Parameters REPLACE WITH USAR Insert 14.0-1

~~For accident evaluation, the initial conditions are obtained by adding maximum steady-state errors to rated values. The following initial conditions and steady-state errors are considered:~~

Average Reactor Coolant Power	1772 ~~1650~~ MWt	± 2% of 1772 ~~1650~~ MWt for calorimetric error
~~Core Inlet~~ Temperature	556.3 - 573.0°F ~~535.5°F~~	± 6 ~~4~~°F for deadband and measurement error
Pressurizer ~~Primary~~ Pressure	2250 psia	± 50.1 [redacted] psi for steady-state fluctuations and measurement error

~~Initial values for power, primary pressure, and core temperature are selected to minimize the initial departure from nucleate boiling ratio (DNBR).~~

~~The initial active core flow rate is conservatively set to account for increased core bypass flow due to thimble plug removal and increased steam generator tube plugging. Unless otherwise stated in the Method of Analysis section for a particular accident the RCS and Core flow rates are set as follows:~~

REPLACE WITH USAR Insert 14.0-2

~~Reactor Coolant System Flow — 83,500 gal/min each loop~~
~~Core Inlet Flow — 63.88E6 lb-m/hr~~
~~Core Bypass Flow — 7.0%~~
~~Effective Core Flow — 59.41E6 lb-m/hr~~

~~**Hot Channel Factors**~~

~~Unless otherwise stated in the sections describing specific accidents, the hot channel factors used are:~~

~~F_q^n (Nuclear Heat Flux Hot Channel Factor) = 2.35~~

~~$F_{\Delta H}^N$ (Nuclear Enthalpy Rise Hot Channel Factor) = 1.70~~

REPLACE WITH
USAR Insert 14.0-3

The movable in-core instrumentation system is employed to verify that actual hot channel factors are, in fact, no higher than the limiting values of the Technical Specifications. These limits on hot channel factors are designed to conservatively bound the assumptions used in the accident analyses.

Add USAR Insert 14.0-4

Reactor Protection System

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanism to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping 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. The time delay and setpoint assumed for each tripping function used in the analysis are as follows:

Reactor Trip Function	Setpoint	Time Delay (sec)
Power Range Negative Rate	N/M*	N/A
Power Range Positive Rate	N/M*	N/A
Power Range Low Setpoint	35%	~~0.5~~ 0.65
Power Range High Setpoint	118%	~~0.5~~ 0.65
Overpower Delta T	~~120%~~ Variable - see Figure 14.0.2	6.0
Overtemperature Delta T	~~119%~~ Variable - see Figure 14.0.2	6.0
RCS Low Flow	86.5 ~~87~~% of loop flow	~~0.6~~ 0.75
High Pressurizer Level	~~100% of level span~~ N/M*	~~1.5~~ N/A
Low Pressurizer Pressure	~~1735 psig~~ 1850 psia	1.0
High Pressurizer Pressure	~~2410 psig~~ 2425 psia	1.0
Low-Low Steam Generator Level	0.0% of level span	1.5
RXCP Undervoltage	N/M*	N/A
RXCP Underfrequency	N/M*	N/A
Turbine Trip	N/M*	N/A

N/M* - not explicitly modeled in safety analysis

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The difference between the limiting trip setpoint assumed for the analysis and the actual trip setpoint represents a conservative allowance for instrumentation channel and setpoint errors. Results of surveillance tests demonstrate that actual instrument errors are equal to or less than the assumed values.

The instrumentation drift and calorimetric errors used in establishing the maximum power range high neutron flux ~~overpower~~ setpoint are presented in Table 14.0.1.

Add USAR Insert 14.0-5

Trip is defined for analytical purposes as the insertion of all full-length rod control cluster assemblies (RCCAs) except the most reactive RCCA, which is assumed to remain in the fully withdrawn position. This is to provide shutdown margin capability against the remote possibility of a stuck RCCA condition existing at a time when shutdown is required.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the control rods and the variation in rod worth as a function of rod position. Control rod positions after trip have been determined experimentally as a function of time using an actual prototype assembly under simulated flow conditions. The resulting rod positions were combined with rod worths to define the negative reactivity insertion as a function of time, as shown in Figure 14.0.1.

In summary, reactor protection is designed to prevent cladding damage in all transients and abnormalities. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. Coincidence of two-out-of-three (or two-out-of-four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria. The reactor protection systems are designed in accordance with Reference 2.

Add USAR Insert 14.0-6

Steam Generator Safety Valves

Unless otherwise stated in the section describing a specific accident, the following Steam Generator safety valve settings with 15% blowdown and rated safety valve flow capacities were assumed:

Valve	Nominal Safety Valve Setting (psig)	Safety Analysis Pressure Setpoint (psig) (Pressure at S/G)
1	1074	1150
2	1090	1167
3	1105	1167
4	1120	1183
5	1127	1193

Calorimetric Error Instrumentation Accuracy

NOT TA SCOPE

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis. The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generator and steam pressure. High accuracy plant instrumentation is provided for these measurements with accuracy tolerances more restrictive than that which would be required to only control the feedwater flow. Each feedwater flow venturi is laboratory calibrated and certified. The expected accuracies are tabulated below with their effect on the overall power measurement.

Variable	Accuracy	Equivalent Percent of Rated Power
Feedwater temperature	± 2°F (½%)	
Feedwater pressure (Small correction on enthalpy)	± 5%	0.3% = Total effect
Steam pressure (Small correction on enthalpy)	± 2%	
Feedwater flow	± 1.25%	1.25%
		1.55% = Total error

NOT TA SCOPE

Note that the errors have been added directly; statistical combination of errors indicate better accuracy. Corrections for moisture carry-over in the steam (0.25% design basis) can be made which would yield a lower measured power level. This effect can be conservatively neglected.

The secondary calorimetric is verified to be conservative as compared to the feedwater bypass line at the beginning of each cycle. The KNPP feedwater bypass line (FBL) is a full flow normal feedwater bypass loop designed to accurately measure total feedwater flow at KNPP. The FBL contains a flow section, which includes a flow straightener and a laboratory calibrated flow nozzle. The flow section is accurate to 0.25%.

The total uncertainty of this feedwater measurement is a function of the uncertainty of the FBL calibration, and the venturi repeatability; the uncertainty of total feedwater flow, as it contributes to the uncertainty of overall reactor power, is significantly less than the required 1.25%.

~~SAFETY ANALYSIS AND~~ CORE RELOAD METHODOLOGY

By letter dated March 27, 1987, WPS submitted for NRC review a topical report entitled "Reload Safety Evaluation Methods for Application to Kewaunee". Additional information was submitted to the NRC on February 12 and March 7, 1988. The report includes methods for analyzing plant accidents, transients, and setpoints excluding the loss-of-coolant accident (LOCA) and the fuel mishandling accident. The NRC Safety Evaluation Report provided in Reference 3 reviewed the description and performance of the DYNODE-P (Version 5.4), the RETRAN-02, the VIPRE-01 and the TOODEE-2 codes employed in the analyses. In addition, the analyses, procedures and the results of specific calculations and reload evaluations were examined. The NRC found that the topical report was acceptable for referencing in KNPP licensing submittals.

The core reload safety evaluation methodology is described in Reference 10.

REFERENCES - SECTION 14.0

1. ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants"

2. IEEE 279, "Standard for Nuclear Plant Protection Systems", August 1968

3. NRC Safety Evaluation Report, JG Giitter (NRC) to DC Hintz (NRC), Letter No. K-88-67 dated April 11, 1988

4. Friedland, A.J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), April 1989.

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

6. Huegel, D.S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), April 1999.

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

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

9. Sung, Y.X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary), October 1999.

10. Bordelon, F.M., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Proprietary), July 1985.

USAR Insert 14.0.-1

For most accidents that are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-11397 (Reference 4). This procedure is known as the "Revised Thermal Design Procedure," and is discussed more fully in Section 3.2.

For accidents in which the Revised Thermal Design Procedure is not employed, the initial conditions are obtained by applying the maximum steady-state errors to the rated values. The following rated values and conservative steady-state errors were assumed in the analyses:

USAR Insert 14.0.-2

Tables 14.0-2 and 14.0-3 summarize initial conditions and computer codes used in non-LOCA accident analyses, and identify which DNB limited transients were analyzed using the Revised Thermal Design Procedure (RTDP).

USAR Insert 14.0.-3

Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. The radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_Q) characterize the power distribution. The peaking factor limits are provided in the Technical Specifications.

For transients that may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 14.0.2. All transients that may be DNB 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 DNB calculations is discussed in Section 3.2. Also, the radial and axial power distributions are input to the VIPRE code as described in Section 3.2.

For transients that may be overpower limited, the total peaking factor (F_Q) is of importance. These transients are assumed to begin with plant conditions, including power distributions, that 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 decrease in the boron concentration of the reactor coolant system, lasting many minutes), the fuel rod thermal evaluations are performed as discussed in Section 3.2. For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled rod cluster control assembly (RCCA) bank withdrawal from subcritical and the RCCA ejection incidents that result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup, and rod power, a typical value at beginning-of-life for high power rods is approximately five seconds.

USAR Insert 14.0.-4

Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Section 3.2.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of coolant from cracks or ruptures in the Reactor Coolant System, do not depend on reactivity feedback effects. 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 represent unrealistic situations.

USAR Insert 14.0.-5

Reference is made above to Overpower and Overtemperature Delta T (ΔT) variable reactor trip setpoints illustrated in Figure 14.0.2. This Figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and power distribution, as described in Section 3.2, as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "Protection Lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.34 for the thimble cell and 1.34 for the typical cell). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the applicable DNBR line at any point does not traverse the area enclosed with the maximum protection lines. The area of permissible operation (power, pressure, and temperature) is bounded by the following combination of reactor trips: high neutron flux (fixed setpoint), high pressurizer pressure (fixed setpoint), low pressurizer pressure (fixed setpoint), overpower ΔT (variable setpoint) and overtemperature ΔT (variable setpoint). The DNBR limit value, which was used as the DNBR limit for all accidents analyzed with the Revised Thermal Design Procedure (see Table 14.0-2), is conservative compared to the actual design DNBR value required to meet the DNB design basis as discussed in Section 3.2.

USAR Insert 14.0.-6

Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, such as those used in the analysis of reactor coolant system pipe ruptures (Section 14.3), are summarized in the respective accident analyses sections. Table 14.0-2 provides a list of codes used for each transient analysis.

FACTRAN (Ref. 5)

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

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

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

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

RETRAN (Ref. 6)

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot and cold leg piping, reactor coolant pumps, steam generators (tube and shell sides), steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The reactor protection system (RPS) simulated in the code includes reactor trips on high neutron flux, overtemperature and overpower ΔT (OTΔT/OPΔT), low reactor coolant system (RCS) flow, high and low pressurizer pressure, high pressurizer level, and lo-lo steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, may also be modeled. RETRAN approximates the transient value of departure from nucleate boiling ratio (DNBR) based on input from the core thermal safety limits.

LOFTRAN (Ref. 7)

Transient response studies of a pressurized water reactor (PWR) to specified perturbations in process parameters use the LOFTRAN computer code. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the reactor protection system (RPS) which includes reactor trips on high neutron flux, OTΔT, OPΔT, high and low pressurizer pressure, low reactor coolant system (RCS) flow, lo-lo steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled. LOFTRAN also approximates the transient value of departure from nucleate boiling ratio (DNBR) based on the input from the core thermal safety limits.

TWINKLE (Ref. 8)

TWINKLE is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8,000 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. The code provides various output, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

VIPRE (Ref. 9)

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and departure from nucleate boiling ratio (DNBR) distributions along flow channels within a reactor core. Additional discussion of the VIPRE code is provided in Section 3.2.

USAR Insert 14.0.-7

The core reload methodology is described in Reference 10.

NMC SCOPE

TABLE 14.0-1 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS NUCLEAR OVERPOWER TRIP CHANNEL		
	Set Point and Error Allowances: (% of rated power)	Estimated Instrument Errors: (% of rated power)
Nominal set point	109	-
Calorimetric error	2	1.55
Axial power distribution effects on total ion chamber current	5	3
Instrumentation channel drift and set point reproducibility	2	1.0
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction	118	-

Table 14.0-2

(New Table to be Added)

Summary of Initial Conditions and Computer Codes Used for Non-LOCA Accident Analyses

Transient/Event	Computer Codes Used	DNB Correlation	Revised Thermal Design Procedure	Initial Core Power (% 1772 MWt)	Vessel Coolant Flow (gpm)	Vessel Avg. Coolant Temp. (°F)	Pressurizer Pressure (psia)
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	W-3[1] WRB-1[2]	No	0	79,922	547.0	2160
Uncontrolled RCCA Withdrawal at Power	RETRAN	WRB-1	Yes (DNB) No (Pressure)	100 (DNB) 60 (DNB) 10 (DNB) 8 (Pressure)	186,000 (DNB) 178,000 (Pressure)	573.0 (100%) 562.6 (60%) 549.6 (10%) 555.6 (8%)	2250 (DNB) 2200 (Pressure)[3]
RCCA Misalignment (Dropped Rod)	LOFTRAN VIPRE	WRB-1	Yes	100	186,000	573.0	2250
Chemical and Volume Control System Malfunction	N/A	N/A	N/A	N/A	N/A	579.0 (Power) 554.3 (Startup) 140.0 (Refueling)	2250 (Power) 2250 (Startup) 14.7 (Refueling)
Startup of an Inactive Reactor Coolant Loop	Event precluded by the Technical Specifications						
Reduction in Feedwater Temperature	Event bounded by the Excessive Load Increase Incident						
Increase in Feedwater Flow	RETRAN VIPRE	WRB-1 (HFP) W-3 (HZP)	Yes (HFP) No (HZP)	100 (HFP) 0 (HZP)	186,000 (HFP) 178,000 (HZP)	573.0 (HFP) 547.0 (HZP)	2250
Excessive Load Increase	N/A	WRB-1	Yes	100	186,000	573.0	2250
Loss of Reactor Coolant Flow	RETRAN VIPRE	WRB-1	Yes	100	186,000	573.0	2250
Locked Rotor	RETRAN VIPRE FACTRAN	WRB-1	Yes (DNB) No (Hot Spot)	100 (DNB) 102 (Hot Spot)	186,000 (DNB) 178,000 (Hot Spot)	573.0 (DNB) 579.0 (Hot Spot)	2250 (DNB) 2300 (Hot Spot)[3]
Loss of External Electrical Load	RETRAN	WRB-1	Yes (DNB) No (Pressure)	100 (DNB) 102 (Pressure)	186,000 (DNB) 178,000 (Pressure)	573.0 (DNB) 579.0 (Pressure)	2250 (DNB) 2200 (Pressure)[3]
Loss of Normal Feedwater	RETRAN	N/A	No	102	178,000	579.0	2300[3]
Anticipated Transients Without Scram	*NMC Scope*						
Loss of AC Power to the Plant Auxiliaries	RETRAN	N/A	No	102	178,000	579.0	2300[3]
Steam Generator Tube Rupture	Not TA Scope						
Steam Line Break	RETRAN VIPRE	W-3	No	0	178,000	547.0	2250
RCCA Ejection	TWINKLE FACTRAN	N/A	No	102 (HFP) 0 (HZP)	178,000 (HFP) 79,922 (HZP)	579.0 (HFP) 547.0 (HZP)	2200[3]

(1) Below the first mixing vane grid. (2) Above the first mixing vane grid. (3) An additional 0.1 psi uncertainty has been evaluated.

New Table to be Added

Table 14.0-3

Nominal Values of Pertinent Parameters for Non-LOCA Accident Analyses

Parameter	Maximum T-avg with RTDP	Maximum T-avg non-RTDP	Minimum T-avg with RTDP	Minimum T-avg non-RTDP
Thermal Output of NSSS (MWt)	1780	1780	1780	1780
Maximum Core Power (MWt)	1772	1772	1772	1772
Vessel Average Coolant Temperature (°F)(1)	573.0	573.0±6.0	556.3	556.3±6.0
Pressurizer Pressure (psia)	2250.0	2250.0±50.1	2250.0	2250.0±50.1
Reactor Coolant Loop Flow (GPM)	93,000	89,000	93,000	89,000
Steam Generator Tube Plugging	0 to 10%	0 to 10%	0 to 10%	0 to 10%
Steam Generator Outlet Pressure (psia)	771 (0% SGTP) 747 (10% SGTP)	771 (0% SGTP) 747 (10% SGTP)	656 (0% SGTP) 634 (10% SGTP)	656 (0% SGTP) 634 (10% SGTP)
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	437.1	437.1	437.1	437.1
Average Core Heat Flux (Btu/hr-ft^2)	206,585	206,585	206,585	206,585

(1) The accident analyses support a full power T-avg range from 556.3°F to 573.0°F.

SCRAM REACTIVITY INSERTION RATE

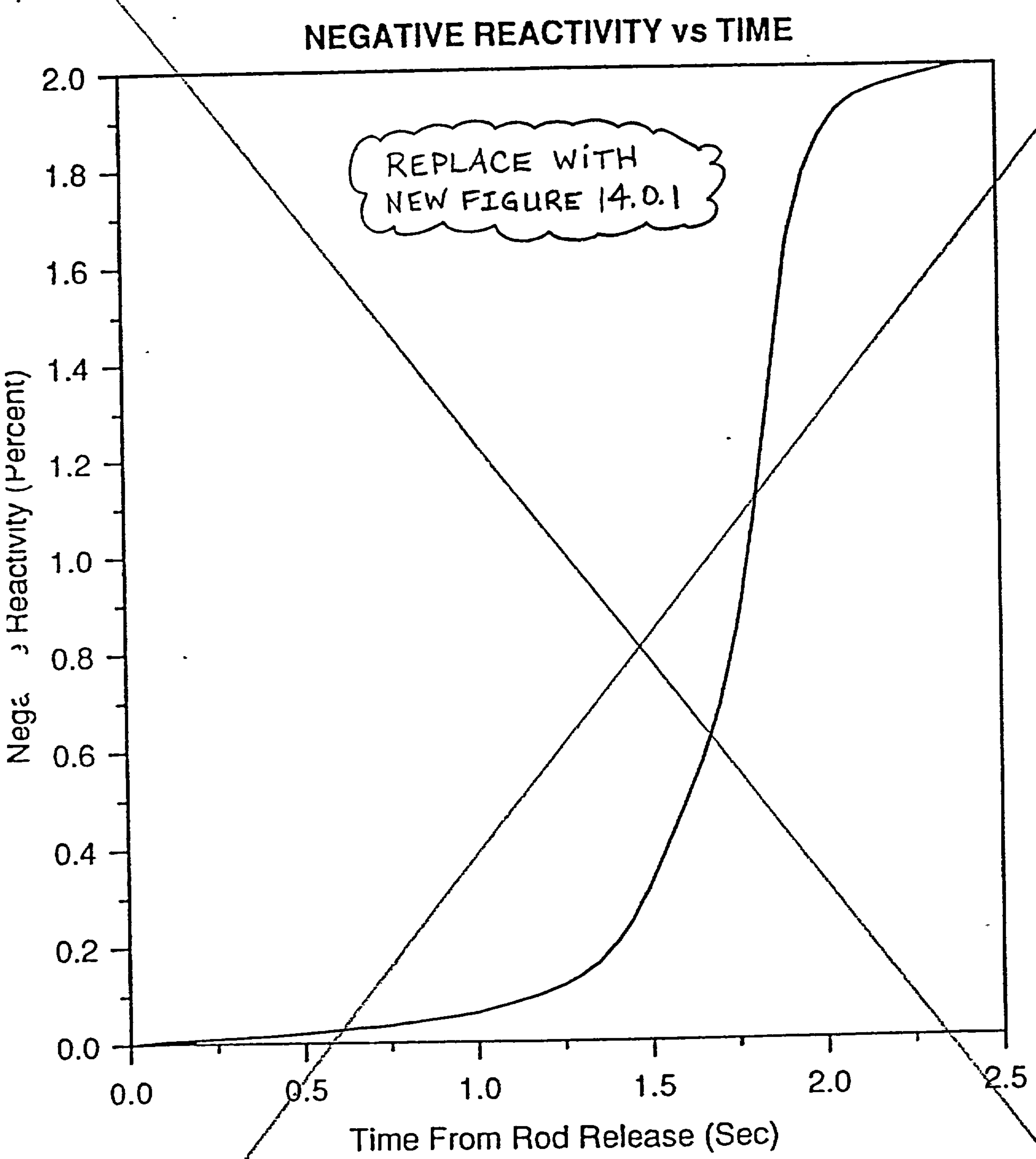

FIGURE 14.0.1

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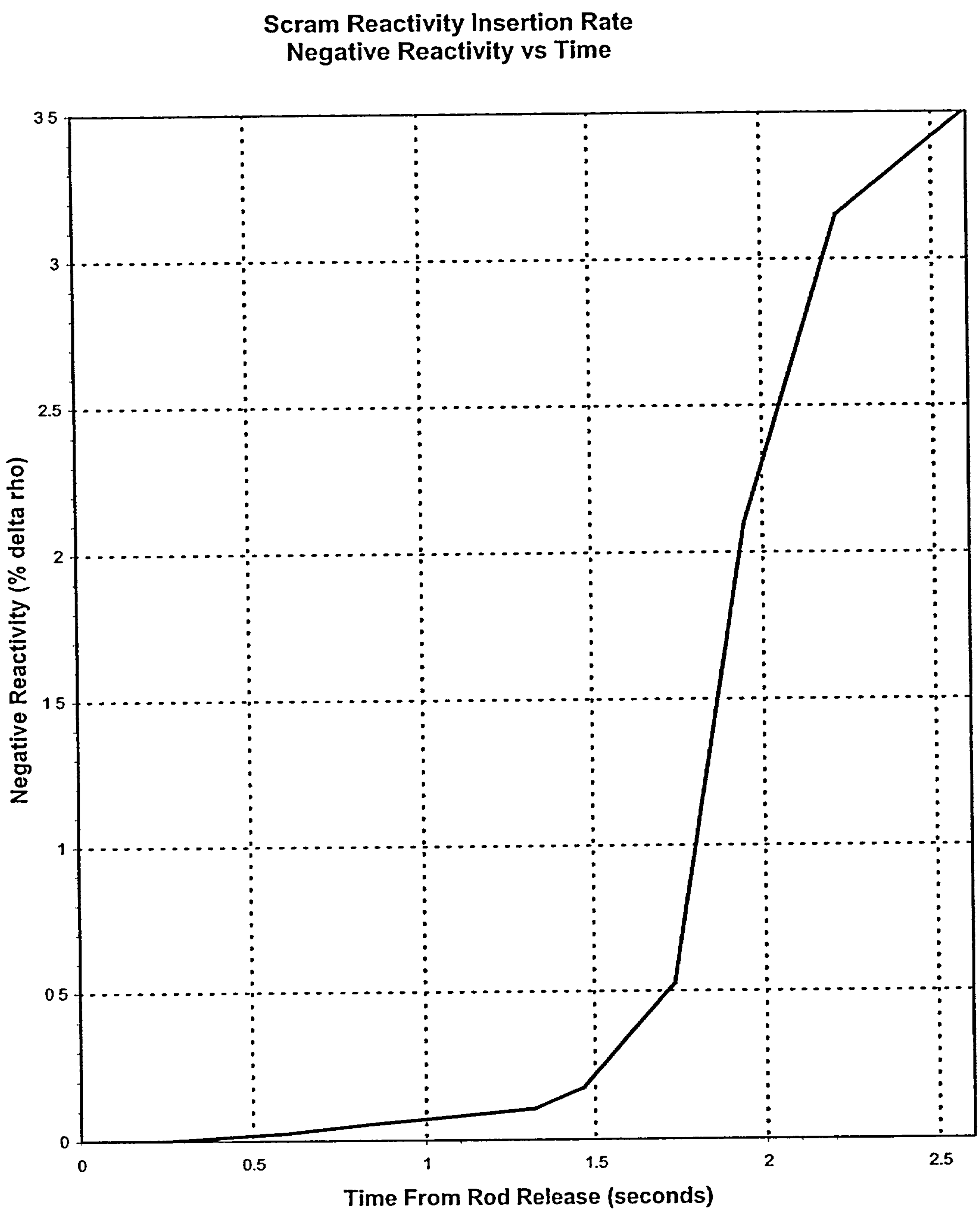
Scram Reactivity Insertion Rate
Negative Reactivity vs Time
Negative Reactivity (% delta rho)
3.5
3
2.5
2
1.5
1
0.5
0
0
0.5
1
1.5
2
2.5
Time From Rod Release (seconds)

Figure 14.0.1

New Figure to be Added

Illustration of Overtemperature and Overpower ΔT Protection

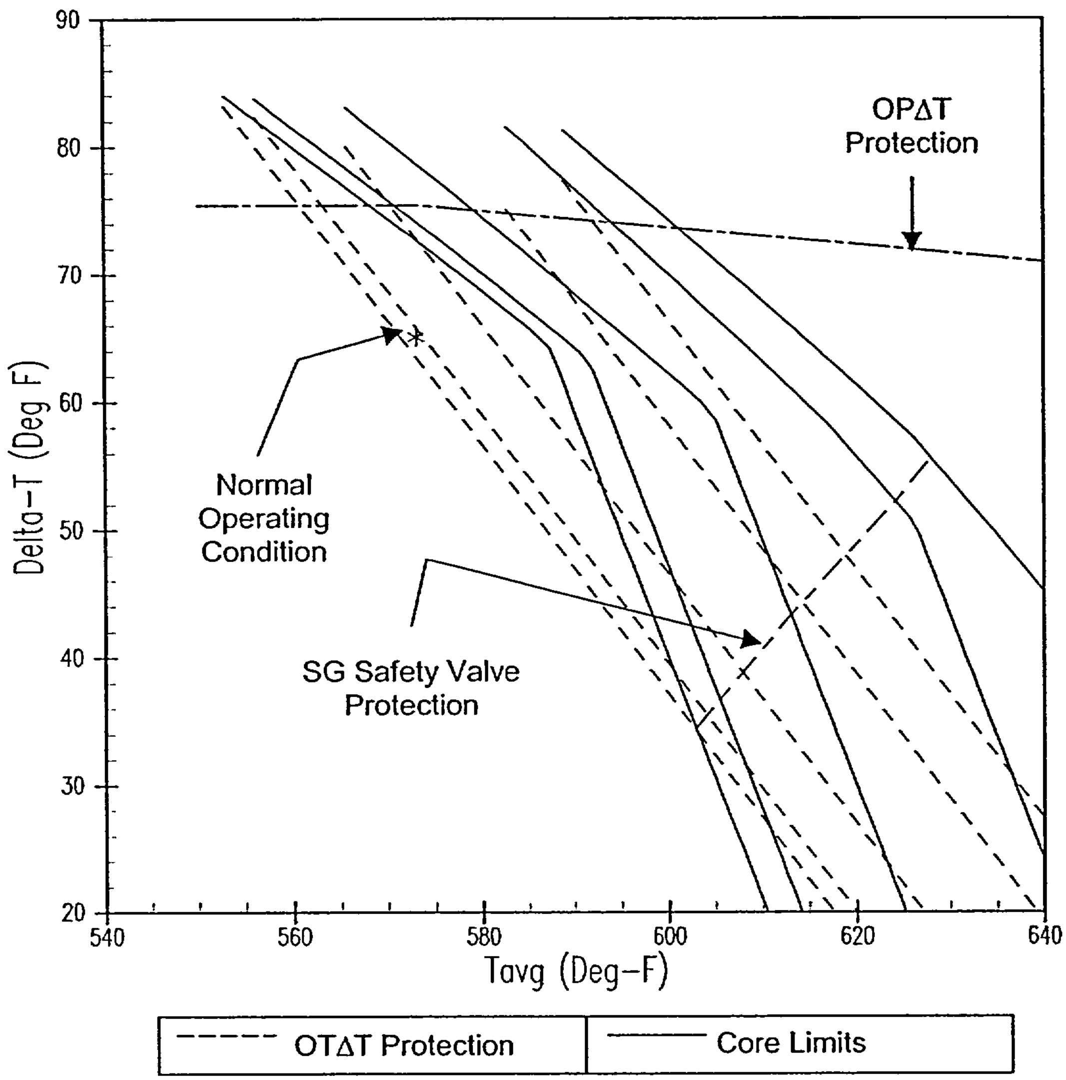

Figure 14.0.2

No changes

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

The following anticipated events are abnormal operational transients resulting from component failure or operator error. They are anticipated to occur sometime in the design life of the plant.

In these events the reactor control and protection system and engineered safeguards are relied upon to protect the core and reactor coolant system boundary from damage.

- Uncontrolled RCCA Withdrawal from a Sub-critical Condition (Section 14.1.1)
- Uncontrolled RCCA Withdrawal at Power (Section 14.1.2)
- RCCA Misalignment (Section 14.1.3)
- Chemical and Volume Control System Malfunction (Section 14.1.4)
- Startup of an Inactive Reactor Coolant Loop (Section 14.1.5)
- Excessive Heat Removal Due to Feedwater System Malfunctions (Section 14.1.6)
- Excessive Load Increase Incident (Section 14.1.7)
- Loss of Reactor Coolant Flow (Section 14.1.8)
- Loss of External Electrical Load (Section 14.1.9)
- Loss of Normal Feedwater (Section 14.1.10)
- Loss of all AC Power to the Plant Auxiliaries (Section 14.1.12)

14.1.1 UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION

Accident Description

A RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of RCCAs resulting in a power excursion. While the probability of this type of a transient is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either sub-critical or at power. The "at power" case is discussed in Section 14.1.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal. Although the initial startup procedure used the method of boron dilution, the normal startup is with RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control rod drive mechanisms are wired into pre-selected bank configurations, which are not altered. The RCCAs are therefore physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel.

The nuclear power response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup accident, since it limits the power to a tolerable level prior to external control action. After the initial power burst, the nuclear power is momentarily reduced. If the accident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Should a continuous RCCA withdrawal be initiated, the transient will be terminated by the following automatic protection or control system actions:

and provides primary protection below the P6 permissive

a. Source Range High Neutron Flux Reactor Trip - This trip is actuated when either of two independent source range channels indicates a flux level above a pre-selected, manually adjustable value. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above ~~a specified setpoint~~. It is automatically reinstated when both intermediate range channels indicate a flux level below ~~a specified setpoint.~~

P6 P6

b. Intermediate Range High Neutron Flux Rod Stop - This rod stop is actuated when either of two independent intermediate range channels indicates a flux level above a pre-selected, manually adjustable value. This rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10% power. It is automatically reinstated when three of the four power range channels are below this value.

c. Although the actuation logic, bypass and automatic reinstatement conditions are the same for the Intermediate Range High Neutron Flux Rod Stop and Intermediate Range High Neutron Flux Reactor Trip, the rod stop is generated at ≤ 35% full power unless manually bypassed above permissive 10 (10% full power). The reactor trip will be actuated at ≤ 40% full power unless it has been manually bypassed above permissive 10.

d. Power Range High Neutron Flux Reactor Trip (low setting) - Trip is actuated when two out of the four power range channels indicate a power level above approximately 25%. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% power and is automatically reinstated when three of the four channels indicate a power level below this value.

e. Power Range High Neutron Flux Rod Stop - This rod stop is actuated when one-out-of-four power range channels indicates a power level above a preset setpoint. This function is always active.

f. Power Range High Neutron Flux Reactor Trip (high setting) - Trip is actuated when two-out-of-four power range channels indicate a power level above a preset setpoint. This trip function is always active.

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

Method of Analysis

REPLACE WITH USAR Insert 14.1.1-1

~~Analysis of this~~ transient is performed by digital computation incorporating the neutron kinetics (including six delayed neutron groups) ~~and the~~ core thermal and hydraulic equations. In addition to the nuclear flux response, the average fuel, clad ~~and water temperatures, and~~ heat flux response were computed.

In order to give conservative results for a startup accident, the following assumptions are made concerning the initial reactor conditions: (absolute magnitude) value for the Doppler power defect is used (1100pcm).

a. Since the magnitude of the nuclear power peak reached during the initial part of the transient is for any given rate of reactivity insertion strongly dependent on the Doppler reactivity coefficient, a conservatively low ~~negative value of −1.0E−5 Δk/°F is used for the startup accident.~~ The less negative ~~Doppler coefficient reduces the~~ Doppler feedback effect~~, thereby~~ increas~~ing~~es the nuclear flux peak.

b. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response time constant. However, after the initial nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value of ~~+1.0E−4 Δk/°F~~ +5 pcm/°F has been used in the analysis ~~since the positive value will~~ to yield the maximum peak core heat flux.

c. The reactor is assumed to be at the hot zero power nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields larger fuel to water heat transfer, larger fuel thermal capacity, and less negative (smaller absolute magnitude) Doppler coefficient. The high nuclear flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. The initial effective multiplication factor ~~(k_{eff})~~ is assumed to be 1.0 since this results in the maximum nuclear flux peak.

d. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint (low setting) raising it from the nominal value of 25% to 35%. Reference to Figure 14.1.1-1, however, shows that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible.

e. A maximum reactivity insertion rate is assumed ~~(8.2E−4 Δk/sec)~~ which is greater than that for the simultaneous withdrawal at maximum speed of the combination of the two RCCA banks having the greatest combined worth.

f. Initial power level of 1.0E−~~10~~ 9 multiplied by the nominal full power level is assumed to maximize the heat flux peak.

Add USAR Insert 14.1.1-2

USAR Insert 14.1.1-1

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the DNBR calculation. The average nuclear power transient with respect to time calculation is performed using a spatial neutron kinetics code, TWINKLE, which includes the various total core feedback effects, i.e., Doppler and moderator reactivity. The FACTRAN code is then used to calculate the thermal heat flux transient, based on the nuclear power transient calculated by TWINKLE. FACTRAN also calculates the fuel and cladding temperatures. The average heat flux is next used in VIPRE for transient DNBR calculations.

USAR Insert 14.1.1-2

g. The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential banks in their highest worth position, are assumed for DNB analysis.

h. One reactor coolant pump is assumed to be in operation. This lowest initial flow minimizes the resulting DNBR.

i. A core flow reduction of 1.1 percent, which addresses the potential reactor coolant flow asymmetry associated with a maximum loop-to-loop steam generator tube plugging imbalance of 10 percent, has been applied.

USAR Insert 14.1.1-3

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.

Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected. The minimum departure from nucleate boiling ratio remains above the limit value and thus, no fuel or clad damage is predicted.

Results

the uncontrolled RCCA bank withdrawal with the

fuel centerline, average fuel

Figures 14.1.1-1 through 14.1.1-5 show the transient behavior of key parameters for ~~a reactivity insertion rate of 8.2E-4 Δk/sec. The~~ accident ~~is~~ terminated by a reactor trip at 35% power.

The nuclear power overshoots nominal full power, but only for a very short time period. Hence, the energy release and the fuel temperature increases are small. The heat flux response, of interest for DNB considerations, is shown in Figure 14.1.1-2. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux that is less than the nominal full power heat flux. There is a large margin to DNB during the transient since the rod surface heat flux remains below the full power design value, and there is a high degree of sub-cooling at all times in the core. Figures 14.1.1-3, 14.1.1-4, and 14.1.1-5 show the response of the ~~core average fuel, coolant~~ and cladding temperatures at the hot spot. The average fuel temperature increases to a value that is lower than the nominal full power value. ~~The average coolant temperature increases to a value that is also less than the full power nominal value.~~ hot spot

REPLACE WITH USAR Insert 14.1.1-3

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Uncontrolled rod withdrawal from sub-critical	3.218/1.14	2358/2750	1156/1210

Conclusions

Considering the conservative assumptions used in the accident analysis, it is concluded that in the unlikely event of a control rod withdrawal accident the core and reactor coolant systems are not adversely affected. The peak heat flux reached remains less than the nominal full power value. DNBR is well above its limiting value. The peak average clad temperature is less than its nominal full power value, and thus there is no possibility of fuel or clad damage.

14.1.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

Section 14.1.2 changes suggested later

Accident Description

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

Replace with New Fig 14.1.1-1

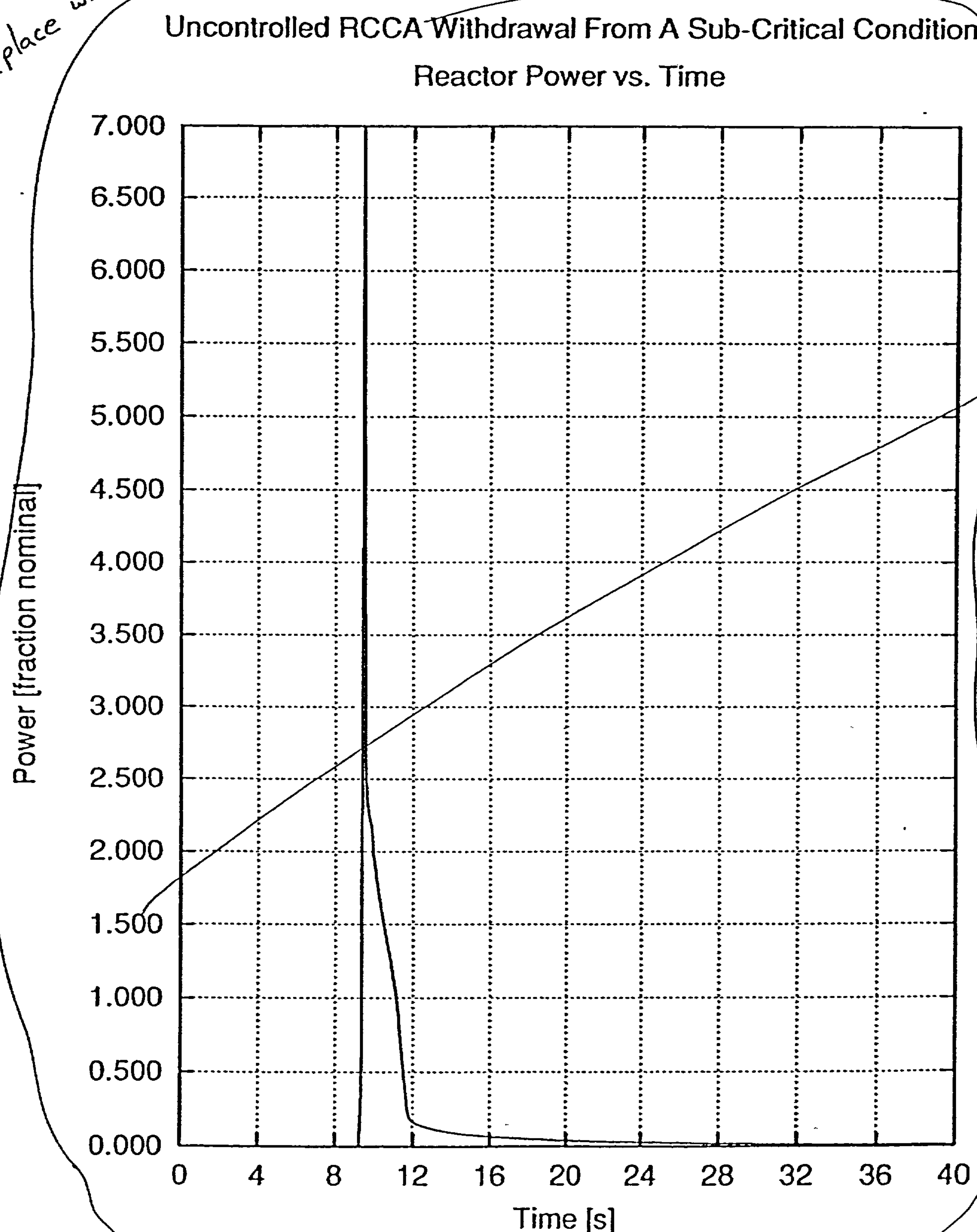

Figure 14.1.1-1

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Reactor Power vs Time

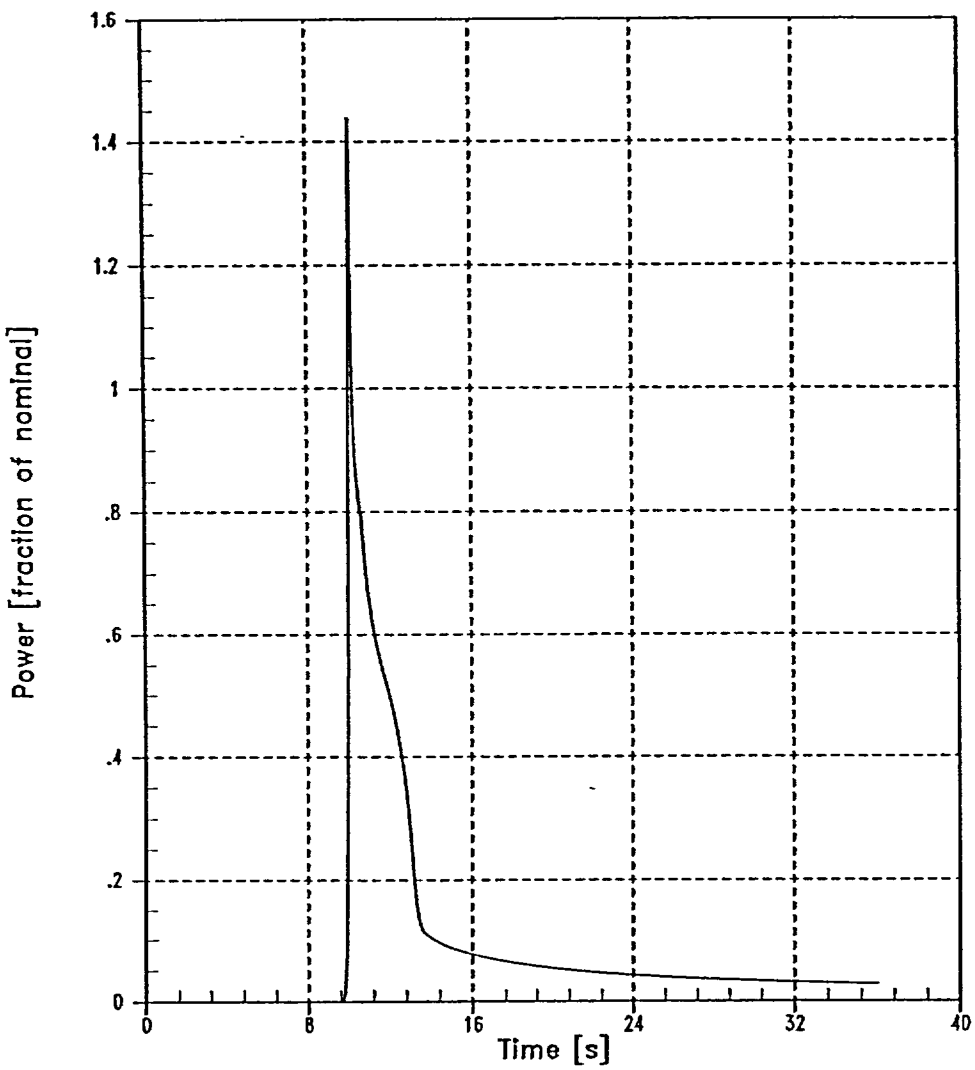

Figure 14.1.1-1

Replace with new Fig. 14.1.1-2

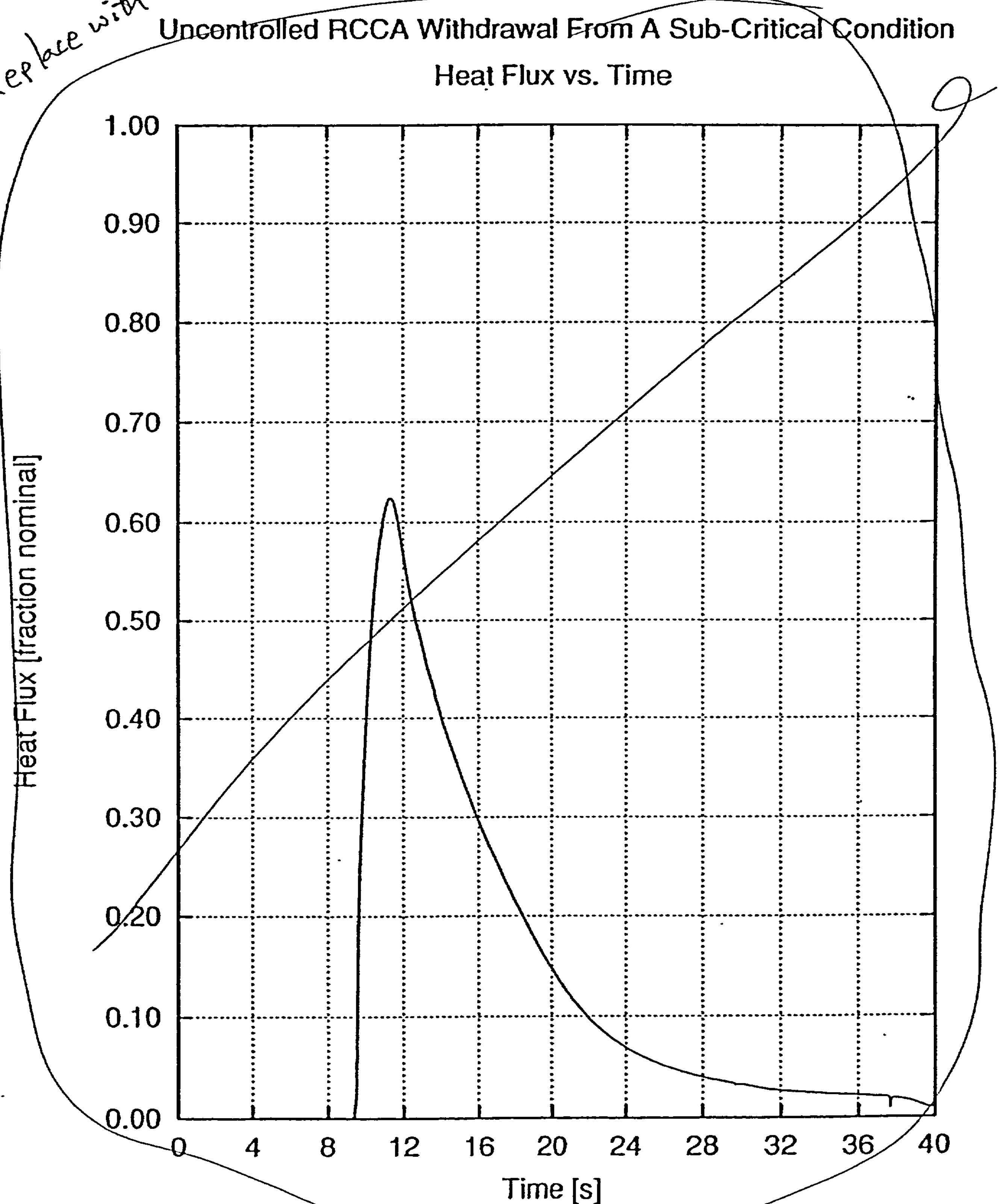

Figure 14.1.1-2

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Heat Flux vs Time

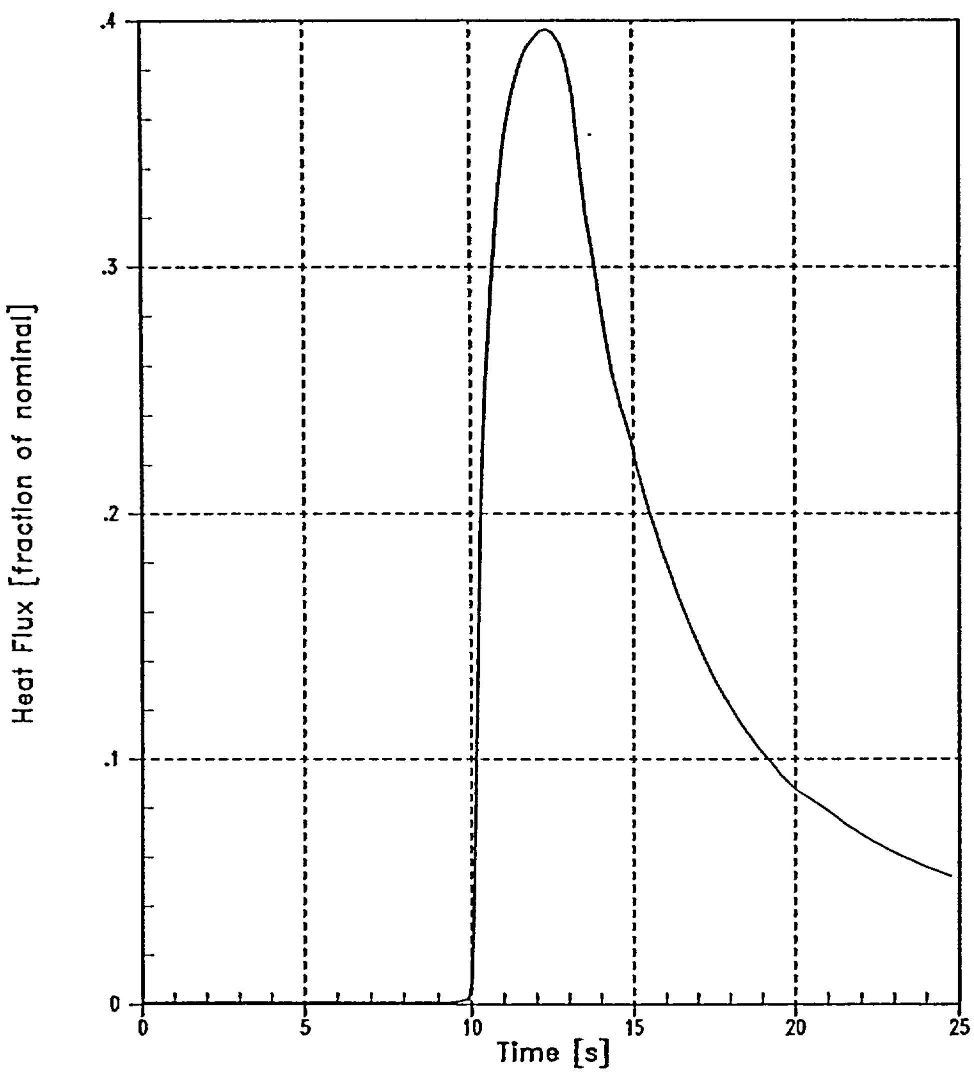

Figure 14.1.1-2

Replace with new Fig. 14.1.1-3

Uncontrolled RCCA Withdrawal From A Sub-Critical Condition

Fuel Temperature vs. Time

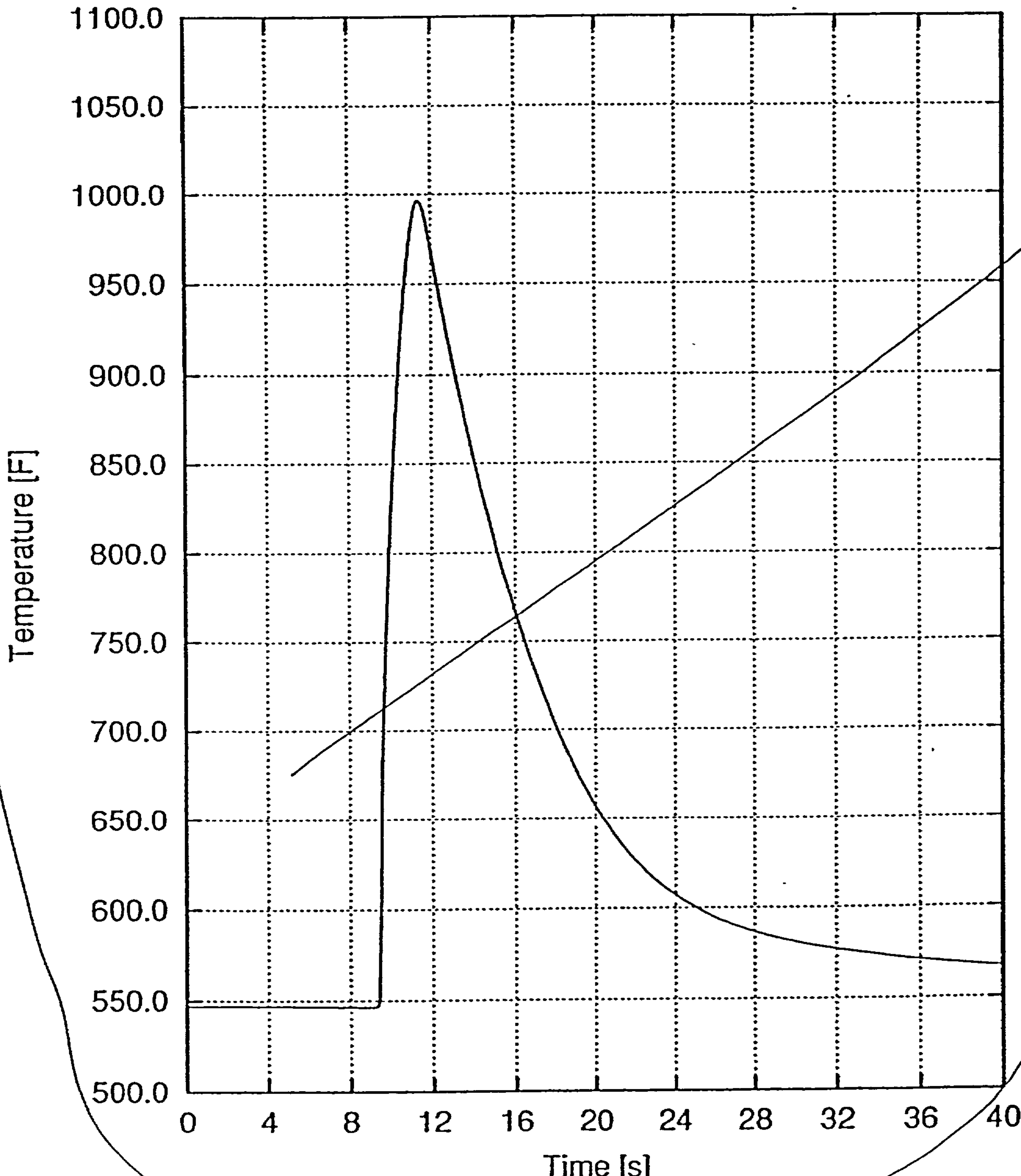

Figure 14.1.1-3

Rev.16
12/01/2000

Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Hot-Spot Fuel Centerline Temperature vs Time

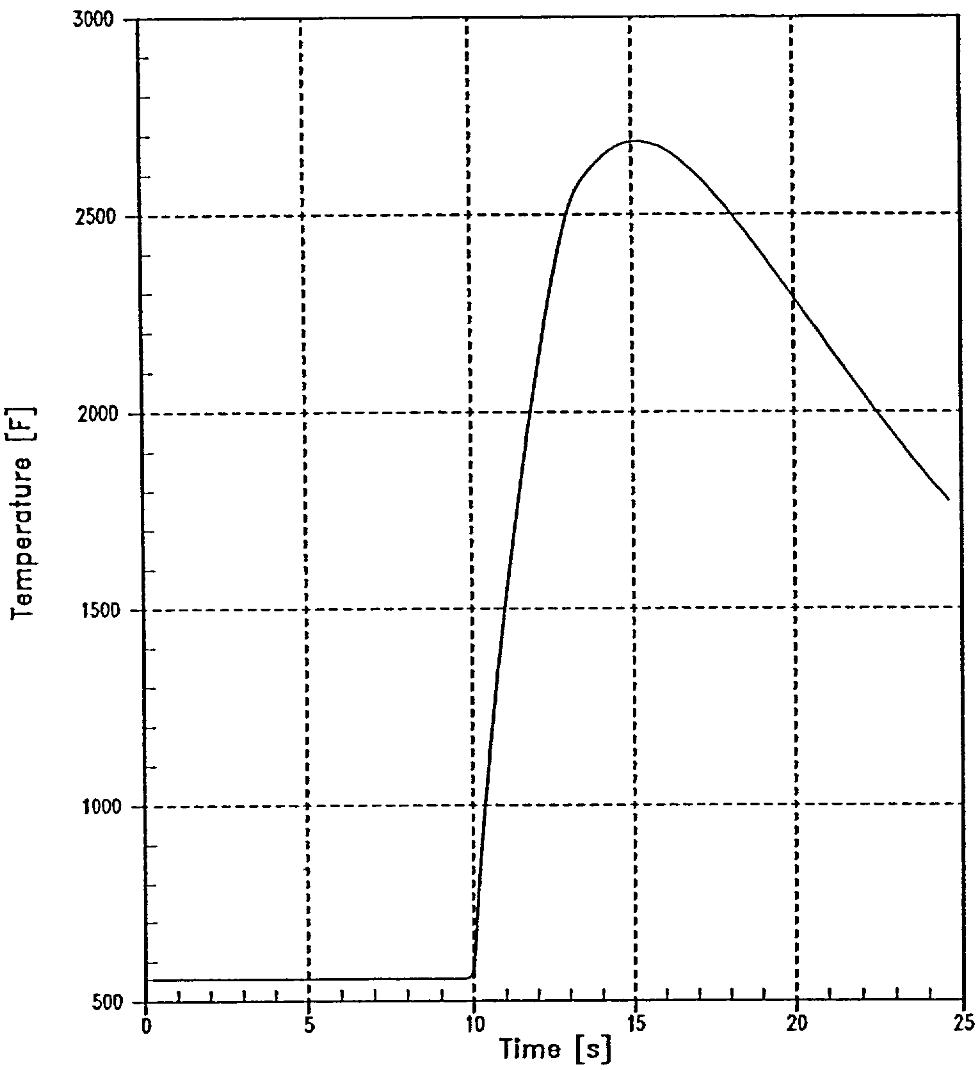

Figure 14.1.1-3

Replace with new Fig 14.1.1-4

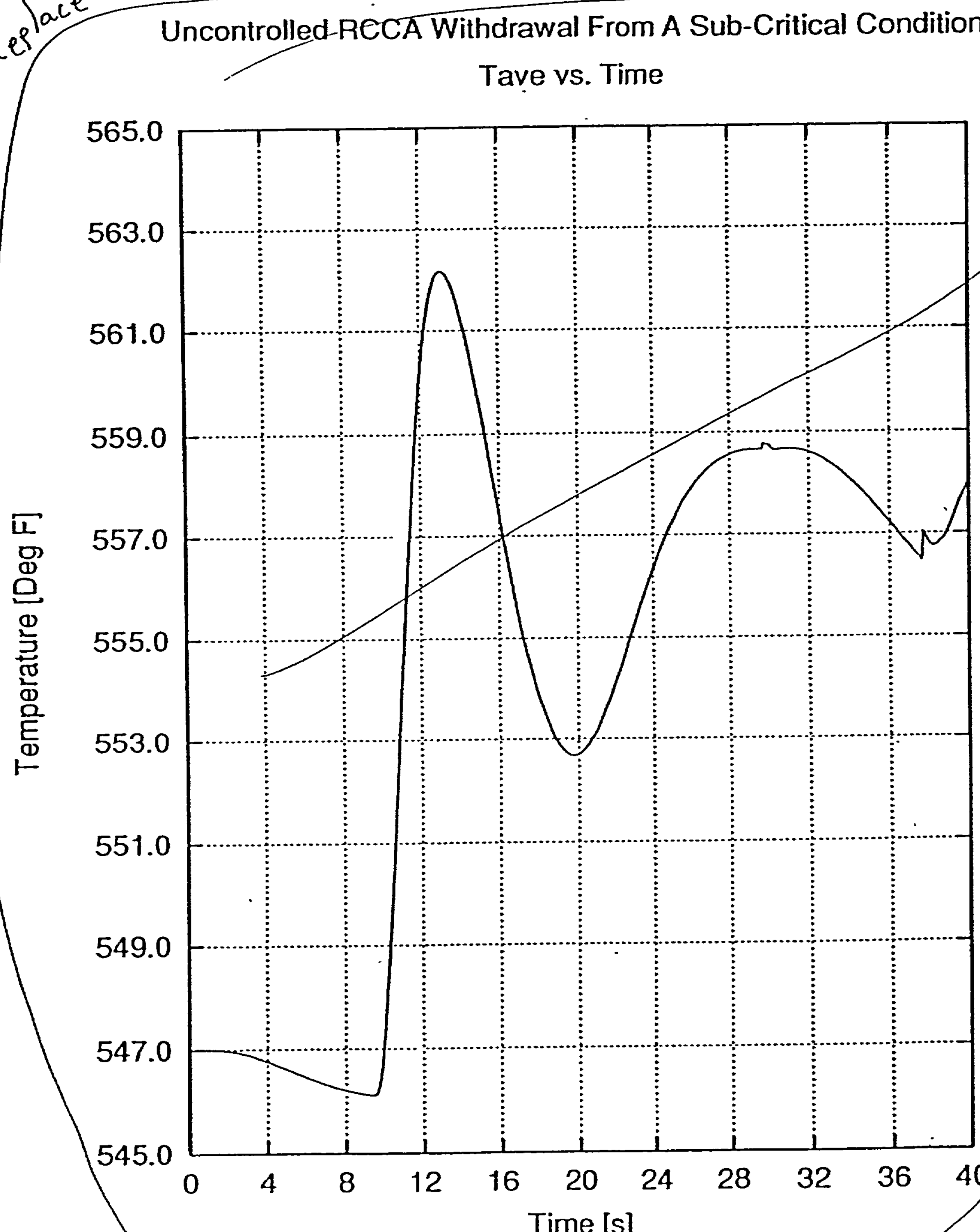

Figure 14.1.1-4

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal From A Sub-Critical Condition
Hot-Spot Fuel Average Temperature vs. Time

