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November 27, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4

Re: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

By letter dated September 10, 1985, Duke proposed an amendment to the Technical Specification for the reactor trip setpoint on lo-lo steam generator water level for Catawba Unit 1. This letter provides supplemental information with respect to the earlier submittal.

As a result of discussions with the NRC staff the following additional information is being provided.

1. The previous Justification and Analysis of No Significant Hazards Consideration discussed changes to the analysis of Loss of Normal Feedwater and Loss of AC Power analyses. These analyses have been further revised to update the decay heat assumption rather than revise the auxiliary feedwater assumption.

2. FSAR pages which will be revised as a result of the proposal Technical Specification changes are attached.

3. The September 10, 1985 amendment request also requested that the proposed changes be incorporated into the proposed Catawba Units 1 and 2 combined Technical Specifications which were previously submitted on March 15, 1985. Because Catawba Unit 2 has Model D-5 steam generators vs. Model D-3 for Unit 1, the reduced low-low level trip setpoint is not needed for Catawba Unit 2. Marked up pages for the Catawba Units 1 and 2 Technical Specifications are attached.

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This letter contains information which supplements that which was provided by my letter of September 10, 1985. As such no additional license fees are necessary.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

ROS:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
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NRC Resident Inspector
Catawba Nuclear Station

Dr. K. Jabbour
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JUSTIFICATION AND ANALYSIS OF NO SIGNIFICANT HAZARDS CONSIDERATION

The steam generator low-low water level trip function protects the reactor from a loss of heat sink in the event of a sustained steam/feedwater flow mismatch. Analyses were performed to justify lowering the programmed steam generator low-low level setpoint for Catawba Unit 1. This setpoint change, along with the addition of a filter to the channel circuitry, will help prevent unnecessary reactor trips as a result of load rejections. This change will also prevent unnecessary actuation of the turbine-driven auxiliary feedwater pump due to spurious steam generator low-low level indications which result from "ringing" in the level transmitters. This benefit has been verified by observation of McGuire reactor trips since implementation of this setpoint modification.

To verify the acceptability of the proposed changes, Westinghouse has reanalyzed the Loss of Normal Feedwater, Loss of AC Power, and Feedwater System Pipe Break transients, which rely on the steam generator low-low level reactor trip for protection. The responses of various system parameters to the above analyzed accidents are given in the attached FSAR pages. Results of these analyses indicate that all applicable safety criteria are met using the revised setpoint and the increased instrument delay time.

For the loss of Normal Feedwater and Loss of AC Power analyses, it was necessary to revise the original FSAR assumption for residual decay heat in order to compensate for the lower low-low level setpoint. For these two analyses, core decay heat generation is based on ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979. The FSAR will be revised to reflect the new assumptions.

The Feedwater Malfunction accident is the only other accident analyzed in the FSAR which takes credit for a rector trip on steam generator low-low level. During this accident a malfunction is postulated which causes an increase in feedwater flow. When the steam generator level in the faulted loop reaches the high-high level setpoint, all feedwater isolation valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps are tripped. This prevents a continuous addition of feedwater and initiates a turbine trip. The analysis does not take credit for a reactor trip on turbine trip. Consequently following the turbine trip, core power stabilizes at a reduced level consistent with the reactivity parameters assumed to maximize the initial increase in power. The reactor is tripped on steam generator low-low level if no action is taken by the operator to terminate the reduced power operation. The revised steam generator low-low level setpoint will only delay the time of reactor trip at the reduced power level. Even assuming a delayed reactor trip, the DNBR limit is not exceeded at any time during the accident.

This evaluation has examined the impact of the proposed steam generator low-low level setpoint on the accident analyses performed in the FSAR. All accidents which take credit for a reactor trip on steam generator low-low level were analyzed. The results of these analyses indicate that all safety criteria are met using the revised setpoint.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The proposed amendment does not involve an increase in the probability or consequences of any previously evaluated accident. The accident analyses have been reviewed and all acceptance criteria has been satisfied.

The proposed amendment does not create the possibility of a new or different kind of accident than any previously evaluated since there will be no physical changes made to any plant system other than the reduction of the trip setpoint and the addition of the filter to the channel circuitry.

The proposed amendment does not involve a significant reduction in a margin of safety. All applicable safety analyses have been reviewed and all acceptance criteria will be met with the revised setpoint.

The Commission has provided guidance concerning the application of standards of no significant hazards determination by providing certain examples (48 FR 14870). This change is similar to example (vi).

For the reasons stated above, it is concluded that the proposed amendment does not involve significant hazards considerations.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water					
a. Unit 1 Level Low-Low	17	14.2	1.5	≥17% of span from 0% to 30% RTP* increasing linearly to ≥ ~~54.9%~~ 40.0 of span from 30% to 100% RTP*	≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥~~53.2%~~ 38.3 of span from 30% to 100% RTP*
b. Unit 2	17	14.2	1.5	≥17% of narrow range span	≥15.3% of narrow range span
14. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
16. Turbine Trip					
a. Low Control Valve EH Pressure	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	≤ 1 second
b. Two Loops (Above P-7 and below P-8)	≤ 1 second
13. Steam Generator Water Level-Low-Low	≤ ~~2.0~~ 3.5 seconds
14. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	≤ 0.6 second
16. Turbine Trip	
a. Low Control Valve EH Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low					
1) Unit 1	17	14.2	1.5	≥ 17% of span from 0% to 30% RTP increasing linearly to ≥ ~~54.9%~~ 40.0 of span from 30% to 100% RTP	≥ 15.3% of span from 0% to 30% RTP increasing linearly to ≥ ~~53.2%~~ 38.3 of span from 30% to 100% RTP
2) Unit 2	17	14.2	1.5	≥ 17% of narrow range span	≥ 15.3% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
e. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low					
1) CAPS 5220, 5221, 5222	N.A.	N.A.	N.A.	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	N.A.	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low	N.A.	N.A.	N.A.	≥ 177.15 inches	≥ 162.4 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

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This time was based on a preliminary value for B_4C rod cluster control assemblies. The final safety analysis value is 3.30 seconds. As transients are reanalyzed for various reasons the 3.30 second drop time will be used. The ~~complete loss of flow~~ reanalyses in Sections ~~15.3.2 used the final value~~. 15.2.6, 15.2.7, 15.2.8, and 15.3.2 used the final value.

~~analyses~~ events, such as the loss of reactor coolant from cracks or ruptures in the Reactor Coolant System, do not depend on reactivity feedback effects. The values assumed for these coefficients are given in Table 15.0.3-2. Reference is made in that table to Figure 15.0.4-1 which shows, as a function of power, the upper and lower bound Doppler power coefficients ~~as a function of power~~, used in the transient analysis. The justification for the use of conservatively large versus small reactivity coefficient values ~~are~~ is treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life, ~~For example, in a load increase transient it is conservative to use a small doppler defect and a small moderator coefficient.~~ although these combinations may not represent possible realistic situations.

15.0.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

The negative reactivity insertion following a reactor trip is a function of the ~~acceleration~~ position versus time characteristic of the rod cluster control assemblies and of the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry, ~~or~~ approximately 85 percent of the rod cluster travel. The rod cluster control assembly position versus time characteristic assumed in the accident analyses is shown in Figure 15.0.5-1. The rod cluster control assembly insertion time to dashpot entry ~~is~~ was taken as 3.05 seconds. ~~unless otherwise noted in the discussion~~. Drop time testing requirements are ~~dependent on the type of cluster control assemblies actually used in the plant and are~~ specified in the plant Technical Specifications, and the initial startup test verified that either safety analysis assumption is conservative.

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

There is inherent conservatism in the use of Figure 15.0.5-2 in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

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

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used.

For the loss of non-emergency AC power to the station auxiliaries, Section 15.2.6, and the loss of normal feedwater, Section 15.2.7, core residual heat generation is based on Reference 5. For the loss of coolant accident, Section 15.6.5, residual heat generation

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 Total Residual Heat

~~Residual heat in a subcritical core~~ is calculated ~~for the loss of coolant accident~~ per the requirements of Appendix K of 10CFR50.46 (Reference 6) as described in References 7 and 8. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the ~~same~~ models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

Section 15.6.5 also

15.0.10.2 Distribution of Decay Heat Following Loss of Coolant Accident

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

For example, consider the transient resulting from the postulated double ended break of the largest Reactor Coolant System pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 COMPUTER CODES UTILIZED

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

15.0.11.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters

Chelemer, H., Boman, L. H., Sharp, D. R., "Improved Thermal Design Procedures", WCAP-8567, July, 1975.

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REFERENCES FOR SECTION 15.0

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

2. ORNL-4628, "ORIGEN - The ORNL Isotope Generation and Depletion Code," M. J. Bell, May 1973.

3. RSIC-DLC-38, "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From END F/B-IV," Radiation Shielding Information Center, Oak Ridge National Laboratory, September 1975.

4. ~~Belle, J., "Uranium Dioxide Properties and Nuclear Applications", Naval Reactors, Division of Reactor Development United Stated Atomic Energy Commission, 1961.~~

~~Deleted~~

5. ~~Booth, A. H., "A Suggested Method for Calculating the Diffusion of Radioactive Rare Gas Fission Products From UO² Fuel Elements," DCI-27, 1957.~~

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

7. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant," WCAP-8302 (Proprietary), and WCAP-8306 (Non-Proprietary), June 1974.

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

9. Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO² Fuel Rod," WCAP-7908, June 1972.

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

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

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

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

Note that this one page replaces both pages of the old Table 15.0.6-1

Table 15.0.6-1

Trip Points and Time Delays To Trip Assumed In Accident Analyses

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delays (Seconds)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
High neutron flux, P-8	85%	0.5
Overtemperature ΔT	Variable see Figure 15.0.3-1	6.0*
Overpower ΔT	Variable see Figure 15.0.3-1	6.0*
High pressurizer pressure	2410 psig***	2.0
Low pressurizer pressure	~~1835~~ 1921 psig***	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage trip	68% nominal	1.5
Turbine trip	Not applicable	2.0
Low-low steam generator level	**	4.0
High steam generator level feedwater pump trip, feedwater isolation, and turbine trip	~~87.4%~~ 93.6% of narrow range level span***	2.0

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

**The numerical setpoint assumed for this trip function varies depending on the accident being analyzed. The values used are given in the descriptions of the various accidents. In all cases which result in a reactor trip on low-low steam generator level, Unit 1 is more limiting and is thus used for the analysis.

***The discrepancies between the values and those given the the Catawba Nuclear Station Setpoint Methodology document are addressed in a June 25, 1984 letter from H. B. Tucker (Duke Power) to H. R. Denton (NRC)

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even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the secondary steam to the atmosphere. The pumps take suction from the auxiliary feedwater storage tank for delivery to the steam generators.

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

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

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

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

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

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

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

The assumptions used in the analysis are as follows:

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

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

3. A heat transfer coefficient in the steam generator associated with RCS natural circulation.

[Core residual heat generation is based on the 1979 version of ANS-5.1 (Reference 6). This method is a conservative representation of the decay energy release rates.

4. Reactor trip occurs on steam generator low-low narrow range level, assumed to be at ~~49.9%~~ 5% below the nominal setpoint. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

5. The worst single failure in the auxiliary feedwater system occurs. 491 gpm of ~~6.~~ a ~~A~~uxiliary feedwater is delivered to two steam generators.

6 ~~7~~. Secondary system steam relief is achieved through the steam generator safety valves.

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

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

Results

The transient response of the RCS following a loss of ac power is shown in Figures 15.2.6-1 through 15.2.6-4~~, 15.2.6-2, and 15.2.6-3~~. The calculated sequence of events for this event is listed in Table 15.2.3-1.

