TENNESSEE VALLEY AUTHORITY
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JAN 27 1987

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. B. J. Youngblood

| | | |
|---|---|---|
| In the Matter of the Application of | ) | Docket Nos. 50-390 |
| Tennessee Valley Authority | ) | 50-391 |

WATTS BAR NUCLEAR PLANT (WBN) - RTD BYPASS LOOP ELIMINATION/UTILIZATION OF EAGLE 21 ELECTRONICS - FSAR CHAPTER 15

The purpose of this letter is to provide NRC with planned FSAR revisions associated with the subject modification. This submittal completes a TVA task as scheduled in my November 3, 1986 letter.

Enclosed are marked-up pages to the WBN FSAR Chapter 15 reflecting the elimination of the RTD bypass loop and the utilization of Eagle 21 system electronics. According to the schedule, it is requested that NRC review these proposed changes and forward questions to TVA the week of February 23, 1987. The enclosed changes, along with any other changes resulting from your review, will be incorporated into a subsequent FSAR amendment.

If you have questions on this topic, please get in touch with Bernie Gergos at (615) 365-8837.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. A. Domer

R. Gridley, Director
Nuclear Safety and Licensing

Enclosures
cc: See page 2



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1/1

U.S. Nuclear Regulatory Commission

JAN 27 1987

cc (Enclosures):

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MARVEL also has the capability of calculating the transient value of DNB ratio on the input from the core limits illustrated on Figure 15.1-1. The core limits represent the minimum value of DNBR as calculated for a typical or thimble cell.

MARVEL is further discussed in Reference [14].

15.1.9.4 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multi-loop system by a lumped parameter single loop model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on neutron flux, overpower and overtemperature reactor coolant delta-T, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The Safety Injection System including the accumulators are also modeled.

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

LOFTRAN also has the capability of calculating the transient value of DNB ratio based on the input from the core limits illustrated on Figure 15.1-1. The core limits reprsent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference [15] and Reference [21]

15.1.9.5 LEOPARD

The LEOPARD computer program determines fast and thermal neutron spectra, using only basic geometry and temperature data. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step. 53

LEOPARD is further described in Reference [16].

10. Stehn, J. R. and Clancy, E. F., "Fission-Product Radioactivity and Heat Generation" and "Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958," Volume 13, pp. 49-54, United Nations, Geneva, 1958.

11. Obenshain, F. E. and Foderaro, A. H., "Energy from Fission Product Decay," WAPD-P-652, 1955.

12. Hunin, C., "FACTRAN, a Fortran IV Code for Thermal Transients in a UO Fuel Rod," WCAP-7908, June, 1972.

13. Geets, J. M. and Salvatori, R., "Long Term Transient Analysis Program for PWR's (BLKOUT Code)," WCAP-7898, June, 1972.

14. Geets, J. M., "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June, 1972.

15. Burnett, T. W. T., McIntyre, C. J., Buker, J. C. and Rose, R. P., "LOFTRAN Code Description," WCAP-7907, June, 1972.

16. Barry, R. F., "LEOPARD, a Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, September, 1963.

17. Barry, R. F. and Altomare, S., "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213-P-A, January, 1975 (Proprietary) and WCAP- 7758-A, January, 1975 (Non-Proprietary).

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

19. Bordelon, F. M., "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7973, September, 1972.

20. Skaritka, J. (Ed.), "Hybrid B C Absorber Control Rod Evaluation Report," WCAB-8846, Rev. 1, February, 1977. 59

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

TABLE 15.[illegible]3

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

| Trip Function | Limiting Trip Point Assumed In Analysis | Time Delays (Second) |
|---|---|---|
| Power Range High Neutron Flux, High Setting | 118% | 0.5 |
| Power Range High Neutron Flux, Low Setting | 35% | 0.5 |
| Overtemperature $\Delta T$ | Variable (see Figure 15.1-1) | ~~6.0*~~ 7.0* |
| Overpower $\Delta T$ | Variable (see Figure 15.1-1) | ~~6.0*~~ 7.0* |
| High Pressurizer Pressure | 2445 psig | 2.0 |
| Low Pressurizer Pressure | 1845 psig | 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.

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

1. High neutron flux (one out of four)

2. Overpower ΔT (two out of four)

3. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is described in Chapter 7. Figure 15.1-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals 1.30. All points below and to the left of a DNB line for a given pressure have a DNBR greater than 1.30. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

### 15.2.2.2. Analysis of Effects and Consequences

#### Method of Analysis

This transient is analyzed by the LOFTRAN [~~5~~ 12] Code. This code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient. The core limits are calculated by applying the "R" grid spacer factor to the W-3 DNB correlation.

In order to obtain conservative values of DNBR the following assumptions are made:

1. Initial conditions of maximum core power and reactor coolant average temperature and minimum reactor coolant pressure, resulting in the minimum initial margin to DNB. (pressurizer pressure -46 psi allowance for steady state fluctuations and measurement error)

2. Reactivity Coefficients - Two cases are analyzed:

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

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

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

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

5. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed. This is also much greater than the maximum reactivity insertion rate associated with withdrawal of a part length RCCA.

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

## Results

The calculated sequence of events for this accident is shown on Table 15.2-1.

Figures 15.2-4 and 15.2-5 show the response of neutron flux, pressurizer pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in $T_{avg}$ and pressure result and a large margin to DNB is maintained.

The response of neutron flux, pressure, average coolant temperature, and DNBR for a slow control rod assembly withdrawal from full power is shown in Figures 15.2-6 and 15.2-7. Reactor trip on overtemperature $\Delta T$ occurs after a longer period of time ~~and for~~ 53
~~the rise in temperature and pressure is consequently larger~~ than for the rapid RCCA withdrawal incident and the rise in temperature is consequently larger.

Following reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator 30
level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stablized condition will be in a time frame in excess of ten minutes following reactor trip.

Figure 15.2-8 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and maximum reactivity feedback. It can be seen that two reactor 53
trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature $\Delta T$ trip channels. The minimum DNBR is never 53
less than 1.30.

Figures 15.2-9 and 15.2-10 show the minimum DNBR as function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature $\Delta T$ trip is effective is increased. In neither case does the DNBR fall 53
below 1.30.

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

Referring to Figure 15.2-9, for example, it is noted that

1. For high reactivity insertion rates (i.e. between ~~2.0~~ 4.0 x $10^{-4}$ $\Delta k/k/sec$ and 8.0 x $10^{-4}$ $\Delta k/k/sec$) reactor trip is initiated 53
by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates while core heat

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

3. With further decrease in reactivity insertion rate, the overtemperature $\Delta T$ and high neutron flux trips become equally effective in terminating the transient (e.g., at $\sim$ ~~2.0~~ 4.0 $\times 10^{-4}$ $\delta$K/sec reactivity insertion rate).

   For reactivity insertion rates between $\sim$ ~~2.0~~ 4.0 $\times 10^{-4}$ $\sim$ K/sec and $\sim 5 \times 10^{-5}$ $\delta$K/sec the effectiveness of the overtemperature $\Delta T$ trip increases (in terms of increased minimum DNB 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.

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

For transients initiated from higher power levels (for example, see Figure 15.2-8) this effect, described in item 4 above, which results in the sharp peak in minimum DNB ratio at $\sim 5 \times 10^{-5}$ $\delta$k/sec, does not occur since the steam generator safety valves are never actuated prior to trip.

Figures 15.2-8, 15.2-9 and 15.2-10 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

### 15.2.2.3 Conclusions

The high neutron flux and overtemperature $\Delta T$ trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than 1.30.

bine trip, no direct reactor trip signal would be generated. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

Onsite power supplies plant auxiliaries during plant operation, e.g., the reactor coolant pumps. Safeguards loads are supplied from offsite power or, alternatively, from emergency diesels. Reactor Protection System equipment is supplied from the 118V AC instrument power supply system, which in turn is supplied from the inverters; the inverters are supplied from a DC bus energized from batteries or rectified AC from safeguards buses. Thus, for postulated loss of load and subsequent turbine generator overspeed, any overfrequency condition is not seen by safety related pump motors, Reactor Protection System equipment, or other safeguards loads. Any increased frequency to the reactor coolant pump motors will result in slightly increased flowrate and subsequent additional margin to safety limits.

Should a safety limit be approached, protection would be provided by high pressurizer pressure and overtemperature $\Delta T$ trip. Power and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of load the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 Hz. This resulting overfrequency is not expected to damage the sensors (Non-NSSS) in any way. However, it is noted that frequent testing of this equipment is required by the Technical Specifications. Any degradation in their performance could be ascertained at that time.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, ~~or~~ the overtemperature $\Delta T$ signal or the low-low steam generator water level signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the Reactor Coolant System (RCS) and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control assembly control nor direct reactor trip on turbine trip.

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

A more complete discussion of overpressure protection can be found in reference [9].

## 15.2.7.2 Analysis of Effects and Consequences

### Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 102 percent of full power without direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins.

The total loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN [5] 12, which is described in Section 15.1. The program simulates the neutron kinetics, RCS, presssurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and variables including temperatures, pressures, and power level.

Typical Assumptions are: (pressurizer pressure -46 psi allowance for steady state fluctuations and measurement error)

1. Initial Operating Conditions - the initial reactor power and RCS temperatures are assumed at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value consistent with the 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 accident.

2. Moderator and Doppler Coefficients of Reactivity - the total loss of load is analyzed for both the beginning-of-life and end-of-life conditions. The least negative moderator temperature coefficients at beginning-of-life and a large (absolute value) negative value at end-of-life are used. A conservatively large (absolute value) Doppler power coefficient is used for all cases.

3. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control.

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

5. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the beginning and end-of-life are analyzed:

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

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

6. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of loss of external electrical load.

Reactor trip is actuated by the first Reactor Protection System trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

Results

The transient responses for a total loss of load from 102 percent of full power operation are shown for four cases; two cases for the beginning of core life and two cases for the end of core life, in Figures 15.2-19 through 15.2-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-19 and 15.2-20 show the transient responses for the total loss of steam load at beginning-of-life with a least negative moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the ~~overtemperature ΔT trip channel~~ high pressurizer pressure signal. The minimum DNBR is well above the 1.30 value.

Figures 15.2-21 and 15.2-22 show the responses for the total loss of load at end-of-life assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. *The reactor is tripped by the low-low steam generator water level signal.*

The pressurizer safety valves are not actuated in the transients shown in Figures 15.2-19 through 15.2-22.