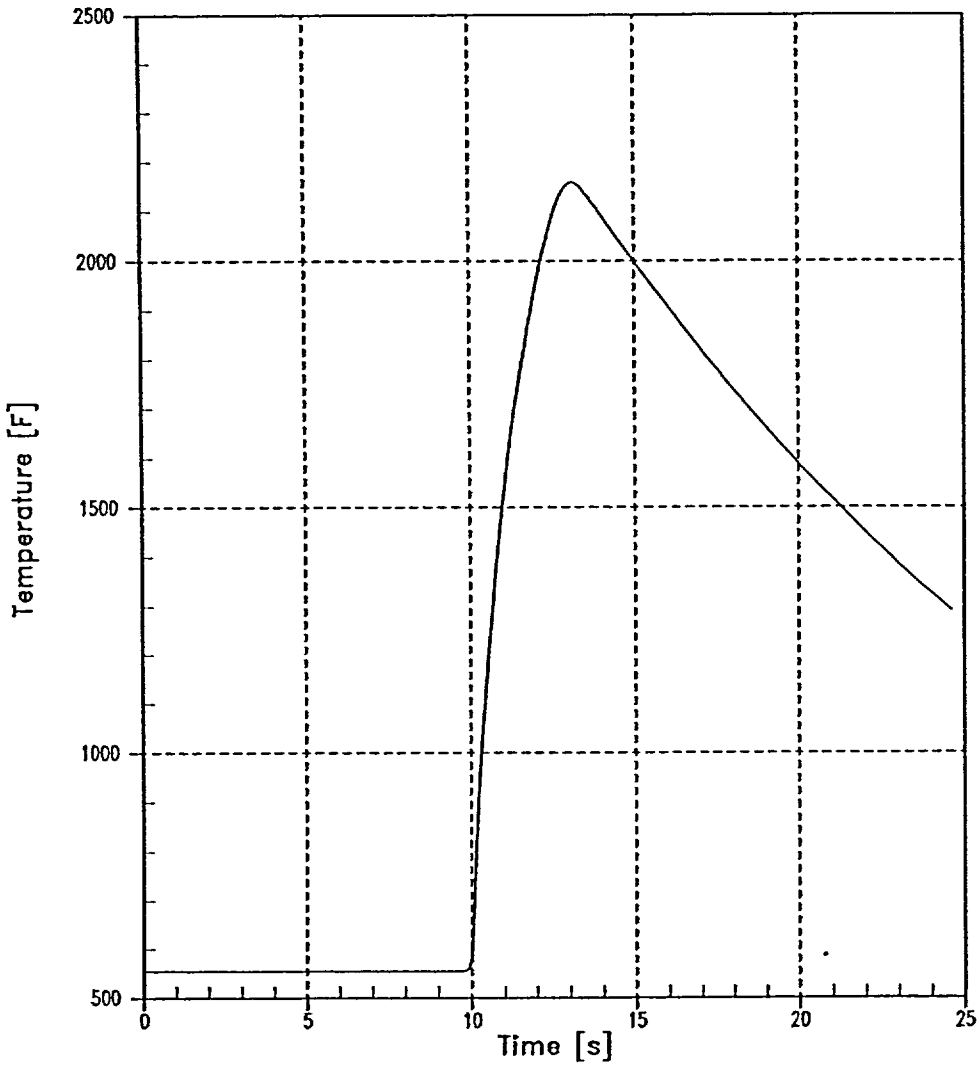

Figure 14.1.1-4

Replace with new Fig 14.1.1-5

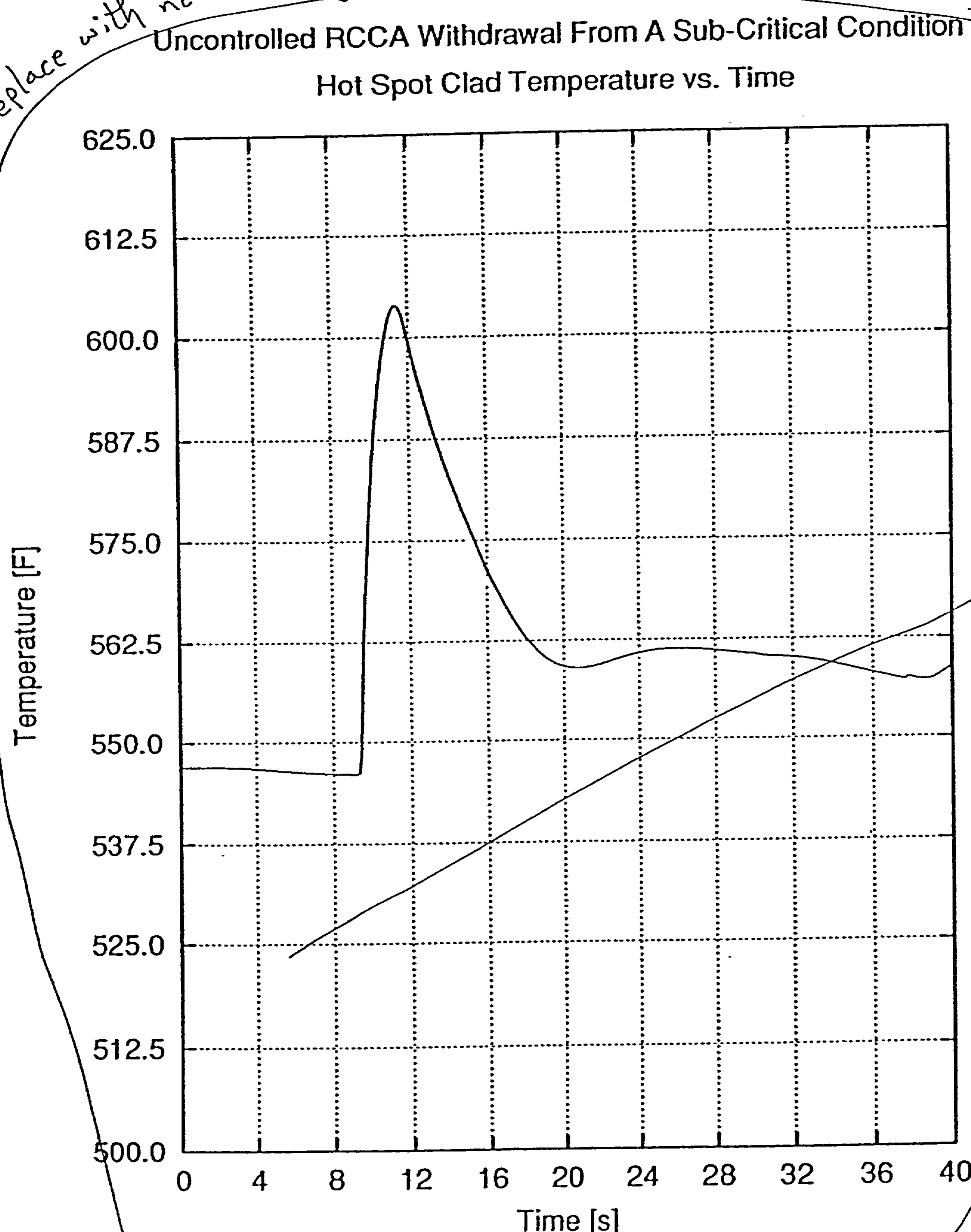

Figure 14.1.1-5

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal From A Sub–Critical Condition
Hot–Spot Cladding Temperature vs. Time

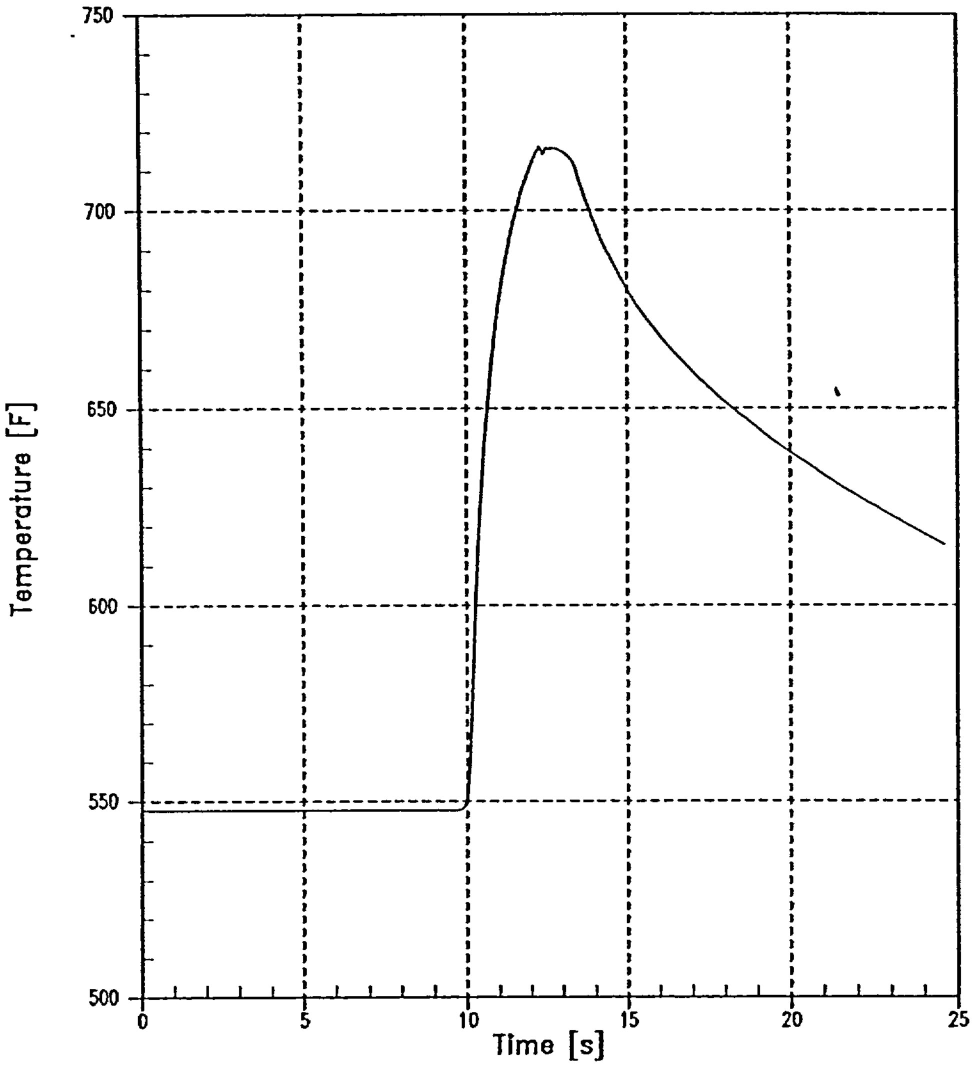

Figure 14.1.1–5

OUT OF SCOPE

Section 14.1.1 Changes Suggested earlier

Results

Figures 14.1.1-1 through 14.1.1-5 show the transient behavior of key parameters for a reactivity insertion rate of 8.2E–4 Δk/sec. The accident is terminated by a reactor trip at 35% power.

The nuclear power overshoots nominal full power, but only for a very short time period. Hence, the energy release and the fuel temperature increases are small. The heat flux response, of interest for DNB considerations, is shown in Figure 14.1.1-2. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux that is less than the nominal full power heat flux. There is a large margin to DNB during the transient since the rod surface heat flux remains below the full power design value, and there is a high degree of sub-cooling at all times in the core. Figures 14.1.1-3, 14.1.1-4, and 14.1.1-5 show the response of the core average fuel, coolant and cladding temperature. The average fuel temperature increases to a value that is lower than the nominal full power value. The average coolant temperature increases to a value that is also less than the full power nominal value.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Uncontrolled rod withdrawal from sub-critical	3.218/1.14	[redacted]/2750	[redacted]/1210

Conclusions

Considering the conservative assumptions used in the accident analysis, it is concluded that in the unlikely event of a control rod withdrawal accident the core and reactor coolant systems are not adversely affected. The peak heat flux reached remains less than the nominal full power value. DNBR is well above its limiting value. The peak average clad temperature is less than its nominal full power value, and thus there is no possibility of fuel or clad damage.

14.1.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

Accident Description

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

The automatic features of the Reactor Protection System which prevent core damage in an RCCA withdrawal incident at power include the following:

1. Nuclear power range instrumentation actuates a reactor trip if two-out-of-four channels exceed an overpower setpoint.

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

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

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

5. A high pressurizer water level reactor trip, actuated from any two-out-of-three level channels, is set at a fixed point. This affords additional protection for RCCA withdrawal incidents.

6. In addition to the above listed reactor trips, there are the following control rod assembly withdrawal blocks:

   - High nuclear power (one-out-of-four)
   - High overpower ΔT (two-out-of-four)
   - High overtemperature ΔT (two-out-of-four)

Method of Analysis

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

Analysis is performed using several digital computer codes. The reactor protection functions are incorporated into the transient analysis digital simulation of the Nuclear Steam Supply System. The system response to the transient is then used as a transient forcing function for the fuel thermal hydraulic analysis and DNBR assessment.

In order to obtain conservatively low DNBRs, the following assumptions are made:

1. Initial conditions assume maximum power and reactor coolant temperatures and minimum pressure; i.e., the power is assumed 2% high, the average temperature is assumed 4°F high, and the pressure is assumed [redacted] psi low. This gives the minimum initial margin to DNB.

Insert A

This transient is analyzed using the RETRAN code. This code simulates the neutron kinetics, RCS, presssurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures and power level. The core limits, as illustrated on Figure 14.0.2, are used to develop input to RETRAN to determine the minimum DNBR during the transient.

In order to obtain a conservative value for the minimum DNBR, the following analysis assumptions are made:

1. This accident is analyzed with the Revised Thermal Design Procedure (Section 3.2). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

2. Reactivity Coefficients - Two cases are analyzed.

    a. Minimum Reactivity Feedback - A zero moderator temperature coefficient of reactivity (0 pcm/°F) is assumed at full power. For power levels less than or equal to 60% power, a positive moderator temperature coefficient of reactivity (+5 pcm/°F) is conservatively assumed, corresponding to the beginning of core life. A conservatively small (in absolute magnitude) Doppler power coefficient is used in the analysis.

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

3. The reactor trip on high neutron flux is actuated at a conservative value of 118% of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and setpoint errors. The delays for trip actuation are assumed to be the maximum values. No credit was taken for the other expected trip functions.

4. 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.

5. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined differential rod worth at a conservative speed (45 inches/minute, which corresponds to 72 steps/minute).

6. Power levels of 10%, 60% and 100% of full power are considered.

7. The impact of a full power RCS vessel T_{avg} window was considered for the uncontrolled RCCA bank withdrawal at power analysis. A conservative calculation modeling the high end of the RCS vessel T_{avg} window was explicitly analyzed.

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

2. A zero ~~moderator~~ coefficient of reactivity is assumed corresponding ~~to the beginning~~ of core life. A conservatively ~~small (in absolute magnitude)~~ Doppler reactivity coefficient is used. ~~The assumed reactivity coefficients result in a minimum~~ of negative feedback ~~reactivity, and~~ therefore, higher peak powers and temperatures.

Results

10.0

Figures 14.1.2-1 through 14.1.2-4 show the response of nuclear power, pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal (~~8.2E–4~~ Δk/sec) incident starting from full power. This reactivity insertion rate is greater than that for the two highest worth banks, both assumed in their highest incremental worth region, withdrawn at their maximum speed. Reactor Trip on high nuclear power occurs less than 2.0 seconds from the start of the accident. Since this is rapid with respect to the thermal time constants small changes in T_{avg} and pressure result. A large margin to the MDNBR limit is maintained.

The response of nuclear power, pressure, average coolant temperature, and DNBR for a slow RCCA withdrawal ([redacted].0E–5 Δk/sec) from full power is shown in Figures 14.1.2-5 through 14.1.2-8. Reactor Trip occurs on overtemperature ΔT. The rise in temperature and pressure is larger than for the rapid RCCA withdrawal. The minimum DNBR reached during the transient is greater than the ~~M~~DNBR limit. Add Insert B

The nuclear power, RCS pressure, coolant average temperature, and DNBR responses for an RCCA withdrawal from 60% power are shown in Figures 14.1.2-9 through 14.1.2-12 for a rapid withdrawal rate (8.2E–4 Δk/sec) and in Figures 14.1.2-13 through 14.1.2-16 for a slow withdrawal rate (1.5E–5 Δk/sec). The results demonstrate that the overtemperature ΔT and high nuclear flux trip functions adequately protect the fuel. The minimum DNBR reached is above the MDNBR limit.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Uncontrolled rod withdrawal [redacted]	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Fast Rate Full Power	[redacted]/1.14	[redacted]/2750	[redacted]/1210
Slow Rate Full Power	1.362/1.14	[redacted]/2750	[redacted]/1210
Fast Rate Intermediate Power	[redacted]/1.14	[redacted]/2750	[redacted]/1210
Slow Rate Intermediate Power	[redacted]/1.14	2350/2750	1182/1210

Conclusions

In the unlikely event of an RCCA withdrawal incident during power operation, the core and Reactor Coolant System are not adversely affected since the minimum value of the DNBR reached is greater than the DNBR limit for all RCCA reactivity rates. Protection is provided by the high nuclear flux, ~~overpower ΔT~~ and overtemperature ΔT trip functions.

Insert B

Figures 14.1.2-9 through 14.1.2-11 show the minimum departure from nucleate boiling ratio as a function of the reactivity insertion rate for the three initial power levels (100%, 60%, and 10%) and minimum and maximum reactivity feedback. It can be seen that the high neutron flux and overtemperature ΔT trip channels provide protection over the whole range of reactivity insertion rates. The minimum DNBR is never less than the limit value.

In the referenced figures, the shape of the curves of minimum departure from nuclear boiling ratio versus reactivity insertion rate is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 14.1.2-11 for example, it is noted that:

1. For high reactivity insertion rates (i.e, between ~100 pcm/second and ~30 pcm/second) when modeling minimum reactivity feedback, reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates, while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum departure from nucleate boiling ratios during the transient. Within this range, as the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (e.g., at an approximately 30 pcm/second reactivity insertion rate).

   The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant trip ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note, however, that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the reactor coolant system in response to power increases.

   For reactivity insertion rates between ~30 pcm/second and ~8 pcm/second, the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum departure from nucleate boiling ratio) due to the fact that, with lower insertion rates, the power increase rate is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

3. For reactivity insertion rates of ~8 pcm/second and lower, the rise in reactor coolant temperature is sufficiently high so that there is more steam relief through the steam generator safety valves prior to trip. This steam relief acts as an additional heat sink on

the reactor coolant system and sharply decreases the rate of rise of reactor coolant system average temperature. This causes the overtemperature ΔT trip setpoint to be reached later with resulting lower minimum departure from nucleate boiling ratios.

REPLACE WITH FIGURE THAT FOLLOWS

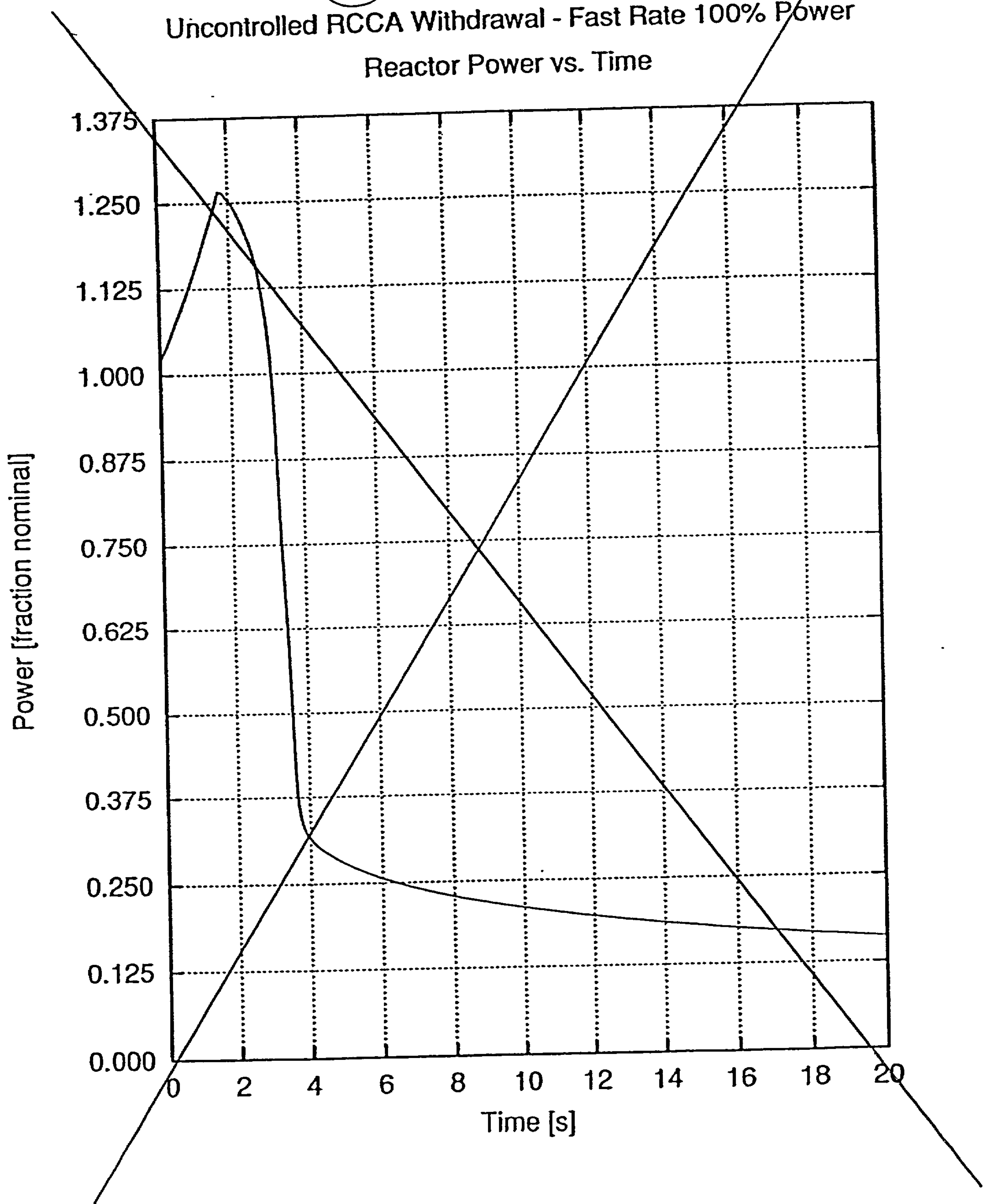

Figure 14.1.2-1

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12/01/2000

Uncontrolled RCCA Withdrawal – Fast Rate 100% Power
Reactor Power vs Time

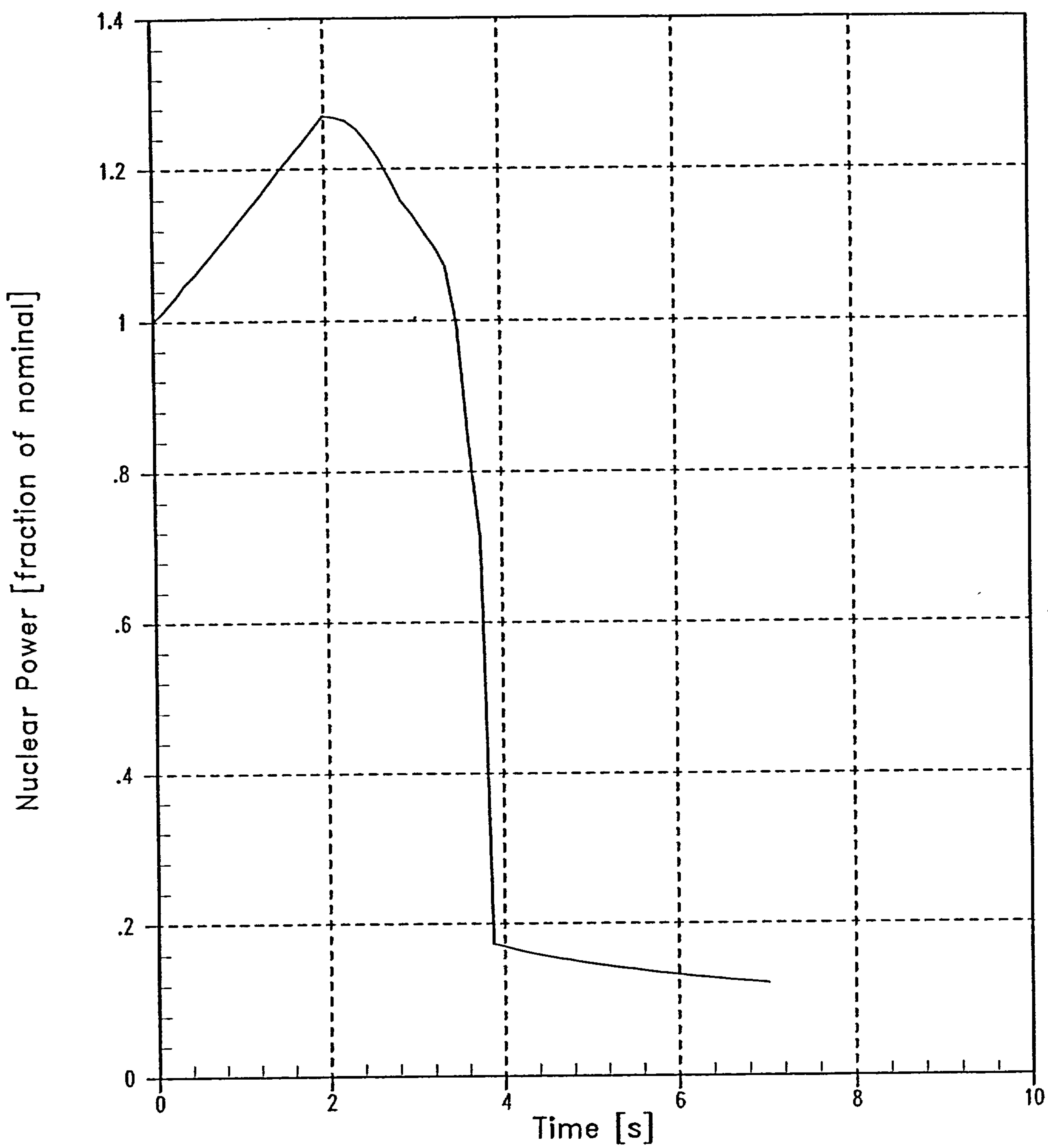

Figure 14.1.2–1

REPLACE WITH THE FIGURE THAT FOLLOWS

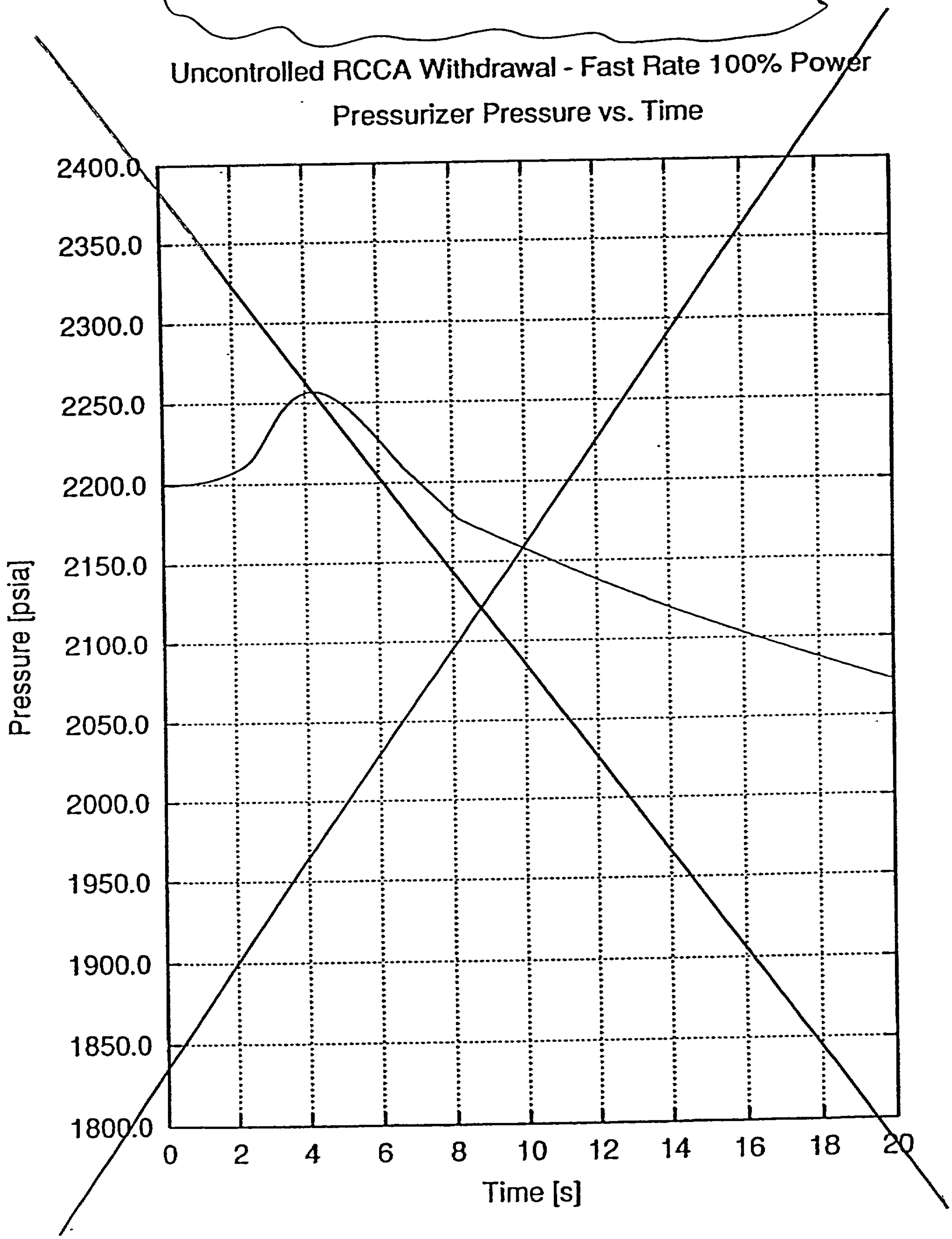

Figure 14.1.2-2

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal — Fast Rate 100% Power
Pressurizer Pressure vs Time

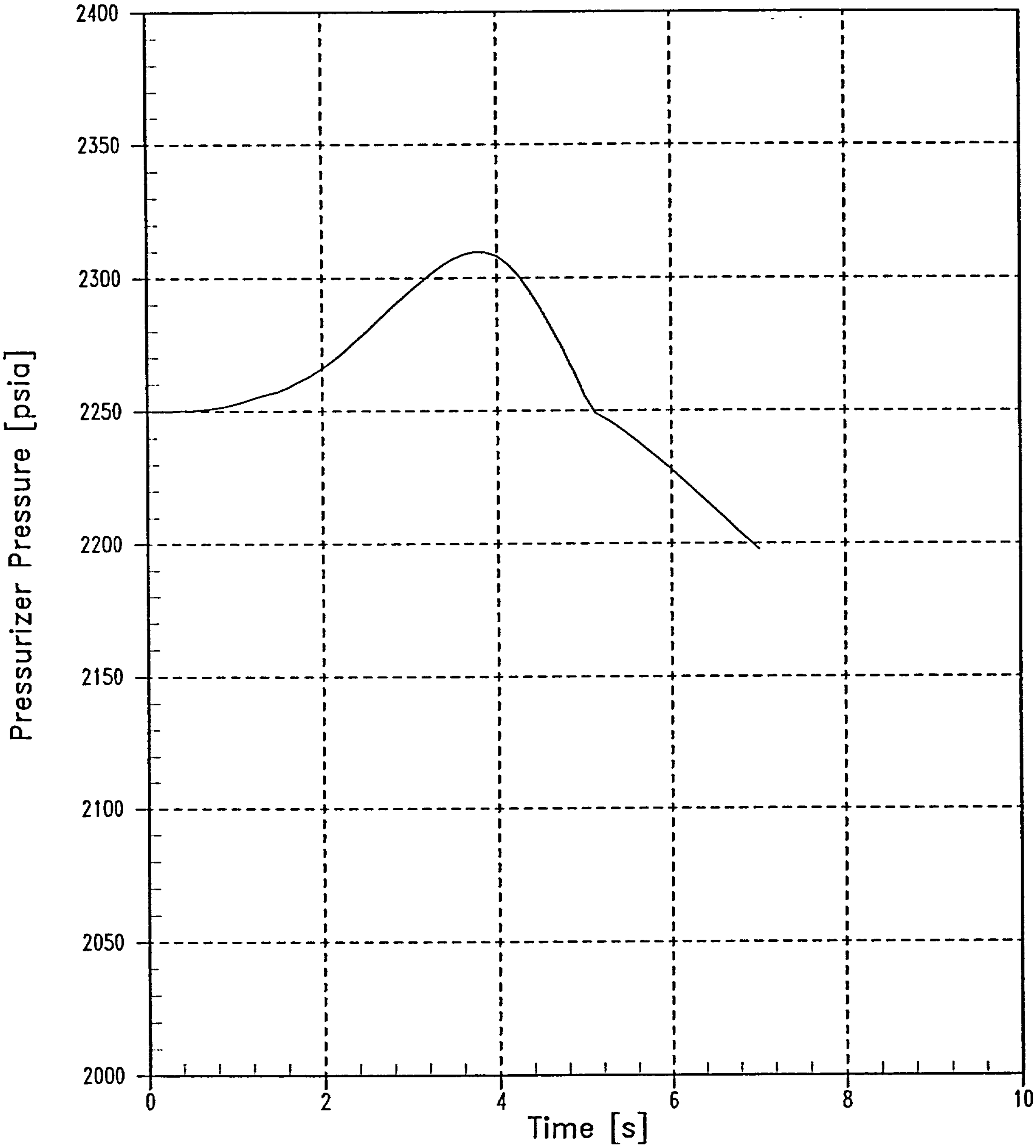

Figure 14.1.2–2

REPLACE WITH THE FIGURE THAT FOLLOWS

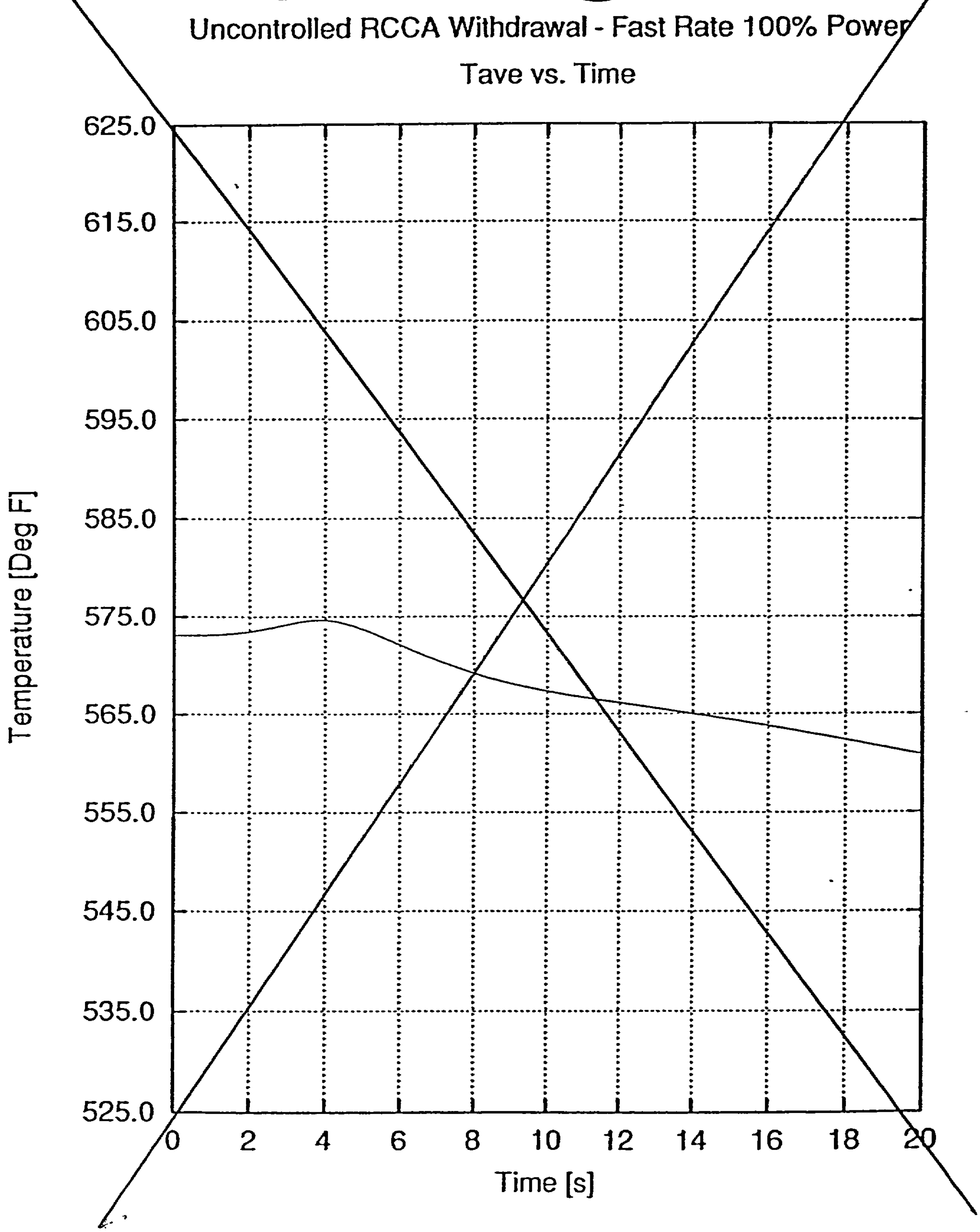

Figure 14.1.2-3

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal – Fast Rate 100% Power
Tave vs Time

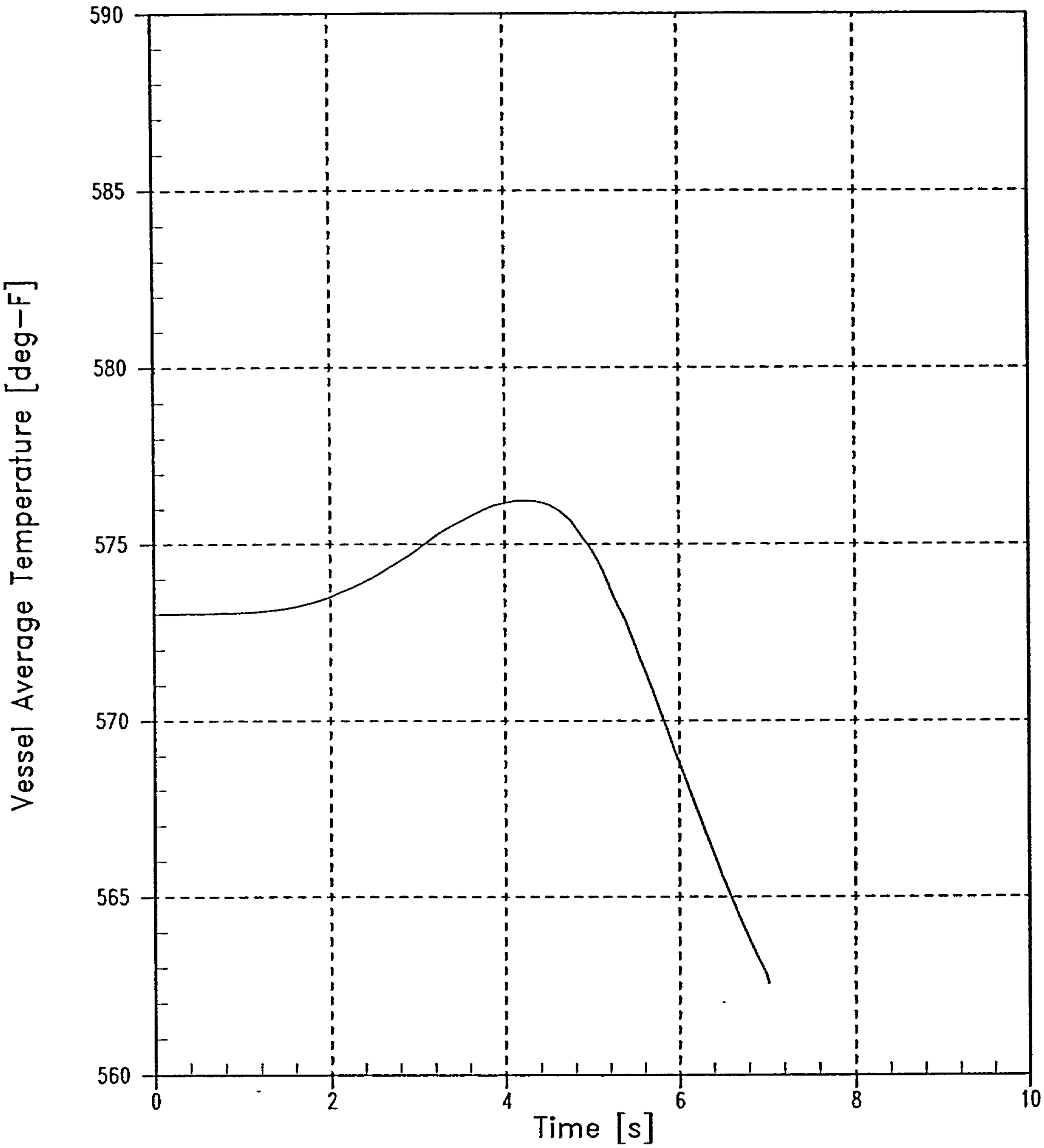

Figure 14.1.2–3

REPLACE WITH THE FIGURE THAT FOLLOWS

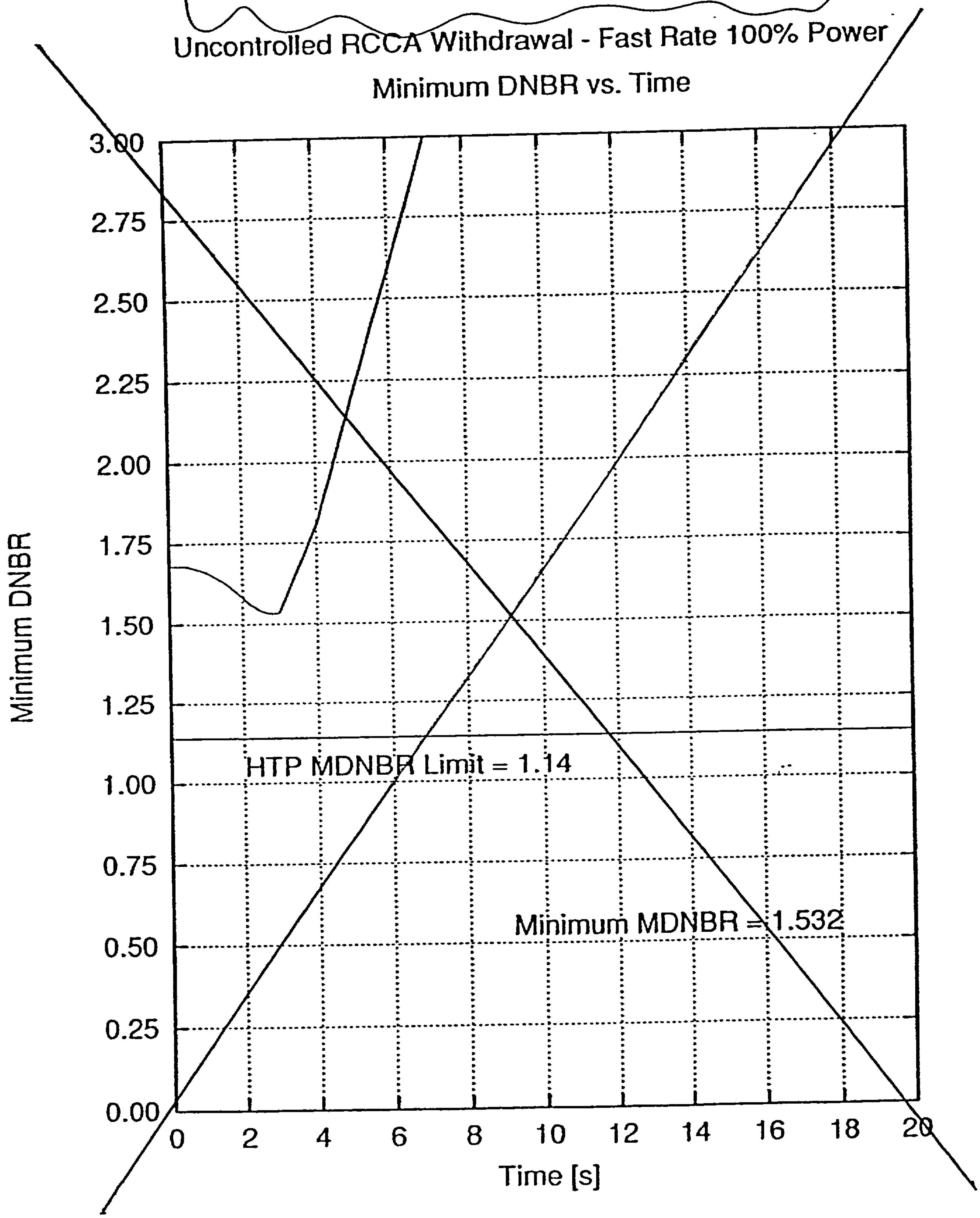

Figure 14.1.2-4

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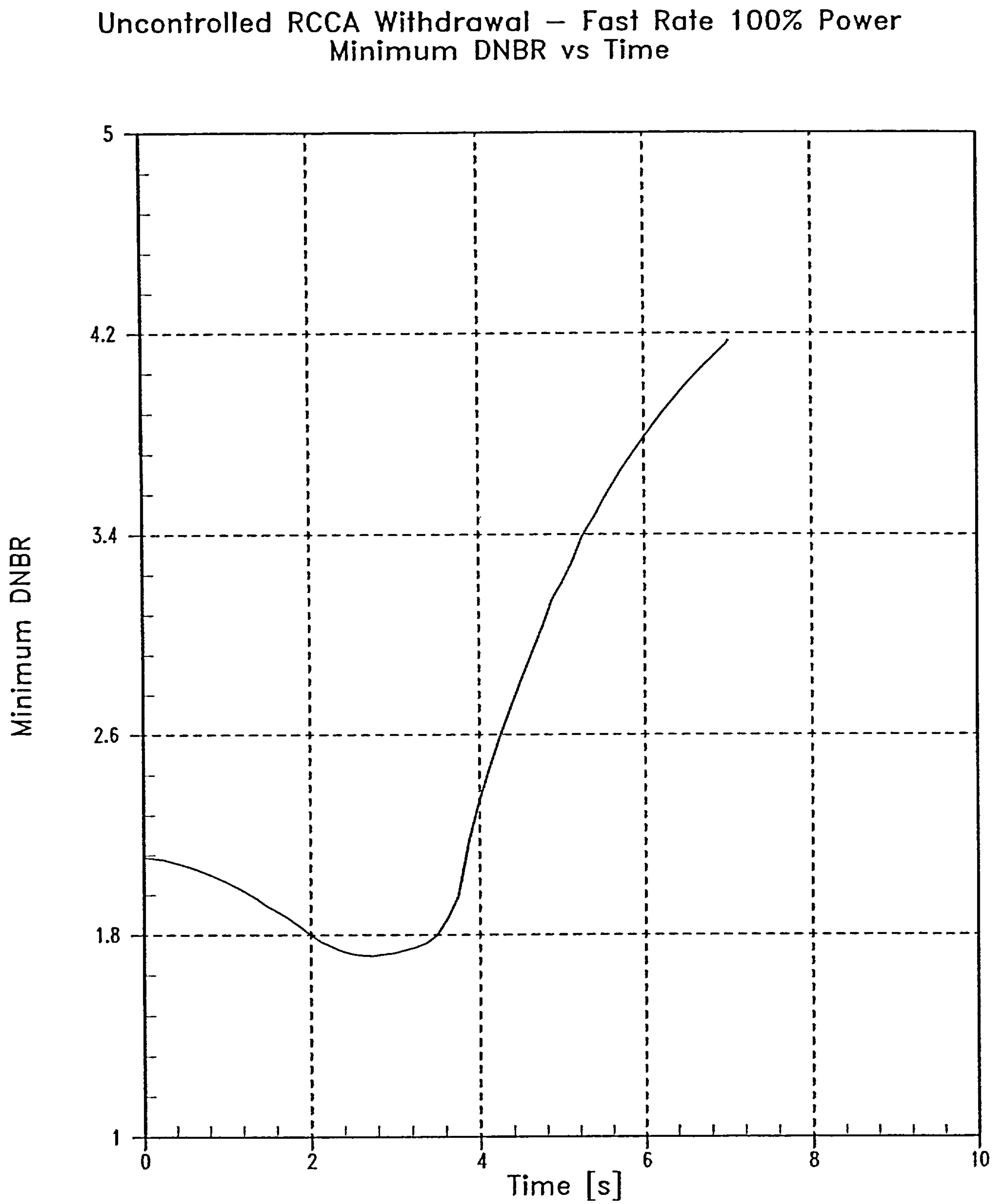

Figure 14.1.2–4

REPLACE WITH THE FIGURE THAT FOLLOWS

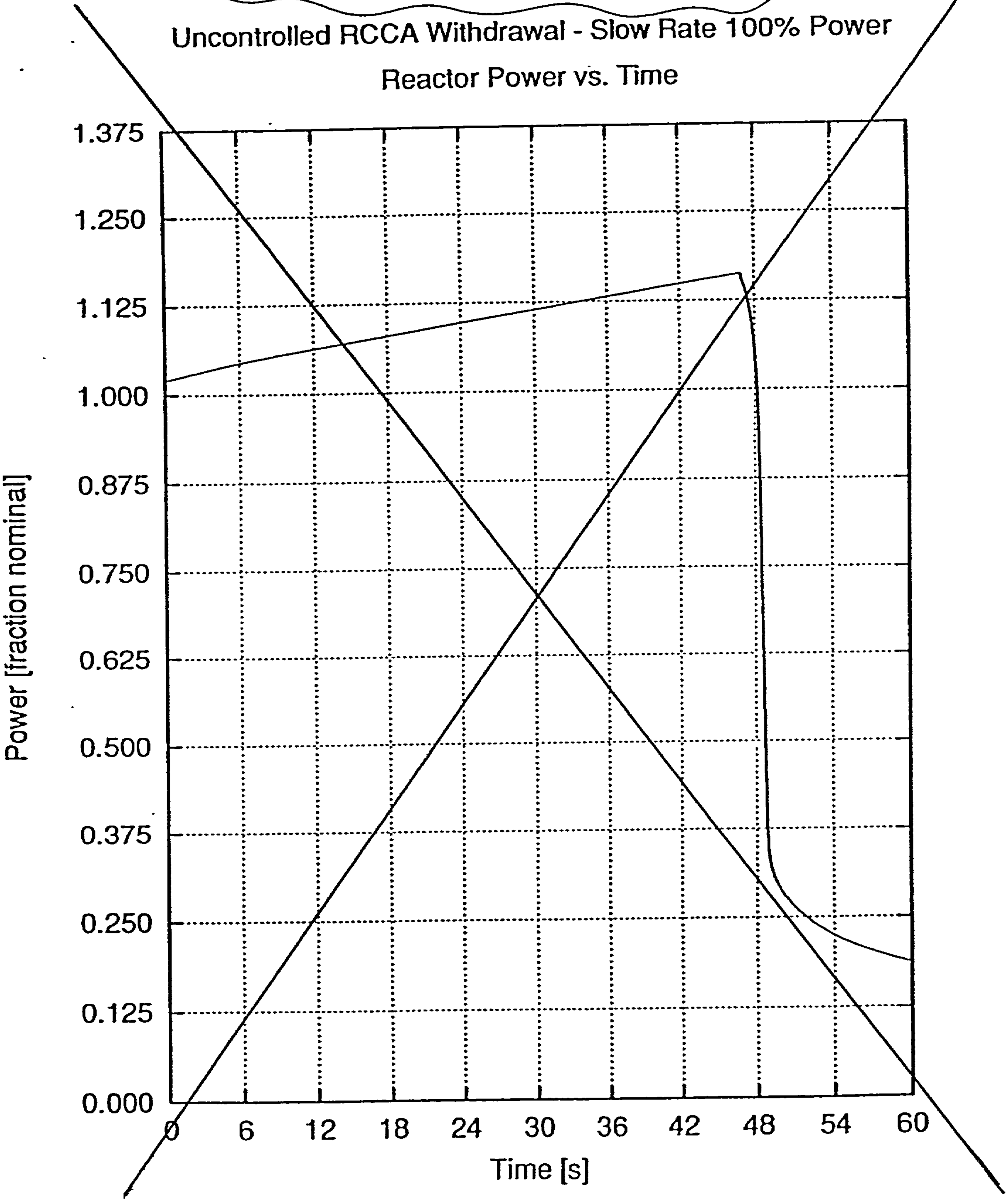

Figure 14.1.2-5

Rev. 16
12/01/2000

Uncontrolled RCCA Withdrawal – Slow Rate 100% Power
Reactor Power vs Time

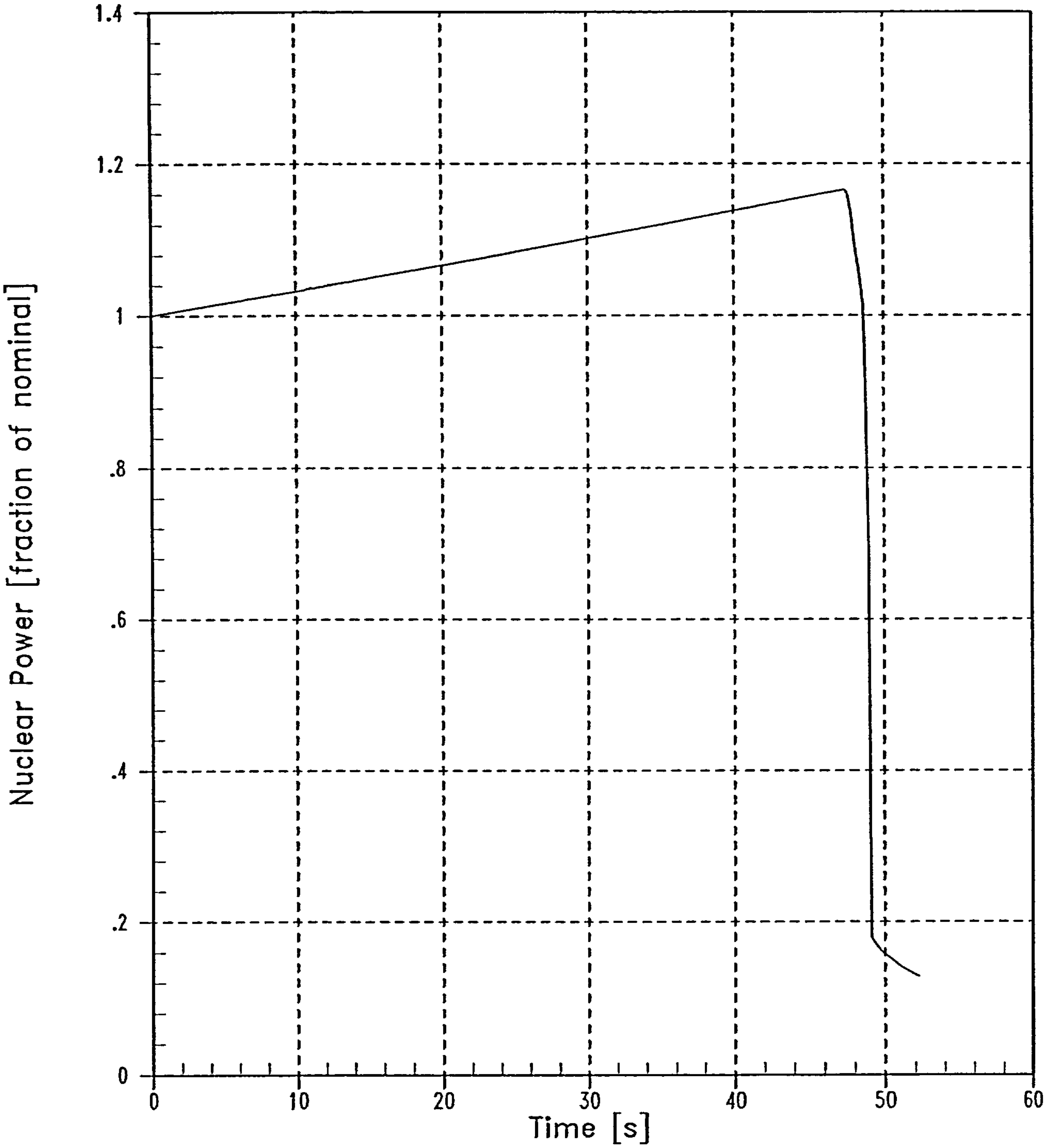

Figure 14.1.2–5

REPLACE WITH THE FIGURE THAT FOLLOWS

Uncontrolled RCCA Withdrawal - Slow Rate 100% Power

Pressurizer Pressure vs. Time

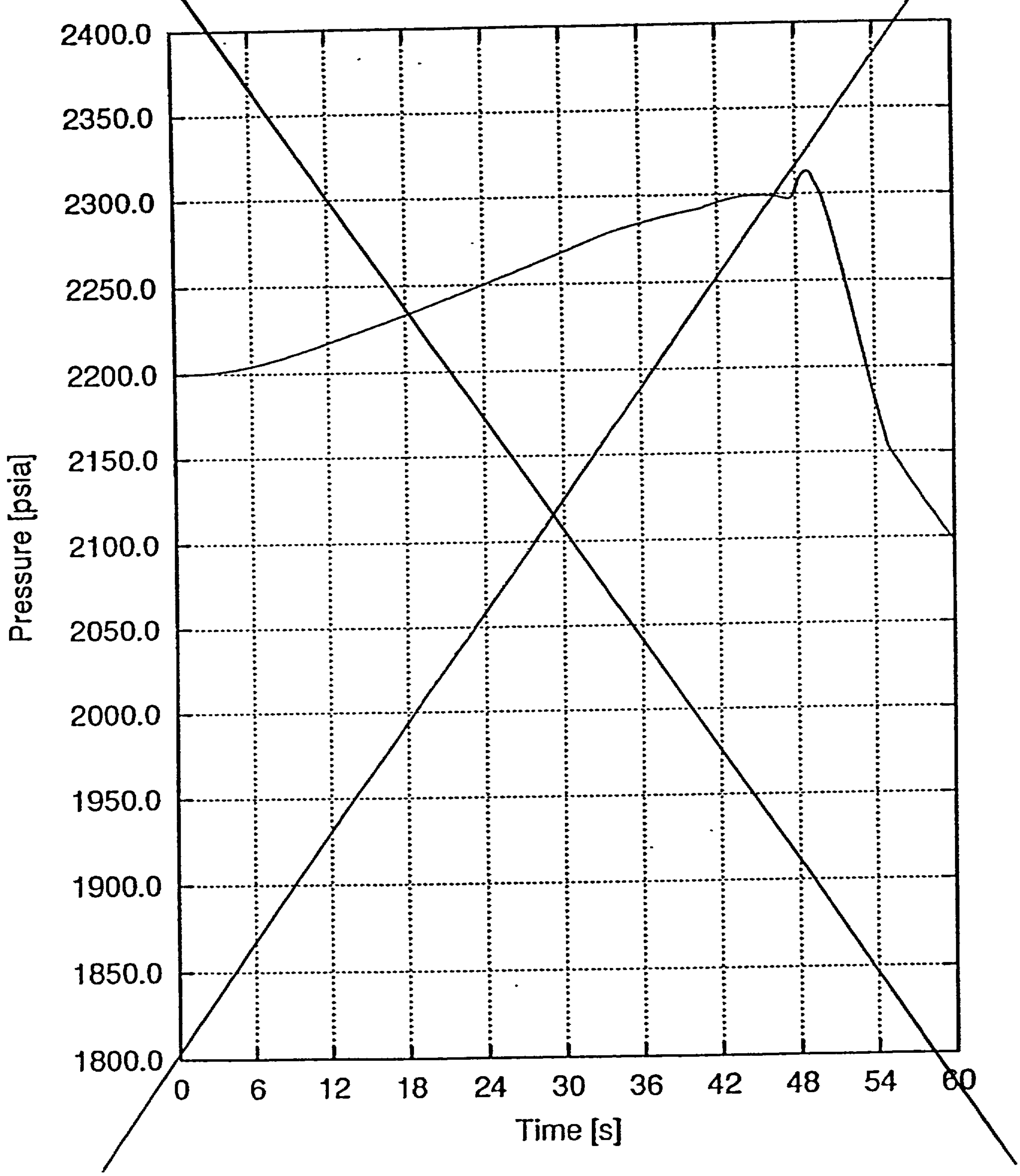

Figure 14.1.2-6

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Uncontrolled RCCA Withdrawal — Slow Rate 100% Power
Pressurizer Pressure vs Time