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

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

15.2.6.3 Environmental Consequences

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented assuming primary to secondary leakage. This analysis incorporates assumptions of 1 percent defective fuel and existence of a 1 gpm steam generator leak rate prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. Three postulated cases are analyzed:

Case 1 (No iodine spike)

Case 2 (With pre-existing iodine spike)

Case 3 (With coincident iodine spike)

15.2.7.2 Analysis of Effects and Consequences

Method of Analysis

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

Assumptions made in the analysis are:

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

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

3. Reactor trip occurs on steam generator narrow range low-low level, assumed to be at 5% below the nominal setpoint ~~17.9%~~.

4. The worst single failure in the auxiliary feedwater system occurs. 491 gpm of ~~5. a~~ auxiliary feedwater is delivered to two steam generators.

5 ~~6~~. Secondary system steam relief is achieved through the steam generator safety valves.

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

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

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

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

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

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

Results

Figures 15.2.7-1~~, 15.2.7-2, and 15.2.7-3~~ through 15.2.7-3 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, at least one auxiliary feedwater pump is automatically started, reducing the rate of water level decrease.

The capacity of the Auxiliary Feedwater System is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS safety valves. Figure ~~15.2.7-1~~ 15.2.7-2 shows that at no time is there water relief from the pressurizer.

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

15.2.7.3 Environmental Consequences

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

15.2.7.4 Conclusions

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

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

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

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

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

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

9. Reactor trip is assumed to be ~~initialed where~~ initiated at 17 percent of narrow range span below the adjusted low-low level trip setpoint ~~minus 10 percent of the narrow range span~~ in the faulted steam generator. ~~is reached.~~

10. The Auxiliary Feedwater System is actuated by the low-low steam generator water level signal. The Auxiliary Feedwater System is assumed to supply a total of 492 gallons per minute (gpm) to the two unaffected steam generators, including allowance for possible spillage through the main feedwater line break. A 60 second delay was assumed following the low-low level signal to allow time for startup of the emergency diesel generators and the auxiliary feedwater pumps. An additional 115 seconds was assumed before the feedwater lines were purged and the relatively cold (134°F) auxiliary feedwater entered the unaffected steam generators.

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

12. No credit is taken for charging or letdown.

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

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

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

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

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

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

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

2. Stop the ~~high head~~ safety injection ~~charging~~ pumps if:

   a. Wide range reactor coolant pressure is stable or increasing. ~~greater than 2000 psig, and is stable or increasing.~~

   b. Pressurizer water level is ~~greater than 50 percent or~~ on span.

   c. RCS is adequately subcooled.

   d. Steam generator narrow sufficient ~~range~~ level indication exists in at least one steam generator or auxiliary feedwater is being injected into ~~at~~ the ~~least one non-faulted~~ steam generators to provide an adequate heat sink.

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

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

No reactor control systems are assumed to function. The Reactor Protection System is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system. A discussion of ATWT considerations is presented in Reference 2.

The engineered safety systems assumed to function are the Auxiliary Feedwater System and the Safety Injection System. For the Auxiliary Feedwater System, the worst case configuration has been used, i.e., two intact steam generators receive auxiliary feedwater following the break. The turbine driven auxiliary feedwater pump is assumed to fail. One motor driven auxiliary

spill its entire flow out the break.
feedwater pump has been assumed to ~~fail;~~ the second motor driven pump ~~together with the turbine driven pump~~ delivers 492 gpm to the two intact steam generators. ~~allowing for spillage out of the break.~~ Only one train of safety injection has been assumed to be available.

For the case without offsite power there will be a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Section 15.2.6, for the loss of AC non-emergency power transient, to be sufficient to remove core decay heat following

REFERENCES FOR SECTION 15.2

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

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

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

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

5. Lang, G. E., Cunningham, J. B., "Report on the Consequences of a Postulated Main Feedline Rupture," WCAP-9230, January 1978.

", WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), April, 1984.

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

Table 15.2.3-1 (Page ~~X~~ 5 ~~of 8~~ 7)

Time Sequence Of Events For Incidents Which Cause A Decrease

In Heat Removal By The Secondary System

Accident	Event	Time (sec)
Loss of Non-Emergency AC Power	Main feedwater flow stops	10
	Low-low steam generator level reactor trip	~~53~~ ~~60~~ 54
	Rods begin to drop	~~55~~ ~~62~~ 58
	Reactor coolant pumps begin to coastdown	~~55~~ ~~64~~ 60
	Peak water level in pressurizer occurs	~~57~~ ~~65~~ 62
	Auxiliary feedwater pumps start	~~120~~ 114
	Two ~~Four~~ steam generators begin to receive auxiliary feedwater from one ~~two~~ motor driven ~~auxiliary feedwater~~ pump~~s~~	~~125~~ ~~176~~ 190
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~ ~~500~~ ~~770~~ 1500
Loss of Normal Feedwater Flow	Main feedwater flow stops	10
	Low-low steam generator level reactor trip	~~53~~ ~~51~~ 54
	Rods begin to drop	~~55~~ ~~53~~ 58
	Peak water level in pressurizer occurs	~~67~~ ~~9446~~ 62
	Auxiliary feedwater pumps start	~~111~~ 114
	Two ~~Four~~ steam generators begin to receive auxiliary feedwater from one ~~two~~ motor-driven ~~auxiliary feedwater~~ pump~~s~~	~~125~~ ~~188~~ 190
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~ ~~1000~~ ~~9346~~ 3300

Table 15.2.3-1 (Page ~~X of X~~ 6) 7)

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

Accident	Event	Time (sec)
Feedwater System Pipe Break		
1. With offsite power available	Main feedline rupture occurs	~~10~~ 10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	~~27~~ ~~31~~ 28
	Rods begin to drop	~~29~~ ~~33~~ 32
	Auxiliary feedwater pumps start	~~31~~ 88
	~~Auxiliary feedwater is delivered to intact steam generators~~ Two steam generators begin to receive auxiliary feedwater from one motor driven pump	~~87~~ ~~168~~ 164
	Low steam line pressure setpoint reached in ruptured steam generator	~~384~~ ~~247~~ 246
	All main steam line isolation valves close	~~391~~ ~~254~~ 253
	Steam generator safety valve setpoint reached in intact steam generators	~~844~~ ~~464~~ 461
	Pressurizer water relief begins	~~1888~~ ~~1476~~ 1461
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~~~5000~~ ~~3980~~ ~4000

Table 15.2.3-1 (Page ~~8~~ 7 ~~of 8~~ 7)

Time Sequence Of Events For Incidents Which Cause A Decrease

In Heat Removal By The Secondary System

Accident	Event	Time (sec)	
Feedwater System Pipe Break			
2. Without offsite power	Main feedline rupture occurs	~~10~~	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	~~27~~ ~~31~~	28
	Rods begin to drop, Power lost to the reactor coolant pumps	~~28~~ ~~33~~	32 34
	Auxiliary feedwater pumps start	~~31~~	89
Two steam generators begin to receive auxiliary feedwater from one motor driven pump	~~Auxiliary feedwater is delivered to intact steam generators~~	~~27~~ ~~166~~	164
	Low steam line pressure setpoint reached in ruptured steam generator	~~451~~ ~~256~~	248
	All main steam line isolation valves close	~~458~~ ~~263~~	255
	Steam generator safety valve setpoint reached in intact steam generators	~~1108~~ ~~568~~	538
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~~2000~~ ~~960~~	~1580

[redacted]

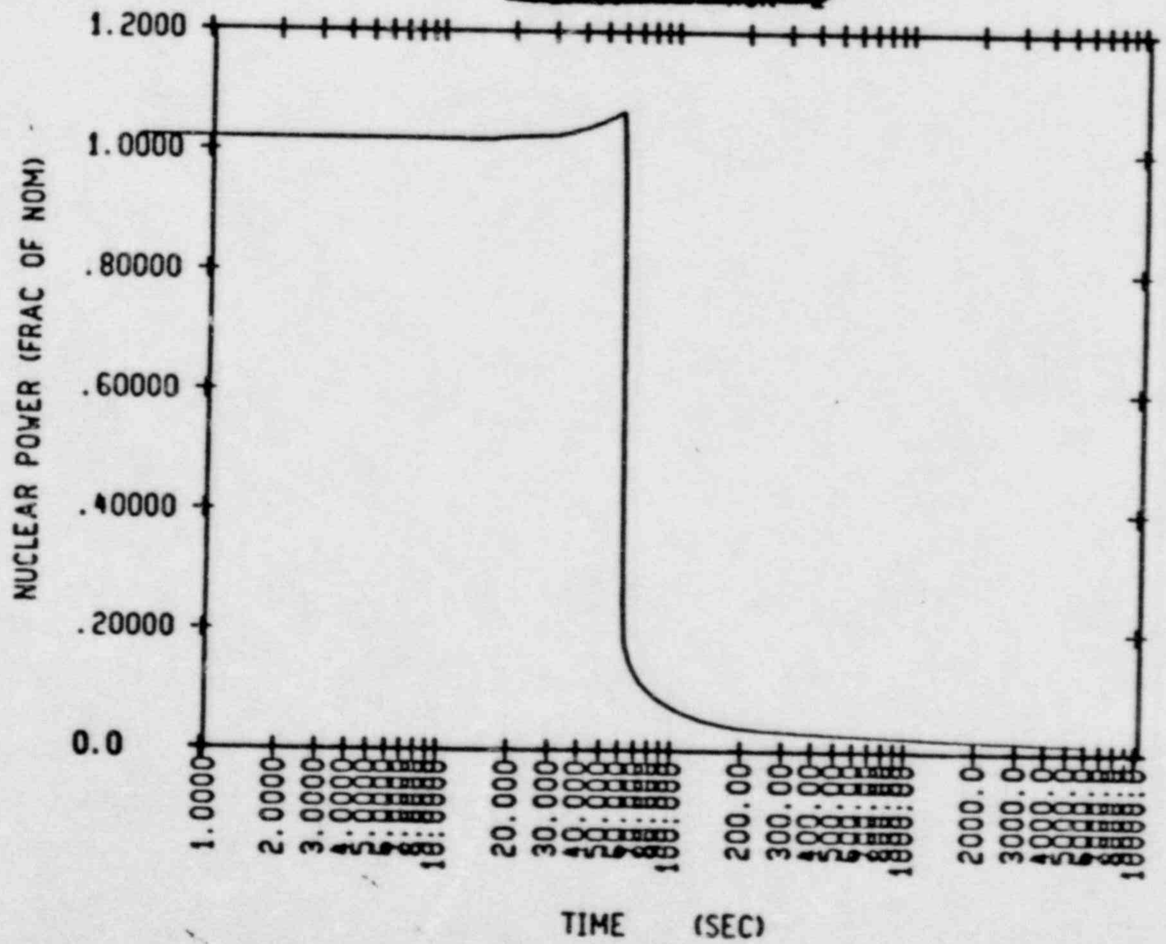

Fig 15.2.6-1 (1 of 2)