The total loss of load accident was also studied assuming the plant to be initially operating at 102 percent of full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2-23 and 15.2-24 show the beginning-of-life transients with

1. Overtemperature $\Delta T$

2. Pressurizer low pressure

15.2.12.2 Analysis of Effects and Consequences

Method of Analysis

1. The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN ~~[5]~~ 12. 53
   The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

In calculating the DNBR the following conservative assumptions are made:

1. Initial conditions of maximum core power and reactor coolant temperatures and minimum reactor coolant pressure (presurizer pressure -46 psi allowance for steady state fluctuations and measurment error) resulting in the minimum initial margin to DNB (See Section 15.1.2.2).

2. A least negative moderator coefficient of reactivity was assumed in this analysis. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. The DNB evaluation is made assuming that core power peaking factors remain constant at their design values while, in fact, the effects of local or subcooled void would have the effect of flattening the power distribution (especially in hot channels) thus increasing the DNB margin.

3. A high (absolute value) Doppler coefficient of reactivity such that the resultant amount of positive feedback is conservatively high in order to retard any power decrease due to moderator reactivity feedback.

Results

Figure 15.2-37 illustrates the nuclear power transient following the accident. Reactor trip on overtemperature $\Delta T$ occurs as shown in Figure 15.2-37. The pressure and core average temperature versus time ~~decay transient~~ following the accident is given in Figure 15.2-38. The resulting DNBR never goes below 1.30 as shown in Figure 15.2-39. The calculated sequence of events for this accident is listed in Table 15.2-1.

Following reactor trip, RCS pressure will continue to fall until flow through the inadvertantly opened valve is terminated. Automatic actuation of the Safety Injection System may occur if the pressure falls to the low pressurizer pressure SI setpoint. 30

September, 1972.

8. Geets, J. M., "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June, 1972.

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

10. Geets, J. M. and Salvatori, R., "Long Term Transient Analysis Program for PWR's (BLKOUT Code)," WCAP-7898, June, 1972.

11. Letter: C. Eicheldinger (W) to D. B. Vassallo (NRC) dated 2/21/75.

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

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

| Accident | Event | Time (sec.) |
|---|---|---|
| Uncontrolled RCCA Withdrawal at power | | |
| 1. Case A | Initiation of uncontrolled RCCA withdrawal at maximum reactivity insertion rate (80 pcm/sec) | 0 |
| | Power range high neutron flux high trip point reached | ~~1.2~~ 1.4 |
| | Rods begin to fall into core | ~~1.7~~ 1.9 |
| | Minimum DNBR occurs | 3.3 |
| 2. Case B | Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/sec) | 0 |
| | Overtemperature ΔT reactor trip signal initiated | ~~24.0~~ 23.2 |
| | Rods begin to fall into core | ~~26.0~~ 24.7 |
| | Minimum DNBR occurs | ~~26.1~~ 25.2 |

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TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

| Accident | Event | Time (sec.) |
|---|---|---|
| Loss of External Electrical Load | | |
| 1. With pressurizer control (BOL) | Loss of electrical load | 0 |
| | High Pressurizer Pressure ~~Overtemperature ΔT~~ Reactor Trip Point Reached | ~~5.3~~ 6.5 |
| | Rods begin to drop | ~~7.3~~ 8.5 |
| | Minimum DNBR occurs | ~~(1)~~ 10.0 |
| | Peak pressurizer pressure occurs | ~~9.0~~ 10.0 |
| 2. With pressurizer control (EOL) | Loss of electrical load | 0 |
| | Low-Low Steam Generator Water Level ~~Overtemperature ΔT~~ reactor trip point reached | ~~5.3~~ 66.4 |
| | Rods begin to drop | ~~7.3~~ 68.4 |
| | Minimum DNBR occurs | (1) |
| | Peak pressurizer pressure occurs | ~~6~~ 7.0 |

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

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

| Accident | Event | Time (sec.) |
|---|---|---|
| 3. Without pressurizer control (BOL) | Loss of electrical load | 0 |
| | High pressurizer pressure reactor trip point reached | ~~3.9~~ 5.0 |
| | Rods begin to drop | ~~5.9~~ 7.0 |
| | Minimum DNBR occurs | (1) |
| | Peak pressurizer pressure occurs | ~~6.5~~ 8.0 |
| 4. Without pressurizer control (EOL) | Loss of electrical load | 0 |
| | High pressurizer pressure reactor trip point reached | ~~3.9~~ 5.0 |
| | Rods begin to drop | ~~5.9~~ 7.0 |
| | Minimum DNBR occurs | (1) |
| | Peak pressurizer pressure occurs | ~~6.5~~ 7.0 |

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

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

| Accident | Event | Time (sec.) |
|---|---|---|
| Accidental depressurization of the Reactor Coolant System | Inadvertent opening of one RCS safety valve | 0 |
| | Reactor Trip | ~~17.6~~ 23.7 |
| | Minimum DNBR occurs | ~~19.6~~ 26.0 |
| Accidental depressurization of the Main Steam System | Inadvertent opening of one main steam safety or relief valve | 0 |
| | Criticality attained | 232 |
| | Pressurizer empties | 130 |
| | UHI injection | 192 |
| | Boron reaches core | 194 |
| Inadvertent Operation of ECCS during power Operation | Charging pumps begin injecting borated water | 0 |
| | Low pressure trip point reached | 51 |
| | Rods begin to drop | 53 |

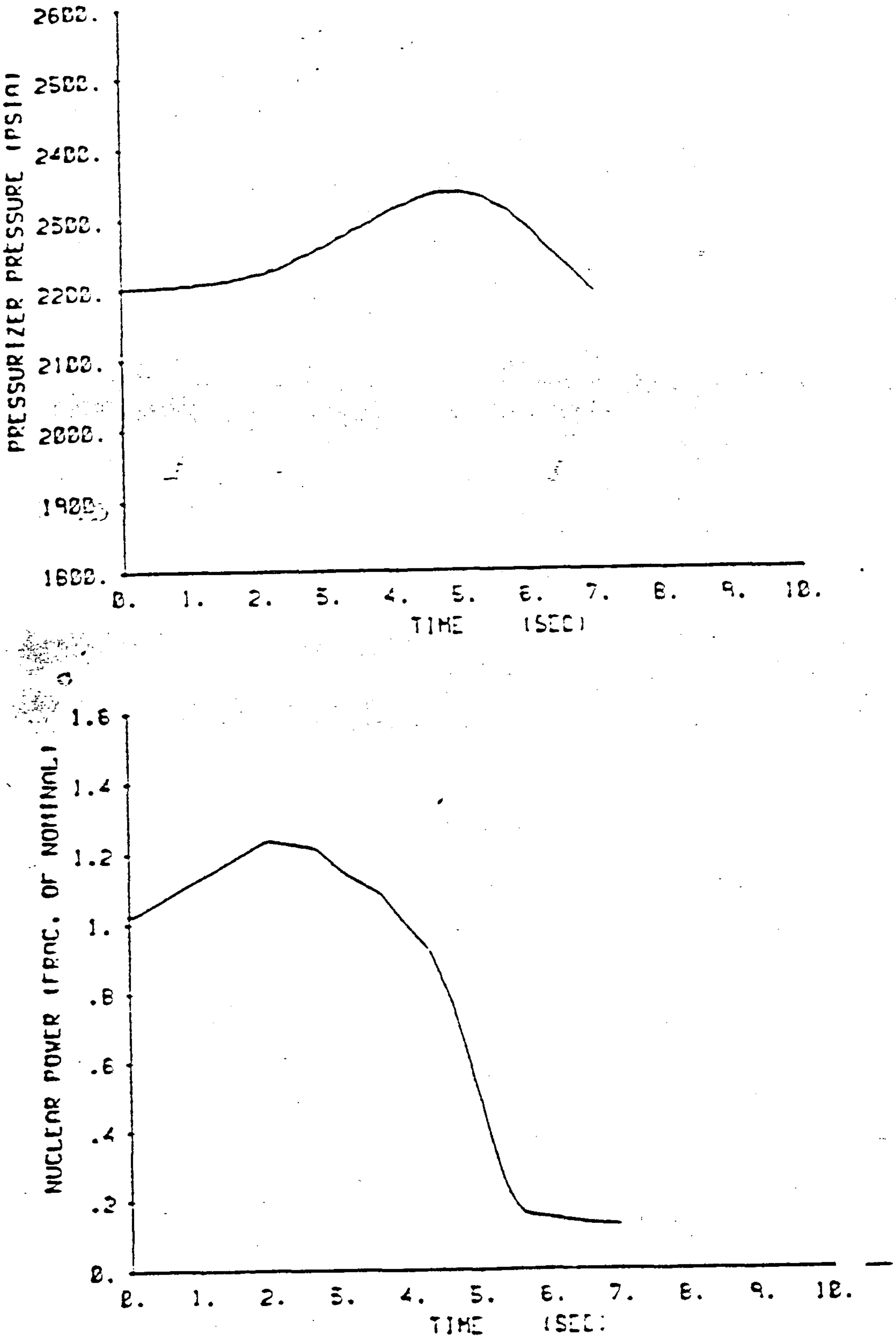


Figure 15.2-4. Pressurizer Pressure Transient and Nuclear Power Transient for Uncontrolled Rod Withdrawal from Full Power with Minimum Feedback and 80 PCM/SEC Withdrawal Rate