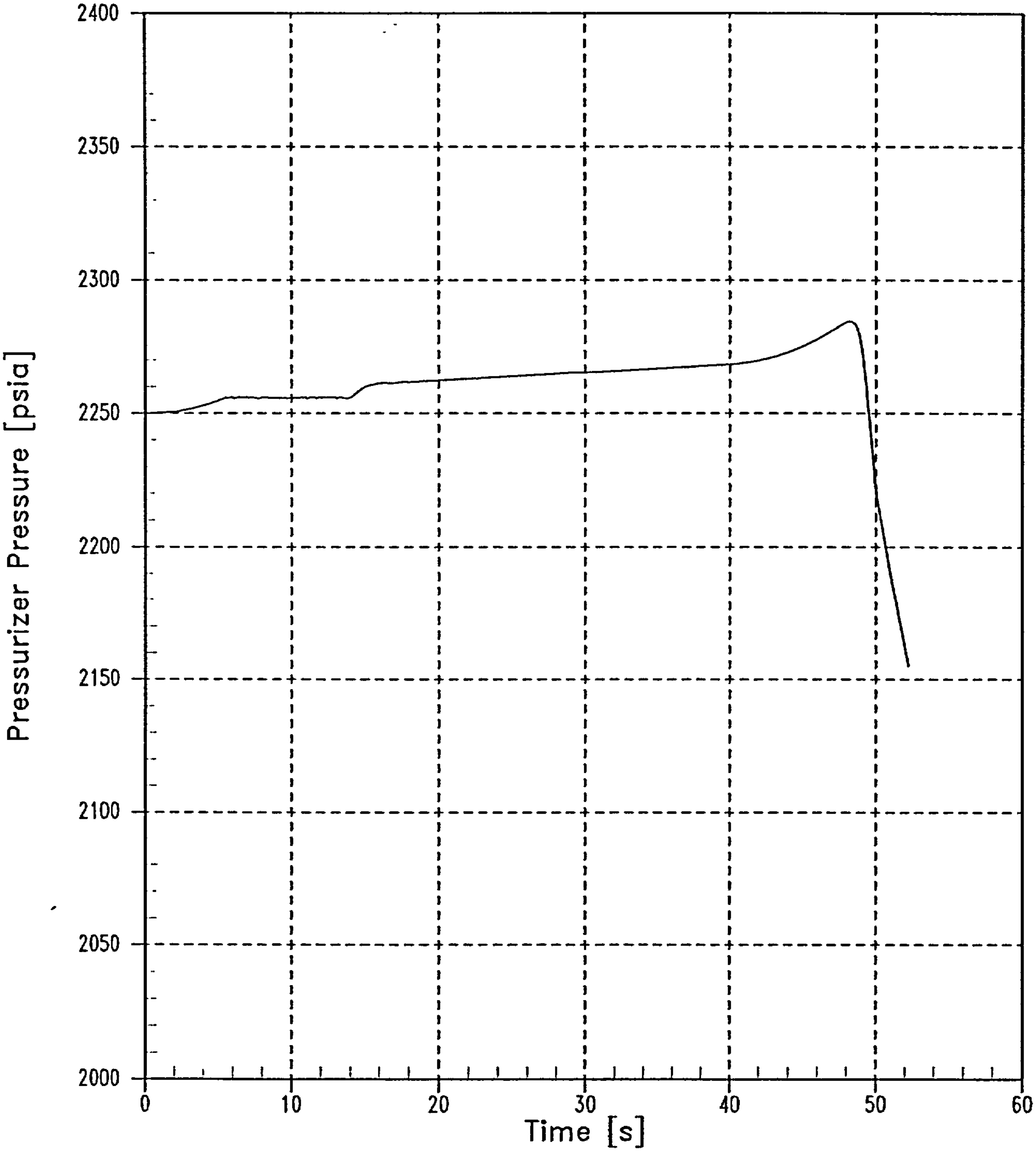

Figure 14.1.2-6

REPLACE WITH THE FIGURE THAT FOLLOWS

Uncontrolled RCCA Withdrawal - Slow Rate 100% Power

Tave vs. Time

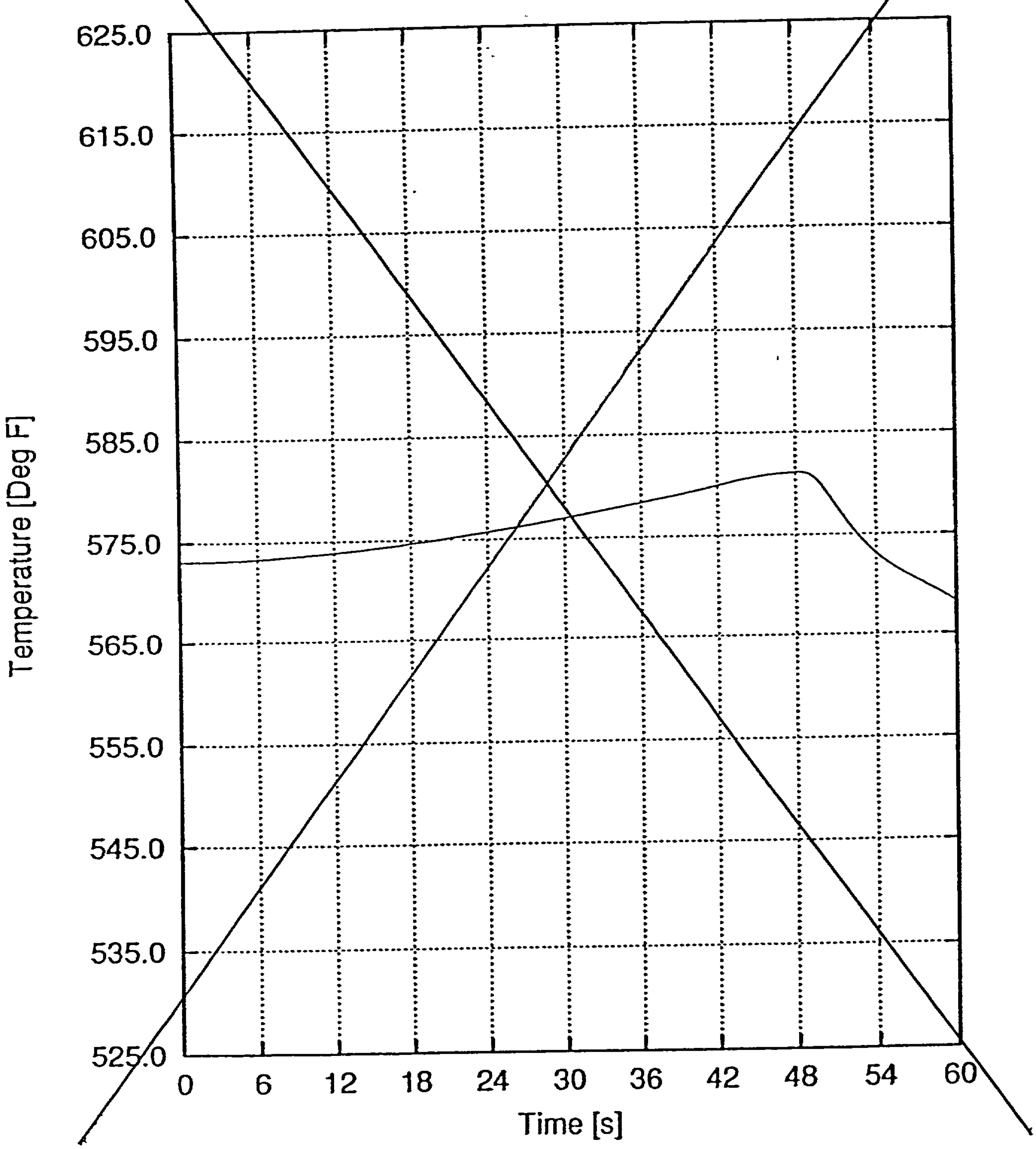

Figure 14.1.2-7

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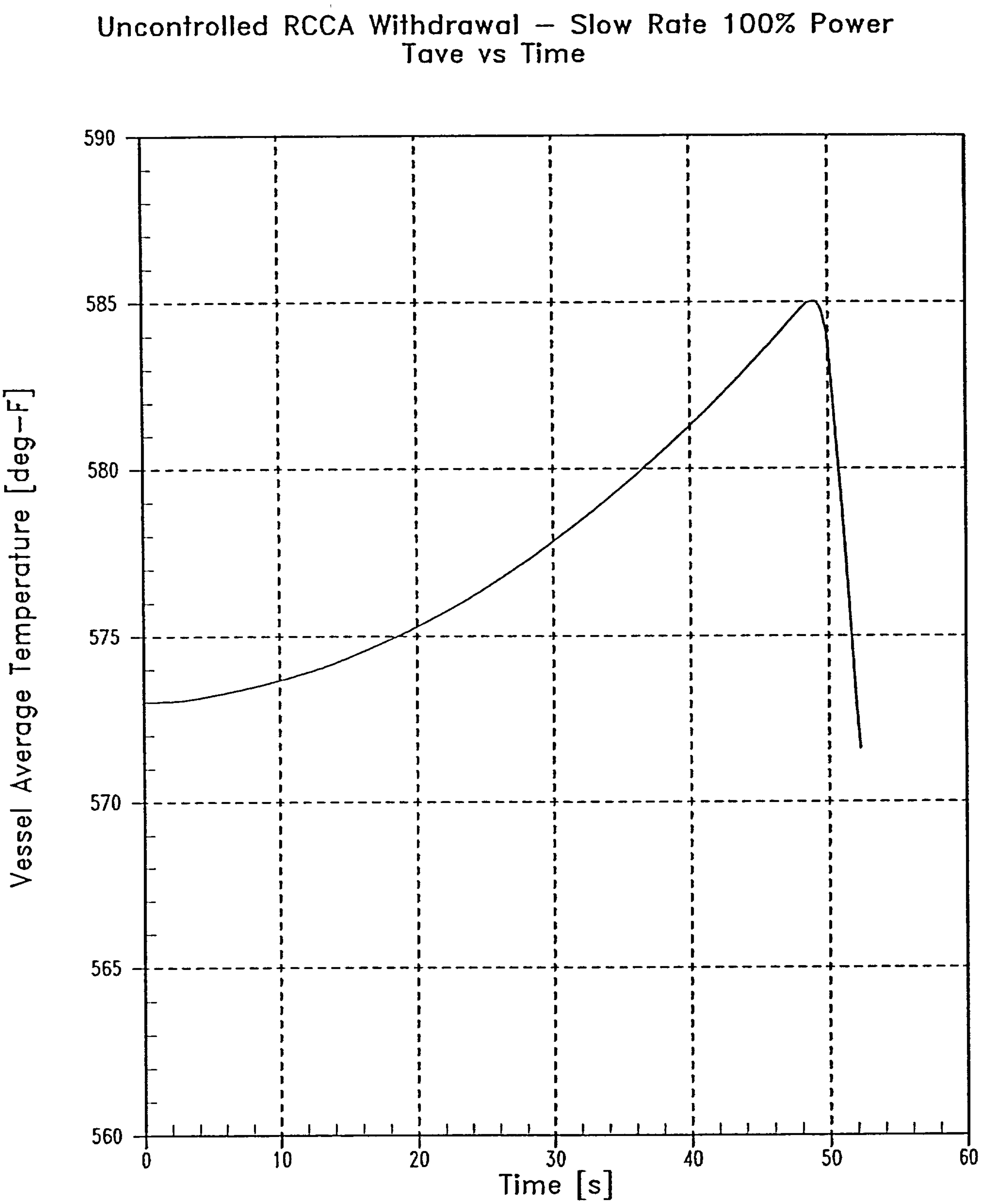

Figure 14.1.2–7

REPLACE WITH THE FIGURE THAT FOLLOWS

Uncontrolled RCCA Withdrawal - Slow Rate 100% Power

Minimum DNBR vs. Time

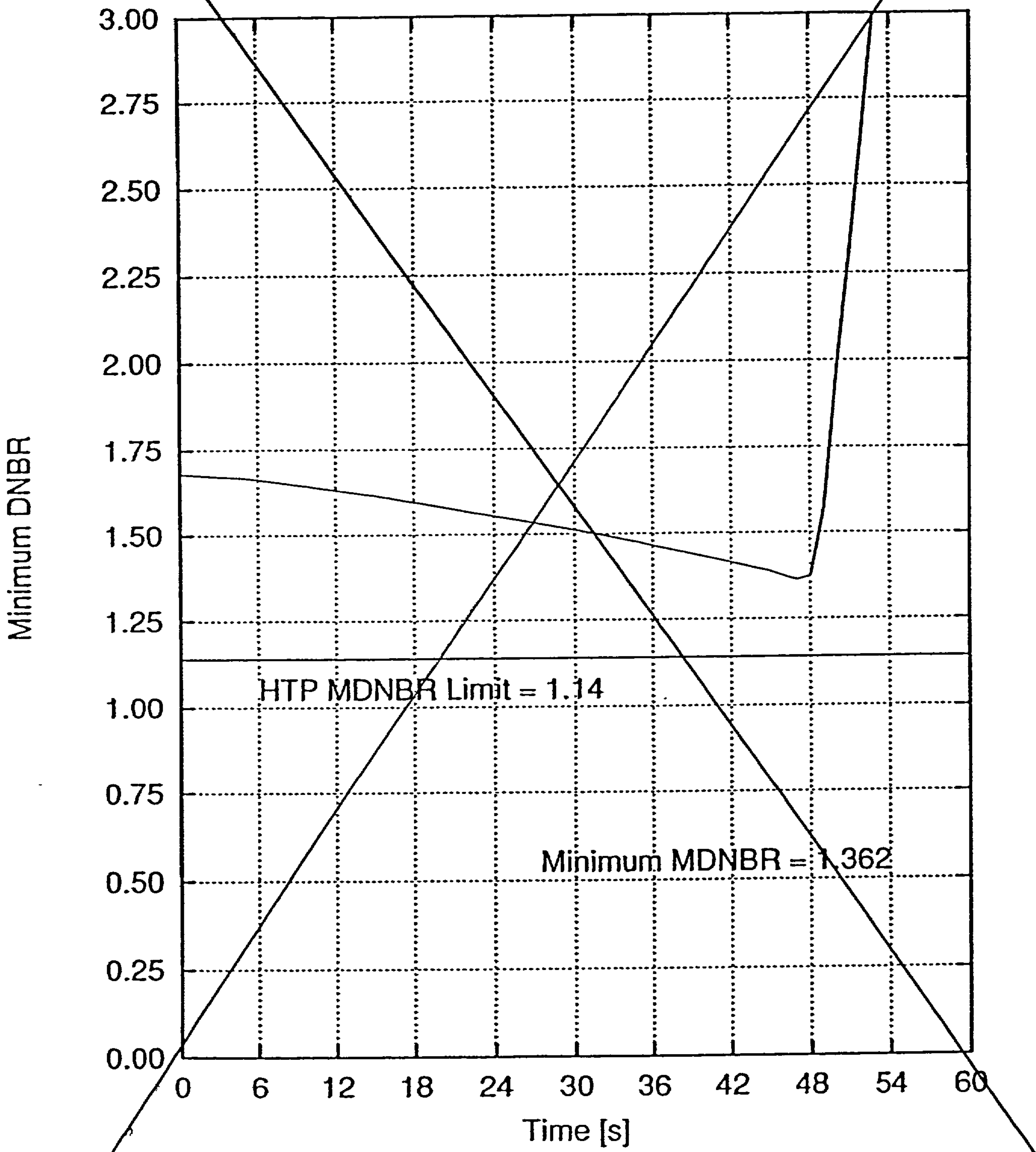

Figure 14.1.2-8

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12/01/2000

Uncontrolled RCCA Withdrawal – Slow Rate 100% Power
Minimum DNBR vs Time

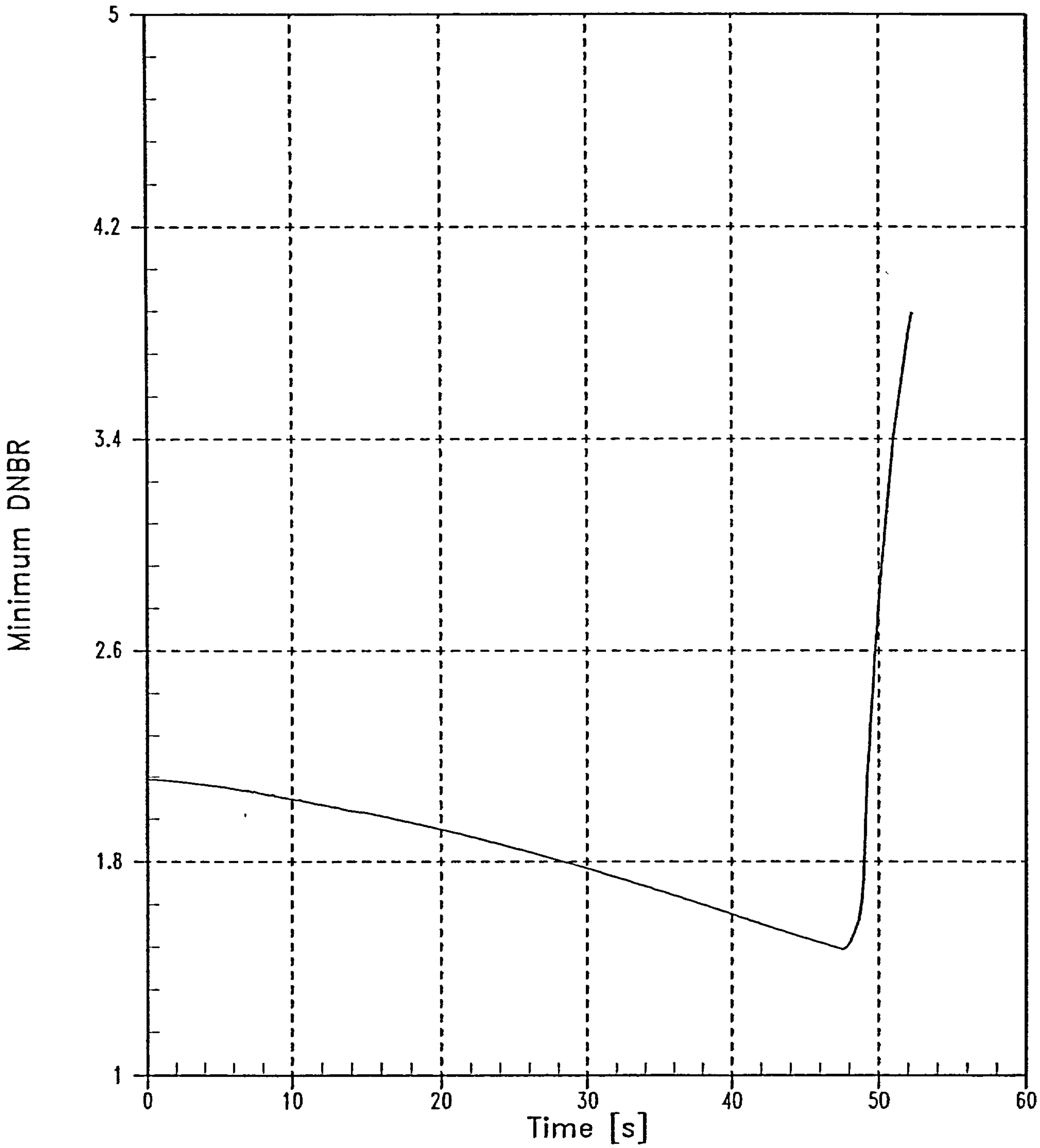

Figure 14.1.2–8

REPLACE WITH THE FIGURE THAT FOLLOWS

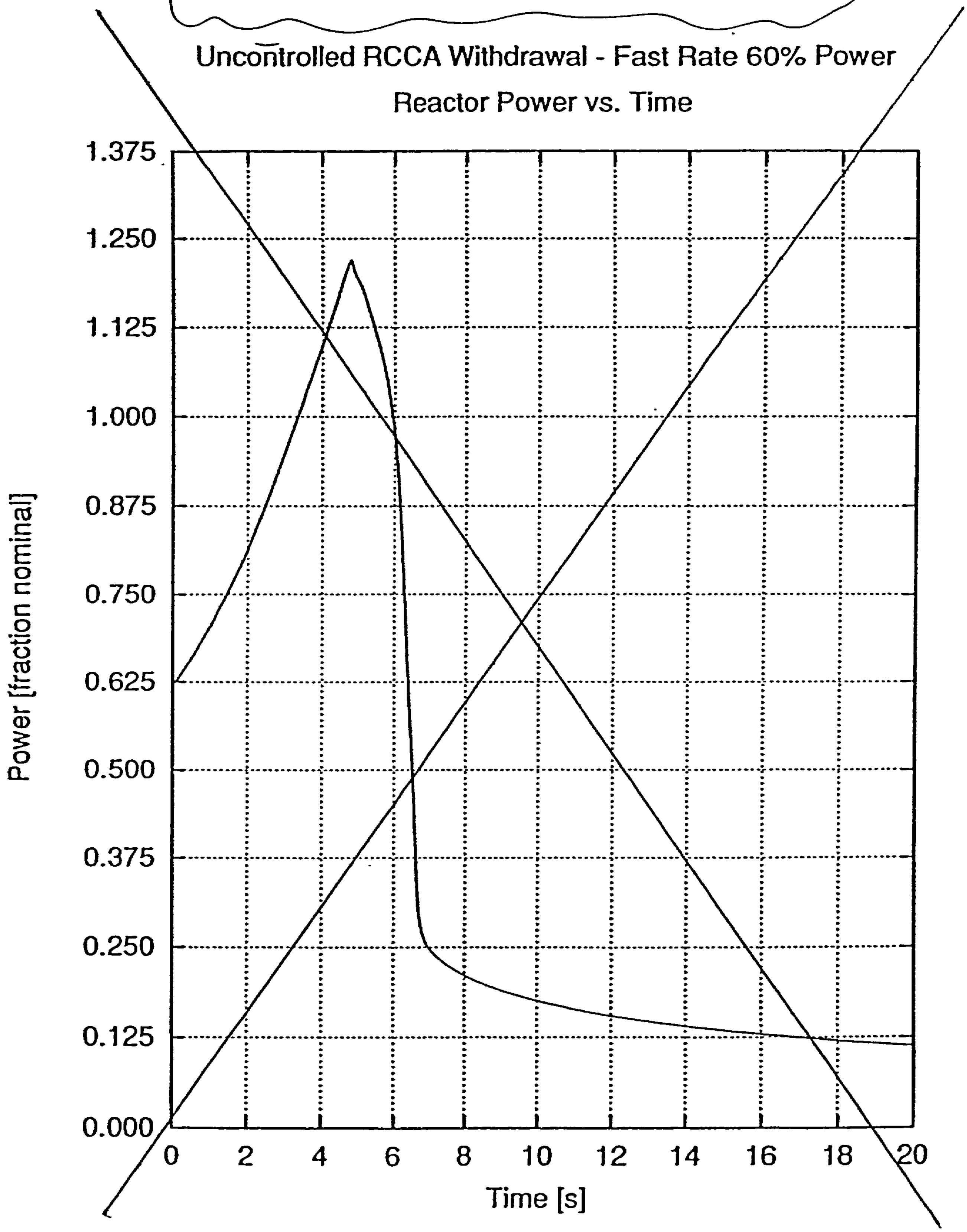

Figure 14.1.2-9

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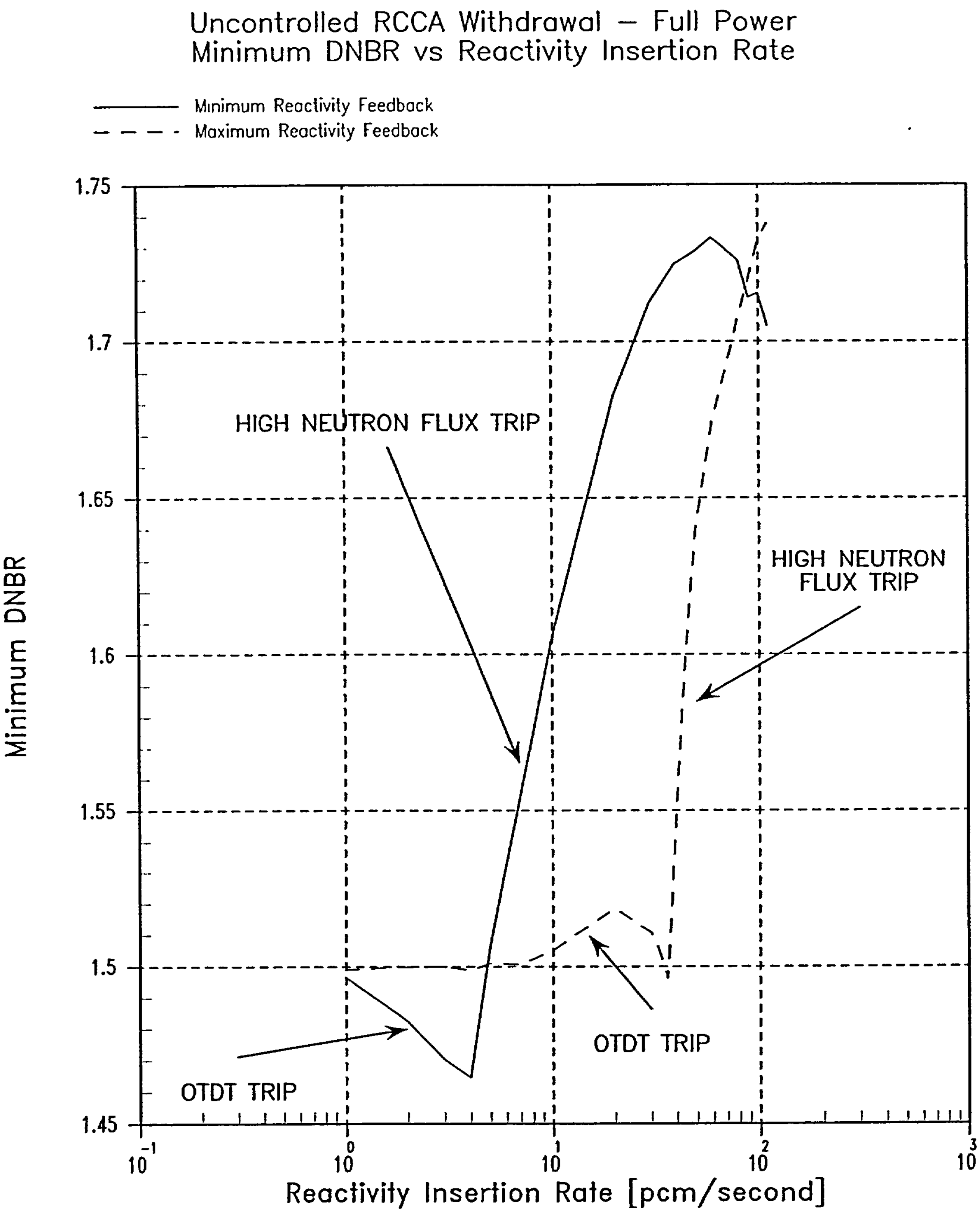

Figure 14.1.2–9

REPLACE WITH THE FIGURE THAT FOLLOWS

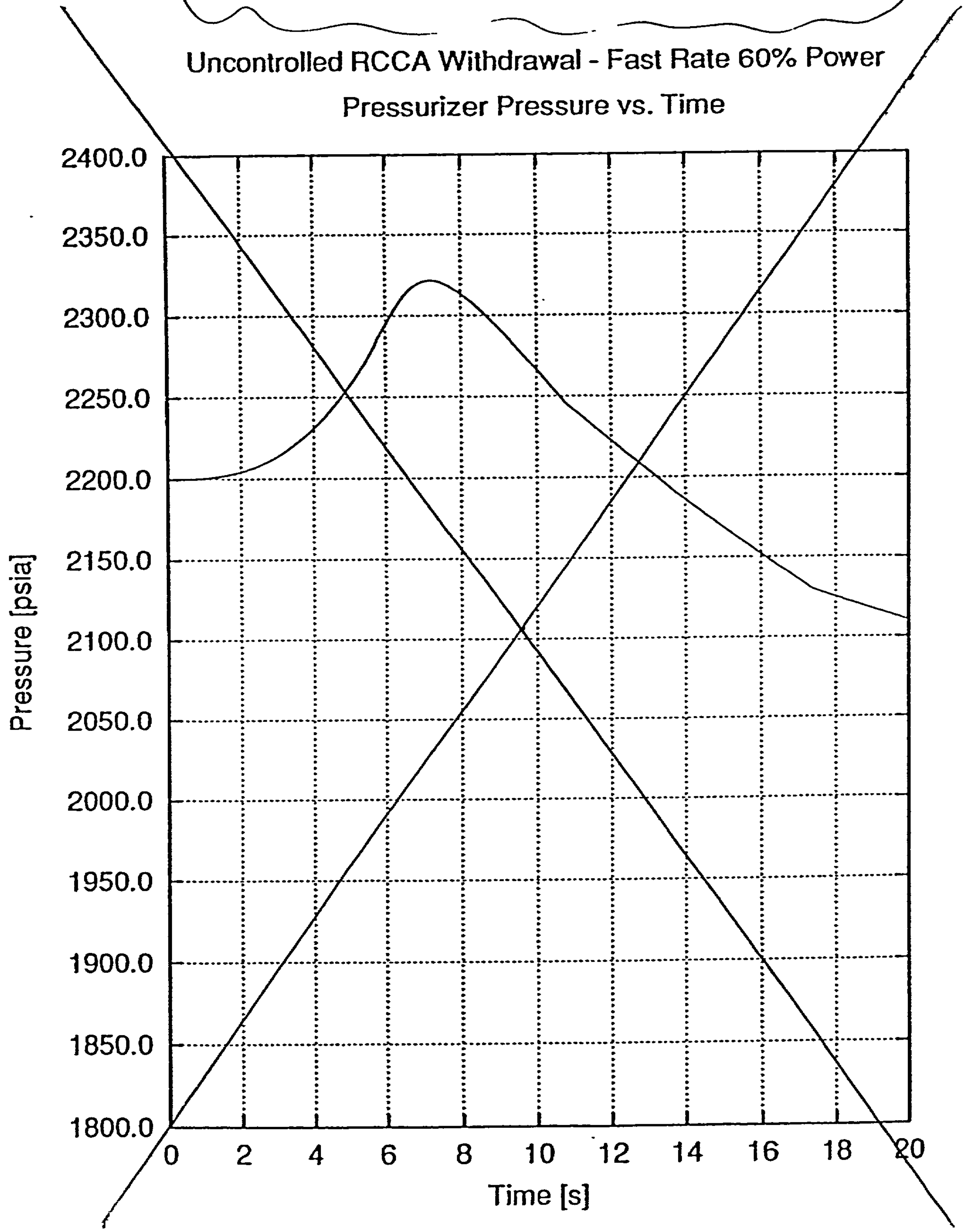

Figure 14.1.2-10

Rev. 16
12/01/2000

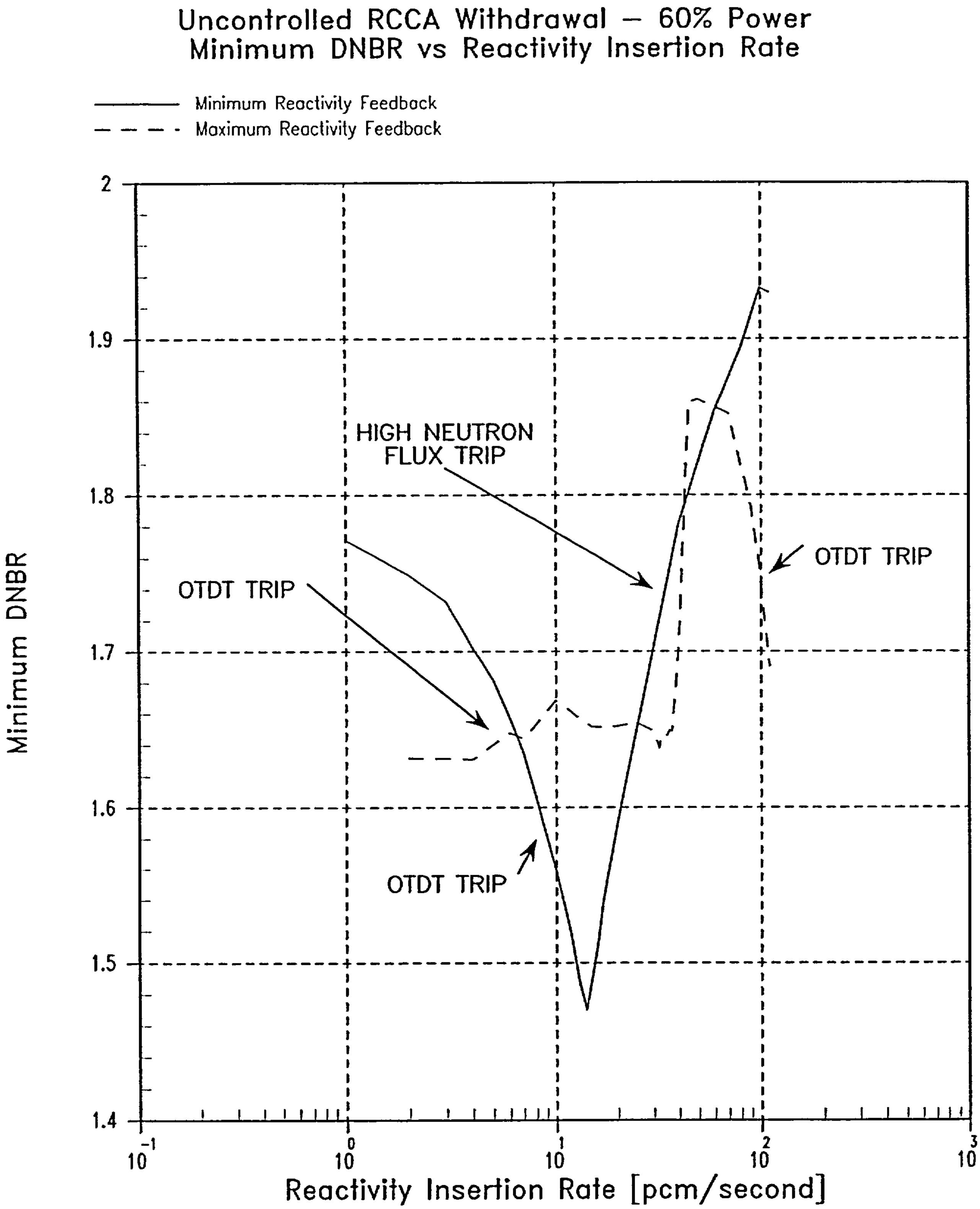

Figure 14.1.2–10

REPLACE WITH THE FIGURE THAT FOLLOWS

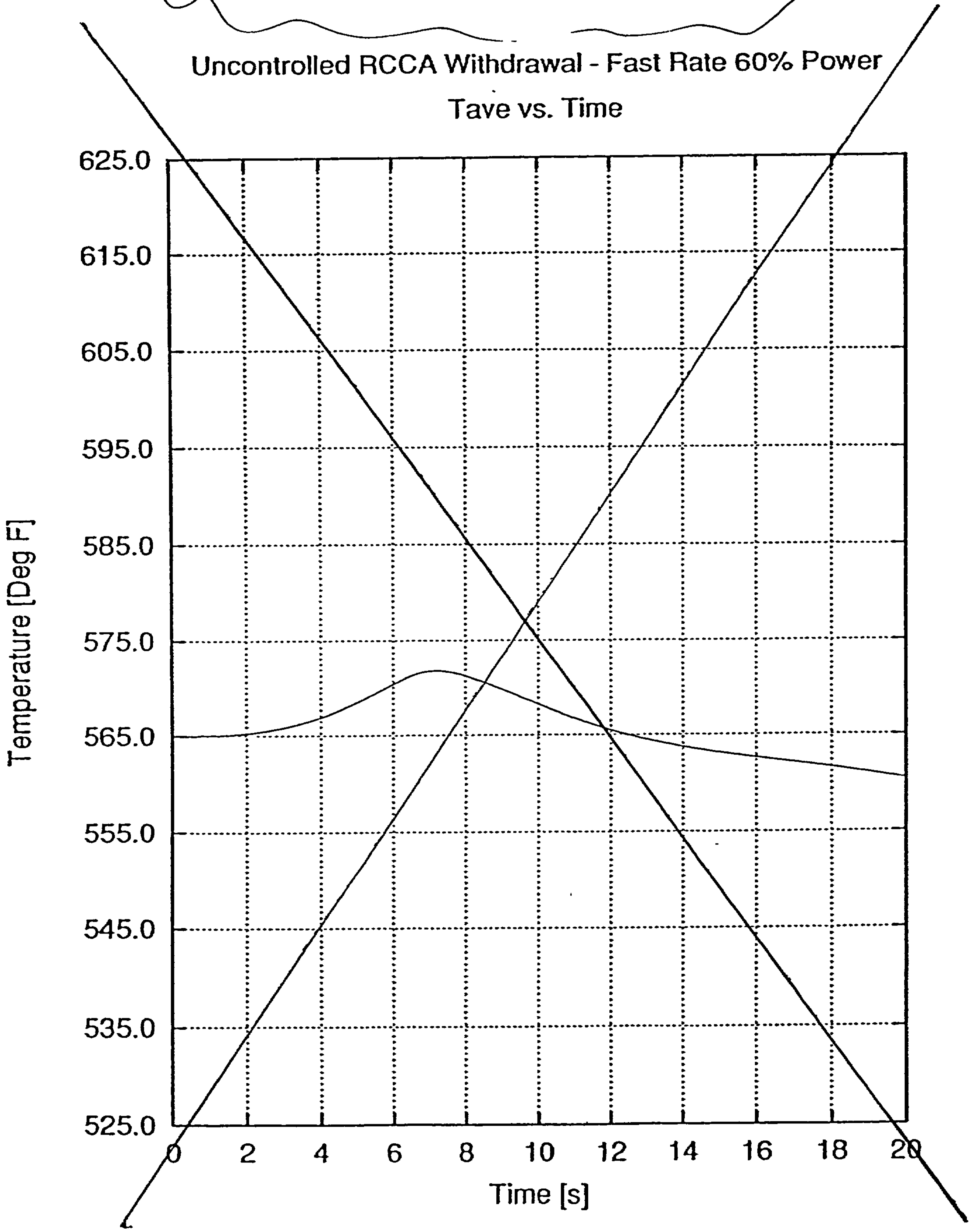

Figure 14.1.2-11

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12/01/2000

Uncontrolled RCCA Withdrawal — 10% Power
Minimum DNBR vs Reactivity Insertion Rate

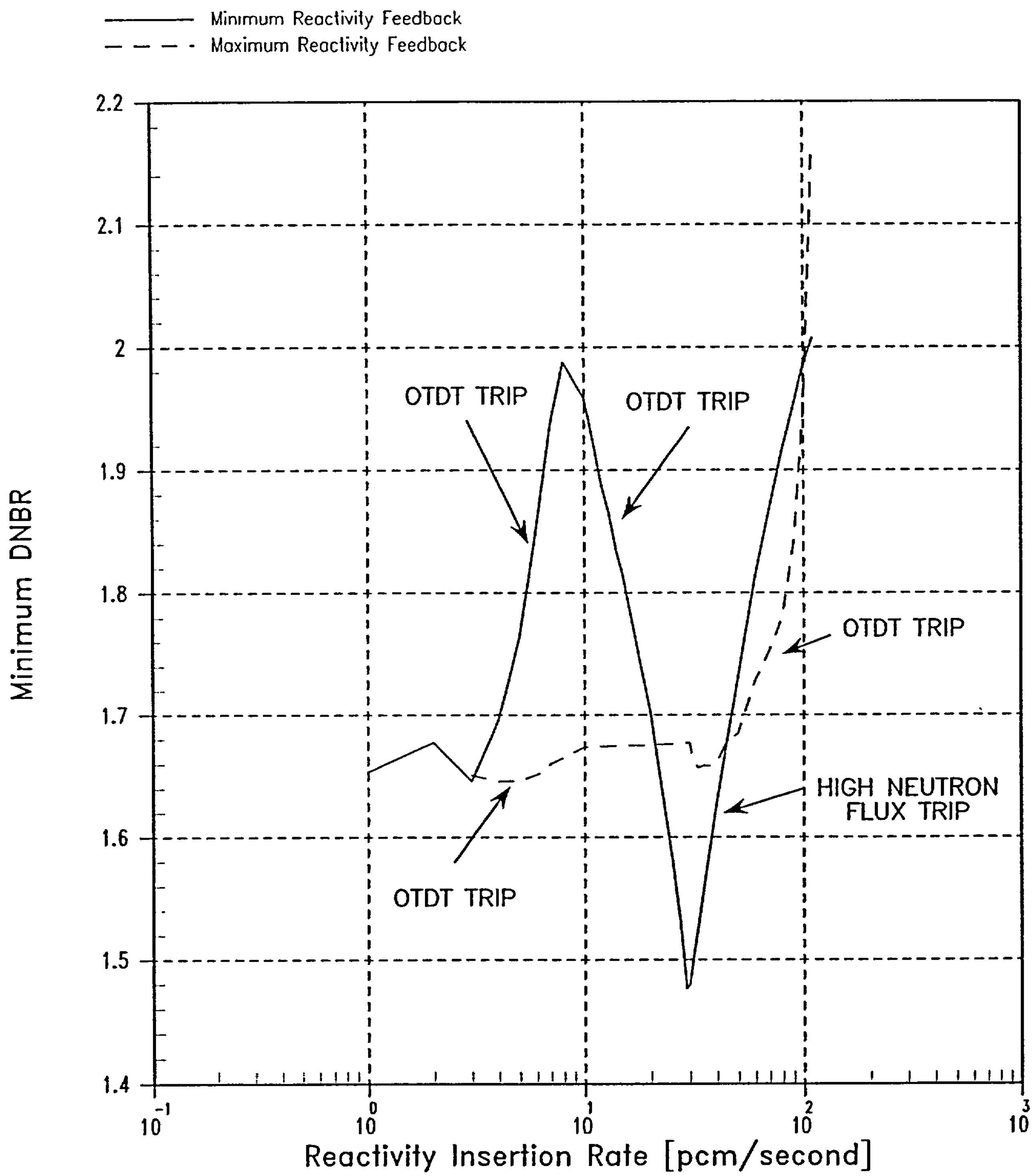

Figure 14.1.2–11

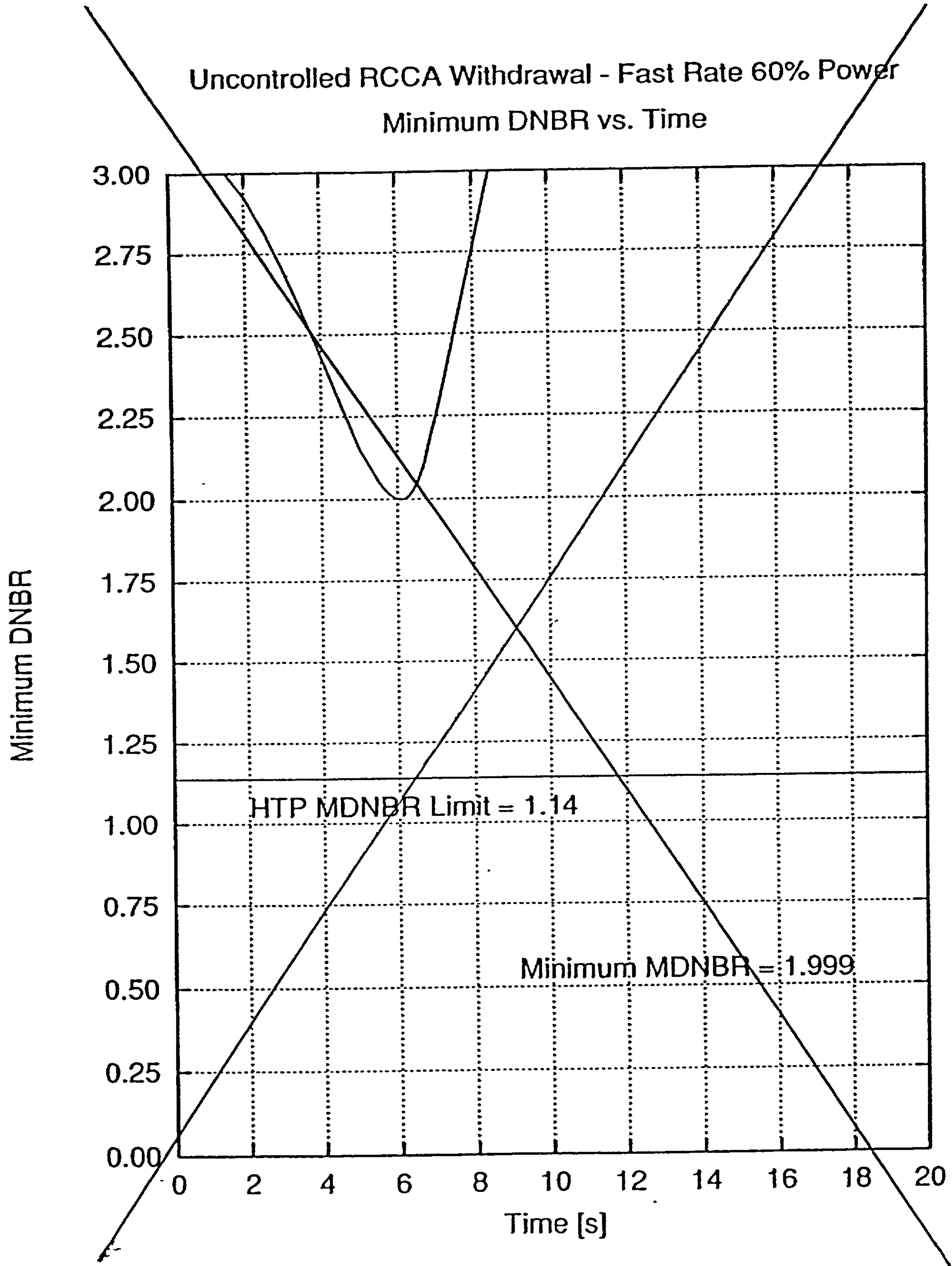

Figure 14.1.2-12

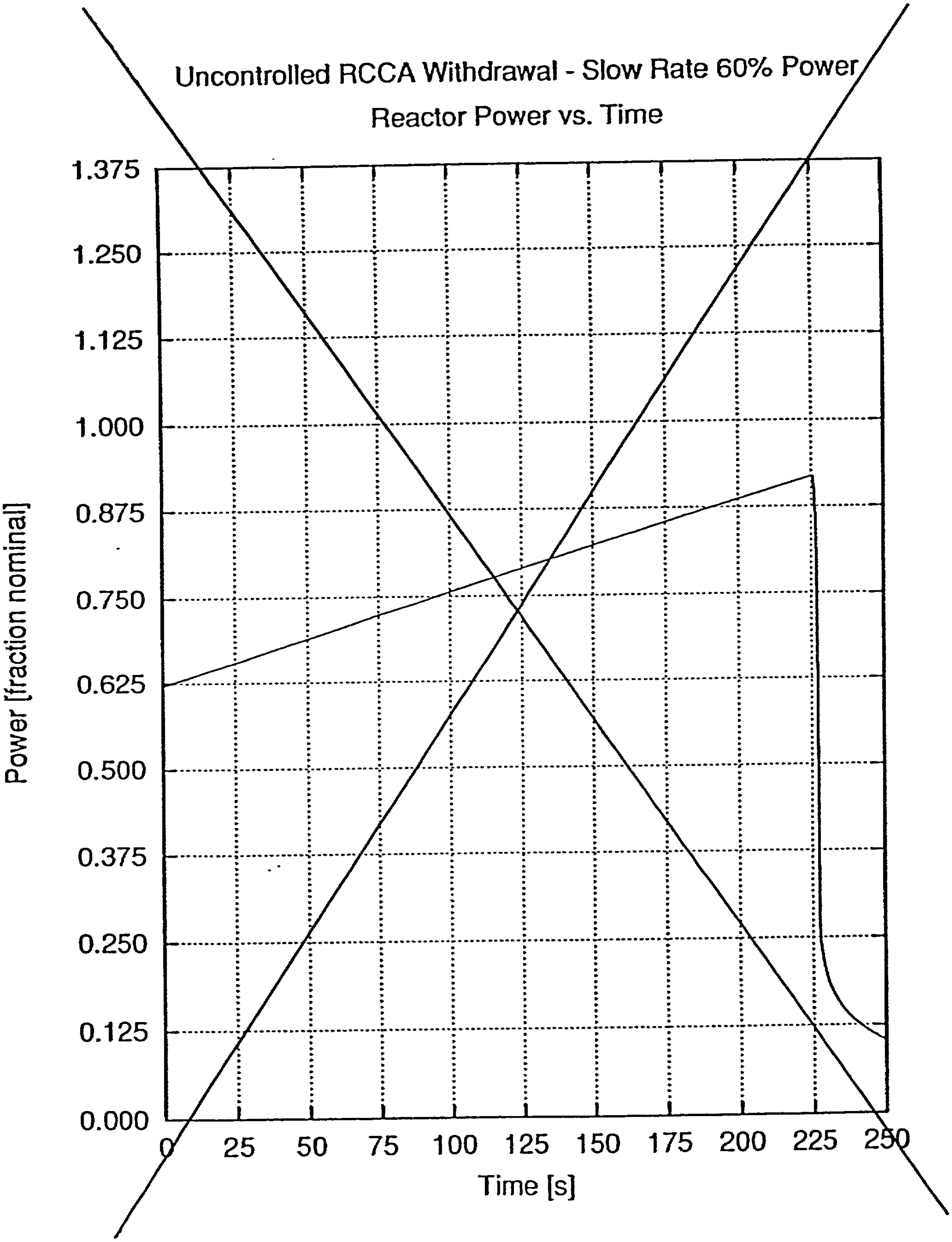

Figure 14.1.2-13

Rev. 16
12/01/2000

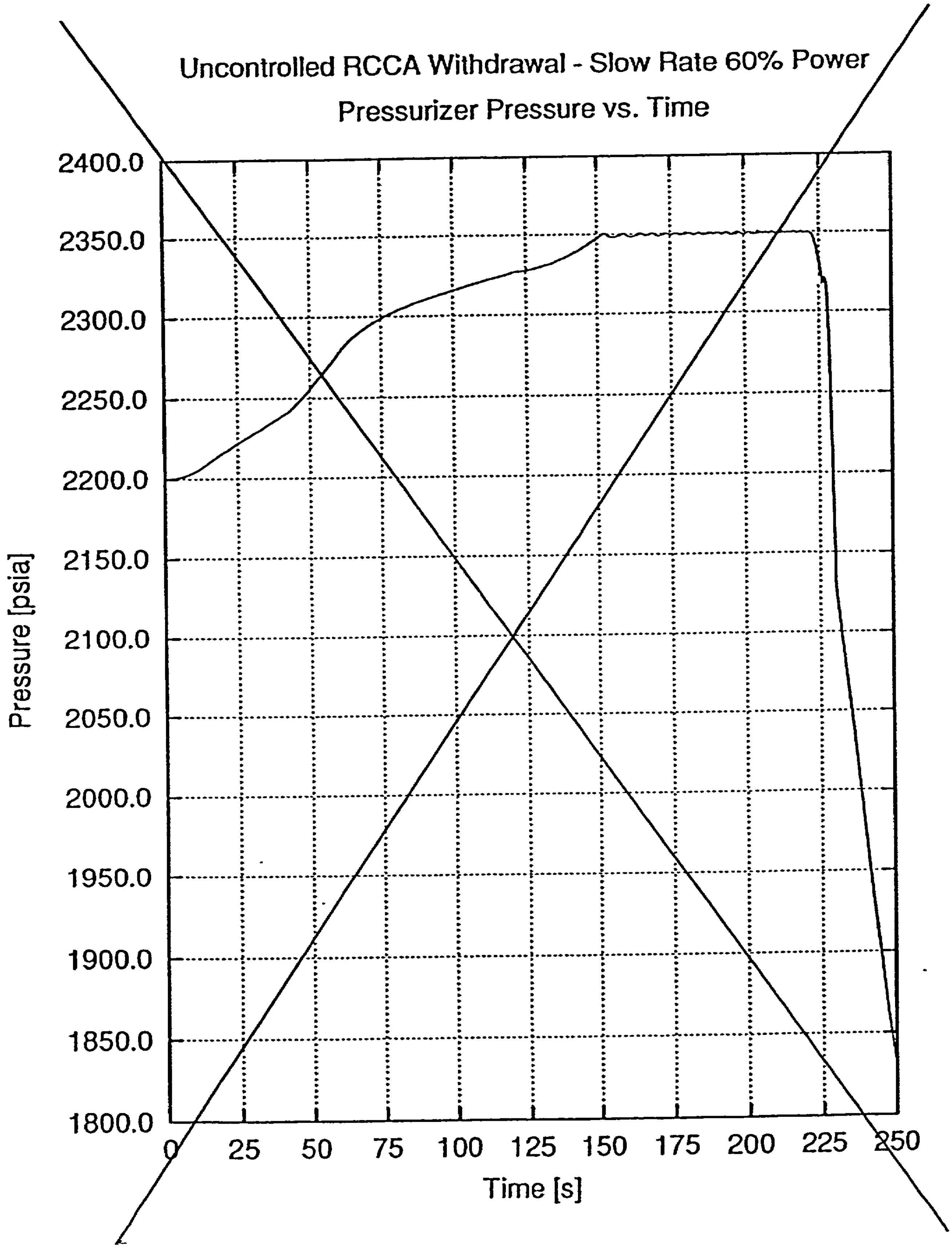

Figure 14.1.2-14

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12/01/2000

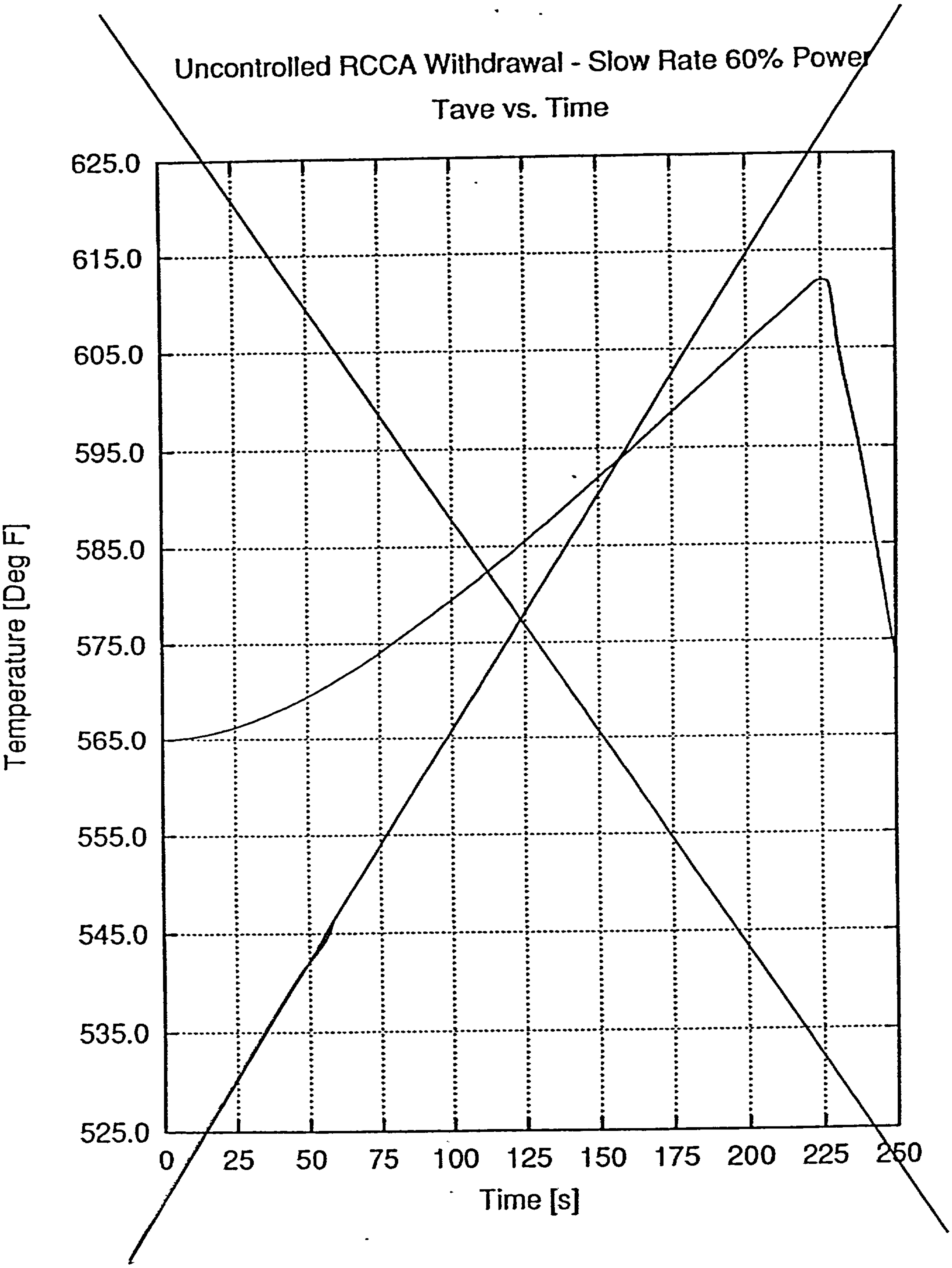

Figure 14.1.2-15

Rev. 16
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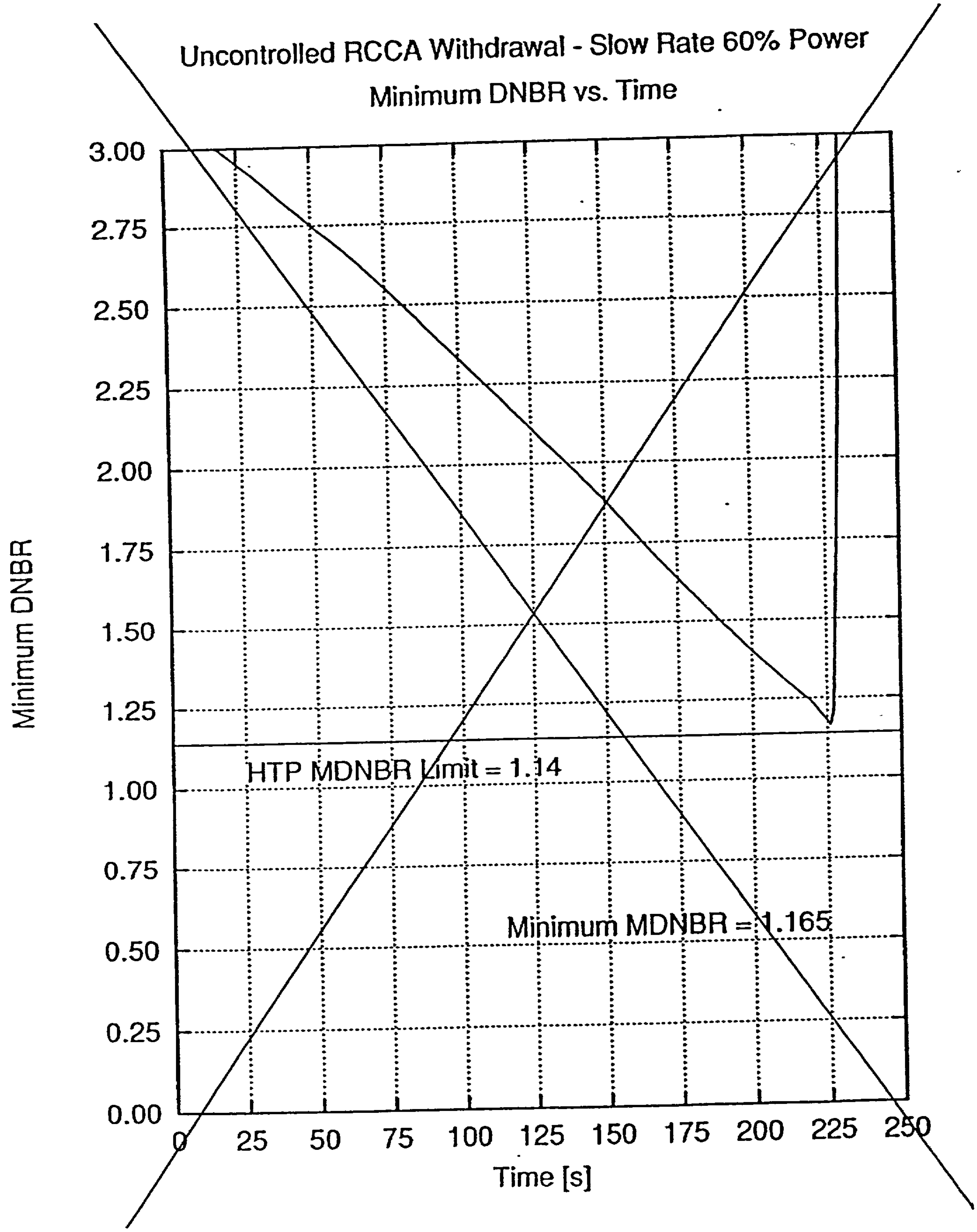

Figure 14.1.2-16

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14.1.3 RCCA MISALIGNMENT

No changes

Accident Description

RCCA misalignment accidents include:

a. dropped full-length RCCAs;

b. dropped full-length RCCA banks; and

c. statically misaligned full-length RCCAs.

Each RCCA has a rod position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by rod bottom lights. Bank demand position is also indicated. The full-length assemblies are always moved in pre-selected banks and the banks are always moved in the same pre-selected sequence.

Dropped assemblies or banks are detected by:

a. sudden drop in the core power level

b. asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

c. rod bottom light(s)

d. rod deviation alarm (if the plant computer is in operation).

Misaligned assemblies are detected by:

a. asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

b. rod deviation alarm (if the plant computer is in operation).