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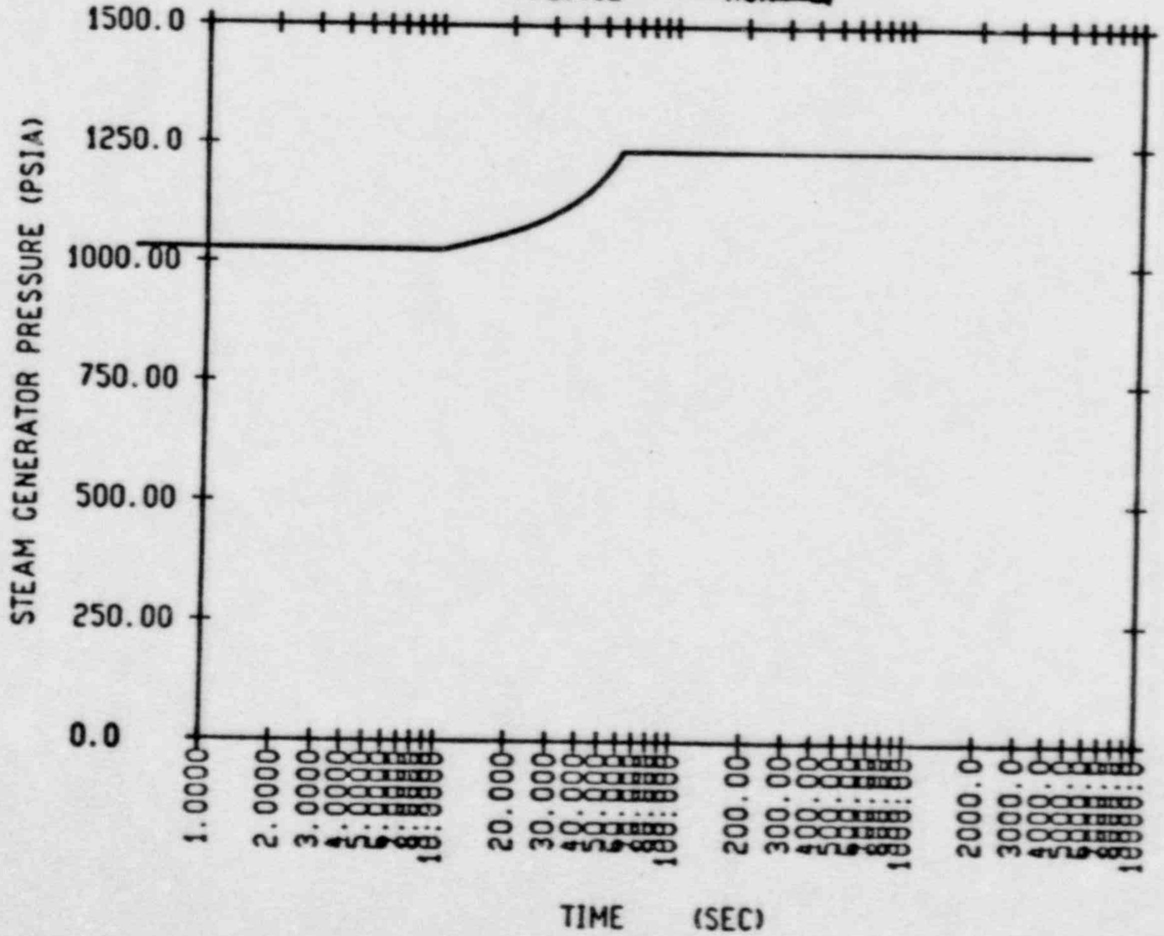

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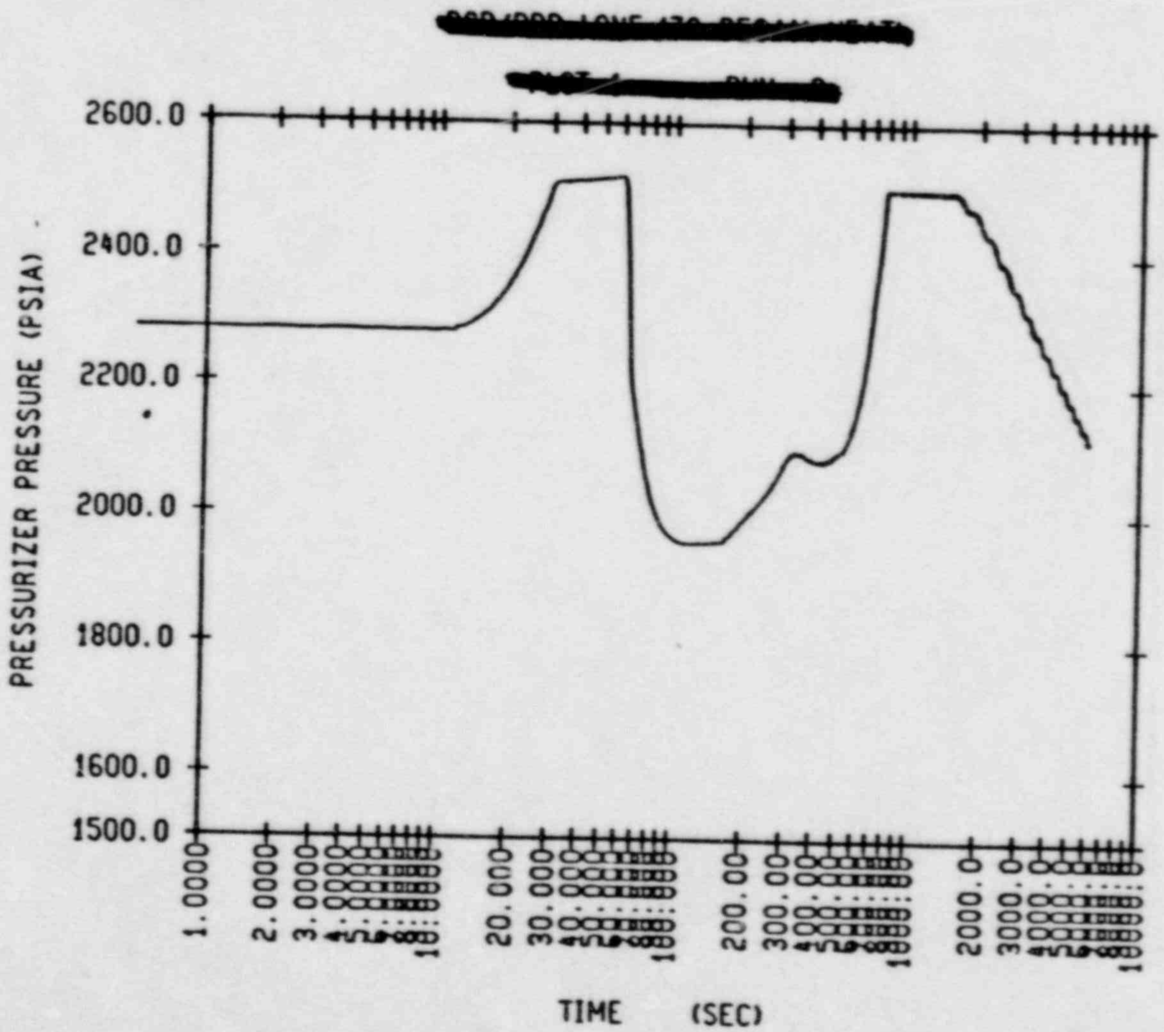

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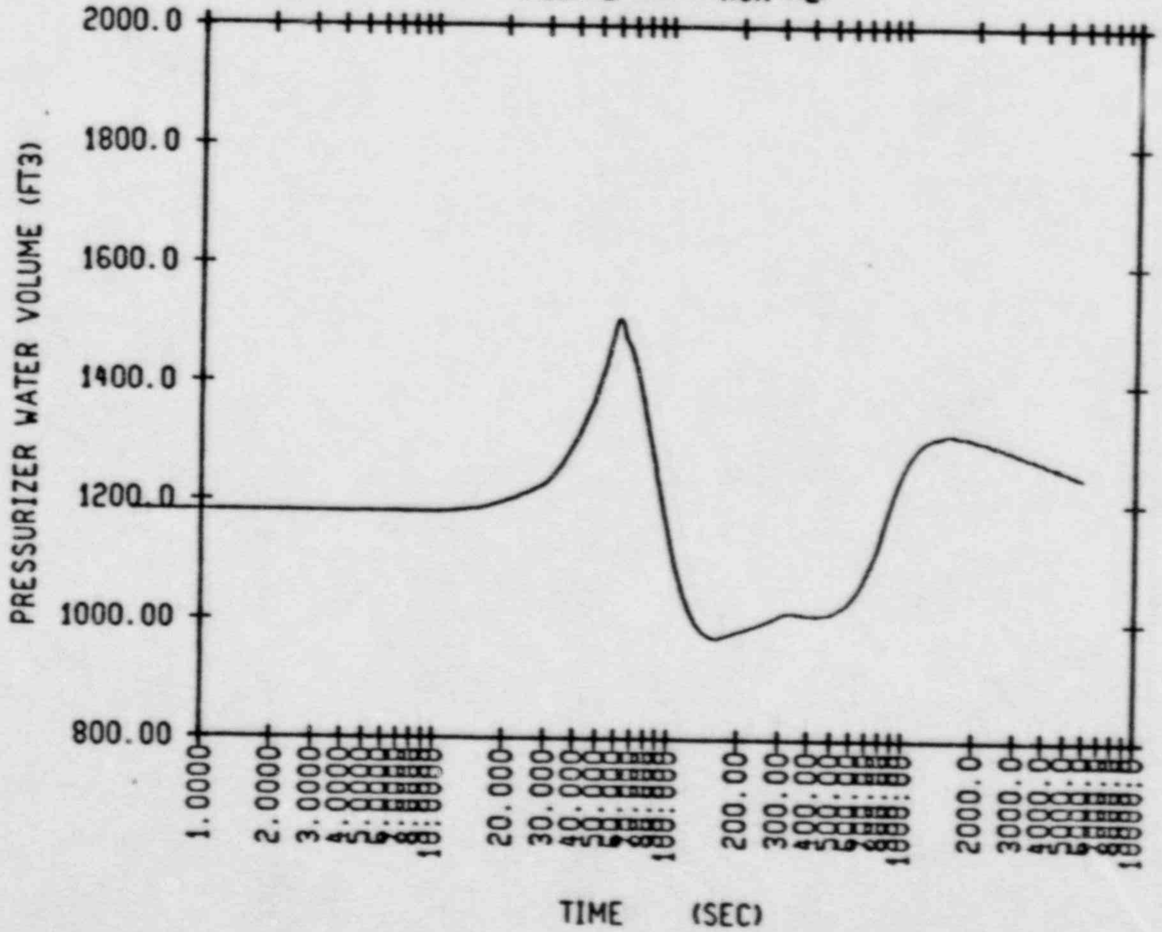

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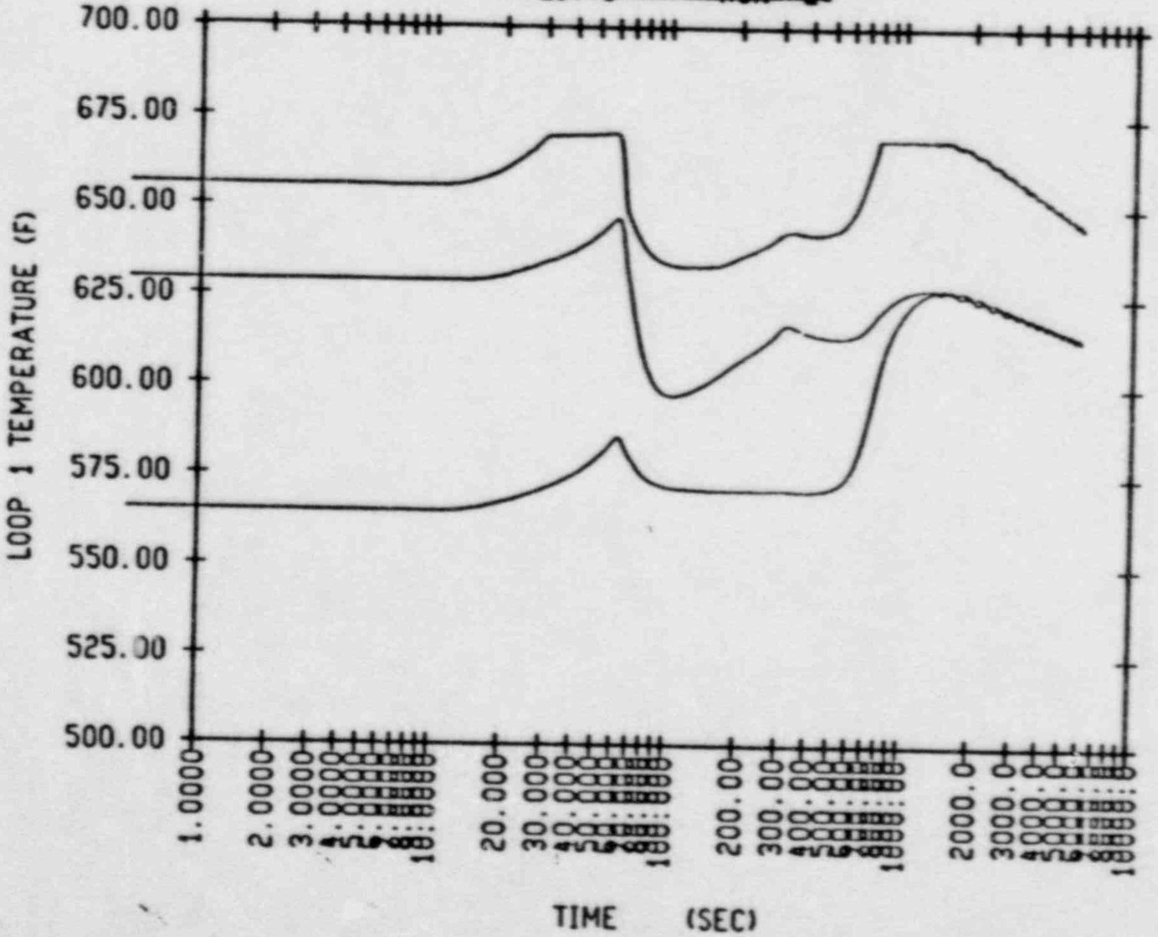