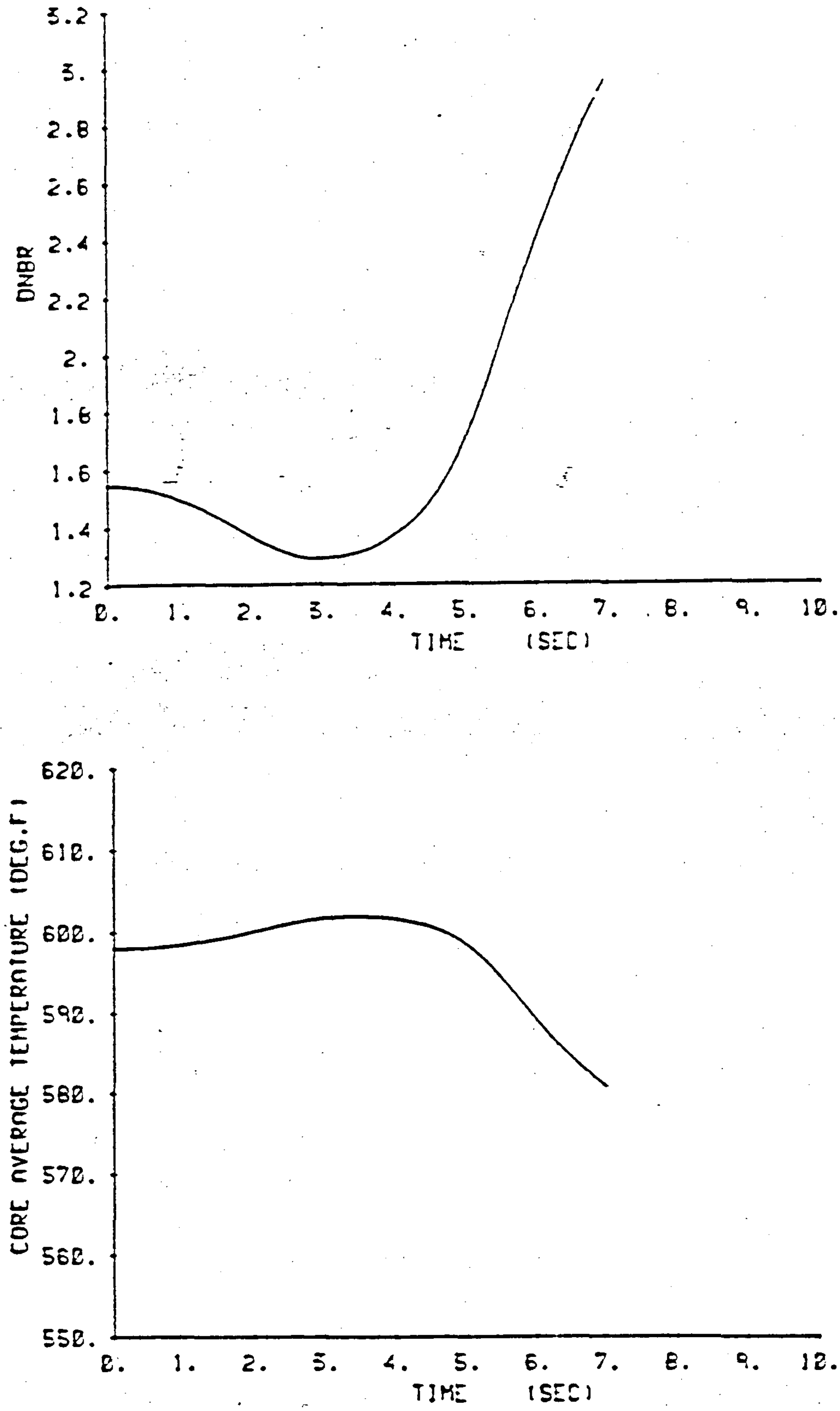


Figure 15.2-5. DNBR Transient and Vessel $T_{AVG}$ for Uncontrolled Rod Withdrawal from Full Power with Minimum Feedback and 80 PCM/SEC Withdrawal Rate

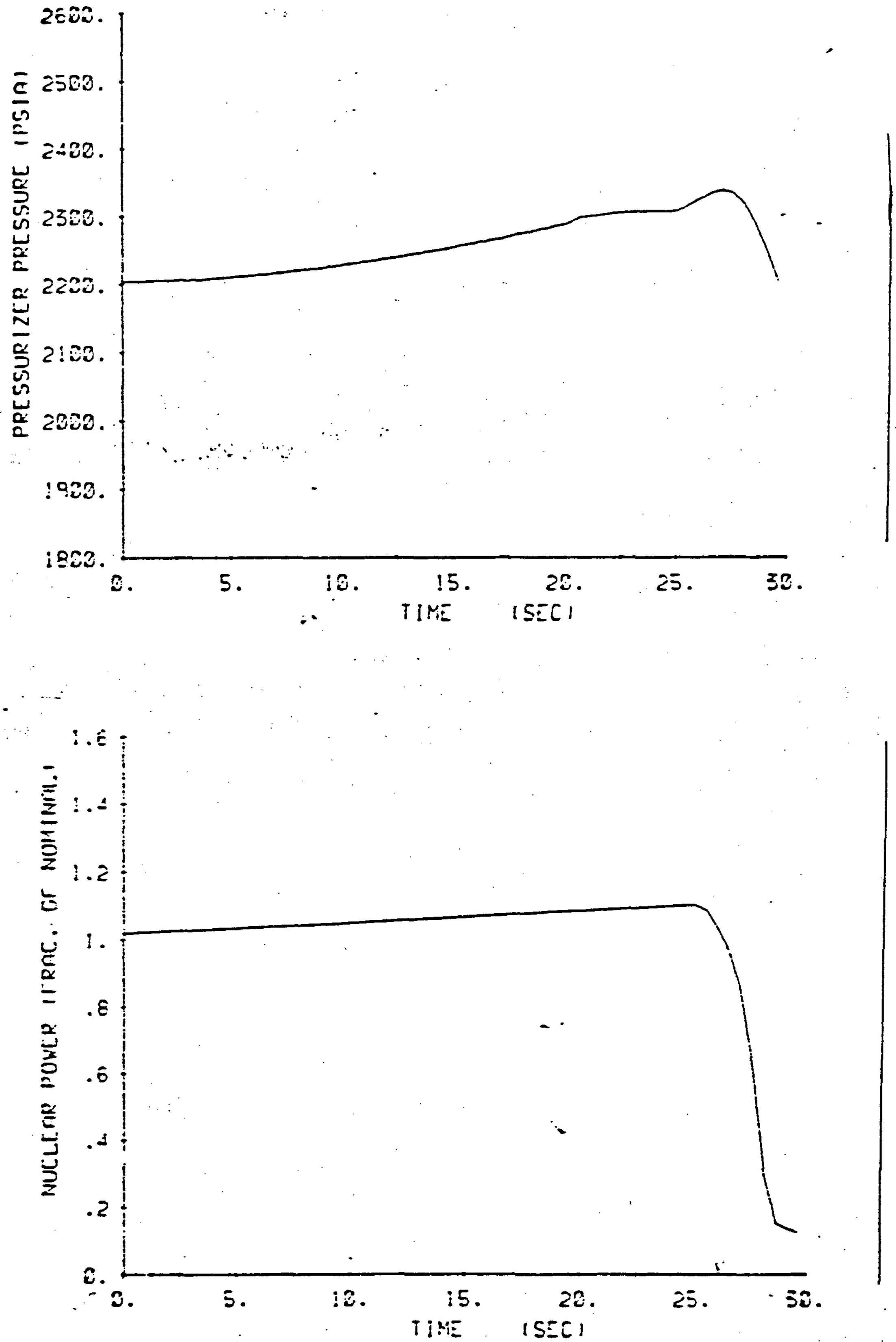


Figure 15.2-6. Pressurizer Pressure Transient and Nuclear Power Transient for Uncontrolled Rod Withdrawal from Full Power with Minimum Feedback and 3 PCM/SEC Withdrawal Rate

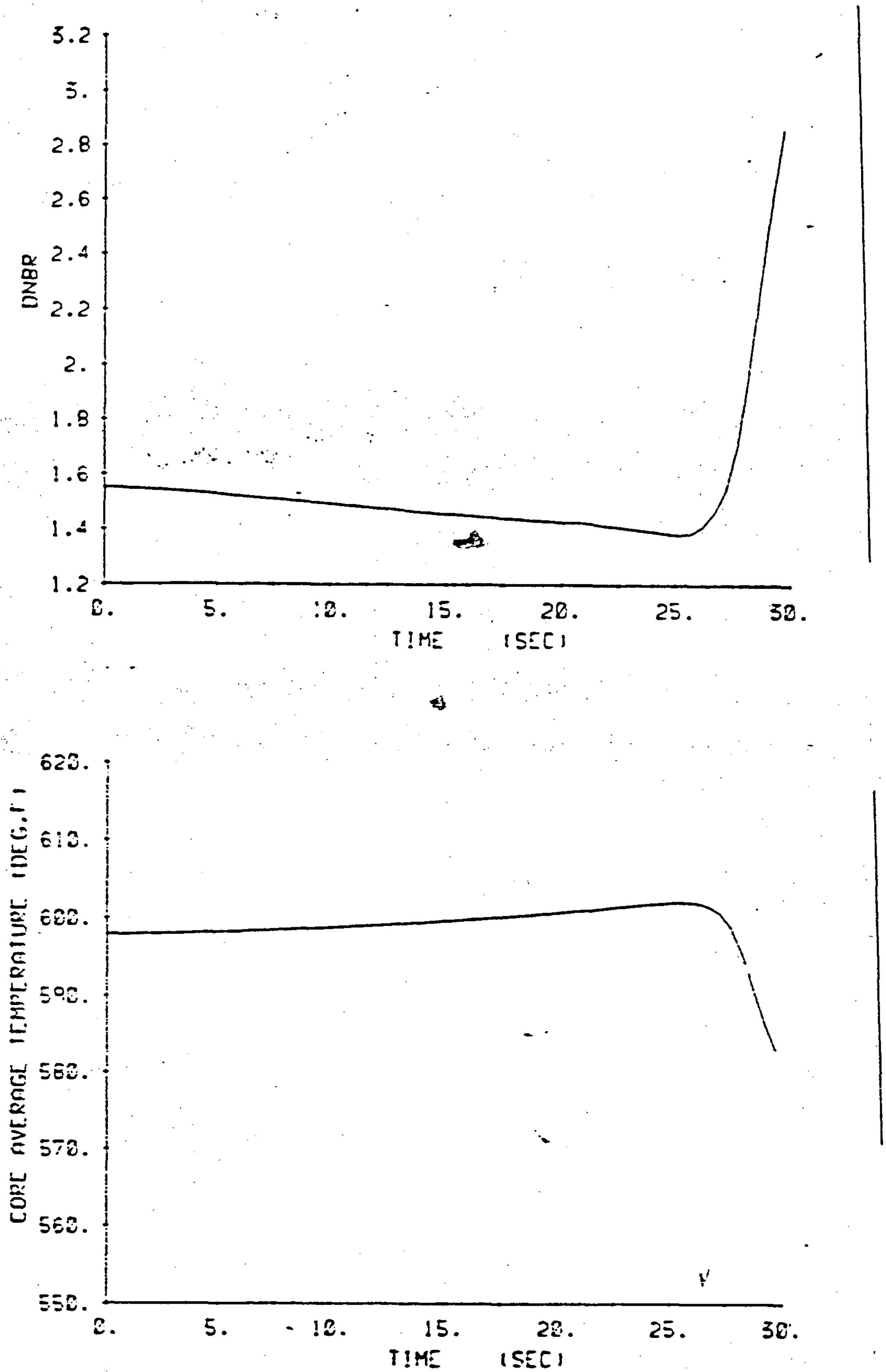


Figure 15.2-7. DNBR Transient and Vessel Average Temperature Transient for Uncontrolled Rod Withdrawal from Full Power with Minimum Feedback and 3 PCM/SEC Withdrawal Rate