The resolution of the rod position indicator channel is ± 5% of span or 7.2 inches (span equals 12 feet). Deviation of any assembly from its bank by twice this distance, 10% of span, or 14.4 inches, will not cause power distributions worse than the design limits.

If one or more rod position indicator channels is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on ex-core nuclear detectors and/or movable in-core detectors, will be used to verify power distribution symmetry.

Method of Analysis

REPLACE WITH USAR Insert 14.1.3-1

The safety analysis for the RCCA misalignment and RCCA drop accidents involves a full power fuel thermal hydraulic analysis. The peak fuel rod $F_{\Delta H}$ in the analysis is increased to adequately bound the core power distribution anticipated during the steady state RCCA misalignment and RCCA drop core conditions. The reactor is assumed not to trip.

It is necessary to show on a reload cycle specific basis that the worst dropped or misaligned RCCA does not result in a peak fuel rod power ($F_{\Delta H}$) that is greater than the $F_{\Delta H}$ assumed in the safety analysis.

Rod drop in automatic control is not analyzed since restrictions on control rods in automatic control are imposed when reactor power is > 90% and control rods are inserted to < 215 steps. This restriction ensures that the rod drop in automatic control accident is bounded by the static RCCA misalignment described above. Removal of the rod drop in automatic control was determined to be acceptable by the NRC in Reference 1.

Results

REPLACE WITH USAR Insert 14.1.3-2

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Control Rod Drop and Misalignment $F_{\Delta H} = 2.02$	1.142/1.14	2250/2750	750/1210

Conclusions

Dropped or misaligned RCCAs are not deemed to be a hazard to the safe operation of the plant because these events are clearly indicated to the operator, and the analyzed cases of the worst misaligned and dropped rod or bank do not result in a DNBR less than the ~~M~~DNBR limit.

~~For all cases of dropped banks, the reactor is tripped by the power range negative neutron flux rate trip and consequently dropped banks do not cause core damage.~~

14.1.4 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Accident Description

Reactivity can be added to the core with the Chemical and Volume Control System by feeding reactor makeup water into the Reactor Coolant System via the Reactor Makeup Control System. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time.

Section 14.1.4 changes suggested later

USAR Insert 14.1.3-1

For the dropped RCCA(s) event, the transient response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code predicts pertinent plant variables including temperatures, pressures and power level.

Dropped RCCA(s) statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. Using the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, the VIPRE code is used to calculate the minimum DNBR to demonstrate that the DNB design basis is satisfied. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 16.

For the RCCA misalignment event, steady-state power distributions are analyzed using the appropriate nuclear physics computer codes. The peaking factors are then used as input to the VIPRE code to calculate the DNB ratio (DNBR). The following cases are examined in the analysis assuming the reactor is initially at full power: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out. It is assumed that the incident occurs at the time in the cycle at which the maximum all-rods-out $F_{\Delta H}$ occurs. This assures a conservative $F_{\Delta H}$ for the misaligned RCCA configuration.

USAR Insert 14.1.3-2

One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will re-establish power.

Following a dropped rod event in manual rod control mode, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Thus, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller, after which the control system will insert the control bank to restore nominal power. Figures 14.1.3-1 through 14.1.3-4 show a typical transient response to a dropped RCCA event with the reactor in automatic rod control. In all cases, the minimum DNBR remains above the limit value.

Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period because power is decreasing rapidly. The transient will proceed similar to that described previously for the one or more dropped RCCAs scenario, but the return to power will be less due to the greater negative reactivity worth of an entire RCCA bank. The power transient for a dropped RCCA bank is symmetric.

Statically Misaligned RCCA

The most severe RCCA misalignment situations with respect to DNB at significant power levels are associated with cases in which one RCCA is fully inserted with either all rods out or bank D at the insertion limit, or where bank D is inserted to the insertion limit and one RCCA is fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions.

The insertion limits in the Technical Specifications may vary from time-to-time depending on several limiting criteria. The full power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle-to-cycle depending on fuel arrangements.

For each case, DNB does not occur for the RCCA misalignment incident, and thus there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation rate is well below that which would cause fuel melting.

New Figure to be Added

Dropped RCCA

Representative Transient Response – Nuclear Power vs. Time

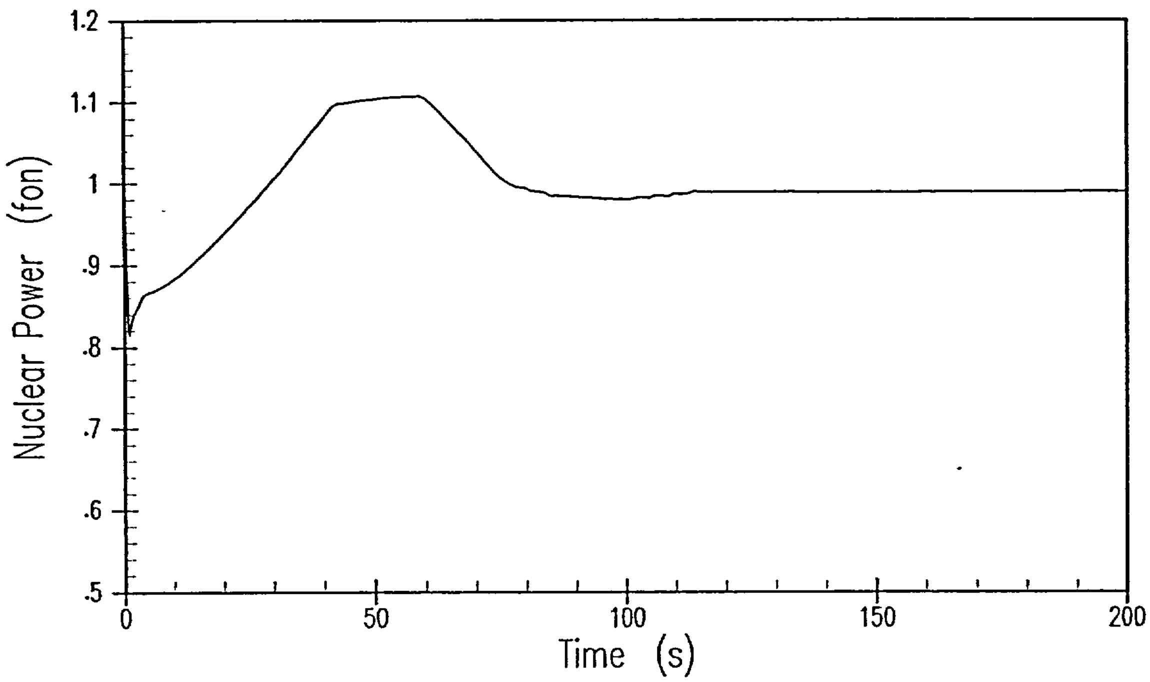

Figure 14.1.3-1

New Figure to be Added

Dropped RCCA

Representative Transient Response – Core Heat Flux vs. Time

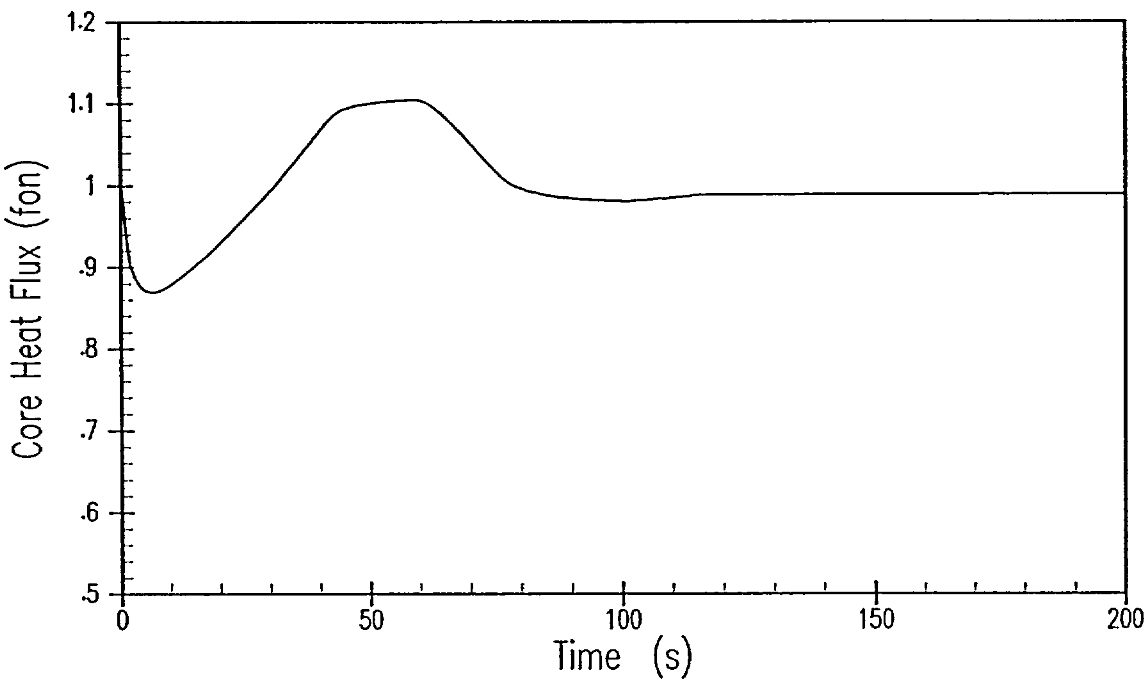

Figure 14.1.3-2

New Figure to be Added

Dropped RCCA

Representative Transient Response – Pressurizer Pressure vs. Time

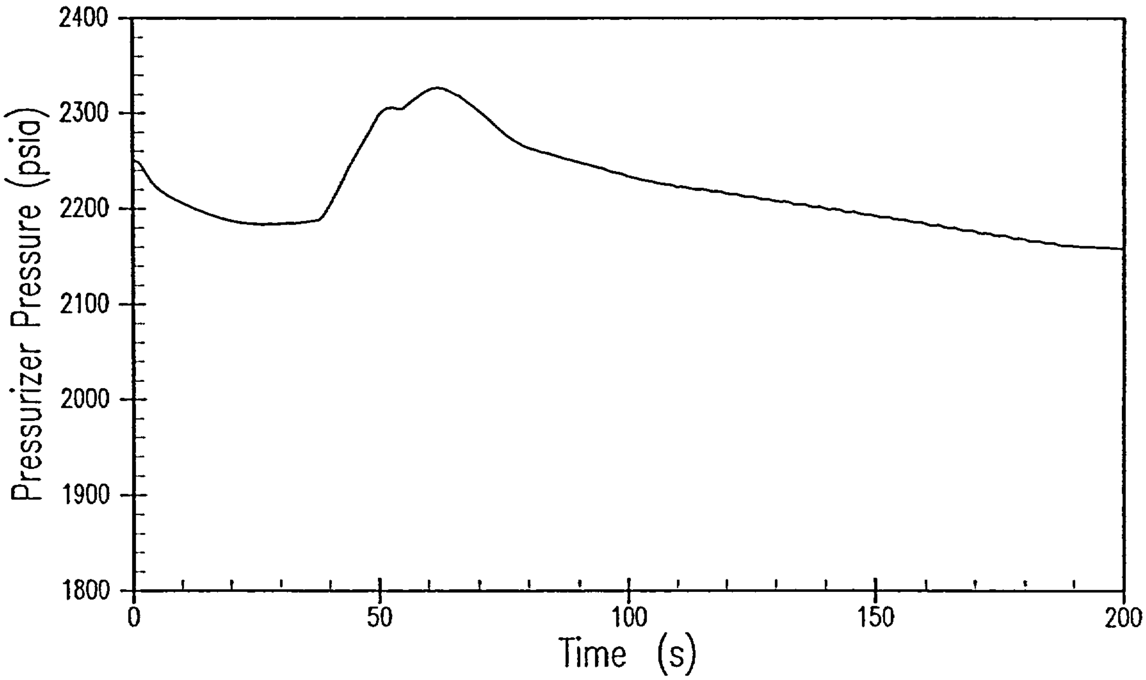

Figure 14.1.3-3

New Figure to be Added

Dropped RCCA

Representative Transient Response – Vessel Average Temperature vs. Time

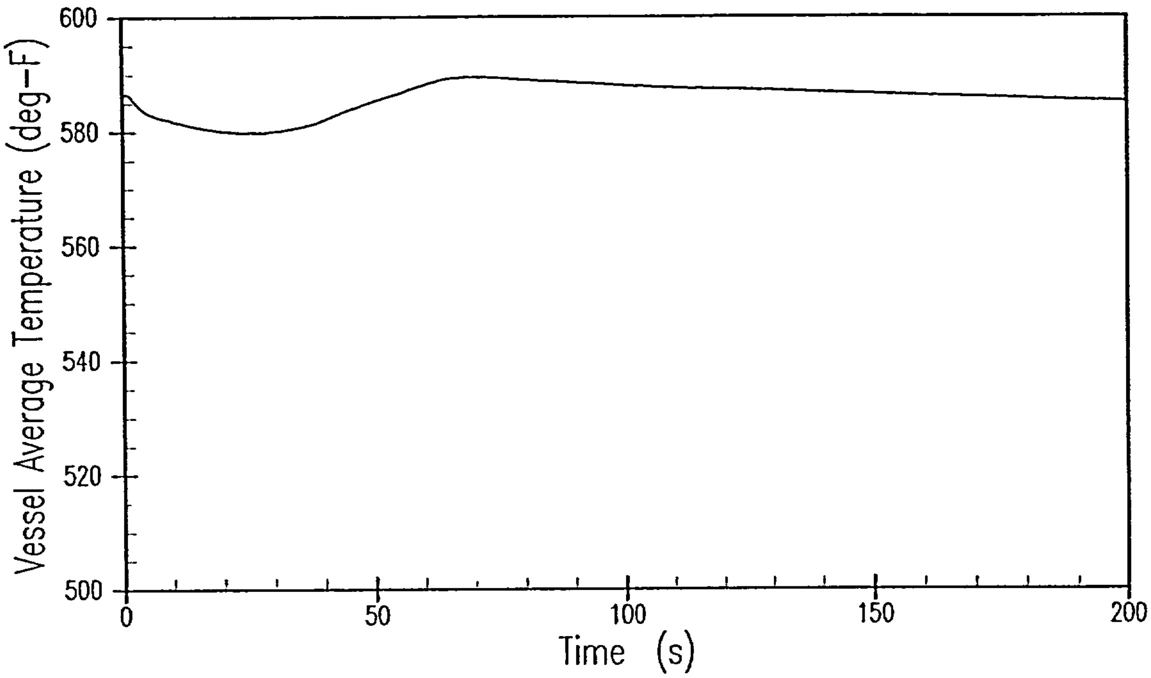

Figure 14.1.3-4

Method of Analysis

The safety analysis for the RCCA misalignment and RCCA drop accidents involves a full power fuel thermal hydraulic analysis. The peak fuel rod $F_{\Delta H}$ in the analysis is increased to adequately bound the core power distribution anticipated during the steady state RCCA misalignment and RCCA drop core conditions. The reactor is assumed not to trip.

It is necessary to show on a reload cycle specific basis that the worst dropped or misaligned RCCA does not result in a peak fuel rod power ($F_{\Delta H}$) that is greater than the $F_{\Delta H}$ assumed in the safety analysis.

Rod drop in automatic control is not analyzed since restrictions on control rods in automatic control are imposed when reactor power is > 90% and control rods are inserted to < 215 steps. This restriction ensures that the rod drop in automatic control accident is bounded by the static RCCA misalignment described above. Removal of the rod drop in automatic control was determined to be acceptable by the NRC in Reference 1.

Section 14.1.3 Changes suggested earlier.

Results

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Control Rod Drop and Misalignment $F_{\Delta H}$ = 2.02	1.142/1.14	2250/2750	750/1210

Conclusions

Dropped or misaligned RCCAs are not deemed to be a hazard to the safe operation of the plant because these events are clearly indicated to the operator, and the analyzed cases of the worst misaligned and dropped rod do not result in a DNBR less than the MDNBR limit.

For all cases of dropped banks, the reactor is tripped by the power range negative neutron flux rate trip and consequently dropped banks do not cause core damage.

14.1.4 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Accident Description

Reactivity can be added to the core with the Chemical and Volume Control System by feeding reactor makeup water into the Reactor Coolant System via the Reactor Makeup Control System. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time.

The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value that after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The source of reactor makeup water for the Reactor Coolant System is the reactor makeup water storage tanks. Inadvertent dilution can be readily terminated by isolating this source. The operation of the reactor makeup water pumps, which take suction from these tanks, provides the only supply of makeup water to the Reactor Coolant System. In order for makeup to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps.

There are three positive displacement variable speed drive charging pumps, manually or automatically controlled. When in automatic, each is provided with a high and low speed alarm. However, only one of them is automatically controlled at any one time, as dictated by procedure.

The rate of addition of unborated makeup water to the reactor coolant system is limited by the capacity of the charging pumps and by the capacity of the control valve between the two makeup water pumps and the three charging pumps. The maximum dilution flow (80 gpm) occurs with two charging pumps operating and three letdown orifices in-service. ~~For the purpose of this analysis, a larger, unrealistic flow rate (180 gpm) is used, corresponding to all three charging pumps operating at full flow.~~ During normal operation, two charging pumps are operated; one in manual and one in automatic control. The speed of the pump selected for automatic control is controlled by the pressurizer level error signal. During load changes the pressurizer level set point varies automatically with T_{avg} such that the charging pump speed remains relatively constant. For this analysis the maximum dilution flow rate corresponding to 80gpm is used.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the Reactor Makeup Controller. Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the control switch must be actuated. Omitting either step would prevent dilution. This makes the probability of inadvertent dilution very small.

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

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

Method of Analysis and Results

Dilution During Refueling

During refueling the following conditions exist:

a. One residual heat removal pump is running to ensure continuous mixing in the reactor vessel,

b. The valve in the seal water header to the reactor coolant pumps is closed,

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

d. The boron concentration of the refueling water is a minimum of 2250 [redacted] ppm, corresponding to a shutdown of at least 5% Δk/k with all control rods in; periodic sampling ensures that this concentration is maintained,

e. The source range detectors outside the reactor vessel are active and provide an audible count rate.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High-count rate is alarmed in the reactor containment and the main control room. The count rate increase is proportional to the inverse core multiplication factor. Assuming the reactor is 5% shutdown at the required refueling boron concentration of 2250 [redacted] ppm, the time to reach critical conditions is > 30 minutes. This is ample time for the operator to recognize the audible high-count rate signal and isolate the reactor makeup water source by closing valves and stopping the reactor makeup water pumps.

Dilution During Startup

During startup the following are assumed for a boron dilution event:

- Core monitoring of neutron flux is provided by the excore detectors.
- Reactor coolant is mixed by operation of the reactor coolant pumps.
- TWO ~~Three~~ charging pumps are running, delivering a maximum dilution flow rate of 80 ~~180~~ gpm.
- The boron endpoint with all rods inserted is ~~1300~~ 1600 ppm.
- Initial reactor boron concentration is [redacted] 1800 ppm.

An evaluation of the reactor shows that the minimum time required to reduce the reactor coolant boron concentration to a concentration at which the reactor could go critical with all RCCAs in is > 15 minutes. This provides adequate time for the operator to respond to the high-count rate signal and terminate dilution flow.

Dilution at Power

two

70

The reactivity addition rate [redacted] a boron dilution flow of ~~180~~ gpm at full power [redacted] all ~~three~~ charging pumps running[redacted] This is a ~~conservatively~~ high [redacted] reactivity insertion rate for the assumed at power boron concentration of ~~1600~~ ppm.

initial 1780

With the reactor in automatic control, at full power, the power and temperature increase from the boron dilution results in the insertion of the controlling RCCA bank and a decrease in shutdown margin. A continuation of the dilution and RCCA insertion would cause the rods to reach the lower limit of the maneuvering band. Before reaching this point, however, two alarms would be actuated to warn the operator of the potential accident condition. These two alarms, the low RCCA insertion limit alarm and the low-low RCCA insertion limit alarm, alert the operator to initiate normal boration. Insert 1

With no boration, the required shutdown margin is maintained for at least 10 minutes during a continuous boron dilution. Therefore, ample time is available following the alarms for the operator to determine the cause, isolate the reactor water makeup source, and initiate reboration.

If rod control is in manual, and the operator takes no action, the power rises to the high neutron flux trip setpoint and the reactor trips. Figures 14.1.4-1 through 14.1.4-5 show the response of nuclear power, pressure, coolant average temperature, heat flux, and DNBR to a boron dilution event in manual control. The boron dilution in this case is essentially identical to a rod withdrawal accident. The reactivity insertion rate due to the boron dilution is within the range of reactivity insertion rates considered in Section 14.1.2 - Uncontrolled RCCA Withdrawal at Power. Assuming a 1% shutdown margin, there is ample time available for the operator to terminate the dilution before the reactor can return to criticality following the trip.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Chemical & Volume Control System Malfunction	1.347/1.14	[redacted]/2750	[redacted]/1210

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition rate due to the dilution is slow enough to allow the operator adequate time to determine the cause of the dilution and take corrective action before required shutdown margin is lost. ~~The dilution event at power is shown to have adequate margin to the MDNBR limit.~~

Insert 1 for FSAR Section 14.1.4:

The low alarm is set sufficiently above the low-low alarm to allow normal boration without the need for emergency procedures. If dilution continues after reaching the low-low alarm, it takes approximately 37.58 minutes before the total shutdown margin is lost due to dilution. Adequate time is therefore available following the alarms for the operator to determine the cause, isolate the reactor makeup water source, and initiate reboration.

With the reactor in manual control, if no operator action is taken, the power and temperature rise causes the reactor to reach the OTΔT trip setpoint. The boron dilution accident in this case is essentially identical to a rod cluster control assembly withdrawal accident at power. Prior to the OTΔT trip, an OTΔT alarm and turbine runback would be actuated. There is time available (~ 34.76 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the reactor makeup water source, and initiate reboration before the reactor can return to criticality.

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CVCS Malfunction - Dilution at Power - Manual Control

Reactor Power vs. Time

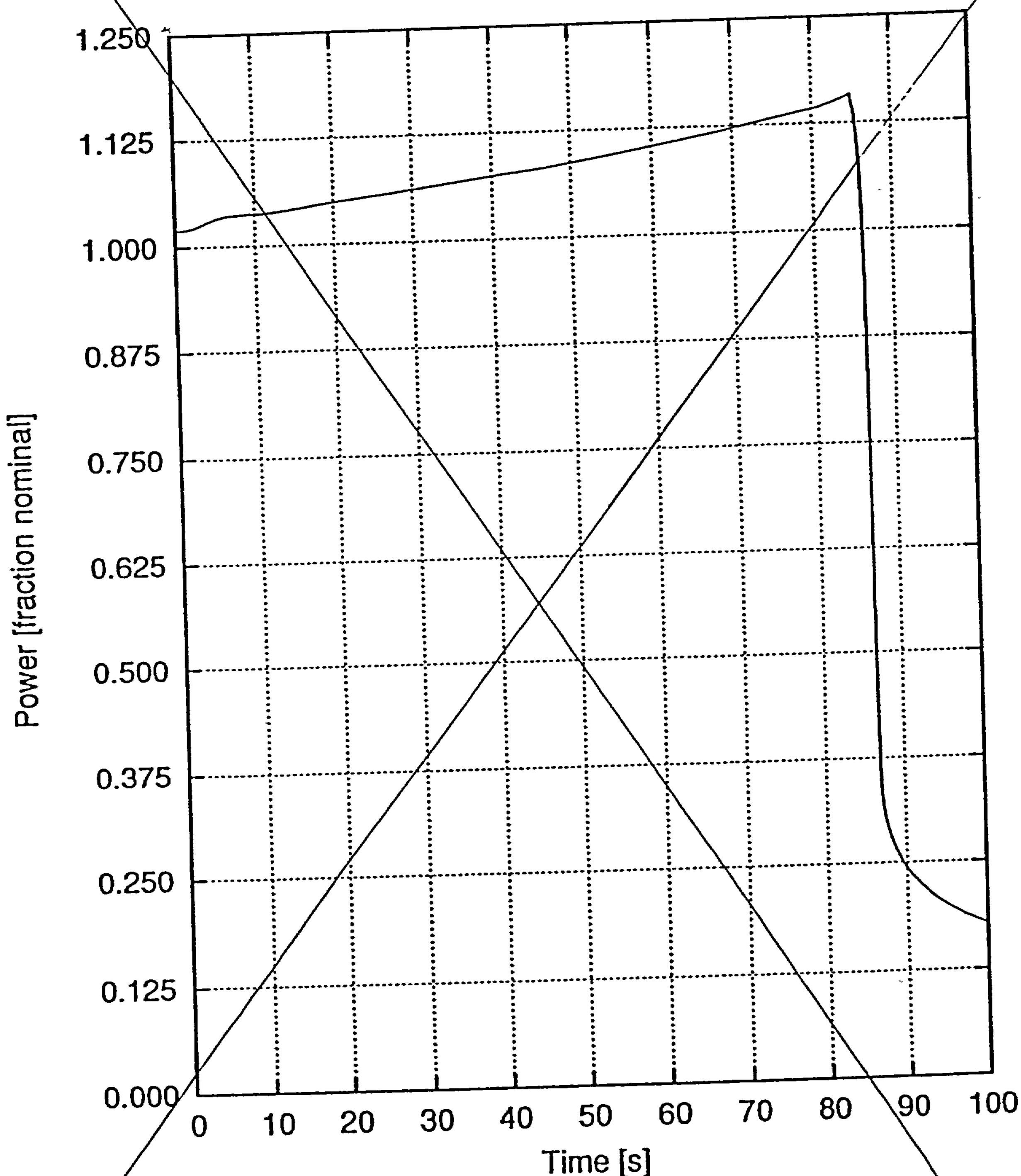

Figure 14.1.4-1

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CVCS Malfunction - Dilution at Power - Manual Control

Pressurizer Pressure vs. Time

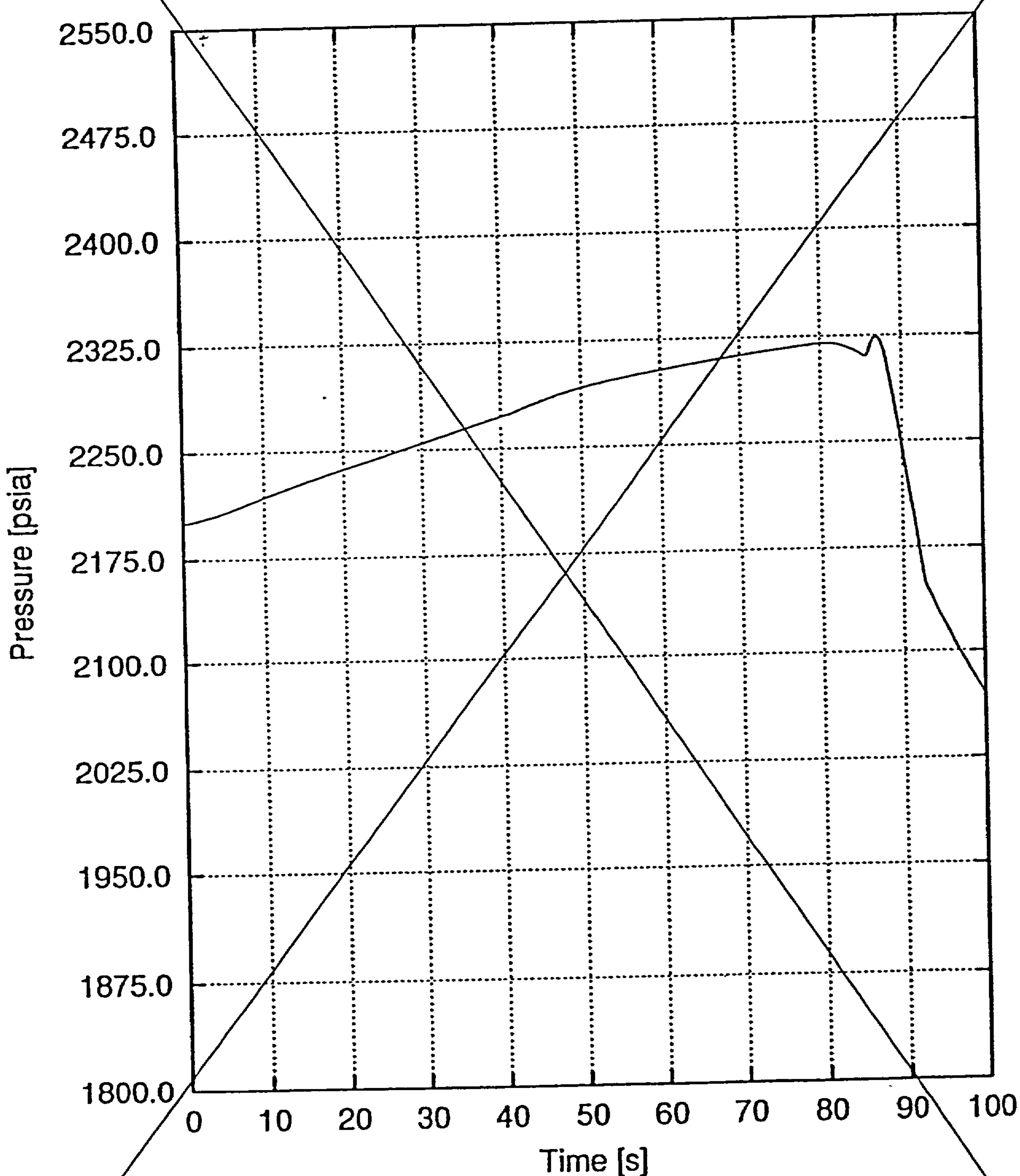

Figure 14.1.4-2

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CVCS Malfunction - Dilution at Power - Manual Control

Tave vs. Time

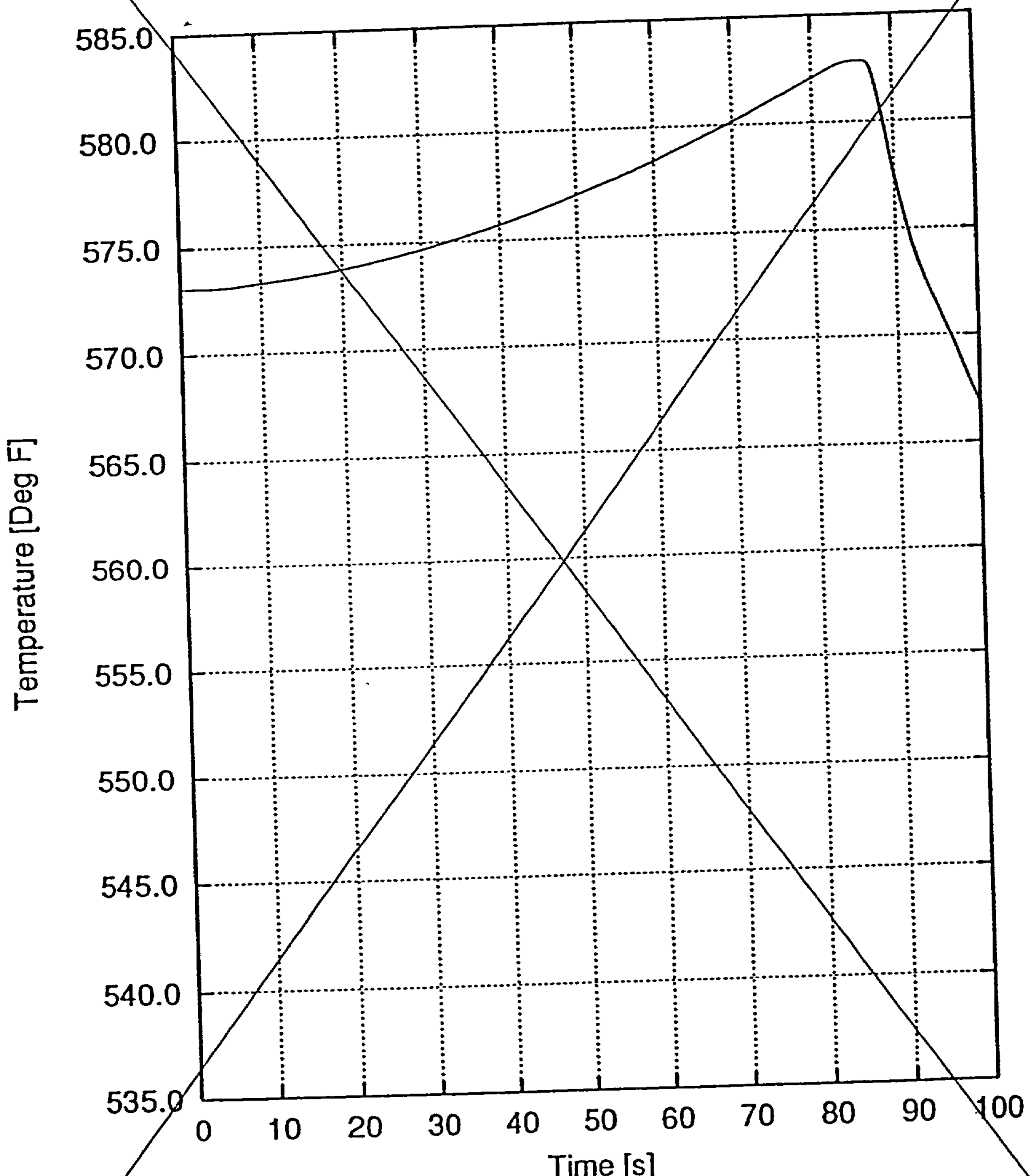

Figure 14.1.4-3

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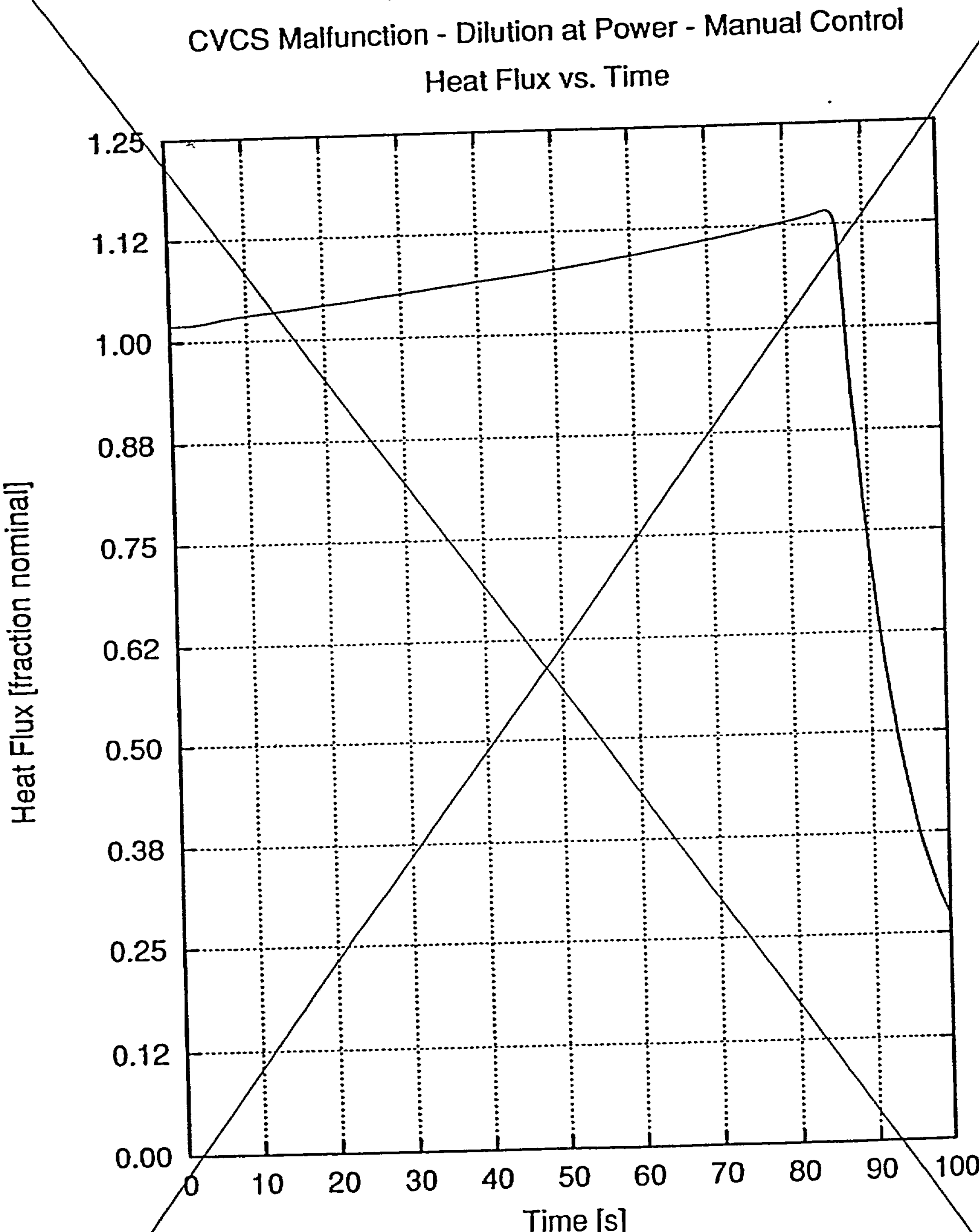

Figure 14.1.4-4

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CVCS Malfunction - Dilution at Power - Manual Control

Minimum DNBR vs. Time

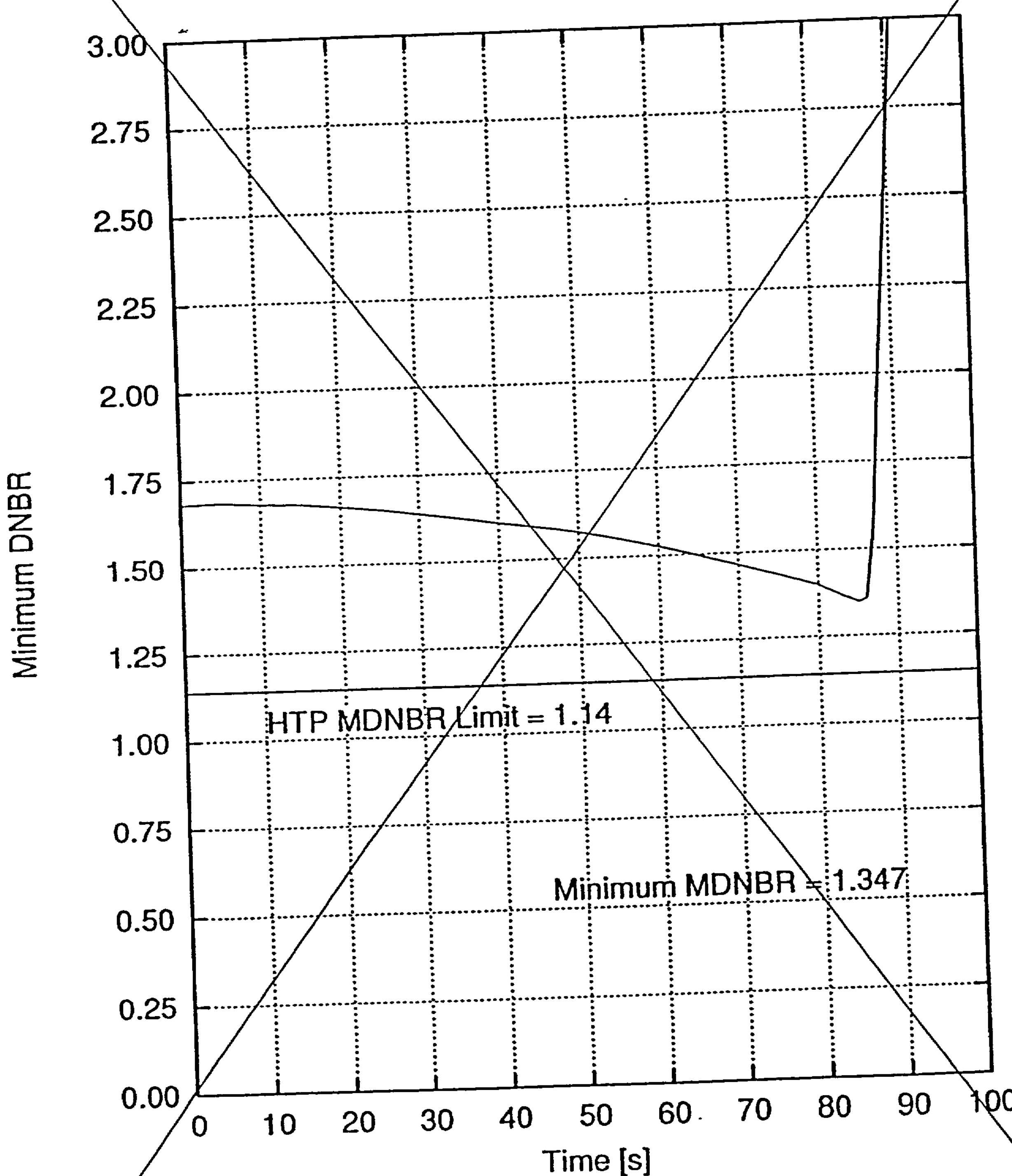

Figure 14.1.4-5

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14.1.5 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

Accident Description

Operation of the plant with an inactive loop causes reversed flow through the inactive loop because there are no isolation valves or check valves in the reactor coolant loops.

If the reactor is operated at power in this condition there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in the inactive loop is identical to the cold leg temperatures of the active loop and to the reactor core inlet temperature. If the reactor is operated at power, there is a temperature drop across the steam generator in the inactive loop, and with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

The protection system prohibits continuous operation of the plant above approximately 10% with one inactive loop. The starting of the idle reactor coolant pump results in the injection of cold water into the core and this causes a rapid reactivity and power increase. However, for power on the order of 10%, the hot leg temperature of the inactive loop is close to the core inlet temperature, thus limiting the severity of the resulting transient.

Add USAR Insert 14.1.5-1

Assumptions and Method of Analysis

The following assumptions are made:

a. Following the start of the idle pump, the inactive loop flow accelerates linearly to its nominal full flow value over a period of 10 seconds.

b. A conservative negative moderator coefficient of −4.0E−4 Δk/°F is assumed.

c. A conservative low Doppler temperature coefficient of −1.0E−5 Δk/°F is assumed.

d. The reactor is assumed to be initially at 12% of 1650 MWt with the secondary side of both steam generators at the same pressure and with reverse reactor coolant flow through the idle loop steam generator. The 12% includes 2% allowance for calibration and instrument errors. The high initial power assumed is conservative since it gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

e. The initial Reactor Coolant System average temperature in the active loops is 4°F above the programmed value for 12% power. This is a conservatively high value for the initial average temperature including instrument errors and results in the minimum margin to core DNB limits.

f. The initial Reactor Coolant System pressure is 50 psi below nominal. This is a conservatively low value for the initial pressure including instrument errors and results in the minimum margin to core DNB limits.

USAR Insert 14.1.5-1

The Kewaunee Nuclear Power Plant Technical Specifications require that both reactor coolant pumps (RCPs) be operating when the reactor is in the OPERATING mode. One pump operation is not permitted except for tests. In the event that one RCP trips with the power being less than 10% of full power, the Technical Specifications require that the core power be reduced to a level below the maximum power determined for zero power testing. If an RCP trips above 10% power, an automatic reactor trip will be initiated. The maximum, initial core power level for the startup of an inactive reactor coolant loop is limited to less than 2%. Under these conditions, there can be no significant reactivity insertion because the reactor coolant system is initially at a nearly uniform temperature. Based on this, an analysis of this event was determined not to be necessary. The discussion presented below corresponds to an analysis previously performed assuming an initial power level of 12% of 1650 MWt and is retained for historical purposes.

USAR Insert 14.1.5-2

The Kewaunee Nuclear Power Plant Technical Specifications require that both reactor coolant pumps (RCPs) be operating when the reactor is in the OPERATING mode. Based on the Technical Specification requirements, an analysis of this event is not required to show that the DNB design basis is satisfied.

A detailed digital simulation of the plant, including heat transfer to the steam generators of the active and inactive loop, and reactor coolant flow transit times, was used to study the transient following pump startup in the inactive loop.

Results

The results following the startup of an idle loop with the assumptions listed above are shown in Figures 14.1.5-1 through 14.1.5-5. The heat flux response, of interest for DNB considerations, indicates that the peak heat flux reaches a value that is less than the nominal full power value. This low heat flux combined with a high degree of sub-cooling in the core at all times results in no adverse effects to the core by the transient. No reactor trip occurs.

It is expected that the actual transient effects would be less severe than those shown because of alleviating factors, which have not been taken into account. For example, the actual starting time of the Reactor Coolant Pump is likely to be about 20 seconds rather than the 10 seconds assumed in the analysis. This means that the change in core temperature would occur more gradually than shown in the figures. Furthermore, the water entering the core is assumed to exhibit the temperature of the water in the inactive loop, providing the analysis with a high degree of conservatism.

The average temperature of the reactor coolant water increases because of the positive reactivity insertion and power increase brought about by the entry into the core of the cold water in the inactive loop. This leads to an increase in pressurizer pressure. The maximum pressure reached is well below the acceptance criteria of 2750 psia.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Valve/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Startup of Inactive Loop	5.878/[redacted]	2313/2750	1153/1210

Conclusions

The results show that for startup of an inactive loop, the power and the temperature excursions are not severe. There is a considerable margin to the limiting MDNBR. Therefore, no undue restriction needs to be placed on the plant when starting a reactor coolant pump at power levels up to 12% power.

Add USAR Insert 14.1.5-2

14.1.6 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

Accident Description

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary system and of the Reactor Coolant System. The Reactor Protection System trip functions prevent any power increase that could lead to a DNBR less than the MDNBR limit.

An extreme example of excessive heat removal from the Reactor Coolant System is the transient associated with the accidental opening of the feedwater bypass valve, which diverts flow around the low-pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., following a large load decrease.

In the event of an accidental opening, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased sub-cooling would create a greater load demand on the Reactor Coolant System due to the increased heat transfer in the steam generator.

Add USAR Insert 14.1.6-1

Another example of excessive heat removal from the Reactor Coolant System is a common mode failure in the feedwater control system, which leads to the accidental opening of the feedwater regulating valves (FW-7A and FW-7B) to both steam generators (see Reference 2).

FW-7A and FW-7B could fail open due to a high output signal to the feedwater control system from any one of the following components:

- PT-485 First Stage Turbine Pressure Transmitter
- PM-485A I/I Converter
- LM-463F Steam Generator Level Auto Programmer Mode Controller
- LM-463H Steam Generator Level Program Median Selector
- LM-463D Current Source for Steam Generator Level Minimum Setpoint
- LM-463C Lead/Lag Circuit

This results in the valves stepping open 20% from their current position followed by a 20% step open every 5 minutes after that until full open.

Accidental opening of the feedwater regulating valves results in an increase of feedwater flow to both steam generators, causing excessive heat removal from the reactor coolant system. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control system feedback.

Continuous addition of cold feedwater after a reactor trip is prevented since the reduction of Reactor Coolant System temperature, pressure, and pressurizer level leads to the actuation of safety injection on low pressurizer pressure. The safety injection signal trips the main feedwater pumps, closes the feedwater pump discharge valves, and closes the main feedwater control valves.

USAR Insert 14.1.6-1

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

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

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive load increase.

USAR Insert 14.1.6-2 NMC Scope/NMC to Confirm

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

The following assumptions are made:

A. Initial power level of 1780 MWt.

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

An evaluation method was applied that demonstrates the decreased enthalpy caused by the feedwater temperature reduction is bounded by an equivalent enthalpy reduction that results from an excessive load increase incident (Section 14.1.7).

USAR Insert 14.1.6-3

The opening of a low pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The reduction in feedwater temperature is less than 33°F resulting in an increase in heat load on the primary system of less than 10% of full power. The reduction in feedwater temperature due to a 10% step load increase is greater than 33°F. The increased thermal load, due to the opening of the low pressure heater bypass valve, thus results in a transient very similar, but of reduced magnitude, to the 10% step load increase incident described in Section 14.1.7. Therefore, the transient results are not presented.

USAR Insert 14.1.6-4

With respect to the feedwater temperature reduction transient (accidental opening of the feedwater bypass valve), it was determined to be less severe than the excessive load increase incident (see Section 14.1.7). Based on results presented in Section 14.1.7, the applicable acceptance criteria for the feedwater temperature reduction transient have been met.

Accidental Opening of the Feedwater Bypass Valve

Method of Analysis

REPLACE WITH USAR Insert 14.1.6-2

Two cases have been analyzed to demonstrate the plant behavior in the event of a sudden feedwater temperature reduction resulting from the accidental opening of the feedwater bypass valve. The first case is for a reactor in manual control with a zero moderator reactivity coefficient since this represents a condition in which the plant has the least inherent transient capability. The second case is for a reactor in automatic control with a conservatively large negative moderator reactivity coefficient (–4.0E–4 Δk/°F).

Initial pressurizer pressure, reactor coolant average temperature, and reactor power are consistent with steady state, full power operation, allowing for calibration and instrument errors. This results in the minimum margin to core DNB at the start of the transient. The analyses are performed using a detailed digital simulation of the plant including core kinetics, Reactor Coolant System, and the Main Steam and Feedwater Systems.

Results

REPLACE WITH USAR INSERT 14.1.6-3

Figures 14.1.6-1 through 14.1.6-5 show the transient without automatic reactor control and with a zero moderator reactivity coefficient representing beginning of cycle conditions. As expected, the average reactor coolant temperature and pressurizer pressure show rapid decreases as the secondary heat extraction remains greater than the core power generation. The core power level increases slowly and eventually comes to equilibrium at a value slightly above the nominal full power value. There is an increased margin to DNB because of the accompanying reduction in coolant average temperature. The reactor does not trip. There is a small increase in core ΔT as the heat transfer increases through the steam generator.

Figures 14.1.6-6 through 14.1.6-10 illustrate the transient with automatic reactor control. A conservatively large negative moderator coefficient (–4.0E–4 Δk/°F) representing end of cycle core conditions is assumed. The large negative moderator coefficient increases reactor power, which reduces the decrease in temperature and pressure. Eventually reactor power comes to equilibrium at a value slightly above the nominal full power value. The minimum DNB ratio decreases slightly but is well above the MDNBR limit.

The reactivity insertion rate at no-load from an excessive feedwater flow increase accident is also analyzed with the following assumptions:

1. A step increase in feedwater flow to one steam generator from zero to the nominal full-load flow.

2. The most negative reactivity moderator coefficient at end-of-life.

3. A constant feedwater temperature of 70°F

4. Heat capacity of the Reactor Coolant System and steam generator shell are not taken credit for

5. Neglect of the energy stored in the fluid of the unaffected steam generator.

The maximum reactivity insertion rate was calculated to be 2.3E-4 Δk/sec which is less than the maximum reactivity insertion rate analyzed in Section 14.1.1, Uncontrolled RCCA Withdrawal from a Sub-critical Condition. It should be noted that if the incident occurs with the reactor critical at no-load, the reactor would be tripped by the power range high nuclear flux trip (low setting) set at approximately 25%. As shown in Section 14.1.1, there is a large margin to DNB with the above calculated reactivity insertion rate.

DELETE

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Feedwater System Malfunction	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
BOC Manual Control	1.681/1.14	2200/2750	751/1210
EOC Auto Control	1.647/1.14	2200/2750	751/1210

Accidental Opening of Feedwater Regulating Valves

Method of Analysis

Add USAR Insert 14.1.6-5

The following assumptions are made for the analysis of for a feedwater malfunction event involving the accidental opening of the feedwater regulating valves:

REPLACE with USAR Insert 14.1.6-6

1) The plant is operating at full power allowing for instrument and calibration uncertainties level with the feedwater control system in automatic mode. The safety analysis uses this power level to give the highest feedwater flow rate.

2) Automatic steam generator level control is functional with the exception of the failed valves.

3) The feedwater in headers A and B is at a temperature of 430°F. This temperature is consistent with normal plant conditions.

4) Feedwater flow increases 50% in both loops from 3.6 to 5.4 MLBM/HR. This is conservative because pump runout flow is 5.0 MLBM/HR.

The analysis is performed using a detailed digital simulation of the plant including core kinetics, Reactor Coolant System and the Main Steam and Feedwater Systems.

Results REPLACE WITH USAR Insert 14.1.6-7

~~Reactor power increases to slightly above the nominal full power value due to the reactor cooldown, which is caused by the excessive feedwater flow to both steam generators. As a result, minimum DNBR decreases slightly but is well above the limiting minimum DNBR.~~

Conclusions

REPLACE WITH USAR Insert 14.1.6-4

Feedwater system malfunction transients involving a reduction in feedwater temperature or an increase in feedwater flow rate have been analyzed. The analyses show an increase in reactor power from the reactor temperature reduction due to the excessive heat removal in the steam generators. ~~The most limiting of the feedwater malfunction transients is the inadvertent opening of a feedwater heater bypass valve at full power conditions.~~ Analyses of the accidental opening of the feedwater regulating valves demonstrate that considerable margin to the safety analysis acceptance criteria, (MDNBR, primary and secondary pressure), exists throughout the transient. Therefore, there is no radioactive release or public hazard in the event of a feedwater malfunction event.

14.1.7 EXCESSIVE LOAD INCREASE INCIDENT

Accident Description

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The Reactor Control System is designed to accommodate a 10% step load increase or a 5% per minute ramp load increase (without a reactor trip) in the range of 15 to 95% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System. If the load increase exceeds the capability of the Reactor Control System; the transient is terminated in sufficient time to prevent the DNBR from being reduced below the MDNBR limit. An excessive load increase incident 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.

For excessive loading by the operator or by system demand, the turbine load limiter keeps maximum turbine load from exceeding 100% rated load.

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

Load increases caused by a hypothetical steam-line break are analyzed in Section 14.2.5.

Add USAR Insert 14.1.7-1

USAR Insert 14.1.6-5

The system is analyzed to demonstrate acceptable consequences in the event of an excessive feedwater addition, due to a control system malfunction or operator error which allows all feedwater control valves to open fully. The following cases have been analyzed:

1a. Accidental full opening of all feedwater control valves with the reactor at full power assuming manual rod control and a conservatively large negative moderator temperature coefficient of reactivity.

1b. Accidental full opening of all feedwater control valves with the reactor at full power assuming automatic rod control and a conservatively large negative moderator temperature coefficient of reactivity.

2. Accidental opening of a feedwater control valve with the reactor at no load (hot zero power) conditions and assuming a conservatively large negative moderator temperature coefficient of reactivity with minimum available shutdown margin.

This accident is analyzed using the Revised Thermal Design Procedure (RTDP) methodology.

USAR Insert 14.1.6-6

1) Initial reactor power, pressure, and RCS temperatures are assumed to be at their conservative nominal values. Uncertainties in initial conditions are included in the MDNBR limit.

2) Feedwater control valves are assumed to malfunction resulting in a step increase from 100% to 150% of nominal feedwater flow delivering flow to both steam generators.

3) For the feedwater control valve accident at full power conditions (cases 1a and 1b), a feedwater temperature of 437.1°F is assumed, consistent with nominal plant conditions.

4) For the feedwater control valve accident at no load conditions (case 2), a feedwater temperature of 198.0°F is assumed, consistent with no load plant conditions.

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

6) The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal that closes the associated feedwater main control and feedwater control-bypass valves, indirectly closes all feedwater pump discharge valves, and trips the main feedwater pumps and turbine generator.

Normal reactor control system and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to power-range high neutron flux, overpower or turbine trip on high-high steam generator water level conditions.

USAR Insert 14.1.6-7

The feedwater flow malfunction at hot zero power conditions result in an increased nominal heat flux and reduced RCS pressure due to the reactor cooldown, which is caused by the excessive feedwater flow to both steam generators. The results of the DNB analysis yielded a minimum DNBR above the safety analysis limit, however it was found that this case is bounded by the excessive feedwater flow cases analyzed at full power initial conditions.

The most limiting case is the excessive feedwater flow from a full power initial condition with automatic rod control. This case gives the largest reactivity feedback and results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in either steam generator reaches the high-high water level setpoint. Assuming the reactor to be in manual rod control results in a slightly less severe transient. The rod control system is not required to function for this event; however assuming that the rod control system is operable, yields a slightly more limiting transient.

For each excessive feedwater flow case analyzed, continuous addition of cold feedwater is prevented by automatic closure of the associated feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. In addition, the feedwater discharge isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Transient results, Figures 14.1.6-1 through 14.1.6-10, show the reactor power, pressurizer pressure, core average temperature, vessel inlet and outlet temperature and minimum DNB conditions throughout the transient for the full power cases (while in manual and automatic rod control). Though the reactor power increases slightly above the nominal full power value during the transient, the DNBR does not drop below the safety analysis limit value.