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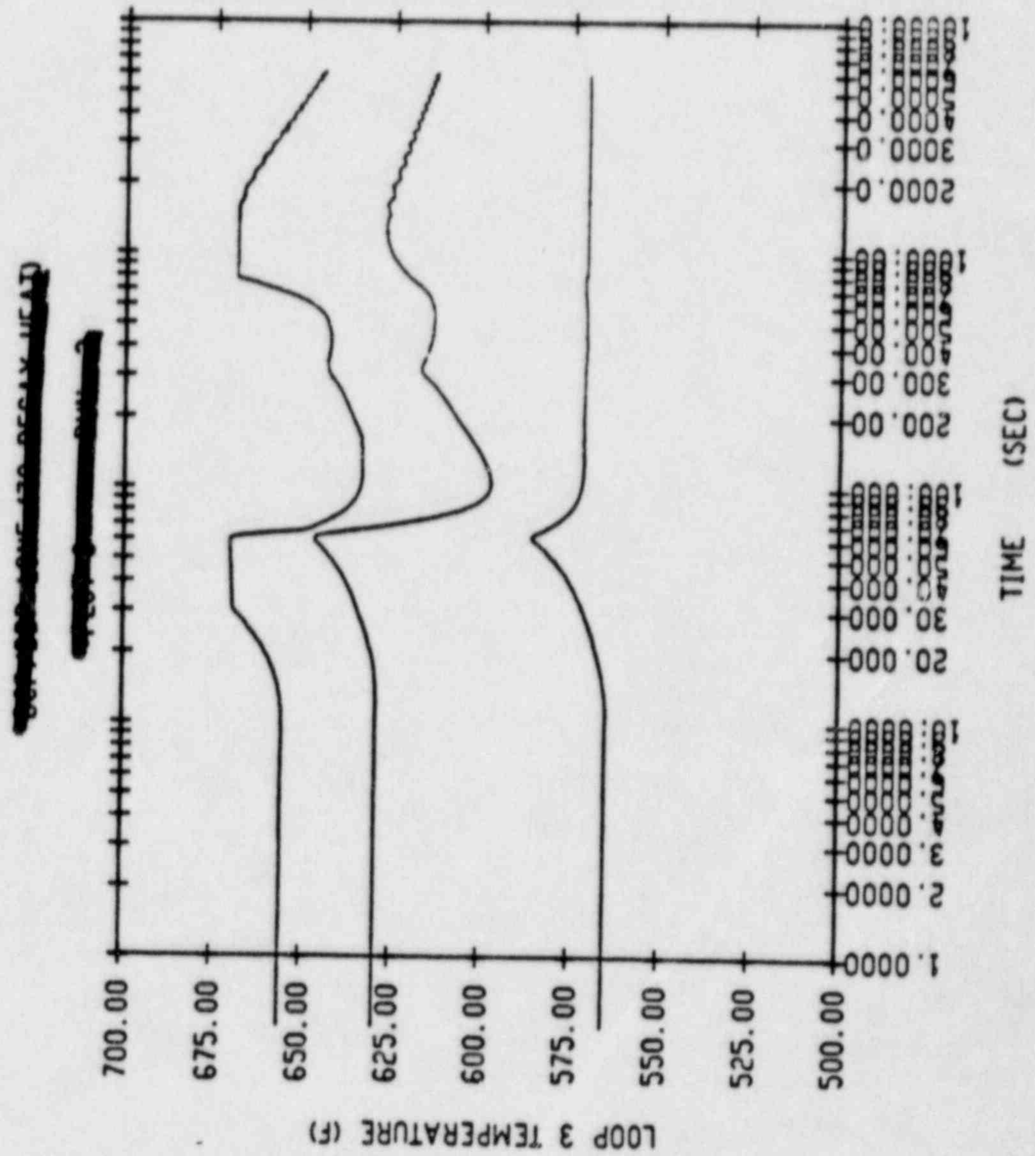

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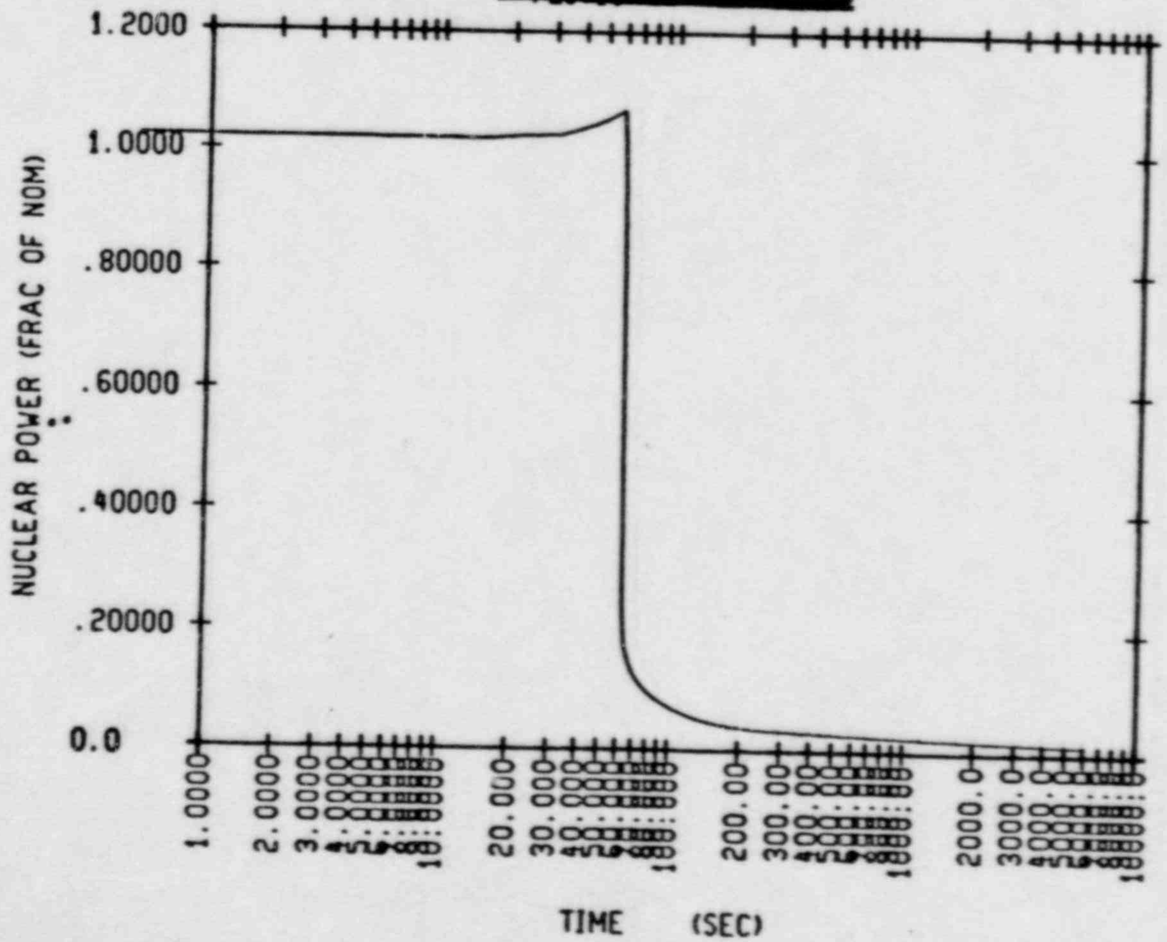

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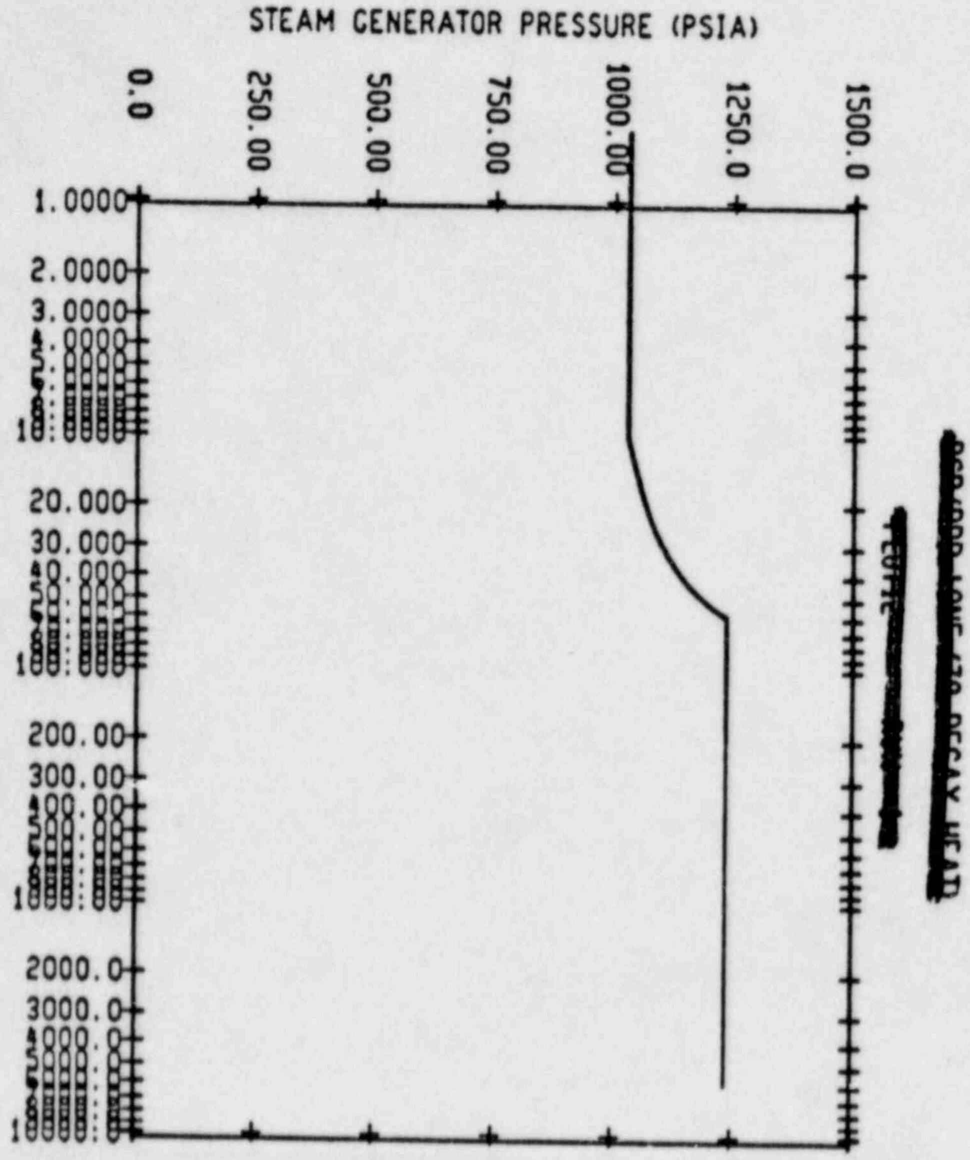
Fig 15.2.7-1 (2 of 2)