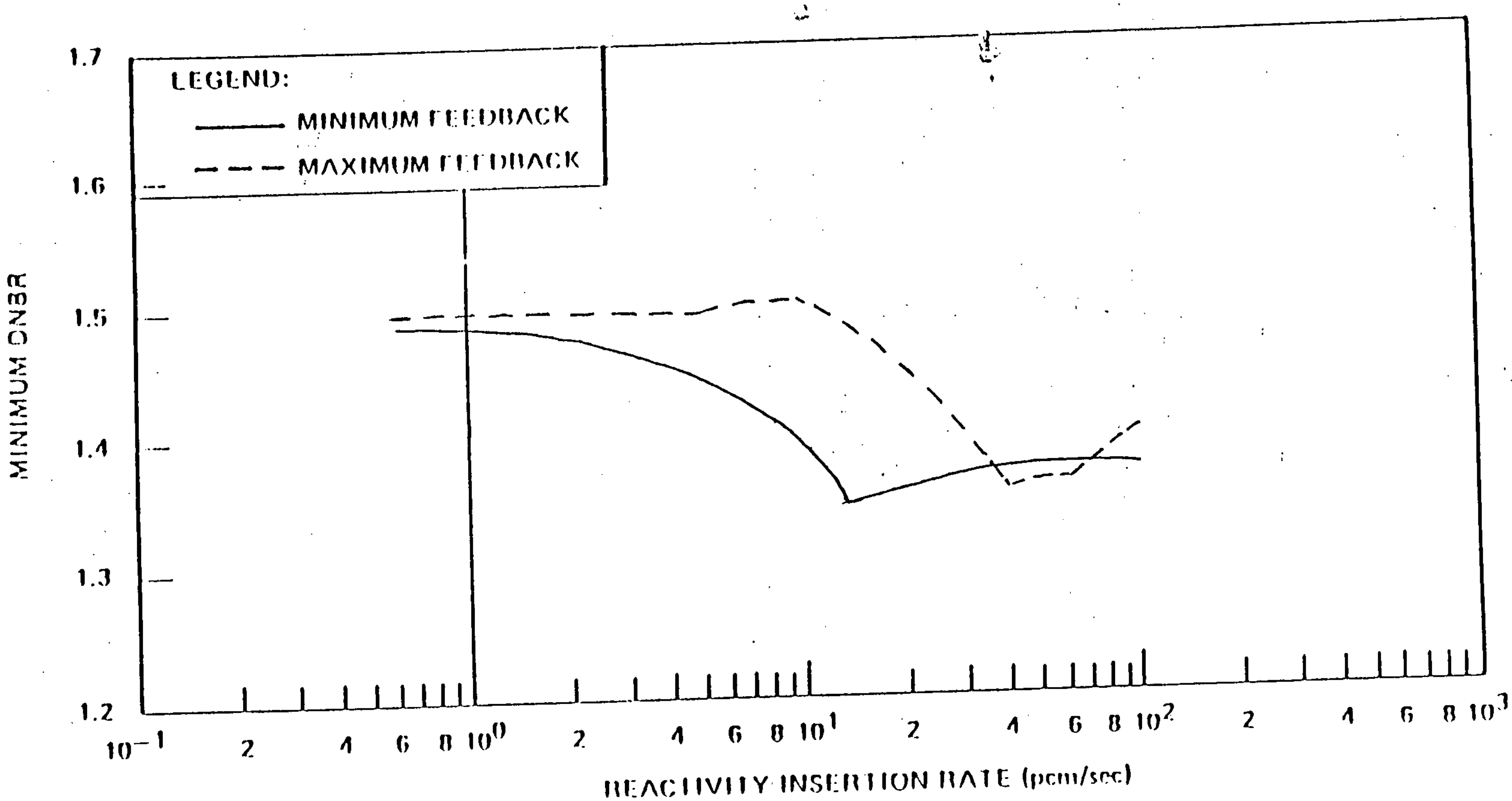


Figure 15.2-8. Effect of Reactivity Insertion Rate on Minimum DNBR for Rod Withdrawal Accident from 100% Power

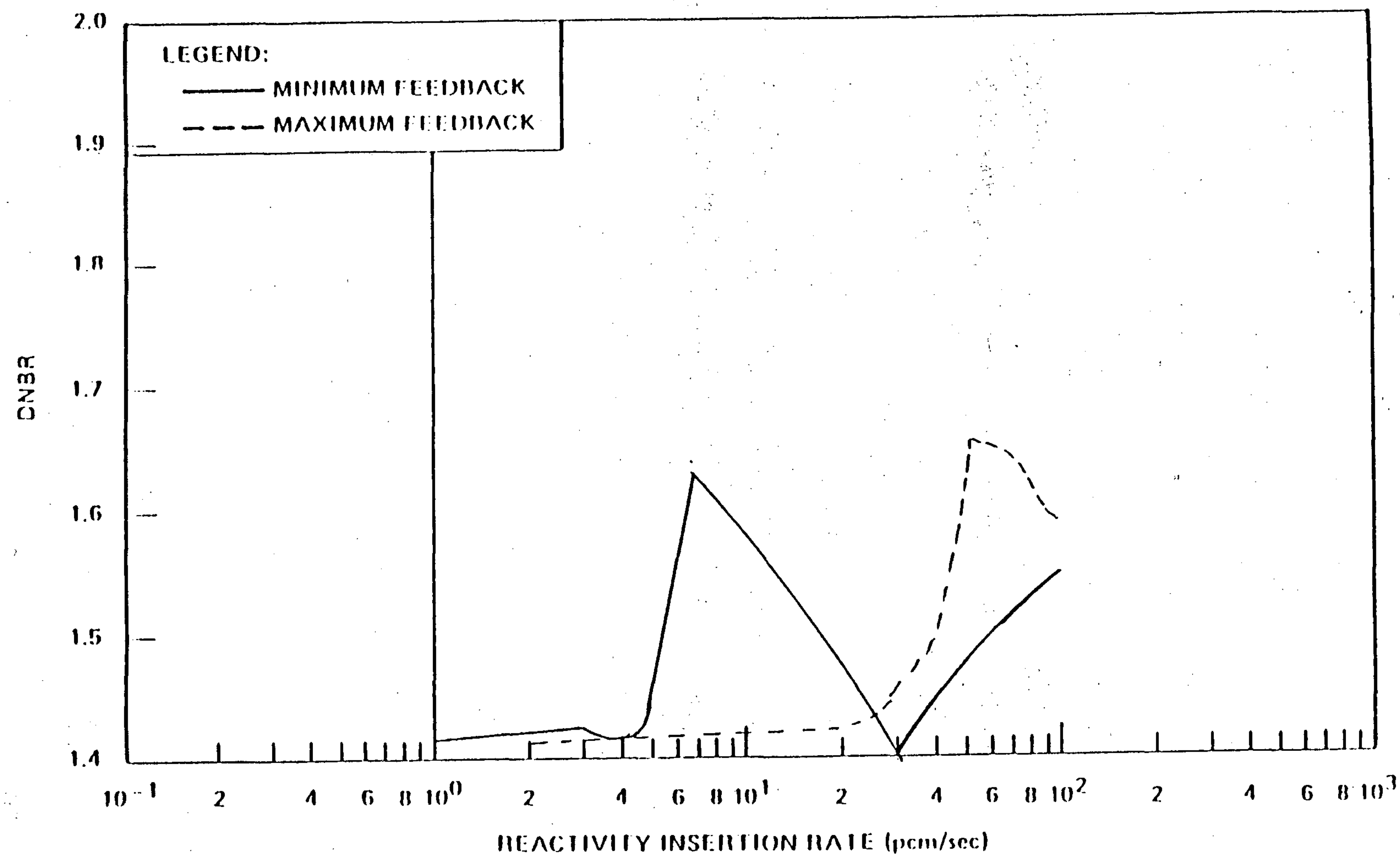


Figure 15.2.9. Effect of Reactivity Insertion Rate on Minimum DNBR for Rod Withdrawal Accident from 60% Power