REPLACE WITH FIGURE 14.1.6-1 (ON FOLLOWING PAGE)

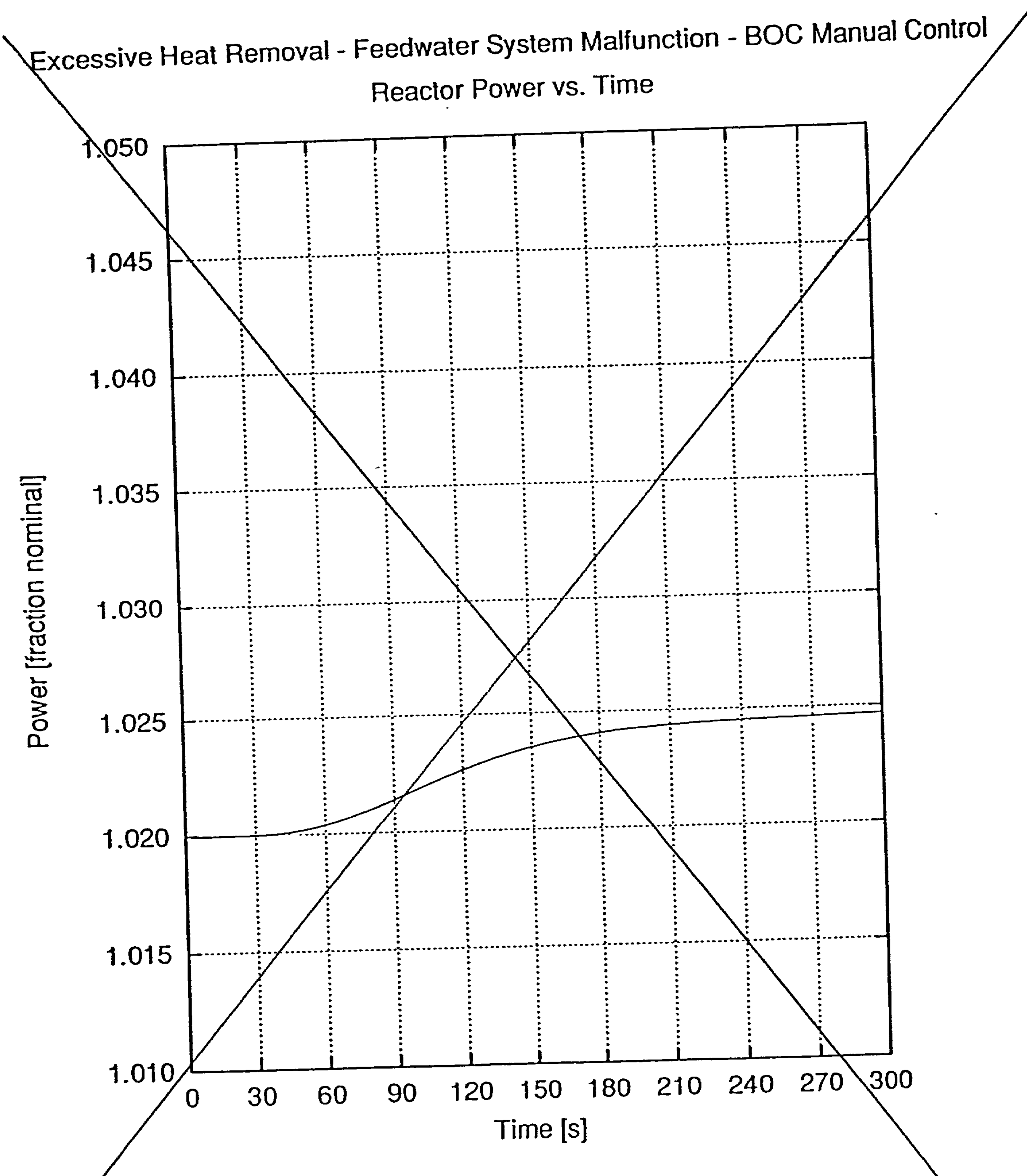

Figure 14.1.6-1

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Excessive Heat Removal – Feedwater System Malfunction – Manual Control
Reactor Power vs. Time

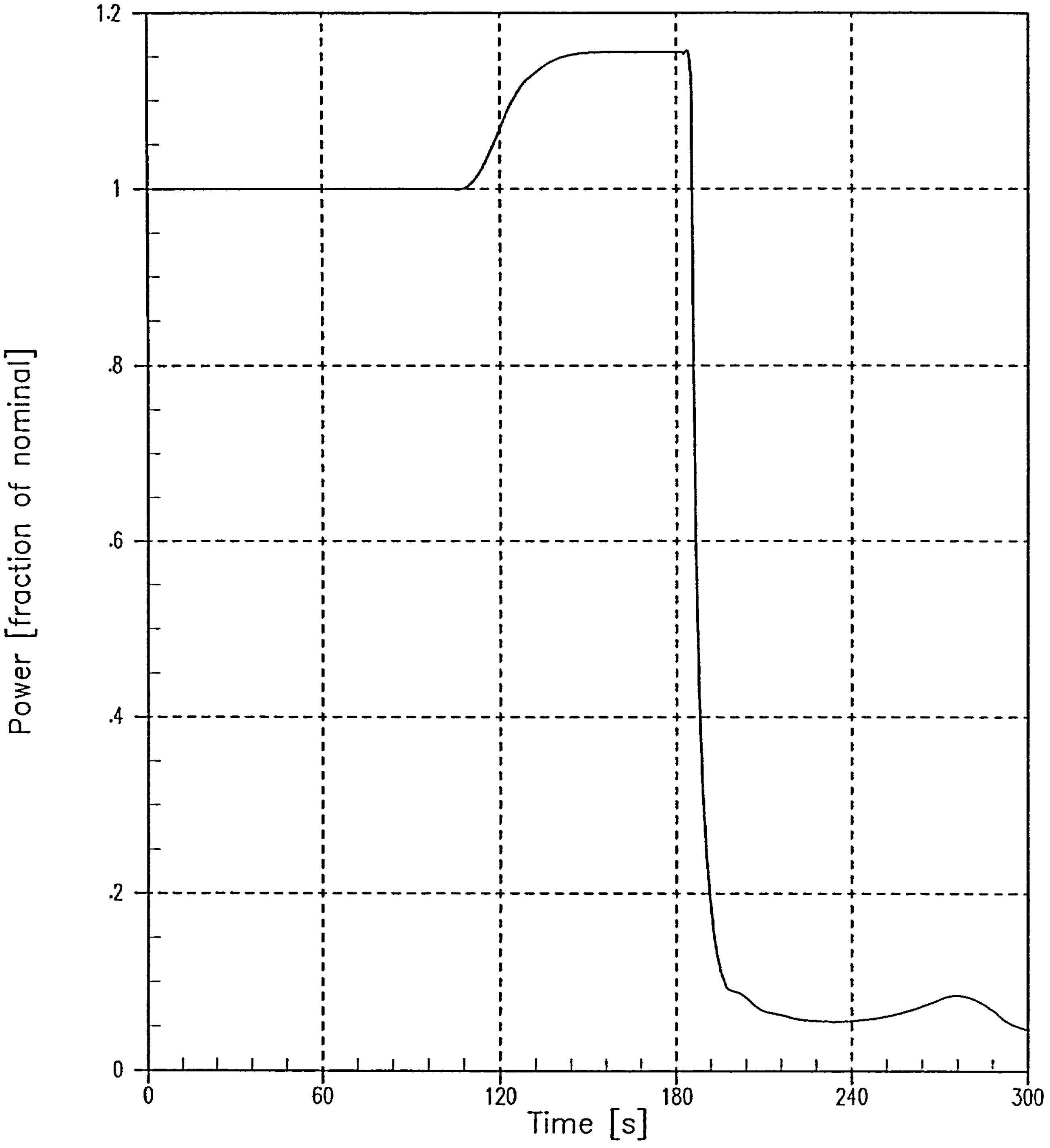

Figure 14.1.6–1

REPLACE WITH FIGURE 14.1.6-2 (ON FOLLOWING PAGE)

Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control

Pressurizer Pressure vs. Time

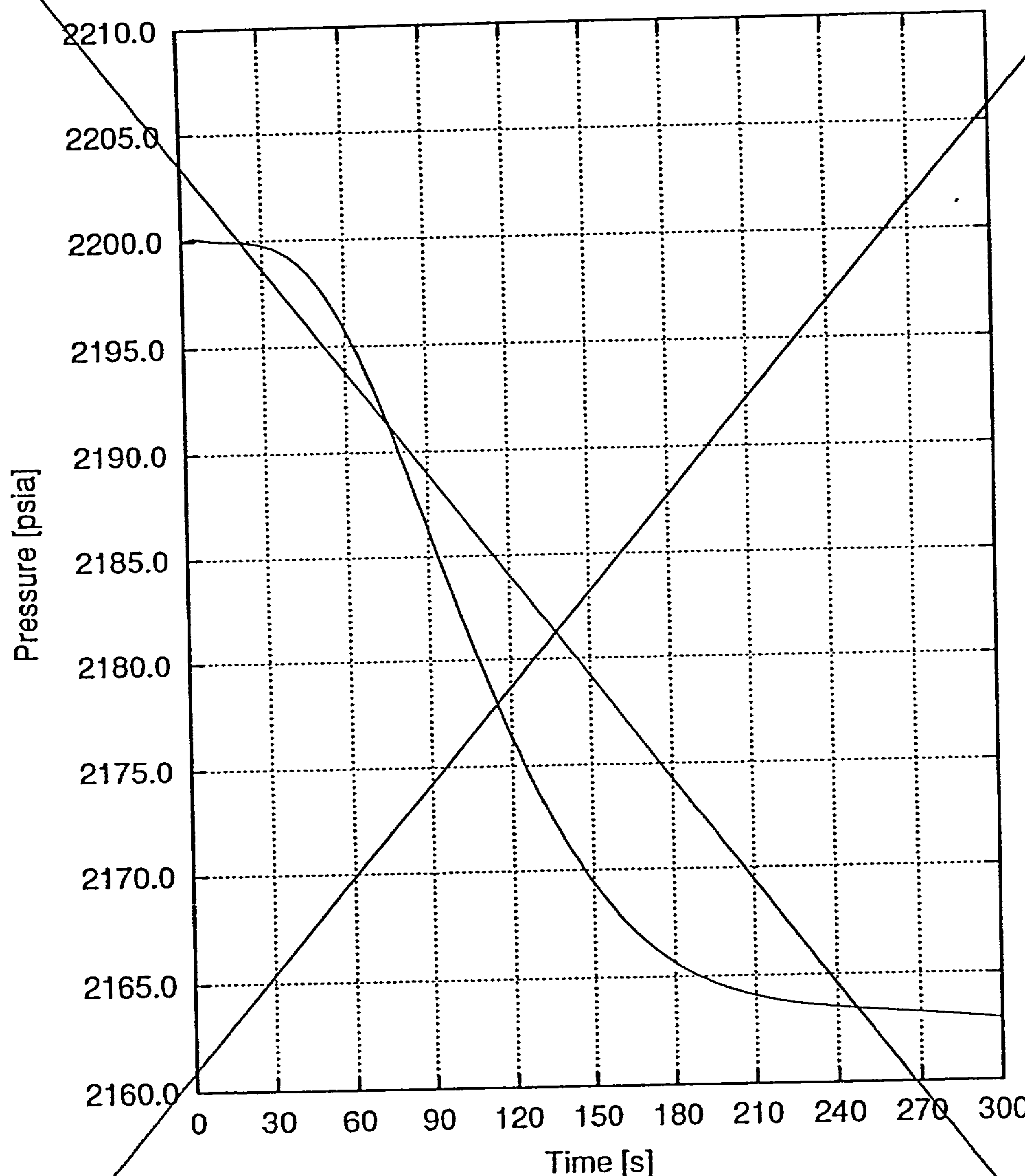

Figure 14.1.6-2

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Excessive Heat Removal – Feedwater System Malfunction – Manual Control
Pressurizer Pressure vs. Time

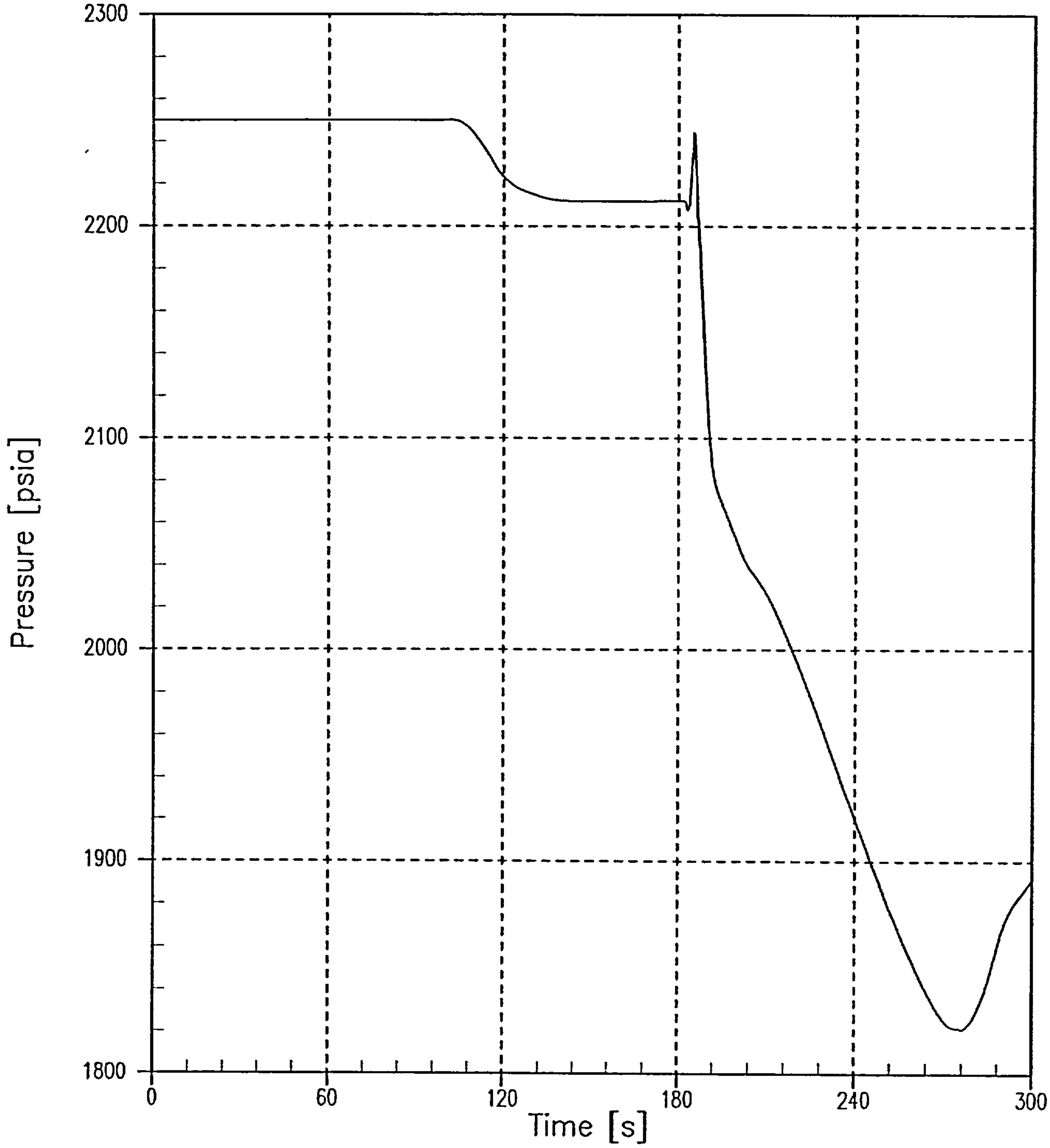

Figure 14.1.6–2

REPLACE WITH FIGURE 14.1.6-3 (ON FOLLOWING PAGE)

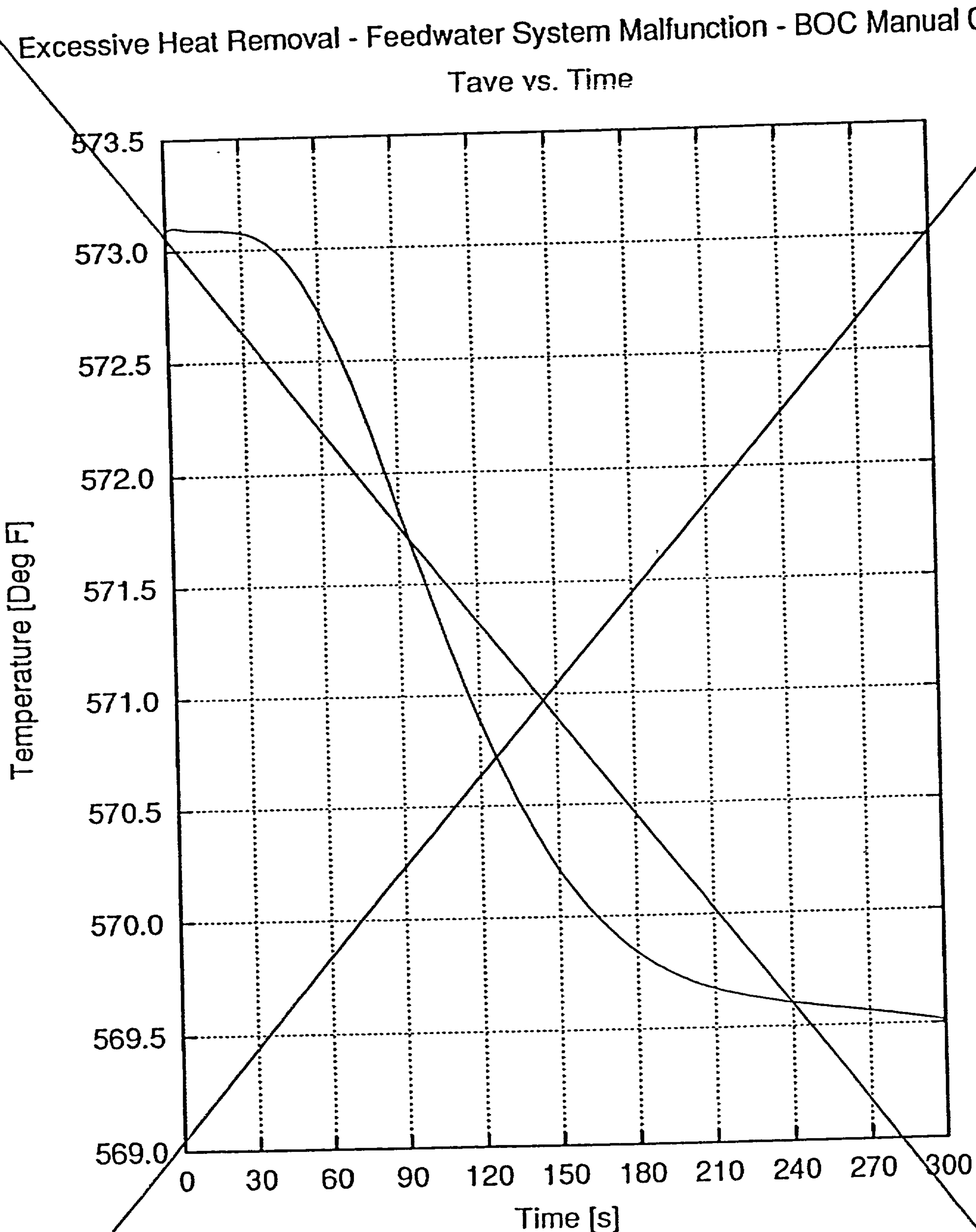

Figure 14.1.6-3

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Excessive Heat Removal — Feedwater System Malfunction — Manual Control
Core Average Temperature vs. Time

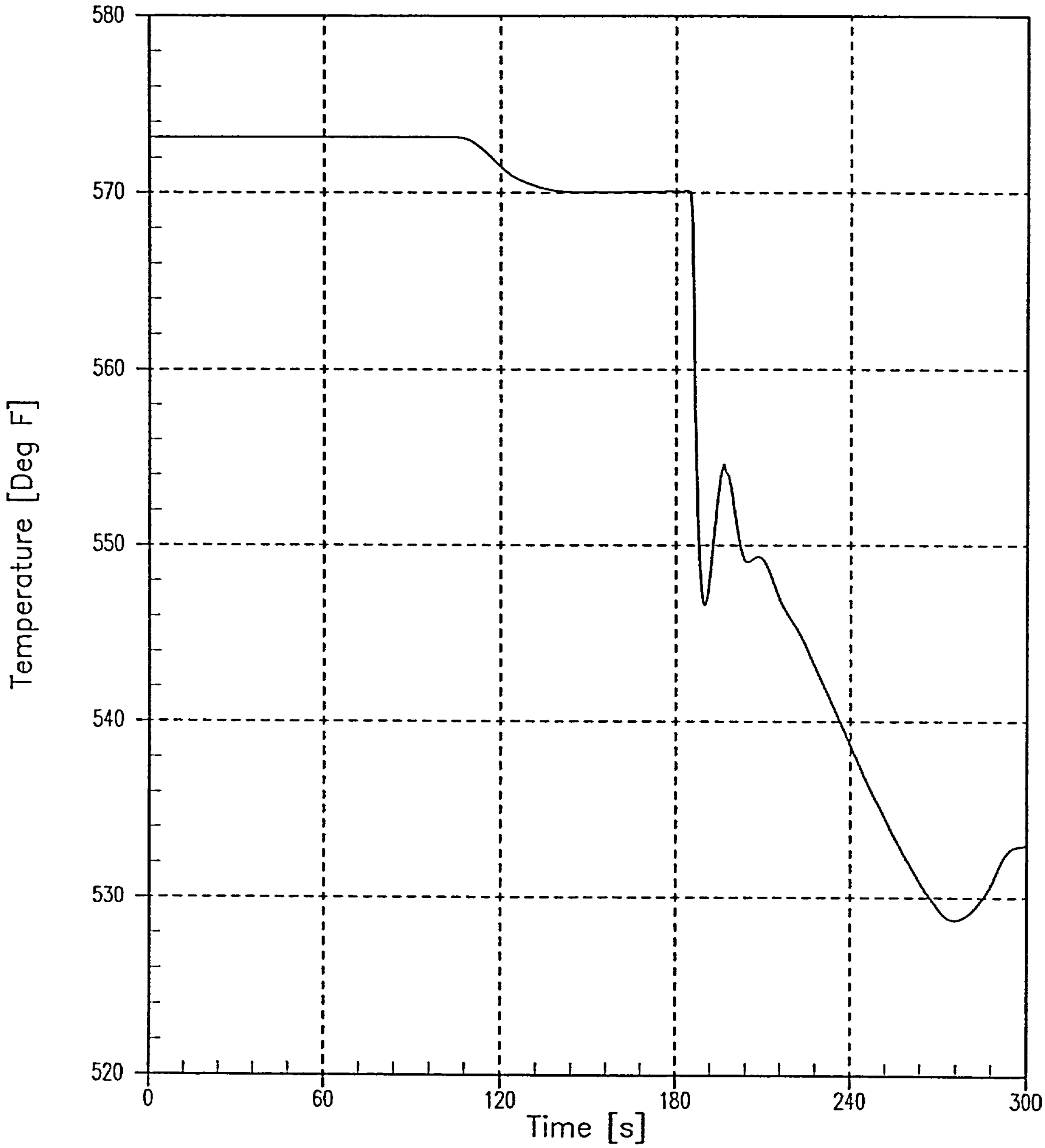

Figure 14.1.6-3

REPLACE WITH FIGURE 14.1.6-4 (ON FOLLOWING PAGE)

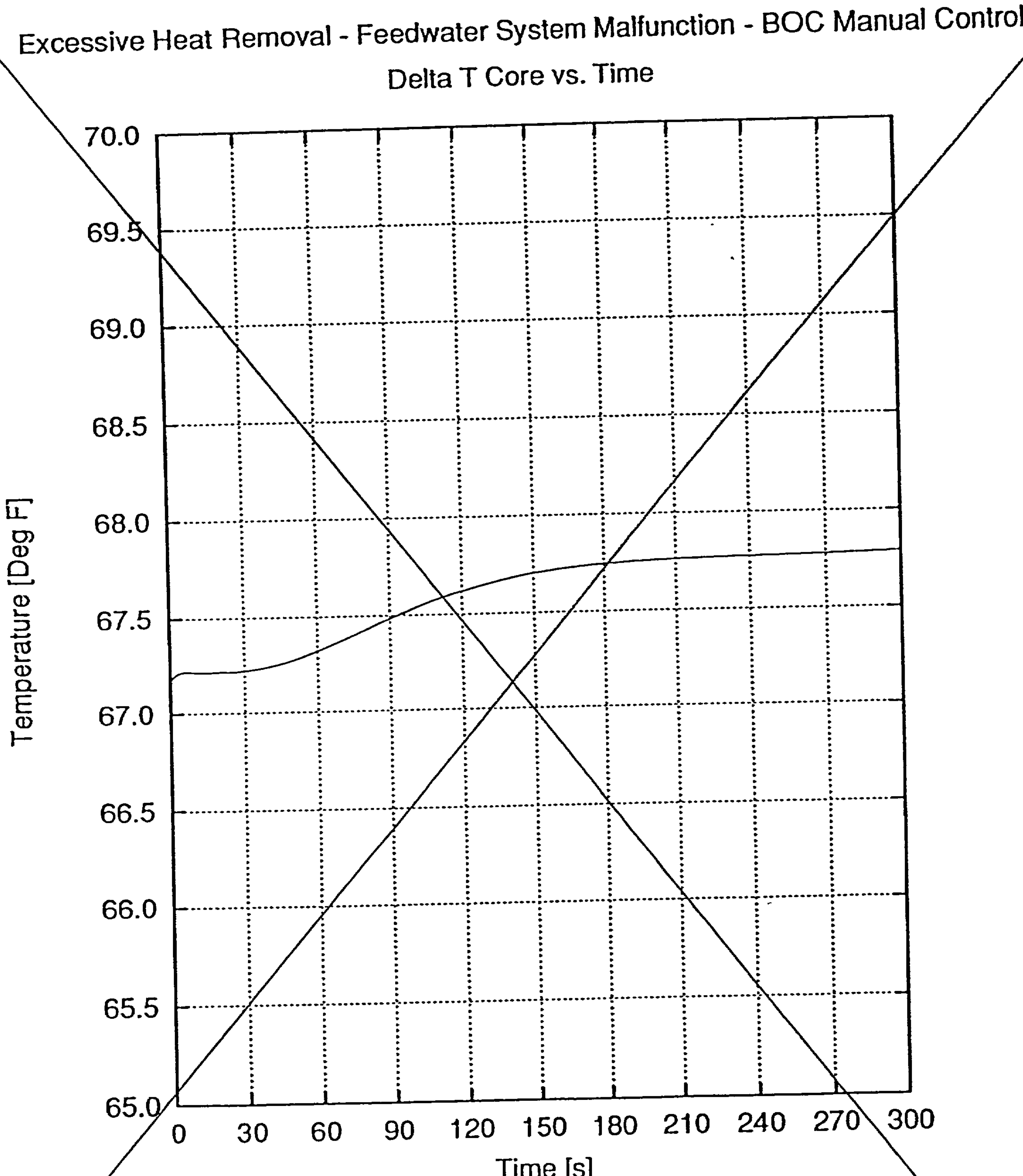

Figure 14.1.6-4

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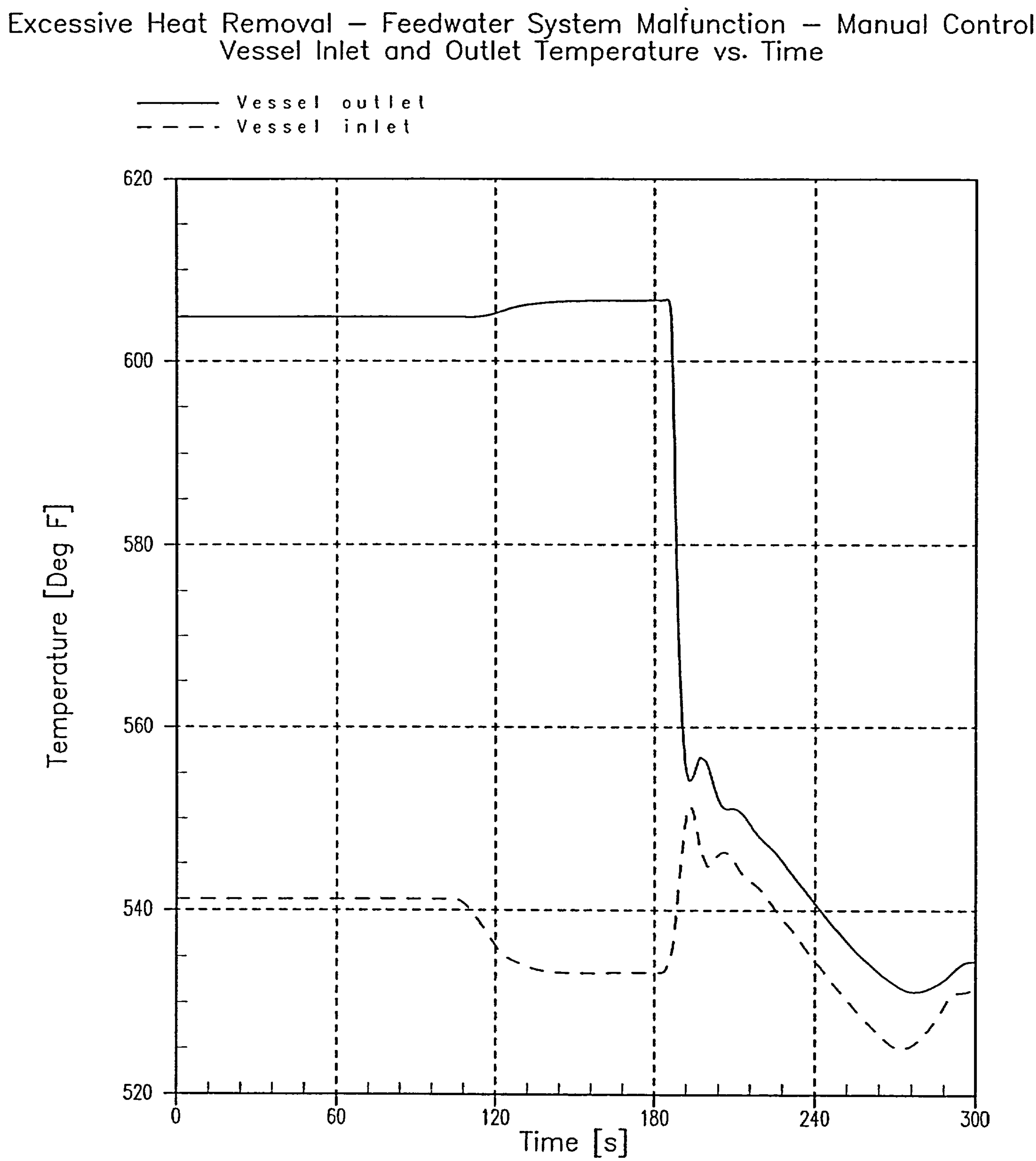

Figure 14.1.6–4

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Excessive Heat Removal - Feedwater System Malfunction - BOC Manual Control

Minimum DNBR vs. Time

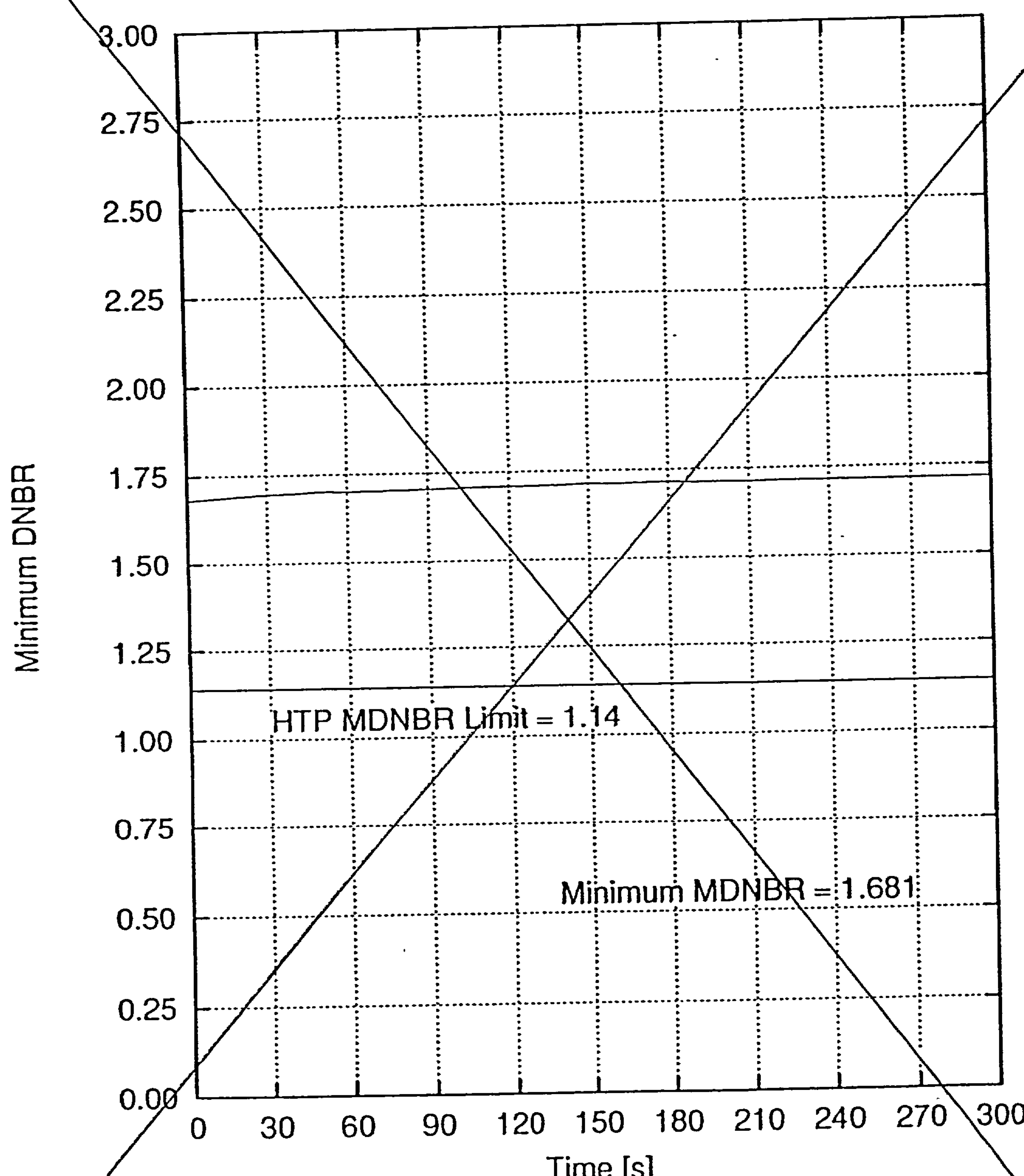

Figure 14.1.6-5

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Excessive Heat Removal – Feedwater System Malfunction – Manual Control
Minimum DNBR vs. Time

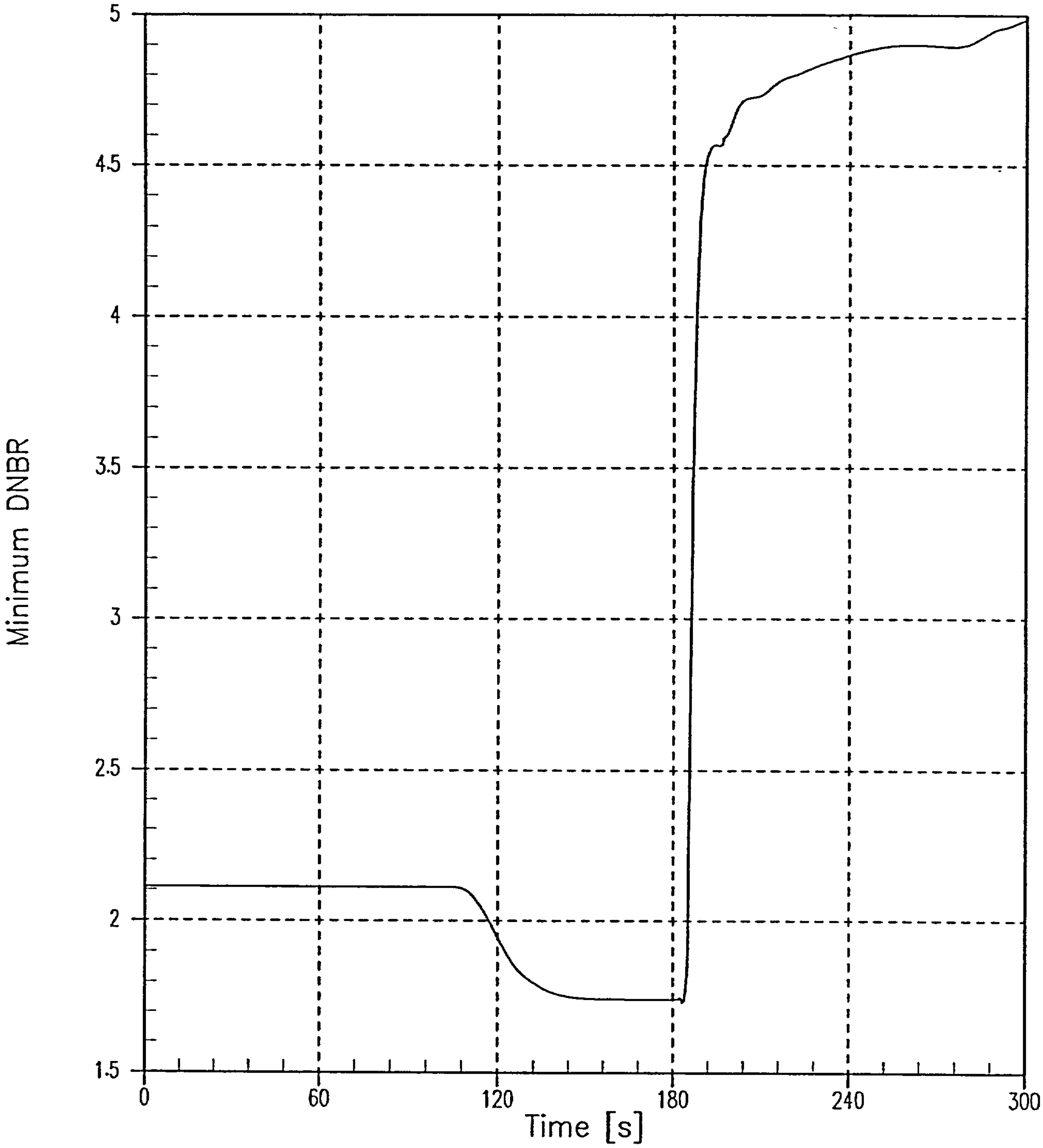

Figure 14.1.6–5

REPLACE WITH FIGURE 14.1.6-6 (ON FOLLOWING PAGE)

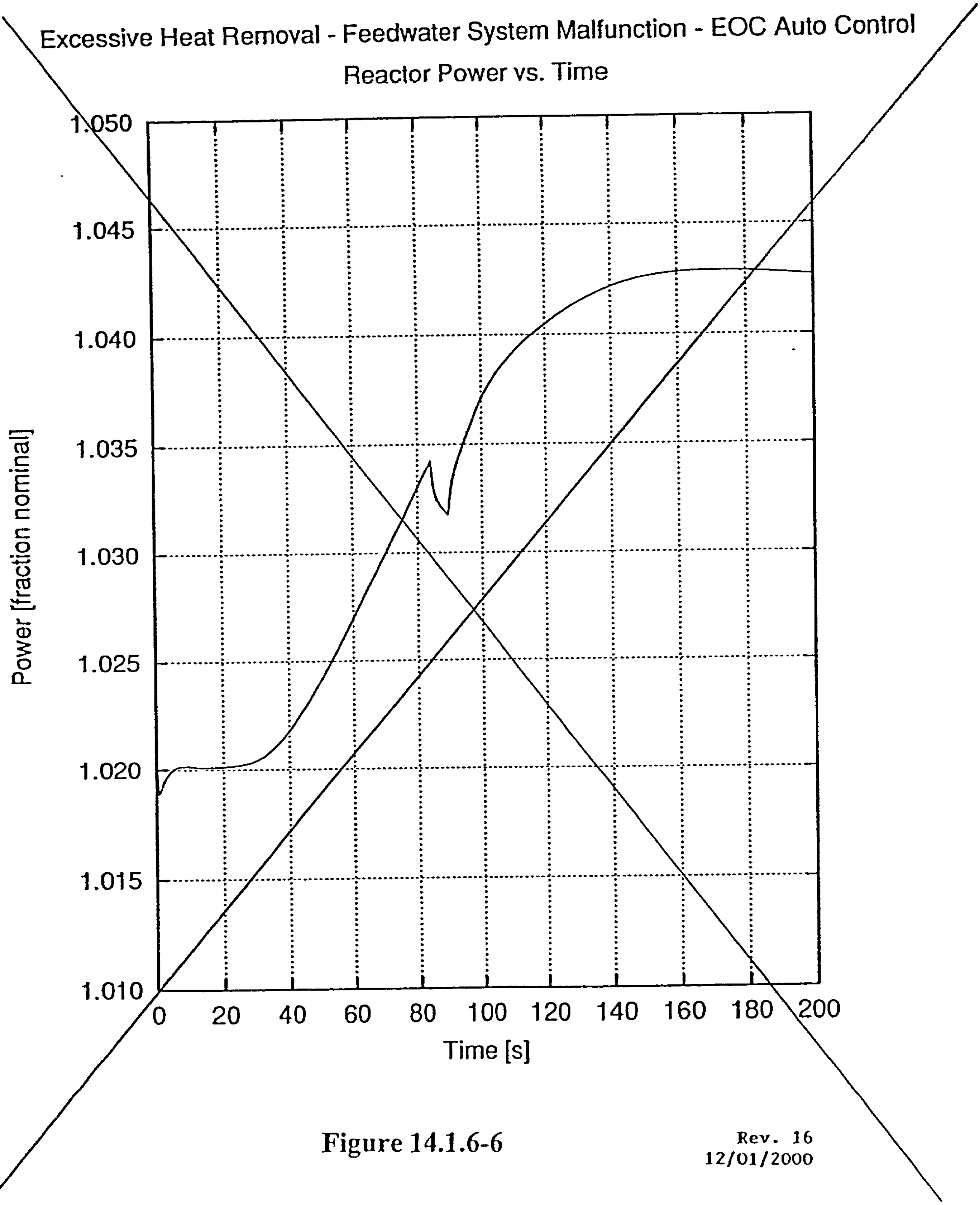

Figure 14.1.6-6

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Excessive Heat Removal — Feedwater System Malfunction — Auto Control
Reactor Power vs. Time

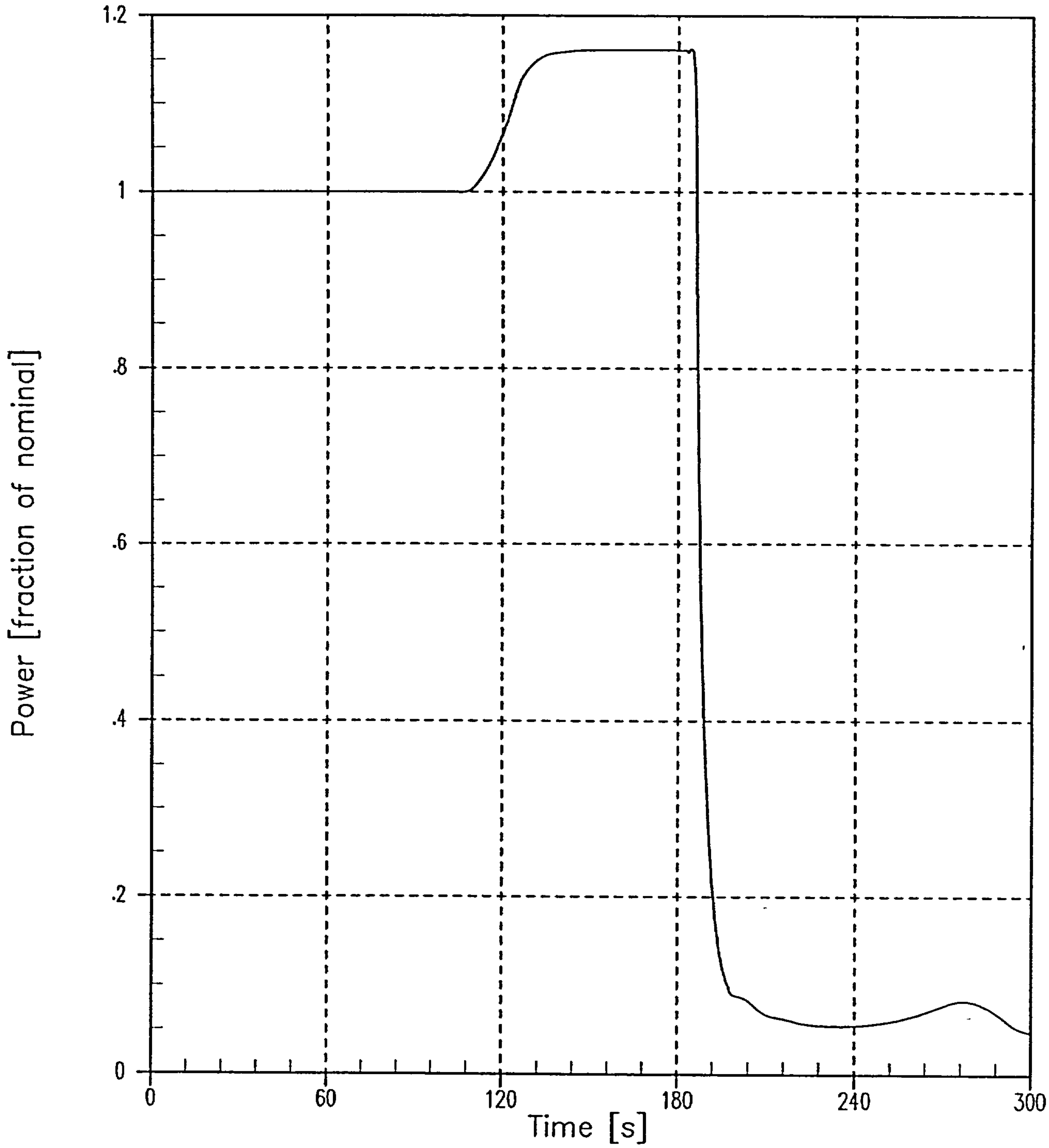

Figure 14.1.6—6

REPLACE WITH FIGURE 14.1.6-7 (ON FOLLOWING PAGE)

Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control

Pressurizer Pressure vs. Time

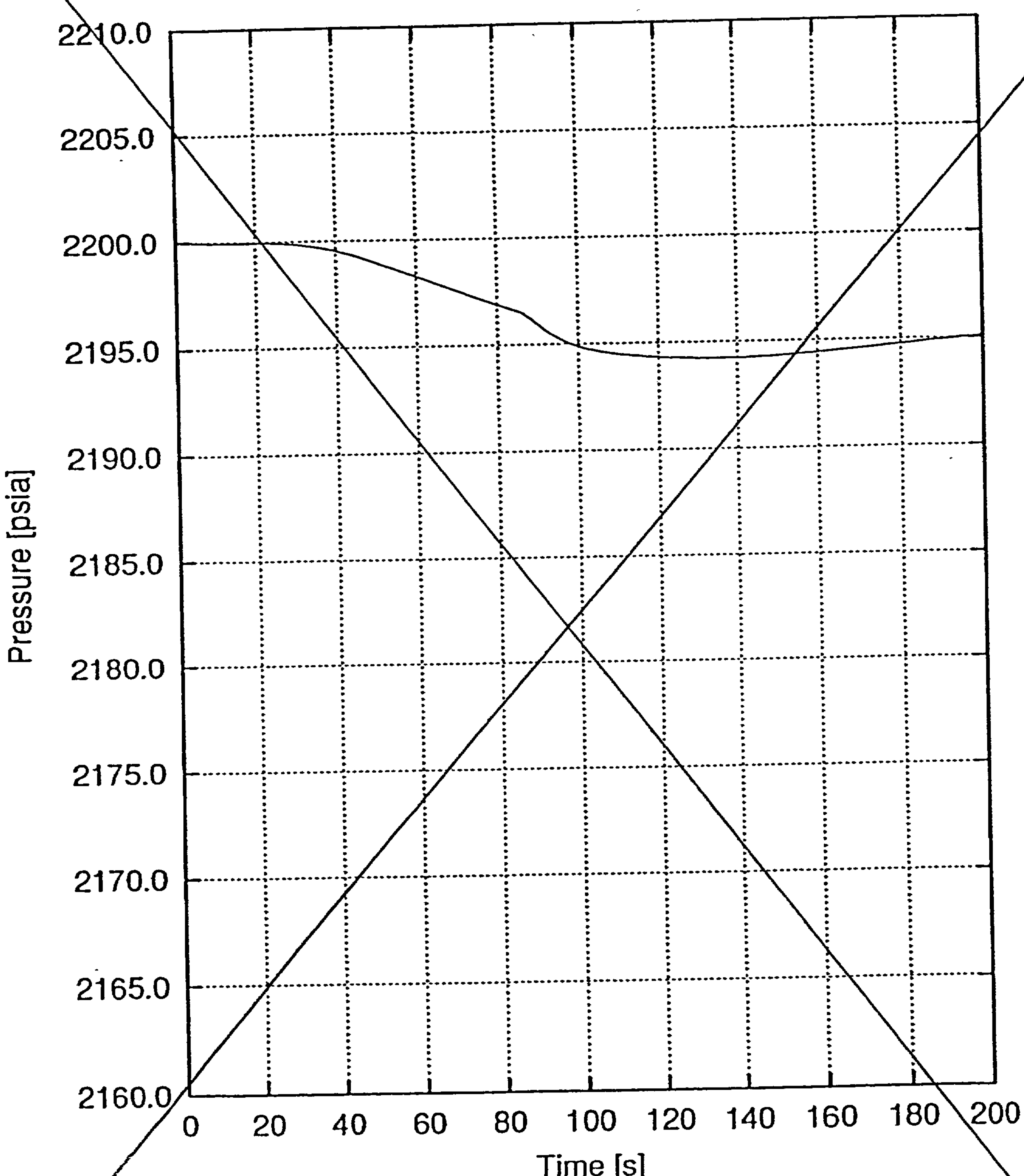

Figure 14.1.6-7

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Excessive Heat Removal — Feedwater System Malfunction — Auto Control
Pressurizer Pressure vs. Time

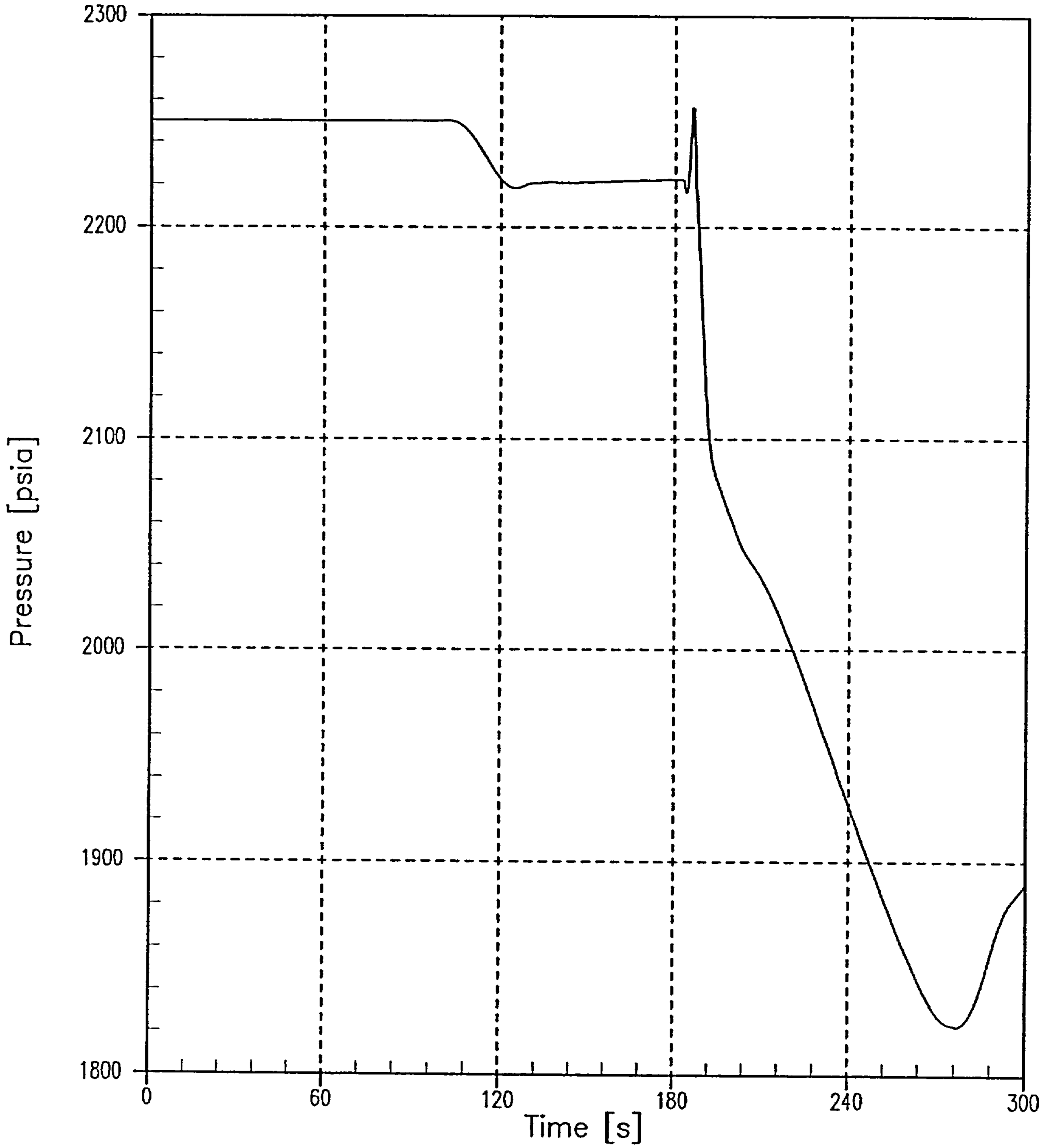

Figure 14.1.6–7

REPLACE WITH FIGURE 14.1.6-8 (ON FOLLOWING PAGE)

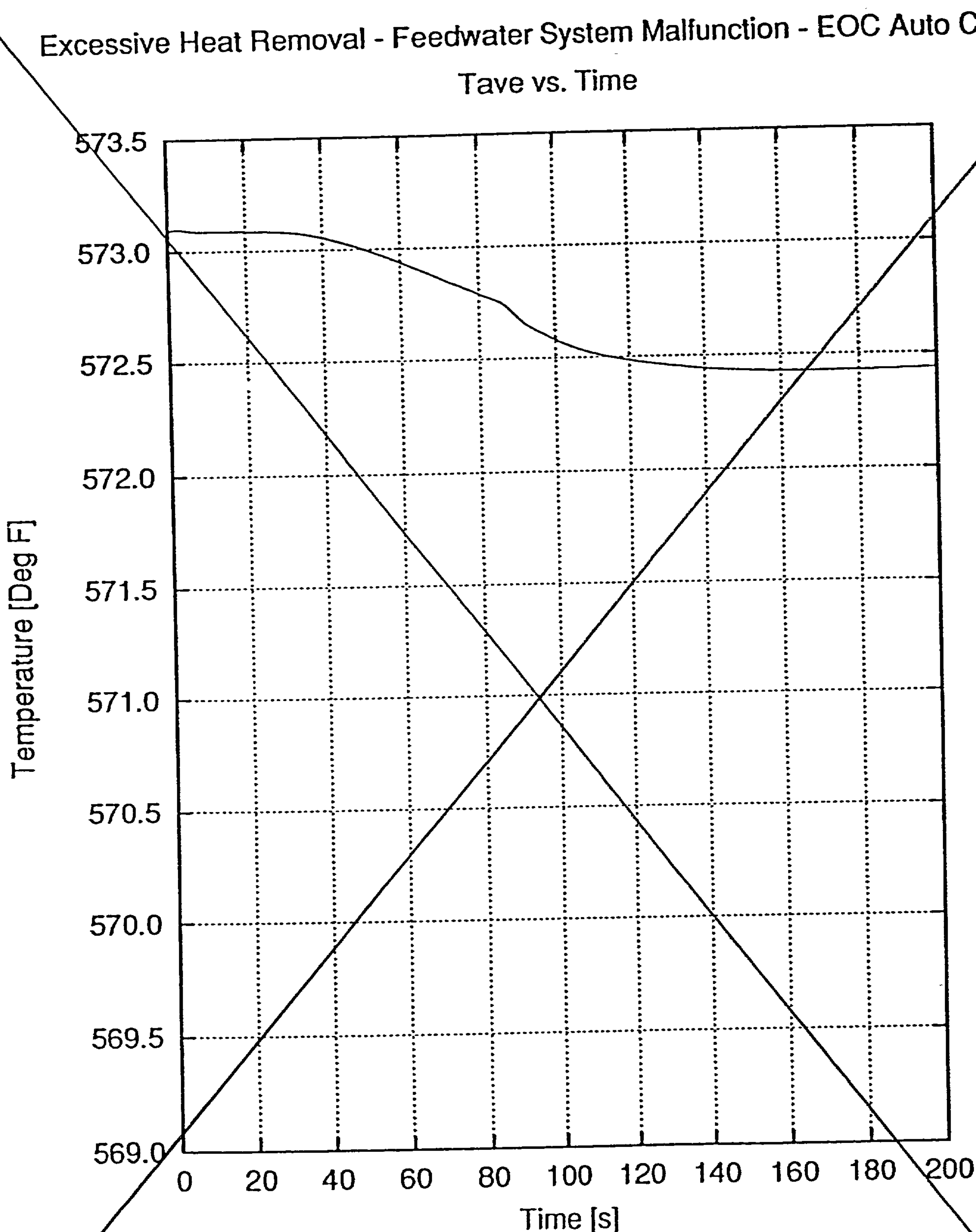

Figure 14.1.6-8

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Excessive Heat Removal – Feedwater System Malfunction – Auto Control
Core Average Temperature vs. Time

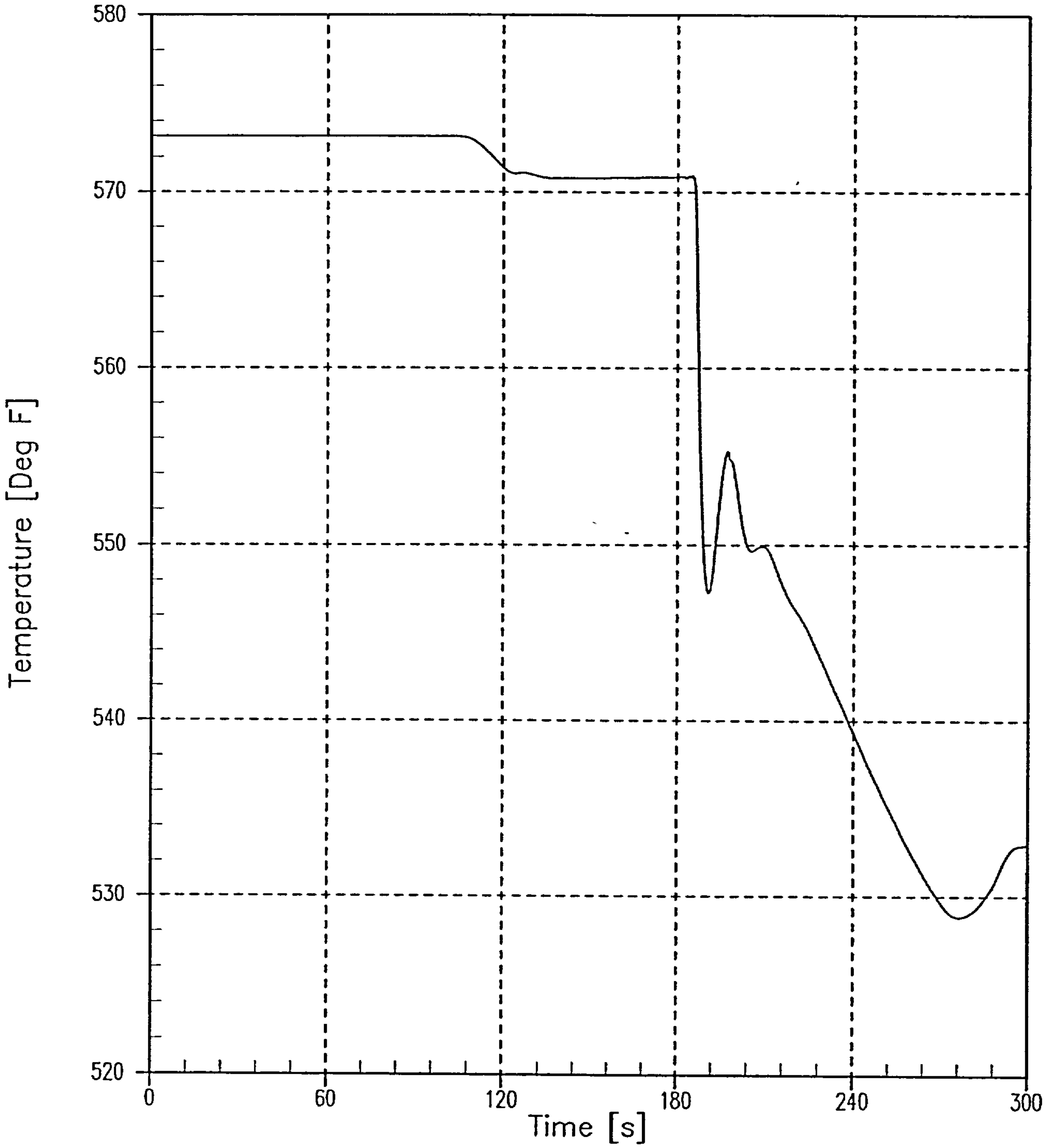

Figure 14.1.6–8

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Excessive Heat Removal - Feedwater System Malfunction - EOC Auto Control

Delta T Core vs. Time

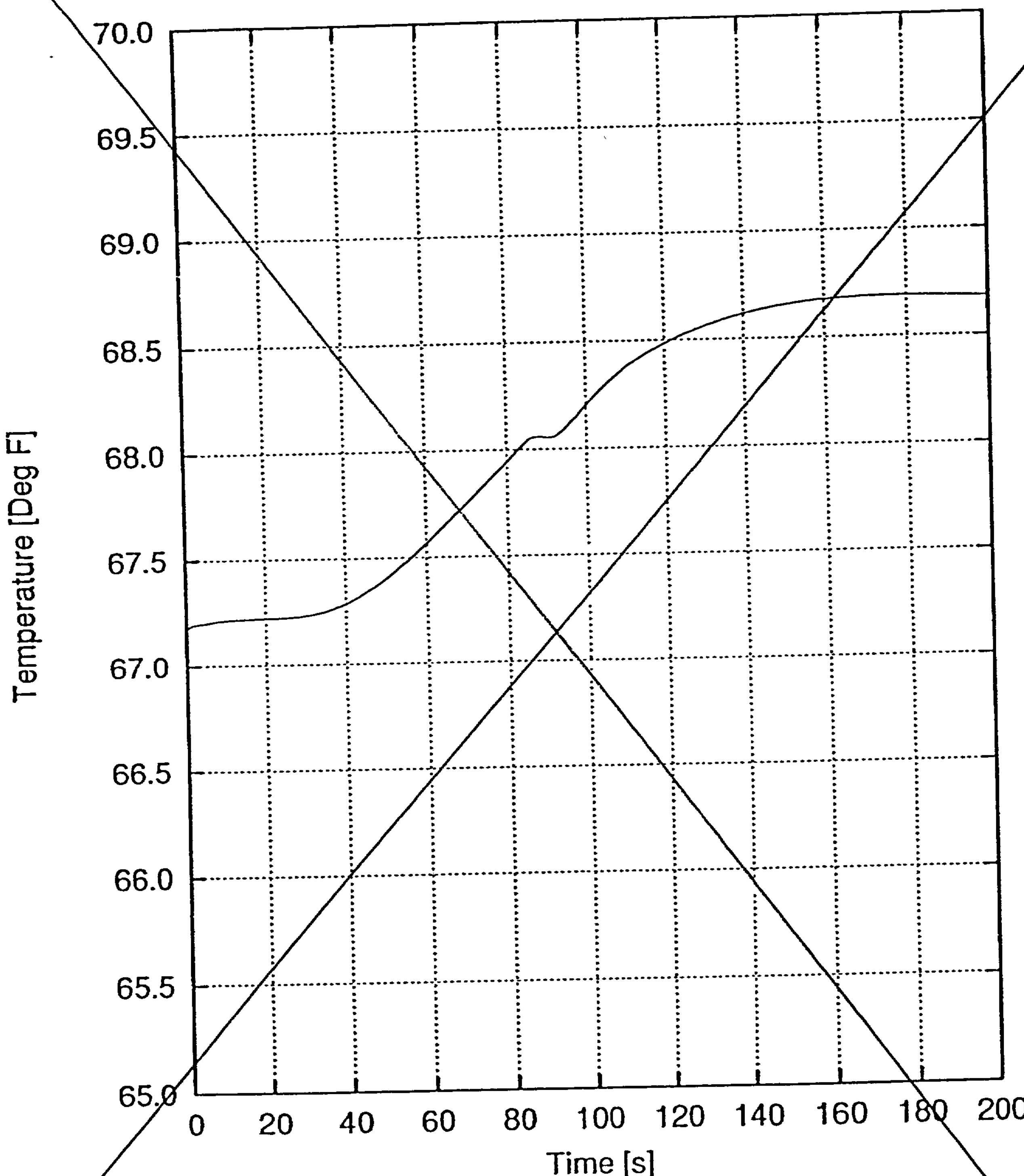

Figure 14.1.6-9

Rev. 16
12/01/2000

Excessive Heat Removal – Feedwater System Malfunction – Auto Control
Vessel Inlet and Outlet Temperature vs. Time