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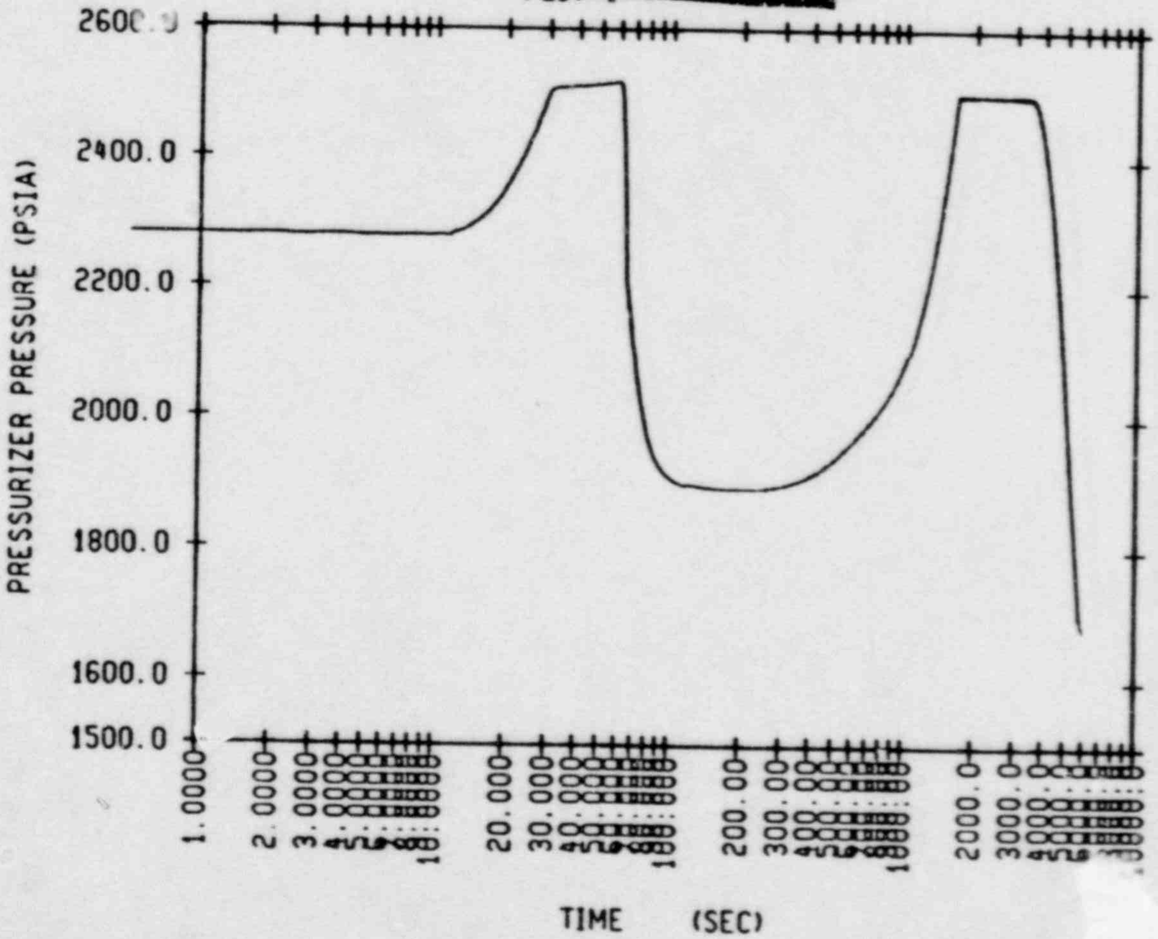

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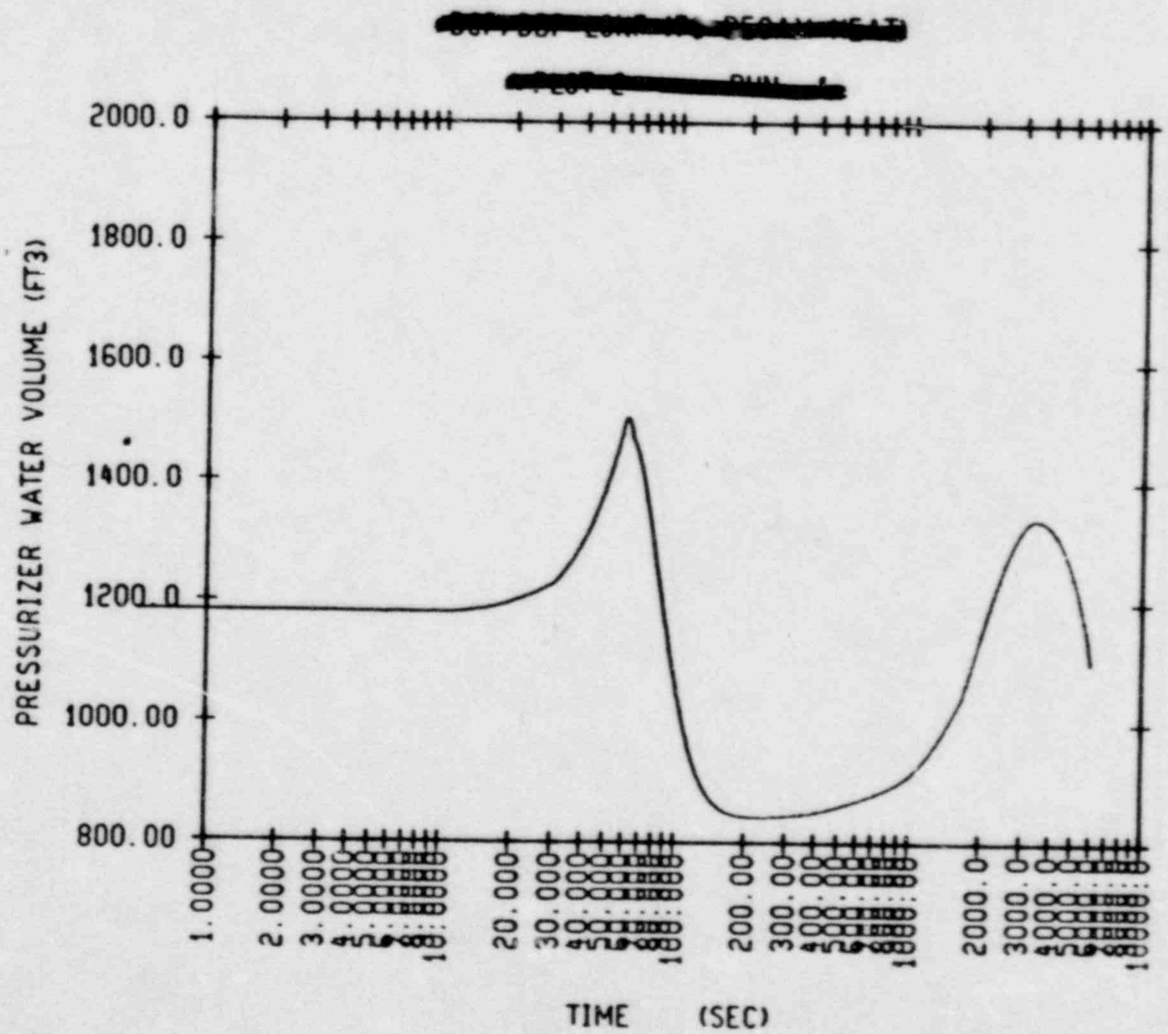

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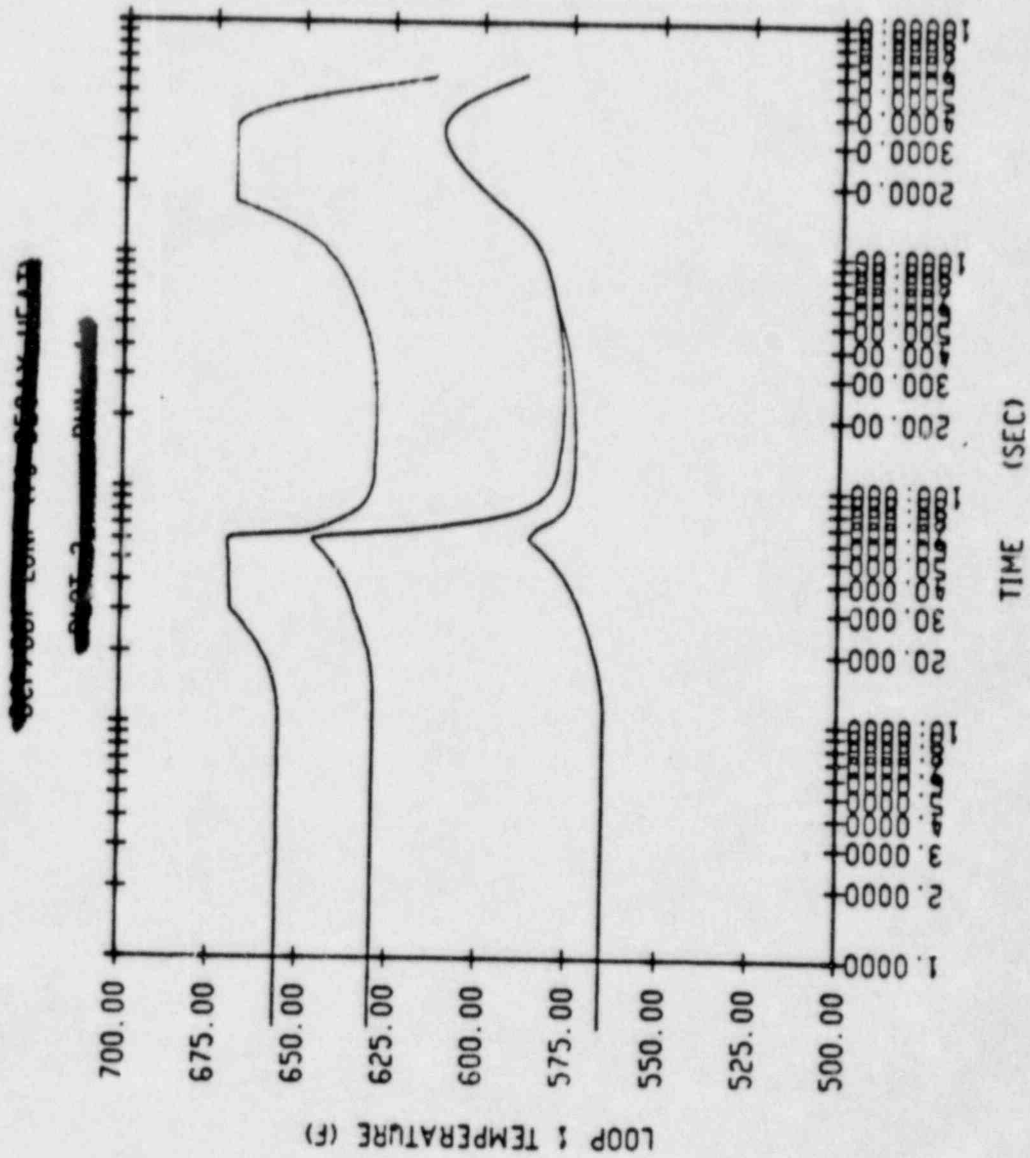