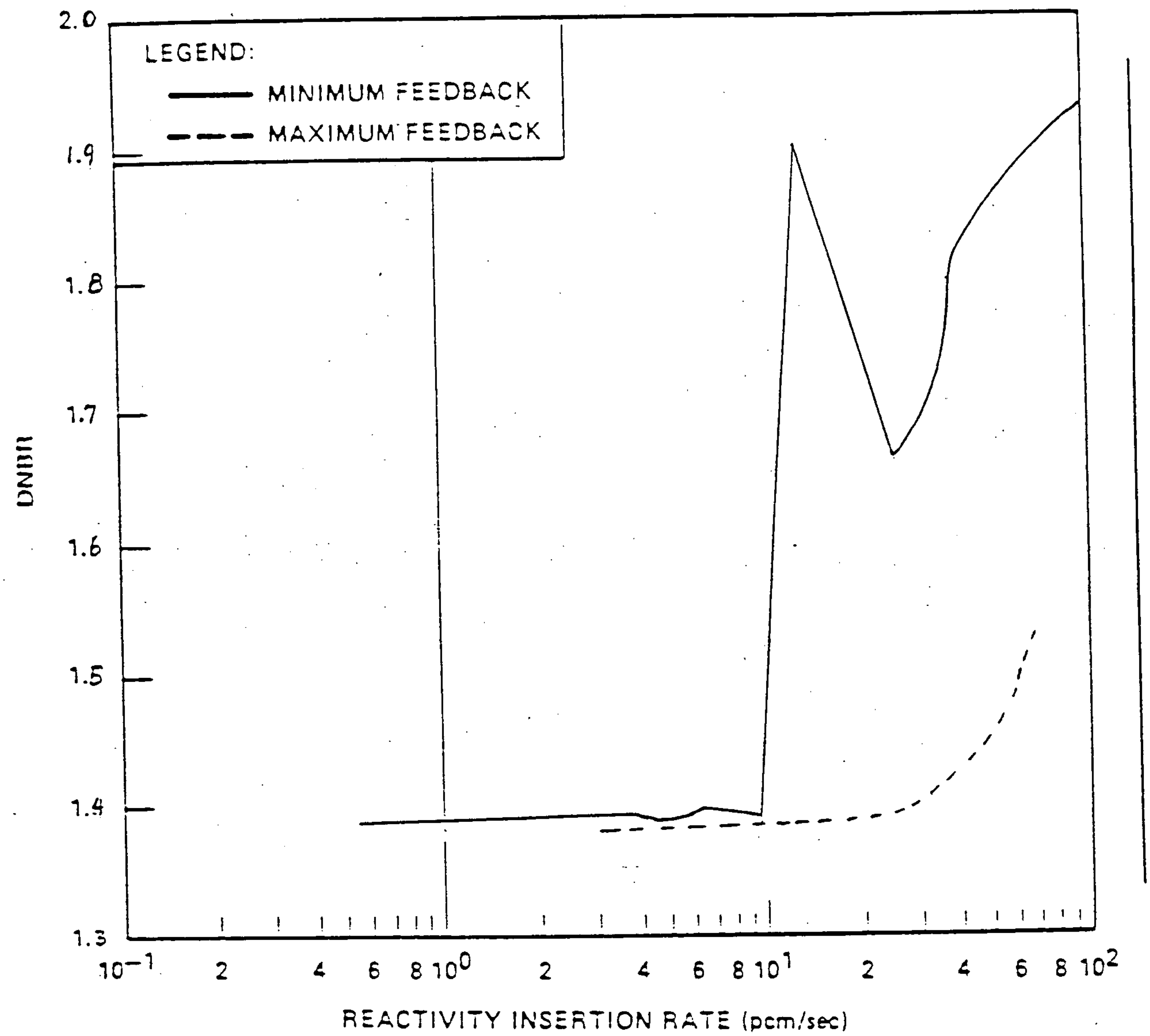


Figure 15.2-10. Effect of Reactivity Insertion Rate on Minimum DNBR for Rod Withdrawal Accident from 10% Power

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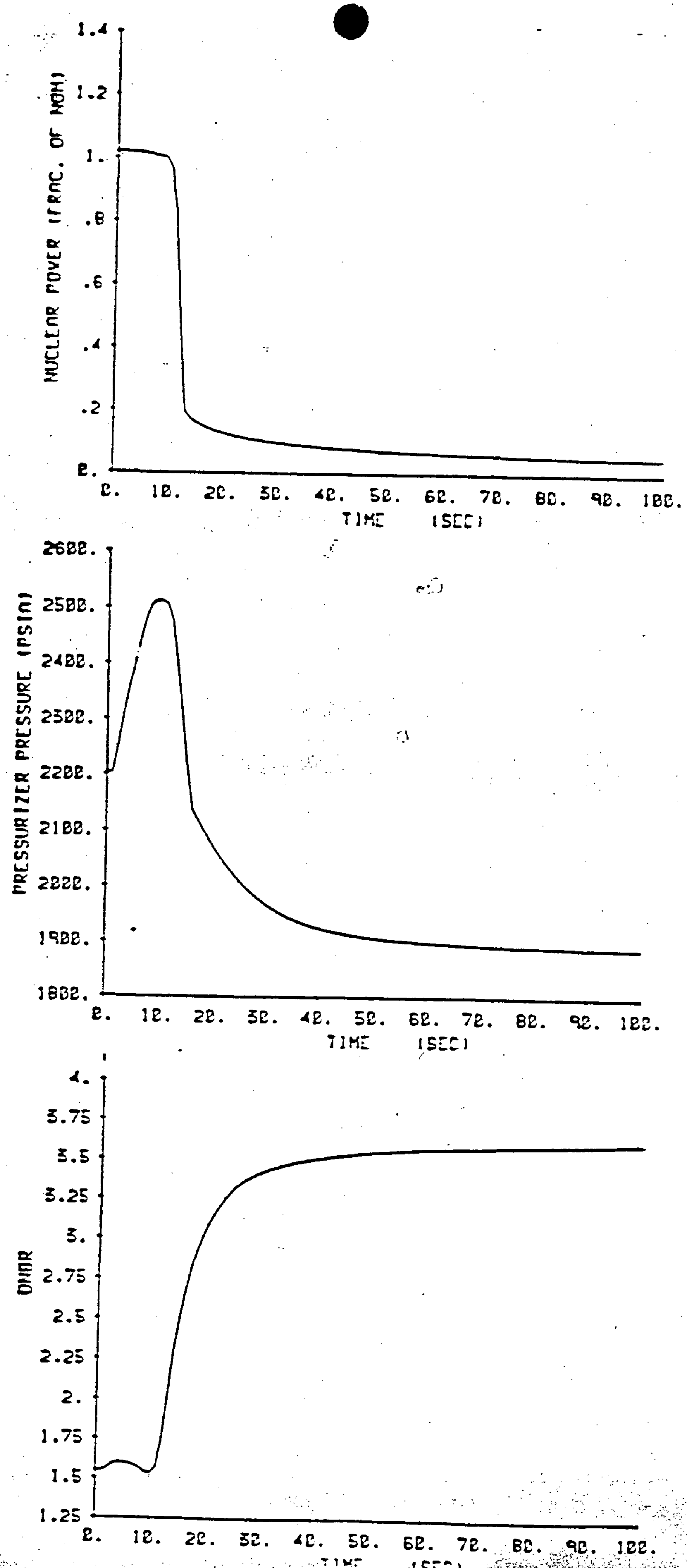



Figure 15.2-19 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves, Beginning-of-Life

Figure 15.2-20 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves, Beginning-of-Life

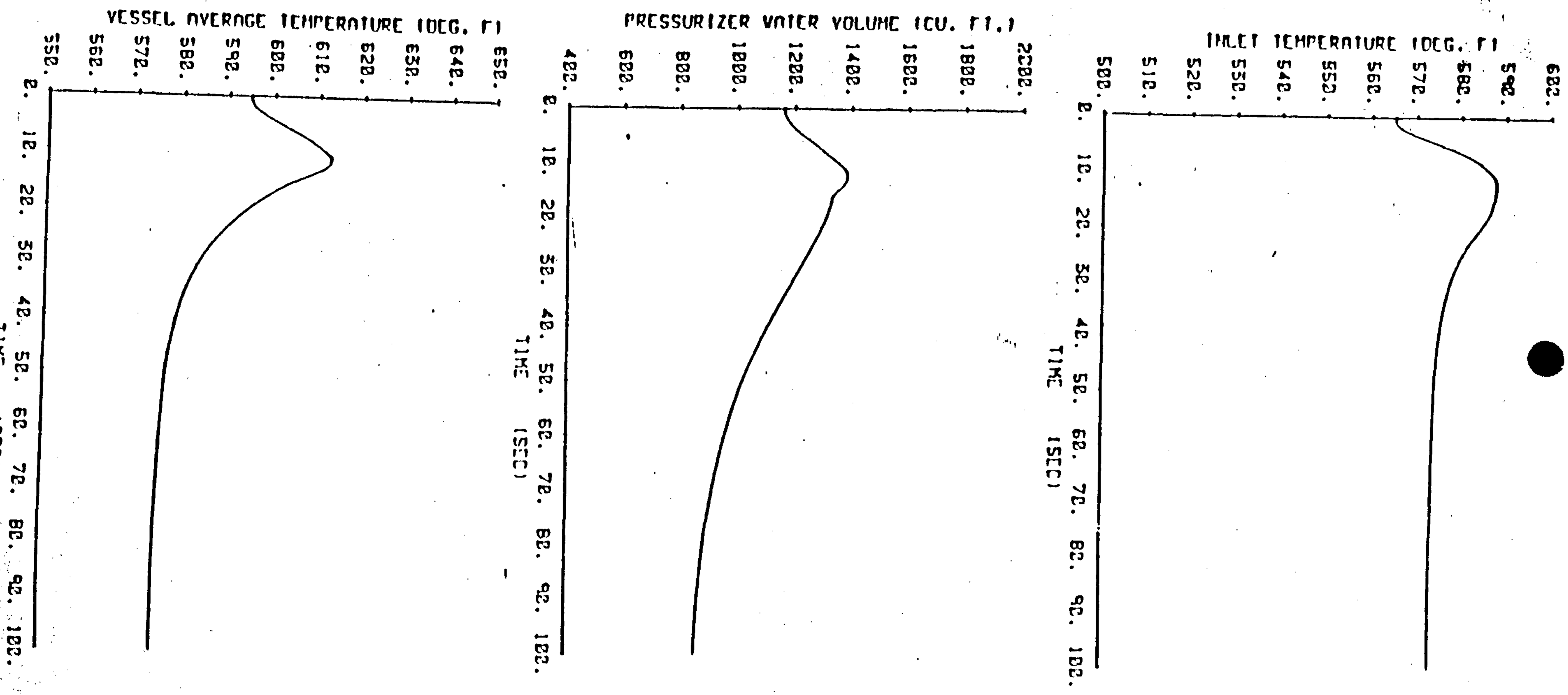


Figure 15.2-21 Loss of Load Accident with Pressurizer Spray and Power–Operated Relief Valves, End-of-Life