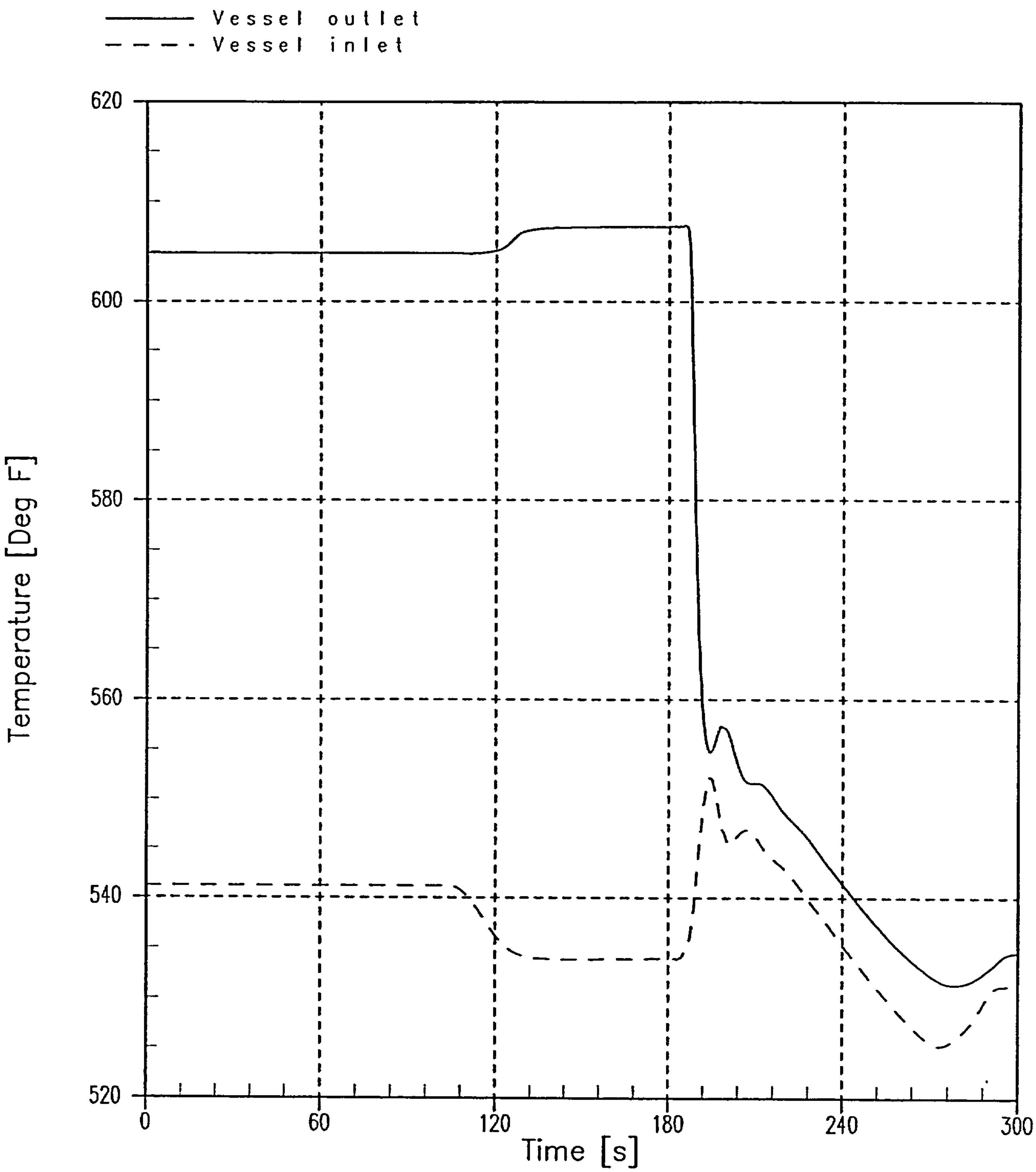

Figure 14.1.6–9

REPLACE WITH FIGURE 14.1.6-10 (ON FOLLOWING PAGE)

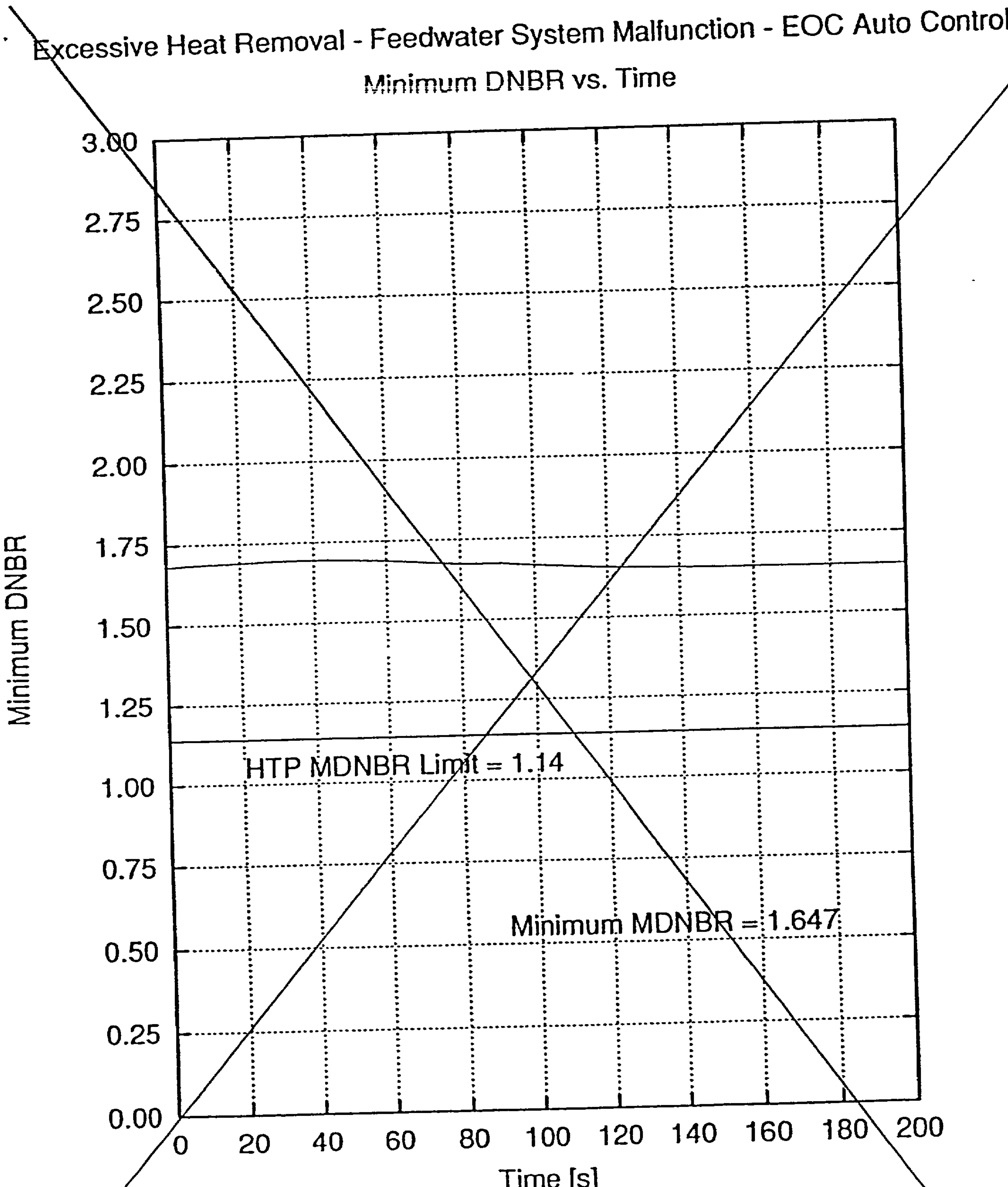

Figure 14.1.6-10

Rev. 16
12/01/2000

Excessive Heat Removal – Feedwater System Malfunction – Auto Control
Minimum DNBR vs. Time

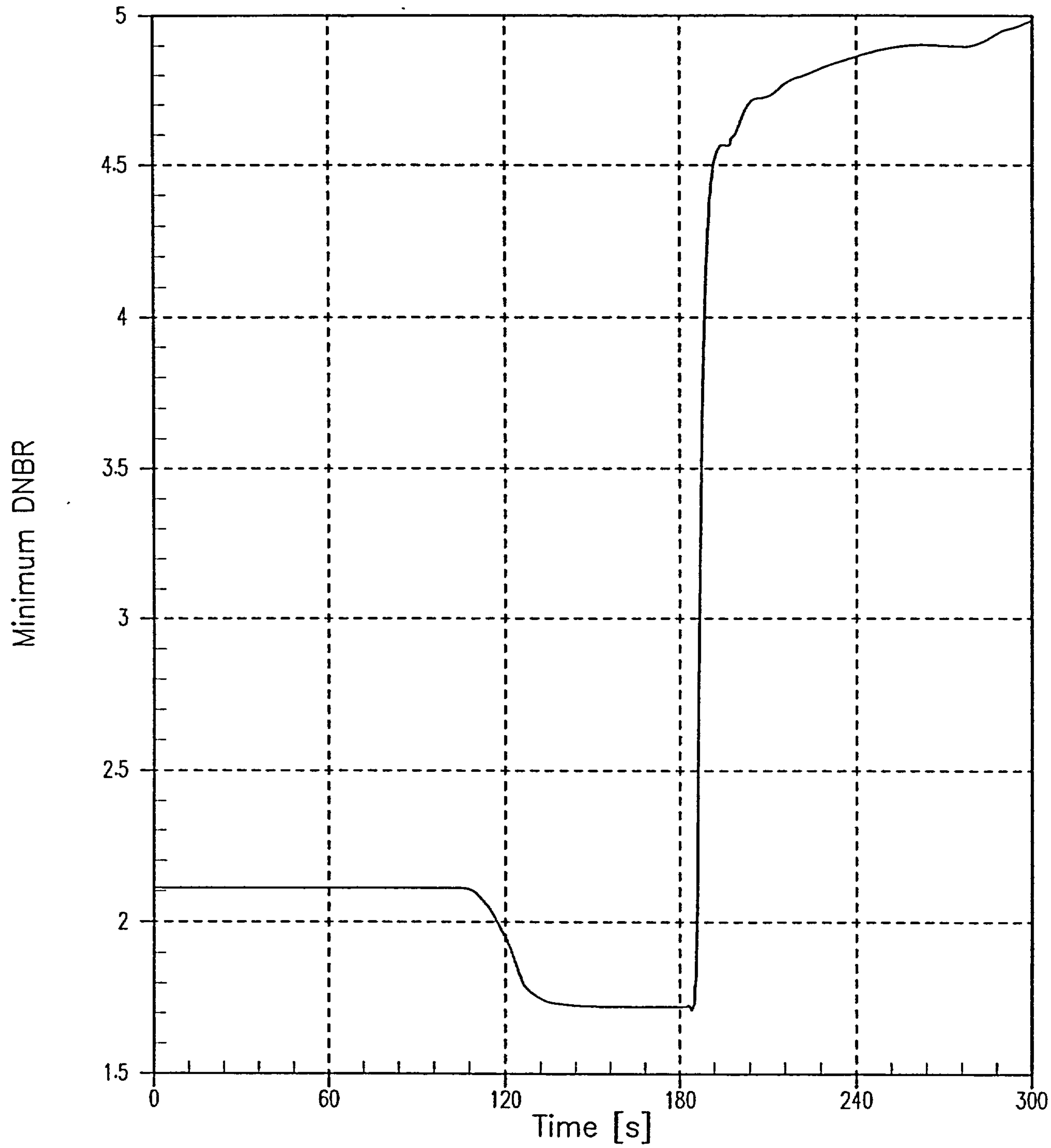

Figure 14.1.6–10

Method of Analysis

Four cases are analyzed to demonstrate the plant behavior for a 20% step increase from rated load. The first two cases are for a manually controlled reactor at beginning of cycle (BOC, α_m = zero Δk/°F) and end of cycle (EOC, α_m = –4.0E–4 Δk/°F) conditions (α_m is the moderator reactivity co-efficient). Beginning of cycle represents a condition when the plant has the smallest moderator temperature coefficient of reactivity and, therefore, the least inherent transient capability. Two cases are analyzed for an automatic control situation at BOC and EOC conditions with control rods initially inserted to the power dependent insertion limits. A conservative limit on the turbine valve opening was assumed corresponding to 1.2 times nominal steam flow at nominal steam pressure. Initial pressurizer pressure, reactor coolant average temperature and power are assumed at extreme values consistent with steady state, full-power operation, allowing for calibration and instrument errors. This results in the minimum margin to core DNB at the start of the transient. The analyses are performed using a detailed digital simulation of the plant including core kinetics, Reactor Coolant System, and the Steam and Feedwater Systems.

Results

Figures 14.1.7-1 through 14.1.7-8 illustrate the transient with the reactor in the manual control mode. As expected, for the BOC case with a very slight power increase, the core average temperature shows a large decrease. For the EOC case, there is a much larger increase in reactor power due to the moderator feedback. Both of the manual control cases demonstrate adequate ~~M~~DNBR margin.

Figures 14.1.7-9 through 14.1.7-18 illustrate the transient assuming the reactor is in automatic control. In automatic control the reactor power transient is greater than for the corresponding case in manual control. The automatic control cases still show adequate margin to the ~~M~~DNBR limit.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Excessive Load Increase	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
BOC Manual Control	1.681/[redacted]	2200/2750	751/1210
BOC Auto Control	1.430/[redacted]	2200/2750	751/1210
EOC Manual Control	1.478/[redacted]	2200/2750	751/1210
EOC Auto Control	1.438/[redacted]	2200/2750	751/1210

Conclusions

Add USAR Insert 14.1.7-2

The four cases analyzed show a considerable margin to the limiting MDNBR. It is concluded that reactor integrity is maintained throughout lifetime for the excessive load increase incident.

USAR Insert 14.1.7-1

Based on historical precedence, this event does not lead to a serious challenge to the acceptance criteria and a reactor trip is not typically generated. As such, it has been determined that a detailed reanalysis of this event is not necessary to support a core power rating of 1772 MWt. A simplified statepoint evaluation, assuming a 10% step load increase, was performed and the results confirmed that core DNB limits are not challenged following this event. The discussion presented below corresponds to the analysis previously performed for this event and is retained for historical purposes.

USAR Insert 14.1.7-2

Furthermore, the results of a simplified statepoint evaluation performed for a 10% step load increase with a nominal core power of 1772 MWt confirm that the core thermal limit lines are not challenged, and that the minimum DNBR during this transient will remain above the safety analysis limit value.

14.1.8 LOSS OF REACTOR COOLANT FLOW

Accident Description

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

- Low voltage on pump power supply bus
- Pump circuit breaker opening (low frequency on pump power supply bus opens pump circuit breaker)
- Low reactor coolant flow

These trip circuits and their redundancy are further described in Section 7.2, Reactor Control and Protection System.

Simultaneous loss of electrical power to all RXCPs at full power is the most severe credible loss-of-coolant flow condition. For this condition, reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent fuel failure, Reactor Coolant System overpressure, and prevent the DNBR from going below its limit.

EPLACE WITH NSERT 4.1.8-1

Two types of flow coastdown accidents were analyzed, loss of two RXCPs at nominal frequency and loss of two RXCPs at low frequency. These two types of flow coastdown analyses are described separately under Loss of Reactor Coolant Flow-Nominal Frequency and under Loss of Reactor Coolant Flow Low Frequency.

Loss of Reactor Coolant Flow-Nominal Frequency

Method of Analysis

The following nominal frequency loss of coolant flow case is analyzed: Loss of two pumps from a Reactor Coolant System heat output of ~~102%~~ 100% of ~~1650~~ 1772 MWt with two loops operating. This case represents the worst credible coolant flow loss.

The normal power supplies for the pumps are the two buses connected to the generator, each of which supplies power to one of the two pumps. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines. Therefore, the simultaneous loss of power to both reactor coolant pumps is a highly unlikely event.

Following any turbine trip, when there are no electrical faults requiring tripping the generator from the grid, the generator remains connected to the grid for at least thirty seconds. Since both pumps are not on the same bus, a single bus fault does not result in the loss of both pumps.

REPLACE WITH INSERT 14.1.8-2

A full plant simulation is used in the analysis to compute the core average and hot spot heat flux transient responses, including flow coastdown, temperature, reactivity, and control rod insertion effects.

These data are then used in a detailed thermal hydraulic computation to compute the margin to DNB. This computation solves the continuity, momentum, and energy equations of fluid flow together with the DNB correlation. The following assumptions are made in the calculations:

a. The initial operating conditions, which are assumed to be most adverse with respect to the margin to DNB, are maximum steady-state power level, minimum steady-state pressure, and maximum steady-state inlet temperature.

b. The largest negative initial value of the Doppler coefficient (–2.32E–5 Δk/°F) and a zero moderator coefficient (0.0 Δk/°F) are assumed since these result in the maximum heat flux during the initial part of the transient, when the minimum DNB ratio is reached.

c. A reactor trip is actuated by low flow. The time from the initiation of low-flow signal to initiation of RCCA motion is 0.6 seconds. The trip signal is assumed to be initiated at 87% of full-loop flow, allowing at least 3% for flow instrumentation errors.

   Upon reactor trip, it is also assumed that the most reactive RCCA is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity.

d. The overall heat transfer between the fuel and the water varies considerably during the transient mostly as a result of the change of fuel gap conductance. A conservatively evaluated overall heat transfer coefficient is used in the analysis.

Results

REPLACE WITH INSERT 14.1.8-3

Reactor coolant flow coastdown curve is shown in Figure 14.1.8-1. Reactor coolant flow is calculated based on a momentum balance in the Reactor Coolant System combined with a pump momentum balance.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Loss of Flow	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
2/2 Pump Trip	[redacted]/1.14	[redacted]/2750	[redacted]/1210

Figures 14.1.8-2 and 14.1.8-3 show the nuclear power and the average heat flux response for the two-pump loss of flow. Figure 14.1.8-4 shows the MDNBR as a function of time.

Underfrequency

Loss of Coolant Flow - ~~Low Frequency~~

Method of Analysis

The underfrequency event is analyzed using a systems analysis that calculates the loop and core flow, nuclear power, and primary system pressure and temperature transients. The MDNBR is calculated by performing a detailed fuel thermal hydraulic simulation using as transient forcing functions the core heat flux, core flow, core inlet temperature, and Reactor Coolant System pressure from the systems analysis.

REPLACE WITH INSERT 14.1.8-4

a. The initial operating conditions, which are assumed to be most adverse with respect to the margin to DNB, are maximum steady-state power level, minimum steady-state pressure, and maximum steady-state average temperature.

b. A conservatively large absolute value of the Doppler only power co-efficient and a zero moderator coefficient (0.0 $\Delta k/°F$) are assumed since these result in the maximum hot channel heat flux during the initial part of the transient, when the MDNBR is reached.

c. A constant frequency decay rate of 5 Hz/sec is assumed. Reference 3 determined that this is the maximum credible frequency decay rate that could occur on a typical electrical grid. Analysis of the Wisconsin-Upper Michigan transmission system indicates that the worst-case frequency decay rate is ~~be~~ approximately 2 Hz/sec (see Reference 4). Therefore, 5 Hz/sec is a very conservative decay rate. In addition, the assumption of a constant rate is conservative, since Reference 3 also shows that the expected grid frequency decay rate actually decreases during the transient.

   Prior to the opening of the RXCP breaker, the RXCP speed is assumed to be directly proportional to the power supply frequency. As discussed in Reference 5, this is a conservative assumption, since the speed coastdown will lag the frequency coastdown due to the effects of pump inertia and induction motor slip. During steady state operation the pump motor speed is below the synchronous speed because of induction motor slip. After the frequency decay starts, the deceleration of the pump-motor-flywheel combination provides a positive driving torque to the pump so that the required electrical torque decreases. The reduction in electrical torque reduces the induction motor slip, thus resulting in a higher speed than that assumed in the analysis. The degree of conservatism varies directly with the assumed decay rate because the inertia torque increases directly with the decay rate. At 5 Hz/sec the expected speed is approximately 1.2% higher than the analysis value.

   Reactor Coolant System flow is calculated based on a momentum balance in the Reactor Coolant System combined with a pump momentum balance.

REPLACE WITH INSERT 4.1.8-5

d. No credit is taken for the RXCP trip on underfrequency.

e. Upon reactor trip, it is assumed that the most reactive RCCA is stuck at its fully withdrawn position, resulting in a minimum insertion of negative reactivity.

REMOVE

In Reference 7, the NRC approved use of the WPS loss of flow underfrequency trip methodology.

Results

REPLACE WITH INSERT 14.1.8-6

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Loss of Flow	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
Underfrequency Trip	[redacted]/1.14	[redacted]/2750	[redacted]/1210

Figures 14.1.8-5 through 14.1.8-8 shows the nuclear power, average channel heat flux, core flow, and MDNBR transient responses for the underfrequency event.

MDNBR is always above the MDNBR limit. Therefore, fuel rod integrity and safe plant shutdown are ensured by an underfrequency trip setting of 54.5 Hz.

Conclusions

Since DNB does not occur, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected the plant can be returned to service in the normal manner. The absence of fuel failures would, of course, be verified by analysis of reactor coolant samples. In the loss of reactor coolant flow accidents, it has been shown that there is adequate reactor coolant flow to maintain a MDNBR greater than the MDNBR limit.

Locked Rotor Accident

Accident Description

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

REPLACE WITH INSERT 14.1.8-7

Method of Analysis

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the RXCPs seizes, the plant is assumed to be in operation under the steady-state operating conditions that are most adverse with respect to MDNBR margin. The plant is assumed to be operating at maximum steady-state power, minimum steady-state pressure, and maximum steady-state core inlet temperature.

After pump seizure, nuclear power is rapidly reduced because of void shutdown and the RCCA insertion upon reactor trip.

No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

REPLACE WITH INSERT 14.1.8-8

The pressurizer safety valves start operating at 2500 psia and relieve steam at their rated capacities. Additional sensitivity analyses were performed at pressurizer safety valve settings of +6% and –4% of the nominal setpoint to account for the effects of steam accumulation and setpoint drift. The critical safety parameters were shown to be acceptable under these assumptions.

Calculations of the extent of DNB in the core during the accident are performed using the heat flux, the coolant flow decay and the coolant pressure and temperature as transient forcing functions.

In order to estimate the severity of the accident in the core as far as the integrity of the fuel rods is concerned the thermal behavior of the fuel located at the hot spot after DNB was investigated. Results obtained from an analysis of this "hot spot" condition represent the upper limit with respect to clad temperature, clad melting and zirconium-steam reaction.

Results

REPLACE WITH INSERT 14.1.8-9

The coolant flow through the core is rapidly reduced to < 50% of its initial value (see Figure 14.1.8-9).

The reactor coolant pressure vs. time for a locked rotor accident is shown in Figure 14.1.8-11. The minimum DNBR for a fuel rod having an initial $F_{\Delta H}$ value of [redacted] is shown in Figure 14.1.8-12. The [redacted] $F_{\Delta H}$ rod reaches a MDNBR of slightly above the MDNBR limit. The MDNBR for the 1.70 $F_{\Delta H}$ fuel rod is less than the MDNBR limit, and the fuel rod is assumed to fail. Up to 40% of the fuel rods in the core can go below the MDNBR limit with acceptable radiological consequences (Reference 8). Fuel rod power census curves are generated for each reload to assess the percentage of fuel rods that are expected to go below the MDNBR limit of this accident.

REMOVE

Figure 14.1.8-13 shows the clad temperature transient at the hot spot. Since in the worst case examined, the clad temperature does not exceed 1800°F, it is not necessary to consider the possibility of a zirconium-steam reaction. The zirconium-steam reaction is only significant above this temperature.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Valve/Acceptance Criterion):

	% Fuel Rods < DNB Limit	Max Clad Temp. (°F)	RCS Pressure (psia)	MS Pressure (psia)
Locked Rotor	[redacted]	[redacted]/2700	[redacted]/2750	1044/1210

* Percentage of Fuel Rods with $F_{\Delta H} \geq$ [redacted]

Conclusions

Since the peak pressure reached during the transient is < 110% of design, the integrity of the Reactor Coolant System is not endangered. The pressure can be considered as an upper limit because of the following conservative assumptions used in the study:

1. Credit is not taken for the negative moderator coefficient.
2. It is assumed that the pressurizer relief valves were inoperative.
3. The steam dump is assumed to be inoperative.

The peak clad temperature calculated for the hot spot, can also be considered an upper limit because of the following:

1. The hot spot is assumed to be in DNB at the start of the accident.
2. A high gap coefficient is used during the transient.
3. The nuclear heat released in the fuel at the hot spot is based on a zero moderator coefficient.

14.1.9 LOSS OF EXTERNAL ELECTRICAL LOAD

Section 14.1.9 Changes suggested later.

OUT OF SCOPE

Accident Description

The loss of external electrical load may result from an abnormal increase in network frequency, opening of the main breaker from the generator, which causes a rapid large Nuclear Steam Supply System load reduction by the action of the turbine control, or by a trip of the turbine generator.

The plant is designed to accept a full-load rejection without actuating a reactor trip. The automatic steam dump system with 85% steam dump capacity (40% to the condenser and 45%

Inserts for USAR Section 14.1.8

Insert 14.1.8-1

Two types of loss of flow accidents were analyzed: complete loss of flow due to the loss of two RXCPs and complete loss of flow due to a frequency decay (underfrequency). These two types of flow coastdown analyses are described separately under Loss of Reactor Coolant Flow – Nominal Frequency and under Loss of Reactor Coolant Flow – Underfrequency.

Insert 14.1.8-2

This transient is analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel.

This event is analyzed with the Revised Thermal Design Procedure (RTDP).

The following assumptions are made in the calculations:

a. Consistent with the RTDP methodology, the initial operating conditions are assumed to be at their nominal values, including the steady-state power level, RCS pressure, and RCS vessel average temperature. Minimum Measured Flow (MMF) is also assumed.

b. The largest negative value of the Doppler Power Coefficient and a zero moderator coefficient are assumed since these maximize the heat flux during the initial part of the transient, when the minimum DNBR is reached.

c. A reactor trip is actuated by low flow. The time from the initiation of the low flow signal to initiation of RCCA motion is 0.75 seconds. The trip signal is assumed to be initiated at 86.5% of full loop flow, allowing 3.5% for flow instrumentation errors.

d Upon reactor trip, it is assumed that the most reactive RCCA is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity.

e. No credit is taken for the reactor trip on reactor coolant pump motor breaker open due to low voltage, or the reactor trip directly on undervoltage.

Insert 14.1.8-3

The reactor coolant flow coastdown is presented in Figures 14.1.8-1 and 14.1.8-2. Reactor coolant flow is calculated based on a momentum balance in the Reactor Coolant System combined with a pump momentum balance. The nuclear power and core average heat flux transients are presented in Figures 14.1.8-3 and 14.1.8-4, and the pressurizer pressure and RCS loop temperature transients are shown in Figures 14.1.8-5 and 14.1.8-6. Finally, the hot channel heat flux and DNBR transients are presented in Figures 14.1.8-7 and 14.1.8-8.

The acceptance criteria for this event, the minimum DNBR limit and the maximum RCS pressure limit of 2750 psia, are met.

Insert 14.1.8-4

As with the complete loss of flow case, the underfrequency transient is analyzed with the RETRAN and VIPRE computer codes, as well as the RTDP methodology.

The following assumptions are made in the calculations:

a. Consistent with the RTDP methodology, the initial operating conditions are assumed to be at their nominal values, including the steady-state power level, RCS pressure, and RCS vessel average temperature.

b. The largest negative value of the Doppler Power Coefficient and a zero moderator coefficient are assumed since these maximize the heat flux during the initial part of the transient, when the minimum DNBR is reached.

Insert 14.1.8-5

d. A reactor trip is actuated by low flow. The time from the initiation of the low flow signal to initiation of RCCA motion is 0.75 seconds. The trip signal is assumed to be initiated at 86.5% of full loop flow, allowing 3.5% for flow instrumentation errors. No credit is taken for the RXCP trip on underfrequency.

Insert 14.1.8-6

The reactor coolant flow coastdown is presented in Figures 14.1.8-9 and 14.1.8-10. Reactor coolant flow is calculated based on a momentum balance in the Reactor Coolant System combined with a pump momentum balance. The nuclear power and core average heat flux transients are presented in Figures 14.1.8-11 and 14.1.8-12, and the pressurizer pressure and RCS loop temperature transients are shown in Figures 14.1.8-13 and 14.1.8-14. Finally, the hot channel heat flux and DNBR transients are presented in Figures 14.1.8-15 and 14.1.8-16, respectively.

The acceptance criteria for this event, the minimum DNBR limit and the maximum RCS pressure limit of 2750 psia, are met.

Insert 14.1.8-7

The Locked Rotor transient is analyzed with three computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code is then used to calculate the thermal behavior of the fuel located at the core hot spot based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The FACTRAN computer code includes a film boiling heat transfer coefficient. Finally, the VIPRE code is used to calculate the "Rods-in-DNB" using the nuclear power and RCS flow from RETRAN.

At the beginning of the postulated RCP Locked Rotor accident, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., a maximum steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature. The analysis is performed to bound operation with a maximum uniform steam generator tube plugging level of 10%. However, a core flow reduction of 1.1 percent, which addresses the potential reactor coolant flow asymmetry associated with a maximum loop-to-loop steam generator tube plugging imbalance of 10 percent, was applied.

A conservatively large absolute value of the Doppler-only Power Coefficient is used, along with the most-positive moderator temperature coefficient limit for full power operation (0 pcm/°F). These assumptions maximize core power during the initial part of the transient when the peak RCS pressures and hot spot results are reached.

A conservatively low trip reactivity value (3.5% $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (1.8 seconds to dashpot).

A loss of offsite power is assumed with the unaffected RCP losing power instantaneously at the time of reactor trip.

Insert 14.1.8-8

The pressurizer safety valves start operating at 2500 psia and relieve steam at their rated capacities. A safety valve set pressure tolerance of +1% and a set pressure shift of +1% are modeled. Also, a sensitivity analysis was performed assuming a pressurizer safety valve tolerance of +6% and a set pressure shift of +1%. The critical safety parameters were shown to be acceptable under these assumptions.

Insert 14.1.8-9

Figures 14.1.8-17 through 14.1.8-25 illustrate the transient response for the Locked Rotor event. The results shown are for the peak RCS pressure/PCT case. The coolant flow through the core is rapidly reduced to less than fifty percent of its initial value (Figure 14.1.8-17). As shown in Figure 14.1.8-22, the peak RCS pressure is less than the acceptance criterion of 2750 psia. Also, Figure 14 1.8-25 shows that the peak cladding temperature is considerably less than the limit of 2700°F. The zirconium-water reaction at the hot spot meets the criterion of less than 16% zirconium-water reaction. This transient trips on a low primary reactor coolant flow trip setpoint which is assumed to be 86.5% of the initial flow.

Calculations performed with the VIPRE code demonstrate that the maximum percentage of rod-in-DNB for this event is less than 50%. This calculation is based upon the RTDP methodology and utilizes a generic rod census curve.

REPLACE WITH NEW FIGURE 14.1.8-1

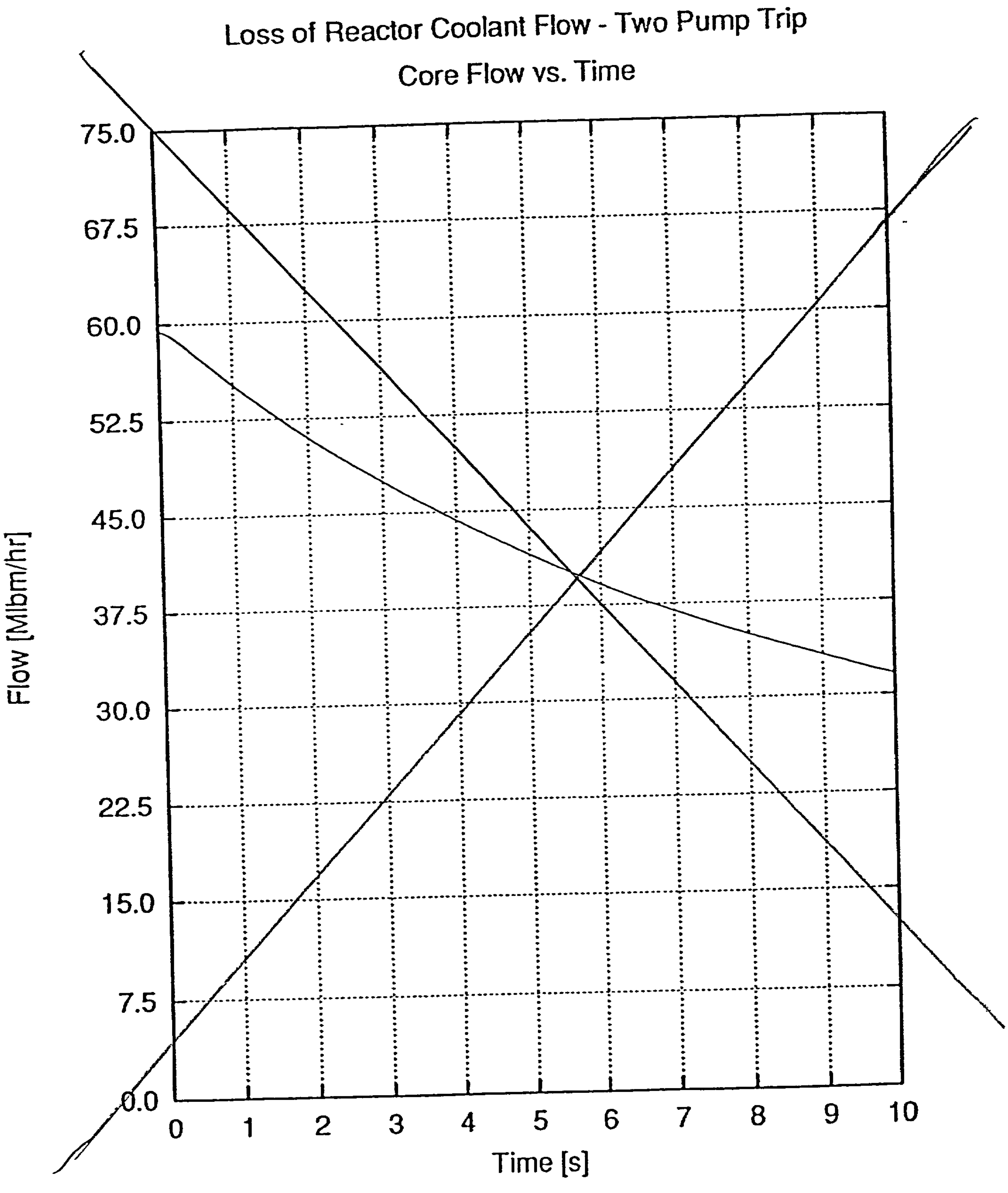

Figure 14.1.8-1

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Complete Loss of Flow - Two Pumps Coasting Down (CLOF)

Total Core Inlet Flow vs. Time

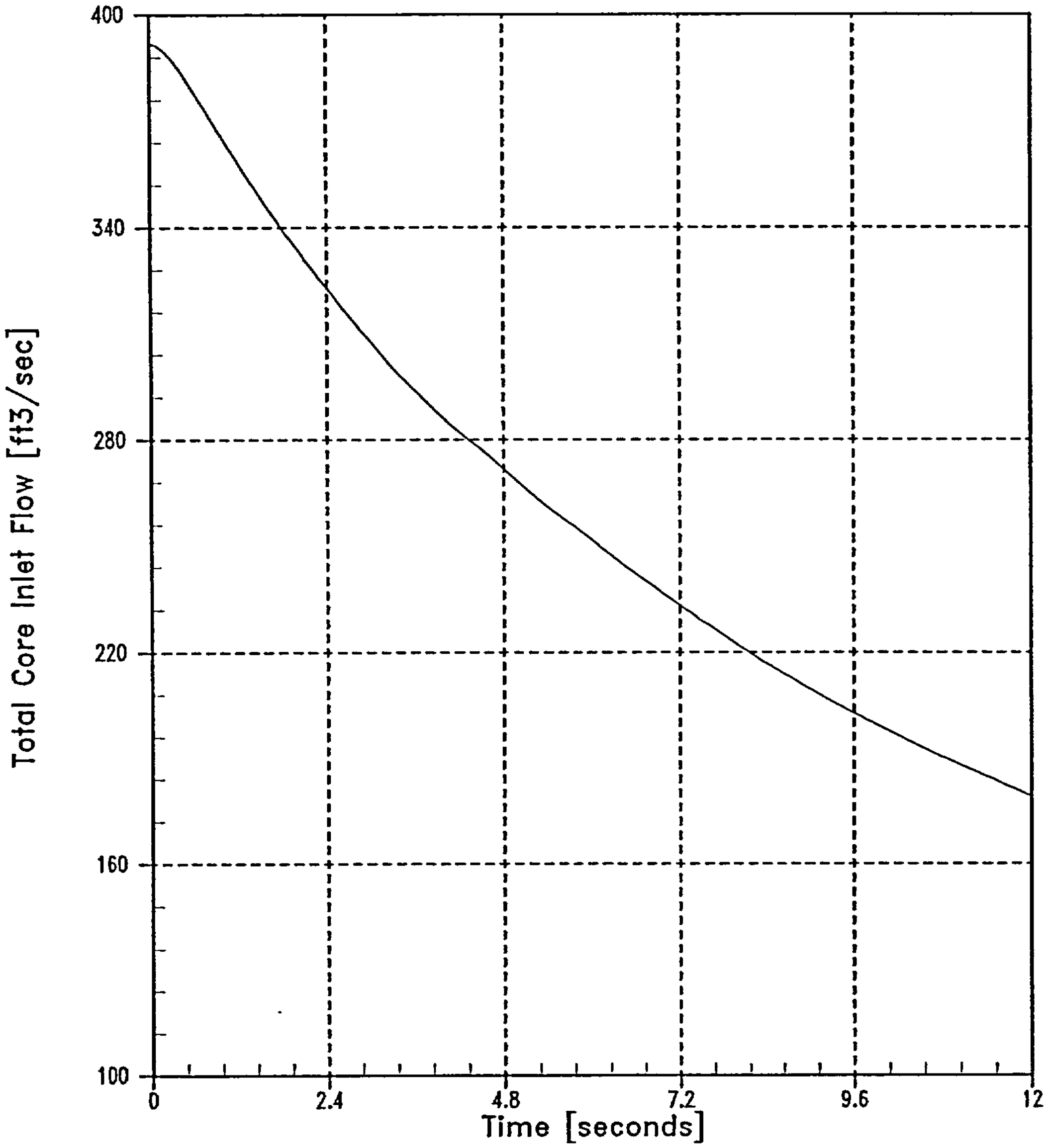

Figure 14.1.8-1

REPLACE WITH NEW FIGURE 14.1.8-2

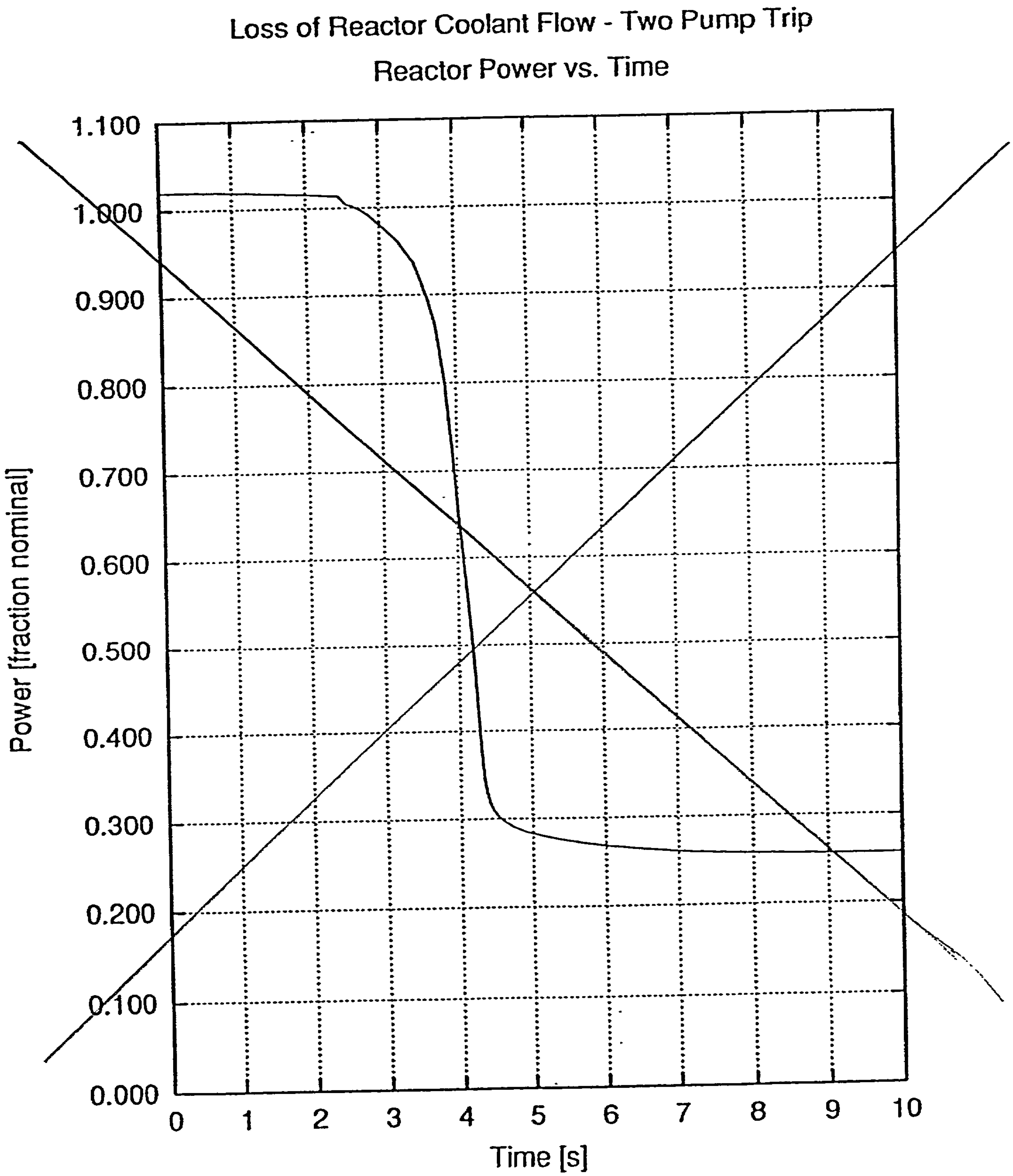

Figure 14.1.8-2

Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

RCS Loop Flow vs. Time

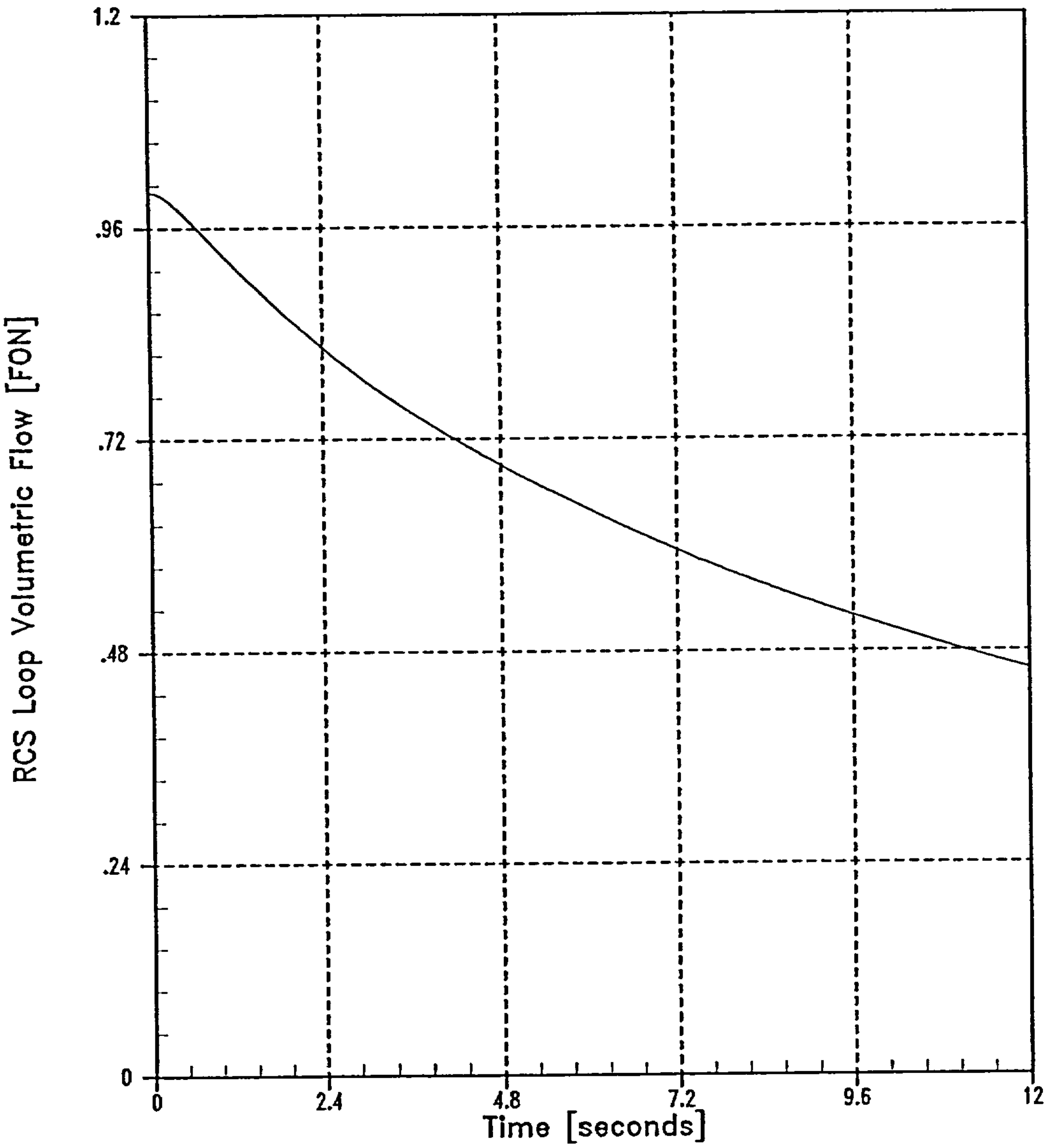

Figure 14.1.8-2

REPLACE WITH NEW FIGURE 14.1.8-3

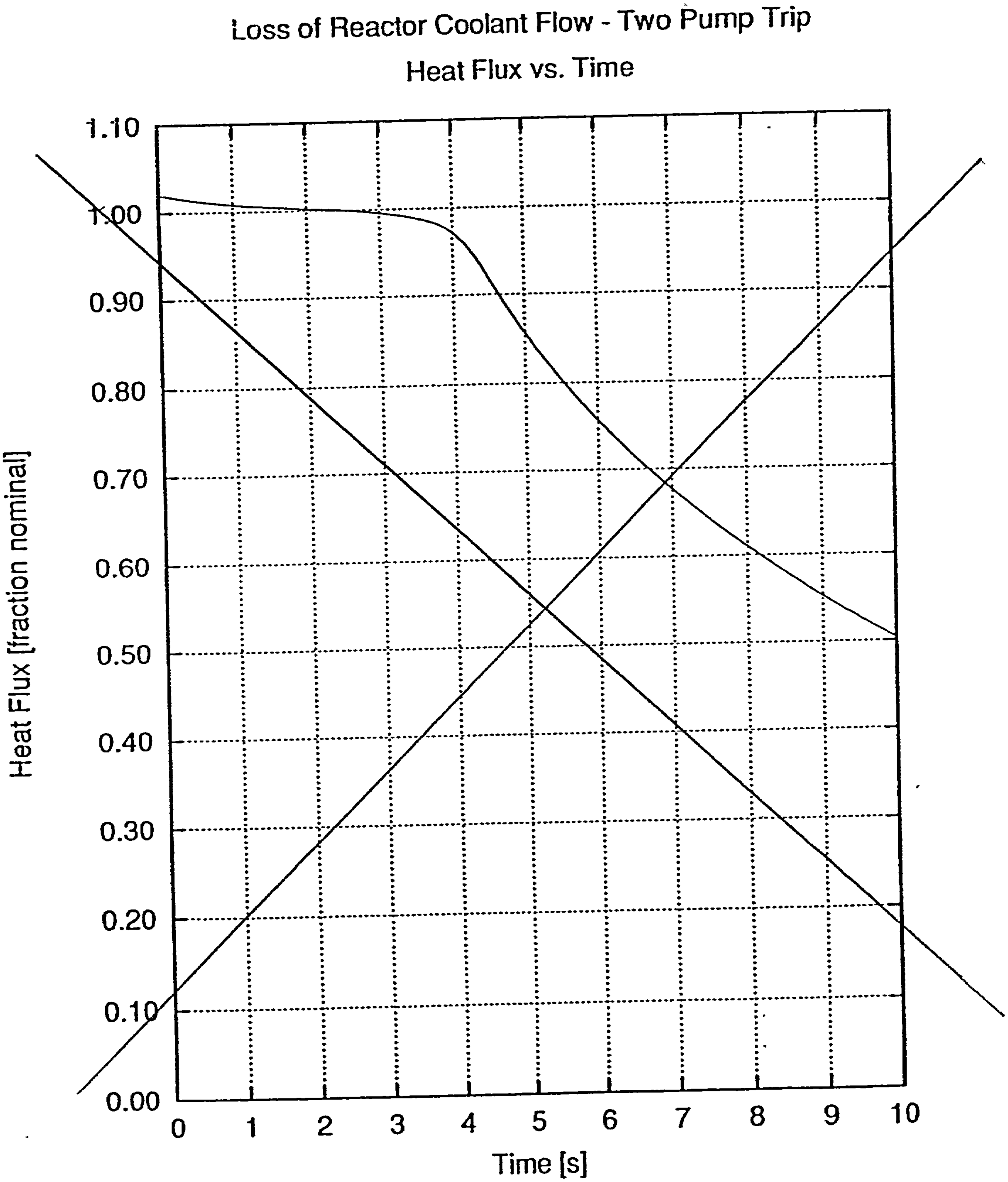

Figure 14.1.8-3

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Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