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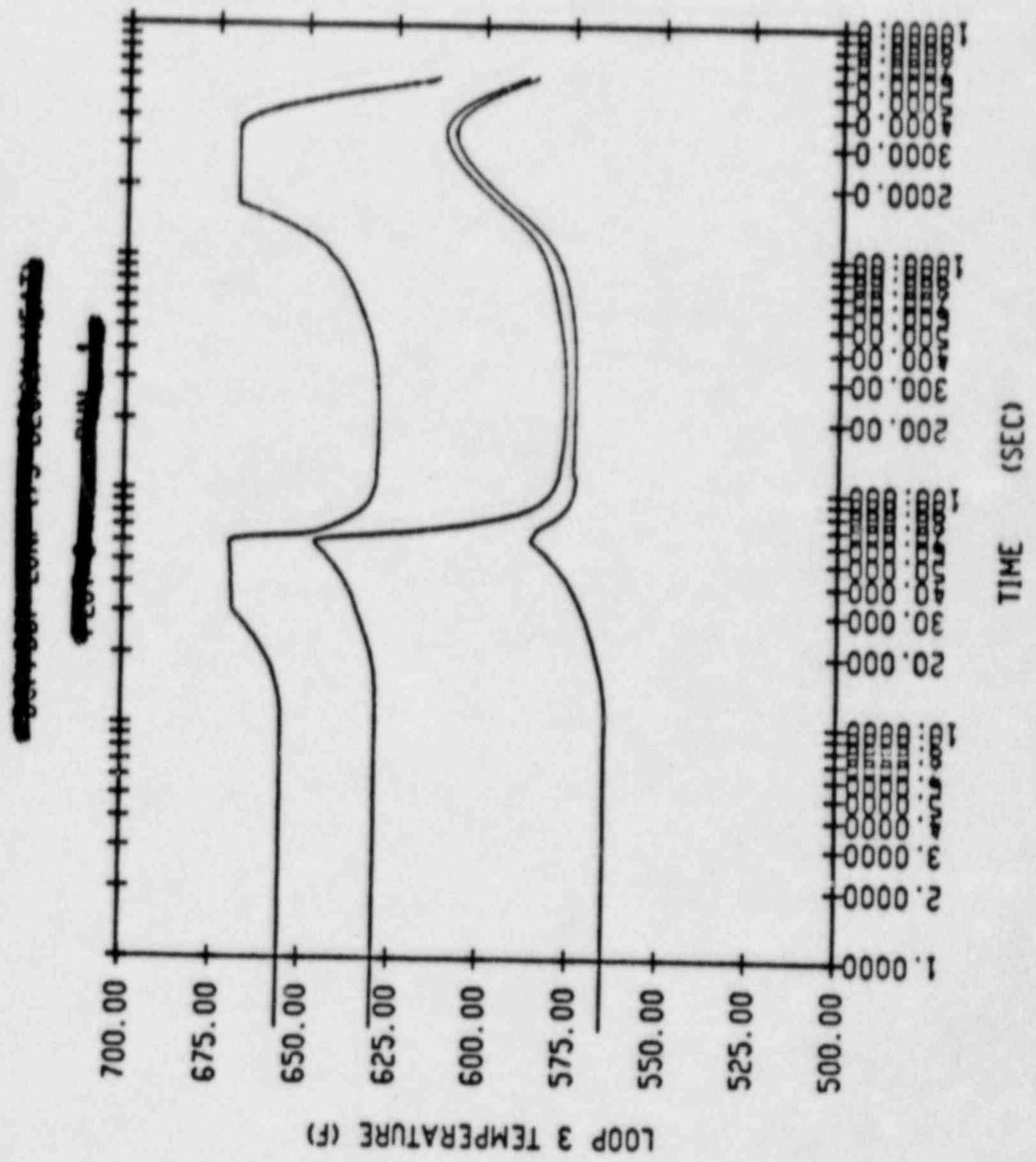

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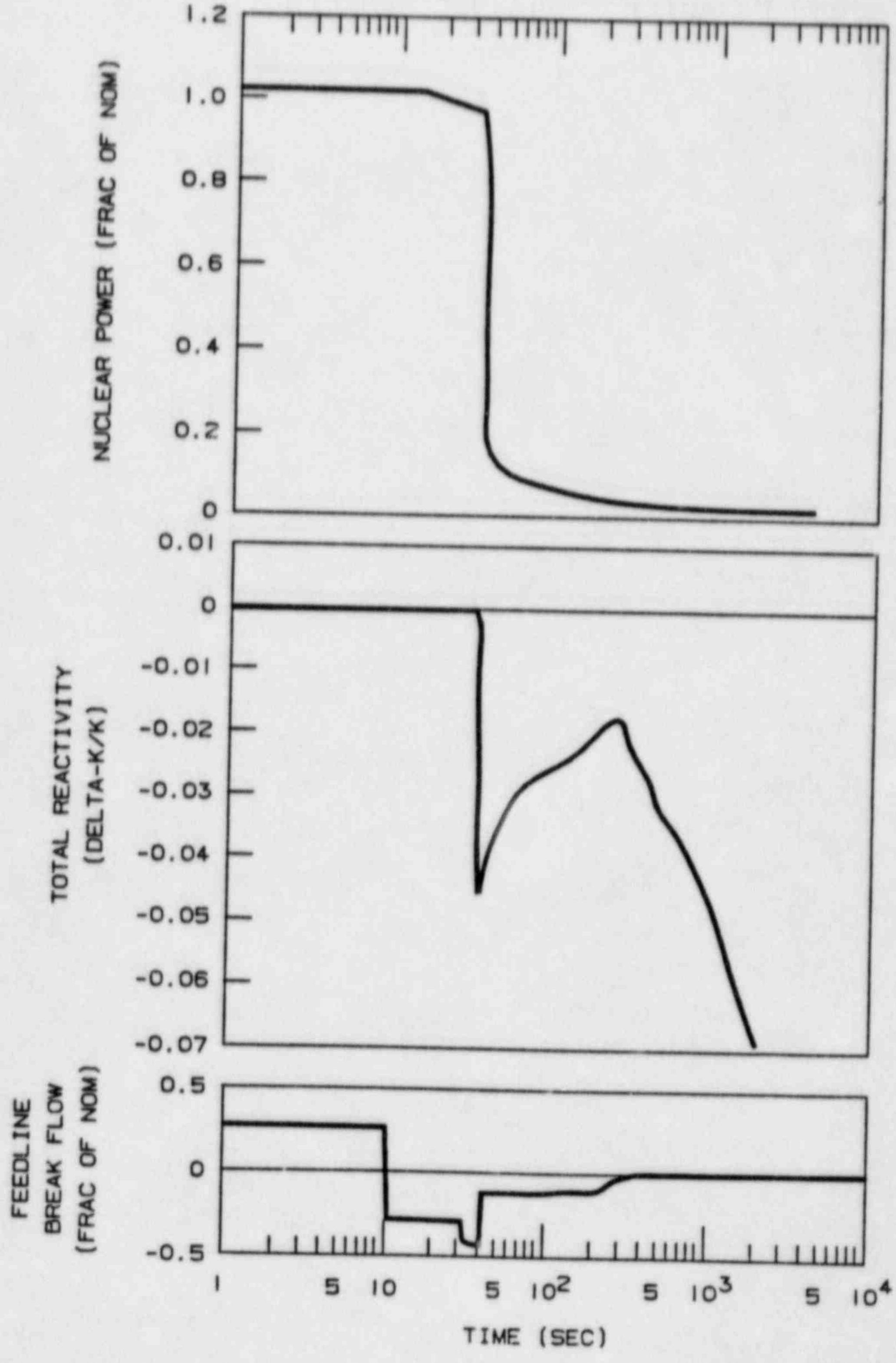
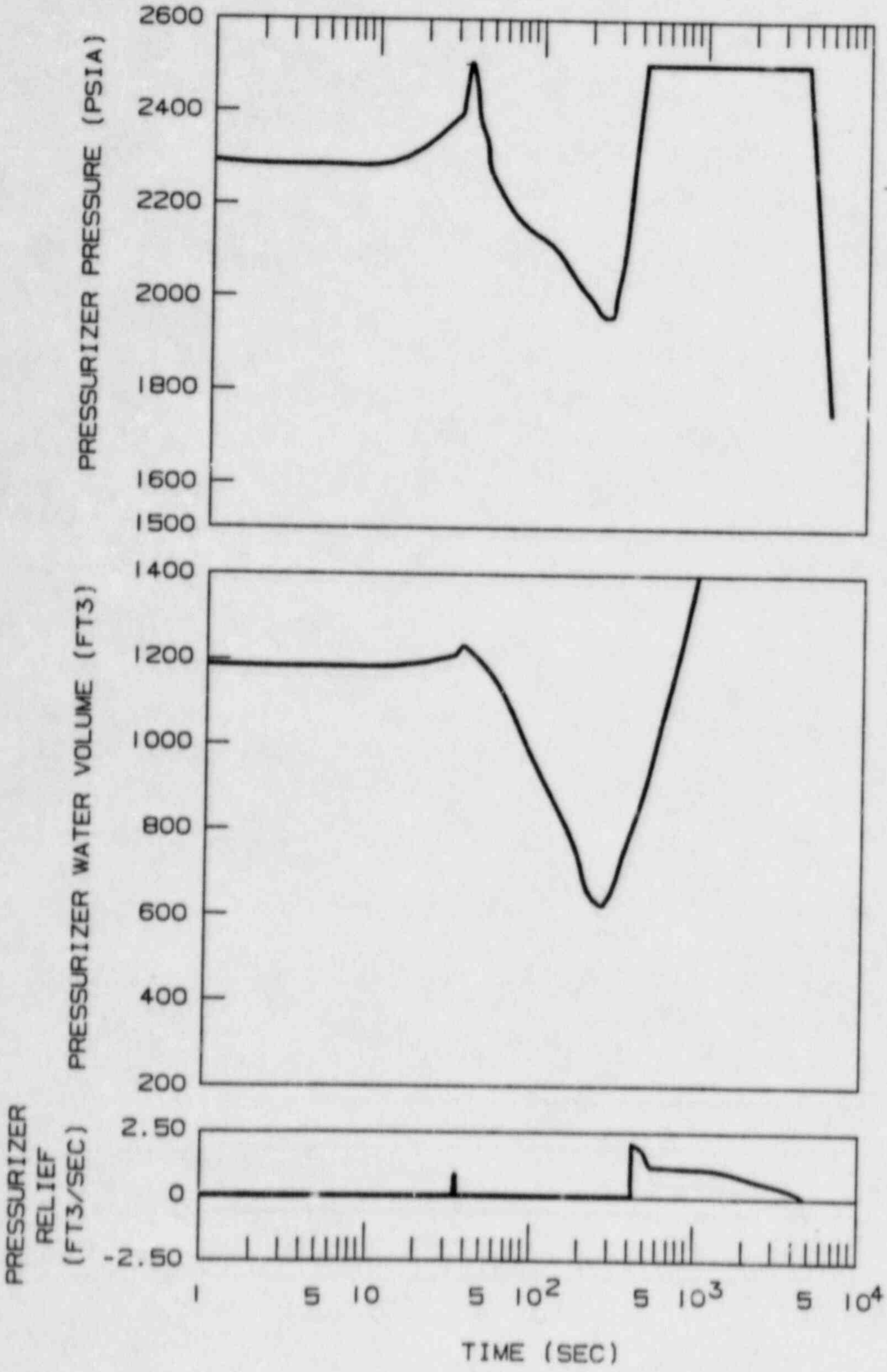
14429-7