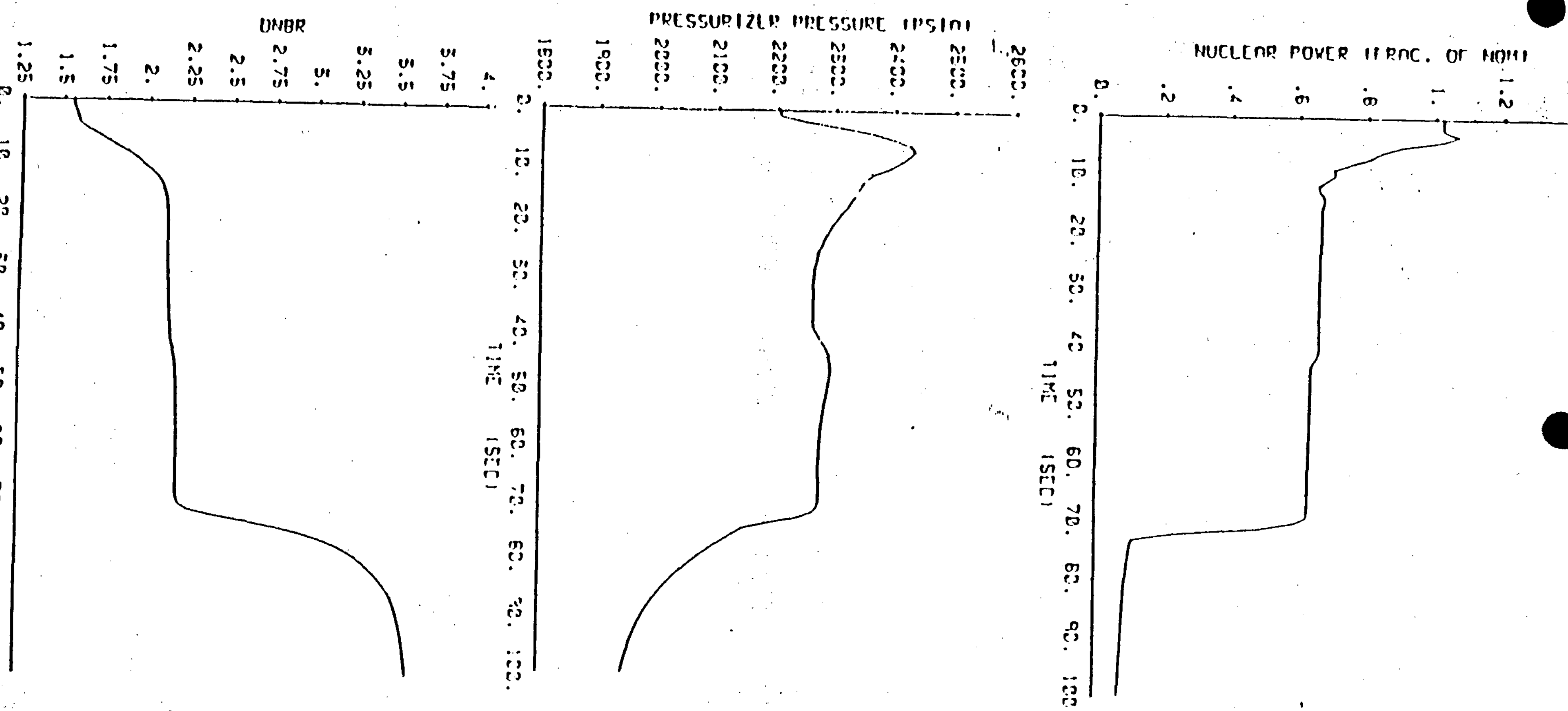


Figure 15.2-22 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves, End-of-Life

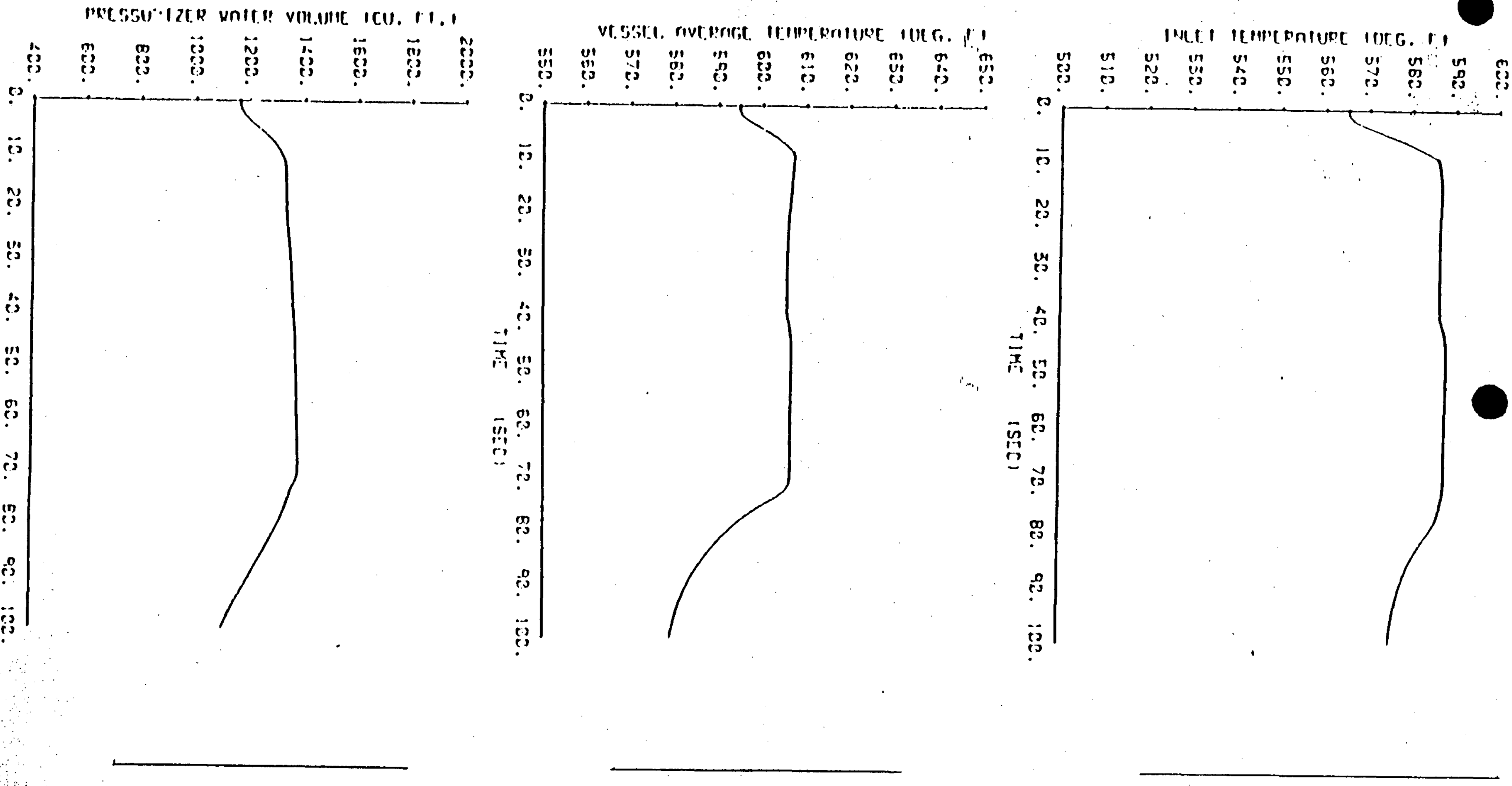
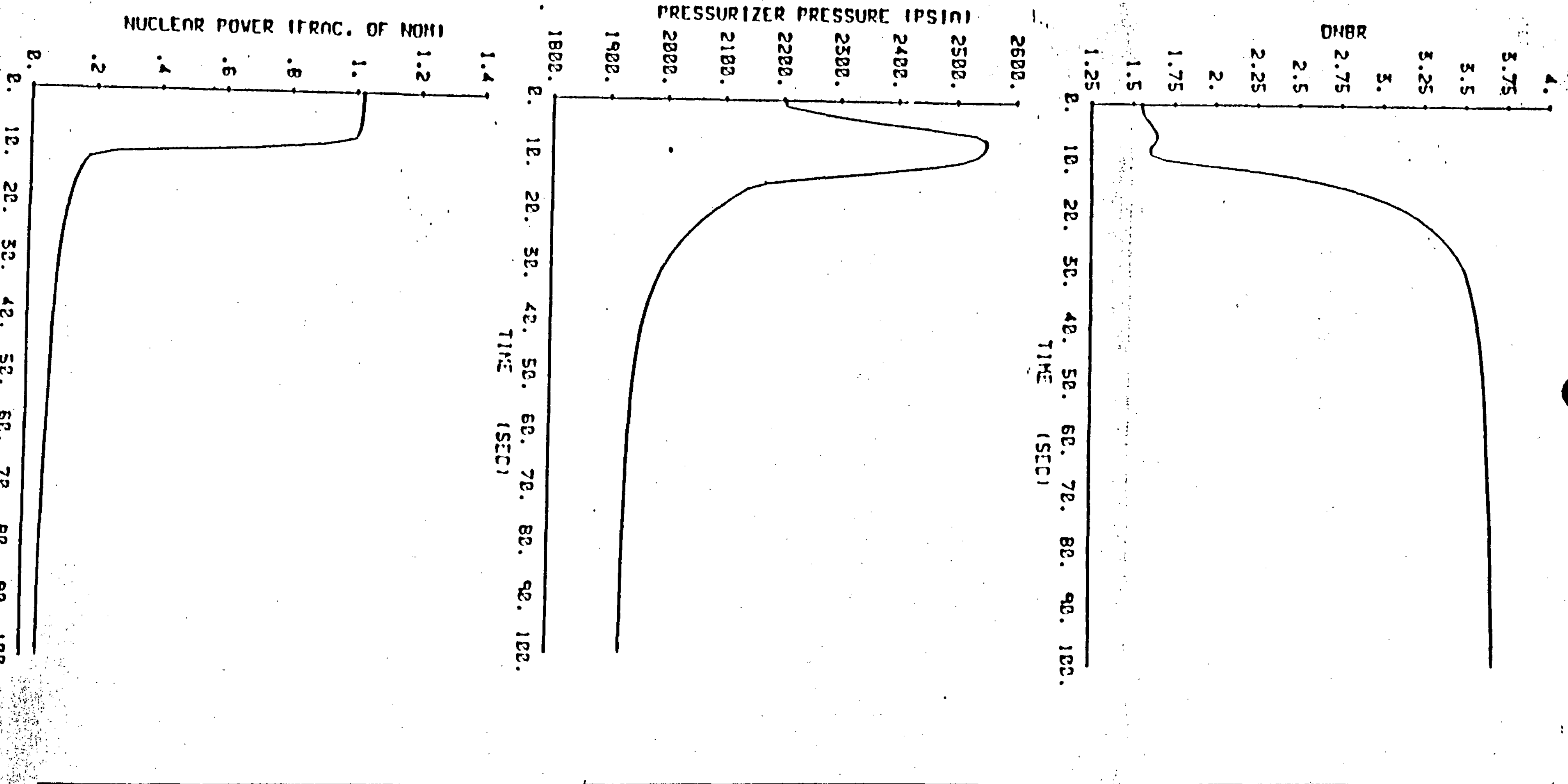