Nuclear Power vs. Time

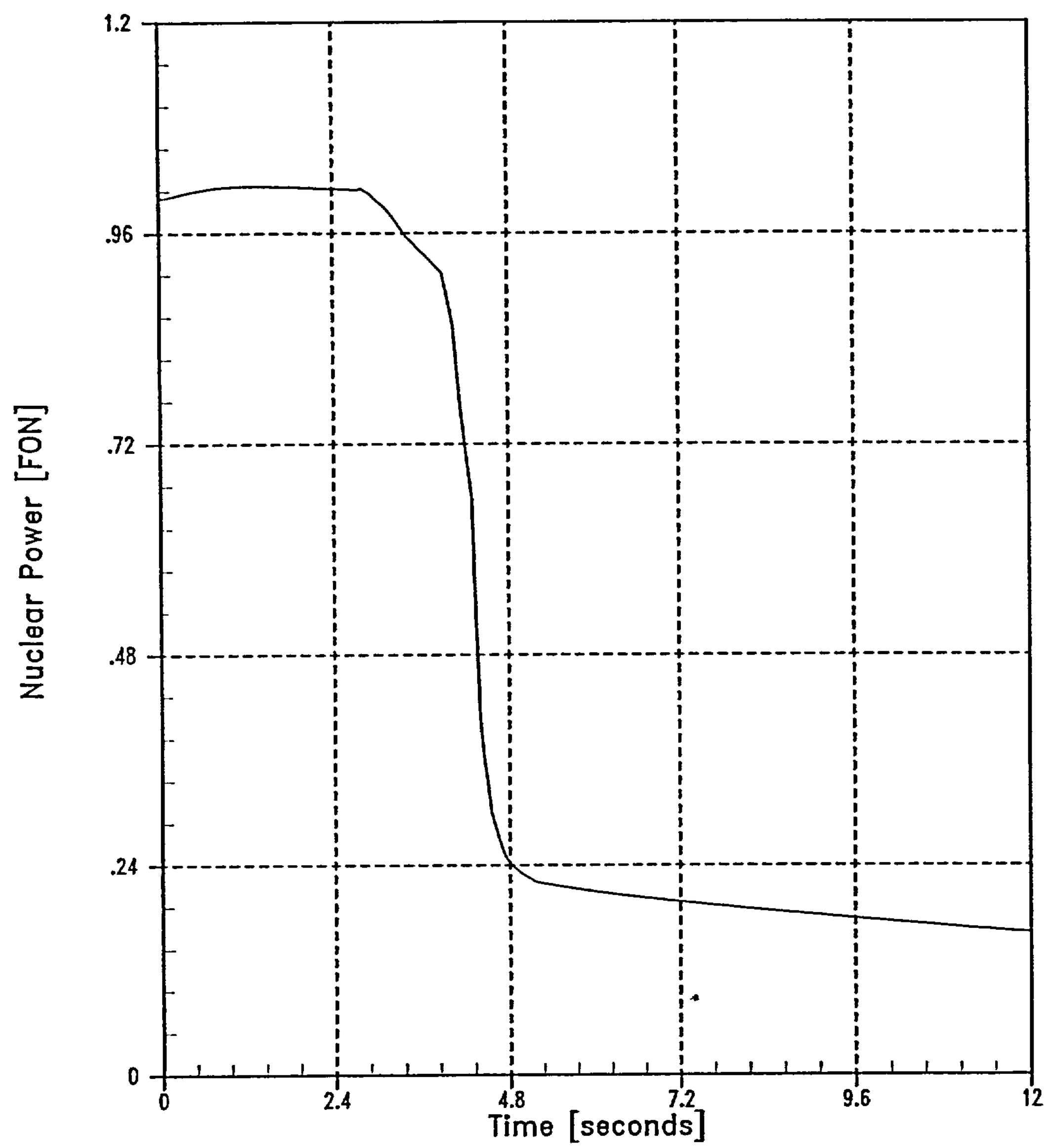

Figure 14.1.8-3

REPLACE WITH NEW FIGURE 14.1.8-4

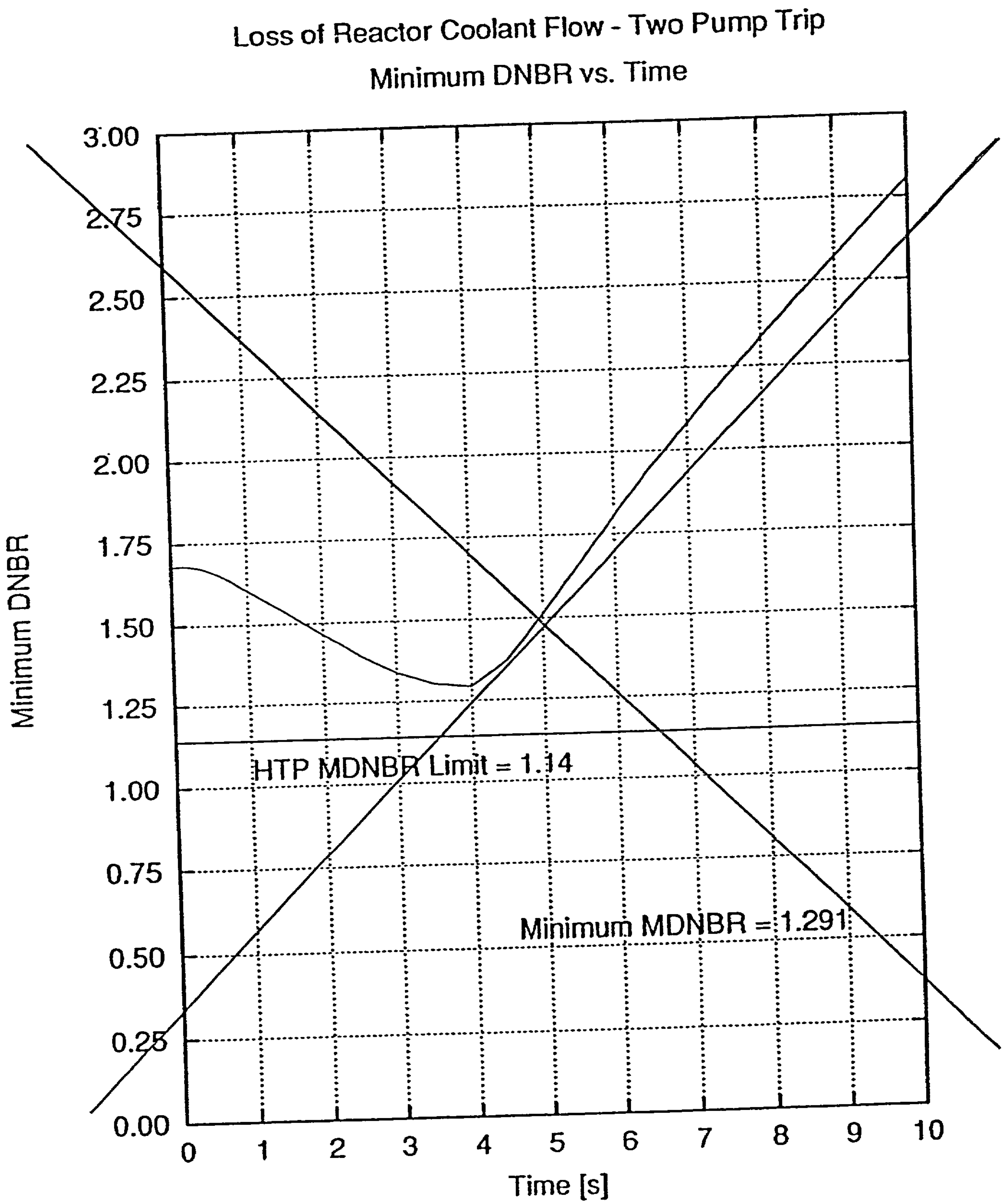

Figure 14.1.8-4

Rev. 16
12/01/2000

Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

Core Average Heat Flux vs. Time

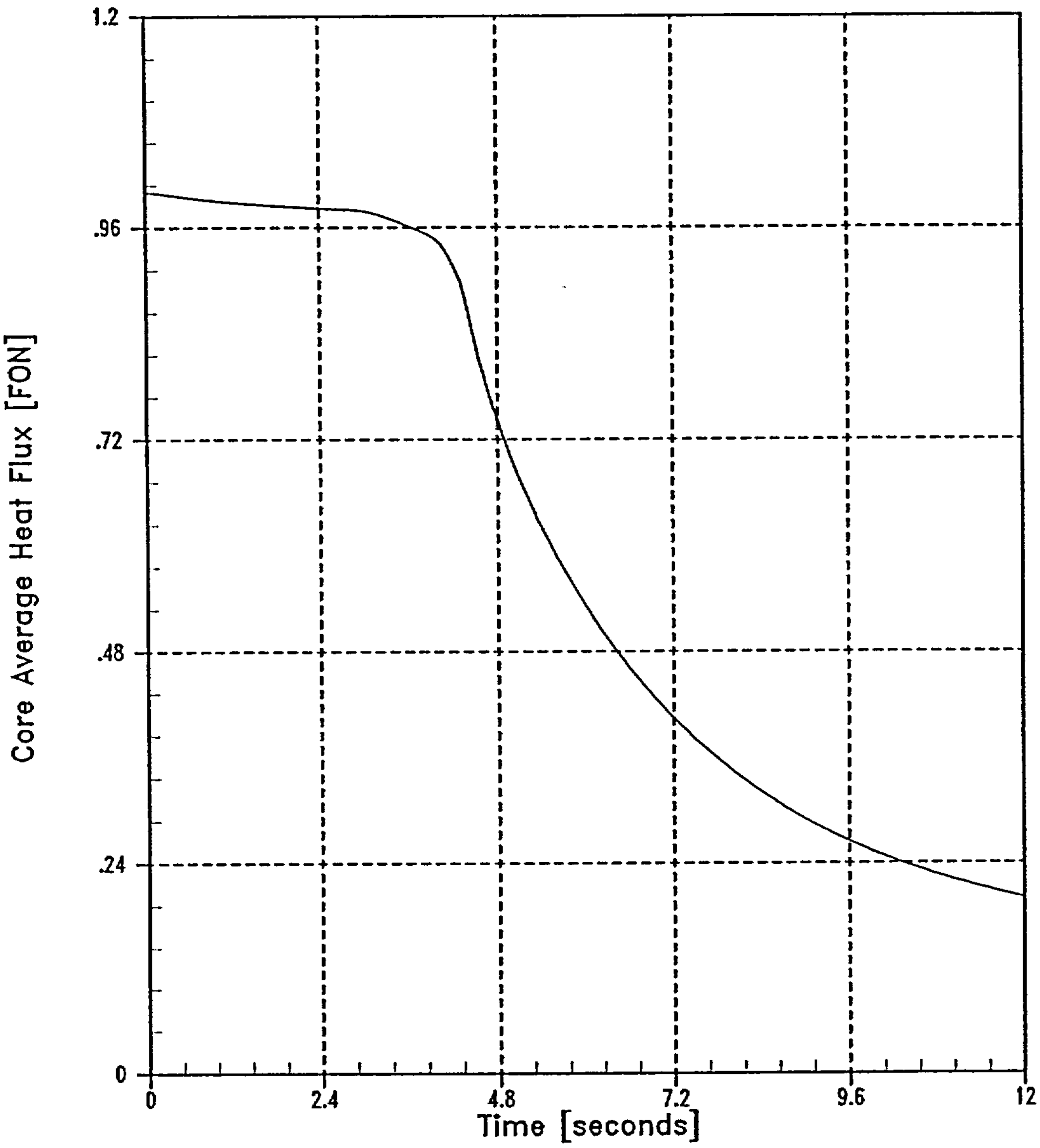

Figure 14.1.8-4

REPLACE WITH NEW FIGURE 14.1.8-5

Loss of Reactor Coolant Flow - Underfrequency Trip

Core Flow vs. Time

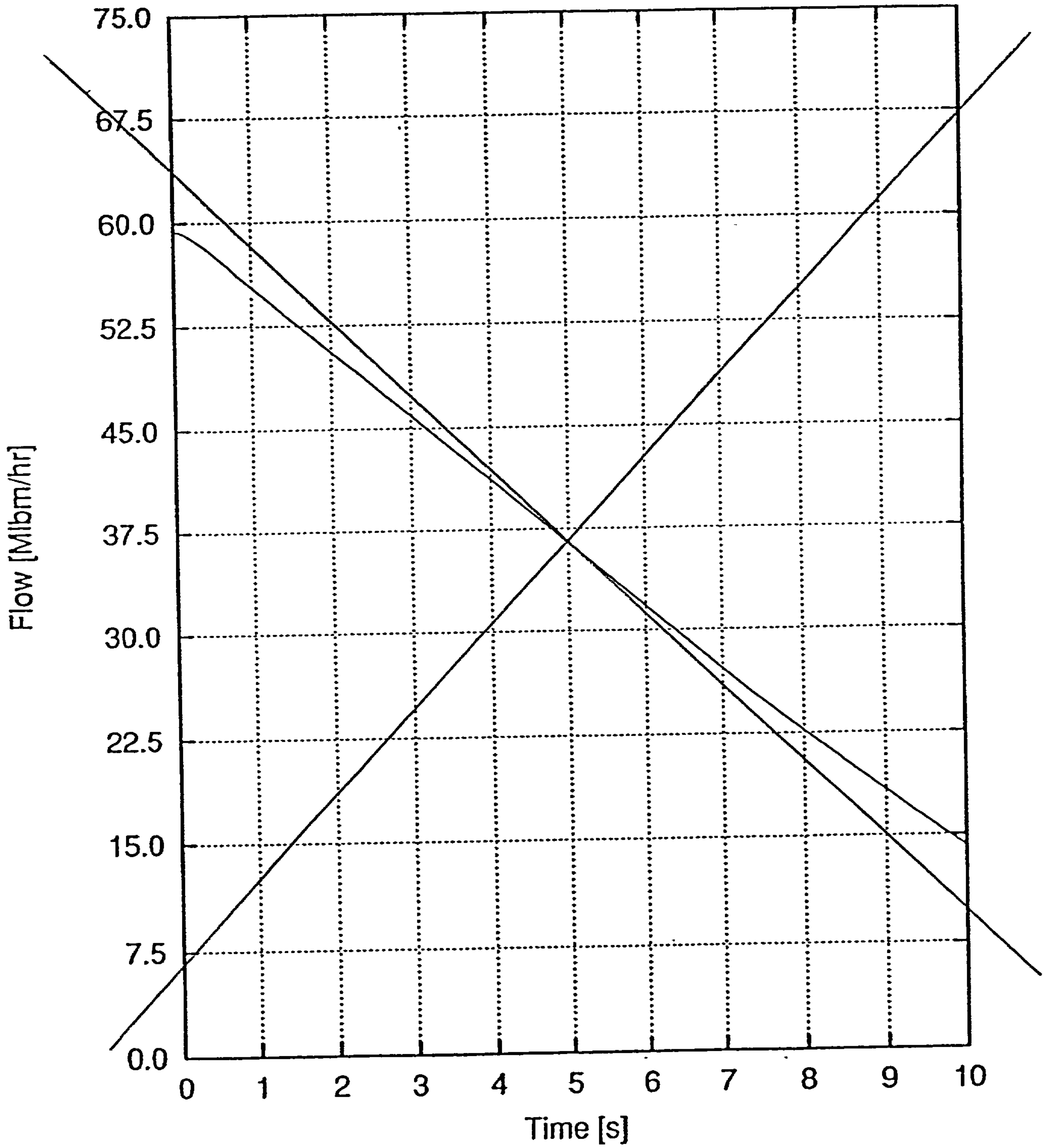

Figure 14.1.8-5

Rev. 16
12/01/2000

Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

Pressurizer Pressure vs. Time

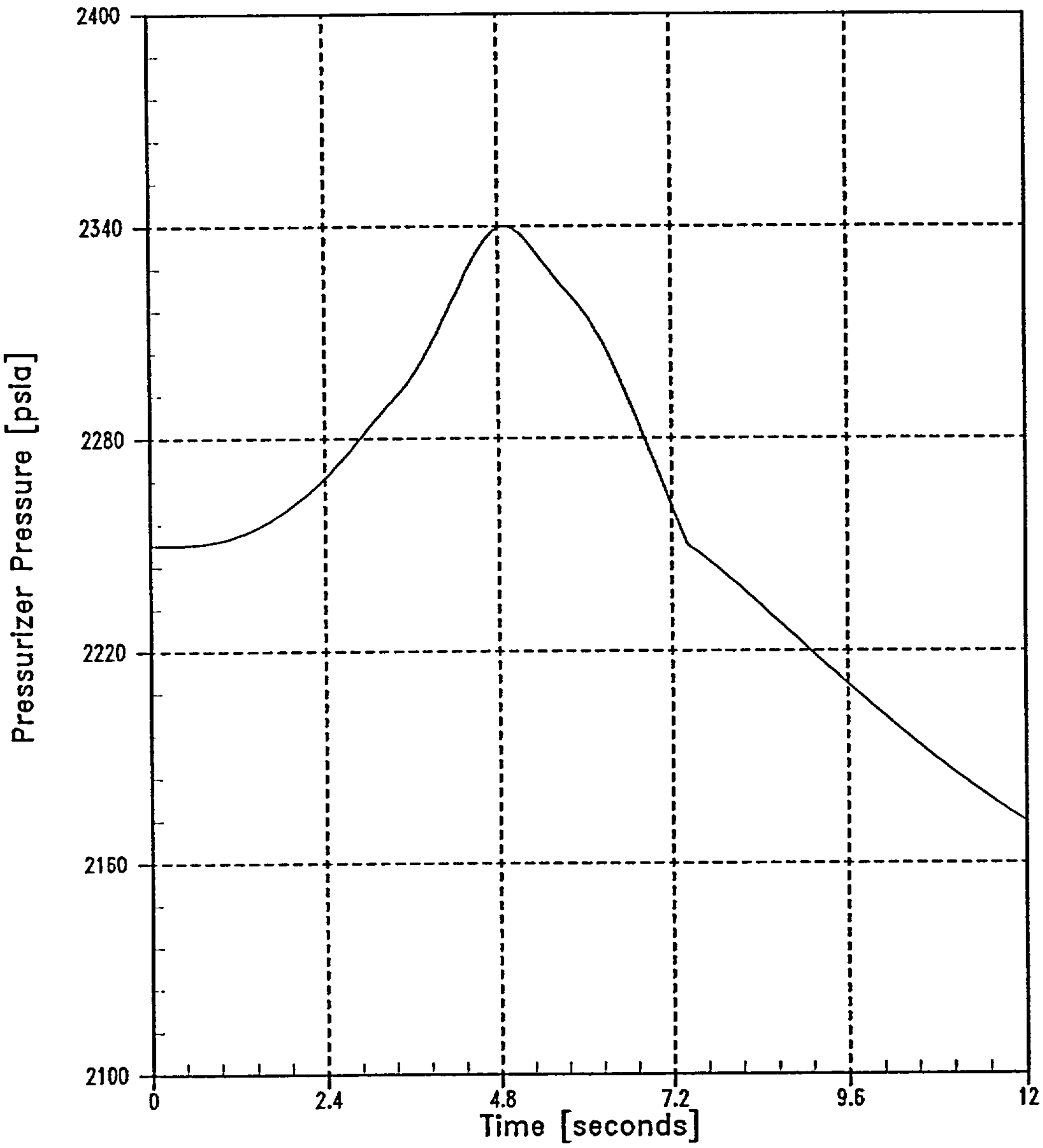

Figure 14.1.8-5

REPLACE WITH NEW FIGURE 14.1.8-6

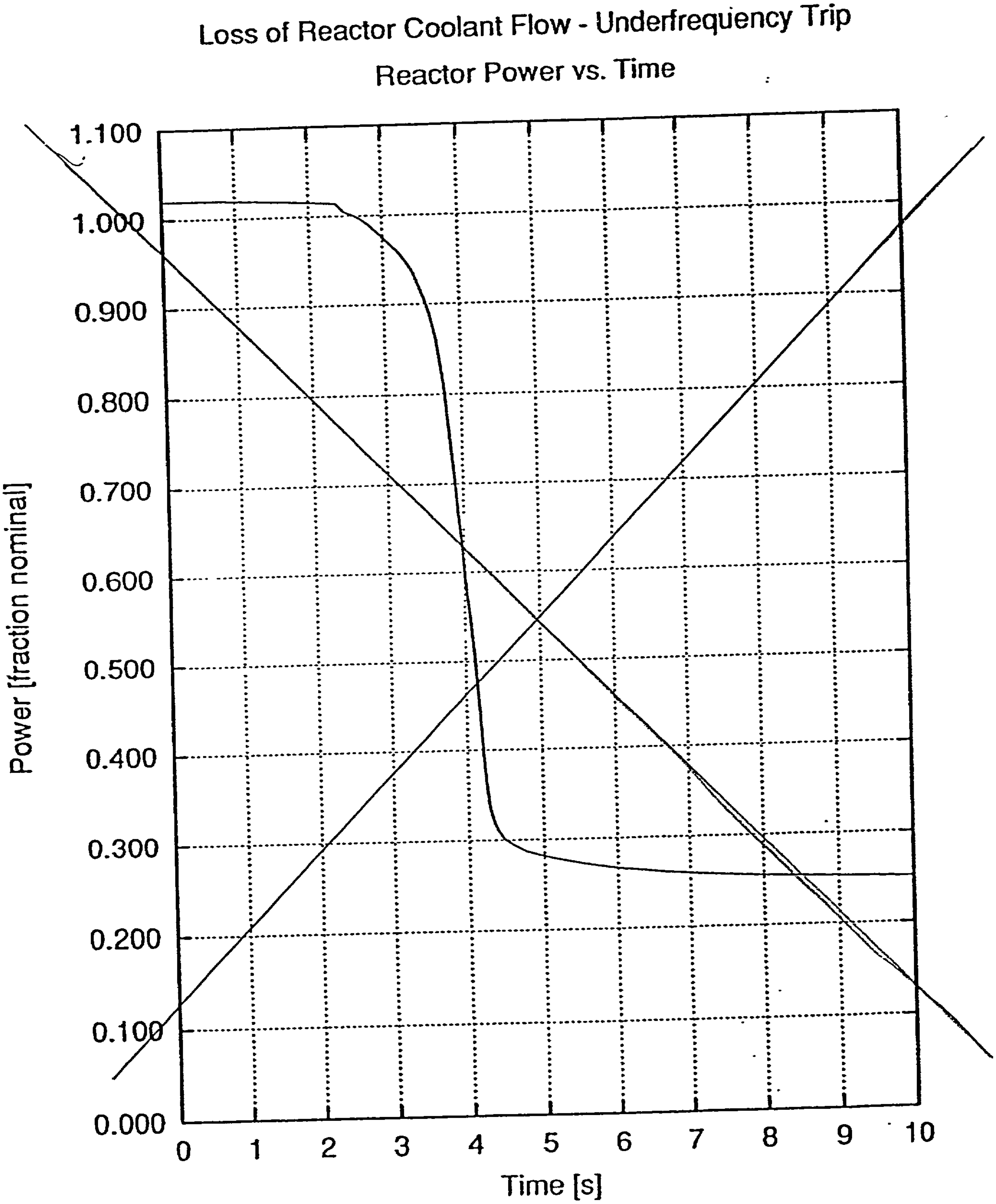

Figure 14.1.8-6

Rev. 16
12/01/2000

Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

RCS Faulted Loop Temperature vs. Time

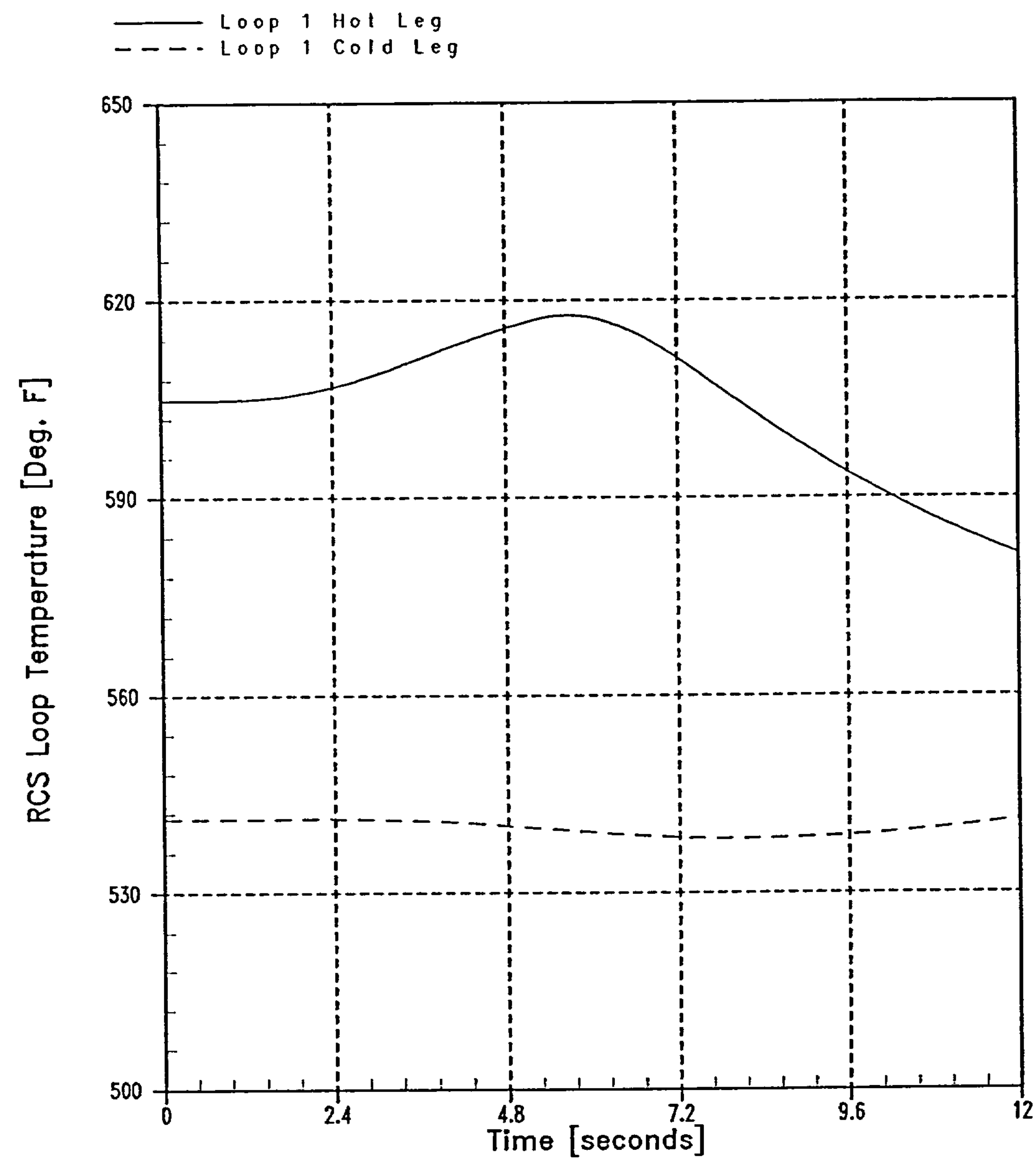

Figure 14.1.8-6

REPLACE WITH NEW FIGURE 14.1.8-7

Loss of Reactor Coolant Flow - Underfrequency Trip

Heat Flux vs. Time

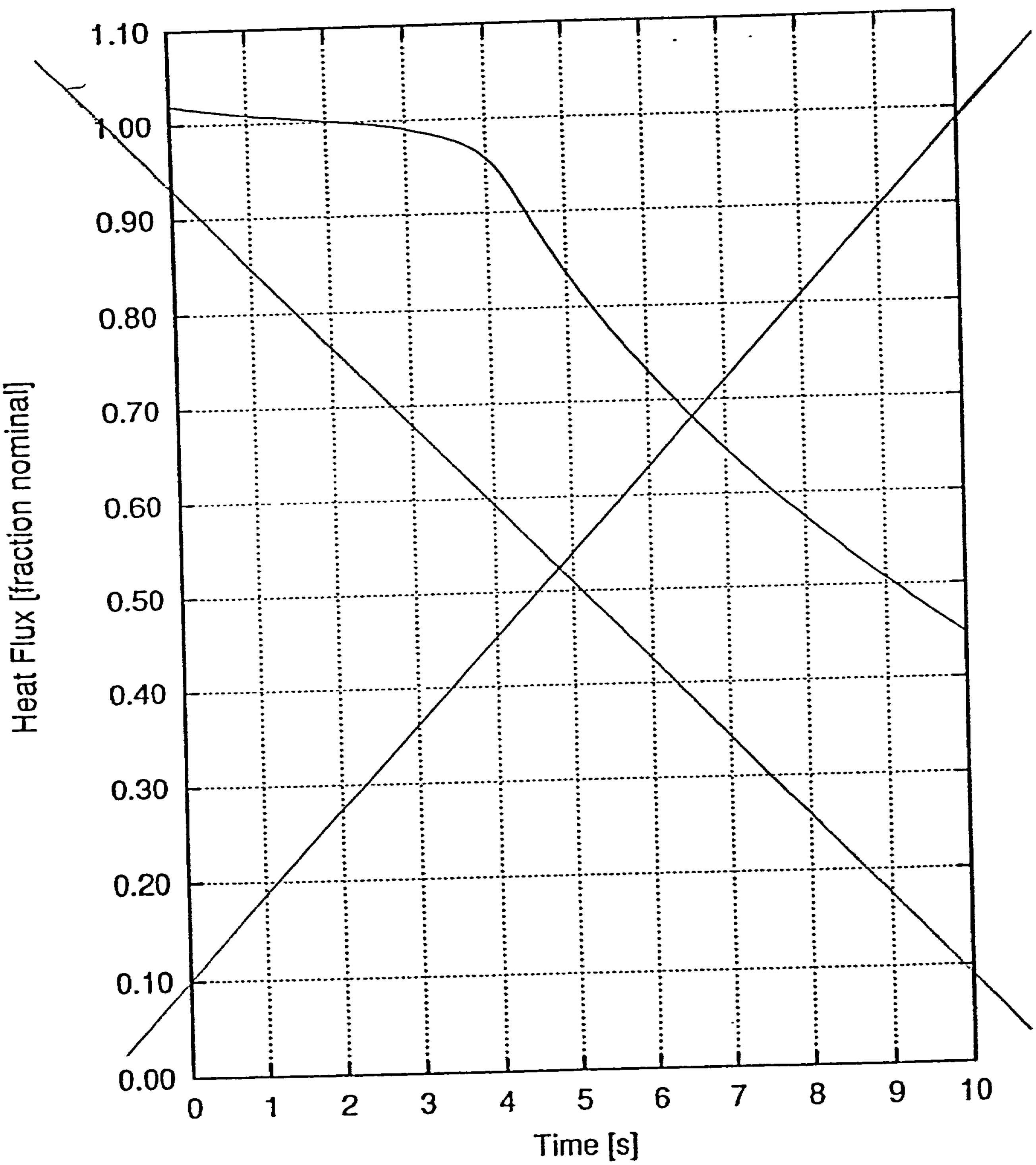

Figure 14.1.8-7

Rev. 16
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Hot Channel Heat Flux vs. Time

Complete Loss of Flow, Two Pumps Coasting Down (CLOF)

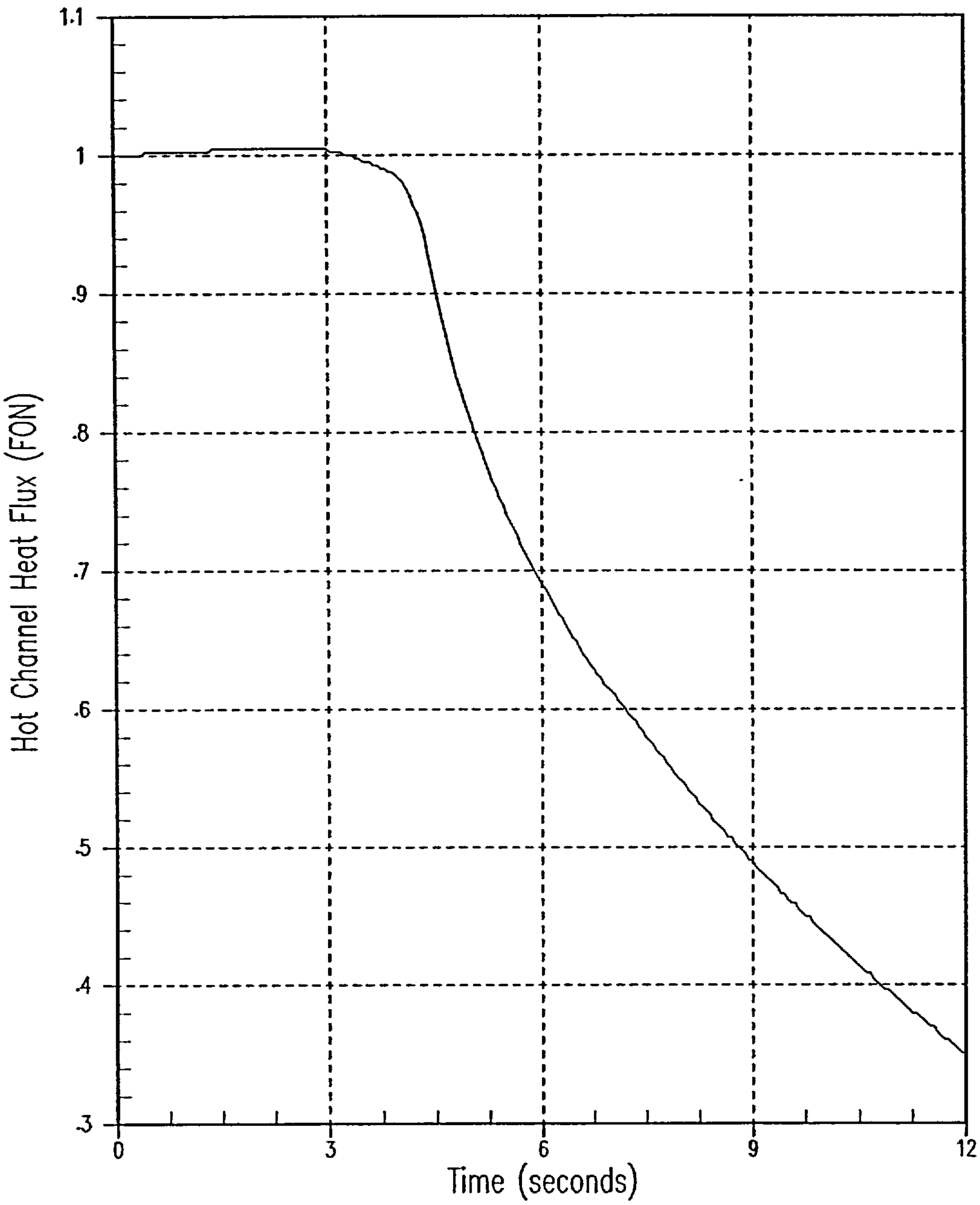

Figure 14.1.8-7

REPLACE WITH NEW FIGURE 14.1.8-8

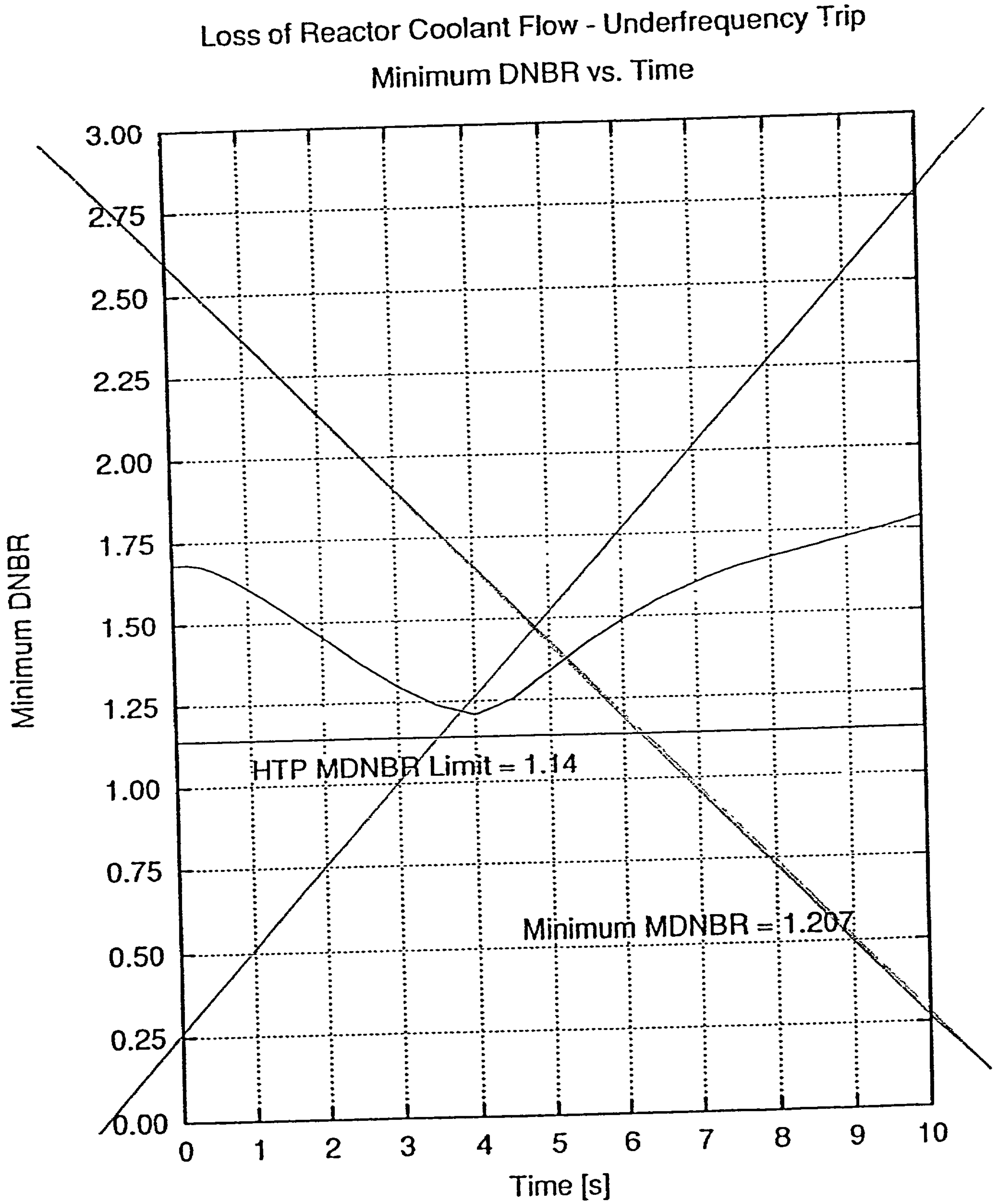

Figure 14.1.8-8

Rev. 16
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Complete Loss of Flow - Two Pumps Coasting Down (CLOF)

DNBR vs. Time

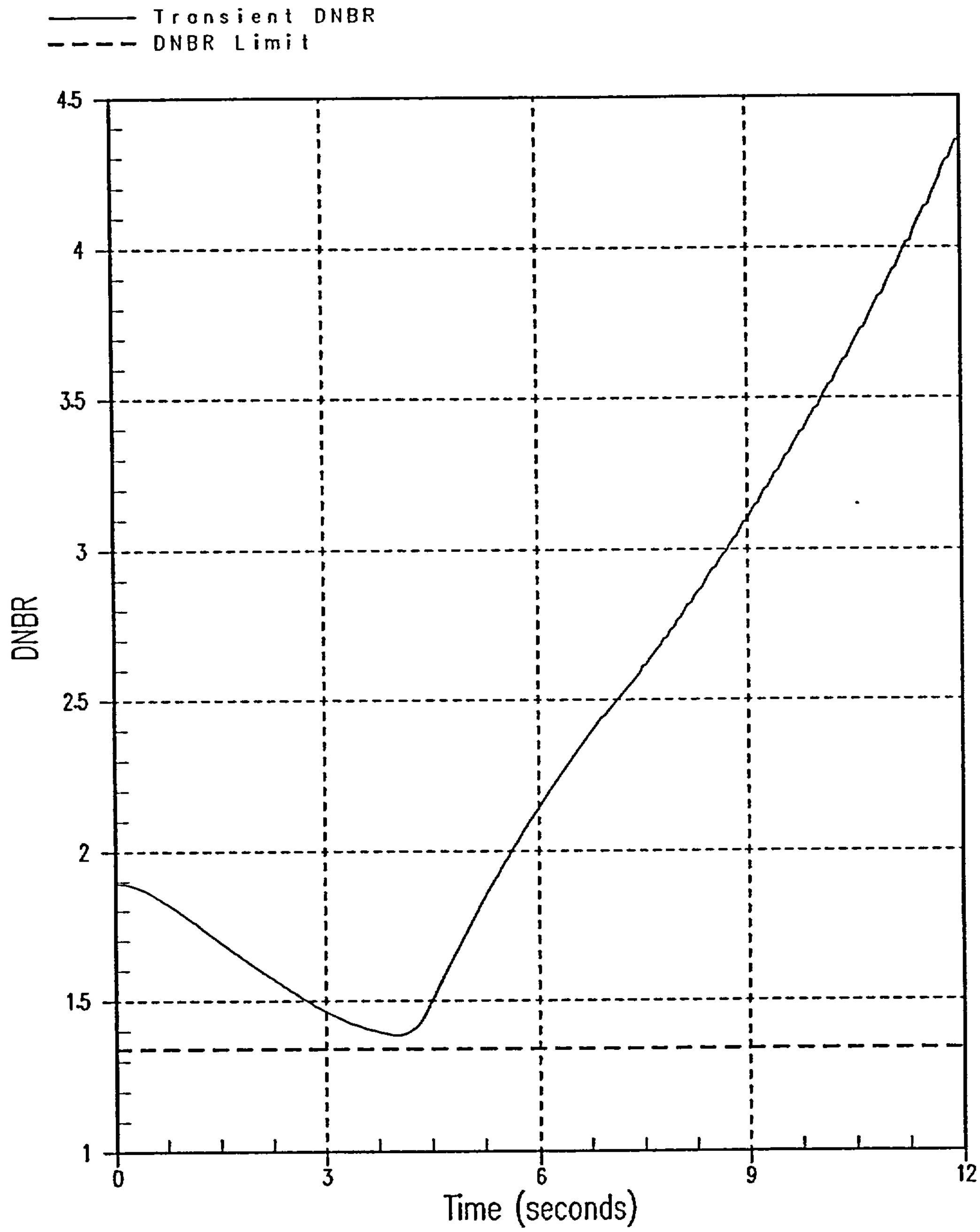

Figure 14.1.8-8

REPLACE WITH NEW FIGURE 14.1.8-9

Loss of Reactor Coolant Flow - Locked Rotor

Core Flow vs. Time

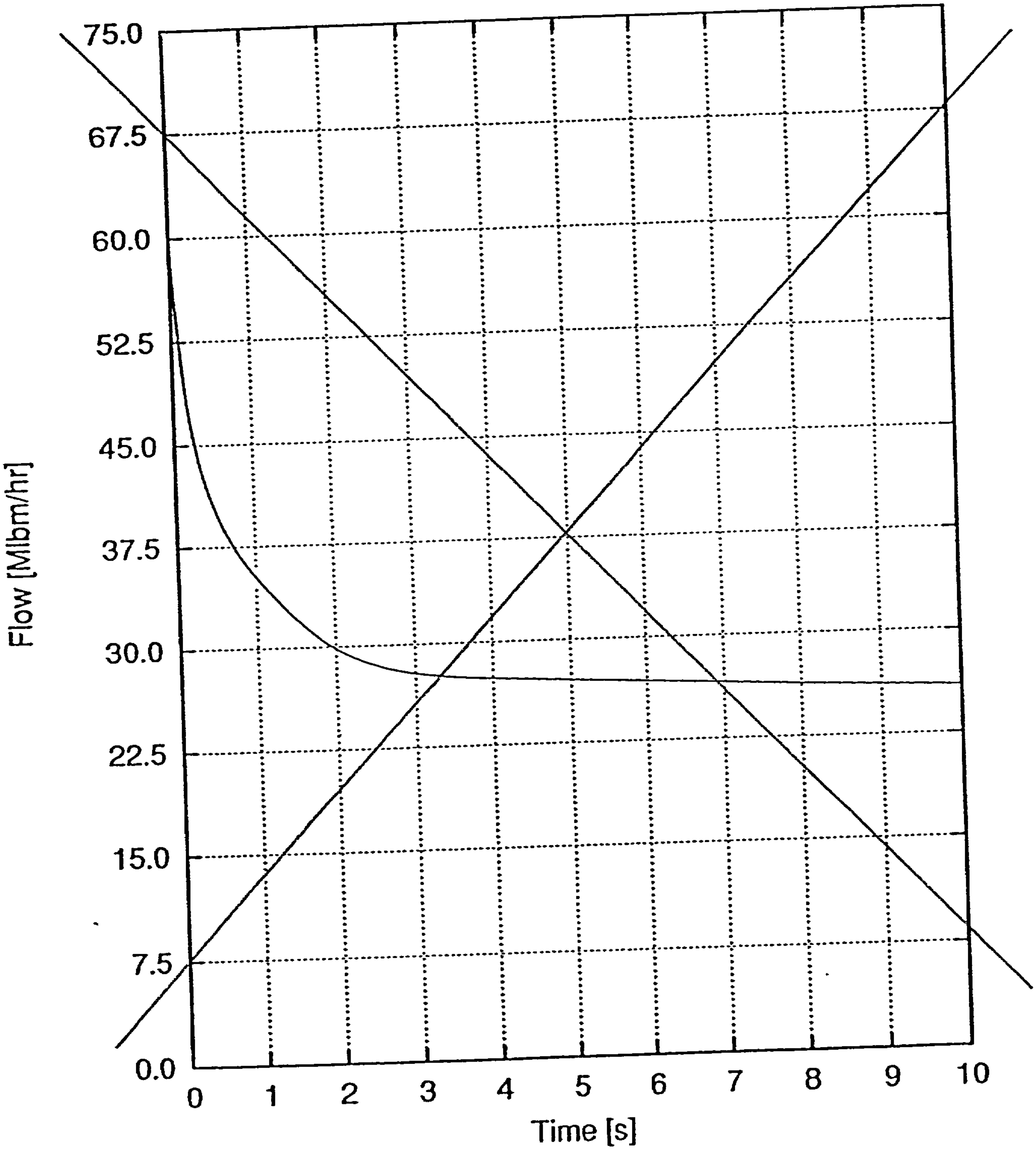

Figure 14.1.8-9

Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

Total Core Inlet Flow vs. Time

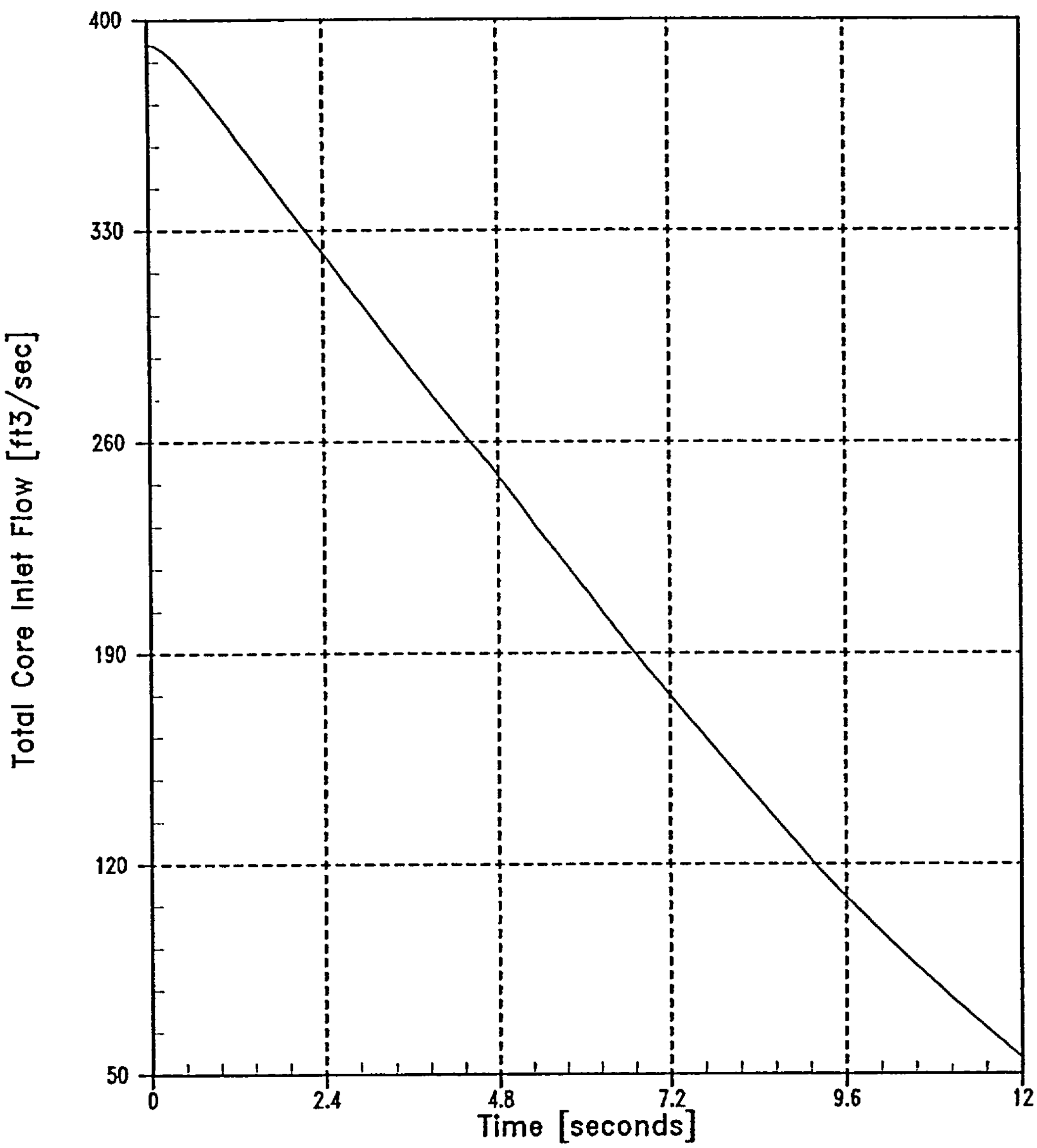

Figure 14.1.8-9

REPLACE WITH NEW FIGURE 14.1.8-10

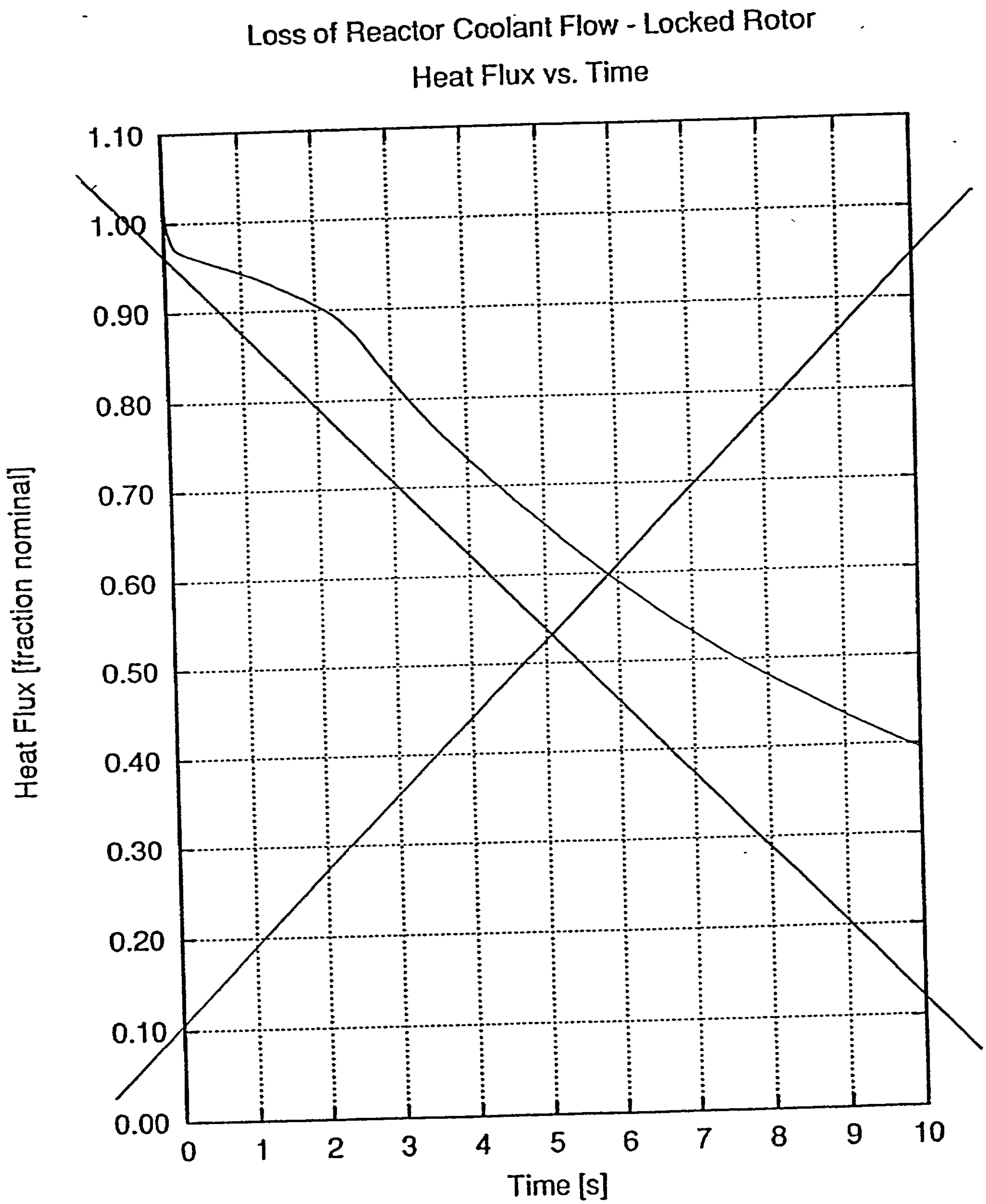

Figure 14.1.8-10

Rev. 16
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Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

RCS Loop Flow vs. Time

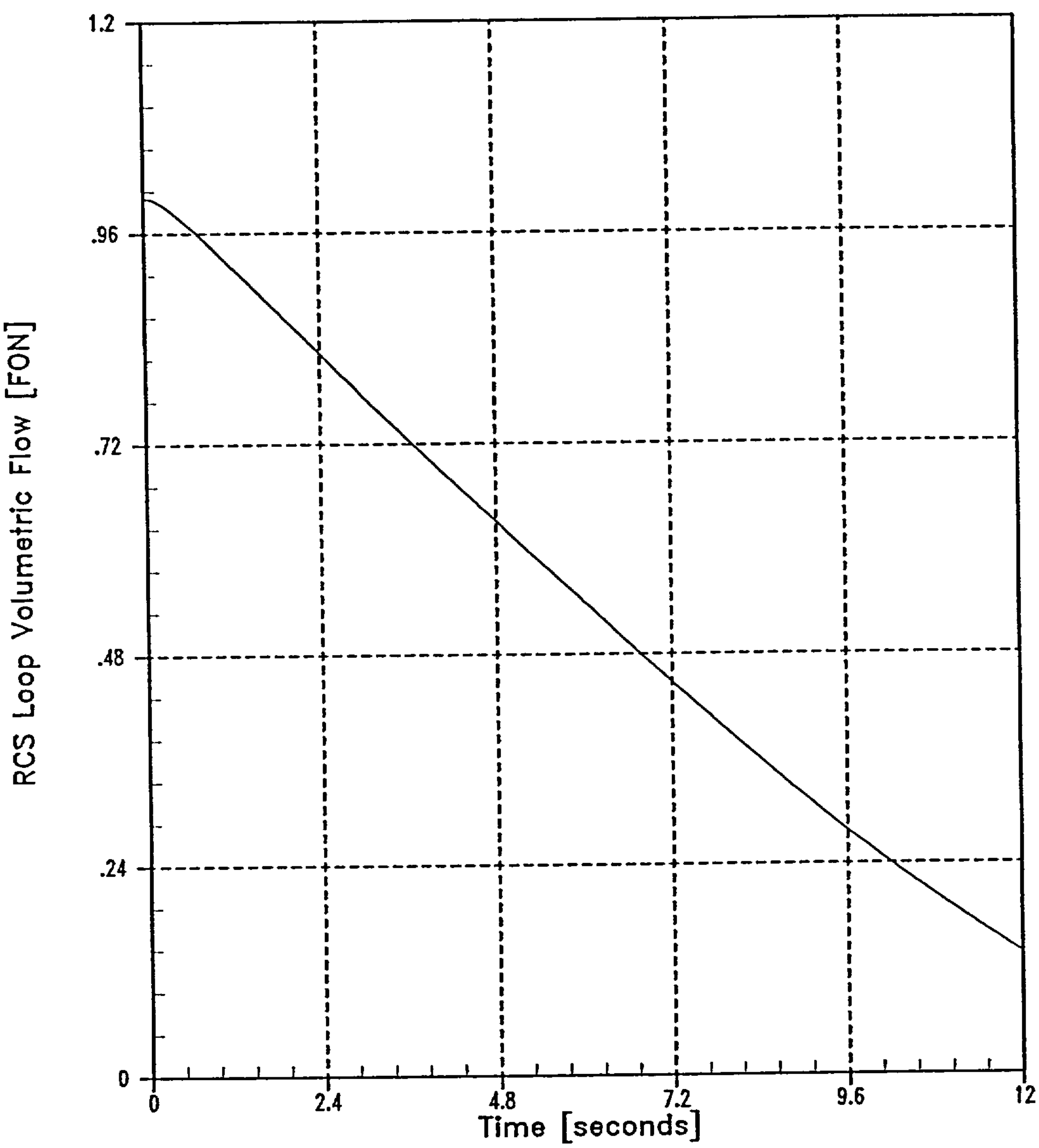

Figure 14.1.8-10

REPLACE WITH NEW FIGURE 14.1.8-11

Loss of Reactor Coolant Flow - Locked Rotor

Pressurizer Pressure vs. Time

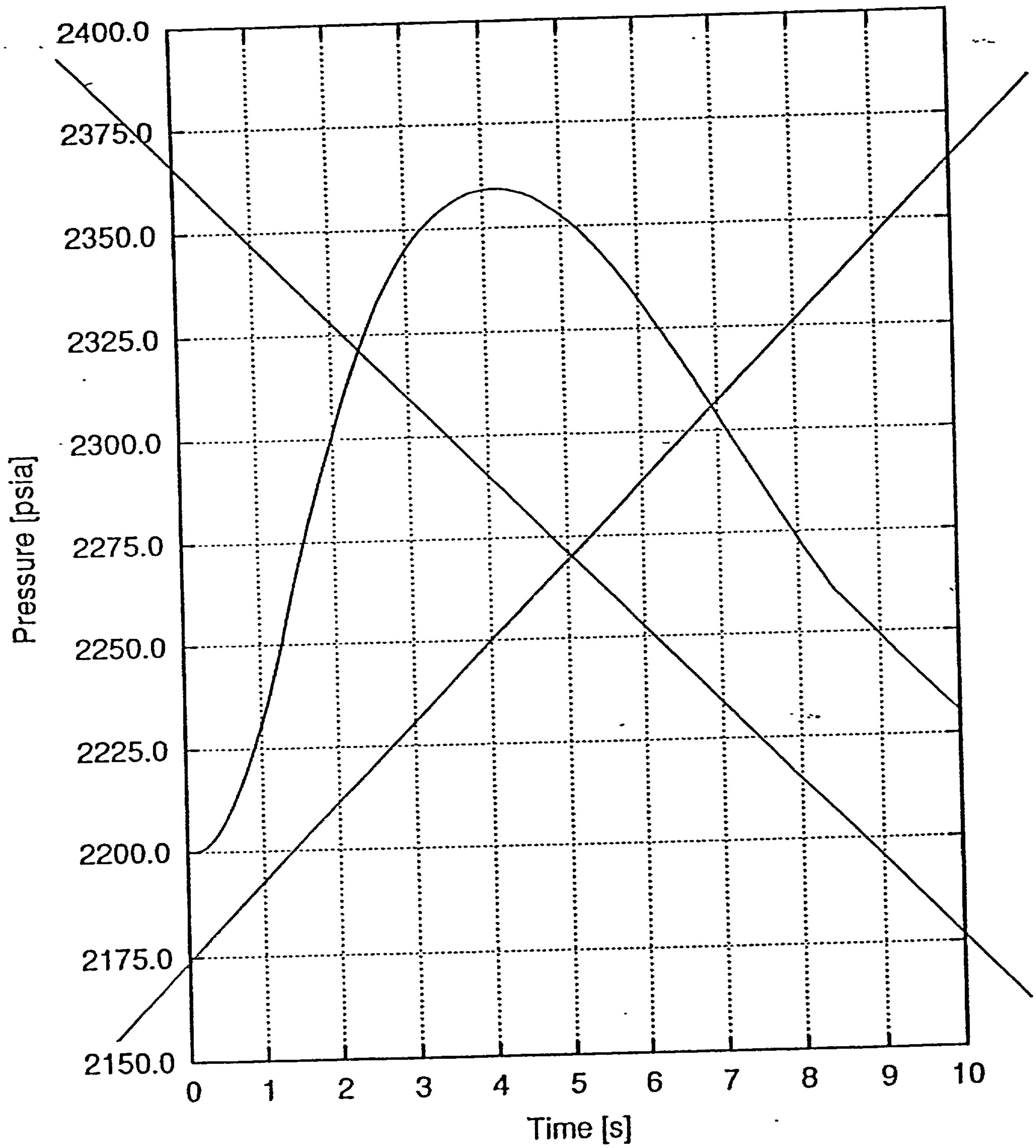

Figure 14.1.8-11

Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

Nuclear Power vs. Time

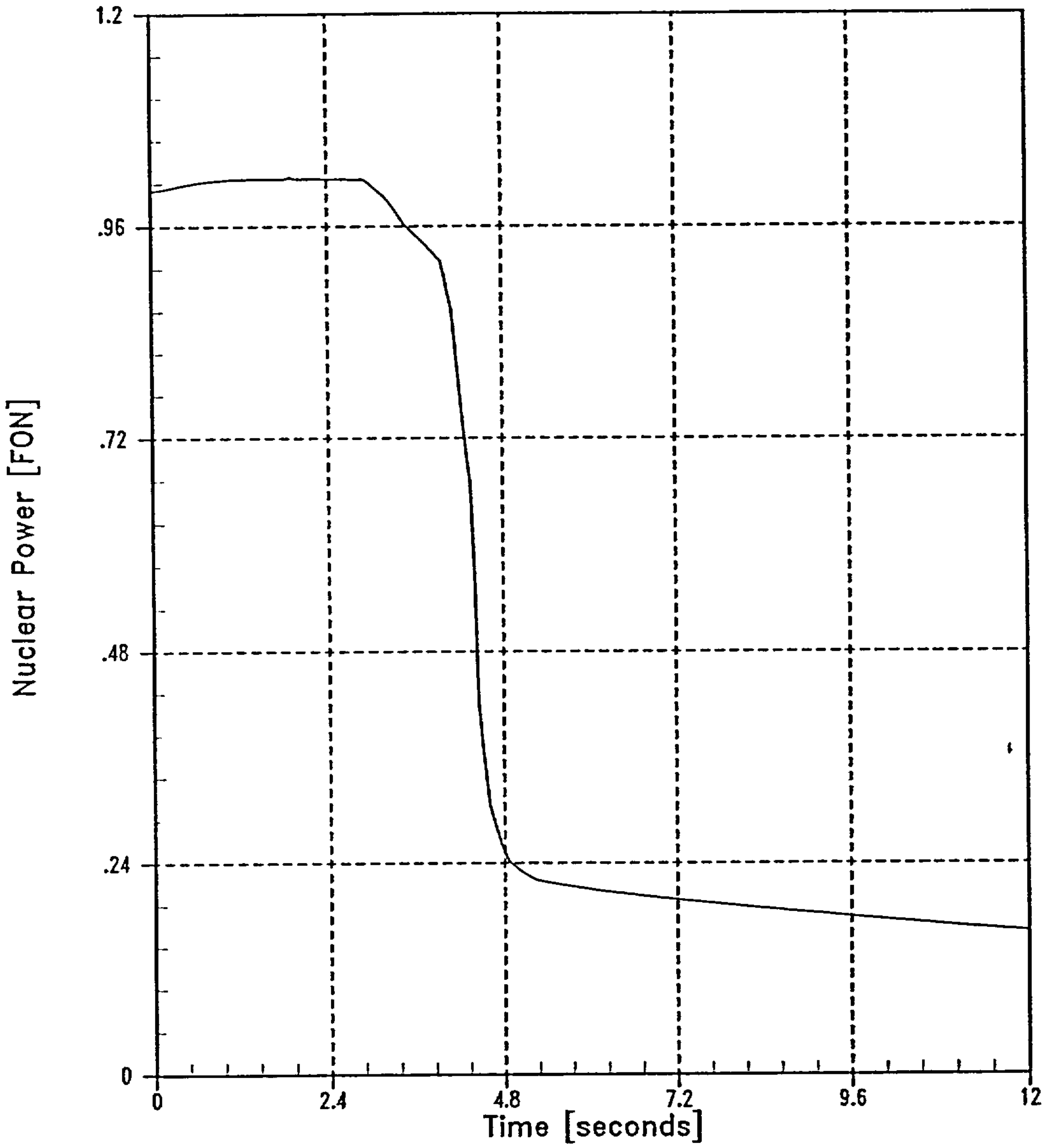

Figure 14.1.8-11

REPLACE WITH NEW FIGURE 14.1.8-12

Loss of Reactor Coolant Flow - Locked Rotor

Minimum DNBR vs. Time

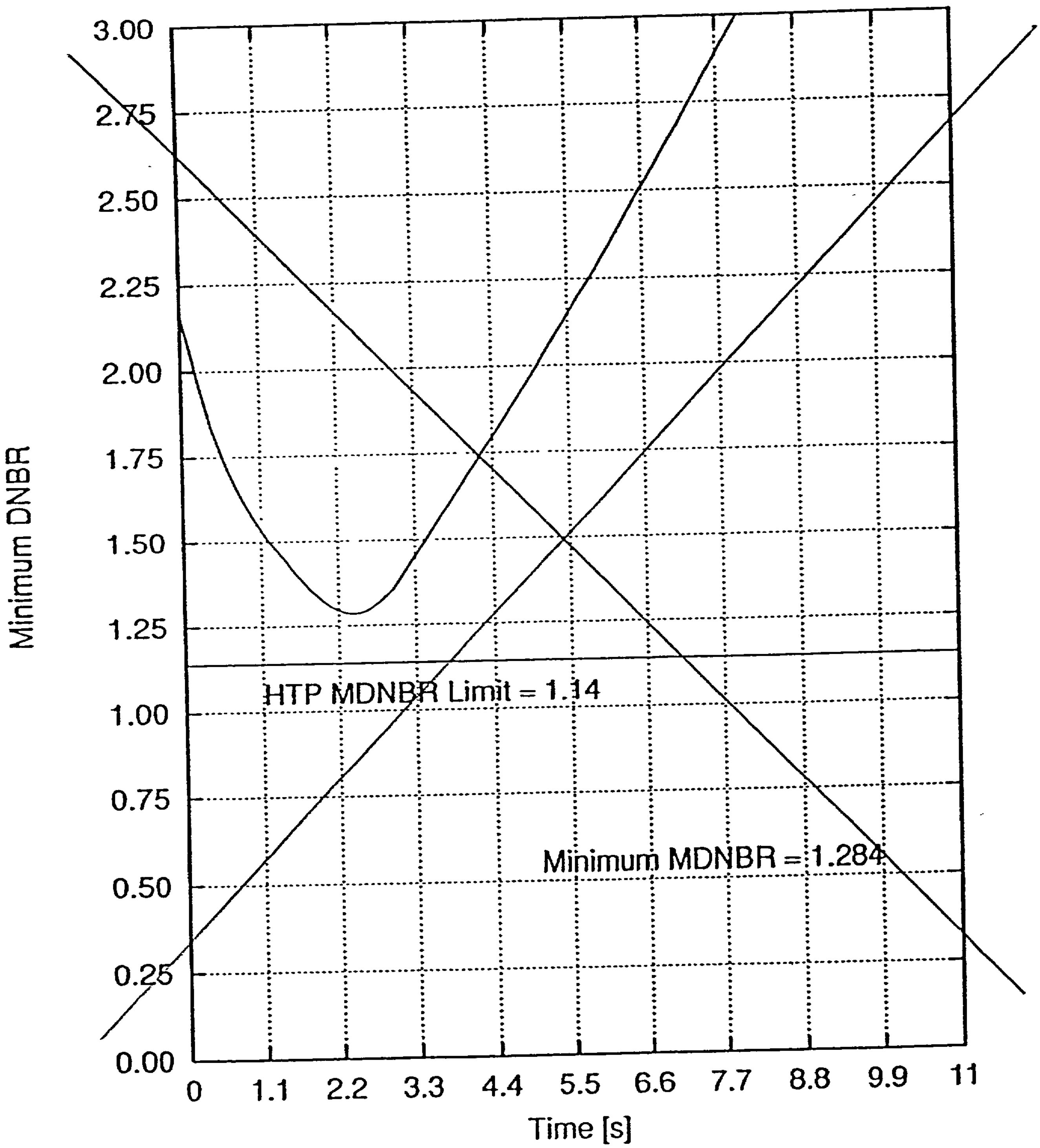

Figure 14.1.8-12

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Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