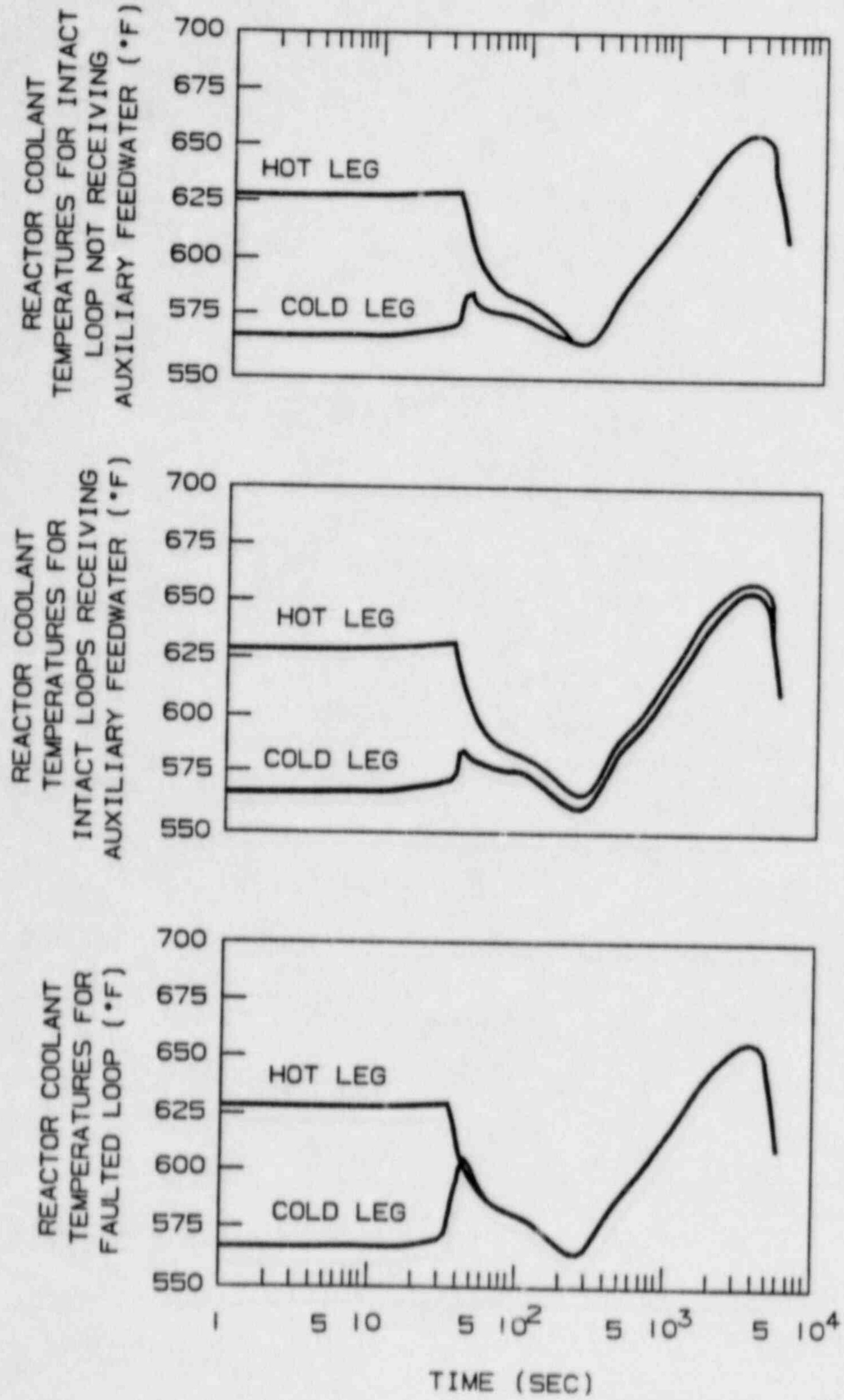
Nuclear Power Transient,
Total Core Reactivity Transient,
and Feedline Breakflow Transient
for Main Feedline Rupture with
Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-1

14429-8



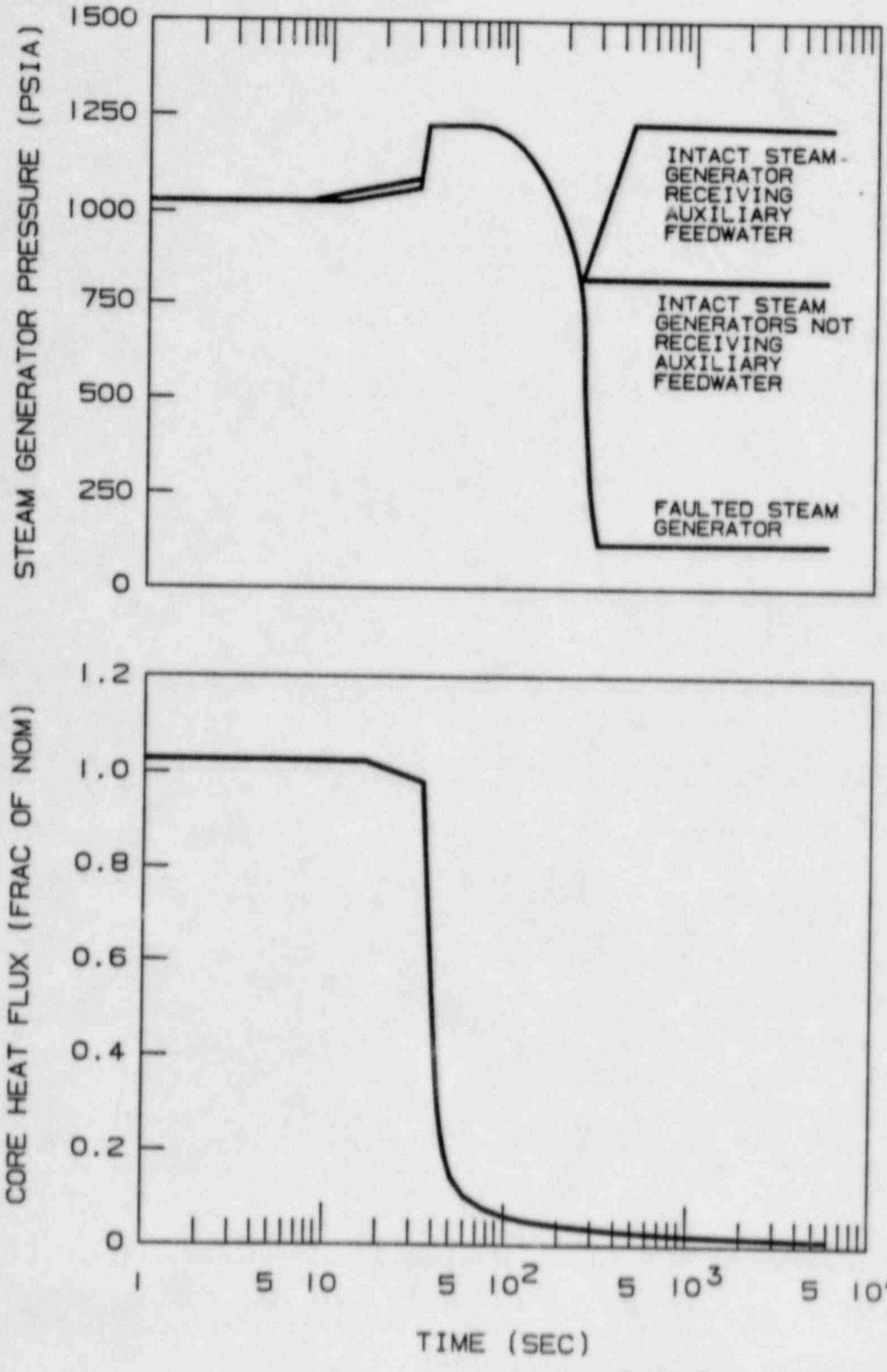
Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture with Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-2

14429-9



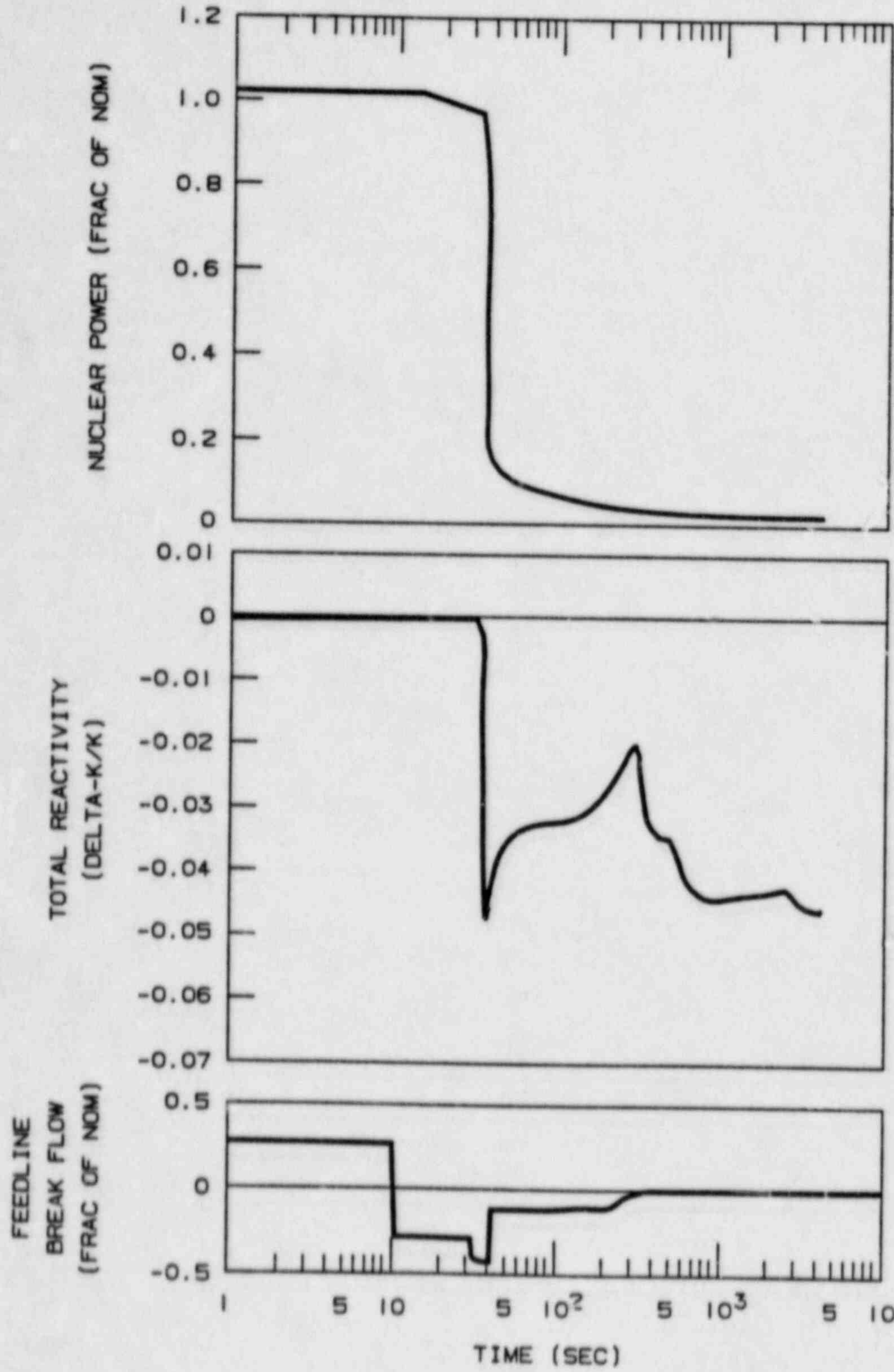
Reactor Coolant Temperature Transients for the Faulted and the Intact Loops for Main Feedline Rupture with Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-3