Figure 15.2-23 Loss of Load Accident Without Pressurizer Spray and Power – Operated Relief Valves, Beginning-of Life



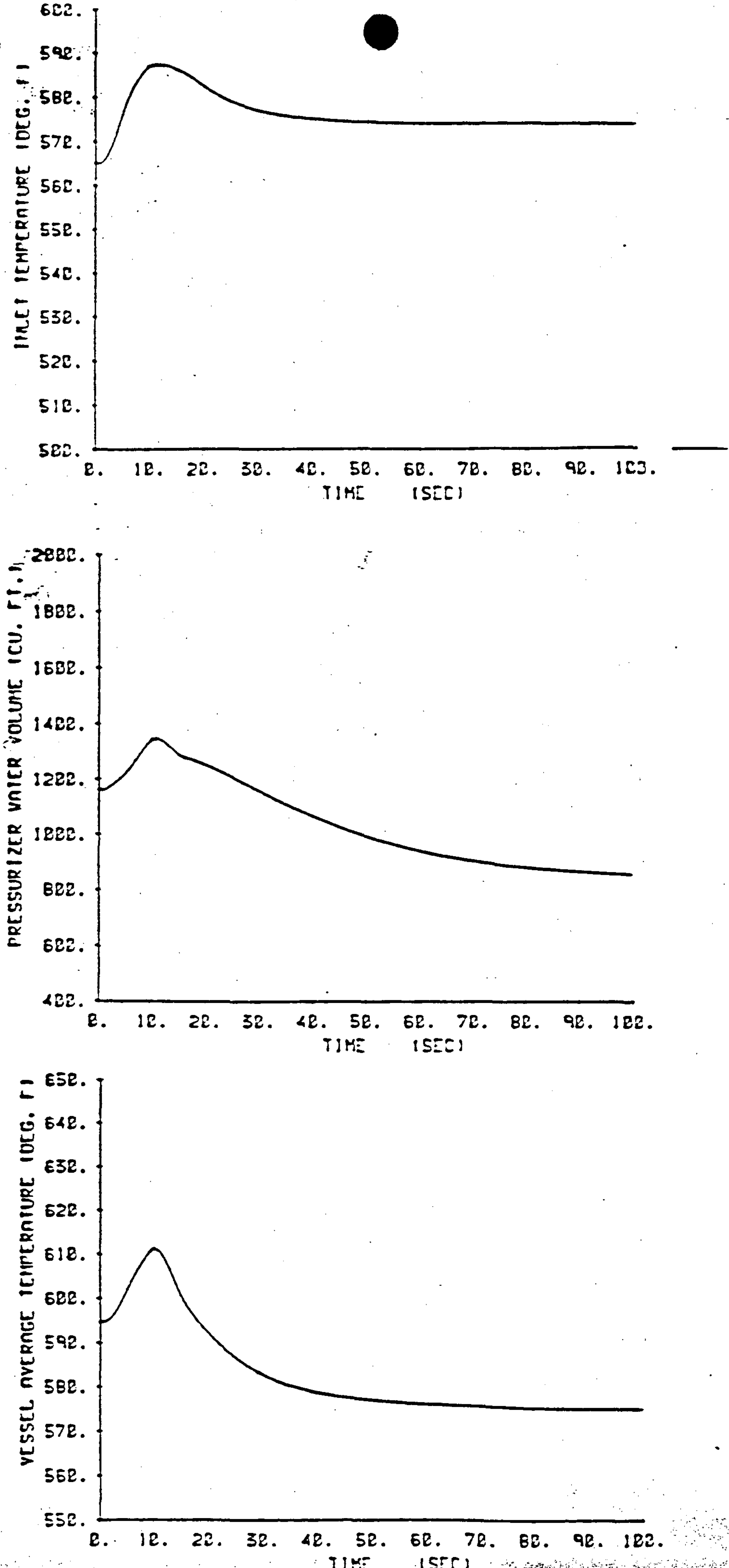



Figure 15.2-24 Loss of Load Accident Without Pressurizer Spray and Power – Operated Relief Valves, Beginning-Of-Life

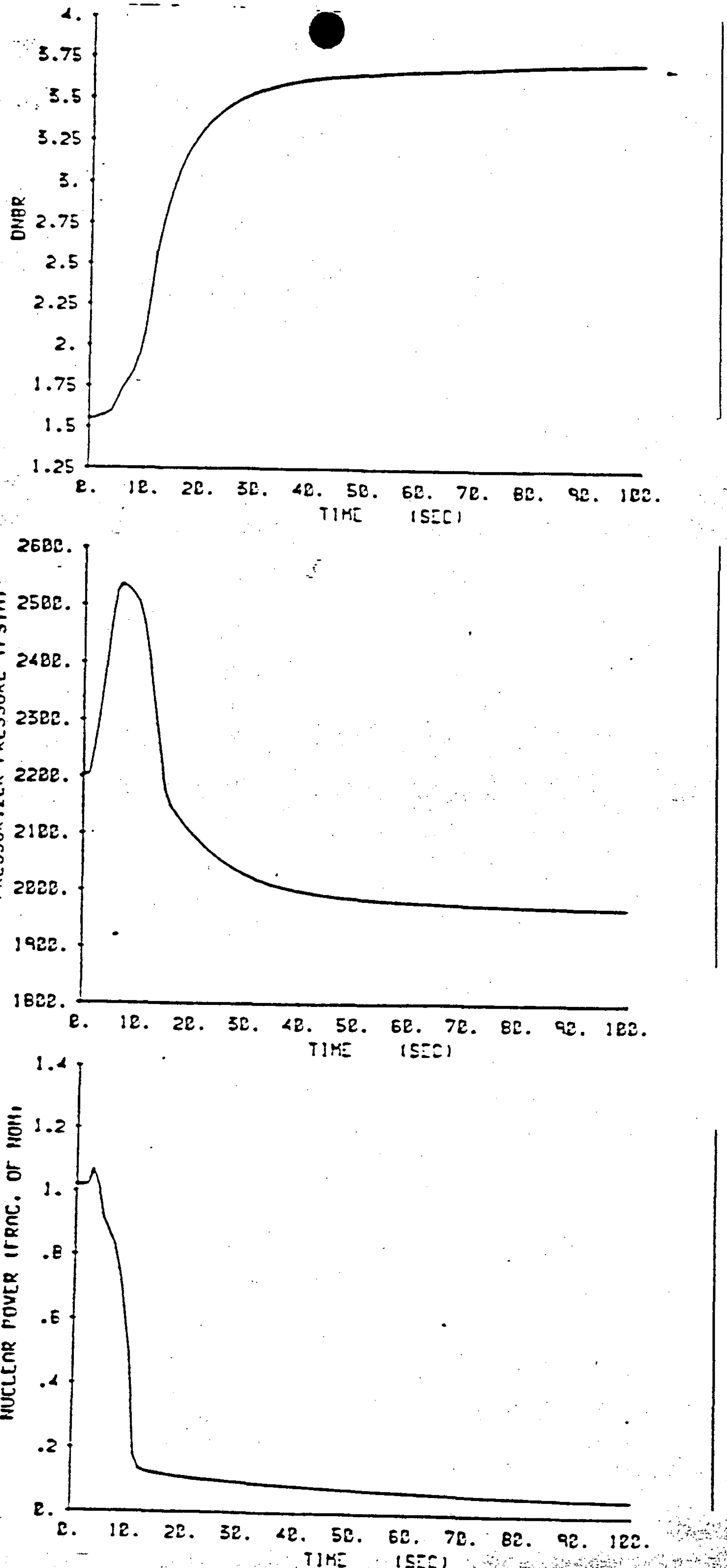



Figure 15.2-25 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves, End-of-Life

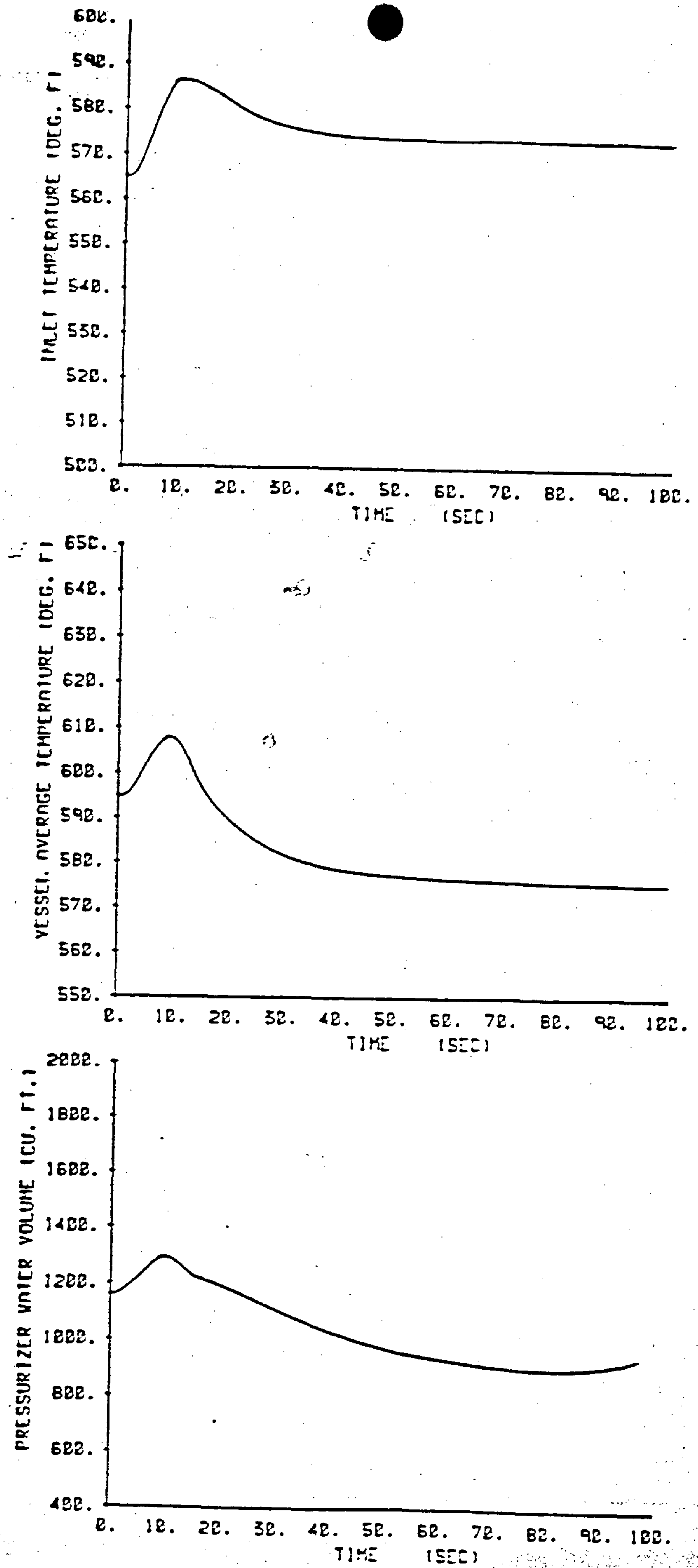



Figure 15.2-26 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves, End-Of-Life