Core Average Heat Flux vs. Time

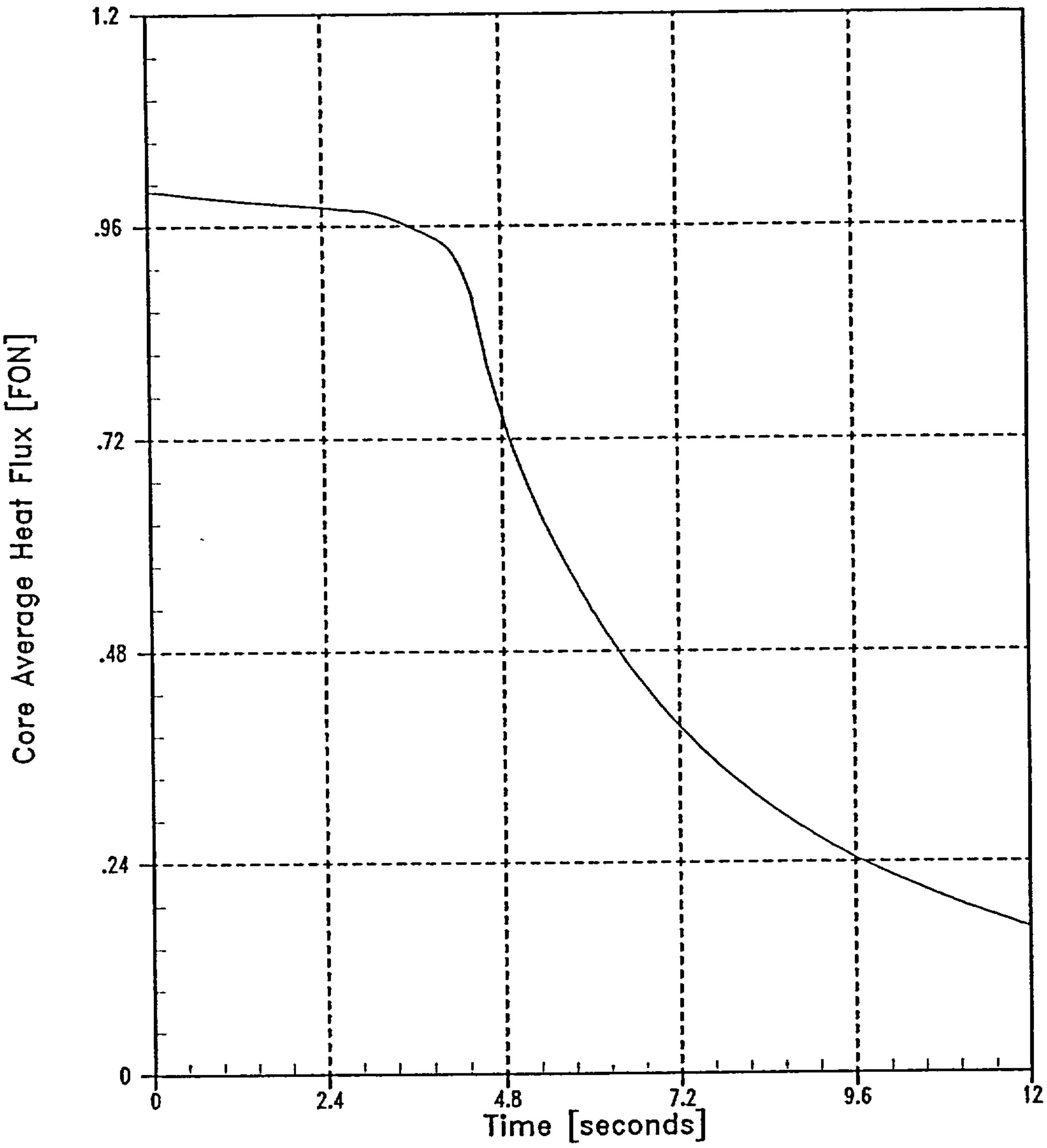

Figure 14.1.8-12

REPLACE WITH NEW FIGURE 14.1.8-13

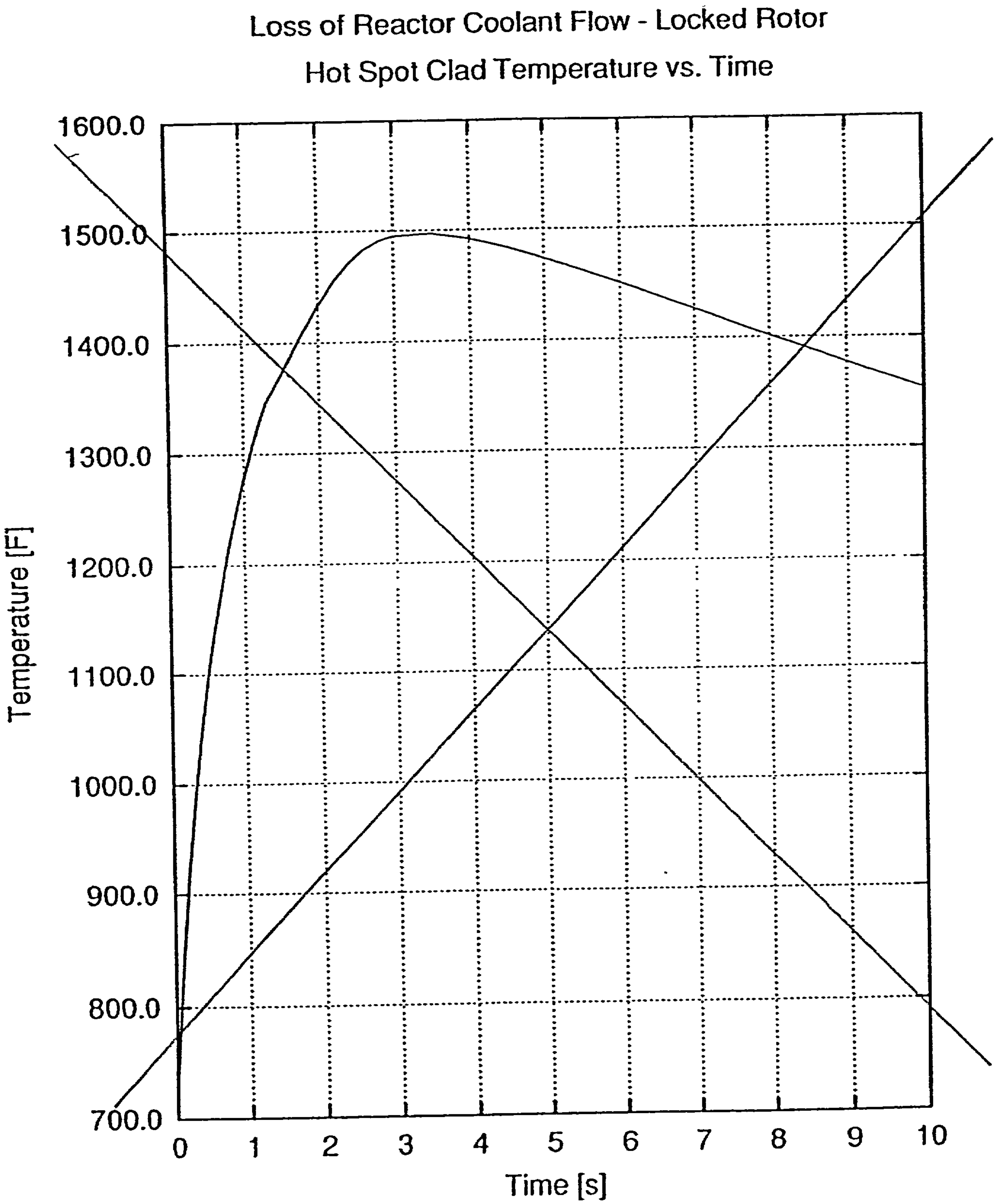

Figure 14.1.8-13

Rev. 16
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Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

Pressurizer Pressure vs. Time

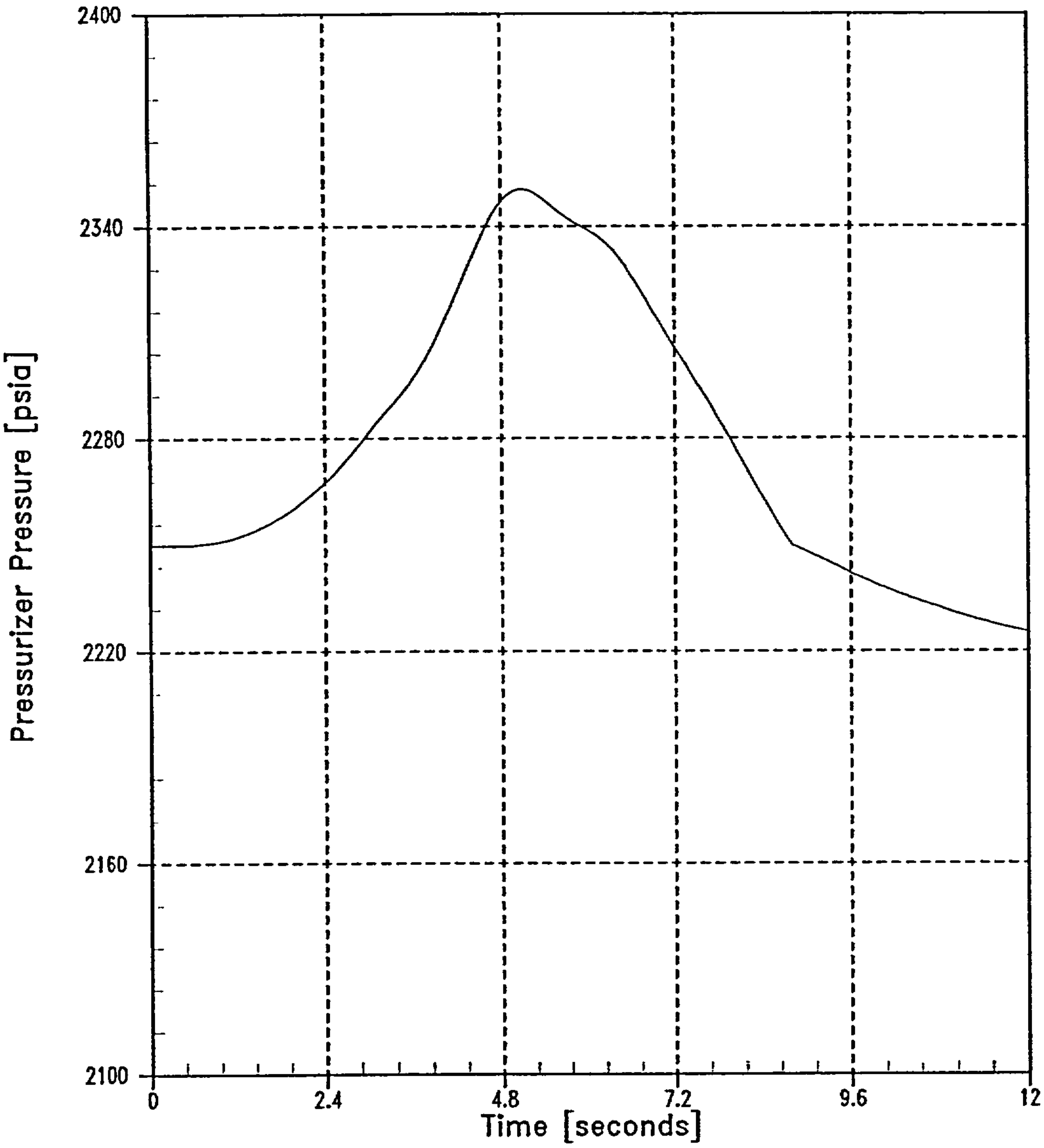

Figure 14.1.8-13

Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

RCS Loop Temperature vs. Time

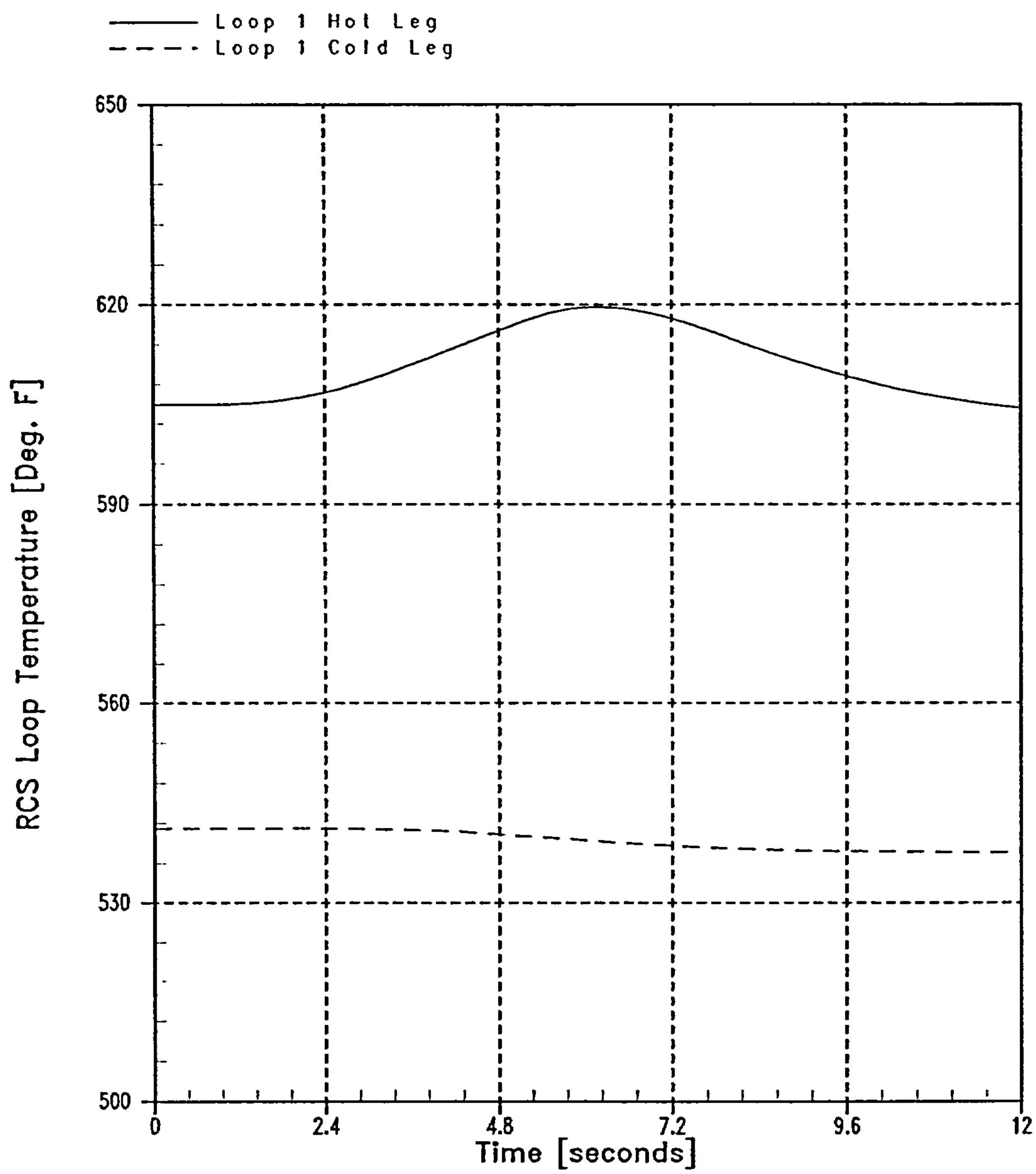

Figure 14.1.8-14

Hot Channel Heat Flux vs. Time

Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

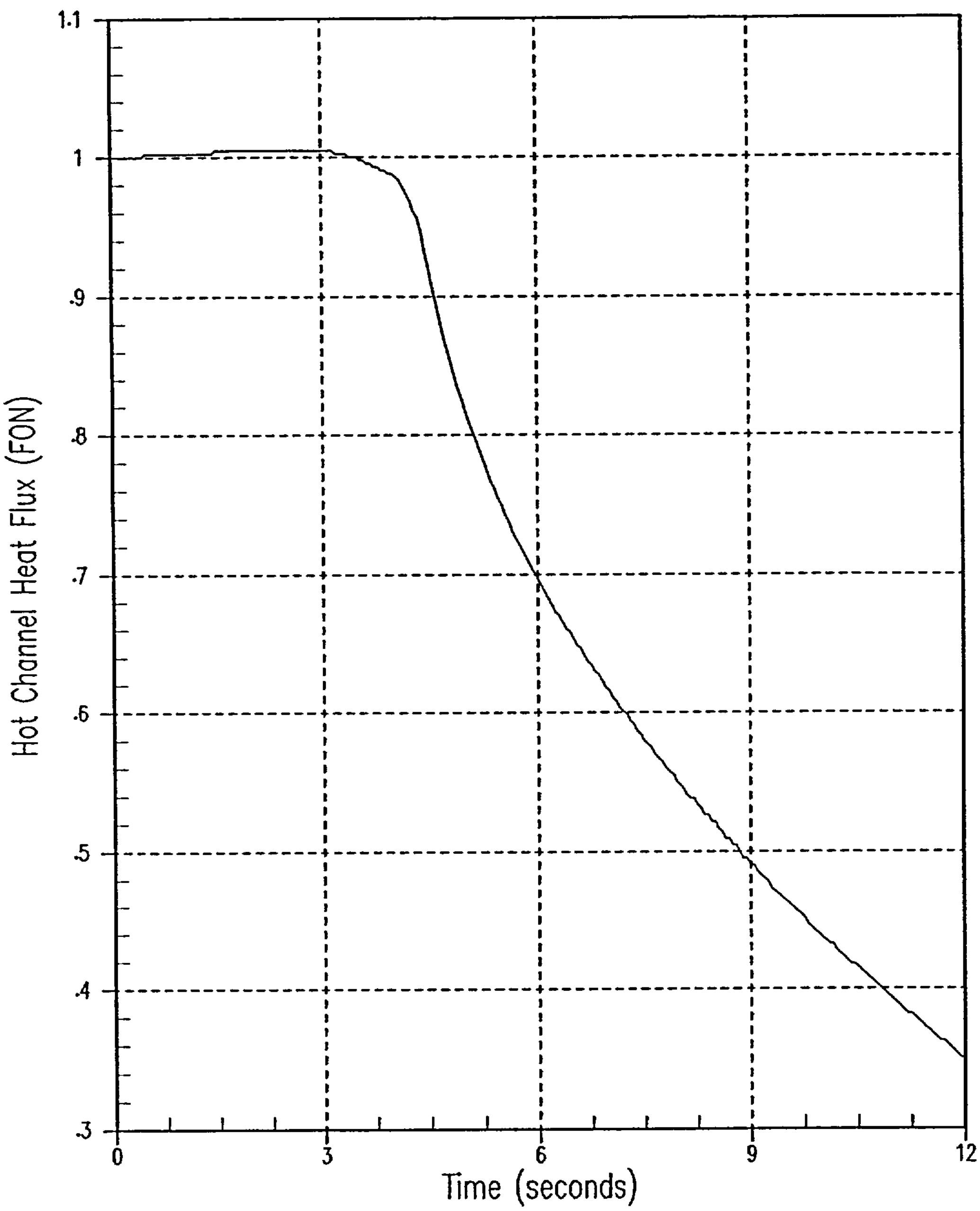

Figure 14.1.8-15

Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

DNBR vs. Time

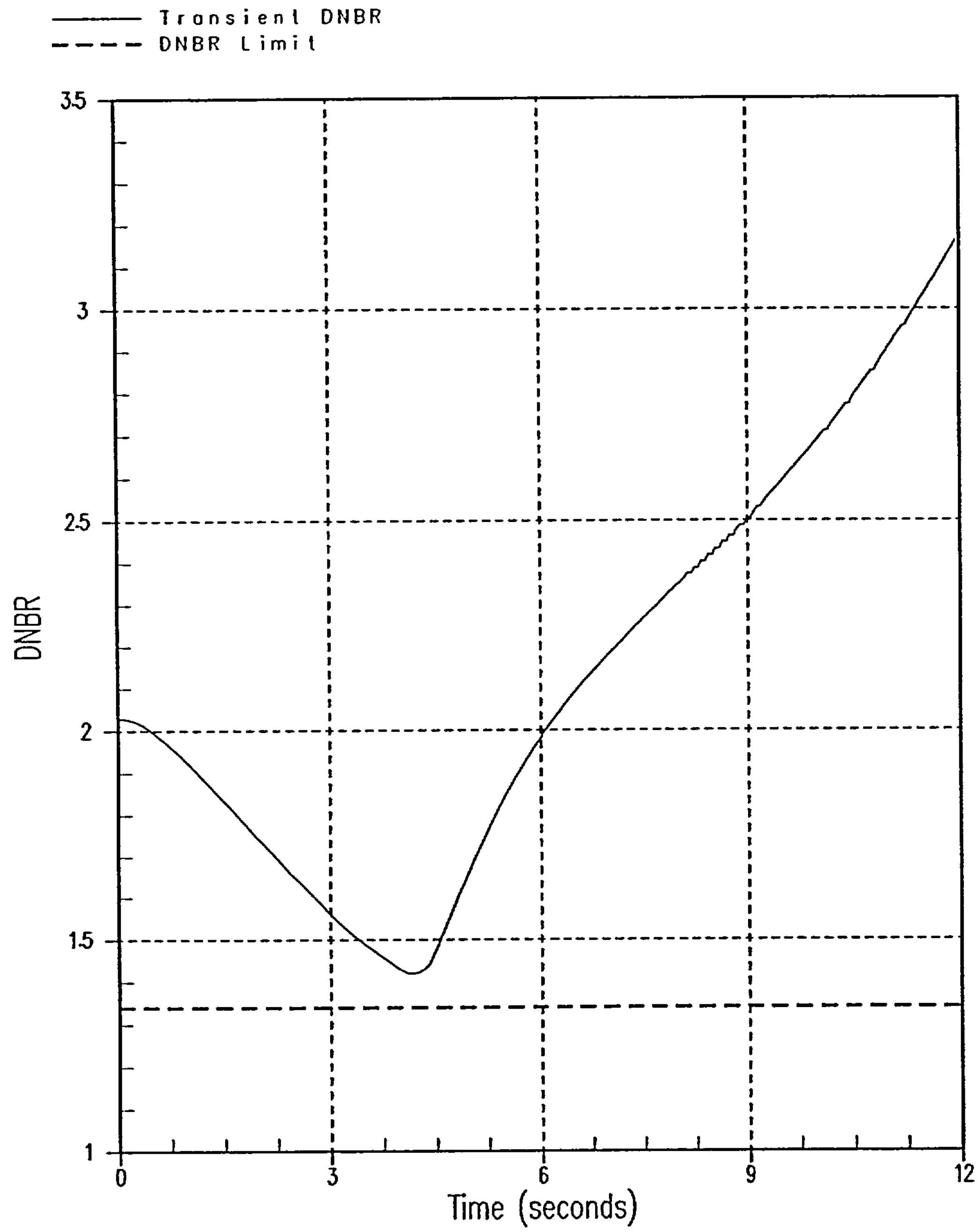

Figure 14.1.8-16

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Total Core Inlet Flow vs. Time

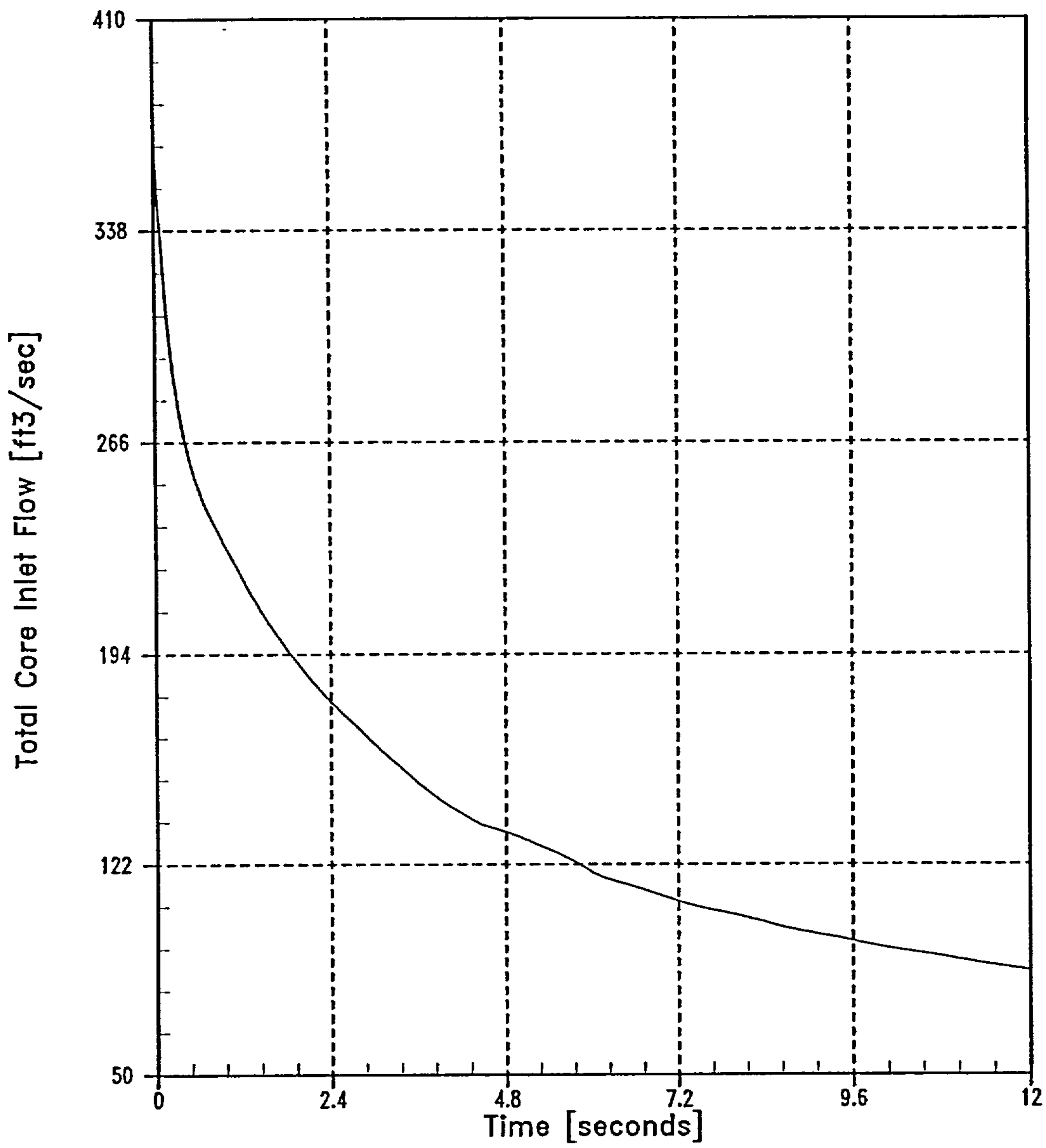

Figure 14.1.8-17

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

RCS Loop Flow vs. Time

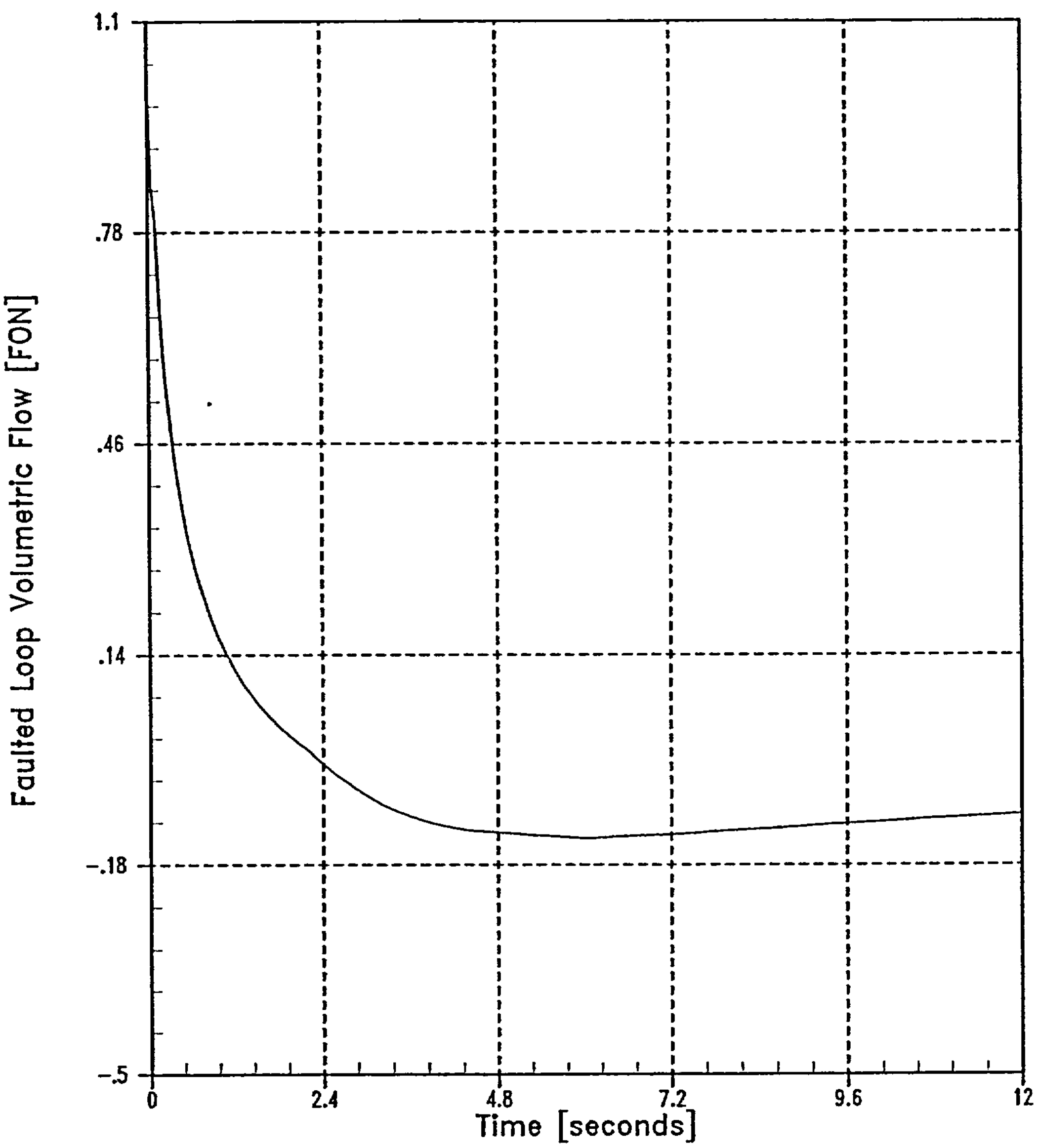

Figure 14.1.8-18

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Nuclear Power vs. Time

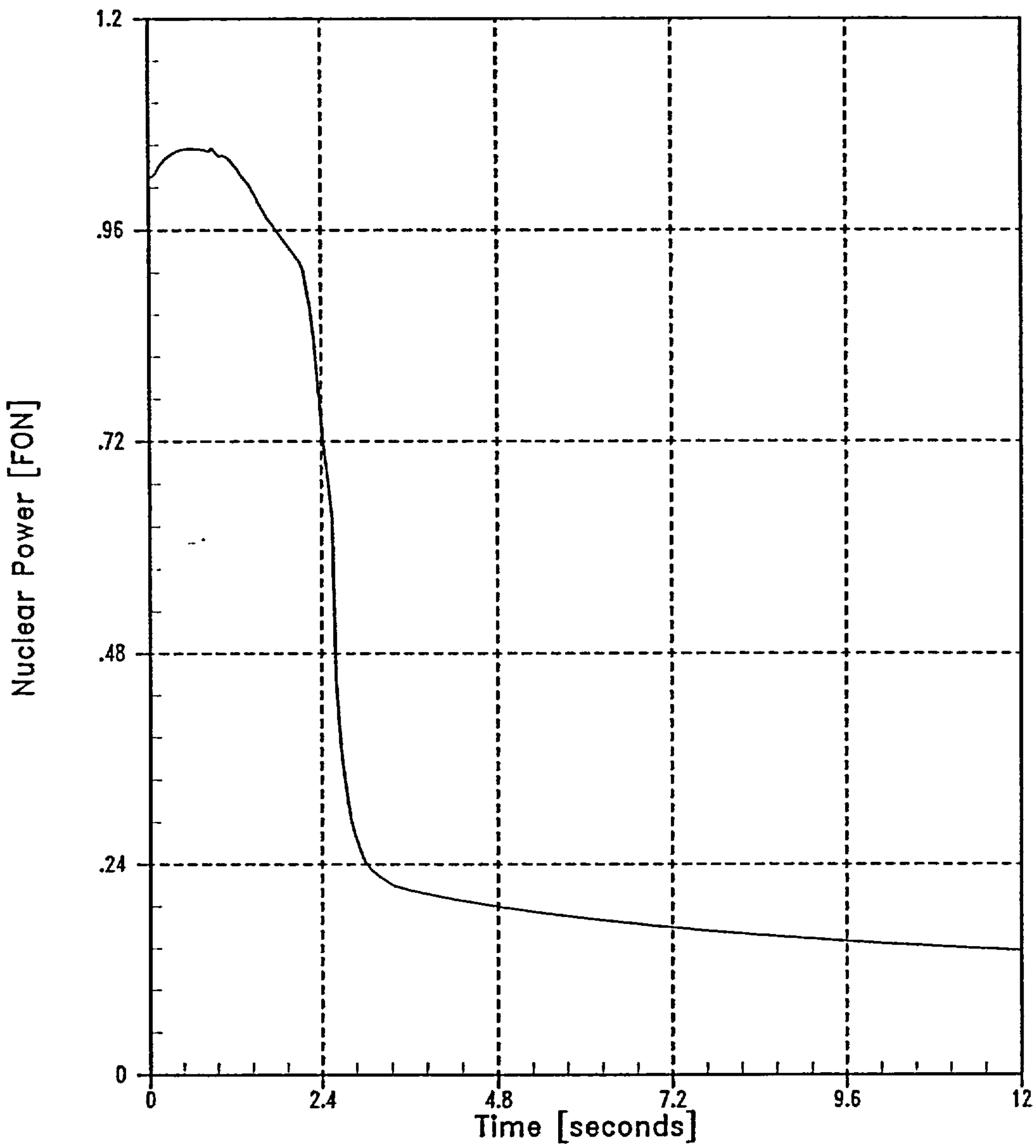

Figure 14.1.8-19

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Core Average Heat Flux vs. Time

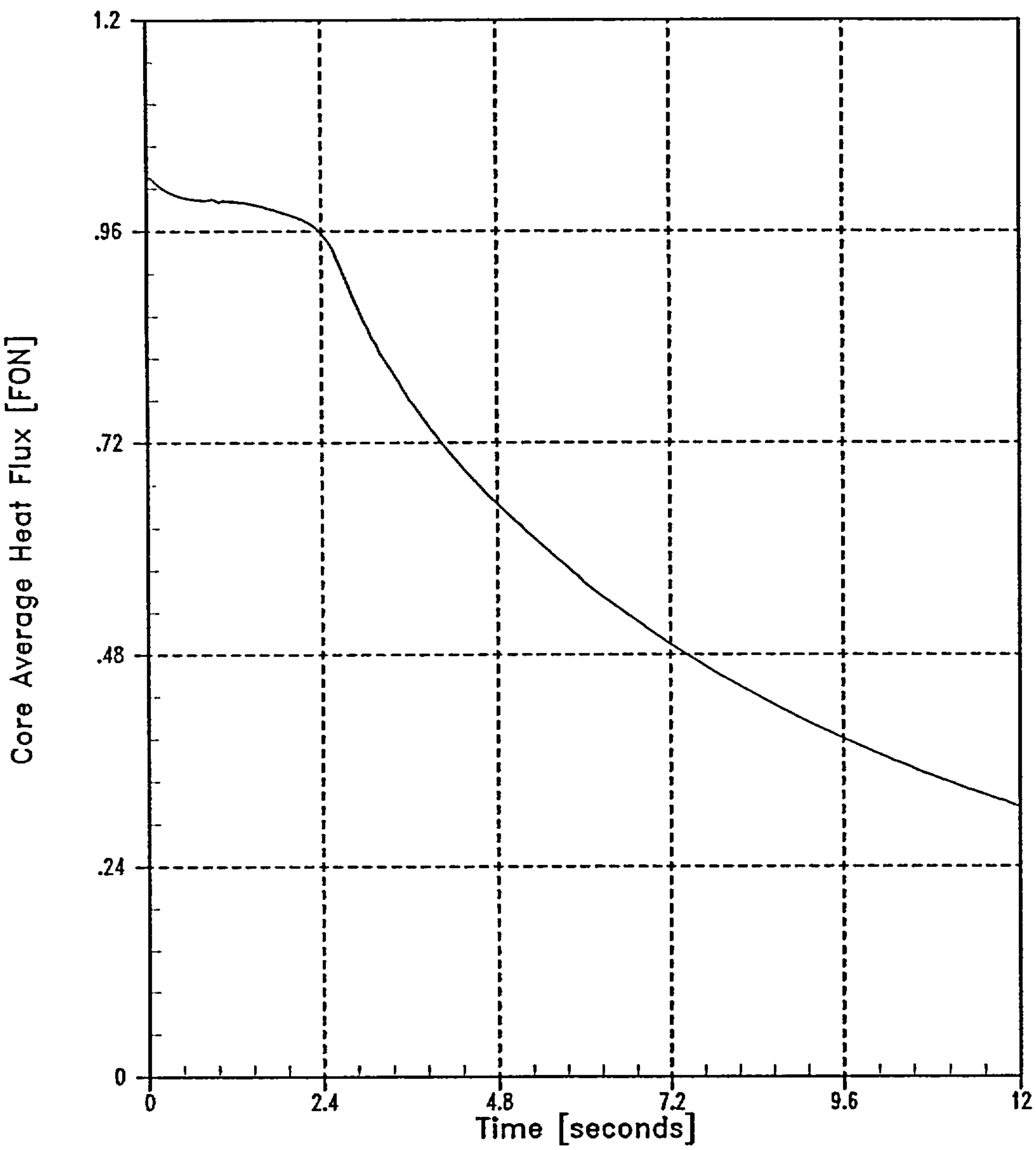

Figure 14.1.8-20

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Pressurizer Pressure vs. Time

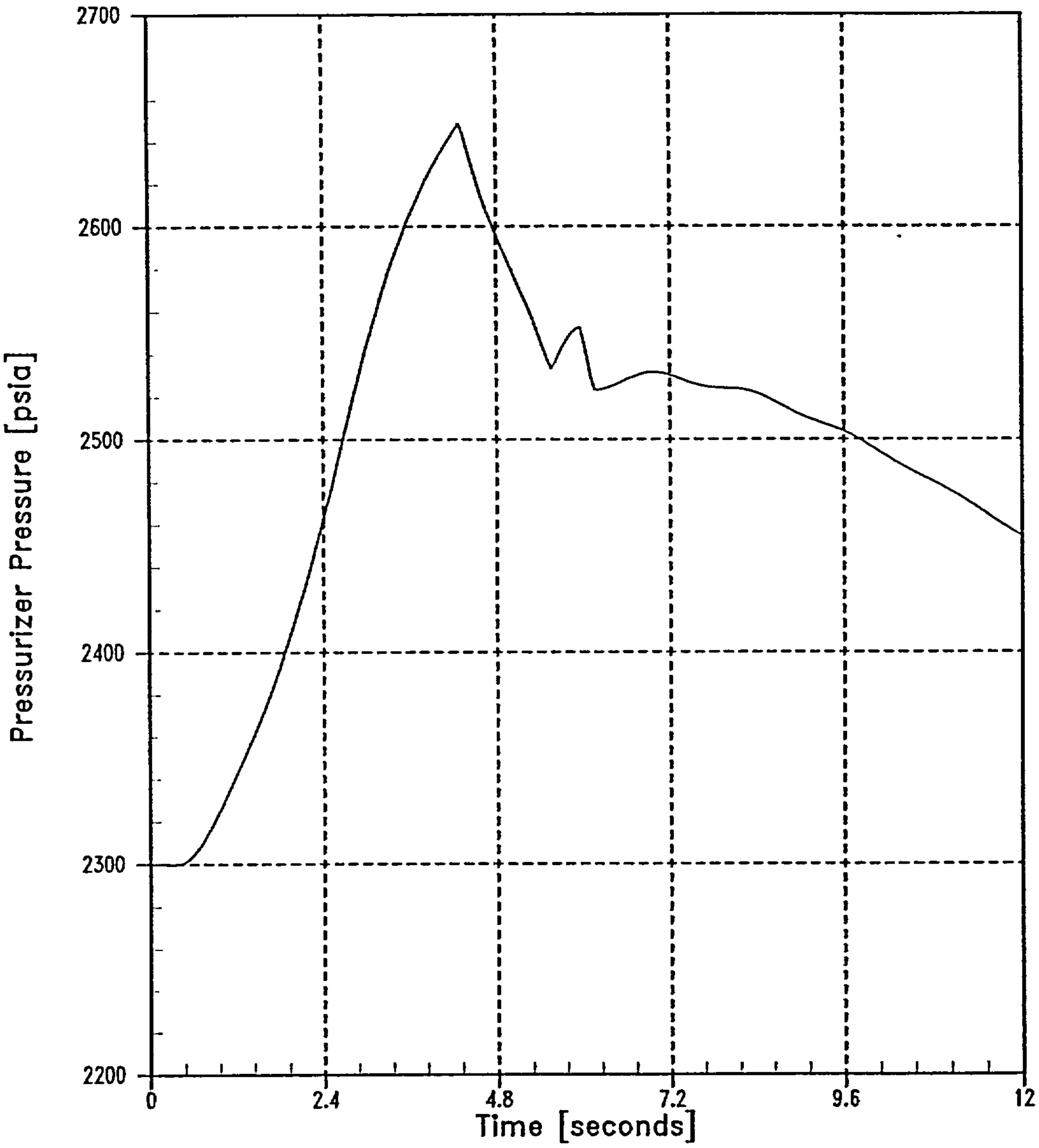

Figure 14.1.8-21

Locked Rotor / Shaft Break – RCS Pressure / PCT Case
Vessel Lower Plenum Pressure vs. Time

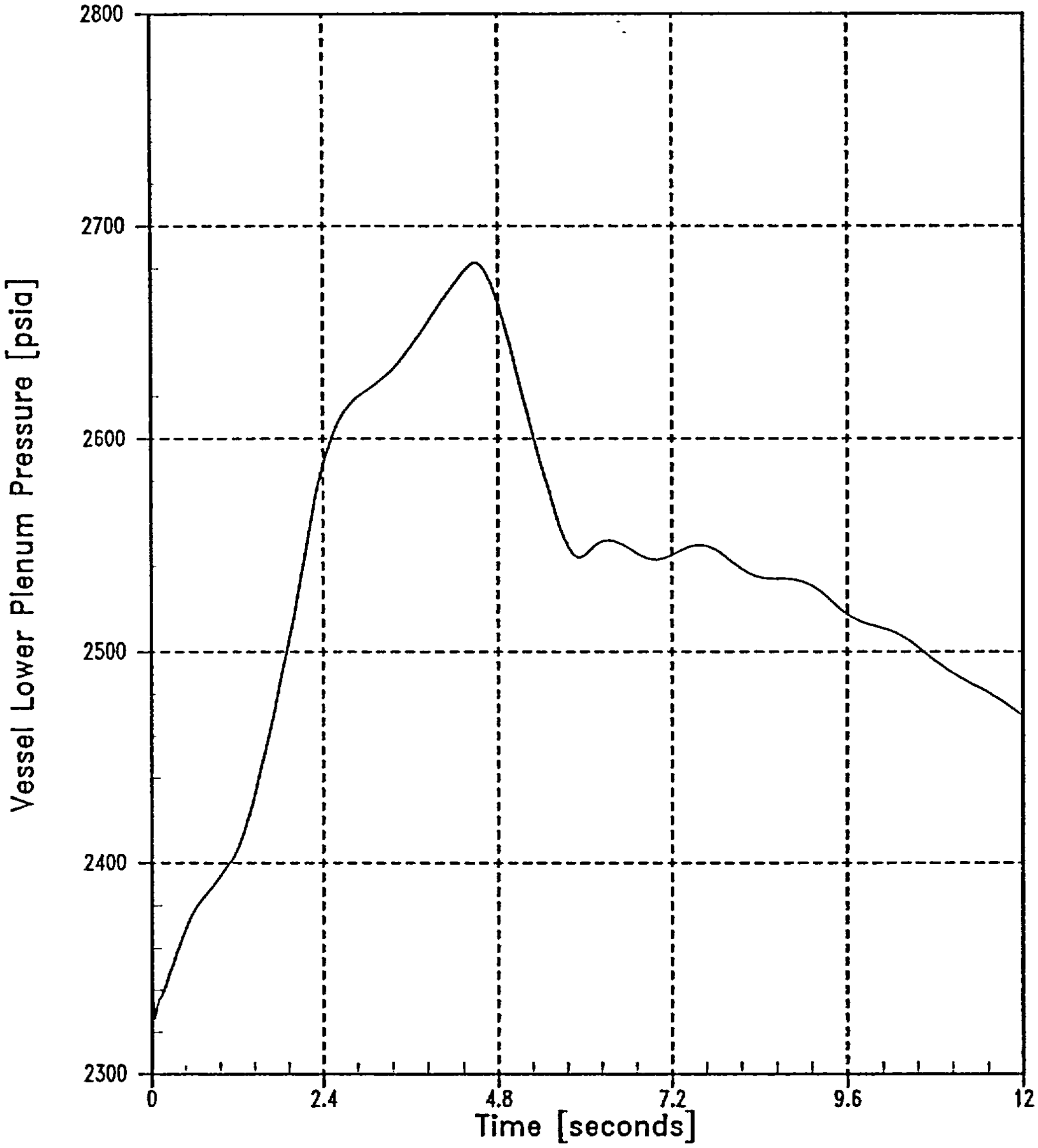

Figure 14.1.8-22

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

RCS Loop Temperature vs. Time

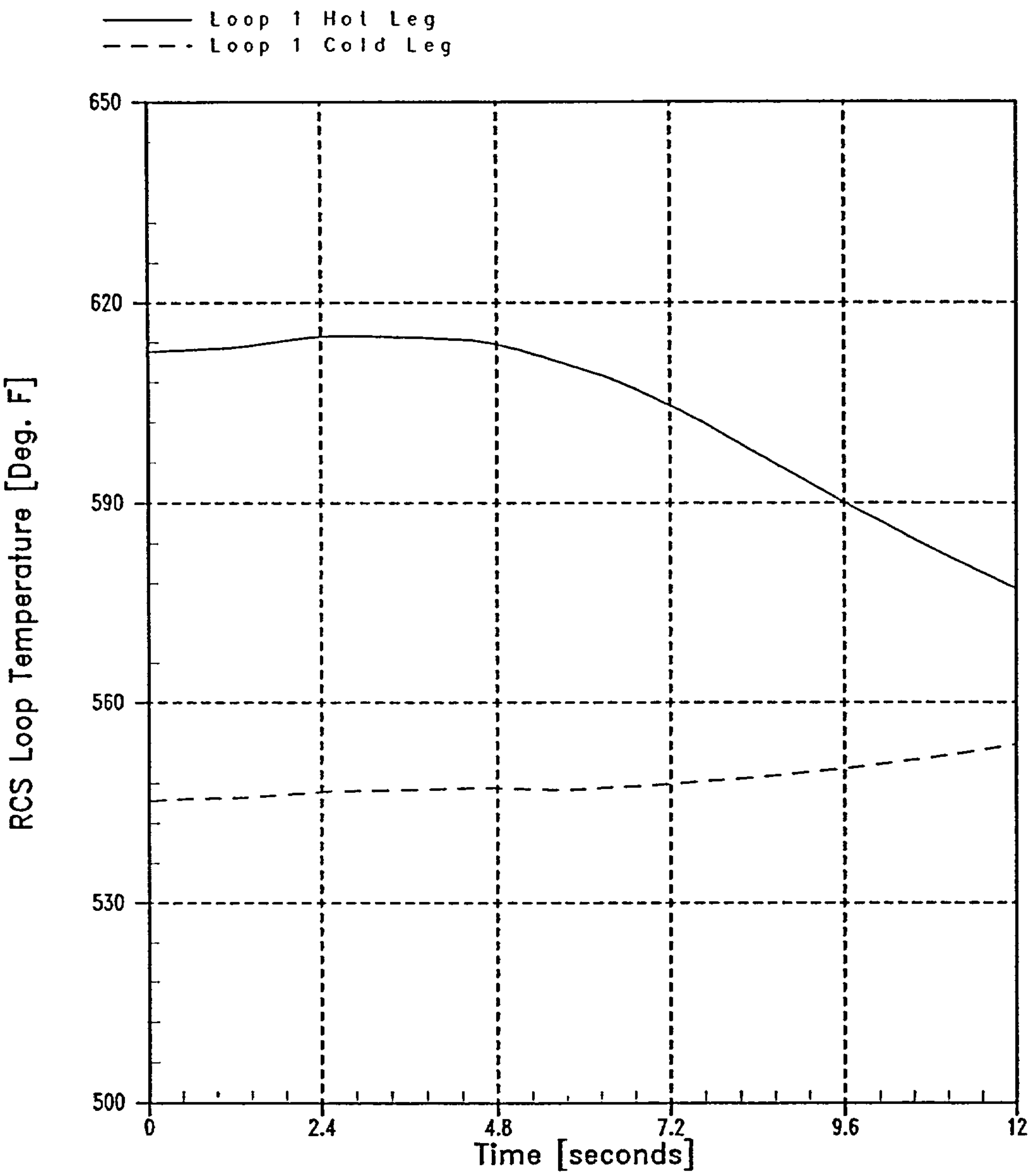

Figure 14.1.8-23

Hot Channel Heat Flux vs. Time

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

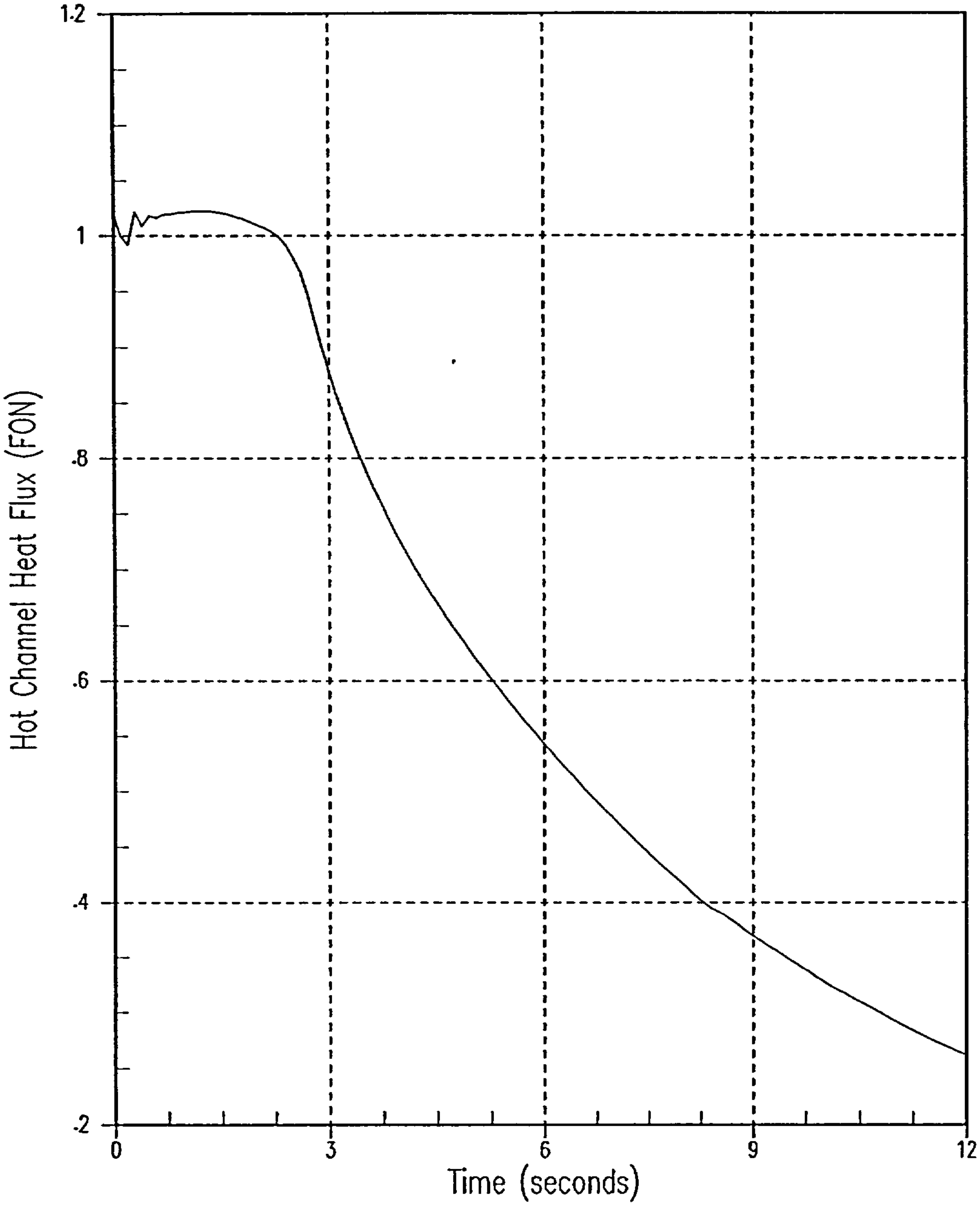

Figure 14.1.8-24

Locked Rotor / Shaft Break – RCS Pressure / PCT Case

Hot Spot Cladding Inner Temperature vs. Time

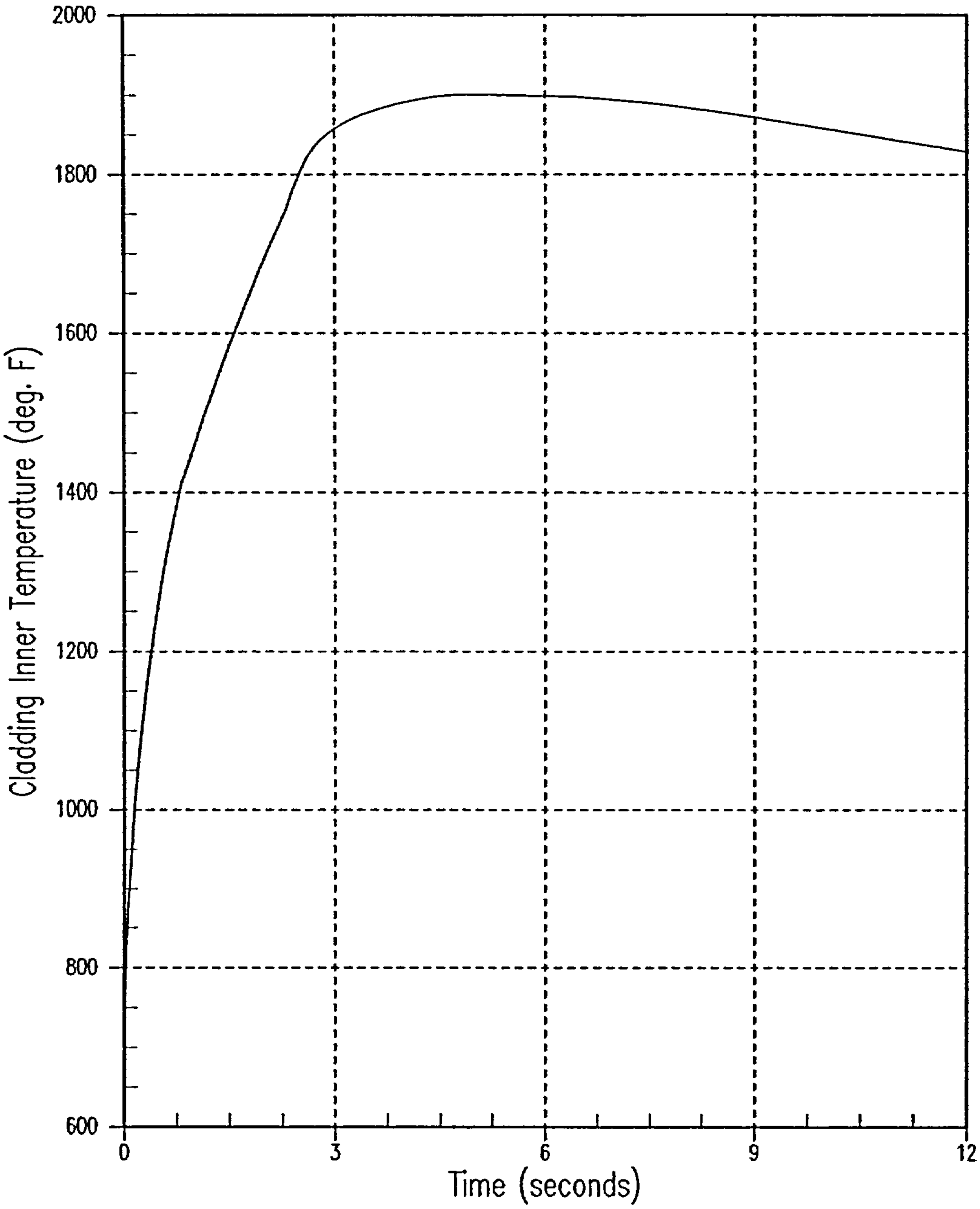

Figure 14.1.8-25

Figure 14.1.8-13 shows the clad temperature transient at the hot spot. Since in the worst case examined, the clad temperature does not exceed 1800°F, it is not necessary to consider the possibility of a zirconium-steam reaction. The zirconium-steam reaction is only significant above this temperature.

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Valve/Acceptance Criterion).

	% Fuel Rods < DNB Limit	Max Clad Temp. (°F)	RCS Pressure (psia)	MS Pressure (psia)
Locked Rotor	[redacted]	[redacted]/2700	[redacted]/2750	1044/1210

* Percentage of Fuel Rods with $F_{\Delta H} \geq$ [redacted]

Outside LOL/TT Scope

Section 14.1.8 changes suggested earlier.

Conclusions

Since the peak pressure reached during the transient is < 110% of design, the integrity of the Reactor Coolant System is not endangered. The pressure can be considered as an upper limit because of the following conservative assumptions used in the study:

1. Credit is not taken for the negative moderator coefficient.

2. It is assumed that the pressurizer relief valves were inoperative.

3. The steam dump is assumed to be inoperative.

The peak clad temperature calculated for the hot spot, can also be considered an upper limit because of the following:

1. The hot spot is assumed to be in DNB at the start of the accident.

2. A high gap coefficient is used during the transient.

3. The nuclear heat released in the fuel at the hot spot is based on a zero moderator coefficient.

14.1.9 LOSS OF EXTERNAL ELECTRICAL LOAD

Accident Description

Replace with following USAR Section 14.1.9

The loss of external electrical load may result from an abnormal increase in network frequency, opening of the main breaker from the generator, which causes a rapid large Nuclear Steam Supply System load reduction by the action of the turbine control, or by a trip of the turbine generator.

The plant is designed to accept a full-load rejection without actuating a reactor trip. The automatic steam dump system with 85% steam dump capacity (40% to the condenser and 45%

Replace with following Section 14.1.9

~~to the atmosphere) is able to accommodate this load rejection by reducing the transient imposed upon the Reactor Coolant System. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the Rod Control System. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for a step loss of load with steam dump to auxiliary load.~~

~~In the event the steam dump valves fail to open following a large load loss, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer level signal. The steam generator shell side pressure and reactor coolant temperatures would increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the Reactor Coolant System and steam generator against overpressure for all load losses without taking credit for the steam dump system.~~

~~The most likely source of a complete loss of load on the Nuclear Steam Supply System is a trip of the turbine generator. In this case, there is a direct reactor trip signal (unless below approximately 10% power) derived from either the turbine auto-stop oil pressure or a closure of the turbine stop valves. Reactor coolant temperatures and pressure do not significantly increase if the steam dump and pressurizer pressure control system are functioning properly. However, in this analysis, the behavior of the plant is evaluated for a complete loss of load from 102% of full power without a direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The Reactor Coolant System and Steam System pressure relieving capacities are designed to insure safety of the plant without requiring the automatic rod control, pressurizer pressure control and/or steam dump control systems.~~

~~Method of Analysis~~

~~The total loss of load transients are analyzed by employing a detailed digital computer program. The program describes the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves.~~

~~The objectives of this analysis are to demonstrate margins to core protection limits and to demonstrate the adequacy of the plant pressure relieving devices.~~

a. ~~The initial reactor power and Reactor Coolant System temperatures are assumed at their maximum values consistent with steady-state full power operation, including allowances for calibration and instrument errors. The initial Reactor Coolant System pressure is assumed at the minimum value consistent with steady-state full power operation, including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the total loss-of-load accident.~~

b. ~~The total loss of load is analyzed for both BOC and EOC conditions. At BOC, a zero moderator coefficient (0.0 Δk/°F) is used; and at EOC, a moderator coefficient value of~~

Replace with following Section 14.1.9

–4.0E–4 Δk/°F is used. A conservatively large absolute value of the Doppler coefficient is used for all cases with a negative moderator coefficient. For the cases in which the moderator coefficient is zero, a conservatively small absolute value of the Doppler coefficient is used.

c. Two cases for both the beginning and end-of-life are analyzed as follows:

- The reactor is assumed to be in normal automatic control with the control rods in the minimum incremental worth region.
- The reactor is assumed to be in manual control with no control rod insertion until a reactor trip occurs.

d. No credit is taken for any of the steam dump valves or power-operated steam generator relief valves. The steam generator pressure rises to the safety valve set point where steam release through safety valves limits secondary steam pressure at the set point.

e. Two cases for both the beginning and end-of-life are analyzed as follows:

- Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting coolant pressure.
- No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting coolant pressure.

  A nominal pressurizer safety valve setpoint of 2500 psia is assumed. Sensitivity analyses were performed at pressurizer safety valve settings of +6% and –4% of the nominal setpoint to account for the effects of steam accumulation and setpoint drift. The critical safety parameters were shown to be acceptable under these assumptions.

Results

Figures 14.1.9-1 through 14.1.9-5 show the transient responses for a total loss of load at beginning of cycle with zero moderator temperature coefficient assuming full credit for the pressurizer spray, pressurizer power-operated relief valves, and automatic control rod insertion. No credit is taken for the steam dump system.

Figures 14.1.9-6 through 14.1.9-10 show the responses for the total loss of load at end of cycle with the most negative moderator temperature coefficient (–4.0E–4 Δk/°F). The rest of the plant operating conditions are the same as the case above.

The loss-of-load accident is also analyzed assuming manual RCCA control. In addition, no credit is taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump system. Figures 14.1.9-11 through 14.1.9-15 show the manual control beginning of cycle transient with zero moderator coefficient. Figures 14.1.9-16 through 14.1.9-20 show the manual control transient results at end of cycle.

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ollowing
ection 14.1.9

The following table shows the comparison of the important calculated safety parameters to their respective acceptance criteria (Calculated Value/Acceptance Criterion):

Loss of Load	MDNBR	RCS Pressure (psia)	MSS Pressure (psia)
BOC Manual Control	1.681/1.14	2501/2750	1182/1210
BOC Auto Control	1.681/1.14	[redacted]/2750	[redacted]/1210
EOC Manual Control	1.681/1.14	[redacted]/2750	[redacted]/1210
EOC Auto Control	1.681/1.14	2377/2750	1198/1210

Conclusions

The safety analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System or the Steam System. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within safety analysis limits. The integrity of the core is maintained by the Reactor Protection System. The MDNBR does not fall below its initial value, which is above the MDNBR limit.

14.1.10 LOSS OF NORMAL FEEDWATER

Section 14.1.10 changes
Suggested later.

Accident Description

A loss of normal feedwater (from a pipe break, pump failure, valve malfunctions, or loss of off-site power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this accident, Reactor Coolant System damage could possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater is not supplied to the plant, residual heat following reactor trip heats the coolant to the point where water relief from the pressurizer occurs. Significant loss of water from the Reactor Coolant System could conceivably lead to core damage.

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scope

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

1. Reactor trip on Low-Low water level in either steam generator.

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

3. Two motor driven auxiliary feedwater pumps which are started automatically on:

   a) Low-Low level in either steam generator, or

   b) Opening of both feedwater pump circuit breakers, or

   c) Safety Injection signal, or