14429-10

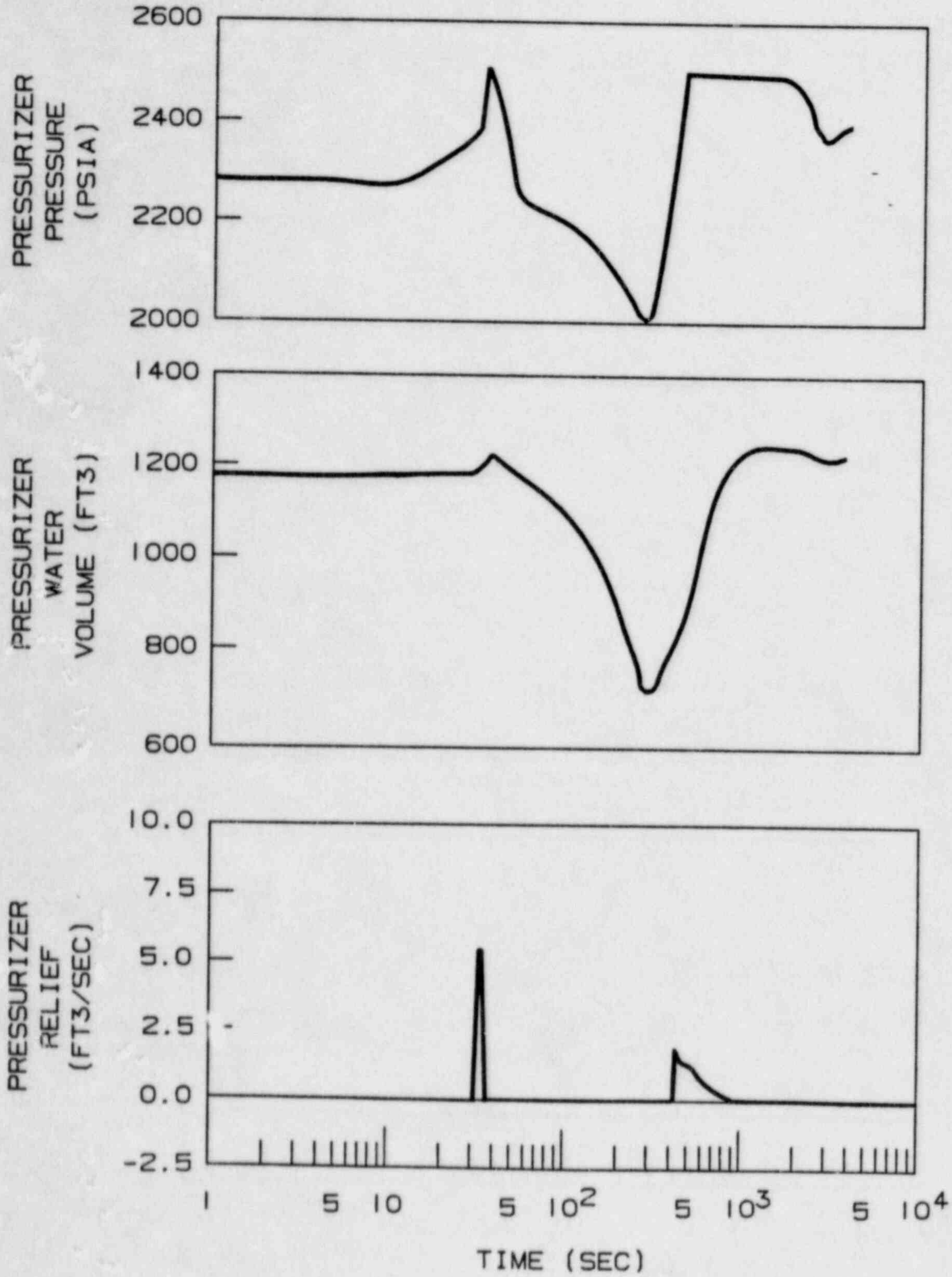


Steam Generator Pressure and Core Heat Flux Transients for Main Feedline Rupture with Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-4

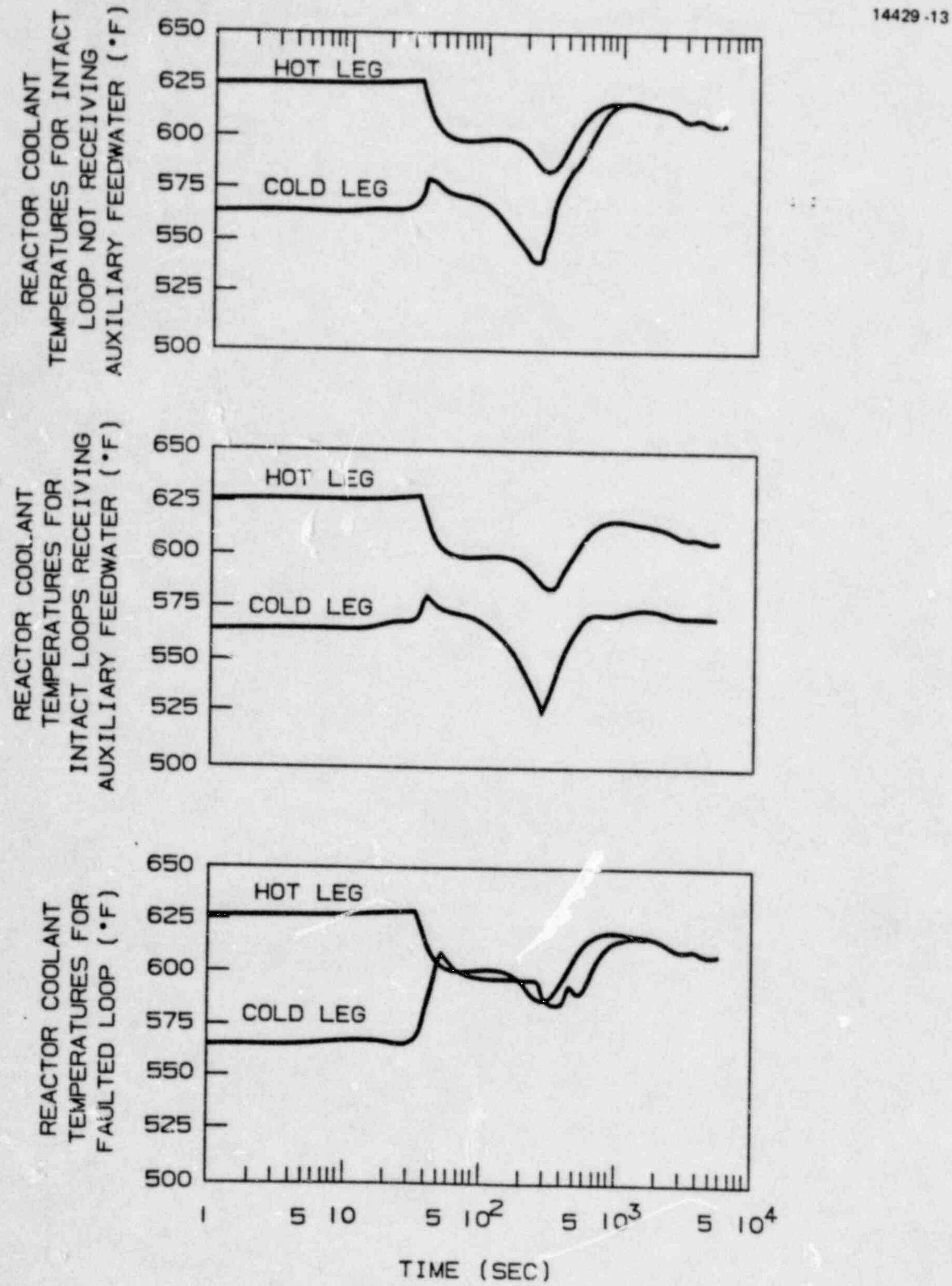
14429-11



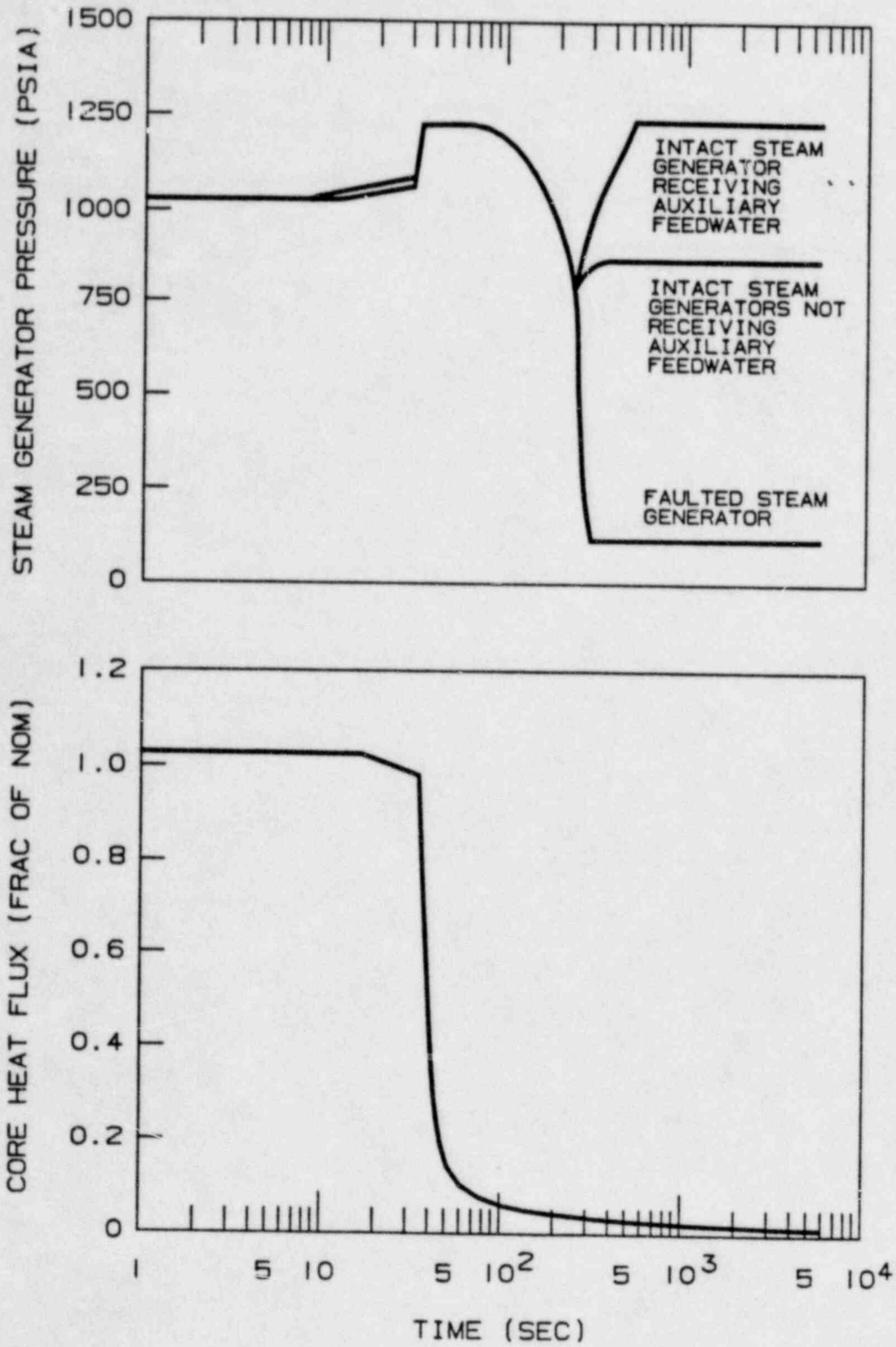
Nuclear Power Transient,
Total Core Reactivty Transient,
and Feedline Breakflow Transient for
Main Feedline Rupture without
Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-5



Pressurizer Pressure, Water Volume, and Relief Rate for Main Feedline Rupture without Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-6



Reactor Coolant Temperature Transients for the Faulted and Intact Loops for Main Feedline Rupture without Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-7



Steam Generator Pressure
and Core Heat Flux Transients
for Main Feedline Rupture
without Offsite Power Available
CATAWBA NUCLEAR STATION
Figure 15.2.8-8