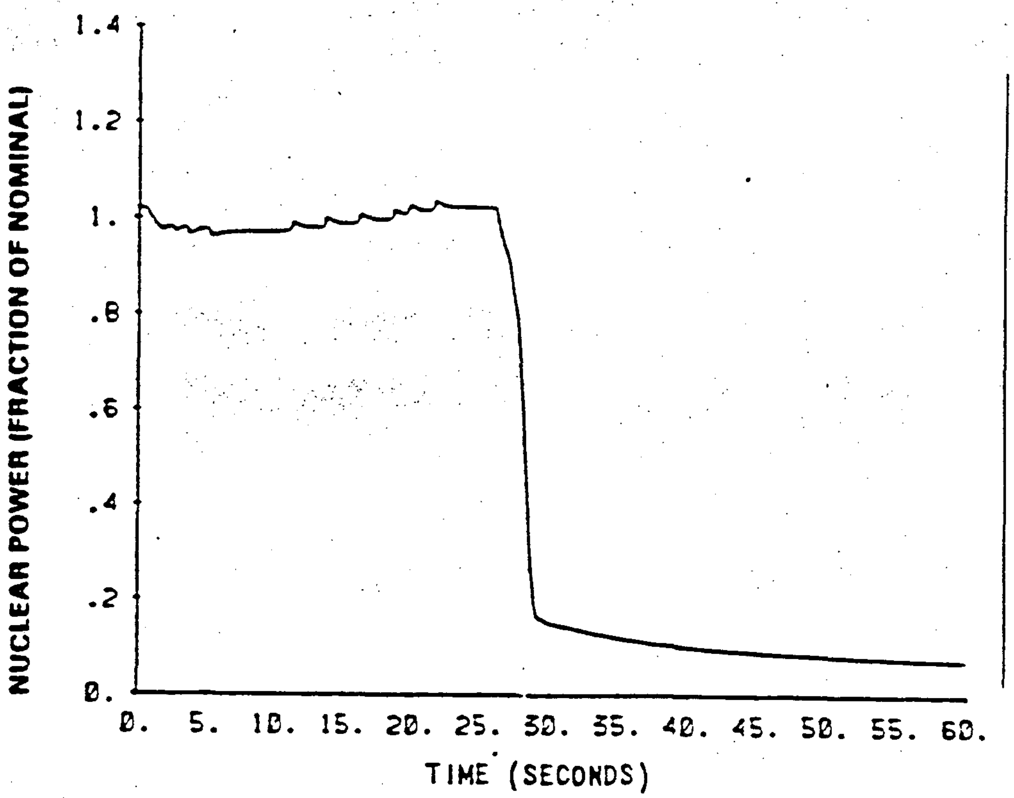



Figure 15.2-37 Power Transient for Accidental Depressurization of the RCS

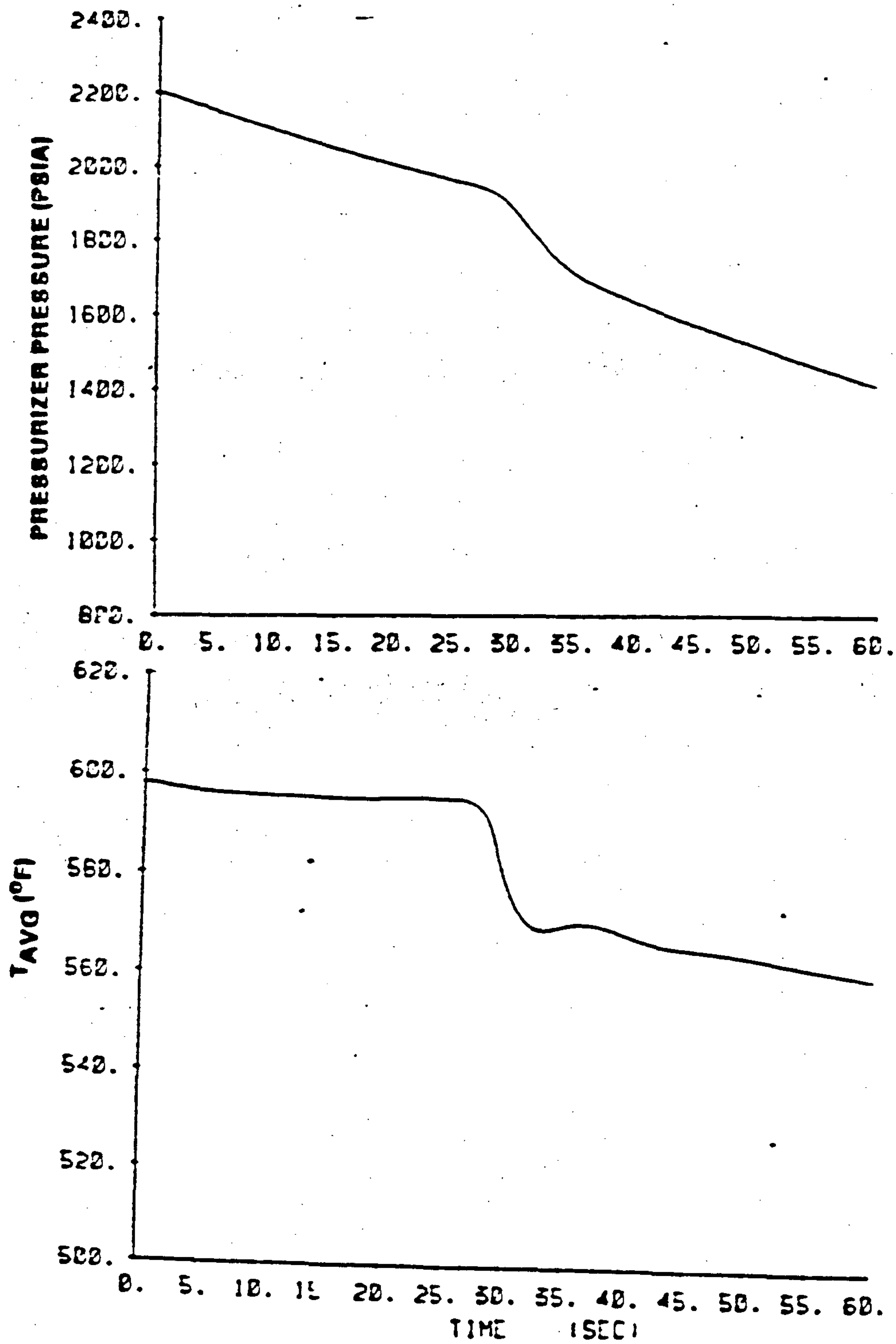



Figure 15.2-38 Pressurizer Pressure Transient and Core Average Temperature for Accidental Depressurization of the RCS

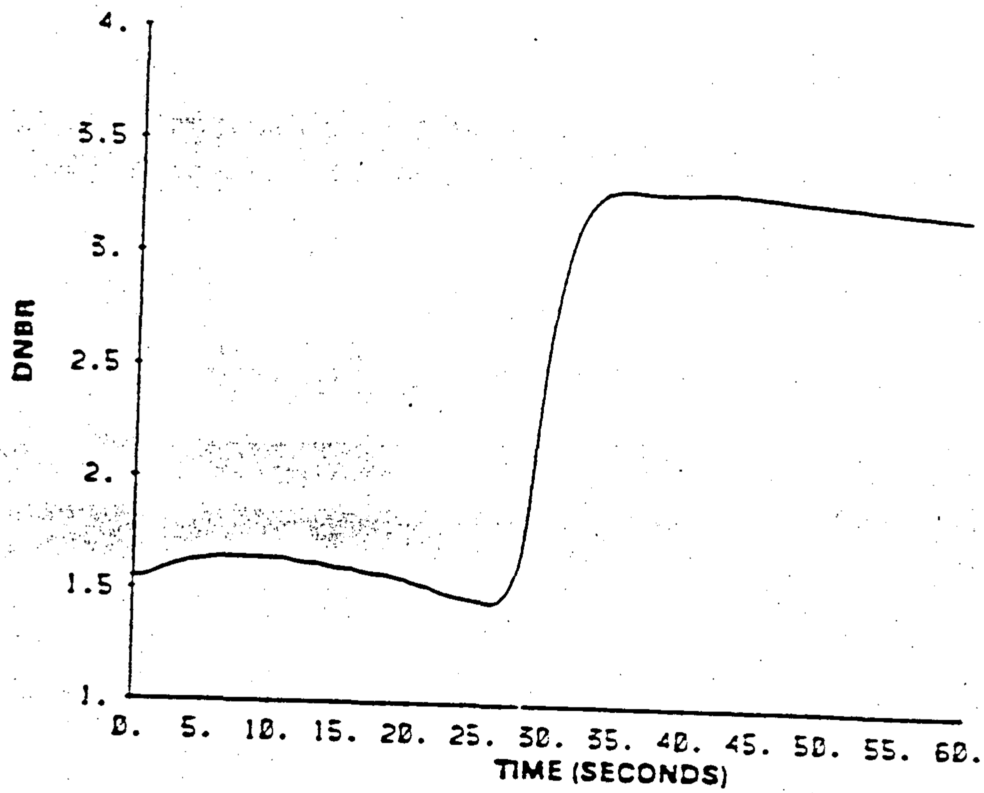



Figure 15.2-39 DNBR Transient for Accidental Depressurization of the RCS